GPU 2494

W 39005

1-C-1332

GPM copies — B/W letter —
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W 39006

W 39007

INDEX

W 39068

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

June 29, 1978

REM-I-361

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Mr. J. P. O'Hanlon
Superintendent, Unit I
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Periodic Incore Detector Leakage Measurements

Gentlemen:

In order to reduce the inaccuracy of the NSS calculational package during cycle 4, due to leakage currents of the SPND's, it is recommended that periodic measurements be made.

Based upon the nominal cycle length of 265 EFPD between refuelings, B&W suggests the following schedule for the inclusion of insulation leakage correction factors to the SPNDI subroutine:

1. Once near the beginning of the core cycle (50 ± 10 EFPD) and once near the end of the cycle (200 ± 10 EFPD), a complete set of insulation leakage data should be taken.

   The newly installed low-leakage detectors are included to verify performance characteristics.

All leakage correction factors equal to or greater than 1.001 should be entered into the computer.

The above calibration rate will assure continued operation within the limits of Technical Specifications with respect to an accurate calculation of time integrated quantities.

W 39010

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/JHF/bay

cc: J. D. Phinney
J. T. Janis
V. P. Orlandi
D. L. Good
R. M. Klingaman
J. G. Herbein
R. A. Kunder

B+W BK

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

December 19, 1978

SOM-II-208
REM-I-371

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

~~Mr. G. P. Miller~~
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: FHE Site Caution Instructions

Gentlemen:

The FHE Site Caution Instructions listed here are supplied by Stearns-Roger and are forwarded for your use:

Attachments:
1. SCI #1 Dillon Calibration
2. SCI #2 Hydraulic Relief Valve Setting Procedure
3. SCI #3 Procedure for Field Repair of Dual Hose Reel
4. SCI #4 Fuel Grapple Disengage Bar
5. SCI #6 Emergency Pull Out Cable

It is suggested that you consolidate all SCI's into a section in the front of the Stearns-Roger FHE Instruction Manual. In addition to this section, SCI's concerning other particular sections of the FHE manual should also be inserted in these sections. In regard to the SCI associated with this transmittal, it is recommended that the following SCI's be included in the following additional sections of the manual:

SCI #1 Dillon Load Cell Section
SCI #2 Procon Pump Co. Section
SCI #3 Aeromotive Section

W 39012

In regard to SCI #1 (Dillon Calibration), the current recommendation to determine the settings for LS-3 (for use in Steps 21, 22, and 23) and LS-4 (for use in Steps 24, 25, and 26) is as follows:

Determine by an actual measurement the moving weight of the fuel assembly without any control components for both up and down directions in the core region.

Set LS-3 (underload trip) at 225 poinds below the minimum downward moving weight recorded in the core region.

Set LS-4 (overload trip) at 325 pounds above the maximum upward weight recorded in the core region.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/LNM/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. D. Phinney
J. T. Janis
J. G. Herbein
R. M. Klingaman
J. B. Logan
G. A. Kunder
J. L. Seelinger

Issued 10/30/78 Rev.	[illegible]	B&W No. 01

[illegible]

SITE CAUTION INST.

W 39614

DILLON CALIBRATION - (Wet)

1. Power switch "off".
2. Remove power cable from Dillon load cell.
3. Meter should read "0".
4. Adjust white adjusting screw on the center of the readout meter until meter pointer rests on "0".
5. Turn power switch "on", meter should still show "0". If not on "0" readjust to "0" using the white adjusting screw.
6. Push "calibrate" button, meter should read 3000#.
7. If meter does not read 3000#, adjust needle to 3000# using "tare adj." screw on cabinet.
8. Release "calibrate" button, meter should read "0".
9. If meter does not return to "0", repeat steps 4, 6 & 7 until both settings are stable and balanced.
10. Turn power switch "off".
11. Connect power cable to Dillon load cell.
12. Turn power switch "on".
13. Lower inner mast 12" to 18" using "jog down" switch.
14. Raise inner mast using "jog up" switch, while hoist is running up meter should read 750#. If meter does not read 750#, adjust tare to 750# while hoist is raising.

    NOTE: All limit switches must be set with the needle traveling. in the direction shown.

15. LS-1 Lowload - Descending -

    Rotate "tare" slowly counter clockwise, low load lite blue should illuminate as needle on readout reaches 400#, traveling toward "0".

16. If low load lite comes on above 400#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw counter clockwise on LS-1. Then repeat Step 15. If required, repeat Step 16 or 17 whichever is appropriate.

W 39615

17. If low load lite comes on below 400#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw first clockwise and then part way back counter clockwise. Then repeat Step 15. If required, repeat Step 16 or 17, whichever is appropriate.

SCI-01

18. LS-2 Engage - Disengage Interlock - Ascending -

Rotate "tare" slowly clockwise, the switch should open as the needle on the readout reaches 1200# in the ascending direction.

19. If the meter opens below 1200#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw clockwise on LS-2. Then repeat Step 18. If required, repeat Step 19 or 20 whichever is appropriate.

20. If the meter opens above 1200#, slowly turn the tare back to 750#, then slowly and minutely turn the adjusting screw counter clockwise on LS-2. Then repeat Step 18. If required, repeat Step 19 or 20, whichever is appropriate.

21. LS-3 Underload ________# - Descending -

Rotate "tare" slowly clockwise, till needle reads 2500#.

Place voltmeter lead on Wire 226 on terminal block in console and on Wire Y on control transformer in motor control center. Meter should read 120V. AC. Then rotate "tare" slowly counter clockwise. The limit switch should open as the needle on the readout reaches ________# in the descending direction.

22. If the meter opens above ____#, slowly turn tare back to 2500#, then slowly and minutely turn the adjusting screw counter clockwise on LS-3. Then repeat Step 20. If required, repeat Step 21 or 22, whichever is appropriate.

23. If the meter opens below ____#, slowly turn tare back to 2500#, then slowly and minutely turn the adjusting screw counter clockwise on LS-3. Then repeat Step 20. If required, repeat Step 22 or 23, whichever is appropriate.

24. LS-4 Overload ________# - Ascending -

Rotate tare slowly clockwise, the "red" overload indicating light should illuminate as the needle on the readout reaches ________#, traveling toward 3000#.

25. If the overload light comes on below ________#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw clockwise on LS-4. Then repeat Step 24. If required, repeat Step 25 or 26 whichever is appropriate.

26. If the overload light comes on above ________#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw counter clockwise on LS-4. Then repeat Step 24. If required, repeat Step 25 or 26, whichever is appropriate.

27. Turn tare back to 750#.

W 39616

28. Dillon is ready for use.

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Rev. 0		No. 02

SITE CAUTION INST.

W 39017

Procedure for Setting the Relief Valves
on the Handling Bridge and Transfer System
Hydraulic Systems.

1. The purpose of this procedure is to describe a method of adjustment which will insure a flow of water at all times through the pump. It is extremely important that the relief valve in the pump is set at a pressure of 10 to 20 psi higher than the system pressure which is controlled by the ½" external relief valve. Normally a system relief pressure of 200 psi and a pump relief pressure of 210 psi is satisfactory.

2. Remove acorn nut that locks the pump relief valve setting.

3. Remove the external relief valve adjusting screw cover.

4. Turn the external relief valve adjusting screw clockwise several turns to prevent by-passing until pump relief valve can be set to desired pressure (210 to 220 psi).

> CAUTION: Do not allow pump to by-pass internally for more than 5 minutes or excessive heat buildup will result which could damage the pump.

5. Turn the pump adjustment screw clockwise to raise or counter- clockwise to lower pressure to a setting of 210 to 220 psi.

6. Turn the external relief valve adjusting screw counter -clockwise to bring system pressure down to 200 psi.

7. Lock the pump adjusting screw by replacing the acorn nut and gasket. Tighten nut firmly.

8. Lock the external relief valve adjusting screw and replace the cover.

9. Allow the pump to run for several minutes and check temperature by feel. The pump body should be cool or slightly warm. Also the pump should not heat up even after extended periods of operation.

10. General Trouble Shooting.

10.1 Excessive noise.

a. Check pump mounting bolts

b. Check for loose drive coupling.

c. Check for improper relief valve setting by performing steps 1 thru 9.

W 39018

10.2 Excessive heat buildup in pump.

a. Check for improper relief valve setting by performing steps 1 thru 9.

10.3 System pressure will not hold at set point.

a. Check the lock nuts which lock the pump adjusting screw and the nut which locks the external relief valve.

b. Check fittings for leakage and tighten if necessary.

W 39019

SITE CAUTION INST.

W 33010

Procedure for Field Repair of Dual Hose Reel #YC1F4-C

General Information:

This procedure covers major field repair work that may be required on the Aero-Motive #YC1F4-C Dual Hose Reel. This reel is mounted on all Stearns-Roger Manipulator Cranes and located on the fuel mast connecting tube.

For major field repair work, the reel must be removed from the mast and taken to the shop for disassembly.

1. Raise fuel mast to an elevation that will allow the hoses to be disconnected from the mast thru the hand holes in the connecting tube. This elevation is usually just below the tube full up position.

2. Disconnect hoses from top of mast and allow hoses to be retracted on reel until all spring motor tension has been removed from reel.

3. Match mark and disconnect supply hoses to reel at termination of tubing on mast structure.

4. Remove reel mounting bolts and take reel to shop.

5. For spring motor repair, replacing main spring, or replacing swivel and shaft seals, see instruction sheet, page 3 and 4 of this procedure.

6. For repair or replacing of shaft assembly, see page 3, 4 and 5 of this procedure.

   Note: The mainshaft is made of stainless steel with brass fittings being silver soldered to seal 3000 psi. See Page 5.

In the event there is a leak between the mainshaft & the brass fittings, where they are silver soldered, they must be resilver soldered or the brass fittings removed, all traces of silver solder removed, both from the brass fitting and the mainshaft. Then they can be welded together. This will be a bimetal weld.

7. Replace reel on crame, mount, and connect hoses. Hoses at top of structure were match marked in Step 3. Retension spring so reel will retract properly at mast full up position.

Estimated Time: W 39621

Remove reel from unit	2 hours
Replace Shaft	4 - 6 hours
Repair Shaft - Add	4 hours
Install reel and test	3 hours

AEROMOTIVE A SUBSIDIARY OF DANIEL WOODHEAD, INC.
AERO-MOTIVE MGF. CO.
P.O. BOX 2478 KALAMAZOO, MICH. 49003
TEL: (616) 381-1242 • TELEX: 224420

INSTRUCTION AND PARTS DATA SHEET YZ-6-72
PART NUMBER 1174-P-32

Y AND Z SERIES
DUAL HYDRAULIC HOSE REEL

NOTE: Before attempting repairs, remove all hydraulic pressure and spring motor tension from reel.

TO REPLACE "Z" SERIES MOTOR:

Remove (4) screws (32-P-103) and (4) washers (101-P-21) holding spring motor to mounting plate (48344). Pull motor from plate disengaging reel shaft drive pin from motor shaft.

TO REPLACE SPRINGS IN "Z" MOTOR:

Place motor on end, back plate down. Remove (6) tie bolts (44894) and nuts (77-P-1). Lift off front cover and outside motor band (40120). Pull the drive shaft, with fixed hub, up and out while holding the center of the exposed spring in position. DO NOT allow the center of the spring to be pulled out. The spring and cup assembly may now be removed and individual parts "unstacked" in order. To reassemble, reverse the above procedure being certain that all parts are replaced in order and are properly aligned.

TO REPLACE "Y" SERIES MOTOR:

Remove retaining ring (580-P-75) and keyed washer (48331-1). Pull motor assembly from shaft. Spring hub will remain on main shaft. When replacing motor, align slot in spring hub with hook on inner end of main spring. Push motor onto reel shaft while aligning the locating pin on the motor housing with the locating pin hole of the frame assembly (44396).

TO REPLACE SPRINGS IN "Y" MOTOR:

Remove motor assembly from reel. Remove (6) screws (2-P-502) and washers (101-P-30) holding cover (44391) to motor housing (44393). Remove cover from housing. Check "holding rivets" at outer end of spring (31158) to be sure they are still secure. If so, lift spring from housing. DO NOT attempt to remove spring if holding rivets are broken. Return motor to factory for servicing. To reassemble motor, reverse the above procedure.

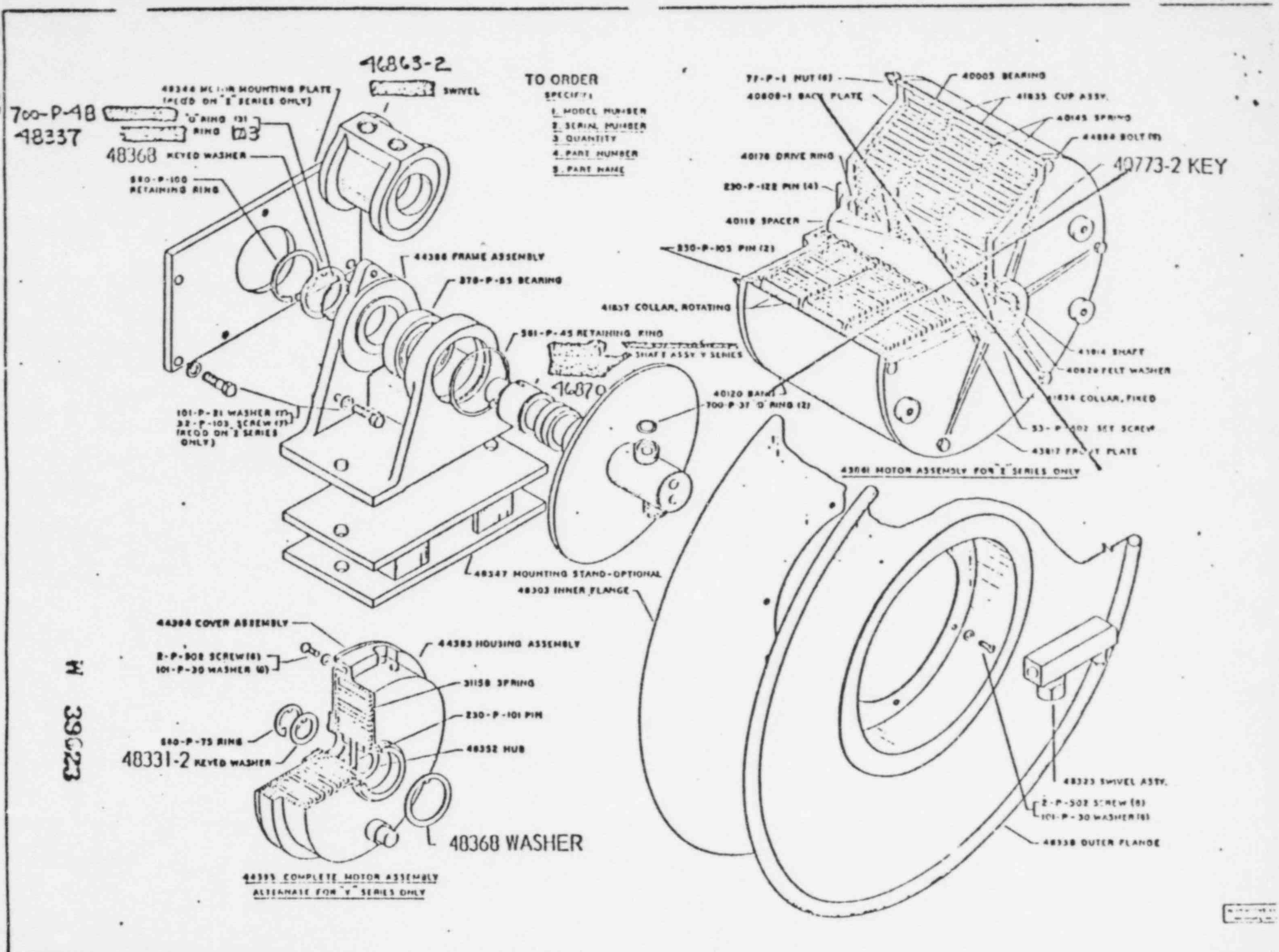
TO REPLACE SWIVEL AND SHAFT SEALS:

Remove spring motor assembly. Remove hose from reel and inner (48303) and outer (48338) flanges from main shaft. Remove retaining ring (580-P-100), (2) keyed washers (48368) and retaining ring (581-P-45) from main shaft and frame assembly. Pull main shaft, by reel end, out of swivel (48361-2) and frame assembly (44396). Remove old seals from shaft. Install (3) new "O" rings [illegible] and (3) new glyd rings (48337), one each, in each seal groove. Lubricate with light oil. Insert main shaft through frame assembly and into swivel being careful to not damage seals. Reverse the above procedure to complete reassembly.

LUBRICATION:

Reel is lifetime lubricated at factory. Further lubrication is neither required nor recommended.

W 39422

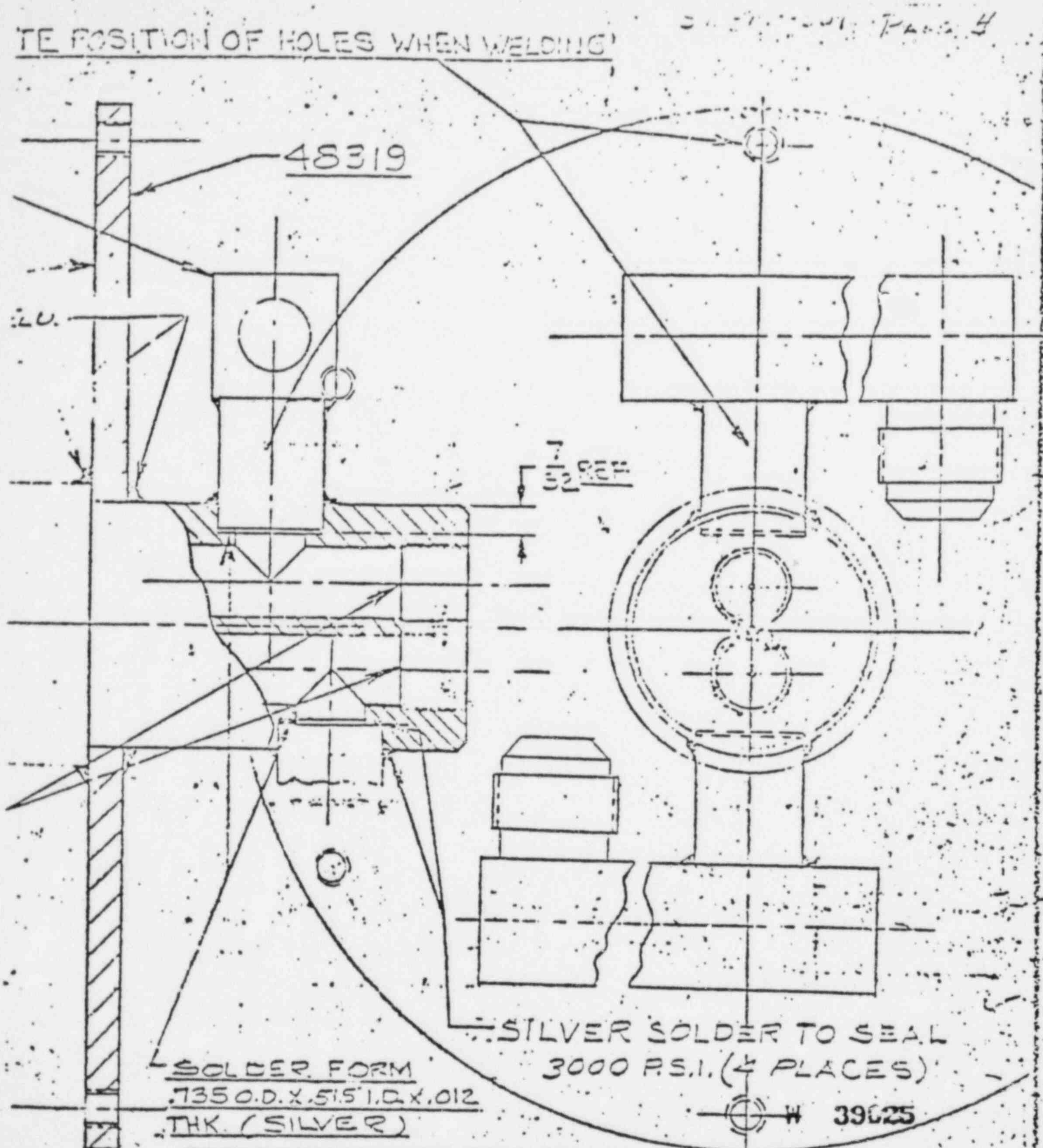

46863-2
SWIVEL
48344 MOTOR MOUNTING PLATE (REQD ON "Z" SERIES ONLY)
700-P-48
"O" RING (3)
48337
RING
48368 KEYED WASHER
880-P-100 RETAINING RING
44386 FRAME ASSEMBLY
370-P-55 BEARING
101-P-31 WASHER (?)
32-P-103 SCREW (?) (REQD ON "Z" SERIES ONLY)
TO ORDER
SPECIFY:
1. MODEL NUMBER
2. SERIAL NUMBER
3. QUANTITY
4. PART NUMBER
5. PART NAME
41837 COLLAR, ROTATING
381-P-45 RETAINING RING
SHAFT ASSY. "Y" SERIES
46870
48347 MOUNTING STAND-OPTIONAL
48303 INNER FLANGE
77-P-1 NUT (8)
40808-1 BACK PLATE
40178 DRIVE RING
230-P-122 PIN (4)
40119 SPACER
230-P-105 PIN (2)
40120 BAND
700-P-37 "O" RING (2)
40005 BEARING
41835 CUP ASSY.
40145 SPRING
44884 BOLT (8)
40773-2 KEY
41814 SHAFT
40820 FELT WASHER
41834 COLLAR, FIXED
53-P-102 SET SCREW
43817 FRONT PLATE
43061 MOTOR ASSEMBLY FOR "Z" SERIES ONLY
44384 COVER ASSEMBLY
2-P-308 SCREW (6)
101-P-30 WASHER (6)
44383 HOUSING ASSEMBLY
31158 SPRING
230-P-101 PIN
880-P-75 RING
48331-2 KEYED WASHER
48352 HUB
48368 WASHER
44395 COMPLETE MOTOR ASSEMBLY ALTERNATE FOR "Y" SERIES ONLY
48323 SWIVEL ASSY.
2-P-502 SCREW (8)
101-P-30 WASHER (8)
48338 OUTER FLANGE
W 39623

SCJ-03

Issued 10/30/78 Rev. 0	[illegible]	B&W No. 04

SITE CAUTION INST.

W 39624



MATERIAL	AERO-MOTIVE MFG. CO. KALAMAZOO, MICHIGAN			MODEL	
				NEXT ASSY	
	DR BY	CHECKED BY	APP'D BY	SCALE	NO. REQ'D
FINISH	DATE	DATE	DATE	FULL	1
	MAIN SHAFT ASSEM.			46870	

SCI-03

SITE CAUTION INST.

W 39026

FUEL GRAPPLE DISENGAGE BAR

This procedure is recommended for use at all sites in the event that the fuel grapple is engaged, unloaded, erroneously.

Required is a 2" x 2" x 12" long type 304 stainless steel bar (provided by site) to be attached to the floor on the reactor side.

Location of the bar should be at an index position on the selsyn dial for bridge position and at any grid location on the trolley positioning scale. A location near the test fixture would be preferable.

Index desired position on bridge and trolley, lower hoist and position bar under fuel grapple, so that two of the alignment blocks on grapple will rest directly on bar, mark location and attach bar to floor.

For the pool side, the same size bar can be clamped or welded to the top of a fuel storage rack, making sure the bar is located in the center of the rack so that it clears the grapple fingers.

Location of the bar should be at an index position on the selsyn dial for bridge position and at any grid location on the trolley positioning scale.

Index desired position on bridge and trolley, lower hoist and position bar under fuel grapple, so that two of the alignment blocks on grapple will rest directly on bar, mark location and attach bar to rack.

W 39027

Emergency Pull Out Cable

In the event it is necessary to use the emergency pull out cable to return the transfer carriage to its storage position, the following equipment and steps are required.

Equipment:

1-Klein Hot Line "Chicago" grip.
Model 1628-5BH (or equal)
1-Load Sensing Device (6000 #CAP)

The load sensing device should be hooked on the hook of the overhead or gantry crane.

The "Chicago" grip should be attached to the load sensing device.

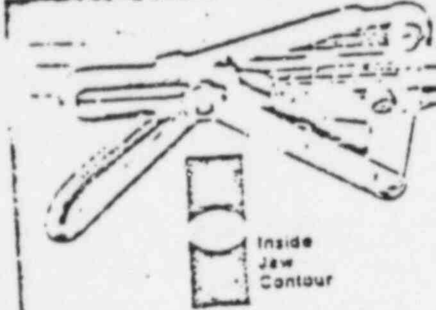
Pull enough cable off of the emergency pullout cable reel to allow the full engagement of the "Chicago" grip on the cable going to the carriage. Make sure that crane is positioned directly in line with the cable to insure a straight pull on the cable.

Slowly raise crane hook, car should start to move at or before 1000# is shown on the load sensing device. As the hook is raising, be sure to watch cable on the reel so that you do not pull it off the reel.

Raise hook 3 or 4 feet, then grab cable leading to transfer car, slide "Chicago" grip down the cable 3 or 4 feet, let cable reel on to the reel. After the first initial pull, the distance the hook can be raised can be increased to 6 or 8 feet.

At no time should the load sensing device read more than 2000# during a carriage retrieval. If a load greater than 2000# is applied, the work should be stopped and the problem evaluated prior to proceeding.

W 39028

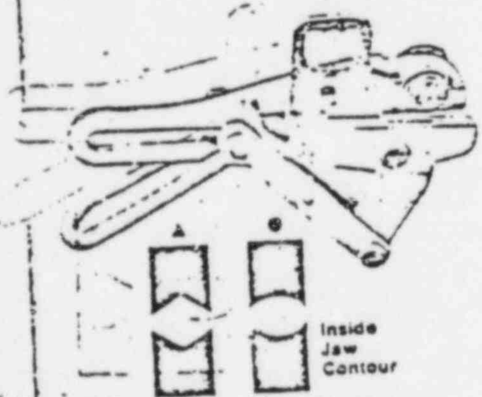


Grips for weatherproof wire

Round inside jaw contour provides maximum protection for weatherproof coatings. Notches in jaw provide firmer grip on insulation.

Cat. No.	Minimum Cable Size—Inches (mm)	Maximum Cable Size—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Jaw Length inches
1611-20	.20 (5.08)	.40 (10.20)	4500 (2041)	3 (1.36)	4
1611-30	.25 (6.35)	.53 (13.46)	4500 (2041)	3¾ (1.70)	4¾
1611-40	.53 (13.46)	.74 (18.80)	8000 (3629)	7¾ (3.52)	[illegible]
1611-50	.78 (19.81)	.88 (22.35)	8000 (3629)	7¾ (3.52)	[illegible]

Note: Due to various types of weatherproof coatings available, selection of proper ... is determined by outside diameter of cable.



Hot Line "Chicago" grips

Klein provides Hot Line Grips for both bare and insulated conductors. Grip may be placed on wire with hot line stick. When stick is removed, safety latch closes automatically to guard against grip accidentally disengaging from wire.

Standard Hot Line Grips are not su... with springs or lock-open featur... these features are desired, pref... "S" before catalog number (ex... S1628-5BH).

For Bare Conductors

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Le... inc...
▲ 1613-40BH	10 B&S Solid .10 (2.54)	1/0 B&S Strand .37 (9.40)	4500 (2041)	3¼ (1.47)	4
▲ 1628-5BH	6 B&S Solid .16 (4.06)	4/0 A.C.S.R. .55 (13.97)	8000 (3629)	6¼ (2.84)	5
● 1656-20BH	6 A.C.S.R. .20 (5.08)	1/0 A.C.S.R. .40 (10.16)	4500 (2041)	3¼ (1.47)	4
● 1656-30BH	4 A.C.S.R. .25 (6.35)	3/0 A.C.S.R. .50 (12.70)	4500 (2041)	4 (1.81)	4
● 1656-40BH	3/0 A.C.S.R. .50 (12.70)	336,400 CM A.C.S.R. .74 (18.80)	8000 (3629)	7¾ (3.52)	[illegible]
● 1656-50BH	397,500 CM A.C.S.R. .78 (19.81)	477,000 CM A.C.S.R. .88 (22.35)	8000 (3629)	7¾ (3.52)	[illegible]
● *1656ABH	3/0 A.C.S.R. .50 (12.70)	500,000 CM A.C.S.R. .90 (22.86)	8000 (3629)	7¾ (3.52)	[illegible]

*For use on aluminum cables where minimum conductor damage is essential, we hold stock of parts, and jaws are m... to ... Range of conductor size is given for general information and is made for one size cable only. *Orders must s... ameter of cable. ... exact outside diameter of cable.*

W 39629

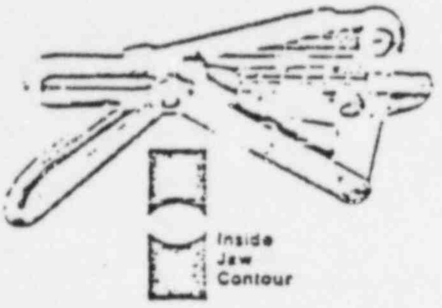
For Insulated Conductors

Same as 1659 series grips on page 42, except for addition of hot line latch

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)
● 1659-20H	.20 (5.08)	.42 (10.67)	4500 (2041)	3¼ (1.47)
● 1659-30H	.31 (7.87)	.50 (12.70)	4500 (2041)	4 (1.81)
● 1659-40H	.49 (12.45)	.70 (20.07)	8000 (3629)	7¾ (3.52)
● 1659-50H	.79 (20.07)	1.01 (25.65)	8000 (3629)	7¾ (3.52)

Note. Duo to various types of PVC and other weatherproof coatings, selection of proper grips is determined by O.D. of cable.

Klein "Chicago"® Grips CONTD

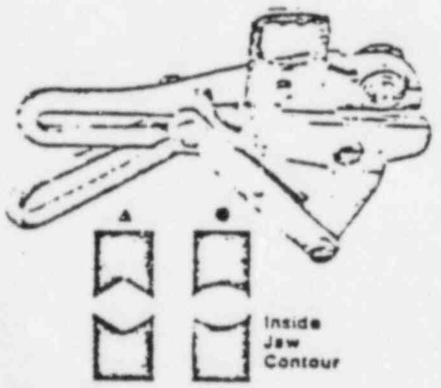


Grips for weatherproof wire

Round inside jaw contour provides maximum protection for weatherproof coatings. Notches in jaw provide firmer grip on insulation.

Cat. No.	Minimum Cable Size—inches (mm)	Maximum Cable Size—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Jaw Length—inches (cm)
1611-20	.20 (5.08)	.40 (10.20)	4500 (2041)	3 (1.36)	4 (10.2)
1611-30	.25 (6.35)	.53 (13.46)	4500 (2041)	3[illegible] (1.70)	4¾ (12.1)
1611-40	.53 (13.46)	.74 (18.80)	8000 (3629)	7¾ (3.52)	5½ (14.0)
1611-50	.78 (19.81)	.88 (22.35)	8000 (3629)	7¾ (3.52)	5½ (14.0)

Note: Due to various types of weatherproof coatings available, selection of proper grips is determined by outside diameter of cable.



Hot Line "Chicago" grips

Klein provides Hot Line Grips for both bare and insulated conductors. Grip may be placed on wire with hot line stick. When stick is removed, safety latch closes automatically to guard against grip accidentally disengaging from wire.

Standard Hot Line Grips are not supplied with springs or lock-open feature. When these features are desired, prefix letter "S" before catalog number (example: S1628-5BH).

For Bare Conductors

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Jaw Length—inches (cm)
▲ 1613-40BH	10 B&S Solid .10 (2.54)	1/0 B&S Strand .37 (9.40)	4500 (2041)	3¼ (1.47)	4 (10.2)
▲ 1628-5BH	6 B&S Solid .16 (4.06)	4/0 A.C.S.R. .55 (13.97)	8000 (3629)	6¼ (2.84)	5 (12.7)
● 1656-20BH	6 A.C.S.R. .20 (5.08)	1/0 A.C.S.R. .40 (10.16)	4500 (2041)	3¼ (1.47)	4 (10.2)
● 1656-30BH	4 A.C.S.R. .25 (6.35)	3/0 A.C.S.R. .50 (12.70)	4500 (2041)	4 (1.81)	4¾ (12.1)
● 1656-40BH	3/0 A.C.S.R. .50 (12.70)	336,400 CM A.C.S.R. .74 (18.80)	8000 (3629)	7¾ (3.52)	5½ (14.0)
● 1656-50BH	397,500 CM A.C.S.R. .78 (19.81)	477,000 CM A.C.S.R. .88 (22.35)	8000 (3629)	7¾ (3.52)	5½ (14.0)
● *1656ABH	3/0 A.C.S.R. .50 (12.70)	500,000 CM A.C.S.R. .90 (22.86)	8000 (3629)	7¾ (3.52)	5½ (14.0)

*For use on aluminum cables where minimum conductor damage is essential, we hold stock of parts, and jaws are milled to fit exact diameter of cable. Range of conductor size is given for general information and is made for one size cable only. *Orders must specify exact outside diameter of cable.*

For Insulated Conductors

Same as 1659 series grips on page 42, except for addition of hot line latch.

W 39030

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Jaw Length—inches (cm)
● 1659-20H	.20 (5.08)	.42 (10.67)	4500 (2041)	3¼ (1.47)	4 (10.2)
● 1659-30H	.31 (7.87)	.50 (12.70)	4500 (2041)	4 (1.81)	4¾ (12.1)
● 1659-40H	.49 (12.45)	.79 (20.07)	8000 (3629)	7¾ (3.52)	5½ (14.0)
● 1659-50H	.79 (20.07)	1.01 (25.65)	8000 (3629)	7¾ (3.52)	5½ (14.0)

Note: Due to various types of PVC and other weatherproof coatings, selection of proper grips is determined by O.D. of cable.

SeI-06

B+W BK

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
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December 18, 1978

REM-I-372
REM-II-36

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Mr. J. L. Seelinger
Superintendent, Unit I
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Nuclear Spare Parts Service

Gentlemen:

With an ultimate goal of improving the availability and capacity factors of Nuclear Steam Supply Systems (NSSS) and facilitating B&W users, Babcock and Wilcox has organized a Nuclear Parts Center with the full status of a department in the Nuclear Power Generation Division.

The objective of the Nuclear Parts Center (NPC) is to provide spare parts for NSSS related equipment most expeditiously and at a competitive price. To achieve the economies of scale and to ensure proper Quality Assurance and Control, the Nuclear Parts Center has developed a good working organization and a parts distribution center.

W 39031

The Nuclear Parts Center has the services of B&W Engineering to determine 10CFR21 Section B applicability for NSSS related parts. The Nuclear Parts Center has existing in-roads into the vendor organizations and by judicious consolidation of the spare parts needs of the different B&W users, the Center can order the NSSS related spare parts in quantities which would accrue the economy of scales, and the resulting savings can be passed on to the B&W users. By passing on the onus for applicability and requirements of 10CFR21 Section B to the Nuclear Parts Center, an avoidable duplication of engineering review can be eliminated. This way B&W users can reduce a substantial load on their engineering staff.

The Nuclear Parts Center has acquired its distribution center where shipments from vendors will be received and passed on to B&W users after scrutiny. Distribution Center will also hold an inventory of spare parts in order to handle plant emergencies. The Center is in the process of being stocked up to the optimal level.

The Nuclear Parts Center is now in the process of updating "Recommended Spare Part Levels" that should be stocked by the NSSS users based upon equipment performance and machinery history. Once updated, the recommended spare part lists will be passed on to the B&W users, and this list may be used as a guideline to build up plant spare part stocks.

To assist the B&W users further and to improve the spare parts service, the Nuclear Parts Center is providing "On Site Representation" on a trial basis. The undersigned has been assigned as the Nuclear Parts Center site representative at the Three Mile Island Site. I will provide Nuclear Parts Center liaison with TMI Site, Engineering, Warehouse, and QA/QC staff.

I am confident that I can be of real help to your staff in the endeavor to build up NSSS spare parts stock to eventually improve the plant availability and capacity.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

SPMaingi

S. P. Maingi

L. C. Rogers
Site Operations Manager

cc: J. G. Herbein
G. A. Orlicki
L. R. Weissert

W 39032

Babcock & Wilcox

Power Generation Group
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May 25, 1979

REM-I-391
REM-II-38

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Fuel Transfer Basket Load Distribution

Dear Gary:

Some B&W sites have reported operation difficulties with the Fuel Transfer System when used for transfer of orifice rods or components other than fuel assemblies or control components. Uneven weight distribution in the basket prevented the basket from lowering fully, resulting in damage or stuck carriage when attempting transfer.

B&W advises that if the Fuel Transfer System is used for transfer of anything other than fuel assemblies or control rod assemblies, the operators should be particularly attentive to ensure that uneven weight distribution does not prevent the basket from lowering to the full down and latched position.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

S P ...

for L. C. Rogers
Resident Engineer Manager

LCR/SPM/lbs

cc: J. D. Phinney
G. T. Fairburn
R. M. Klingaman
J. G. Herbein
J. L. Seelinger
L. L. Lawyer
W. E. Potts
L. R. Pletke
W. H. Spangler
G. A. Kunder

W 39033

Babcock & Wilcox

Power Generation Group

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Telephone: (804) 384-5111

May 25, 1979

REM-I-392
REM-II-39

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Fuel Transfer System Cable Takeup Reels

Dear Gary:

Operating problems with Fuel Transfer System cable takeup reels at some of the B&W sites have been the result of a breakdown of lubricants in the subject reels. Some instances have resulted in cables being cut by the transfer carriage when the reel did not take up properly.

B&W recommends the use of "Never Seez" as an approved lubricant for this application. Additionally, stripes should be painted on all transfer system takeup reels in such a manner that the operator can see, at a glance, that the reel is turning and taking up properly.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

for L. C. Rogers
Resident Engineer Manager

LCR/SPM/bay

cc: J. D. Phinney
G. T. Fairburn
R. M. Klingaman
J. G. Herbein
J. L. Seelinger
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L. R. Pletke
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W 39634

Babcock & Wilcox

Power Generation Group

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May 25, 1979

REM-I-393
REM-II-40

Mr G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Fuel Handling Bridge Hydraulic Hose Deterioration

Dear Gary:

At some of the B&W plants there has been some deterioration at the outer layer of material of the rubber hydraulic hoses on the Fuel Handling Bridges. This deterioration can cause flaking of the outer layer of material which may lead to contamination of the refueling canal and ultimately the RCS. These were Weatherhead brand hoses which had been as replacements for the originally installed Gates rubber hoses.

The rubber hydraulic hoses on all Fuel Handling Bridges should be inspected at the next available opportunity and replaced if evidence of deterioration is noted. The approved replacement material is Gates rubber hose #119-B, and may be obtained through the Spare Parts Center. Additionally, these rubber hoses should be replaced every two years thereafter as a preventive maintenance measure.

If you have any further questions, please do not hesitate to contact me.

Very truly yours

Stamainger

for L. C. Rogers
Resident Engineer Manager

LCR/SPM/lbs

cc: J. D. Phinney
G. T. Fairburn
R. M. Klingaman
J. G. Herbein
J. L. Seelinger
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W. E. Potts
L. R. Pletke
W. H. Spangler
G. A. Kunder

W 39035

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W 39039

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W 39041

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#	Description

W 39042

SOM-II
B&W LETTERS

W 39043

[redacted]

Babcock & Wilcox

Power Generation Group

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March 20, 1978

REM-I-342 Rev. 1
SOM-II-107 Rev. 1

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Mr. J. P. O'Hanlon
Superintendent, Unit I
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Mr. R. J. Toole
Test Superintendent
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Subject: Revision to OTSG Level Requirements During Plant Shutdown and Startup

Gentlemen:

As you are aware, Three Mile Island Unit I and II Once-Through Steam Generators (OTSG) have seal welded feedwater nozzles; and these nozzles are susceptible to circumferential thermal shocks whenever feedwater flow is very low. To minimize the thermal cycles on these nozzles, B&W previously advised Met-Ed/GPU to maintain OTSG feedwater level high enough to keep the feedwater nozzles flooded at a reactor power less than 5 percent of rated thermal power.

W 39044

It has now come to B&W's attention that this practice does not satisfy the initial conditions of the accident analysis and is, thus, an unanalyzed situation. The steam-line-break analysis assumes a minimum OTSG inventory at low power levels while flooded nozzles constitute a maximum inventory. Accordingly, revised operating guidelines are provided in Attachment 1. These guidelines represent a compromise that satisfies the initial conditions of the accident analysis while minimizing the cycles on the feedwater nozzles.

Note that before OTSG level is raised to flood the feedwater nozzles, the reactor must be subcritical by at least 4% $\Delta k/k$ to ensure maintaining subcriticality in the event of a steam line break. Since the boron concentration to be subcritical by 4% $\Delta k/k$ at 532°F never exceeds that concentration needed to get a 1% $\Delta k/k$ shutdown margin at 70°F, no additional boration is required for cooldowns. No additional boron is required for maintaining the plant >500°F in hot standby (Mode 3) for short periods (<24 hours); however, if the plant is to be maintained >500°F beyond the point of Xenon decay below its 100% equilibrium level, additional boron may be required to maintain 4% subcriticality.

The feedwater bypass valves (FW-V-66A&B) should remain open to provide a 32 gpm bypass flow anytime the OTSG is not in full wet layup with level control via lower tube sheet drains and sample lines (i.e., continuous blowdown).

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/JV/bay

cc: L. R. Pletke
W. H. Spangler
T. F. Scott
J. G. Herbein
R. M. Klingaman
J. T. Janis
J. D. Phinney

W 39045

ATTACHMENT I

OPERATING GUIDELINES FOR MAINTAINING OTSG LEVEL

SHUTDOWN TO HOT STANDBY (RCS $>$500°F)

1. Decrease reactor power to 0% Rated Thermal Power (RTP). The OTSG level should be at low level setpoint.
2. Insert all Control Rod Groups (CRG) 1 through 7, verify all rods at 0% WD.
3. If necessary, borate RCS to 4% Δk/k subcritical at 532°F.
4. Raise the OTSG levels to 97-99% on the operate range.

SHUTDOWN AND COOLDOWN (RCS $<$500°F)

1. Decrease reactor power to 0% RTP.
2. Insert CRG-2 through 7 to 0% WD, retain CRG-1 at 100%WD.
3. Verify CRG-2 through 7 are 0% WD, verify CRG-1 is movable
4. Commence cooldown, makeup with boric acid to reach the 70°F, 1% Δk/k shutdown margin concentration (this exceeds the 532°F, 4% Δk/k subcritical concentration). This concentration should be reached while above 500°F.
5. Raise OTSG levels to 97-99% on operate range, then continue cooldown.

STARTUP

1. Increase T_{ave} to 532°F (CRG-1 is 100% WD, CRG-2 through 7 is 0% WD, boron concentration is at the 70°F, 1% Δk/k shutdown margin value).
2. Deborate from the 70°F, 1% Δk/k shutdown margin value to the 532°F, 4% subcritical value or the desired critical boron value, whichever is higher.
3. Lower OTSG level to low level set (this is first step if starting up from hot standby, RCS $>$500°F)
4. Deborate as needed to the desired critical boron value.
5. Withdraw control rods to achieve criticality

Cc: JLS File 351.1
G. Miller

Babcock & Wilcox

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March 9, 1978

SOM-II-117

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Mr. R. J. Toole
Station Superintendent
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Subject: Removal of Conax Fitting Protectors

Gentlemen:

Attached are the procedures for removal and replacement of the conax fitting protector cans for both the manway cover and the inspection cover on the TMI-2 Steam Generator (2-B). This procedure is applicable anytime the cans need to be removed for inspection of the conax fittings or for any other reason. Also included in the attached procedure are torque values for the conax fittings to be used should retorquing be necessary.

B&W would appreciate your providing a description of any activities associated with the use of the attached procedures.

If you have any further questions, please do not hesitate to contact me.

W 39047

Very truly yours,

E B [signature] for

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

cc: L. R. Pletke
W. H. Spangler
T. F. Scott
J. G. Herbein
R. M. Klingaman

W 39048

TMI-2 OTSG INSTRUMENTATION PROGRAM

CONAX FITTING TORQUEING PROCEDURE

FOR INSPECTION OPENING

Prepared By *[signature]* Date 2-21-78

Reviewed By *[signature]* Date 2-21-78

Approved By *[signature]* Date 2-21-78

Issued 2-21-78

W 39049

CONAX FITTING TORQUEING PROCEDURE

FOR INSPECTION OPENING

1.0 Introduction

1. In using this procedure to torque conax fittings, a Babcock & Wilcox engineering representative should be notified and the same should be on site to supervise unless this authority is otherwise delegated.

2. Conax fittings are required to be torqued per this procedure after each thermal cycle as defined in 2.0 below.

3. The thermal cycle definition does not exclude that this procedure may be used during intermediate cycles of less temperature and pressure than that described below.

2.0 Definition of a Thermal Cycle

1. The thermal cycle is defined as reaching a primary temperature of 532 degrees F and returning to room temperature, and simultaneously reaching a primary pressure of 1400 psig at temperature.

W 39050

3.0 Parts and Equipment Required

1. Torque Wrench - Range includes 45 ft-lbs to 250 ft-lbs
2. 3/4 inch Hex Socket - Fits above Torque Wrench
3. 9/16 inch Hex Socket - Fits above Torque Wrench
4. Split Socket Assembly (Part No. 98)
5. Tin Snips
6. Screw Driver - For adjusting Hose Clamps (Part No. 188)
7. Banding Equipment
8. Nylon Ties (Part No. 186)
9. Tuct Tape (Part No. 212)
10. Fiber Glass Tape (Part No. 137)
11. 1-5/8 inch Hex Socket (for 1 inch Heavy Hex Nut) - Fits above Torque Wrench

4.0 Procedure

Reference Drawings:

1003170C Instrumentation Lead Routing from Inspection Opening

1003009D Inspection Opening Cover Assembly

NOTE: Part Nos. referred to in this procedure are identified on the above reference drawings.

1. Remove OTSG mirror insulation from around inspection opening cover assembly (Part No. 139) to allow access for removing the cover.
2. Remove nylon cable ties (Part No. 186) that are tied around the split hose conduit (Part No. 197) for approximately three feet out from the inspection cover assembly.

W 39051

3. Remove tuct tape (Part No. 187) from the split hose conduit for approximately 3 feet out from the inspection cover assembly.

4. Using a screw driver, remove two hose clamps (Part No. 183) from the split hose conduit at the inspection cover assembly.

5. By hand, pull open the split hose conduit and peel off and away from the hard line instrumentation cables. Approximately 3 feet of peel back is required to allow easy removal of the inspection cover assembly.

   NOTE: In removing the inspection cover assembly, the initial movement is perpendicular to the inspection penecration cover.

6. Using tin snips, cut the steel banding away from around the external inspection cover assembly.

7. Remove two 3/8 inch hex nuts (Part No. 210) from end of inspection cover assembly.

8. Remove the cover (Part No. 143) from over the slot in the inspection cover assembly.

9. Remove the cover adapter (hose adapter) (Part No. 196) from the inspection cover assembly.

   NOTE: Several turns of fiber glass tape may have to be removed first.

10. Remove four 1 inch Hex nuts (Part No. 110) which connect the inspection cover assembly to the inspection penetration cover.

11. Remove the inspection cover assembly (Part No. 139). Lift first perpendicularly away from the inspection penetration cover to clear the conax fittings and converging instrumentation leads. Then manipulate so the slot in the cover is guided over the lead bundle.

W 39052

NOTE: Cover weighs about 30 lbs and can be handled easily by one man.

12a. At this stage, all conax fittings are accessible for torqueing.

b. If there are any indications that any conax fitting is leaking, contact a B&W engineering representative, NPGD Lynchburg, (J. W. Mitchem) prior to proceeding further.

13. Using the torque wrench, 3/4 inch hex socket, and split socket assembly, torque all conax fittings (12 total) to 200 ± 20 ft-lbs.

14. Replacement of the inspection cover assembly and conduit accessories will be the reverse sequence of the preceeding steps, and is simplified in the following steps.

15. Replace inspection cover assembly.

16. Replace the four 1 inch hex nuts and torque to 175 ± 10 ft-lbs.

17. Replace the cover adapter (hose adapter) and wrap with several turns of fibe glass tape to hold in position.

18. Replace the cover over the inspection cover slot.

19. Replace the two 3/8 inch hex nuts and torque to 50 ± 5 ft-lbs.

20. Replace the steel banding, one or two turns strapped tightly, so that the cover over slot cannot vibrate.

21. Replace the split hose conduit over the instrumentation leads.

22. Replace the two hose clamps.

23. Replace the tuct tape fully wraping the conduit area where tape had been previously removed.

24. Replace the nylon ties as required.

25. Replace the OTSG mirror insulation.

W 39053

TMI-2 OTSG INSTRUMENTATION PROGRAM

CONAX FITTING TORQUEING PROCEDURE

FOR MANWAY OPENING

Prepared By *[signature]* Date 2-21-78

Reviewed By *[signature]* Date 2-21-78

Approved By *[signature]* Date 2-21-78

Issued 2-21-78

W 39054

CONAX FITTING TORQUEING PROCEDURE

FOR MANWAY OPENING

1.0 Introduction

1. In using this procedure to torque conax fittings, a Babcock & Wilcox engineering representative should be notified and the same should be on site to supervise unless this authority is otherwise delegated.

2. Conax fittings are required to be torqued per this procedure after each thermal cycle as defined in 2.0 below.

3. The thermal cycle definition does not exclude that this procedure may be used during intermediate cycles of less temperature and pressure than that described below.

2.0 Definition of a Thermal Cycle

1. The thermal cycle is defined as reaching a primary temperature of 532 degrees F and returning to room temperature, and simultaneously reaching a primary pressure of 1400 psig at temperature.

3.0 Parts and Equipment Required

1. Torque Wrench - Range includes 45 Ft-Lbs to 250 Ft-Lbs.
2. 3/4 inch Hex Socket - Fits above Torque Wrench
3. 9/16 inch Hex Socket - Fits above Torque Wrench
4. Split Socket Assembly (Part No. 98)
5. Tin Snips
6. Screw Driver - For adjusting Hose Clamps (Part No. 188)
7. Banding Equipment
8. Nylon Ties (Part No. 186)
9. Tuct Tape (Part No. 212)
10. Fiber Glass Tape (Part No. 187)

4.0 Procedure

Reference Drawings:

1003169C Instrumentation Lead Routing from Manway Opening

1002988D Manway Cover Assembly

1002987C Cover Assembly

NOTE: Part Nos. referred to in this procedure are identified on the above reference drawings.

1. Remove OTSG mirror insulation from around manway cover assembly (Part No. 132) to allow access for removing the cover.
2. Remove nylon cable ties (Part No. 186) that are tied around the split hose conduit (Part No. 197) for approximately three feet out from the manway cover assembly.

W 39056

3. Remove tuct tape (Part No. 187) from the split hose conduit for approximately 3 feet out from the manway cover Assembly.

4. Using a screw driver, remove two hose clamps (Part No. 188) from the split hose conduit at the manway cover connection.

5. By hand, pull open the split hose conduit and peel off and away from the hard line instrumentation cables. Approximately 3 feet of peel back is required to allow easy removal of the manway cover assembly.

   NOTE: In removing the manway cover assembly, the initial movement is perpendicular to the manway penetration cover.

6. Using tin snips, cut the steel banding away from around the external manway cover assembly.

7. Remove two 3/8 inch hex nuts (Part No. 210) from end of manway cover assembly.

8. Remove the cover (Part No. 198) from over the slot in the manway cover assembly.

9. Remove the cover adapter (hose adapter) (Part No. 196) from the manway cover assembly.

   NOTE: Several turns of fiber glass tape may have to be removed first.

10. Remove four 1/2 inch hex head cap screws (Part No. 211) which connect the manway cover assembly to the manway penetration cover.

11. Remove the manway cover assembly (Part No. 132). Lift first perpendicularly away from the manway penetration cover to clear the conax fittings and converging instrumentation leads. Then manipulate so the slot

in the cover is guided over the lead bundle.

NOTE: Cover weighs about 50 lbs and can be handled easily by two men.

12\. a. At this stage, all conax fittings are accessible for torqueing.

b. If there are any indications that any conax fitting is leaking, contact a B&W engineering representative, NPGD Lynchburg, (J. W. Mitchem) prior to proceeding further.

13\. Using the torque wrench, 3/4 inch hex socket, and split socket assembly, torque all conax fittings (12 total) to 200 ± 20 ft-lbs.

14\. Replacement of the manway cover assembly and conduit accessories will be the reverse sequence of the preceeding steps, and is simplified in the following steps.

15\. Replace manway cover assembly.

16\. Replace the four 1/2 inch hex head cap screws and torque to 100 ± 5 ft-lbs.

17\. Replace the cover adapter (hose adapter) and wrap with several turns of fiber glass tape to hold in position.

18\. Replace the cover over the manway cover slot.

19\. Replace the two 3/8 inch hex nuts and torque to 50 ± 5 ft-lbs.

20\. Replace the steel banding, one or two turns strapped tightly, so that cover over slot cannot vibrate.

21\. Replace the split hose conduit over the instrumentation leads.

22\. Replace the two hose clamps.

23\. Replace the duct tape fully wraping the conduit area where tape had been previously removed.

24\. Replace the nylon ties as required.

25\. Replace the OTSG mirror insulation.

10
3-17

cc. JLS
JFH
B&W Letter Book
(U-2)

[illegible]

Power Generation Group
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March 16, 1978

SOM-II-[illegible]

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Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
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Post Office Box 480
Middletown, PA 17057

Subject: Test Program Modifications

Gentlemen:

A review of the zero power physics testing program with three reactor coolant pumps has been completed. B&W comments and recommendations are provided in the attached site instruction, which is addressed to the B&W TMI Site Office.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

[signature]

L. C. Rogers
Site Operations Manager

[illegible]

cc: [illegible]

W 39059

SITE INSTRUCTION NO. 6/32

DISTRIBUTION [illegible]

NO RESPONSE REQUIRED

[illegible] SIGNATURE [illegible]

APPROVED [illegible]

[illegible] (Cat. 2010)

[illegible] March 15, 1978

[illegible] for qualifications which [illegible] [illegible] recommendations [illegible] separately.

1. [illegible] to performing zero power physics testing [illegible] pumps operating.

2. [illegible] testing which requires [illegible] reactor coolant pump [illegible] as soon as practicable after [illegible] are available.

3. [illegible]

4. [illegible] pressure control should be returned to the [illegible]

5. [illegible] criteria.

[illegible]

W 39060

[illegible]	Total [illegible] Flow Rate [illegible]
0	[illegible]
1/2	[illegible]
1	[illegible]
2	[illegible]
3	[illegible]
4	83.0
5	76.0

[illegible] the acceptance criterion used for the three pump coastdown [illegible] to zero [illegible] test should be performed as soon as four [illegible] available.

[illegible] the pressurizer [illegible] was set during Hot Functional Testing, no [illegible] checks or adjustments are necessary. Experience has shown that [illegible] is acceptable.

[illegible]

cc: [illegible]

W 39061

10
41-6

BEW/ke
JLS

Babcock&Wilcox

Power Generation Group
P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

April 4, 1978

SOM-II-120
Addendum

Mr. R. J. Toole
Test Superintendent
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Test Program Modifications

Gentlemen:

In paragraph 3 of the site instruction attached to SOM-II-120, dated 16 March 1978, an acceptance criterion of 1.32 seconds was specified as the maximum CRD trip time for testing with three reactor coolant pumps. Subsequent discussions have raised the question of whether or not 1.32 includes breaker delay time. It does not. The acceptance criterion including CRD breaker delay time should 1.40.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/JV/bay

cc: L. R. [illegible]
[illegible]
T. F. Scott

C. E. Harbein
R. M. [illegible]

W 39062

8/3-27

B&W LTR BK
DMS
J Fi[illegible]

G Miller

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March 22, 1978

SOM-II-122

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Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Limits and Precautions, OP 1101-01

Gentlemen:

The Limits and Precautions for the makeup and purification seal return and seal injection filters recommend cleaning the filter at a 5 psi differential for seal return and 25 psid for seal injection. Based on an engineering evaluation, these limits can be increased to 25 psid for seal return and 50 psid seal injection.

One concern in increasing the seal return filter ΔP is that the seal return line back pressure would increase and possibly cause erratic RCP seal staging at low RC system pressures. Therefore, the back pressure in the seal return line measured at the discharge of the third stage pressure breakdown coil should be ≤100 psig.

W 39063

R. J. Toole
L. L. Sawyer
G. P. Miller

3/22/78

The following corrections should be made to DP 1101-01, Rev. 00 dated 31 May 1977:

1) Page 24 Item 12: add "The maximum back pressure of the controlled bleed off line measured at the discharge of the third stage pressure breakdown coil is 100 psig."

2) Page 49 Item 2.5.2 Normal Operating Conditions-B - change pressure drop, 1 to 5 psi, to read, "1 to 25 psig."

   Item 2.5.2.02 - change 5 psi to read "5 to 25 psi"

3) Page 50 Item 2.5.3 Normal Operating Conditions-B - change pressure drop, 5 to 25 psi, to read, "5 to 50 psi"

   Item 2.5.3.02 - change 25 psi to read "25 to 50 psi"

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGC/bay

cc: L. R. Fletke
W. H. Spangler
T. F. Scott
J. G. Herbein
R. M. Klingaman
J. L. Seelinger
T. A. Mackey

W 39C64

cc JFH
JLS
[redacted]

Babcock & Wilcox

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March 20, 1978

SOM-II-121

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Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

[redacted]
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Movable Incore Detector System (MIDS)

Gentlemen:

As you are aware, B&W has installed a Movable Incore Detector System (MIDS) in TMI-2 core location N-8 in order to monitor the performance of a gadolinium demonstration assembly.

The attached procedure will be used to collect data from the MIDS equipment and the plant computer for transmittal to Lynchburg. The PDO data collection will be coordinated with the PDO data requirements for physics testing and Met-Ed surveillance requirements as much as possible throughout the startup test program and through Cycle 1 operations. B&W site personnel will collect the MIDS data.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/JV/bay

W 39065

cc: L. R. Pletke (w/o attachment)
W. H. Spangler "
T. F. Scott "
J. G. Herbein (w/attachment)
R. M. Klingaman "
R. J. Toole (w/attachment)

BWNP-20004 (6-76)

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TECHNICAL DOCUMENT

TEST PROCEDURE

63 - 1002906 - 00

Doc. ID – Serial No., Revision No.

for

MOVEABLE INCORE DETECTOR SYSTEM

TP 2483

W 39C66

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

BWNP-20005 (6-76)

NUMBER 63-1002906-00

RECORD OF REVISION

REV. NO.	CHANGE SECT/PARA.	DESCRIPTION/CHANGE AUTHORIZATION
00		Original Issue - J. Veenstra, Nuclear Services

PREPARED BY James Veenstra (NAME) [illegible] (TITLE) DATE 10/5/77

REVIEWED BY [illegible] (NAME) Supervisory Engineer Fuel Engineering (TITLE) DATE 10/5/77

REVIEWED BY [illegible] (NAME) [illegible] (TITLE) DATE [illegible]

APPROVED BY [illegible] (NAME) [illegible] (TITLE) DATE [illegible]

W 39067

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TABLE OF CONTENTS/EFFECTIVE PAGE LIST

NUMBER
63-1002906-00

W 39C68

1.0 TEST OBJECTIVE

The data collected by the Moveable Incore Detector System (MIDS) will provide information on the axial power distribution in the gadolinium demonstration assembly located in Core Location N-8 at TMI-Unit 2. Data taken from the plant computer at the same time will be used for correlation with the MIDS data.

2.0 ACCEPTANCE CRITERIA

No specific acceptance criteria are applicable for this test.

3.0 CONDITIONS PRIOR TO TEST

3.1 Plant Activities

3.1.1 The incore monitoring system and the plant computer are operational.

NOTE: If either of the above systems is inoperable for an extended period and the recommended data interval of Section 6.1 is about to be exceeded, record the MIDS data without taking a Performance Data Output (PDO).

3.1.2 The MIDS is operational.

3.1.3 The plant should be operating at a steady state power level (preferably above 70% FP). Three dimensional xenon equilibrium conditions should exist as indicated by the traces on the backup recorders showing parallel straight lines.

3.2 Prerequisite Tests

None.

3.3 Configuration of the System

3.3.1 Axial Power Shaping Rods (APSR's) must be at a constant position for at least 15 minutes prior to the test.

3.3.2 No control rod or APSR motion should occur when the MIDS recorder trace is being taken and the PDO is being compiled.

3.3.3 The MIDS should be checked out and calibrated in accordance with Reference 7.1 before recording the MIDS data.

3.4 Environmental Conditions

Ambient.

W 39069

4.0 SPECIAL PRECAUTIONS

4.1 Voltages applied to control and signal conductors for test and check-out purposes must not exceed fifty (50) volts.

4.2 Do not disconnect the detector leads unless the detector is parked (withdrawn to the out-limit).

4.3 Do not drive the detector into the reactor if the neutron or background signal circuits are open.

4.4 Never leave the system with the detector unparked. If the detector is left in the reactor, its operational lifetime will be shortened.

4.5 When the reactor is operating, do not leave the detector in the reactor unnecessarily.

4.6 Do not leave the MIDS energized when the system is not in use.

4.7 As long as the equipment is energized, power is applied to the drive motor whether or not motion is commanded.

4.8 To allow for thermal stabilization, the signal amplifier must be given a minimum warmup time of thirty (30) minutes.

4.9 At drive velocities in the range of 10 to 45 inches per minute, gear noises emanate from the drive units. These are normal and are due to the fact that anti-backlash gears are not used.

4.10 Depressing the "zero button" of the signal amplifier opens the neutron and background signal circuits. The switches on the load resistor-selection box should be turned to "short" before depressing the zero button.

5.0 DATA

All data collected should be coordinated through the shift supervisor and control room operators. Whenever possible, the PDO requirements should coincide with PDO's taken during the initial startup testing power escalation sequence or taken to fulfill the technical specification requirements.

5.1 The data collected by the MIDS is described in Reference 7.1 and should be labeled by attaching a copy of Data Sheet 1.

5.2 For each set of MIDS data, obtain a copy of PDO Tape Segments 1 thru 6 on magnetic tape for transmittal to B&W, Lynchburg.

W 39070

6.0 PROCEDURE

6.1 This procedure should be used to collect MIDS data at the following times during Cycle 1:

NOTE: Stable plant conditions and equilibrium xenon are prime concerns. These factors outweigh the exact Effective Full Power Days (EFPD's) listed below:

6.1.1 Every EFPD up to 5 EFPD.

6.1.2 Every 5 ± 1 EFPD up to 35 EFPD.

6.1.3 Every 25 ± 5 EFPD beginning at 50 ± 5 EFPD.

6.2 Ensure that operations personnel are familiar with the special precautions of Section 4.0.

6.3 Verify that all plant conditions of Section 3.0 are satisfied before recording any data.

NOTE: For data recorded during the initial startup testing, MIDS traces will only be available at lower power levels for the first few EFPD's.

6.4 Startup and calibrate the MIDS equipment as required by Reference 7.1.

6.5 Obtain a MIDS recorder trace in accordance with Reference 7.1. Ensure that chart speed and drive speed are set up per Reference 7.1 and recorded on Data Sheet 1.

6.6 Verify that no control rod or APSR motion occurred while the trace was taken and that no discontinuities exist in the trace.

6.7 Repeat Step 6.5 if rod motion secured during the recorder trace.

6.8 Obtain the PDO data of Step 5.2.

6.9 As soon as the PDO data is compiled, notify the shift supervisor that data collection is complete.

6.10 Label the MIDS trace by attaching and completing Data Sheet 1.

6.11 Shutdown the MIDS equipment per Reference 7.1.

6.12 Transmit the MIDS and PDO data to B&W, Lynchburg.

7.0 ENCLOSURES

7.1 References

W 39071

MIDS Instruction Manual

7.2 Special Test Equipment

MIDS and associated recording equipment.

7.3 Data Sheets

Attached.

W 39072

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

March 23, 1978

SOM-II-123

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Operating Plant Continuing Service

Dear Gary:

The critical operation's sequence for TMI-II is rapidly moving towards performance, and all site organizations are supporting the program, as necessary, towards this major milestone. The thought may be premature, but passing this milestone practically describes the end of B&W contract personnel in residence at the TMI site.

In accordance with earlier discussions between Mr. Herbein, you, and I at separate and widely separated times, my impression is that Met-Ed will be very interested in a Master Service agreement with B&W to provide continuing operating plant service to Units I and II at TMI following commercial commissioning of Unit II. Therefore, because of the necessary time required to establish agreements such as these with our respective organizations, it seems appropriate to initiate specific task descriptions of the services available that may be supplied to Met-Ed to allow review, comment, and incorporation of specifics from all organizations.

The broad scope of service that your people should be interested in picking up would encompass coverage of:

a. supervisor - includes any or all of the following capabilities
b. reactor core and plant performance support engineer
c. mechanical equipment (pumps, valves, special equipment, system performance) engineer
d. electrical/instrument/controls equipment and system performance engineer
e. chemo-nuclear engineer, water chemistry and radio-chemistry
f. as proposed, most engineers would be capable of more than one area of support expertise

W 39073

These capabilities would be aimed toward the achievement of improved plant reliability and performance leading to higher unit availability performance factors.

The site team concept allows for increased observations, direct assistance, and special study capabilities available to the Met-Ed staff. This program would be similar to the type of assistance performed for Unit I during the preceding three cycles of operation, with heavier emphasis during the cycle I period.

With the commercial operation of Unit II, one should expect a heavier work load towards refueling cycles every six months and preparations for the "No Name Outages" as used currently at the site which would lead to successful maintenance periods. A B&W site team as conceptualized could provide very broad and when needed specific assistance to the TMI site needs that would cover operating periods, scheduled and non-scheduled maintenance outage periods and normal refuelings.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/bay

cc: L. R. Pletke
W. H. Spangler
T. F. Scott
J. G. Herbein
R. M. Klingaman
J. D. Phinney
J. T. Janis
J. P. O'Hanlon
L. L. Lawyer
L. F. Taynton
R. E. Wascher

6
3-27

CC JLS
J. Floyd
B&W ...

Babcock & Wilcox

Power Generation Group
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March 24, 1978

SOM-II-124

Mr. R. J. Toole
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Manager, Generation Operations
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Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Power Escalation With Less Than Four RCP's
Reference: 1. Letter L. C. Rogers to Messrs. Toole, Lawyer, and Miller, SOM-II-120, "Test Program Modifications," dated 16 March 1978
2. Unit Startup and Power Escalation Test, SP 800/21

Gentlemen:

As requested, B&W has evaluated a power escalation test program with three reactor coolant pumps (RCP) for TMI-2. B&W will support initial escalation with three reactor coolant pumps as per Enclosure I, with subsequent operation at 40% RTP up to a cumulative 30 EFPD. Operation beyond this point will require a reassessment of technical specifications and possible changes if deemed necessary. Note that when requested, B&W will prepare the 40% core power distribution values needed for the acceptance of that test with three reactor coolant pumps.

Testing recommended subsequent to the fourth reactor coolant pump being available (assuming Enclosure I has been completed) is outlined in Enclosure II. Note that additional analysis may be required for completion of the test program with four reactor coolant pumps.

W 39074 A

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/JV/bay

cc: L. R. Pletke
W. H. Spangler
T. F. Scott
J. G. Herbein
R. M. Klingaman

ENCLOSURE I

INITIAL ESCALATION WITH THREE RCP's

1. Precritical and Zero Power testing should be completed as modified by Ref. 1

2. Collect Unit Load Steady State baseline data (TP 800.12) and set the NI over-power trip setpoint to 50% RTP.

3. Insure the plant computer is prepared to calculate accurate heat balances.

   A. CTPA partial pump flow values (Tab 1-8) should be set based on the RCS flow test (TP 200.11). This is the normal procedure, but is particularly important for partial pump operation.

4. Increase reactor power to 5-7% RTP as per Ref. 2 then tune the turbine bypass valves to maintain 885 psig as per TP 800.08.

5. Alternate heat balance calculations and NI calibrations to 11.25% RTP. (11.25% RTP equals 22.5% steam generator load and should be divided 15% load on OTSG B and 7.5% load on OTSG A.)

6. Adjust the OTSG-B low level setpoint to control T_{ave} in loop B at 582°F (TP 800.12).

7. Adjust the OTSG-A low level setpoint equal to the OTSG-B setpoint.

8. Load the turbine-generator and assume all load on the turbine.

9. Record ULSS data (TP 800.12).

10. Increase reactor power to 15% RTP as per Ref. 2

11. The 15% RTP plateau and the escalation from 15% to 40% RTP should be performed as per Ref. _ with the addition of ULSS data at 20% and 30% RTP.

12. Conduct 40% RTP testing as per Ref. 2 except:

   A. The Dropped Control Rod Test (TP 800.31) should not be performed until four RCP's are operating.

   B. Unit Load Transient Testing (TP 800.23) should be restricted to the minimum transients necessary to ascertain acceptable feedwater response.

   NOTE: This three RCP escalation requires B&W to prepare a 40% RTP Core Power Distribution based on three RCP's, which will take 1-2 weeks after the customer decides on a three RCP startup and requests this data.

W 39C76

ENCLOSURE II

RECOMMENDED TESTING AFTER RESTORING FOUR RCP OPERATION

1. Precritical testing as mentioned in Ref._ must be done before the reactor is operated critical with four RCP's.

   A. RC Flow and Flow Coastdown Test (TP 200.11) as presently written

   B. Record Unit Load Steady State (TP 800.12) base line data with four RCP's at zero power.

2. Startup to 15% RTP per normal plant startup procedures. At 15% RTP:

   A. Perform a heat balance (TP 800.22) and NI calibration (TP 800.02) (Insure computer heat balance is restored to normal four RCP mode.)

   B. Check and adjust as necessary turbine bypass valve tuning (TP 800.03) and OTSG low level setpoints (TP 800.12).

   C. Record ULSS data (TP 800.12) and tune the ICS as necessary (TP 800.03).

3. Increase power to 25% RTP and record ULSS data (TP 800.12).

4. Increase power to 40% RTP:

   A. Perform a heat balance (TP 800.22) and NI calibration (TP 800.02).

   B. Record ULSS data (TP 800.12) and tune the ICS as necessary (TP 800.03).

   C. Establish 3-D equilibrium Xenon and conduct a Core Power Distribution (TP 800.11).

   D. Conduct the Dropped Rod Test (TP 800.31).

   E. Complete the Unit Load Transient Test (ULTT) (TP 800.23).

5. Continue escalation above 40% RTP as per Ref. _.

   NOTE: This additional testing may require re-analysis of the predicted core power distributions at 40%, 75%, 100% RTP, depending on the actual burnup achieved with partial pump operation. These analyses, if run, will require 1-2 weeks after the customer decides how many EFPD will be burned at partial RCP operation.

W 39077

3-27

Babcock & Wilcox

Power Generation Group

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Telephone: (804) 384-5111

March 24, 1978

SCM-II-125

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Manager, Generation Operations
Metropolitan Edison Company
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Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: RCS "Noise" During Single RCP Operation

Gentlemen:

On February 28, 1978, Walt Merciez and Joe Cates arrived on site with the deployable reactor diagnostic system to further investigate a primary system noise heard during initial reactor coolant pump operation at reactor coolant system pressure of 400 lbs. and temperature 150°. An accelerometer was installed on the motor support stand of the 1A reactor coolant pump. Data was taken during single pump operation of all four reactor coolant pumps, with loop A pumps (1A and 2A) and again with both loop B pumps (1B and 2B).

Based on time-delay matrix analysis, the reduced data indicated that the noise was in the reactor vessel. The noise was not periodic and was low frequency. This indicated that the noise was not caused by any reactor coolant pump and confirmed earlier suspicions that the cause of the noise was an internal vent valve in the X-Y quadrant. This vent valve action was not considered to be in a damaging energy impact level.

W 39078

Since the noise was more evident during single pump operation of RCP-1A, the following sequence for starting the pumps is recommended to avoid the vent valve movement.

1. Start RCP-2A or 2B first followed by the second pump in the same loop, i.e., 2A then 1A or 2B then 1B.

2. Secure the pumps in reverse order.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
T. F. Scott
J. G. Herbein
R. M. Klingaman
J. L. Seelinger

[redacted]

G. Miller

Babcock & Wilcox

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March 20, 1978

REM-I-346
SOM-II-126

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Mr. G. P. Miller
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Metropolitan Edison Company
Post Office Box 480
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Mr. J. P. O'Hanlon
Superintendent, Unit I
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: CRDM Stator O-Ring Replacement

Gentlemen:

Operating experience at some B&W plants has shown that the control rod drive stator ethylene-propylene o-rings may take a permanent set and crack after some period of time. This condition could violate the sealing capability and allow moisture to enter the stator cavity. This moisture could lead to a deterioration of the insulation and eventual stator failure.

Investigation by Babcock & Wilcox/Diamond has determined that a silicone o-ring will have a longer service life. Although our investigation is continuing, we are recommending that all ethylene-propylene o-rings be replaced with silicone o-rings during your next refueling outage. This is a precautionary measure since a non-sealing o-ring by itself does not cause a stator failure; however, the combination of a system leak or spill and a faulty o-ring can lead to a stator failure.

W 39C80

Attachments I, II, and III give the general procedures for Pre-installation Preparation, Vacuum Drying of Stators, and Pre-installation Precautions. The steps to be followed during stator changeout and/or ring replacement are as follows:

A. Use silicone o-rings
B. Use Attachment I drying procedure
C. Follow Attachment II precautions
D. Megger stators for insulation resistance

    1. If > 200 megohms, install stator
    2. If < 200 megohms, contact B&W

The following checks should be made as a normal refueling outage maintenance item:

A. Megger stators for insulation resistance

    1. If > 200 megohms, stator is considered dry
    2. If < 200 megohms, use Attachment II drying procedure

A corroded or wet connector can cause a wet stator indication. A stator with a low resistance reading should have the connector electrically isolated from the power conduct and the megger test re-run on the stator to verify the low resistance reading.

When stators are removed from the CRDM for o-ring changeout, the stator filler bushings should be inspected for evidence of corrosion. If any corrosion is found, the filler bushings and end turns should be lightly coated with "Belco Silicone Grease" using a suitable brush or lint-free cloth.

To facilitate the vacuum drying operation, the stator vent plug can be replaced with a vent adapter. Diamond Power has these adapters available for purchase.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager/Manager, Site Services

LCR/SPM/bay

W 39081

cc: J. D. Phinney
J. T. Janis
L. R. Pletke
J. G. Herbein
R. M. Klingaman
W. H. Spangler
T. F. Scott

ATTACHMENT I

2/1/78

Attachment I

Pre-installation Preparation (Drying)

1. Remove the moisture barrier from the stator just prior to installation of stator on mechanism.

2. Be certain that all moisture and foreign material are removed from the motor tube of the mechanism. A special tool should be constructed and used to remove moisture and foreign material.

3. Be certain that the ID "O" rings are properly installed in the upper and lower mounting flanges of the stator.

4. Connect the stator to the service power supply without cooling water, energizing any two (2) windings in the hold mode for a minimum time period of one (1) hour at 50 volts. A simple voltage limiting device should be installed between the power supply and stator to provide automatic shutdown if the voltage exceeds 50 volts.

5. Immediately after stator has been disconnected from the power supply, place it on the mechanismin the warm condition.

W 39C82

2/1/78 Attachment II

VACUUM DRYING OF STATORS

Objective

The objective of this procedure is to establish a method of removing moisture from the stator while it is mounted on the CRDM. Drying of suspect stators during refueling will remove moisture present in the stator assembly and reduce the deterioration of the stator insulation system caused by corrosion.

2.0 Discussion

The incidents of stator failures in the field involving varnish insulated stators represents a nuisance factor that should be minimized if possible. Each of the varnished stators which have failed to date have revealed evidence of moisture being present within the winding area resulting in corrosive action which has damaged the stator insulation systems. Instructions concerning the packing, shipping, storing and installation procedures for the stators to minimize the moisture introduced into the stator have previously been issued. This procedure will implement the removal of any moisture which may have been introduced after the stator was installed.

3.0 References

3.1 Stator Water Jacket Assembly Type A - 703806-1052, 703806-3058, 703808-1050, 703808-3056

3.2 Stator Water Jacket Assembly Type B - 703811-2053, 703811-4059, 703810-2054

3.3 Stator Water Jacket Assembly Type C - 709287-1057, 709287-2055

4.0 Equipment

4.1 Service Power Supply - 706185-2047, 706185-1049

4.2 Vacuum Pump * - With a capacity for a change of 28.5 in. Hg.

4.3 Vacuum Gage - Measures in inches of Mercury

Note: Do not use a gage containing Mercury.

4.4 Vacuum Hose and fittings to connect intake of vacuum pump (4.2) to opening in top of power connector conduit where the vent valve is located. The vacuum gage (4.3) must also be connected into this hook-up.

4.5 Necessary tools to remove and install the vent valve at the top end of the stator power connector conduit.

4.6 500 Volt DC Megger.

W 39083

* Welsh Scientific Co. Skokie, Ill.
Model 1397 Dual Seal -- or equivalent

5.0 Procedure

5.1 Use a 500 volt DC megger (4.6) to test the insulation resistance of each stator. Connect one lead of the megger to the neutral pin(s) of the stator power receptacle, and the other lead to the power connector conduit. Any stators which have an insulation resistance of less than 200 meg. ohms at room temperature shall be submitted to vacuum drying.

5.2 Connect the service power supply to the stator power connector and energize the stator windings A&B with 50 volts DC. Cooling water shall be turned off during this time.

5.3 Remove the vent valve from the power connector conduit.

5.4 Connect the vacuum gage and hose between the vacuum pump intake and the vent valve opening of the power connector conduit.

5.5 Activate the vacuum pump and monitor the vacuum gage to ascertain that proper vacuum is being developed in the stator assembly. This value should reach a minimum of 28.5 inches of Hg. If the vacuum fails to develop examine all hose and gage connections to ascertain that no leaks exists. If no leaks are found in the connections and vacuum does not develop, indications are that either the top or bottom "O" ring seal on the state is leaking. In this event disconnect the power supply and vacuum from the stator, mark the stator for latter removal and go on to the next stator. If the stator does develop a vacuum leave the vacuum pump in operation until the pressure is stabilized, or until the stator has been energized for 30 minutes, which ever is longer.

5.6 At the end of the prescribed time, turn off the power supply but leave the vacuum pump operate. Disconnect the power supply from the stator power connector and test the insulation resistance of the stator using the 500 volt DC megger (4.6). An insulation resistance greater than 200 meg. ohm indicates the stator is dry. An insulatin resistanceless than 200 meg. ohm indicates additional testing is necessary. This should be done as described in 5.4, 5.5 and 5.6.

5.7 When the stator is dry, connect the power supply to the next stator as described in 5.2. Remove the vent valve from thepower connector conduit.

5.8 Turn off the vacuum pump, disconnect it from the one stator and connect it to the next stator and dry it as described in 5.4, 5.5 and 5.6 Replace the vent valve in the stator assembly that has been dried. Continue this procedure until all the stators designated for vacuum drying are completed.

W 39084

5.9 Any stators which would not hold the proper vacuum shall be removed from the mechanism and the top and bottom "O"ring replaced. Also examine the end turn areas of the stator for evidence of corrosion.

5.10 While the stators are removed from the mechanism they shall be connected to the service power supply and windings A&B energized at 50 volts. Allow each stator to remain energized for one hour.

5.11 After drying the stators and while they are still warm replace them on the CRDM. Make certain that all moisture is completely removed from the motor tube before replacing the stator.

5.12 Connect the vacuum pump to each of the stators, at the vent valve opening, after they have been replaced on the CRDM. Energize the pump and determine that a vacuum is developed and maintained on each stator. Any stators which do not develop a vacuum will have to be submitted to further inspection of all areas to determine the cause and correction of leaking.

W 39085

ATTACHMENT III

2/1/78

Attachment II

Pre-Installation Precautions

1. Store stators in shipping crates as received.

2. Reseal any moisture barriers penetrated for incoming inspection measurements. Check pliofilm bag for tears and repackage if necessary.

3. Storage shall be in the following environment:

   a. 40 Deg. F to 170 Deg. F room temperature.

   b. Below 90% relative humidity

4. Remove the moisture barrier from the stator just prior to installation of stator on mechanism.

5. Be certain that all moisture and foreign material are removed from the motor tube of the mechansim. A special tool should be constructed and used to remove moisture and foreign material.

6. Be certain that the ID "O" rings are properly installed in the upper and lower mounting flanges of the stator.

W 39086

ATTACHMENT III

2/1/78

Attachment III

Pre-installation Precautions

1. Store stators in shipping crates as received.

2. Reseal any moisture barriers penetrated for incoming inspection measurements. Check pliofilm bag for tears and repackage if necessary.

3. Storage shall be in the following environment:

   a. 40 Deg. F to 170 Deg. F room temperature.

   b. Below 90% relative humidity

4. Remove the moisture barrier from the stator just prior to installation of stator on mechanism.

5. Be certain that all moisture and foreign material are removed from the motor tube of the mechansim. A special tool should be constructed and used to remove moisture and foreign material.

6. Be certain that the ID "O" rings are properly installed in the upper and lower mounting flanges of the stator.

W 39087

ATTACHMENT III

2/1/78

Attachment II

Pre-installation Precautions

1. Store stators in shipping crates as received.

2. Reseal any moisture barriers penetrated for incoming inspection measurements. Check pliofilm bag for tears and repackage if necessary.

3. Storage shall be in the following environment:

   a. 40 Deg. F to 170 Deg. F room temperature.

   b. Below 90% relative humidity

4. Remove the moisture barrier from the stator just prior to installation of stator on mechanism.

5. Be certain that all moisture and foreign material are removed from the motor tube of the mechansim. A special tool should be constructed and used to remove moisture and foreign material.

6. Be certain that the ID "O" rings are properly installed in the upper and lower mounting flanges of the stator.

W 39098

DEQA RRS
~~JSS~~
TAM

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April 4, 1978

SOM-II-127

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Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: RCS Chloride Contamination Evaluation

Gentlemen:

It is the opinion of B&W (NPGD Technical Staff) that the high chloride contamination of the RCS will have no deleterious effect upon the structural integrity of the RCS or associated auxiliary systems and equipment. Therefore, the RCS remains acceptable for continued operation (heatup and startup). This instruction constitutes the required engineering evaluation in accordance with Plant Technical Specification (3/4 4-17).

This evaluation is based on the conditions bounded as follows:

Chlorides	4.0	ppm (maximum)
Oxygen	10	ppb (maximum)
Sodium	430	ppm

and the RCS at Hot Standby Conditions.

This evaluation is specifically based on the basic pH associated with the sodium hydroxide contamination and the presence of low oxygen levels.

If you have any further questions, please do not hesitate to contact me.

W 39089

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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9
H-24

B&W
Book

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April 14, 1978

SOM-II-127
Addendum

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Station Superintendent
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Subject: Recommendations for Removal of Sodium and Chloride Contamination from the Reactor Coolant System

Gentlemen:

This is to formally transmit the verbal recommendations given to you concerning removal of sodium and chloride contamination from the Reactor Coolant System (including pressurizer and CRDM motor tubes).

1. Continue cooldown of the Reactor Coolant System to cold shutdown conditions to be in compliance with technical specifications for chloride contamination.

   NOTE: It is B&W's opinion that the plant should also be cooled down based upon concern for the sodium contamination. The Reactor Coolant System should be < 10 ppm sodium within 48 hours or cooldown to cold shutdown conditions commenced.

2. Continue cleanup of the sodium and chloride contamination in the Reactor Coolant System via the makeup system demineralizers.

W 39091

3. During the cooldown (when < 400 F) add hydrazine in quantity just sufficient to scavenge the oxygen present in the Reactor Coolant System as a result of the required additions of fresh makeup water. It is estimated that no more than 1-2 ppm hydrazine will be required.

4. Cleanup should continue until the sodium is < 10 ppm and chloride is < .15 ppm at cold shutdown conditions. While the Reactor Coolant System is being cleaned up, all auxiliary systems and connections to the Reactor Coolant System (such as instrument lines, dead legs, BWST, etc.) which were possibly contaminated should be sufficiently flushed to prevent recontamination of the Reactor Coolant System. The pressurizer water should be circulated as necessary to reduce contamination to the same limits as the Reactor Coolant System.

5. Reactor Coolant System boron concentration should be maintained at all times, to be in accordance with cold shutdown requirements, to prevent inadvertent criticality.

6. When Item 4 conditions are met and prior to heatup, Reactor Coolant System water samples from one center and three peripheral CRDM's (at different relative RV head locations) should be obtained and analyzed for sodium contamination. The samples should be about two (2) gallons each. If any one of the samples obtained have > 10 ppm sodium contamination, each of the CRDM's should be flushed approximately one volume (∿ 5 gallons). If all CRDM samples have < 10 ppm sodium contamination, no flushing of the CRDM's is necessary.

   NOTE: Samples should be obtained from any CRDM's which have vent valve leaks.

7. When Items 4 and 6 conditions are met, heatup may commence. The chloride contamination limits should meet the requirements of SCM-II-129.

8. Prior to increasing system pressure above 500 psig or prior to exceeding Mode 4 (hot shutdown) conditions, appropriate data from the contamination incident (including a chronology of events for Reactor Coolant System temperature and pressure and the sodium and chloride contamination levels) must be forwarded to and evaluated by the Technical Staff (NPGD). This is per Plant Technical Specification requirements (3/4 4-17).

9. Prior to criticality, the sodium concentration in the Reactor Coolant System must be < 2 ppm.

   NOTE: The decay heat system must have less than 2 ppm sodium prior to subsequent use.

It should be emphasized that the sodium hydroxide and chloride contamination could lead to serious stress corrosion cracking in stainless steel, Inconel, and Zircaloy if not removed prior to heating up.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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W 39093

8/4-6

G. Miller

DEW BK.
JWS
TAM

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April 5, 1978

BCM-II-128
RLM-I-347

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Middletown, PA 17057

Subject: Condensate Polishing & Feedwater Requirements

Gentlemen:

It is becoming more and more apparent that our customers are operating their secondary cycle with a philosophy that 100% condensate polishing is not always necessary assuming the feedwater is within B&W specifications. The purpose of this letter is to negate that position.

The B&W feedwater requirements are based on the premise that 100% of the condensate flow is being "polished" using the condensate polishing demineralizers (CPD). Analysis for several ions are not specifically run since they are assumed to be removed with 100 percent CPD. These include, but are not limited to, sodium, calcium, magnesium, aluminum silicate, carbonates, nitrates, and chlorides. Our experience has been that with 100% CPD usage, very little of these ions past through the CPD. Without 100% CPD utilization, however, traces or sometimes substantial levels of these have been found in steam turbine deposits. Certain anions can lead to corrosive damage in the OTSG's. Cations can deposit on OTSG tubing with a subsequent lowering of the heat transfer

W 39094

R. J. Toole
W. L. Lawyer
J. P. Miller

coefficient. Transport of either cations or anions through the OTSG can promote both to steam turbine and other secondary system components corrosion damage as well as to a loss of turbine aerodynamic efficiency.

The "pumping forward" of moisture separator reheater drains (MSR) to achieve a higher thermal efficiency can also be a significant factor in OTSG and steam turbine contamination. Sufficient data exists to show that the MSR drains are a concentrating of contaminant loop during the normal operation of the plant secondary cycle. Sodium, for example, has been shown at TMI-I to concentrate by a factor of 18:1 MS to CPD effluent and has been calculated to be as high as 200:1. Therefore, at anytime of CPD bypass, low level contaminants can concentrate in the MSR cycle and thus can deposit in the OTSG and/or the steam turbines. The latter can also happen when 100% CPD are used but breakthrough of the polishers is occurring due to exhaustion because of a condenser leak.

It remains B&W's position that:

a. 100% CPD usage is required at all times for OTSG feedwater flow
b. MSR drains be routed directly to the hotwell at anytime of condenser leakage and/or CPD breakthrough

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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W 39095

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April 7, 1978

SOM-II-129
REM-I-348

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Mr. G. P. Miller
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Subject: RCS Chemistry Control

Gentlemen:

B&W Water Chemistry Guide Specification 2050 and Water Chemistry Manual 1385 restrict chloride and fluoride concentrations to <.1 ppm prior to heatup and dissolved oxygen to <.1 ppm prior to heatup above 250F. In order to make these requirements conform with the Standard Technical Specifications (Section 3/4-7), the chloride, fluoride and oxygen restrictions have been modified as follows:

1. If either Cl^- or F^- is greater than 0.15 ppm but less than 1.0 ppm with the RCS at ≤200F and the pressurizer in a pressurized condition at >200F, restore the contaminant level to <0.15 ppm within 24 hours or reduce the pressurizer temperature to ≤200F within the next 36 hours and restore the contaminant concentration to <0.15 ppm prior to increasing the pressurizer temperature above 200F.

2. If either Cl^- or F^- is greater than 1.0 ppm with the RCS at ≤200F and the pressurizer in a pressurized condition at >200F, reduce the pressurizer temperature to <200F within 36 hours and restore the contaminant concentration to <0.15 ppm prior to increasing the pressurizer temperature above 200F.

W 39096

3. If O_2 is greater than 0.1 ppm but less than 1.0 ppm with the RCS at ≤250F and the pressurizer at >250F, restore the O_2 to 0.1 ppm within 24 hours or reduce the pressurizer temperature to <250F within the next 36 hours and restore the O_2 to <0.1 ppm prior to increasing the temperature of the pressurizer above 250F.

4. If the O_2 is greater than 1.0 ppm with the RCS at ≤250F and the pressurizer at >250F, reduce the pressurizer temperature to <250F within 36 hours and restore the O_2 to <0.1 ppm prior to increasing the temperature of the pressurizer above 250F.

5. In preparing for startup with the RCS and the pressurizer at ≤200F if either Cl^- or F^- is greater than 0.15 ppm but less than 1.0 ppm, the pressurizer pressure may be allowed to increase to the minimum pressure to operate the RC pumps in order to affect cleanup of the RCS and pressurizer and reduce the contaminant level below 0.15 ppm. If the contaminant concentration is not reduced to <0.15 ppm within 24 hours, reduce the pressurizer temperature to ≤200F within 36 hours and restore the contaminant to <0.15 ppm prior to increasing the temperature of the RCS and/or pressurizer above 200F.

6. In preparing for startup with the RCS and pressurizer at ≤200F if either Cl^- or F^- is greater than 1.0 ppm, the pressurizer pressure shall not be increased until the contaminant concentration is reduced to below 1.0 ppm. When the contaminant level is below 1.0 ppm, then Item (5) above applies.

7. In preparing for startup with the RCS and pressurizer at ≤250F, if the O_2 is greater than 0.1 ppm but less than 1.0 ppm, the pressurizer pressure may be allowed to increase to the minimum pressure to operate the RC pumps in order to affect cleanup of the RCS and pressurizer and reduce the oxygen to< 0.1 ppm. If the oxygen is not reduced to <0.1 ppm within 24 hours, reduce the pressurizer temperature to ≤250F within 36 hours and restore the O_2 to <0.1 ppm prior to increasing the temperature of the RCS and/or pressurizer above 250F.

8. In preparing for startup with the RCS and pressurizer at ≤250F, if the O_2 is greater than 1.0 ppm, the pressurizer pressure shall not be increased until the O_2 is reduced to below 1.0 ppm. When the O_2 is below 1.0 ppm, then Item (7) above applies.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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April 11, 1978

SOM-II-130

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~~Mr. G. P. Miller~~
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Reactor Coolant Oxygen Chemistry Control

Gentlemen:

The Reactor Coolant System has been operated at normal operating temperatures and pressures during the last several days, supporting the reactor physics testing sequences. For a significant portion of the recent period, the hydrogen (H_2) injection system has been inoperable due to system mechanical problems. The inability to control the hydrogen inventory during this period may have serious deleterious effects on future Reactor Coolant System operations.

Results from the chemistry analysis of the Reactor Coolant System conditions tend to lull one into a false sense of security because of the low indicated O_2 in the analytical results. Previous experience has shown that using oxygenated makeup water (typically 300 PPB, TMI-2), especially during this phase of testing which requires large amounts of borating and deborating operations, results in a significant level of free O_2 in the Reactor Coolant System water for a short period of time. The B&W specification that requires a given inventory of H_2 in solution typically reduces the O_2 level by the $H_2 \rightarrow O_2$ in a gamma flux type reaction. Without the H_2 presence, O_2 rapidly reacts with the Reactor Coolant System steel constituents in the hot environment and forms crud, which allows for low indicated dissolved O_2 in the samples.

W 39C99

B&W's concern is that the crud level in the system will lead to letdown filter problems, seal injection filter problems, and increased radiation levels during subsequent TMI-2 operations. The consequences of these problems cannot be evaluated easily with unknown continued hot critical plant operations without the proper chemistry control.

Discussions with plant personnel have also identified another potential problem area created by operating personnel in an effort to reduce the O_2 concentrations. Reportedly, hydrazine has been introduced into the makeup water in an effort to reduce the O_2 in the system. Again, chemistry sample results tend to falsify real conditions when an action has been taken to improve plant conditions. Specifically, experience has shown that the hydrazine would decompose to ammonia and nitrogen in the above 400°F environment of the Reactor Coolant System prior to being able to scavenge any free O_2 present. By itself this condition would not be a problem, but when the ammonia level is higher than normal, it tends to cause rapid elution of the contaminants in the purification system resins. This could very easily lead to out-of-specification conditions in the Reactor Coolant System with contaminants such as chloride and sodium, which the plant has just recently completed a recovery program from inadvertent injection of such contaminants. The condition could be considered extremely damaging since the discovery of the out-of-spec condition may not be evident until the next routine sample analysis is made (as long as 24 hours).

Therefore, it is imperative to expedite repair and return to service the auxiliary support systems such as the H_2 system whenever the Reactor Coolant System is continuing to be operated at full hot pressurized conditions. The consequences of not evaluating the need for such support systems can easily cause serious delays in the plant startup sequences. Also, to a greater extent, can affect the plant availability following commercial operation in order to correct or reduce the plant conditions, which were caused early in the startup through incomplete interpretation of the need for such support systems.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/bay

W 39100

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April 12, 1978

SOM-II-131
REM-I-349

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Mr. R. J. Toole
Test Superintendent
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Subject: Control Rod Drive Operation

Gentlemen:

B&W Engineering has reviewed the control rod drive operation modes in the case of a suspected stuck control rod and feels that the possibility exists of overloading the control rod drive (CRD) spider in case the control rod drive is operated in JOG speed under a stuck rod condition.

Therefore, B&W Engineering recommends that both TMI-I and TMI-II, which use Mark B fuel and control components, "A control rod not known to be free running may be operated only in run speed except for the purpose of latching the control rod drive mechanism to the leadscrew when the rod is known to be at the lower limit of travel."

W 39101

We suggest that your make appropriate changes to your control rod drive operating and emergency procedures to reflect this operating philosophy.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

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April 13, 1978

SOM-II-132

Mr. W. T. Gunn
Project Site Manager
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Subject: Procedure for Installation of RCP-2A Anti-Rotation Device

Dear Mr. Gunn:

The recommended steps for the installation of the rebuilt anti-reverse clutch are enclosed (including motor shaft hub and motor housing top hat). These steps have been reviewed and approved by B&W Engineering and Siemens-Allis Engineering.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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W 39103

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4-25

J F 4

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April 20, 1978

SOM-II-133

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Mr. G. P. Miller
Station Superintendent
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Post Office Box 480
Middletown, PA 17057

Subject: Core Power Distributions for Three Reactor Coolant Pump Operation

Gentlemen:

The attached figures are the predicted radial and total core power distributions for three reactor coolant pump operation for TMI-2, Cycle 1 at 40% RTP at 4 and 10 EFPD. These figures are provided, as requested, to supplement the information in the Physics Test Manual and should be used as the basis for comparison with measurements performed at 40%FP.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/JV/bay

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W 38104

FIGURE 1 Radial Power Distribution, 40% FP, 4 EFPD, 3 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	1.457	1.227	1.144	1.089	* 1.017	1.152	1.405	1.007
K		1.204	1.127	1.103	1.038	1.116	1.127	0.921
L			1.116	1.052	0.920	1.039	1.197	0.687
M				1.010	0.964	0.951	0.903	
N					0.912	0.809	0.557	
O						0.581		
P								
R								

Group 6	87	%WD
Group 7	87	%WD
Group 8	19	%WD
Imbalance	-.25	%FP
% Offset	-.63	

*Location of Symmetric Gadolinium Assembly

W 39105

FIGURE 2 Total Power Distribution,
40% FP, 4 EFPD, 3 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	1.987	1.642	1.552	1.441	* 1.361	1.528	1.917	1.363
K		1.604	1.489	1.482	1.422	1.485	1.515	1.243
L			1.503	1.465	1.454	1.426	1.645	0.935
M				1.438	1.354	1.284	1.215	
N					1.240	1.069	0.742	
O						0.765		
P								
R								

*Location of Symmetric Gadolinium Assembly

Group 6	87 %WD
Group 7	87 %WD
Group 8	19 %WD
Imbalance	-.25 %FP
% Offset	-.63

W 3S1C6

FIGURE 3 Radial Power Distribution, 40% FP, 10 EFPD, 3 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	1.466	1.242	1.159	1.103	* 1.025	1.150	1.383	0.988
K		1.219	1.142	1.116	1.046	1.114	1.115	0.906
L			1.130	1.062	0.925	1.037	1.180	0.678
M				1.017	0.968	0.948	0.893	
N					0.913	0.806	0.554	
O						0.578		
P								
R								

*Location of Symmetric Gadolinium Assembly

Group 6	87	%WD
Group 7	87	%WD
Group 8	19	%WD
Imbalance	.19	%FP
% Offset	.47	

W 39107

FIGURE 4 Total Power Distribution,
40% FP, 10 EFPD, 3 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	2.021	1.684	1.593	1.479	* 1.392	1.544	1.905	1.351
K		1.645	1.531	1.519	1.453	1.503	1.517	1.236
L			1.541	1.501	1.481	1.443	1.640	0.933
M				1.469	1.380	1.300	1.216	
N					1.260	1.082	0.748	
O						0.774		
P								
R								

*Location of Symmetric Gadolinium Assembly

Group 6	87	%WD
Group 7	87	%WD
Group 8	19	%WD
Imbalance	.19	%FP
% Offset	.47	

W 3S108

17
5-8

B&W letter book

Babcock & Wilcox

Power Generation Group

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Telephone: (804) 384-5111

April 25, 1978

SCM-II-134

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Mr. R. J. Toole
Test Superintendent
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Subject: Fuel Handling Equipment Problems

Gentlemen:

In order to help analyze the problems experienced with the fuel handling equipment during fuel load, B&W sent Ron Pillow of their mechanical systems section to the site for a follow-up study. He feels that although both transfer systems operated successfully in a dry environment after shimming, both transfer systems should be tested with the spent fuel pool and the reactor pool full of water and the carriage loaded with the full-weighted dummy fuel assembly to ensure confidence in system operability.

The drive sprocket design modification which would provide a positive means of keeping the drive sprocket engaged with the carriage chain should be handled through our B&W Project Management Department.

If you have any further questions, please do not hesitate to contact me.

W 39109

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/JMP/bay

cc: L. R. Pletke
W. H. Spangler
T. F. Scott
J. G. Herbein
R. M. Klingaman
J. B. Logan
J. L. Seelinger

W 39110

#29 G. Miller B&W Site

Babcock & Wilcox

Power Generation Group

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April 27, 1978

SOM-II-135

Mr. W. T. Gunn
Project Site Manager
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Subject: RCPM-1A Noise; RCPM-2A Lower Bearing Housing Insulation

Dear Mr. Gunn:

At the completion of HFT, the source of a noise in the 1A reactor coolant pump motor was determined to be from vibration in the lower bearing oil cooler. The coils of the cooler were strapped, under the direction of the Allis-Chalmers representative, in order to prevent tube fin vibration. During subsequent motor operation, the noise was still heard; but the intensity of vibration was decreased considerably.

During the present outage, it is recommended that the oil cooler be inspected and, possibly, additional strapping be added. Enclosed are the general steps involved in obtaining access to the oil cooler. This work should be done under the supervision of an Allis-Chalmers representative.

RCPM-2A:

During upper bearing housing insulation check-out, it was discovered that the insulation for the lower bearing housing had failed. This insulation was installed on Unit II motors to facilitate check-out for grounds. During normal operation, the lower bearing housing is grounded with a grounding strap.

Materials to replace the lower bearing housing insulation have been ordered, and one shipment has been received. It is recommended that this insulation (old lower bearing housing insulation) be replaced when all material is received on site. The scope of work involved will be outlined by the Allis-Chalmers representative when he arrives on site. This work should be done under his supervision.

W 39111

At the present time, plans have been made for an Allis-Chalmers representative to arrive on site Sunday, 30 April for security training. Monday he will begin his work on the RCPM-2A anti-rotation device clutch installation, in addition to the aforementioned items.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E S Schaedel
for
L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
L. L. Lawyer
G. P. Miller
J. B. Logan
J. L. Seelinger
R. J. Toole
S. Levin

Attachment I

RCPM-1A

Reference: Drawing 05-507-989-404

1. Drain oil from lower bearing oil reservoir
2. Drain cooling water (needed for cooler removal)
3. Disconnect oil drain line and fill line
4. Remove oil drip pan and standpipe bolts
5. Lower pan and standpipe
6. Split 2 halves of pan and remove
7. Fix or remove cooler
8. Reverse the above steps for reassembly

NOTE: This work should be done under the supervision of an Allis-Chalmers representative

5-18

Babcock & Wilcox

Power Generation Group

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May 2, 1978

SOM-II-137

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~~Mr. G. P. Miller~~
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Review of Zero Power Physics Test Results

Gentlemen:

B&W has reviewed the results of the zero power physics test program conducted at TMI-2, cycle 1. The measurement uncertainty for the shutdown margin test was calculated, based on the actual test data, to be less than 20%. Based on this calculation, the current 50% uncertainty value in the test procedure is hereby revised to 20%.

With this change, all zero power physics test results are within the stated acceptance criteria; and B&W supports escalation of TMI-2 to 15% RTP for additional testing.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39114

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6″

LCR/JV/bay

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5-9

B+W lk

Babcock & Wilcox

Power Generation Group

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May 2, 1978

SGM-II-138

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Middletown, PA 17057

Subject: Revised Rod Worths

Gentlemen:

New predicted rod bank worths for TMI-2, cycle 1 are given below. These predictions are from a calculational model developed since the original TMI-2 PTM was prepared. The effect of the 16 gadolinia pins in the TMI-2 core are included in this calculation. The gadolinia effect is small, except for Group 5 rod worths. The primary reason for differences in rod worths is the use of the newer calculational model.

Bank	Worth % $\Delta K/K$
1	0.21
2	2.00
3	1.03
4	1.07
5	1.99
6	1.89
7	1.45
1-7	9.64

W 39116

Bank	Worth % ΔK/K
1-4	4.31
5-7	5.33

B&W has reviewed the remaining PTM data to determine if other PTM sections should be revised. The only data which now appears to need revision is the 40%, 75%, and 100% FP power distributions. The revised power distributions for 40% FP, 3-pump operation has already been transmitted. The 4-pump, 40%, 75%, and 100% FP power distributions will be revised and will reflect the current control rod operating band with Group 6/7 at 95% withdrawn.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers

L. C. Rogers
Site Operations Manager

LCR/JV/bay

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29
5-8

B+W Book

G. Miller

Babcock & Wilcox

Power Generation Group

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May 3, 1978

SOM-II-141
REM-I-353

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Subject: Torque Values for Incore Monitor Closures

Gentlemen:

In response to the request from your instrument shop personnel for torque values for the incore monitor assemblies, the following recommendations apply:

1. The normal torque value for nut ring assembly bolts is 50 ft/lbs; however, to stop leakage the torque may be increased to 60 ft/lbs. This will give a metal to metal fit, and any higher torque may result in thread galling. If leakage does not stop after torquing to 60 ft/lbs, then the gasket should be replaced and seating surfaces inspected and/or repaired as necessary.

2. The normal torque value for the vent plug and hydro test plug blind flange bolts is 10 ft/lbs. If leakage continues, increase torque by 5 ft/lbs, up to 15 ft/lbs, and then up to 20 ft/lbs, if necessary. The maximum allowable torque is 20 ft/lbs based on thread galling.

W 39118

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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May 2, 1978

SOM-II-139
REM-I-352

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Mr. R. J. Toole
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Subject: Inspection of Ratchet Tripped Control Rod Drive Mechanisms

Gentlemen:

A question presented by another B&W customer, relative to inspection of control rod drive mechanisms suspected of having experienced a ratchet trip, has been evaluated. In response, Diamond Power has taken a two-phase position relative to this inspection. B&W concurs with Diamond's position and suggests that Met-Ed evaluate the following:

1. If there is evidence that a control rod drive mechanism (CRDM) has been subjected to a ratchet trip, the leadscrew of that mechanism should be removed for inspection if the length of the outage is compatible with such a program. Inspection will reveal one of the following conditions:

W 39120

(a) No damage to leadscrew. This indicates that the suspected ratchet trip did not occur or occurred only from a near full insertion. No further corrective action is required.

(b) Marks on the leadscrew thread face or slight rolling on the edge of the leadscrew thread. This slight damage can be polished out with crocus cloth.

(c) Significant rolling on the edge of the leadscrew thread. This will necessitate filing with a stainless steel file, followed by polishing with crocus cloth.

(d) Chips out of the leadscrew. This condition requires significant repair or possible replacement of the leadscrew. Also, the entire drive mechanism must be pulled from the head and cleaned to remove the metallic chips.

Inspection of the leadscrew should be made with the presence of either a Diamond Power or a Babcock & Wilcox representative.

It should be noted that if an entire group of control rod drive mechanisms is suspected to have experienced a ratchet trip, it might not be necessary to inspect each leadscrew in the group. If inspection of the first two or three leadscrews indicates no damage, it is apparent that the suspected group ratchet trip did not occur.

2. If Met-Ed is pressed for time during an outage, the above steps need not be taken unless there has been a problem with latching and/or driving out of a drive mechanism.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SFM/bay

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W 39121

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5-8

P+W.
Book 2

Babcock & Wilcox

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May 4, 1978

SOM-II-142

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Subject: Sodium and Chloride Contamination of the Reactor Coolant System

Gentlemen:

As a continuing effort, B&W's Technical Staff Chemistry Section has reevaluated the NaOH injections into the Reactor Coolant System (both occurrences). As a result, B&W's positions are as follows:

1. The cleanup and handling of the sodium and chloride contamination prior to and during heatup and startup shall be in accordance with the intent of the requirements in SOM-II-127, dated 4 April 1978; SOM-II-127 addendum, dated 14 April 1978; and SOM-II-129, dated 7 April 1978.

2. An engineering evaluation was made of the chloride contamination in accordance with Technical Specification (3/4 4-17), and it was determined that the high chlorides should not have a deleterious effect on the structural integrity of the Reactor Coolant System or associated systems. Thus, based on the chlorides, the Reactor Coolant System remains acceptable for continued operation.

W 39122

3. In regard to the sodium contamination, the effects of the sodium hydroxide were greatly reduced by the fact that the Reactor Coolant System was at elevated temperatures for a short period of time and boric acid was present to provide some neutralization effects. However, we consider the introduction of sodium into the Reactor Coolant System a matter of very serious concern, primarily because it might have collected in crevices and dead areas and may create problems when temperatures and pressures are increased.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers

L. C. Rogers
Site Operations Manager

LCR/bay

cc: L. R. Pletke
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R. M. Klingaman
J. B. Logan
J. L. Seelinger
E. G. Schaedel
K. H. Frederick
K. L. Harner
W. F. Pitka
B. Allen
B. Hopkins

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5-3

B&W
Book

Babcock & Wilcox

Power Generation Group

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May 5, 1978

SOM-II-144

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Station Superintendent
Metropolitan Edison Company
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Middletown, PA 17057

Subject: TMI-2, Cycle 1 Restart with Four Reactor Coolant Pumps

Gentlemen:

In preparation for the impending startup following the current maintenance period which includes full repairs to RCP-2A, B&W has determined that the following tests and guidelines should be included in the site planning program.

During plant startup, while in Mode 3 (hot standby) 532F, 2155 psia, the following tests should be completed:

1. TP 200/6, RCP Tests and Operation

2. TP 200/11, RCP Flow Tests

3. TP 200/12, RCP Flow Coastdown

4. SP 365/2, Loose Parts Monitor Baseline Data Acquisition

W 39124

Prior to entering Mode 2, TP 600/13, Pressurizer Operational and Spray Flow Test, Section 9.3 should be re-run. This will test the operation of the pressurizer level controller. The acceptance criteria (per Section 11.2 of TP 600/13) is controlled within 16 inches. On 4 September 1977 the pressurizer level controller was successfully tested with results well within the acceptance criteria. A change has occurred since that time. During the 15 minute period immediately prior to the cooldown transient on 23 April 1978, pressurizer level was oscillating with a peak to peak amplitude of 62 inches and a half cycle period of roughly 9 minutes. During this time, Reactor Coolant System pressure was only varying between 2205 psi and 2130 psi. Peak to peak variations in OTSG levels were only 7.5 inches in A and 6.5 inches in B. Tave was oscillating with a period equal to that of pressurizer level with corresponding modes and a peak to peak amplitude of 6°F. It would appear that Tave is the driving force of level swings, however, with no response from the level controller, a 5"/°F change in level will occur. This would induce a swing of only 30", one half of that observed. When properly controlled, MU-V17 should handle 30" swings (peak to peak) induced by a change in Tave especially over a period of 9 minutes. The present state of pressurizer level control at TMI-2 is unacceptable. Fortunately, the trip on 23 April 1978 occurred with an initial pressurizer level some 40" above a minimum recorded just 3 minutes earlier.

Due to recent problems measuring reactivity at TMI-2, it is essential that Section 9.3 of TP 710/1, Reactimeter Checkout, be performed at zero power. Also, perform Section 9.1 of TP 800/12, Unit Load Steady State Tests.

Since TMI-2's NI's have been changed out, the escalation from 0% to 15% power should be performed per Section 9.1 of SP 800/21, Power Escalation Test.

The 15% Power Plateau should be performed per Section 9.2 of SP 800/21 with the exception of steps 9.2.4.2, 9.2.4.3, 9.2.10, 9.2.14, and 9.2.15.

Operation above 15% power is not recommended until the above tests have been satisfactorily completed.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers

L. C. Rogers
Site Operations Manager

cc: L. R. Pletke
W. H. Spangler
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W 39125

1[?]
5-15

B+W BR

Babcock & Wilcox

Power Generation Group

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May 11, 1978

SGM-II-145

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Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: OTSG Internal Accelerometer Signal Recovery

Gentlemen:

As Unit II proceeds along toward resuming the test program, it is apparent that the complicated, expensive OTSG special instrumentation program has been affected by the recent electrical pressure penetrations leakage problems.

B&W has expressed a strong desire to attempt to derive a recovery program which can return the system to a sensible, effective data information collection system.

Problem

Presently, there are no meaningful internal accelerometer signals from the accelerometers installed as part of the OTSG Instrumentation Program. The electrical characteristics of the signals are unstable with a high 60 cycles level indicating the sensor signals are "grounding out." After a week of signal recovery activity, the problem has been isolated to a probability of moisture entering the hardline cable through the connector in the junction boxes. An investigation has indicated moisture has most likely "backed up" the hardline all the way to the accelerometer crystal. The presence of excessive moisture in the junction boxes was caused by leaking conax fittings on both the manway and inspection opening.

W 38126

Proposed Recovery Action

It is imperative that all possible signal recovery action be taken now prior to the next power escalation. Per discussions with S.E. Reed, Alliance Research Center (ARC), the following action is recommended in order to "dry out" the hardline cables. It is possible to "drive" the moisture out of the cables by heating the hardline cables. During the next heatup, the temperature of the Reactor Coolant System should be raised to 250°F to 300°F and stabilized to "soak" the cables, "driving" the moisture out of the OTSG upper head. Then by using a heat gun, heat the hardline cables outside the upper head (from the conax fitting to the junction boxes), "driving" the moisture out of the cables completely.

To help this "drying out" process, a vacuum pump will be attached to the connector area of each hardline connector. The progress of this activity can be monitored periodically by taking resistance measurements of the cables.

As a last resort, it is possible to cut off the connector from each hardline cable thus minimizing the resistance to moisture flow out of the cable. The same process of the Reactor Coolant System at 250°F to 300°F, heat gun external to the upper head, and vacuum pump on the hardline cable will be used to "drive" the moisture out of the hardline. The risk involved in this process is in reattaching a connector to the hardline. It is estimated that a "field fix" of the connector is 50% effective.

In addition to the above described "drying out" process, in order to preclude steam from potential leaking conaxes from entering the junction boxes, action needs to be taken to remove the external cover material routing from the conaxes on the OTSG to the junction boxes.

B&W requests that the site organizations support the field work program in an effort to recover usage of as many of the accelerometer channels as is practical. The result may be that an undetermined period of time with the Reactor Coolant System at ≈ 250°F to 300°F may be required to recover a reasonable number of instrument channels, which allow proper data collection during the remainder of the proposed OTSG special testing period.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39127

LCR/bay

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O+W ck

Babcock & Wilcox

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May 11, 1978

SOM-II-146

Mr. G. P. Miller
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Subject: Reactor Trip/E.S./Cooldown Incident of 23 April 1978

Dear Gary:

As a continuing follow-up of the information regarding B&W's evaluations of the subject transient, please include the following statement with your records of the evaluation results.

A preliminary review of the transient history and sequence of events relating to the rapid cooldown transient experienced at TMI-2 (NSS-6) on 23 April 1978 has been completed. The results of this preliminary evaluation are presented in SOM-II-140, dated 2 May 1978; and SOM-II-143, dated 5 May 1978. The preliminary evaluations performed to date have specifically addressed violations of Plant Technical Specification Items 3.4.9.1 (b) and 3.4.9.2 (a). These preliminary evaluations indicate no deleterious effects on the RCS loop pressure boundary (including pressurizer and surge line) that would prevent continued plant operation. A formal evaluation of the above noted transient will be performed in the near future in order to support the above statements.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/bay

W 39128

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1
5-22

Babcock & Wilcox

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May 18, 1978

SOM-II-148

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Subject: Reactor Coolant Pump Lube Oil and Backstop Oil Systems

Gentlemen:

This letter will provide a summary of events pertaining to the reactor coolant pump problems starting from 14 May at 0158.

On 14 May a low oil reservoir level was noticed on RC-P-2A; RC-P-2A, 1B, and 2B were operating at this time. About 40 minutes after the first alarm, a high upthrust bearing temperature alarm was received, and the 2A was shut off. An excessive amount of oil was seen dripping down from the upper bearing area of the 2A. The plant was subsequently cooled down and cleanup operations commenced. The following items were checked in conjunction with this cleanup effort:

W 39129

1. The oil leak originated from the backstop lube oil filter. A total of ≈ 80 gallons of oil was required to refill the reservoir, which holds ≈ 120 gallons. This filter and gasket were replaced. Leak integrity of the system was then verified.

2. The oil was cleaned from all affected areas. However, all of the oil may not have been recovered from the inside of the J-leg piping insulation.

3. The backstop was visually inspected for evidence of heat checking or oil lacquering. Neither were found.

4. Shaft rotation and backstop backlash were inspected for proper operation (of the backstop) and verified to be operating correctly.

5. The stator area of the motor was inspected for oil and was found to be clean.

6. Terminal boxes and lube oil motors were inspected for oil and found to be clean.

The motor was turned over the operations on the afternoon of 15 May.

On 16 May (≈ 0800), it was reported that the backstop lube oil system alarms were not clearing on the 1A, 2A, and 1B pumps. Inspections of the backstop pumps revealed that four of the eight backstop pumps were rotating backwards. This condition may have existed since the motors were reinstalled following reactor coolant pump casing gasket repairs, due to rewiring errors which were subsequently never checked. This is further amplified because some problems are still being encountered with lube oil alarms during startup and shutdown of the pumps.

The backstop pump rotation problem has been corrected. Rotation of the high pressure lube oil motors does not matter. Each of the four lube oil systems has been operated and found to be operating correctly and leak tight. Some adjustment of the 1B oil regulating valve may be required as it now appears to be relieving an excessive amount of oil.

B&W suggests that the relief valve problem be investigated; however, it should not affect operation of the 1B pump as long as the lube oil header pressure remains > 1800 psig. At some point it would be advisable to check all filters and strainers on all four pumps.

B&W also suggests that all instrumentation to all four pumps and motors be verified for correct termination and contact status. This will provide full assurance that these pumps and motors are wired correctly and that the operators may rely fully on their digital and analog indications on this equipment.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/WDC/bay

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GAK

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May 16, 1978

SOM-II-147
REM-I-354

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Subject: CRDM Stator O-Ring

Gentlemen:

Recently, at an operating B&W plant, during the removal of the CRDM stators concurrent with a refueling, it was reported that some of the bottom o-rings for the stator were damaged.

The failure mode suggests that damage to the o-rings was caused during stator-water jacket installation, most likely because the vendor recommended procedure for installation was not followed.

W 39132

B&W, therefore, would like to take this opportunity to remind your staff that it is imperative that the stator-water jacket installation procedure, as suggested in Diamond Power Speciality Corporation's Control Rod Drive Mechanism instruction manuals, is meticulously followed. This procedure is covered under Section 4.4.3 in the Unit I instruction manual and under Section 4.4.4 in the Unit II instruction manual.

If you or your staff have any questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

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B+W BR

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

May 19, 1978

SOM-II-149

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Subject: TMI-2 Cycle 1 Core Power Distributions for Four Reactor Coolant Pump Operation

Gentlemen:

Attached are the Radial and Total Peak Core Power Distributions at 40%FP for TMI-2, Cycle 1 as revised to reflect the presence of the gadolinium demonstration assemblies and the new rod insertion operating band.

A revision to the Physics Test Manual is being prepared, which will include the core power distributions attached and revised values for 75%FP and 100%FP.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39134

LCR/JV/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
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Radial Power Distribution, 40% FP, 2 EFPD

4 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	1.453	1.206	1.119	1.067	* .997	1.171	1.479	1.030
K		1.163	1.098	1.070	1.030	1.118	1.170	.956
L			1.079	1.036	.920	1.054	1.252	.701
M				1.000	.957	.938	.911	
N					.890	.797	.544	
O						.566		

Group 6/7 91.5 % WD

Group 8 28.8 % WD

Imbalance .69

% Offset 1.72

W 39135

*Gadolinium Assembly Symmetric Location

Figure 2

Steady-State Total Peak Power Distribution,

40% FP, 2 EFPD, 4 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	1.937	1.593	1.505	1.440	* 1.363	1.564	1.994	1.410
K		1.540	1.471	1.468	1.439	1.517	1.566	1.283
L			1.482	1.469	1.454	1.468	1.703	.946
M				1.431	1.368	1.296	1.234	
N					1.241	1.065	.738	
O						.770		

Group 6/7 93.5 % WD

Group 8 28.8 % WD

Imbalance .69

% Offset 1.72

*Gadolinium Assembly Symmetric Location

W 39136

Figure 3

Radial Power Distribution, 40% FP,

4 EFPD, 4 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
E	1.465	1.217	1.127	1.073	* .999	1.170	1.473	1.044
K		1.172	1.107	1.074	1.033	1.117	1.167	.991
L			1.085	1.041	.924	1.053	1.245	.697
M				1.001	.957	.934	.805	
N					.886	.793	.541	
O						.563		

Group 6/7 93.5 % WD

Group 8 28.8 % WD

Imbalance .71

% Offset 1.77

W 3S1C7

*Gadolinium Assembly Symmetric Location

Figure 4

Steady-State Total Peak Power Distribution

40% FP, 4 EFPD, 4 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	2.010	1.655	1.565	1.494	1.409*	1.610	2.037	1.438
K		1.604	1.530	1.521	1.488	1.562	1.603	1.309
L			1.537	1.520	1.499	1.510	1.733	.965
M				1.476	1.408	1.328	1.290	
N					1.272	1.111	.754	
O						.787		

Group 6/7 93.5 % WD

Group 8 28.5 % WD

Imbalance .71

% Offset 1.77

*Gadolinium Assembly Symmetric Location

W 39138

Figure 5

Radial Power Distribution, 40% FP,

10 EFPD, 4 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	1.458	1.220	1.132	1.080	* 1.005	1.168	1.451	1.027
K		1.176	1.114	1.081	1.039	1.116	1.158	.938
L			1.092	1.046	.929	1.053	1.232	.691
M				1.008	.963	.936	.900	
N					.892	.797	.542	
O						.566		

Group 6/7 93.5 % WD

Group 8 28.8 % WD

Imbalance .72

% Offset 1.80

W 39139

*Gadolinium Assembly Symmetric Location

Figure 6

Steady-State Total Peak Power Distribution.

40% FP, 10 EFPD, 4 Reactor Coolant Pump Operation

	8	9	10	11	12	13	14	15
H	1.986	1.650	1.564	1.498	* 1.411	1.599	1.989	1.401
K		1.601	1.532	1.522	1.490	1.552	1.379	1.280
L			1.539	1.526	1.499	1.503	1.706	.948
M				1.479	1.411	1.325	1.244	
N					1.275	1.112	.751	
O						.788		

Group 6/7 93.5 % WD

Group 8 28.8 % WD

Imbalance .72

% Offset 1.80

W 39140

*Gadolinium Assembly Symmetric Location

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May 19, 1978

SOM-II-150

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Subject: TMI-2 Revised Error Adjusted Imbalance Alarm Limits

Gentlemen:

As a result of the completion of B&W's incore uncertainty studies, the error adjusted imbalance alarm limits were revised. The attached limits should be used to update your operating procedures and computer alarm package as required.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/JV/bay

cc: L. R. Pletke
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W 3S141

Table 1

THI-2 CYCLE 1 0 TO 200 EFPD

ERROR ADJUSTED LOCA LIMITS

POWER % OF 2772MW	IMBALANCE LIMIT TECH SPECS	ALARM SETPOINTS FULL INCORE	MINIMUM INCORE	OUT-OF-CORE
0.	-28.10	-20.06	-19.25	-19.56
43.	-28.10	-20.06	-19.25	-19.56
82.	-16.70	-11.96	-8.31	-8.77
92.	-9.30	-5.72	-1.74	-2.12
102.	-9.00	-6.09	-1.14	-1.74
102.	15.80	12.41	6.85	8.10
92.	16.60	12.51	7.87	8.94
80.	19.20	14.16	10.48	11.37
40.	25.40	17.35	16.85	16.85
0.	25.40	17.35	16.85	16.85

W 39142

Table 2

THI-2 CYCLE 1 200 EFPD TO EOC

ERROR ADJUSTED LOCA LIMITS

POWER % OF 2772MW	IMBALANCE LIMIT TECH SPECS	FULL INCORE	ALARM SETPOINTS MINIMUM INCORE	OUT-OF-CORE
0.	-29.60	-21.45	-20.51	-20.95
43.	-29.60	-21.45	-20.51	-20.95
82.	-18.20	-13.36	-9.57	-10.13
92.	-10.80	-7.12	-3.00	-3.49
102.	-10.50	-7.48	-2.40	-3.12
102.	13.70	10.46	5.09	6.16
92.	12.20	8.42	4.18	4.87
73.	20.20	14.64	11.56	12.40
40.	25.50	17.45	16.95	16.95
0.	25.50	17.45	16.95	16.95

3
5-22

Babcock & Wilcox

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May 19, 1978

SOM-II-152
REM-I-355

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Subject: Makeup Pump Dowelling
Reference: GPU Problem Report #2694, dated 22 March 1978

Gentlemen:

The following information is provided with regard to dowelling of the makeup pumps, motors, and high speed gear drive for Units I and II. This information was requested in GPU Problem Report #2694, dated 22 March 1978 and is provided for your use as recommended by Bingham and with B&W approval.

1. Verify alignment of the pump/gear/motor while at normal operating temperatures. Adjust alignment by shimming as required.

2. Dowel the pump using #9 dowels in the holes provided on opposite corners of the casing.

3. Dowel the gear using #9 dowels in the holes provided on the input (motor) shaft side of the gear.

W 39146

4. Dowel the motor on two diagonal feet using #9 dowels.

The B makeup pump on Unit I already has a dowel in place on the gear under the driver end of the output (pump) shaft. Although its placement is not incorrect, B&W recommends that the dowel be removed and all six pumps be dowelled in the fashion outlined above.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/WDC/bay

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JPC - ACTION

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June 6, 1978

SOM-II-154
REM-I-356

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Subject: Customer Questionnaire for Refueling Outage Scheduling

Gentlemen:

This questionnaire is being sent to you with the hope that your knowledge and experience within a nuclear station environment will aid us in gathering the information we need to determine system requirements for a computerized refueling outage scheduling system. We are sending this questionnaire to all of the nuclear stations which B&W services. This will enable us to get a cross section of responses, which will give us a better feel of what this system should do in order to help you during future refueling outages. Your input now will ultimately help you later if the system we develop reflects the needs of your nuclear station.

We will examine all responses very carefully and write up a document showing the results of the survey. These results will be forwarded to all interested sites requesting this feedback. After reviewing the results of our survey, we invite any sites interested in supplying further input into the development of this system to contact A.A. Bovino, Nuclear Services Department in Lynchburg, Virginia. We will welcome any such participation.

If you have any further questions, please do not hesitate to contact me.

W 39148

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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G. A. Kunder
L. L. Lawyer

1. What scheduling periods do you currently use?

A. Hours
B. Shifts
C. Days
D Weeks

2. What criteria determines how each work request is scheduled?

A. Complete Date Only
B. Start & Complete Date
C. Start Date by Crew/Section
D. Complete Date by Crew/Section
E. Start & Complete Date by Crew/Section

3. How and how rapidly is new work identified during the outage passed on to the planning team so that they can make adjustments to the schedule correctly?

4. What increments of time should a computerized scheduling system be able to handle?

5. How important is the ability to plot a full network and/or subnets of a network in a computerized scheduling system? Please give reasons for your answer.

6. How important is the ability to show short duration network plots based upon user requested time frames? Please give reasons for your answer.

W 39150

7. What factors do you evaluate when determining the optimum way of planning the schedule before the outage and adjusting the schedule during the outage? Please briefly explain what you look at.

8. What information should be on a feedback document which sends work status data from crew supervisors to the planning team?

9. How often should work status feedback be received?

   A. Hourly C. Daily
   B. Shift D. Other __________

10. What information would you like to see reported to adequately show the status of work to the people who need to know?

11. Explain the benefits of keeping a planned vs. as-built schedule on the computer.

12. How important is it to highlight work being done by one section which is restraining work to be done by another section?

W 39151

13. Is there a need to factor into the schedule shifts of different durations?

14. Does radiography need to be scheduled as separate work items? Please explain.

15. Where do manpower estimates originate for each work request?

16. Is the source of these estimates different during the outage than when you are in a non-outage situation?

17. Do you feel there is a need to maintain a standardized set of repetitive work done every refueling outage?

18. How would this standardized work help the site plan and control the outage?

19. What information would help you maintain the standard work packages and make it useful in the planning process?

20. Is non-maintenance work sufficiently broken down so that it can be factored into the schedule in a meaningful way?

21. Would the site be willing to identify non-maintenance work so that it could be factored into the schedule properly?

22. Is there a lack of communication between sections concerning work of one section restraining work to be started by another section?

23. How is communication between sections handled during the outage?

24. Do you feel there is a feedback problem from the maintenance crews to the planning team concerning work status data?

25. Do you feel the planning team keeps all support areas up to date with changing schedule conditions?

26. If support areas are not properly kept up to date, what information needs to be communicated?

27. Is there a need to monitor the percent complete of predecessor work which is restraining a successor work event?

28. Who makes your manpower and time estimates for a work item?

29. Where do you feel manpower and time estimates for a work item should originate?

30. How could your current method of determining manpower needs be improved?

31. Where do the planned material requirements for a work item originate?

32. Are the planned material requirements accurate assuming that the original scope of work has not changed?

33. Who should be determining materials needed for a work item?

34. Do the work crews adhere to the schedule as planned or do they take it upon themselves to divert from the schedule without informing the planning team?

35. Why don't the work crews follow the schedule as it is presented to them?

36. Do you feel that work items needing H/P or QA support are properly identified to these groups prior to the initiation and completion of the work? If not, why not?

W 39155

37. How many work crews exist during the outage?

38. Is more than one crew, section or department required for each work request?

39. If there is more than one crew, section or department per work request, could this work be broken out into separate work orders for each section or department?

40. What is the current method of determining the resource requirements for an outage?

41. How does the site currently factor previous outage experience into the resource estimating?

42. What historical information would help you factor previous outage experience into your future resource estimating?

43. Explain the need for knowing manpower allocation status during an outage?

44. Do we need to show resource availability and utilization for each work item?

45. How would you like this system to display resource leveling? (Linear vs. Non-Linear)

46. List the resources we need to consider when planning an outage.

47. List the manpower classifications used at your station during an outage (i.e., craft - or any other classification used).

48. List special tools (e.g., stud tensioner) which are used during an outage and classify them, if applicable.

49. List special equipment (e.g., polar crane) which is used during an outage and classify them, if applicable.

50. Would it be practical to have a method of assigning individuals to work items in the planning phases as opposed to supervisor assignment at work execution time? Explain how this would or would not help.

51. Do you feel that radiation exposure as a constraint to the schedule is feasible? Please explain.

52. How could radiation exposure be a constraint to the schedule?

53. At what level (individual, work item, craft) should radiation exposure be tracked so as to factor it into the schedule as a constraint?

54. Do scheduling constraints for each work item need to be identified and, if so, to whom should they be identified?

W 39158

55. List any other constraints to an outage schedule besides time, manpower, equipment and parts availability.

56. What is the average number of work activities initiated in a day?

57. What is the average number of work activities going on in a day?

58. What percent of work requests are performed during every refueling outage? Please show this (1) relative to total work items, (2) relative to time spent on the work.

59. On the average, how many work requests are executed per refueling outage?

60. What is the average time span for a work request to be completed?

W 39159

61. How would you like to track work requests?

A. By System
B. By Department
C. By Vendor
D. Crew
E. All of the Above
F. Other ________

62. Do you have a difficult time keeping up with new work items once the outage starts? Please explain.

63. What information would help you keep track of new work items?

64. Is there a problem with non-critical work items being overlooked and not recognized until they become critical? Please explain.

65. Is all work, even non-maintenance type work, governed by a work order?

66. If non-maintenance work is not governed by a work order, is there a problem? If so, how can it be remedied?

67. Should there be a procedure verification process for safety related work?

68. What is the present method of verifying a procedure exists for a work item?

69. How is the status of work activities reported and used while the work is in progress?

70. How are completed work items reported?

71. How and when are QA and H/P notified as to when they are needed?

72. What information relative to procedures would help you plan and control your outage?

W 39161

73. Do you feel that a cost accounting function is a necessary ingredient to the type system which we are addressing here? Please explain.

74. How are work activity actuals gathered and used under current situations?

75. Does the site know what costs were included during a refueling outage for labor and materials?

76. Is material and labor data available at the work order level?

77. Is material and labor data accurate enough to be used in refining work estimates for the future?

78. Should manhour and material budgets be maintained?

W 39162

79. How will the site use this budgetary information?

80. What degree of accuracy do you feel is required in the collection of actuals?

81. How should manhour actuals be collected?

82. Do budget overruns and underruns need to be shown?

83. At what various levels of management do you feel cost accounting reports should be aimed?

84. Currently, can the nuclear station show total cost per work activity?

85. Do you feel that the current collection of time/labor actuals incurred against a work order is accurate?

W 39163

86. Do you feel that a spare parts inventory function would be valuable to a computerized refueling outage scheduling system? Please explain.

87. How is an inventory currently kept to reflect planned requirements during a refueling outage?

88. How do you handle spare parts requirements for unplanned work items?

89. What method are you using to determine material requirements?

90. What type of inventories do you currently maintain?

91. Are you able to appropriately keep up with items which are frequently depleted and must be reordered during the outage?

W 39164

92. How does the nuclear station currently monitor procurement dates and insure that they are met?

93. Does the method used for monitoring procurement dates work?

94. Should material procurement status be a constraint to the schedule?

95. How does procurement status presently get factored into the schedule?

96. Do you feel the method used in No. 95 is a good one?

97. Does work identified to be done in pre-outage planning but never accomplished during the outage need to be highlighted in the post-outage audit?

98. What information is needed for a post-outage audit?

99. Who needs to see the post-outage information?

100. How will this information be used?

101. How many work schedules per outage are not being met and how many of those are critical path items?

102. How many work orders which start out as non-critical end up on the critical path?

103. I would like to receive a summary of the survey results.

☐ Yes Address____________________________________

☐ No

104. Would your nuclear station be interested in being involved in the definition of a Computerized Refueling Outage Scheduling System and being kept abreast of its development?

☐ Yes

☐ No

W 39166

B-X
B2

Babcock & Wilcox

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June 8, 1978

SQM-II-155
REM-I-357

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Subject: 1. Non-Condensable Gases in Pressurizer Steam Space
2. Chloride Specific Ion Electrode Procedure

Gentlemen:

Per Section 2.15.5, Part 1, of BAW-1385, the pressurizer steam space should be analyzed for non-condensable gases once per week, and the gases should be maintained at a partial pressure of less than 10 psi. An alternative to weekly analyses is routine venting of the pressurizer gas space through the sample lines. Most plants can vent the pressurizer to the letdown storage tank gas space; in this way, venting does not cause gaseous activity releases.

At steady-state conditions, the dissolved gases in the spray line should be the main source for the non-condensable gas buildup in the steam space; and the venting time can be calculated by the following equation:

$$\frac{c_1}{c_2} = e^{\frac{qt}{V}}$$

W 39167

where: c_1 = initial non-condensable gas volume in steam space, std. ft^3, or partial non-condensable gas pressure in steam space, psia

c_2 = final non-condensable gas volume in steam space, std. ft^3, or partial non-condensable gas pressure in steam space, psia

q = venting rate, lbs steam/hour

t = venting time, hours

v = pressurizer steam space steam inventory, lbs steam

The venting time necessary to maintain proper non-condensable gas concentrations depends on the actual conditions that are determined to exist between venting periods and at the time of venting; and venting time should be adjusted accordingly. For information purposes, the following table gives the venting time required for once per week venting, based on the assumptions listed below the table:

Average Non-Condensable Gas Concentration in RC System, std. cc/kg water	Venting Time for Once Per Week, Hour
30	5.5
40	7.5
50	10
60	16.6
100	28

Assumptions

1. Steady-state operation with normal water level in pressurizer, i.e., 700 ft^3 steam space containing 45 lbs steam.

2. 1.5 gpm continuous flow through spray bypass line.

3. Vent rate of 200 lbs/hr, based on design rating of pressurizer steam space sample cooler.

4. All dissolved gases in spray bypass line water (reactor coolant gas concentration) collect in pressurizer steam space.

W 39168

5. Partial pressure at start of venting is 10 psia, and sufficient venting is performed to limit buildup to 10 psia prior to next weekly venting.

In reference to subject number two, transmitted herewith is a new procedure for inclusion into the BAW 1385, Water Chemistry Manual.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
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SUBJECT: DETERMINATION OF CHLORIDE IN HIGH PURITY WATER AND BORIC ACID SOLUTIONS USING A CHLORIDE SELECTIVE ION ELECTRODE

1. Scope

This method utilizes the Ionel HgS/Hg_2Cl_2 specific ion electrode for the determination of chloride in high purity water and in water containing boric acid and lithium at concentrations of 13,500 ppm and 2 ppm, respectively.

The range of chloride concentration covered by this method is 10 to 500 ppb as Cl^-. Higher concentrations can be determined simply by running higher standards and extending the calibration curve.

2. Principle of Method

The HgS/Hg_2Cl_2 solid state chloride ion sensitive electrode develops a potential proportional to the activity of the chloride ion according to the Nernstian equation. Potentials are increasingly negative with increased chloride concentrations. Lower limit of detection is determined by the solubility product of Mercurous Chloride ($Hg_2Cl_2 \rightleftharpoons 2Hg^+ + 2\ Cl^-$). As chloride ions present in standards or samples come in contact with the sensor element, the above reaction goes to the left, creating a change in potential across the electrode.

3. Definitions

3.1 Reagent Grade - Conforming to ASTM specification E-200.

3.2 Reagent Water - Conforming to ASTM specification D 1193, Type I.

4. Interferences

4.1 The electrode will malfunction if ions which form very insoluble salts with mercury are present at sufficiently high concentrations (S^{2-}, I^-, Br^-, CNS^-).

4.2 The electrode will function satisfactory in the presence of the following contaminants:

4.2.1 6 ppm morpholine and 15 ppm hydrazine

4.2.2 6 ppm morpholine, 15 ppm hydrazine, and 1 ppm fluoride

4.2.3 200 ppm hydrazine and 20 ppm ammonia

W 39170

4.2 Electrode potential is affected by total ion strength, therefore pH and total ionic strength of both samples and standards must be kept constant.

4.3 Electrode potentials are affected by temperature; standards and samples should be approximately the same temperature.

5. Solutions

5.1 Concentrated chloride ion standard - 0.1 M (Orion Research, Cambridge, MA).

5.2 Ionic Strength and pH Adjuster - 2 M HNO_3, 126 g or 84 ml concentrated reagent grade nitric acid (HNO_3)/liter.

5.3 Internal filling solution for reference electrode (1 M KCL saturated with AgCl).

5.4 External filling solution for reference electrode (1 M KNO_3).

6. Apparatus

6.1 Chloride Ion-Selective Electrode (Ionel Co., Mount Hope, Ontario, Canada, Phone 416-632-7212).

6.2 Double Junction Reference Electrode (Model 90-02-00, Orion Research, Inc.).

6.3 High input impedance (10^{12}) specific ion meter or pH meter with expanded mv scale.

6.4 Mixer, magnetic with a TFE fluorocarbon-coated stirring bar.

7. Calibration

7.1 Prepare a series of chloride standards to cover the range of 10 to 500 ppb; at least four (4) standards should be determined. Prepare standards by appropriate dilution of the 0.1 M chloride solution described in Section 5.1.

7.2 Carry standards through Step 8.

7.3 Plot mV readings vs. chloride concentration on semilogarithmic graph paper (4 cycles).

7.4 Follow the directions of the manufacturer for the operation of the specific ion meter and electrodes.

8. Procedure

8.1 Rinse all items that will be in contact with blanks, standards and samples thoroughly with reagent water.

8.2 Add 100 ml of reagent water to a 150 ml pyrex beaker and immerse the electrode (at a reproducible depth and position) in the solution.

8.3 Add 1 ml of 2 M nitric acid (HNO_3), set the meter to mV output and stir the solution at medium speed (magnetically) for exactly three (3) minutes. Record the mV readings on the expanded mV scale.

8.4 Repeat this procedure with standard solutions and samples. Both electrodes should be thoroughly rinsed with reagent water after each reading.

8.5 Chloride concentration is read directly from the calibration curve plotted in Step 7.3.

9. Notes

9.1 Should slow electrode response persist, polish the sensing element of the chloride electrode with alumina powder.

9.2 The specific ion meter should be zeroed on the millivolt scale by shorting the reference and test jacks. A meter reading of ±0.1 millivolts is acceptable. With the meter zeroed, reagent water with 1.0 ml HNO_3 should read +360 to 375 millivolts. Readings of standards and samples will be increasing negative from this point.

W 39172

9.3 A pH of 2.0 is optimum for minimizing ion interference with the sensing element. Samples suspected of having high pH should be checked and additional acid added, if required.

9.4 Mercuric salts are poisonous. After handling the chloride electrode, hands should be washed prior to eating or smoking.

W 39173

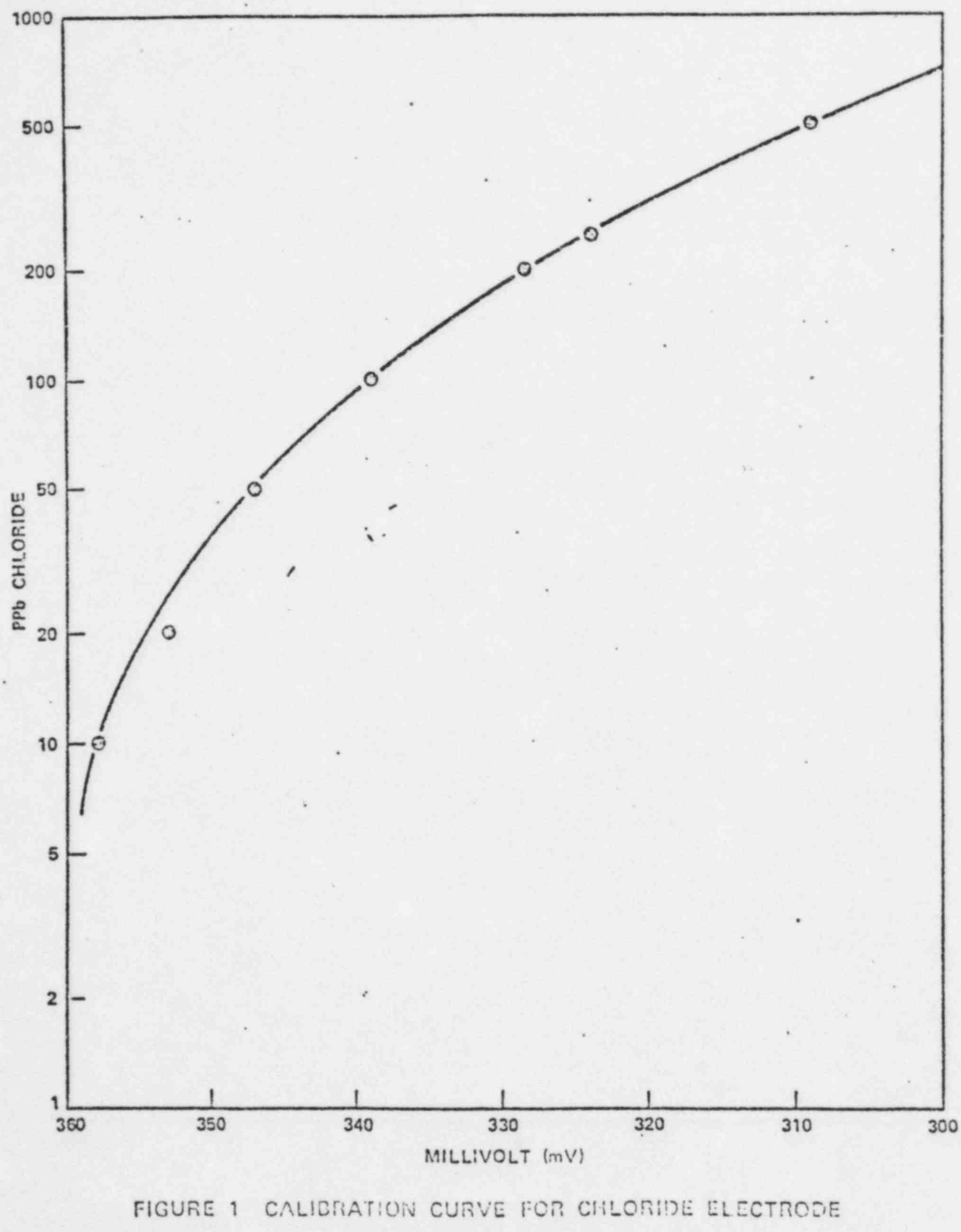

FIGURE 1 CALIBRATION CURVE FOR CHLORIDE ELECTRODE

W 39174

B&W Book

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
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June 9, 1978

SOM-II-156

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Manager, Generation Operations
Metropolitan Edison Company
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Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: ORA/BPRA Retainer Installation: B&W Support

Gentlemen:

The long-handled tool (5 sections) for ORA removal, arrived on site Tuesday, 6 June 1978. Attached is a procedure for the assembly of this tool.

Also attached is a procedure to be used for fuel assembly alignment verification after the BPRA and modified ORA retainers have been installed.

During the steps outlined in SOM-II-153, B&W advise and consultation will include shift coverage with two people from Lynchburg and one person from the site office. During the inspection of the fuel assembly locking cups, B&W personnel will handle and operate the special inspection equipment. This equipment, which is contaminated, will be brought on site Sunday, 11 June 1978. Please make the necessary arrangements for receipt on site.

The following four people are scheduled to arrive on site late Sunday or Monday (11 June or 12 June). Since they will be subject to 12-hour shifts, it is advisable for them to have basic and RWP training to eliminate the need for escorts. Please make the necessary arrangements and advise this office of the RWP training schedule.

W 39175

John Mayer	-	B&W Fuels Engineering
Bob Copeland		"
Bill Shield	-	B&W LRC
Dave Kimmel		"

The above dates of arrival may change due to site schedule changes.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
J. L. Seelinger
E. G. Schaedel

5/1/78

BPRA/ORA Grappling Tool
Assembly Procedures (Vertical)

1. Assemble bracket and attach it to guard rail of fuel handling bridge.
2. Hang lower section from bracket with coupling down.
3. Couple inner tube at 1st joint and mate flanges.
4. Install and tighten bolts (with lock washers) and install safety wire. Hex head = 9/16 inch.
5. Repeat steps 3 and 4 at all but upper section joint..
6. Rotate feed knob to show indicator pin at engaged ("ENG") position.
7. Couple inner tube at upper joint.
8. Rotate upper tube section clockwise to mate flanges. Align flats. (CAUTION: Do not over-engage.)
9. Install bolts (with lock washers) hand tight and verify proper travel of indicator pin. Pin should travel to within 1/16 inch of slot ends. Repeat step 8 if necessary.
10. With indicator pin at "TEST" position, tighten bolts at upper joint and install safety wire.
11. Rotate feed knob to show indicator pin at release ("REL") position.

TOOL IS NOW READY FOR USE!

W 39177

5/1/78

BPRA/ORA Grappling Tool
Assembly Procedures (Horizontal)

1. Lay inner and outer tube sections on floor in proper sequence.

2. Couple inner tube at lower (1st) joint and mate flanges.

3. Install bolts and tighten (with lock washers). Hex head = 9/16 inch.

4. Repeat steps 2 and 3 at all but upper section joint.

5. Rotate feed knob to show indicator pin at engaged ("ENG") position.

6. Push inner tube toward lower coupling and verify that plunger protrudes approximately 3/4 inch from end.

7. Couple inner tube at upper joint.

8. Lift upper tube section off floor and rotate it clockwise to mate flanges. Align flats. Support upper joint off floor and apply a force toward lower end of tool while rotating. (CAUTION: Do not over-engage.)

9. Install bolts (with lock washers) hand tight and verify proper travel of indicator pin. Pin should travel to within 1/16 inch of slot ends. Repeat step 8 if necessary.

10. With indicator pin at "TEST" position, tighten bolts at upper joint and install safety wire at all joints.

11. Rotate feed knob to show indicator pin at release ("REL") position.

TOOL IS NOW READY FOR USE!

W 39178

1.0 TEST OBJECTIVE

To verify that the alignment of fuel assembly upper end fittings relative to the reactor internals core basket and adjacent fuel assemblies is adequate to permit the installation of the plenum assembly. If excessive misalignment exists, this specification covers the corrective actions to be taken.

2.0 ACCEPTANCE CRITERIA

No gaps or misalignments greater than one-quarter (1/4) inch between adjacent fuel assemblies exist.

3.0 CONDITIONS PRIOR TO TEST

3.1 Core loaded and ready for final verification that all fuel assembly core positions are correct.

3.2 Water level sufficient for shielding during inspection.

3.3 Water turbulence reduced to a minimum.

4.0 SPECIAL PRECAUTIONS

All mechanisms operated over the fuel transfer canal must have all small tools and loose parts secured by a safety line or some other positive means to prevent their falling into the reactor or fuel transfer canal.

5.0 DATA

Record data on the core map (Data Sheet 7.3.1) and data work sheet (Data Sheet 7.3.2) as required by Section 6.0 of this specification. Inspections should be recorded on videotapes.

6.0 PROCEDURE

6.1 Visually scan the top of the core. A video camera and monitor are recommended. If the gaps between end fitting assemblies appear to be uniform and form straight lines, the position check is complete.

6.2 Record locations (if any) where the Step 6.1 criteria are not met on the core map (Data Sheet 7.3.1) and proceed with Step 6.3.

6.3 Closely examine gaps where the end fittings do not form straight lines. Visually estimate maximum gap dimensions at these locations. Use known upper end fitting dimensions from Figure 7.4.1 as an aid when estimating. A ruler can be used on the video monitor screen to adequately ratio the maximum gap dimensions to a known upper end fitting dimension.

W 39179

6.3.1 If the maximum gap dimension is less than one-quarter (1/4) inch, the inspection is complete.

6.3.2 If the maximum gap dimension exceeds one-quarter (1/4) inch, proceed to Step 6.4.

6.4 Fuel Assembly - When a gap greater than one-quarter (1/4) inch is present, one or both fuel assemblies on the sides of the gap are misaligned. The customer must exercise judgment to identify the misaligned assembly. Generally, it is expected one assembly will deviate more from the straight lines on the rows or columns running parallel to the gap than will the other assembly. The assembly with the larger deviation is considered misaligned. Further, it is expected the misaligned fuel assembly will have no gap on the side opposite the large gap, plus no gap or a smaller-than-normal gap between other assemblies along the row or column opposite the gap. When a misaligned fuel assembly is identified, record dimensions of gap around the misaligned assembly(s) on Data Sheet 7.3.2, and contact Babcock & Wilcox for resolution.

7.0 ENCLOSURES

7.1 List of References

7.1.1 Fuel Loading and Refueling Limits and Precautions

7.1.2 Initial Fuel Loading

7.1.3 Fuel and Control Component Handling

7.1.4 Fuel Handling Administration

7.1.5 Technical Specifications

7.2 Equipment

7.2.1 Variable Intensity Underwater Lighting

7.2.2 Video or optical equipment which will permit accurate measurement of clearances.

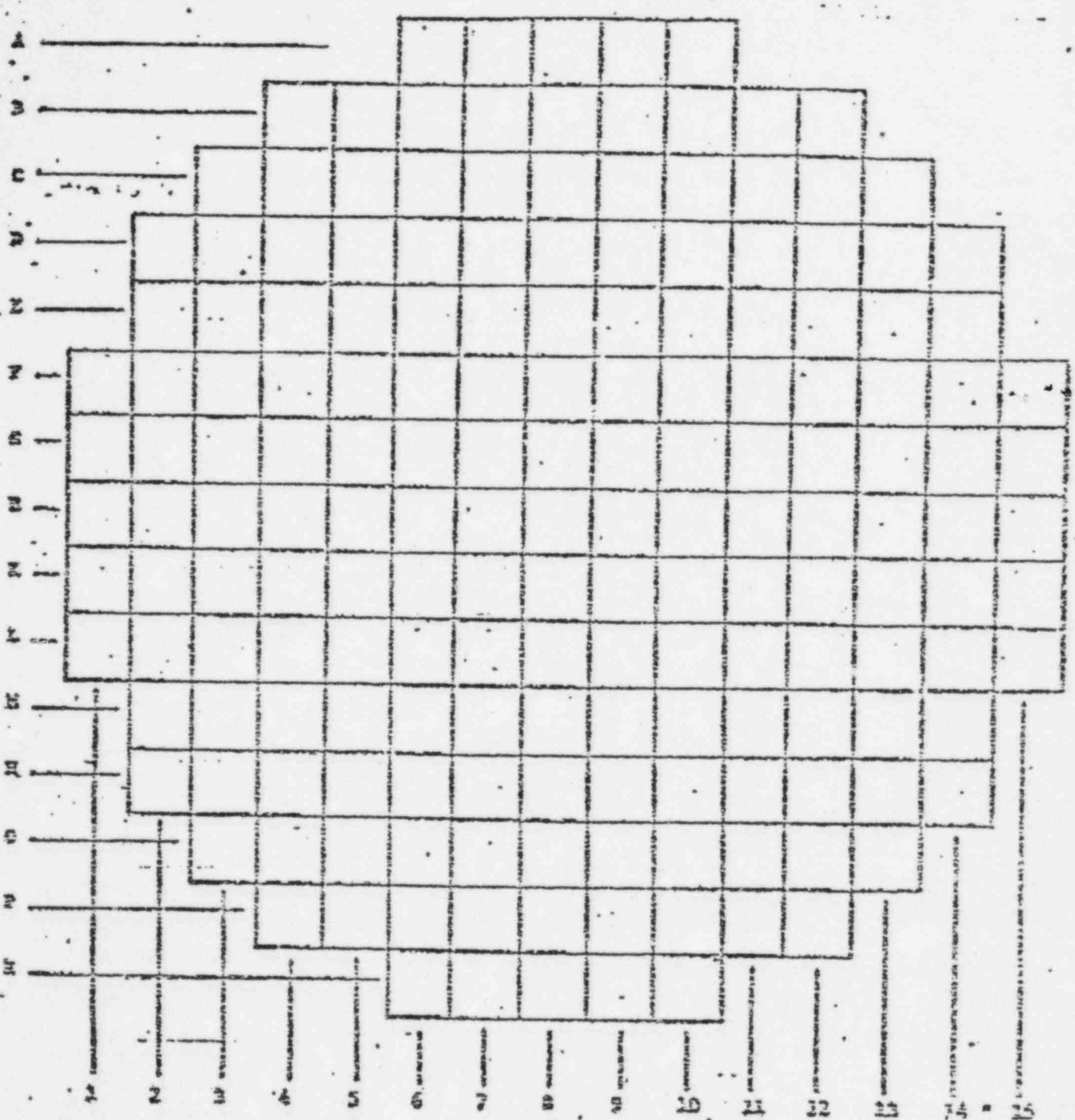
7.3 Data Sheets

7.3.1 Core Map

7.3.2 Dimensions for End Fitting Gap

7.4 Figures

7.4.1 Mark B Upper End Fitting Dimensions

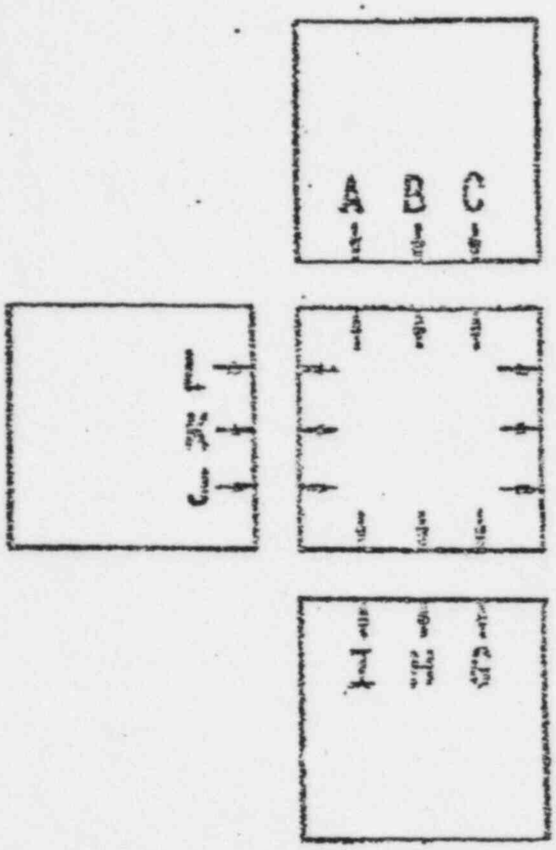


RECORD CORE ORIENTATION BY LABELING SIDES WXYZ OR DIRECTION OF NORTH

DATE: PAGE SERIAL: W 39181

Data Sheet 7.3.2. Dimensions for End Fitting Gaps

Column (number) ____________



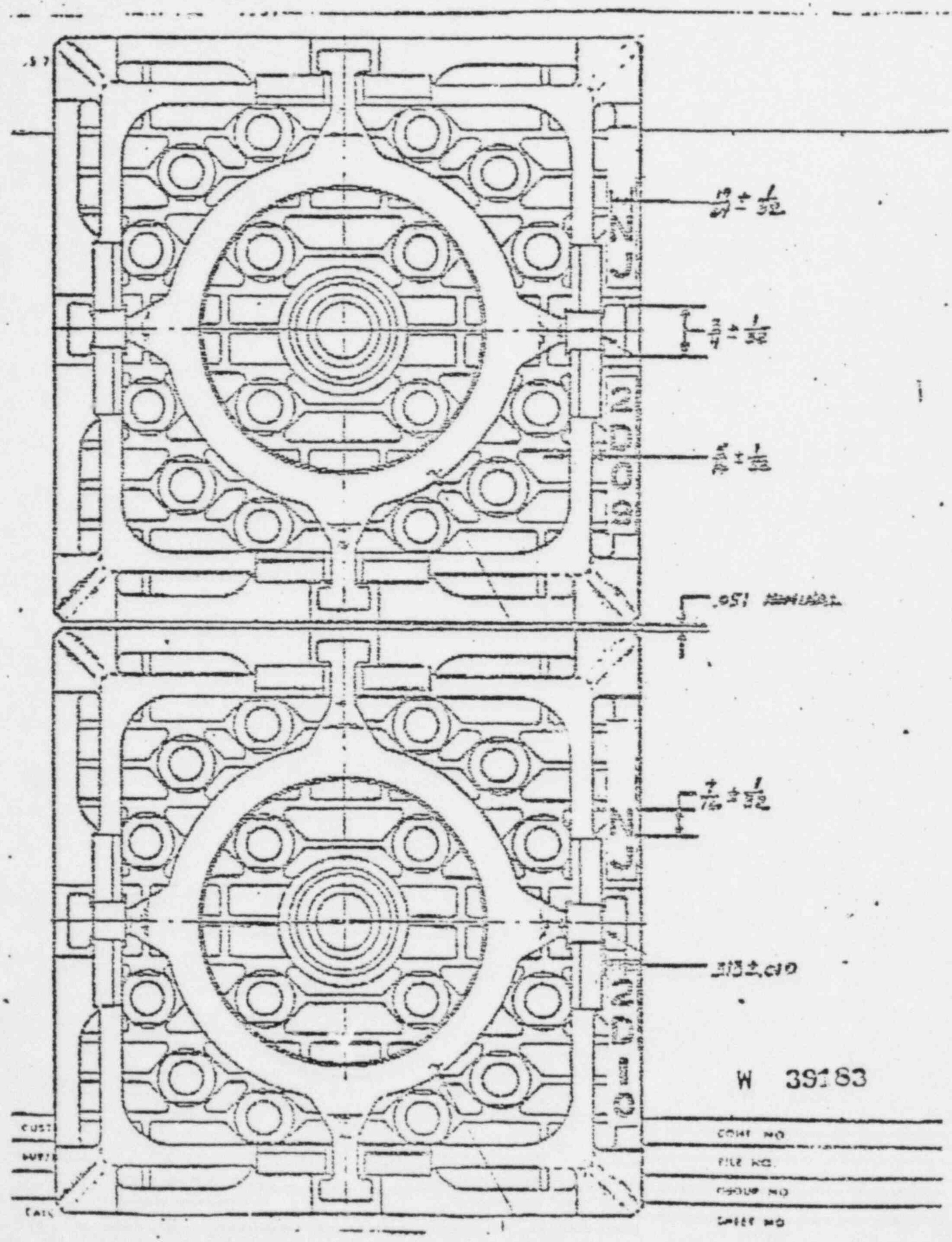
Row (letter) ____________

Center assembly is misaligned assembly

Gap Location	Estimated Gap Width, Inches
A	______
B	______
C	______
D	______
E	______
F	______
G	______
H	______
I	______
J	______
K	______
L	______

W 39182

FIGURE 7.4.1. MARK B UPPER END FITTING DIMENSIONS



Pole Book

Babcock & Wilcox

Power Generation Group

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June 5, 1978

SOM-II-153

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~~Mr. G. P. Miller~~
Station Superintendent
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Post Office Box 480
Middletown, PA 17057

Subject: Core Inspection - ORA and BPRA Locations
Reference: Letter, L. Pletke to J. Barton, "Plan for ORA's and BPRA's in TMI-2," dated 31 May 1978

Gentlemen:

As outlined in the referenced letter, and agreed upon at a meeting held on site 2 June, the steps involved in the ORA/BPRA program are as follows:

1. Remove all 40 ORA's from the core and store (storage location at Met-Ed's discretion).

2. Inspection of the fuel assembly holddown latch assemblies for:

    a. The 40 ORA assemblies
    b. 20 designated BPRA assemblies

3. Inspection of designated fuel assembly holddown latch assemblies, from Unit I, located in Unit I spent fuel pool. This will be done after steps 1 and 2 above or during a 1 to 2 day delay in step 2 above.

W 39184

4. Installation of 2 modified ORA's in the fuel assemblies containing the sources.

5. Inspection of fuel assemblies, installation and position verification of the 70 BPRA and modified ORA retainers.

6. Identify all additional testing due to the above work and incorporate into the test program.

This letter formally transmits procedures for steps 1, 2 and 5. These procedures were hand delivered on Friday, 2 June. A procedure for step 3 and a letter summarizing step 6 will be sent at a later date.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
J. L. Seelinger
J. F. Hilbish

ORA REMOVAL PROCEDURE

1. Receive manual tool on site (from TECO). The tool is contaminated.
2. Move tool into the Reactor Building.

   NOTE: This tool has five (5) sections.
3. Assemble the tool on the operating level of the Reactor Building. This will consist of bolting sections together and lock wiring the bolts. A B&W Engineer will assist with this assembly. The polar crane should be used to assemble the tool in a vertical position and then transferred to the fuel transfer canal.
4. The tool should be connected to the auxiliary fuel handling bridge monorail hoist, or equivalent, with a spring scale sufficient to read 200 lbs.
5. Position tool over the first ORA to be removed. Record tool weight one time (≈ 80 lbs). Assure tool position is in "release" position.
6. Lower tool and insert into ORA spider coupling. The spring scale will show engagement when the reading is less than the tool weight.
7. Rotate knob on tool to "engage" position.
8. Raise tool but do not exceed 200# lifting force above combined tool and ORA static weight (ORA weight = 12 lbs wet).

8A. If ORA does not lift with a 200# force, shake the tool or attempt to twist (rotate) the tool (not the engage-release knob) while maintaining the 200# upward force on the tool. Continue to do this until the ORA spider assembly frees itself from the fuel assembly latching cup.

9. Raise ORA out of core and move to storage container position.
10. Inspect, by use of television camera, the ORA ball locking device. A video tape recording should be made (include ORA serial number on the video tape).

W 39186

11. Place ORA in storage container and rotate knob on tool to the "release" position.

12. Repeat for all 40 ORA's.

INSPECTION OF FUEL ASSEMBLY LOCKING CUPS

A. 40 Fuel Assemblies that originally contained ORA's

1. Verify and designate fuel assembly location.
2. A fuel bridge will be used as a platform for the inspection equipment. Position the inspection equipment over the fuel assembly locking cup. A B&W Engineer will operate the inspection equipment.
3. Rotate the inspection equipment at least 360° and record on video tape while viewing on a television monitor.
4. Remove inspection equipment from the fuel assembly.
5. Repeat for all 40 ORA locations.

B. 20 Fuel Assemblies that contain BPRA

6. The attached core map shows the location of the 20 BPRA for inspection of locking cups.
7. Inspection equipment will be located on the main fuel handling bridge.
8. Index control rod mast over first BPR fuel assembly to be inspected.
9. Remove BPRA from the fuel assembly and raise into the mast. (This can be accomplished using existing Met-Ed procedures.)
10. Move bridge to allow access of the inspection equipment to the fuel assembly locking cup.
11. Follow steps 2 through 4 above.
12. Repeat for all 20 designated locations.
13. If any BPRA cannot be removed using the control rod mast, record and continue to next location.
14. After all 20 locations have been inspected, report to B&W any assemblies that were bypassed because the BPRA could not be removed.

TMI-1 INITIAL CORE LOADING PLAN
REVISION 1

FUEL TRANSFER CANAL →

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						TP OXX	TY OXX	UC OXX	TC OXX	TW OXX					
B				UT OXX	UX OXX	UG C139	RL B153	UB C124	RF B158	UA C131	U1 OXX	U4 OXX P			
C			TS OXX	T7 B164	PW C163	SN B174	Q8 C179	SY B177	QB C172	SC B173	PY C156	TB B163	TF OXX		
D		TR OXX	TA B168	PV C155	T5 B197	Q4 A024	SW B160	RC C144	S1 B183	Q2 A017	T4 B200	PT C152	T8 B169	TD OXX	
E		U2 OXX	QS C162	PN B198	QN C131	RX B195	QG C171	SZ B187	QJ C164	TO B204	R8 C128	PF B206	PZ C157	U5 OXX	
F	TN OXX	UR C138	RW B171	Q5 A023	RN B202	R6 C151	RS B146	QZ C140	S4 B155	QK C148	S7 B201	QQ A018	RH B182	UQ C133	TG OXX
G	TM OXX	SQ B147	QV C178	SO B185	R2 C170	RQ B145	QX C183	RP B149	QE C180	SM B139	QH C165	SS B176	QA C173	RK B157	US OXX
H	UU OXX	UF C127	S2 B186	RD C147	T1 B189	QY C143	SH B143	RR C123	SJ B142	QF C141	T2 B188	RA C145	SX B184	UE C125	UY OXX
K	UW OXX	RM B152	Q7 C177	RU B161	R4 C169	T3 B141	QD C182	RT B144	QM C181	SL B154	QC C166	SB B179	Q9 C174	RJ B151	UV OXX
L	TQ OXX	UK C137	RG B162	QP A022	SR B192	QL C150	SK B156	RO C142	S3 B150	QW C149	RV B199	Q3 A019	SA B165	UL C134	TJ OXX
M		UO OXX	QR C161	PM B193	Q6 C130	S8 B194	R3 C168	S6 B159	R1 C167	RY B191	R9 C129	T6 B205	QT C158	TU OXX	
N		TZ OXX	U9 B175	PU C154	SG B203	Q1 A021	SV B180	RB C146	RZ B181	QU A020	PQ B196	PS C153	UH B190	TE OXX	
O			U8 OXX	UJ B166	QO C160	S9 B167	R7 C176	ST B172	R5 C175	SU B178	PX C159	T9 B170	TT OXX		
P				TV OXX P	U6 OXX	UP C136	RE B148	UM C126	SP B140	UN C135	U7 OXX	TX OXX			
R						TH OXX	TL OXX	UD OXX	U3 OXX	TK OXX					

Z

W 39189

Fuel Assembly ID (PS-RD: Batch 1; RE-T6 & PM-PQ: Batch 2; T7-UY: Batch 3)
Control Component ID (CXXX - CRA; AXXX - APSRA; BXXX - BPRA; OXX - ORA)
Sources (P - Primary)

NOTE: NJ00 precedes all fuel assembly ID's.

Installation of Burnable Poison
and Modified Orifice Rod Assembly Retainers
at Three Mile Island II

1. This operation may be performed with water level in the refueling canal at normal refueling level or at the RV flange. The retainers should be installed just prior to installation of the plenum. An underwater TV camera will be required for installation and verification of the retainers.

Procedure

1.0 Initial conditions

1.1 Reactor head and plenum have been removed.

1.2 Water level is at predetermined level.

1.3 A fuel bridge is available as a work platform.

1.4 The underwater TV camera is available.

1.5 Retainers baskets with retainers have been positioned on canal floor next to RV flange.

1.6 HP coverage is available.

1.7 Orifice Rod Assemblies have been removed, AND THE (2) TWO MODIFIED ORA'S HAVE BEEN INSTALLED

1.8 ½ to 1 ton chain fall available.

2.0 Installation of the BPRA and Mod. ORA retainers

2.1 Attach a retainer to the Special Handling Tool and insure that the retainer is locked on to the tool.

2.2 Slowly lower the retainer onto the desired BPRA or Mod. ORA.

2.3 Maneuver the tool until the retainer is completely seated on the spider. This is confirmed when the four (4) cutouts on the flange of retainer engage with the inner most row of nuts on spider.

2.4 Verify with underwater camera that retainer is engaged and seated on spider.

2.5 Verify orientation of retainer feet. Position of feet should be 45° (degrees) clockwise from W or Y axis.

2.6 Disengage the retainer tool lock and remove form RV.

2.7 Repeat steps 2.1 through 2.6 until all retainers have been installed.

W 39190

2.8 Confirmation that the retainer is properly positioned shall be accomplished by viewing and video taping with underwater TV camera.

The four (4) cutouts on the retainer flange are engaged with the four (4) inner most row of nuts on the spider.

The orientation of retainer feet are as stated in step 2.5.

The top plate of the retainer is approximately even with the top of the female coupling of spider.

2.9 Remove Special Handling Tool and baskets from canal.

3.0 Final conditions

3.1 All retainers have been installed and verified to be properly positioned.

Babcock & Wilcox

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June 13, 1978

SOM-II-157
REM-I-353

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Mr. J. P. O'Hanlon
Superintendent, Unit I
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Middletown, PA 17057

Subject: Preventive Maintenance of CRDCS Breakers

Gentlemen:

In the past, some of B&W's plants have experienced problems with CRDCS trip breakers. The problems have been traced to lack of preventive maintenance. B&W suggests that a planned, carefully executed, maintenance program be established using the maintenance program outlined in the Diamond Power CRDC System Vendor Manual. Particular attention should be directed to proper cycling, cleaning, and lubrication of the breakers. B&W further recommends that this program be scheduled at a minimum frequency of every refueling cycle and more frequently for plants during startup when the equipment is subjected to adverse environmental conditions.

We understand that you had a vendor representative on site to service these breakers during the past refueling outage of Unit I, and you intend making this preventive maintenance of the CRDCS breaker a regular feature of subsequent refueling outages.

W 39192

B&W suggests that a similar preventive maintenance program be instituted for TMI-2 CRDCS breakers.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
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J. B. Logan

W 39193

B+W Book.

Babcock & Wilcox

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June 19, 1978

SOM-II-159

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Middletown, PA 17057

Subject: Turbine Bypass Valve Setpoint Change

Gentlemen:

Recently, the turbine bypass valve (TBV) overrride setpoints were scrutinized by B&W Control Analysis in regard to some of the transients experienced at TMI-2.

In order to improve plant transients involving load rejections and turbine trips, it was suggested in GPU Startup Problem Report 2753 that this setpoint should be reduced to 1025 psig. This setpoint change should provide faster turbine bypass valve response without exceeding the Reactor Coolant System cooldown rates following a reactor trip.

It is suggested that you change the setpoints on modules IC 21.5 and IC 21.13 to 1025 and update page 34 of the plant setpoints document, OP 2101-02 to show the new setpoints.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E. S. Schaedel
for
L. C. Rogers
Site Operations Manager

W 39194

LCR/SFM/bay

cc: L. R. Pletke
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R. J. Toole

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June 13, 1978

SOM-II-158

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Mr. G. P. Miller
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Metropolitan Edison Company
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Middletown, PA 17057

Subject: ORA/BPRA: Material Receipt

Gentlemen:

The 72 BPRA retainers, 3 modified ORA's, installation tool, and 2 aluminum racks arrived on site 13 June. The QC data packages for the retainers and modified ORA's were included with the above material. A formal site receipt and inspection has been tentatively arranged for 14 June at 1 p.m.

A question was raised concerning the boron concentration required during movement of the BPRA's. A boron concentration of 2000 ppm will be adequate during the current ORA/BPRA outage to insure that Keff $\leq$ .95. This assumes that there will be at no time more than one BPRA withdrawn from the core.

Three (3) of the seventy-two (72) retainers have been specified for use with the modified ORA's. They are: L004, L005, and L006.

Attached is a suggested core loading plan. Please send this Office a copy of the actual core loading plan after installation.

If you have any further questions, please do not hesitate to contact me.

W 38195

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
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J. F. Hilbish

TMI-II INITIAL CORE LOADING PLAN
REVISION 2

FUEL TRANSFER CANAL →

X

					TP	TY	UC	TC	TW					
			UT	UX	UG C139	RL B153 L007	UB C124	RF B158 L008	UA C132	U1	U4 03AT & L004 P			
		TS	T7 B164 L009	PW C163	SN B174 L010	Q8 C179	SY B177 L011	QB C172	SC B173 L012	PY C156	TB B163 L013	TF		
	TR	TA B168 L014	PV C155	T5 B197 L015	Q4 A024	SW B160 L016	RC C144	S1 B183 L017	Q2 A017	T4 B200 L018	PT C152	T8 B169 L019	TD	
	U2	QS C162	PN B198 L020	QN C131	RX B195 L021	QG C171	SZ B187 L022	QJ C164	TO B204 L023	R8 C128	PP B206 L024	PZ C157	U5	
TN	UR C138	RW B171 L025	Q5 A023	RN B202 L026	R6 C151	RS B146 L027	QZ C140	S4 B155 L028	QK C148	S7 B201 L029	QQ A018	RH B182 L030	UQ C133	TG
TM	SQ B147 L031	QV C178	SO B185 L032	R2 C170	RQ B145 L033	QX C183	RP B149 L034	QE C180	SM B139 L035	QH C165	SS B176 L036	QA C173	RK B157 L037	US
UU	UF C127	S2 B186 L038	RD C147	T1 B189 L039	QY C143	SH B143 L040	RR C123	SJ B142 L041	QF C141	T2 B188 L042	RA C145	SX B184 L043	UE C125	UY
[illegible]	RM B152 L044	Q7 C177	RU B161 L045	R4 C169	T3 B141 L046	QD C182	RT B144 L047	QM C181	SL B154 L048	QC C166	SB B179 L049	Q9 C174	RJ B151 L050	UV
TQ	UK C137	RG B162 L051	QP A022	SR B192 L052	QL C150	SK B156 L053	RO C142	S3 B150 L054	QW C149	RV B199 L055	Q3 A019	SA B165 L056	UL C134	TJ
	UO	QR C161	PM B193 L057	Q6 C130	S8 B194 L058	R3 C168	S6 B159 L059	R1 C167	RY B191 L060	R9 C129	T6 B205 L061	QT C158	TU	
	TZ	U9 B175 L062	PU C154	SG B203 L063	Q1 A021	SV B180 L064	RB C146	RZ B181 L065	QU A020	PQ B196 L066	PS C153	UH B190 L067	TE	
		U8	UJ B166 L068	QO C160	S9 B167 L069	R7 C176	ST B172 L070	R5 C175	SU B178 L071	PX C159	T9 B170 L072	TT		
		L005 &	TV 03AU P	U6	UP C136	RE B148 L073	UM C126	SP B140 L074	UN C135	U7	TX			
					TH	TL	UD	U3	TK					
1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

Z

W 39197

Fuel Assembly ID (PS-RD:Batch 1; RE-T6 & PM-PQ:Batch 2; T7-UY:Batch 3)
Control Component ID (CXXX - CRA; AXXX - APSRA; BXXX - BPRA; OXXX-MOD.ORA; LOXX -RETAINER)
Sources (P - Primary)

NOTE: NJ00 precedes all fuel assembly ID's

B+W [illegible]

Babcock & Wilcox

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June 21, 1978

SOM-II-160

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Reading, PA 19603

[redacted]
Station Superintendent
Metropolitan Edison Company
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Middletown, PA 17057

Subject: Final Inspection & Completion of ORA/BPRA Program for Unit II

Gentlemen:

On Friday afternoon, 16 June, the decision was made to use the manual tool in order to loosen two BPRA's that could not be removed from the fuel assembly with the fuel handling control rod mast. This involved removal of four BPRA retainers, usage of the manual tool only to unlatch the BPRA and then removal of the BPRA with the control rod mast.

This letter formally transmits the procedure, Alternate Procedure for Removal of Lumped Burnable Poison and Orifice Rods from Fuel Assemblies, that was given to you on Friday, 16 June.

The above mentioned work, including the inspection of the fuel assemblies and reinstallation of the two BPRA's and four retainers, was completed at 2100 on Friday, 16 June. This concluded the ORA/BPRA Program for Unit II.

W 39198

Also attached is the recommended inspection plan for Unit I fuel assemblies. This was given to you on Tuesday, 20 June. When the spent fuel bridge (Unit I) is operational, the B&W inspection team will return to the site to complete this phase of the program.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E. G. Schiedel
for
L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
J. L. Seelinger
J. F. Hilbish

W 39199

Subj. ALTERNATIVE PROCEDURE FOR REMOVAL OF LUMPED BURNABLE POISON AND ORIFICE RODS FROM FUEL ASSEMBLIES

Date JUNE 1,1978

This letter to cover one customer and one subject only.

1) Using manual tool, unlatch LBPR/OR from fuel assembly, raise about one inch, then lower such that locking balls rest on top of FA locking cup.

2) Set CR mast lowload limit at 50 pounds less than the downward moving weight of the unloaded grapple (this limit may be incrementally increased to 100 pounds if required because of excessive hoist cutoff, but <u>should</u> <u>not</u> be increased for convenience only).

3) Position Fuel Handling Bridge Control Rod mast over applicable core position.

4) Lower inner mast (jogging final 2 inches) onto fuel assembly.

5) Read and record Z-Z tape.

6) Raise inner mast 1/2 inch. Bypass lowload interlock.

7) Lower telescopic cylinder to full down.

8) Set CR grapple switch to "engage".

9) Withdraw LBPR/OR to full up position inside inner mast.

10) Raise inner mast.

W 39200

TMI-1 LATCH TUBE AND BPR INSPECTION PLAN
(Spent Fuel Pool)

Introduction

In order to further characterize the latch tube wear phenomenon, several fuel assemblies and burnable poison rod assemblies (BPRA's) currently in the TMI-1 spent fuel pool will be examined. This examination will provide detailed information on the effects of flow variations and hardness differences on latch tube wear. It will also aid in the diagnosis of the loose ball problem discovered at Oconee-2, thought to be a result of the staking process used in the manufacture of Oconee-2 and TMI-1 BPRA couplings.

Scope and Method

There are currently 96 fuel assemblies in the TMI-1 spent fuel pool (SFP) which contained either a BPRA or an orifice rod assembly (ORA) for one or more cycles. These assemblies are listed in Table I. The actual number to be examined will depend on the results of other examinations, but it is expected that a representative sampling from each of the following "classes" will be done:

Batch	Component Cyc 1/Cyc 2/Cyc 3	Purpose
2	BPRA/None/—	Characterize cycle 1 BPRA wear
2	BPRA/CRA/—	
3	BPRA/CRA/CRA	
3	CRA/ORA/CRA	Isolate cycle 2 ORA wear
3	None/CRA/ORA	Isolate cycle 3 ORA wear
2	BPRA/ORA/—	Characterize combined BPRA & CRA wear
3	None/ORA/ORA	Characterize 2-cycle ORA wear
3	ORA/ORA/ORA	Characterize 3-cycle ORA wear

A minimum examination scope is indicated by the asterisks in Table I. Depending on the degree of wear seen, additional assemblies may be done to increase the data base.

In addition to the fuel assembly latch tube examination, approximately 20 BPRA couplings will be examined. These couplings had the same manufacturing (especially ball staking) characteristics as those used in the Oconee-2 core. Examination of Oconee-2 couplings showed several empty ball holes. The purpose of this examination is to obtain data on the similar TMI-1 couplings under different operating conditions. A list of 20 BPRA's to be examined is given in Table II. Depending on the results, others may be added to the list. A map of assemblies and components currently in the TMI-1 spent fuel pool is shown in Figure 1.

W 39201

Latch tubes will be examined using a specially-constructed underwater TV system. In most cases the examination will simply be a matter of positioning the system (from the SFP bridge) over the latch tube and rotating to obtain a 360-degree view (and videotaping). Each operation should take about five or ten minutes. If the assembly contains a component, it must be removed (using the SFP mast) prior to the examination and returned afterwards. BPRA examination will be done by raising the component a few inches with a manual tool (provided) and viewing the ball areas with underwater TV.

The TMI-1 SFP examination is currently scheduled to be done following the TMI-2 core inspection of latch tubes. The latter is tentatively scheduled to begin about June 18. If an opportunity for the SFP exam presents itself during the TMI-2 inspection, it will be done at that time.

W 39262

TABLE I - TMI-1 FUEL ASSEMBLIES IN SPENT FUEL POOL

Component (Cyc 1 - Cyc 2 - Cyc 3)	Fuel Assemblies	No.
BPRA-None-Out	1B06, 19, 20, 45, 52*, 59*, 60*, 61*	8
BPRA-CRA-Out	1B04, 05, 07, 08, 13, 14, 15*, 16*, 17, 18*, 21, 22, 34, 37, 38, 39, 40, 41*, 42*, 44*, 46*, 47, 49, 53*, 55, 56, 57, 58	28
BPRA-CRA-CRA	1C26*, 27*, 28, 31, 32, 33*, 34*, 47	8
CRA-ORA-CRA	1C01, 02*, 03*, 04*, 05, 12, 13*, 50*, 54*, 56*, 57*, 60	12
None-CRA-ORA	1C19*, 23*, 25*, 36*, 42*, 52*, 53*, 55*	8
BPRA-ORA-Out	1B01*, 02*, 03*, 09*, 10*, 11*, 12*, 23*, 24*, 26*, 28*, 32*, 48*, 50*, 51*, 54*	16
None-ORA-ORA	1C06*, 07*, 08*, 11*, 14*, 16*, 17*, 38*, 43*, 51*, 58*, 59*	12
ORA-ORA-ORA	1C10*, 21*, 22*, 41*	4
		96

*Minimum inspection scope (65 assemblies).

W 39203

TABLE II - TMI-1 BPRA's TO BE EXAMINED

BPRA No.	FA Cycle 1	Core Location (Cycle 1)	
·B12	1B56	B7	20D
.B40	1C31	C4	21E
.B45	1B29	C10	18H
.B42	1B25	L3	26F
·B51	1C32	D13	17K
B64	1B37	E4	19K
·B29	1B40	D7	26E
·B14	1B16	G2	17D
.B13	1B61	G6	22K
·B18	1B60	G10	19C
·B19	1B15	G14	~~[illegible]~~ 14K
·B10	1B52	K6	22H
·B03	1B59	L9	20G
.B56	1B23	L11	19D
B71	1B27	L13	17H
·B35	1C34	N3	12C
·B58	1B22	N5	13L
·B28	1C28	N13	18K
.B27	1B33	O10	24E
.B06	1B18	P7	25D

W 39204

FIGURE 2

TMI-1 SPENT FUEL POOL A

INVENTORY 7/7/77 (JTM CON)

32	1C18 047	1C70	1C27	1C47		201A		
31	1C11	1C69	1C01	1C40	1C49	1C24	1C33	
30	1C23	1C12	1C33	1C23	1C57	1C13	1C29	
29	1C52	1C56	1C21	1C08	1C21	1C54	1C39	1C51
28	1C50	1C16	1C59	1C07	1C43		1C36	1C10 1C26 510
27	1C05	1C14	1C58 C03	1C06	1C04	1C17	1C53	1C44
26	1C35	1A51 B39	1A52 B29	1A53 B42	1A31	1A54	1A56 B11	1C42
25	1A42	1A43 B06	1A44 B24	1A45 B61	1A46 B33	1A47	1A48 B47	1C22 C04 511
24	1C41 C38	1A49	1A36 B27	1A37	1A38	1A39 B09	1A55	1A41
23	1A26 B02	1A27 B54	1A28 B46	1A29 B21	1A30 B01			1A33 B22
22	1C15	1A19 B43	1A16 C08	1C09	1C48 C18	1A23 [illegible]	1A24 B13	1A25 B08
21	1A10 B67	1C03	1A12 B40	1C31	1C26	1A15 B33	1C46	1A17
20		1A03 B12	1A04	1A05	1A06 B03	1A07 B07		1C45
19	1B17 [illegible]	1B15 B51	1B13 B50	1B19 B17	1B21 B16	1B09 B24	1B02 B64	1A01 B15
18	1B15	1B56	1B12 B43	1B36 B04	1B16 B52	1B03 B45	1B49 [illegible]	1B28 B32
17	1B46	1B53 B14	1B34	1B38 B33	1B64 B25	1B24 B01	1B61 B51	1B47
16	1B40 B60	1B05	1B22	1C32	1B55	1B10 C47	1B58 B55	1B35
15	1B39	1C27	1B23 B57	1B04 B41	1B01 C64	1B37	1B44 B62	1B07
14	1B08 B05	1A40	1C34	1A20	1B60 022	1B41	1B11 B19	1C66
13	1B48 B35	1B19	1B45	1B42 C07	1B52		1B50 B36	1B32 B58
12	1B57 021		1C25	1C[illegible]	1B57	1B20 C64	1B51 C59	FA[illegible](*) B30
11	1C30	1C02						
10								DFA DER 04
9								
8								
7								
6								
5								
4	FA[illegible](*) C103	FA[illegible](*)						
3								
2	RSPA	RSPA B23	RSPA [illegible]	RSPA B64	RSPA B24	RSPA C49		
1	RSPA B45	RSPA C21	RSPA B17	RSPA [illegible]	RSPA [illegible]	RSPA [illegible]		

NOTE: 1A10, 1A12, 1A33 & 1A19 ARE SUPPOSED TO BE IN CORE (CYCLE 4). 1A21 & 1A22 ARE NOT SHOWN IN POOL OR CORE MAP (CYCLE 4). TELECOPY (FROM WHICH THIS COPY WAS MADE) WAS NOT TOO CLEAR. SHOULD BE CROSS-CHECKED.

W 39205

34W BR

Babcock & Wilcox

Power Generation Group

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Telephone: (804) 384-5111

June 21, 1978

SOM-II-161
REM-I-359

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[redacted]
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Subject: Limits & Precautions and Setpoints - 1. DH Cooler Outlet Lo Temp Alarm, 2. Plant Cooldown

Gentlemen:

Section 3 (Page 19) Item 2 of Plant Setpoints lists a low alarm of 180°F for the decay heat injection flow. This alarm was intended to prevent the decay heat cooling system from exceeding a 75°F ΔT between the Reactor Coolant System and the Decay Heat System's return to the Reactor Coolant System at the time the reactor coolant pumps were stopped. Since this limit is only required at one point in time (i.e., stopping the reactor coolant pumps), the alarm would become a "nuisance alarm." Therefore, it is recommended that this alarm be deleted from the setpoint section and be made an administrative limit in Limits & Precautions as follows:

1101-02 - Plant Setpoints:

Section 3, Page 19, Line 2: Delete annunciator low alarm of 180°F.

W 39206

1101-01 - Limits & Precautions:

Section 3, Page 55: Add Item 12 as follows:

> The Decay Heat System return temperature to the Reactor Coolant System should be limited to a maximum of 75°F below the reactor coolant temperature at the time that all reactor coolant pumps are stopped. See Plant Technical Specification Figure 3.4-3.

In reference to subject (2), Section 1.3, Item 10 (Page 21) limits the ΔT between the "dry" OTSG shell and that loop Reactor Coolant System temperature to 100°F. It is recommended that the wording be changed to expand this limit for all non-normal plant conditions rather than just the "dry" OTSG condition. The definition of "normal" is two OTSG's operational. Therefore, the following change to Limits and Precautions is recommended:

Section 1.3, Page 21, Item 10: Delete the existing wording and replace it with the following:

> When cooling down the Reactor Coolant System under other than normal conditions (normal - two OTSG's operating), the maximum allowable temperature difference between the average OTSG shell and its associated reactor coolant loop average temperature is <u>100°F</u>.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E G Schuedel
for
L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
J. L. Seelinger
J. P. O'Hanlon
J. D. Phinney
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G. A. Kunder
GPU PR 5340

W 39207

Babcock & Wilcox

Power Generation Group

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June 26, 1978

SOM-II-162

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Subject: Reactor Vessel Head O-Ring (Stainless Steel) Qualification

Gentlemen:

At your request, an effort was made to provide "traceability" for a set of reactor vessel head stainless steel o-rings. The B&W QA section at Mt. Vernon traced the serial numbers S-3 and L-3 to the listed internal purchase order #635175-MK. This purchase order contains the B&W Engineering specifications and QA requirements.

Attached is a letter that certifies o-rings, S-3 and L-3, that conform to the listed B&W specifications and requirements.

No "traceability" between the purchase order #635175-MK and a corresponding purchase order from either Met-Ed, GPU, or JCP&L could be found.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39208

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
L. L. Lawyer

June 26, 1978

SUBJECT: Certification of O-Rings S-3 and L-3

This letter is to certify that o-rings S-3 and L-3 were purchased for B&W purchase order #635175-MK. These o-rings meet all purchase order requirements as follows:

1. Material shall be in accordance with SA-213 Type 304 Stainless Steel and Section III 1965 Edition, with Addenda through Summer 1967, ASME Code as modified by this order.

2. Parts shall be silver plated 0.004" - 0.006" thick. No plating of the rings should be performed until the results of all tests have proved satisfactory.

3. Ultrasonic inspection of the tubing used to fabricate the o-rings shall be in accordance with Paragraphs N-324.3 of ASME Code Section III, 1965 Edition and Addenda through Summer 1967, Article 3 with the following modifications.

   A. U.T. shall be to a 3% notch sensitivity or a notch depth of 0.004", whichever is greater.

   B. Shear wave shall be performed in two opposite circumferential directions to detect discontinuities in a longitudinal direction relative to the axis of the pipe. Also, shear wave shall be performed in a longitudinal direction to detect transverse discontinuities relative to the axis of the pipe. Discontinuities exceeding the 3% notch or .004" whichever is greater shall be rejected.

   C. Vendor's ultrasonic test report shall indicate the test frequencies employed, type of instrument, couplant and transducer employed.

   D. Before wrapping o-rings for shipment, vendors must contact B&W so B&W can perform a UT inspection of the finished o-ring. Vendors must guarantee the o-ring will pass a UT inspection based on the same acceptance standards used by the mill on the original tubing.

4. Cobalt content shall be .20 percent maximum. Vendor should aim for 0.10 percent maximum cobalt.

W 39210

5. In forming the o-ring, one weld is preferred but two welds are permissible if required by tubing stock lengths. If o-ring is made from two (2) sections of tubing, the outside diameters must match within .002".

6. All welds shall be polished and ground flush with OD of tubing. Reduction in the tube diameter at weld joint shall not exceed .004" in order to ensure smooth blend at joint. Each weld shall be inspected for thickness. The weld thickness shall not exceed 25% of the tube wall thickness. The weld thickness shall be measured by a videgage or other acceptable procedure which must be approved by B&W, Quality Control. The minimum and maximum tube wall and weld thickness shall be recorded and reported to B&W.

7. Radiographic inspection shall be performed in accordance with ASME Code Section III except as modified by this order.

8. The penetrameter dimensions (size) shall be in accordance with Section III as modified. The thickness shall be .005", the hole diameters shall be .010", .020", and .040". The sensitivity level shall demonstrate the .010" hole.

9. Two radiographs 90° apart (double wall) shall be taken of each weld. The acceptance standards shall be as follows:

   A. Any porosity, visible on the radiograph, shall be considered as an unacceptable defect.

   B. Defects such as cracks, incomplete penetration, incomplete fusion or other linear defects are not acceptable.

10. Liquid penetrant examination shall be performed in accordance with B&W Quality Control specification 04-402-02 prior to plating and cutting of any holes.

W 39211

Dr... EK

Babcock & Wilcox

Power Generation Group

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June 27, 1978

SOM-II-163
REM-I-360

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Mr. J. P. O'Hanlon
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Subject: Evaluation of Control Rod Drive Cable Assemblies

Gentlemen:

As an on-going service, of B&W Nuclear Service, to improve the NSSS availability, B&W has recently investigated the loss of plant availability due to failures of control rod drive mechanism cable assemblies used between the transfer canal connectors and the control rod drive mechanism connectors.

Attached is a copy of the results and recommendations derived from a study of the operating history and problems associated with control rod drive mechanism cable assemblies. This study has focused on the modes of failure of the cables and cable connectors and on the causes for the failures.

W 39212

The recommendations offered in this report are intended to improve reliability and extend the useful life of the cable assemblies. Our vendor, Bendix, has indicated a willingness to provide training to B&W and utility personnel in the use and maintenance of the cable assemblies, if there is need and interest.

Please advise if there is interest within your organization for training or if you have questions on the content of the report.

As these plants are aging, the availability of the spare parts is going to be scarce. With this general consideration in mind, Diamond Power Specialty Corporation has released Service Tip No. 1-78 relating to the availability of replacement parts for the General Electric AK-2A-25 circuit breakers used as control rod drive AC breakers.

Five copies of the service tip are attached for distribution amongst your engineering and operating staff.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

cc: L. R. Pletke
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W 39213

CONTROL ROD DRIVE MECHANISM CABLE ASSEMBLIES

B&W has investigated the loss of plant availability due to failures of CRDM cable assemblies used between the transfer canal connectors and the CRDM connectors. The results of this investigation of both known and assumed modes of failure are presented below.

The known modes of failure are insulation breakdown and low resistance between conductors. A third, but assumed, mode of failure is excessive resistance of the conductors.

The investigation revealed that insulation breakdown is due primarily to excessive heat. There are several causes of excessive heat including 1) Loss of cooling to the service structure and 2) high resistance in the connections caused by dirty, bent, or corroded contacts. Another cause of insulation breakdown is physical damage, usually resulting from improper handling or installation.

Low resistance between conductors can be a result of insulation breakdown, however, a separate mode of failure is moisture between conductors at the connections. Any improper mating of the plug and receptacle can allow moisture to enter and provide a potential cause for failure. Dirty, bent, or corroded contacts can result in misleading torque readings during connection makeup. Improper, or the lack of, lubricant can also give misleading torque readings. Damaged or dirty threads on the connectors can prevent proper seating. Connector O-rings that are damaged or that have lost their resilience also can prevent proper seating.

High resistance of the conductors can result in excessive heat and insulation breakdown or can lead to an open conductor. High resistance can be the result of improper handling techniques such as: pulling the cable by the connector, hanging additional weight on connected cables, excessive flexing of the cable, and treading on the cables.

B&W supplied cable assemblies are designed for the PWR containment environment and are fabricated to the current state of the art. It is evident from this investigation that to maximize the useful life of the cable assemblies, they must be carefully handled and protected to prevent damage. Specifically, to maximize the useful life of the cable assemblies, B&W recommends the following:

1) The connector O-rings should be replaced each time the connection is broken at the stators and periodically replaced at the bulkhead based on the service history.

2) Use the recommended lubricant and torque the connection to the value recommended by the cable assembly manufacturer (See Attachment 1).

3) Unmated connectors should be protected and maintained dry by the use of protective caps.

4) Do not allow personnel to walk on the cables.

5) If a liquid spill occurs, assure that the connectors are thoroughly dried.

6) Prior to making connections, inspect the connectors for dirt, damage, or corrosion. Correct any anomalies.

7) Develop a quality assurance/quality control program for maintenance and operation.

TORQUE VALUES

ELECTRICAL CONNECTORS

Bendix Electrical Components Division

Sidney, N.Y. 13838

November 1972 L-725-2

1. INTRODUCTION.

2. This publication contains minimum and maximum torque values recommended by The Bendix Corporation, Electrical Components Division, Sidney, New York 13838.

3. Maximum torque values are primarily governed by strength of material used to fabricate threaded components. Maximum values listed in this publication were computed to protect threads and other bearing surfaces which could be damaged by excessive torque.

4. Minimum torque values were computed to assure proper mating of connectors and/or accessories (providing threaded areas are properly lubricated). Connectors procured less lubrication on the thread are controlled by special part numbers. For these connectors, the user is responsible for proper thread lubrication. Lubricants should be compatible with the connector and its application. Two such applications would be for use in a N_2O_4 (nitrogen tetroxide) atmosphere and a LOX atmosphere. Lubricants which are N_2O_4 resistant and may be used in this atmosphere are Molykote X-15 and Fluorocarbon Lubricant 95-1. Molykote X-15 is available from the Alpha Molykote Corporation, Stamford, Connecticut 06904, and is recommended for use with the JTN series connector. Fluorocarbon Lubricant 95-1 is available from the Dixon Corporation, Bristol, Rhode Island 02809, and is recommended for use with AN/MS type alumilite finish connectors. Lubricants which are LOX compatibile and may be used in a LOX atmosphere are Molykote X-15 and KELF-90, available from the Minnesota Mining & Manufacturing Co., St. Paul, Minnesota 55101.

5. Metal to metal seating of mated parts should occur where a flat gasket is used for an end seal. In some instances, when a cable accessory or MS3057B type clamp is used with maximum diameter cable, a metal to metal seating may not occur on the initial tightening. A second tightening is therefore necessary after cable has been allowed to cold flow (approximately 12 hours).

6. Torque values are listed by thread sizes. We recommend the Bendix 3C Connector Catalog for mating and accessory thread diameters for each connector shell size.

NOTE: See torque value notes following paragraph 10 for accessories having three threads or less.

7. TORQUE VALUES. W 39216

8. The torque values listed in Table I are for UN (Handbook H-28) threads. These values apply to connector's coupling nut, jam nut, cable clamp, back shell or accessories.

9. Column "A" of Table I lists the minimum and maximum values for threads on aluminum die cast parts such as MS or MS Modified connectors.

TORQUE VALUES FOR

Thread Size	COLUMN "A" Threads On Die Cast Aluminum Parts				COLUMN "B" Threads on Extruded Or Machined Aluminum And Steel Parts			
	Inch Lbs.		Foot Lbs.		Inch Lbs.		Foot Lbs.	
	Min.	Max.	Min.	Max.	Min.	Max.	Min.	Max.
0.3125	20	26			70	75		
0.3750	20	26			70	75		
0.4375	20	26			70	75		
0.5000	20	26			70	75		
0.5625	26	32			80	85		
0.6250	26	32			90	95		
0.6875	30	36			100	110	8	9
0.7500	34	40			110	120	9	10
0.8125	40	46			120	130	10	11
0.8750	46	50			140	150	11	12
0.9375	50	55			150	160	12	13
1.0000	55	60			160	170	13	14
1.0625	60	65			190	200	16	17
1.1250	70	75			210	230	18	19
1.1875	75	80			240	260	20	21
1.2500	80	85			260	280	21	23
1.3125	85	90			280	300	24	25
1.375	90	95			300	320	24	25
1.4375	100	110	8	9	300	325	26	28
1.5000	100	110	8	9	325	350	28	30
1.625	110	120	9	10	350	375	28	30
1.750	120	130	10	11	375	400	30	32
1.875	140	150	11	12	400	425	32	34
2.0000	150	160	12	13	425	450	36	38
2.0625	160	170	13	14	450	475	38	40
2.1250	170	180	14	15 ①	475	500	40	42
2.2500	170	180	14	15	500	525	42	44
2.3125	180	190	15	16	525	550	44	46
2.375	190	200	16	17 ②	550	575	46	48
2.500	200	210	17	18	575	600	48	50
2.525	210	220	18	19 ③	600	650	50	55
2.750	220	230	18	19	650	700	55	60
2.875	230	240	19	20	700	750	55	60
3.000	240	250	20	21	750	800	60	65

10. Column "B" of Table I lists the minimum and maximum values for threads on extruded or machined aluminum and steel parts. The torque value for hex mounting nuts on all jam nut receptacles is listed in column "A" of Table I. The hex nuts are machined but the torque value is reduced due to the limited amount of threads.

NOTE: Torque value for all sizes of accessories having three threads, or less and involving a modified major/minor diameter (crest removed) is 30-35 in. lbs. This applies to many PT, JT, and LJT accessories.

NOTE: Torque values for accessories having three threads or less, but not having a modified major/minor diameter (crest removed) are as follows:
Shell size 8 through 19; 50 ± 5 in. lbs.
Shell size 20 through 25; 100 ± 10 in. lbs.

W 39217

11. CONNECTORS WITH DOUBLE STUB THREADS (MIL-S-23747). Minimum torque values in Table II will assure main joint sealing for connectors, such as QWL and QWLD series, incorporating double stub threads (MIL-S-23747).

① PI CABLE BULKHEAD JAM NUT TORQUE
② POWER CABLE BULKHEAD JAM NUT TORQUE
③ PRIMARY T.C. CABLE BULKHEAD JAM NUT TORQUE

TABLE II

TORQUE VALUES FOR DOUBLE STUB THREADS MAIN JOINT SEALING*				
Thread Size	Inch Lbs.		Foot Lbs.	
	Min.	Max.	Min.	Max.
0.875	150	170	12	14
1.000	170	190	14	16
1.125	230	260	19	21
1.250	280	300	23	25
1.375	325	350	26	28
1.500	350	400	30	32
1.750	400	425	32	36 ⑤
2.000	425	450	38	42 ⑥
2.250	500	600	44	48 ⑥
2.500	600	700	50	55 ⑦
2.750	700	800	60	65
3.000	800	850	65	70

*Table of values applies except for those series mentioned in NOTE below.

NOTE: Die cast single key of the AN double stub thread (65- & 66- series connectors) requires using column "A" of Table I for torque values. The small single key of the PC double stub thread connectors requires using column "B" of Table I for torque values.

12. CONNECTORS WITH BAYONET COUPLING. All Bendix bayonet coupling connectors (Pygmy) including those with special coaxial arrangements will meet the coupling and uncoupling forces of specification MIL-C-26482. Before mating, receptacles must be suitably mounted. Coupling nuts on plugs must be properly lubricated.

Maximum engagement and minimum disengagement forces applied to the coupling ring shall be within the limits specified in Table III.

TABLE III

TORQUE VALUES FOR BAYONET COUPLING		
Connector Size	Inch Lbs. Max. Engagement	Inch Lbs. Min. Engagement
8	8	1
10	12	1
12	16	2
14	20	4
16	24	4
18	28	4
20	32	6
22	36	7
24	44	7

④ DRIVE T.C. CONNECTION TORQUE
⑤ PI CONNECTION TORQUE
⑥ POWER CONNECTION TORQUE
⑦ PRIMARY T.C. CONNECTION TORQUE

W 39218

CRDC SERVICE TIP NO. 1-78

RENEWAL PARTS FOR AK-2A-25 CIRCUIT BREAKERS

General Electric has advised Diamond Power that complete breakers as renewal items will be obtainable through 1986. All parts making up these breakers will also be available through 1986. In addition, all recommended spare parts will be available through 1996. These recommended spare parts are shown in General Electric bulletin GEF-4149G, pages 3 and 4.

In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39219

NUCLEAR EQUIPMENT SERVICE TIPS BULLETIN

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

CRDC SERVICE TIP NO. 1-78

RENEWAL PARTS FOR AK-2A-25 CIRCUIT BREAKERS

General Electric has advised Diamond Power that complete breakers as renewal items will be obtainable through 1986. All parts making up these breakers will also be available through 1986. In addition, all recommended spare parts will be available through 1996. These recommended spare parts are shown in General Electric bulletin GEF-4149G, pages 3 and 4.

In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39220

NUCLEAR EQUIPMENT SERVICE TIPS BULLETIN

DIAMOND POWER SPECIALTY CORP.
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CRDC SERVICE TIP NO. 1-78

RENEWAL PARTS FOR AK-2A-25 CIRCUIT BREAKERS

General Electric has advised Diamond Power that complete breakers as renewal items will be obtainable through 1986. All parts making up these breakers will also be available through 1986. In addition, all recommended spare parts will be available through 1996. These recommended spare parts are shown in General Electric bulletin GEF-4149G, pages 3 and 4.

In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39221

CRDC SERVICE TIP NO. 1-78

RENEWAL PARTS FOR AK-2A-25 CIRCUIT BREAKERS

General Electric has advised Diamond Power that complete breakers as renewal items will be obtainable through 1986. All parts making up these breakers will also be available through 1986. In addition, all recommended spare parts will be available through 1996. These recommended spare parts are shown in General Electric bulletin GEF-4149G, pages 3 and 4.

In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39222

NUCLEAR EQUIPMENT **SERVICE TIPS** BULLETIN

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

CRDC SERVICE TIP NO. 1-78

RENEWAL PARTS FOR AK-2A-25 CIRCUIT BREAKERS

General Electric has advised Diamond Power that complete breakers as renewal items will be obtainable through 1986. All parts making up these breakers will also be available through 1986. In addition, all recommended spare parts will be available through 1996. These recommended spare parts are shown in General Electric bulletin GEF-4149G, pages 3 and 4.

In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39223

Babcock & Wilcox

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June 26, 1978

SOM-II-164

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Mr. G. P. Miller
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Middletown, PA 17057

Subject: Beff Values for Physics Testing

Gentlemen:

The reactivity program used by the reactimeter is referenced to an effective delayed neutron fraction which is a function of core burnup. To account for burnup effects, the system tape has been expanded to include revised values every 10 EFPD for the first 50 EFPD. Thus depending on the particular core burnup (i.e., 0-10, 10-20, 20-30, etc.), the person operating the reactimeter should select the applicable program file. To ensure that the correct value of Beff is employed during a physics measurement at power, the appropriate test procedures (i.e., TP 800/5, TP 800/20, and TP 800/31) should reflect this modification.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39224

LCR/JMP/bay

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J. L. Seelinger
G. K. Wandling
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June 29, 1978

SOM-II-165

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Subject: Plant Limits & Precautions and Plant Setpoints

Gentlemen:

1. Several times during the Unit II startup program, problems were encountered in venting control rod drive mechanisms and eliminating leakage past the control rod drive mechanism vent valves. Paragraph 1.6-16 of Plant Limits and Precautions states that control rod drive mechanisms should not be vented unless the Reactor Coolant System pressure is less than 300 psig and Reactor Coolant System temperature is less than 200°F. The limit on pressure is based on removal of a sufficient amount of gas from solution.

A method for stopping vent valve leakage is to revent the control rod drive mechanism. It is acceptable to vent the control rod drive mechanism up to the Reactor Coolant System pressure limit established in the control rod drive mechanism instruction manual provided that the venting is not for removal of dissolved gases and Reactor Coolant System temperature is less than 200°F.

W 39225

It is suggested to add the following to Item 16, Page 33 of Limits and Precautions:

> NOTE: The 300 psig applies to venting for removal of gases. It is acceptable to vent the control rod drive mechanism up to the pressure limit established in the control rod drive mechanism manual to eliminate vent valve leakage.

2. Item 11, Section 3, Page 55 of Limits and Precautions states:

> "Route the reactor coolant pump seal return to waste disposal prior to Decay Heat/Reactor Coolant Pump concurrent operation.
>
> NOTE: This will maintain No. 1 seal delta P."

This precaution is only applicable to Westinghouse reactor coolant pumps. Since Unit II has Bingham pumps, this precaution can be deleted for Unit II.

3. Plant Setpoints, Page 15, instrument MU7 "seal inlet line flow" low flow alarm is set at 6 gpm. This setpoint also causes the individual seal return valves (MU-V33A,B,C.D) to close. Plant Limits and Precautions, Page 9, Item 19, recommends reducing seal injection flow to 6 gpm during certain conditions. In order to avoid spurious alarms and isolation of the seal return flow, which destages the three-stage seal, the low flow setpoint should be changed from 6 gpm to 5 gpm. The minimum allowable seal injection flow limit still remains at 6 gpm.

Page 15 of Plant Setpoints instrument MU7:

> Change low setpoint from 6 gpm to 5 gpm.

4. Plant Limits and Precautions, Page 25, Item 20 - Based on past experienced, this item is incorrect and should be changed to read:

> "The pump should not be uncoupled from the motor until the Reactor Coolant System pressure is less than or equal to ________ psig. Do not start the injection system when the pump is uncoupled. 40 psig"

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

[signature] for

L. C. Rogers
Site Operations Manager

W 39226

LCR/EGS/bay

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W 38227

DAOOD

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June 29, 1978

SOM-II-166
REM-I-362

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Subject: Recommended Actions for Suspected Loose Parts in the Reactor Coolant System

Gentlemen:

The attached contains standard recommendations to the B&W Nuclear Steam Supply System operating utility. It consists of immediate actions to be taken should the loose parts monitor indicate a loose part in the Reactor Coolant System.

These recommendations apply regardless of the supplier of the loose parts monitors.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

W 39228

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J. D. Phinney
G. A. Kunder

RECOMMENDED ACTIONS FOR A SUSPECTED LOOSE PART

Due to the serious consequences of potential material damage in the event of a loose part inside the Reactor Coolant System, it is strongly recommended that a plan of action be developed by each utility to cover this situation. Quick and decisive action by the operations personnel is required in the event of a suspected loose part to place the unit into a safe condition to avoid the potential of several months of repair.

Loose Parts Monitoring (LPM) Systems supplied by B&W are designed and calibrated to detect unusual noises above the normal system background. The Loose Parts Monitoring System is set to alarm for detected noises having the characteristics of metal-to-metal impacts. Regardless of the supplier of the Loose Parts Monitoring System, B&W recommends that immediate action be taken to determine the validity of the alarm. The alarm should be assumed to be the result of a loose part until proven to be otherwise. If the alarm cannot be invalidated, the plant should be shutdown, cooled down, and placed into the decay heat removal mode to minimize the damage that could be sustained due to the presence of a loose part(s).

The types of questions that must be answered to determine the validity of the alarm include but are not limited to:

1. What were the plant operations immediately prior to the Loose Parts Monitoring alarm? Did anything change abruptly or in a manner that could have caused the noise or that could have dislodged a loose part?

2. Was the plant undergoing a significant evolution at or near the time when the alarm occurred (reactor trip, turbine trip, feedwater flow transient, etc.)? Could the evolution attribute to the noise or cause a loose part to become dislodged?

3. Do other plant instruments indicate abnormal readings that may be related to the noise?

4. Are other Loose Parts Monitoring channels alarming? Do the relative magnitudes of the signals from the various Loose Parts Monitoring channels indicate the location of the noise?

If the answers to the above questions confirm a loose part or fail to promote justification to ignore the alarm, the plant administrative procedures should call for (a) notification to key plant personnel of the presence of a potential loose part and (b) the timely shutdown of the plant. The manner in which the reactor is shutdown (i.e., orderly, reactor trip, reactor and reactor coolant pump trip, etc.) should be dictated by the potential consequences of the particular alarm situation.

W 39230

RECOMMENDED ACTIONS FOR A SUSPECTED LOOSE PART

The preservation of plant integrity should take precedence over data acquisition in all cases; however, during the evaluation period, it is desirable that the following data be taken to provide a historical record of the alarm for action, justification, and later comparisons and analyses:

1. Log all alarm conditions, high or low, on an alarm record sheet (see Attachment 1).

2. Reset the alarm and log the tape recorder digital counter.

3. If required, adjust the amplifier gain for the maximum output without overload (adjust so average signal is 20-40 percent full scale). Tag the new gain setting and tape digital counter.

4. For a low alarm condition, defeat the low alarm option of the specific alarmed module.

5. If any individual high alarm occurs within 30 minutes of a previous high alarm, the shift supervisor and technical staff engineer should be notified.

6. Upon notification of the Loose Parts Monitoring System alarms, the shift supervisor or the technical staff engineer should review the Loose Parts Monitoring System alarm record. The locations of the alarmed sensors should be cross-referenced with the plant conditions at the time of the alarm.

7. Review the automatically recorded tapes and determine the peak-to-peak "G" levels of the recorded impacts. The operability and calibration of the tape auto start system should also be periodically checked.

8. Use the audio module, oscilloscope, or spectrum analyzer to characterize the signals from the alarmed sensors. The points of interest should include metal-to-metal impact noises, the delay time matrix if more than one sensor indicates the noise, and the amplitude of the impacts (i.e., in "G's").

9. Evaluate the cause of the alarm:

   a. Electrical: Most of the electrical noises are found to be periodic in nature and usually have individual wave forms with spike shapes and no amplitude decay.

   b. Mechanical, and if so, what is the probability that it was caused by normal plant operation?

W 39231

RECOMMENDED ACTIONS FOR A SUSPECTED LOOSE PART

c. Loose Part: Take immediate action to preclude further damage and contact B&W.

10. Log the sensor gain setting, if changed, on the alarm record sheet to allow accurate correlation with the recorded tape.

W 39232

ATTACHMENT I

EXAMPLE

LOOSE PARTS MONITORING SYSTEM ALARM RECORD SHEET

TIME OF ALARM	CHANNELS ALARMED	REMARKS	RECORDED BY
MONTH/DAY/HOUR/MINUTE	FIRST OUT/OTHERS ALARMED	OPERATOR ACTIONS, SENSOR GAIN & TAPE DIGITAL COUNTER	SIGNATURE

W 39233

Dřw DK

Babcock & Wilcox

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June 30, 1978

SOM-II-167

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Subject: RCP-2A Lower Bearing Housing Insulation

Gentlemen:

While performing the upper bearing insulation checks during reactor coolant pumps casing gasket replacement, a grounded condition was discovered on the 2A lower bearing. Allis-Chalmers recommended that this condition be repaired. They claimed that the work could be done with the motor in place by unbolting the lower bearing housing and jacking it up. The insulating strips could then be removed and replaced with new material.

The anticipated vertical clearances required to do the work, as outlined above, are not present. The insulating strips cannot be removed. Allis recommends that the best way to gain additional vertical clearance is to release the bearings and anti-rotation device at the top of the motor and jack the shaft up to obtain the required clearances.

This additional work presents several problems which must be weighed against the benefit of having an insulated lower bearing.

W 39234

1. Since all of the bearings will be disconnected, the motor could lie against the stator, which could damage both.

2. Again, since all of the bearings will be disconnected, realignment of those bearings will be difficult and not obtaining the original bearing locations is a virtual certainty.

3. In general, any operation involving major disassembly of equipment opens the door for not being able to reassemble the motor to its original factory-shipped condition.

The purpose of the lower bearing insulation is to provide a boundary for testing the upper bearing insulation. The lower bearing insulation is grounded out by a ground strap during all operating conditions. This boundary can also be created by dropping the lower oil pan, lowering the bearings and insulating the air seal. The latter method is performed on Unit I motors which do not have insulated lower bearings. This method will cost about one 10 hour day of additional work for disassembly and likewise for reassembly each time the upper bearing housing insulation is checked (once every two years?).

Based on a verbal discussion with Jim Seelinger, Met-Ed has determined that this further disassembly is not warranted, and the motor should be reassembled. Please be aware that this motor, RC-P-2A, will henceforward require a Unit I style upper bearing housing insulation check for which procedures have been prepared.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/WDC/bay

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DMS
G+W OK

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July 18, 1978

SOM-II-169
REM-I-363

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Subject: Crane Brakes

Gentlemen:

The purpose of this letter is to forward information, based on our experience at other sites, concerning the possible failure of crane brakes due to loss of power while lifting heavy loads such as the closure head.

For your guidance, enclosed herewith is a report concerning this problem.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39236

LCR/WHA/bay

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FAILURE OF CRANE BRAKES

Recently, at a manufacturing shop, a 500 ton bridge crane malfunctioned while in use. The crane, using two 250 ton capacity trolleys, supported a heavy load for transfer from the manufacturing shops to a shipping barge. Both trolleys were attached to the load through lifting slings such that one trolley was carrying 232 tons and the other trolley was carrying 188 tons. While lowering the load, it was observed that the trolley supporting the heavier end of the load was lowering faster than the other trolley. This necessitated a jogging operation in order to keep the load level while it was lowered. About 15 feet above the barge, the heavier end of the load began an uncontrolled descent. Two holding brakes on the crane were applied but failed to stop the descent, and the end of the load hit the barge deck, causing minor damage to the barge and significant damage to the load.

An investigation was conducted immediately into the cause of this incident and revealed the following:

- The two holding brakes on the crane were factory set in accordance with manufacturer's specifications; nevertheless, the brakes were badly damaged by heat during this incident.
- The crane was being operated according to established procedures and by experienced operators.
- The crane had been certified in February, 1977 by an agency accredited by the Department of Labor as required by OSHA. The certification procedure includes a complete inspection of the hoist brakes and a dynamic load test of 110% of rated capacity.
- Under the Preventive Maintenance Program, the crane and its hoist brakes were inspected in March, 1977 and again in June, 1977. Included in the June inspection was a hoist motor locked-rotor test. During these inspections no deficiencies were noted in the brake assemblies or hoist controls.
- A representative of the crane manufacturer examined the crane, brake and controls and found no abnormal conditions other than the burned brake lining.

Load tests were conducted on the crane after the brake linings were replaced to determine if excessive heat build-up had caused the brakes to fail. Test weights of up to 248 tons were used, exceeding the load that either hoist was required to lift when the incident occurred. The heat build-up during these tests was relatively small and well within the allowable range of the lining material.

After additional evaluations and testing, it was concluded that the incident was caused by failure of the dynamic braking system. A test was performed in which the overload relay was manually tripped, thus defeating the dynamic braking system. In this test the holding brakes could not stop the motor with the simulated load and, had the test been run for a few more seconds, the high heat build-up in the brakes would have been duplicated. It was subsequently determined that a defective overload relay had caused the temporary loss of dynamic braking.

A similar type of failure could occur in the event that a power failure occurred during handling of a heavy load. In this situation, the static brakes of the crane would be called upon to stop and hold the load. Therefore, the brakes of this crane and all cranes in the facility were reset to higher torque values, within the manufacturer's allowable range, to assure that the brakes were capable of stopping a heavy load during descent.

In order to prevent incidents of this type from occurring, at customer's or supplier's facilities, this report is issued with a suggestion that heavy cranes, if equipped with dynamic braking, be checked to assure that the cranes are capable of stopping a load in the event of loss of the dynamic braking system.

8408 [illegible]

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July 21, 1978

SOM-II-170
REM-II-364

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Subject: Turbidity Procedure for BAW 1385 Water Chemistry Manual

Gentlemen:

Attached herewith is a new procedure for inclusion in Section 15 of the subject manual. The procedure is number 14.a, and the page numbers should be 15.14a.1 through 15.14a.4.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
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W. F. Pitka

W 39240

14.a MEASURING THE TURBIDITY OF WATER

1. SCOPE

1.1 This procedure is used for the determination of low levels of turbidity in high-purity water samples [0.5 to 25 Jackson Turbidity Units (JTU)].

2. PRINCIPLE

2.1 The turbidity is measured by nephelometric methods and compared to permanent standards and secondary standards prepared from formazine polymer and from diatomaceous earth. The procedure is similar to ASTM D-1889-71, Standard Test Methods for Turbidity of Water.

3. DEFINITIONS

3.1 The term turbidity in these methods is defined in accordance with ASTM definitions D1129, Terms Relating to Water, as follows:

3.1.1 Turbidity - Reduction in transparency of a sample due to the presence of particulate matter.

3.1.2 Jackson Candle Turbidity - An empirical measure of turbidity in special apparatus, based on the measurement of the depth of a column of water sample that is just sufficient to extinguish the image of a burning standard candle observed vertically through the sample.

3.1.3 Nephelometric Turbidity - An empirical measure of turbidity based on a measurement of the light-scattering characteristics (Tyndall effect) of the particulate matter in the sample.

4. PURITY OF REAGENT WATER

4.1 Turbidity-Free Water - Pass reagent water through a 0.45 mμ millipore filter if such filtered water shows a lower turbidity than the deionized water. Discard the first 200 ml collected. The water to be used shall conform to ASTM Specification D 1193, Reagent Water.

W 39241

5. REAGENTS AND SOLUTIONS

5.1 Hydrazine Sulfate, $(NH_2)_2 \cdot H_2SO_4$, Reagent Grade

5.2 Hexamethylene Tetramine, $(CH_2)_6N_4$, Reagent Grade

5.3 Mercuric Chloride, $HgCl_2$, Reagent Grade

5.4 Diatomaceous Earth, 325 mesh. Dicalite Speed Plus, 325 mesh, has been found satisfactory for this purpose.

5.5 Formazine Solution I - Dissolve 1.000 grams hydrazine sulfate in distilled water and dilute to 100 ml in a volumetric flask.

5.6 Formazine Solution II - Dissolve 10.000 grams hexamethylene tetramine in distilled water and dilute to 100 ml in a volumetric flask.

5.7 Formazine Stock Suspension, 400 JTU - In a 100 ml volumetric flask, mix 5.0 ml Formazine I solution with 5.0 ml Formazine II. Allow to stand 24 hours at 25 ± 3°C, then dilute to the mark and mix. The turbidity of this suspension is 400 JTU. Prepare the formazine solutions and suspensions monthly.

5.8 Formazine Stock Suspension, 40 JTU - Dilute 10.00 ml of the 400 JTU formazine stock suspension to 100 ml with turbidity-free water; prepare weekly. The turbidity of this suspension is defined as 40 JTU.

5.9 Diatomaceous Stock Suspension - Add 5 grams diatomaceous earth to a 1000 ml volumetric flask, add 100 ml deionized water, swirl to mix, and then dilute to 1000 ml. Invert flask 30 times and let settle for 24 hours. Decant the top layer to another volumetric flask, add 0.8 grams mercuric chloride to preserve the suspension, and use to prepare secondary standards. This solution has a turbidity of approximately 40 JTU.

6. EQUIPMENT

6.1 Coleman Model 9 Nephelometer - Colorimeter or equivalent.

6.2 Coleman certified Nephelo Standard Set 7-720 or equivalent.

6.3 Cuvettes, 19 x 105 mm round, or equivalent for nephelometer used.

6.4 Jackson Candle Turbidimeter.

W 39242

7. STANDARDIZATION

7.1 Determine the turbidity of the 40 JTU formazine suspension and/or the diatomaceous earth suspension on the Jackson candle turbidimeter.

7.2 Dilute these suspensions to the working range of the nephelometer. The upper limit for the Coleman instrument for linear relationship between turbidity and instrument output is 20 JTU. Other instruments may have different linear ranges.

7.3 Place the various Coleman standards in the nephelometer and read the galvanometer deflection. Calculate the equivalent scale deflection factor per nephelo unit by dividing the nephelo value for the standard by the instrument deflection.

7.4 Place the appropriate formazine and diatomaceous earth suspensions in the cuvette and read the instrument deflection. Convert this value to nephelo units by multiplying it by the conversion factor determined in 7.3.

7.5 Plot the nephelo values found for the formazine and diatomaceous earth standards against their nephelo unit values.

8. ANALYSIS PROCEDURE

8.1 Transfer a portion of the sample to a dry 19 x 105 mm cuvette.

8.2 Read the instrument deflection for the sample.

8.3 Convert the deflection value to nephelo units by multiplying by the deflection to nephelo unit factor.

8.4 Convert the nephelo unit to JTU using the standardization curve. Report the value to the nearest 0.1 JTU.

9. CALCULATIONS

$$(CNU)_1 = \frac{(CNU)_2}{(SCD)_2} \times (SCD)_1$$

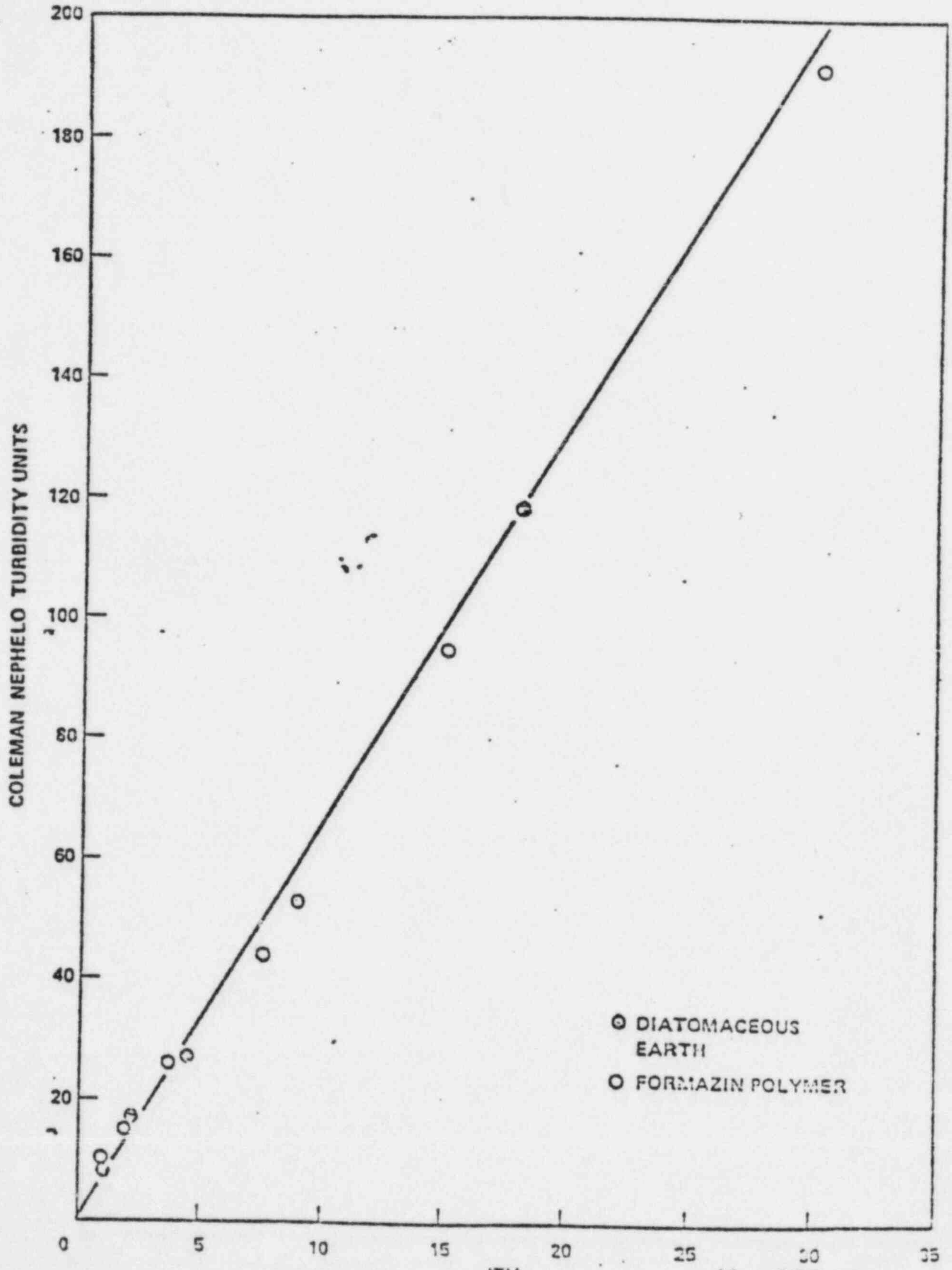
Where:

$(CNU)_1$ = Coleman Nephelo Value for Formazine Standard

$(CNU)_2$ = Coleman Nephelo of Coleman Nephelo Standard

$(SCD)_1$ = Scale Deflection for Formazine Standard

$(SCD)_2$ = Scale Deflection for Coleman Nephelo Standard



W 39244

FIGURE 5.2 RELATIONSHIP OF COLEMAN NEPHELO UNITS TO JTU

0400 0 4

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July 21, 1978

SCM-II-171

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Subject: Examination of Surveillance Specimen Holder Tubes (SSHT)

Gentlemen:

Surveillance specimen capsule(s) must be removed during the first refueling outage. Prior to the removal of the SSHT closure(s) and the capsule(s), it is desired to visually inspect the outside of all SSHT's and closures for anomalies. These inspections may be accomplished by using a television camera lowered through appropriate access holes.

Specifically, the following visual inspections are requested:

1. Confirm that the closures are properly latched by viewing the upper end of the SSHT from a near horizontal position. The closure is properly latched if the uppermost part of the finger latch assembly is approximately centered in the 1/2 inch deep slot on the SSHT.

2. Locate the SSHT viewing slot and observe the Belleville spring configuration. The springs should be approximately centered in the viewing slot if the closure and specimen capsules are properly positioned.

W 39245

3. Visually inspect each SSHT bracket attachment bolt (MK 27-48), drawing number 238780E, lockweld for evidence of cracks.

4. Visually inspect each SSHT bracket attachment bolt locking cup (MK 27-254), drawing number 238780E, for indications of any anomaly.

Any anomalies noted in the above inspections should be brought to the immediate attention of B&W NPGD, Lynchburg. The results of the inspections are requested to be documented and transmitted to B&W NPGD, Lynchburg. Video tapes of the inspections would be one acceptable means of documentation.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/WHA/bay

cc: L. R. Pletke
W. H. Spangler
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W 39246

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SOM-II-172

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Subject: Specimen Capsule "Core Side" Markings

Gentlemen:

The purpose of this letter is to provide information concerning the inscription "Core Side" marked on the specimen capsules for TMI-II.

The capsules were manufactured before the final design of the end fittings was approved. The end fittings orient the capsule with relation to the core, and the word "opposite" was to be added to "Core Side" on the capsules. This was not done in all cases. No functional deficiency will be experienced by this situation.

The addendum to the TMI-II data package will contain revised loading diagrams with changes and corrections indicated thereon.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager W 39247

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Subject: Surveillance Capsule Insertion/Withdrawal Schedule Revision

Gentlemen:

The Surveillance Capsule Insertion/Withdrawal Schedule, which was forwarded by SOM-II-038 dated 8 June 1977, should read,

1. End of 13th Cycle, remove TMI-2E instead of TMI-2F

2. End of 19th Cycle, remove TMI-2F instead of TMI-2E

The capsule identification numbers may be written as TMI-2A or TMI2-A; TMI-2B or TMI2-B; etc., and may be interchanged. The Users' Group capsules may be identified and may be interchanged as TMI-L1 or TMI-2LG1; TMI-L2 or TMI-2LG2.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

W 39249

LCR/WHA/bay

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July 31, 1978

SOM-II-174

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Subject: Fuel Storage Pool Level Detectors

Gentlemen:

At one of the B&W plants, Fisher 2340-24913 Level-trol Displacer Assembly used for fuel storage pool level detector had a rusting problem associated with it. TMI-2 level detectors were inspected for any apparent signs of rusting, but no rusting was identified.

To preclude the possibility of rusting at a later stage, B&W Engineering recommends that the Torque Arm, Displacer Assembly and the Detector Housing of the level detector should have "Plasite 7155NP Primer" and "Plasite 9009 Hi-Resistant Protective Coating" applied. The instructions for application of the "Plasite" are attached for your reference.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39250

LCR/SPM/bay

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1. SCOPE

THE INTENT OF THIS SPECIFICATION IS TO PROVIDE A PAINT PROCEDURE FOR PAINTING WITH "PLASITE 7155NP PRIMER" AND "PLASITE 9009 HI-RESISTANT PROTECTIVE COATING".

2. GENERAL

2. PAINT

2.1.1 PRIMER

2.1.1.1 COLOR - MOSS GREEN

2.1.1.2 STORAGE TEMPERATURE - 70°F (APPROX.)

2.1.1.3 SHELF LIFE

A. MIXED (PARTS I, II & III COMBINED) - 8 HOURS MAXIMUM
B. SEPARATE CONTAINERS - 12 MONTHS MAXIMUM.

2.1.2 PROTECTIVE COATING

2.1.2.1 COLOR - GRAY

2.1.2.2 STORAGE TEMPERATURE - 70°F (APPROX.)

2.1.2.3 SHELF LIFE

A. MIXED (PARTS I & II COMBINED) - 6 HOURS MAXIMUM.
B. SEPARATE CONTAINERS - 24 MONTHS MAXIMUM.

2.2 THINNER

2.2.1 ONLY PLASITE NUMBER 71 THINNER SHALL BE USED.

2.3 SPRAY EQUIPMENT

STANDARD TYPE SPRAY GUNS ARE ACCEPTABLE.

3. SURFACE PREPARATION

3.1 GENERAL SURFACE PREPARATIONS

3.1.1 ALL WELDS AND SHARP EDGES MUST BE ROUNDED.

3.1.2 ALL PAINT, PLATING AND OTHER FINISHES SHALL BE REMOVED PRIOR TO GLASS BEAD BLASTING.

W 39252

B.M. ALLIS DATE [illegible] -69 CHKD. [illegible] APVD. [illegible]

3.1.3 SURFACES SHALL BE DEGREASED PRIOR TO BLASTING. ORGANIC SOLVENTS, ALKALINE SOLUTIONS, VAPOR OR HOT WATER WITH DETERGENTS MAY BE USED.

3.1.4 TRACES OF DUST MAY BE REMOVED BY BRUSHING. CARE MUST BE TAKEN TO AVOID CONTAMINATING SURFACE WITH FINGER PRINTS.

3.1.5 PRIMER COAT MUST BE APPLIED SAME DAY BLASTING IS PERFORMED TO PREVENT CONDENSATION OR VISIBLE OXIDATION OF PREPARED SURFACES.

3.2 SURFACE PREPARATION FOR STEEL PARTS.

3.2.1 AFTER GENERAL SURFACE PREPARATIONS HAVE BEEN PERFORMED GLASS BEAD BLAST ALL SURFACES TO BE PAINTED

3.3 SURFACE PREPARATION FOR ALUMINUM PARTS

3.3.1 AFTER GENERAL SURFACE PREPARATIONS HAVE BEEN PERFORMED GLASS BEAD BLAST ALL SURFACES TO BE PAINTED

3.3.2 FOLLOWING REMOVAL OF DUST, PARTS ARE TO BE GIVEN AN "ALODINE 1200 SYSTEM" TREATMENT. NO SUBSTITUTE IS PERMITTED.

4. MIXING

4.1 PRIMER

4.1.1 PLASITE 7155NP PRIMER IS A 3 COMPONENT COATING CONSISTING OF THE FOLLOWING:

A. PART I - PIGMENTED RESIN.
B. PART II - MEDIUM VISCOSITY RESIN.
C. PART III - CATALYST.

4.1.2 MIXING PROCEDURE FOR PRIMER

4.1.2.1 MIX PART I WITH PART II. ADD PART III AND STIR MIXTURE THOROUGHLY. IF NECESSARY, PAINT MAY BE THINNED WITH PLASITE 71 THINNER TO OBTAIN GOOD WORKABILITY WITHOUT SAGGING. CONTENT OF THINNER IN MIXTURE SHALL NOT EXCEED 15% WEIGHT OF PARTS I, II AND III COMBINED.

4.1.2.2 LET STAND FOR APPROXIMATELY 30 MINUTES BEFORE USING.

4.2 PROTECTIVE COATING

4.2.1 PLASITE 9009 PROTECTIVE COATING IS A 2 COMPONENT PAINT CONSISTING OF THE FOLLOWING:

A. PART I - COATING
B. PART II - CATALYST

W 39253

4.2.2 MIXING PROCEDURE FOR PROTECTIVE COATING.

4.2.2.1 MIX PART I WITH PART II THOROUGHLY. IF NECESSARY, PAINT MAY BE THINNED WITH PLASITE 71 THINNER TO OBTAIN GOOD WORKABILITY WITHOUT SAGGING. CONTENT OF THINNER IN MIXTURE SHALL NOT EXCEED 15% WEIGHT OF PARTS I AND II COMBINED.

4.2.2.2 LET STAND FOR APPROXIMATELY 30 MINUTES BEFORE USING.

PROCEDURE

5.1 PRIMER AND PROTECTIVE COATING SHALL BE APPLIED BY SPRAYING.

5.2 APPLY A "MIST" BONDING PASS AND ALLOW 5 MINUTES MINIMUM TO DRY. NEVER ALLOW FILM TO DRY COMPLETELY.

5.3 AFTER COMPLETING PARAGRAPH 5.2, APPLY CRISS-CROSS MULTI-PASSES MAINTAINING WET APPEARING FILM. REQUIRED THICKNESS IS ACHEIVED WHEN FILM APPEARS TO BE FLOWING TOGETHER.

5.4 DRYING

5.4.1 AFTER PAINTING PER PARAGRAPH 5.2 & 5.3 THE PARTS SHALL BE DRIED IN A CLEAN ATMOSPHERE.

5.4.2 DRYING TIME FOR PRIMER SHALL BE AS FOLLOWS:

A) DRY "TACK FREE" - 30 TO 45 MINUTES @ 70°F APPROXIMATELY.
B) DRY BEFORE APPLYING PROTECTIVE COATING - 1 TO 4 HOURS.

5.4.3 DRYING TIME FOR PROTECTIVE COATING SHALL BE AS FOLLOWS:

A) DRY "TACK FREE" - 1.5 HOURS @ 70°F APPROXIMATELY
B) DRY HARD - 10 HOURS MINIMUM

INSPECTION

6.1 TOTAL FILM THICKNESS SHALL BE .005 INCHES MINIMUM. PLASITE PRIMER 7155NP SHALL BE .003 INCHES MINIMUM AND PLASITE PROTECTIVE COATING SHALL BE .003 INCHES MINIMUM.

6.2 THE COMPLETED PARTS SHALL BE VISUALLY INSPECTED BY A PERSON WITH NORMAL VISION ACUITY, NATURAL OR CORRECTED UNDER NORMAL OVERHEAD LIGHTING FOR THE FOLLOWING DEFECTS.

A) INCOMPLETE COVERAGE.
B) DEEP SCRATCHES, CHIPS, BLISTERS, FLAKING AND EXCESSIVE AMOUNT.

6.3 MINOR DEFECTS SUCH AS SLIGHT PAINT RUNS, SCUFF MARKS, OR SHADE VARIATIONS ARE ACCEPTABLE.

W 39254

DATE 12-11-68 CHKD. HF APVD.

6.4 PARTS REJECTED FOR DEFECTS IDENTIFIED IN PARAGRAPH 6.2 CAN BE TOUCHED UP.

7. DESIGN CONDITIONS

THE FOLLOWING PARAGRAPHS IDENTIFY DESIGN CONDITIONS FOR REFERENCE ONLY AND SHALL NOT BE USED FOR QUALITY CONFORMANCE INSPECTIONS AND/OR CRITERIA FOR ACCEPTANCE.

7.1 SHIPPING CONTAINER:

7.1.1 PRIMER

PLASITE 7155NP PRIMER AVAILABLE IN THE FOLLOWING KITS:

2-PINT KIT - 1 PINT EACH PART I AND PART II PLUS SMALL CONTAINER CATALYST PART III.
2-QUART KIT - 1 QUART EACH PART I AND PART II PLUS SMALL CONTAINER CATALYST PART III.
2-GALLON KIT - 1 GAL. EACH PART I AND PART II PLUS SMALL CONTAINER CATALYST PART III.
10-GALLON KIT - 1-5 GAL. EACH PART I AND PART II PLUS SMALL CONTAINER CATALYST PART III.

EXAMPLE: 24 GALLONS EQUIVALENT WOULD REQUIRE:

2 -	10 GAL. KIT	20 GAL. TOTAL
2 -	2 GAL. KIT	4 GAL. TOTAL
		24 GALLONS

7.1.2 PROTECTIVE COATING

PLASITE 9009 COATING IS AVAILABLE IN GALLON QUANTITIES ONLY. CATALYST AND PAINT COME IN SEPARATE CONTAINERS, WHEN MIXED WITH EACH OTHER WILL EQUAL ONE GALLON.

7.1.3 THINNER

PLASITE THINNER NO. 71 IS AVAILABLE IN QUARTS AND GALLONS. QUANTITY OF THINNER SHOULD EQUAL 20% OF PAINT ORDER.

7.2 USAGE

THE COMBINATION OF PLASITE 7155NP PRIMER AND 9009 PROTECTIVE COATING SHALL BE USED ON PARTS WHICH REQUIRE RESISTANCE TO RADIATION AND A DECONTAMINATION CAPABILITY.

W 39255

B.M. ARTIS DATE 12-?-69 M CHKD. J.H.F. APVD. [illegible]

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August 1, 1978

SOM-II-175

Mr. R. J. Toole
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Subject: Power Imbalance Detector Correlation Test, TP 800/18

Dear Ron:

As a result of recent studies of the incore to out-of-core detector correlation slopes, it has been determined that a slope of 1.15 is necessary to assure that the correlation remains conservative through the entire cycle of operation. Therefore, B&W recommends revising your test procedure, TP 800/18, such that the "desired slope" is $\geq$ 1.15. Attachment 1 is provided as an updated data sheet for your procedure.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/WHA/bay

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J. L. Seelinger
L. L. Lawyer
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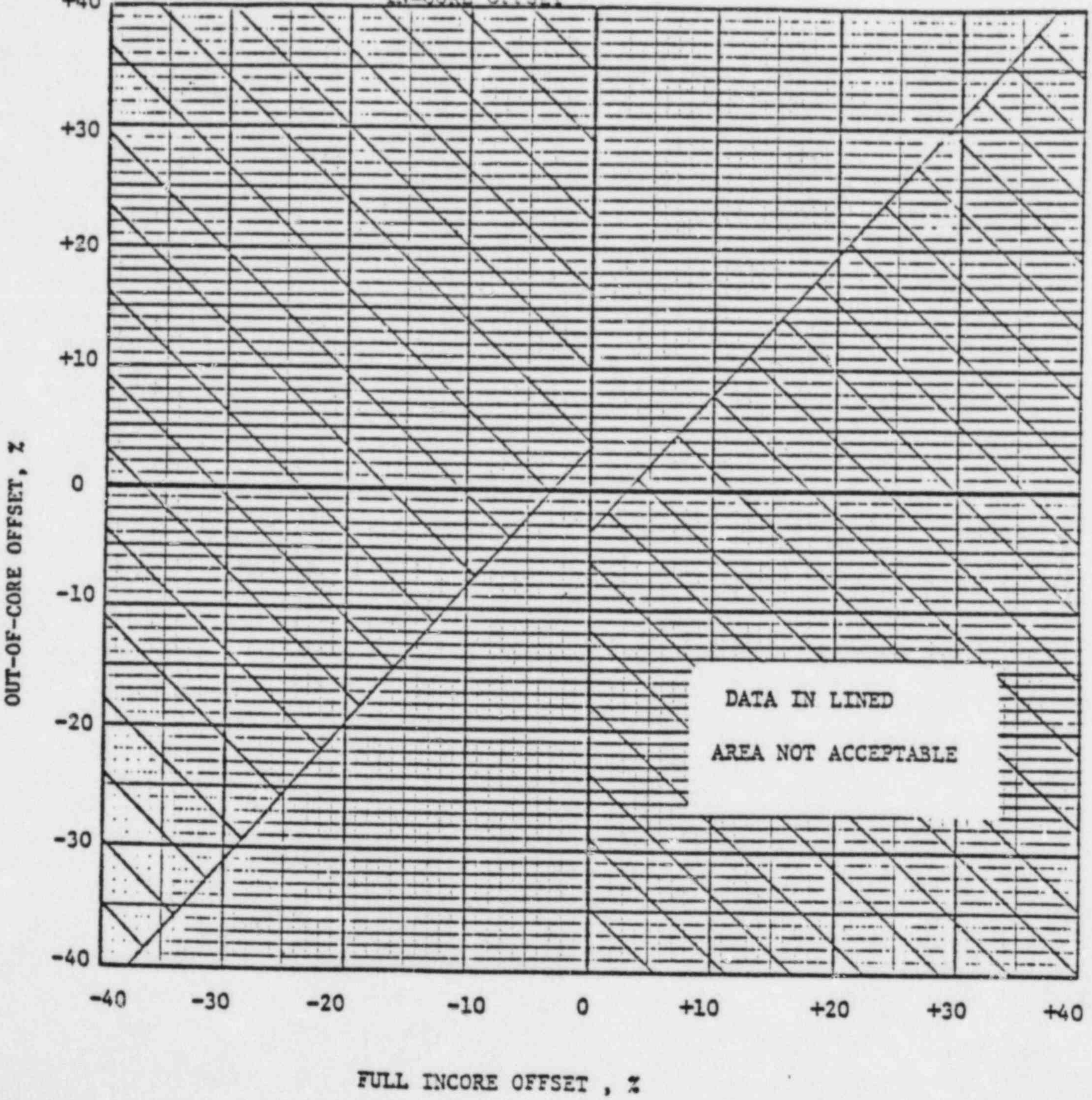
W 39256

ATTACHMENT 1

OUT-OF-CORE OFFSET

VS

IN-CORE OFFSET *



* For OCD-ICD correlation slope of 1.15:1

W 39257

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August 1, 1978

SOM-II-176

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Middletown, PA 17057

Subject: TMI-2, Cycle 1 Startup Testing Following Main Steam Safety Valve Replacement Outage

Gentlemen:

Following the main steam safety valve modifications, B&W recommends that:

a. Prior to achieving criticality, 1) all of the main steam safety valves lift at their proper respective pressures, and 2) actual valve blowdown is within the code limits.

b. Because ORA's have been removed, the following tests should be performed while in Mode 3 at 532°F and 2155 psig:

1. TP 200/11 - RCP Flow Tests (four pump only)

2. TP 200/12 - RCP Flow Coastdown (four pump and one pump)

3. TP 330/5 - CRD Trip Tests (record trip times only)

4. SP 365/2 - Loose Parts Monitor Baseline Data Acquisition (as required)

W 39258

The attached figures of Flow Coastdown Acceptance Criteria are provided for use in performing (2) above. Note that both curves start at time zero at a value of 376,640 gpm. This is the current value (with orifice rods removed) of minimum acceptable four pump flow. The maximum acceptable value of four pump flow remains unchanged (410,000 gpm). Also, please note that these curves include the effect of the snubbers on the DP transmitters while in position number 4 but that they do not account for any measurement error. Therefore, test data should be decreased by the maximum measurement error prior to comparison with the acceptance criteria.

c. The Reactimeter Checkout Test, TP 710/1, Section 9.3, should be performed while at zero power.

d. Escalation from 0% to 15% power should be performed per Section 9.1 of SP 800/21

e. The 15% power plateau testing should be performed per Section 9.2 of SP 800/21, with the exception of Steps 9.2.14 and 9.2.15

f. Operation above 15% power is not conducted until the above testing has been satisfactorily completed.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
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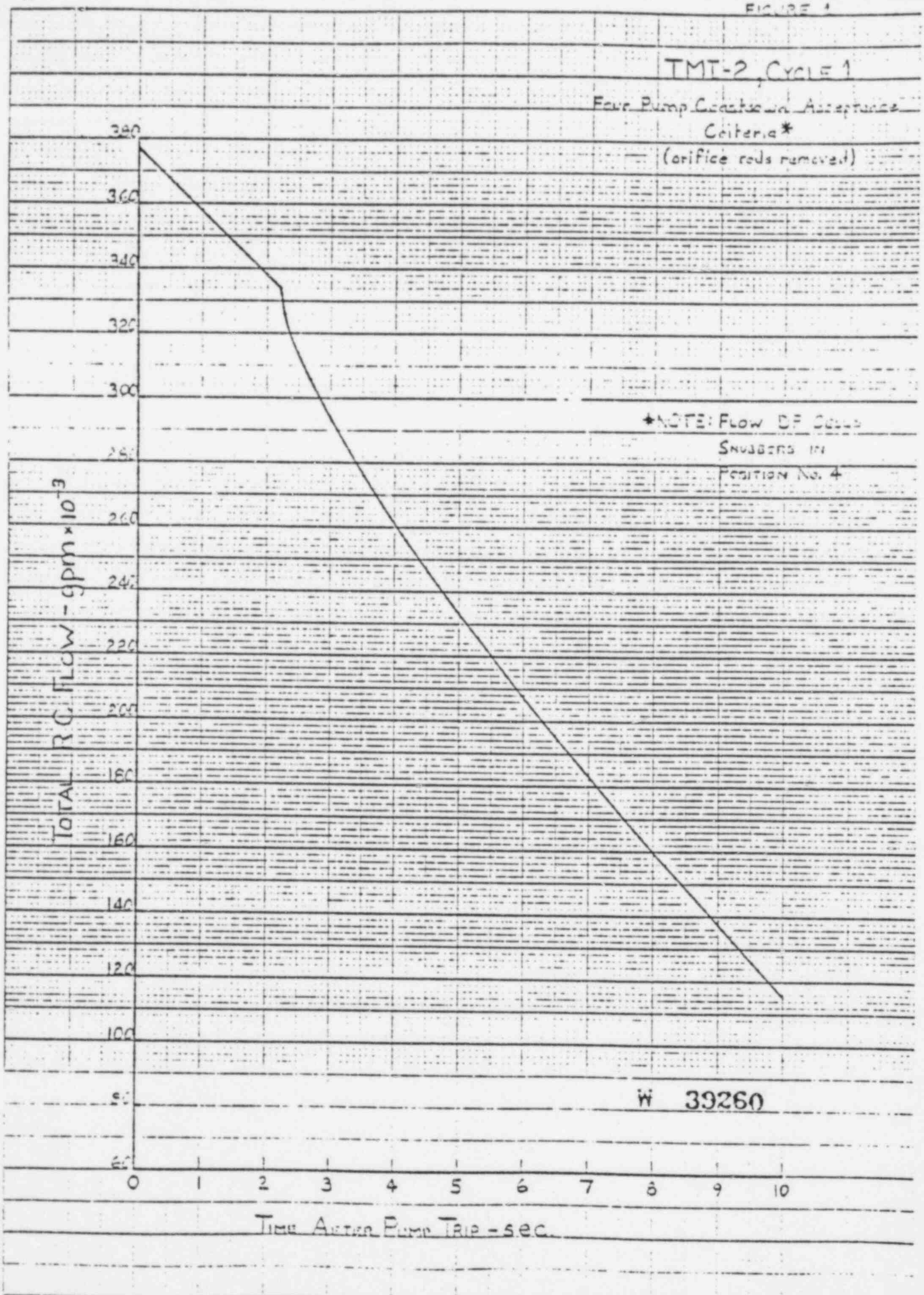
FIGURE 1

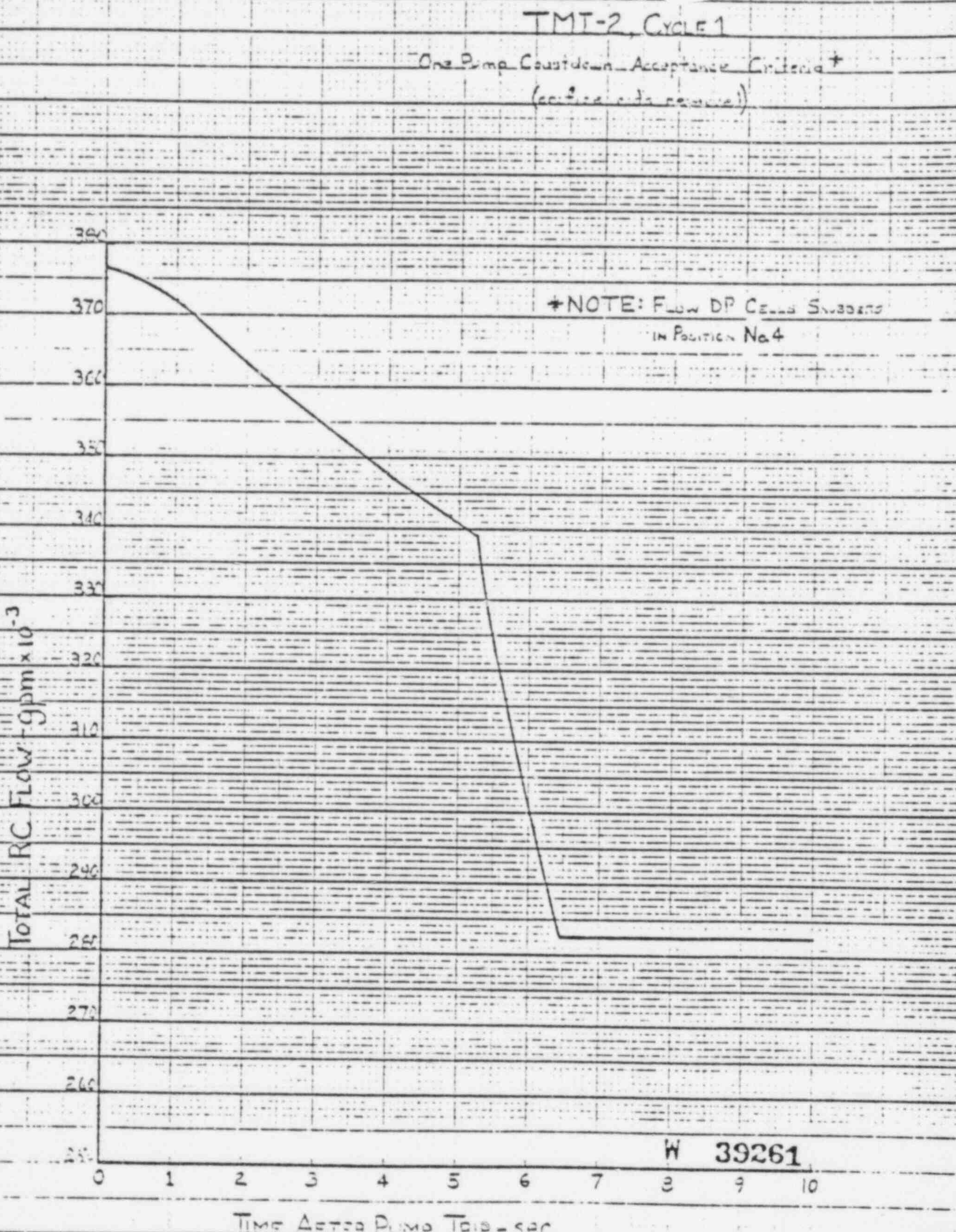


W 39260

K-E 10 X 10 TO 1/2 INCH 7 X 10 INCHES KEUFFEL & ESSER CO. MADE IN U.S.A.

46 1320

FIGURE 2



W 39261

46 1320

B&W

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August 9, 1978

SOM-II-177
REM-I-365

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Subject: Turbine Stop Valve Testing Recommendations

Gentlemen:

W 39262

B&W Engineering's review of the steam generator tube failures with some B&W units has resulted in investigations into the possible cause(s) for such failures. One such potential candidate is turbine stop valve testing. Frequent testing at high power levels (high rates of steam flow), especially with rapid valve stroke times, has been postulated as one of the potential contributors. The following recommendations are developed, for all operating B&W nuclear units, to reduce the potentially adverse effects of turbine stop valve testing on the once-through steam generators.

1. Reactor power should be $\leq$65% of rated power. If a down power maneuver is required, the time spent at this reduced power should not exceed approximately 15 minutes. Operation at a reduced power level significantly beyond 15 minutes (e.g., greater than 1 hour), followed by a rapid return to full power, can result in a larger kw/ft, which is not the mode of most prudent operation for the fuel.

2. Turbine stop valve stroke times during the testing mode should be reasonably slow (e.g., 15 to 20 seconds opening or closing).

3. The frequency at which the turbine stop valves are tested should be minimized as much as practical.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

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August 14, 1978

SOM-II-178

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Subject: Reactor Vessel Material Irradiation Surveillance Schedule

Gentlemen:

In a previous letter, SOM-II-173, you were notified of a change for removal of two surveillance capsules. This information is correct but incomplete. The complete change to the existing schedule (Table 4.4-5 page 4-28 of the Technical Specifications) should be as follows:

> Change capsule TMI-2F to capsule 2E and change capsule TMI-2E to capsule TMI-2F.

This means that capsule TMI-2E will be inserted at the end of cycle 1, in the top position of holder tube XW, and removed at the end of cycle 13. Capsule TMI-2F will be inserted at the end of cycle 6, in the bottom position of the XW holder tube, and removed at the end of cycle 19.

The reason for this correction is that capsule 2E is designated as a top position capsule, and the specimens contained in this capsule are oriented for this top position. Due to this reverse in order, capsule 2F is designated as a lower position capsule. It should be noted that physically the

W 39264

capsules can be installed in either the top or bottom position. Therefore, the surveillance schedule must be corrected.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
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J. L. Seelinger

W 39265

Babcock & Wilcox

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August 21, 1978

SOM-II-181

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Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: DP 1101-01, Plant Limits and Precautions/Decay Heat Removal Pumps

Gentlemen:

The following corrections should be made to DP 1101-01, Rev. 00, dated May 31, 1977. This will eliminate the discrepancy between the SAR, equipment spec, and the subject document.

1. On page 56, paragraph 3.1, "Design Conditions," change subparagraphs A and B to:

   A. Pressure 520 psig

   B. Temperature 300°F

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E S [signature]

L. C. Rogers
Site Operations Manager

LCR/MLN/bay

W 39266

R. J. Toole
L. L. Lawyer
G. P Miller

8/21/78

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August 23, 1978

SOM-II-182

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Subject: OTSG Accelerometer Testing During Heatup

Gentlemen:

B&W Component Engineering has been working closely with Endevco in order to upgrade the accelerometer channels signal recovery for the instrumentation installed at the "B" Once-Through Steam Generator at TMI-2.

They have recently completed most of the field modifications at the hardliner cables of the instrumentation. They displayed a high level of confidence in this recovery program. However, in order to verify the recovery of the signals, a comparative evaluation of the accelerometer data channels would have to be made.

This comparative analysis will require that certain data recordings be re-recorded (during the initial heatup to 532°F) and that a detailed analysis be performed to assure that the accelerometers are operating within specification.

W 39268

Test Conditions:

The following test conditions should be re-recorded:

1. Two pump steady-state operation (B1, B2), 150°F

2. Three pump steady-state operation (B1, B2, A1) 250°F

3. Three pump steady-state operation (B1, B2, A1) 350°F

4. Three pump steady-state operation (B1, B2, A1) 450°F

5. Three pump steady-state operation (B1, B2, A1) 532°F

6. Impact (Bong) test at 150°F

The recording time at each plateau would be 30 minutes and considering the time required to achieve equilibrium, etc., it may amount to a 1 hour delay at each temperature plateau.

Considering that some variance in signals may creep up after the hardliners have been "baked out" following operation at 532°F, it may be necessary to re-record the accelerometer channel data at the following test conditions to optimize the calibration.

1. Three pump steady-state operation (B1, B2, A1) 532°F

2. Three pump steady-state operation (B1, B2, A1) 450°F

The calibration runs, if required, could be completed during the subsequent testing related with reactor coolant flow coastdown and main steam relief valves.

The recommendations concerning reactor coolant pump starting sequences in SOM-II-125 are waived for this testing only. In the event further calibration runs are necessitated at a later stage, these could be scheduled for turbine screen outage or some unplanned outage.

B&W would deeply appreciate your cooperation in carrying out this program.

R. J. Toole
L. L. Lawyer
G. P. Miller

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E. L. Selsedel
for
L. C. Rogers
Site Operations Manager

LCR/SFM/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
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D. Slear

W 39270

ب ق اردیس

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August 29, 1978

SOM-II-183

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Subject: OTSG Oscillation Data

Gentlemen:

With the resumption of plant startup testing on TMI-2, B&W considers that the opportunity to collect performance characteristics of the OTSG and feed system will be available. The intent of such a study is to determine the value of adjusting the OTSG feed flow restricting orifice plate position which may allow reduction of plant pressure oscillations during operations at less than full power. A similar adjustment of the orifice plate has been accomplished in other B&W units with very encouraging preliminary results. B&W considers that adjustment at TMI-2 should not be made until feed system and steam generator performance characteristics at the current orifice plate position are evaluated.

In order to obtain sufficient baseline data on plant oscillations in the 20% to 90% FP range, B&W requests that information be collected and forwarded to Lynchburg as outlined in the enclosed procedure. This data will then serve to help determine what adjustments will be required, if any, to the OTSG orifice plates on TMI-2. This data can be collected during a scheduled power transient and it need not be scheduled as a complete maneuver from 20% to 90% FP.

W 39271

Dupe

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/JV/bay

cc: L. R. Pletke
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W 39272

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TECHNICAL DOCUMENT

TEST SPECIFICATION

62 - 1005379 - 00

Doc. ID — Serial No., Revision No.

for

OTSG OSCILLATION DATA

W 39273

Dupe

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

RECORD OF REVISION

NUMBER
62-1005379-00

REV. NO.	CHANGE SECT/PARA.	DESCRIPTION/CHANGE AUTHORIZATION
00		Original Issue - J. Veenstra Nuclear Service

PREPARED BY James Veenstra (NAME) Engineer (TITLE) DATE 8/21/78

REVIEWED BY Robert Wenks (NAME) Principal Engineer (TITLE) DATE 8/23/78

APPROVED BY [signature] for W.H. Spangler (NAME) Mgr. PSS (TITLE) DATE 8/23/78

W 39274

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TABLE OF CONTENTS/EFFECTIVE PAGE LIST

NUMBER 62-1005379-00

W 38275

1.0 TEST OBJECTIVE

1.1 To measure selected plant parameters while unit is slowly increasing/decreasing power level from 20 to 90%.

2.0 ACCEPTANCE CRITERIA

2.1 The fluid temperature in the upper downcomer shall not fall below the temperature limits determined by Attachment 1.

3.0 CONDITIONS PRIOR TO TEST

3.1 All four RC pumps are in operation.

3.2 The plant will operate normally with the ICS in the automatic mode.

3.3 Plant power level will be increased/decreased at nominal rates with 5 minute holds at 5% FP increments from 20% to 90% FP to attain quasi-steady state conditions.

3.4 Plant Computer data groups and Brush recorders are set up to record the data required in Section 5.0.

4.0 SPECIAL PRECAUTIONS

4.1 In the event of undesirable system oscillations at a particular partial power level, the operator is free to place any ICS station in manual control mode to increase plant stability.

4.2 Rather than terminate the test, the operator should provide brief periods of normal system operation (ICS in automatic mode) at selected partial power levels if the system oscillations are still unacceptable.

4.3 The upper downcomer temperature should not exceed the limits established by Paragraph 2.0.

5.0 DATA ACQUISITION

5.1 Quasi-steady state performance data to be recorded (B&W Reactimeter or equivalent)

1. OTSG A steam pressure
2. OTSG B steam pressure
3. OTSG A steam temperature
4. OTSG B steam temperature
5. OTSG A operate level
6. OTSG B operate level
7. Loop A T hot leg
8. Loop B T hot leg
9. Loop A T cold leg A-1
10. Loop A T cold leg A-2

W 39276

11. Loop B T cold leg B-1
12. Loop B T cold leg B-2
13. Loop A feedwater flow
14. Loop B feedwater flow
15. Loop A feedwater temperature
16. Loop B feedwater temperature
17. Loop A lower downcomer temperature
18. Loop A upper downcomer temperature
19. Reactor power either NI-5,6,7,8
20. Loop A startup level
21. Loop B startup level
22. Loop A ~~upper downcomer temperature~~ shell Temperature
23. Loop B lower downcomer temperature
24. Loop B upper downcomer temperature

Record every .2 sec. for 2 minutes at every 5% power plateau between 20 and 90%. In addition, obtain a Heat Balance power at each test data acquisition plateau.

5.2 Reactor power is to be increased or decreased in 5% FP increments with a 5 minute hold at each level from 20% to 90% FP.

5.3 The following channels are to be recorded on Brush recorder strip charts:

1. OTSG A steam pressure
2. OTSG A operate level
3. OTSG B operate level
4. Loop A feedwater flow
5. Loop B feedwater flow
6. NI power

6.0 RECOMMENDED METHOD

6.1 Power Increase

6.1.1 At a plant power level of 20%, begin to record the system oscillations data.

NOTE: It is not necessary to record data continuously for the duration of the entire test.

6.1.2 At each 5% change in plant power level, obtain a complete set of measured data.

6.1.3 Proceed at a nominal rate from 20 to 90% or to the power level cutoff point.

W 39277

6.1.4 The test is complete when the system oscillations have been measured at the highest plant power level not exceeding the power level cutoff point. However, as a minimum, data should be obtained up to 85% power.

6.2 Power Decrease

6.2.1 At a plant power level of 90%, begin to record the system oscillations data.

NOTE: It is not necessary to record data continuously for the duration of the entire test.

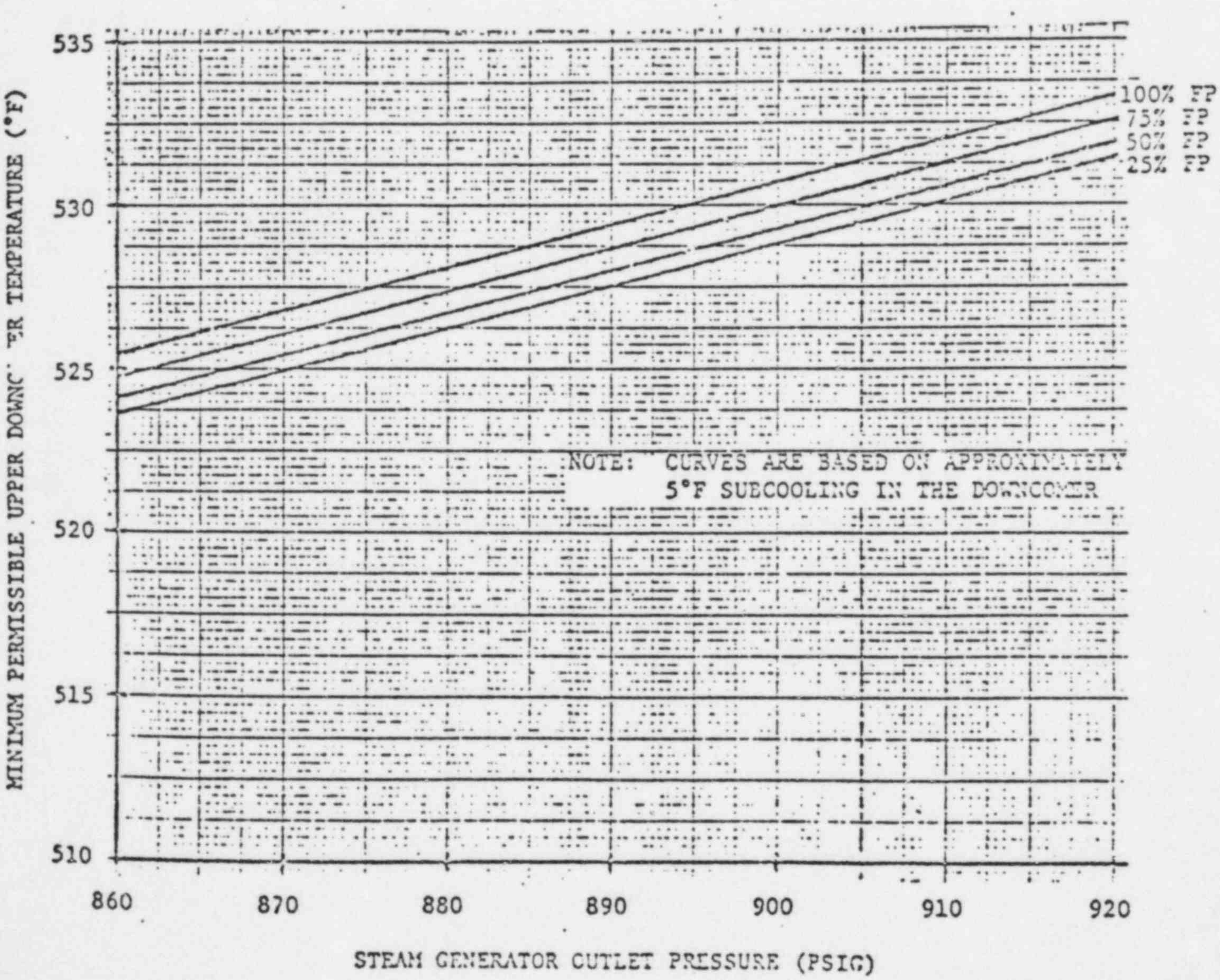
6.2.2 At each 5% change in plant power level, obtain a complete set of measured data.

6.2.3 Proceed at a nominal rate from 90 to 20%.

6.2.4 The test is complete when the system oscillations have been measured at the 20% power plateau.

W 39278

Attachment 1: Minimum Permissible Upper Downcomer Temperature as a Function of Power and Steam Generator Outlet Pressure



W 39279

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August 29, 1978

SOM-II-184

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Subject: Measurement Uncertainty for Shutdown Margin

Gentlemen:

In order to demonstrate satisfactory test results for the shutdown margin measurement made as part of TP 710/1, Zero Power Physics Test, an uncertainty analysis was performed using the test results for TMI-2. This analysis supports the use of 22.06% as the uncertainty to be applied to the test results prior to comparison with the acceptance criterion. Previously, a figure of 20% (SOM-II-137) was given as a result of the preliminary analysis. The detailed analysis is enclosed for your records and for information.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39280

LCR/JV/bay

cc: L. R. Pletke
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MEASUREMENT UNCERTAINTY FOR SHUTDOWN MARGIN - TMI-2

The determination of measurement uncertainty for shutdown is dependent on the plant design and actual measured data. Since these considerations are significantly different between TMI-2 test and the original test at Arkansas Nuclear One, we have reanalyzed the measurement uncertainty based specifically on the TMI-2 test.

Method

The shutdown margin is determined from a series of rod drops as shown in Figure 1. The minimum shutdown margin is extrapolated on the inserted worth axis from the rod insertion limit to the all rods less stuck rod inserted position using a least-square fit line. The inserted worth is determined by boron swap while the rod drop worth is measured by the reactimeter following the drop.

The uncertainty is determined in three steps:

1. The random error in the x-intercept of the least-squares fit line
2. Combination of the error in the least-square line with error terms for nonlinearity in extrapolation and errors in the boron swap measurement. This gives the error in the total-less-stuck rod worth.
3. The error in the reactivity worth from step 2 is combined with the error in the rod worth - inserted at the in limit. The result is expressed in percentage of shutdown margin.

W 39281

1. The calculation of the least-squares fit line is given in step-by-step fashion.

Inserted Worth x	Drop Worth y	Σx	Σy	Σx^2	Σy^2	Σxy
0.055	6.325					
0.584	5.722					
2.312	4.061					
4.011	2.318					
Totals		6.96	18.43	21.78	94.61	22.38

where $y = a + bx$

$$a = \frac{\Sigma x^2 \Sigma y - \Sigma x \Sigma xy}{N \Sigma x^2 - (\Sigma x)^2} = 6.353$$

$$b = \frac{N \Sigma xy - \Sigma x \Sigma y}{N \Sigma x^2 - (\Sigma x)^2} = -1.0035$$

x-intercept

$$-\frac{a}{b} = 6.33$$

Error in fit

$$S^2 = \frac{1}{N-2} \Sigma (y - bx - a)^2 = .001824$$

Error in x-intercept (-a/b)

$$S^2(a/b) = \frac{S^2 \left[(\Sigma x)^2 - N(\Sigma x^2)\right]^2}{(N \Sigma xy - \Sigma x \Sigma y)^4} \left[N(\Sigma xy)^2 - 2 \Sigma x \Sigma y \Sigma xy + \Sigma x^2 (\Sigma y)^2\right]$$

$$= .004404$$

Expressed as a percentage of x-intercept

$$\left(\frac{S_{a/b}}{a/b}\right) \cdot 100\% = 1.0484$$

W 39282

2. The error due to possible nonlinearity is estimated to be 5%. The error due to the boron swap measurement is 7.1%. Hence, the error in the total-less-stuck-rod worth (P_x) is:

$$\frac{\varepsilon(P_x)}{Px} \cdot 100\% = \sqrt{(1.0484)^2 + (5)^2 + (7.1)^2}$$

8.747%

3. The error in shutdown margin expressed as a percentage is:

$$\frac{\varepsilon(P_{SDM})}{P_{SDM}} = \frac{\sqrt{\frac{(\varepsilon(P_x))^2}{(P_x)^2} + f^2\frac{(\varepsilon(P_{BS}))^2}{(P_{BS})^2}}}{1-f}$$

where $f = \frac{P_x - P_{SDM}}{P_x} = \frac{3.57}{6.33} = .564$

$$\frac{\varepsilon(P_{BS})}{P_{BS}} = \text{Boron Swap Error} = 7.1\%$$

The total error in shutdown margin is then:

$$\frac{\varepsilon(P_{SDM})}{P_{SDM}} = 22.06\%$$

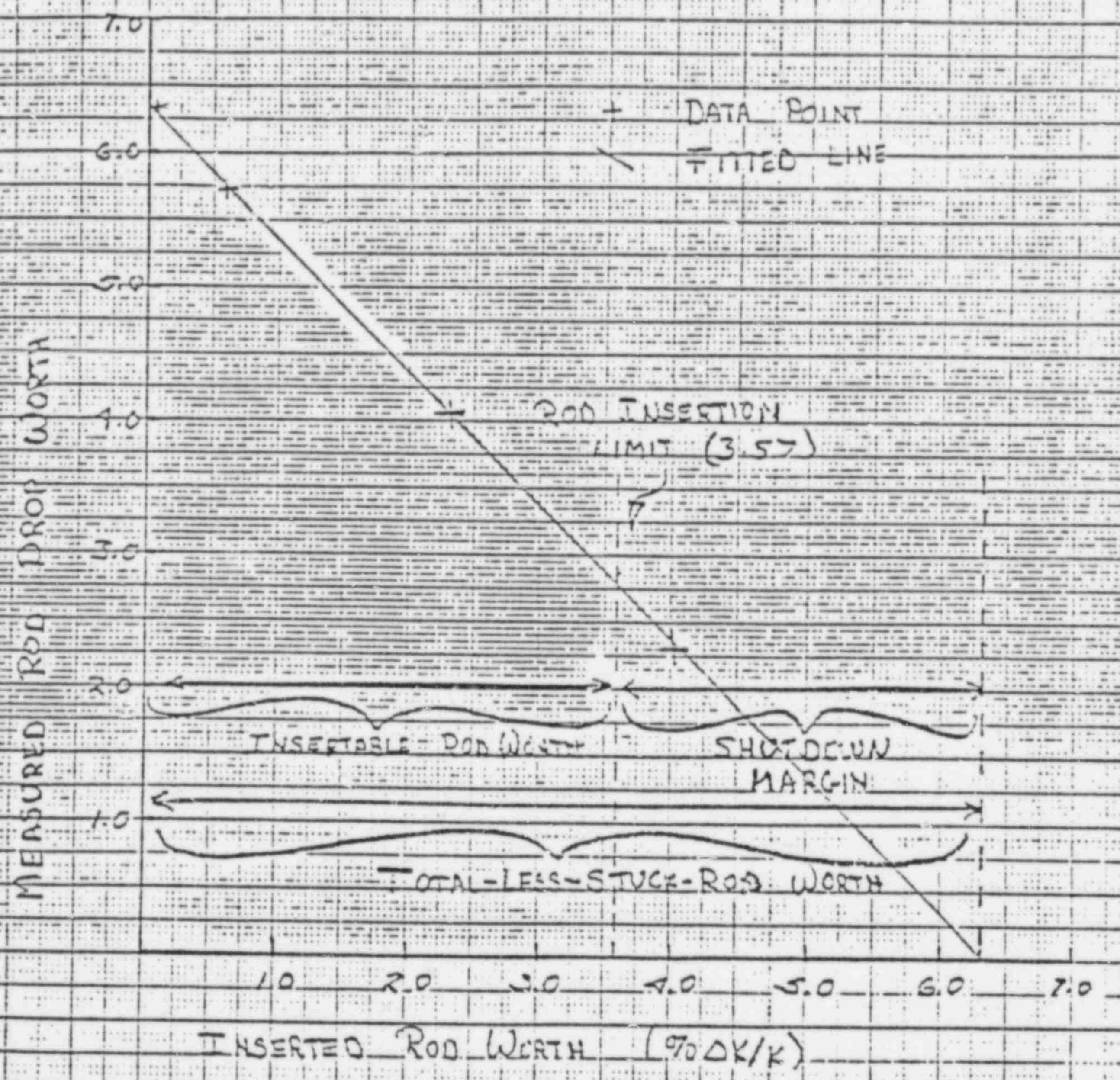
Thus the error adjusted shutdown margin

$$P_{EASDM} = P_{SDM} - \varepsilon(P_{SDM}) = (1-.2206)\ P_{SDM}$$

$$= 2.15\%\ \Delta k/k$$

W 39283

FIGURE I: SHUTDOWN MARGIN TEST
TMI-2 BOL



W 39284

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September 7, 1978

SOM-II-185

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Reading, PA 19603

~~Mr. G. P. Miller~~
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: MIDS Data Retrieval at TMI-2

Gentlemen:

The schedule for collection of the moveable incore detector system (MIDS) data during the power escalation and first core cycle at TMI-2 as recommended by B&W Engineering is as follows:

Every EFPD up to 5 EFPD

Every 5 ± 1 EFPD up to 35 EFPD

Every 5 - 7 EFPD from 35 EFPD to 200 EFPD or until the gadolinium is essentially burned out

Every 25 ± 3 EFPD until end of cycle

NOTE: The point of gadolinium burnout will be determined by analysis of the data by B&W Engineering in Lynchburg

The data would be collected by the B&W personnel on site at equilibrium conditions; and for the period of data collection, a PDO of the reactor would be required in order to perform a more meaningful analysis of the data. The data would be shared with Met-Ed/GPU, and in case Met-Ed/GPU

W 39285

wants MIDS data at any particular condition other than scheduled above, B&W site personnel would be glad to collect the data for them.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

cc: L. R. Pletke
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September 8, 1978

SOM-II-186

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Subject: Reactor Coolant System Flow and FlowCoastdown Test Results

Gentlemen:

A review of TP 200/11 and 200/12 test results has been conducted by B&W, Lynchburg, and the test results are unacceptable.

It is B&W's recommendation that the four pump, steady-state flow test be repeated using the plant computer flow values reduced by 2% for instrumentation uncertainties for comparison to the 377,000 gpm technical specification limit. The NNI flow values should be recorded also to determine the offset between NNI and RPS flow indications.

In addition, the trip of the B-2 reactor coolant pump should be repeated from the four pump operation mode under the following conditions:

1. That two hydraulic snubbers on one ΔP transmitter in each loop be made inoperative (remove internal parts) to provide unsnubbered Loop A and B reactor coolant flow rates. The ΔP transmitter output signals should be transmitted to the plant computer and the calculated flow rate values should be transmitted to the B&W reactimeter.

W 39287

2. That the reactimeter record at the 0.2 second rate the unsnubbered reactor coolant flow, the snubbered (Position #4) RPS reactor coolant flow, and the snubbered NNI reactor coolant flow for both Loop A and B.

3. That the unsnubbered total reactor coolant flow rate must be equal to or exceed the following values of the minimum acceptable flow rate after total reactor coolant flow decreases to 355,500 gpm (the flux-to-flow setpoint):

   Ø/W Trip + 0.5 seconds = 350,500 gpm
   Ø/W Trip + 1.0 seconds = 346,500 gpm
   Ø/W Trip + 1.5 seconds = 342,500 gpm
   Ø/W Trip + 2.0 seconds = 339,000 gpm

4. Reduce the RPS reactor coolant flow signal by 2% and the NNI flow signal by 2.5% for measurement uncertainty when comparing the unsnubbered total reactor coolant flow rate to the acceptance criteria above.

5. Calibrate reactor coolant flow ΔP transmitters prior to repeating the test (unsnubbered reactor coolant flow transmitters only).

If this test will cause an unacceptable delay in plant heatup and startup, B&W approves TMI-2 continuing up in power to 90% without resolving this problem. This permission is based on the expectation that total reactor coolant flow rate, as determined by a plant heat balance calculation, will exceed 112% times a design flow rate of 352,000 gpm. This total reactor coolant flow rate is required in order not to violate a DNBR of 1.30 following a one pump trip if the flow coastdown curve from the 31 August 1978 test is used. NOTE: The problem will have to be resolved prior to full power operation (> 90%).

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

G. Veenstra for

L. C. Rogers
Site Operations Manager

LCR/JV

cc: L. R. Pletke
W. H. Spangler
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W 39288

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September 25, 1978

SOM-II-189

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Mr. G. P. Miller
Station Superintendent
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Post Office Box 480
Middletown, PA 17057

Subject: TP 200/12 Flow Coastdown Test Results

Gentlemen:

An analysis of the one-pump trip performed on 11 September 1978 at TMI-2, as part of TP 200/12, was conducted by B&W, Lynchburg. The unsnubbered data was smoothed to a least squares fit curve and compared to the acceptance criteria. This procedure was necessary because of the excessive noise on the unsnubbered signal and produced a coastdown curve consistent with precise coastdown information recorded by incore ΔP instrumentation at Oconee I.

Attachment I shows the unsnubbered acceptance criteria and the "smoothed" test results (reduced by 2.5%) for the above test. B&W has concluded, therefore, that the acceptance criteria is met.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

W 39289

LCR/JV/bay

cc: L. R. Pletke
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G. K. Wandling
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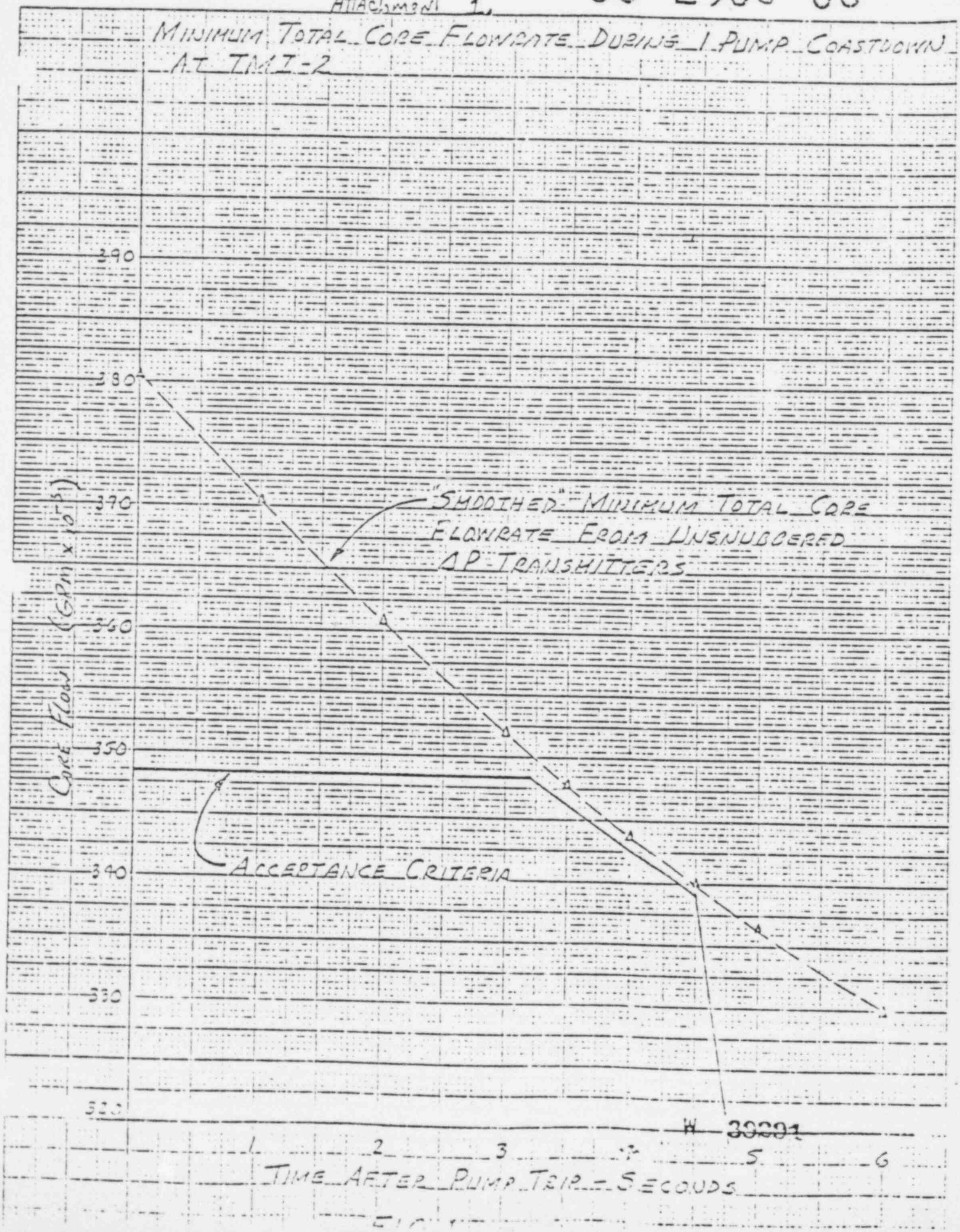
W 39290

86-2306 00

Attachment 1.



Babcock & Wilcox

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September 29, 1978

SCM-II-188

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Station Superintendent
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Subject: Reactor Coolant Pump Operation Summary and Limits & Precautions

Gentlemen:

The purpose of this letter is to summarize the operating restrictions for the reactor coolant pumps and to update these restrictions in the Limits & Precautions document. Please make the following changes to DP 1101-01, Rev. 00, dated 31 May 1977:

1. Replace Figures 1.0-05.1; 1.0-05.2; 1.0-05.3; and 1.0-05.4 with the new figures attached to this letter.

   Add the following to Section 1.4.01 (Page 23)

   "It is permissible to start one pump outside of the NPSH limit for a single pump in any loop, but within the NPSH limit for 2-0 and 0-2 combinations, provided that the second pump in that loop is started as soon as feasible-and not to exceed 10 minutes."

   "Should it become necessary to operate a single reactor coolant pump for an extended period (greater than 10 minutes), reactor coolant system pressure must be maintained at a minimum of 600 psig in order to provide a margin of safety for NPSH requirements; and B&W should be promptly notified."

W 39292

The new figures, 1.0-05.1 through 1.0-05.4, include the information contained in the curves (Figure 1-3) attached to the SOM-II-106 letter dated 31 January 1978. The information in SOM-II-106 is still valid.

The reason for the 600 psig single pump limit is as follows:

> "The B&W supplied reactor coolant pumps were designed to operate most efficiently at the duty point of four (4) pump operation. Any operation of combinations of pumps other than four (4) means that the pumps are not operating at optimum conditions. While this type of operation is not considered detrimental to the pumps during normal heatup and cooldown of the plant, it should be realized that continuous operation under any mode other than four (4) pump could shorten the projected life of those components."

> "B&W is particularly concerned about the operation of a single pump (1/0 or 0/1 pump combination) as this is so far away from the point of best efficiency. Single pump operation promotes recirculation within the impeller passages and mismatched flow and vane angles which in turn can cause localized cavitation. This localized cavitation may cause deterioration of the impeller vanes and shorten the projected life of the impeller. Therefore, single reactor coolant pump operation should be restricted as much as possible. In addition, should it become necessary to operate a single reactor coolant pump for an extended period (greater than 10 minutes), reactor coolant system pressure must be maintained at a minimum of 600 psig in order to provide a margin of safety for NPSH requirements; and B&W should be promptly notified."

SOM-II-125, dated 24 March 1978, recommends the optimum sequence for starting and stopping reactor coolant pumps. It is not a limit but only a recommendation to minimize internals vent valve movement.

SOM-II-31, dated 20 May 1977, has been superseded by the issuance of the Limits & Precautions document, dated 31 May 1977.

SOM-II-52, dated 3 August 1977, no longer applies since the plenum flow restrictor has been removed and the core installed.

All time limit restrictions of 2, 3, or 4 pump operation, due to BPRA and ORA wear problems, have been rescinded. This is due to the removal of the ORA's and installation of the BPRA retainers.

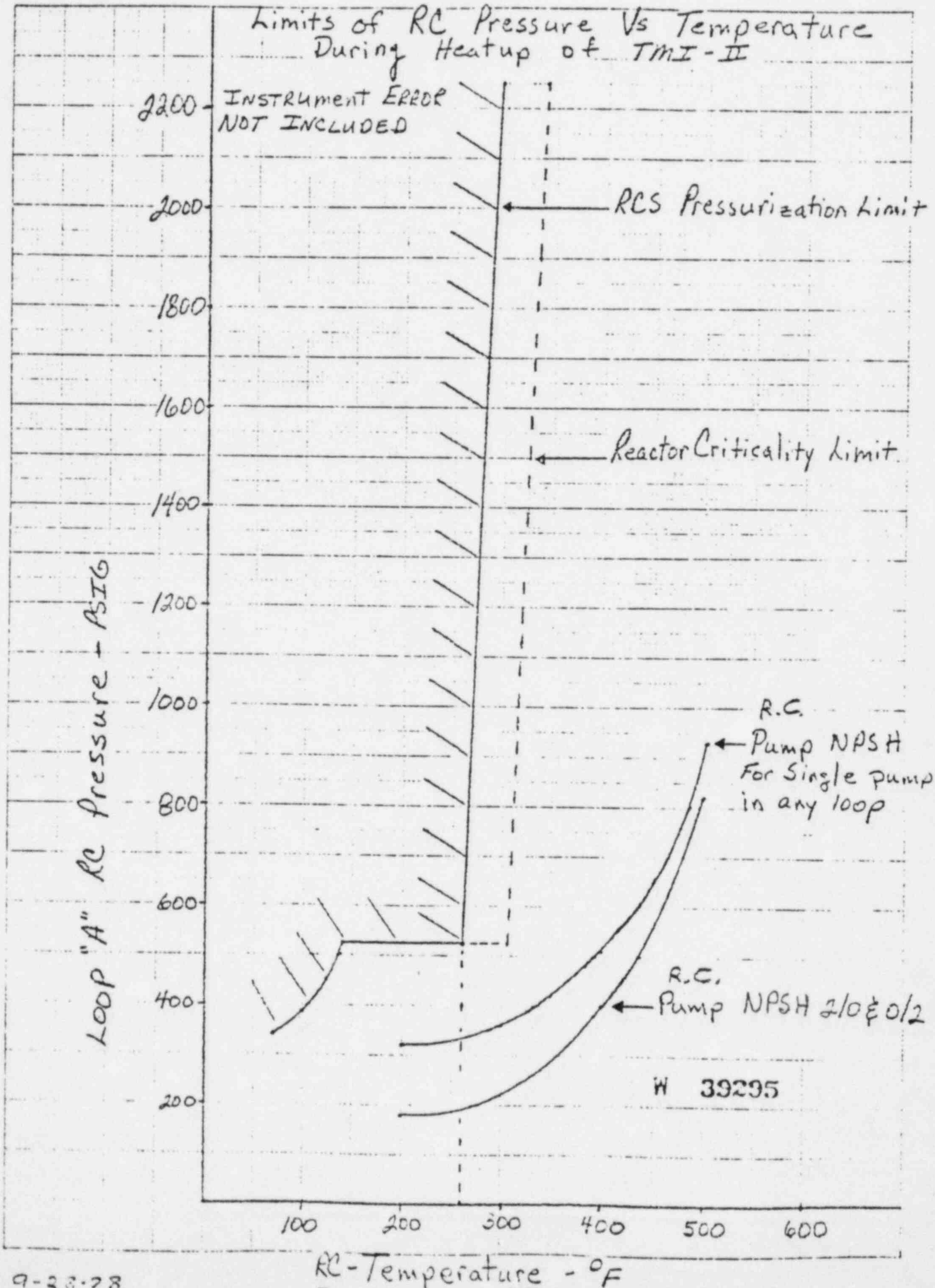
If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

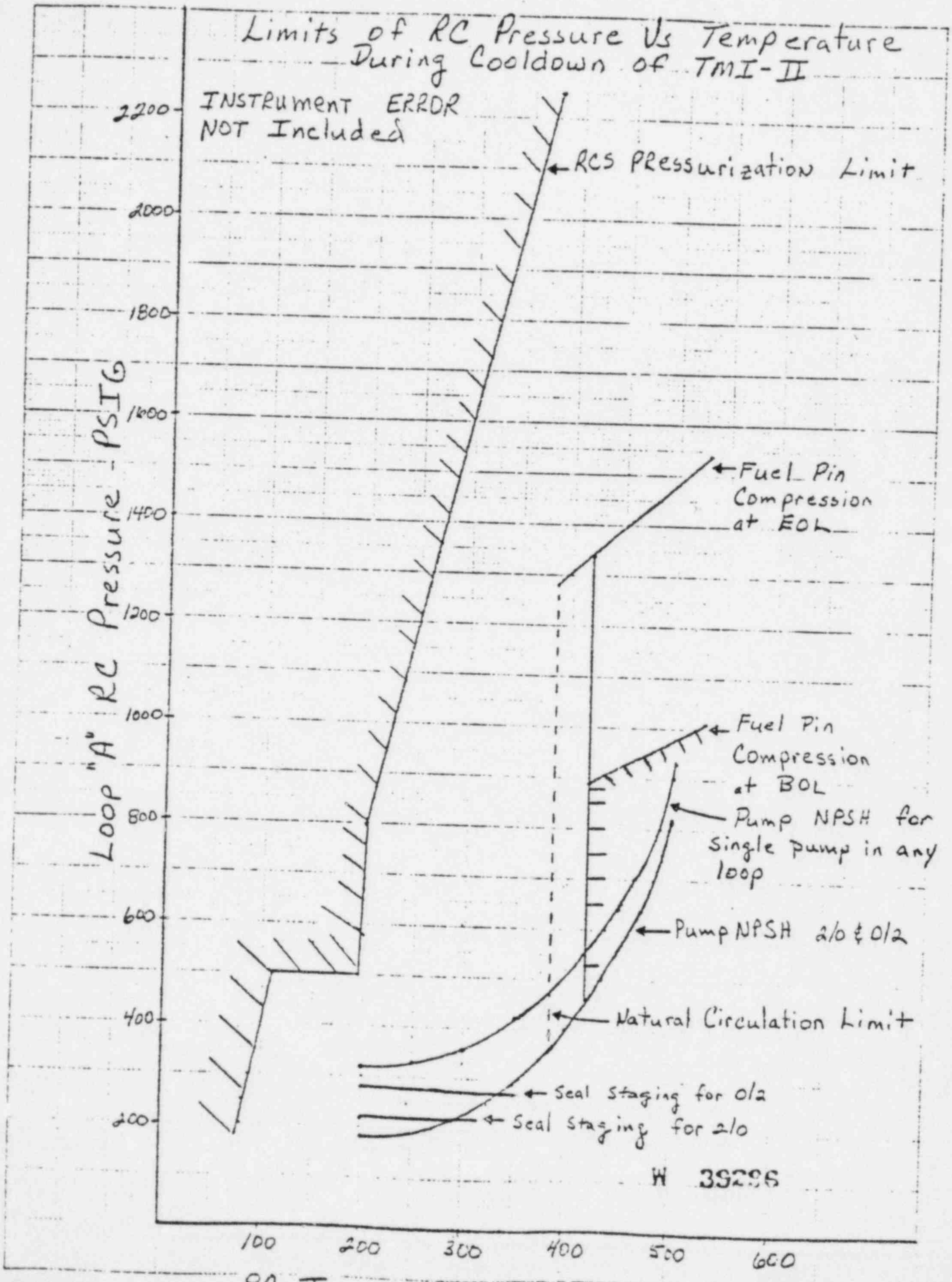
LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
J. L. Seelinger



Limits of RC Pressure Vs Temperature During Heatup of TMI-II
INSTRUMENT ERROR NOT INCLUDED
RCS Pressurization Limit
Reactor Criticality Limit
R.C. Pump NPSH For Single pump in any loop
R.C. Pump NPSH 2/0 & 0/2
W 39295
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
100
200
300
400
500
600
LOOP "A" RC Pressure - PSIG
RC-Temperature - °F

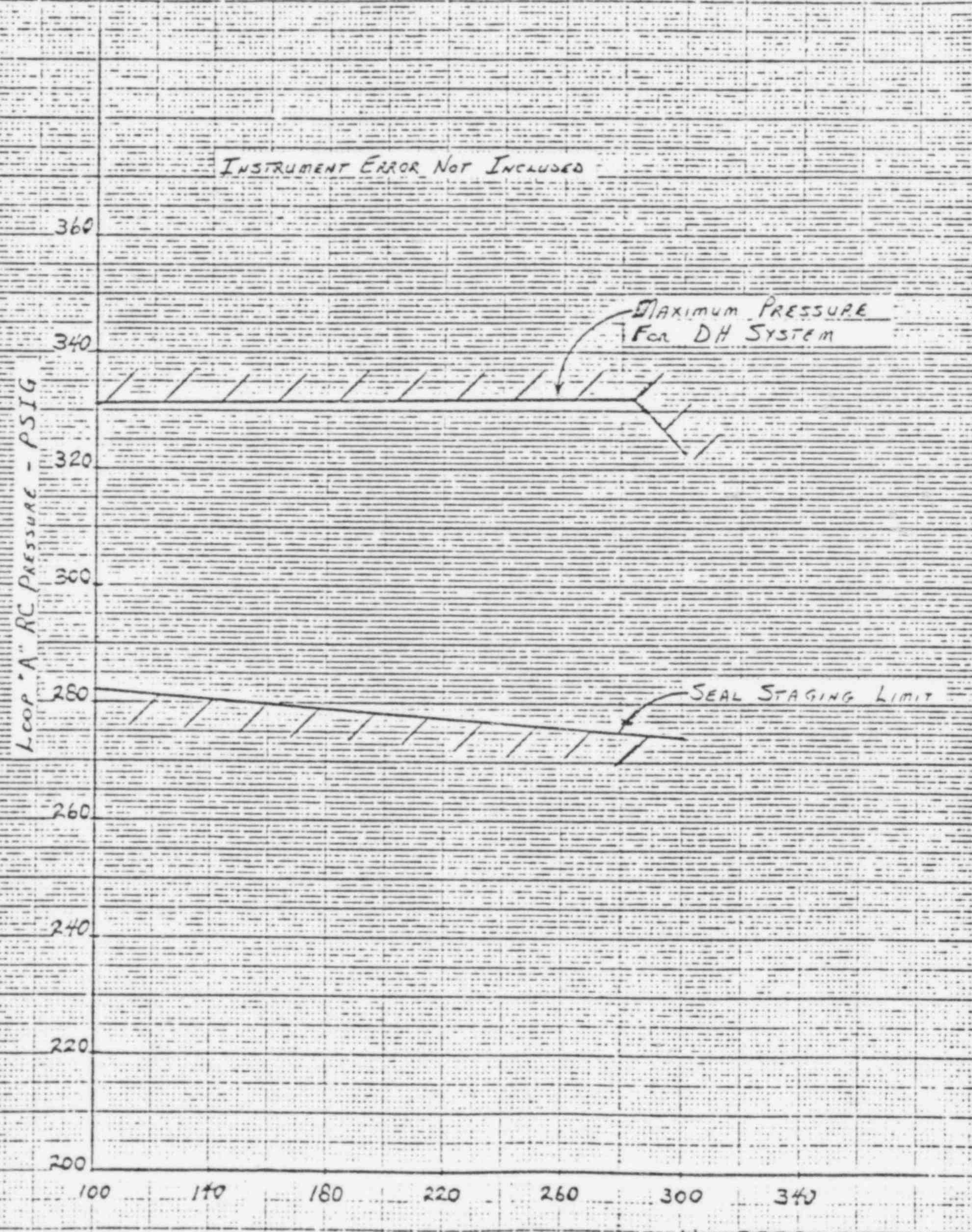
9-23-78



9-29-73

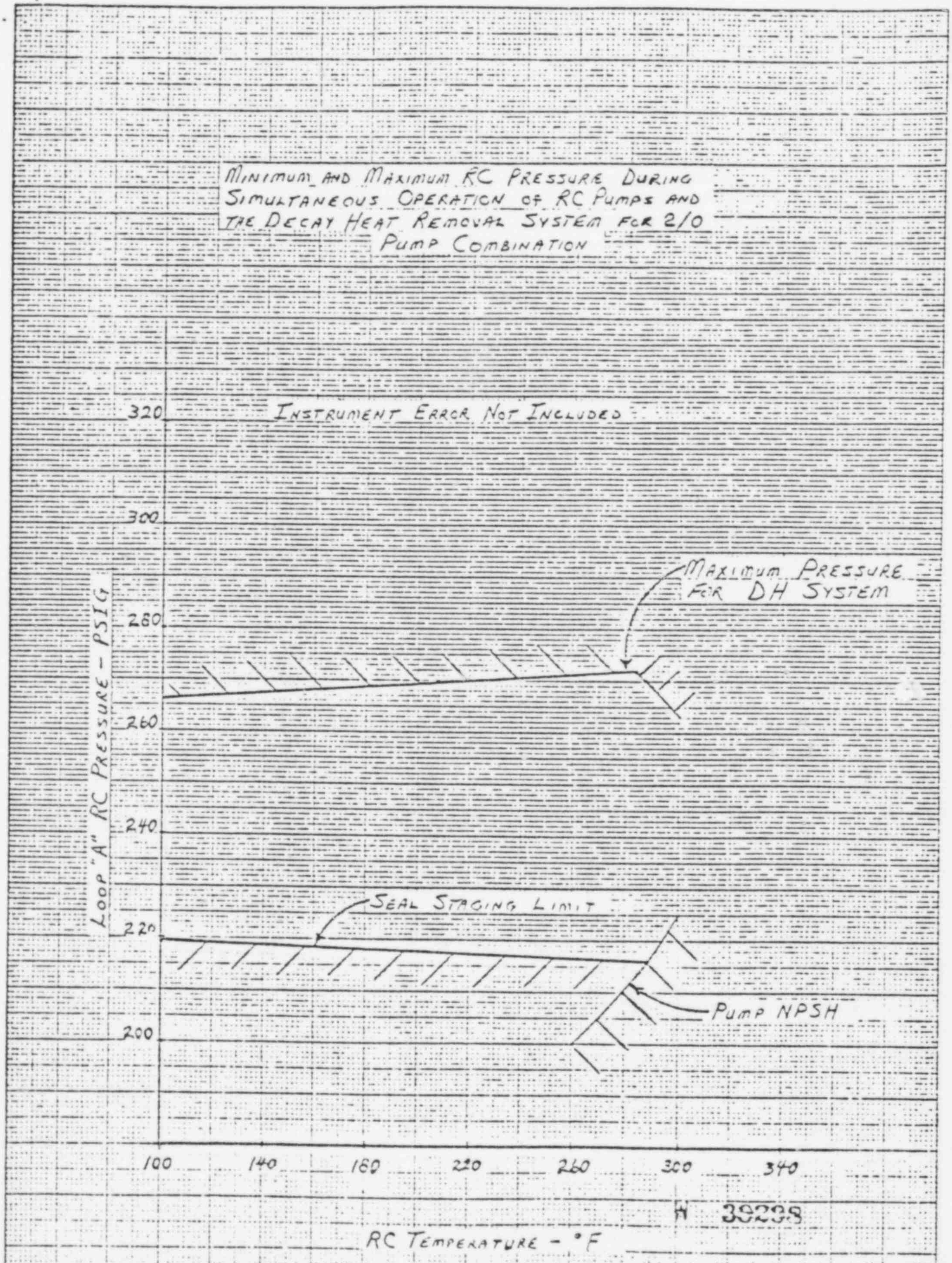
FIGURE 1.0-65.2

Minimum and Maximum RC Pressure During Simultaneous Operation of RC Pumps and the Decay Heat Removal System for 0/2 Pump Combination



39297

5-23-78 Figure 1.0-05.3



5-23-78

FIGURE 1.0-05.4

Please incorporate these changes in your Plant Setpoint document, DP 1101-02, and reset the respective alarms to the new values.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

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W 39300

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September 29, 1978

SOM-II-190

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Subject: Changes in Plant Setpoints Document (DP 1101-02) - Reactor Coolant System Lo-Lo Pressure Alarm & Electromagnetic Relief Valve Reset Pressure

Gentlemen:

Since the high pressure injection (HPI) setpoint for TMI-2 is set at 1640 psig, because of apparent advantages of forewarning about ESFAS actuation, B&W recommends as follows:

1. The reactor coolant low pressure computer alarm presently set at 1700 psig should be raised to 1750 psig.

2. The reactor coolant lo-lo pressure annunciator alarm presently set at 1650 psig should be raised to 1700 psig.

   Both of these alarms are covered on page 6 of DP 1101-02, dated June 7, 1977.

Additionally, the close setpoint for the electromagnetic relief valve on the pressurizers seems to have been incorrectly listed at 2230 psig on page 8 of DP 1101-02, dated June 7, 1977. The recommended close setpoint is 2205 psig, and the open setpoint for the electromagnetic is 2255 psig.

W 39299

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October 4, 1978

SOM-II-193

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Subject: DNBR Extrapolation

Gentlemen:

The Unit Startup and Power Escalation Test, SP 800/21, requires that the measured minimum DNBR be extrapolated to ensure that the limit of 1.30 is not exceeded. The curve in Attachment 1 should be used to extrapolate the measured DNBR as indicated. This extrapolated value should then be corrected as follows and compared to 1.30.

1. Record the level where the total peak occurs and the plant computer value of HCMPK for that fuel assembly (in segment 5 of the PDO or axial peak value from computer group 20 - "Worst Case Thermal" printout).

2. Record the extrapolated minimum DNBR

3. DNB1 = DNBR + ΔDNBR1 where:

$$\Delta DNBR_1 = 1.375\ HCMPK - 2.06$$

DNB2 = DNB1 + ΔDNBR2, where:

$$\Delta DNBR_2 = 0.0 \text{ for } HCMPK < 1.5$$

W 39301

$$\Delta DNBR_2 = -.8 * \frac{\text{Axial Position}}{144} + .4$$

4. DNB2 must be > 1.30

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

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Site Operations Manager

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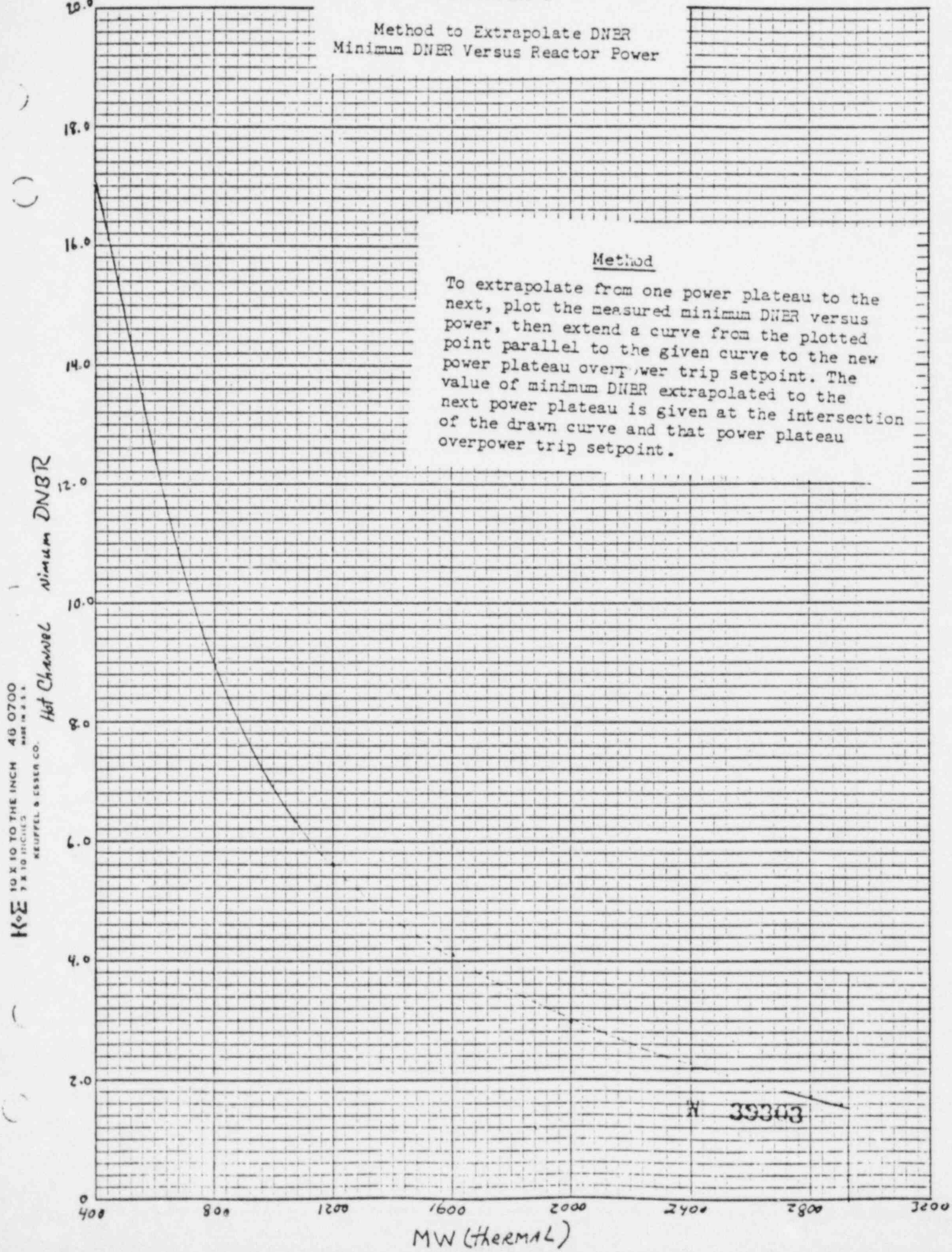
W 39302

Attachment 1

Method to Extrapolate DNBR
Minimum DNBR Versus Reactor Power



Method

To extrapolate from one power plateau to the next, plot the measured minimum DNBR versus power, then extend a curve from the plotted point parallel to the given curve to the new power plateau overpower trip setpoint. The value of minimum DNBR extrapolated to the next power plateau is given at the intersection of the drawn curve and that power plateau overpower trip setpoint.

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October 5, 1978

SOM-II-194

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Subject: TP 800/31, Pseudo Dropped Rod Test

Gentlemen:

B&W has reevaluated the worst case dropped rod power distribution at 40% FP for TMI-2 and recommends that the Group 7 control rod in core location H-14 be the pseudo dropped rod for TP 800/31 instead of the Group 5 rod in core location H-12, as specified in the Physics Test Manual. The power peak will occur in H-2, so the Group 7 control rod in location P-8 may be substituted in order to increase the quality and reliability of the test results. This will cause the peak to occur in an instrumented fuel assembly (B-8).

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/JV/bay

W 39304

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October 6, 1978

SCM-II-195

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Subject: DHV-4A Packing Leakage

Gentlemen:

B&W has been asked to resolve a problem with packing leakage from DHV-4A. It has been reported that the valve was repacked numerous times without any success in stopping the leakage. The problem was then assumed to be a misalignment of the gland follower, and GPU Problem Report #2676 resolved this problem by enlarging the follower bolt hole. Subsequently, the valve packing continued to leak.

Upon researching the history of this valve, the only written record (equipment history) shows that DHV-4A was repacked only once, using grafoil packing, on December 8, 1977, and adjusted again on April 26, 1978. Based on these facts, B&W recommends the following:

1. The valve, as supplied by B&W, did not have grafoil packing and was not designed and manufactured for grafoil. We do not recommend the use of 100% grafoil in this application.

2. We recommend a return to John Crane 187-IX or some combination of packing such as described in the article by Robert T. Wilson of Fisher Controls Company. The grafoil, if used, should be of a type compatible with Reactor Coolant System use. If only John Crane packing is used, drawing interpretations show seven rings of packing below the latern gland and three rings of packing above.

W 39305

3. At present, there isn't any method known to us on how to completely remove the old grafoil packing - short of disassembly of the valve. To ensure satisfactory operation of the new packing, the old grafoil packing must be removed to the maximum extent possible.

4. No additional reaming of gland follower holes or any other alterations should be done at this time.

5. Detailed drawings, that would give dimensions of the gland follower, are not made available to us by the vendor.

6. A complete inspection of packing related components, with special attention to the stem, should be made.

7. A B&W site representative should be present during the repacking of DHV-4A.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers

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W 39306

New Control Valve Packing System May Match Valve Life

A combination of braided graphite filament rings and laminated or ribbon graphite rings in high temperature packing systems can remain essentially leak-tight through almost 50,000 valve strokes

Greg SCHAEDEL
TMI

By ROBERT T. WILSON, Fisher Controls Co.

Many industrial control valves of the sliding stem type use some form of graphite to form the seal between the stem and the packing box. This is particularly true of valves designed for high-temperature steam or water. Some recent testing of several types of preformed graphite rings, however, have shown that conventional methods of using these rings have been far from optimum.

For example, a reciprocating valve of standard construction with a ½-in.-diam stem and conventional laminated graphite ring packing exhibited an unacceptably high leakage rate after 800 cycles against 600 F, 400-psi steam. With a newly developed packing system, this same valve, operating under the same conditions, was well within leakage specifications after 45,000 cycles. This represents an increase in packing life of more than 5000%.

The secret to this success lies in the physical structure of the rings and the order in which they are placed on the stem during assembly. To understand the wide differences in performance between the conventional and the new systems, first consider the differences in physical structure of commercially available graphite packing rings.

Laminates, ribbons, and braids

Laminated graphite rings are made from a large number of thin sheets of laminated graphite, stacked to the desired thickness and compressed to force the laminates tightly together.

Packing made from graphite ribbon is formed by winding the ribbon in a spiral around a mandrel which is the same size as the valve stem until it is large enough to loosely fill the packing box. Pressure is then applied to the two ends of the roll to compress it into an accordion-like structure. In theory, the accordion pleats will give a good seal with lower friction than laminated rings. These rings can be wound from either smooth or crinkled graphite ribbon, but the latter is used much more often.

Braided graphite filament rings are just that: graphite filaments are braided into a rope-like structure that looks very much like black sash cord. The braid is cut to the proper length to make one turn around the valve stem; pressure then forces the two ends together and establishes the necessary seal at the OD and ID.

Each of these ring types has its advantages and disadvantages. Laminated rings give a good seal because the only possible leak paths are along the OD and ID surfaces. However, they sometimes extrude into the small space between the packing box and the packing follower, or between the valve stem and the follower. When this happens, the packing pressure is lessened at the OD and ID with greater tendency toward leakage. This can be prevented, however, by using anti-extrusion rings above and below the laminated graphite ring. These are designed to fit snugly in the box and around the stem to eliminate the "cracks" which allow the extrusion effect.

Graphite ribbon wound rings, either crinkled or smooth, have the same disadvantages as laminated rings plus the additional disadvantage of leakage possibilities if the turns of the ribbon windings are not kept tightly compressed together. This will happen when wear at the ID allows them to come apart. Once a leakage path is well established between ribbon turns, it is difficult to close even with increased tightening of the packing follower bolts.

Braided graphite filament rings, under pressure, expand to fill the space between box and stem quite well, but the tiny openings inherent in the braided structure cannot provide the seal against leakage that either the laminated or ribbon rings exhibit. For this reason, their use has been limited.

Graphite particle transfer

When ribbon graphite packing material was introduced to the market, there were some unanswered questions about it. One of them concerned the reason why it didn't give the expected reduction in friction. One of the first steps in a program designed to answer this question was to observe what happened when a newly packed valve was stroked, cycle by cycle.

The packing used for this experiment was a set of standard ribbon-wound rings (crinkle ribbon) precompressed to increase their density. The rings were installed in a packing box using a standard packing ring sequence. In the first few strokes of the stem, friction became very high; however, on continued stroking, friction dropped. At this point leakage occurred. Further tightening of the packing follower stopped the leakage, but friction was then undesirably high in the many uneventful strokes that followed.

W 39307

On the premise that some kind of wear was occurring quickly in the break-in period, valves were taken apart at various stages of break-in and the packing rings and stem examined. Particles of graphite had deposited on the polished stem surface, darkening and dulling the surface area where ring and stem had been in sliding contact. Surprisingly, this graphite transfer was observed even when the stem had not been stroked: each ring left its own imprint of graphite where it had been compressed against the stem.

Examinations of valve stems that had been in service in the field for extended periods also ...

Figure 1. Typical packing rings. Three rings in upper row are standard graphite types. At left foreground is a laminated graphite type with the laminates spread open. Ring standing on edge is a ribbon graphite type that has suffered extrusion; note raised rims around both OD and ID. The split ring in right foreground is a braided graphite filament type.

Figure 2. Graphite ribbon rings. At left is a smooth ribbon and ring; at right is the crinkled ribbon starting material and a ring made from it. Both rings have been cut open to show direction of the laminations.

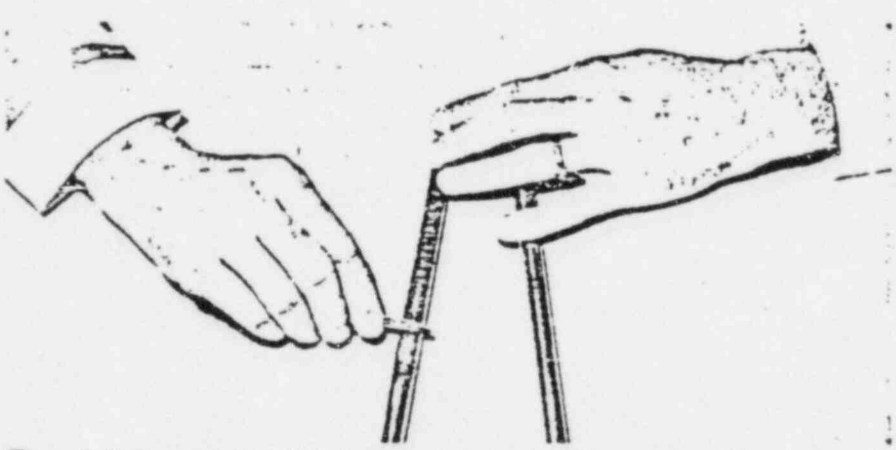


Figure 3. Valve stem on left is from a valve that had been been laboratory tested, using laminated rings. Graphite particles can be scraped off. Stem at right had been tested with the braided graphite filament/laminate graphite combination. Stem is smooth and free of particles.

W 39308

effect; the graphite buildup on the stem was so thick it could be scraped off with a knife. The surface of the deposit was much rougher than the underlying stem surface—a condition which had to be conducive to wear. Further investigation revealed that both graphite ribbon-wound rings and laminated graphite rings exhibited the effect.

At this point there seemed to be two approaches to the problem: either prevent the particle transfer in the first place, or find some means of cleaning the deposit off the stem surface as fast as it formed. Tight-fitting solid graphite rings were used above and below laminated or ribbon rings; they behaved as very good anti-extrusion rings but didn't help the particle transfer problem and showed no tendency to clean particles off the stem.

Braided graphite filament

Concurrent with the particle transfer studies, an investigation was under way to check out various types of anti-extrusion rings. One such ring was the graphite filament type—chosen because it could be fitted tightly to the stem and box surfaces. This set the stage for discovery.

Rings made of braided graphite filament were found to have a very important property: they didn't show the graphite particle transfer effect. A systematic investigation was then set to test all reasonable combinations of graphite ring structures to see if the non-particle-transfer properties of braided graphite filament could be combined with the good sealing properties of laminated or ribbon graphite rings, thereby achieving the best of both worlds.

The result of this study was even more rewarding than expected: not only did the braided graphite filament exhibit its non-transfer property, it also was found to be an excellent "cleanup" material for the particles transferred from the laminated or ribbon rings. The remaining studies were designed to verify this observation in practical field use and to optimize the number of rings needed and their position on the valve stem.

To allow direct comparison of the great many packing systems tested, the dimensions of stem and packing box were standardized and, with a few justifiable exceptions, the number of rings used was kept constant. The packing system schematics shown in Figures 4 through 7 indicate the packing ring sequences and the resulting cycles vs leak rate plots.

The lantern ring shown in each schematic separates the four (or five) ring major seals located at the bottom of the stem from the minor, or secondary, seals located above the lantern ring. The lantern ring forms the chamber which is tested for leakage by means of the side vent shown. The major seals were always tested against 600 F, 400-psi steam. Minor seals merely prevent steam that penetrates the major seal from leaving the lantern ring chamber. The maximum allowable leakage rate typically specified high temperature steam service is 0.5 cc/hr for a 0.5-in-diam stem.

Dramatic effects revealed

Results with conventional laminated graphite and ribbon graphite packing rings are shown in Figure 4. The packing sequence, starting from the high-pressure end of the stem, is four graphite rings, a lantern ring, and three graphite rings. As expected, leakage occurred when the original packing ring stress was not followed by the "post-break-in" packing-follower tightening normally required by this packing scheme. Even after normal tightening of the follower, the leak rate limit was quickly exceeded. Tightening the packing follower bolts again may have brought the valve stem leakage within specifications, but friction would have increased above the starting level.

Figure 5 shows the cycles vs leak rate obtained when braided graphite filament rings were used to replace some of the laminated graphite rings. The 0.5 cc/hr leak rate specification was not exceeded until after 19,000 cycles—an improvement of about 2000% in packing life. Even at 30,000 cycles, leak rate was only 1.0 cc/hr. For this test, the laminated graphite rings (one above the lantern ring, one below) were positioned between braided graphite filament rings. Packing ring stress on these was the same as for the tests shown in Figures 4, 5, and 7.

Another variation was to use laminated rings and ribbon rings reduced in thickness by half, in conjunction with graphite filament as shown in Figure 5. Two ⅛-in.-thick rings were used below the lantern ring chamber, and one was used above. At the 0.5-cc/hr leak rate, reached after 30,000 cycles with laminated rings, packing life was increased about 3600%. The theory here was to keep the graphite particle transfer to a minimum, consistent with adequate sealing, by keeping the seal ring/stem area of surface contact low.

The best of the variations tested to date is shown in Figure 7. Over 47,000 cycles were logged using laminated rings with braided graphite filament rings before the 0.5-cc/hr leak rate specification was exceeded. After 50,000 cycles the leak rate was still only 0.75 cc/hr. (These may be considered near the end of their mechanical life due to general wear and tear.) The lower curve plotted in Figure 7 shows the effect when using ribbon rings instead of laminated rings. The leakage specification is exceeded at about 23,000 cycles, and 1.0-cc/hr rate is exceeded at about 28,000 cycles. All packing cycle life listed, for all tests, is with no adjustment made to the packing bolts after start of test.

Stem roughening vs polishing

Valve stems that had been in extended use in the field were brought into the laboratory and compared in surface appearance and roughness with the stems used in the above tests. Field valves had used standard graphite ribbon or laminated graphite packing rings. In all cases, their sliding contact area was blackened, and the deposit was thick enough that the graphite coating could be scraped into a pile

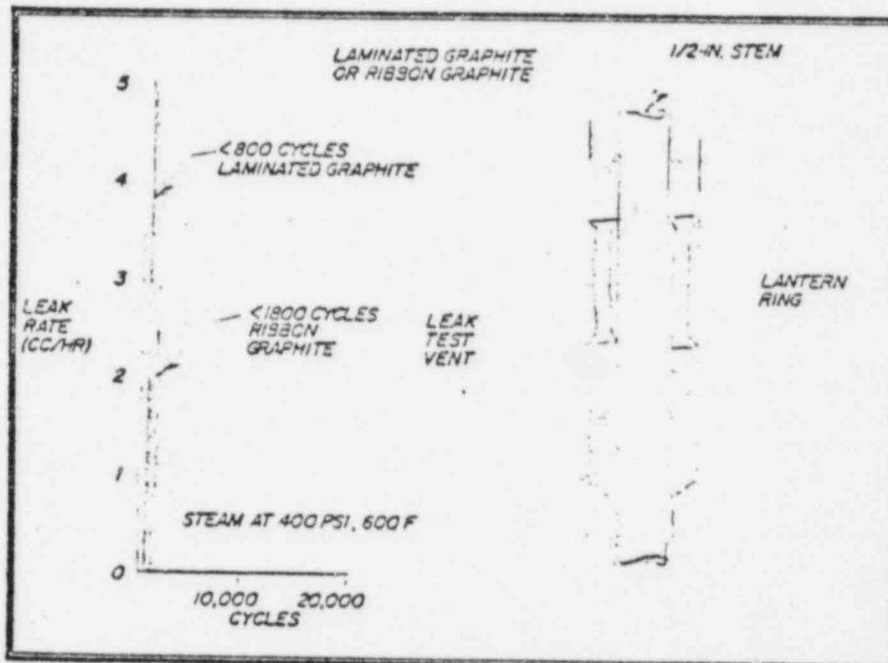
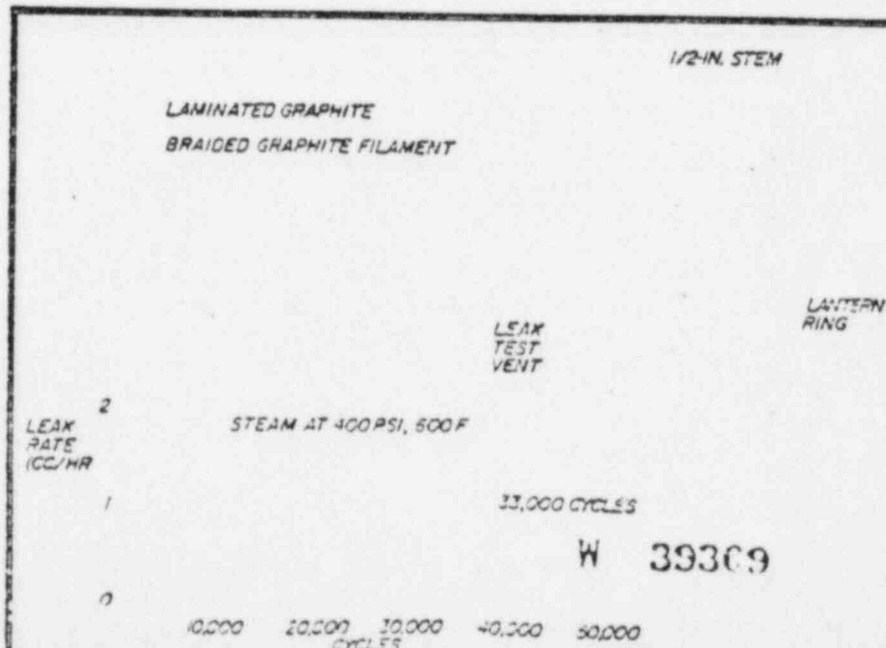


Figure 4. Plot of valve stem cycles vs leak rate through packing for laminated graphite or ribbon graphite only.



W 39369

Figure 5. Results [illegible]

with a knife blade.

On the other hand, the stems used in the laboratory tests that had gone over 20,000 cycles were still extremely smooth with only slight discoloration. Interference colors were visible, which suggested that any coating present was very thin and very smooth. The slight coating seemed to be polished to a smoothness better than that of the original stem.

The contact surfaces of the field examples seemed to become blacker and rougher the longer they were used, suggesting that graphite particle transfer continues as long as the stem is stroked. In addition, putting new packing rings on a stem with this built-up deposit could lead to shorter operating life compared to that provided by the originals.

Another factor related to stem surface roughness is the force necessary to start relative motion between stem and packing rings when the packing ring stress has been adjusted to allow for long cycle life and low leakage. In laboratory tests, the required force to start sliding motion for the laminated ring system shown in Figure 4 was 25% greater than for the systems containing braided graphite filament rings.

During these hysteresis tests, another interesting fact emerged. Ribbon graphite packing rings made from crinkled ribbon didn't show the lower friction expected of them when compared to laminated rings.

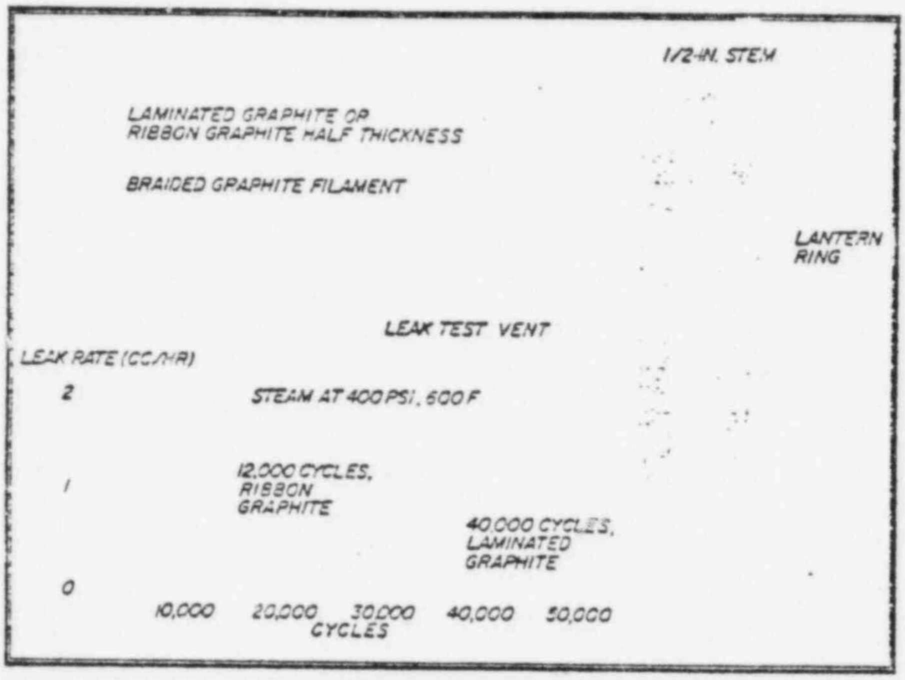


Figure 6. Results with a combination packing of half-thickness laminated or ribbon graphite rings and braided graphite filament rings.

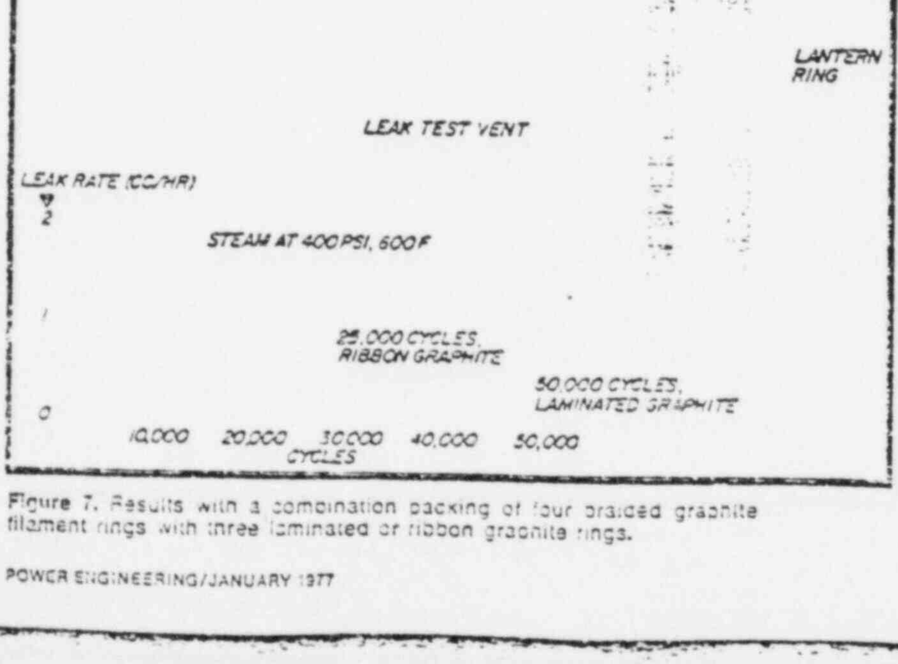


Figure 7. Results with a combination packing of four braided graphite filament rings with three laminated or ribbon graphite rings.

Proof will be in the field

The results of the study are consistently repeatable in the laboratory. This, together with past experience in applying practical research to field use, provides a high degree of certainty that the packing system will be superior to those currently used.

The specific valve application which prompted these studies was nuclear service, but the results can be applied in non-nuclear high-temperature systems as well. For instance, the upper temperature limit for the graphite filament is about 700 F when used in oxidizing service and 1000 F when used in non-oxidizing service.

For nuclear service it is recommended that the low chloride content graphite be used along with the sacrificial zinc washer supplied with this type of graphite. Packing costs will be higher, but it is good insurance against corrosion.

By using the new graphite filament packing technique to advantage and by using TFE packing for non-radioactive conditions below 450 F, a valve user now can standardize on two packing styles.

One thing should be kept in mind if valves are not to be installed immediately. Some valves, especially those destined for nuclear service, may remain in storage up to three years. For this reason, new valves should be shipped without packing rings in place. If the rings are in place, care should be taken to keep the packing box completely dry. When a valve is installed, any hydrostatic testing should be carried out with rubber O-ring seals, after which the O-ring gland can be removed, the box thoroughly dried, and the graphite rings installed. END.

W 39310

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V.13 Torquing of bolting must be performed in strict accordance with Torquing Procedure VEL-P-586 Rev. 1 *(See Section VIII)*

V.14 If a torque wrench is not available, the following is a guideline that can be used on bolt sizes up to and including 1-1/8" in emergency:

Assuming that an average man can develop about 100 ft. lb. torque on a 12" wrench, a 6" wrench should be used on 1/2" bolts; 9" wrench on 9/16" bolts; 12" wrench on 5/8" bolts; 18" wrench on 3/4" bolts; 24" wrench on 7/8" bolts; and 36" wrench on 1-1/8" bolts. On any larger sizes special torque wrenches must be used with a ratio of 1 – 4 or 1 – 6 for bolts 1-1/4" – 2" size.

Tightness of Packing Chamber:

V.15 The following new tips are being made available to all operators of Velan valves.

V.16 Packing rings used in high temperature service as received from packing manufacturers are porous and should be compressed as much as possible initially to provide a good seal.

V.17 Packing Rings of the type "John Crane 187 – I" or equivalent, (Asbestos and Inconel core) used extensively in high temperature service will remain tight for extensive periods of time only after an initial compression of each individual ring to 10,000 – 15,000 psi. This is done presently in the Velan plant for nuclear valves only.

V.18 Hydrostatic shell test pressure in the plant also conditions the rings, especially the 2 – 3 lower ones. Gland bolts however must be re-torqued after the test to maintain tightness to torque values shown in TABLE B.

TABLE B
PACKING GLAND BOLT TORQUES

SIZE OF BOLT INCHES	NUT TORQUE IN FT./LBS. ASTM DESIGNATION OF BOLT MATERIAL	
	B6, B7, B16, 630, 660 (1)	B8 (2)
1/4"	6	3
5/16"	9	5
3/8"	14	7
7/6"	22	11
1/2"	33	16
9/16"	48	22
5/8"	60	30
3/4"	100	50
7/8"	160	100
1"	250	130

(1) Min. Yield 100.000 psi (2) Yield 30.000 psi

V.19 It is also required to retorque the bolting after the initial 5 – 7 valve cycles.

V.20 **Valves packed in the plant**

All Velan nunclear valves models 1971 will have packing rings conditioned to the newly developed technology as described in V.15 – V.19.

All that is required therefore is to tighten up gland bolts before start up and re-tighten again after the initial 5 – 7 valve cycles to torque value shown in TABLE B.

V.21 **Valves packed in the field**

W 39311

Users who pack new valves in the field must follow instructions given in paragraph VI.f.1, "Repacking."

VI.e.8 Follow instructions in par. VI.e.1 – VI.e.6 for Gate Valves.

VI.f.1 **Repacking of Gate, Globe and Stop Check Valves.**

If moisture or dripping occurs around the stem or the O.D. of packing chamber which cannot eliminated by pulling down on the gland bolt nuts, to the maximum torque shown in Table B, the following points should be investigated before deciding to replace old packing:

VI.f.2 If the gland bushing has been fully taken up and is touching the bonnet, add additional packing rings.

VI.f.3 If the stem is deeply scratched or otherwise damaged from careless handling and has worn tapered or has been bent, the stem should be replaced.

VI.f.4 If the split packing rings have not been staggered properly around the stem (each consecutive ring should be staggered 120° apart so that the fourth ring installed has its split back at the starting point), the packing rings should be inserted properly.

VI.f.5 If the gland bushing is binding against the packing chamber wall or stem and does not compress the packing properly, the gland bushing has to be aligned and tightened down equally on each side. If points VI.f.2 – 5 did not remedy leakage, packing ring must be replaced.

VI.f.6 Removal of packing rings

First check if repacking can be down underline pressure *(VI.f.10)*. With packing flange and gland bushing lifted and held in position by a wire holding the bushings to handwheel after gland bolts have been removed (swing bolts, hang down), the packing rings should be removed individually by using special flexible removal tools *(Fig. 21)*. The removal tools have special hooks which screw into the packing ring. Fig 22 shows typical engagement and removal.

Removal of packing rings is a difficult and time consuming operation and care has to be taken not to scratch the stem or the walls of the packing chamber during the removal of the packing rings.

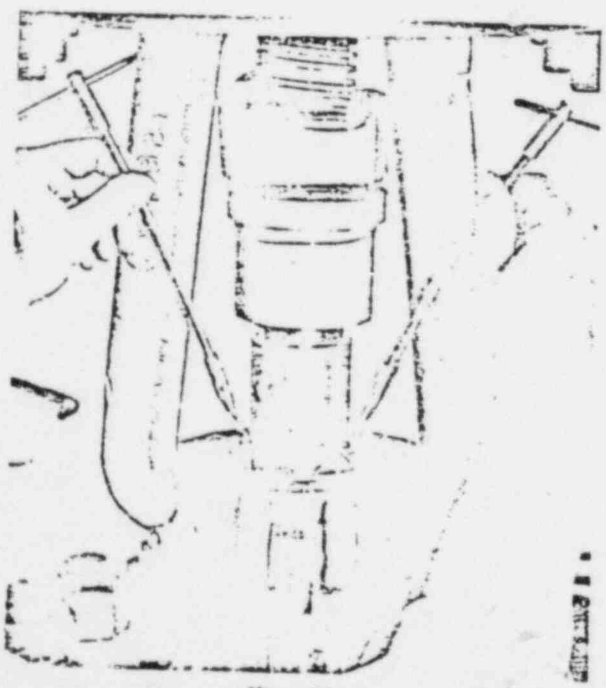


Fig. 21

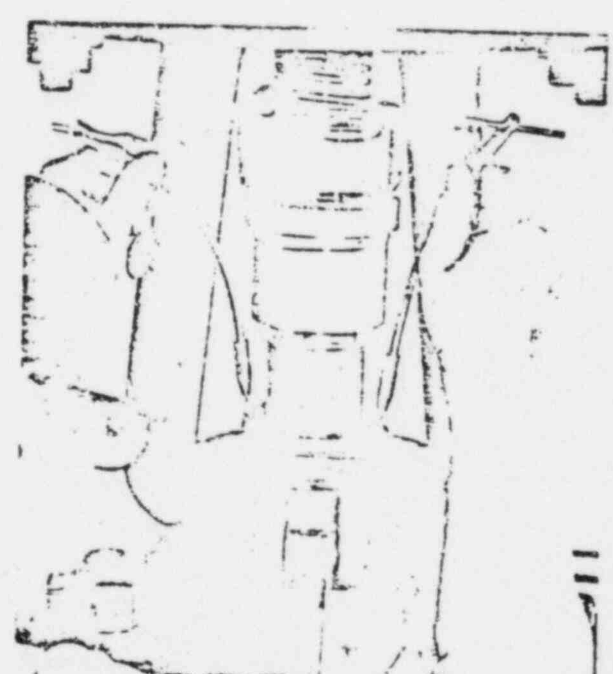


Fig. 22

W 39312

VI.f.7
In packing chambers with 2 packing sets, lantern ring and leak-off, the lantern ring can be removed with 2 wires hooked into the holes at the top or the lantern ring or with screwed in extracting wires, where tapped holes are provided. *(Fig. 23).*

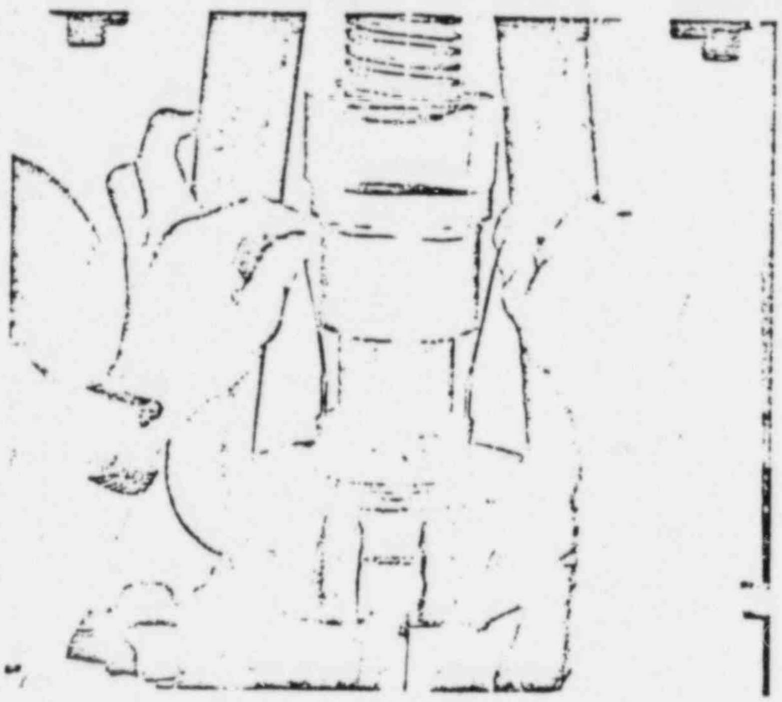


Fig. 23

VI.f.8
In designs where a special "blow out" connection is provided packing rings can be removed by applying gas or hydraulic pressure from below as shown schematically on Fig. 24.

You proceed as follows:

Removed bolting so that gland bushing and gland flange is free.

Place stem in backseat position and tighten up properly.

Attach pressure source to the "Blow out connection" and apply pressure.

All packing rings and lantern ring can be removed simultaneously.

This is a fast and efficient operation.

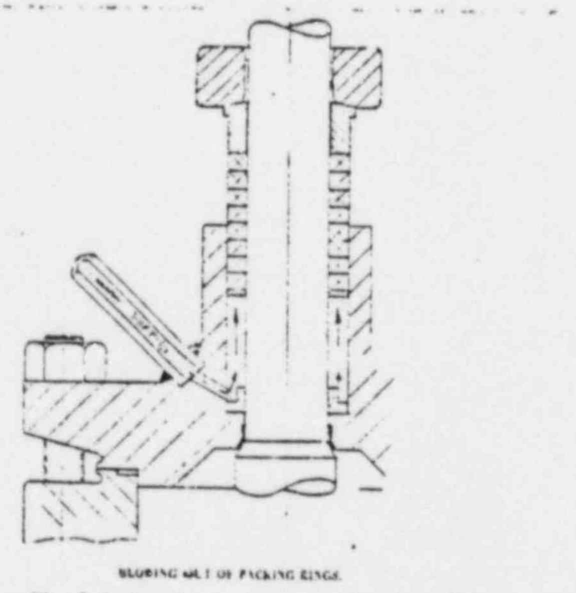


Fig. 24

VI.f.9 Repacking
Before repacking check stem and packing chamber wall for damages. Scratches no greater than .101" can be removed by polishing the surface with a fine emery cloth. Taper worn or bent stems must be replaced or if scoring, scratches and pits are deeper that .010".

In rare cases, when walls of the packing chamber are damaged by wire drawing and cannot be repaired by polishing with emery paper or special lapping the bonnet must also be replaced.

VI.f.10 It is practical to repack Velan valves under line pressure as all backseats are lapped and factory leak tested. To do this, move valve stem to backseat position and tighten up seat firmly. If, however, leakage occurs during the repacking process, line pressure must be shut off, so that repacking can proceed.

W 39313

V1.f.11 Repacking Procedure (Sequence shown on Fig. 25)

Insert first packing ring by squeezing the ring by hand as deep as possible. *(Fig. 25-1).*

V1.f.12 Insert split packing adaptors and push packing ring to the bottom of the chamber, making sure that the lap joint does not become reversed during the operation.

V1.f.13 Place gland bushing in position and compress the bottom packing by tightening the gland flange nuts in the normal manner by using maximum torque shown in Table B. *(Fig. 25-2)*

V1.f.14 Remove nuts and split packing, adpators, insert next packing ring and repeat procedure as above.

V1.f.15 Subsequent packing rings are repacked in the same manner until the point is reached whereby the special packing adaptors are not required anymore and the standard gland bushings can be used.

Split lap joints of each consecutive ring must be staggered 120° apart so that the fourth ring installed has its lap back at the starting point. *(Fig. 25-3).*

V1.f.16 In packing chambers with 2 sets of packing, lantern ring and leak-off, the lower set of packing rings must have sufficient packing height to place the lantern ring at the exact same height as the leak-off connection. This must be checked before inserting the lantern ring.

When the packing chamber is full, align carefully the gland bushing and secure gland flange with two gland bolts and nuts. Apply the maximum torque shown in Table B.

V1.f.17 Cycle valve at least 5 times from full closed to full open position, whenever possible under line pressure, and re-tighten again gland nuts to the maximum torque.

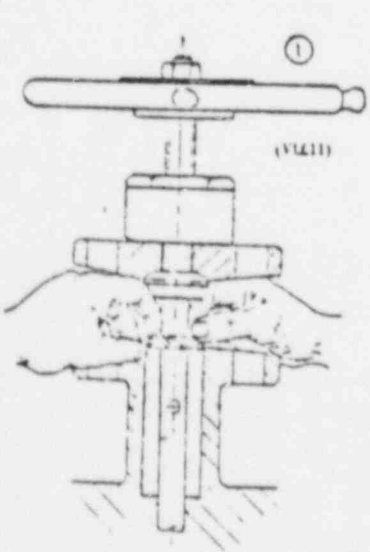
V1.f.18 WARNING:

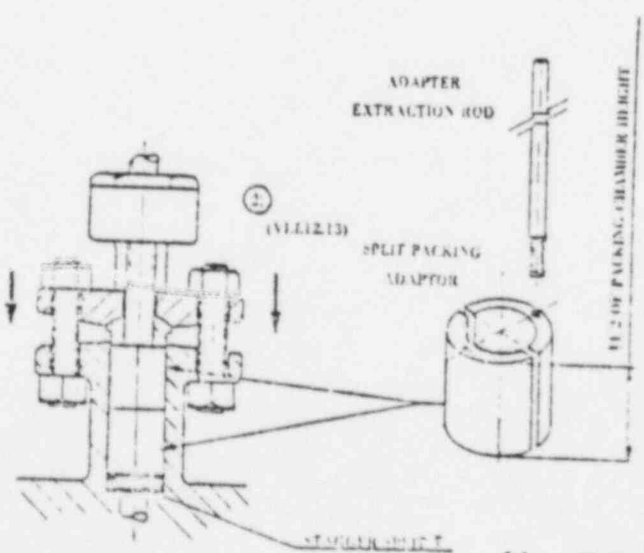
Under no circumstances should more than 1 packing ring be inserted and each packing ring must be compressed to the maximum torque (Table B).

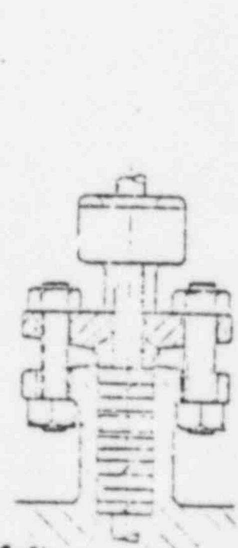
V1.f.19 Strict adherence to the above instructions will provide a tight packing gland.

V1.f.20 Remark:

Velan's R & D department is developing a better and more efficient way of repacking. As soon as satisfactory results are available, customers will be advised accordingly.







W 39314

Fig. 25-1

Fig. 25-2

Fig. 25-3

Babcock & Wilcox

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September 29, 1978

SOM-II-191

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Subject: Periodic Incore Detector Leakage Measurements

Gentlemen:

In order to reduce the inaccuracy of the NSS calculational package during cycle 1, due to potential leakage currents of the SPND's, it is recommended that periodic measurements be made.

Based upon the nominal cycle length of 421 EFPD until initial refueling, B&W suggests the following schedule for the inclusion of insulation leakage correction factors to the SPNDI subroutine:

1. Once near the beginning of the core cycle (50 ± 10 EFPD).

2. Once near the middle of the core cycle (200 ± 10 EFPD).

3. Once near the end of the core cycle (350 ± 10 EFPD).

A complete set of insulation leakage data should be taken based upon the above schedule, and all leakage correction factors equal to or greater than 1.001 should be entered into the computer.

W 39315

The aforementioned calibration rate will assure continued operation within the limits of Technical Specifications with respect to an accurate calculation of time integrated quantities.

Reactor power level should be equal to or greater than 75% Rated Thermal Power, and equilibrium xenon conditions should be maintained during the recording of data. Control rod movement should be minimal during this period to insure a stable flux distribution during the recording of data.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/JHF/bay

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SOM-II-192

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Subject: Power Imbalance Detector Correlation Test
Reference: 1. Letter, W. R. Gibson to G. R. Bond, "Report on Power Imbalance Detector Correlation Test (PSC 9-78)," dated June 21, 1978
2. Letter, L. C. Rogers to R. J. Toole, "Power Imbalance Detector Correlation Test, TP 800/18," dated August 1, 1978 (SOM-II-175)

Gentlemen:

During the course of the power escalation test sequence at the 40% plateau, a question was raised in reference to conflicting recommended values for the incore to out-of-core detector correlation slopes.

Reference (1) essentially recommended the following alternatives concerning the power imbalance detector correlation test:

<u>Alternative 1</u>:

1. Perform the normal power imbalance detector correlation test at 40% FP and set the gain such that the slope is $\geq$ 1.25.

2. Remain at this gain for the entire cycle.

W 39317

Alternative 2:

1. Perform the power imbalance detector correlation test at 40% FP and set the gain such that the slope is ≥ 1.25.

2. Reperform the test at 75% FP and reset the gain such that the slope is ≥ 1.15.

3. Each 75 EFPD thereafter - reperform the test to verify the 1.15 slope.

In contrast, Reference (2) recommended that the test procedure (800/18) be revised such that the slope would be ≥ 1.15 and that this slope would remain applicable through the entire cycle of operation.

We have reevaluated these recommendations and additional information. A special test was performed at SMUD which showed that the gain for the slope did not change through 169 EFPD of burnup thus currently concludes that push-pull plants need not verify the 1.15 slope each 75 EFPD. Therefore, Alternative 2 of Reference (1) would no longer require the 75 EFPD retest on push-pull plants. In addition, since the power imbalance detector correlation test performed at 40% will be verified at 75% before the plant achieves 100% power, it is satisfactory to set the 40% slope at 1.15 rather than the 1.25 specified in Reference (1).

In summary, the recommendation forwarded per Reference (2) is still valid for TMI-2's present power escalation testing sequence program.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

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W 39318

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October 10, 1978

SOM-II-196

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Subject: Reactor Coolant Pump Restart Interlock

Gentlemen:

B&W Engineering has reviewed the requirements for the neutron power > 22% interlock on restarting a reactor coolant pump (RCP) and has concluded that the setpoint may be increased to some higher power level-probably 50 to 60%FP. There are two reasons for increasing the setpoint. At 50%FP, both OTSG's are not on minimum level control, which allows more precise control of the transient. In addition, the temperature difference between the circulating coolant and the coolant in the idle pump is less than originally estimated, which reduces temperature effect on the neutronics in the core.

The attached figures, based on the B&W Lynchburg simulator, show that the transient of starting a fourth reactor coolant pump at 50%FP is less severe than the transient from 30%FP.

It would be advantageous to Jersey Central, TMI-2, to raise the interlock setpoint. B&W could provide a permanent change to this setpoint by confirming the simulator curves with operational data. Should Jersey Central wish to pursue an increase in the interlock setpoint, the following test should be performed to confirm simulator results:

W 39319

1. Establish reactor power between 28% and 30% with three reactor coolant pumps running
2. Start the fourth reactor coolant pump
3. Trip the fourth reactor coolant pump
4. Establish reactor power at 50 ± 2% with three reactor coolant pumps running
5. Start the fourth reactor coolant pump

Equilibrium conditions should be established at each step by taking readings on measured parameters at one (1) minute intervals until three (3) consecutive readings are within two (2) percent of each other, and temperatures are within two (2) degrees F. Record data at 0.2 second intervals on the B&W reactimeter. Forward data tape to Lynchburg for processing. Record the following parameters:

1. Reactor Coolant System pressure
2. Pressurizer level
3. Neutron power (four power range channels)
4. Loop 2 Reactor Coolant System flow
5. Loop 1 Reactor Coolant System flow
6. Loop 2 Cold Leg Temperature (Pump 1-1)
7. Loop 2 Cold Leg Temperature (Pump 1-2)
8. Loop 1 Cold Leg Temperature (Pump 2-1)
9. Loop 1 Cold Leg Temperature (Pump 2-2)
10. Loop 2 Feedwater Flow
11. Loop 1 Feedwater Flow

Prior to performing the suggested reactor coolant pump restart testing, personnel involved should be satisfied that the ICS is properly tuned in order to control the resulting transient.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
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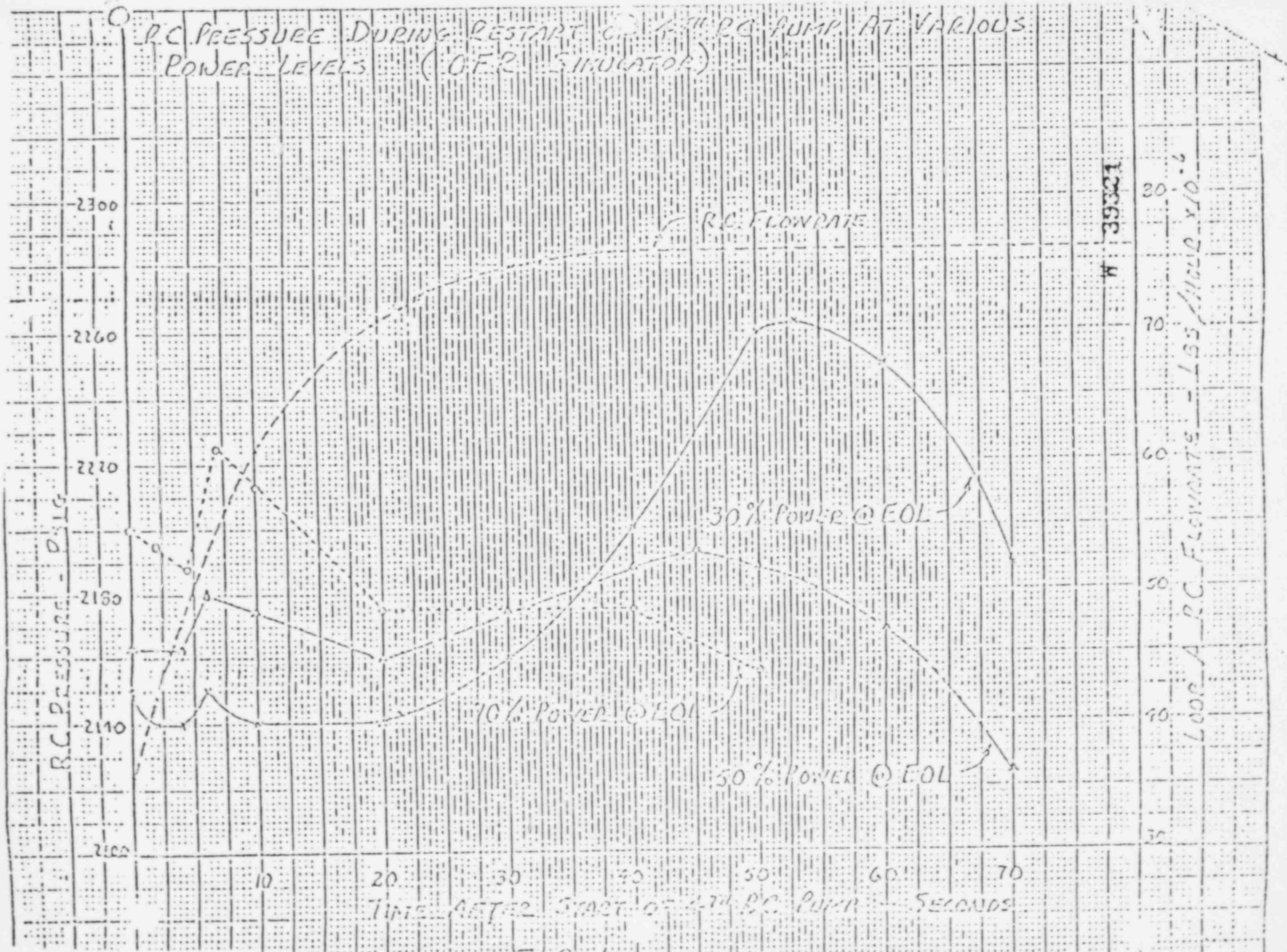
W 39320


R.C. PRESSURE DURING RESTART OF 4TH R.C. PUMP AT VARIOUS
POWER LEVELS (OFR SIMULATOR)
R.C. PRESSURE - PSIG
2300
2260
2220
2180
2140
2100
R.C. FLOWRATE
30% POWER @ EOL
70% POWER @ EOL
50% POWER @ EOL
10
20
30
40
50
60
70
TIME AFTER START OF 4TH R.C. PUMP - SECONDS
80
70
60
50
40
30
LOOP A R.C. FLOWRATE - LBS/HOUR x10^-6
W 39321

NEUTRON POWER DURING RE[illegible] OF [illegible] RC PUMP AT VARIOUS POWER LEVELS (OFR SIMULATOR)

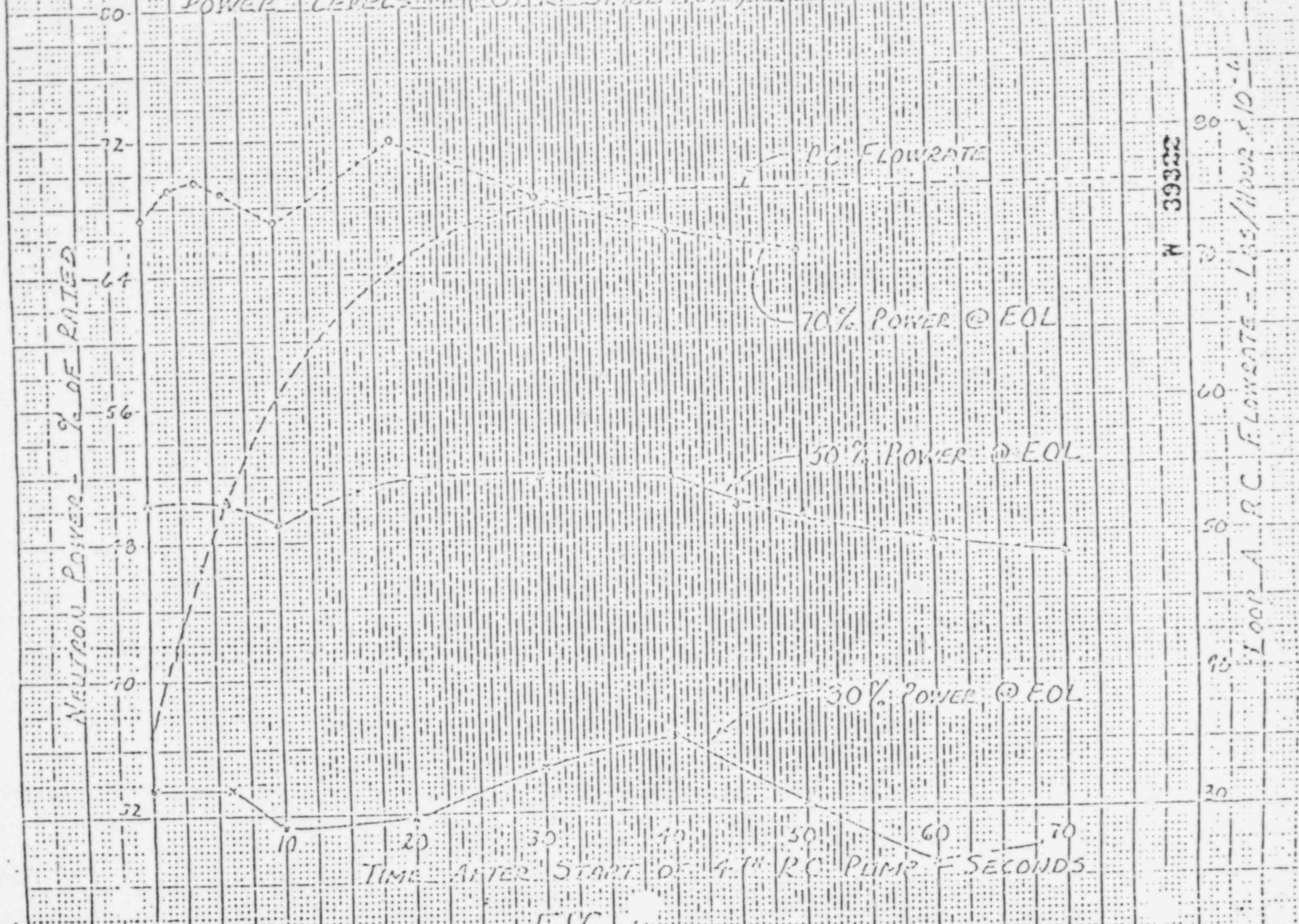
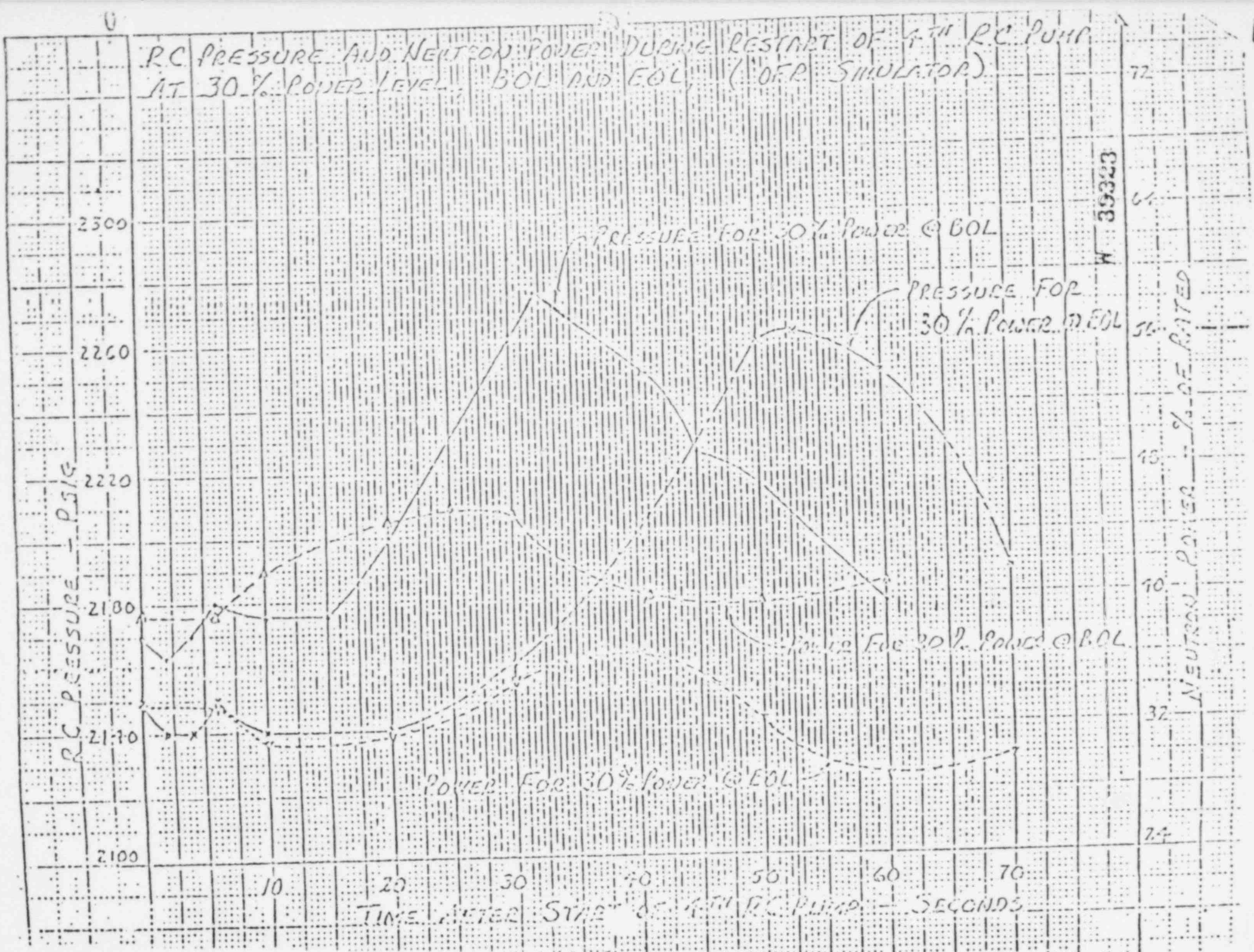


FIG 2



RC PRESSURE AND NEUTRON POWER DURING RESTART OF 4TH RC PUMP
AT 30 % POWER LEVEL BOL AND EOL (OFP SIMULATOR)
PRESSURE FOR 30 % POWER @ BOL
PRESSURE FOR 30 % POWER @ EOL
POWER FOR 30 % POWER @ EOL
RC PRESSURE — PSIG
NEUTRON POWER — % OF RATED
2300
2260
2220
2180
2100
10
20
30
40
50
60
70
SECONDS
W 39323

R.C. PRESSURE AND NEUTRON POWER DURING RESTART OF 4TH R.C PUMP AT 70 % POWER LEVEL, BOL AND EOL (AOFR SIMULATOR)

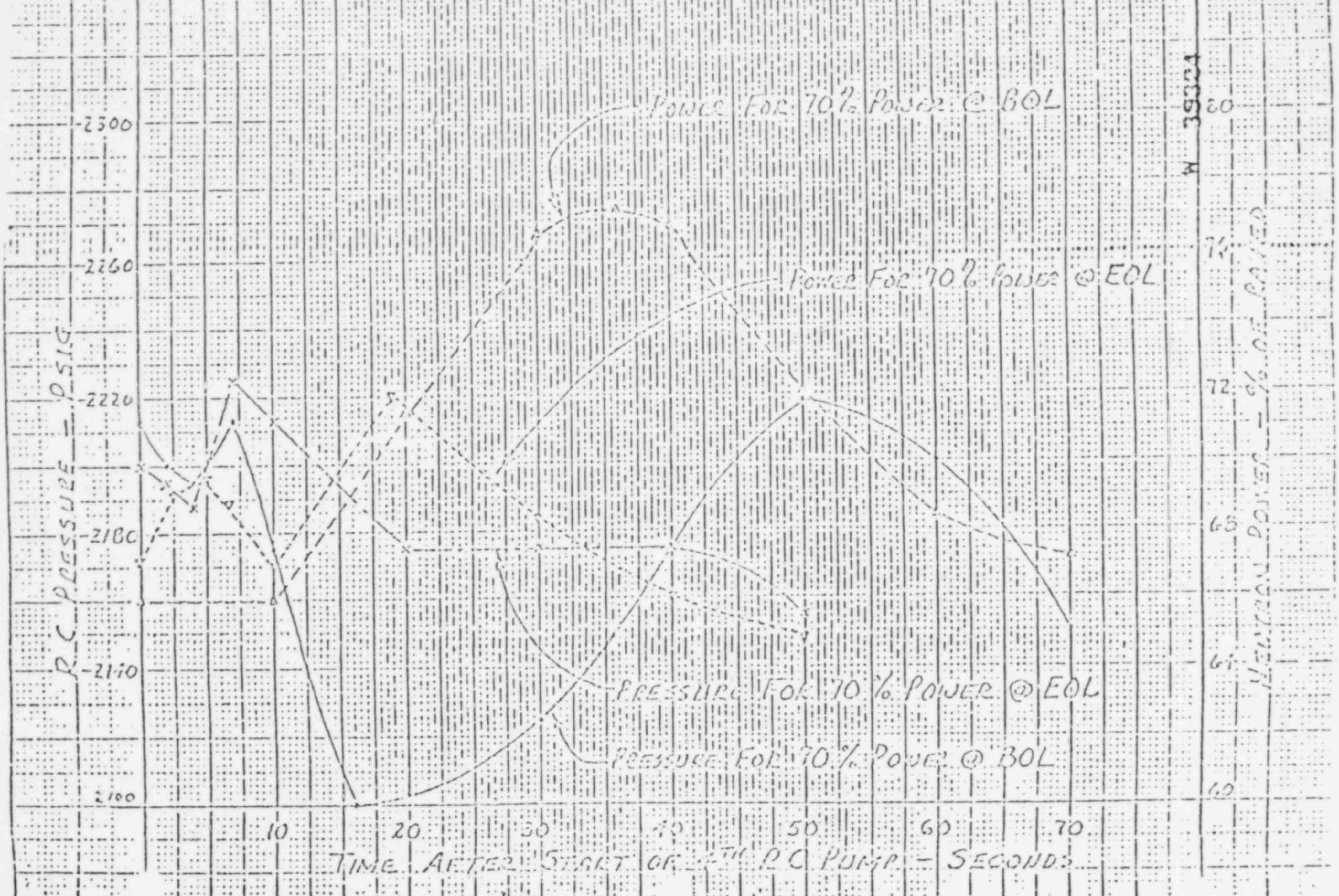


FIG. 4

6+0e2

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October 19, 1978

SOM-II-196
Revision 1

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Subject: Reactor Coolant Pump Restart Interlock
Reference: TMI-2 FSAR, Section 15.1.6

Gentlemen:

This revision completely replaces the original SOM-II-196.

B&W Engineering has reviewed the requirements for the neutron power > 22% interlock on restarting a reactor coolant pump (RCP) and has concluded that the setpoint may be increased to some higher power level - probably 50 to 60%FP. There are two reasons for increasing the setpoint. At 50%FP, both OTSG's are not on minimum level control, which allows more precise control of the transient. In addition, the temperature difference between the circulating coolant and the coolant in the idle pump is less than originally estimated, which reduces temperature effect on the neutronics in the core.

The attached figures, based on the B&W Lynchburg simulator, show that the transient of starting a fourth reactor coolant pump at 50%FP is less severe than the transient from 30%FP.

W 39325

It would be advantageous to Jersey Central, TMI-2, to raise the interlock setpoint. B&W could provide a permanent change to this setpoint by confirming the simulator curves with operational data. Should Jersey Central wish to pursue an increase in the interlock setpoint, the following test should be performed to confirm simulator results:

1. Establish reactor power between 28% and 30% with three reactor coolant pumps running
2. Start the fourth reactor coolant pump
3. Trip the fourth reactor coolant pump
4. Establish reactor power at 50 ± 2% with three reactor coolant pumps running
5. Start the fourth reactor coolant pump

Equilibrium conditions should be established at each step by taking readings on measured parameters at one (1) minute intervals until three (3) consecutive readings are within two (2) percent of each other, and temperatures are within two (2) degrees F. Record data at 0.2 second intervals on the B&W reactimeter. Forward data tape to Lynchburg for processing. Record the following parameters:

1. Reactor Coolant System pressure
2. Pressurizer level
3. Neutron power (four power range channels)
4. Loop A Reactor Coolant System flow
5. Loop B Reactor Coolant System flow
6. Loop A Cold Leg Temperature (Pump 1-1)
7. Loop A Cold Leg Temperature (Pump 1-2)
8. Loop B Cold Leg Temperature (Pump 2-1)
9. Loop B Cold Leg Temperature (Pump 2-2)
10. Loop A Feedwater Flow
11. Loop B Feedwater Flow

Prior to performing the suggested reactor coolant pump restart testing, personnel involved should be satisfied that the ICS is properly tuned in order to control the resulting transient.

B&W Safety Analysis has reviewed this analysis of reactor coolant pump starts at power. Reference (1) shows that the transient from two reactor coolant pumps started at 60% FP does not violate the safety of the plant. The two-pump start transient analysis is conservative relative to a single pump start.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

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Site Operations Manager

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R.C PRESSURE DURING ...
POWER LEVELS (OFR SIMULATOR)

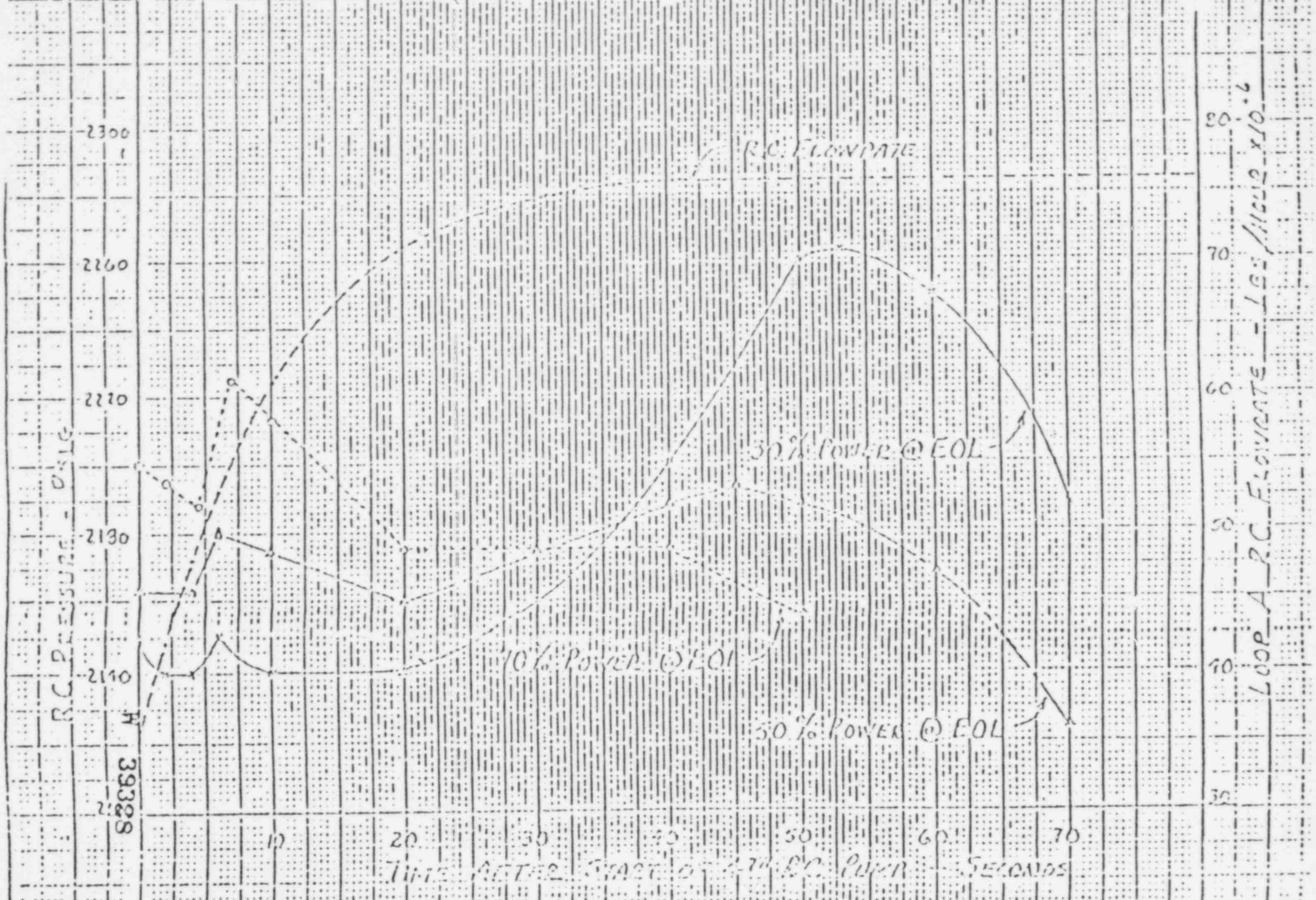
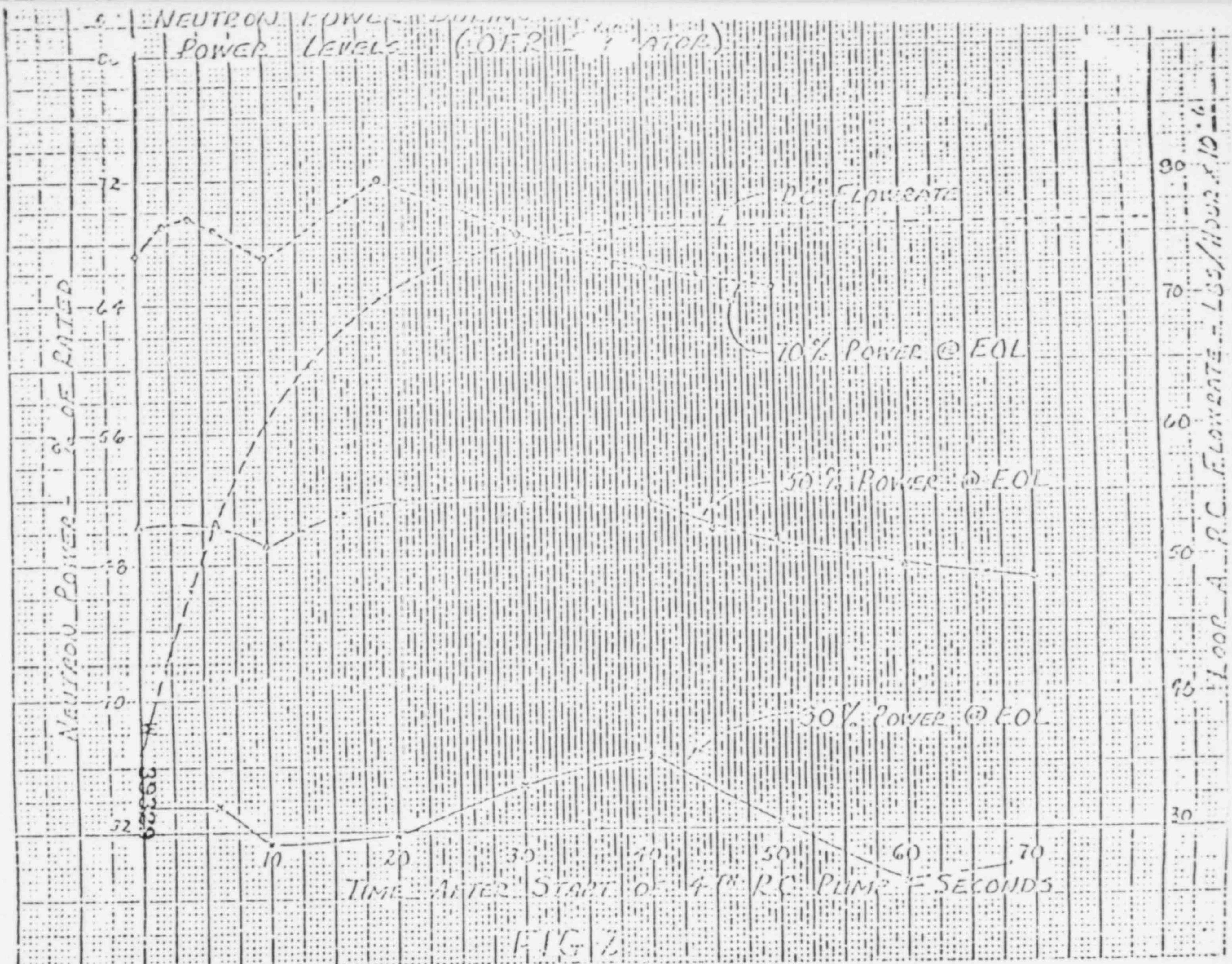
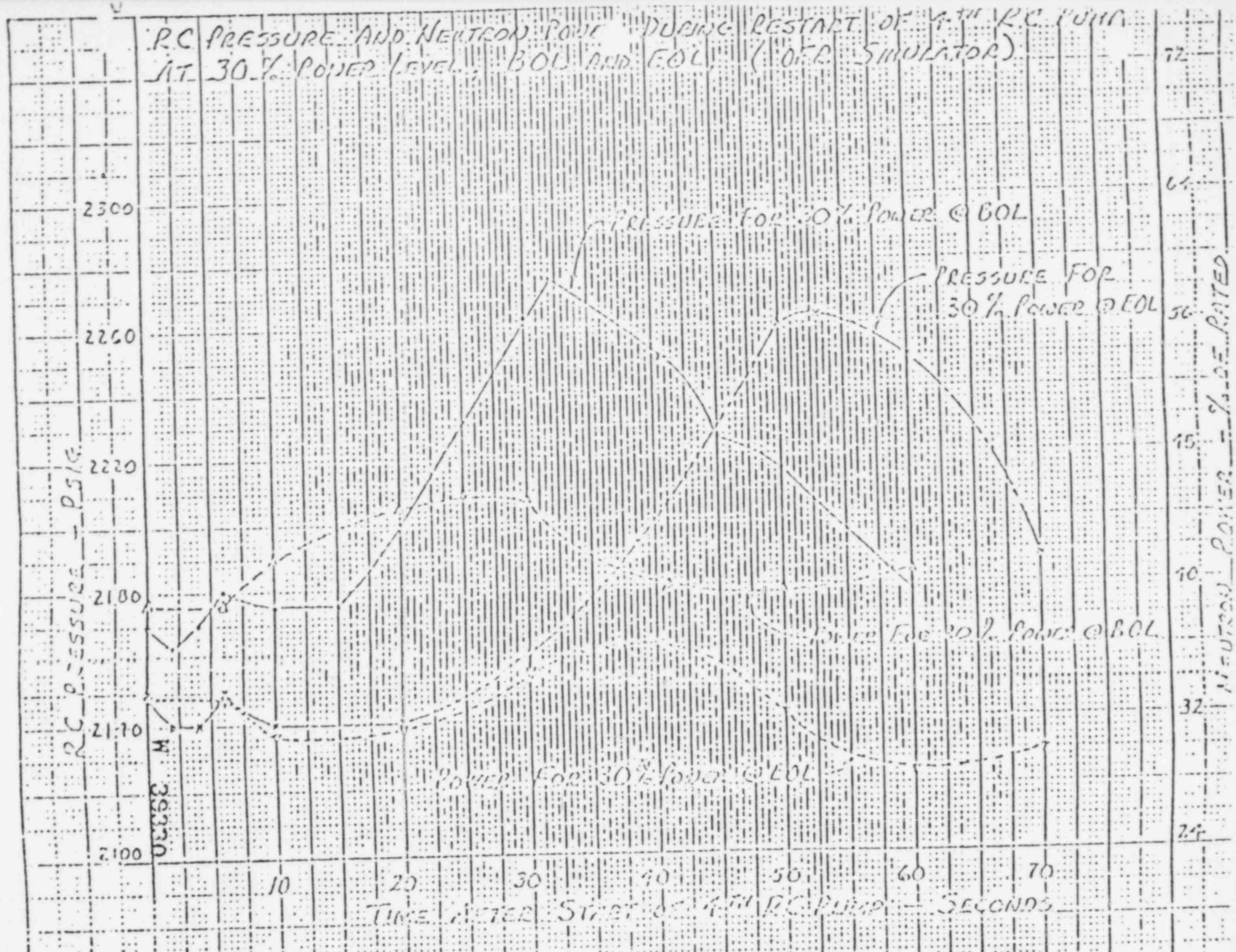


FIG 1



Power Levels
RC Flowrate
70% Power @ EOL
50% Power @ EOL
30% Power @ EOL
Neutron Power - % of Rated
Loop A RC Flowrate - Lbs/Hour x 10^-6
Time After Start of 4th RC Pump - Seconds
80
72
64
56
48
40
32
10
20
30
40
50
60
70
90
39329

FIG 2


R.C. PRESSURE AND NEUTRON POWER DURING RESTART OF 4TH R.C. PUMP
AT 30 % POWER LEVEL, BOL AND EOL (OFR SIMULATOR)
PRESSURE FOR 30 % POWER @ BOL
PRESSURE FOR 30 % POWER @ EOL
POWER FOR 30 % POWER @ BOL
POWER FOR 30 % POWER @ EOL
2300
2260
2220
2180
2140
2100
R.C. PRESSURE — PSIG
72
64
56
48
40
32
24
NEUTRON POWER — % OF RATED
10
20
30
40
50
60
70
TIME AFTER START OF 4TH R.C. PUMP — SECONDS
39330
FIG. 3

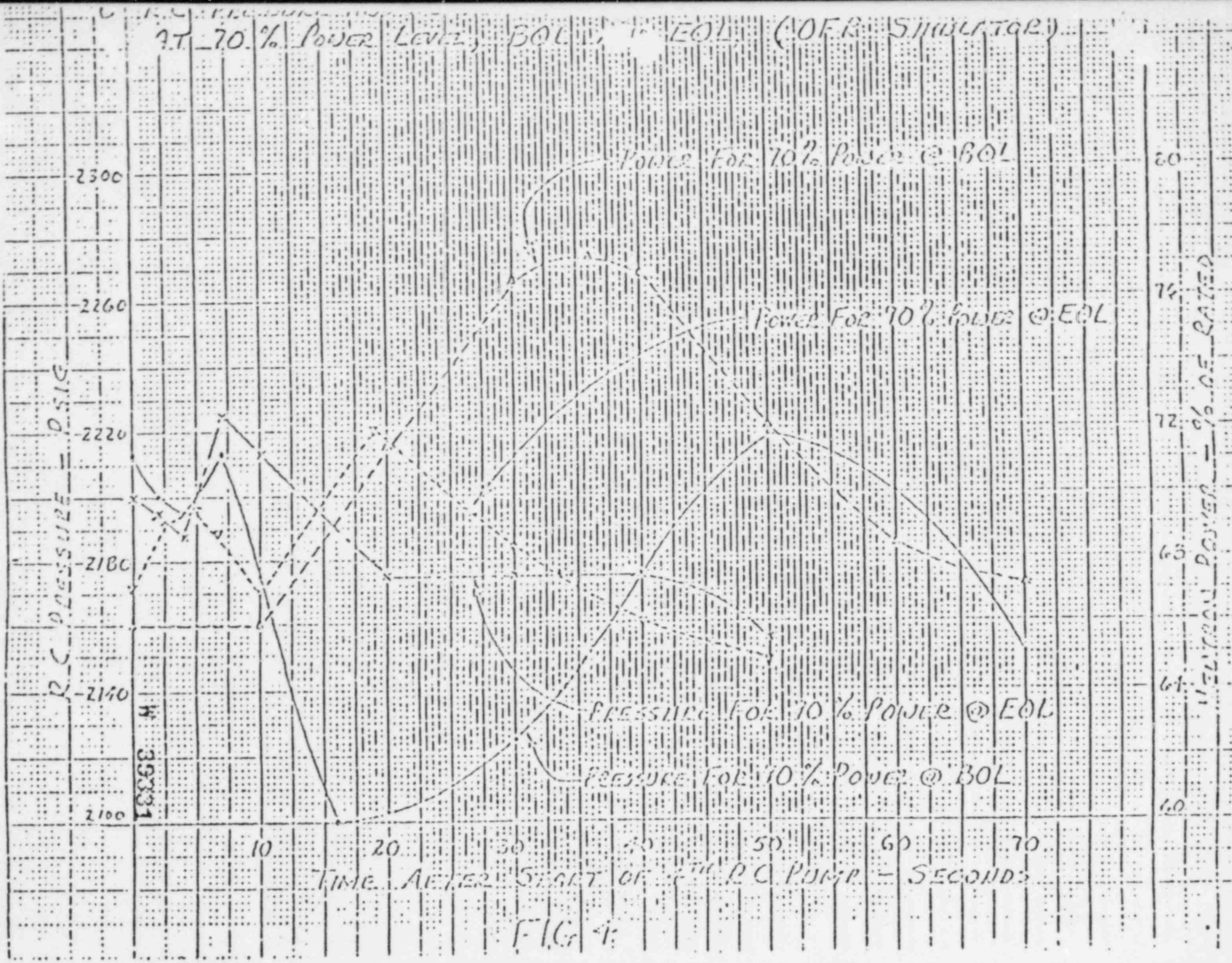

AT 70 % POWER LEVEL, BOL & EOL (OFR SIMULATOR)
POWER FOR 70% POWER @ BOL
POWER FOR 70% POWER @ EOL
PRESSURE FOR 70% POWER @ EOL
PRESSURE FOR 70% POWER @ BOL
R C PRESSURE — PSIG
2300
2260
2220
2180
2140
2100
NEUTRON POWER — % OF RATED
80
76
72
68
64
60
10
20
30
40
50
60
70
TIME AFTER START OF 4TH R C PUMP — SECONDS
39331

FIG 4

8410 B...

Babcock & Wilcox

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October 30, 1978

SOM-II-198

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Subject: All Rods Out Boron Prediction

Gentlemen:

B&W has reviewed the as-built fuel data sheets for differences between the SMUD and TMI-2 fuel. As a result, a new all rods out, HZP critical boron concentration, was derived as follows:

1553 ppmb HZP, ARO (SMUD measured)
\- 12 ppmb Gadolinium presence in 4 FA
\- 18 ppmb Fuel Density & Enrichment Differences
1523 ppmb Best estimate for TMI-2 HZP, ARO

This value should replace the current value of 1566 in the Physics Test Manual.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39332

LCR/JV/bay

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October 10, 1978

SOM-II-197

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Subject: Four Pump Flow Acceptance Criteria

Gentlemen:

The acceptance criteria for the four pump flow coastdown test is that the error adjusted measured flow, as a function of time, must be greater than that shown by the curve on the attached figure. The steady state, error adjusted measured flow must also be greater than the steady state value shown at t = 0 on the attached figure (W = 376,640 gpm).

The error adjusted measured flow is measured using unsnubbered ΔP transmitters and is reduced by 2.5% from the best estimate of measured flow. The use of 2.5% error adjustment is related specifically to using the unsnubbered ΔP data. The steady state flow should be computed using the average ΔP from 10 seconds of data (0.2 seconds recording frequency). The steady state flow is also error adjusted by reducing the best estimate by 2.5%.

The snubbers should be set at #4 following the coastdown test.

If you have any further questions, please do not hesitate to contact me.

W 39334

Very truly yours,

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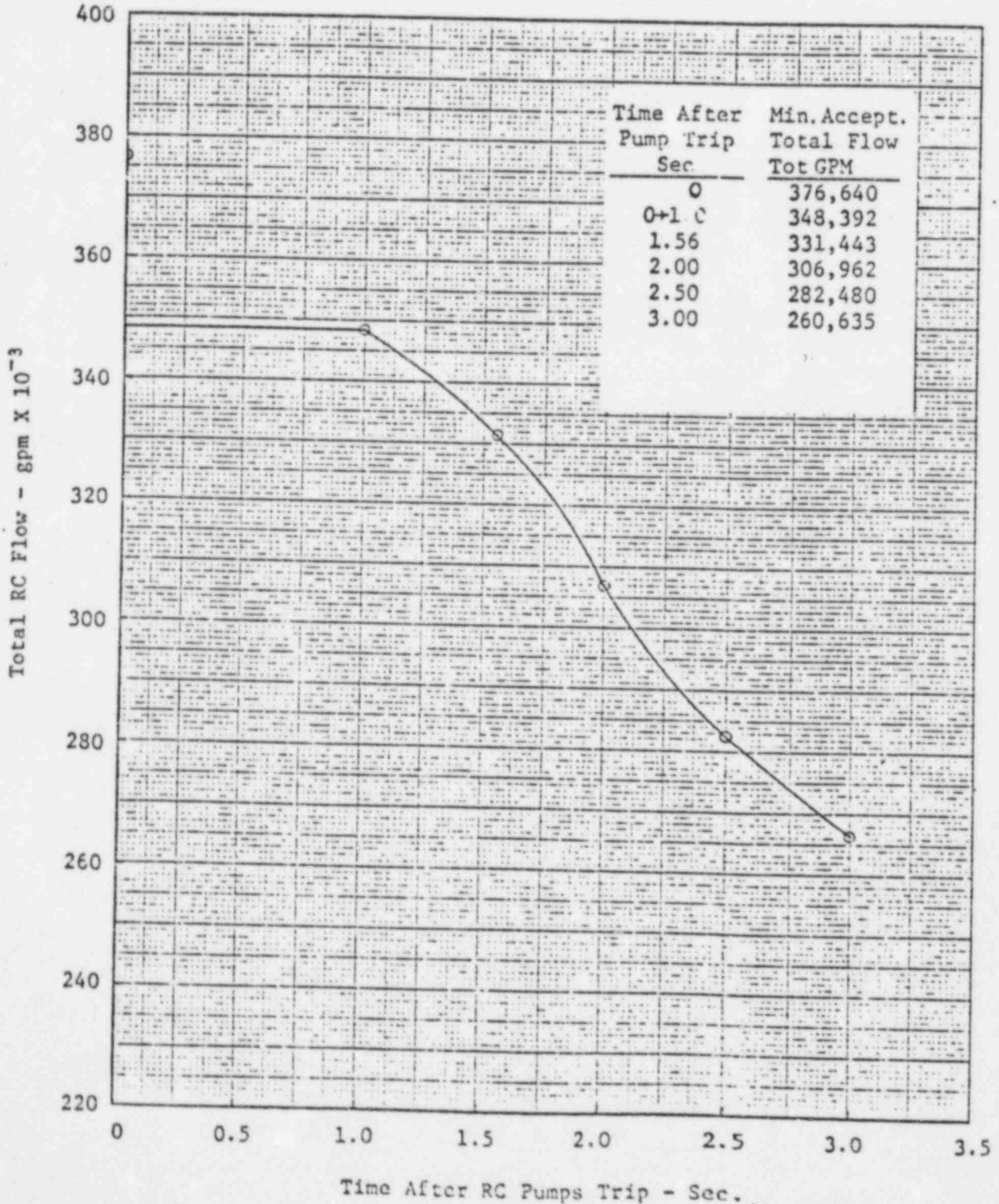
TMI-2, Cycle 1
Four Pump Flow Coastdown
Acceptance Criteria
(orifice rods removed, unsnubbered values)



Time After Pump Trip Sec	Min. Accept. Total Flow Tot GPM
0	376,640
0+1.0	348,392
1.56	331,443
2.00	306,962
2.50	282,480
3.00	260,635

W 39336

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November 6, 1978

SOM-II-201

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Mr. G. P. Miller
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Subject: Unit Acceptance Test

Gentlemen:

The TMI-2 startup testing program has proceeded with relative success to the 90% testing plateau. Upon achieving this power plateau, the warranted output described in the Nuclear Steam Supply System contract has been delivered from the Babcock & Wilcox supplied systems to the balance of plant systems.

In accordance with the terms of the contract between Babcock & Wilcox Co. and Jersey Central Power & Light Co., Babcock & Wilcox is so notifying Jersey Central Power & Light (and its agents) that the B&W supplied equipment is declared ready for performance tests for the warranted output. The test to qualify the warranted output is the appropriate sections of the Unit Acceptance Test, TP 800/17.

This test performance should be accomplished as soon as practical, compatible with existing policies and procedures. If for any reason you do not intend to perform the aforementioned test within a reasonable time, I specifically request that you inform me in writing of any such reason or reasons.

W 39337

R. J. Toole
L. L. Lawyer
G. P. Miller

The B&W Site Manager or his appointed representatives shall have access to all applicable test records, including a complete copy of the test data and associated results.

Your published current schedule for the TMI-2 startup sequence shows TP 800/17 to be performed after the Turbine Generator and Secondary Plant Screen Outage following operation and other testing at the 100% power plateau.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
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November 21, 1978

SOM-II-202

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[redacted]
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
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Subject: Power Doppler Coefficient of Reactivity

Gentlemen:

The power doppler coefficient limit for TMI-2 has been recalculated using assumptions consistent with the FSAR. The new acceptance criteria is that the measured value must be more negative than -0.688×10^{-4} ΔK/K/%FP.

The power doppler measurements to be done at 90% and 100% FP during TMI-2 power escalation testing shall also utilize the power range NI's as input to the reactimeter.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/RL/bay

W 39339

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November 22, 1978

SOM-II-203

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Subject: Hydrogen Addition to the Makeup Tank

Gentlemen:

Due to the problems in adding hydrogen to the makeup tank, B&W was asked to look at the operation of the system. The following problems associated with the hydrogen addition system were noted:

It was attempted to add hydrogen on 1 November 1978. Pressure did not increase in the makeup tank. A check of the hydrogen manifold revealed no upstream pressure on the regulator, and downstream pressure was the same as the makeup tank pressure. A fitting near the hydrogen bottle allowed hydrogen leakage to the atmosphere. This fitting has been repaired with electrical tape and will need a proper permanent fix.

Pressure downstream of the regulator bled down to zero over approximately six (6) hours. This indicates, (1) leakage in the system, possibly downsteam of the regulator, and (2) backflow in the system past the check valve (MU-V171).

W 39340

The system was operated in such a manner as to require the hydrogen regulators to act as non-leakage isolation valves for the hydrogen bottles with no flow in the system. This was not a design feature of the B&W supplied regulators (MU-V29A&B - B&W) HY-V44A&B.

Based on the above observations, the following recommendations are made:

(1) The operating procedure, 2103-1.5, should be modified to place HY-V44A&B in service only during hydrogen addition.

(2) Repair the electrical taped fitting located between the hydrogen bottle isolation valves and the regulators (work request submitted).

(3) If the system is to be operated according to procedure 2103-1.5, replace HY-V44A&B with regulators designed for no flow operation. B&W is able to supply such regulators for the hydrogen addition system.

(4) A method to ensure sufficient hydrogen pressure in the bottles prior to attempting to add hydrogen to the makeup tank. B&W suggests the procedure change mentioned above include steps to ensure sufficient hydrogen bottle pressure. Alternatives are electrical interlocks with MU-V28 and HY-V55 for low bottle pressure preventing their operation or pressure indication in the control room of hydrogen bottle pressure.

(5) Filter HY-U9 should be checked to ensure that boron crystals have not been backflushed into the system.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E B Schield
for

L. C. Rogers
Site Operations Manager

LCR/LAP/bay

cc: J. G. Herbein
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W 39341

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November 29, 1978

SOM-II-204

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Subject: Flow Value Used in Safety Analysis
Reference: Justification for Removal of Orifice Rod Assemblies in Three Mile Island Unit 2, Cycle 1, BAW 1497, June 1978

Gentlemen:

Table 3.2-2 of the Technical Specifications for TMI Unit 2 defines limiting conditions for operation such that the margin to DNB (departure from nucleate boiling) will be equivalent to that assumed in the transient and accident analyses. The transient analyses performed in support of BAW 1497 (referenced above), which were the basis for a change in the minimum required Reactor Coolant System flowrate from 369,600 gpm to 377,000 gpm, assumed a Reactor Coolant System flowrate of 377,000 gpm with no allowance for instrument inaccuracy. Therefore, the measured flowrate which is used to demonstrate that this requirement is met must be 377,000 gpm plus the minimum justifiable flow measurement error.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

for
L. C. Rogers
Site Operations Manager

LCR/JV/bay

W 39342

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November 29, 1978

SOM-II-205

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~~Mr. G. P. Miller~~
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
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Subject: Reactor Coolant System Chloride Contamination Evaluation

Gentlemen:

It is the opinion of B&W (NPGD Technical Staff) that the high chloride contamination of the Reactor Coolant System will have no deleterious effect upon the structural integrity of the Reactor Coolant System or associated auxiliary systems and equipment. Therefore, the Reactor Coolant System remains acceptable for continued operation (heatup and startup). This instruction constitutes the required engineering evaluation in accordance with Plant Technical Specification 3.4.7 (Page 3/4 4-17).

This evaluation is based on the conditions bounded as follows:

Chlorides	1.9 ppm (maximum)
Oxygen	Zero ppb

and the Reactor Coolant System at Hot Standby Conditions.

This evaluation is specifically based on the presence of low oxygen levels during the out-of-specification chloride condition and the fact that the chlorides were reduced to below the steady-state limit within the time period allotted by Plant Technical Specifications.

W 39344

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E G Schaedel
for
L. C. Rogers
Site Operations Manager

LCR/EGS/bay

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December 5, 1978

SOM-II-206
REM-I-369

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Subject: Retorquing Primary and Secondary System Bolted Closures

Gentlemen:

Attached are the latest recommendations for retorquing the primary and secondary system bolted closures (manways and handhole covers, etc.). These recommendations also address new installation components that have not been field hydrotested. These sections (Section 2 and 3) do not apply to either Unit I or Unit II.

These recommendations are provided to help minimize the problems of leaking closures and at the same time will help reduce personnel exposure by revising the relubrication requirement which is currently found in the OTSG Instruction Manual. These recommendations should be included with the current instructiors of the component instruction manual.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

W 39346

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R. E. Sieglitz
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J. T. Janis

RECOMMENDATIONS FOR RETORQUING PRIMARY AND SECONDARY BOLTED CLOSURES

These recommendations are based on B&W shop hydrostatic test experience and are intended to help minimize problems of leaking bolted closures. These recommendations apply to the following situations:

A. New Installation
B. After Installing New Gaskets
C. After Installing New Studs

1. Initial torquing of bolted closures should be performed in accordance with the procedure outlined in the B&W component manual.

2. During the initial field hydrostatic test of the OTSG (secondary hydro), recheck the final torque when the system reaches hydro temperature (approximately 120° - 150°F) but prior to any increase in system pressure. If necessary retorque in sequence until nuts do not turn at the final torque value.

3. Following the initial field hydrostatic test of the RCS (primary hydro) and OTSG (secondary hydro) recheck the final torque. If necessary, retorque in sequence until nuts do not turn at the final torque value. To minimize personnel safety hazard, ensure system is not pressurized.

4. After the first pressurized heatup and cooldown cycle following gasket or stud replacement recheck the final torque value. If necessary, retorque in sequence until nuts do not turn at the final torque, torque value. If there is any indication of insufficient lubricant on the studs, remove one nut at a time, lubricate, replace the nut and retorque to the final torque value as described above. If studs are adequately lubricated, the nuts need not be removed for relubrication.

W 39348

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December 15, 1978

SOM-II-207

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Subject: Hydrogen Supply to the Makeup Tank
Reference: SOM-II-103, "Hydrogen Addition to the Makeup Tank," dated November 22, 1978

Gentlemen:

The work request to repair a fitting at the hydrogen bottle manifold and to install a full hydrogen bottle was completed and the system put into operation. Hydrogen was determined to be going into the makeup tank but at a very slow rate. At that time, it was suspected that blockage in the 1/2" line was the cause of the low flow rate.

A work request was submitted to clean the filter (HY-V9). This filter was replaced on 13 December 1978. On 14 December 1978, the system was put into operation and again the flow rate of hydrogen was very low. Adjustment of the regulators (HY-V44A and B) and corresponding pressures observed on the PI's at the regulator and MU-PI-20 indicated that the blockage was between the regulator and the filter (HY-V46; 30; and 31 were verified open). Based on these observations of the system, it is recommended that work requests be initiated for the following items.

W 39349

1. HY-V33 packing leak - This leakage necessitates isolation of the hydrogen bottle.

2. Both regulators, HY-V44A and B, should be removed and inspected for cleanliness; also inspect the inlet screen. If no obvious dirt or foreign material is found that would cause the regulators to leak through, then the regulators will be returned to the supplier for rebuilding. Please contact E. G. Schaedel (extension 237) when the regulators are removed for inspection.

3. HY-PS-1473 and HY-PS-1474 are lo pressure and hi pressure switches with 12# and 55# setpoints respectively. Neither alarm in the control room actuated (Panel 8 Column 16) when the system pressure was below 10# or at 75#. MU-PI-20 pressure gauge did respond to pressure changes but did not appear to go to zero. All three items should be recalibrated.

4. The 1/2" pipe, from the regulators to the filter, should be flushed to insure that it is not plugged and will allow sufficient flow.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS

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W 39350

B&W BK

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December 19, 1978

SOM-II-208
REM-I-371

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Subject: FHE Site Caution Instructions

Gentlemen:

The FHE Site Caution Instructions listed here are supplied by Stearns-Roger and are forwarded for your use:

Attachments:
1. SCI #1 Dillon Calibration
2. SCI #2 Hydraulic Relief Valve Setting Procedure
3. SCI #3 Procedure for Field Repair of Dual Hose Reel
4. SCI #4 Fuel Grapple Disengage Bar
5. SCI #6 Emergency Pull Out Cable

It is suggested that you consolidate all SCI's into a section in the front of the Stearns-Roger FHE Instruction Manual. In addition to this section, SCI's concerning other particular sections of the FHE manual should also be inserted in these sections. In regard to the SCI associated with this transmittal, it is recommended that the following SCI's be included in the following additional sections of the manual:

SCI #1 Dillon Load Cell Section
SCI #2 Procon Pump Co. Section
SCI #3 Aeromotive Section

W 39351

Dup

In regard to SCI #1 (Dillon Calibration), the current recommendation to determine the settings for LS-3 (for use in Steps 21, 22, and 23) and LS-4 (for use in Steps 24, 25, and 26) is as follows:

Determine by an actual measurement the moving weight of the fuel assembly without any control components for both up and down directions in the core region.

Set LS-3 (underload trip) at 225 poinds below the minimum downward moving weight recorded in the core region.

Set LS-4 (overload trip) at 325 pounds above the maximum upward weight recorded in the core region.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/LNM/bay

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Issued 10/30/78 Rev. 0	[illegible]	BSW No. 01

[illegible]

SITE CAUTION INST.

W 39353

Dupe

DILLON CALIBRATION - (Wet)

1. Power switch "off".

2. Remove power cable from Dillon load cell.

3. Meter should read "0".

4. Adjust white adjusting screw on the center of the readout meter until meter pointer rests on "0".

5. Turn power switch "on", meter should still show "0". If not on "0" readjust to "0" using the white adjusting screw.

6. Push "calibrate" button, meter should read 3000#.

7. If meter does not read 3000#, adjust needle to 3000# using "tare adj." screw on cabinet.

8. Release "calibrate" button, meter should read "0".

9. If meter does not return to "0", repeat steps 4, 6 & 7 until both settings are stable and balanced.

10. Turn power switch "off".

11. Connect power cable to Dillon load cell.

12. Turn power switch "on".

13. Lower inner mast 12" to 18" using "jog down" switch.

14. Raise inner mast using "jog up" switch, while hoist is running up meter should read 750#. If meter does not read 750#, adjust tare to 750# while hoist is raising.

NOTE: All limit switches must be set with the needle traveling in the direction shown.

15. LS-1 Lowload - Descending -

Rotate "tare" slowly counter clockwise, low load lite blue should illuminate as needle on readout reaches 400#, traveling toward "0".

16. If low load lite comes on above 400#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw counter clockwise on LS-1. Then repeat Step 15. If required, repeat Step 16 or 17 whichever is appropriate.

W 39354

17. If low load lite comes on below 400#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw first clockwise and then part way back counter clockwise. Then repeat Step 15. If required, repeat Step 16 or 17, whichever is appropriate.

18. LS-2 Engage - Disengage Interlock - Ascending -

Rotate "tare" slowly clockwise, the switch should open as the needle on the readout reaches 1200# in the ascending direction.

19. If the meter opens below 1200#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw clockwise on LS-2. Then repeat Step 18. If required, repeat Step 19 or 20 whichever is appropriate.

20. If the meter opens above 1200#, slowly turn the tare back to 750#, then slowly and minutely turn the adjusting screw counter clockwise on LS-2. Then repeat Step 18. If required, repeat Step 19 or 20, whichever is appropriate.

21. LS-3 Underload ________# - Descending -

Rotate "tare" slowly clockwise, till needle reads 2500#.

Place voltmeter lead on Wire 226 on terminal block in console and on Wire Y on control transformer in motor control center. Meter should read 120V. AC. Then rotate "tare" slowly counter clockwise. The limit switch should open as the needle on the readout reaches ________# in the descending direction.

22. If the meter opens above ____#, slowly turn tare back to 2500#, then slowly and minutely turn the adjusting screw counter clockwise on LS-3. Then repeat Step 20. If required, repeat Step 21 or 22, whichever is appropriate.

23. If the meter opens below ____#, slowly turn tare back to 2500#, then slowly and minutely turn the adjusting screw counter clockwise on LS-3. Then repeat Step 20. If required, repeat Step 22 or 23, whichever is appropriate.

24. LS-4 Overload ________# - Ascending -

Rotate tare slowly clockwise, the "red" overload indicating light should illuminate as the needle on the readout reaches ________#, traveling toward 3000#.

25. If the overload light comes on below ________#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw clockwise on LS-4. Then repeat Step 24. If required, repeat Step 25 or 26 whichever is appropriate.

26. If the overload light comes on above ________#, slowly turn tare back to 750#, then slowly and minutely turn the adjusting screw counter clockwise on LS-4. Then repeat Step 24. If required, repeat Step 25 or 26, whichever is appropriate.

W 39355

27. Turn tare back to 750#.

28. Dillon is ready for use.

Issued 10/30/78	[illegible]	B&W
Rev. 0		No. 02

SITE CAUTION INST.

W 39356

Dupe

Procedure for Setting the Relief Valves
on the Handling Bridge and Transfer System
Hydraulic Systems.

1. The purpose of this procedure is to describe a method of adjustment which will insure a flow of water at all times through the pump. It is extremely important that the relief valve in the pump is set at a pressure of 10 to 20 psi higher than the system pressure which is controlled by the ½" external relief valve. Normally a system relief pressure of 200 psi and a pump relief pressure of 210 psi is satisfactory.

2. Remove acorn nut that locks the pump relief valve setting.

3. Remove the external relief valve adjusting screw cover.

4. Turn the external relief valve adjusting screw clockwise several turns to prevent by-passing until pump relief valve can be set to desired pressure (210 to 220 psi).

CAUTION:	Do not allow pump to by-pass internally for more than 5 minutes or excessive heat buildup will result which could damage the pump.

5. Turn the pump adjustment screw clockwise to raise or counter- clockwise to lower pressure to a setting of 210 to 220 psi.

6. Turn the external relief valve adjusting screw counter -clockwise to bring system pressure down to 200 psi.

7. Lock the pump adjusting screw by replacing the acorn nut and gasket. Tighten nut firmly.

8. Lock the external relief valve adjusting screw and replace the cover.

9. Allow the pump to run for several minutes and check temperature by feel. The pump body should be cool or slightly warm. Also the pump should not heat up even after extended periods of operation.

10. General Trouble Shooting.

    10.1 Excessive noise.

    a. Check pump mounting bolts

    b. Check for loose drive coupling.

    c. Check for improper relief valve setting by performing steps 1 thru 9.

W 39357

10.2 Excessive heat buildup in pump.

a. Check for improper relief valve setting by performing steps 1 thru 9.

10.3 System pressure will not hold at set point.

a. Check the lock nuts which lock the pump adjusting screw and the nut which locks the external relief valve.

b. Check fittings for leakage and tighten if necessary.

W 39358

ah 5cI-02

Procedure for Field Repair of Dual Hose Reel #YC1F4-C

General Information:

This procedure covers major field repair work that may be required on the Aero-Motive #YC1F4-C Dual Hose Reel. This reel is mounted on all Stearns-Roger Manipulator Cranes and located on the fuel mast connecting tube.

For major field repair work, the reel must be removed from the mast and taken to the shop for disassembly.

1. Raise fuel mast to an elevation that will allow the hoses to be disconnected from the mast thru the hand holes in the connecting tube. This elevation is usually just below the tube full up position.

2. Disconnect hoses from top of mast and allow hoses to be retracted on reel until all spring motor tension has been removed from reel.

3. Match mark and disconnect supply hoses to reel at termination of tubing on mast structure.

4. Remove reel mounting bolts and take reel to shop.

5. For spring motor repair, replacing main spring, or replacing swivel and shaft seals, see instruction sheet, page 3 and 4 of this procedure.

6. For repair or replacing of shaft assembly, see page 3, 4 and 5 of this procedure.

   Note: The mainshaft is made of stainless steel with brass fittings being silver soldered to seal 3000 psi. See Page 5.

In the event there is a leak between the mainshaft & the brass fittings, where they are silver soldered, they must be resilver soldered or the brass fittings removed, all traces of silver solder removed, both from the brass fitting and the mainshaft. Then they can be welded together. This will be a bimetal weld.

7. Replace reel on crame, mount, and connect hoses. Hoses at top of structure were match marked in Step 3. Retension spring so reel will retract properly at mast full up position.

   Estimated Time:

W 39369

Remove reel from unit	2 hours
Replace Shaft	4 - 6 hours
Repair Shaft - Add	4 hours
Install reel and test	3 hours

AERO-MOTIVE A SUBSIDIARY OF DANIEL WOODHEAD, INC.

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INSTRUCTION AND PARTS DATA SHEET YZ-6-72
PART NUMBER 1174-P-32

Y AND Z SERIES
DUAL HYDRAULIC HOSE REEL

NOTE: Before attempting repairs, remove all hydraulic pressure and spring motor tension from reel.

TO REPLACE "Z" SERIES MOTOR:

Remove (4) screws (32-P-103) and (4) washers (101-P-21) holding spring motor to mounting plate (48344). Pull motor from plate disengaging reel shaft drive pin from motor shaft.

TO REPLACE SPRINGS IN "Z" MOTOR:

Place motor on end, back plate down. Remove (6) tie bolts (44894) and nuts (77-P-1). Lift off front cover and outside motor band (40120). Pull the drive shaft, with fixed hub, up and out while holding the center of the exposed spring in position. DO NOT allow the center of the spring to be pulled out. The spring and cup assembly may now be removed and individual parts "unstacked" in order. To reassemble, reverse the above procedure being certain that all parts are replaced in order and are properly aligned.

TO REPLACE "Y" SERIES MOTOR:

Remove retaining ring (580-P-75) and keyed washer (48331-1). Pull motor assembly from shaft. Spring hub will remain on main shaft. When replacing motor, align slot in spring hub with hook on inner end of main spring. Push motor onto reel shaft while aligning the locating pin on the motor housing with the locating pin hole of the frame assembly (44396).

TO REPLACE SPRINGS IN "Y" MOTOR:

Remove motor assembly from reel. Remove (6) screws (2-P-502) and washers (101-P-50) holding cover (44394) to motor housing (44393). Remove cover from housing. Check "holding rivets" at outer end of spring (31158) to be sure they are still secure. If so, lift spring from housing. DO NOT attempt to remove spring if holding rivets are broken. Return motor to factory for servicing. To reassemble motor, reverse the above procedure.

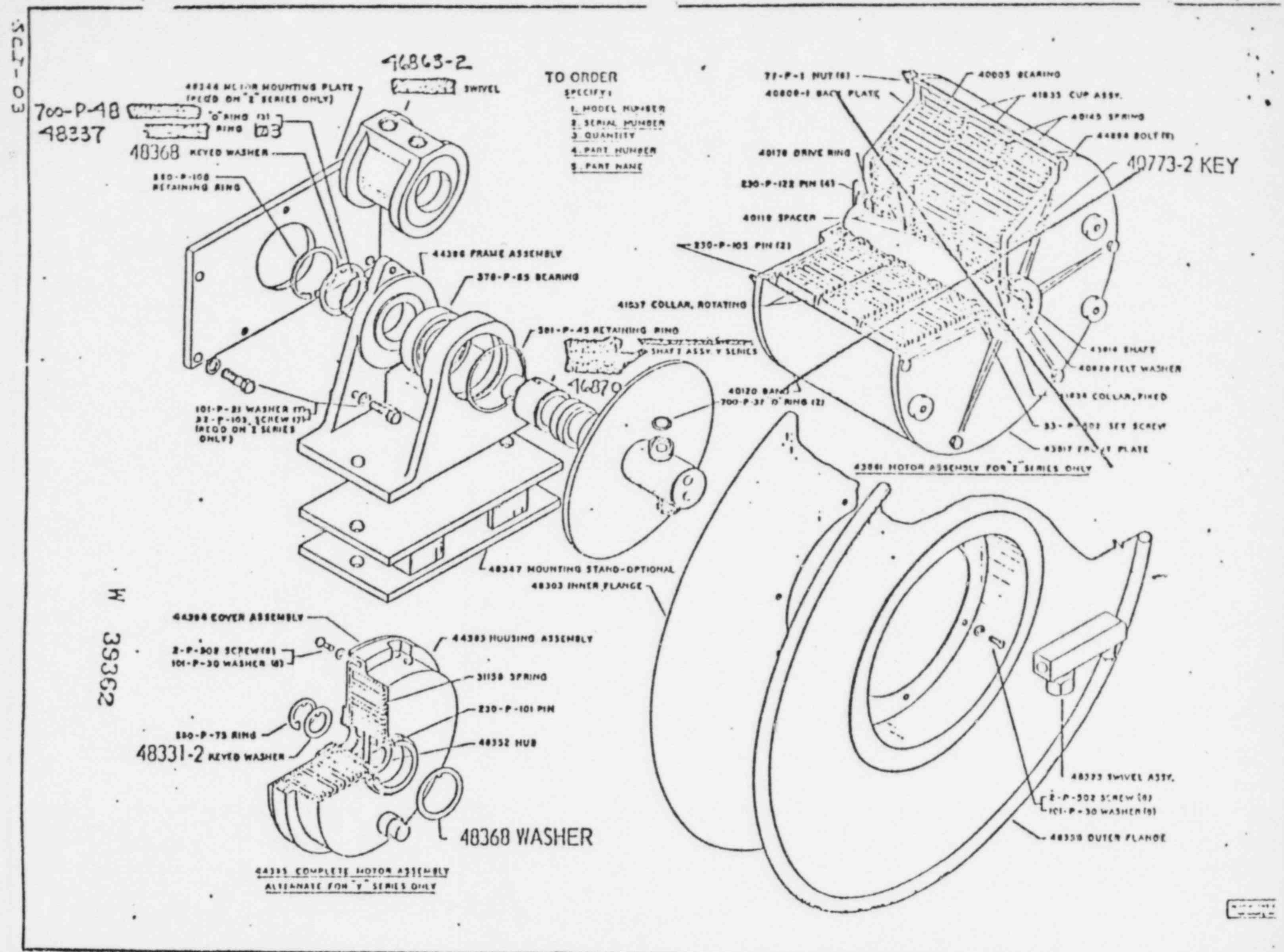
TO REPLACE SWIVEL AND SHAFT SEALS:

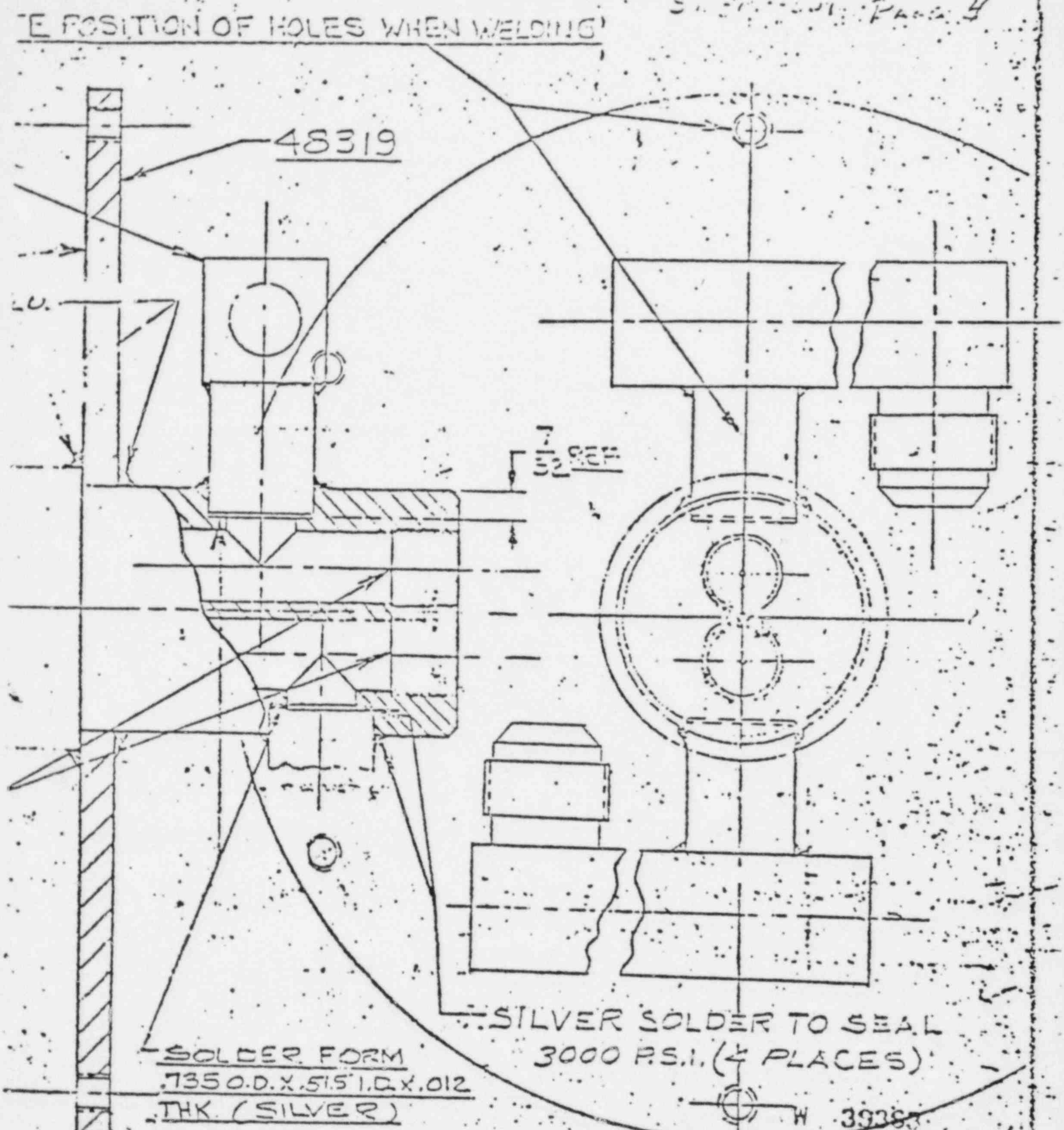
Remove spring motor assembly. Remove hose from reel and inner (48303) and outer (48338) flanges from main shaft. Remove retaining ring (580-P-100), (2) keyed washers (48308) and retaining ring (581-P-45) from main shaft and frame assembly. Pull main shaft, by reel end, out of swivel (48361-2) and frame assembly (44396). Remove old seals from shaft. Install (3) new "O" rings [illegible] and (3) new glyd rings (48337), one each, in each seal groove. Lubricate with light oil. Insert main shaft through frame assembly and into swivel being careful to not damage seals. Reverse the above procedure to complete reassembly.

LUBRICATION:

Reel is lifetime lubricated at factory. Further lubrication is neither required nor recommended.

W 39361


TO ORDER
SPECIFY:
1. MODEL NUMBER
2. SERIAL NUMBER
3. QUANTITY
4. PART NUMBER
5. PART NAME
46863-2
SWIVEL
48344 MOTOR MOUNTING PLATE (REQD ON "Z" SERIES ONLY)
700-P-48
48337
"O" RING (2)
RING
48368 KEYED WASHER
380-P-108 RETAINING RING
44386 FRAME ASSEMBLY
370-P-85 BEARING
301-P-45 RETAINING RING
46879
101-P-21 WASHER (2)
32-P-103, SCREW (2) (REQD ON Z SERIES ONLY)
48347 MOUNTING STAND-OPTIONAL
48303 INNER FLANGE
44384 COVER ASSEMBLY
44383 HOUSING ASSEMBLY
2-P-302 SCREW(8)
101-P-30 WASHER (8)
31158 SPRING
230-P-101 PIN
380-P-73 RING
48331-2 KEYED WASHER
48332 HUB
48368 WASHER
44385 COMPLETE MOTOR ASSEMBLY ALTERNATE FOR "Y" SERIES ONLY
77-P-1 NUT (8)
40005 BEARING
40800-1 BACK PLATE
41835 CUP ASSY.
40145 SPRING
44884 BOLT (8)
40773-2 KEY
40170 DRIVE RING
230-P-122 PIN (4)
40118 SPACER
230-P-105 PIN (2)
41537 COLLAR, ROTATING
43814 SHAFT
40820 FELT WASHER
40120 BAND
700-P-37 "O" RING (2)
33-P-302 SET SCREW
43817 FRONT PLATE
43861 MOTOR ASSEMBLY FOR "Z" SERIES ONLY
48323 SWIVEL ASSY.
2-P-302 SCREW (8)
101-P-30 WASHER (8)
48338 OUTER FLANGE
SCJ-03
W 39362



MATERIAL	AERO-MOTIVE MFG. CO. KALAMAZOO, MICHIGAN			MODEL NEXT ASSY
	DR BY	CHECKED BY	APP'D BY	SCALE: FULL
FINISH	DATE	DATE	DATE	NO. REQ'D: 1
	MAIN SHAFT ASSEM.			46870

...ERWISE SPECIFIED
...TOLERANCES ARE
... ANGULAR ±...

SCI-03

Issued 10/30/78 Rev. 0	[illegible]	B&W No. 04

SITE CAUTION INST.

W 39364

Dupe

FUEL GRAPPLE DISENGAGE BAR

This procedure is recommended for use at all sites in the event that the fuel grapple is engaged, unloaded, erroneously.

Required is a 2" x 2" x 12" long type 304 stainless steel bar (provided by site) to be attached to the floor on the reactor side.

Location of the bar should be at an index position on the selsyn dial for bridge position and at any grid location on the trolley positioning scale. A location near the test fixture would be preferable.

Index desired position on bridge and trolley, lower hoist and position bar under fuel grapple, so that two of the alignment blocks on grapple will rest directly on bar, mark location and attach bar to floor.

For the pool side, the same size bar can be clamped or welded to the top of a fuel storage rack, making sure the bar is located in the center of the rack so that it clears the grapple fingers.

Location of the bar should be at an index position on the selsyn dial for bridge position and at any grid location on the trolley positioning scale.

Index desired position on bridge and trolley, lower hoist and position bar under fuel grapple, so that two of the alignment blocks on grapple will rest directly on bar, mark location and attach bar to rack.

Issued 10/30/78 Rev. 0	[illegible]	B&W No. 06

SITE CAUTION INST.

W 39366

Emergency Pull Out Cable

In the event it is necessary to use the emergency pull out cable to return the transfer carriage to its storage position, the following equipment and steps are required.

Equipment:

1-Klein Hot Line "Chicago" grip.
Model 1628-5BH (or equal)
1-Load Sensing Device (6000 #CAP)

The load sensing device should be hooked on the hook of the overhead or gantry crane.

The "Chicago" grip should be attached to the load sensing device.

Pull enough cable off of the emergency pullout cable reel to allow the full engagement of the "Chicago" grip on the cable going to the carriage. Make sure that crane is positioned directly in line with the cable to insure a straight pull on the cable.

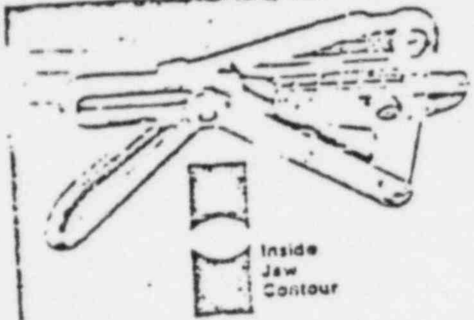
Slowly raise crane hook, car should start to move at or before 1000# is shown on the load sensing device. As the hook is raising, be sure to watch cable on the reel so that you do not pull it off the reel.

Raise hook 3 or 4 feet, then grab cable leading to transfer car, slide "Chicago" grip down the cable 3 or 4 feet, let cable reel on to the reel. After the first initial pull, the distance the hook can be raised can be increased to 6 or 8 feet.

At no time should the load sensing device read more than 2000# during a carriage retrieval. If a load greater than 2000# is applied, the work should be stopped and the problem evaluated prior to proceeding.

W 39367

Klein "Chicago"® Grips CONT'D

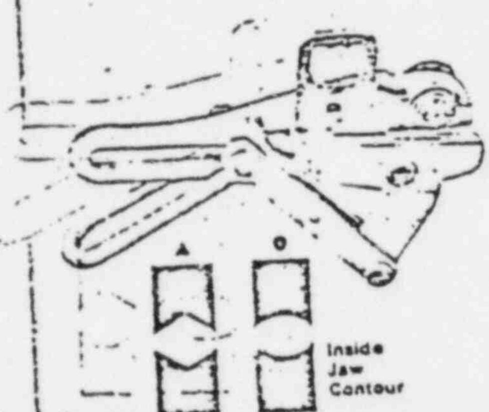


Grips for weatherproof wire

Round inside jaw contour provides maximum protection for weatherproof coatings. Notches in jaw provide firmer grip on insulation.

Cat. No.	Minimum Cable Size—inches (mm)	Maximum Cable Size—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)
1611-20	.20 (5.08)	.40 (10.20)	4500 (2041)	3 (1.36)
1611-30	.25 (6.35)	.53 (13.46)	4500 (2041)	3¾ (1.70)
1611-40	.53 (13.46)	.74 (18.80)	8000 (3629)	7¾ (3.52)
1611-50	.78 (19.81)	.88 (22.35)	8000 (3629)	7¾ (3.52)

Note: Due to various types of weatherproof coatings available, selection of proper
is determined by outside diameter of cable.



Hot Line "Chicago" grips

Klein provides Hot Line Grips for both bare and insulated conductors. Grip may be placed on wire with hot line stick. When stick is removed, safety latch closes automatically to guard against grip accidentally disengaging from wire.

Standard Hot Line Grips are not s with springs or lock-open featur these features are desired, pref "S" before catalog number (e S1628-5BH).

For Bare Conductors

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)
▲ 1613-40BH	10 B&S Solid .10 (2.54)	1/0 B&S Strand .37 (9.40)	4500 (2041)	3¼ (1.47)
▲ 1628-5BH	6 B&S Solid .16 (4.06)	4/0 A.C.S.R. .55 (13.97)	8000 (3629)	6¼ (2.84)
● 1656-20BH	6 A.C.S.R. .20 (5.08)	1/0 A.C.S.R. .40 (10.16)	4500 (2041)	3¼ (1.47)
● 1656-30BH	4 A.C.S.R. .25 (6.35)	3/0 A.C.S.R. .50 (12.70)	4500 (2041)	4 (1.81)
● 1656-40BH	3/0 A.C.S.R. .50 (12.70)	336,400 CM A.C.S.R. .74 (18.80)	8000 (3629)	7¾ (3.52)
● 1656-50BH	397,500 CM A.C.S.R. .78 (19.81)	477,000 CM A.C.S.R. .88 (22.35)	8000 (3629)	7¾ (3.52)
● *1656ABH	3/0 A.C.S.R. .50 (12.70)	500,000 CM A.C.S.R. .90 (22.86)	6000 (3629)	7¾ (3.52)

*For use on aluminum cables where minimum conductor damage is essential, we hold stock of parts, and jaws are m [illegible] to ameter of cable. Range of conductor size is given for general information and is made for one size cable only. *Orders must exact outside diameter of cable.*

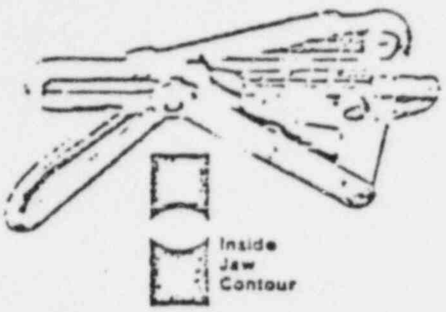
W 39368

For Insulated Conductors

Same as 1659 series grips on page 42, except for addition of hot line latch

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)
● 1659-20H	.20 (5.08)	.42 (10.67)	4500 (2041)	3¼ (1.47)
● 1659-30H	.31 (7.87)	.50 (12.70)	4500 (2041)	4 (1.81)
● 1659-40H	.49 (12.45)	.79 (20.07)	8000 (3629)	7¾ (3.52)
● 1659-50H	.79 (20.07)	1.01 (25.65)	8000 (3629)	7¾ (3.52)

Klein "Chicago"® Grips CONTD

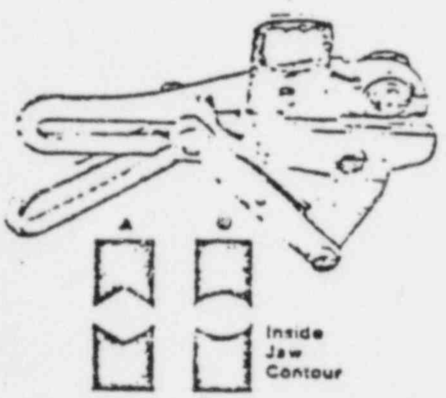


Grips for weatherproof wire

Round inside jaw contour provides maximum protection for weatherproof coatings. Notches in jaw provide firmer grip on insulation.

Cat. No.	Minimum Cable Size—inches (mm)	Maximum Cable Size—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Jaw Length—inches (cm)
1611-20	.20 (5.08)	.40 (10.20)	4500 (2041)	3 (1.36)	4 (10.2)
1611-30	.25 (6.35)	.53 (13.46)	4500 (2041)	3¾ (1.70)	4¾ (12.1)
1611-40	.53 (13.46)	.74 (18.80)	8000 (3629)	7¾ (3.52)	5½ (14.0)
1611-50	.78 (19.81)	.88 (22.35)	8000 (3629)	7¾ (3.52)	5½ (14.0)

Note: Due to various types of weatherproof coatings available, selection of proper grips is determined by outside diameter of cable.



Hot Line "Chicago" grips

Klein provides Hot Line Grips for both bare and insulated conductors. Grip may be placed on wire with hot line stick. When stick is removed, safety latch closes automatically to guard against grip accidentally disengaging from wire.

Standard Hot Line Grips are not supplied with springs or lock-open feature. When these features are desired, prefix letter "S" before catalog number (example: S1628-5BH).

For Bare Conductors

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Jaw Length—inches (cm)
▲ 1613-40BH	10 B&S Solid .10 (2.54)	1/0 B&S Strand .37 (9.40)	4500 (2041)	3¼ (1.47)	4 (10.2)
▲ 1628-5BH	6 B&S Solid .16 (4.06)	4/0 A.C.S.R. .55 (13.97)	8000 (3629)	6¼ (2.84)	5 (12.7)
○ 1656-20BH	6 A.C.S.R. .20 (5.08)	1/0 A.C.S.R. .40 (10.16)	4500 (2041)	3¼ (1.47)	4 (10.2)
○ 1656-30BH	4 A.C.S.R. .25 (6.35)	3/0 A.C.S.R. .50 (12.70)	4500 (2041)	4 (1.81)	4 (10.2)
○ 1656-40BH	3/0 A.C.S.R. .50 (12.70)	336,400 CM A.C.S.R. .74 (18.80)	8000 (3629)	7¾ (3.52)	4¾ (12.1)
● 1656-50BH	397,500 CM A.C.S.R. .78 (19.81)	477,000 CM A.C.S.R. .88 (22.35)	8000 (3629)	7¾ (3.52)	5½ (14.0)
● *1656ABH	3/0 A.C.S.R. .50 (12.70)	500,000 CM A.C.S.R. .90 (22.86)	8000 (3629)	7¾ (3.52)	5½ (14.0)

*For use on aluminum cables where minimum conductor damage is essential, we hold stock of parts, and jaws are milled to fit exact diameter of cable. Range of conductor size is given for general information and is made for one size cable only. *Orders must specify exact outside diameter of cable.*

W 39369

For Insulated Conductors

Same as 1659 [illegible] grips on page 42, except for addition of hot line latch.

Cat. No.	Minimum Conductor—inches (mm)	Maximum Conductor—inches (mm)	Maximum Safe Load—lbs (kg)	Approx. Weight Each—lbs (kg)	Jaw Length—inches (cm)
● 1659-20H	.20 (5.08)	.42 (10.67)	4500 (2041)	3¼ (1.47)	4 (10.2)
● 1659-30H	.31 (7.87)	.50 (12.70)	4500 (2041)	4 (1.81)	4¾ (12.1)
● 1659-40H	.49 (12.45)	.79 (20.07)	8000 (3629)	7¾ (3.52)	5½ (14.0)
● 1659-50H	.79 (20.07)	1.01 (25.65)	8000 (3629)	7¾ (3.52)	5½ (14.0)

Note: Due to various types of PVC and other weatherproof coatings, selection of proper grips is determined by O.D. of cable.

ScT-06

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

December 21, 1978

SOM-II-209

Mr. L. L. Lawyer
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Post Office Box 542
Reading, PA 19604

[redacted]
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Turbine Bypass to Atmosphere Valves

Gentlemen:

There has been a great deal of concern in the operation of MS-V3A&B at TMI-II. Both valves have been "stroked" by Met-Ed I&C personnel and by a Fisher Valve Company representative. "Weeping" through the valve has continued.

On November 17, 1978, the actuator for MS-V3B was removed in order to replace a body to bonnet gasket on MS-V3B. While the actuator was removed, it was disassembled for maintenance and lubrication. Upon disassembly, two major problems were discovered. The first problem observed was that the operator's lubrication was dried out and caked on various surfaces inside the gear box. This problem was corrected by cleaning the operator and lubricating with a high temperature lubricant.

The second problem was more serious and could easily occur again. The operator shaft was discovered to be binding inside the leadscrew assembly, which operates the valve when in manual operation. When the two shafts were separated, it was observed that in the area around the manual operation engagement pin hole, the metal was "upset." This upset was caused by (1) inserting the tapered locking pin from the wrong direction, and (2) use of a "ground down" or modified locking pin. Either or both of these two items allowed a very large force to be exerted on a very small surface area; this upsets the metal and causes binding.

W 39370

The actuator for MS-V3A was removed for inspection after finding the two (2) problems with the MS-V3B actuator. The lubricant in MS-V3A actuator was in a deteriorated condition but in much better condition than the lubricant in MS-V3B operator.

The metal upset observed on the MS-V3B actuator was also found on MS-V3A actuator and of the same order of magnitude. The upset metal was removed from the shafts of both operators, and the operators were reassembled and installed.

To minimize the possibility of a reoccurrence of this problem, B&W recommends that maintenance and operations personnel be directed to use only the proper tapered pin and to ensure that the pin is fully inserted (it will not be possible to insert the pin fully if the pin is inserted from the wrong direction). It should be noted that the operators for the turbine bypass to condenser valves, MS-V25A&B and MS-V26A&B, are similar and the recommendation stated above applies to these valves also. It is also recommended that the lubricant in MS-V3A&B be inspected the next time maintenance is performed on these valves to ensure that the high temperature lubricant used is providing adequate lubrication at the elevated temperature these actuators are exposed to.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/LNM/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
G. A. Kunder
J. L. Seelinger

B+W Book

Babcock & Wilcox

Power Generation Group

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January 2, 1979

SCM-II-210
REM-I-273

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Mr. J. L. Seelinger
Superintendent, Unit I
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: DP 1101-01, Limits & Precautions - Boric Acid Addition to Primary System

Gentlemen:

During the refilling of a B&W operating plant's primary system, a solution of 7 wt% boric acid was added followed by demineralized water. This procedure combined with the boric acid concentration and an ambient temperature of approximately 70°F resulted in a seal injection line becoming plugged with boric acid crystals. The crystallization point for 7 wt% boric acid is about 95°F.

W 39372

Based on this incident, the following observations have been made:

1. The fact that it did occur during a filling operation clearly shows that it could occur again under similar circumstances at other times and at other plants.

2. In our plants we have based our designs and operations on the arrangement that any water drained from the Reactor Coolant System for maintenance, such as refueling, is pumped to the reactor coolant bleed holdup tank for temporary storage and then transferred back to the Reactor Coolant System when it is refilled. In this way, it was expected that the need to add concentrated boric acid during a fill operation would be a very infrequent operation.

3. The manner in which the boric acid and demineralized water were added does not violate any of our present operating instructions. In fact, if the additions are made separately, the boric acid must be added first to assure that the proper boric acid concentration is maintained in the coolant.

4. The boric acid could also possibly crystallize in the pump seals and damage the seals.

5. Our designs and operations are now primarily based on the use of 5 wt% boric acid which has a crystallization temperature of about 72F and which reduces the possibility of crystallization from occurring.

6. The possibility of such an incident occurring while the Reactor Coolant System is pressurized with the makeup pumps operating is considered more remote. The reason is that the capacity of the makeup pumps is much larger than that of the boric acid addition pumps. Hence, it is not possible to operate the makeup pumps with only the boric acid addition pumps connected to the suction. The suction must be open to the makeup tank which contains a large volume of water.

7. The situation where it could possibly occur with the Reactor Coolant System pressurized and the makeup pumps operating is when the Reactor Coolant System is cooled down for refueling and similar maintenance that requires large quantities of boric acid to be added in order to compensate for the contraction of the coolant and to borate the coolant up to 13,000 ppm boric acid. If the letdown flow was stopped for some reason, it is

7. conceivable that the concentrated boric acid flow could run long enough for the concentration in the makeup tank and makeup lines to approach that in the concentrated boric acid storage tank (BAST).

As a precaution to prevent this from occurring at TMI, make the following addition to Plant Limits and Precautions, Section 4.3.2-05 (Page 74).

> Boric acid additions to the Reactor Coolant System shall not exceed a concentration of 5 wt%. However, higher concentrations may be used if it is first diluted with unborated water or if makeup flow has been established to provide dilution prior to entering the makeup or seal injection lines.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
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G. K. Wandling
J. D. Phinney
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W 39374

B+W Book

Babcock&Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

January 9, 1979

SOM-II-211

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Reading, PA 19603

[redacted]
Station Superintendent
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Post Office Box 480
Middletown, PA 17057

Subject: RPS Setpoints and Allowable Reactor Power Versus Reactor Coolant Flow

Reference: 1. SOM-II-204, "Flow Value Used in Safety Analysis," dated 29 November 1978
2. Technical Specification revision to Table 3.2-2 (DNB Margin)

Gentlemen:

An analysis has been performed which shows that limiting reactor power to that allowed by Technical Specification Table 3.2-2 is conservative with respect to maintaining the same DNBR margin as that calculated for licensed conditions. Therefore, the overpower trip setpoint must be reduced by an amount equal to the amount allowable operating power is reduced below 100%.

Limitations of Technical Specification Table 3.2-2 also affect the flux/flow trip. Since a change to the flux to flow trip setpoint would require a change to the Technical Specification, it is advisable to limit the output voltage from the flow buffer amplifier instead. The equation which normally describes this output voltage is:

$$V_{out} = 0.08 \frac{\text{volts}}{\%\ \text{flow}} \times (1.05) \times (100-x)$$

where 1.05 = flux/flow setpoint and

x = the amount allowable operating power is reduced below 100% by Technical Specification Table 3.2-2

W 39375

It is recommended that using the best flow rate estimate to date, determine the maximum allowed power level. With this value, calculate the maximum output voltage and normalize the curve provided in TP 800/22. This normalized curve can then be used to check the output voltage of the flow buffer amplifier at any power level. It is recommended that you check and, if necessary, adjust the output voltage at 90% RTP and, of course, reset the output voltage at the maximum allowed power.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E G Schiedel
for
L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
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B+W Book

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

January 10, 1979

SOM-II-212

Mr. L. L. Lawyer
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Post Office Box 542
Reading, PA 19603

[redacted]
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Reactor Coolant System Flow via Secondary Heat Balance

Gentlemen:

Enclosed is B&W's recommended procedure for calculating primary (Reactor Coolant System) flow based on a secondary heat balance.

At this time, data from TMI-2 exists to establish the primary enthalpy difference correction factor, ϵ, in the enclosed procedure. The data was taken on December 21, 1978, between 1216 and 1301 hours. In addition, Lynchburg Engineering believes this offset in primary ΔT is constant over the range of temperatures encountered in the performance of this procedure. Further, the offset is expected to be constant for at least two to three months.

In order to monitor the possibility of a "degrading" Reactor Coolant System flow situation, please send a copy of the periodic heat balance calculations to this office for transmittal to Lynchburg for engineering review.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

for

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

W 39377

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
G. A. Kunder
J. L. Seelinger

(2)

RCS FLOW DETERMINATION BY HEAT BALANCE

1.0 TEST OBJECTIVE

The purpose of this procedure is to measure the reactor coolant system flow rate (i.e., total reactor vessel flow rate) by the use of a secondary plant heat balance.

2.0 PREREQUISITES

2.1 Prior to reactor startup, the NSS will be maintained at the hot zero power condition (HZP) long enough to establish a thermal equilibrium between the primary and secondary plants. In addition, the reactor will have been shutdown long enough, prior to this time, for decay heat to be negligible (less than 0.2% FP).

2.2 Following the establishment of this equilibrium, at least 15 minutes of reactor coolant hot leg and cold leg temperatures shall be recorded at intervals not in excess of one minute to verify that the NSS is in thermal equilibrium. No heatup or cooldown trend should be indicated by this data. During this period, letdown flow rate should be reduced to the minimum practical flow.

NOTE: All narrow range cold leg and all hot leg RTD's should be recorded. These readings will be used to determine the magnitude of the calibration correction (Enclosure 01) which should be applied to the RCS temperature readings during the heat balance calculations.

2.3 Adequate data recording equipment should be set up to take the data. Where possible, the plant computer should be utilized to avoid the use of NNI strings which introduce additional error.

2.4 The same RCS temperature channels used to determine the loop temperature detector calibration errors shall be recorded during the heat balance data collection.

2.5 During the heat balance run, feedwater flow should be recorded continuously on a strip chart recorder or the reactimeter so that a rapid sampling rate can be maintained. This rate should be at least once per second.

2.6 All other parameters may be recorded once per minute.

3.0 SPECIAL PRECAUTIONS

If the plant has been operating for a considerable length of time and recent isothermal (HZP) data is not available, the heat balance may be performed without making this correction. However, increased accuracy may be obtained by accounting for the temperature detector calibration errors prior to performing the heat balance.

W 39379

4.0 INITIAL CONDITIONS

4.1 Steady State Conditions Established in NSS

NOTE: Relative errors on key plant parameters used in the heat balance calculation are minimized at full load. Therefore, the most reliable flow determination will be achieved at the highest power level.

4.1.1 Steady state conditions, for the purpose of this test, may be assumed to exist when data recorders (i.e., reactimeter, strip chart recorders, or computer samplings) show no increasing or decreasing trends for a period not less than 15 minutes with a minimum of one data point recorded per minute on the following parameters:

4.1.1.1 hot leg temperatures

4.1.1.2 cold leg temperatures

4.1.1.3 steam pressures (OTSG outlet)

4.1.1.4 steam temperatures (OTSG outlet)

4.1.1.5 feedwater flow rate

4.1.1.6 feedwater temperatures

4.2 Minimum practical letdown flow rate should be established.

5.0 DATA REQUIRED

5.1 Collect three consecutive sets of data, each set being ten minutes in duration, and each parameter to be recorded at least once per minute with the exception of feedwater flow which shall be continuously recorded by strip chart or otherwise as described in Sections 2.5 and 2.6. The following parameters are required both for Enclosures 01 and 02 separately:

5.1.1 Loop A RC Outlet Temp 1 NNRW

5.1.2 Loop A RC Outlet Temp 2 NNRW

5.1.3 Loop A Inlet Temp NREW 1

5.1.4 Loop A Inlet Temp NREW 2

5.1.5 Loop A RC Pressure NREW

5.1.6 OTSG Steam Pressure A

5.1.7 Main Steam Temperature A

W 39380

5.1.8 Average Feedwater Temperature A

5.1.9 Feedwater Flow Rate A

5.1.10 The same data is required for Loop B

5.2 Processing of Data

Each recorded parameter should be averaged over the ten minute period. In addition, the primary loop hot and cold leg temperature readings should be averaged to form a resultant Loop A average hot leg temperature, Loop A average cold leg temperature, Loop B average hot leg temperature, and Loop B average cold leg temperature (for example, the average Loop A RC outlet temperature is actually the average of both Loop A outlet temperature detectors). These values will be referred to as $\overline{T}_{h_A}$, $\overline{T}_{c_A}$, $\overline{T}_{h_B}$, and $\overline{T}_{c_B}$, respectively.

5.3 The feedwater flow data should be examined critically and an average value determined for the applicable period of time for each loop in terms of lbm/hr.

W 39381

ENCLOSURE 01

DETERMINATION OF CORRECTION FACTOR FOR RCS ΔT

Based on Isothermal Condition

W 39382

ENCLOSURE 01

LOOP A ZERO POWER TEMPERATURE DATA

(Identical data to be supplied for Loop B)

*Time	TE A1	TE A3	TE A5	TE A6
Average Values:	$\overline{TE\ A1}$	$\overline{TE\ A3}$	$\overline{TE\ A5}$	$\overline{TE\ A6}$

*Minimum of 10 readings, one minute apart

Note: "TE A " designation is arbitrary to describe the respective RCS RTD's.

let $T_{c_A} = \frac{\overline{TE\ A1} + \overline{TE\ A3}}{2}$ = true average cold leg temperature

let $T_{h_A} = \frac{\overline{TE\ A5} + \overline{TE\ A6}}{2}$ = true average hot leg temperature

then define $\varepsilon_A = T_{h_A} - T_{c_A}$ as the difference between hot and cold leg temperatures for loop A.

Since there are temperature changes due to pump work and pressure drops around the loop, the final calibration error is determined by:

$$\varepsilon'_A = T_{h_A} - T_{c_A} - 0.25°F$$

W 39383

ENCLOSURE 02

CALCULATIONS OF PRIMARY SYSTEM FLOW RATE

W 39384

ENCLOSURE 02

Basic Equation: $$W_A = \frac{W_{Fdw_A}(h_{stm} - h_{fdw}) + 1/2\ Q_{losses}}{h_{hot} - h_{cold}}$$

Where:

W_A	=	Loop A RC flow rate
W_{Fdw_A}	=	Loop A feedwater flow rate
h_{stm}	=	OTSG A outlet steam enthalpy
h_{fdw}	=	OTSG A inlet feedwater enthalpy
Q_{losses}	=	Ambient heat losses
h_{hot}	=	Loop A hot leg enthalpy
h_{cold}	=	Loop A cold leg enthalpy

Q_{losses} for full power operation is approximately 1.86×10^6 BTU/hr.

From the data collected in Section 5, determine the four enthalpies required by the above equation and an average feedwater flow rate:

W_{fdw} = ______________________ lb/hr

h_{stm} = ______________________ BTU/lbm

h_{fdw} = ______________________ BTU/lbm

h_{hot} = ______________________ BTU/lbm

h_{cold} = ______________________ BTU/lbm

W 39385

ENCLOSURE 02 Continued

Correction factor for temperature detector calibration error:

To arrive at the correct primary side enthalpy difference, the calibration error is accounted for as follows:

$$\frac{\overline{Th} - \overline{Tc} - \varepsilon}{\overline{Th} - \overline{Tc}} = f\text{, the calibration correction factor}$$

This correction must be calculated for each loop.

Final loop flow calculation: substitute the appropriate values for each parameter into the following:

$$W_A = \frac{FW_{fdw}(h_{stm} - h_{fdw}) + 0.93 \times 10^6}{f(h_{hot} - h_{cold})}$$

Where: W_A is in lbm/hr

To convert to volumetric flow rate:

$$0.1247 \frac{W_A}{\rho cold\ leg} = Q_A \text{ (gpm)}$$

Perform duplicate calculation for the B loop. Then total RCS flow is $Q_A + Q_B$.

$$\%\text{ of design flow} = \frac{Q_A + Q_B}{352{,}000} \times 100$$

W 39386

346 BX

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
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January 10, 1979

SOM-II-213

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: TMI-2 Unusual Noise OTSG B (Channel 12)

Gentlemen:

B&W personnel listened to the tape on which the TMI Unit 2 unusual noise was recorded and monitored the external OTSG test program sensor channels. As a result, we considered the following as possible sources of the noise:

1. Loose part
2. Water hammer
3. Thermal expansion
4. External impacting (particularly to the OTSG skirt)
5. Water dripping on the accelerometer cover box

Since the low frequency of the impact is indicative of a massive object and the noise was heard clearly on only one channel, a loose part was eliminated as a likely source. Originally, we were under the impression that the OTSG lower sensors were mounted on the OTSG sensing lines; therefore, an external noise, possibly a water hammer, could have explained the noise. After finding that the sensors are mounted on the OTSG skirt, we feel that a water hammer is an unlikely noise source; water hammers do produce a muffled, low frequency noise similar to the noise recorded on tape. The possibility of noise from thermal expansion was eliminated since no severe thermal transients were involved and since the noise was heard over a period of several hours.

W 39387

Impacts (particularly from a massive object) to the OTSG skirt near the LPM channel 12 sensor appear to be the most probable noise source. We performed a simulation in our lab during which we dripped water on a metal box covering an accelerometer mounted on a steel beam. The noise produced was similar to the noise recorded on the cassette tape if the amplifier output was low pass filtered at less than 1.5 KHz. It is our understanding that the low pass filters in the TMI Unit 2 LPMS are set much higher than 1.5 KHz; therefore, water dripping on the accelerometer cover box is not considered a probable source. During the test we tapped very lightly on the steel beam with a sledge hammer and were able to produce a noise comparable to the one recorded on the tape.

As a course of action, we suggest the following:

1. TMI personnel could enter containment and inspect the OTSG skirt area (particularly near the channel 12 sensor) for any object which might impact the skirt, supports, etc.

2. While in containment, personnel should impact the OTSG skirt, with personnel in the control room, to verify the LPMS installation.

3. If the noise returns, someone should enter the containment and listen with a stethoscope for the noise and then try to locate the source.

4. In addition to or instead of step 3, a test program requiring the installation of additional sensors could be used to locate the noise source.

Thursday, January 4, 1979, was the last day that B&W site personnel heard the noise. Again on Friday, January 5, 1979, and Monday, January 8, 1979, the site personnel listened for the noise but could not hear it.

If the noise is assumed to be from an external source, it is recommended that plant operating conditions and equipment status from Thursday and Friday (January 4 and 5, 1979) be reviewed to determine what could have caused the noise. Any situation or equipment configuration that may be suspected should be re-enacted to verify if it was the cause of the noise.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E. G. Schneider
for

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
G. A. Kunder
J. L. Seelinger
J. A. Brummer
I. Porter

B+W Book

~~G. Miller~~

Babcock & Wilcox

Power Generation Group

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January 11, 1979

SOM-II-214

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: OTSG B Instrumentation: Completion of Data Acquisition Phase

Gentlemen:

This letter is to advise you that all of the data acquisition requirements of the Test Guide and the Test Guide Deviations have been fulfilled. Effective Wednesday, January 10, 1979, the data acquisition phase of the instrumentation program is terminated.

The data tapes have been reviewed and accepted by the B&W Engineering Department. The DPS trailer will be transferred to Lynchburg in order to continue with the data analysis. GPUSC has approved the termination of the data acquisition and the removal of the DPS trailer from the site.

Preparations are now being made to remove the data tapes and electronic equipment from the site. It is tentatively scheduled for Catalytic to start the OTSG trailer removal preparations on Monday, January 15, 1979. This will be confirmed on Monday morning. The trailer removal will be scheduled for some time between January 17, 1979, and January 19, 1979.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

E G Schedel
for

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

W 39390

L. L. Lawyer
G. P. Miller

cc: L. Pletke
W. Spangler
G. Wandling
J. Herbein
R. Klingaman
J. Logan
G. Kunder
J. Seelinger
L. Zubey
U. Lamay
J. Blanton
R. Fite
J. Olszewski

W 33391

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P.O. Box 1260, Lynchburg, Va. 24505
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January 15, 1979

SOM-II-215

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Limits & Precautions: Restart of Fourth Reactor Coolant Pump at 50% Power

Gentlemen:

This letter supersedes the memo dated January 9, 1979, that was given to J. Logan and the Shift Supervisor concerning the same subject. A copy of the memo is attached for reference.

Based on the evaluation of test data from the TMI-2 restart of a fourth reactor coolant pump at 50% FP, B&W approves an interim revision to the existing limit of 30% FP supplied in Plant Limits and Precautions (DP 1101-01) Section 1.1.03. This new limit of 50% FP is allowable until the final recommendation is provided on a generic basis to all B&W plants. This approval is also based on the fact that analyses (which appear in the TMI-2 FSAR) have been performed which show that starting two reactor coolant pumps at 60% FP at EOL does not violate any safety limits.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

W 39392

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
G. A. Kunder
J. L. Seelinger

L.C.Rogers, Site Operations Manager, TMI 2

FROM

J. Veenstra, Plant Performance Services Section

TMI 2

SUBJ

Restart of Fourth RCP at 50% Power

DATE 1/9/79

Based on the evaluation of test data from the TMI-2 restart of a fourth RCP at 50%FP; B&W approves an interim revision to the existing limit of 30%FP supplied in Plant Limits and Precautions Section I.D.33. This new limit of 50%FP is allowable until the final recommendation is provided on a generic basis to all B&W plants. This approval is based on the fact that the above test demonstrated satisfactory plant response under more severe thermal feedback conditions than will exist at any time in the future on 177 FA plants.

Engineering: Robert Wright 1/9/79

Nuclear Services: J. Veenstra 1/9/79

W 39394

B&W Book

G. Miller

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
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January 15, 1979

SOM-II-216
REM-I-374

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Internal Vent Valves

Gentlemen:

Disassembly and examination/inspection of the internal vent valves at Oconee II and Rancho Seco indicates there is a probability of wear on the jackscrew locking mechanism between the pressure plate and retainer plate, particularly the valves adjacent to the outlet nozzles and the jackscrews closest to the outlet nozzles.

Recommendations:

1. Inspection procedures for use at the next refueling are in preparation, along with modification recommendations if determined to be required by the inspection. These will be supplied at a later date.

2. Prudent operations would dictate that the loose parts monitoring system should be verified operable and should be closely monitored at all times during reactor coolant pump operations. Particularly, the sensors on the reactor vessel head and the OTSG upper head should be closely monitored for unusual noises.

B&W has concluded that this does not pose a safety problem. Justification for this position is supported by Attachment 1.

W 39395

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/EGS/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
G. A. Kunder
J. L. Seelinger
W. E. Potts
J. D. Phinney
J. T. Janis

ATTACHMENT #1

CONSEQUENCES OF VENT VALVE JACKSCREW LOCKING MECHANISM WEAR

In order to determine the worst situation that could occur due to wear of the locking mechanism, a series of worst case assumptions were made and the consequences evaluated. The conclusion reached, even with these worst case assumptions, is that the operability of the valve would not be affected and thus, does not pose a safety problem.

1) Assume that the retainer plate and locking cup do wear to the point than a portion of the retainer plate is loose and the pressure plate escapes. Based on the worst wear at SMUD, it can be concluded that at this point 1/3 to 1/2 of the pressure plate mass would be worn away. The worn pressure plate of about 3 to 4 oz. mass would travel to the top of the steam generator. If the loose part were to impact the upper tube-sheet with sufficient energy to damage the tube ends or tube-to-tubesheet weld it would in all probability be detected by the loose parts monitor, the plant shutdown and the part removed. The other possibility is that the loose part would stay in the upper head of the steam generator until it beat itself into smaller pieces and thus, pass through the generator tubes. The part(s) would then lodge in the bottom of the fuel assemblies and be removed from the reactor when the fuel is removed. This later possibility, although remote could result in primary to secondary tube-to-tubesheet weld leaks which exceed Tech. Spec. Limits and requires plant shutdown and repair of the generator.

2) It is very probable that even with the pressure plate missing from one jackscrew, the screw would not loosen in that it has a significant amount of preload. However, assuming that it does loosen, the maximum amount that it can rotate is 1¼ turns due to the fact that the threads will bind-up. This is due to the fact that the two jackscrews have to be run together in order to collapse the wedge rings. Because of the 10^{o} wedge angle this would result in a 0.033" maximum gap between the valve body and the core support shield mounting ring. This gap would only exist on one side in that the jackscrew on the opposite side of the valve would not allow the wedge ring to collapse. The 0.033" gap would result in a 0.05% increase in core bypass flow. Further, making the worse case assumption that all four (4) valves near the outlet nozzle were to reach this condition, the increase in core bypass flow would be 0.2% which is significantly less than the current flow margins for the 177 FA plants. At this point, the valve assembly may rattle in the mounting ring and thus, be detected by the LPM and the reactor shutdown or simply remain stationary against the wedge ring due to the reactor vessel inlet to outlet pressure drop. In either event the function and operability of the valve will not be affected.

W 39397

In summary, even with several very conservative assumptions which have an extremely low probability of occurrence, the discovered problem of wear on the jackscrew locking mechanisms near the outlet nozzles cannot lead to a reactor safety problem. The worse thing that can happen is damage to the generator upper tubesheet and perhaps primary to secondary leaks.

W 39398

1-B+W
Book
1-351

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Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

January 23, 1979

SOM-II-219
REM-I-376

Mr. C. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Locking Tab Washer for CRDM Closure

Dear Gary:

Diamond Power Specialty Corporation has upgraded the locking arrangement for the CRDM top closure, which facilitates the easy installation and removal of the locking tab washers. A copy of the CRDM Service Tip Number 4-78 concerning the installation of the CRDM closure locking washers is attached for your reference.

The Diamond part number for the locking tab washer is 707755-1138, and the delivery of the locking washers and tool can be effected in about four (4) weeks after the receipt of the order.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. D. Phinney
J. T. Janis
J. G. Herbein
R. M. Klingaman
L. L. Lawyer
J. B. Logan
J. L. Seelinger
G. A. Kunder
W. E. Potts

W 39399

INSTRUCTION FOR UPDATING

CRDM NUCLEAR EQUIPMENT BULLETINS

CRDM SERVICE TIP NO. 4-78

NOVEMBER 1978

Add the attached Service Tip
to your maintenance procedure
and file as a permanent record.

W 39400

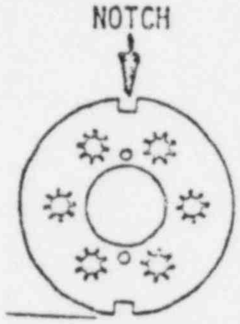
NUCLEAR EQUIPMENT BULLETIN

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

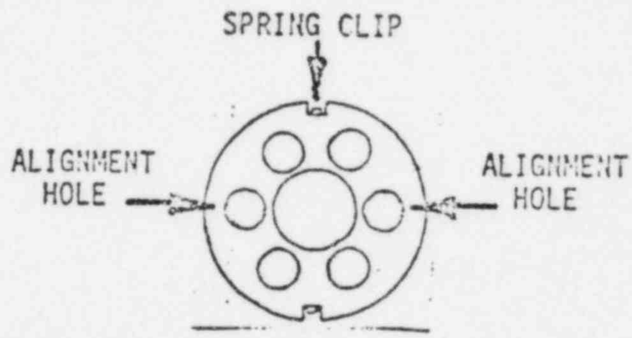
CRDM SERVICE TIP NO. 4-78

INSTALLING THE CRDM CLOSURE LOCKING TAB WASHER

When electing to use the Closure Locking Tab Washer (707753-1138) instead of lockwire, a four to one time savings will be experienced. It will be necessary to use a washer installation tool (707918-1033) for proper installation.



LOCKING WASHER



INSTALLATION TOOL

To install the closure locking washer, position the washer on the installation tool. Align the washer notches with the spring clips on the tool, and press the washer onto the tool. Using the alignment holes in the tool, align the tool with the squarehead screws on top of the drive. Strike the tool with a rubber mallet until the washer is flush with the top of the motor tube, taking care not to mar the motor tube with the spring clips on the tool. At this point the washer will have engaged the squarehead screws. Remove the tool, the washer will remain in place.

To remove the locking washer from the drive, use two screwdrivers, 180° apart. Insert the screwdrivers from the outside toward the middle and raise the washer, moving the screwdrivers around the motor tube until the washer is clear of the squarehead screws.

Parts and installation tool are available from B&W Nuclear Parts Center, P.O. Box 1260, Lynchburg, Virginia 24505.

W 39401

Prz Level
U1-2

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

[redacted]

1-JLS – followup
1-45
1-Lee R
I Be back to you
1-JH
I'd Like to buy this
1-GAK To follow/action U2&1
1-PHH
1-SS1

1- Lee R — followup next meeting 2/22
1- B+W Book

January 18, 1979

SOM-II-217

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Analysis Work to Allow Pressurizer Level Changes

Dear Gary:

Following your request for an estimate of the costs relating to analytical work supporting an increase in nominal pressurizer operating level, B&W Lynchburg has responded with a program which should support the planned operations.

The scope of the work to be performed by B&W's Safety Analysis Group will include:

a. Rod Withdrawal from Startup Accident

b. Rod Ejection Accident

c. Loss of Feedwater Accident

These accidents constitute the maximum pressure cases and should be compared for impact of raising the nominal pressurizer operating level from 220 inches to 240 inches. The analysis results will include the range of operation between the stated ranges. The B&W Control Analysis Group will also QA the results prior to being submitted to Met-Ed.

The expected charges for the above task are approximately 400 man-hours level of effort and about 2.0 hour computer time. The Met-Ed expense will be invoiced at the Master Service rates applicable at the time of task performance. The man-hour estimate given does not include the Control Analysis work, but that review effort is minimal by comparison.

If you have any further questions, please do not hesitate to contact me.

W 39402

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
L. L. Lawyer
J. B. Logan
G. A. Kunder
J. F. Hilbish - Met-Ed Reading

B&W Book

Babcock & Wilcox

Power Generation Group

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March 9, 1979

SOM-II-226

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Fourth Reactor Coolant Pump Startup Temperature
Reference: REM-I-221, "Fourth Reactor Coolant Pump Startup Temperature," dated February 12, 1976

Dear Gary:

As part of B&W's periodic review of applicable operational conditions, Unit II testing results have shown that the minimum temperature limit for starting the fourth reactor coolant pump may be relaxed from the previous requirements. Met-Ed should change the Unit II operating limits document for the fourth reactor coolant pump start. It should state that the required reactor coolant system temperature be ≥ 500°F. Flow measurement results have shown that this minimum temperature insures an acceptable margin against core lift when the fourth pump is started.

This position is based on previous analysis which has shown that spring stiffness in the fuel assemblies, spring configuration, flow uncertainty, irradiation and other accumulated effects are within acceptable boundaries.

The startup temperature limit may be subject to change in the future, based upon irradiation exposure and new fuel assembly design and measured reactor coolant flow conditions.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/bay

W 39404

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
L. L. Lawyer
J. B. Logan
G. A. Kunder

INFO LETTERS

W 39406

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

May 31, 1979

TMI 79-79

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
P. O. Box 480
Middletown, PA 17057

Subject: Natural Circulation Operating Guidelines

Dear Mr. Miller:

Attached for your information and use is a copy of Revision 8 of the Operating Guidelines for a Controlled Transition to Natural Circulation for Lower Loop Plants. This revision was prepared based on comments received from some of our customers and based on further B&W analysis. It should be noted that forced circulation is still preferred when available and that if natural circulation cannot be established and reactor coolant pumps cannot be restored, plant decay heat must be removed by HPI cooling.

If you have any questions, please advise.

Very truly yours,

G. T. Fairburn
Service Manager

GTF:NF
Attachment

cc: (w/attachment)
R. M. Klingaman
J. F. Hilbish
L. L. Lawyer
J. F. Fritzen
R. F. Wilson (3) - GPUSC
R. W. Heward - GPUSC
S. L. Seelinger
L. C. Rogers
S. L. Smith

W 39407

Rev. 0E
05/29/79

OPERATING GUIDE

Controlled Transition to Natural Circulation for Lower Loop Plants

NOTE: This Procedure is to be utilized in the event there is a need to stop reactor coolant pumps. The use of reactor coolant pumps is preferable to natural circulation for maintaining RCS flow.

1.0 Initial Conditions

1.1 Reactor has been tripped for at least 10 minutes

1.2 One or more RCP's operating

1.3 Feedwater available

1.4 HPI available

1.5 Reactor coolant pressure stable

1.6 T_c is stable (cooldown rate not to exceed 10°F/hr, no heat up) with steam pressure being maintained by the turbine bypass system

2.0 Limits and Precautions

2.1 Maintain RCS subcooled

2.1.1 Before RCP's tripped (refer to Figure 1).

2.1.2 After RCP's tripped (refer to Figure 2).

2.2 Normal cooldown limits should be maintained (including Appendix G NDT limits).

2.3 Maintain pressurizer level above pressurizer heaters.

2.4 The number of available pressurizer heaters is not limiting for the initiation of natural circulation. However, an effort should be made to maximize the number of available pressurizer heaters to provide the operator with additional pressure control capability.

2.5 Prior to stopping the last operating RCP, OTSG level must be established at or above 50% on the operating range and maintained there while on natural circulation.

2.6 The conditions of pressure and temperature allowed by Figure 2 must be maintained at all times or RCP operation and/or HPI cooling established.

W 39408

3.0 Immediate Actions: None

4.0 Long Term Actions

4.1 Establish subcooling in RCS per Figure 1, using the following as required:

4.1.1 Use available pressurizer heaters.

4.1.2 Secondary pressure using turbine bypass system to control T_c.

4.1.3 Use additional makeup.

4.1.4 Initiate HPI if required.

Caution: If adequate subcooling per Figure 1 cannot be established, continue to run one reactor coolant pump in each loop.

4.2 Establish OTSG level at 50% of the operating range.

Note: Emergency feedwater through the emergency feedwater nozzles is the preferred configuration prior to stopping RCP's. However, main feed through the main or emergency feedwater nozzles (where available) is permitted provided the OTSG level is established and maintained at or above 50% on the operating range before stopping RCP's.

4.3 Establish and control pressurizer level between 100 and 200 inches.

4.4 Stop operating RCP's.

Note: The operator must account for the loss of pump heat input to the system to prevent overfeeding the OTSG after the RCP's have been stopped.

Caution: Stopping the last operating RCP causes the OTSG level to shift control to the emergency feedwater system and/or the feedwater nozzles. The operator should be prepared to maintain control of feedwater after the RCP's are stopped.

4.5 Maintain constant or slowly decreasing (cooldown rate not to exceed 10°/hr, no heatup) T_c by controlling steam pressure with the turbine bypass system.

4.6 Continuously monitor T_h in both loops to assure the pressure and temperature allowed by Figure 2 are maintained. If the conditions of Figure 2 cannot be maintained, go immediately to step 4.9.

4.7 Verify natural circulation by one or more of the following methods.

Note: Indication of natural circulation may not stabilize for 15 to 30 minutes

W 39409

4.7.1 RCS ΔT increases and stabilizes.

4.7.2 Verify heat removal from OTSG's.

a. Turbine bypass valve positions.
b. Atmospheric dump valve positions.
c. Feedwater valve positions.
d. Feedwater flow.

Note: May not indicate for low decay heat case.

4.7.3 Incore thermocouple temperatures stabilize.

4.8 If natural circulation is confirmed by step 4.7, continue to remove decay heat with natural circulation.

4.9 If natural circulation cannot be confirmed by step 4.7, maintain the pressure and temperature limits of Figure 2 or restart one RCP in each loop.

4.10 If the limits of Figure 2 are exceeded and at least one RCP cannot be started, initiate HPI cooling per the guideline for small break LOCA.

4.11 If feedwater flow is lost:

4.11.1 Start an RCP.

4.11.2 Immediately attempt to restore feedwater flow. If feedwater flow is restored, transition to natural circulation may be attempted again starting at step 4.2.

4.11.3 If the limits of Figure 2 are exceeded or OTSG level drops below the low level limit, initiate HPI cooling per the guidelines for small break LOCA.

5.0 Expected Plant Response

5.1 General

This procedure is to be utilized in the event that there is a need to secure reactor coolant pumps with initial conditions as defined in Section 1.0. The principles which form the basis of this procedure are:

1. It is preferred to use reactor coolant pumps to supply core flow.

2. The hot leg of the RCS must be maintained subcooled to prevent formation of a steam bubble which could inhibit natural circulation.

W 39410

3. With natural circulation or reactor coolant pumps unavailable, the alternative is the HPI cooling mode.

5.2 Plant response to long term actions

5.2.1 Establishing desired conditions in the RCS Paragraphs 4.1 through 4.3 will establish conditions in the RCS favorable to the initiation of natural circulation.

First adequate subcooling must be established in the RCS (per Figure 1) preferably by raising RC pressure using pressurizer heaters or by lowering primary temperature using turbine bypass system valves.

Next, steam generator level must be raised to a level known to promote natural circulation. Caution must be exercised during the feeding of the generators since overcooling may result in a fairly rapid decrease in pressurizer level and RC pressure. The attached Figures 3 and 4 provide guidance on the expected system response during the feed operation.

It should be noted that it may be desirable to raise OTSG level above the 50% level in the operating range. Since it is known that the 50% level is adequate for initiation of natural circulation, this procedure was based on that level. If the decision is made to raise level above the 50% level caution must be exercised to prevent overcooling the RCS causing loss of RC pressure control or pressurizer level.

The final action prior to tripping the RC pumps is to stabilize pressurizer level. After the pumps are tripped, T_h will increase with T_c remaining essentially constant, resulting in a pressurizer level increase.

5.2.2 Securing the RC pumps

With steam generator pressure being automatically controlled, T_c can be expected to remain essentially constant after the RC pumps are secured. With the decrease in RC flow, core T_h will increase. As a point of reference, a typical increase of 25 - 35°F in T_h has been noted for past natural circulation transients. Figure 1 is intended to provide

W 39411

adequate subcooling before the RC pumps are secured, so that the hot legs will be at least 20°F subcooled after the pumps are off.

Stopping the last RCP causes the OTSG level control to shift to the emergency feedwater system and/or the emergency feedwater nozzles.

5.2.3 Verifying natural circulation

Natural circulation may be verified by:

1. Monitoring that T_c stays essentially constant with T_h increasing and then leveling out. The hot leg should remain subcooled.
2. Monitoring the heat removal from the OTSG's signifying that primary flow is available to remove core heat production.
3. Monitoring incore thermocouples (if available). The temperatures should rise and then level off and stabilize. The final temperature will be a function of many variables including decay heat level.

5.2.4 Contingencies

The actions in 4.8 are applicable if feedwater is available. With feedwater unavailable, the actions in 4.11 are applicable.

In Section 4.11, starting an RCP will provide additional time for the plant operator to try to reestablish feedwater flow by increasing core flow. If feedwater cannot be reestablished, HPI cooling must be initiated.

W 39412

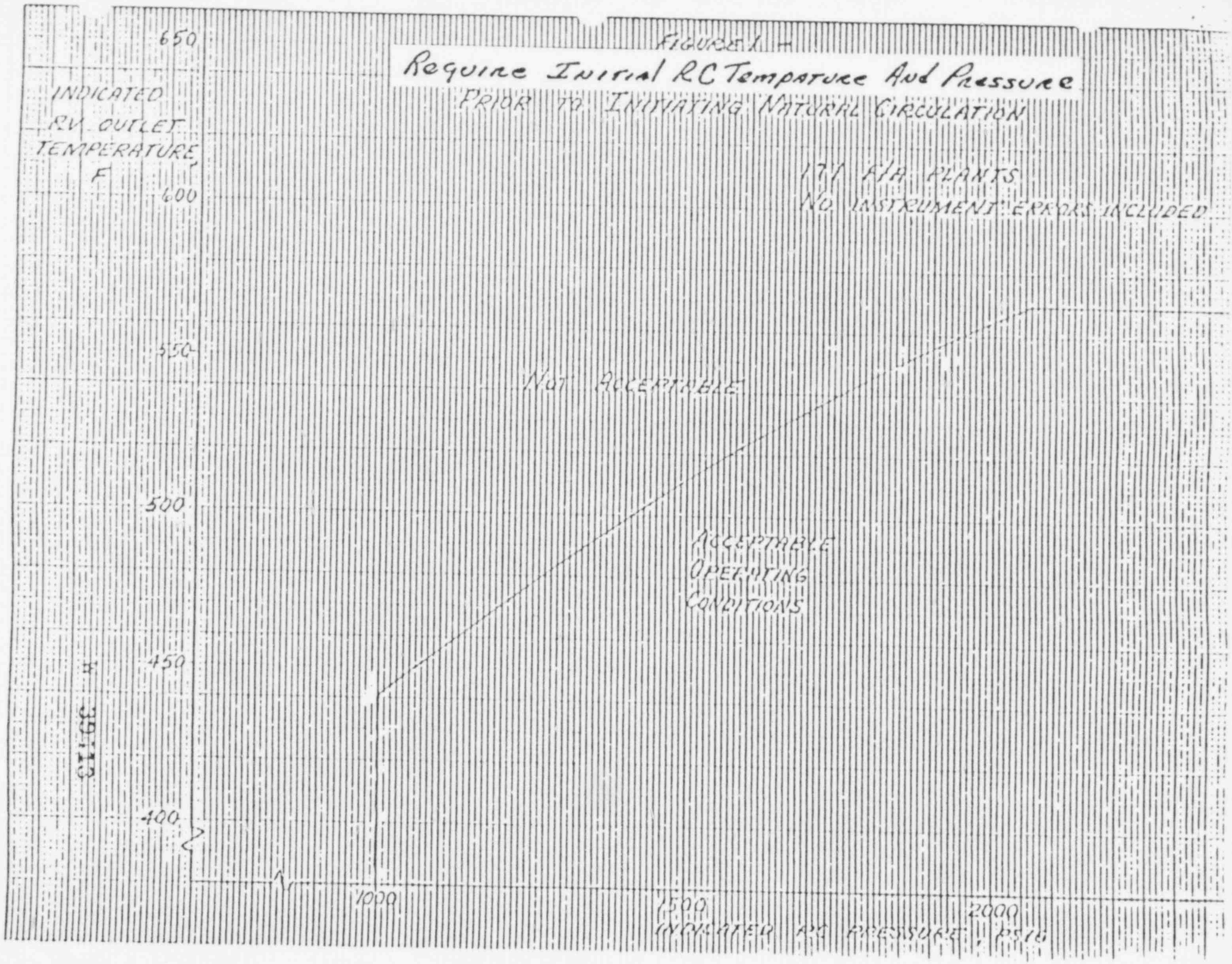
FIGURE 1
Require Initial RC Temparture And Pressure
Prior to Initiating Natural Circulation
177 FA Plants
No Instrument Errors Included
INDICATED RV OUTLET TEMPERATURE, F
650
600
550
500
450
400
Not Acceptable
Acceptable Operating Conditions
1000
1500
2000
INDICATED RC PRESSURE, PSIG

39113

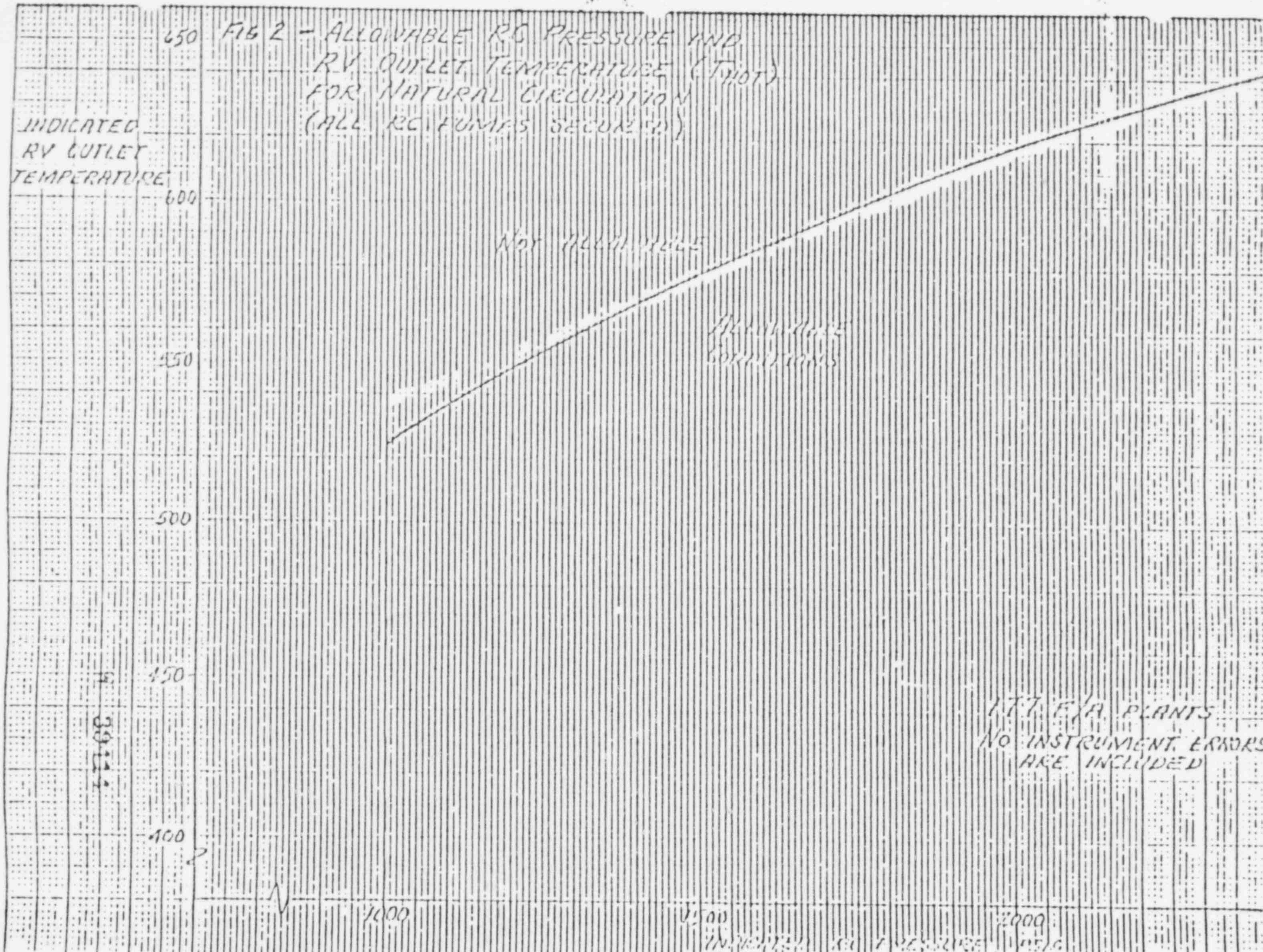
FIG 2 - ALLOWABLE RC PRESSURE AND
RV OUTLET TEMPERATURE (THOT)
FOR NATURAL CIRCULATION
(ALL RC PUMPS SECURED)
INDICATED
RV OUTLET
TEMPERATURE
650
600
550
500
450
400
NOT ALLOWABLE
ALLOWABLE
CONDITIONS
177 FA PLANTS
NO INSTRUMENT ERRORS
ARE INCLUDED
1000
1500
2000
39114

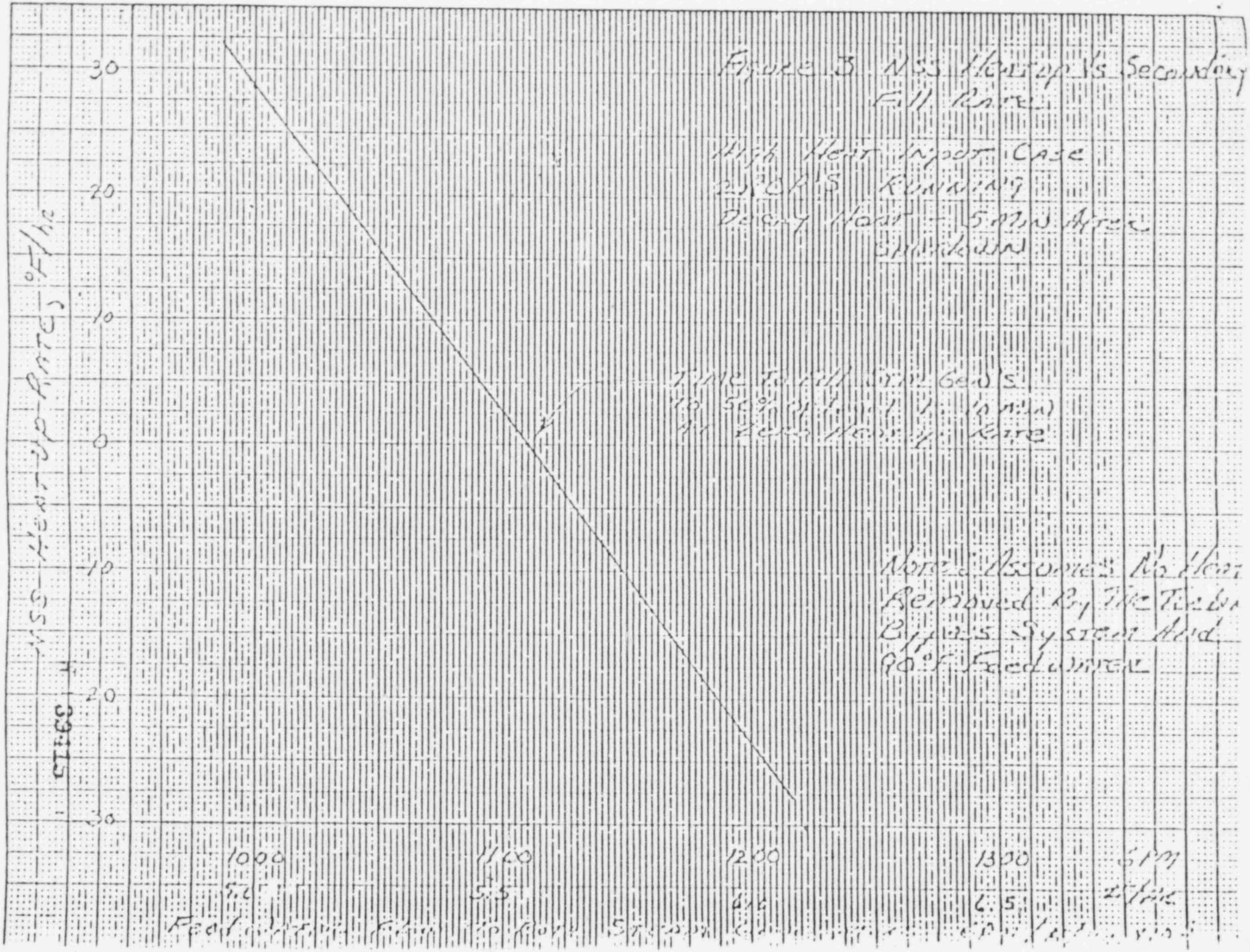

Figure 3 NSS Heatup Vs Secondary Fill Rate
High Heat Input Case
2 RCP'S Running
Decay Heat - 5 Min After Shutdown
Time To Fill OTSG's To 50% Level Is (min) At Zero Heatup Rate
Note: Assumes No Heat Removed By The Turbine Bypass System And 90°F Feedwater
30
20
10
0
10
20
30
NSS Heat-up Rate, °F/hr
1000
1100
1200
1300
GPM
5.5
6.5
Feedwater Flow
33415

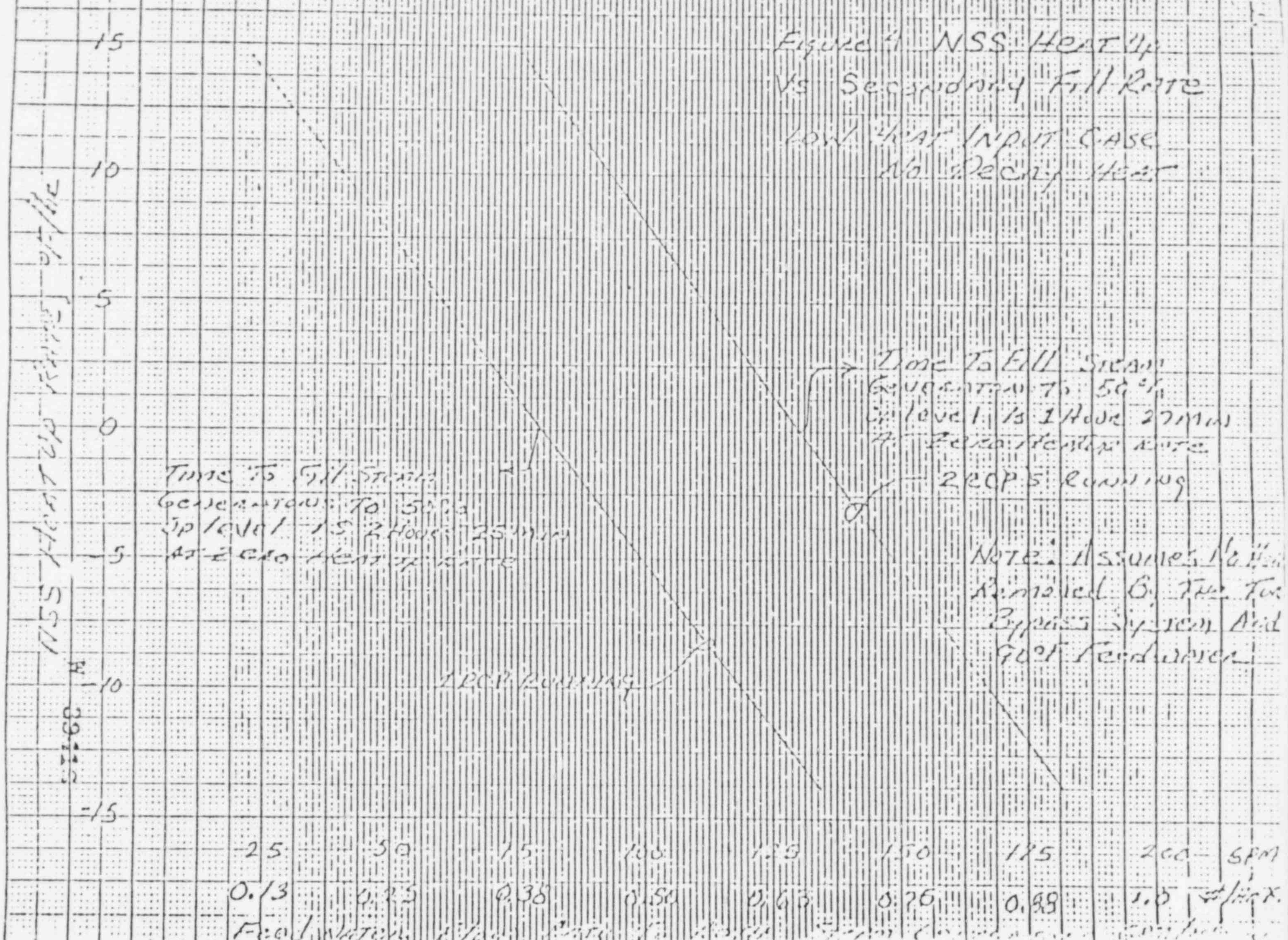

Figure 4 NSS Heat Up
Vs Secondary Fill Rate
Low Heat Input Case
No Decay Heat
Time To Fill Steam
Generator To 50%
level Is 1 Hour 27 Min
At Zero Heatup Rate
2 RCPS Running
Time To Fill Steam
Generators To 50%
Op level Is 2 Hours 25 min
At Zero Heatup Rate
Note: Assumes No Heat
Removed By The Turbine
Bypass System And
Cool Feedwater
1 RCP Running
NSS Heat Up Rate °F/Hr
15
10
5
0
-5
-10
-15
25 50 75 100 125 150 175 200 GPM
0.13 0.25 0.38 0.50 0.63 0.76 0.88 1.0 #/Hr X
Feed Water Flow Rate

1

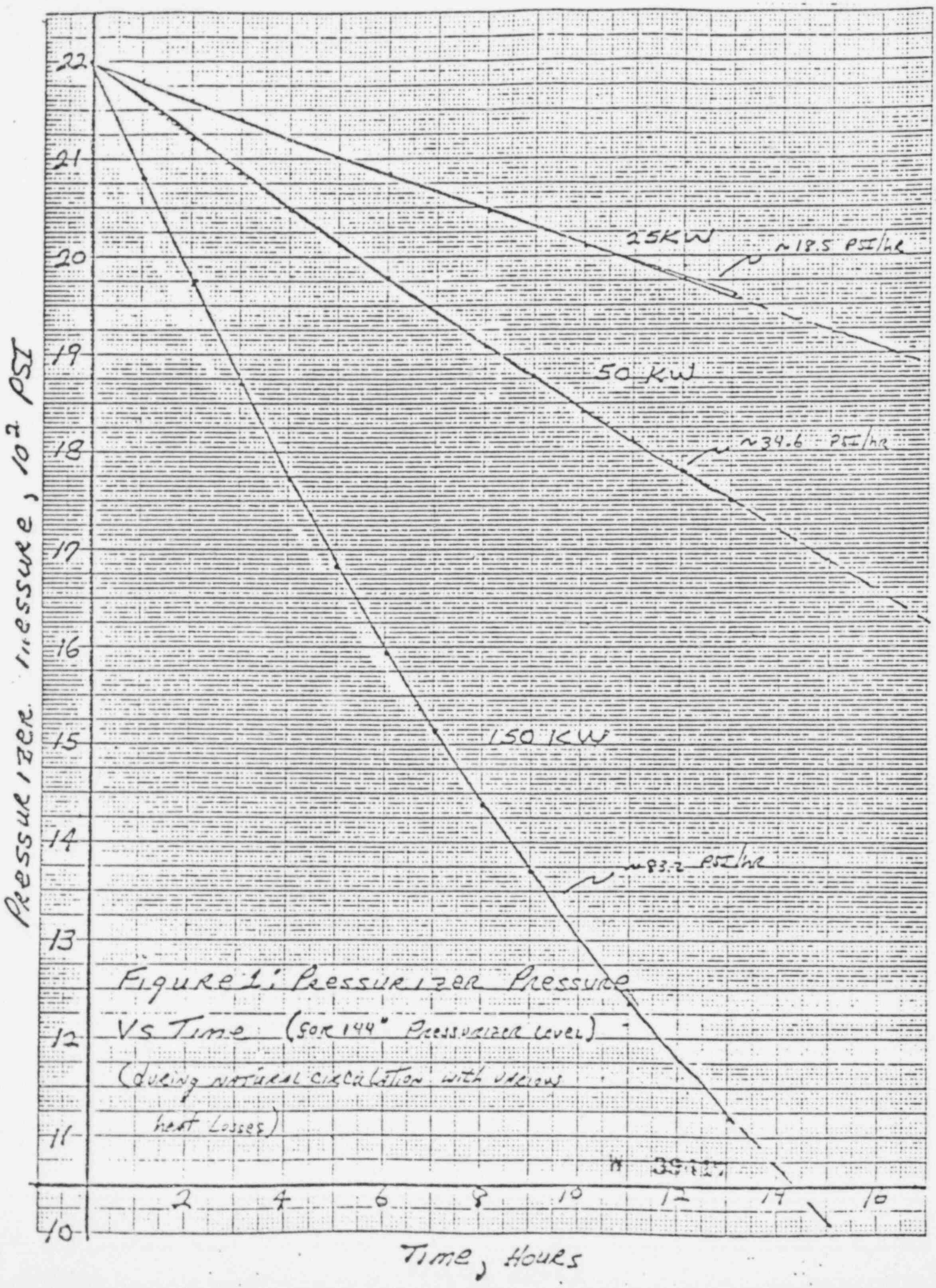

Figure 1: Pressurizer Pressure Vs Time (for 144" Pressurizer level)
(during natural circulation with various heat losses)

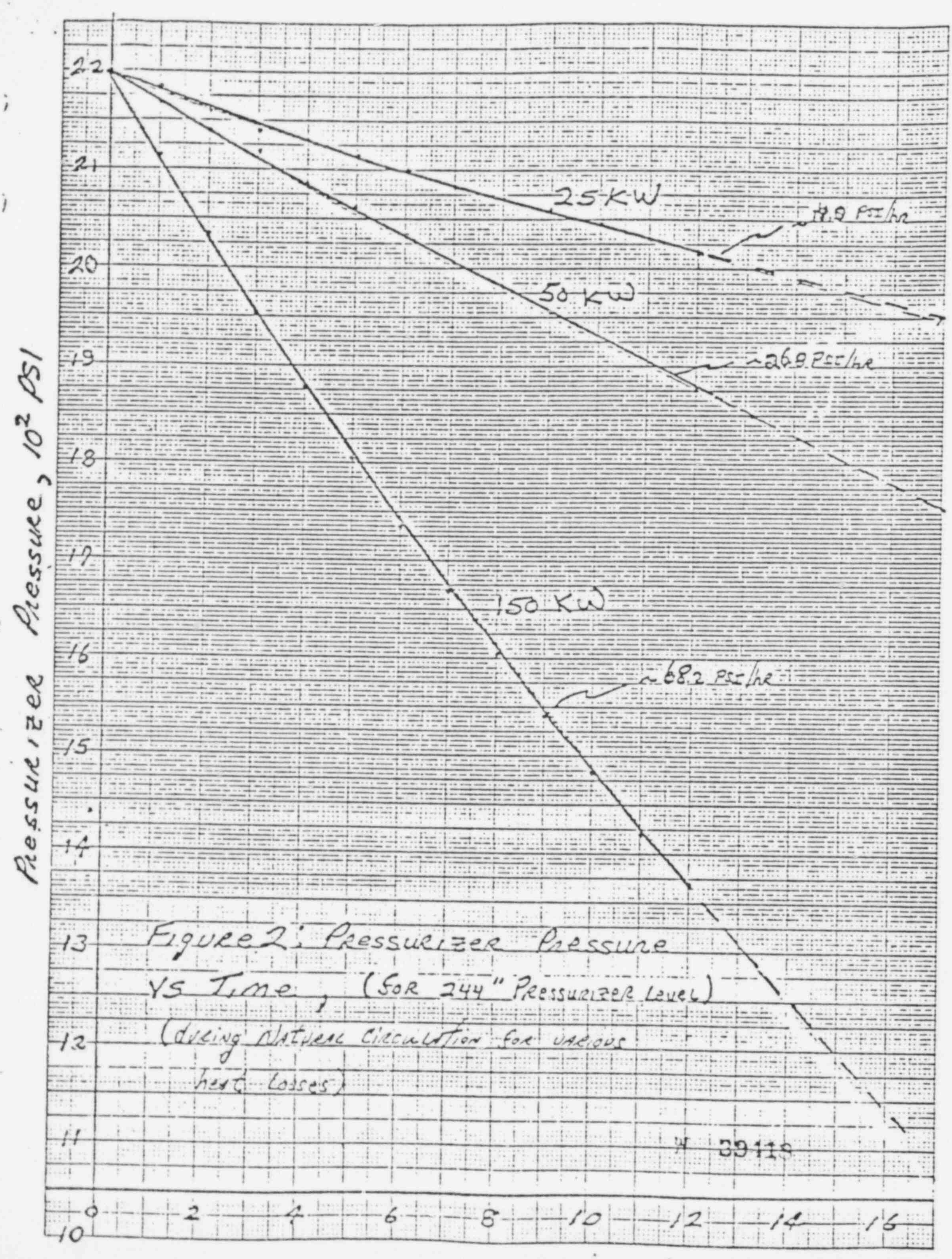

Figure 2: Pressurizer Pressure vs Time, (for 244" Pressurizer Level) (during Natural Circulation for various heat losses)

W 39119

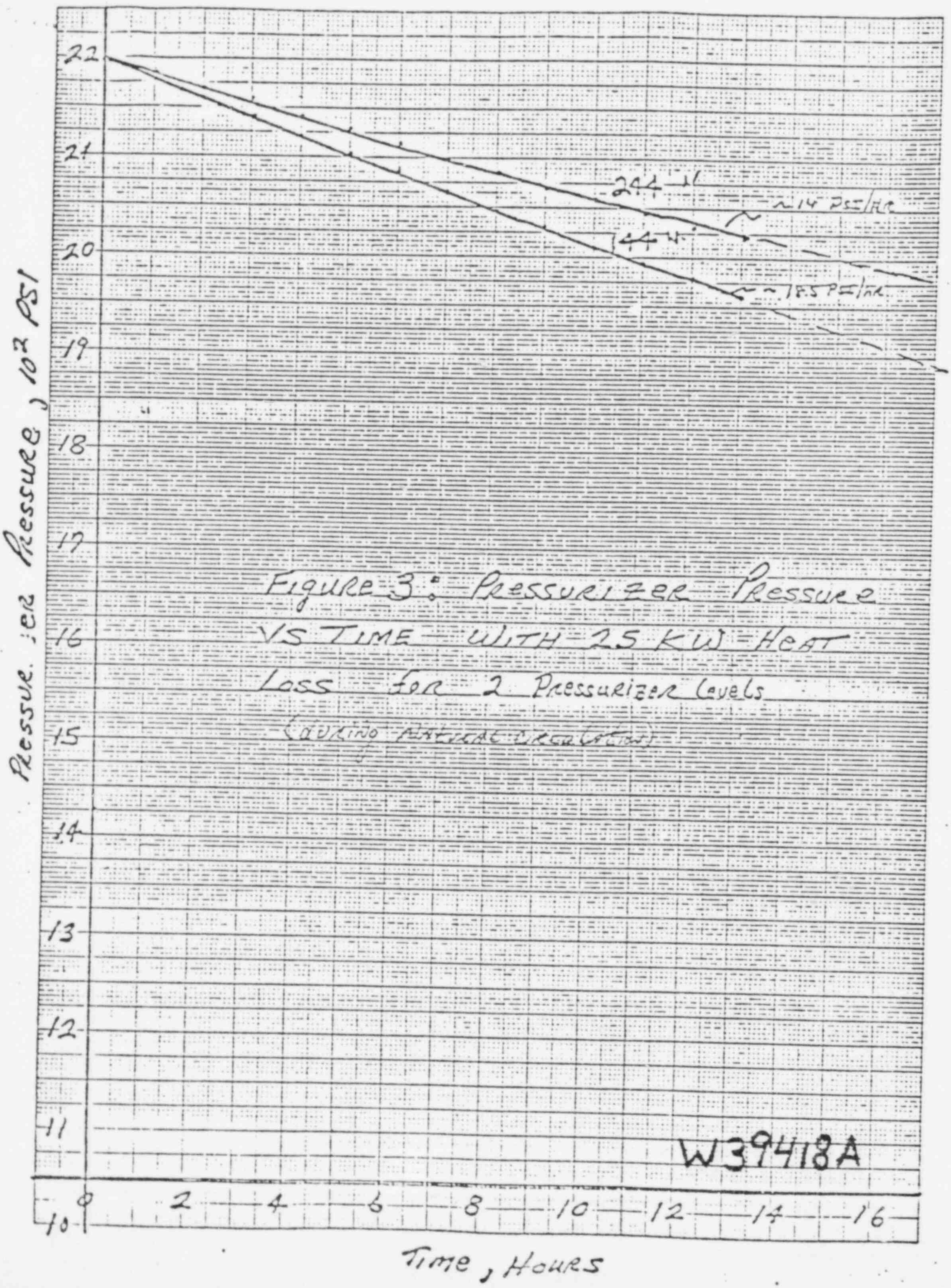

Figure 3: Pressurizer Pressure VS Time with 25 KW Heat Loss for 2 Pressurizer Levels

(During Natural Circulation)

W39418A

DA... from: [illegible] cc Warren
J. L. SEEL... ER

Bezilla - note
~~[illegible] - notes~~
[redacted] - note

3/12

Terry Mackey

1. Insure this guidance on Conax connectors is incorporated in our procedures.

2. Also insure appropriate flags are included in controlling procedures.

Jim

W 39419

MEMO from:

J. L. SEELINGER

SCM-II-124

cc B[illegible]
[illegible]

3/30/78

Bill Iels

To: John Wilburt
John Brummer
Terry Mackey
Dick Bensel

Subj: Power Escalation testing with 3 RCP's

1. Be aware of how this affects your respective areas.

Jim

W 39420

MEMO from: SON-II-125 CC Moore
J. L. SEELINGER
Bezilla
[illegible]

3/28/78

Terry Mackey

I believe this TCN & guidance has
been incorporated into our procedures.
Ensure it has. However, it should
be noted that if both pumps in a given
loop are secured with the rods energized
(even safeties cocked), the reactor will
trip. Ensure the B&W guidance is
tempered with this fact.
Ensure completed by 3/31

Jim

W 39421

MEMO from: SCM-II-126

J. L. SEELINGER

cc Skoolin
Sieglitz
Pittle
Rippon
Benzilla - task
Duffy - info

4/11/78

To: Dick Bensel

Subj: CRD O Rings

1. Ensure silicon O rings are on order. Initiate the paper or have electrical maintenance initiate the paper to order them.
2. Create generic maintenance procedures for stator replacement and/or vacuum drying of stators. Work with electrical maintenance as appropriate on this.
3. If procedures already exist, modify them as appropriate.

Target date - July 15

Jim

W 39422

SOM-III-127
(Addendum)

MEMO from:

J. L. SEELINGER

cc ~~[illegible]~~ B&W Lead
Bezilla
Floyd
Dubiel

4/18/78

Terry Mackey

Subj: B&W Chemistry Recommendations

1. BWST still has Na in it.
2. I see a problem for upcoming ISI tests on DH & BS pumps, i.e., we could contaminate previously flushed paths
3. Investigate
4. What is the late date we must have BWST cleaned up by
5. What are the alternatives
6. Continue to maintain the colored flow diagram of what is where and what water ~~outside of the R.B.~~ in (i.e. DH piping from BWST) to be flushed non used legs remains
7. Brief at morning meeting on 4/19

Jim

W 39423

MEMO from: CON-II-139 CC Bezella ~~[illegible]~~ Dubiel Lloyd

J. L. SEELINGER

4/10/78

Cover memo
To B&W SR

To: Terry Mackey

Subj: B&W Chemistry Numbers.

1. I'm giving this to you based upon your tech spec involvement.
2. Read this very carefully. In some cases it fits into the tech spec. In other cases it does not.
3. Mark up this copy to make it agree with ~~the~~ T.S. If ~~the~~ T.S. is more limiting than the letter so note. If the converse is true, flag this also.
4. Construct a cover letter for my signature to Lee Rogers. I would like to send this back to B&W & have them get it right or at least the story straight.

Note: the differences are subtle, but they are there.

Eventually we should incorporate the most limiting B&W or T.S. guidance into our startup procedures.

Jim

W 39424

MEMO from: SCM-II-131

J. L. SEELINGER

CC Beyilla
~~Staff~~ info

4/12/78

Dick Bensel

CRD Procedure Change per attached B+W letter

Make appropriate change to our procedures. Note that in the E.P. for CRD motor fault we used to go to jog. We were going to change this by going to sequence bypass and then going to jog.

Insure changes get made

Target date: Apr 20

Jim

W 39425

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

March 20, 1978

Mr. J. D. Luoma
GPU Service Corporation
260 Cherry Hill Road
Parsippany, New Jersey 07054

Subject: Thoria Fuel Fabrication and Irradiation Program

Dear John:

In response to your request for background information on thorium fuel cycles and the Indian Point 1 reactor, copies of references 1 and 2 are enclosed. A bibliography of reference material on thorium that may be useful in evaluating the Thorium Fuel Fabrication and Irradiation Program is also included.

Reference 3 is a comprehensive assessment of thorium fuel cycles in pressurized water reactors. Reference 1 was provided as a summary of this material, because reference 3 is quite lengthy.

Reference 4 is a good source book on thorium fuel fabrication and also contains an extensive bibliography on early thorium related work.

Reference 5 contains several recent papers on thorium fuel cycles. In particular, pages 957-970 discuss various thorium fuel cycle schemes.

References 6-9 provide additional discussion of thorium fuel cycles.

Other references are also available and can be provided if you desire further information.

Sincerely,

T. A. Coleman

T. A. Coleman

TAC/ahg

W 39426

Enclosures

cc: G. R. Bond R. M. Klingaman
[redacted] J. D. McCarthy

References:

1. Matzie, R. A. Rec, J. R. "Assessment of Thorium Fuel Cycles in Pressurized Water Reactors" - Summary of Reference 3 presented at International Conference on World Nuclear Power, Sponsored by American Nuclear Society and European Nuclear Society, Washington, D. C., November 14-19, 1976.

2. Kerr, J. M., et. al. Indian Point I-A ThO_2-UO_2 Fueled Reactor, AIChE National Meeting, February, 1978.

3. Shapiro, N. L., Rec., J. R. Matzie, R. A., "Assessment of Thorium Fuel Cycles in Pressurized Water Reactors". EPRI NP-359, Project 515-1, Final Report, February, 1977.

4. Weissert, L. R. and Schielc, G., Fabrication of Thorium Fuel Elements, American Nuclear Society publication, 1968.

5. Transactions American Nuclear Society, Vol. 27, 1977 Winter Meeting.

6. The Use of Thorium in Nuclear Power Reactors, USAEC, June, 1969 (WASH 1097).

7. Oosterkamp, W. J. (IEA-Brazil) The Potential of the Thorium Cycle in PWRs, Trans. Am. Nucl. Soc. (24), p. 220, (1976).

8. Matzie, R. A. and Rec, J. R., (C-E). Practical Considerations in the Use of a Thorium Cycle in PWRs. Trans. Am. Nucl. Soc. (24), p. 220-222, (1976).

9. Hellens, R. L. (C.E.) et. al. A Survey of Thorium Fuel Cycles in PWRs, Trans. Am. Nucl. Soc. (23), p. 272-273, (1976).

Babcock&Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

March 24, 1978

Mr. R. M. Klingaman
Manager, Generation Engineering
Metropolitan Edison Company
P.O. Box 542
Reading, PA 19603

Subject: Three Mile Island Generating Station - Unit No. 1
OTSG Orifice Plate Settings

Dear Mr. Klingaman:

During a January 1978 meeting held among B&W, Duke Power and GPU, there was considerable discussion regarding the pressure/power oscillations which have been observed in B&W 177 FA reactor plants and their possible effect on steam generator performance. B&W has had a continuing program to understand the nature of these oscillations, pinpoint the cause, and recommend actions to eliminate or reduce them. The purpose of this letter is to bring you up to date on the status of these efforts and to make recommendations for changes to the OTSG downcomer orifice plate settings which we believe will reduce or eliminate the oscillations.

The status of B&W efforts can be summarized as follows:

1. Review of data from some of our operating plants confirms the existence of steam pressure and power oscillations of the characteristic quarter hertz frequency. Further, a review of laboratory test results on model OTSG's confirms the existence of similar pressure oscillations. These pressure oscillations also result in a periodic variation in the boiling length in the OTSG.

2. We have performed stability analyses of both the OTSG itself as well as the secondary systems. These analyses confirm the susceptibility of the steam to oscillations at a quarter hertz. Further, the analytical work shows that an increase in the OTSG downcomer resistance can damp out the oscillations. Laboratory tests in the model boiler have confirmed that increased downcomer resistance will reduce the magnitude of the pressure oscillations.

W 39428

3. B&W has performed analyses to show the effect of increasing downcomer resistance on steam generator performance. We have concluded from this analytical work that an increase in the downcomer resistance to damp out pressure oscillations will not produce a detrimental effect on the margin to flooding the downcomer, the state of the fluid entering the bottom of the tube bundle, nor the integrity of the tubes themselves.

4. Past attempts to adjust the orifice plates in operating plant OTSG's have not been successful because the existing tooligm could not exert the required force to move plates which have been in service for some time. We have undertaken a re-design of the tooling and currently have in fabrication tools which will be capable of moving orifice plates that have been in service. These tools will be mockup tested prior to use in an actual OTSG.

While we have not yet conclusively shown that the pressure oscillations being observed can throw water high into the generator as postulated by MPR, we concur that subjecting the OTSG to such oscillations places an unnecessary burden on the generator and the elimination of these oscillations could result in improved performance of both the generator and overall plant. In view of the data we have accumulated as summarized above, we believe that increasing the downcomer resistance by closing down the orifice plates will reduce these pressure oscillations without producing any other undesirable effect on the steam generator. Accordingly, we recommend that Metropolitan Edison take action to close the orifice plates during the current TMI-1 outage.

We are prepared to meet with you on March 29, 1978 at the TMI site to elaborate on the items discussed above and present the technical information upon which this recommendation is based. We suggest the agenda shown in Attachment 1 which covers a review of the current oscillation experience in both operating plants and laboratory tests, the analytical and experimental verification that increasing the downcomer resistance will reduce the oscillation without causing other problems in the OTSG, and the status of tooling to accomplish the hardware modification. Included in the discussion of tooling would be our estimate of start date and schedule for this work for your consideration in the TMI-1 outage.

Because this subject is common to both TMI and Oconee, we would recommend representatives from Duke Power attend the meeting if it is agreeable to GPU and Metropolitan Edison. Please confirm the meeting date as satisfactory to your and do not hesitate to contact us if you have further questions.

Very truly yours,

Joel T. Janis
Service Manager

W 39429

JTJ/hh
Attachment
cc: JF Fritzen, JG Herbein, LL Lawyer, GP Miller
JP O'Hanlon, R. Wilson, D. Slear, LC Rogers, SL Smith

AGENDA

Met Ed/GPU/Duke/B&W Meeting
Regarding OTSG Orifice Plate Settings
TMI-1 Site

March 29, 1978

B&W Attendees

C. W. Pryor
D. H. Roy
L. H. Bohn
R. M. Gribble
L. E. Johnson
P. E. Perrone

Subjects of Discussion

1. Power/Pressure Oscillation Experience Overview - Operating Plant Experience . . . . D.H. Roy, R.M. Gribble

2. Pressure Oscillation Experience in Model Boiler Tests, Observed Effect of Increased Downcomer Resistance . . . . . L.E. Johnson

3. Effect of Closing Orifice Plate on Operational OTSG . . . . R.M. Gribble

4. Status of Field Change, Testing Procedure, Tooling Design, and Checkout. Estimated Time to Change Setting - Relation to Site Schedule . . . . L.H. Bohn

W 39430

CC Bezille - task end of May
~~[illegible]~~ - info

5/6/78

To: John Brummer

Subj: Incore Procedure Addition

1. Make appropriate procedure changes.

Jim Seelinger

W 39431

MEMO from:

J. L. SEELINGER

cc M. Bezilla 5/21/8

~~[illegible]~~

To: Rick Noll

Subj: B&W Letter on CRD Stator O-Ring.

1. Ensure U2 procedures reflect the proper guidance

Target date: June 2

Jim

W 39432

MEMO from:

J. L. SEELINGER

CC M Bozylla 5/21/78

[redacted]

"

To: Tom Morck

John Brummer

Subj: RCP Motor Oil Problems - B&W Letter

Take action indicated in the margins

Jim

W 39433

MEMO from:
J. L. SEELINGER

cc Bozilla
[redacted]

5/21

John Hilbish

Subj: B&W Letter on Core Power Distribution

1. Make appropriate procedure / Physics Test Manual Changes.

Target Date: 5/27

Jim

W 39434

MIMO from:

J. L. SEELINGER

cc Delt
Bezilla
[redacted]

5/21

John Hilbish

Subj: B&W Letter on Imbalance Alarm Limits

1. Make necessary procedure changes
2. Insure computer gets appropriately changed.

Target Date 5/27

Jim

W 39435

MEMO from:
J. L. SEELINGER

cc Gary Miller — recd 5/21
M. Bojilla
M. Shaffer - B&W item

Dick Dubiel

Subj: B&W Letter on Inadequate Chemistry Coverage.

1. Concerns point to significant generic problems.
2. I request you call Mike & set-up a time to brief GPM & me on 5/23 or 5/24 unless GPM has given you other direction.
3. Also I request you address the specific concerns in the B&W letter at the 5/23 morning meeting

Jim

W 39436

MEMO from:
J. L. SEELINGER

cc Bevilla 5/81

[redacted]

Tom Morck

Subj: B&W PR Answer on MCP's

1. What action is there for U2
2. Do the pumps need to be ~~taken~~ out 1 at a time now or can this be done at a no ~~mo~~ norme outage
3. Let me know by May 26

Jim

W 39-137

MEMO from:

J. L. SEELINGER

-II-157

cc Beyilla - text
[redacted]
Beiivel - info
Hartman - info

6/13/78

B+W B.R.

To: Ivan Porter

Subj: CRD Breaker PM System

1. Set up system similar to VI's per the attached B+W letter.
2. Due to the forecasted length of this outage the program should perhaps begin during it.
3. Charlie Hartman should be able to give you the details of the VI program.
4. Coordinate as necessary with Rippon & Rittle to get it started

Target date by which to establish program - July 14

Jim

W 39438

D. Seelinger

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

June 13, 1978

SOM-II-157
REM-I-353

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Mr. J. P. O'Hanlon
Superintendent, Unit I
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Preventive Maintenance of CRDCS Breakers

Gentlemen:

In the past, some of B&W's plants have experienced problems with CRDCS trip breakers. The problems have been traced to lack of preventive maintenance. B&W suggests that a planned, carefully executed, maintenance program be established using the maintenance program outlined in the Diamond Power CRDC System Vendor Manual. Particular attention should be directed to proper cycling, cleaning, and lubrication of the breakers. B&W further recommends that this program be scheduled at a minimum frequency of every refueling cycle and more frequently for plants during startup when the equipment is subjected to adverse environmental conditions.

We understand that you had a vendor representative on site to service these breakers during the past refueling outage of Unit I, and you intend making this preventive maintenance of the CRDCS breaker a regular feature of subsequent refueling outages.

W 39439

Dupe

B&W suggests that a similar preventive maintenance program be instituted for TMI-2 CRDCS breakers.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. L. Seelinger
R. J. Toole
J. B. Logan

DHO D2

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505

Telephone: (804) 384-5111

June 27, 1978

SOM-II-163
REM-I-360

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

[redacted]
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Mr. J. B. Logan
Superintendent, Unit 2
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Mr. J. P. O'Hanlon
Superintendent, Unit I
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: Evaluation of Control Rod Drive Cable Assemblies

Gentlemen:

As an on-going service, of B&W Nuclear Service, to improve the NSSS availability, B&W has recently investigated the loss of plant availability due to failures of control rod drive mechanism cable assemblies used between the transfer canal connectors and the control rod drive mechanism connectors.

Attached is a copy of the results and recommendations derived from a study of the operating history and problems associated with control rod drive mechanism cable assemblies. This study has focused on the modes of failure of the cables and cable connectors and on the causes for the failures.

Dupe

W 39441

The recommendations offered in this report are intended to improve reliability and extend the useful life of the cable assemblies. Our vendor, Bendix, has indicated a willingness to provide training to B&W and utility personnel in the use and maintenance of the cable assemblies, if there is need and interest.

Please advise if there is interest within your organization for training or if you have questions on the content of the report.

As these plants are aging, the availability of the spare parts is going to be scarce. With this general consideration in mind, Diamond Power Specialty Corporation has released Service Tip No. 1-78 relating to the availability of replacement parts for the General Electric AK-2A-25 circuit breakers used as control rod drive AC breakers.

Five copies of the service tip are attached for distribution amongst your engineering and operating staff.

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L C Rogers

L. C. Rogers
Site Operations Manager

LCR/SPM/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. L. Seelinger
J. D. Phinney
J. T. Janis
G. A. Kunder

W 39442

CONTROL ROD DRIVE MECHANISM CABLE ASSEMBLIES

B&W has investigated the loss of plant availability due to failures of CRDM cable assemblies used between the transfer canal connectors and the CRDM connectors. The results of this investigation of both known and assumed modes of failure are presented below.

The known modes of failure are insulation breakdown and low resistance between conductors. A third, but assumed, mode of failure is excessive resistance of the conductors.

The investigation revealed that insulation breakdown is due primarily to excessive heat. There are several causes of excessive heat including 1) Loss of cooling to the service structure and 2) high resistance in the connections caused by dirty, bent, or corroded contacts. Another cause of insulation breakdown is physical damage, usually resulting from improper handling or installation.

Low resistance between conductors can be a result of insulation breakdown, however, a separate mode of failure is moisture between conductors at the connections. Any improper mating of the plug and receptacle can allow moisture to enter and provide a potential cause for failure. Dirty, bent, or corroded contacts can result in misleading torque readings during connection makeup. Improper, or the lack of, lubricant can also give misleading torque readings. Damaged or dirty threads on the connectors can prevent proper seating. Connector O-rings that are damaged or that have lost their resilience also can prevent proper seating.

High resistance of the conductors can result in excessive heat and insulation breakdown or can lead to an open conductor. High resistance can be the result of improper handling techniques such as: pulling the cable by the connector, hanging additional weight on connected cables, excessive flexing of the cable, and treading on the cables.

B&W supplied cable assemblies are designed for the PWR containment environment and are fabricated to the current state of the art. It is evident from this investigation that to maximize the useful life of the cable assemblies, they must be carefully handled and protected to prevent damage. Specifically, to maximize the useful life of the cable assemblies, B&W recommends the following:

1) The connector O-rings should be replaced each time the connection is broken at the stators and periodically replaced at the bulkhead based on the service history.

2) Use the recommended lubricant and torque the connection to the value recommended by the cable assembly manufacturer (See Attachment 1).

3) Unmated connectors should be protected and maintained dry by the use of protective caps.

4) Do not allow personnel to walk on the cables.

5) If a liquid spill occurs, assure that the connectors are thoroughly dried.

6) Prior to making connections, inspect the connectors for dirt, damage, or corrosion. Correct any anomalies.

7) Develop a quality assurance/quality control program for maintenance and operation.

TORQUE VALUES

ELECTRICAL CONNECTORS

Bendix Electrical Components Division

Sidney, N.Y. 13838

November 1972 L-725-2

1. INTRODUCTION.

2. This publication contains minimum and maximum torque values recommended by The Bendix Corporation, Electrical Components Division, Sidney, New York 13838.

3. Maximum torque values are primarily governed by strength of material used to fabricate threaded components. Maximum values listed in this publication were computed to protect threads and other bearing surfaces which could be damaged by excessive torque.

4. Minimum torque values were computed to assure proper mating of connectors and/or accessories (providing threaded areas are properly lubricated). Connectors procured less lubrication on the thread are controlled by special part numbers. For these connectors, the user is responsible for proper thread lubrication. Lubricants should be compatible with the connector and its application. Two such applications would be for use in a N_2O_4 (nitrogen tetroxide) atmosphere and a LOX atmosphere. Lubricants which are N_2O_4 resistant and may be used in this atmosphere are Molykote X-15 and Fluorocarbon Lubricant 95-1. Molykote X-15 is available from the Alpha Molykote Corporation, Stamford, Connecticut 06904, and is recommended for use with the JTN series connector. Fluorocarbon Lubricant 95-1 is available from the Dixon Corporation, Bristol, Rhode Island 02809, and is recommended for use with AN/MS type alumilite finish connectors. Lubricants which are LOX compatibile and may be used in a LOX atmosphere are Molykote X-15 and KELF-90, available from the Minnesota Mining & Manufacturing Co., St. Paul, Minnesota 55101.

5. Metal to metal seating of mated parts should occur where a flat gasket is used for an end seal. In some instances, when a cable accessory or MS3057B type clamp is used with maximum diameter cable, a metal to metal seating may not occur on the initial tightening. A second tightening is therefore necessary after cable has been allowed to cold flow (approximately 12 hours).

6. Torque values are listed by thread sizes. We recommend the Bendix 3C Connector Catalog for mating and accessory thread diameters for each connector shell size.

NOTE: See torque value notes following paragraph 10 for accessories having three threads or less.

7. TORQUE VALUES.

8. The torque values listed in Table I are for UN(Handbook H-28) threads. These values apply to connector's coupling nut, jam nut, cable clamp, back shell or accessories.

W 39445

9. Column "A" of Table I lists the minimum and maximum values for threads on aluminum die cast parts such as MS or MS Modified connectors.

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TORQUE VALUES FOR

Thread Size	COLUMN "A" Threads On Die Cast Aluminum Parts				COLUMN "B" Threads on Extruded Or Machined Aluminum And Steel Parts			
	Inch Lbs.		Foot Lbs.		Inch Lbs.		Foot Lbs.	
	Min.	Max.	Min.	Max.	Min.	Max.	Min.	Max.
0.3125	20	26			70	75		
0.3750	20	26			70	75		
0.4375	20	26			70	75		
0.5000	20	26			70	75		
0.5625	26	32			80	85		
0.6250	26	32			90	95		
0.6875	30	36			100	110	8	9
0.7500	34	40			110	120	9	10
0.8125	40	46			120	130	10	11
0.8750	46	50			140	150	11	12
0.9375	50	55			150	160	12	13
1.0000	55	60			160	170	13	14
1.0625	60	65			190	200	16	17
1.1250	70	75			210	230	18	19
1.1875	75	80			240	260	20	21
1.2500	80	85			260	280	21	23
1.3125	85	90			280	300	24	25
1.375	90	95			300	320	24	25
1.4375	100	110	8	9	300	325	26	28
1.5000	100	110	8	9	325	350	28	30
1.625	110	120	9	10	350	375	28	30
1.750	120	130	10	11	375	400	30	32
1.875	140	150	11	12	400	425	32	34
2.0000	150	160	12	13	425	450	36	38
2.0625	160	170	13	14	450	475	38	40
2.1250	170	180	14	15 (1)	475	500	40	42
2.2500	170	180	14	15	500	525	42	44
2.3125	180	190	15	16	525	550	44	46
2.375	190	200	16	17 (2)	550	575	46	48
2.500	200	210	17	18	575	600	48	50
2.525	210	220	18	19 (3)	600	650	50	55
2.750	220	230	18	19	650	700	55	60
2.875	230	240	19	20	700	750	55	60
3.000	240	250	20	21	750	800	60	65

10. Column "B" of Table I lists the minimum and maximum values for threads on extruded or machined aluminum and steel parts. The torque value for hex mounting nuts on all jam nut receptacles is listed in column "A" of Table I. The hex nuts are machined but the torque value is reduced due to the limited amount of threads.

NOTE: Torque value for all sizes of accessories having three threads, or less and involving a modified major/minor diameter (crest removed) is 30-35 in. lbs. This applies to many PT, JT, and LJT accessories.

NOTE: Torque values for accessories having three threads or less, but not having a modified major/minor diameter (crest removed) are as follows:
Shell size 8 through 19; 50 ± 5 in. lbs.
Shell size 20 through 25; 100 ± 10 in. lbs.

11. CONNECTORS WITH DOUBLE STUB THREADS (MIL-S-23747). Minimum torque values in [Ta]ble II will assure main joint sealing for connectors, such as QWL and QWLD series, incorporating double stub threads (MIL-S-23747).

(1) PI CABLE BULKHEAD JAM NUT TORQUE

(2) POWER CABLE BULKHEAD JAM NUT TORQUE

(3) PRIMARY T.C. CABLE BULKHEAD JAM NUT TORQUE

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TABLE II

Thread Size	Inch Lbs.		Foot Lbs.	
	Min.	Max.	Min.	Max.
TORQUE VALUES FOR DOUBLE STUB THREADS MAIN JOINT SEALING*				
0.875	150	170	12	14 ④
1.000	170	190	14	16
1.125	230	260	19	21
1.250	280	300	23	25
1.375	325	350	26	28
1.500	350	400	30	32
1.750	400	425	32	36 ⑤
2.000	425	450	38	42 ⑥
2.250	500	600	44	48
2.500	600	700	50	55 ⑦
2.750	700	800	60	65
3.000	800	850	65	70

*Table of values applies except for those series mentioned in NOTE below.

<u>NOTE</u>: Die cast single key of the AN double stub thread (65- & 66- series connectors) requires using <u>column "A" of Table I</u> for torque values. The small single key of the PC double stub thread connectors requires using <u>column "B" of Table I</u> for torque values.

12. CONNECTORS WITH BAYONET COUPLING. All Bendix bayonet coupling connectors (Pygmy) including those with special coaxial arrangements will meet the coupling and uncoupling forces of specification MIL-C-26482. Before mating, receptacles must be suitably mounted. Coupling nuts on plugs must be properly lubricated.

Maximum engagement and minimum disengagement forces applied to the coupling ring shall be within the limits specified in Table III.

TABLE III

Connector Size	Inch Lbs. Max. Engagement	Inch Lbs. Min. Engagement
TORQUE VALUES FOR BAYONET COUPLING		
8	8	1
10	12	1
12	16	2
14	20	4
16	24	4
18	28	4
20	32	6
22	36	7
24	44	7

④ DRIVE T.C. CONNECTION TORQUE
⑤ PI CONNECTION TORQUE
⑥ POWER CONNECTION TORQUE
⑦ PRIMARY T.C. CONNECTION TORQUE

W 39447

NUCLEAR EQUIPMENT SERVICE TIPS BULLETIN

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

CRDC SERVICE TIP NO. 1-78

RENEWAL PARTS FOR AK-2A-25 CIRCUIT BREAKERS

General Electric has advised Diamond Power that complete breakers as renewal items will be obtainable through 1986. All parts making up these breakers will also be available through 1986. In addition, all recommended spare parts will be available through 1996. These recommended spare parts are shown in General Electric bulletin GEF-4149G, pages 3 and 4.

In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

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Babcock & Wilcox
Diamond Power

DP-7096

NUCLEAR EQUIPMENT **SERVICE TIPS** BULLETIN

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

CRDC SERVICE TIP NO. 1-78

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Dupl

W 39449

NUCLEAR EQUIPMENT **SERVICE TIPS** BULLETIN

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

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In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39450

CRDC SERVICE TIP NO. 1-78

RENEWAL PARTS FOR AK-2A-25 CIRCUIT BREAKERS

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In general, those parts that will be available through 1996 are the parts which would tend to wear through mechanical operation, carry electrical current or be required for the successful operation of the breaker. Such parts as backplates and support parts would probably not be available through 1996.

It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39451

NUCLEAR EQUIPMENT **SERVICE TIPS** BULLETIN

DIAMOND POWER SPECIALTY CORP.
LANCASTER, OHIO

CRDC SERVICE TIP NO. 1-78

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General Electric has advised Diamond Power that complete breakers as renewal items will be obtainable through 1986. All parts making up these breakers will also be available through 1986. In addition, all recommended spare parts will be available through 1996. These recommended spare parts are shown in General Electric bulletin GEF-4149G, pages 3 and 4.

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It is of the utmost importance to include the summary number found on the breaker nameplate with any request for spare parts. This summary number will define the exact makeup of the circuit breaker in question.

W 39452

MEMO from:

J. L. SEELINGER

cc Beyilla
Shaffer B&W item 8/29/78

Ron Warren

Subj: Attached B&W Letter on
OTSG Oscillation

1. Have one of your people get with tech.
2. See if GPU will incorporate in test procedure or if they want Met Ed SOP; if SOP have prepared.
3. Let me know which way we will go by Sept 6.

Jim

W 39153

G. Seelinger

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

August 29, 1978

SOM-II-183

Mr. R. J. Toole
Test Superintendent
GPU Service Corporation
Post Office Box 480
Middletown, PA 17057

Mr. L. L. Lawyer
Manager, Generation Operations
Metropolitan Edison Company
Post Office Box 542
Reading, PA 19603

Mr. G. P. Miller
Station Superintendent
Metropolitan Edison Company
Post Office Box 480
Middletown, PA 17057

Subject: OTSG Oscillation Data

Gentlemen:

With the resumption of plant startup testing on TMI-2, B&W considers that the opportunity to collect performance characteristics of the OTSG and feed system will be available. The intent of such a study is to determine the value of adjusting the OTSG feed flow restricting orifice plate position which may allow reduction of plant pressure oscillations during operations at less than full power. A similar adjustment of the orifice plate has been accomplished in other B&W units with very encouraging preliminary results. B&W considers that adjustment at TMI-2 should not be made until feed system and steam generator performance characteristics at the current orifice plate position are evaluated.

In order to obtain sufficient baseline data on plant oscillations in the 20% to 90% FP range, B&W requests that information be collected and forwarded to Lynchburg as outlined in the enclosed procedure. This data will then serve to help determine what adjustments will be required, if any, to the OTSG orifice plates on TMI-2. This data can be collected during a scheduled power transient and it need not be scheduled as a complete maneuver from 20% to 90% FP.

W 39454

Dupe

If you have any further questions, please do not hesitate to contact me.

Very truly yours,

L. C. Rogers
Site Operations Manager

LCR/JV/bay

cc: L. R. Pletke
W. H. Spangler
G. K. Wandling
J. G. Herbein
R. M. Klingaman
J. B. Logan
J. L. Seelinger

W 39455

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TECHNICAL DOCUMENT

TEST SPECIFICATION

62 - 1005379 - 00
Doc. ID – Serial No., Revision No.

for

OTSG OSCILLATION DATA

W 39456

Dupe

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

RECORD OF REVISION

NUMBER 62-1005379-00

REV. NO.	CHANGE SECT/PARA.	DESCRIPTION/CHANGE AUTHORIZATION
00		Original Issue - J. Veenstra Nuclear Service

PREPARED BY James Veenstra (NAME) Engineer (TITLE) DATE 8/21/78

REVIEWED BY Robert Winks (NAME) Principal Engineer DATE 8/23/78

APPROVED BY [illegible] for W.H. Spangler (NAME) Mgr. PSS (TITLE) DATE 8/23/78

W 39457

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TABLE OF CONTENTS/EFFECTIVE PAGE LIST

NUMBER 62-1005379-00

W 39458

1.0 TEST OBJECTIVE

1.1 To measure selected plant parameters while unit is slowly increasing/decreasing power level from 20 to 90%.

2.0 ACCEPTANCE CRITERIA

2.1 The fluid temperature in the upper downcomer shall not fall below the temperature limits determined by Attachment 1.

3.0 CONDITIONS PRIOR TO TEST

3.1 All four RC pumps are in operation.

3.2 The plant will operate normally with the ICS in the automatic mode.

3.3 Plant power level will be increased/decreased at nominal rates with 5 minute holds at 5% FP increments from 20% to 90% FP to attain quasi-steady state conditions.

3.4 Plant Computer data groups and Brush recorders are set up to record the data required in Section 5.0.

4.0 SPECIAL PRECAUTIONS

4.1 In the event of undesirable system oscillations at a particular partial power level, the operator is free to place any ICS station in manual control mode to increase plant stability.

4.2 Rather than terminate the test, the operator should provide brief periods of normal system operation (ICS in automatic mode) at selected partial power levels if the system oscillations are still unacceptable.

4.3 The upper downcomer temperature should not exceed the limits established by Paragraph 2.0.

5.0 DATA ACQUISITION

5.1 Quasi-steady state performance data to be recorded (B&W Reactimeter or equivalent)

1. OTSG A steam pressure
2. OTSG B steam pressure
3. OTSG A steam temperature
4. OTSG B steam temperature
5. OTSG A operate level
6. OTSG B operate level
7. Loop A T hot leg
8. Loop B T hot leg
9. Loop A T cold leg A-1
10. Loop A T cold leg A-2

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11. Loop B T cold leg B-1
12. Loop B T cold leg B-2
13. Loop A feedwater flow
14. Loop B feedwater flow
15. Loop A feedwater temperature
16. Loop B feedwater temperature
17. Loop A lower downcomer temperature
18. Loop A upper downcomer temperature
19. Reactor power either NI-5,6,7,8
20. Loop A startup level
21. Loop B startup level
22. Loop A ~~upper downcomer temperature~~ shell Temperature
23. Loop B lower downcomer temperature
24. Loop B upper downcomer temperature

Record every .2 sec. for 2 minutes at every 5% power plateau between 20 and 90%. In addition, obtain a Heat Balance power at each test data acquisition plateau.

5.2 Reactor power is to be increased or decreased in 5% FP increments with a 5 minute hold at each level from 20% to 90% FP.

5.3 The following channels are to be recorded on Brush recorder strip charts:

1. OTSG A steam pressure
2. OTSG A operate level
3. OTSG B operate level
4. Loop A feedwater flow
5. Loop B feedwater flow
6. NI power

6.0 RECOMMENDED METHOD

6.1 Power Increase

6.1.1 At a plant power level of 20%, begin to record the system oscillations data.

NOTE: It is not necessary to record data continuously for the duration of the entire test.

6.1.2 At each 5% change in plant power level, obtain a complete set of measured data.

6.1.3 Proceed at a nominal rate from 20 to 90% or to the power level cutoff point.

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6.1.4 The test is complete when the system oscillations have been measured at the highest plant power level not exceeding the power level cutoff point. However, as a minimum, data should be obtained up to 85% power.

6.2 Power Decrease

6.2.1 At a plant power level of 90%, begin to record the system oscillations data.

NOTE: It is not necessary to record data continuously for the duration of the entire test.

6.2.2 At each 5% change in plant power level, obtain a complete set of measured data.

6.2.3 Proceed at a nominal rate from 90 to 20%.

6.2.4 The test is complete when the system oscillations have been measured at the 20% power plateau.

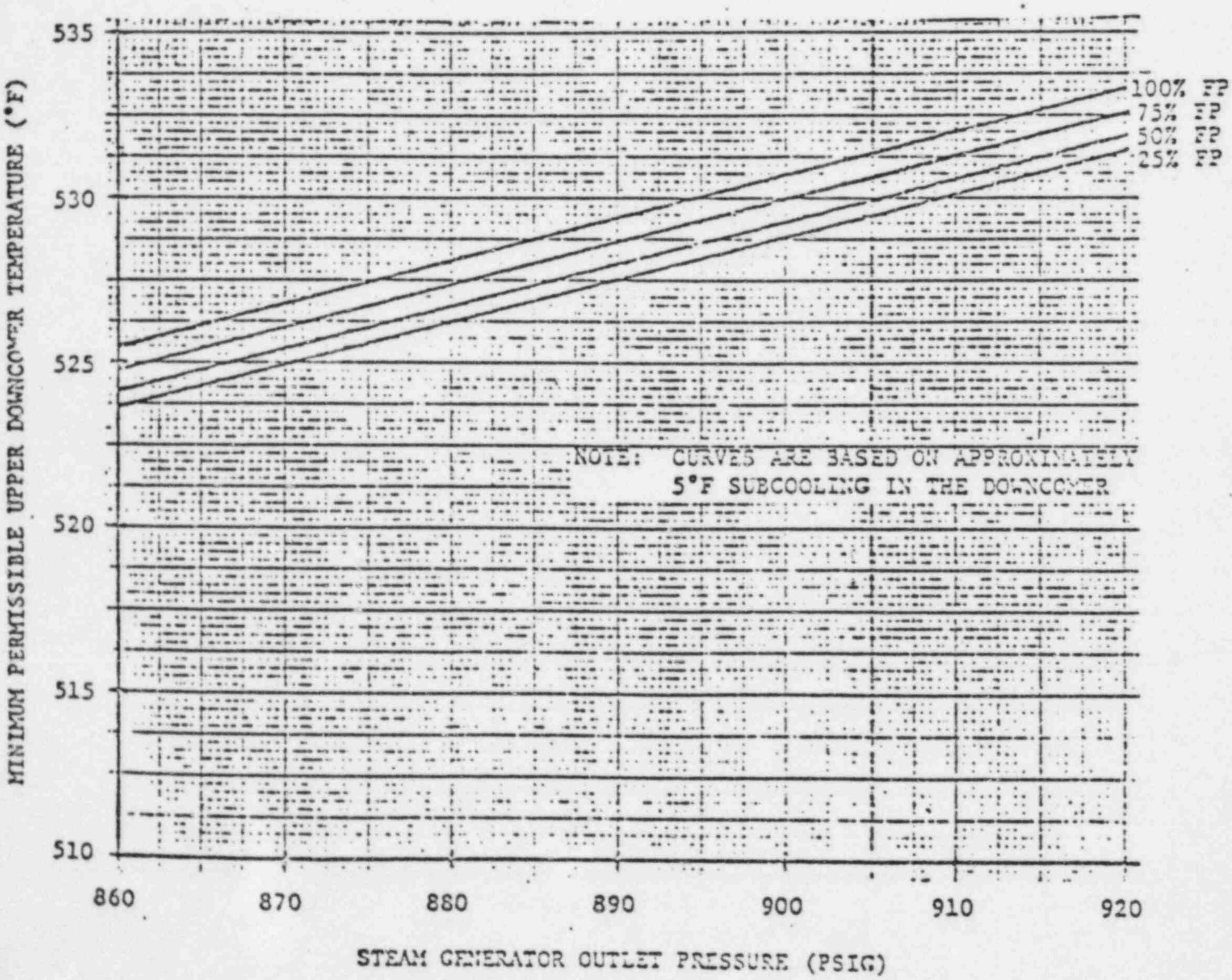
W 39461

BABCOCK & WILCOX
NUCLEAR POWER GENERATION DIVISION

TECHNICAL DOCUMENT

NUMBER
62-1005379-00

Attachment 1: Minimum Permissible Upper Downcomer Temperature as a Function of Power and Steam Generator Outlet Pressure



W 39462.