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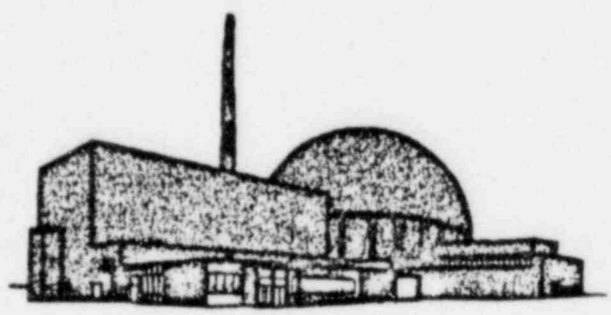
ANNUAL REPORT

YEAR 1967



DRESDEN NUCLEAR POWER STATION

Commonwealth Edison Company

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION

ANNUAL REPORT OF STATION OPERATION
FOR THE YEAR 1967

January 23, 1968

DRESDEN NUCLEAR POWER STATION ANNUAL REPORT

I. INTRODUCTION

This sixth annual report is submitted in compliance with paragraph 3.2 (2) of Utilization Facility License DPR-2, as amended, and covers operation of Dresden Nuclear Power Station during the year 1967.

II. SUMMARY OF OPERATIONS

A. Scope of Operations

Operation of the plant continued from the preceding year until January 13, 1967, when it was shutdown until May 29, 1967, for the fourth partial reactor refueling and high pressure turbine inspection. During this outage, considerable effort was devoted to fuel cleaning and testing and primary system piping inspections.

The plant continued operation through December 31, 1967, with a total of nine shutdowns during the year. Additions to and changes in facility design were made by: addition of an electro-chemical test flange in "B" reactor recirculation loop; installation of an off-gas filter in the stack; installation of an air operated valve in the off-gas drain line; installation of an area radiation monitor in Environs Station No. 2; relocation of the Clay Products Environs Station; discontinuance of samples at the Hansel and Breen Environs Stations; modifications to the fuel building and reactor canal crane controls; additions of mechanical stops to the fuel handling grapples; and erection of a concrete block wall in the fuel building.

A total of six shipments, consisting of twenty-one spent fuel assemblies, were shipped for the U.S.A.E.C. (Savannah River Operations Office) to the Chemical Processing Plant of Nuclear Fuel Services, Inc. at West Valley, New York.

B. Shutdowns

The plant was shutdown nine times during the year as shown in Table 1. Three of these were forced outages: one due to a fault on a phase potential transformer and eventual generator inspection when the unit was synchronized out of phase with the system; another, due to a manual reactor scram, caused by the tripping of the primary feedwater pumps due to low suction pressure; and the last, due to turbine secondary control valve vibrations during startup leading to repairs.

There were six scheduled outages: the first outage was for the fourth partial refueling, high pressure turbine inspection, fuel assembly cleaning and testing, and primary system piping inspections; the second was for a turbine overspeed test; the third for feedwater heater tube leaks; the fourth for the Unit 3 primary containment overpressure test and A.E.C. License examinations; the fifth for reactor operator training, and the sixth for 138 KV switchyard modifications and A.E.C. License examinations, which was extended into a forced outage for primary system piping inspections.

C. Load Restrictions

The load restrictions imposed during the year are listed in Table 2. Restrictions were due to fuel depletion, incore stabilization and calibration, condensate demineralizer inspection and regeneration, feedwater heater outages, and reactor recirculation loop outages.

TABLE 1

OPERATING PERFORMANCE 1967

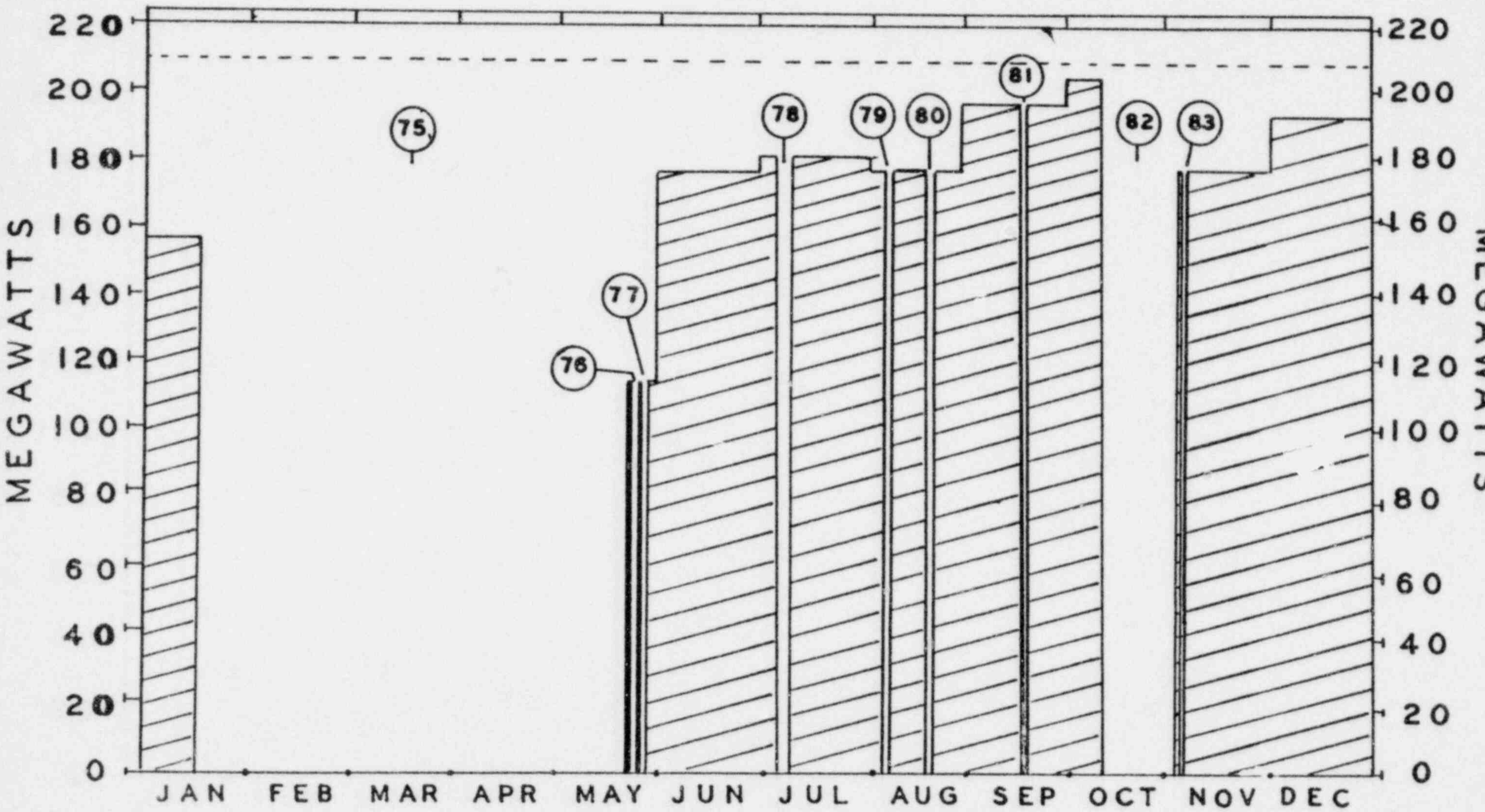
No. Of	Off System		On System			
Outage	Date	Time	Date	Time	Outage Hours	Reason For Outage
75	1/13/67	9:37 p.m.	5/23/67	7:00 p.m.	3,116 Hrs. 23 Min.	Fourth partial refueling outage and high pressure turbine inspection (1,849 Hrs. 24 Min.); fuel cleaning (441 Hrs. 59 Min.); and primary system piping inspection (825 Hrs. 0 Min.).
76	5/23/67	8:31 p.m.	5/28/67	9:06 p.m.	120 Hrs. 35 Min.	Potential transformer failure.
77	5/29/67	1:13 a.m.	5/29/67	1:51 a.m.	0 Hrs. 38 Min.	Turbine overspeed test.
78	7/ 5/67	5:18 p.m.	7/ 9/67	3:29 p.m.	94 Hrs. 11 Min.	Repair leaks in "C" primary feedwater heater.
79	8/ 4/67	6:51 a.m.	8/ 6/67	5:02 a.m.	46 Hrs. 11 Min.	Unit No. 3 containment over-pressure test and reactor operator licensing.
80	8/18/67	4:25 a.m.	8/20/67	10:25 a.m.	54 Hrs. 0 Min.	Reactor scram - low suction to the primary feedwater pumps.
81	9/17/67	1:53 a.m.	9/18/67	1:46 a.m.	23 Hrs. 53 Min.	Operator training.
82	10/11/67	5:25 p.m.	11/ 4/67	8:56 p.m.	580 Hrs. 31 Min.	138 KV switchyard modifications and reactor operator licensing and training (199 Hrs. 36 Min.) and primary system piping inspections (380 Hrs. 55 Min.).
83	11/ 5/67	4:47 a.m.	11/ 6/67	8:24 a.m.	27 Hrs. 37 Min.	Turbine-generator secondary control valve repairs.
TOTAL OUTAGE HOURS FOR YEAR					4,063 Hrs. 59 Min.	

TABLE 2

LOAD RESTRICTIONS FOR 1967

Date	Reduction From Maximum Capability of 210 MWe	Condition
January 1 - January 13	50	Fuel Depletion
May 23	100	Incore Stabilization
May 28 - May 29	100	Incore Stabilization
May 29 - June 2	100	Incore Calibration
June 2 - June 7	50	Incore Calibration
June 7 - June 16	20	Incore Calibration
June 19	10	Condensate Demineralizer Inspections
July 1 - July 5	18	"C" Primary Heater Tube Leaks
July 29	18	"E" Primary and Secondary Heaters (Union and Vent Repairs)
August 20 - August 25	18	"E" Primary Heater Diaphragm Repair
September 19 - September 22	18	"E" Primary Heater Drain Valve Repair
November 4 - November 5	25	"D" Reactor Recirculating Loop Out of Service
November 6 - December 12	25	"D" Reactor Recirculating Loop Out of Service
December 15	60	Condensate Demineralizer Regeneration
December 17 - December 19	18	"A" Heater Extraction Valve Diaphragm Repairs
December 22	60	Condensate Demineralizer Regeneration
December 25	60	Condensate Demineralizer Regeneration
December 28	60	Condensate Demineralizer Regeneration

PLANT ELECTRICAL LOADING
YEAR 1967
DRESDEN NUCLEAR POWER STATION



III. DISCUSSION

A. Operating Experience

1. Generation

The total reactor operating (critical) time during the year was 4,905 hours and the total power for the period was 115,362.5 MWDt. The gross electrical generation during the year was 853.566.53 MWHe; net generation was 807,028.4 MWHe. As of December 31, 1967, the total gross generation since commencement of power operation, April 15, 1960, was 7,454,328.04 MWHe.

2. Scrams

a. At 1:15 p.m., on August 4, a reactor scram was initiated by a spurious signal from the reactor out-of-core power range monitors system. The scram was due to a high neutron flux indication on Channel 4 out-of-core while the reactor was subcritical. The spurious signal was caused by maintenance personnel working in the cableway.

b. On August 4, at 5:50 p.m., while conducting operator training criticals, the reactor was scrammed by moving the reactor control switch from "start" to "shutdown", instead of from "start" to "refuel". Fifty-two rods were withdrawn prior to the scram.

c. At 3:55 p.m., on August 5, while pulling a critical, Control Rod C-2 was withdrawn from Position 1 to Position 2. It subsequently went past Notch 2 and latched at Position 3, causing a high neutron flux on Channel 16. Forty-five rods were withdrawn prior to the scram.

d. At 4:25 a.m., on August 18, the reactor was manually scrammed after loss of suction pressure to the primary feedwater pumps. Investigation revealed a flow control valve, FCV-357, on the condensate recirculation line, to be leaking through to the condenser. The valve was repaired and the unit returned to service.

e. At 11:30 a.m., on October 20, the reactor scrammed due to a drop in condenser vacuum below 23" Hg with greater than 200 psi on the primary system. The scram occurred while heating the system with the reactor control switch in the "start" position and a condenser vacuum of 21" Hg.

f. At 12:36 p.m., on November 4, the reactor scrammed due to high neutron flux. The reactor was at rated temperature and pressure with forty-two control rods withdrawn. Preparations were being made for the turbine roll. The primary drum level was low and the control operator had opened the motor operated valve in the primary feedwater line. He then commenced to open the primary feedwater regulating station to bring up his drum level. No response was noticed to the initial loading pressur , so the operator began to decrease the loading signal to the valve, but Channels 1 to 6 spiked upward beyond 120% of power on the 70 MWt range, and the reactor scrammed on high flux on Channels 2 and 3. Investigation revealed the primary feedwater regulating valve stuck in the closed position.

g. At 1:55 p.m., on November 4, the reactor scrammed due to low vacuum and 200 psi on the primary system. The reactor inlet temperature was 390° F and was just brought critical. Rod verification was in process. The turbine gland seal regulator steam was being bypassed to maintain gland sealing steam on the turbine glands. An increase in the gland sealing steam pressure caused the gland seal dump to the 23rd stage of the turbine to open. The pilot valve regulating the operation of the regulator stuck holding the gland dump valve open. Seals were lost on the turbine and the reactor scrammed at 23" Hg.

h. At 1:13 a.m., on November 6, the reactor scrammed due to low vacuum and 200 psi on the primary system. The scram occurred while heating the system, with the reactor at 390° F. The mechanical pump was in service with the recirculation valve partially open. The shift operators were about to put an evactor in service, but were completing work in checking out the turbine before cutting in the evactor. The reactor scrammed during this period with a condenser vacuum of 21" Hg.

3. Incidents

On February 1, 1967, the fuel building control circuit experienced two grounds, which shorted out the "raise" circuit and caused an uncontrolled, continuous upward drive of the grapple hoist. The grapple held no load and no personnel radiation exposures resulted; however, the hoist was being used at the time to channel new fuel elements.

The prescribed pre-startup checkout had been successfully performed prior to the hoist being used to confirm the operability of all control and safety functions, but the two grounds in the circuit bypassed both control and safety functions and caused the grapple hoist to operate in its upward movement at maximum speed. An operator opened a nearby main power disconnect to stall the hoist; however, by then, the empty hoist had already reached its maximum up-travel.

To prevent recurrence of a similar incident, mechanical stops and additional electrical stops were added to both the fuel building and reactor canal fuel handling grapples. These are further discussed under "Changes to Facility Design" in this report.

4. Control Rod Drives

a. Control Rod Drive Operation

1. On May 22, while conducting reactor criticality tests, Control Rod B-3 drifted out from Position 6 to Position 7. Several attempts were made to lock the drive at the fully inserted position, but it drifted out in each instance. The drive was subsequently scrammed in, locked, and deactivated. The malfunction was attributed to foreign material buildup between the collet and shuttle piston.

   On October 11, prior to Unit I shutdown for electrical yard work, the Asco Valve on Drive B-3 was adjusted and the drive exercised and flushed. The drive functioned normally on all following operations and was returned to service.

2. During control rod worth tests, July 23, Control Rod K-4 was inserted from Position 12 to Position 6. When the control switch was released, the drive drifted out to Position 12. The drive was inserted to Position 5 and drifted out to Position 12 again. Exercise and flushing freed the collet assembly and shuttle piston and normal latching was accomplished in all subsequent operations. The malfunction is attributed to foreign material buildup within the drive collet assembly.

   On August 20, Drive K-4 failed to withdraw by normal operation. In order to withdraw the drive, it was given a quick insert signal to unlatch the collet, and the switch was then moved quickly back to withdraw. The drive was operated in this manner until it was to Position 0, where it was given an insert signal to unlatch and then a withdraw signal. The drive was then withdrawn satisfactorily. Drive K-4 has a history which indicates that some foreign material is preventing the shuttle piston from engaging the collet fingers and thus prevents the drive from withdrawing properly.

3. On July 27, Drive J-2 was inserted from Position 12 to Position 11 and then withdrawn to Position 12. When the control switch was released, the drive inserted rapidly to Position 0. The drive was withdrawn to Position 5 and as soon as the control switch was released, the drive inserted to zero. Further operations with the drive were all successful.

   Similar experiences of Drive J-2 inserting occurred on August 4, August 14, October 1, and October 8. On each of these occurrences, Drive J-2 inserted following previous movements. Further exercising was accomplished with no problems. The cause for these malfunctions was considered to be due to the scram inlet valve leaking, as indicated by higher temperatures on the scram inlet line relative to others. On October 17, during shutdown, the scram inlet valve was repaired and Drive J-2 was returned to normal operation.

4. During routine timing tests on August 19, Drive J-9 was withdrawn from Position 0 to Position 12 and then re-inserted to Position 0. Without further operator action, the drive drifted out to Position 12. The drive was inserted to Position 0 and it again drifted to Position 12. This was repeated three additional times with the same results. The drive was then operated satisfactorily ten additional times. Foreign material buildup between the shuttle piston and collet assembly is attributed to the drive malfunction. The repeated operation of the drive flushed the foreign material from the area, and thus cleared the malfunction.

   On September 23, while moving Drive J-9 from Position 12 to 0, the drive stopped at Position 2 and would not move in or out. The flow orifices were opened wide and the drive check valve vibrated. Vibrating this check valve evidently caused the valve to reseat, with the resultant effect of stopping probable leakage and allowing for normal operation.

   During shutdown on November 5, Drive J-9 would not insert from Position 10. Pr + 200 was increased to 250 psi, but the drive would not move. Insertion of the drive was accomplished by scramming Accumulator 3. Further operation since insertion has been successful. Friction tests have indicated that this drive has worn seals, which would contribute to its failure to insert.

5. During control rod worth tests on September 23, Drive B-4 would not latch with any regularity after insertion to various notch positions. This resulted in the drive drifting to Position 12. The drive was exercised, flushed, and functioned normally in all other operations. The problem is attributed to foreign material between the shuttle piston and collet assembly.

6. On October 9, when the selector switch was moved to the withdraw position, the red withdraw light came on, but as Drives D-7, E-7, and D-5 were operated individually, each inserted until the selector switch was released. The Barksdale Valve was inspected and found to be working stiffly. It was taken out of service and the alternate Barksdale put in its place. On October 17, after repairs, the original Barksdale was placed back in service.

7. On October 14, during friction tests, Drive F-1 was withdrawn to Position 12. It was again inserted, but would not go beyond Position 11. Pressure was raised to 300 psi, but to no avail. The drive was then scrammed in. The actual failure to insert was attributed to worn seals, as indicated by the friction and timing tests. Increasing the Asco orifice opening permitted rod insertion with normal Pr + 200 pressure. The drive operated satisfactorily on all subsequent operations.

8. On November 5, while shutdown, Drive A-6 would not insert from Position 12. Pr + 200 was increased to 250 psi and the drive inserted to Position 6. Total insertion of the drive was accomplished by scramming Accumulator 26. The drive was exercised with all subsequent operations accomplished with no problems. The actual failure of A-6 to insert is attributed to worn seals, as indicated by previous friction tests.

9. On November 6, following a reactor scram, Drive H-7 could not be withdrawn. Varying pressures on Pr + 200 did not move the drive, so it was decided to wait until the system reached normal operating pressures and temperatures. Later the same day, with the system at normal operating pressure and temperature, Drive H-7 was withdrawn and has been successfully exercised each night thereafter. The drive tests on this drive show that seals and operation are normal. It was concluded that foreign material may have temporarily hung up in the collet assembly and cleared when the drive was exercised.

10. On November 7, Drive G-9 position indicator showed the drive to be at Position 12, but by all instrument responses, the drive appeared to be at Position 0. Investigation revealed the instrument probe to be malfunctioning. The drive was deactivated and is awaiting repairs.

11. On December 30, while inserting Drive H-4 from Position 10 to 9, the drive continued to insert to Position 1. The drive was then inserted to Position 0 and given two one minute flushes. After the flush, H-4 was stepped out to Position 10. Since this malfunction, Drive H-4 has functioned normally. The problem is attributed to either the Barksdale Valve or scram inlet line leaking through to the drive.

b. <u>Control Rod Drive Tests</u>

On January 14, all control rod drives, except SN 1290 (Core Position C-9) and SN 1272 (Core Position J-2), were scram and friction tested and timed for normal insertion and withdrawal. The two drives not tested were considered stuck during Cycle IV and were valved out of service during most of that cycle. Both drives, after overhaul, were returned to service during the fourth partial refueling outage. Scram, friction, and timing tests were also conducted on March 31 and October 13 and 14. The data obtained from all the above tests were satisfactory on all drives, although seal wear was evident on five drives during the October tests. These drives will be removed from the reactor for inspection during the 1968 refueling outage.

c. Inspection

Eleven control rod drives were removed and replaced with repaired drives during the fourth partial refueling outage. These drives had exhibited abnormalities during Cycle IV.

Prior to shutdown for refueling, twelve drives were selected for removal. After shutdown, some difficulty was encountered in removing the guide tubes from B-8, G-1 and D-1 cells. This necessitated revising the original list of drives to be inspected. The following table shows the eleven drives removed, reason for selection, and three drives which were not removed because of difficulty in pulling the guide tube.

Serial Number	Drive	Reasons for Removal
1290	C-9	In attempting to withdraw C-9 from Position 9 on June 11, 1965, the rod inserted to Position 8 and remained there. Further withdrawal attempts were not successful. Further testing on June 15 and 16, 1965, revealed that the problem was in the drive itself which remained at Position 8 and was disarmed June 16, 1965.
1272	J-2	Failed to insert while in shutdown on October 11, 1965. Testing on J-2 was continued on October 12, 13, and 14, 1965. During startup on October 16, 1965, J-2 would not withdraw until reactor pressure was lowered to 211 psig at which point it was deactivated at Position 12.
1251	E-5	Drive had not been removed since 12/15/62. Exhibited shorter than average time in buffer during scram tests.
1250	H-6	Drive had not been removed since 12/15/62. Exhibited shorter than average time in buffer during scram tests.
1245	F-6	Drive had not been removed from reactor since drive modification in February, 1961. Exhibited shorter than average time in buffer during scram tests.
1229	H-7	Drive had not been removed from reactor since drive modification in February, 1961. Exhibited shorter than average time in buffer during scram test.

Serial Number	Drive	Reasons for Removal
1250	H-7	Guide tube dropped on drive spud during installation.
1291	B-2	Drive had not been removed from reactor since drive modification in February, 1961. Exhibited shorter than average time in buffer during scram tests.
1273	E-10	Drive had not been removed from reactor since drive modification in February, 1961. Exhibited longer than average insertion time during scram tests.
1265	B-8	Drive had exhibited longer than average insertion time, but was not removed because of difficulties in removing guide tube.
1297	C-1	Drive had exhibited longer than average insertion time, but was not removed because of difficulties in removing guide tube.
1233	A-5	Drive had not been removed since drive modification in February, 1961. Exhibited longer than average insert time during scram tests.
1257	C-2	Drive had not been removed since drive modification in February, 1961. Exhibited longer than average insert time during scram tests.
1248	D-1	Drive had exhibited longer than average insertion time, but was not removed because of difficulties in removing guide tube.
1243	D-10	Drive had not been removed since drive modification in February, 1961. Exhibited longer than average insert time during scram tests.

Following drive removal, a visual inspection was made by direct viewing whenever radiation exposure levels permitted the use of distance and/or face shielding as protection. Most parts could be inspected at close range. Only the roller mount assembly required a lead cave and quartz viewing window for additional shielding during close observations.

All magnet housings were given a 212° F boiling test to check the integrity of the magnet housing seal.

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

The following components of the eleven drives were subjected to the fluorescent dye penetrant examination:

Index tube (585D288)
Piston head assembly (192C554)
Shuttle piston (115A8600)
Collet assembly (693C827)
Roller mount assembly (932C149)
  a. Weld
  b. Spud
  c. Anti-rotational roller
  d. Guide roller
  e. Roller housing
Spring (111A3298)
Spring washers (145A5454)
Guide plug (856B397)
Drive housing welds (585D289)

The results of individual drive inspections are listed below:

1. <u>Drive Number 1290 (Cell C-9) - Failure to Withdraw</u>

   The problem associated with this drive was found to be caused by foreign material between the shuttle piston and the collet assembly. Apparently the shuttle piston would not move to unlock the collet fingers so the drive could not be moved. During scram and friction testing on 1/13/67, C-9 was scrammed in and subsequently drifted back out, indicating that the fingers were being held open by the shuttle piston, and the collet fingers would not latch at the notches on the index tube. When the drive was being disassembled, the shuttle piston had to be tapped to free it, and it was removed without being separated from the collet assembly. Some foreign material was found between the piston and collet fingers when they were pulled apart. No permanent damage was visible on either the shuttle piston or the collet assembly. A leak was found in the magnet weld when it was boiled - a new magnet was installed.

   All other component parts were visually inspected and found to be in good condition. The dye penetrant inspection revealed no cracks.

2. <u>Drive Number 1251 (Cell E-5) - Short Buffer Time</u>

   The outer filter screen had a kink in it and there was foreign material on the screen. The screen was replaced.

   The threads on the guide plug galled during disassembly and a new guide plug was installed.

   There were some shallow lengthwise scratches on the index tube, which were cleared up by honing.

All stop piston seals were worn and seal No. 4 was broken. This accounts for the short buffer time.

The inner seals on the drive piston exhibited some slight scoring on the inside.

Visual inspection and dye penetrant checks revealed no other defects or damage to any of the other parts of this drive.

3. Drive Number 1250 (Cell H-6) - Short Buffer Time

One seal on the stop piston was broken. The main piston seals were all scored, and outer seal No. 1 and inner seal No. 8 were broken. The inner and outer bushings on the main piston were also heavily scored from foreign particles.

All other drive components parts were inspected visually and with dye penetrant and were found to be in good condition.

The drive was reinstalled in reactor Cell H-7. When lowering the guide tube into the cell, the spud end of the drive got jammed into one of the slots in the guide tube. A pole had to be used to free the spud, and the drive was removed again.

Preliminary inspection of the drive revealed that the spud was slightly cocked. Attempts to unthread the spud proved futile. The index tube was cut and both spud and index tube were replaced with new parts. The drive was again disassembled and inspected visually and all other parts were still in good condition.

4. Drive Number 1245 (Cell F-6) - Short Buffer Time

Some of the chrome plating on the inside of the guide sleeve portion of the collet assembly was damaged and had chipped off. The collet assembly and adapter sleeve were both replaced with new parts.

Some foreign material was found in the guide bushing slot on the stop piston. All four seals were quite worn, especially at the corners, and seals No. 3 and 4 were broken. The main piston was deformed during disassembly to such an extent that there was difficulty removing the internal seals and bushings which, when removed, were found to be in reasonably good condition. The outer seals and bushings, however, were heavily scored on the outside, and both bushings were chipped. The condition of the seals and bushings in the stop piston accounts for its short buffer time.

All other component parts were visually inspected and found to be in good condition. The dye penetrant inspection revealed no cracks.

5. Drive Number 1229 (Cell H-7) - Short Buffer Time

The threads on the guide plug galled in disassembly and it was eventually replaced with a new one.

Stop piston seals No. 1 and 2 were worn and seals No. 3 and 4 were broken. Outer guide bushing No. 2 on the main piston was scored on the outside. The inner main piston seals were in good condition except for seal No. 4 which was chewed up on the outer edges. The broken stop piston seals accounted for the short buffer time on this drive.

All other component parts were visually inspected and found to be in good condition. The dye penetrant inspection revealed no cracks.

6. Drive Number 1291 (Cell B-2) - Short Buffer Time and Long Insert

Stop piston seal No. 4 was broken, and the other three seals were slightly scored on the outside. The outer guide bushings and the outer piston seals on the main piston were heavily scored and nicked on the outside. The inner seals and bushings on the main piston were worn and had small nicks in some cases. The condition of the stop piston and main piston seals accounts for the short buffer time and long insert time characteristic of this drive.

All other component parts were in good condition. The dye penetrant inspection revealed no cracks.

7. Drive Number 1272 (Cell J-2) - Stuck Drive

One of the rollers on the roller mount assembly was missing when the drive was removed from the reactor. The entire roller mount assembly was replaced with one from another drive. Apparently, when the roller came loose from the spud, it jammed between the spud and the guide tube, causing the drive to malfunction. This is the only acceptable reason for the malfunction since all other component parts of the drive were inspected and found to be in good condition. Also, the dye penetrant inspection revealed no cracks.

Additional inspections were conducted to determine if the failure of the roller pin was the result of wear during operation. The pins from four drives removed during this outage were inspected and measured for wear with a micrometer. Examination of these pins revealed that the chrome plating was partially worn, the wear taking place primarily on the side of the pin nearest the guide tube. All 16 pins examined were worn to essentially the same extent.

The rollers on the six drives which previously had been returned to the reactor had been checked for looseness and were found to be satisfactory.

As a result of the inspection of Drive J-2 and roller pins on the other four drives, the following conclusions were reached:

a. The cause of failure was clearly due to overload of the roller pin; however, the origin of the overload is not absolutely clear.

b. Movement of the drive resulted in the roller assuming different positions between the control rod guide tube and drive roller mount, and accounts for the erratic results of the drive malfunction and the excessive wear of the other three pins.

The pin failure of Drive J-2 was unique. None of the pins on other drives inspected showed indication of impending failure similar to that experienced by Drive J-2. Although inspection of other drives revealed roller pin wear, the amount of wear was far less than on Drive J-2.

8. <u>Drive Number 1233 (Cell A-5) - Long Insert Time</u>

The four stop piston seals and bushing were all lightly scored on the outside, and there were some nicks apparent on the tangentially cut inner drive piston seals. A small amount of foreign material was found in the shuttle piston.

This would account for the long insert time experienced on this drive. The original spud associated with 1233 was used on Drive SN 1272, and a new spud will be installed on 1233 later.

All other component parts were in good condition. The dye penetrant inspection revealed no cracks.

9. <u>Drive Number 1243 (Cell D-10) - Long Insert Time</u>

The inner screen on the outer filter was pulled away from the weld, and a new screen was installed.

Drive piston seal No. 4 was cracked and the other inner and outer seals were lightly scored. The condition of these seals accounts for the long insert time on this drive.

New rollers and pins were installed on the roller mount assembly. The original rollers and pins had been removed for inspection.

All other component parts were in good condition. The dye penetrant inspection revealed no cracks.

10. Drive Number 1257 (Cell C-2) - Long Insert Time

The cap had come off the inner filter allowing a large amount of crud to accumulate inside the drive index tube. A new inner filter was installed. New pins and rollers were also installed on the roller mount assembly. One outer drive piston seal was broken and the other inner and outer drive piston seals and bushings were slightly scored, accounting for the long insert time of this drive.

All other component parts were in good condition. The dye penetrant inspection revealed no cracks.

11. Drive Number 1273 (Cell E-10) - Long Insert Time

Outer bushing No. 1 and inner bushing No. 3 were broken, while the other seals and bushings were slightly scored, accounting for the long insert time associated with this drive.

New rollers and pins were installed on the roller mount assembly.

All other component parts were in good condition. The dye penetrant inspection revealed no cracks.

5. Control Rod Blades

During periods of operation, control rods have been verified for blade following on a weekly basis. Monthly control rod worth tests were conducted regularly after the refueling outage except during the month of October, when the unit experienced a short period of operation. During each startup, control rod patterns for criticality have been predicted and all blade following verified.

6. System Stability

a. Dual Cycle Operation

The secondary steam generation was reduced to zero via the control valves at 9:15 p.m., on January 13, 1967. The load dropped from 140 MWe to 81 MWe with little or no change in primary steam flow. The initial and final conditions as experienced are tabulated below:

	Before	After
MWe	140	81
PWF	0.94	0.94×10^6 lbs/hr.
SWF	0.88	0.00×10^6 lbs/hr.

b. Turbine Trip

In preparation for a turbine trip, the load was adjusted to 68 MWe by the insertion of 14 notches and an increase in secondary steam flow to accomplish the initial conditions exhibited in Figure 1. The rod patterns and critical conditions subsequent to the turbine trip and in use during the training of operators are also indicated in Figure 1.

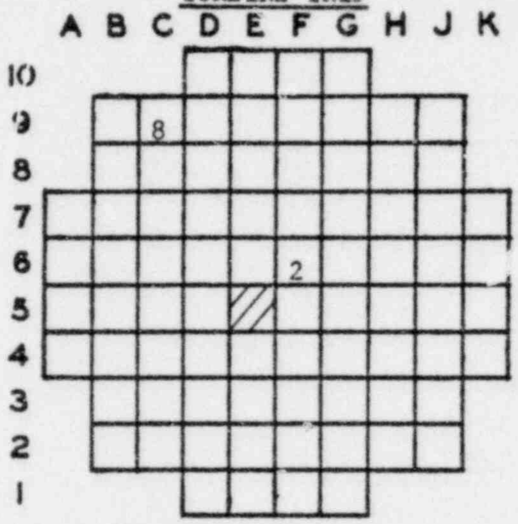
The transients experienced as exhibited in Table 3 are typical of those experienced during such trips conducted previously. The reactor experienced an initial increase in pressure of two psi as the control valves were tripped and a ten psi drop as the bypass valves opened. The primary steam flow dropped 24.3 percent, rose 42 percent and settled back to the initial conditions.

The power level indicators experienced a 20 percent increase as the pressure increased and a subsequent drop of 34 percent as the bypass valves opened.

FIGURE 1

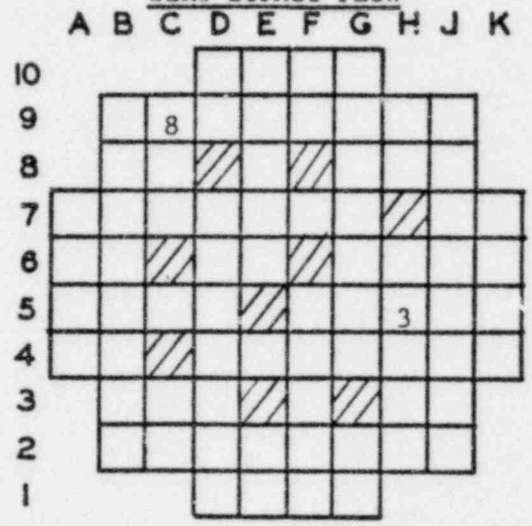
TURBINE TRIP AND REACTOR SHUTDOWN AT END OF CYCLE IV
JANUARY 13, 1967

TURBINE TRIP



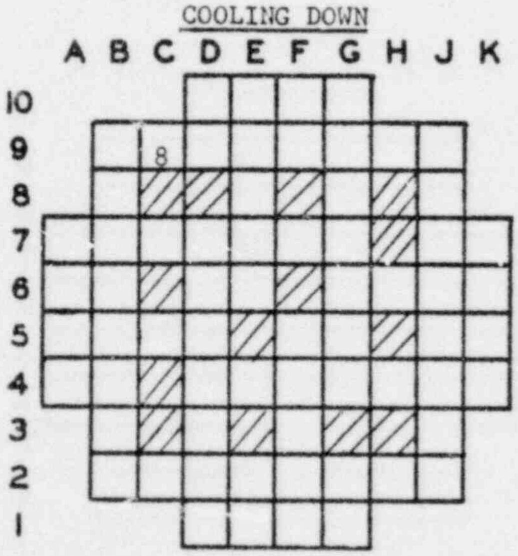
Conditions prior to trip
Time - 9:33 P.M.
MWe - 70
PSF - .74 x 10^6 lb/hr.
SSF - .10 x 10^6 lb/hr.

ZERO BYPASS FLOW

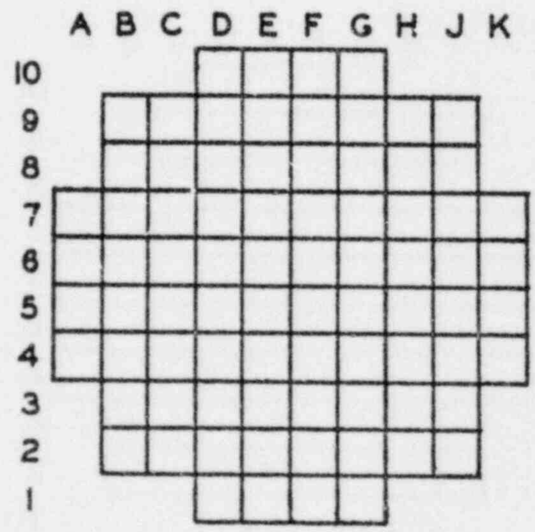


1/13/67 Time - 10:30 P.M.
Residual - 10 rods - 1 notch

COOLING DOWN



1/14/67 Time: 1:30 A.M.
Rt - 450° F
Residual - 14 rods - 4 notches



▨ INDICATES FULLY INSERTED

TABLE 3

TURBINE TRIP TEST
JANUARY 13, 1967
9:33 P.M.

	Initial Conditions	Transient Conditions		
Power Level	70 MWe	0	0	0
Primary Steam Flow - 10^6 lb/hr.	.74	.56	1.05	.73
Reactor Pressure - PSI	1000	1002	992	992

Power Channel	Percent of Power Initial	Peak	Final
1	40	48	30
2	39	46	30
3	39	46	32
4	39	47	31
5	39	47	31
6	39	47	31

7. Changes in Facility Design

a. Electro-Chemical Test Flange Installation

An electro-chemical test flange was installed by General Electric on a four-inch decontamination flange in "B" secondary steam generator primary loop on February 16. The purpose of this test is to obtain basic corrosion data on various series of austenitic stainless steels. The data will be used to determine: (1) if there is any change in the primary loop water chemistry, (2) if there is any correlation between in-pile and out-of-pile test data, and (3) the corrosion rate of various metal specimens.

b. Off-Gas System

1. Stack Filter Replacement

The temporary off-gas filter was replaced with a permanant system during the month of April. The new system maintains the four filter idea, which consists of two parallel outlets with two filters in each outlet. In operation, only one set of filters is used at a time. two air operated butterfly valves for filter change-over replaced the old damper system. The new valves are operated from the fan room where the control air is tapped from an instrument air line. The filters are shielded by two sliding lead doors. Accessibility to each filter was improved by installing covers which can be removed quickly for changing used filters.

General Electric also installed a facility to measure the pressure drop across any one filter or set of filters. Filter selection instrumentation and pressure gauges are located in the fan room.

2. Off-Gas Drain Valve Installation

During April, an air operated valve was installed in the off-gas drain line. This valve will be used to isolate the off-gas holdup line from turbine building sump. In the event of high off-gas flow, this line would remain open while the holdup line valve would trip closed.

c. Environs Monitoring

1. Environs Station No. 2

During October, a new General Electric area radiation monitor was installed in on-site Environs Station No. 2. No. 2 Environs Station is located in a temporary building near the construction area at the discharge sample station for Unit No. 1. The operating conditions for instruments at this station are poor because of the high dust content in the air. The installation of the area monitor at the station will give the unit an operational test under adverse conditions.

2. Clay Products Environs Station Relocation

The Clay Products Environs Station was relocated in June, as it was judged unsafe to service. In addition to being located in an inaccessible location, this station was also troubled by a heavy dust load in the area due to operations at the Clay Products Plant. The new site for the station is approximately one mile south and one-half mile east of its original location. This relocation monitors the same quadrant as the Clay Products location.

3. Deletion of Hansel and Breen Environs Stations

Sample collections were discontinued at the Hansel and Breen Environs Stations during the year, as a result of Environs Contractor Suggestions and Commonwealth Edison Review Board concurrence.

d. Fuel Building and Reactor Canal Crane Controls

During the refueling outage, modifications were made to the fuel building and reactor canal crane controls by installing electrical interlocks on each crane. With this new system, one radiation sensor or three trip switches will open the main breaker on the reactor or fuel building cranes.

e. Fuel Handling Grapples

To prevent accidentally lifting a fuel element above a prescribed five foot water depth, mechanical stops were added to both the fuel building and reactor canal fuel handling grapples. An additional electrical stop was also added to both grapples to prevent lifting an element above a six foot water depth and also to prevent contacting the mechanical stop during normal operation. Both grapple control circuits still contain the original limit switches, whose function is to prevent the grapple hook from raising above a ten foot water depth.

f. Fuel Building Concrete Shield Wall

During the refueling outage, a sand filled concrete block wall was erected in the fuel building to shield the fuel pool coolers. The purpose of the wall is to reduce radiation exposure to workers in the fuel building.

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 May 17, May 19, October 19, and October 20, 1967. 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 .354 x 10^{-7} uc/ml (35.4 uuc/Liter) compared to an average limit of 1.00 x 10^{-7} uc/ml (100 uuc/Liter) for unidentified mixtures containing no radium 226 or radium 228 as specified in 10 CFR Part 20.

Solid radioactive wastes were stored on-site pursuant to License DRP-2. Table 4 shows the content, shipment locations, and dates of radioactive waste shipments made during the year. A total of ten radioactive waste shipments were made in 1967.

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 11,300 uc/second.

During July and August, 1967, General Electric personnel prepared selected fuel rods from various fuel assemblies for shipment to San Jose, California for isotopic and metallurgical analyses. With the exception of the first shipment, all rods came from fuel assemblies which were discharged during the fourth partial refuel-ing. The three fuel shipments are tabulated in Table 5. Each shipment was made via a General Electric series 700 cask. Six General Electric-Knapp Mills truck cask shipments, totaling twenty-one spent fuel assemblies, were made during January to the Chemical Processing Plant of Nuclear Fuel Services, Inc. in West Valley, New York. This fuel is the property of the U.S.A.E.C. per contract number AT (38-1)-315 and was shipped for and upon instruction from its Savannah River Operations Office. Table 6 is a breakdown of all the shipments made since the initiation of such shipments in June, 1965.

TABLE 4

RADIOACTIVE WASTE SHIPMENTS - 1967

Content	Date	Volume	Total Activity (Millicuries)	Location of Shipment
Dry Radioactive Waste	January 3	1,043.70 ft^3	178.35	Nuclear Fuel Services, Inc., West Valley, New York
Dry Radioactive Waste	January 6	994.70 ft^3	347.98	Nuclear Fuel Services, Inc., West Valley, New York
Dry Radioactive Waste	June 22	1,082.90 ft^3	436.69	Nuclear Fuel Services, Inc., West Valley, New York
Dry Radioactive Waste	June 26	1,116.90 ft^3	111.59	Nuclear Fuel Services, Inc., West Valley, New York
Dry Radioactive Waste	June 27	1,296.00 ft^3	55.57	Nuclear Fuel Services, Inc., West Valley, New York
Dry Radioactive Waste	November 21	1,296.00 ft^3	36.09	California Nuclear, Inc., Sheffield, Illinois
Dry Radioactive Waste	November 22	477.75 ft^3	178.35	California Nuclear, Inc., Sheffield, Illinois
Dry Radioactive Waste	November 27	448.00 ft^3	0.64	California Nuclear, Inc., Sheffield, Illinois
Dry Radioactive Waste	November 28	448.00 ft^3	0.64	California Nuclear, Inc., Sheffield, Illinois
Spent Resins	September and October	4,110.00 ft^3	12,000.00	Contracted To: California Nuclear, Inc., Sheffield, Illinois

TABLE 5

SUMMARY OF GENERAL ELECTRIC FUEL SHIPMENTS

Shipment Number	Date Shipped	Content	Total Activity
Val Shipment No. 1	July 19, 1967	Fuel Segments and Damaged Channel	35,000 Curies
Val Shipment No. 2	August 15, 1967	Fuel Segments and Eleven B4C Tubes From Control Blade B-38	91,300 Curies
Val Shipment No. 3	August 17, 1967	Fuel Segments	110,900 Curies

TABLE 6

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	[illegible]		48
3		7/16/65	24	0	0	0	0	[illegible]		72
4		8/ 3/65	24	0	0	0	0	[illegible]		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
	[illegible]	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
	3[illegible]	12/11/66	0	0	4	0	0		4	447
	3[illegible]	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

11. Plutonium Rod Installation

At the end of the fourth fuel cycle, four Type III-F assemblies (G-77, G-90, G-92 and G-111) were removed from the core for plutonium rod installations. One gadolina rod was removed from each of these four assemblies and plutonium rods, containing 1.2 to 1.4% plutonium, were installed in their places. The four assemblies were then returned to the core for Cycle 5.

12. Fuel Assembly Cleaning and Testing

a. Fuel Cleaning

During General Electric's inspection of the seven hole orifice nose piece of XE-103, considerable crud buildup was noted. The lower tie plates of the Type I fuels also were found heavily coated with crud.

Fuel cleaning was initiated on February 21, 1967, immediately after the completion of the initial critical test. All fuel, excluding two "special" assemblies (SA-1 and PF-10), was cleaned before reuse in Cycle 5. This totaled 356 assemblies cleaned. The two "special" assemblies had been prepared by General Electric for reuse and were found to be acceptable by flow testing.

b. Bowed Assemblies

After noting a few "bowed" assemblies during fuel movements, two bow checking gauges were fabricated, one for the fuel building and one for the reactor. Eighty-six Type III and the 106 new Type V assemblies were bow checked at the reactor, and the remainder of the assemblies used in the Cycle 5 core were checked in the fuel building. On March 3, 1967, a bowed assembly, E-24 was given a series of tests to prove definitely that the "bowing" was due to warped channels and not to warped assemblies. Twenty-one assemblies, including one Type V, were found to have "bowed" channels and had to be rechanneled for use in the core.

c. Fuel Assembly Flow Testing

On March 4, 1967, a fuel assembly flow test fixture was installed in the fuel transfer pool. Measurements of flow and pressure drop were taken on both Type I and Type III assemblies in the clean and dirty condition.

The flow test fixture is illustrated in Figure 2. Information obtained from the flow test fixture for a typical Type III fuel assembly with a "D" type orifice and with an "E" type orifice, both before and after orifice cleaning, is exhibited in Figure 3. A new Type V fuel assembly, which is very similar to the Type III, was flow tested and the results are also shown in Figure 3 for comparison. After orifice cleaning, the Type III fuel assemblies, flow vs. pressure drop curves approached that of the Type V assembly. Similar information for Type I fuel assemblies is exhibited in Figure 4. Estimated data for clean Type I fuel assemblies is also shown for comparison. The percent increase in flow after cleaning, at a pressure drop of 8 psi, for typical fuel assemblies is shown in the table below:

Assembly	Orifice	Installed in Reactor	% Increase In Flow As A Result of Cleaning Orifice
Type III	D	Start of Cycle III	240%
Type III	D	Start of Cycle IV	150%
Type III	E	Start of Cycle III	45%
Type III	E	Start of Cycle IV	30%
Type I	C	Start of Cycle I	170%
Type I	B	Start of Cycle II	72%

' IGURE 2

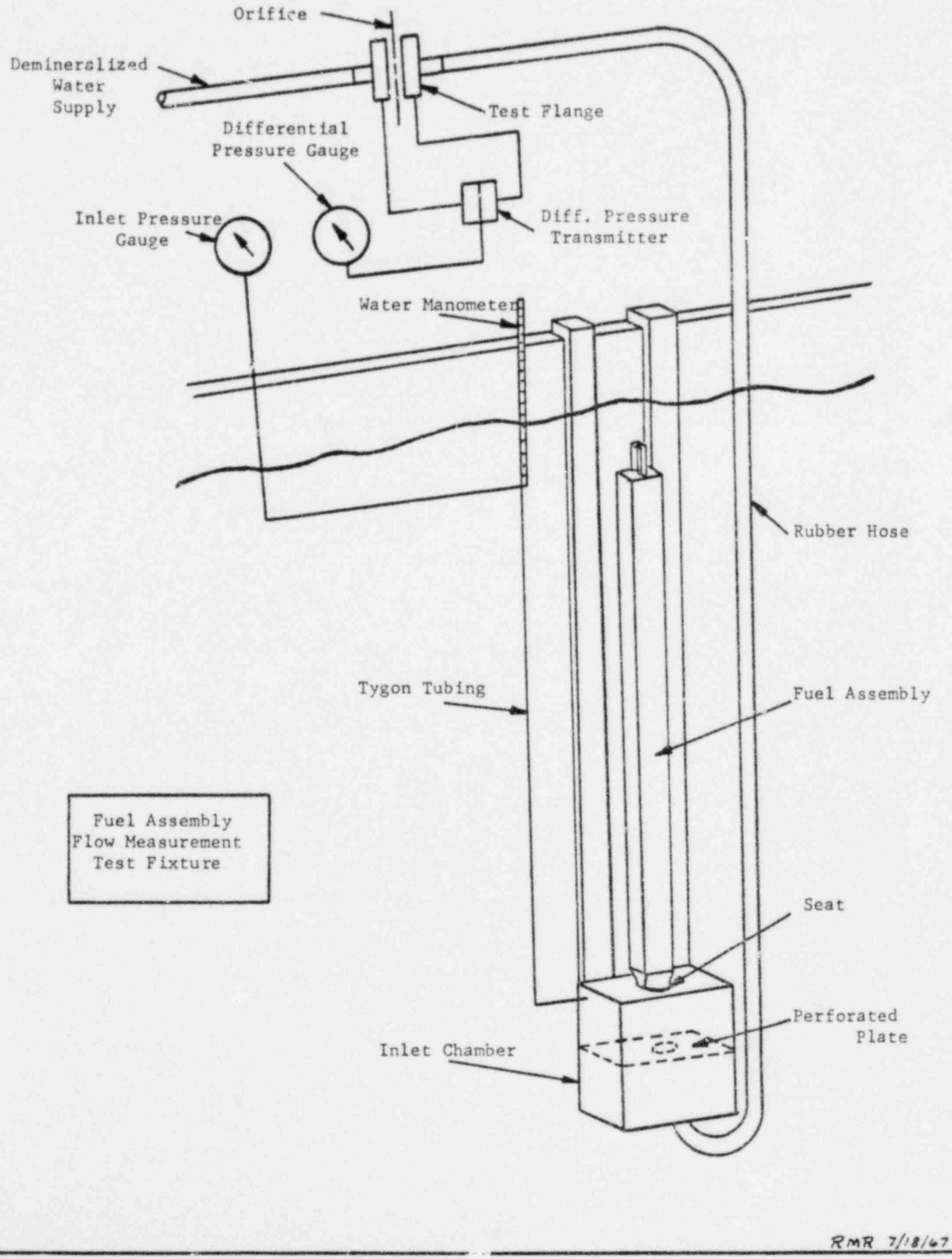
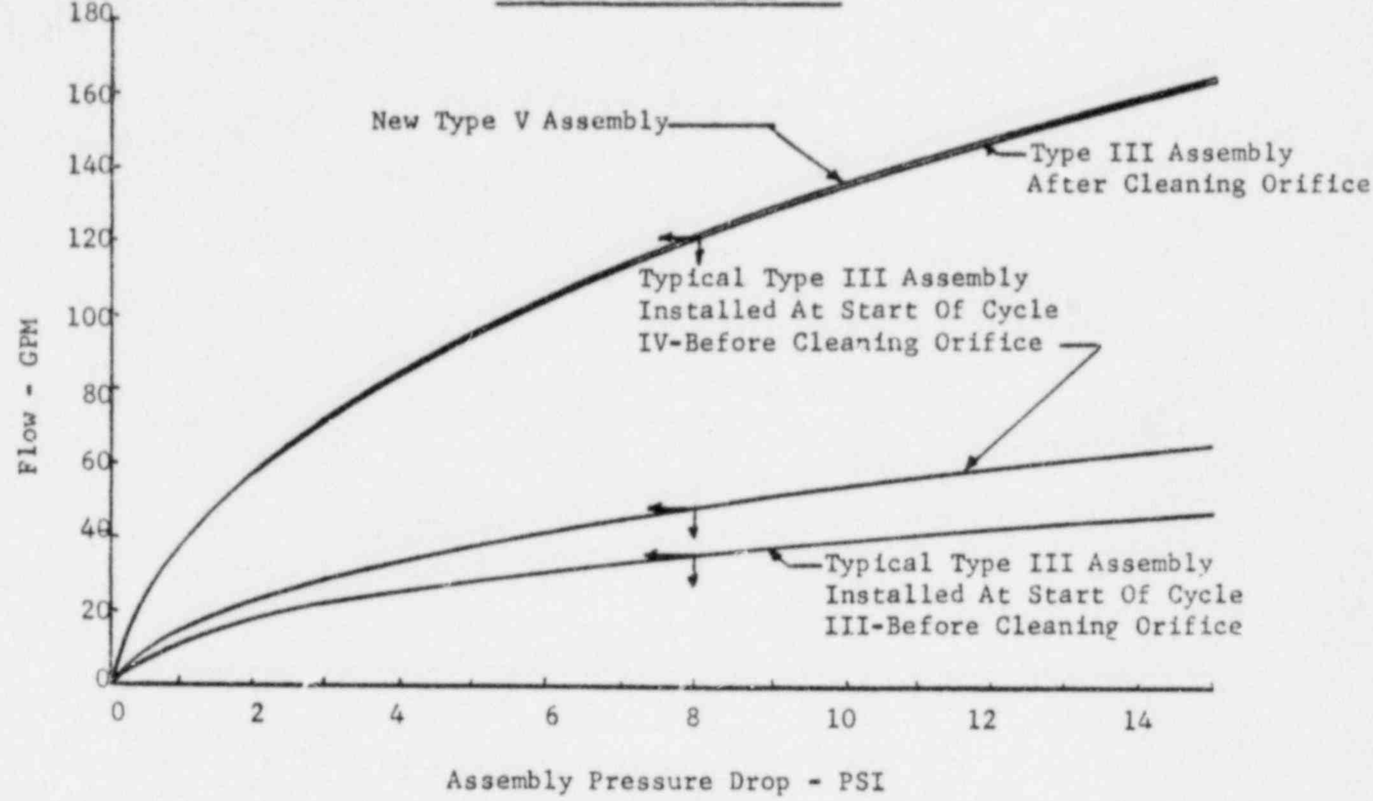

Orifice
Demineralized Water Supply
Test Flange
Differential Pressure Gauge
Inlet Pressure Gauge
Diff. Pressure Transmitter
Water Manometer
Rubber Hose
Tygon Tubing
Fuel Assembly
Fuel Assembly Flow Measurement Test Fixture
Seat
Perforated Plate
Inlet Chamber
RMR 7/18/67

FIGURE 3

FUEL ASSEMBLY
PRESSURE DROP VS. FLOW
TYPE "D" ORIFICE



FUEL ASSEMBLY
PRESSURE DROP VS. FLOW
TYPE "E" ORIFICE

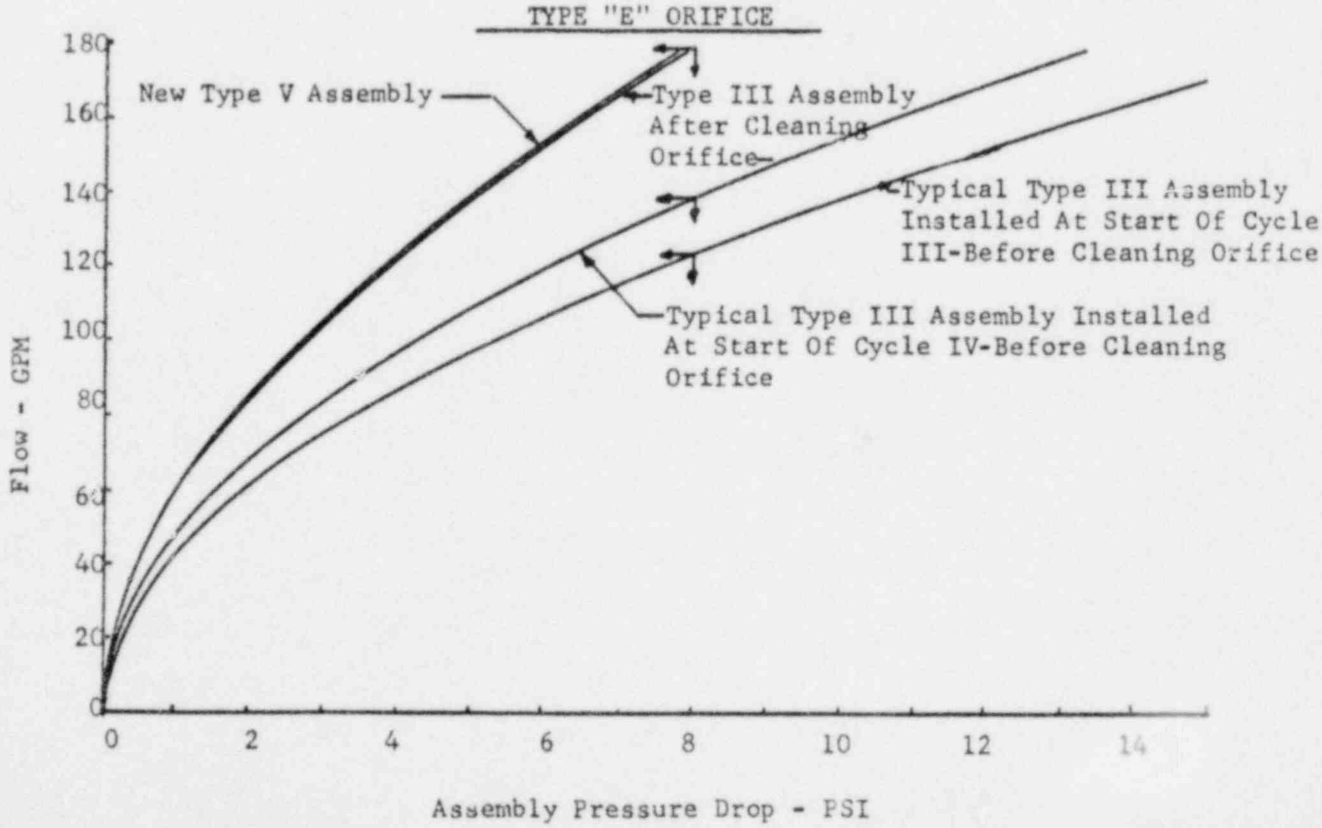
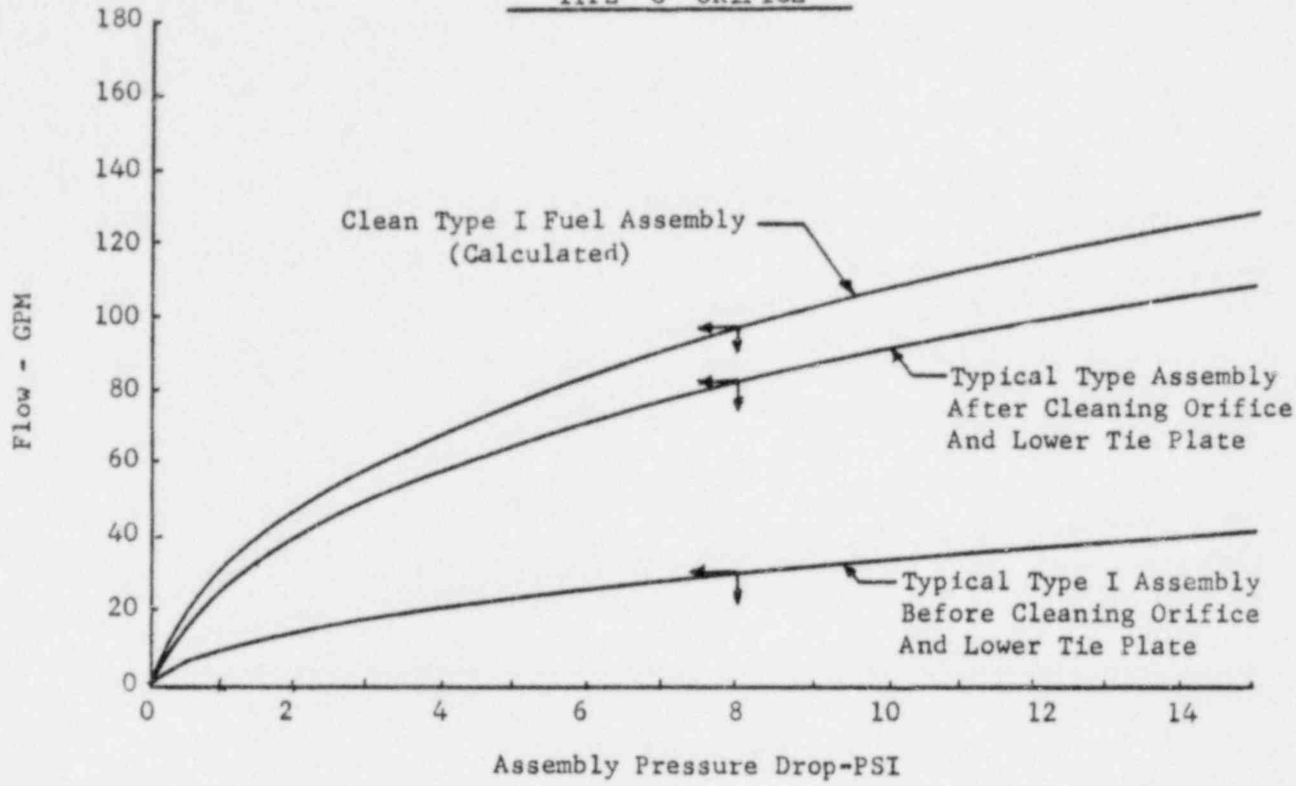
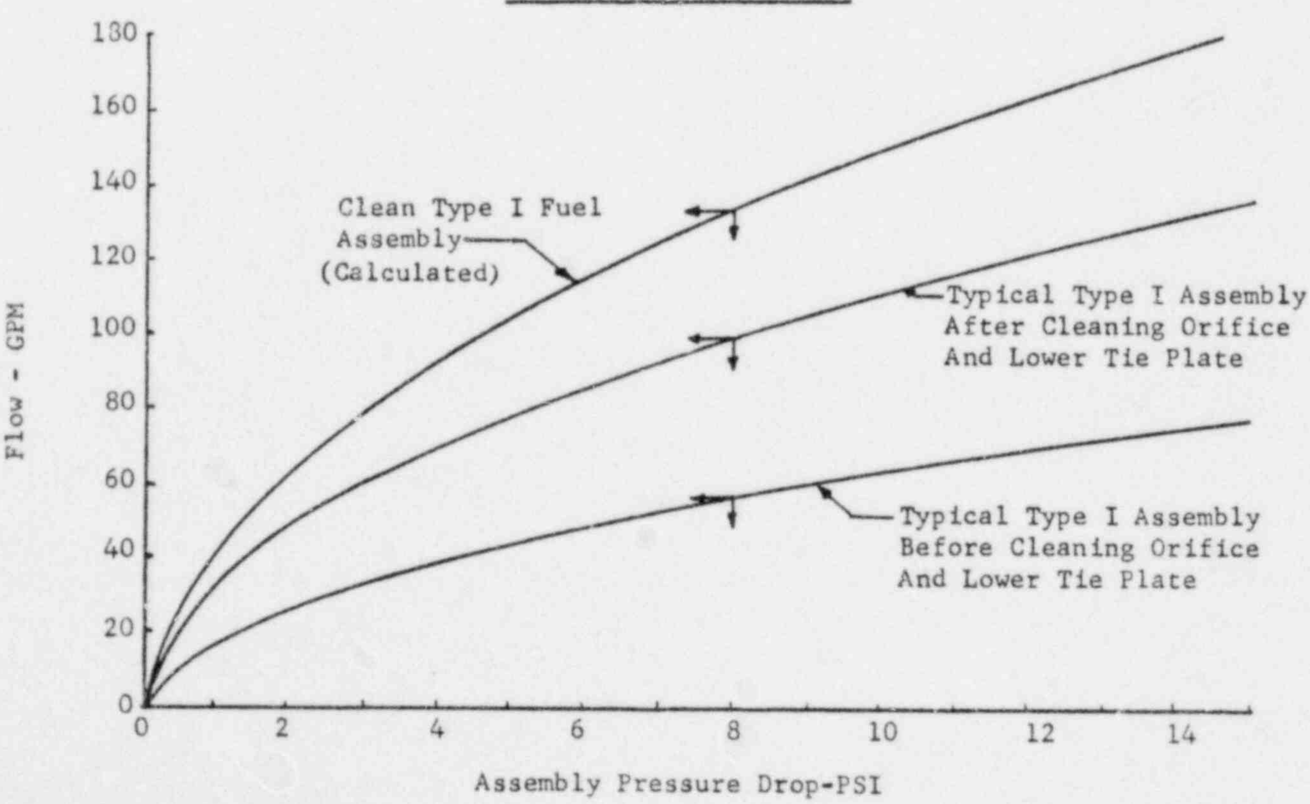


FIGURE 4

FUEL ASSEMBLY
PRESSURE DROP VS. FLOW
TYPE "C" ORIFICE



FUEL ASSEMBLY
PRESSURE DROP VS. FLOW
TYPE "B" ORIFICE



13. H.P. Turbine Inspecti n

An overhaul of the high pressure turbine and associated parts was performed during the 4th partial refueling outage. Radiation levels encountered were considerably higher than those encountered during the 1962 inspection. The results of the inspection are as follows:

a. Turbine

The high pressure rotor was in satisfactory condition, showing no evidence of erosion or rubbing. The shaft packing, the high pressure shell joint, and the coupling were all in satisfactory condition; while both main H.P. bearings and the thrust bearing were in excellent condition.

b. Valves

The primary and secondary stop valves were modified to reduce closing time on trip.

New stems were installed on numbers two and three primary control valves. The operating mechanisms were also inspected and cleaned.

c. Governor

The governor drive assembly was found worn, with worm wheel bearings having loose babbitts and excessive clearance. In addition, the thrust runner was rough and the I.D. bore was sticking on the shaft. It was found that the worm gear was too low.

The worm gear was shimmed, whereas no spares were available. The governor stand bearings were reassembled in reverse order to place the worn section at the top.

The piston rings in the primary operating cylinder were worn and the bottom one stuck in its groove. Due to lack of spares, the rings were cleaned and reinstalled.

d. Gland Seal Regulator

The main gland seal regulator was extensively repaired. The safety valve was inspected and pop tested satisfactorily.

e. Main Oil

The main oil pump was found to be in satisfactory condition. The main oil tank was cleaned and found to be in excellent condition.

f. Turbine Extraction Piping

Corroded and eroded sections of the turbine extraction piping were replaced by General Electric during the refueling outage. Those sections replaced were:

1. "D" extraction 16" line.

2. "C" extraction 20" line.

3. "C" extraction 8" bypass line.

4. Twenty-second stage 14" drain line.

g. Generator

Routine electrical tests were conducted on the stator and field windings during the outage.

The generator was "bumped" while attempting to synchronize with the system on May 23, 1967. This was caused by damaged secondary connections on the A phase potential transformer on the main generator winding that caused the synchroscope to be offset by roughly 40 - 60 electrical degrees.

The unit was removed from the system and an inspection was conducted on the end turns of the stator windings. No signs of damage were found. Overspeed tests were satisfactorily completed May 29, 1967, and the unit returned to the system.

14. Inspections

a. Control Rod Drive Thimble Welds

Eleven control rod drive thimble welds were ultrasonically inspected during the refueling outage. No indications of defects were found.

b. Reactor Flange and Head

During the refueling outage, with the turning vane installed in the reactor and shielded, the main flange of the reactor vessel was dye penetrant checked and ultrasonically by B & W. No defects were detected.

Both dye penetrant and magnetic particle inspections were conducted on the reactor head. No defects were found. These tests were performed by B & W non-destructive testing specialists.

The stud bolts on the reactor head flange were also ultrasonically tested. Test results were satisfactorily on all studs.

c. Drum Downcomers, Risers and Reactor Inlet Nozzles

Ultrasonic tests and dye checks were conducted on shop and field welds on the reactor inlet nozzles, drum risers, and drum downcomers during February, March and April, 1967. No defects were noted in any of these inspections.

d. Primary System Piping

Ultrasonic testing of welds in the secondary steam generating rooms was started by General Electric on January 30, 1967. This phase of tests was completed on February 3, 1967, and included:

"A" Secondary Steam Generator Room:	All welds on 6" bypass line plus decontamination stubs on 6" line and suction header.
"B" Secondary Steam Generator Room:	All welds on 6" bypass line, both decontamination flanges and welds on reactor return line.
"C" Secondary Steam Generator Room:	All welds on 6" bypass line plus decontamination stub weld on reactor return line and weld on the return line at the secondary steam generator.
"D" Secondary Steam Generator Room:	All welds on 6" bypass line plus decontamination stub welds on 6" and suction lines. Suction header field welds, and reactor return line field welds were also inspected.

Three downcomer tee's were inspected from the pipe side only and two downcomer tee's were inspected on all three legs of the tee's. The vertical leg was tested only on one downcomer. One length of shop horizontal welds was also ultrasoncially tested in "D" secondary steam generator room.

The following defects were noted in the above tests: One 30% deep by 1" crack in the 6" bypass weld in "A" secondary steam generator room; one 10% deep by 3" crack in the 6" bypass in "C" secondary steam generator room; two 10% deep by 1" crack in the 6" bypass loop in "D" secondary steam generator room. Two defects noted in "A" and "C" secondary steam generator rooms 6" bypass loops were later cut out and replaced.

While grinding out the 30%, 1" crack in the weld on the 6" loop in "A" secondary steam generator room, a 3/4" 50% deep longitudinal crack was found in this weld. General Electric then ultrasonically tested longitudinally all welds in the 6" bypass lines in the four secondary steam generator rooms on March 28, 1967, and no further crack indications were found.

A hydro test on the primary system at the completion of the refueling, April 19, 1967, revealed two leaks on the six inch stainless steel cross tie connecting the two 22" recirculation pump suction headers. This line supplies the reactor cleanup demineralizer, and is made from 304 stainless steel, Schedule 80, seamless pipe. Stainless steel lines, from two inch to ten inch in diameter, in the primary system, are made of seamless pipe, the larger lines being made of seam-welded pipe.

As a result of these leaks, it was decided to ultrasonically inspect a minimum of: 100% of the three, four, six and eight inch pipe, and 85% of the two and ten inch pipe. This inspection program was completed in April. However, only 94% of the welds (32 out of 34) were inspected in the three inch poison feed line. The two welds not inspected are located in the sand filled biological shield around the reactor. No defect indications were found in the 32 welds inspected.

Defects were found in Lines 0137, 0301 and 0406. Discussing each line separately, the results are as follows:

Line 9137 (Lower Header Tie) - This six-inch line contains 18 welds, and all were inspected. Seven defect indications by ultrasonic tests were found. All defect indications were circumferential and in the heat affected zone in the pipe adjacent to the weld. The line was entirely replaced with 304 extra low carbon stainless steel.

Line 0301 ("A" Clean-up Demineralizer Supply) - This four-inch line contains six welds, and all were inspected. One defect indication was found. However, this defect indication was in the pipe, longitudinal, and quite small. The defect indication was about 10% of the wall thickness in depth and 1/4 inch long. The defect occurred in a straight piece of pipe next to a 90° elbow. One elbow was replaced.

Line 0406 (Poison Tank Vent or Equalizing Line) - This three inch line contains 21 welds, and all were inspected. Three defect indications were found. All were in the welds themselves and were grouped in one area. Three elbows and a straight section ras replaced.

With all piping replacement completed, sonic tested, and inspected as needed, preparations for a hydro test on the primary system was started on May 15. Hydrostatic tests were made at 700#, 900# and 1200# on May 17, 1967, on the primary system. All tests were okay, no leaks were noted in the new piping or new welds.

Word was received from General Electric Co. on October 20, that ultrasonic inspection of sixty feet of the six-inch stainless steel piping, removed from the cleanup loop suction line during the refueling outage, revealed nine flaws in one twenty foot section. It was decided that an extensive ultrasonic inspection of the small diameter piping in the primary system would be conducted.

Ultrasonic testing began on October 23, with all reasonably accessible 2", 3", 4", 6" and 8" piping being inspected. The tabulation below shows the lengths of these various pipes and also the percentage inspected.

Pipe Size	Total Length (Ft.)	% Inspected	% Unavailable
4 Inch	270	68	32
6 Inch	203	94	6
8 Inch	154	97	3
3 Inch	231	32	68
2 Inch	136	65	35
TOTAL	994 Ft.	69%	31%

Inspection was completed on October 27, 1967, and no flaws or cracks were found, although one of the two known cracks in "D" secondary steam generator bypass loop was found to have propagated slightly. This entire loop was isolated until repairs could be made.

On December 12, 1967, after replacing the entire 6" bypass loop and completing successfull radiograph, ultrasonic, and hydrostatic tests, "D" loop was returned to service.

15. Tests

a. Power and Temperature Coefficient of Reactivity

Reactivity measurements were taken at xenon free conditions during the startups at temperatures of 113, 122, 247 and 371° F. The resultant power and temperature coefficients of reactivity are listed in Table 7 and Figure 5 and graphically illustrated in Figure 6. These also include one point obtained November 4, at 398° F, which is shown in Figure 7. The experimentally determined power and temperature coefficient of reactivity shows that the coefficient initially changed from positive to negative at a temperature below 113° F. This value is consistent with the measured values at the beginning of previous cycles and, combined with the predicted changes during the cycle, meets the license requirement at being negative at operating tempe ture at any time during the operating cycle.

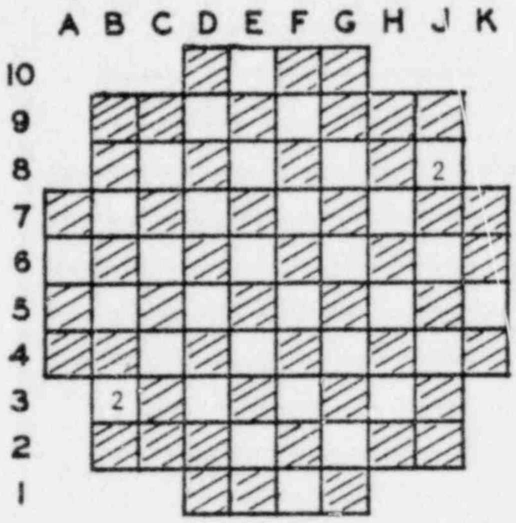
TABLE 7

TEMPERATURE COEFFICIENT OF REACTIVITY, BOC 5

Temperature	% $\Delta K/K$ - ^{o}F
113^{o}F	-0.33×10^{-3}
122^{o}F	-0.42×10^{-3}
247^{o}F	-2.22×10^{-3}
371^{o}F	-6.11×10^{-3}

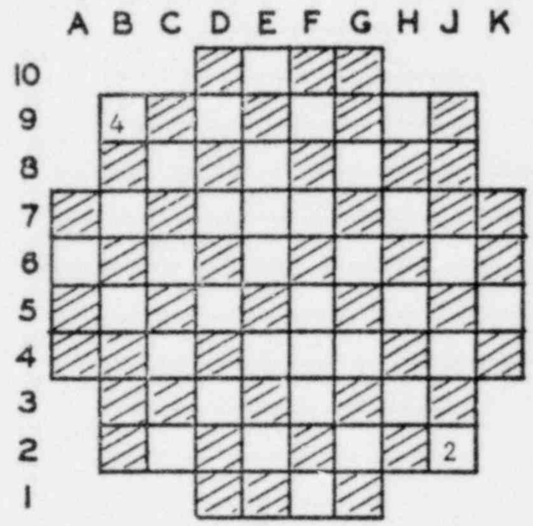
FIGURE 5

ROD PATTERNS AND STATE CONDITIONS DURING
TEMPERATURE COEFFICIENT OF REACTIVITY MEASUREMENTS
ROD WITHDRAWAL PATTERN 5-B



5/22/67

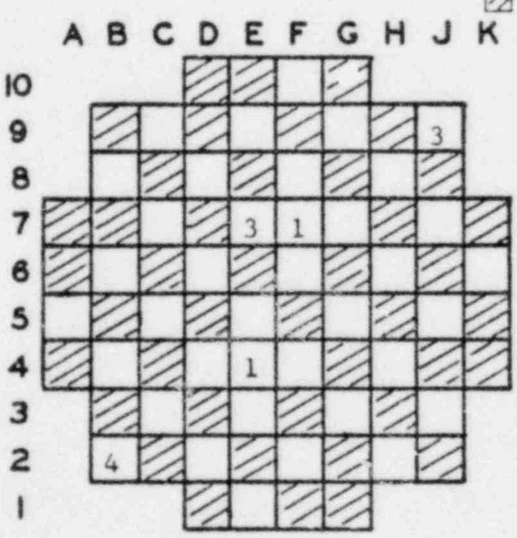
Time	Temp. °F	τ (Sec.)	% ρ
1830	110	109	0.058
2100	116	114	0.056
2142	128	127	0.051



5/23/67

Time	Temp. °F	τ (Sec.)	% ρ
0915	362	23.2	0.164
0950	380	121	0.053

□ Withdrawn
▨ Inserted



5/28/67

Time	Temp. °F	τ (Sec.)	% ρ
1045	242	74	0.078
1120	251	108	0.058

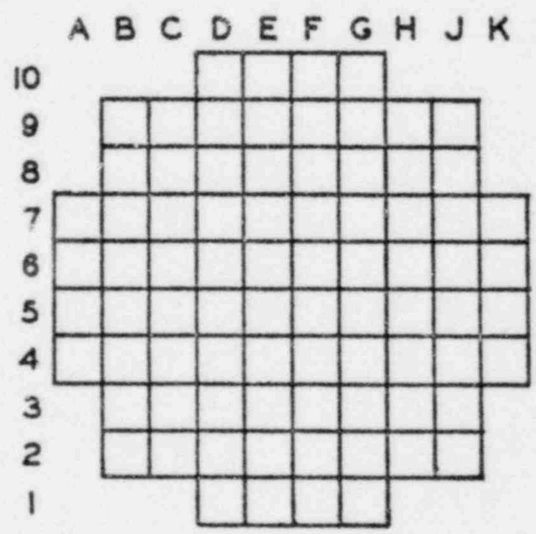


FIGURE 6

TEMPERATURE COEFFICIENT OF REACTIVITY

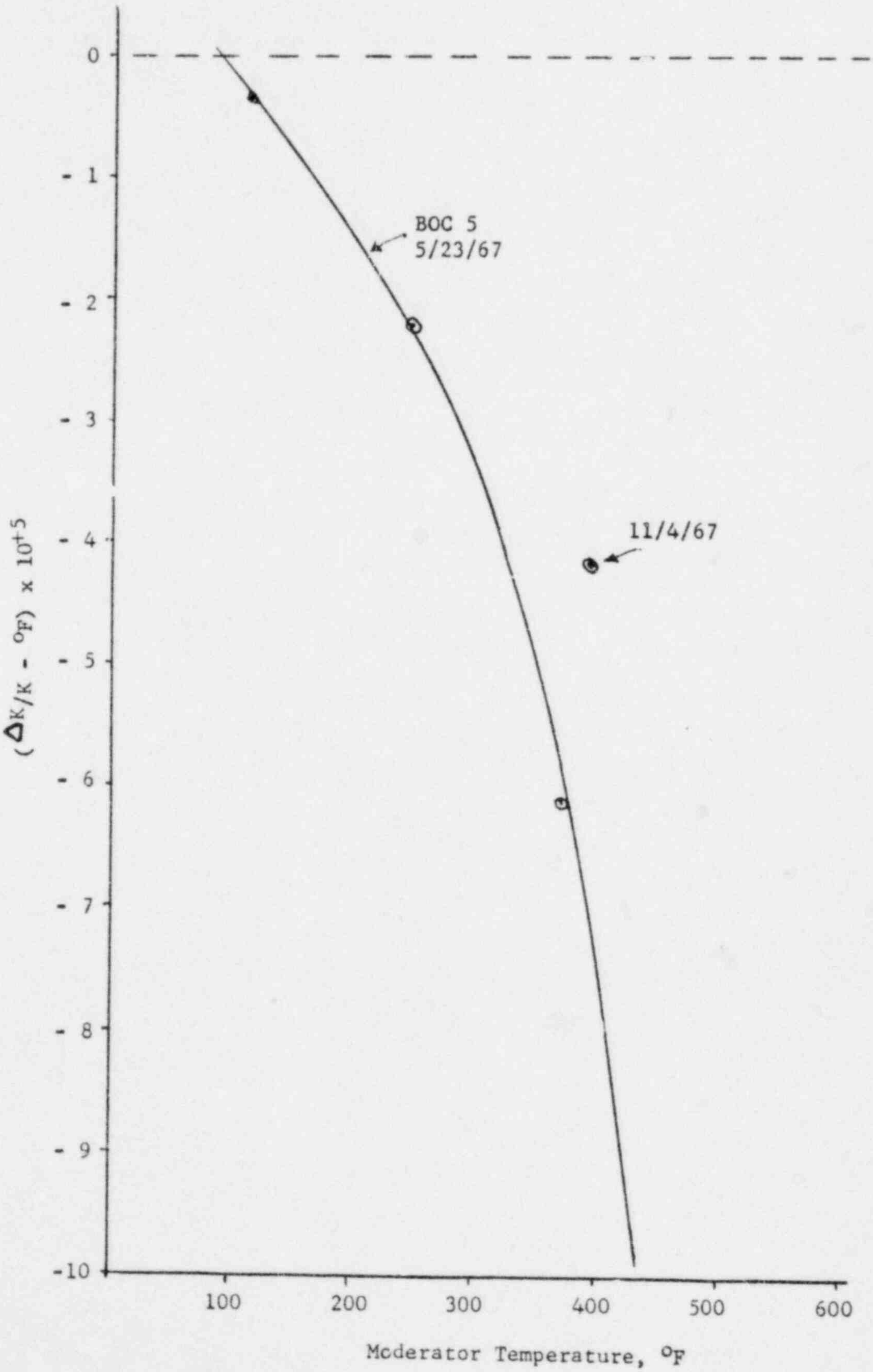


FIGURE 7

ROD PATTERN, STATE CONDITIONS AND RESULTS OF TEMPERATURE COEFFICIENT OF REACTIVITY CHECK, NOVEMBER 4, 1967

	A	B	C	D	E	F	G	H	J	K
10					12					
9		2		12		12		12		
8			12		12	5	12		12	
7		12		12		12		12		
6	12		12	12	12		12		12	
5		12		12		12	12	12		12
4			12		12		12		12	
3		12		12	5	12		12		
2			12		12		12		2	
1						12				

Temperature (^oF)	Period (sec.)	Reactivity P
391	59	0.0918
405	260	0.0337

Temperature Coefficient = $\Delta P / \Delta T = -4.15 \times 10^{-5} \ \Delta K/K - {}^oF$ at 398 ^oF

b. Fuel Sipping and Leaker Detection

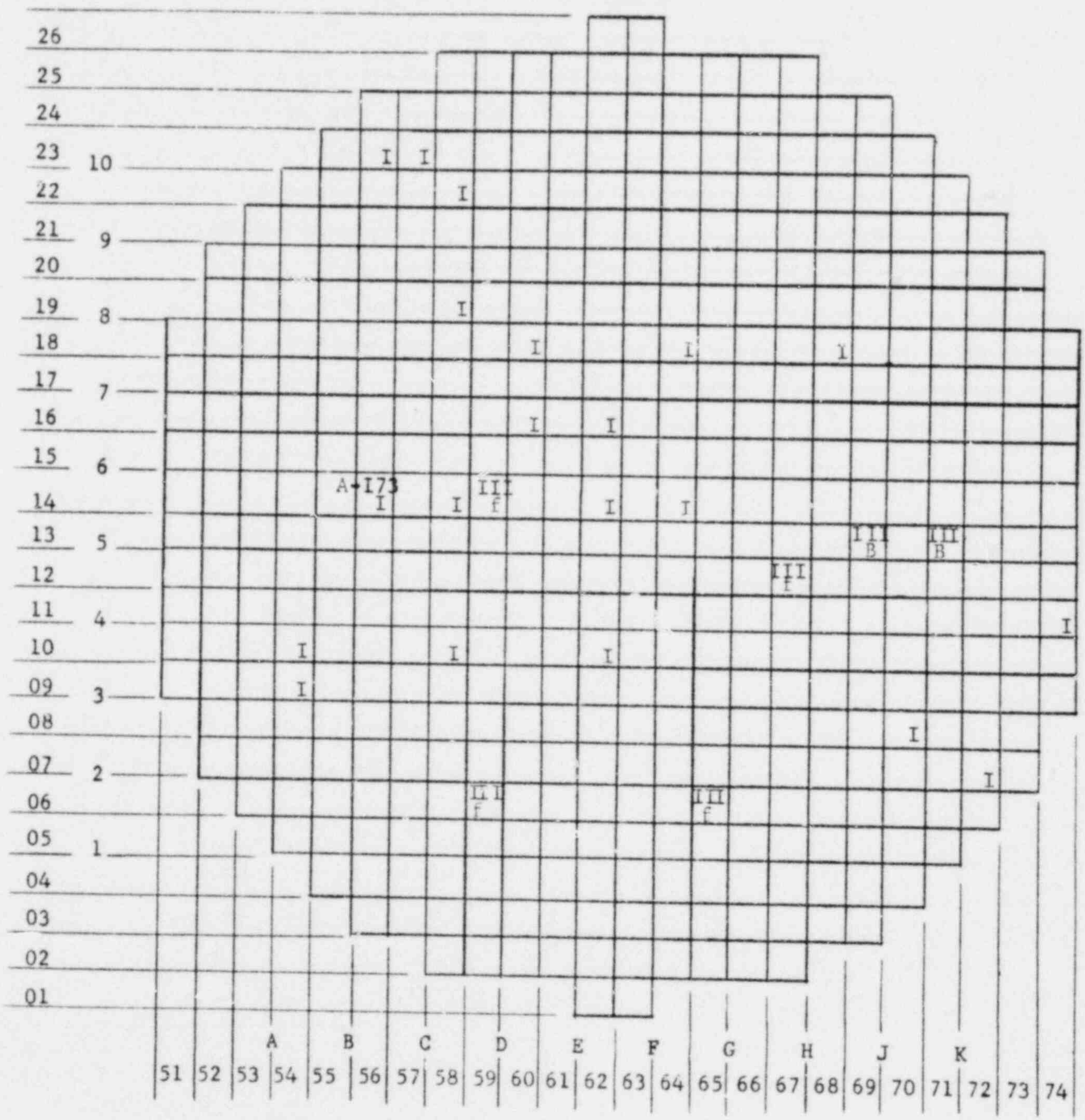
Four Hundred Twenty-Nine (429) fuel assemblies were sipped at the reactor and eight assemblies were sipped in the fuel building between the periods of January 21 and January 29. From the results of the sipping technique, a total of twenty-five assemblies were classified as leakers. Nineteen of these twenty-five leakers were of the one hundred sixty-six (166) Type I fuels in the core at the end of Cycle 4. Seven of these had been installed in the core at the beginning of Cycle 1.

Not included in these results was Type I assembly A-173, which was found to be ruptured during the dechanneling of the discharged Type I fuel in October. Additional data on this assembly includes an exposure of 13,235 MWD/T and an initial installation date of the beginning of Cycle 2.

The core positions, during Cycle 4, of the twenty-five confirmed leakers plus the defective assembly A-173, is shown by type in Figure 8.

FIGURE 8

LOCATIONS OF DEFECTIVE FUEL ELEMENTS BY TYPE



c. Shutdown Margin Checks

Reactor shutdown margin checks were conducted on February 15, 1967, to demonstrate that the refueled core met the license requirements with regard to the "stuck rod" criterion and that the margin is in excess of one percent throughout the core. The margins found were in excess of 1.1 percent throughout and in excess of 1.7 percent across the north half of the core.

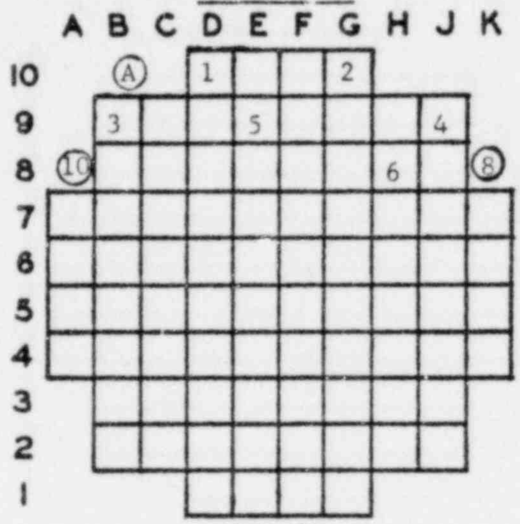
Following several fuel rearrangements further shutdown margin checks were made. These further checks, conducted on April 2nd and 3rd, used a rod withdrawal sequence which, when successfully withdrawn, indicated a subcritical margin in excess of 1.7% in the periphery and 1.9% in the central region. Several fuel rearrangements were required to approximate equal subcritical margins on all sides of the periphery. Shutdown margin checks are exhibited in Figures 9, 10 and 11.

FIGURE 9

SHUTDOWN MARGIN CHECKS - PERIPHERAL ZONE
1.25% FOR 6 RODS

No. Rods	Margin (%)
4	1.10
6	1.25

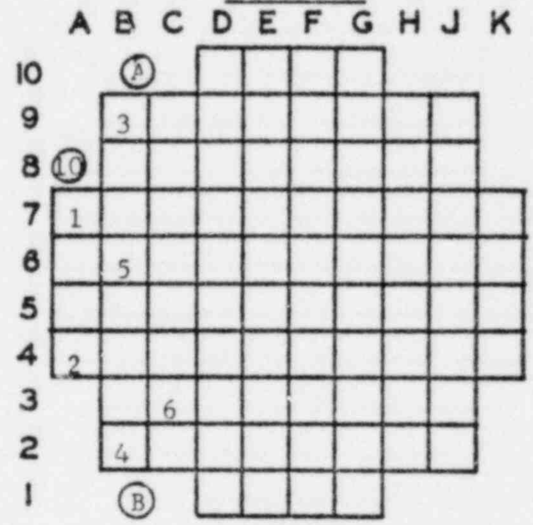
NORTH SIDE



4-1-67

* C	A	8	10
M	417	286	71

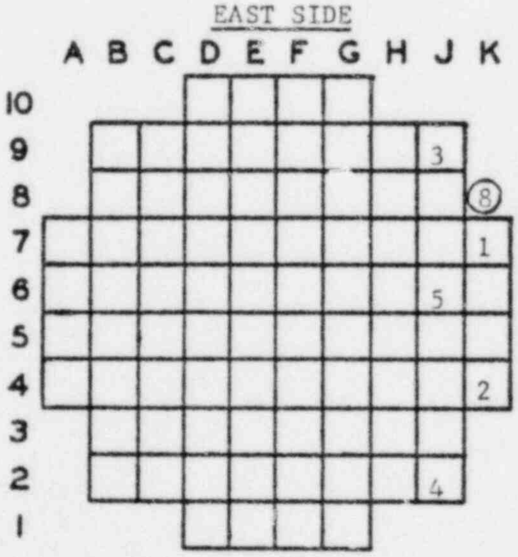
WEST SIDE



4-1-67

* C	A	B	10
M	3.3	3.1	9.3

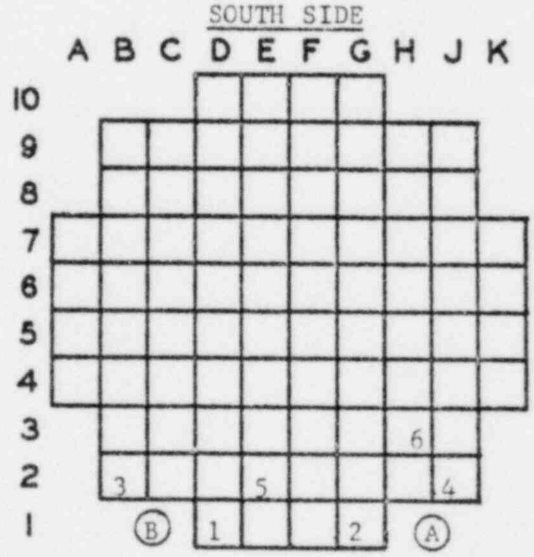
EAST SIDE



4-1-67

* C	8
M	9

SOUTH SIDE



4-1-67

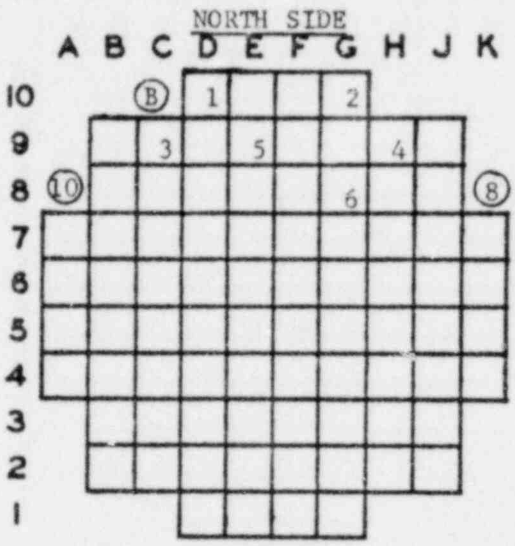
* C	A	B
M	13.3	13.1

* C = Chamber
M = Multiplication

FIGURE 10

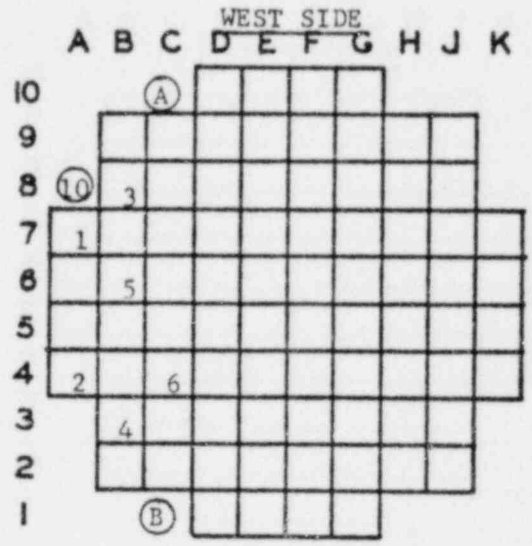
SHUTDOWN MARGIN CHECKS - PERIPHERAL ZONE
1.7% FOR 6 RODS

No. Rods	Margin (%)
4	1.5
5	1.6
6	1.7



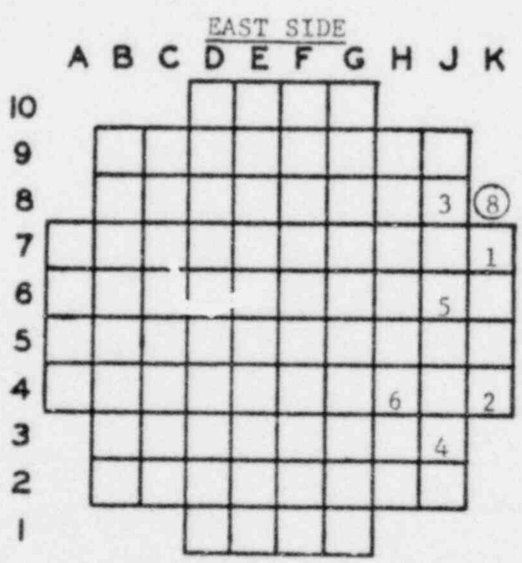
4-2-67

* C	B	8	10
M	100	24	6.7



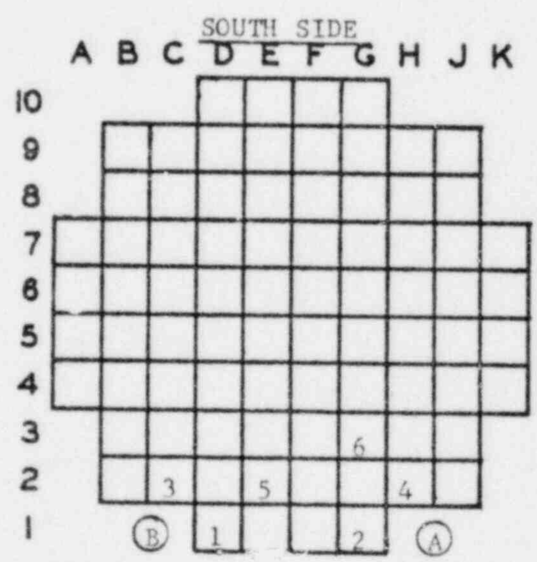
4-1-67

* C	A	B	10
M	58	192	400



4-1-67

* C	8
M	30



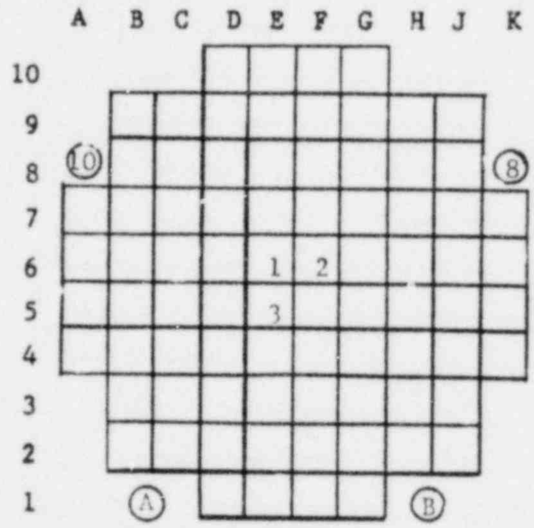
4-1-67

* C	A	B
M	113	58

* C = Chamber
M = Multiplication

FIGURE 11

SHUTDOWN MARGIN CHECKS
CENTRAL ZONE
1.9% FOR 2 RODS



4-1-67

NO MULTIPLICATION DETECTED

C = Chamber

d. Critical Testing of Refueled Core

The whole core was critically tested on February 20, 1967. The control rod withdrawal sequence, 5B, was used to attain a critical state after the partial withdrawal of the thirty-first rod. Criticals attained during this and previous refuelings are summarized in the following tabulation.

The insertion of one or more control rods in the center of the core appeared to have little or no effect on the critical state indicating that the critical was essentially attained by the withdrawal of the peripheral control rods.

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)

* Initial loading - 448 assemblies

e. Control Rod Calibration

The incremental power and steam generation rate of H-7 was determined on January 11, 1967, by progressive insertion of the rod, notch by notch. Incore data were recorded for the initial conditions and after the insertion of each notch thereafter.

The power, steam and incore changes as affected by the rod are exhibited in Figure 12. The responses obtained are typical of that experienced during previous end-of-life checks.

The rod pattern in use at power and the variations within it as experienced during the month are exhibited in Figure 13.

FIGURE 12

CONTROL ROD H-7 POWER CALIBRATION
JANUARY 11, 1967

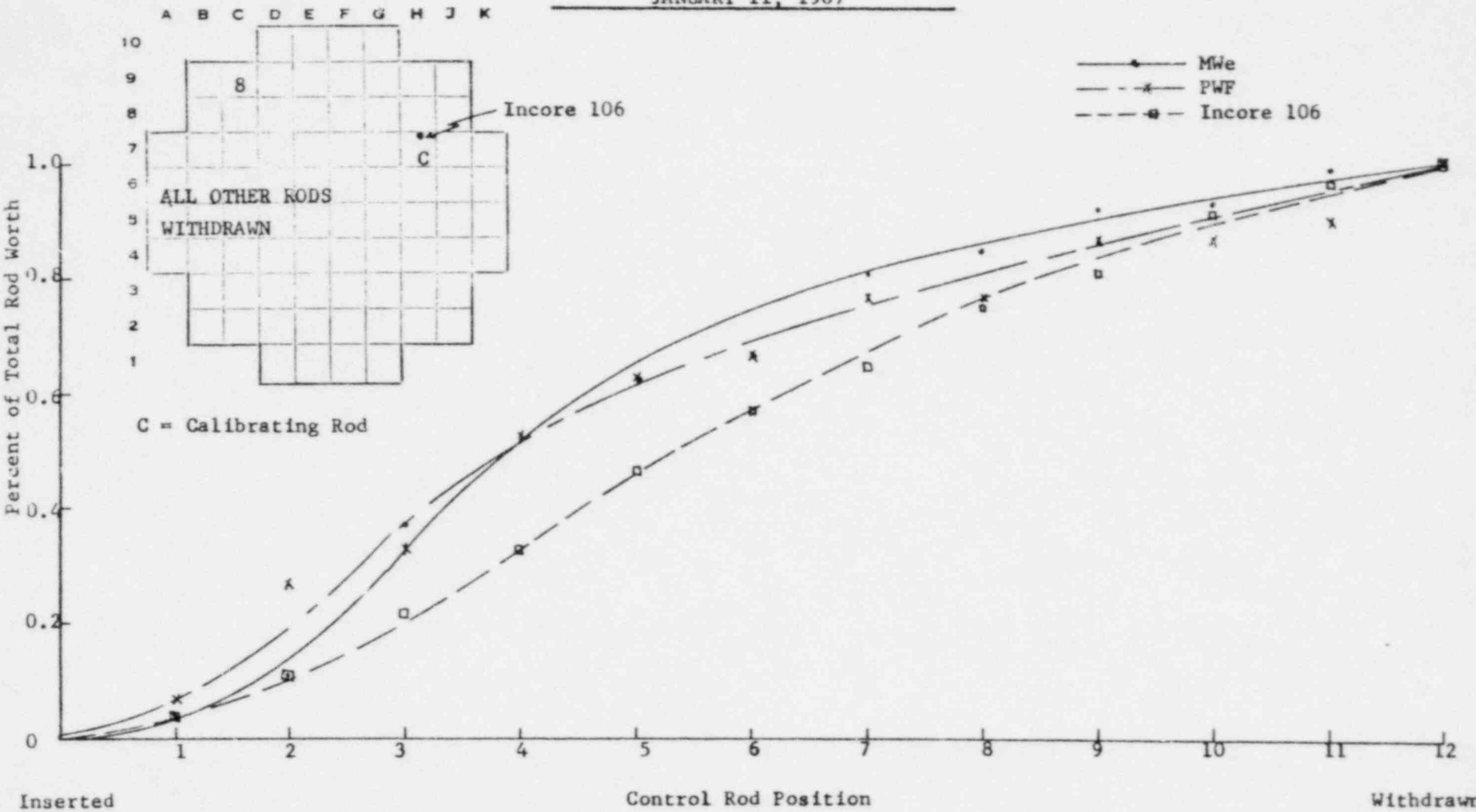
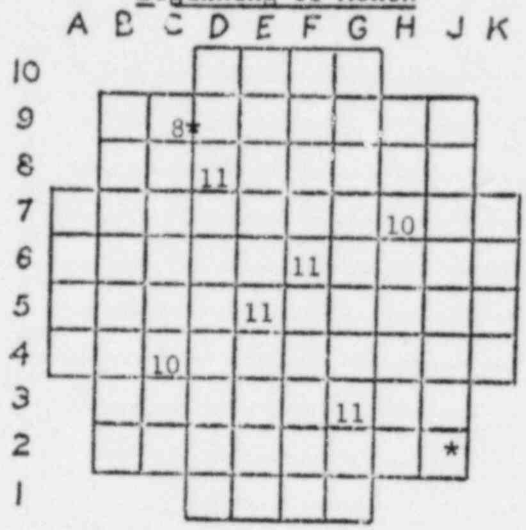


FIGURE 13

ROD PATTERNS DURING JANUARY

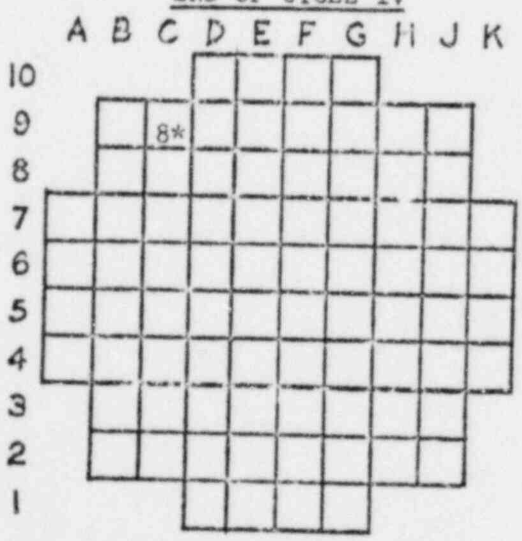
Beginning of Month



* Stuck Rod Valved Out

1/1/67
MWe - 160
MWt - 517
PWF - .97 x 10^6 lb/hr.
SWF - 1.14 x 10^6 lb/hr.
Residual - 1 rod

END OF CYCLE IV



* Stuck Rod Valved Out

1/13/67
MWe - 140
MWt - 453
PWF - .94 x 10^6 lb/hr.
SWF - .87 x 10^6 lb/hr.
Residual - 4 notches

f. Primary System Hydrostatic Tests

On April 19, 1967, a hydrostatic test of the primary system was conducted, resulting in two leaks being found at welds in the lower header tie serving the reactor clean-up demineralizer system. A full scale weld inspection, on primary piping 8" and under, followed.

At the completion of this inspection, May 17, the primary system was again tested at pressures of 700, 900 and 1,200 psig. The test resulted in the packing of several valves and the replacement of a pressure seal on the primary system isolation valve, MO-169. All other major components inspected were in satisfactory condition.

Following the October 23, through November 2, piping inspection, a successful hydrostatic test of the primary system was conducted at 1,205 psig. All inspections were satisfactory.

"D" loop remained out of service until December 12. At this time, with replacement of the 6" bypass loop complete, a hydrostatic test of 1,205 psig was conducted on the loop to demonstrate the integrity of the repairs. Results of the test and inspections were satisfactory.

g. Safety Valves

The five spare reactor drum safety valves were tested for relief pressure on the safety valve test fixture. (This fixture was described in the 1964 Annual Report). All valves were set at their respective design pressures $\pm$ 10 psi. Relief pressures were checked by repeated popping.

h. Air Locks

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

i. Spent Fuel Torque Tests and Capscrew Inspection

Assembly A-294 was dechanneled on March 25, 1967. The bale came loose as the element was placed into storage position F-30. The decision was then rendered to check the bale capscrews on a number of Type I spent fuel assemblies. This check was performed on April 2, 1967, and the results are tabulated in Table 8 with a guide to the capscrew location shown in Figure 14.

Another phase of the same check was to inspect all Type I fuels remaining in the reactor for missing capscrews. The results of this check are tabulated in Table 9.

j. Channel Inspection

The inspection of some of the spent fuel Zircaloy-2 channels was initiated on May 2, 1967. A total of 51 channels were inspected, including six which were closely examined. The close examination consisted of a visual inspection of the top and four sides of the channel with the channel as close to the top of the fuel pool as radiation levels would permit.

No damaged tabs or other defects were observed in the six closely examined channels.

An inspection of the other 45 channels showed four damaged channels. Two of the four had all four tabs broken, one had two broken tabs and one had one broken tab. All four cases showed the tabs broken in the weld area.

TABLE 8

TYPE I SPENT FUEL CAPSCREW TORQUE CHECK

Assembly Number	Added to Core	Exposure MWD/T	Results
A-203	BOC 1	10,684	All 8 capscrews torqued to 75 in lb. and OK
A-83	BOC 1	8,871	All 8 capscrews torqued to 75 in lb. and OK
A-155	BOC 1	9,995	All 8 capscrews torqued to 75 in lb. and OK
A-43	BOC 1	9,090	All 8 capscrews torqued to 75 in lb. and OK
A-352	BOC 1	9,669	All 8 capscrews torqued to 75 in lb. and OK
A-464	BOC 1	17,138	B-3 capscrew broken previously other seven torqued OK at 75 in lb.
A-359	BOC 1	9,950	B-4 capscrew broke with no torque other seven torqued OK at 75 in lb.
A-318	BOC 2	12,965	B-1 capscrew broke at 10 in lb. B-3 capscrew broke at 70 in lb. Other 6 torqued OK at 75 in lb.
A-333	BOC 2	11,621	A-2 capscrew broke at 20 in lb. Other seven torqued OK at 75 in lb.
A-284	BOC 2	11,404	All 8 capscrews torqued to 75 in lb. and OK
A-6	BOC 2	8,916	All 8 capscrews torqued to 75 in lb. and OK
A-8	BOC 2	11,399	B-3 capscrew broken previously other seven torqued OK at 75 in lb.
A-302	BOC 2	12,661	A-3 capscrew broke at 25 in lb. B-3 capscrew broke at 30 in lb. A-1, A-2, B-1, B-2 OK at 75 in lb. A-4 and B-4 not checked
A-328	BOC 2	12,947	A-3 capscrew broke at 10 in lb. B-1, B-2, B-3, B-4 OK at 75 in lb. A-1, A-2, A-4 not checked

TABLE 9

CYCLE V, TYPE I FUEL

In-Vessel Cap Screw Inspection

Assembly	Core Location	Exposure (MWD/T)	Capscrews Missing
A-60	7410	8853	B-2, A-3
A-111	7120	7234	B-4, A-4
A-40	6325	8625	B-2
A-97	6905	8142	A-1
A-74	5803	9083	1 Missing, Location Not Recorded
A-64	5703	7557	1 Missing, Location Not Recorded
A-82	5208	7445	B-3, B-4, A-1, A-2

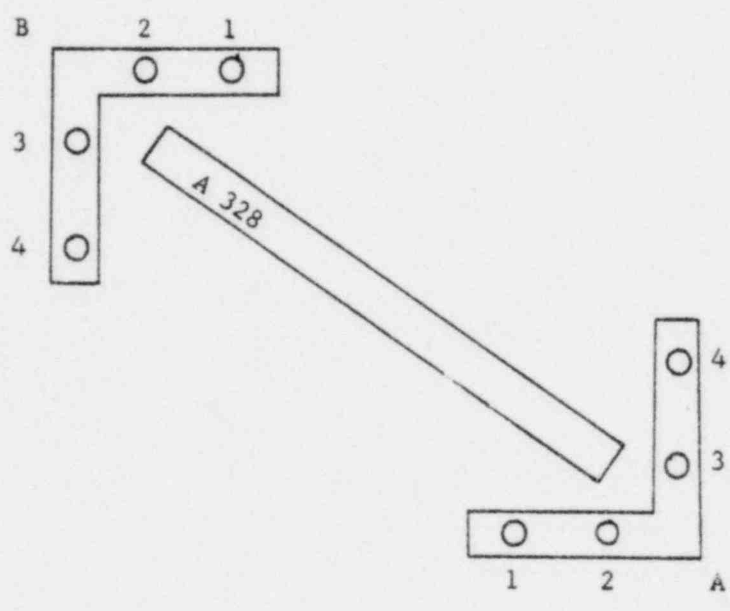
TABLE 9

TYPE I SPENT FUEL CAPSCREW TORQUE CHECK

Assembly Number	Added to Core	Exposure MWD/T	Results
A-15	BOC 1	12,643	All 8 capscrews torqued to 75 in lb. and OK
A-27	BOC 1	10,075	All 8 capscrews torqued to 75 in lb. and OK. A-1 capscrew was found backed off 3/4 turn.
A-18	BOC 1	8,830	A-4 capscrew broke with no torque A-3, B-1, B-2, B-3, B-4 OK at 75 in lb. A-1, A-2 not checked
A-146	BOC 1	10,463	B-1 capscrew broke with no torque A-1, A-2, A-3, A-4 OK at 75 in lb. B-2, B-3, B-4 not checked

FIGURE 14

CAPSCREW LOCATION KEY



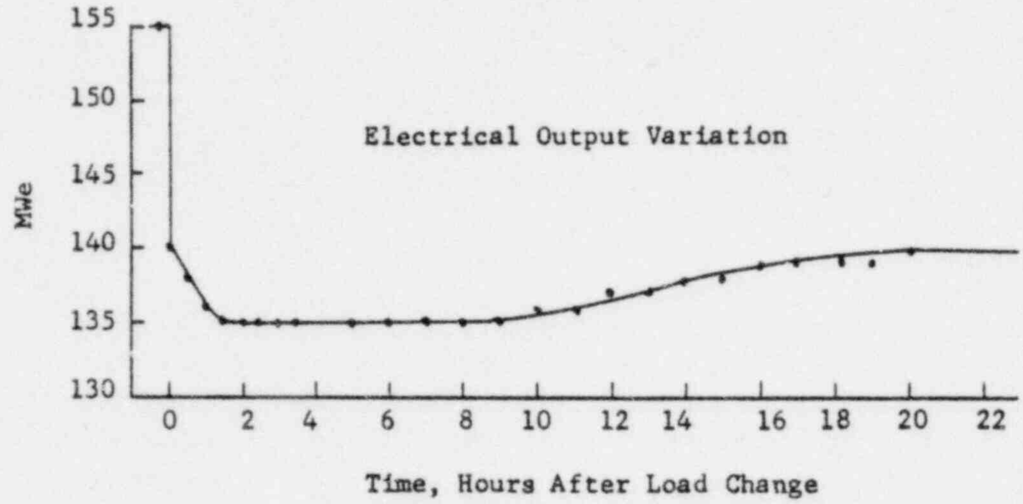
k. Reactor Stability and Xenon Oscillation

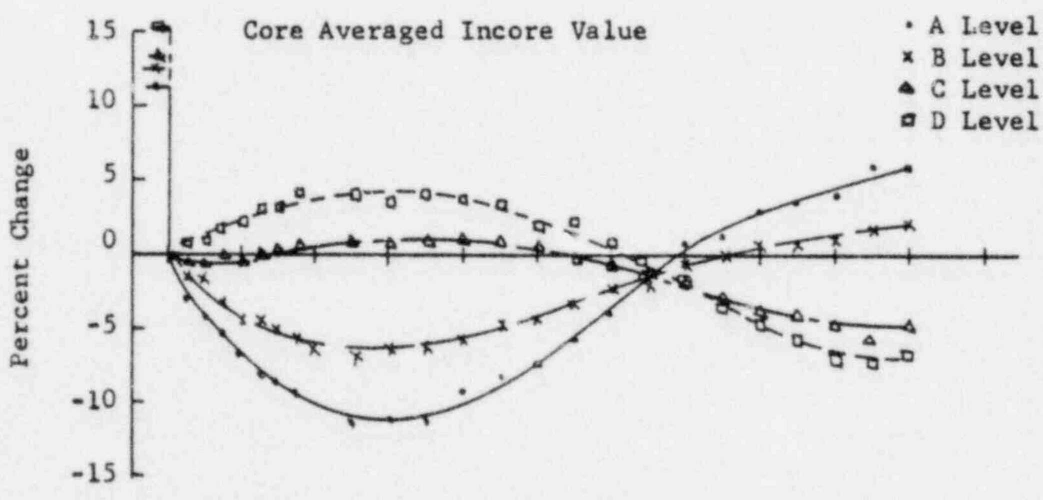
The effect of a load change on reactor stability and xenon oscillation was tested on January 12, 1967, by initiating a 15 MWe or 9.7 percent load reduction via use of the secondary control valves.

The secondary steam flow and rod pattern were kept constant over the next 24 hours. The power level and primary steam flow decreased, during the first two hours, remained essentially constant over the next seven and slowly returned toward the initiating load level in ten hours, as exhibited in Figure 15. The axial distribution in the meantime shifted toward the top of the core and reversed itself. The maximum average incore swing was experienced on "A" level approaching 12 percent of the level set by the load reduction. The maximum power level drop approached 33 percent of its initiating power step.

FIGURE 15

REACTOR STABILITY
LOAD STABILITY AND INCORE VARIATIONS RESULTING
FROM A STEP CHANGE ON POWER LEVEL AND XENON OSCILLATIONS





B. License DPR-2

Table 10 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 10

SUMMARY OF LICENSE AMENDMENTS
PENDING DURING 1967

	Date	
	Request	Authorization
Request to amend License DPR-2 to permit operation with 106 Type V Fuel Assemblies. (Change No. 13)	10/ 6/66	2/23/67
Request to amend License DPR-2 to permit operation with four Type III-F fuel assemblies using plutonium rods substituted for gadolinia urania rods. (Change No. 13)	1/ 9/67	2/23/67
Request to amend License DPR-2 to remove selected rods from Type SA-1 fuel assembly. (Change No. 13)	1/31/67	2/23/67
Request to amend License DPR-2 to permit operation with 96 Type VI fuel assemblies. (Change No. 14)	9/14/67	

Correspondence References - 1967:

(1) Letter to AEC dated January 9, 1967, requesting amendment of Appendix "A" of Operating License DPR-2 to allow operation of four Type III-F fuel assemblies with a plutonium rod substituted for gadolinia-urania rods. (Change No. 13)

(2) Letter to AEC dated January 13, 1967, regarding Summary of Operating Experience for 1966.

(3) Letter to AEC dated January 31, 1967, requesting change of Appendix "A" of Operating License DPR-2 to allow operation of fuel assembly SA-1, modified by removal of selected rods. (Change No. 13)

(4) Letter to Commonwealth Edison, dated February 1, 1967, notifying expiration of Byproduct Material License 12-5650-3.

(5) Letter to Commonwealth Edison, dated February 23, 1967, authorizing Change No. 13 to Operating License DPR-2.

(6) Letter to AEC dated March 7, 1967, applying for renewal of Byproduct Material License 12-5640-3.

(7) Letter to AEC dated April 12, 1967, regarding 1967 Refueling and Inspection Outage.

(8) Letter to AEC dated April 19, 1967, claifying points in letter of April 12, 1967.

(9) Letter to AEC dated June 12, 1967, regarding expansion of "Operating Experience Program" and Dresden Fuel Hoist Event.

(10) Letter to Commonwealth Edison Company dated June 26, 1967, granting extension of expiration dace for Byproduct Material Licnese 12-5650-3 to June 30, 1972.

(11) Letter to AEC dated September 14, 1967, requesting amendment of Appendix "A" of Operating License DPR-2 to provide operation of Dresden Unit 1 with Type VI replacement fuel batch. (Change No. 14)

(12) Letter to AEC dated November 27, 1967, regarding provisions for resumption of Reactor Operations following primary system piping inspections.