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June 16, 1969

DOUGLAS UNITED NUCLEAR

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SUMMARYREACTOR PLANT OPERATIONSProduction Statistics

	<u>KE</u>	<u>KW</u>	<u>N</u>
Input Production - Pu (KMWD)	99.2	60.0	44.0
- U-233 (Equiv. KMWD)	18.3	14.3	-
Time Operated Efficiency - %	83.1	51.5	38.6
Steam Availability to WPPSS - %	-	-	37.6

C & K Reactors

Maximum power levels at KE (4000 MW) and KW (4010 MW) continued to be limited by inlet water piping brittle fracture considerations. The low TOE at KW reflects a scheduled outage for charge-discharge, maintenance, and modification of the A-B riser cross-tie (Project DCP-522). At both Ks the program to produce fuel grade plutonium is continuing and thorium core loads are being irradiated to make U-233.

Fuel discharge from C Reactor was completed on May 4, and process water to the reactor was cut off the next day.

N Reactor

Maximum power level was 4000 MW, the administrative limit. There were three reactor outages, one scheduled and two unscheduled.

The charge-discharge and maintenance outage was extended for (1) cleanup of the graphite cooling system following an inadvertent raw water dump, and (2) the discharge, inspection, and recharging of 54 fuel columns to assure replacement of all newly-charged fuel which might have sustained damage due to the improper assembly of tube charging equipment. Following startup it was necessary to shut down briefly to remove debris from a few graphite cooling system tubes which became plugged during the startup. The other unscheduled outage resulted from a flow monitor trip.

FUEL AND TARGET FABRICATIONProduction Statistics (tons)

	<u>For KE & KW</u>	<u>For N</u>
Billets Extruded	-	34.9
Finished Fuel Produced	142.9	37.0
Thorium Canned	4.8	-

DECLASSIFIEDK Fuels

AlSi canning operations continued on the basis of two lines per day, five days per week. Production of hot-die-sized fuel was interrupted to permit relocation of the nickel plating machine from Building 3716 to Building 313.

N Fuels

Output production was slightly below forecast because quality control personnel were diverted from fuel assembly to N Reactor to assist with the inspection of fuel potentially damaged in charging.

TECHNICAL ACTIVITIESK Reactors

In the brittle fracture program, ten fracture toughness tests were completed by the Lawrence Radiation Laboratory using test temperatures of 33 F and 70 F on wedge opening loading samples precracked at rates no greater than 1 microinch per cycle as recommended in ASTM-STP-410. Samples were previously precracked at rates of 6 to 7 microinches per cycle. Some preliminary test results showed that (1) fracture toughness values do not appear to be influenced by precracking rates in the range of less than 1 to 7 microinches per cycle, (2) temperature dependence of fracture toughness of the parent metal of the riser steel was consistent with that shown by the Charpy V-notch data, (3) samples fractured at -76 F (-76 F data considered representative K_{Ic} values) with no apparent yielding, indicating that plane strain conditions were satisfied for the sample thickness, and (4) the data from the three risers sampled varied no more than 17 percent at -76 F, 13 percent at 33 F, and less than 5 percent at room temperature.

Radiochemistry results on the five thorium elements recently sampled from the 2586-KW monitor column discharged in October 1968 agreed closely with the U-233 and U-232 data obtained earlier from column 2886-KW.

Loading changes made in the 10 kg plutonium irradiation test block at KE during April have accomplished their purpose. Driver tube powers at the edge of the block, suppressed after the initial loading, are at near-normal levels and the power in the PuAl columns has increased to the intended range.

Twelve target samples of americium for the production of medical-grade Pu-238 were charged into a single column at KE Reactor on May 27. The charging of neptunium in a four-column cluster configuration and in a fringe blanket column is planned for the June outage at KW; these elements will be used to evaluate techniques for reducing the Pu-236 contaminant in Pu-238.

A reliability analysis of the total K reactor process coolant system was completed and documented. Included were the BPA-powered primary system, the steam-powered secondary system, and the diesel-powered tertiary system

(common to both Ks). The quantitative probability of failure values calculated for each case are intended primarily for case comparisons; their use as precise indications of system performance is limited by some variability in the quality of available plant equipment data. These probabilities can be used as accurate bases for evaluating proposed modifications to the K process coolant system.

N Reactor

The first large central zone loading of Mark IV fuel columns (132) was charged and 17 columns were discharged. The latter were part of the initial Mark IV production model test charged in July 1968.

Radiometallurgical examination of the incipient Mark IV fuel failure discharged in February shows that the clad cracking on the inner bore of the outer element apparently was the result of discharge damage. Fuel discharge damage has increased in frequency, and this may result from the added step to the discharge procedure whereby the new fast-cart system dumps fuel assemblies down a 12-foot chute.

An outer fuel element component was heated inductively to the point of clad failure (but below the uranium melting point) in a flowing steam atmosphere, and a continuous measurement of the hydrogen evolved as a result of Zr-H₂O and U-H₂O reactions was obtained. Flow and percent hydrogen recorders indicated that uranium metal was continuously being exposed to the oxidizing atmosphere for at least the first 20 minutes of the test. Results have been compared with analytical studies for unfailed fuel, and with a molten pool metal-water reaction model used in N Reactor safety analyses. Future tests will be performed in the molten temperature range and will investigate the full spectrum of fuel conditions.

Test results look very promising on a new device for tube flow throttling and orificing which may eliminate the troublesome V-11 valves. This device is a movable sleeve designed for installation in the tube inlet nozzle. The sleeve can be adjusted for full flow or reduced charge-discharge flows.

FEATURE REPORT

This month the appended special summary report describes the successful development of an uncooled horizontal control rod for the K reactors. The prototype of this rod was installed in KE in January, and its performance to date has been excellent.

GENERAL

The reduction in force as the result of the shutdown of C Reactor is proceeding according to schedule. Of the 74 employees removed from the rolls

thus far, 60 percent have been by voluntary layoff. An additional 20 percent have been placed in jobs with other Hanford contractors.

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

Charles D. Harrington
Charles D. Harrington
President

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REACTOR PLANT OPERATIONS - KE & KWPRODUCTIONGeneral

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

The production of nondefense plutonium in the 8-10 percent Pu-240 range is continuing with enriched uranium at the K reactors. Thoria loads are being irradiated in both Ks.

Statistical Summary

	<u>KE</u>	<u>KW</u>	<u>Total</u>
Input Production - Pu (KMWD)	99.2	60.0	159.2
- U-233 (Equiv. MWD)	18,333	14,333	32,666
Power Level (MW) - Maximum	4,000	4,010	8,010
- Average	3,851	3,757	7,608
Time Operated Efficiency - %	83.1	51.5	67.3
Number of Outages	2	2	4
Number of Startup Interruptions	0	1	1
Operating Coolant Flow - M gpm	189.6	190.0	379.6

	<u>KE</u>	<u>KW</u>	<u>Total</u>
Fuel Charge (Tons) - Natural U	3.3	3.1	6.4
- Enriched U	356.0	359.4	715.4
Fuel Element Failures	0	0	0
Helium Losses - M cu.ft.	85.6	71.0	156.6

OPERATING EXPERIENCE

Reactor Loadings

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

The discharge of fuel from C Reactor was completed on May 4, and water to the reactor was cut off the next day.

Power Levels

Maximum power levels shown for the K reactors represent continuing operation under precautions taken to insure adequate coolant flow in the event of a brittle fracture failure of inlet piping. The PuAl fuel charged under PTA-150 also reduced the KE power level slightly because of less than normal heat in the test block.

Reactor Outages

Four reactor outages were experienced as summarized below:

<u>Date Down</u>	<u>Reactor</u>	<u>Outage Hours</u>	<u>Remarks</u>
May 3	KE	41.1	High speed scanner trip caused by an RTD failure. Miscellaneous maintenance work was accomplished.
May 3	KW	41.9	Malfunction of power supply on high speed scanner. Miscellaneous maintenance work was accomplished.
May 17	KW	309.8	Scheduled outage for charge-discharge, installation of DCP-522 (A-B crosstie revision), improvements to the high speed scanner, and miscellaneous maintenance work.
May 24	KE	84.6	Scheduled outage for charge-discharge, charging of americium (PTA-171), and miscellaneous maintenance work.

EQUIPMENT EXPERIENCEZircaloy Tube Replacement - KW Reactor

KW tubes 1354 and 1355 were removed for examination. Tube 1355 was replaced with a new Zircaloy tube sleeved with a cobalt alloy tube for test irradiation purposes (see Mission 2, Section D). The graphite in channel 1354 was badly broken and on two occasions the broach became stuck; on the second, the broach could not be recovered without prolonging the outage. The channel was temporarily "buttoned up" with the broach near the rear gunbarrel.

DCP-522 A-B Crosstie Modification - KW Reactor

The A-B crosstie modification was completed during the May 17 outage. Inspection, flushing, flow tests and ATPs were performed with satisfactory results. Crossheader screens on all four risers were inspected prior to startup.

Ball 3X System - KW Reactor

Seven additional 3X bellows were shrouded at KW Reactor, completing the program planned through calendar year 1970. Also at KW, the debris was removed from hopper No. 55 and it was returned to service.

Header No. 59 Check Valves - KW Reactor

Stainless steel check valves have been reinstalled on crossheader No. 59 at KW. These replaced the carbon steel valves that have been in use since the original valves were removed for destructive testing in December as a part of the brittle fracture program.

Deactivation - C Reactor

Deactivation work has progressed steadily since the reactor was shut down on April 25.

Construction forces have completed filling the bottom of the 107-B north and south basins and are presently filling the bottom of the 107-CE basin with dirt.

It is estimated that the overall reactor deactivation work is 60% complete and water facility deactivation work is 50% complete.

Columbia River Telemetering

A Minor Design Change is being prepared for relocation of the river telemetering equipment presently located in the 183-C Building. Since there are no operating personnel in this building, the receiver and recorder for river temperature and level will be moved to the 165-KW Building.

ADP Equipment

The movement of Production Computing personnel and auxiliary equipment to the

UNIVAC 9300 facility in the 1101-N Building was completed. The leased ADP machines were delivered to the AEC for return to the IBM Company. All Production Computing work is now being performed on the UNIVAC 9300 system.

PROCESS ASSISTANCE & CONTROL

Operational Physics

Flattening efficiency was generally 2-3 percent higher during May than in April, thus allowing slightly increased equilibrium power levels with the tube power limits. The 225-tube PuAl test block in KE is operating very near the intended power, but splining for long-term gains is still done outside the block. The rod configuration in the top of the reactor near the block is normal. Several loading changes (thoria to 94 Metal) will be made during the next scheduled outage to bring the block to the intended level.

There continue to be no significant operational physics problems with the E-Q loads. Considerable effort has been made to optimize charge-discharge timing to minimize the spline requirements for total control. Certain restrictive regions of the reactor can be discharged early in the outage while credit can be taken for xenon.

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

	Reactor	
	KE	KW
Effective Central Tubes (ECT)*	2250	2271
Flattening Efficiency** - May	0.72	0.73
- 12-mo. average	0.70	0.71
Maximum Operating Time Permitting Scram Recovery - Hours***	10	10

*Reactor power level divided by the average power of the 10 most productive tubes which are representative of the reactor loading.

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

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

Production Fuel Performance

There were no production fuel element failures during May.

The following table shows production fuel failure frequencies, as number per

million elements discharged, for the 12- and 24-month periods ending April 20:

	<u>12 Months</u>	<u>24 Months</u>
C Reactor - Natural Uranium	15	32
- 94 Metal	32	28
K Reactors - Natural Uranium	7	7
- 94 Metal	3	6

U-233 Yield and Quality Monitoring

Radiochemistry results on the five thoria elements recently sampled from column 2586-KW, discharged in October 1968, agreed closely with the U-233 and U-232 data obtained earlier from column 2886-KW and reported in DUN-5625, "Interim Report, PTA-137, E-Q Monitor Columns".

The detailed comparisons are shown below (corrected to total Pa-233 decay; the exposure of column 2586 was approximately 2 percent higher than that of 2886):

Piece No.	<u>Column 2886-KW</u>		<u>Thoria Column 2586-KW</u>	
	<u>Gm U-233</u>	<u>U-232</u>	<u>Gm U-233</u>	<u>U-232</u>
	<u>Ton Thoria</u>	<u>U-233 (ppm)</u>	<u>Ton Thoria</u>	<u>U-233 (ppm)</u>
11	1730	10.44	1780	11.07
15	2033	10.81	2115	10.87
21	2018	11.34	2239	10.67
30	1637	9.40	1657	9.45
34	1241	7.78	1240	6.29

Analysis is continuing on the interpretation of central element data from fringe column 0167-KE, in which the source for U-232 buildup would be expected to be more dependent on fissions in the thoria column itself relative to the source from other columns.

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

Loading Pattern - KE Reactor

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

Loading Pattern - KW Reactor

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REACTOR PLANT OPERATIONS -N

PRODUCTION

General

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

Statistical Summary

Input Production - Pu (KMWD)	44.02
Electrical Generation - KMW	206.29 (per WPPSS)
Power Level (MW) - Maximum	4,000
- Average	3,678
Time Operated Efficiency - %	38.6
Steam Availability - %	37.6
Number of Shutdowns - Scheduled	1
- Unscheduled	2
Fuel Failures	0
Fuel Charge (Tons) - 94 Metal	307.6
- 125 Metal	68.3
- Natural U	0.4
Total	376.3

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Helium Losses - M cu.ft.	239.3
Fuel Oil Usage - bbl.	15,339

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-6.

Charge-Discharge

During the scheduled outage started on May 8, 204 tubes of fuel were charge-discharged. During this work, the 3-inch monotube liner inserts failed on eight monotubes. Following charge-discharge, it was discovered that the nozzle adapter transition piece was oriented 180 degrees from its proper position. Push-pole discharge and inspection of five of the newly-charged, unirradiated fuel columns which took high charging pressures showed damage to fuel supports and support shoes missing. A total of 54 columns of unirradiated fuel, selected on the basis of increased charging pressure, were push-pole discharged and inspected to provide assurance that the remaining newly-charged fuel was usable. Also, one process tube, 2661-N, was borescoped and showed no serious damage.

Graphite cooling tube 1588 containing a cobalt target was scheduled for discharge during the May outage. The charge stuck in the tube and was not removed. Coolant flow through the tube is normal. Discharge of the target is to be rescheduled.

Power Level

Power level continued to be administratively limited at 4000 MW. It was possible to maintain main steam header pressure (MSHP) at approximately 123 psig during the month, thus maximizing the electrical generation capability of WPPSS.

Reactor Outages

The reactor was operating at maximum authorized power level at the beginning and at the end of the month. The three reactor outages and their principal causes were as follows:

<u>Date Down</u>	<u>Outage Hours</u>	<u>Cause</u>
May 8	450.4	Scheduled outage for refueling and extensive maintenance outage extended for cleanup of the graphite cooling system following an inadvertent raw water dump.

<u>Date Down</u>	<u>Outage Hours</u>	<u>Cause</u>
May 27	4.6	Unscheduled outage to clean debris from a few graphite cooling system tubes.
May 29	1.8	Automatic scram due to a flow monitor trip on process tube 0458 caused by improper setting.

EQUIPMENT EXPERIENCE

Isolation of A-Bus

The A-bus 230 KV, 13.8 KV, and 4.16 KV electrical systems were removed from service for four days to perform Equipment Maintenance Standards work. All A-bus Standards work has now been completed for 1969 with the exception of one 230 KV lightning arrester on the 230 KV to 13.8 KV transformer. This will be scheduled when parts are available.

Primary Loop

During the scheduled outage, a number of primary system leaks were repaired. This work included: (1) repacking and installing Mark IV operators on 17 V-11 valves, (2) repacking and installing manual operators on 7 V-11 valves, (3) replacing 3 V-12 valves, (4) installing a rebuilt actuator on the V-4-6 valve, then repacking and returning it to service, and (5) adjusting the packing on all pressure relief valves in the pressurizer penthouse.

In an effort to determine the extent of the stress corrosion which resulted in a leak on the high pressure impulse line from the venturi to the Barton transducer (causing the April 10 scram), two sections of flow monitor sensing lines were removed during the scheduled outage for further engineering evaluation.

Work continued on Cell 2 retubing.

The turning gear on No. 6 drive turbine was damaged when the unit was test run following the A-bus outage work. The pony motor used to power the unit for the test turned backwards. The motor had been reconnected in accordance with normal markings which are in reverse for the N application.

Gas Atmosphere

Helium loss during equilibrium operation in May prior to the scheduled outage averaged 17,700 cubic feet per day compared to 14,500 cubic feet per day in April and 12,300 cubic feet per day in March. A major gas leak was found on the rear Omega seal. This leak was repaired, as was also a small leak on the gas system makeup flow transmitter. Losses following the repairs appeared to be leveling off at about 7,000 cubic feet per day.

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Graphite Cooling System

Graphite cooling system heat exchanger No. 4 was taken out of service and moved to the 189-F Building for decontamination by plant forces and retubing by Combustion Engineering.

A reactor building B-bus electrical outage occurred on May 13 while A-bus was isolated for maintenance work. As a result, the graphite dump system was operated, allowing raw water to flow single-pass through the graphite cooling tubes for about five minutes. Debris in the raw water caused plugging of orifices in the graphite system and required a major effort and an outage extension to correct the problem. All graphite cooling system tubes were checked, some several times, before the system was ready for startup. The primary shield system was also checked for cleanliness prior to startup, including flow transmitter orifices.

Inspection of other raw water users was also made to assure systems were clean. This included drive turbine and TG set surface condensers, graphite heat exchangers, raw water storage tank, and the raw water screen wash system at Bldg. 181. Based on the possibility that trash carry-over at the screens may contribute to the debris problem, procedures were instituted to provide more visual observation of the operation. No. 1 graphite cooling system heat exchanger sustained a high differential pressure during its return to service (following inspection) which resulted in several tube leaks. These leaks were repaired prior to startup. The right side primary shield flows were erratic during startup, and the flow on the right side was subnormal; however, outlet tube temperatures remained about normal.

Ball Safety System

Twenty-seven selected hoppers were test dropped as part of the ball system reliability program. All worked satisfactorily. During the May outage hoppers 17, 97, and 100 were repaired and returned to service. Also, five hopper mechanisms were reworked because of rusty appearance or some indication of erratic opening during the testing. Radioactive balls from the ball system and ball storage were placed in shielded containers and stored inside the earthen shielding wall of the 1310-N storage tank. This move reduced radiation levels around the ball recovery system.

Nuclear Flux Monitor System

During the May outage, both sub-critical monitor channels were repaired and returned to service. New chambers were installed for the 3A high-level channel and the 3A galvanometer channel. The chamber was removed from 1B galvanometer channel and 4B galvanometer channel was returned to service.

Confinement System

The confinement system integrated system test was completed and exceptions corrected. There were no major deficiencies. D-cell filters at the 117 Building were differential pressure tested satisfactorily.

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Boiler Experience

Combustion Engineering boiler No. 1 was shut down on May 3 for tube leak repairs. It was returned to service on May 6. The Foster Wheeler was shut down for tube leak repairs on May 18 and returned to service on May 19. Combustion Engineering boiler No. 1 was shut down again for flue gas leak repairs on May 28. A leak in a floor tube was also located. Repair of leaks and installation of new refractory in No. 1 boiler continued through month's end. Production Test NR-115, "Establishing CE Boiler Performance to Permit Reactor Startup on CE Boiler Steam Supply," was initiated during the plant startup on May 27 but was aborted due to the failure of No. 1 CE boiler. The Foster Wheeler boiler was given a full inspection during the May scheduled outage.

Circulating Raw Water System

No. 2 circulating raw water pump was removed from service to permit pump and motor overhaul. In order to have two B-bus powered pumps for the ensuing operating period, the power supply for No. 2 CRW pump was jumpered over to CRW pump No. 1. The condition will remain until No. 2 unit is returned to normal service.

Equipment Modifications

The following equipment modifications were completed during the month:

Minor Design Change N-69-27, "Intermediate Range Flux Monitor Power Supply Current Meter," which authorized replacement of the existing intermediate range power supply meters with a selectable range ammeter in channels 1, 2, and 3.

EMP-353, "Installation of Fixed Area Radiation Monitoring System."

EMP-441, "105 and 109-N Steam Vent Circuit Changes," which changed portions of the steam vent automatic closure circuits to increase the reliability of the steam vent control system.

PROCESS ASSISTANCE & CONTROL

Operational Physics

With the charging of spike material in the left side, top, and bottom portions of the spike ring, the flattening imbalance of last month was alleviated. Last month's imbalance was caused when almost the entire right side only of the spike ring was recharged with spike material. The excess reactivity low point will be maintained at about 3 mk, now that operating experience has shown this to be an acceptable minimum. The transition from inboard to outboard rod patterns as required has been very successful for operation with a low excess reactivity.

The 180 Mark IV fuel columns recently charged into the central zone are operating near the reactivity and power generation levels expected. Conversion of

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the central zone to this fuel model is therefore not expected to cause any significant operating physics problems. The calculated 6 percent increase in power generation of Mark IV fuel in a Mark I lattice has proven to be only about 3 percent and, therefore, no power spikes have occurred.

Some operational physics parameters of interest are:

Effective Central Tubes (ECT)	803
Flattening Efficiency - May	0.80
- 10-mo. average	0.82
Maximum operating time permitting scram recovery - hours	24

Operations Analysis

The report DUN-5685, "Basic Time Operating Efficiency Characteristics of N Reactor," was completed and issued. Included in the report are the development and analysis of a theoretical time operating efficiency model of N. Variables considered are the fuel charge-discharge rate, the quantity of fuel charged-discharged per outage, critical path maintenance time, critical path checkout time, and the number of forced outages per year.

Control Rod Pitting

The program to remove two N Reactor control rod seal section tubes, one chromic acid cleaned and one not cleaned, for destructive examination to determine the extent of aluminum pitting and the contribution of the cleaning to pitting, was partially completed during the May outage. Control rod 41 was cleaned and a seal section tube removed for examination; examination is in progress. The uncleaned tube was not removed, however, for lack of a replacement.

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	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	
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Fuel Code	No. Tubes	Description	PT-NR No.	No. Tubes	Description
C	94	Mk-IC (94 Metal Fringe)	76	4	Fuel Meltdown Test
CE	61	Mk-IC (94 Metal Fringe)	94+	6	Mk-IV Demonstration
D	164	Mk-IC (94 Metal - High U-236)	96	16	Mk-I From Upset-Forged Billets
E	219	Mk-IC (94 Metal - Fringe - Central)	01	12	Mk-I From Direct-Cast Billets
F	112	Mk-IV (94 Metal - High U-236)	07	47	Initial Full Length Mk-IV Columns
G	21	Mk-IV (94 Metal - Central)	■	1	Blank Channel or Empty Tube
N	1	Mk-I (Natural U)			
X	246	Mk-IA (1.25 - 94 Metal)			
	918	Total		86	Total
	86	Total PTs			
	1004	Grand Total			

+ Includes Mk-IV High U-236 Content Fuel and 2 Tubes w/Mk-IA 1.25-94 Metal and Mk-IV 94 Metal.

LOADING PATTERN - N REACTOR

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FUEL AND TARGET FABRICATION - K REACTORS

PRODUCTION

General

Production of AlSi-bonded and hot-die-sized (HDS) fuel for the K reactors was 102.1 percent of forecast. Sixty-five percent of these elements had bumpers or self-supports attached.

HDS fuel production was shut down at mid-month to provide additional manpower for the thorium wafer program, and to allow construction forces to relocate the nickel plating machine from the 3716 Building to Building 313.

Acceptable Elements Produced

	Finished Production (tons)	Yield - Percent	
		Current Month	FYTD
AlSi-Bonded Fuel	134.3	95.3	96.1
Hot-Die-Sized Fuel	8.6	83.1	82.1
Thorium	4.8	87.1	96.1

Month-End Inventories

	Tons
Bare Uranium Cores	955*
Finished Fuel: AlSi-Bonded	1,259*
Hot-Die-Sized	56
Thorium Elements	5

*These totals include 120 tons of cores and 130 tons of finished fuel made for the smaller reactors.

OPERATING EXPERIENCE

Overall operating efficiency of the AlSi-bonding lines was 98.5 percent. Downtime was assigned 43 percent to equipment malfunctions and 57 percent to operations. AlSi canning operations continued on the basis of two lines on the day shift, five days per week.

EQUIPMENT EXPERIENCE

Several modifications were made to the HDS fuel welder to improve throughput and quality for the final installation. An improved, permanently mounted

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mandrel was installed on the spindle, resulting in a 50 to 100 percent increase in throughput, dependent upon operator capability. The spindle speed was increased to 30 rpm, allowing a reduction in weld cycle time with a consequent increase in throughput. A change was made to control circuit to permit striking the arc off the weld area. This will improve initial melt-together of the can base and spire, and eliminate the possibility of weld voids at the arc strike point.

PROCESS ASSISTANCE AND CONTROL

Neptunium Target Elements

In support of the medical grade Pu-238 program and the current neptunium irradiations authorized under a Production Test, work was conducted during this period on both Np-Al alloy target fabrication and Np-graphite target development. Twelve Np-Al alloy targets were completed through helium leak testing, hydrostatic sizing, and radiographing. These targets are awaiting shipment to KW Reactor.

Certification of the Np-graphite fabrication techniques were completed. Three types of wafers for this program are in the final stages of assembly and are scheduled to be shipped to the reactor early in June.

FUEL AND TARGET FABRICATION - N REACTORPRODUCTIONInput Production

Total billets extruded	245
Tons extruded	34.9
Percent of forecast	134.2

Output Production

Total finished fuel assemblies	2,016
Tons output	37.0
Percent of forecast	94.9

<u>Uranium Utilization - %</u>	73.6
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Month-End Inventories - Tons

Bare uranium billets	231
Finished fuel	287

OPERATING EXPERIENCE

Personnel from the Fuels Section assisted N Reactor personnel by inspecting discharged fuel thought to have been damaged during charging. This accounts for the below-forecasted output production.

EQUIPMENT EXPERIENCE

The extrusion press was shut down and a scheduled four-day overhaul performed. The major items recommended during the last visit of the Loewy representative were completed.

PROCESS ASSISTANCE AND CONTROLCopper Coating of Zircaloy

Each of the three methods presently used to encase Zircaloy extrusion billets in copper or copper-silicon has certain disadvantages:

- a) Flame-sprayed copper coatings - These coatings are not metallurgically bonded to the Zircaloy. Higher upset pressures are also encountered, probably because the copper is highly oxidized during the flame spraying operation.

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- b) Electroplated copper coatings - Although upset pressures are not appreciably increased by this type of coating, the electro-deposited copper is porous and does not adhere well to Zircaloy. Billet preheating often results in spalling and blistering.
- c) Canning in copper-silicon - The main disadvantages of this method are high cost, and loose fit between the can and billet.

Although these processes produce good quality extrusions, the disadvantages noted have created a desire to develop other means of applying copper to Zircaloy. Attempts to hot-dip Zircaloy into molten copper-silicon have failed. During one of these tests, however, it was observed that copper plated out of molten cuprous-chloride (which was being used as a flux) onto the Zircaloy.

Subsequent laboratory tests have resulted in a process which deposits 0.0005 inch of uniform, tightly bonded copper on Zircaloy surfaces. The process consists of immersing the Zircaloy in molten cuprous-chloride at 400-500 C, and the following reaction takes place: $4 \text{CuCl} + \text{Zr} = 4 \text{Cu} + \text{ZrCl}_4 \text{ (vapor)}$. Billet preheating temperatures do not adversely affect the coating. Further investigation of this process is being conducted.

TECHNICAL ACTIVITIES - C & K REACTORSRESEARCH AND DEVELOPMENTMission 1 - Basic Production1-A. Brittle Fracture ProgramTechnical Bases for K Reactor Tube Power Limits

Analysis of all of the significant data from the laboratory experiments conducted in the fall of 1968 to simulate K reactor inlet riser failures has been completed. The principal information determined from the experiments that is of value to the analysis is the relationship between the pressure differential and film heat transfer coefficient for fuel surfaces in a process channel which has undergone a flow instability transient and is operating in two-phase flow. The relationship is shown graphically in Figure D-1, (next page). In the analysis of tube power limits, this correlation is utilized for calculation of the maximum temperature that would be reached on the surface of the fuel in a process channel during the transient that would follow a supply piping failure.

Even without consideration of the fact that the experiments were not conducted for the purpose of obtaining film coefficients, the scatter of data is remarkably small. This consistency, together with the points from other sources that are shown on the plot, substantiates the applicability of the lower envelope line for use in the accident analysis. A thorough search of the literature has failed to locate general correlations that would be applicable to the low pressures and high temperatures that would take place during the conditions associated with the accident cases under consideration.

High-Lift Pump Tests

DUN-5042, "Production Test Authorization 161, Operation Test of 190-K High-Lift Pumps," has been completed using the No. 1 pump at KE. The cavitation cut-off flow for full speed operation was determined to be 42,000 gpm. The cut-off flow at 1585 rpm was measured at 41,000 gpm. At 1485 rpm the flow was 39,500 gpm. During the performance of this test, approximately 37 minutes of operation at cut-off flow rates was accumulated on the pump.

During these tests, vibration readings were taken on the pump outboard bearing, inboard bearing, and casing. The maximum vibration amplitude measured was 0.003 inch; the maximum vibration velocity was 1.8 inches per second. Conversations with representatives of Byron Jackson Pumps, Inc. have led to the conclusion that these values are not excessively high for this large equipment. A report on these tests is being prepared and will be issued in the near future.

DUN-5673, "Production Test Authorization 173, Gross Cavitation of 190-K High-Lift Pumps," has been cancelled. Past experience and testing have

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Figure D-1. Heat Transfer Coefficient vs. Pressure Differential

resulted in about four hours of operation of these high-lift pumps in cavitation cut-off flow conditions. Further, in April 1960, an incident occurred which resulted in the failure of one-half of the reactor pumping plant, the same end result with respect to the surviving pumps as would be expected with a dual piping failure. The above incident was successfully handled, and three high-lift pumps provided enough coolant to prevent fuel damage during and after the shutdown transients. No evidence of pump damage was evident in subsequent inspections. Since this same sequence of events was to be simulated by the gross cavitation test, the test was concluded to be unnecessary. A report detailing past operating tests and incidents is in preparation.

Fracture Mechanics Testing Program

Ten fracture toughness tests were completed at test temperatures of 33 F and 70 F on WOL (wedge opening loading) specimens precracked at rates no greater than 1 microinch per cycle as recommended in ASTM-STP-410. In previous tests at the Lawrence Radiation Laboratory, the specimens were precracked at rates of 6 to 7 microinches per cycle. Some preliminary results of these tests are as follows:

- The temperature dependence of fracture toughness of the parent metal of the riser steel was consistent with that shown by the Charpy V-notch data. In the region below 33 F, K_{IC} shows a small temperature dependence; but, within a rather narrow temperature range above 33 F, the values increase rapidly. The 15 foot-pound transition temperature for the steel is in the range of 33 to 50 F.
- At -76 F, the WOL specimens fractured with no apparent yielding indicating that plane strain conditions were satisfied for the sample thickness. The -76 F data are considered representative K_{IC} values. The lowest value, 47 ksi-in^{1/2}, was obtained on a D riser specimen. The fracture faces of the 33 F and room temperature parent metal specimens were observed to be brittle in appearance. Thus, although there was yielding at the edges of the specimens, there was little plasticity in the fracture plane.
- The D riser K_{IC} values form a lower-bound curve for the data. The data from the three risers sampled varied 17 percent at -76 F, 13 percent at 33 F, and less than 5 percent at room temperature. The range of K_{IC} values measured at 33 F, the lowest operating temperature, was 52 to 65 ksi-in^{1/2}. Critical through-the-thickness crack lengths based on the lower K_{IC} value are approximately 12 inches for the nominal piping stress of 6000 psi, and approximately 1.5 inches for stresses just under the yield stress for the material. Thus, under expected conditions, leak-before-break behavior would be anticipated by growth of small flaws as indicated by a critical through-the-thickness crack length which is large compared to the wall thickness.

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- The fracture toughness values do not appear to be influenced by precracking rates in the range of less than 1 to 7 microinches per cycle.
- Longitudinal weld specimens yielded K_{IC} values about 30 percent higher than base metal specimens at a given test temperature. Tests on circumferential field weld specimens also yielded values higher than base metal specimens, but showed poor fusion in some locations as evidenced by the weld pulling apart along the fusion line. Specimens machined from heat-affected zones yielded K_{IC} values comparable with the highest values obtained from parent metal specimens. The results from heat-affected zones are tentative because there was insufficient material to assure testing the heat-affected zones directly.

Experiments to determine fatigue crack growth rates under cyclic loadings, and the effect of sustained load on crack growth behavior, were initiated. Both experiments are being performed in reactor process water.

Missile Velocity Calculations

A series of conservative calculations have been made to estimate the velocity of missiles released from a riser upon brittle fracture. Very small missiles are assumed to be propelled by a jet-like stream of water which would reach a velocity of 260 ft/sec. For large missiles, the energy imparted to the missile is the total energy of compression of the water in the riser plus the energy in the riser walls due to internal pressurization. This would be approximately 58,000 ft-lb and could cause a missile weighing 5000 pounds to have a velocity of 27.5 ft/sec. Figure D-2 (next page) shows the range of missile velocities as a function of missile weight for K riser fragments.

1-B. Zircaloy Process Tube Hydriding

Stress Analysis

Investigation of the stresses induced in Zircaloy process tubes when a case layer is formed on the inner surface continued. Using the electrochemical removal technique, the stress relaxation as the case layer was removed from several tube samples was determined using strain gages. These data will be compared with dimensional changes obtained before and after the case-layer removal. Preliminary observations are that case-layer induced stresses are higher than hydrostatic or thermal stresses from normal operation. It has also been noted that the case-layer induced strains are about twice as great in the longitudinal direction as in the circumferential.

Electrochemical Removal of Hydride

Investigation of the proposed electrochemical method of removing zirconium hydride from process tubes is being developed as a major program. Equipment

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Figure D-2. Velocity of Fragments Released From Fracture of a K Reactor Riser

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for building the test mockup in the 108-D Laboratory is being assembled and the mockup layout has been completed. A PERT chart indicating the timing and schedule for the initial phase of the program--laboratory testing and initial on-reactor trial--has been prepared.

Tube Examinations

Tubes 1354- and 1355-KW, which were grit blasted on September 10, 1966, were removed for analysis on May 20. These will provide additional data on the ability of anodized spacers to prevent hydriding in cleaned tubes.

1-C. Corrosion Studies

Corrosion Testing of Oralloy Fuels

Five columns of oralloy fuel charged under PTA-138 were discharged from C Reactor after 449 to 504 MWD/T exposure. These fuels will be weighed and the data utilized in updating parameters in the fuel corrosion equation.

Aluminum Corrosion Mechanisms - Erosion-Corrosion Investigation

Laboratory tests to investigate the mechanisms of erosion-corrosion on Hanford aluminum alloy fuel cladding are progressing. Preliminary tests conducted for 48 hours in pH 6.6 water show that (1) severe erosion-corrosion occurs with zero and 0.2 ppm dichromate, (2) erosion-corrosion occurs only at the outlet end of the test sample with 0.5 ppm dichromate, and (3) essentially no erosion-corrosion occurs at 1.0 ppm.

Half-Plant Low Dichromate Evaluation

Although a half-plant test at B Reactor in 1967-1968 showed 0.5 ppm dichromate addition to the process water was as effective as 1.0 ppm in preventing non-uniform corrosion of the fuel cladding, experience has been mixed at the K reactors. There have been no fuel failures attributed to non-uniform corrosion in KE Reactor where the dichromate concentration is 0.5 ppm, but examination of fuel elements has provided conflicting results.

A Production Test Authorization (PTA-176) is being prepared to permit a half-plant evaluation of fuel cladding corrosion in water treated with 0.5 and 1.0 ppm dichromate at the KW Reactor. Groups of fuel elements exposed to these concentrations of dichromate inhibitor will be irradiated in process tubes operating at nominally 95, 105, and 115 C outlet temperatures. Fuel examinations will be made after 800, 1200 and 1600 MWD/T exposure.

Mission 2 - Coproduct

2-A. Cobalt Tubes

One of the available cobalt-bearing (Haynes alloy 25) tubes was installed in channel 1355-KW on May 20. It was inserted into a new Zircaloy tube which

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was then installed in this channel as the replacement for one of the sand-blasted tubes removed to study hydriding phenomena. The tube is charged with aluminum dummy elements for the duration of the next operating period after which it will be fueled.

Mission 3 - Transplutonium Technology

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

The loading changes made in and adjacent to the PuAl test block during the April KE outage accomplished their purpose. The power in the PuAl columns themselves increased an average of more than 50 percent, and the tube powers in the driver ring at the edge of the block were restored to normal levels. In view of the currently favorable operating levels within the block, and the exposure of the thorium columns in the adjacent buffer layer, present plans call for no further loading adjustment within the block until the KE outage in late June.

Mission 4 - Pu-238

4-A. Americium Irradiation

PTA-171, authorizing the irradiation of 12 target samples of americium in a single column, was approved and the material was charged on May 27.

A version of the MOFDA lattice analysis code is being prepared to compute production of clean Pu-238 from Am-241 irradiation. Due to the long irradiation time of the americium route, flux and spectrum changes should be recalculated as isotopes in the chain build up and burn out during the irradiation cycle. A buildup and burnout routine will be added to MOFDA to permit isotope inventory and flux changes to be handled by the code in one run, thus making the version of MOFDA rapid and consistent for survey purposes. Additional work has been done on updating the Np-237 and Pu-238 cross sections in the MOFDA code.

4-B. Neptunium Irradiation

The authorization document for irradiation of neptunium in a four-column cluster and in a fringe blanket column has been routed for approval signatures. The charging of this test is planned for the June KW outage. The use of a graphite diluent in the elements of two of the cluster columns and in the fringe column is expected to demonstrate the production of Pu-238 with even less Pu-236 than the 0.3 ppm demonstrated previously at Hanford.

Mission 10 - Columbia River

10-A. High Turbidity Coolant

The half-plant test at KE Reactor with high turbidity (0.3 JTU) has not been conducted during this report period due to high basin turbidity. The test will be resumed as soon as conditions permit.

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10-B. Columbia River Water Analyses

Results from the neutron activation analyses of Columbia River water samples collected on February 12 and March 11 above the Hanford reactors are:

	<u>February 12</u>	<u>March 11</u>
	<u>pp/b</u>	<u>pp/b</u>
Na	1365	2290
K	0.4	1060
Rb	-	1.0
Cs	0.03	0.1
Cr	<0.05	2.2
Mn	5.5	15.5
Fe	<10	-
Co	0.04	0.8
Cu	<0.03	<0.07
Ag	<0.01	<0.01
Au	<0.005	-
Zn	44.2	190
Hg	<0.1	-
As	1.4	2.5
Sb	0.4	0.7
Se	0.2	<0.1
Br	5.5	5.1
U	0.5	0.8
Sc	0.001	0.003
Ca	510	-
La	0.09	10

The results are considered normal for those times of the year.

The neutron activation of the above samples was conducted in facilities at C Reactor. Due to the deactivation of C, additional analyses will be delayed until the facilities can be moved to the K reactors.

10-C. Effluent Manganese Activity

Routine effluent samples taken at C Reactor on April 22 exhibited higher than normal gross activity levels. Analyses indicated 63,000 to 78,000 pCi/ml Mn-56 was present, and these levels persisted until C Reactor was shut down. Analyses at the K reactors indicated similar levels during the remainder of April at KW, but lower levels (15,000 to 24,000 pCi/ml) at KE. Up to May 8, the Mn-56 levels at KW had gradually fallen to 46,000 pCi/ml while those at KE rose to 38,000 pCi/ml. Comparison of these data with data taken in 1967-1968 indicates the present Mn-56 levels are much higher. The previous data show that a peak Mn-56 concentration occurs in the spring of the year; a maximum value of 16,881 pCi/ml was obtained in May 1967. Values for the remainder of 1967 ranged from 2300 to 6000 pCi/ml. Comparison of parent manganese concentrations in the Columbia River water in 1962, 1968 and 1969 indicates no significant change. Samples of clearwell filter plant water are being analyzed to compare clearwell parent manganese concentration with raw water parent manganese.

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ENGINEERING AND TECHNOLOGY - REACTORS

K Reactor Total Coolant System Reliability Analysis

A reliability analysis of the total K Reactor coolant system was completed and issued as document DUN-4461, Supplement 1. The total process coolant system for a K Reactor includes the BPA-powered Primary Process Coolant System, the steam-powered Secondary Coolant System, and the diesel-powered Tertiary ("Last Ditch") Coolant System which is common to both KE and KW Reactors.

This reliability analysis includes the effects of active and passive component failure, failure detection, and component repair times as related to the probability of failure of each coolant system and to the probability of failure of the total coolant system. A general case representing the existing total coolant system was analyzed, and three additional major cases were considered: (1) increased frequency of testing the tertiary system diesels, (2) increasing the reliability of certain limiting components, (3) A-B coolant supply riser isolation as being accomplished by Project DCP-522, and (4) the combined effects of (1), (2), and (3).

The quantitative probability of failure values calculated for each case are intended primarily for case comparisons. The use of these failure probabilities as precise indications of the K Coolant System performance is limited by some variability in the quality of available plant equipment data. These probabilities can be used as accurate bases for evaluating proposed modifications to the K Coolant System.

Side-to-Side Test Hole Enlargement

With the issuance of document DUN-5643, "Enlargement of Side-to-Side General Purpose Test Hole", engineering work on this program has been completed until such time as the on-reactor effort is scheduled. The document describes the equipment and outlines the procedure for its use.

Project Engineering - Reactor Facilities

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

ENGINEERING AND TECHNOLOGY - FUELS & TARGETS

Reclamation of Spireless Components

Development and testing for reclamation of defective K5AE-A spireless impact cladding components was concluded. Special processes were developed including precleaning in $\text{HNO}_3\text{-NiSO}_4\cdot 6\text{H}_2\text{O}$ solution which reduced the fuel reject rate from about 20% to 4%. Information from this test established the extra handling procedures and determined the reject rate to be incurred by the use of these 28,900 components. Investigation with the vendor on the cause of the defect led to a tooling correction which has eliminated the problem. This defect was

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limited to the punching operation in the base of the spireless impact which resulted in aluminum galling and tearing.

Hot-Die-Size (HDS) Fuel Production

A project proposal for Replacement Fuel Production (HDS) was issued on May 7, and approved by the AEC-RL Project Review Board on May 15. Total capital project estimate was \$395,000. In addition, \$204,000 in expense funds will be required to do connected work.

The proposed HDS facility, when operated three shifts, five days per week, will fabricate sufficient fuel elements up to 12 inches long to supply two K reactors producing weapons grade plutonium. A right-of-way for the new process equipment has been cleared, and the radiograph machine and related equipment relocated and isolated from the construction zone. The tool cleaning AlSi recovery line has been relocated and is 85% installed in the 313 Building south of the pickling machines.

The special component cleaning line is being relocated to the 304 Building and is about 20% installed. Excavation of the foundation for the nickel plating machine has started.

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TECHNICAL ACTIVITIES - N REACTOR

RESEARCH AND DEVELOPMENT

Mission 1 - Basic Production

1-A. Mark IV Fuel Development

Irradiation Testing

The first large increment of central zone loading of Mark IV fuel was charged in April. A total of 132 columns were charged and 17 columns discharged, bringing the in-reactor total to 184. The discharged columns were part of the initial Mark IV production model test columns charged in July 1968. Two of the 17 columns were monitor columns which will be evaluated for dimensional changes and volume increases. Eight other monitor columns will be used for obtaining fuel performance data at exposures up to 7000 MWD/T for comparing with Mark I performance at high exposures.

Evaluation of the Mark IV demonstration test fuels is continuing. The test was discontinued at approximately 3000 MWD/T exposure; however sufficient data have been generated to provide a performance model for Mark IV fuel with respect to dimensional changes and volume increases for nondefense plutonium production.

Flow Loop Testing

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

<u>Fuel Type</u>	<u>Support System</u>	<u>No. of Assemblies</u>	<u>Loop Time (Hours)</u>
26" Mark IV	End-spiders	8	4594
26" Mark IV	End-spiders with one or two legs removed	3	2280
12" Mark IV	Spring-stop	6	1336
		4	738

The last inspection showed that an assembly with two spider legs cut on each end (adjacent cuts) had failed on the upstream spider. The inner tube displaced approximately 70 mils without failing the downstream spider. Further displacement was restrained by the downstream assembly. Total loop hours of exposure were between 2071 and 2370 hours. One end-spider with a loop exposure of 4318 hours or 8.3×10^8 cycles was sent to the laboratory for photomicrographs and die penetrant testing of the high stress areas. No indications of fatigue cracking were observed. The 12-inch Mark IV fuel continues to show no signs of detrimental effects due to placing the outer supports close to the ends.

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1-B. Mark IV Fuel Failure

A Mark IV fuel failure caused the reactor shutdown on April 26. Inspection of the assembly revealed two small blisters on the inner bore of the outer element at or near an upstream solid support contact point. The tube, 2146, was operating at 4200 kw and had accumulated an average exposure of 598 MWD/T at the time of the rupture, which was in position 9 from the downstream end. Based on circumstantial evidence and the history of previous N fuel failures, this failure is thought to be caused by fretting of debris lodged beneath a support. The assembly will be sent to BNW Radiometallurgy facility for confirmation of the failure mechanism.

The incipient Mark IV failure from tube 2662, discharged in February 1969, has been examined in the Radiometallurgical laboratory. It appears that the clad cracking on the inner bore of outer element was the result of discharge damage. Subsequent examinations of fuel in the observation basin has shown that discharge damage to fuels is occurring at a higher frequency than in the past. This may result from the new fast-cart system which adds a step to the discharging procedure in which the fuel assemblies are dumped down a 12-foot chute. The increase in discharge damage appears to be independent of fuel geometry.

1-C. Mark I-C Studies

Six more columns of upset forged test material were discharged during the May outage, making a total of 15 columns discharged to date. Examination of the monitor columns is in progress and results should be available within a month. In addition, 16 columns remain in the reactor and are at approximately 50 percent of goal exposure. The 12 columns of direct cast material are being irradiated without difficulty. Discharge is scheduled for November 1969.

1-D. Mark IV-AA Fuel Development

Production Test PT-NR-94 SUP5, authorizing the irradiation of two prototypic Mark IV-AA (both inner and outer 1.25 percent enriched) spike columns, was charged into fringe locations during the May outage. Column instrumentation will provide the thermal hydraulic operating data necessary to evaluate the Mark IV-AA fuel design. Normal goal exposures are planned for these columns.

Assemblies fabricated with flat cap, brazed closures were alternated in the two test columns with assemblies fabricated with chevron cap, brazed closures. Profilometer measurements of the end closures will be compared for both the pre-and post-irradiation conditions in order to examine the effect of cap geometry on the dimensional stability of the closure.

1-E. Corrosion Studies

Examination of the Inconel-600 tube removed from steam generator 4A in January 1969 has revealed no damage mechanism which would impair the life of the tube. The results are summarized as follows:

- Secondary side (outside surface) intergranular attack about 1/4 mil deep (maximum observed penetration of 1 mil) was uniformly distributed over the tube. The corrosion is similar to that observed on tubing removed in 1966 from steam generator 4A and that observed in samples of as-received tubing. No effects of tube sheet crevices, deposited crud, or tube supports on the attack were observed.
- Primary side (inside surface) intergranular attack about 1/4 mil deep occurred at the high temperature inlet end of the tube and decreased uniformly toward the U bend, where it had essentially vanished. Only slight intergranular attack had been observed on the inside surface of the tube removed in 1966 and even less on the preoperational samples, indicating a slow, progressive intergranular attack in high temperature sections.
- Sectioning of the tube showed that the source of the previously reported ultrasonic indication 12 mils below the surface was a 1-mil diameter spherical void, presumably a manufacturing defect.

Mission 8 - Nuclear Safety

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

An N outer fuel element component was heated inductively to the point of clad failure (but below the uranium melting point) in a flowing steam atmosphere. A continuous measurement of the hydrogen evolved as a result of Zr-H₂O and U-H₂O reactions was obtained. The fuel was heated to rupture at 1025 C in about 2.75 minutes, then held at 1045 C for 33 minutes in a flowing steam atmosphere. Rupture first occurred by separation of the inner cladding from the downstream end cap. Post-heating examination showed that most of the extruded uranium was contained in the Zircaloy tube used to provide longitudinal restraint to the end caps and in the center hole of the fuel. A combination of cladding distortion and uranium oxidation had plugged the hole. A section of cladding about 4 inches long and 3/4 inch wide was missing from the side of the fuel (the uranium in this area was completely oxidized). Three other relatively small breaks in the cladding were observed and the diameter of the fuel in the region of greatest distortion had increased to 2.75 inches (process tube ID is 2.700 inches minimum).

During the test, the flow and percent hydrogen recorders indicated that uranium metal was continuously being exposed to the oxidizing atmosphere for at least the first 20 minutes of the test. This was confirmed by a plot of the total hydrogen released versus time, which was linear for the first 18 minutes and then approached a parabolic rate function. Since the U-H₂O rate function was previously shown to be parabolic, the linear portion of the curve indicates a continuous exposure of new uranium metal during that period of the test. A total of 51.9 ± 5 percent liters of hydrogen was produced during the 33 minute period after rupture for this particular fuel.

The distortion of the Zircaloy-2 cladding precludes an accurate calculation of the hydrogen produced by the Zircaloy-2-H₂O reaction. A rough approximation shows about 25 to 30 percent of the total hydrogen measured resulted from this reaction.

Since the volume of hydrogen is principally a function of the amount of uranium exposed to the furnace atmosphere, caution should be exercised in using the following approximate equations developed from this single test--the next fuel may or may not approximate this one relative to the amount of uranium oxidized:

$$0 - 18 \text{ min. } V_{H_2} = 1.91 t \quad (1)$$

$$18 - 33 \text{ min. } V_{H_2} = 10.2 (t-6.9)^{1/2} \quad (2)$$

where

V_{H_2} = hydrogen volume in liters, and

t = time in minutes

The second equation does not apply beyond 33 minutes since it is not a true parabolic rate function between 18 - 33 minutes.

The results from this test may be compared with previous analytical studies in RL-GEN-1541, "Metal-Water Reactions during a Loss-of-Coolant Accident in N Reactor," by T. W. Evans, March 23, 1967:

RL-GEN-1541

Molten pool of uranium held in process tube. Steam reacting with surface of pool according to Wilson-Barnes parabolic rate law. Steam reacting with exposed Zircaloy fuel clad and process tube (above pool) according to Baker-Just parabolic rate law.

Predict 1.25 ft³ hydrogen per 7.44" of fuel in 33 minutes at 1045 C.

[This model used as basis for evaluating potential effects of metal-water reaction in hypothetical accident (see RL-GEN-1375).]

Experiment SNH #1

Uranium extruding from cladding, with expansion of fission gases as a driving force. No inner fuel piece as either a source of more uranium or a physical barrier to limit degree of uranium extrusion from cladding. No process tube for either restriction of swelling or a source of Zircaloy - steam reaction.

Measured 1.83 ft³ H₂ per 7.44" outer element in 33 minutes at 1045 C.

Further tests in the 292-T Building will utilize physical restraint of the elements to simulate the effect of the process tube and inner element on the amount of uranium extrusion that occurs.

Figure DN-1 (next page) shows the hydrogen generated as a function of time for the test SNH #1, and compares it with the analytical model used to assess the potential metal-water reaction effects for a hypothetical loss-of-coolant accident. Also shown is the minimum potential metal-water reaction if the element had not failed and exposed the large uranium surface area. It is apparent that the extrusion (or foaming) of the uranium from breaches in the cladding result in a greater surface area for metal-water reaction than was assumed in the model. These tests at clad failure temperature may well be the worst-case condition, since higher temperatures would cause melting and a probable reduction in surface area as the uranium pooled in the process tubes. The fact that this is the first test with a new experimental technique precludes application of the data to accident analyses at this time. Future tests in the new 324-D facility will be performed in the molten uranium temperature range and will investigate the full spectrum of fuel conditions (exposure, geometry effects, etc.).

The "D" cell of 324 Building was declared operational on May 12, 1969. Initiation of the first test with irradiated fuel is now awaiting delivery of a fuel element from the 100-N storage basin. The fuel shipment has been delayed due to interferences with a scheduled reactor charge-discharge outage at the reactor.

ENGINEERING & TECHNOLOGY - N REACTOR

Flow Throttling and Orificing Device for V-11 Valve Elimination

A test is being conducted to determine the feasibility of performing the V-11 valve function (throttling process tube flow during charge-discharge) with a movable sleeve in the inlet nozzle at N Reactor. The sleeve is designed to pass full flow during operating conditions. During reactor shutdown, the nozzle cap may be loosened and the sleeve rotated to a throttling position in which a series of holes are provided to reduce tube flow. These holes have been experimentally sized to deliver 14 to 20 gpm depending on the tube position in the unit; the larger flow corresponds to the greater head on the lower tube rows.

An additional series of holes is being drilled in the prototype sleeve to ensure that at any position of the sleeve in the nozzle the flow can never fall below process standards limits.

Test results to date look very promising. A patent application is being submitted.

Ball Trip Mechanism

Design and detailed drawings of two new ball trip mechanism concepts were completed and prototype models are currently being fabricated in the 189-D

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Figure DN-1. Metal-Water Reaction Data

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Development Laboratory. Both designs incorporate a magnetic clutch as the latching mechanism. However, they differ in that in one design the clutch is coupled directly to the gate shaft whereas the second design incorporates a mechanical lever arm arrangement.

ENGINEERING & TECHNOLOGY - FUELS & TARGETS

Uranium Sleeve Manufacture

The remainder of the uranium sleeves necessary to convert the excess Mark I outer billets into Mark IV outer billets were extruded this month. This represents the second production campaign of these components. The sleeves were made by machining Mark I outer uranium billets to an OD of 5.850 inches, sleeving the billets in .065-inch copper silicon followed by extrusion into the sleeve components. A total of 25 extrusions was made, yielding 17 components per extrusion or a total of 425 pieces. Process steps included cut-up, end squaring, identification, and copper removal. No OD or ID machining was required.

Segmented Extrusion Stems

A new, two-piece segmented stem for both the 6-inch and 5-inch tool size is now available for use on special or regular extrusions that require high strength material. The stem has been designed in two pieces so that only the relatively small cross section portion is made from the high strength, high cost alloy, while the heavy stem butt is made from the conventional H-12 alloy steel. The stems are manufactured from M-300 type maraging steel forgings with a compression strength approximately 35% greater than the steel normally used in the manufacture of extrusion stems. The new 6-inch stem also incorporates a larger ID than the standard 6-inch size allowing for a larger mandrel clearance for special large ID thin wall tubes from a 6-inch liner employing the traveling mandrel technique. The 6-inch diameter stem with 2-1/2 inch ID hole has been tested at a compressive force of 2475 tons and has been certified for use at 2150 tons continuous and 2400 tons intermittent use on special engineering programs.

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IRRADIATION SERVICES

FUEL TECHNOLOGY

Fast Reactor Oxide and Nitride Fuels - Battelle-Northwest

The second in a series of (U-Pu) O_2 fuel capsules was irradiated in the Snout facility at KW. The fuel in this capsule did not reach the desired core-melting temperatures although cooling water temperatures indicated sufficient heat generation to achieve core-melting.

MATERIALS DEVELOPMENT

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

The in-reactor test section of the CPTF, which had a successful 200-hour run before C was deactivated on April 25, was discharged and shipped to the BNW Radiometallurgy Laboratory for examinations. The CPTF equipment has been removed from the 105-C Building and is being installed in the 105-KE Building.

Nuclear Graphite Program - Battelle-Northwest

One BNW "cold seeding" effect graphite capsule was irradiated in the KW Reactor Snout facility.

Eight boats containing graphite samples were discharged from the B test hole at C Reactor. These boats will be shipped to BNW for sample measurement and then recharged into the 4B test channel at KE.

Irradiation Damage to Reactor Metals - Battelle-Northwest

One boat containing nickel-base Charpy impact specimens was discharged from the E test hole channel at C Reactor for BNW.

ISOTOPE PRODUCTION

C-14 Production - Oak Ridge National Laboratory

Following the shutdown of C Reactor, three process columns and one test hole loaded with Be $_3$ N $_2$ pieces were discharged. These 240 pieces will be recharged into KE Reactor for further irradiation.

ROUTINE IRRADIATIONS

One hundred and eight Quickie activation analysis capsules were irradiated for BNW.

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ADMINISTRATION - GENERALFY 1971 BUDGET AND REVISION OF FY 1970 BUDGET

The revised FY 1970 budget and the budget for FY 1971 have been completed and transmitted officially to AEC-RL. End program costs are budgeted at \$41,550,000 and \$46,315,000 for FY 1970 and FY 1971, respectively.

DISPOSAL OF RESIDUAL INSTALLED EQUIPMENT - F AND H AREAS

A sale of equipment and associated piping and power wiring in 190-F Annex was recently completed and the purchaser is removing the equipment. On May 26, bids were opened in the sale of 184-F and 184-H steam generation buildings and equipment.

AEC-RL recently informed DUN of the following tentative bid opening dates for the sale of equipment in other F Area buildings:

July 28	Equipment in 181-F, 182-F, and 183-F raw water pumping and filter equipment.
September 24	Process water equipment in 190-F.
January 26, 1970	Equipment in 105-F (the reactor building).

AUDITING

Scheduled internal audits of Travel, Living and Moving Expenses, and Contracting and Procurement were completed and reports were issued. Procedures and controls were appropriately documented and, with only minor exceptions, activity reviewed was found to be in compliance. The Contracting and Procurement audit was coordinated with an AEC appraisal in order to avoid duplication of areas reviewed. Both the appraisal and the audit indicated completely satisfactory performance.

TRAVEL AND LIVING EXPENSES

A recent analysis of the per diem travel and living variation revealed that for the period July through December, 1968, the average per diem costs were \$17.02 and those for the period January through April, 1969, were \$18.41. AEC-RL has agreed to increase the per diem allowance from \$15.00 to \$17.00 effective July 1, 1969.

APPROVAL LETTERS

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

<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-112 Add. #2	Reclassification of the C Reactor Plant	May 23, 1969
ATD-180	Lay Off for Lack of Work Appendix "B" Modification MO-23	February 24, 1969
ATD-184	McDonnell Douglas Astronautics Company Development Program for DUN Management	May 9, 1969

SECURITY AUTOMATIC DATA INFORMATION EXCHANGE (SADIE)

In compliance with an AEC-RL request for information on long-range usage of the SADIE System, a potential use survey was conducted within DUN. Results indicated that local usage for classified data transmission within the nationwide network of AEC sites and its contractors will continue to expand as the capabilities of the system are fully explored.

MANPOWER LEVELS

The reduction in force as the result of the shutdown of C Reactor is proceeding according to schedule. Of the 74 employees removed from the rolls thus far, 60 percent have been by voluntary layoff. An additional 20 percent have been placed in jobs with other Hanford contractors.

PLANS FOR PROGRESS - EEO

By month-end, four minority employees had left the Company's payroll due to the C Reactor closure.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - May	0
- Year to date	0
Days since last disabling injury	212

Man-hours since last disabling injury 2,080,000

The 2-million man-hours mark was passed at midnight on May 21, and appropriate recognition programs are planned for all plant areas on June 6.

EMPLOYMENT SUMMARY

DUN personnel totals and employee allocations as of April 30 and May 31 are shown in Appendix B.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

<u>Number & Title</u>	<u>Authorized Funds - \$</u>	<u>Percent Complete</u>		<u>Status</u>
		<u>Design</u>	<u>Construction</u>	
<u>Single-Pass Reactors</u>				
DCE-505, Boiler Control Improvements - 165-KE & KW	410,000	100	75	Acceptance testing of No. 1 KW boiler was completed May 1, and beneficial use obtained. Construction on No. 2 KE boiler was completed and unfired phase of acceptance testing started the week of May 12. Work on the fifth boiler (No. 2 KW) was started the same week.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	200,000	100	96	Revised design is complete. The first phase, relocation of the deflector plate on the KE units, is complete. The second phase for the KE units is to lower the bucket guides and move them a little closer to the conveyors.
DAP-516, Storage Building Addition - 105-KE and KW	142,000	5	0	No progress since January 17 when design was halted on account of production cut-back planning.
DAE-518, Effluent Radioiodine Monitor - C, KE & KW Reactors	100,000	100	0	Work authority for \$27,800 received from AEC April 24 for equipment fabrication (by Reactor Plant Services) and engineering services. Offsite procurement of engineered equipment is underway--bid closing date is June 5.

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PROJECT STATUS SUMMARY - REACTOR FACILITIES (contd.)

Number & Title	Authorized Funds - \$	Percent Complete		Status
		Design	Construction	
DCP-522, Modification of Reactor Coolant Crosstie Piping - 105-KE & KW	163,000	100	51	All shop work and procedural preparations for the installations are complete. The KW installation started on May 19 and was completed on May 29.
<u>N Reactor</u>				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	96	Installation of reactor platforms in progress, as outage time permits. Revised proposal in progress.
GCE-408, W, C & D Elevator Safety Improvements - 105-N	90,000	100	33	Installation scheduled for the extended outage in July.
GCP-411, Effluent Control Program - 100-N Area	1,830,000	100	15	Work progressing on Sections I and III. Silo poured to elevation 478' on May 20. Detailed design for foundation of the dump tank and calculations for the tank have been evaluated and are being returned to the construction subcontractor for correction and resubmittal. Design criteria for Diesel Fire Protection have been approved.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin	269,000	40	0	Project proposal revision is in process to increase authorization to \$400,000 and extend completion date.

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APPENDIX BEMPLOYMENT SUMMARY
(with employee allocations)

	<u>4/30/69</u>	<u>5/30/69</u>
<u>CONTRACT PERSONNEL</u>		
<u>02 Programs</u>		
Douglas United Nuclear	1794	1735
Assisting other Contractors	<u>14</u>	<u>14</u>
Total - 02	1808	1749
<u>Other Programs under AEC Contract</u>		
Assisting other Contractors and WFPSS	43	43
Special Irradiations	<u>10</u>	<u>6</u>
Total - Other Programs	53	49
Total Contract Personnel	<u>1861</u>	<u>1798</u>
COMMERCIAL ACTIVITIES PERSONNEL		
	18	23
TOTAL FORCE	<u>1879</u>	<u>1821</u>

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FEATURE REPORT

UNCOOLED HORIZONTAL CONTROL ROD - K REACTORS

INTRODUCTION

Last month's Feature Report discussed the severance and shrouding of Ball 3X bellows to alleviate one of the problems at the K reactors which has resulted from continued distortion of the graphite moderator. The present report relates to another problem created by the graphite contraction; namely, the binding of horizontal control rods (HCRs) in the most grossly distorted channels at the top of the reactors (see Figure 1, appended).

There are twenty HCRs in each K reactor, arranged as shown in appended Figure 2. They ride in side-to-side channels having openings as sketched in appended Figure 3. The original HCRs are water cooled, boron carbide filled, aluminum rods about 2-5/8 inches in diameter with a 45-foot active tip. These round rods are quite stiff, and have very little vertical clearance in the original channels.

In 1963 a program was initiated to provide solutions to the HCR binding problem. A cooled rod having a flat cross-section was developed which is more flexible than the round rod, and the half-inch spacer plates devised for use with it provide ample channel clearance (see Figure 3). Four such rods are currently in service at the two K reactors. In addition, one round rod has been provided with the shallower track plates to improve its channel clearance.

While the cooled flexible HCR has afforded the necessary relief to date, and may suffice for the further graphite distortion foreseen, an uncooled segmented rod was seen as the long-term solution to this problem and work on its development has been conducted simultaneously. The purpose of this summary report is to describe this successful development effort, including (1) the in-reactor performance of the initial segment and the first prototype of the uncooled rod, and (2) the broader significance of the uncooled rod and the outlook for its future use.

BACKGROUND FOR UNCOOLED ROD DEVELOPMENT

Offsite work on the development of uncooled control rods for various reactors dates from 1957. This work included the following:

- Oak Ridge National Laboratory developed a special clad-rolling technique to produce europium oxide (Eu_2O_3) poison sections for the Army Package Power Reactor (APPR-I).
- In 1958, the Knolls Atomic Power Laboratory and the Battelle Memorial Institute, independently, built control rods made of Eu_2O_3 poison dispersed in stainless steel and clad with titanium. These rods,

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which were for shutdown purposes only, were made by a co-extrusion process which involved a complex billet extruded through a round die.

- Also in 1958, the Argonne National Laboratory worked on the problem of covering gadolinium oxide (Gd_2O_3) ceramic shapes with a stainless steel jacket to form a cruciform control rod for the Experimental Boiling Water Reactor.

Upon irradiation, the rare earth poison in each of these rods reacted chemically with the silicon in the stainless steel to cause swelling, cracking, embrittlement, and ultimate failure of the control rod.

In 1961, the General Electric Company developed an articulated erbium oxide - nickel control rod clad with a low-silicon 80 percent nickel - 20 percent chromium alloy for operation at 1650-1800 F in an air-cooled reactor.

UNCOOLED ROD DEVELOPMENT BY GE-NMPO

In 1963 the General Electric Company's Nuclear Materials & Propulsion Operation (GE-NMPO) at Cincinnati, Ohio, was asked by General Electric, Hanford Atomic Products Operation, to assist in the development of an articulated uncooled HCR for high temperature operation. The resultant three-phase development program (established under Contract ATH-IPD-4-65) is described below. The first two phases of the program were conducted concurrently and overlapped the third phase.

Phase I - Cladding Material Evaluation

The criteria employed in evaluating and selecting the rod cladding material were as follows:

- Operating Conditions
 1. Temperature - 1800 F (later raised to 1900 F)
 2. Contact with reactor graphite
 3. He-CO₂-N₂ atmosphere (-15 to -36 F dew point)
- Design life - 10 years
- Cobalt content of the cladding must be <0.05%

Initial stability tests of various materials in the above operating conditions indicated that nickel-base alloys would be the most suitable. Individual sections of twelve prospective cladding alloys were tested for long-term stability at operating conditions in contact with reactor grade graphite on one face and with a sintered Ni-Dy₂O₃ pellet on the other face. A weight was placed on top to force mechanical contact. Samples were removed periodically and those alloys indicating poor metallographic results were eliminated from further testing. Examination of the specimens after 5,000 hours indicated

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that Uniloy 19-9 DL and Inconel 600 were the preferred cladding materials, and these specimens were continued on test for an additional 3,000 hours. Microstructure examination revealed an oxidation-type scaling phenomena with Uniloy 19-9 DL and a carbide precipitation in Inconel 600. However, Inconel 600 was the most thermally stable from a metallographic standpoint.

Elevated temperature tensile tests on Uniloy 19-9 DL and Inconel 600 were conducted in air. These tests indicated that Inconel 600 samples had a higher average yield strength and less spread in yield strength.

Phase II - Composite Core-Cladding Evaluation

At an early stage in the metallurgical studies, dysprosium oxide was chosen as the nuclear poison. The choice of Ni-Dy₂O₃ cermet was influenced by the chemical, thermal, and dimensional stability under reactor operating conditions. During this early period of the development program, a final choice of cladding material had not been made; therefore, testing continued on both the Inconel 600 and the Uniloy 19-9 DL candidates.

Early physics calculations indicated a need for 40% Dy₂O₃ core, although later in the program this requirement was reduced to 30%. Initial attempts to blend the 60% Ni-40% Dy₂O₃ mixture were unsuccessful because of agglomeration of both the nickel and Dy₂O₃ to form large coarse grains. The homogeneity of the blend was improved by wet milling in methanol, followed by dry blending, isostatic compaction, and vacuum sintering (at 2500 F). Subsequently, propanol was substituted for methanol in the last milling step.

Test samples of Ni-Dy₂O₃ cores were sealed inside cans of the two cladding candidates. The core was bonded to the cladding by a hot gas isostatic pressure technique. Approximately 35 rectangular component test samples (3/4" x 1/2" x 3") were placed in a furnace at 1800-1900 F. Metallographic examinations of the test specimens were conducted after various test periods. After 5,000 hours, the Uniloy 19-9 DL cladding exhibited exterior oxidation-type scaling, and a uniform carbide precipitation was evident towards the core. After the same period the Inconel 600 specimen showed only minor surface pits. Principally because of the scaling problem, Uniloy 19-9 DL was eliminated from further consideration.

Concurrently with the above tests at Cincinnati, Uniloy 19-9 DL and Inconel 600 clad component test specimens were placed in separate graphite boats and positioned in a test hole at KW Reactor to determine irradiation and temperature effects. After 6 months the Uniloy 19-9 DL specimen was removed, and after one year the Inconel 600 specimen was removed. Neither specimen showed visible change.

Phase III - Rod Design and Evaluation

This phase of the development program consisted of (1) rod joint design and testing, (2) design and fabrication of rod segments, and (3) testing of a simulated rod in a channel mockup.

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The rod joint design had to fulfill several criteria: (1) each joint must provide about 10-1/2° of vertical angulation, (2) it must be capable of taking a 300-lb. load at 1900 F, and (3) it must prevent accidental disconnection while in the reactor. Joint connections considered included a solid strap between two sections, hooks, ball and sockets, and threaded bolts; however, all of these were eliminated due to one design deficiency or another.

The joint design chosen was the hinge arrangement shown in Figure 4, appended. It consists of a slug hinge and pivot pin. To assemble the hinge, the two halves are brought together at right angles to each other and the pivot pin is put in place. When the hinge is brought back into alignment, as for in-reactor use, the flanged tip of the pivot pin is held in place by the locking pin in the last lug of the hinge. The flat edge of the pivot pin head rests against the locking shoulder of the hinge body, preventing rotation of the pivot pin.

Several tests were performed to determine a suitable hinge pin material and lubricant. After more than 10,000 cycles, the tests indicated that both Inconel 600 and Uniloy 19-9 DL are satisfactory materials with a colloidal graphite lubricant applied to the hinge, and chrome carbide (with a chrome-nickel binder) flame-sprayed onto the pin. Inconel 600 was chosen on the basis of Phase I and II results.

The design studies recommended a 40" segment length for the uncooled HCR in order to achieve the required flexibility. Co-extrusion of the cladding and core was utilized to create a metallurgical bond for maximum heat transfer. Seven segments were extruded in developing the final technique. The last sample segment had an extruded cermet core density of 8.28 gms/cm³, and a dysprosium content slightly in excess of the specified 1.95 gms/cm³ minimum. This segment was completed to the final design and delivered to DUN at Hanford for in-reactor evaluation. A typical segment is shown sectioned in Figure 5, appended.

A full-scale mockup rod made of solid stainless steel sections was fabricated and tested in a distorted channel mockup (in Bldg. 189-D) as shown in appended Figure 6. This testing indicated that the maximum rod stress of 3,300 psi occurred in the rod section within the step plug. This value is within the capability of the cladding. Load tests indicated that the withdrawal forces (up to 275 lbs.) were greater than the insertion forces (up to 160 lbs.). The channel was purposely blocked and the rod driven into the obstruction without any damage to the rod or channel.

FABRICATION OF THE PROTOTYPE ROD

The processing sequence used by GE-NMPO in fabricating each segment for the uncooled rod prototype can be followed on Figure 7, appended. The principal steps are summarized below.

Preparation of Ni-Dy₂O₃ Cores

A dry powder mixture of 30% Dy₂O₃ and 70% Ni, by weight, was slurried with propanol, ball milled for 16 hours, and then vacuum dried at 250 F for 24 hours.

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The mixed powder was then sieved through a -50/+125 mesh, vibrationally compacted into a rubber mold, and isostatically compressed at 12,000 psi. The resultant compaction was hand granulated, vacuum dried, and again vibrationally compacted into a rubber mold and isostatically pressed at 60-70,000 psi. The compact was next vacuum sintered at 2400 F for three hours.

Fabrication of the Extrusion Billet

Each Ni-Dy₂O₃ compacted core was dry machined to close tolerances and fitted into an Inconel 600 extrusion billet. The two parts were sealed in place by welding of the end cap. The close fit of the core to the clad, and the uniform density of the core, influence the core-clad thickness ratio in the as-extruded condition.

Billet Extrusion

The extrusion operation is the most critical phase of the fabrication process. Its reproducibility is highly dependent upon the flame-sprayed ZrO₂ extrusion die and the flow of glass-pad lubricant between the die and the work piece.

The extrusion billet was induction heated in an argon atmosphere to 2,000 F and held at this temperature for a 10-minute soak period. Using an average force of 800 tons, the billet then was extruded with a 6/1 area reduction at a rate of 2-1/2 inches/sec. The result was a 48" long section with a core density of 8.440 gms/cm³, which is almost exactly the Inconel 600 density of 8.439 gms/cm³. The excellent bond and the very similar densities of core and cladding made ultrasonic determination of clad thickness very difficult. To compensate, the finished dimension was increased slightly to enable final machining without any exposure of the cermet core.

Finishing

The extruded segments were cut to length and rough machined on all surfaces. The ends were then machined to accept the Inconel 600 hinge blanks. These blanks were welded to the segment and the composite piece machined to final dimensions. A full-length rod consists of twelve such segments. One complete prototype rod and two extra segments were fabricated by GE-NMPO and delivered to DUN.

IN-REACTOR TESTS

Segment Test

The purpose of this in-reactor test was to determine (1) the operating temperature of an uncooled rod segment, and (2) any possible damage resulting from the irradiation and high temperature exposure. The test segment was instrumented with thermocouples along its sheath and in the core, and placed in a side-to-side test hole channel at KW Reactor as shown in Figure 8, appended. Temperatures were recorded continuously during reactor operation. The maximum sheath temperature experienced was 1875 F during a high heat cycle.

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After 960 hours of reactor operation, the segment was removed and visually examined in the Radiometallurgy Laboratory. Structural failure was evidenced by the 700-mil uniform sag noted in Figure 8, but there were no areas of corrosion or scaling. Subsequent laboratory furnace testing determined conclusively that the sag was caused primarily by creep of the Inconel 600 cladding. The experimentally determined creep strength values were an order of magnitude higher than those published by the material manufacturer.

Although the unusually high reactor temperature cycles may have caused the test segment to sag, there was concern for the effect of such bowing in an installed HCR. However, subsequent laboratory tests indicated that the distortion does not interfere significantly with rod insertion or withdrawal.

Prototype Rod Test

The prototype uncooled rod shown in appended Figure 9 was installed in the No. 2 HCR channel at KE Reactor in January 1969. The performance of the rod to date has been excellent from both operational and maintenance standpoints. Induced radiation levels are as predicted; they are substantially higher than with the original rods because of the nickel content of both sheath and core.

UNCOOLED ROD SIGNIFICANCE AND OUTLOOK

The basic R&D efforts of the GE-NMPO were funded through both AEC-RL and the AEC's Division of Reactor Development & Technology. The prototype rod and the two finished segments were funded through a memorandum purchase order (R6S-555-12218) for approximately \$41,000. The high costs are largely due to the expensive co-extrusion process, and the significant machining effort required after extrusion. Substantial savings could be achieved by hot-forming a pre-fabrication sheath over the core, thereby eliminating the machining step.

Based on the tests and operating data to date, the uncooled rod is a technical success. Furthermore, elimination of the cooling requirement not only simplifies the HCR system and increases its reliability, but also eliminates the personnel radiation exposure associated with maintenance work on the HCR coolant system. Although induced radiation levels are higher with the uncooled rod, no maintenance would be expected with this rod for the life of the reactor.

Other potential uses for an uncooled rod certainly include N Reactor, particularly under "power only" operation. While this specific design can be considered only marginal for this application, the concept of an uncooled rod is highly desirable. Additional R&D effort would probably solve creep strength-temperature problems. A primary concern for N application stems from the fact that the HCRs there function also as safety rods, and therefore "scram" into the reactor at a high rate of speed. Buckling of the segmented rod could be a serious problem.

At present, there are no plans to procure additional uncooled rods for the K reactors. The cooled flat HCR, although much less flexible than the uncooled rod, satisfies the current requirements and does so at a much lower price (~\$7,000 per rod). However, if the HCR channels become distorted beyond the capability of these cooled rods, a proven solution in the form of the uncooled, segmented rod is readily available.

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Figure 1. HCR Channel Traverse - No. 2 Rod at KE

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Figure 2. HCR Pattern - K Reactors
(Uncooled rod in No. 2 Channel)

Figure 3. HCR Channels - K Reactors

KEY:

- 1 - Hinge, left half
- 2 - Hinge, right half
- 3 - Pivot pin
- 4 - Locking pin
- 5 - Pivot pin, flanged tip
- 6 - Pivot pin, head
- 7 - Hinge, pin locking shoulder

Figure 4. Uncooled HCR - Hinge Design

Figure 5. Uncooled HCR - Typical Segment

Figure 6. Mockup of Distorted HCR Channel
(Arrow on simulated shield shows rod centerline)

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Figure 7. Uncooled HCR - Fabrication Flowsheet

Figure 8. Segment of Uncooled HCR
(Showing sag developed in test)

Figure 9. Prototype Uncooled HCR
(Showing maximum horizontal flexure)