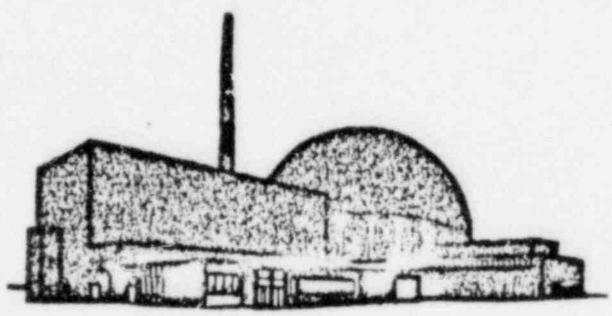
ANNUAL REPORT

YEAR 1968



DRESDEN NUCLEAR POWER STATION

Commonwealth Edison Company

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION

ANNUAL REPORT OF STATION OPERATION
FOR THE YEAR 1968

January 20, 1969

DRESDEN NUCLEAR POWER STATION ANNUAL REPORT

I. INTRODUCTION

This seventh annual report is submitted in compliance with paragraph 3.C (2) of the Utilization Facility License DPR-2, as amended, and covers operation of Dresden Nuclear Power Station during the year 1968.

II. SUMMARY OF OPERATIONS

A. Scope of Operations

Dresden Station operated during 1968 with a total of 15 shutdowns. Included in this total was the fifth partial refueling, inspection and maintenance outage, which extended from February 3, until June 2, 1968. This outage was quite extensive and included items such as: Overhaul of the main turbine-generator; removal of turbine crossover grating; refueling of 96 fuel assemblies and cleaning of all fuel returning to the core for Cycle 6 operation; overhaul of 40 control rod drives; inspection and testing of major primary system welds; and general maintenance and inspections made available by the shutdown.

During the year, additions to and changes in facility design were made by: Addition of a steam sample station on "E" turbine extraction line; modification of the "trip" circuit on the reactor canal crane; addition of a new radiation monitor in the fuel building; installation of a materials test loop in "B" secondary steam generator compartment; revision of the mechanical vacuum pump six inch discharge line; erection of control rod drive thimble support fixtures; modifications to No. 1 deepwell pump; addition of mechanical stops to the off gas monitor range switch; erection of an on-site 400 foot meteorological tower; reactivation of Hansel and Breen Environs Stations; and the addition of tie connections between Units #1, #2, and #3 service air and water systems.

A total of eight shipments, consisting of 184 spent fuel assemblies, was shipped to the Chemical Processing Plant of Nuclear Fuel Services, Inc., at West Valley, New York during the period.

B. Shutdowns

The plant was shutdown 15 times during 1968 as shown in Table 1 and Figure 1. Twelve of these were forced outages, all of which were due to turbine condenser tube leak repairs. There were three scheduled outages: Two for operator training and one for the major inspection, refueling and maintenance outage.

Three of the 12 forced outages were temporarily extended during the year. The first of these occurred on October 30, for repairs to "B" clean-up demineralizer pump; the second on November 10, for operator training; and the last, on December 26, for a leak check in "A" unloading heat exchanger oil cooler.

C. Load Restrictions

The load restrictions imposed during the year are listed in Table 2. Restrictions were due to reactor recirculation loop outages, feedwater heater outages, clean-up demineralizer pump outages, incore instrument stabilization and calibration, and end of core life physics tests.

TABLE 1

OPERATING PERFORMANCE 1968

No. Of Outage	Off System Date	Off System Time	On System Date	On System Time	Outage Hours	Reason For Outage
84	1/13/68	2:36 a.m.	1/15/68	5:34 a.m.	50 Hrs. 58 Min.	Operator training.
85	2/ 3/68	12:56 a.m.	6/ 2/68	5:52 a.m.	2,883 Hrs. 56 [illegible].	Fifth partial refueling, control rod drive overhaul, fuel cleaning, generator overhaul.
86	7/10/68	6:31 p.m.	7/12/68	12:22 a.m.	29 Hrs. 51 Min.	Condenser tube leak repair.
87	7/20/68	1:33 a.m.	7/21/68	6:50 a.m.	29 Hrs. 17 Min.	Condenser tube leak repair.
88	8/24/68	1:42 a.m.	8/26/68	12:27 a.m.	46 Hrs. 45 Min.	Condenser tube leak repair.
89	9/ 6/68	11:18 p.m.	9/ 8/68	1:49 p.m.	38 Hrs. 31 Min.	Operator training.
90	9/20/68	6:27 a.m.	9/22/68	7:54 a.m.	49 Hrs. 27 Min.	Condenser tube leak repair.
91	10/28/68	1:36 a.m.	10/29/68	9:45 a.m.	32 Hrs. 9 Min.	Condenser tube leak repair.
92	10/29/68	12:17 p.m.	11/ 1/68	1:51 a.m.	61 Hrs. 34 Min.	Condenser tube repairs (15 Hrs. 43 Min.), "B" clean-up pump repair (45 Hrs. 51 Min.).
93	11/ 9/68	1:38 a.m.	11/10/68	10:39 a.m.	33 Hrs. 1 Min.	Condenser tube repairs (17 Hrs. 7 Min.), Operator training (15 Hrs. 54 Min.).
94	11/23/68	5:06 a.m.	11/25/68	5:08 a.m.	48 Hrs. 2 Min.	Condenser tube leak repairs.
95	12/ 8/68	9:23 a.m.	12/ 9/68	11:40 a.m.	26 Hrs. 17 Min.	Condenser tube leak repairs.
96	12/16/68	9:08 p.m.	12/18/68	5:47 a.m.	32 Hrs. 39 Min.	Condenser tube leak repairs.
97	12/25/68	8:26 a.m.	12/26/68	8:42 p.m.	36 Hrs. 16 Min.	Condenser tube leak repairs (24 Hrs. 4 Min.), "A" unloading heat exchanger oil cooler leak check (12 Hrs. 12 Min.).

TABLE 1

OPERATING PERFORMANCE 1968
(Continued)

No. Of	Off System		On System			
Outage	Date	Time	Date	Time	Outage Hours	Reason For Outage
98	12/28/68	10:22 p.m.	12/30/68	2:45 a.m.	28 Hrs. 23 Min.	Condenser tube leaks repairs.
TOTAL OUTAGE HOURS FOR YEAR					3,427 Hrs. 5 Min.	

TABLE 2

LOAD RESTRICTIONS FOR 1968

Date	Reduction From Maximum Capability of 210 MWe	Condition
January 16 - January 29	25	"B" Secondary Steam Generator Tube Leak
January 29 - January 31	10	"A" Drain Cooler Tube Leak
January 31 - February 3	30	End of Life Core Physics Test
June 2 - June 6	100	Reactor Incore Calibration
June 6 - June 11	50	Reactor Incore Calibration
June 11 - June 15	20	Reactor Incore Calibration
June 15 - June 25	18	"E" Primary Heater Tube Leak
July 12 - July 13	110	"B" Reactor Clean-up Pump Failure
July 13 - July 17	10	"A" Secondary Feedwater Heater Tube Leak
October 3 - October 16	35	"C" Secondary Steam Generator Flange Leak
October 16 - October 21	10	"C" Primary and "A" Secondary Feedwater Heater Tube Leak
October 21 - October 29	25	"C" and "D" Primary Heater Tube Leak and "A" Secondary Heater Tube Leak
November 1 - November 4	25	"C" and "D" Primary, "A" Secondary Feedwater Heater Tube Leak
November 4 - November 9	18	"C" Primary and "A" Secondary Feedwater Heater Tube Leak
November 10	110	Condensate Demineralizer High Differential Pressure

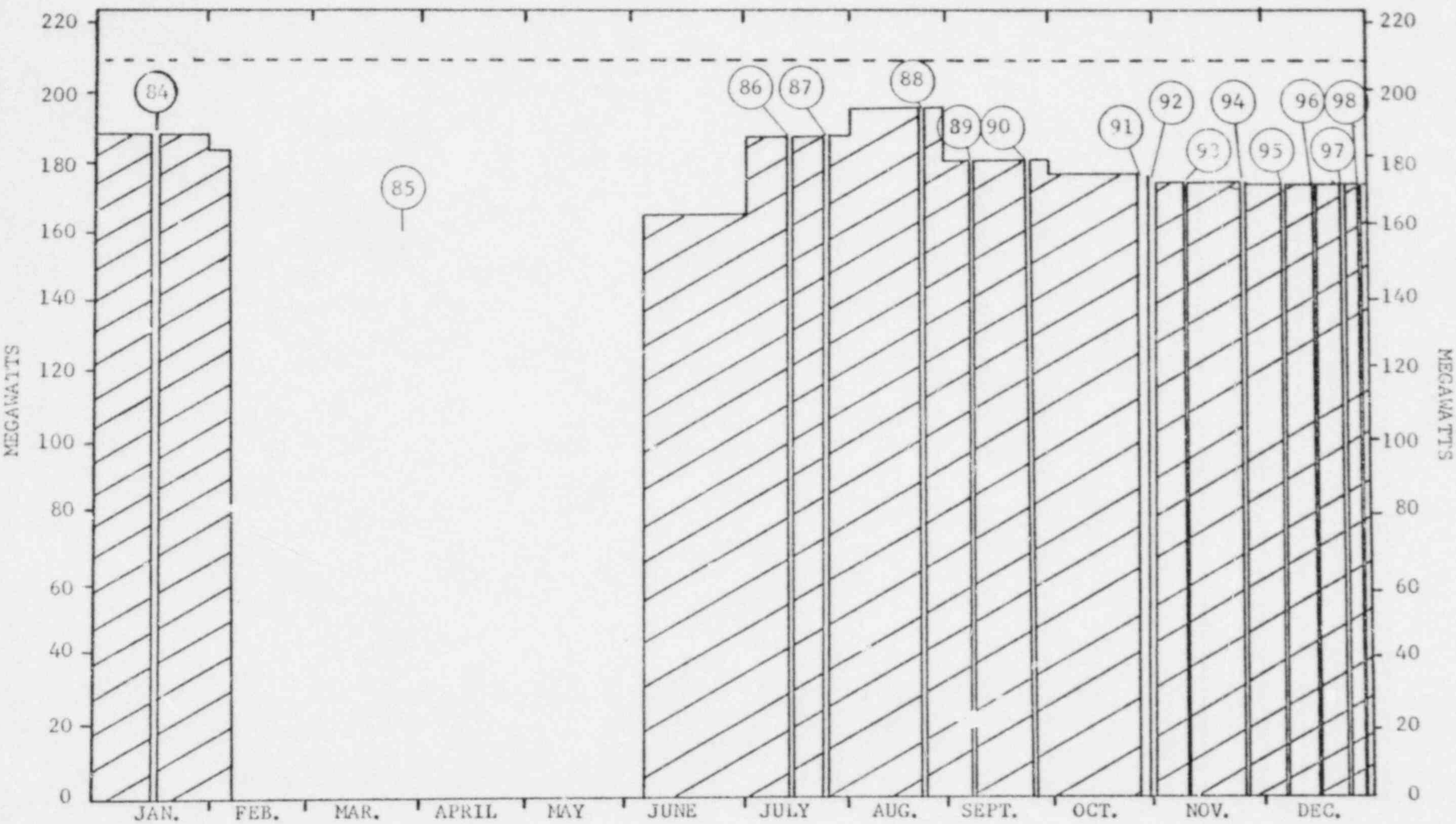
TABLE 2

LOAD RESTRICTIONS FOR 1968
(Continued)

Date	Reduction From Maximum Capability of 210 MWe	Condition
November 11 - December 21	18	"C" Primary and "A" Secondary Feedwater Heater Tube Leak
December 21 - December 30	45	"B" Secondary Steam Generator Tube Leak Repair
December 30 - December 31	60	"B" Secondary Steam Generator Tube Leak Repair and Feedwater Heater Tube Leak Repair

FIGURE 1

PLANT ELECTRICAL LOADING
YEAR 1968
DRESDEN NUCLEAR POWER STATION



III. DISCUSSION

A. Operating Experience

1. Generation

The total reactor operating (critical) time during the year was 5,583 hours and the total power for the period was 133,307.7 MWDt. The gross electrical generation during the year was 966,792.23 MWHe; net generation was 915,930.00 MWHe. As of December 31, 1968, the total gross generation since commencement of power operation on April 15, 1960, was 8,421,120.27 MWHe.

2. Scrams

a. At 6:50 p.m., on November 9, during operator training criticals, a spurious signal on Channel 3 out-of-core initiated a scram due to the coincidence/non-coincidence switch not being locked in the full coincidence position. The reactor power was approximately 200 watts with 40 rods and four notches withdrawn prior to the scram.

b. At 11:33 p.m., on December 17, the reactor was scrammed by the plant safety system from the No. 1 vacuum trip. The scram occurred while heating the primary system at approximately 50° F per hour, with the reactor power at approximately 20 MWt, as the unit was returning to service following a condenser tube repair outage. The scram was due to the approach of reactor pressure to 200 psig before the No. 1 vacuum trip had reset at the setpoint of 23" Hg.

c. At 11:50 a.m., on December 26, a reactor scram was initiated by a spurious signal from Channel 4 out-of-core while down ranging the micro-micro ammeters. The reactor was on a post outage heating rate, at 376° F and approximately 700 watts, with 42 rods and five notches withdrawn prior to the scram.

3. Incidents

a. During scram testing on February 5, Accumulator No. 17 failed to scram. Investigation of the system revealed that the plastic core pilot seat in the scram solenoid valve had broken off and blocked the bleed-off, preventing the scram valves from opening.

The seat was repaired and ten additional seats from the 54 scram solenoid valves were picked at random and inspected. From the conditions observed, no further replacements were necessary. This has been the second seat to malfunction during the history of these valves.

b. On September 23, 1968, the post incident system valves became inoperative from the remote location after rain water leaked into the electrical distribution panel. The leakage originated from an accumulation of rain water on the roof above the panel due to a plugged roof drain.

Repair of the distribution panel was completed within three hours, load was reduced to 135 MWe and shutdown requirements were evaluated vs. repair time. As a result of the occurrence, permanent protection of the existing panel has been completed and all other distribution panels for plant systems, both for safety and operating functions, have been reviewed by station management and found to be adequate.

4. Control Rod Drives

a. Control Rod Drive Operation

1. While exercising control rod drives on January 5, Drive A-5 inserted to Position 4 during attempts to withdraw it from Position 8. The drive was moved to the fully inserted position and flushed. It was then exercised and found to be working properly. The malfunction was attributed to a sticking Barksdale valve. The Barksdale was taken out of service for inspection and repair and returned to service on January 16 with no further problems being experienced.

2. On April 20, while the reactor was in the refuel mode, control rod Drive E-5 could not be withdrawn from Position 0. Investigation led through a series of orifice and drive water regulator setting changes, inspection of the Barksdale and solenoid valves, and checking the conditions of the seats on the Asco valves. Although no apparent reason for the problem was found, the drive operated properly when it was returned to service. It was believed that the malfunction was caused by a blocked solenoid orifice, which was relieved when the solenoids were disassembled. Although the problem would cause the drive to malfunction in this manner, it would not impair the drive's ability to scram.

3. While attempting to use control rod Drive F-5 as a cocked rod during reactor fuel loading on April 20, the drive failed to withdraw at normal or increased hydraulic pressures. Investigation of the occurrence revealed that a fuel element nosecone had parted from the fuel assembly during movement and fell into the guide tube opening on the core lower grid, thereby obstructing control rod blade movement. No further problems were experienced with Drive F-5 after the element nosecone was removed from the control cell.

4. During daily control rod drive exercises on May 10, control rod E-6 was withdrawn to Position 12. It then inadvertently drifted to Position 11. The malfunction of the drive is attributed to a sticking Asco valve, which when slow or reluctant to move, would cause the drive to drift after previous insert or withdraw movements. Abnormal operations of the drive were cleared when the Asco's were changed on August 27.

5. On June 7, during rod withdrawal, the position indicator for Drive H-9 was noticed to jump to Position 12, hold for a time, then drop back again. The drive was then inserted with the indicator again jumping to Position 12, this time remaining there. The cause of the malfunction was due to a faulty probe, which after replacement on July 11, revealed no further abnormalities.

6. On June 8, control rod Drive H-4 position indicator was noticed to jump off scale (past 12) when the drive was inserted from Position 12 to Position 11. The drive was withdrawn to Position 12 and again inserted to Position 11. Deflection was observed once more. From the irratic response of the indicator, it was assumed that Switch 12 on the drive probe was remaining closed and the resulting combination of Switches 11 and 12 caused the full scale reading. The faulty probe was replaced on July 11.

7. On June 12, with the drive at Position 0 and the green light on, the drive position indicator of H-5 drive was noticed to suddenly jump to Position 3, hold for a time, then drop back. Observation resulted in the conclusion that Switch 3 on the drive probe was shorting out on occasion, thus moving the indicator to Position 3. The faulty probe was replaced on July 11.

8. During daily control rod drive exercises on June 12, the green position indication light for Drive K-5 was observed to remain on at positions other than 0. The transmitter for Drive K-5 was changed, but the green light remained on. The conclusion therefore, was that the drive probe S. G. switch was probably shorted out. The probe was changed on July 11, alleviating the problem.

   On July 15, when the transmitter was replaced, a relay fuse blew in the drives' control circuit. Because of the abnormal response of this transmitter, and similar responses to a later transmitter, it was concluded that either the probe or probe connector was shorted. The probe was replaced on July 20, after which normal operation resumed.

9. During reactor startup on July 20, prior to becoming critical, control rod drives on Accumulator 21 (E-8, A-7 and C-3) and Accumulator 17 (D-10, D-7 and C-2) drifted in following previous withdraw movements. Extensive investigation of piping, valves, pressure settings, etc. revealed no explicit cause of the malfunction. However, it is believed that the scram inlet valves from both accumulators were not seated properly at the lower temperatures and thus caused the drives to insert. When the reactor was at rated temperature and pressure, and following extensive movements and testing for drift, all six drives on the two accumulators began operating normally, and have been operating properly since the occurrence.

10. During control rod drive exercises on September 4, control rod Drive D-7 would not unlatch from Position 0 with normal or increased hydraulic pressures. The Asco orifices to Drive D-7 were checked and the insert orifice found fully closed. Normal position of this orifice is completely open. The system was returned to correct settings and normal operation was continued.

11. On December 20, with the reactor at normal operating pressures and temperatures, control rod Drive H-7 failed to withdraw from Position 0. Varied pressures and Asco settings resulted in only intermittent movement of the drive. Finally, after successfully moving the drive from Position 0 to Position 1, H-7 drifted on to Position 2 after a 4 - 5 minute settling period. The drive was left at this location with no further movement attempted. Additional exercising and flushing of H-7 will continue until the problem is resolved. It is believed that a crud buildup is temporarily hung up in the collet piston - shuttle piston area and is therefore preventing normal unlatching of the collet fingers.

b. Control Rod Drive Tests

On February 5, all control rod drives, except SN 1290 (Core Position F-6), were scram and friction tested and timed for normal insertion and withdrawal. Drive 1290 was not scram tested because of its previous history of short buffer. This testing was performed as a routine check of drives to be overhauled during the spring refueling outage.

Following the overhaul and replacement of 40 drives during March and April, 1968, initial scram and timing tests were performed on the 80 drives on April 14, to insure the operability of each drive prior to loading fuel. The tests were conducted using "dummy" fuel assemblies in the fuel cells. These tests were followed by normal timing and latching tests and scram and friction tests on May 10 and 11, and on October 29 - 31. The data obtained from all the above tests were satisfactory on all drives.

c. Inspection

The technical specifications to Dresden's License DPR-2, as amended by Change Nc. 7, dated April 9, 1964, state that during major outages, "Not less than two control rod drive mechanisms shall be removed, disassembled, and thoroughly inspected at intervals not to exceed 24 months." In addition to license requirements, drives removed for inspection are selected on the basis of drive test results and malfunctions experienced during operation.

Forty control rod drives were removed and replaced with repaired drives during the fifth partial refueling outage. Many of these drives had exhibited long insert times during Cycle V. Four spare drives, SN 1233, SN 1273, SN 1257 and SN 1243, were overhauled prior to the outage and were tested satisfactorily in early February, 1968, in the drive test facility. These drives replaced four drives removed from the reactor, of which two of the replaced drives plus 38 others removed during the outage were overhauled, inspected returned to the reactor, and satisfactorily tested before reloadings started. Figure 2 shows the location of the 40 drives removed.

After removal of reactor head, turning vane and fuel, the control blades were removed and index tubes withdrawn. The drives were removed from core and transported to the 565' elevation.

1. Visual Inspection

All parts were visually inspected as closely as radiation levels would permit. A 1/4" plexiglass face shield was used to protect against beta emission when viewing parts. All moving parts on roller mounts were actually moved in order to verify freedom of movement. Visual inspection of roller mount assemblies was done at about a three foot distance or viewed underwater.

All canned magnets were dropped into hot water and heated to about 200° F. Leakers were found by streams of air bubbles being forced from small racks.

2. Flourescent Dye Penetrant Check

A fluorescent dye penetrant Zyglo examination was made on components of drives. A penetrant (ZL-2), developer ZP-5, and ultraviolet light were used for the inspection.

After ultrasonic cleaning, parts were painted with penetrant and allowed to set for 20 minutes to allow capillary action time to draw penetrant into any cracks. The dye was then wiped off with lint free rags removing all dye except that in small scratches and cracks.

The part was dusted lightly with developer in order to draw dye from any cracks. The dye fluoresces under ultraviolet light and cracks will stand out readily against the background.

3. Results of Inspection

a. Of the 40 drives removed, only 38 were overhauled for lack of spare parts. The remaining two drives will be overhauled at a later date.

b. Nitrided guide roller pins were used to replace all chrome plated pins (11A3448) in the roller mount assembly. Practically all of the pins removed exhibited some wear in the form of loss of chrome. Of the 152 pins only one had failed and two were severely worn. All others had worn an average of six thousandths on the 250 thousandths diameter.

c. All rollers were checked with a go-no-go gage. This gage has an outside diameter of .260". The maximum tolerance for the hole of a new roller is .259", thus only one thousandth was acceptable for roller wear. Only sixteen rollers were worn an amount greater than this. No effort was made to determine the actual wear of rejected rollers. The replacement rollers were nitrided inside the pin hole to minimize wear.

d. Dye checking revealed chrome cracking or flaking on seven guide plugs very similar to conditions found on inspection of December, 1962. (See comments on individual drives for specific details.) Aside from guide plug chrome plating, no cracking was found in drive components.

e. All seals were replaced on every drive overhauled. In the absence of gross seal wear or breakage, the condition of seals causing long insert times is described as "normal wear" in the report.

f. Eleven drives had scratches on the inside of the shuttle piston (856B398P1) of magnitude sufficient to cause interference with the collet assembly and in some cases, malfunctions of the collet.

Table 3 contains a summary of the 1968 drive inspection and overhaul.

FIGURE 2

LOCATION OF CONTROL ROD DRIVES OVERHAULED
DURING 1968 MAJOR OUTAGE

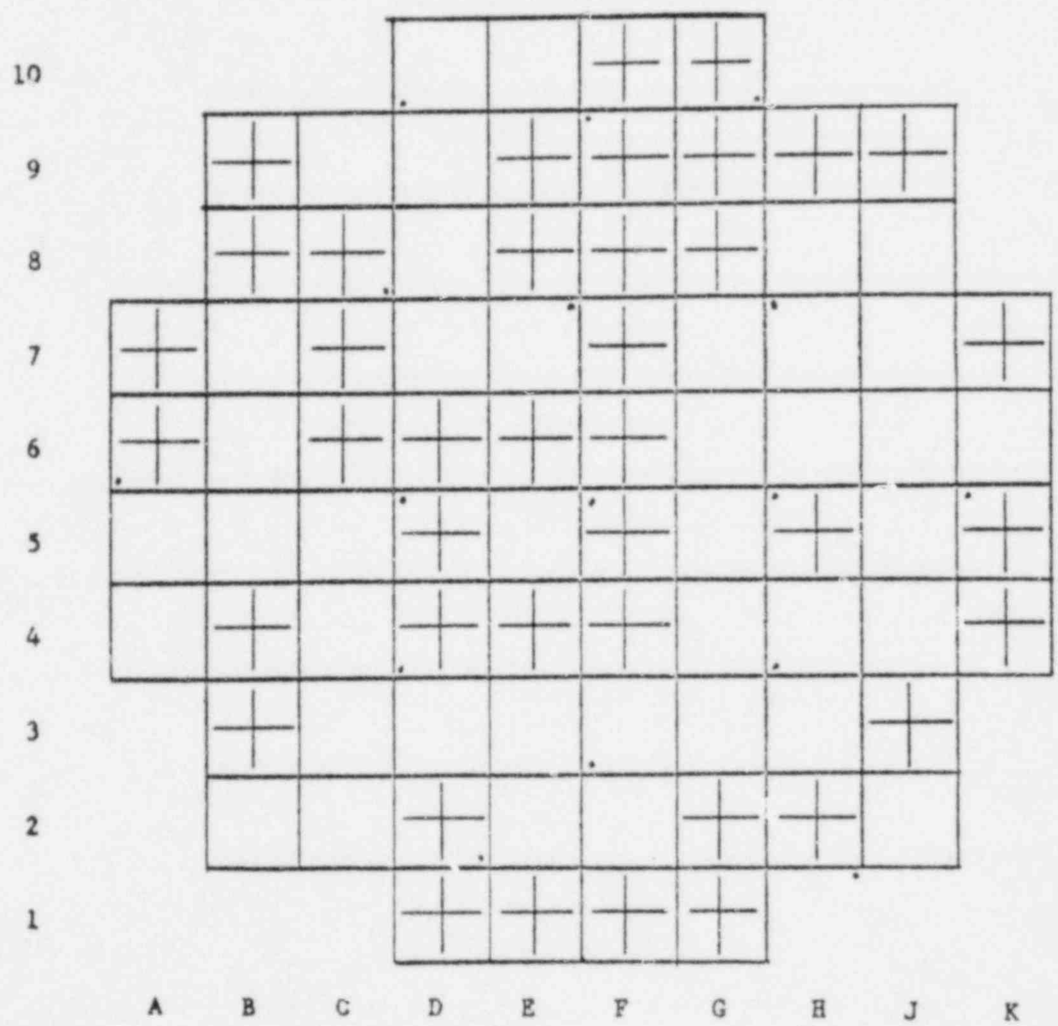


TABLE 3

1968 DRIVE INSPECTION RECAP

DRIVE NUMBER	CORE POSITION	DEFICIENCY *	SEVERLY WORN ROLLER PINS	EXCESSIVE WEAR ON ROLLERS	CHROME IMPERFECTIONS ON GUIDE PLUG O.D.	SCRATCHES ON SHUTTLE PISTON ID	DAMAGED STRAINER (8858170)	BENT FINGERS ON COLLET ASS'Y	WORN OR BROKEN DRIVE PISTON SEALS	WORN OR BROKEN STOP PISTON SEALS	DYE CHECK RESULTS	PARTS REPLACED **
1294	B3	DRIFT SLOW			X				X	X	OK	
1292	A6	STUCK SLOW				X			X	X	OK	Magnet, collet piston rings
1279	K4	STUCK DRIFT		2		X				X	OK	Magnet
1254	J9	DRIFT SLOW	1			X			X		OK	Index tube
1278	E4	SLOW						X	X		OK	Collet adaptor sleeve, collet assembly
1262	B4	DRIFT							X	X	OK	Collet assembly
1244	F1	SLOW					X		X	X	OK	Guide plug, strainer
1290	F6	SHORT BUFFER			X	X			X	X	Guide plug cracked	Guide plug, collet adaptor sleeve
1284	E9	SHORT BUFFER							X	X	OK	
1252	E6	SHORT BUFFER							X	X	OK	Roller mount assembly
1226	D5	SHORT BUFFER							X	X	OK	
1259	C6	SHORT BUFFER								X	OK	Guide plug, magnet
1306	H9	SHORT BUFFER	1	4	X			X	X	X	OK	Guide plug, collet assembly
1317	F8	SHORT BUFFER				X			X	X	OK	Magnet
1237	F4	SHORT BUFFER		2	X	X			X	X		Drive piston & piston head
1267	F5	SHORT BUFFER								X	OK	Filter (8568985)
1263	F10	SLOW							X	X	OK	Magnet
1310	A7	SLOW		2					X	X	OK	Guide plug, filter (8568985)
1248	D1	SLOW		1					X	X	OK	
1297	G1	SLOW		1	X				X	X		Guide plug
1238	F9	SLOW							X	X	OK	
1286	B9	SLOW	1 broken	3					X		OK	
1316	D6	SLOW				X			X	X	OK	
1265	B8	SLOW					X		X	X	OK	Strainer
1315	C8	SLOW				X			X	X	OK	
1231	K7	SLOW							X	X	OK	
1313	F7	SLOW							X	X	OK	
1293	E1	SLOW							X		OK	
1288	C7	SLOW							X	X	OK	
1227	G9	SLOW				X			X	X	OK	Magnet, index tube
1260	E8	SLOW				X			X	X	OK	
1242	D4	SLOW				X			X	X	OK	
1256	D2	SLOW					X	X	X	X	OK	Magnet, strainer, collet assemblies
1239	H5	SLOW							X	X	OK	Filter
1309	G10	SLOW			X				X	X		Guide plug
1270	H2	SLOW				X			X		OK	
1230	K5	SLOW							X	X	OK	
1268	G2	SLOW		1	X				X	X		

* "DRIFT" means drive failed to latch
"STUCK" means drive could not be moved
"SLOW" means drive had long insert time, ie greater than 26 seconds
"SHORT BUFFER" means drive did not slow up sufficiently at end of scram stroke (travel time in the buffer less than .3 seconds)

**In addition to parts replaced as noted, all roller pins, drive piston seals and stop piston seals were replaced on all overhauled drives

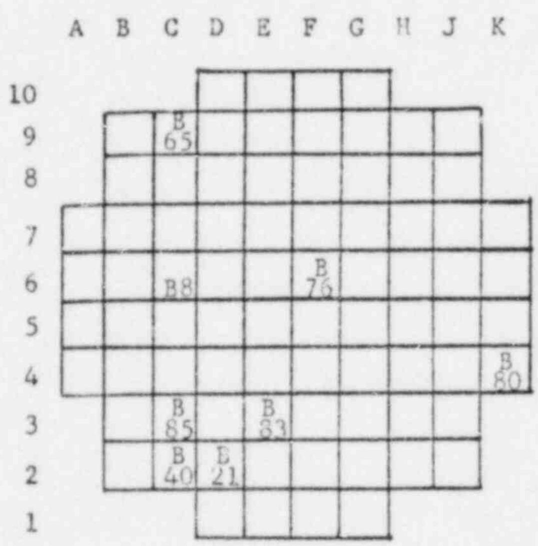
5. Control Rod Blades

a. Blade Following Checks

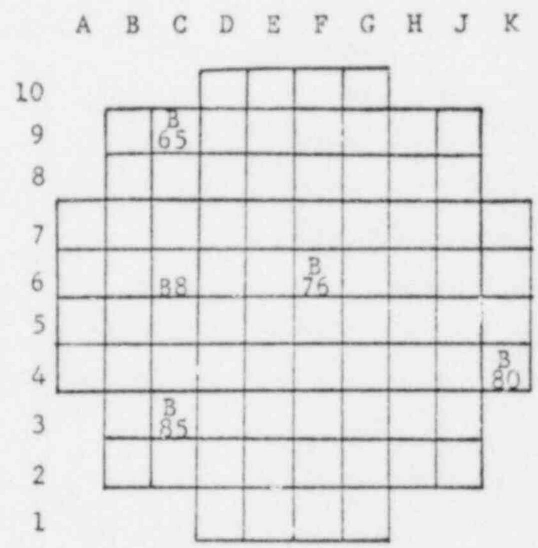
During periods of operation, control rods have been verified for blade following on a weekly basis. Control rod worth tests were also conducted prior to and following the refueling outage and during each month of operation. During each startup, control rod patterns for criticality have been predicted and all blade following verified.

b. Control Blade Inspection

Control blades were inspected in the fuel building storage pool during the refueling outage on April 7 and 8, 1968. Eight blades were checked with go-no-go gages for dimensional variations and five of the eight by underwater TV for signs of visual wear. All blades inspected were found to be in acceptable condition. The blades examined and their locations in the core are exhibited below:



Control Blades
Examined By
Go-No-Go Gage



Control Blades
Examined By
Underwater TV

6. Reactor Clean-up Demineralizer System

"A" clean-up demineralizer loop has been out of service since November, 1966. The loop was taken out of service due to apparent tube leaks in the regenerative heat exchangers.

During the month of August, 1968, a decontamination project was undertaken to reduce the high contamination levels inside of "A" clean-up loop piping and heat exchangers. Prior to the clean-up project, work had been seriously hampered by the high radiation fields near the demineralizers.

The decontamination project was under the direction of the Dow Chemical Company, Dow Industrial Services, and Atcor, Inc., who, with the loop isolated, circulated an acid solvent through the "A" system. The decontamination of the clean-up loop was completed in September. As a result of the cleaning, the contamination levels were reduced to levels low enough to allow access to the loop for inspection and repair work. At the close of the year, this work was still progressing on the repair of "A" clean-up demineralizer loop heat exchangers.

On July 12, 1968, during a plant startup, "B" clean-up demineralizer pump tripped out of service after a control rod drive spud guide roller wedged between the impeller and pump casing. The unit was derated 100 MWe for 42 hours with no primary clean-up system in service while the pump from "A" clean-up loop was installed in "B" clean-up circuit.

On October 29, during a condenser tube leak repair outage, "B" clean-up demineralizer pump again tripped out of service after a small piece of metal wedged between the impeller and the pump casing. The pump was repaired and the unit returned to service November 1.

7. Changes in Facility Design

a. Fuel Building Radiation Monitor

During the month of January, an Area Radiation Monitor was installed in the fuel building. The monitor alarms in the control room and fuel building when the sensor, located at the north end of building adjacent to the storage pool, sees more than 10 mR per hour. In addition, the monitor trips all power to the overhead crane, thus preventing any further movement. As was the original Riggs monitor, the new Area Radiation Monitor is tested each day when men are working in the area.

b. Reactor Canal Crane Circuit Trip Modification

A trip circuit was installed on the reactor canal crane during the month of March. Any one of three radiation monitors (set to alarm at 10 mR/Hr.) will now trip the reactor canal crane main contactor during refueling operations. Hand operated trip buttons were also placed at the north and south ends and center of the pool area.

c. Material Test Loop Installation

During the fifth partial refueling outage, General Electric Company initiated installation of a materials test loop in "B" reactor recirculation loop. This installation was completed on May 28, 1968. The purpose of the facility is to determine whether corrosion by a reactor water environment can significantly reduce the fatigue life of austenitic stainless steels. A secondary objective is to determine whether metal fatigue in a boiling water reactor coolant can change mode of propagation, from transgranular to an intergranular mode.

The facility includes 1 1/2 inch supply and return lines from "B" loop which will supply approximately 1 GPM of reactor water to an autoclave outside the generator room. Both lines of the loop are terminated outside of "B" compartment with a valve and blind flange pending installation of the autoclave later in 1969. The autoclave will contain stainless steel metal samples for strain cycling and associated instrumentation and control. The supply line connects to a cross flange installed on the six inch bypass loop decontamination flange. The return line connects to a second cross flange installed on the decontamination flange on the generator inlet line. Both lines have motor operated isolation valves inside the compartment.

The two cross flanges house four electrochemical corrosion test flanges which were installed during the outage. The test flanges are a continuation of a corrosion rate test program initiated during 1967.

d. Vacuum Pump Discharge Line Revision

The six inch discharge line from the mechanical vacuum pump was rerouted from the 30 inch air ejector holdup to the 24 inch gland exhauster holdup on May 12-15. The revision restores the mechanical vacuum pump tie to the 24 inch holdup line as it was originally designed into the plant. During Unit #1 startup testing in 1960, the piping had been routed to the 30 inch holdup because of backpressure on the turbine gland seal system. The backpressure problem was eliminated by a connection directly into the 24 inch holdup header rather than the original six inch gland exhauster discharge line.

A pre-operational test was made on this new arrangement on May 27. Test data, taken from various vacuum pump and gland exhauster operational combinations, indicate that no problems exist with the revision.

e. Control Rod Drive Thimble Supports

Support structures, for the reactor control rod drive thimble housings, were installed during May by Shear Connectors, Inc., under the supervision of the Commonwealth Edison Mechanical and Structural Department. The supports, which were installed below the 80 control rod drive flanges at the base of the reactor, are designed to prevent drive ejection from the vessel in the event of a thimble weld failure.

f. Off Gas Monitor Range Switch

A mechanical stop was installed on the off gas monitor range switch during August. The mechanical stop prevents ranging the monitor above values which correspond to off gas release limits.

g. Extension of No. 1 Deep Well

The No. 1 deep well was modified during the latter part of 1968 to accommodate the additions of Dresden 2 and 3. The deep well, which is located approximately 400 feet directly south of the station for supply of well water to Dresden Unit No. 1, was extended from the original depth of 788 feet to a new, more abundant supply depth of 1,500 feet. Digging of the well was essentially completed on December 6, and capacity and water sampling tests were in progress at the end of the year.

h. Reactivation of Hansel and Breen Environs Stations

On August 25, the Hansel and Breen Environs Stations were returned to active service. Both stations were returned for operation as they were originally before deactivation in 1967, although both have been modified for use only in direct radiation monitoring of the environs.

i. Meteorological Survey Tower Erection

For determination of climatological conditions at the station, a site meteorological survey tower was constructed and put into operation during the early part of 1968. The facility consists of a 400 foot tower with four distinct elevations for measurement of wind speed, direction and temperature. The data obtained from these elevations is fed to a computer at the base of the tower for computation and recording for contractor analysis. The data will be used for determination of:

1. Climatological conditions

2. Ground level concentrations from stack gas emissions

j. Dresden Units No. 1, 2 and 3 System Connections

Connections between operational systems on Units No. 1, 2 and 3 were made possible during 1968 in the following locations.

1. Unit #1 make-up pumps are available to supply Units 2 and 3 heating boilers in the event of loss of the Unit 2 clean demineralized water pumps. The crosstie connects Unit 1 to Unit 2 via a three inch line with manual isolation valves. A check valve, also installed, prevents the normally higher Unit 2 pressure from supplying Unit 1.

2. A tie into Unit 1 clean demineralized water storage tank was made for the Unit 2 and 3 clean demineralized water pumps. The tie is a three inch line with manual isolation valves for normal supply to Units 2 and 3 pumps.

3. A six inch tie into the Unit 1 well water tank was made for Units 2 and 3 make-up feed pumps.

4. A crosstie between Units 1 and 2 clean demineralized water header exists on Unit 2 in order to supply demineralized water to Unit 2 should the Unit 2 clean demineralized water pumps be out of service. The crosstie is made via a three inch line with manual isolation valves.

5. Units 1, 2 and 3 instrument and service air systems can be interconnected.

   Service Air is crosstied to Unit 2 by a two inch line. The cross connection is automatic at 95 psi on Unit 2 by action of a pressure regulating valve between the systems. The crosstie is to act as a backup for Unit 2 service air in as much as Unit 2 has only one air compressor.

   Instrument Air is crosstied between Units 1 and 2 through a two inch line. The cross connection is automatic via a pressure control valve set at 92 psi. It is used to supply instrument air to Unit 2 if the Unit 2 air driers were out of service or instrument air pressure dropped below the required setpoint. The crosstie is limited by a 1/4 inch orifice which prevents a break in the Unit 2 instrument air header from dropping Unit 1 instrument air pressure.

All systems have been tested and are operational. The connections have all been used intermittently as construction on the two newer units progresses.

k. Installation of Unit #1 Evacuation Sirens

Two Unit 1 evacuation alarm sirens were installed during August in the Unit 2 and 3 construction area. One siren was installed on the cribhouse roof in the Unit 2 area and the other at the south-west corner of the main floor of the Unit 3 turbine building. The control key and control relays for these two alarms are mounted on a wall south of Unit 1 generator exciter. A 4,800 volt feed is supplied from CTR 26, bus 5A (Unit 1) at 100 AMPS. The sirens are actuated automatically by Unit 1 reactor sphere pressure or by a control key for testing.

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7. "A" Clean-up Loop Decontamination Piping Isolation

During August, two isolation valves (AO 600 and AO 601) were installed in "A" clean-up demineralizer pump room to facilitate decontamination of the "A" clean-up loop non-regenerative heat exchangers. Both valves are two inch air operated valves and are installed on a spare two inch stainless steel line (5639) to the radwaste area. The valves are controlled from the control room, with each valve having position indicating lights on Bench Panel B-3. In the event of a scram, both AO 600 and AO 601 will close automatically through the plant safety system relays.

8. Personnel Radiation Exposure

Personnel exposures to radiation were within limits specified in 10 CFR Part 20.

9. Liquid Poison System

The liquid poison system was operative at all times during the year. The boron poison was sampled on March 29, May 29, May 30, September 21, and October 31. There were no conditions which would indicate a loss of boron from the solution tank. Boron concentrations in the reactor water remained low throughout the year.

10. Radioactive Waste Disposal

Release of radioactive liquid waste was accomplished in batch quantities at controlled release flow rates according to established procedures. The contribution to the activity of dilution water was always maintained within the limits specified in the applicable federal regulations. The average contribution to the unidentified activity in the water utilized for radioactive liquid waste dilution during the year was calculated to be $.189 \times 10^{-7}$ uc/ml (18.9 uuc/l) compared to an average limit of 1.00×10^{-7} uc/ml (100 uuc/l) for unidentified mixtures containing radium 226 or radium 228 as specified in 10 CFR Part 20.

Solid radioactive wastes were stored on-site pursuant to License DPR-2. Table 4 shows the content, shipment locations, and dates of radioactive waste shipments made during the year. A total of 11 radioactive waste shipments were made in 1968.

The concentration of noble fission gases in the stack discharge to atmosphere was maintained well within license limits of 700,000 microcuries per second. The average activity release rate for the year while the plant was operating was approximately 12,500 uc/second.

Eight Nuclear Fuel Services, Inc. - Stanray rail cask shipments, totaling 184 spent fuel assemblies, were made during the year to the Chemical Processing Plant of Nuclear Fuel Services, Inc., in West Valley, New York. Table 5 is a breakdown of these shipments in addition to all shipments made since initiation in June, 1965.

TABLE 4

RADIOACTIVE WASTE SHIPMENTS - 1968

Content	Date	Volume	Total Activity (Millicuries)	Location of Shipment
Dry Radioactive Waste	May 1	286.65 ft^3	318.96	Nuclear Engineering Co.; Sheffield Nuclear Center; Sheffield, Illinois
Dry Radioactive Waste	May 2	499.80 ft^3	142.21	"
Dry Radioactive Waste	May 2	405.00 ft^3	600.53	"
Dry Radioactive Waste	May 3	1,116.00 ft^3	16.48	"
Dry Radioactive Waste	July 9	1,374.60 ft^3	147.96	"
Dry Radioactive Waste	July 10	730.50 ft^3	105.06	"
Dry Radioactive Waste	November 13	1,523.85 ft^3	141.85	"
Dry Radioactive Waste	November 14	342.00 ft^3	57.20	"
Dry Radioactive Waste	November 15	321.00 ft^3	58.96	"
Dry Radioactive Waste	November 18	1,147.45 ft^3	251.27	"
High Level Waste	August 23	60.00 ft^3	207,000.00	Atcor, Inc.; Hawthorne, New York

TABLE 5

SPENT FUEL SHIPMENT SUMMARY

Shipment Number		Date	Number of Assemblies or Containers							
			Batch					Total		
Rail	Truck	Shipped	1	2	3	4	5	Rail	Truck	To Date
1		6/11/65	24	0	0	0	0	24		24
2		6/30/65	24	0	0	0	0	24		48
3		7/16/65	24	0	0	0	0	24		72
4		8/ 3/65	24	0	0	0	0	24		96
5		8/16/65	24	0	0	0	0	24		130
6		9/ 2/65	24	0	0	0	0	24		154
7		9/22/65	24	0	0	0	0	24		168
	1	8/ 1/66	0	0	4	0	0		4	172
8		8/ 5/66	16	0	6	0	0	22		194
	2	8/15/66	0	0	4	0	0		4	198
	3	8/24/66	0	0	4	0	0		4	202
	4	8/28/66	0	0	4	0	0		4	206
9		8/31/66	0	0	12	8	0	20		226
	5	9/ 5/66	0	0	4	0	0		4	230
	6	9/12/66	0	0	4	0	0		4	234
	7	9/14/66	0	0	4	0	0		4	238
10		9/16/66	0	0	0	20	0	20		258
	8	9/19/66	0	0	4	0	0		4	262
	9	9/21/66	0	0	4	0	0		4	266
	10	9/25/66	0	0	4	0	0		4	270
	11	9/26/66	0	0	4	0	0		4	274
	12	9/28/66	0	0	4	0	0		4	278
	13	10/ 2/66	0	0	4	0	0		4	282
	14	10/ 3/66	0	0	4	0	0		4	286
11		10/ 7/66	0	0	0	24	0	24		310
	15	10/11/66	0	0	4	0	0		4	314
	16	10/12/66	0	0	4	0	0		4	318
	17	10/20/66	0	0	4	0	0		4	322
	18	10/23/66	0	0	4	0	0		4	326
	19	10/26/66	0	0	4	0	0		4	330
12		10/28/66	0	0	3	16	0	19		349
	20	10/30/66	0	0	4	0	0		4	353
	21	11/ 1/66	0	0	4	0	0		4	357
	22	11/ 6/66	0	0	4	0	0		4	361
	23	11/ 8/66	0	0	4	0	0		4	365
	24	11/10/66	0	0	4	0	0		4	369
	25	11/13/66	0	0	4	0	0		4	373
13		11/14/66	0	0	0	23	0	23		396
	26	11/15/66	0	0	4	0	0		4	400
	27	11/17/66	0	0	4	0	0		4	404
	28	11/20/66	0	0	4	0	0		4	408
	29	11/27/66	0	0	4	0	0		4	412
	30	11/29/66	0	0	4	0	0		4	416

Shipment Number		Date	Number of Assemblies or Containers							
			Batch					Total		
Rail	Truck	Shipped	1	2	3	4	5	Rail	Truck	To Date
14		12/ 2/66	0	0	4	19	0	23		439
	31	12/ 6/66	0	0	4	0	0		4	443
	32	12/11/66	0	0	4	0	0		4	447
	33	12/15/66	0	0	4	0	0		4	451
	34	12/18/66	0	0	4	0	0		4	455
	35	12/20/66	0	0	4	0	0		4	459
	36	12/27/66	0	0	4	0	0		4	463
	37	1/ 3/67	0	0	4	0	0		4	467
	38	1/ 5/67	0	0	4	0	0		4	471
	39	1/ 8/67	0	0	4	0	0		4	475
	40	1/10/67	0	0	4	0	0		4	479
	41	1/15/67	0	0	4	0	0		4	483
	42	1/22/67	0	0	1	0	0		1	484
15		7/15/68	0	0	0	24	0	24		508
16		8/ 9/68	0	0	0	24	0	24		532
17		8/29/68	0	0	0	21	0	21		553
18		9/19/68	0	0	0	21	0	21		574
19		10/ 9/68	0	0	0	24	0	24		598
20		10/30/68	0	0	0	24	0	24		622
21		11/20/68	0	0	0	24	0	24		646
22		12/11/68	0	0	0	22	0	22		668

11. Partial Refueling and Fuel Inspection

a. Fifth Partial Refueling

To facilitate fuel cleaning and drive removals, all fuel assemblies and control blades were removed from the core and stored in the fuel building.

All fuel assemblies were removed from the core, 2/18/68 through 3/1/68. All control blades were removed from the core, 2/23/68 through 2/29/68. Forty drives were removed from the core, 3/6/68 through 4/1/68. Repaired 40 drives reinstalled in core, 3/14/68 through 4/2/68. All control blades reinstalled in core, 4/12/68 through 4/15/68. All fuel assemblies reinstalled in core, 4/19/68 through 5/9/68. Metal specimens (removed on 3/2/68) were reinstalled in the core on 5/9 and 5/10/68.

1. New Fuel Characteristics

Ninety-six Type VI fuel assemblies were purchased from United Nuclear Corporation of Elmsford, New York. This fuel is similar in nuclear and thermal hydraulic characteristics to the General Electric Type III-F fuel.

Type VI fuel consists of 84 "normal" assemblies and 12 instrumented assemblies. The "normal" assembly consists of a six by six array of fuel rods containing UO_2 with one of the rods being a removable poison rod containing Gd_2O_3 mixed with UO_2. The instrumented assemblies consist of a six by six array without a poison rod. An instrument tube is inserted in place of the removable poison rod leaving 35 rods containing UO_2.

Thirty-two Type VI assemblies have a peripheral orifice and 64 assemblies have a central region orifice. The peripheral orifice has a single hole, 1.49 inches in diameter. The central orifice has a single hole, 1.81 inches in diameter.

Critical experiements, using Type VI fuel, were performed at Pawling, New York, laboratory of United Nuclear Corp. A loading to critical was performed first using only unpoisoned assemblies and then with only poisoned assemblies.

A combination of poisoned and unpoisoned assemblies were made critical with the water height near the top of the fuel for the uniformity check. Eleven assemblies were checked for uniformity with a total spread of 7.2¢ in reactivity. The critical core configurations are shown on the following page.

	X		X[1]
X	X	X	X
X	X	X	X
X	X	X	X

Unpoisoned Critical Array - 14 Assemblies

	X	X	
X	X	X	X
X	X	X	X
X	X	X	X
X	X	X	X

Poisoned Critical Array - 18 Assemblies

	X	X	
X[2]	X	X	X
X	X[23]	X[2]	X
X	X	X	X

Uniformity Test Array

1 - Last Assembly Added
2 - Poisoned Assembly
3 - Substitution Location

2. Fuel Assembly Cleaning and Testing

a. Fuel Cleaning

Fuel assembly inlet orifice cleaning was initiated on March 4. Ultrasonic cleaning of the inlet portion of the fuel assemblies was attempted in an effort to develop a faster and more efficient method of crud removal. After three days of only partial success with ultrasonic cleaning, it was decided that mechanical cleaning by manually brushing should be initiated in order to complete the job in the allotted outage time.

Manual brushing began on March 6, and was completed on March 22. Inlet orifices on all assemblies scheduled for return to the reactor for Cycle VI, with the exception of assemblies PF-10, SA-1, and A-465, were manually brushed and the effectiveness of the cleaning determined by flow testing. A total of 365 assemblies were cleaned, including 163 Type III-B's, 96 Type III-F's and 106 Type V's.

b. Fuel Assembly Flow Testing

All but four of the assemblies in the reactor during Cycle V were flow tested in the "as found" or "dirty" condition and all were tested again following cleaning. Flow testing of the fuel assemblies was accomplished in the flow test fixture constructed during the 1967 refueling outage.

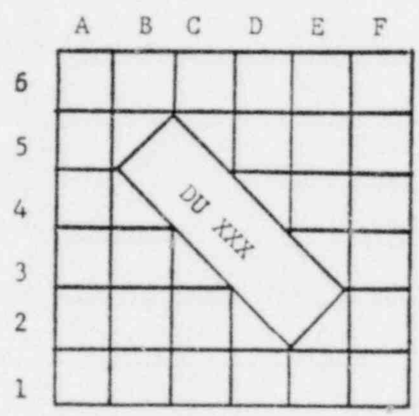
Data obtained by flow tests of one new Type V assembly with an "E" orifice and one with a "D" orifice during the 1967 outage were used as the criteria for cleaning effectiveness. A comparison of this data with data obtained from cleaned assemblies during the present outage is shown in Figure 3.

Three new Type VI assemblies with Type I orifices and three with Type II orifices were flow tested to obtain a reference point for future cleaning of Type VI fuel. Data obtained are also shown in Figure 3. The orifice of the assembly was removed, cleaned, replaced and the assembly was flow tested again to see that it had been cleaned sufficiently.

3. <u>Gadolinia - Alumina Rod Installation</u>

a. <u>Gadolinia Rod Inspection</u>

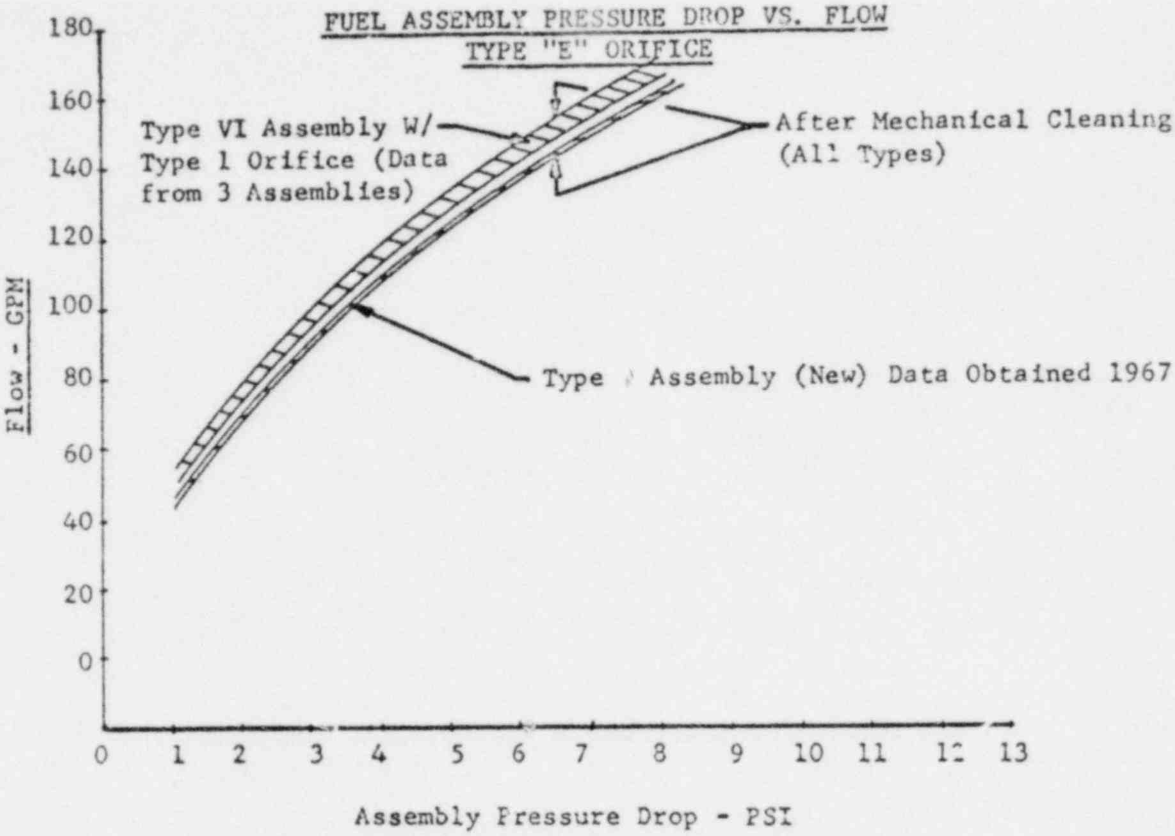
Gadolinia-urania rods were removed from the fuel assemblies DU-102 and DU-56. They were replaced with gadolinia-alumina rods. The removal and replacement scheme was as follows:

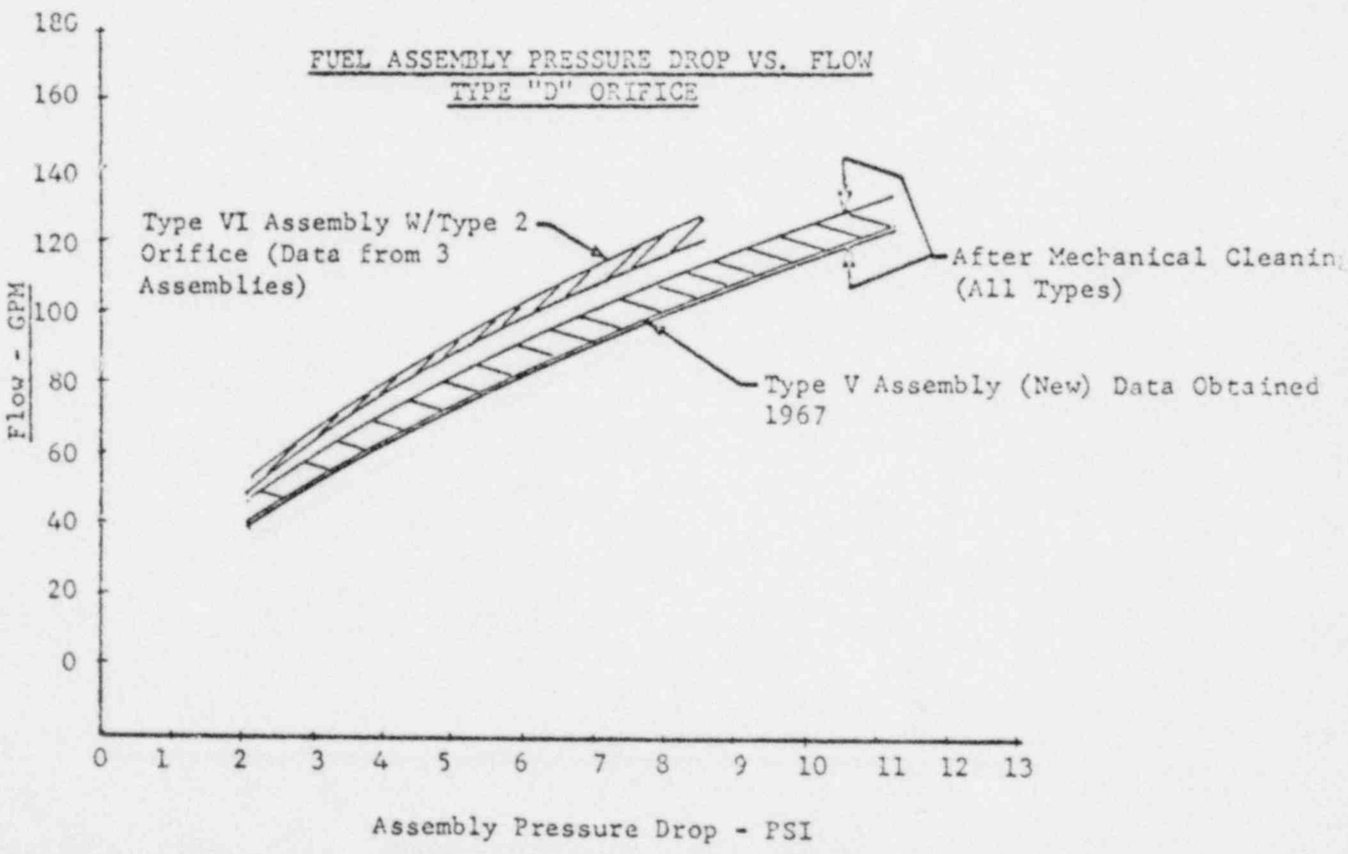


ASM	Exchange
DU-56	Gadolinia-urania rod, from E-5 to storage. Gadolinia-alumina rod (S. N. DS013) to E-5.
DU-102	Gadolinia-urania rod from B-2 to storage. Gadolinia-urania rod from E-5 to B-2. Gadolinia-alumina rod (S. N. DS041) to E-2.

A visual inspection of the spacer contact elevations disclosed no fretting or wear.

FIGURE 3


FUEL ASSEMBLY PRESSURE DROP VS. FLOW
TYPE "E" ORIFICE
Type VI Assembly W/ Type 1 Orifice (Data from 3 Assemblies)
After Mechanical Cleaning (All Types)
Type Assembly (New) Data Obtained 1967
Flow - GPM
0
20
40
60
80
100
120
140
160
180
0 1 2 3 4 5 6 7 8 9 10 11 12 13
Assembly Pressure Drop - PSI


FUEL ASSEMBLY PRESSURE DROP VS. FLOW
TYPE "D" ORIFICE
Type VI Assembly W/Type 2 Orifice (Data from 3 Assemblies)
After Mechanical Cleaning (All Types)
Type V Assembly (New) Data Obtained 1967
Flow - GPM
0
20
40
60
80
100
120
140
160
180
0 1 2 3 4 5 6 7 8 9 10 11 12 13
Assembly Pressure Drop - PSI

b. Fuel Inspection

1. Fuel Capscrew Inspection

The Type I fuel was inspected for missing capscrews using the underwater T.V. camera. This check was made prior to scheduling their removal from the core. The results agreed with the inspection performed during the 1967 outage with the exception of one assembly. It appears that an error was made in recording the data during the 1967 inspection.

The results of the inspection are shown below:

TYPE I FUEL CAPSCREW INSPECTION

Assembly	Core Location	Capscrews Missing
A-40	6325	B-2
A-111	7120	A-2, B-2
A-60	7410	B-2, A-3
A-82	5208	A-3, A-4, B-1, B-2
A-97	6905	A-3
A-64	5703	A-4
A-49	6724	B-4

2. General Electric Fuel Inspection

General Electric Company conducted inspection work on several selected fuel elements during the months of March and April. The inspection employed television, boroscope, ultrasonic, eddy currents and profilometry methods. The following fuel elements were inspected:

Assembly	Type	Description
G-42	III-F	A leak exposure
DU-100	V	A thin clad pilot assembly
A-465	I	Highest exposure
E-111	III-B	A lead exposure
DU-46	V	A lead exposure
G-3	III-F	A leaker

Assembly	Type	Description
G-103	III-F	A leaker
G-102	III-F	A leaker
G-99	III-F	A leaker
DU-45	V	Tubing manufactured by G. E.
DU-102	V	Zircaloy spacers
G-93	III-F	Revised rod cleaning process
E-120	III-F	Springless spacers
E-149	III-B	Revised autoclave process

The results of the inspection showed the assemblies to be in satisfactory mechanical condition for continued operation with the exception of the known leakers.

12. Generator No. 1 Overhaul, Inspection and Testing

The Unit No. 1 main turbine generator was overhauled, inspected and tested during the fifth partial refueling outage. The armature winding of the stator was inspected and appeared to be in excellent condition. The repair work on the armature included anchoring several loose slot wedges and varnishing portions of the stator end turns. Following the repair work, the armature was meggered and a high potential test performed on the armature winding of the stator. All tests were satisfactory.

The field rotor and winding was inspected and appeared to be in very good condition except for minor migration out of the block support on the rotor core.

The mica insulating blocks were replaced and several retaining wedges in the field winding slots were driven back into place and relocked in position. The field winding was meggered and satisfactory high potential tests were performed.

The generator was reassembled and an air leakage test was performed on the generator casing. The leakage was within the allowable limits. Following the air leakage test, the collector rings were machined, ground, and polished. The final stages of the work was then completed and the startup procedures were initiated.

13. Inspections

a. Reactor Vessel Inspection

On April 16, a visual inspection was made of the reactor vessel internals with all the fuel and blades removed from the core.

Typical areas observed during the inspections included; the hold down lugs and lifting eyes on the upper grid, the "T" handles on the hold down lugs for the bottom core support plate, the eight baton strips which hold the segments of the thermal shield, the poison sparger ring in the diffuser basket and the thimbles and guide tubes in the bottom of the vessel. From the conditions observed, no problems exist in the integrity of the vessel internals.

b. Reactor Flange Inspection

The reactor flange was dye penetrant and ultrasonically inspected by Pittsburgh Testing Laboratories on May 16, 1968. The tests were conducted the same as those initiated during the 1967 refueling outage inspections, using the same calibration standards and methods employed at that time. Testing was performed under the supervision of the Commonwealth Edison Company's Operational Analysis Department and was witnessed by representatives from the Atomic Energy Commission, Travelers Insurance Company, Babcock and Wilcox Company and the State of Illinois Boiler Inspection Division.

The ultrasonic and dye penetrant examinations were performed at the inside diameter and between the vessel stud holes on the cladding on the face of the flange. The dye penetrant inspection was conducted through two quandrants of the flange, whereas the ultrasonic inspection included an area 360° around the inside surface of the flange and at 15 various locations between the vessel stud holes. No indications of defects were found.

c. Reactor Flange Stud Bolts

In addition to the weld inspections, ultrasonic tests were conducted on the 56 vessel stud bolts by the Commonwealth Edison Operational Analysis Department. No indications of defects were found. Tests were made on April 2, 1968.

d. Reactor Thimble Weld Testing

Thirty-six of 40 control rod drive housing thimble welds were ultrasonically inspected by Pittsburgh Testing Laboratories following removal of the control rod drives for overhaul during the refueling outage. The inspections were conducted using the ultrasonic fixture and reference standard constructed for the 1967 refueling outage inspections. Of the 36 thimble welds ultrasonically inspected, no indications of discontinuities were found.

Ultrasonic testing of thimble J-welds was initiated during the 1967 refueling outage. A summary of the locations of the

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thimbles inspected during the 1967 and 1968 outages, including locations of the four sleeved thimbles not ultrasonically inspected, is shown in Figure 4.

e. Primary System Weld Inspections

The 1968 primary system stainless steel piping weld inspection on Dresden Unit No. 1 was completed during the refueling, maintenance and inspection outage, February 3 through June 2, 1968. This intensified inspection duplicated the ultrasonic testing of welded areas performed in 1967 with an increase in percentage of large plate pipe welds inspected, and a more extensive use of the dye check technique.

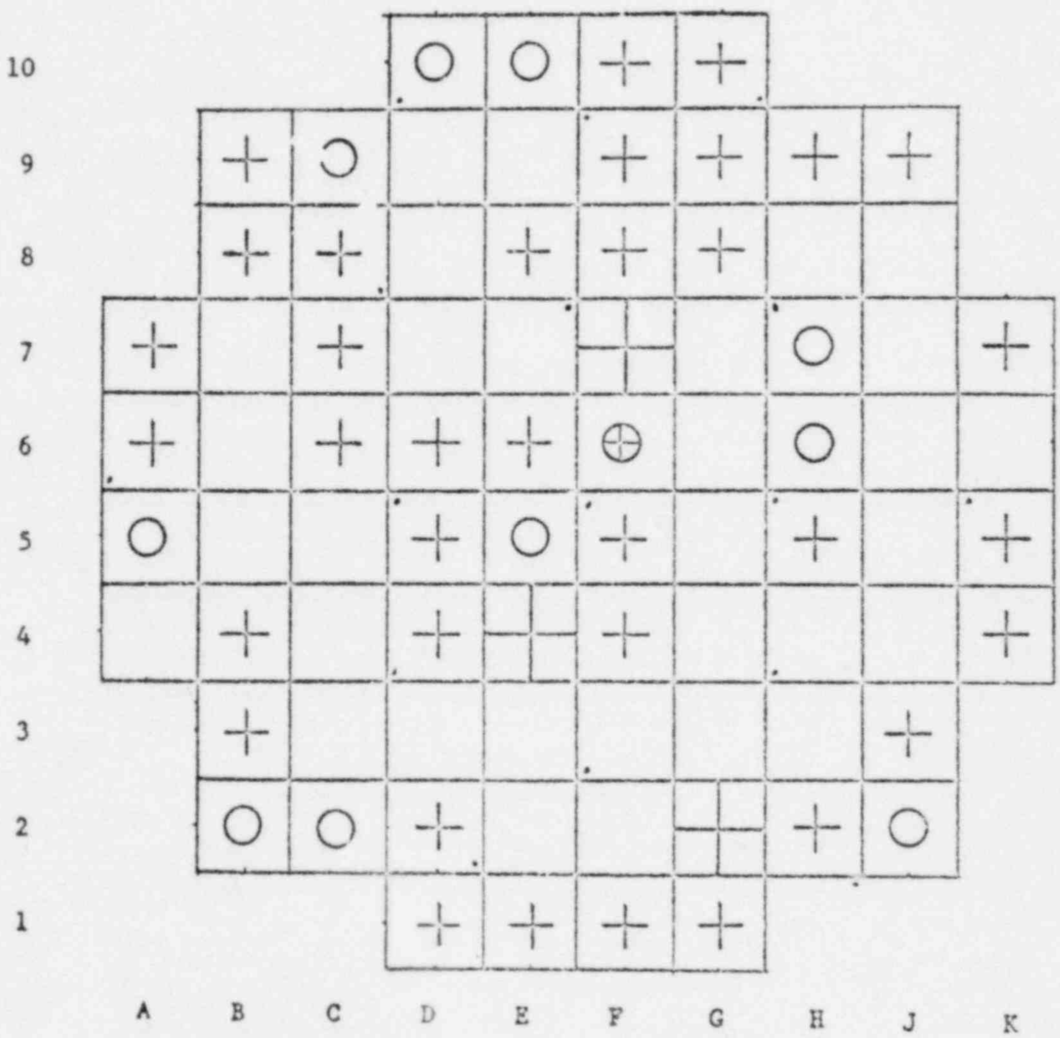
Dye penetrant checking revealed a number of minor surface imperfections. None of these were of significance to nuclear safety and none resulted from operation of the plant. These surface imperfections were largely due to undercutting and poor weld finishing techniques, and they were all removed by filing. None of these imperfections were deep enough for detection by ultrasonics. A total of 20 dye check indications were found and corrected.

Ultrasonic checking revealed a total of 13 indications in the piping in the range of 3% to 10% wall thickness in depth. A defect indication equal to or greater than 3% of the wall thickness and 1" in length, detected by ultrasonic techniques, was the basis for repair or replacement. This standard is described in Section III of the ASME Boiler & Pressure Vessel Code, Nuclear Vessels, in paragraph N-322. One weld in a section of 6" pipe ("B" unloading heat exchanger return line) had a defect indication of 10% of pipe thickness which extended approximately 4" in a longitudinal direction on the inside of the pipe. This same 20' section of pipe had three other defects and was replaced with 304 L stainless. The pipe containing the defects was sent to the General Electric Company for metallurgical examination.

In addition to replacing the 6" unloading heat exchanger discharge line, an 8" valve (MO-100) on the emergency condenser steam supply line was replaced. This was done because of ultrasonic indications noted in the valve during the 1967 refueling outage inspections. The replacement of this valve was made in accordance to the ASME pressure vessel codes. Radiographic and ultrasonic tests were conducted on the welds following installation. One small ultrasonic indication was detected. The indication was located at weld 223 on the body of the valve. It approached 3% of wall thickness near the outer surface of the valve, with very little detectable length to it. Since the discontinuity was well within the allowable code tolerance, no repairs were necessary.

FIGURE 4

CONTROL ROD DRIVE HOUSING
THIMBLE WELD INSPECTIONS
1967 - 1968



Ultrasonically Inspected, February - March, 1967

Ultrasonically Inspected, March, 1968

Sleeved Drive Supports - Not Tested

The 6" stainless steel bypass loop in "B" secondary steam generator compartment was also replaced, although no indications of defects were found in this line. The replacement was made because of problems encountered previously with 304 stainless steel in the other three loops. The bypass was replaced with 304 L stainless steel and was radiographic and ultrasonically tested prior to and following installation.

No other significant indications of defects were noted in the remainder of the piping and equipment inspections. A few defects of 10% of wall thickness or less were found on the exterior surface of the pipe. These were all repaired by filing, rewelding and dressing the pipe to full wall thickness.

Table 6 summarizes the scope of the inspection as compared to the 1967 inspection.

The integrity of the entire primary system was confirmed by a 1205 psig hydrostatic test following replacements and repairs to the primary system piping. Mr. James Longbucco, of Travelers Insurance Company, witnessed the test on the system.

f. Emergency Condenser

The emergency condenser was inspected during May by climbing down the vent. The interior of the condenser was found in satisfactory condition with no measurable amount of deterioration since the last inspection.

g. Turbine Crossunder/Extraction Piping Inspection

Two windows were cut in the north and south crossunders to inspect for piping erosion during the February - June refueling and maintenance outage. There appeared to be little or no change in condition of the pipe from the 1967 inspection. It should be noted that the unit was in operation only about seven months between inspections. A hole drilled in an eroded area showed remaining metal was 11/16" thick. A ten inch square area in a reducer was padded with stainless steel weld.

A window was also cut in the "D" extraction 16 inch piping which was replaced in 1967. No erosion was observed in either the copper bearing (0.3% to 0.5%) steel or the 2 1/4% chrome, 1% molybdenum alloy.

The pipe test segment which was installed in 1967 was removed for inspection. None of the six materials in the test segment exhibited any significant erosion. Extraction valves on "B" and "C" extractions were inspected through windows cut into extraction lines. All valves appeared to be okay.

TABLE 6

ULTRASONIC INSPECTION OF
PRIMARY SYSTEM PIPING WELDS

Welded Plate

	Total Welds		1968 Weld Inspections		Previous Weld Inspections*	
	Field	Shop	Field	Shop	Field	Shop
Risers (16")	56	36	14	9	4	0
Downcomers (16")	50	20	13	5	5	0
Suction Piping (22")	38	32	9	9	9	2
Return Piping (18")	32	22	8	6	5	2
Return Piping (22")	4	4	1	0	4	0

* All ultrasonic inspections conducted during 1967.

Number of indications (By U.T.) during 1968 = 0.

Seamless (10" and Under)

Size	Total Welds	1968 Weld Inspections	Previous Weld* Inspections	10/67 Total Pipe Length Inspection (% of Pipe Length)
10 "	12	12	10	--
8 "	29	29	29	97
6 "	96	96	126 **	94
4 "	60	60	60	68
3 "	61	61	61	32
2 "	34	29	23	65
1½"	15	10	10	--

* All ultrasonic inspections conducted during 1967, except for 1965-66 inspections of the six-inch bypass loops.

Number of indications (By U.T.) during 1968 = 13.

**All 96 welds were inspected during 1967.
30 of the 96 were inspected during 1965 and 1966.

14. Tests

a. Shutdown Margin Checks

Two shutdown margin checks were performed at the end of Cycle V on February 17. The results, compared to the beginning of Cycle V, are exhibited in Figure 5. The north edge of the reactor has a shutdown margin in excess of 1.7%, which is greater than it was at the beginning of Cycle V.

Reactor shutdown margin checks were again conducted on May 9, 1968, to demonstrate that the refueled core met license requirements with regard to the "stuck rod" criterion and that the margin is in excess of one percent throughout the core. The reactivity worths of the control rods used during the shutdown margin checks are exhibited in Figure 6. The margins were found to be in excess of 1.2% on the periphery of the core and in excess of 1.3% in the center of the core.

b. Fuel Sipping and Leaker Detection

On February 10, 1968, eight days after the reactor was shutdown for refueling and overhaul, a program of sipping individual fuel assemblies in their Cycle V locations within the reactor vessel was begun. The detection and location of defective fuel assemblies took precedence over all activities involving fuel handling at the reactor. The general order of sipping and number sipped are indicated below. The Type I, SA-1 and PF-10 assemblies, not sipped at the canal, were sipped in the fuel building.

SIPPING AT THE CANAL

	Number in Core	Number Sipped
Type I	66	59
Type III-B (Installed BOC 3)	94	94
Type III-B (Installed BOC 4)	96	96
Type III-F (Installed BOC 4)	100	100
Type V (Installed BOC 5)	106	74
TOTAL SIPPED		423

Four defective fuel assemblies were identified by sipping at the canal. The location of the defective assemblies is shown in Figure 7. A summary of the defective assemblies, including exposure is tabulated on the following page. Three of the four assemblies are powdered, bringing the total defective powdered assemblies, to date, to five out of ten. The ten powdered assemblies were installed in the core at the beginning of Cycle IV (BOC 4), May, 1965.

SUMMARY OF DEFECTIVE FUELS (EOC 5)

Element	Cycle V Location	Exposure (EOC 5)	Type of Construction
G-3	6112	10,740	Pellet
G-99	7112	11,252	Powder
G-102	6718	11,693	Powder
G-103	5908	11,439	Powder

Seven Type I assemblies and two special General Electric assemblies were later sipped in the fuel building. The Type I's were not sipped at the reactor because each had one or more bale capscrews missing and it was thought inadvisable to handle them at the core. The special assemblies were sipped in the fuel building at General Electric's request. None of the nine assemblies sipped indicated leakage.

c. Critical Testing of Refueled Core

The whole core was critically tested on May 9, 1968. The control rod withdrawal sequence, 5C, was used to attain a critical state after the partial withdrawal of the 17th rod. Criticals attained during this and previous refuelings are summarized in the following tabulation.

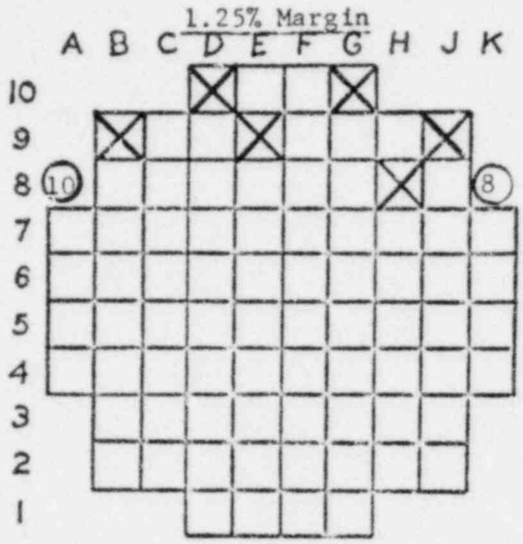
INITIAL CRITICAL AND REFUELING DATA SUMMARY

Fuel Cycle	Date	Critical Rods Withdrawn	Core Fraction Refueled	Fuel Type Added	Minimum Critical Size
1	3/30/60	18 1/2	5/5*	I	28
2	3/ 8/63	36 1/2	2/5	I - II	28
3	6/ 7/64	43 1/2	1/5	III-B	24
4	4/27/65	20 1/2	2/5	III-B - III-F	16
5	2/20/67	30 1/2	1/5	V	24
6	5/ 9/68	17	1/5	VI Unpoisoned (Instrumented)	14

* Initial loading - 448 assemblies

FIGURE 5

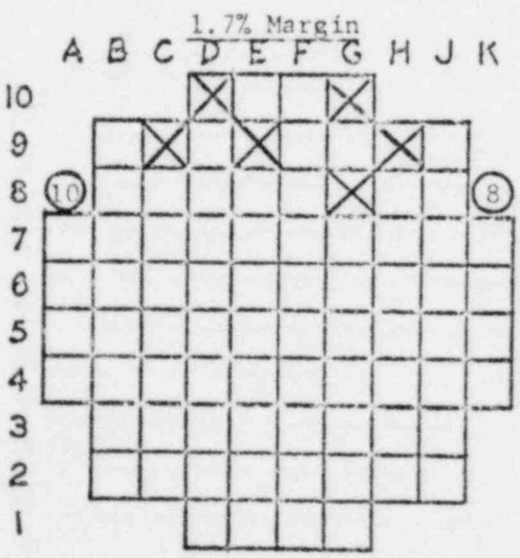
END OF CYCLE V SHUTDOWN MARGIN CHECKS



C	8	10	
M	286	71	4/1/67 (BOC 5)

C	8	10	
M	1.4	1.63	2/17/68 (EOC 5)

☒ Rod Withdrawn



C	8	10	
M	24	6.7	4/2/67 (BOC 5)

C	8	10	
M	1.2	1.14	2/17/68 (EOC 5)

☒ Rod Withdrawn

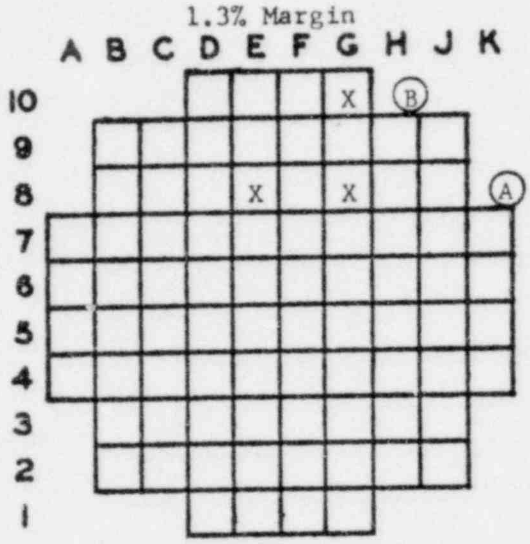
C = Chamber
M = Multiplication

BOC 5 = Beginning of Cycle V
EOC 5 = End of Cycle V

FIGURE 6

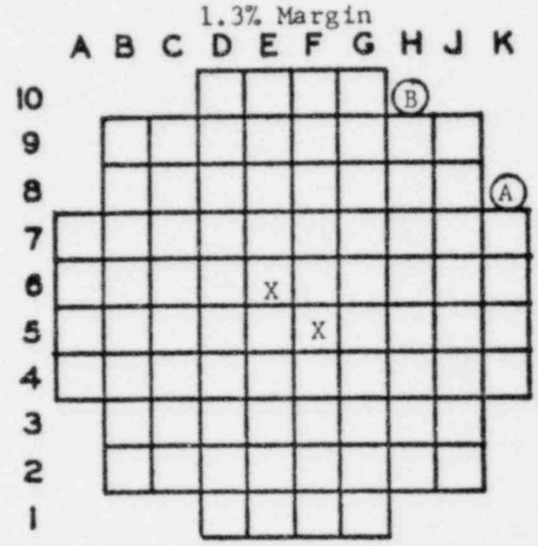
SHUTDOWN MARGIN CHECKS
5/9/68

1
1.3% Margin



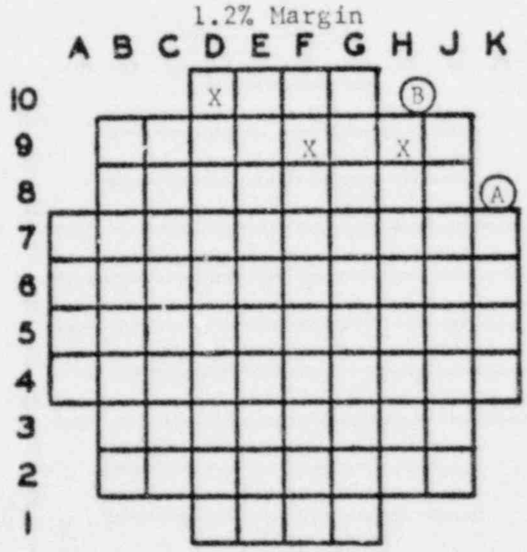
M 8 = 1.03
M10 = 1.00
M A = 1.13
M B = 2.5

2
1.3% Margin



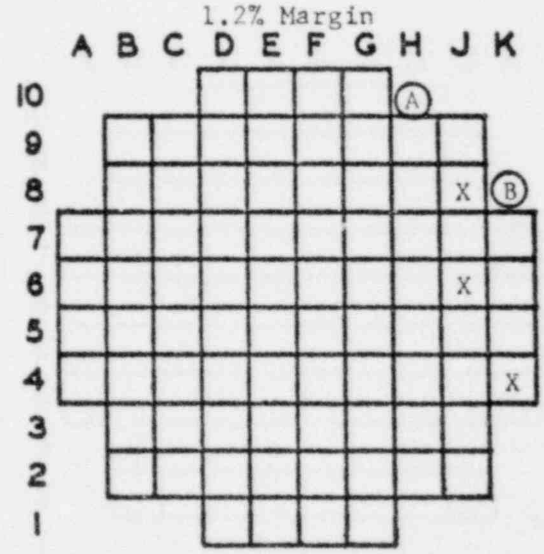
M 8 = +1.00
M10 = 1.00
M A = 1.00
M B = 1.00

3
1.2% Margin



M 8 = 1.04
M10 = 1.00
M A = 1.17
M B = 1.91

4
1.2% Margin



M 8 = 1.00
M10 = 1.11
M A = 1.06
M B = 1.21

$\boxed{X}$ = Indicates Rod Full Out

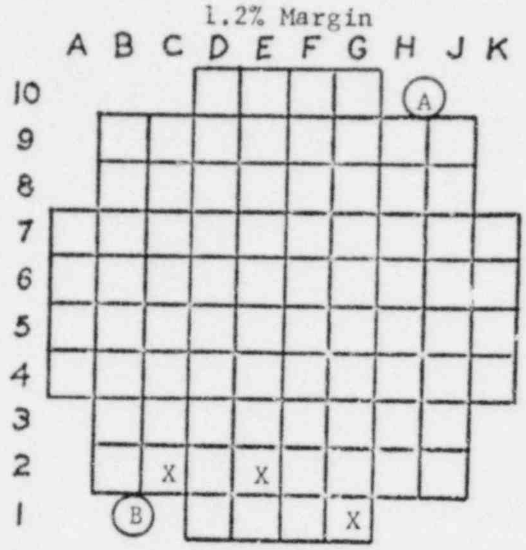
A = In-Vessel Fission Chamber Locations

B "

FIGURE 6A

SHUTDOWN MARGIN CHECKS
5/9/68

5
1.2% Margin



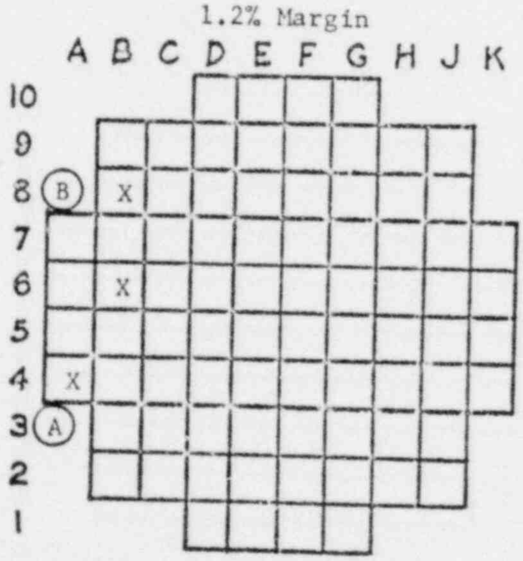
M 8 = 1.00
M10 = 1.00
M A = 1.55
M B = 2.54

X = Indicates Rod Full Out

A = In-Vessel Fission Chamber Locations

B = In-Vessel Fission Chamber Locations

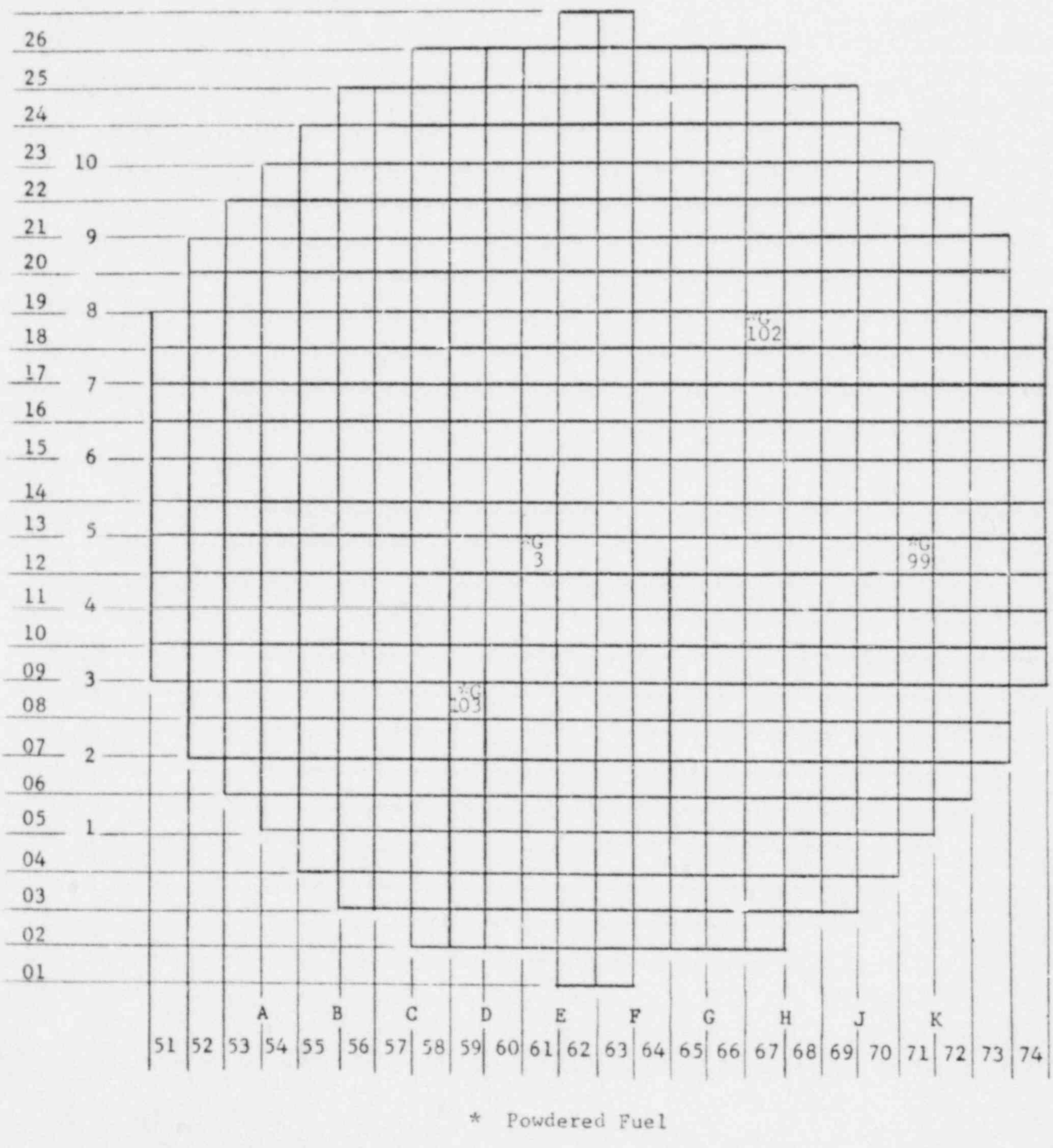
6
1.2% Margin



M 8 = 1.00
M10 = 1.11
M A = 1.06
M B = 1.21

FIGURE 7

DEFECTIVE ASSEMBLY LOCATION (EOC 5)



* Powdered Fuel

d. Control Rod Calibration

Immediately following initial criticality, a control rod calibration was performed on control rod B-7. The reactivity worth of rod B-7 was determined to be .3% $\frac{\Delta K}{K}$. The integral worth curve and the state conditions are exhibited in Figure 8. The procedure followed and the basic data obtained during the calibration is exhibited in Table 7.

e. Temperature Coefficient of Reactivity Measurements

The moderator temperature coefficient of reactivity as a function of moderator temperature was measured during the initial startup, May 30 - June 1, and is exhibited in Figure 9. The rod patterns associated with these measurements are exhibited in Figure 10. The temperature coefficient is negative at all temperatures of interest.

f. Control Rod Zones of Influence Tests

Control rod zones of influence tests were performed June 5, 1968, on two control rods, E-10 in the periphery, and D-6 in the central region of the core. These tests were performed to demonstrate compliance with AEC regulations, which specify that if two adjacent control rods are withdrawn to demonstrate a flux distortion, this distortion must be detected by at least two incore monitors. The test results showed that there is a response from at least the four incore monitors nearest the withdrawn rod for any single control rod movement.

g. Primary Steam Drum Safety Valves

The five spare primary safety valves were installed on the primary steam drum on February 24, thus replacing five of the safety valves previously in service. All five safety valves were tested for relief pressure on August 14, 1967, and were set at their respective design pressures $\pm$ 10 psi. Relief pressures were checked by repeated popping. The valves installed were cleaned, set and leak-checked in the shop facility.

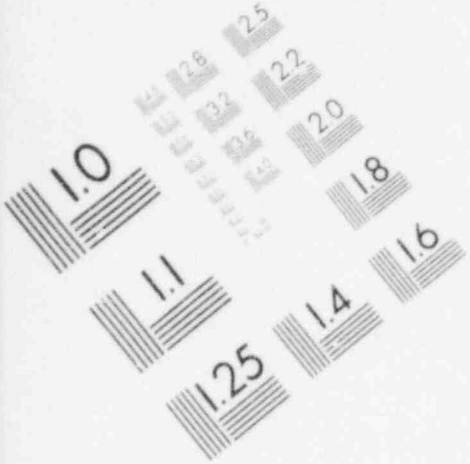
h. Air Locks

All air locks, ventilating valves, and process isolation valves were tested periodically during the year and found to be within the licensed allowable leakage rate.

i. General Electric Extraction Steam Quality Test

During August, General Electric Company installed test equipment and performed tests to determine the feasibility of using radioactive tracers to calculate flow through a turbine extraction line. The extraction flow tests were conducted on August 19 through August 23, on "A" and "E" extraction lines. The flow was determined by injecting a sodium-24 tracer of known concentration into the extraction line, sampling it down stream, and subsequently, conducting a tracer balance from which wet steam quality and heater heat balances could be calculated.

Preliminary results of the tests compare favorably with previous turbine performance tests. The results show that the method is feasible and General Electric has indicated they will continue work in this area.



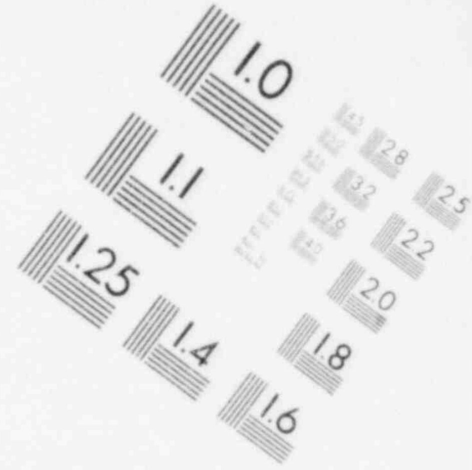
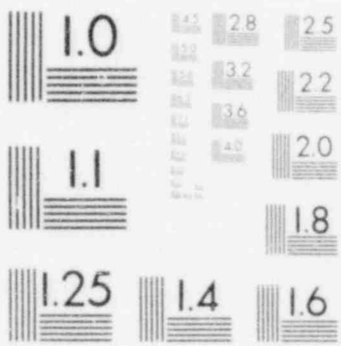
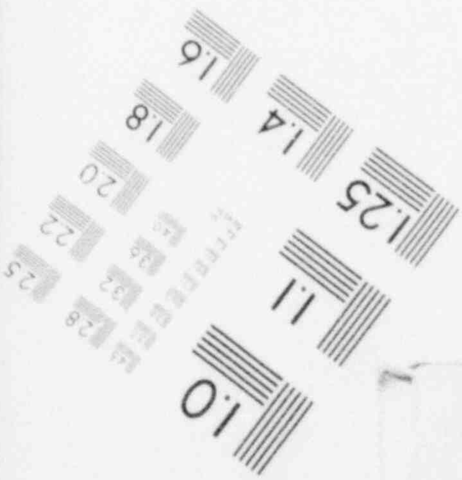


IMAGE EVALUATION
TEST TARGET (MT-3)



6"

MICROCOPY RESOLUTION TEST CHART



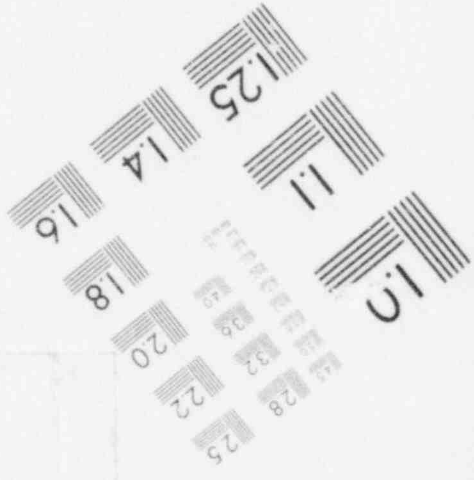
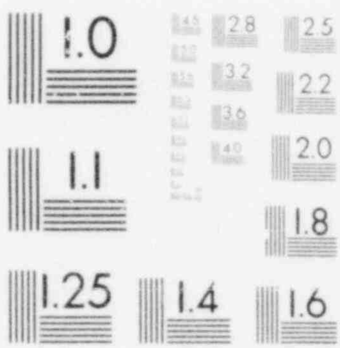


IMAGE EVALUATION
TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART

FIGURE 8

CONTROL ROD CALIBRATION

Rod Calibrated B-7, 5/30/68

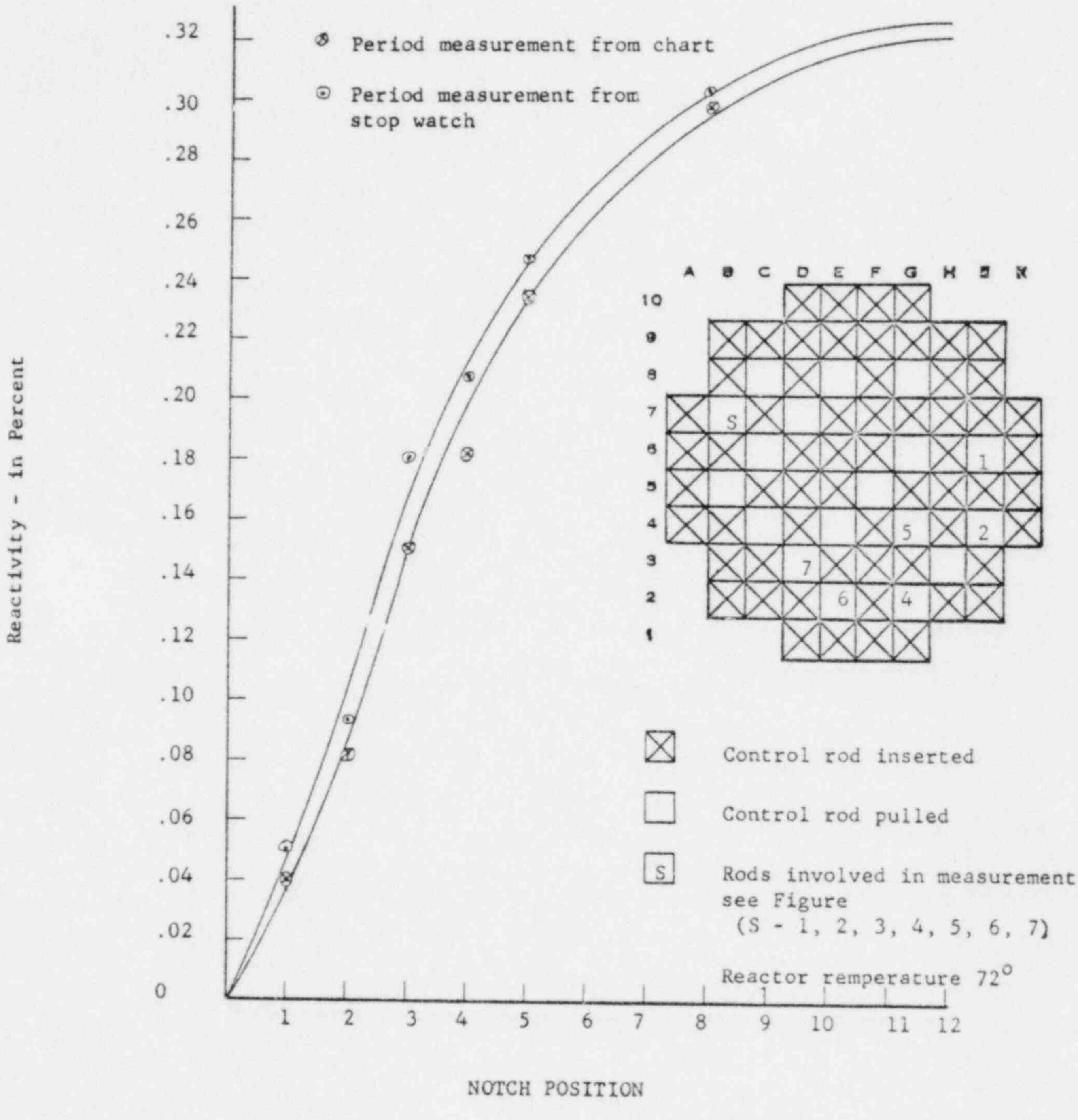


TABLE 7

CONTROL ROD CALIBRATION

Rod Calibrated B-7, 5/30/68

	Control Rod Position								Period		Reactivity Added	
Step	B-7	J-6	J-4	H-3	G-2	G-4	E-2	D-3	Stop Watch	Chart	Stop Watch	Chart
0	0	12	12	12	12	12	12	12				
1	1	12	12						135	176	.04977	.03988
2	1	2	12								.0	
3	2	2	12						160	163	.0424	.04256
4	2	0	4									
5	3	0	4						65	92	.08730	.06748
6	3		0	6								
7	4		0	6					280	230	.0[illegible]57	.03164
8	4		0	0	0	4						
9	5		0			4			173	126	.04047	.05265
10	5		0			0	2					
11	8		0			0	2		118	101	.05551	.06278
12	8		0			1	0	3				
13	12					1	0	3				

FIGURE 9

MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY

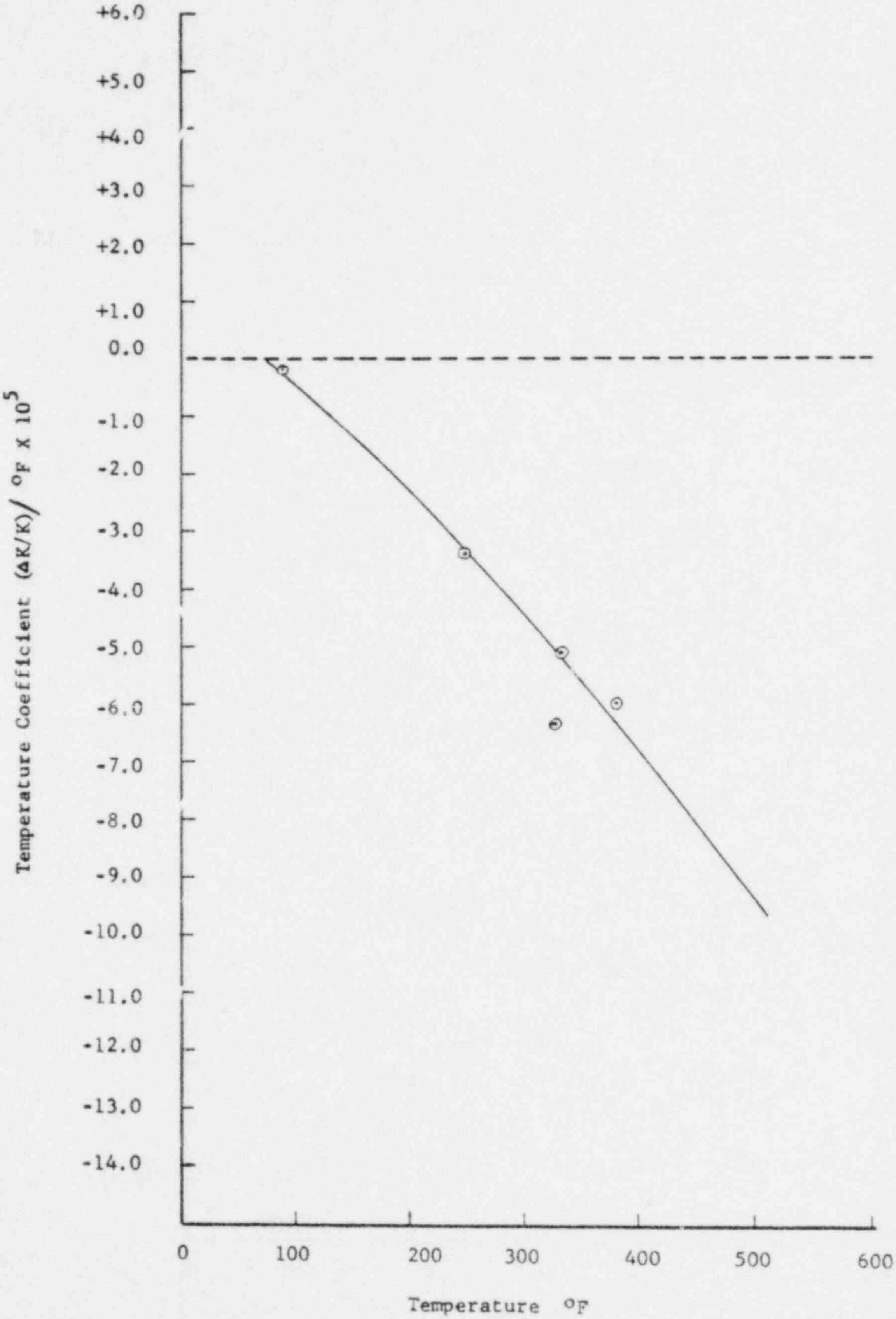
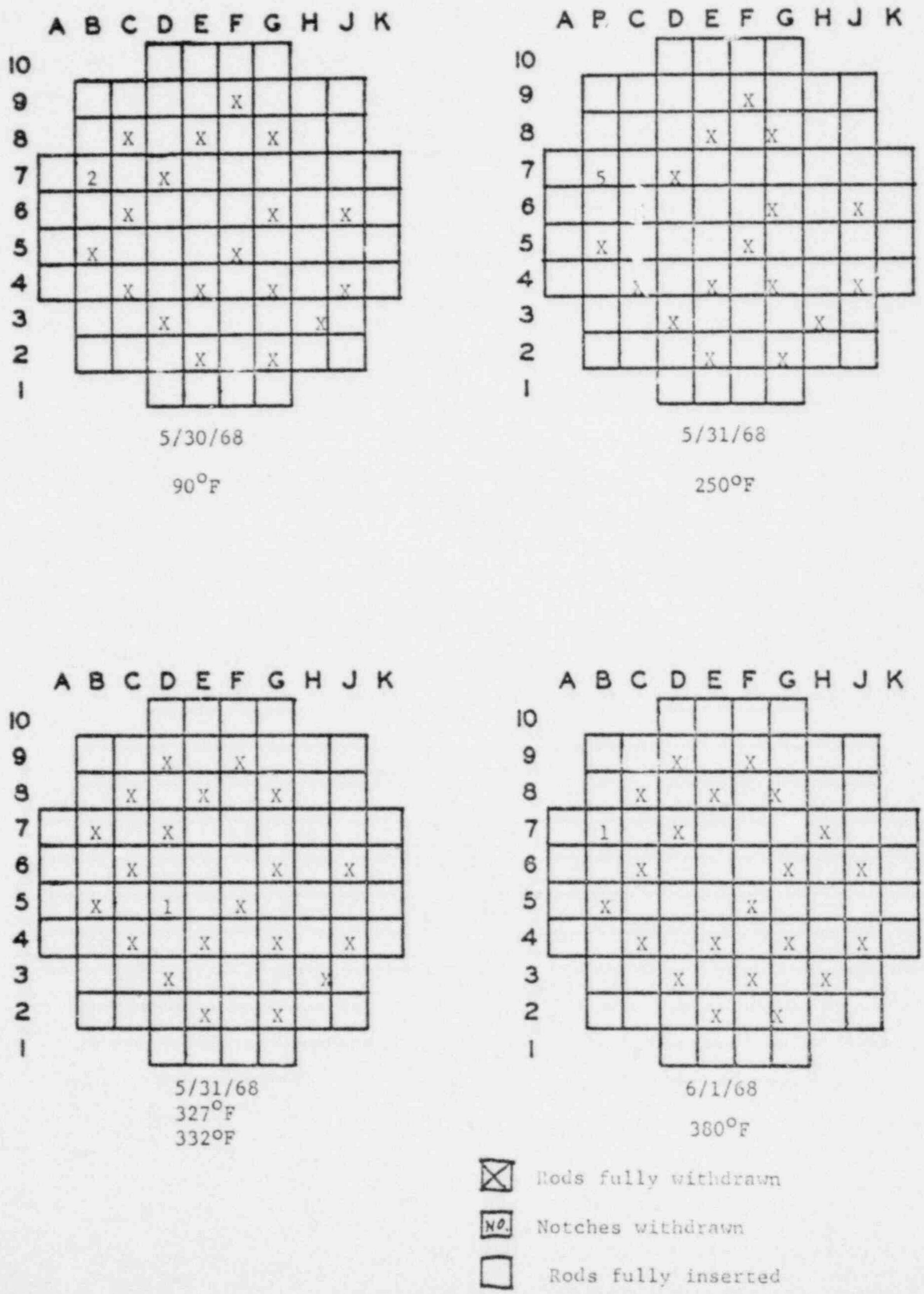


FIGURE 10

Moderator Temperature Coefficient Tests



B. License DPR-2

Table 8 lists the amendments to our license requested and/or authorized during the year. Pertinent correspondence pertaining to these requests are listed in the Correspondence References.

TABLE 8

SUMMARY OF LICENSE AMENDMENTS
PENDING DURING 1968

	Date	
	Request	Authorization
Request to amend License DPR-2 to permit operation with 96 Type VI fuel assemblies. (Change No. 14)	9/14/67	4/22/68
Request to amend License DPR-2 to authorize elimination of the cocked control rod during fuel assembly additions and impose a condition requiring withdrawal and reinsertion of a control rod in the vicinity of the core position being refueled before and after each fuel assembly is inserted into the core. (Change No. 15)	1/17/68	5/10/68
Request to amend License DPR-2 to authorize replacement of one of the gadolinia-urania poison fuel rods with unirradiated gadolinia-alumina poison rods in two of the Type V high gadolinia fuel assemblies. (Change No. 16)	5/ 9/68	5/17/68
Request to amend License DPR-2 to permit the installation of the Emergency Core Cooling System. (Change No. 17)	10/10/68	

Correspondence References - 1968

(1) Letter to AEC dated January 17, 1968, submitting additional information and analysis pertinent to Change No. 14.

(2) Letter to Commonwealth Edison Company dated April 22, 1968, authorizing Change No. 14 to the Operating License DPR-2.

(3) Letter to AEC dated January 17, 1968, applying for an amendment to authorize elimination of the cocked rod during fuel assemble additions. (Change No. 15)

(4) Letter to Commonwealth Edison Company dated May 10, 1968, authorizing Change No. 15 to the Operating License DPR-2.

(5) Letter to AEC dated May 9, 1968, requesting authorization to replace one gadolinia-urania rod from each of two exposed Type V high gadolinia assemblies with unirradiated gadolinia-alumina rods. (Change No. 16)

(6) Letter to Commonwealth Edison Company dated May 17, 1968, authorizing Change No. 16 to the Operating License DPR-2.

(7) Letter to Commonwealth Edison Company dated May 29, 1968, regarding non-compliance with Sections D5b and E3 of Appendix A to the Operating License DPR-2.

(8) Letter to AEC dated June 15, 1968, concerning facts of non-compliance with Sections D5b and E3 of Appendix A to the Operating License DPR-2.

(9) Letter to AEC dated October 10, 1968, requesting amendment to Licence DPR-2 to permit the installation of the Emergency Core Cooling System. (Change No. 17)