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DOUGLAS UNITED NUCLEAR MONTHLY REPORT

FEBRUARY 1970

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DOUGLAS UNITED NUCLEAR

MONTHLY REPORT

February 1970

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SUMMARY

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REACTOR PLANT OPERATIONS

Production Statistics

	KE	N
Input Production - Pu (KMWD)	1.2	81.9
Time Operated Efficiency - %	1.1	80.8
Steam Availability to WPPSS - %	-	78.3

K Reactors

As scheduled, the KW Reactor was shut down for deactivation at 12:01 a.m. on February 1. That same day, KE Reactor began an extended outage for the replacement of wye-joints in the coolant supply piping, for system proof testing, and for other maintenance. The wye-joint work was about 90 percent complete by month end.

N Reactor

Operation was at the 4000 MW maximum authorized power level at the start of the month, but subsequently was lowered to 3800-3900 MW to provide a small additional margin between tube temperatures and their limits because of reactivity considerations. Reactor stability has been excellent while operating with less than 1 milli-k excess reactivity in rods. Longitudinal traverse data continue to show a strong reactivity depression in the vicinity of a central ball channel.

There were four outages, all unscheduled. Two were manual scrams upon confirmation of fuel failures, one was an automatic trip of the safety circuit caused by a simultaneous trip of four Zone Temperature Monitor channels, and the fourth was a manual shutdown to permit the testing and adjusting of valves in the sixteen safety flow control legs.

Steam generator retubing work continued in Cell 1.

FUEL AND TARGET FABRICATION

Production Statistics (tons)

	For KE	For N
Billets Extruded	-	0
Finished Fuel Produced	205.0	13.8

K Fuels

AlSi canning operations continued on the basis of four lines except for two days of five-line operation.

N Fuels

Because of realignment and overhaul of the extrusion press, no input production was scheduled.

TECHNICAL ACTIVITIES

K Reactors

Physics calculations in support of the dual-riser failure analysis under current operating conditions have been completed and sensitivity calculations for the various parameters are under way.

In another phase of the brittle fracture program, applicability of the proof test to dynamic loads was discussed with the AEC Division of Reactor Licensing. The contention that proof testing can be used to demonstrate protection against dynamic loads in the straight pipe runs appeared to be acceptable. It was agreed that the nozzle-to-riser welded joint is acceptable for static conditions, and that successful completion of the proof test would permit operation at full pressure on dual-riser failure limits.

Graphite stack measurements at KW show that "turnaround" (change from contraction to expansion) occurred in the high-flux region at 7×10^{21} nvt exposure and 2.4% contraction. KW core samples irradiated in the ETR had indicated that turnaround would not occur until 11×10^{21} nvt and 3.5% contraction. Turnaround has not been observed at KE, which continues to contract linearly with exposure.

Calculations have been made to determine physics limitations to the amount of plutonium-aluminum fuel that could be charged under existing nuclear safety criteria. Within a normal uranium loading, speed-of-control considerations limit the allowable Pu loading to 50 ± 5 kg and total control considerations limit it to 75 ± 5 kg.

Eight NpO_2-graphite wafered target elements, irradiated in January to determine the cause of growth failures with elements of this type tested last year, were discharged from KW Reactor. No failures were experienced with these eight elements; growth analysis of the individual wafers is in progress.

N Reactor

As a check of the models used in the MIMIC simulation of the primary coolant high pressure injection/pressure control systems, a recent scram was used as a comparison. Close agreement between the simulation and observed data for pressurizer level, pressurizer pressure, and total injection pump flow was noted.

Mark IV fuel end-spider support life tests in the 189-D loop have shown that support performance is acceptable with one leg intentionally defected on both ends of an assembly, but that total support failure occurs when two legs are severed on both ends.

Two additional metal-water reaction tests using irradiated N Reactor fuel were performed at temperatures in excess of the uranium melting point. Significant hydrogen evolution was observed, and the extent of the metal-water reaction appears to be strongly influenced by the large sloping molten uranium surface that develops at each end of the element. In an actual fuel column, the adjacent elements would be expected to reduce this surface area significantly.

An accelerated program of graphite stack inspection and measurement has been developed and work on means to correct the gaps in the ball channel liners has been initiated.

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IRRADIATION SERVICES

Trial separation runs using natural xenon and elemental iodine for the Xe-128 development and production work requested by Argonne National Laboratory indicate that xenon recovery > 99% can be obtained with product impurities of < 1%.

FEATURE REPORT

This month the appended summary report describes the fog spray subsystem at N Reactor, presents a ranking of critical failure modes for various configurations and operational alternatives, and illustrates the magnitude of subsystem reliability improvements available for each alternative considered. These analyses typify the current effort to evaluate the reliability of critical N Reactor systems.

GENERAL

The force reduction is proceeding on schedule. Total employment dropped from 1675 on January 31 to 1595 on February 28. By June 30 about 45% of the Company's minority employees probably will have been lost.

A dispute concerning the seniority status of previously promoted employees who are returned to the unit has been presented to arbitration and a decision is expected on March 10. Another dispute concerning pro-rata vacation entitlement of laid-off employees is now being scheduled for arbitration.

A disabling injury was incurred on February 12 when an N Plant pipefitter suffered a loss of bone from the tip of one finger while operating a magnetic drill. No radiation exposures exceeded operational control.

Charles D. Harrington

Charles D. Harrington
President

REACTOR PLANT OPERATIONS - KE & KW

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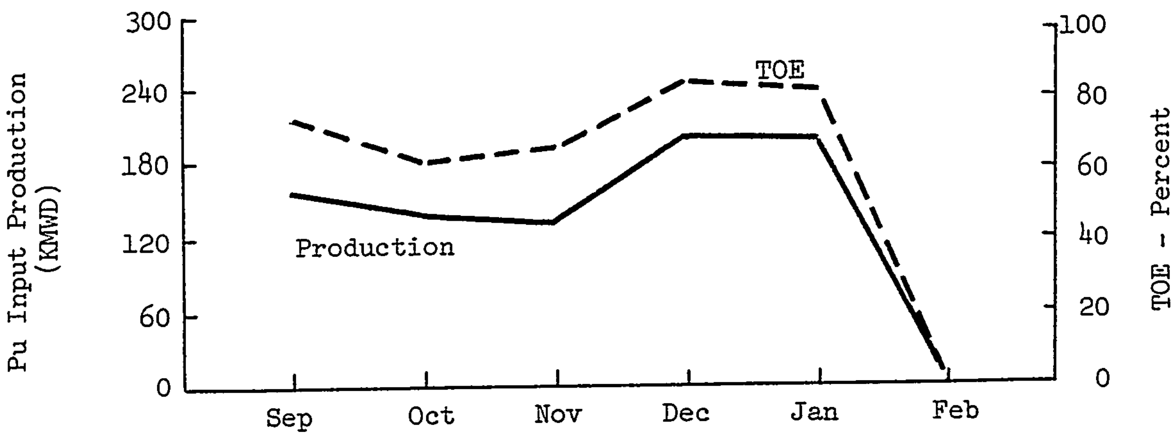
PRODUCTION

General

The KW Reactor was deactivated on schedule at 0001 hours February 1. On that same day, KE Reactor began an extended scheduled outage for the replacement of wye-joints in the process water supply piping and for other maintenance work. KE is scheduled to resume operation in late March.

Combined KE and KW input production and time operated efficiency (TOE) for the past six months are shown on the following chart:



Statistical Summary

	KE Reactor
Input Production - Pu (KMWD)	1.2
- U-233 (Equiv. KMWD)	-
Power Level (MW) - Maximum	4,000
- Average	4,000
Time Operated Efficiency - %	1.1
Number of Outages	1
Number of Startup Interruptions	0
Operating Coolant Flow - 1000 gpm	199.5
Fuel Charge (Tons) - 94 Metal	89.3
- Natural U	344.9

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	KE Reactor
Fuel Element Failures	0
Helium Losses - 1000 cu. ft.	35.8

OPERATING EXPERIENCE

Reactor Loadings

KW Reactor discharge commenced six hours following the shutdown and was completed at 0510 hours on February 15. The coolant water supply was removed from the reactor at 1030 hours on February 16; the remaining water was subsequently drained and then blown from the tubes with compressed air.

The front face map showing the loading in KE Reactor is reproduced on the page following B-3. The tonnages listed are approximate; actual fuel charge totals are tabulated above.

Reactor Outages - KE Reactor

Date Down	Outage Hours	Remarks
February 1	664.7	Scheduled wye-joint replacement and other maintenance work.

Boiler Operation - 165 Buildings

Steam generation at both 165-KE and KW was reduced to one boiler throughout the month. KE was reduced for economic gains during the wye-joint outage and KW due to deactivation. During this period, the one boiler in KW supplied the glycol system heating steam.

EQUIPMENT EXPERIENCE

Deactivation of KW Reactor

At the time of shutdown, the graphite in KW was dry and no equipment problems were evident. Deactivation work is progressing on schedule. At month end Processing work was approximately 72 percent complete, Power work was 60 percent complete, and Maintenance work was 44 percent complete.

Wye-Joint Outage

Good progress was made on replacement of the four large wye-joints. At month end this work (being performed by J. A. Jones forces) was estimated to be 90 percent complete. Proof-testing will be conducted in March.

Other maintenance work at KE included the following:

- Five flexible vertical safety rods were installed. All five

[redacted]

channels were borescoped prior to installing the rods. The need for sleeve repair work in No. 49 was confirmed.

- No. 12 HCR was replaced with a cooled flat rod. The step-plug for No. 7 HCR was replaced with one of modified design, and the tip section of the rod was replaced because of a deformation. No. 8 HCR has been removed and its channel is being prepared for a cooled flat rod.

- Seventeen thin-walled aluminum process tubes were replaced.

- Forty-nine RTDs were replaced, and 75 more are scheduled for replacement during the current outage.

Boiler Tests - 165-KE

The No. 1 KE boiler was in service during the week of February 16 for tests following instrumentation changes. This boiler is the last of six to be thus modified. Tests were conducted under the direction of the Beckman Instrument Company. The unit was removed from service on February 20 pending additional Visacorder tests.

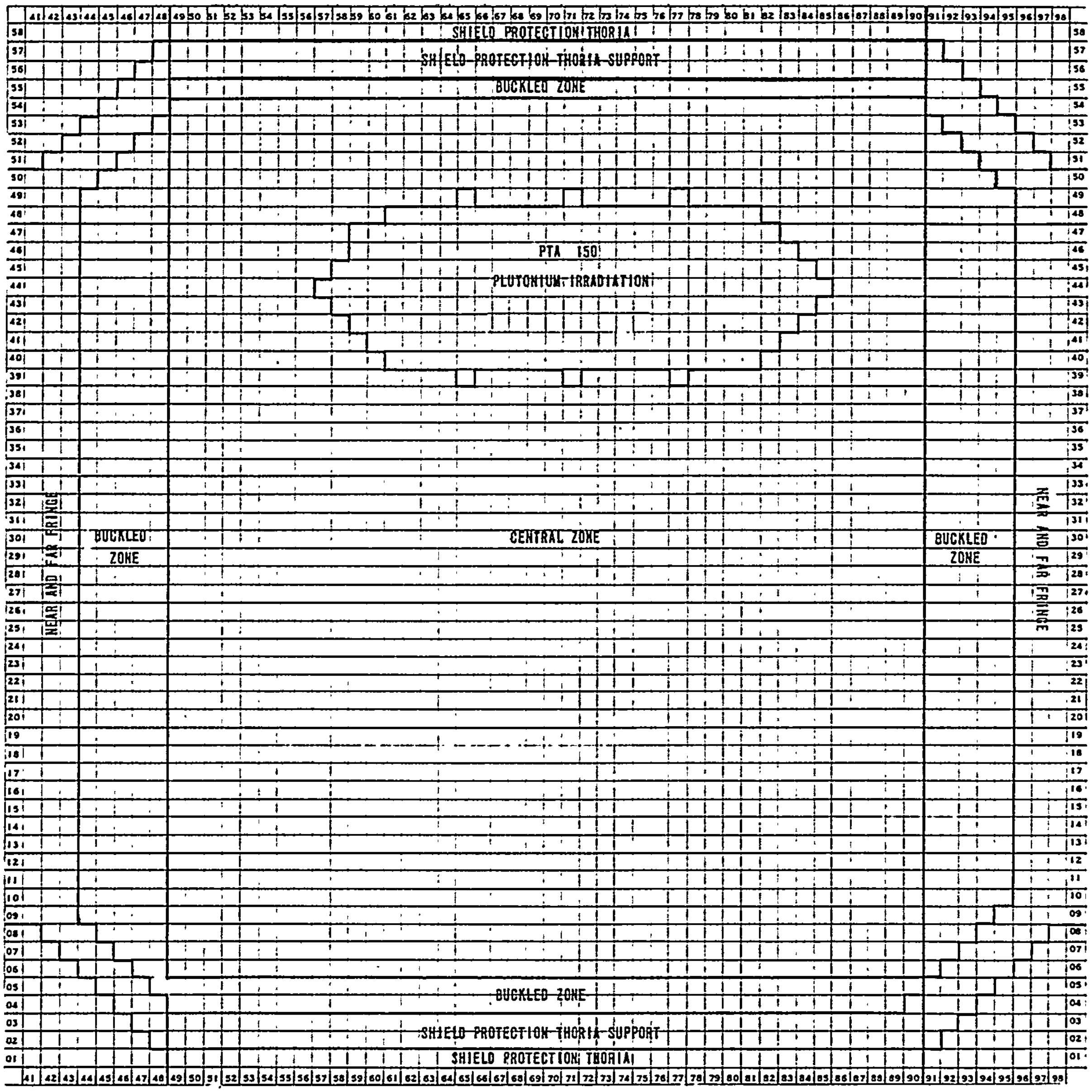
PROCESS ASSISTANCE AND CONTROL

Operational Physics

Nothing to report.

HCR Poison Burnout

Following the replacement of HCRs 8 and 12 during the current KE Reactor outage, burnout to the 90 percent effectiveness level is calculated to occur in only two of the remaining rods before late 1972; these are HCRs 10 and 18.



Zone	Tons	Material
Central	212 66	Natural Uranium 94 Metal
Buckled	82	Natural Uranium
Near and Far Fringe	48	Natural Uranium
Shield Protection	23 7	94 Metal (Thoria Support) Thoria

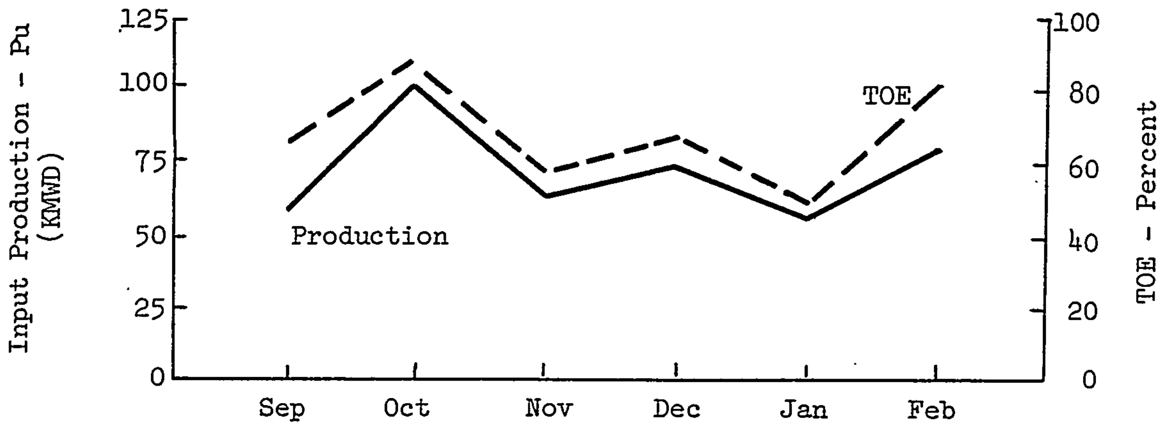
Loading Pattern - KE Reactor

REACTOR PLANT OPERATIONS - N

PRODUCTION

General

Reactor production (a combination of fuel grade and blend material), 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)		81.96
Electrical Generation (KMWH)	- WPPSS	400.77
	- 184-N	5.79
	- Total	406.56
Power Level (MW)	- Maximum	4,000
	- Average	3,622
Time Operated Efficiency - %		80.8
Steam Availability - %		78.3
Number of Shutdowns	- Scheduled	0
	- Unscheduled	4
Fuel Failures		2

Fuel Charge (Tons)	- 94 Metal	322.00
	- 125 Metal	65.35
	- Natural U	.37
	- Total	387.72
Helium Losses - 1000 cu. ft.		332.10
Fuel Oil Usage - bbl.		17,501

OPERATING EXPERIENCE

Reactor Loading

The reactor loading at month end is shown on the front face map which follows page BN-6. The only change during the month resulted from the charge-discharge of two tubes in which fuel failures occurred (see page BN-5). In each case, the element cladding fractured suddenly which resulted in some fission product contamination of the primary loop.

Power Level

Power level was administratively limited throughout the month, except during periods of nonequilibrium operation. The administrative limit was at 4000 MW during February 1, 2, 16, 17, 18 and 19. During the balance of the month it varied from 3800 to 3900 MW to (1) provide a small additional margin between tube temperatures and their operating limits, and (2) avoid the reactivity transient which would result if a power cut was necessary while operating with less than 1 milli-k of reactivity in control rods (i.e., if sufficient reactivity could not be maintained by decreasing steam pressure).

Because of the shortage of reactivity, it was necessary to reduce main steam header pressure (MSHP) to approximately 119 psig starting on February 1 and until the unscheduled shutdown on February 2. This resulted in a 35 MWe reduction in electrical generation by WPPSS during this period. For the balance of the month, it was possible to maintain MSHP at approximately 123 psig, thus enabling electrical generation by WPPSS at their station's design power of 800 MWe.

Reactor Outages

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

Date Down	Outage Hours	Cause
February 2	24.2	Manual shutdown due to a fuel failure in process tube 1046.
February 10	25.6	Automatic shutdown from a Zone Temperature Monitor trip when four ZTM channels tripped simultaneously.

Date Down	Outage Hours	Cause
February 19	56.4	Manual shutdown to check and adjust valves in the safety flow control legs.
February 23	22.6	Manual shutdown due to a fuel failure in process tube 2456.

EQUIPMENT EXPERIENCE

Primary Loop

The V-11 valve operators on tubes 1046 and 2456 were converted to manual. These valves were not repacked.

A flow monitor sensing line leak (tube 3056) was located and repaired.

Primary loop leak rates continued high, increasing from about 220 gpm at the beginning of the month to about 400 gpm at the end. No significant primary loop leaks were repaired during February other than seating of the V-21-5 valve.

The multiple trip of Zone Temperature Monitor channels on February 10 was believed to have resulted when the power supply voltage dropped, due either to too many channels being placed on bypass or to overloading due to too many grounded RTDs. As a consequence, bypassed ZTM channels are currently disconnected from the power supply in order to reduce the load on the system, and the system status is being recorded formally.

Due to an increased inleak rate on steam generator 5B, it was inspected during the February 19 outage and one broken tube (row 34, line 8) was plugged.

Cell 1 retubing by Combustion Engineering progressed about on schedule. Installation of new tubing began on February 27.

Boiler Experience

Investigation of the CE-1 and CE-2 boiler flameouts that resulted in loss of B bus during startup from the February 19 outage revealed few possible causes of failure. A relay mounting screw on the forced draft fan breaker was found loose. The screw was tightened and the breaker contacts cleaned. A 3-foot section of wiring between the low-level water column and a terminal block was found to be rather severely charred from high temperature. This section was replaced with high temperature wiring, and the terminal block was tightened. The boilers were returned to service and tested at 200,000 pounds per hour load without further incident.

The FW boiler was shut down on February 20, following observation of tube leakage. Two tubes were plugged, and leaks on three other tubes were welded. The boiler was returned to service on February 22.

Horizontal Control Rods

Difficulty was experienced in operating left-side rods during operation. It was necessary to backflush the hydraulic lines from the pump suction into the oil reservoir through the filters.

During the February 19 outage, the left-side oil reservoir was cleaned, the filters replaced, and the system returned to operation. Scram accumulators on rods 73 and 97 were removed and examined for fouling. Both were determined to be in good condition. O-rings removed from the control valves were found to be soft and tacky. Scram solenoid valves were checked and determined to be satisfactory. All rods were bled, exercised, scram tripped, and timed successfully.

The problem appears to be the result of bacteria growth in the emulsion. Further study by Engineering is underway.

Circulating Raw Water System

During the February 10 outage, the failed brush holder on the motor of CRW pump No. 2 was repaired. The badly burned brush holder was first noticed during operation on February 4. This failure, coupled with the fact that inadvertently one brush had not been installed, led to the need for constant monitoring by an electrician until an automatic trip device was installed to protect the motor in case of additional circuitry problems.

No. 1 CRWP motor was inspected during the February 19 outage and it was found that the slip rings were rough. This condition was corrected by honing.

Emergency Raw Water Cooling System

The reactor shutdown on February 19 was made to determine the status of the CV-2 and V-10 valves in the 16 safety flow control legs.

The CV-2 valves are two-way tilting disc type valves with adjustable stops provided to permit limited reverse flow for emergency use, or for flushing the lines. These valves are located in the four inch leg between the bottom of each primary coolant loop inlet header and the emergency cooling water manifold downstream of the V-3s (emergency coolant supply valves). The safety question related to how far the check valves were unseated by the adjusting pins.

Backflow tests run after shutdown indicated that CV-2-3, 4, 5, and 7 on the left side, and CV-2-11 and 13 on the right side, had more flow than proper. The maintenance stop positions were determined, and the pin adjustment procedure was then applied to get all the valves correctly set. Two of the six valves with high backflow had pins set at about the right place, which indicated that the swinging discs may not have seated on the pins. The other four were found to be improperly adjusted.

All 16 of the CV-2s were adjusted and backflow tested satisfactorily before startup. In addition, the positions of all 16 of the V-10 valves were verified as being full open.

Equipment Modifications

MDC-N-69-72 "Ball Hopper Gate-Closed Indicating Light Color" was completed This Minor Design Change provided for white instead of green hopper gate-closed lights to increase the illumination and permit rapid determination of the ball hopper gate status.

PROCESS ASSISTANCE AND CONTROL

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Operational Physics

Stable reactor operation has continued with less than 1 mk excess reactivity absorbed in control rods. The potential reactivity gain of up to about one-half mk from cutting steam pressure has been withheld to keep available fine shim control for maintaining the control configuration in the desired minimum-reactivity pattern.

Flux traverses continue to indicate a strong control effect from residual balls in the vicinity of channel 49.

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

Effective Central Tubes*	815
Flattening Efficiency** - February	0.81
- 12-Month Average	0.81
Maximum Operating Time Permitting Scram Recovery - Hours***	24

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

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

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

Production Fuel Performance

The Mark I-C outer fuel element which failed on February 2 was in position 8 from the rear of tube 1046 (average column exposure - 1705 MWD/T).

The point of water entry was at the mid-point of the element at a buggy spring in line with the locking clip. The inner diameter of the element had a large swelled area that had pushed the inner to one side. One side of the swelled area had ruptured, and the ensuing reaction perforated the element wall. The outer cladding has a swelled area with two blow-out holes and a crack approximately seven inches long. Initiation of the failure is suspected to have been fretting of the inner clad of the outer element by a foreign object or the inner buggy spring. Radiometallurgical examination is planned to confirm the mechanism involved.

The Mark IV outer element which failed on February 23 in position 4 from the rear of tube 2456 was at approximately 9 percent of goal exposure. The failure mechanism has not yet been determined.

	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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14	E	E	94	X	X	G	E	G	E	F	F	F	G	G	G	E	G	E	F	G	E	01	G	G	01	E	F	G	G	X	X	X	ES	E	14
13	E	ES	X	X	X	G	F	E	E	G	F	F	G	F	E	F	F	F	G	G	F	F	G	F	G	F	G	F	F	X	X	X	ES	ES	13
12	E	E	X	X	X	F	F	F	F	F	F	F	G	F	E	G	E	F	94	F	G	F	G	F	F	F	G	G	G	X	X	X	E	ES	12
11	E	ES	X	X	X	E	F	F	F	F	E	E	F	F	F	F	G	G	F	D	D	G	F	F	G	F	F	G	F	X	X	X	ES	ES	11
10	E	E	X	X	X	G	F	E	F	F	F	F	F	G	F	F	F	G	G	E	F	G	F	F	G	G	F	G	G	X	X	X	ES	E	10
09	ES	E	X	X	X	X	E	F	F	E	E	F	F	F	E	E	F	F	F	F	G	G	G	F	F	F	G	G	X	X	X	X	E	CE	09
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03						E	76	ES	E	E	E	E	E	E	E	ES	E	E	E	E	E	E	E	E	E	E	ES	ES	E						03
02							E	CE	E	E	E	E	E	E	ES	E	ES	ES	ES	ES	ES	ES	E	E	E	E	ES	E							02
01																																			01
	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65	66	67	68	69	70	71	72	73	74	

Fuel Code	No. Tubes	Description
CE	5	Mk-IC (94 Metal - Fringe)
D	2	Mk-IC (94 Metal - High U-236)
E	257	Mk-IC (94 Metal - Fringe & Central)
ES	60	Mk-IC (94 Metal - Fringe)
F	250	Mk-IV (94 Metal - High U-236)
G	183	Mk-IV (94 Metal - Central)
N	1	Mk-IB (Natural U)
X	230	Mk-IA (125 - 94 Metal)
	988	Total
	16	Total PTs
	1004	Grand Total

PT-NR No.	No. Tubes	Description
76	1	Fuel Meltdown Test
94*	10	Mk-IV Demonstration
01	4	Mk-I & Mk-IV from Direct Cast Billets
■	1	Graphite Samples Channel
	16	Total PTs

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*Includes Mk-IV high U-236 content fuel, 1 tube with Mk-IA 125-94 Metal and Mk-IV 94 Metal, and 6 tubes with Mk-IV-AA 125 Metal and Mk-IV 94 Metal.

Loading Pattern - N Reactor

FUEL AND TARGET FABRICATION - K REACTORS

DECLASSIFIED

PRODUCTION

General

Production of AlSi-bonded fuel for KE Reactor was 131.4 percent of forecast. Canning operations continued on the basis of four lines per day on the day shift, five days per week, except for two days of five-line operation to utilize available manpower. All of the elements produced had self-supports or bumpers attached.

Acceptable Elements Produced

	Tons	Yield - Percent Current Month	Yield - Percent FYTD
AlSi-Bonded Fuels	205.0	95.9	95.7

Month-End Inventories

		Tons
Bare Uranium Cores		778*
Finished Fuel:	AlSi-Bonded	1,288*
	Hot-Die-Sized	19
Thoria Elements		14

*These totals include 132 tons of bare cores and 110 tons of finished fuel of the sizes used in the smaller reactors.

OPERATING EXPERIENCE

Canning line efficiency of the AlSi-bonding lines was 97.8 percent. Downtime was assigned 68 percent to equipment malfunctions and 32 percent to operations.

EQUIPMENT EXPERIENCE

Nothing of significance to report.

PROCESS ASSISTANCE AND CONTROL

AlSi process model K5N spires received during the month were found to have OD defects up to five mils deep. A statistical sample of these spires was used in a control canning test and results of the total ID bond count from the UE-1 internal test probe on these canned elements indicated the spires were acceptable since the total count was well within process limits. Hence the total lot of spires was approved for use.

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FUEL AND TARGET FABRICATION - N REACTOR

DECLASSIFIED

PRODUCTION

Input Production

Total billets extruded	0
Tons extruded	0
Percent of forecast	-

Output Production

Total finished fuel assemblies	540
Tons output	13.8
Percent of forecast	106.2

Uranium Utilization - % (No input)

Month-End Inventories - Tons

Bare uranium billets	268
Finished fuel	251

OPERATING EXPERIENCE

No input production was scheduled because the extrusion press was down for overhaul and realignment during most of the month.

EQUIPMENT EXPERIENCE

A four-week overhaul and realignment of major components of the 2750-ton Loewy Hydropress began February 12. To realign the platen, the four press columns or tie-rods must be evenly stressed and locked at a pressure in excess of that encountered during the normal extrusion cycle. This is accomplished by applying and holding a maximum pressure of 3200 psi to the main and side rams, while the inner compression nuts are torqued and locked against the platen. Since 3200 psi exceeds the main cylinder prefill valve maximum working pressure rating by 7 percent, the valve manufacturer's concurrence was obtained before the work was done. No difficulties were experienced and the platen was successfully realigned.

PROCESS ASSISTANCE AND CONTROL

A survey of the literature indicated that a rough uranium surface could enhance wetting of the uranium surface during brazing and hence reduce cap-to-core bond rejects. Pilot-plant tests using a six-minute prebraze etch bath rather than the standard three-minute bath confirmed that the rougher surface was beneficial and reduced these rejects from 1.1 percent to 0.2 percent. Process Standards will be changed to specify a minimum of six minutes for prebraze etching.

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TECHNICAL ACTIVITIES - K REACTORS

RESEARCH AND DEVELOPMENT

Basic Production

Brittle Fracture Program - K Inlet Piping

K Limits Physics Studies

Physics calculations in support of the dual riser failure analysis under current operating conditions have been completed and sensitivity calculations for the various parameters are under way. Efficiency improvements in the riser evaluation method include a kinetics-equivalence determination for smeared VSR strength top-to-bottom, the documentation of void coefficient values employed as anticipated in the January report, and modification of CLUMSY input handling to eliminate hand calculations between feedback steps.

Proof Testing vs. Dynamic Loading

Applicability of the proof test to dynamic loads was discussed with DRL and their consultants on February 4. The calculational procedures and reasoning to support the contention that the proof test can be used to demonstrate protection against dynamic loads in the straight pipe runs appeared to be acceptable. One area of concern was that the design of the nozzle-to-riser welded joint provided a built-in one-inch-deep defect which might be extended by fatigue to become a sharp-edged flaw. Depending on the applied stress and dynamic toughness of the material, this flaw might become unstable in the event of dynamic loading. It was agreed, however, that the joint was acceptable for static conditions and that successful completion of the proof test would permit operation at full pressure on the dual-riser failure limits.

Segmental Discharge Studies

Recent calculations of potential advantages of segmental discharge in a K reactor have been documented in DUN-6694, "Production Data for 26-Piece KVN Segmental Discharge". In addition, CAGE data are being prepared for the segmental discharge cases for 4, 5, and 6 wt% Pu-240 plutonium.

Through-Reactor Decontamination

The Production Test (PTA-158, Supplement B) of through-reactor decontamination with a full crossheader at KE Reactor, employing preheated decontamination solutions, was performed on February 19. Results will be analyzed and reported in March.

Zircaloy Process Tube Hydriding

Production Test PTA-204, for removing the hydride case layers from ten Zircaloy process tubes in KE Reactor using the electrochemical method, was conducted on February 27. Results will be analyzed and reported in March.

Rear-Face Dose Rate Study

Investigations are being conducted to determine the source of Sc-46, an assumed major contributor to increased rear-face radiation dose rates experienced over the past few years at the K reactors. KE Reactor was operated for one day with one-half of the reactor cooled with water treated with liquid alum and the other half treated with the normal flocculating agent. Samples of coolant were taken from each half of the reactor after equilibrium conditions were established, and are being analyzed by neutron activation to determine if bauxite is contributing parent scandium.

Half-Plant Dichromate Test

The weight loss data for aluminum fuel jackets exposed to coolant containing 0.5 ppm and 1.0 ppm sodium dichromate solution in a half-plant test at KW Reactor correlate well with temperature and dichromate concentration. The fuel cladding corrosion loss with 0.5 ppm dichromate is about 1.45 times the corrosion loss with 1.0 ppm dichromate. A clear-cut correlation of pit depth was not obtained. However, for exposures of about 100 operating days, it appears that (1) below a fuel surface temperature of about 102 C, pit depths will be less than 20 mils, but (2) pit depths to the AlSi layer can be expected at surface temperatures greater than 120 C. Analyses are being conducted to determine the effect of dichromate concentration on type and frequency of attack.

"Turnaround" of KW Graphite Stack

In December 1966, 18 graphite samples were cored from the filler blocks in the center of KW Reactor. These samples were subsequently irradiated in the ETR at various temperatures to determine the exposure at which "turnaround" (change from contraction to expansion) occurred. Turnaround on samples irradiated at typical K graphite temperatures was observed at an exposure of approximately 11 x 10^{21} nvt which would have corresponded to a projected turnaround in the K reactors by 1975-1976.

Contrary to predictions based on the core samples, KW Reactor has shown a decreasing contraction rate during the past year. Recent measurements confirm that turnaround occurred in the high flux center region and expansion there had begun. This KW turnaround took place at an exposure of 7 x 10^{21} nvt and 2.4 percent contraction compared to turnaround of the core samples at 11 x 10^{21} nvt and 3.5 percent contraction.

No such turnaround has been observed at KE Reactor which continues to contract linearly with exposure. The exposure of the KE stack is similar to that of KW. At this time no explanation can be made for the difference in behavior of the two stacks.

Product Flexibility

High Enrichment Utilization Studies

PuAl Loading Limitations

Calculations have been made to determine physics limitations to the amount of plutonium-aluminum fuel that could be charged under existing nuclear safety criteria. Within a normal uranium loading, speed-of-control considerations limit the allowable Pu loading to 50 ± 5 kg and total control considerations limit it to 75 ± 15 kg. Provided that the PuAl columns would be scattered sufficiently to preclude post-meltdown conglomerations, the net meltdown effect of both the normal uranium loading and the PuAl diffusion into the graphite stack would be strongly negative. Effects of PuAl element slumping and of flowing to the center of zirconium tubes would not add inordinately to total control requirements.

Ten-Kilogram PuAl Irradiation (PTA-150)

Decreasing blackness of the PuAl elements with plutonium burnup, combined with loading changes during the January KE Reactor outage, resulted in a reduction of approximately 30 percent in column heat generation rates compares to early December. Replacement of five additional thoria columns in rows 43 and 45 during the current extended outage is being considered.

Oralloy Test Planning

Work continued on preparation of the Production Test Authorization for irradiating oralloy in the top of the KE enrichment ring. The technical bases document (DUN-6618) was issued.

Pu-238

Some of the NpO_2-graphite wafer elements irradiated in 1969 as part of PTA-163 (DUN-5113, "PTA-163 - Clean Pu-238") failed from longitudinal growth of the wafers after about 300 MWD/AT exposure. To determine the cause of these failures, eight target elements were irradiated (under PTA-194) in KW Reactor last month to about 270 MWD/AT; they were discharged shortly after reactor shutdown on February 1. No failures were experienced with the eight elements. The individual wafers in each element are being analyzed for growth in relationship with the various fabrication and irradiation parameters used. This work should be completed during the next report period.

Development work is proceeding on the advanced thin-annulus (tubular) elements in both the graphite and water core designs. Pressing and sintering trials on cermet rings are progressing well using UO_2 as a stand-in for NpO_2. Design of cladding and end caps, plus the assembly methods for both element designs, is nearing completion. A 2 x 2 cluster irradiation of prototype elements is planned for KE Reactor in April.

Environmental and Regulatory Technology

Nothing to report.

Waste Management

K Reactor Effluents

DUN-6660, "Radiologic Aspects Associated with the Ground Disposal of K Reactor Effluent", contains the results of an evaluation of the radiological implications of the production, disposal, and potential future release of radioactive material attendant with ground disposal of K reactor effluent. Particular emphasis was placed on the consequences of future release of stored radioactive material to uncontrolled areas as a result of a "once in a century" accident.

Data in the report show that soil retention of radionuclides associated with a 10-year continuous operation span of a K reactor and subsequent complete release to the river in one day would not, after dilution, produce concentrations in the river that would exceed the limits for uncontrolled release as specified in 10-CFR-20 limits for routine releases.

It is concluded that, in terms of radiologic considerations, the potential for a significant uncontrolled release of stored radionuclides does not preclude the concept of ground disposal of effluent from operation of a K reactor.

ENGINEERING AND TECHNOLOGY - REACTORS

Prototype Uncooled HCR

The uncooled horizontal control rod installed in No. 2 channel at KE Reactor in January 1969 has been visually examined after a year of service through remote TV and photography. Figure 1, appended to this Section on page D-A, shows the second segment from the connector block end of this prototype rod. This rod segment is unirradiated and still retains its original shiny condition. Photographs of segments No. 6 and No. 7 (Figure 2 on page D-A) revealed 3/8 inch bows in each of these segments. Also, the Inconel-600 sheath has tarnished due to surface oxidation in the high temperature environment.

This rod operates with about 20 percent out of the reactor, and segments 6 and 7 are normally positioned in the most severely distorted section of the channel. Some bowing due to creep was expected and presents no problem. No operating difficulties have been encountered and the in-reactor test has been considered completely satisfactory. A Design Change has been prepared to authorize permanent use of this prototype rod.

At this time there are no plans to procure and install additional uncooled rods, as the flat Inconel-sheathed water cooled rods which were introduced as an interim measure appear adequate for the expected channel distortion for several more years.

Project Engineering

The status of construction projects relating to K reactor facilities is summarized in Appendix A. As formalized, the impact of the KW shutdown on these projects will be covered by appropriate entries in this Appendix.

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Nothing to report.

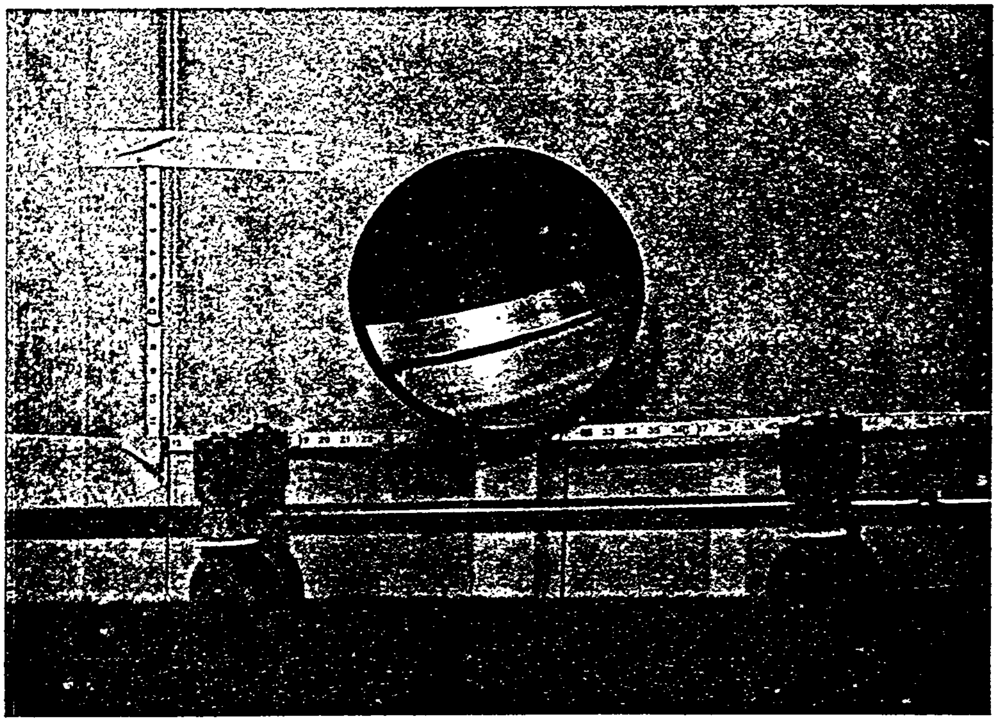

Figure 1. Uncooled HCR - Segment No. 2
(Unirradiated; original shiny condition)

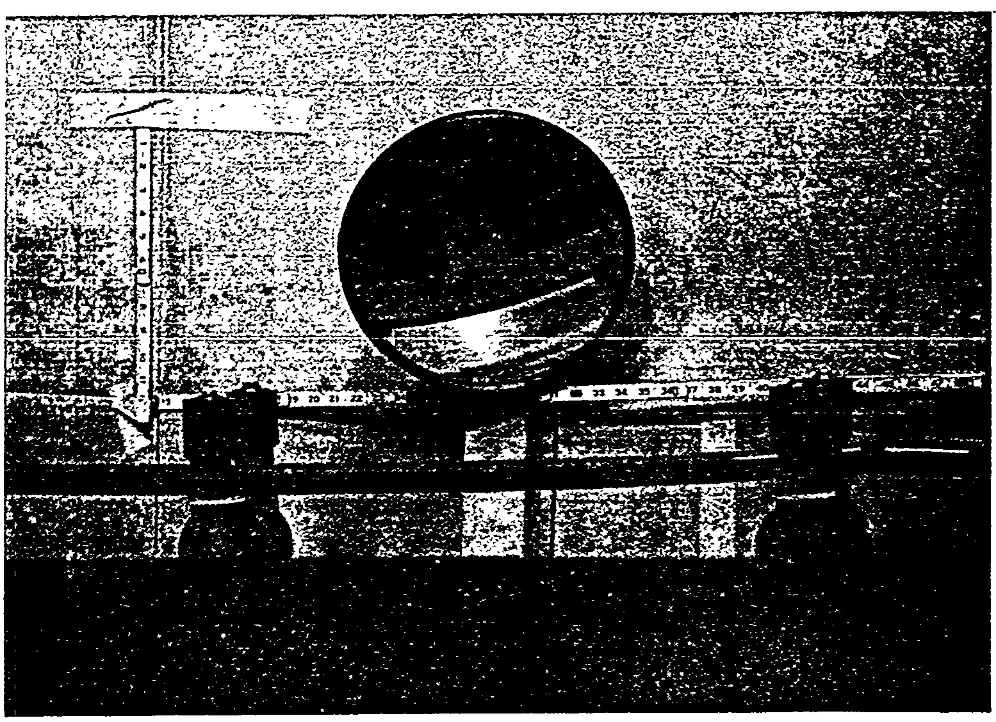

Figure 2. Uncooled HCR - Segment No. 7
(Irradiated; bowed & oxidized)

TECHNICAL ACTIVITIES - N REACTOR

DECLASSIFIED

RESEARCH AND DEVELOPMENT

Basic Production

Yield and Burnout Sampling (PT-NR-95)

Analysis was started by ARHCO on the dissolver samples obtained in recent small batch separations of N Reactor fuel. In addition, three samples have been shipped to BNW for analysis.

Nuclear Protection Systems

CLUMSY kinetics cases using the side-to-side model have been run to better define the effects of out-of-service high level neutron chambers. Front-to-rear and side-to-side cases are also contemplated.

Primary Coolant Control System Simulations

As a check of the models used in the MIMIC simulation of the high pressure injection/pressure control systems, a recent scram was used as a comparison. The primary coolant shrinkage rate for this particular scram was determined from the high speed recordings of pressurizer level and total injection flow. Using this shrinkage rate as an input, and initial conditions corresponding to those observed before the reactor scram, a MIMIC simulation was made. To improve the match between observed scram data and the simulation, two modifications to the simulation model were made as follows:

1. A 5-second time delay was added to the pressurizer level instrumentation and standby high pressure injection pump (HPIP) startup circuits to get a match for the initial increase in HPIP flow.

2. The injection valve flow coefficient was adjusted as required to get a best fit for the maximum injection pump flow rate.

Figures 1, 2, and 3 (on next three pages) show the close agreement between the simulation and observed data for pressurizer level, pressurizer pressure and total injection pump flow, respectively.

Production Tests are being planned to determine the cause of the additional 5-second delay observed in injection flow response.

All-Tube Temperature Monitor System

As a result of favorable response to the report DUN-6632, "Reduction of Exposure and Costs at N Reactor through Replacement of RTDs with Thermocouples", a Production Test is being prepared to assess the performance ade-adequacy of chromel-alumel thermocouples.

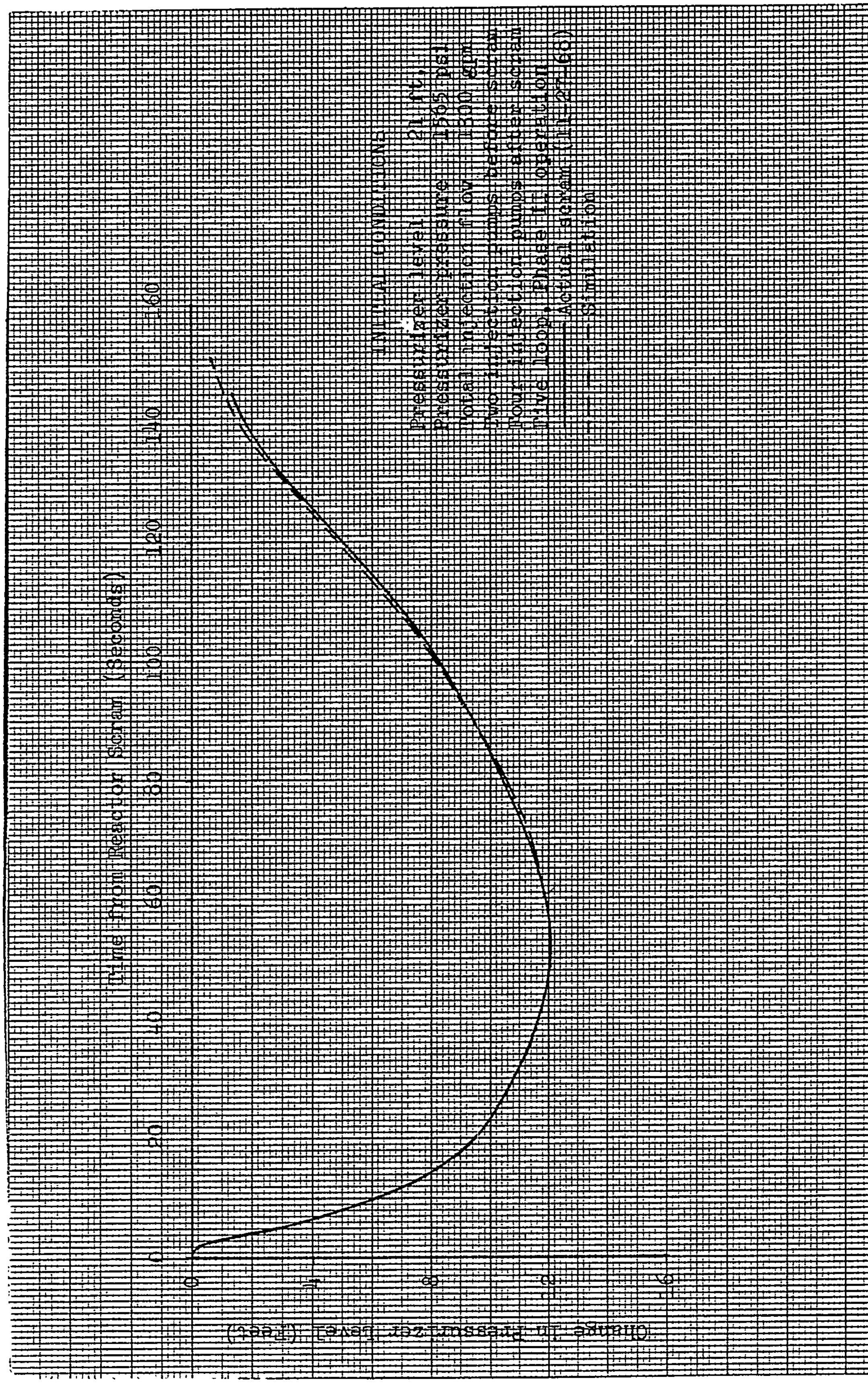

Figure 1. Comparison of Simulation and Observed Scram, Pressurizer Level

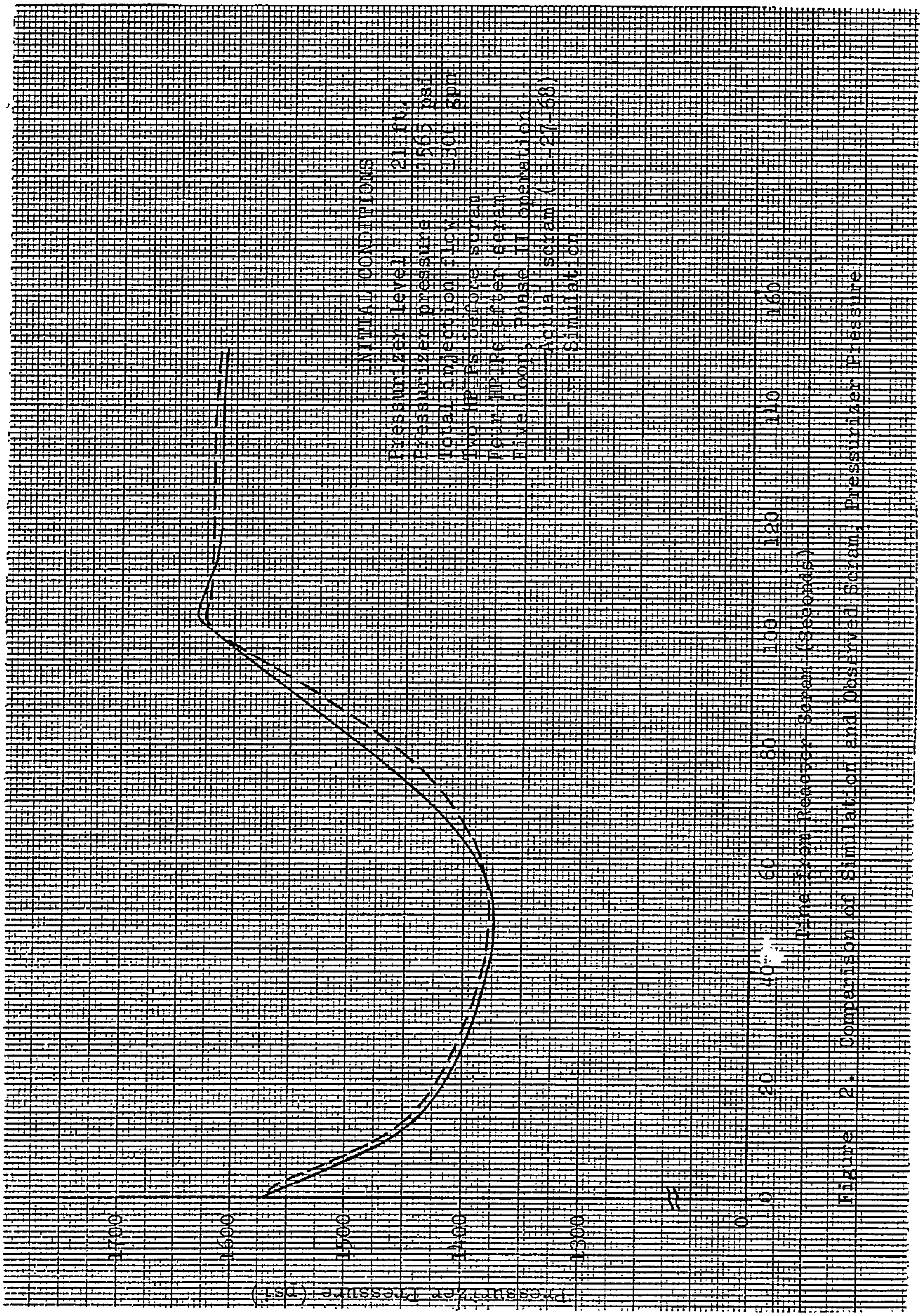

Figure 2. Comparison of Simulation and Observed Scram, Pressurizer Pressure

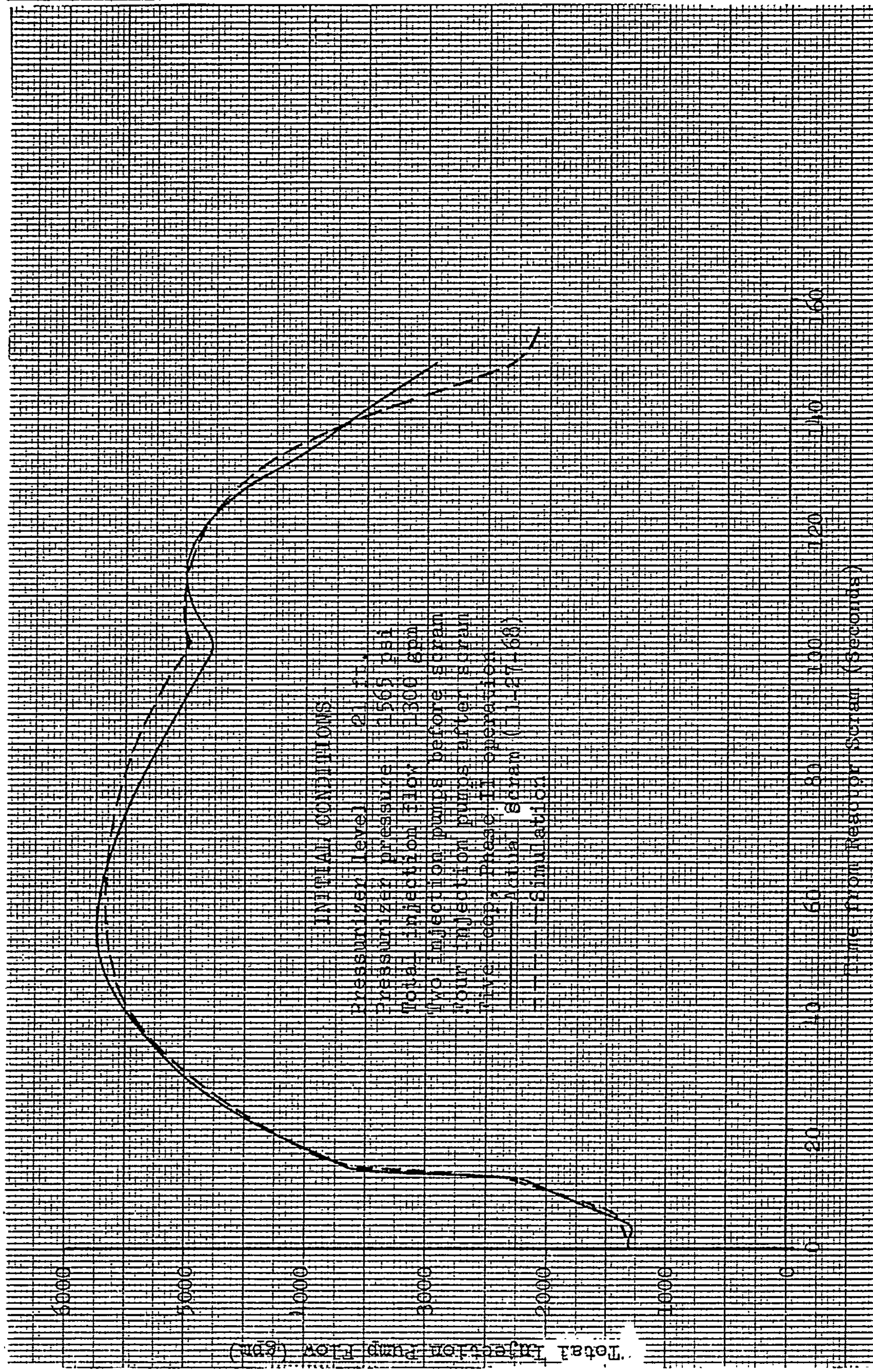

Figure 3. Comparison of Simulation and Observed Scram, Total Injection Pump Flow

Fuel Development - End-Spider Supports

The current status of end-spider support life tests in the 189-D loop facility is as follows:

End-Spider Condition	No. of Elements	No. of Cycles*
As per specification	7	13.1×10^8
1 severed leg - one end	1	9.15×10^8
2 severed legs - one end	1	9.15×10^8
1 drilled weld - both ends	1	2.8×10^8
2 drilled welds - both ends	1	2.8×10^8
Axially deformed - 100 mils	2	1.8×10^8
- 60 mils	2	1.8×10^8

*10×10^8 cycles = 2200 MWD/T in a 4000 kw tube

These tests are all being conducted on 26-inch Mark IV fuel assemblies to develop maximum cyclic loading conditions. Support performance has been acceptable when one support leg was intentionally defected on both ends of an assembly. However, when two legs are severed on both ends, total support failure occurred.

End-spider supports are particularly sensitive to shock loading. Inner fuel displacement during charging was evaluated by ex-reactor impact tests. The results indicated that the relative axial displacement of the inner fuel would be less than 0.010 inch. Normal fabrication and handling procedures also plastically deform the spider support. To assess the effect of prior deformation on the support fatigue life, loop tests have been initiated on four assemblies that were axially deformed first in one direction and then the other as follows:

+X, -2X, +2X, -2X, +X, where X = 50 mils for two assemblies
and X = 30 mils for two assemblies

Production test PT-NR-113, which will authorize the irradiation of five columns of 26-inch Mark IV end-spider supported assemblies, has been drafted and circulated for comments. Fuel fabrication is scheduled to begin in March.

Graphite Distortion

Although the N Reactor graphite has a relatively low contraction rate, it has recently become evident that significant distortion of the stack has occurred. This has manifested itself first as separation and misalignment of the ball channel liners to the extent that safety balls can now escape from a few centrally located channels.

An accelerated program of stack inspection and measurement has been developed, and efforts on means to correct the gaps in the ball channel liners have been initiated.

Inconel-600 Steam Generator Tube Surveillance

An Inconel-600 tube removed from steam generator 6A in November 1969 is being decontaminated prior to nondestructive and metallographic examinations. The purpose of these examinations is to detect any progressive damage to the Inconel-600 steam generator tubes as a result of continued operation A previous examination of two tubes from steam generator 4A showed no damage mechanisms that would reduce the design life of the tube. However, steam generator 4A has been showing indications of a primary-to-secondary side leak; this emphasizes the need for continuing surveillance.

Product Flexibility

Nothing to report.

Environmental and Regulatory Technology

Cesium Ratio Analyses

An earlier disagreement between calculated Cs-137 to Cs-134 ratios and N Reactor meltdown test results is being resolved. Employment of recent resonance integral data for Cs-133 brings the RIBD calculations into reasonable agreement with the test results. Results of fission product data from the analyses currently under way of N Reactor small-batch sampling will be included in the final conclusions, and in the cross section library information used with the RIBD code.

Criticality Code Availability

A working version of the criticality code super-KENO is now available on site for use with the UNIVAC 1108. Adaptation of this code for Hanford use was done by the BNW theoretical physics group with funding provided in part by DUN.

Metal-Water Reaction

The sixteenth and seventeenth metal-water reaction tests with irradiated N Reactor fuel were performed. These tests (SNH-16 and SNH-17) were the fourth and fifth tests at temperatures in excess of the uranium melting point. Significant hydrogen evolution under these conditions was observed. Test SNH-17 showed the highest hydrogen evolution measured to date. The extent of metal-water reaction on these molten uranium tests of outer elements appears strongly influenced by the large sloping molten uranium surface that develops at each end of the element. In an actual fuel column, the physical restrictions from adjacent elements would be expected to reduce this surface area significantly. Methods to improve the simulation of these configurational effects are being investigated.

Waste Management

Nothing to report.

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ENGINEERING AND TECHNOLOGY - REACTOR

Safety Flow Control Leg - CV-2 Stop Setting and V-10 Settings

A search for information concerning the basis for the original CV-2 and V-10 valve settings yielded laboratory data and documentation on the flow characteristics (HW-78524, "Design Test 1152 Flow Calibration Test NPR Safety Flow Control Leg Report of Test Data", August 21, 1963) and analysis of the lab data (HW-76495, "105-N Safety Flow Control Leg Flow Resistance Study", February 6, 1963) for the safety flow control leg. An analysis shows that conservative data were used to determine the final recommendations for the valve settings.

Briefly, the previous analysis combined an analytical analysis with the experimental laboratory data to optimize the hydraulic resistance of the safety flow control leg. To minimize the loss of emergency coolant following an inlet riser rupture, the optimum V-10 setting was determined as 100 percent lift with 80 percent lift adequate. To prevent reverse flow coolant loss in excess of the makeup system capacity following a flush header rupture, the data indicated that the check valve disc limit stop must not exceed two revolutions from the fully closed position, but 1-1/2 revolutions was recommended.

Technical Bases and Process Standards have been issued to provide formal specifications on settings of the valves in the safety control leg. On February 19, N Reactor was shut down to permit inspection of the valves for compliance with these specifications (see report Section BN).

Tube Scratching Problem

While the wear test machine is being completed, materials are being investigated for potential use as fuel support shoes. A sheet of 0.050-inch 17-4 Ph stainless steel has been rolled to about 0.015 inch and H1150 hardened. This treatment results in a hardness and tensile strength about three times that of 1010 steel (311 versus 110 DPH and 145,000 versus 47,000 psi), and an equivalent ductility of 1010 steel. Two strips of this material will be coated with chrome (Microlube MM-11) and one strip will not. Support shoes will be fabricated from all three strips for testing on the wear machine.

Three fuel elements in the end-spider flow test have been fitted with chrome-plated shoes to obtain wear and corrosion data on these shoes.

Control Rod Examinations

Samples of melted portions of the control rod (No. 34) which was severely damaged by excessive heating in January have been sent to Radiometallurgy for further analysis. Examination of the rod channel and subsequent operation has shown that no significant loss of poison from the rod occurred.

Control rod 33 was monitored on closed-circuit TV for evidence of damage caused by debris from rod 34. The only indication that debris may have fallen onto this rod was some tracks in the graphite coating on the rod. These tracks were evident on the portion of the rod extending from about the edge of the reflector (with the rod fully in) to within six to eight feet of the rod tip. There was no damage to the titanium sheath.

Process Tube Examinations

Three process tubes were borescoped during the January outage. Two of these (0867 and 1550) are monitor tubes last examined in early 1968. The rolled joints of both of these tubes were replicated to evaluate the current status of pitting at the joint and as a base for future comparisons. Visual examination did not show any significant changes in the tubes' appearance since the last inspection. Analysis of the rolled joint replicas has not been completed. Tube 0643 also was borescoped because of high charging forces. The tube was found to be undamaged and was returned to service.

A-B Electrical Bus Reliability Analysis

The fault tree models for A bus and B bus have been completed and the criteria for the combined A-B bus fault tree model of the total electrical AC power system at 100-N has been determined. The combined A-B bus fault tree model contains approximately 1700 component failure modes and 1000 failure events which describe system failures at the various power levels. Collection and tabulation of the necessary failure data is under way in preparation for computer simulation of the A-B bus base case fault tree model. Some analyses of variations of the base case are expected and will then be performed and compared to the base case results.

Project Engineering

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

ENGINEERING AND TECHNOLOGY - FUELS AND TARGETS

Nothing to report.

IRRADIATION SERVICES

PRODUCTION OF Xe-128

Trial separation runs using natural xenon and elemental iodine have been performed for the Xe-128 development and production work requested by Argonne National Laboratory. Results of these tests indicate that xenon recovery greater than 99 percent can be attained with product impurities of less than 1 percent. The target capsule containing 450 grams of iodine has been discharged from the reactor and transferred to the storage basin. Final preparations are now being completed to perform the separation and recovery run using the irradiated target capsule. About 150 ml of Xe-128 is expected from this source.

ISOTOPE PRODUCTION

Hafnium-181 for Portland State University

A small sample of hafnium oxide was irradiated in a Quickie facility to produce 16 millicuries of hafnium-181.

MATERIALS TESTING

The second in a series of uranium oxide creep rate tests was run in a Snout facility.

ROUTINE IRRADIATIONS

The last two of a series of boron carbide capsules were discharged from a KW Reactor general purpose facility. Capsule number two had reached goal exposure, but capsule number three was discharged early due to the KW Reactor shutdown.

Three cold tensile specimen capsules were removed from the KW Reactor Snout and shipped to the customer for analysis. The remaining six capsules will be recharged in KE Reactor.

One hundred sixty-nine activation analysis capsules were irradiated in the K reactor Quickie facilities.

ADMINISTRATION - GENERAL

BUDGET PRESENTATIONS TO F. P. BARANOWSKI

FY-1970 and FY-1971 budget information was presented to F. P. Baranowski on February 10. The presentation included estimated costs, tons of finished fuel production, and personnel levels. The FY-1971 budget was based on a funding level of $35.5 million. Alternate cases were developed in which reactor TOE and product assay varied.

REACTOR SUPPORT PERSONNEL

At the request of AEC-HQ, a study was made in October 1969 of operation and engineering support personnel requirements for equilibrium dual-purpose operation of N Reactor. During his recent visit, Mr. Baranowski requested estimates of other support personnel requirements in dual-purpose operation, with KE Reactor also operating. These estimates, which included the transition period from January 1, 1971 to July 1, 1973, were forwarded to AEC-RL on February 16.

KW REACTOR STARTUP COST ESTIMATE

A cost estimate, based on the assumptions that (1) KW was in equilibrium standby condition and (2) 18 months were allowed to complete startup activities, was prepared at the request of AEC-HQ for inclusion in planning estimates.

EMPLOYMENT

Reduction of Force

The force reduction is proceeding on schedule. As shown in Appendix B, total employment dropped from 1675 on January 31 to 1595 on February 28.

Plans for Progress - EEO

The Company's progress in employment of minorities is being severely set back as the result of budget cuts and force reductions. It is expected that by June 30 about 45% of the minority employees in DUN will have been lost.

UNION RELATIONS

A dispute concerning the seniority status of previously promoted employees who are returned to the unit has been presented to arbitration and a decision is expected on March 10, 1970. Another dispute concerning pro-rata vacation entitlement of laid-off employees is now being scheduled for arbitration.

SAFETY

No personnel radiation exposures exceeded operational control.

Month-end safety statistics were:

Disabling injuries - February	1
- CY to date	1
Days since last disabling injury	16
Man-hours since last disabling injury	152,000

The disabling injury was incurred on February 12 when an N Plant pipefitter suffered a loss of bone from the tip of his left ring finger while operating a magnetic drill.

APPENDIX A

PROJECT STATUS SUMMARY - REACTOR FACILITIES

AUTHORIZED PROJECTS

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
Single-Pass Reactors				
DCE-505, Boiler Control Improvements - 165-KE & KW	410,000	100	98	Acceptance testing of the No. 1 KE facility was started February 10. Project proposal revision requesting extension of the project completion date to April 1 was submitted to AEC-RL for approval February 9.
DAP-510, Discharge Chute Clearing Equipment - K Reactors	220,000	100	100	Construction was completed February 18. Minor underwater testing remains to be done. KW work was removed from the project scope.
DAP-516, Storage Building Addition - 105-KE & KW	142,000	100	0	Awaiting AEC-RL decision to proceed with the KE portion of the work.
DAE-518, Effluent Radio-iodine Monitor - KE & KW Reactors	100,000	100	100	Construction was completed January 30. Acceptance testing has been delayed pending availability of normal effluent from KE Reactor.
DCP-526, Deactivation of Hanford Production Reactor C	75,000	97	44	No change in status. Awaiting release of additional funds by AEC-RL.
DCP-535, Sodium Sulfite System - 182-K Building	25,300	100	75	Directive was issued February 11. This work was previously funded by AR-B99656.

Number & Title	Authorized Funds - $	Percent Complete Design	Percent Complete Construction	Remarks
N Reactor				
GCP-406, Improved Safety Platforms and Accesses - 100-N Area	300,000	100	79	1310-N circular stairway installed February 12.
GCP-411 Effluent Control Program - 100-N Area	1,830,000	100	65	Section II - Charpy tests on wrapped 30" pipe in process by vendor. Section IV - Held in abeyance pending approval of increased funding requested in Rev. 2 to the project proposal. Section V - Title III construction tests of tank in progress.
DCE-519, Replacement of Bridge Crane and Hoist System with New Crane System - 105-N Storage Basin Area	465,000	96	0	The directive increasing authorized funds to $465,000 was issued January 28. The purchase order contract for crane systems was mailed to Crane Hoist Engineering February 5.
DCP-528, Fire Protection System Improvements - 100-N	40,000	2	0	A planning meeting to initiate design was held with Vitro/HES February 5.

PROJECT PROPOSALS AWAITING AUTHORIZATION

Number & Title	Funds Requested	Date to AEC-RL
GCP-411, Rev. 2 - Effluent Control Program - 100-N	$2,010,000 (new total)	9/24/69
DCE-505, Rev. 3 - Boiler Control Improvements - 165-KE & KW	None additional	2/9/70
DCP-525, Fire Protection Improvements - KE	225,000	5/2/69

Number & Title	Funds Requested	Date to AEC-RL
DCP-527, Graphite Cooling & Fog Spray - N	97,000	5/9/69
DCP-528, Fire Protection System Improvements - 100-N	290,000	6/4/69
DCP-529, Gravity Drainage System and Disposal Basin - 100-N	200,000	7/10/69
DAP-530, Upgraded Electrical Services and Lighting 1100-N and 1101-N Buildings	78,000	10/2/69
DAP-531, Establish 1102-N Office Building - 100-N Area	45,000	8/19/69
DCE-532, Isolation of Process Coolant Risers - K Reactors	270,000*	9/25/69

*Additionally, this project requests the use of $240,000 of operating funds.

PROJECT PROPOSAL PREPARATION

Number & Title	Design Criteria	Project Proposal
DCP-527, Rev. 1 - Graphite Cooling & Fog Spray - N	Completed	Completed - in approval status.
DCP-528, Rev. 1 - Fire Protection System Improvements - 100-N	Completed	Completed - in approval status.
Export Water System Backup - 182-D (for 200 Areas)	Completed for Design Council approval.	Comment draft issued.
Stack Monitoring Improvements - 100-N Plant	Revision completed for Design Council approval.	Awaiting approval of revised design criteria.

Number & Title	Design Criteria	Project Proposal
Upgrade Flux Monitor System - N Reactor	Engineering study is in progress.	
High Pressure Injection and Seal Water Improvements - 109-N	Preliminary engineering study completed.	

APPENDIX B

EMPLOYMENT SUMMARY
(with employee allocations)

	1/31/70	2/28/70
CONTRACT PERSONNEL		
02 Programs		
Douglas United Nuclear	1594	1515
Assisting Other Contractors	13	15
Total - 02	1607	1530
Other Programs Under AEC Contract		
Assisting Other Contractors and WPPSS	44	35
Special Irradiations	8	8
Other Programs - Standards		3
Total - Other Programs	52	46
Total Contract Personnel	1659	1576
COMMERCIAL ACTIVITIES PERSONNEL	16	19
TOTAL FORCE	1675	1595

FEATURE REPORT

RELIABILITY ANALYSIS OF N REACTOR'S FOG SPRAY SUBSYSTEM

INTRODUCTION

Performance reliability of the fog spray subsystem, an important element of N Reactor's engineered safeguard systems, was evaluated from both nuclear safety and operability standpoints. Since the design of safeguard system elements is primarily based on nuclear safety requirements, it follows that reliability criteria from this standpoint are significantly more important than those developed from operability requirements. However, to provide assurance that the fog spray subsystem meets nuclear safety requirements without seriously degrading operability, two separate but closely coordinated analytical efforts were conducted.

The nuclear safety analysis employed the fault tree analytical technique to identify reliability limitations and provide comparative bases for subsystem reliability improvements. This objective was established within the overall licensability program to evaluate existing critical systems for the integrity and redundancy required for coolant supply continuity. The operability analysis used the fault tree technique in conjunction with other more qualitative analyses to rank potential reliability problems according to their likelihood of degrading operability. This objective was established within the overall plant improvement program to improve the operating efficiency of N Reactor. Careful integration of both analyses enabled common use of data and minimized duplication of effort.

This summary report describes the fog spray subsystem, presents a ranking of critical fog spray failure modes for various configuration and operational alternatives, and illustrates the magnitude of subsystem reliability improvements available for each alternative considered. These analyses typify the effort currently in progress to evaluate the reliability of critical N Reactor systems.

FOG SPRAY SUBSYSTEM DESCRIPTION

The fog spray subsystem, shown schematically in appended Figure 1, is designed to spray fine droplets of water within the Zone I area of the 105-N and 109-N Buildings upon demand. This spray will (1) condense high pressure steam emanating from a primary loop pipe rupture, and (2) scrub fission products emanating from the pipe break should fuel jacket melting occur. The subsystem is designed to supply water on demand for extended periods of time in response to certain reactor incidents. Figure 2, appended, illustrates the two types of fog spray nozzles used; Figure 3 shows a demonstration test of the 2" Spraco nozzle at 116 gpm and 57 psi.

The 105-N fog spray consists of nine fog spray stations located over the reactor and the inlet and outlet pipe spaces. One of these stations is shown in appended Figure 4. Water is released to each station when 105-N Zone I pressure exceeds a preset level. The main control is by diaphragm-operated butterfly valves held closed by compressed air from the building compressed air system and DC powered solenoid control valves. The 105-N fog spray is designed so that upon loss of either (1) compressed air supply, or (2) DC power, all fog spray valves open.

The 109-N fog spray consists of seven fog spray stations, one in each of the steam generator cells, and one covering the pipe gallery. Water is released to the stations in a manner similar to the 105-N stations. The 109-N fog spray is designed so that upon loss of either compressed air supply or DC power, all fog spray valves open. Current operating practices allow the cell fog spray stations to be removed from the automatic trip circuitry, although they may be opened manually from the 105-N Control Room. The pipe gallery station is the only 109-N station on automatic control.

Water supply for the fog spray subsystem is high pressure raw water supplied by two diesel-driven fog spray pumps (see appended Figure 4) and three electric motor-driven HPRW pumps. The three motor-driven HPRW pumps are supplied by the circulating raw water system, and the two fog spray diesel-driven pumps can be supplied by either the circulating raw water system or the emergency raw water system. The HPRW pump motors are supplied by A and B 480-V buses. The fog spray diesel engines are started by 600 psi compressed air upon fog spray accumulator low level or low pressure signals. Both sets of pumps supply the high pressure raw water system (RWS-2) which supplies water to the rupture monitor system, graphite loop once-through backup, rod and shield backup systems, 105-N fire protection system, etc., in addition to the fog spray.

NUCLEAR SAFETY ANALYSIS

Failure Definition

The nuclear safety analysis of the fog spray subsystem is based on conditions that lead to insufficient fog spray protection under incident situations. Out-of-service or unavailable fog spray station combinations that result in insufficient fog spray capacity are:

- The 109-N pipe gallery fog spray station fails.
- 105-N station No. 2 and any other 105-N stations fail.
- Any three 105-N stations (except 105-N Station No. 2) fail.

The failure of the fog spray subsystem is described by any of the above three conditions. Figure 5, appended, is a fault tree which illustrates these top failure conditions; for example purposes, it describes the failure combinations that will result in the failure of 105-N station No. 2.

Of equal operational importance is the fog spray water supply system. Fog spray water supply system supply and demand curves were used to determine pump combinations that would adequately supply water to the fog spray subsystem. Under the assumption that no emergency users (graphite loop backup, rod and shield backup, etc.) were drawing from the RWS-2 supply, one HPRW pump and one fog spray diesel-driven pump are required as a minimum to adequately supply the 109-N and 105-N fog spray stations on automatic control.

In the event of simultaneous loss of both A and B electrical buses, the emergency users would also be drawing from the RWS-2 supply. In this situation, both fog spray diesel-driven pumps are required to handle the additional flow requirement. Thus, total fog spray pumping requirements can be met by either both fog spray diesel pumps or one diesel pump and one HPRW pump for A and B bus loss. Fog spray subsystem failure occurs (due to insufficient pumping capacity) should either of the following conditions occur:

- Both diesel-driven pumps fail.
- One Diesel-driven pump fails and all three HPRW pumps fail.

Although functional testing of the fog spray diesel pumps is required on a semiannual basis, credit is given in this analysis to monthly diesel tests, a current operational practice that is assumed to continue.

The fog spray subsystem is part of the confinement system, which is considered Class I. Under this classification, all components of the system are to be designed to meet a 0.2 g seismic criterion. However, certain components are made of cast iron and were not specifically designed to this criterion. In an attempt to weigh the importance of this suspected deficiency, this analysis assumes that the cast iron components are more susceptible to rupture by a factor of 10. The effect of this assumption will be shown in the detailed case comparisons that follow.

Nuclear Safety Analysis Conclusions

Five comparative case analyses of the N Reactor fog spray subsystem were made to evaluate the magnitude of reliability improvement possible if certain changes were made to subsystem components, Process Standards, and test frequencies. These cases were:

Case I - The existing fog spray subsystem before completion of Project GCP-411 (Effluent Control Project).

Case II - Case I after completion of the Effluent Control Project.

Case III - Case II under a modified Process Standard equating the 109-N pipe gallery station with any of the 105-N stations except No. 2.

Case IV - Case III with improved pump discharge butterfly and check valves.

Case V - Case IV with monthly testing of all station control valves.

The results of these case analyses are shown in Table 1, below. The probability of failure is the probability that the system will be in a failed state, as previously defined, before the end of a one calendar year period of operation. The factor of improvement is the ratio of the probability of failure for Case I to the probability of failure of the represented case. The mean downtime is the average time required, for each system failure, to detect that system failure has occurred and to restore the system to an unfailed state.

TABLE 1

Fog Spray Analysis Results

Case	Probability of Failure[a]	Mean Downtime (hours)	Reliability[b]	Factor of Improvement[c]
I	0.491	1474	0.509	1.00
II	0.426	1949	0.574	1.15
III	0.335	320	0.665	1.47
IV	0.253	422	0.747	1.94
V	0.0726	261	0.9274	6.77

[a] Based on a one year (8760 hours) period of operation
[b] Reliability = 1.0 minus probability of failure.
[c] Probability of failure (Case I)/probability of failure (case represented).

The details of each case presented in the next section include many improvements that are already in various stages of completion. Other improvements suggested have been considered primarily from the reliability viewpoint and will not be acted upon until other related parameters have also been evaluated. When appropriate studies on the entire confinement system are complete, the significance of fog spray reliability improvements will be established in proper system perspective for further management action.

Detailed Case Comparison

In each of the five cases discussed in the following paragraphs, only the five most critical failure modes are listed. For each failure mode, the percent contribution of that failure mode to subsystem failures is shown. This figure, obtained from fault tree computer simulation runs, is a relative statistic for the particular case being evaluated and is presented to place subsystem failure modes in perspective. Input data used to exercise the fault tree computer simulation model were obtained from on-plant experience, when such data existed in a statistically meaningful form, and from similar industrial applications otherwise.

Case I

This base case considers the existing fog spray subsystem before modifications made in the Effluent Control Project are completed. The functional requirements for the subsystem are stated in the N Confinement System Process Standard B-290. Test frequencies are:

- Fog spray diesel pumps - monthly starts.
- Integrated subsystem test - annual test required by Equipment Maintenance Standard (EMS) No. 13.1.
- Zone I pressure sensors - semiannual calibration check required by EMS No. 13.1.

Table 2, below, ranks the five major component contributors to subsystem failure for Case I in terms of percent. This ranking shows that the 109-N pipe gallery station components are more critical to subsystem performance than like components of the 105-N stations, due to a greater importance placed on the operation of the pipe gallery station than on any other subsystem station.

The failure of the pipe gallery station leads to an immediate subsystem failure, whereas coincident 105-N station failures are required for the same event. The conclusion to be drawn from this ranking is that if subsystem improvement is required, special consideration should be given to the pipe gallery station (special testing procedures, redundant control, ultra-high reliability components, etc.), or operational requirements should be relaxed.

TABLE 2

Case I - Critical Component Ranking

Failure Mode	Percent of Subsystem Failures
Rupture of HPRW and diesel pump discharge butterfly valves and diesel discharge check valves	18.8
Loss of starting air to fog spray diesels	11.0
Rupture of fog spray accumulator	8.0
109-N flow controller failure	4.8
109-N flow control butterfly valve (RWV-208-1) fails closed	4.8

The assumption of increased seismic rupture potential for the pump discharge check and butterfly valve bodies is clearly evident when compared to other subsystem components (not shown) in this ranking. If this assumption is valid, the valves seriously compromise subsystem reliability. Rupture of any of these valves would cause loss of water supply to not only the fog spray subsystem, but also backup supply to systems such as the graphite coolant and rod and shield coolant systems.

Two of the prime reliability offenders in the existing subsystem are the starting air to the fog spray diesels and the fog spray accumulator. Completion of the Effluent Control Project will eliminate these contributors (as is discussed in Case II).

Case II

This case considers the effects of modifications made to the fog spray subsystem by the Effluent Control Project. Significant reliability improvements to two of the top five Case I critical components, diesel starting air and the fog spray accumulator, are enumerated below:

- Isolation of the starting air supply to the fog spray diesels. This eliminates the previous interdependency of the two diesels on the starting air supply by providing check valve isolation for each diesel starting air accumulator.

- Removal of the fog spray accumulator from service. Diesel start would then be initiated by low RWS-2 header pressure (redundant pressure switches) rather than by accumulator low level or pressure.

With the exception of these two modifications, Case II is identical to Case I in testing frequencies and component failure rates. Table 3 ranks the five major contributors to subsystem failure for Case II.

TABLE 3

Case II - Critical Component Ranking

Failure Mode	Percent of Subsystem Failures
Rupture of HPRW and diesel pump discharge butterfly valves and diesel discharge check valves	23.2
109-N pipe gallery station globe valve fails closed	8.0
109-N pipe gallery station piston operator failure	7.2
109-N flow control butterfly valve (RWV-208-1) fails closed	5.2
Either 109-N pipe gallery station control valve stuck	4.6

This ranking of the five critical components of Case II follows a trend similar to that of Case I, with the components of the 109-N pipe gallery station and the 105-N station No. 2 being more critical to system performance than the same components of the other 105-N stations.

Case III

Critical component rankings from Cases I and II clearly show the criticality of the pipe gallery fog spray station components to subsystem reliability. In a protective system such as the fog spray subsystem, where reliability is of prime importance, multiplicity or redundancy is highly desirable since then single isolated failures will not cause subsystem failure. The 105-N fog spray is fairly successful in meeting such a requirement since the loss of any single station does not result in failure. However, the 109-N fog spray does not meet this requirement since failure of the pipe gallery station causes fog spray subsystem failure, as defined by Process Standard B-290.

If an analysis could show that the pipe gallery station were no more important to the system safety functions than any of the 105-N stations, excluding station No. 2 due to its prime area of coverage, the Process Standard could be relaxed. Thus, by definition, an element of redundancy could be added with a corresponding increase in subsystem reliability.

In such a situation, inadequate station coverage would result if either of the following situations occurred:

- 105-N station No. 2 and any other 105-N station (including the 109-N pipe gallery station) fail.
- Any three 105-N stations (including the 109-N pipe gallery station, but excluding the 105-N station No. 2) fail.

Case III considers the effect of using this modified subsystem criterion on subsystem reliability. All other conditions of Case II (component failure rates, testing frequencies, Effluent Control Project modifications, etc.) are retained in Case III. Table 4 ranks the five major contributors to subsystem failure in Case III.

TABLE 4

Case III - Critical Component Ranking

Failure Mode	Percent of Subsystem Failures
105-N station No. 2 fog spray butterfly valve	5.4
Any HPRW pump or diesel pump discharge butterfly valve rupture	4.6
105-N station No. 2 diaphragm operator failure	4.0
Either diesel discharge check valve ruptures	3.5
105-N station No. 2 manual gate or butterfly valve closed	3.4

Case IV

Critical component rankings for Case III show that the integrity of the HPRW pump discharge butterfly valve bodies and the fog spray diesel pump discharge butterfly and check valve bodies is critical to reliability. The sensitivity of valve integrity is due to the seismic deficiency assumption previously stated. Case IV is identical to Case III with the exception that the HPRW pump discharge butterfly valves and the fog spray diesel pump discharge butterfly and check valves are of integrity equal to that of the RWS-2 piping.

Critical component rankings for Case IV are the same as those of Case III except that the percent contribution of the discharge butterfly valves to subsystem failure drops from 4.6 percent per valve (Case III) to 0.6 percent per valve (Case IV), while the diesel pump discharge check valves drop from 3.5 percent per valve (Case III) to 0.3 percent per valve (Case IV). Overall reliability is improved by a factor of 1.33 over Case III.

Case V

Case V is identical to Case IV, but with the testing interval of the 105-N and 109-N station valves lowered from one year to one month. By increasing component test frequency, greater confidence in the operational status of the component is gained. Components in the fog spray subsystem are subject to failure while in a standby state and unless such failures can be detected and repaired, the subsystem's probability of functioning properly if required would decrease with time. Under these conditions, subsystem reliability is a function of test frequency as well as time. Thus, by increasing the frequency of testing, the subsystem reliability is increased although the relative magnitude of the increase diminishes as test frequency increases. Another contingency to the benefits of increased test frequency is that such increases should not significantly increase the likelihood of test-induced failures.

The modified test procedure used in this analysis involves exercising all station control valves from the 105-N Control Room. This testing sequence is being considered for inclusion in the Equipment Maintenance Standards. It would be performed on a monthly basis during an outage or while operating (the 105-N stations can be tested individually during operation without violating Process Standards).

Table 5 ranks the five major contributors to subsystem failure for Case V.

TABLE 5

Case V - Critical Component Ranking

Failure Mode	Percent of Subsystem Failures
Failure of either fog spray diesel engine	19.5

Failure Mode	Percent of Subsystem Failures
Failure of either diesel-driven pump	7.5
105-N station No. 2 fog spray butterfly valve fails closed	6.3
109-N flow controller failure	4.8
105-N station No. 2 diaphragm operator failure	4.6

This ranking departs significantly from Case IV with the diesel engines and diesel-driven pumps comprising the major limitations to subsystem performance. Components of 105-N station No. 2 remain more significant than components of other stations due to the performance requirements on that station. 109-N flow control components are significant because they remain on an annual test frequency. When considering the ranking of station components, it must be realized that the subsystem failure potential of these items is approaching that of the RWS-2 piping integrity, considered the limiting factor on system performance.

The fog spray diesels, though ranked top on the critical component listing, have demonstrated a reliability which may be as good as can ever be expected from engines of this type. An alternative approach therefore may be the installation of a third diesel-driven pump. This may also be justified by the fact that two diesels are required to adequately supply water to the fog spray subsystem when emergency users demand water; in such an event, two diesel-driven pumps offer no pumping capacity redundancy.

OPERABILITY ANALYSIS

Failure Definition

The fog spray subsystem function is not required for day-to-day plant operations. Capability of the fog spray system to function as required in the event of a primary coolant system pipe break is verified in accordance with previously discussed Process Standards requirements. From the operability standpoint, reactor operations are affected by fog spray capability to the extent that this capability can be measured through inspection and tests. Since it is not practicable to completely test the entire subsystem, many of the failure modes which might degrade fog spray performance during an accident condition have no effect on operability because the existence of these failure modes cannot be determined.

Failure of the fog spray subsystem is defined from the operability standpoint in terms of those failure modes which, upon discovery, force a reactor shutdown in accordance with Process Standard B-290. Thus, the reliability of the fog spray subsystem from the operability standpoint carries a significantly different connotation than reliability in the nuclear safety sense.

Operability Analysis Conclusions

The consequences of determinable (through test and inspection) failures upon reactor availability and day-to-day operating reliability were evaluated. The following conclusions are based on a review of over 500 subsystem elements for functional need and performance adequacy, operability critical failure modes and mechanisms, and criticality ranking:

- The potential of the fog spray subsystem to degrade operability appears to be minimal relative to other plant systems, although this judgment cannot be completely supported now by quantitative analyses. Operability critical elements are ranked in order of their relative contribution to subsystem failure:

Element	Percent of Subsystem Failures
Fog spray diesel engine assembly	34.5
Diesel-driven pump	10.6
109-N pressure switches	4.8
105-N pressure switches	4.6
Diesel-driven gear increasers	4.5
109-N solenoids	4.2
Diesel-driven pump butterfly valves	2.2
Electric motor-driven pump butterfly valves	2.1
Diesel-driven pump expansion joint	1.9
105-N solenoid control valves	1.8

- The most critical failure mechanisms from the operability standpoint are brittle fracture and DC electrical grounds. Plans for improving many of the brittle-fracture prone components are being formulated while corrective effort on DC grounds continues.

- Rigorous application of the fog spray Equipment Maintenance Standards will improve the adequacy of preventive and control measures in mitigating the effects of operability critical failure modes and mechanisms.

FUTURE WORK

The fog spray reliability analyses reported here are part of an overall confinement system nuclear safety and operability improvement effort. Although some of the reliability improvements identified in this summary report are already in various stages of completion, many of them will be considered in conjunction with reliability improvements identified for the remainder of the confinement system prior to initiating further improvement efforts. Comparative cases are now being evaluated for the purpose of determining which reliability improvements will best ensure the adequacy of the total confinement system to reliably perform its function.

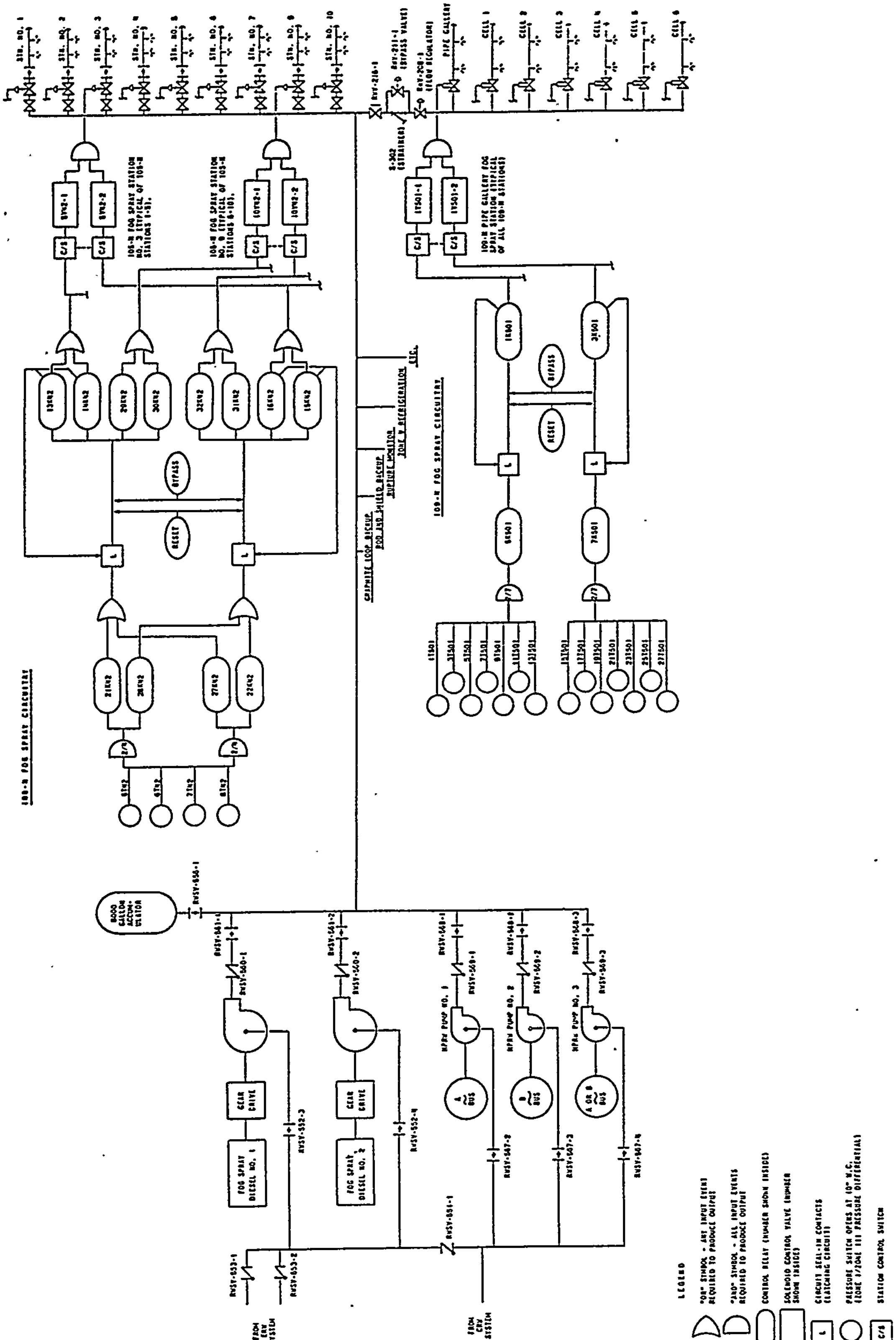
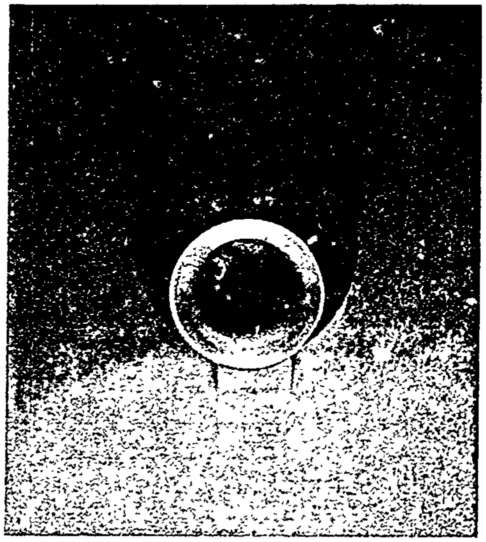
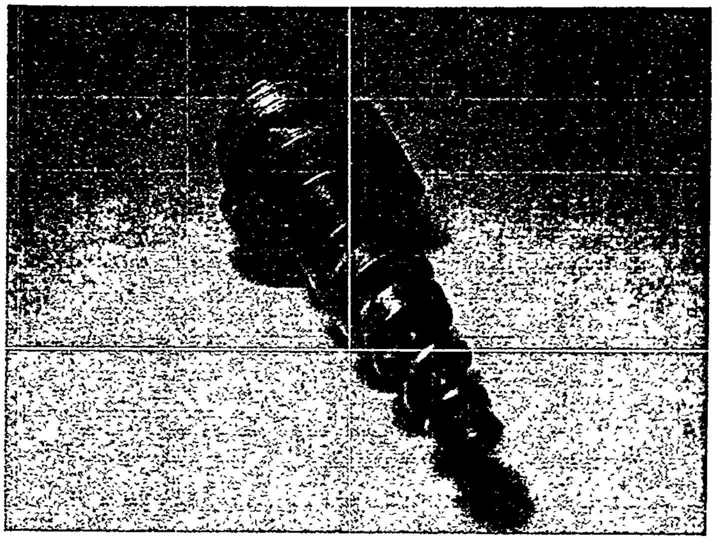


Figure 1. Fog Spray Subsystem - Functional Schematic



2" Spraco Nozzle (as used in 109-N)



Bete Nozzle (as used in 105-N)

Figure 2. Fog Spray Nozzles

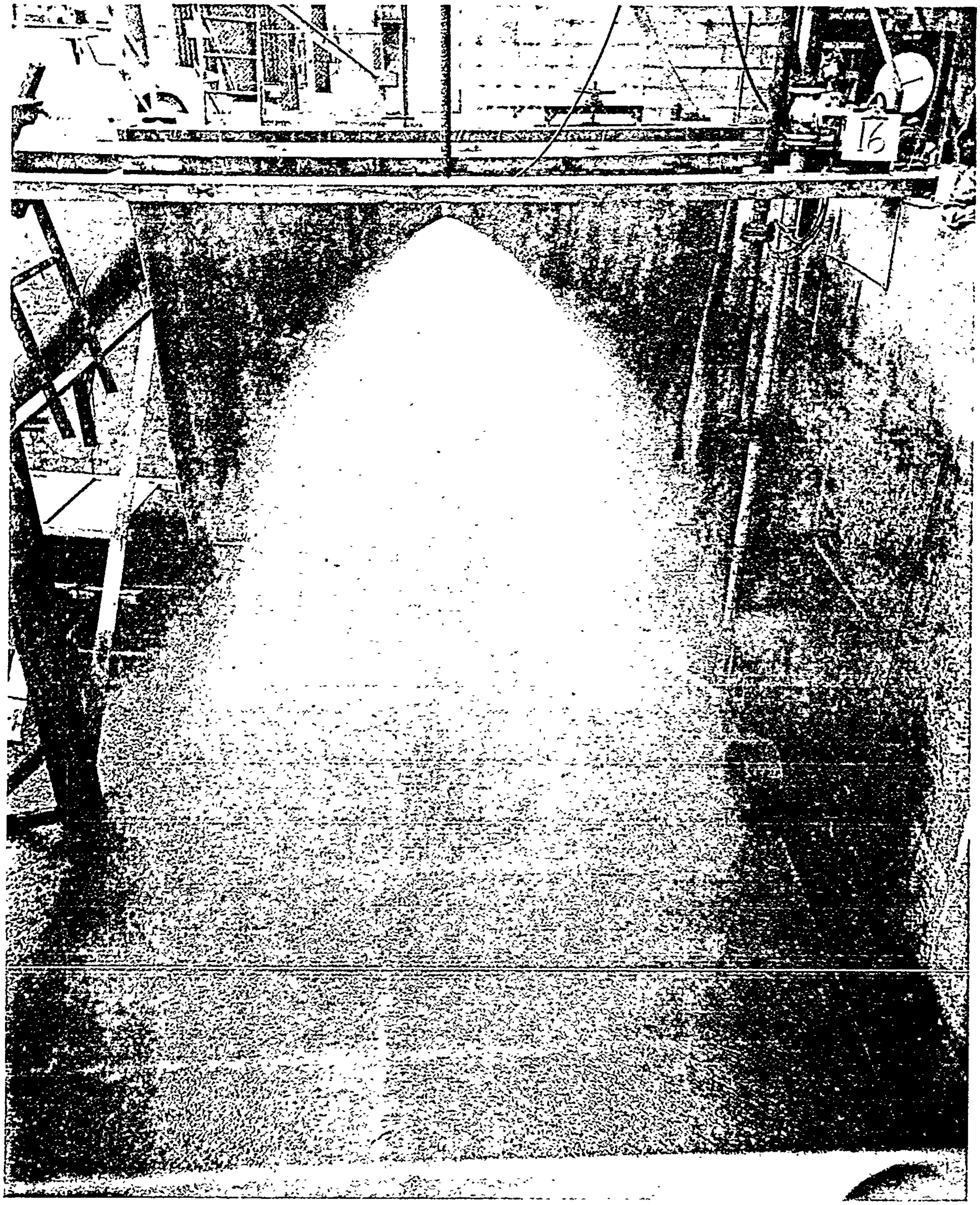
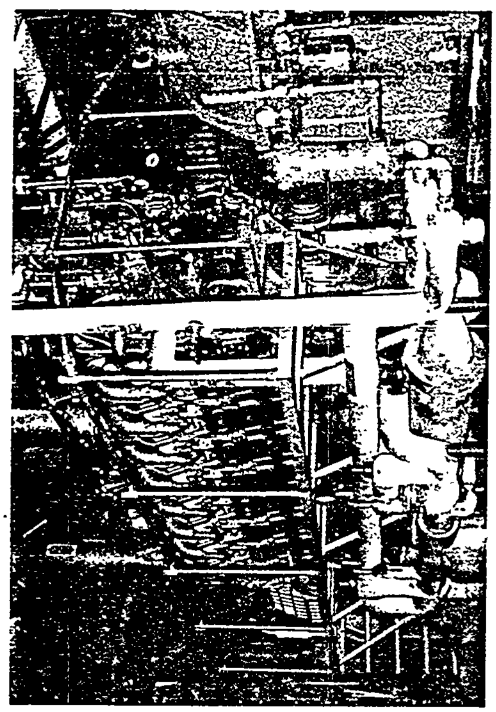
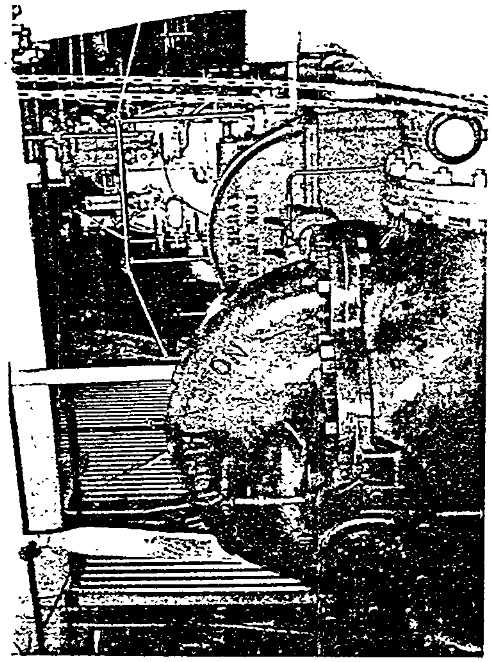


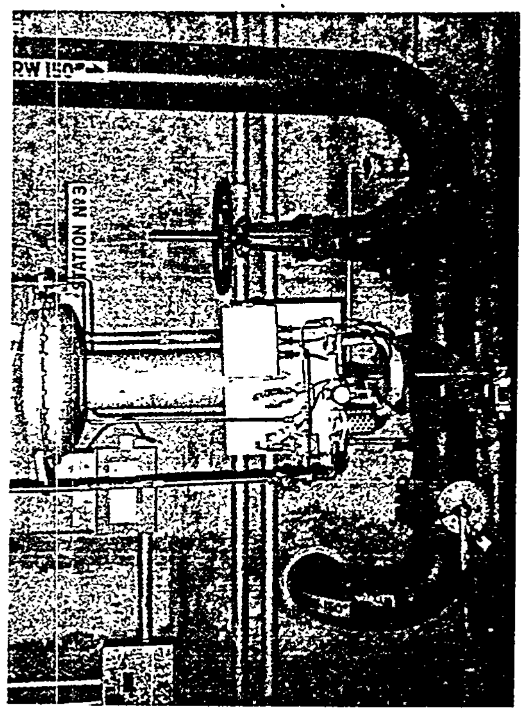
Figure 3. Fog Spray Test
(2" nozzle, 57 psi, 116 gpm)



Diesel Engine No. 1



Diesel Pump No. 1



105-N Control Station (Typical)

Figure 4. Fog Spray Subsystem Elements

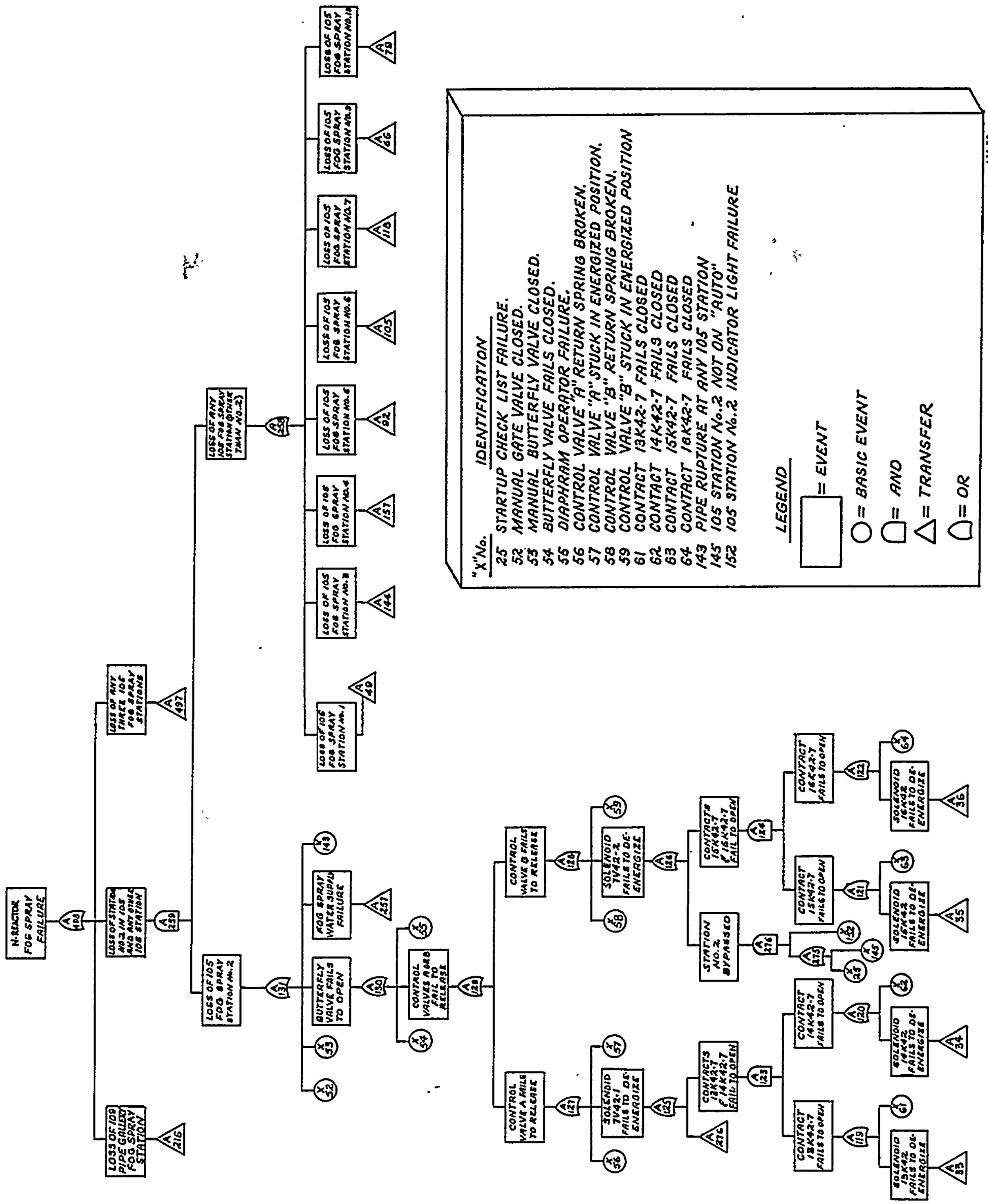


Figure 5. Partial Fog Spray Fault Tree