LONG ISLAND LIGHTING COMPANY
SHOREHAM NUCLEAR POWER STATION
P.O. BOX 618, NORTH COUNTRY ROAD • WADING RIVER, N.Y. 11792

Direct Dial Number

June 28, 1983 SNRC-924

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Revised Responses to Acceptance
Review Requests 212.14, 212.76,
212.104, 223.38, 223.72 and 223.73
Final Safety Analysis Report (FSAR)
Shoreham Nuclear Power Station - Unit 1
Docket No. 50-322

Dear Mr. Denton:

LILCO's efforts are directed at resolving all outstanding NRC Staff concerns associated with their acceptance review. Therefore, we are providing as an enclosure to this letter, current and complete information to the above subject acceptance review requests. These six requests fall into three subject areas: 1) Level 8 instrumentation (212.104) 2) Recirculation Pump Trip electrical schematics (212.14, 212.76 and 223.73) and 3) Rod Sequence Control System electrical schematics (223.38 and 223.72).

These revised responses and the associated figures will be included in the next FSAR update and are provided herein to allow for a timely staff review. Should you have any questions concerning the enclosed subject material do not hesitate to call this office.

Very truly yours,

J. L. Smith

J. L. Smith
Manager, Special Project
Shoreham Nuclear Power Station

B001 1/9 Limited Dist

GJG:bc

Enclosure

cc: J. Higgins
All Parties Listed in Attachment 1

FC-8935.1

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PDR ADOCK 05000322
A PDR

ATTACHMENT 1

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Request 212.104 (RSP) (15.0):

In analyzing anticipated operational transients, the applicant has taken credit for plant operating equipment which has not been shown to be reliable as required by General Design Criterion 29. The staff has discussed the application of this equipment generically with General Electric. In these discussions General Electric has stated that the most limiting transient that takes credit for this equipment is the excess feedwater event. Further, General Electric has stated that the only plant operating equipment that plays a significant role in mitigating this event is the turbine bypass system and the Level 8 high water level trip (closes turbine stop valves). We will allow the use of the turbine bypass and Level 8 high water level trip systems in mitigating transients except for the turbine trip and generator load rejection without bypass transients which are currently minimum critical power ratio—limiting.

To assure an acceptable level of performance, it is the staff's position that this equipment be identified in the plant Technical Specifications with regard to availability, set points, and surveillance testing. The applicant must submit his plan for implementing this requirement along with any system modifications that may be required to fulfill this requirement.

Response:

In discussions between GE and the NRC on November 20 and 21, 1978, GE presented the results of transient analysis performed to design basis accident condition assumptions (i.e., equipment availability). The analysis indicated that failure to give credit to the Level 8 turbine trip and the main turbine bypass system could result in a difference in the critical power ratios of 0.02 and 0.08, respectively. Therefore, these postulated conditions could not result in unacceptable impacts on the health and safety of the public.

The Level 8 instrumentation will be subject to technical specification requirements associated with the feedwater system/main turbine trip system. The proposed Shoreham technical specification addresses this concern in Section 3.3.9 of the limiting conditions for operation.

The turbine steam bypass system and stop valves are furnished with the main turbine generator and have exhibited high reliability on existing nuclear and fossil fueled operating units.

Normal operating procedures require that the valves be functionally exercised periodically in accordance with vendor recommendations. This effort will ensure valve operability and provide adequate assurance that the valves will operate when required.

Request 212.14 (5.2.2.7.6 15.1.1.1):

In order to assess the effectiveness of PRT, the following information, as it relates to the Shoreham plant, is required.

a. Provide turbine trip transient analyses assuming scram on turbine stop valve, position, recirculation pump trip, and no turbine bypass. The results should include pressure, CPR, neutron flux, surface heat flux and fuel pellet temperature as a function of time for the following conditions:

   1) without PRT, using design conservatism factors on void coefficients and scram reactivity curve.

   2) without PRT, using expected operational factors on the scram reactivity curve and void coefficient (i.e., best estimate of actual factors at EOC for an equilibrium core).

   3) part 1) with PRT

   4) part 2) with PRT

b. Provide generator trip transient analyses assuming scram on turbine control valve position, with bypass, without recirculation pump trip, and using design conservatism factors on the scram reactivity curve and the void coefficient (i.e. more mild, more frequently expected transient). The results should include pressure, CPR, neutron flux, surface heat flux and fuel pellet temperature for the following conditions:

   1) with PRT

   2) without PRT

c. On currently operating plants, what is the relief valve operation frequency that has been experienced? That is, what is the number of individual valves that will be expected to open per unit time of operation, counting an event that causes 3 valves to open as 3 instances of operation, etc?

d. What is the expected increase in the part c) frequency with PRT for the Shoreham plant?

e. What does available data show regarding failure frequency of S/R valves to reseat properly following opening?

f. Is there any reason to expect that the S/R valves designed for the Shoreham plant will have a different failure (to close) frequency?

g. Will the specified number of operating cycles for the lifetime of the Shoreham S/R valves be increased to account for the additional duty due to PRT operation. If so, what is the increase both in number and as a percent of total duty cycles?

h. Will the specified lifetime number of rapid depressurization cycles for the reactor pressure vessel be increased to allow for improper PRT operations (i.e. stuck open relief valves)? If so, by how many cycles?

i. Will the specified lifetime total number of thermal cycles for the suppression pool be increased to account for both partial (proper PRT operation) and complete (improper PRT operation, stuck relief valve) blowdowns due to PRT? If so, by how many cycles?

j. Will the specified number of dynamic pressure cycles for the suppression pool and associated piping be increased to account for PRT operation? If so, by how many cycles?

k. Provide analyses for all accidents for which the PRT system is expected to function. Provide information to show whether criteria which must be met for each accident are still satisfied assuming the PRT functions as designed. Accidents considered should include LOCA (steam line and recirculation line breaks) and ATWS.

l. Provide a summary table of advantages and disadvantages of the PRT system. For example, our tentative and incomplete table included: ADVANTAGES; decreased duty cycle (power, temperature, pressure, etc.) on fuel, decreased duty cycle (pressure) on RPV assuming proper PRT operation. DISADVANTAGES; increased plant complexity, increased duty cycle (cooldown transient for improper PRT operation), increased duty cycle on suppression pool.

m. Provide the results of comparative analyses (with and without PRT) of expected radiation exposures, both to plant operating personnel and offsite, for normal operations, transients and accidents.

n. What scram reactivity (or related) experiments have been run? How do the results of any such tests compare quantitatively with calculational results from the latest models? Is there operating plant transient information which would be relevant to the assessments for the need for PRT? If so, please identify and discuss.

Response

PRT has been removed from the Shoreham plant design due to technical and licensing concerns. As an alternative to PRT, General Electric has designed a recirculation pump trip (RPT) feature to operate on turbine trip and generator load rejection transients.

A description of the recirculation pump trip system is contained in Section 7.6.1.3. An analysis of conformance to Regulatory Guides and Industry Standards is found in Section 7.6.2.3. The drawings which are applicable to the RPT system are specified in the response to Request 212.76.

Request 212.76(15.1):

Several transient analyses in this section take credit for the recirculation pump trip (RPT) function. Description of these trip circuits is not presented in the FSAR. Discuss operation of this system as it was assumed to function to mitigate the transients analyzed in Section 15. Describe the signals or mechanisms that actuate the tripping devices and include setpoint values.

If the recirculation pump trip is initiated by tripping the M-G set field breaker or a generator output breaker (rather than the drive motor breaker), the coastdown apparently would be more rapid than that presented in NEDO-10802. Provide details of the analytical models and methods used to predict recirculation system behavior during transients which take credit for RPT. Justify the time delay used for two-pump trip. Discuss plans for startup tests which will justify the analytical methods and verify that core flow during the coastdown is in agreement with that predicted.

A loss of off-site power (all grid connections) associated with a LOCA may cause a generator load rejection and RPT at the time of LOCA initiation. With regard to a concern that installation of RPT may compromise the recirculation pump coastdown assumed in the LOCA analysis, show that RPT would not cause a coastdown of the recirculation pumps more rapidly than that assumed.

Response:

1. The RPT system design and functional requirements are described in Sections 7.6.1.3 and 7.6.2.3. Figure 7.6.1-3 is a simplified functional control design which shows the actuation signals and the permissive functions for recirculation pump trip.

   Figures 4.2.3-8A to -C, 5.1.2-1A to -C, 5.2.2-2A and -B, 7.7.1-1A to -C, 7.2.1-1A to -D, 7.3.1-7A to -E, 7.3.1-12A to -E, 7.7.1-2A to -G, and 7.7.1-5A to -E have been revised to include the RPT system.

2. The RPT system trips the redundant line breakers which are located between the motor generator and the pump motor. Thus, for events during which the RPT system is outlined, the coastdown characteristic of the pump, pump motor, and the pump shaft is an important aspect of the transient. A description of these characteristics follows:

   Section 2.12 of NEDO-10802 includes the following equation:

$$\frac{\pi J}{30\ g_c} \quad \frac{dn}{dt} = \Delta T$$

Request 223.73:

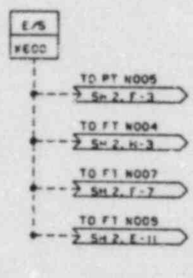
Supplement the description of the recirculation pump trip presented in Revision 5 to the FSAR with revised electrical schematics.

Response:

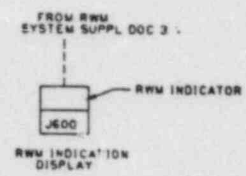
Revised drawings to reflect the incorporation of recirculation pump trip are specified in the response to Request 212.76.

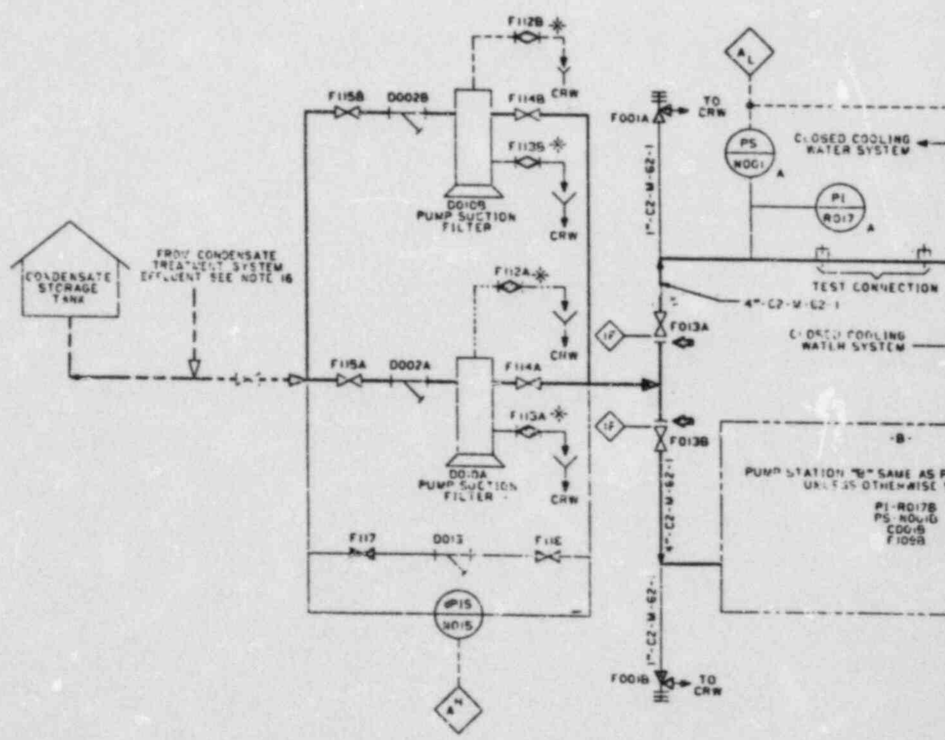
SUPPLEMENTAL DOCUMENTS

1. NUCLEAR BOILER SYS P&ID
2. CRD HYDRAULIC SYS PD
3. CRD HYDRAULIC SYS FCD
4. CRD HYDRAULIC SYS PIPING ARGMT
5. REACTOR PROTECTION SYS (RPS) IED
6. PIPING & INSTRUMENT SYMBOLS
7. PROCESS INSTRUMENT PIPING AND TUBING SPECIFICATION
8. PRESSURE INTEGRITY OF PIPING AND EQUIPMENT PRESSURE PARTS
9. CRD HYDRAULIC SYS DESIGN SPEC
10. REACTOR RECIRC SYS P&ID
11. CRD INSTRUMENT DATA SHEETS
12. REACTOR MANUAL CONTROL ELEM DIAG
13. WATER SAMPLING
14. WATER QUALITY
15. REACTOR WATER CLEANUP SYS P&ID
16. REACTOR RECIRC PUMP AND MG SET ELEM DIAG
17. CRD HYD SYS INSTR ELEM DIAG



POWER DISTRIBUTION





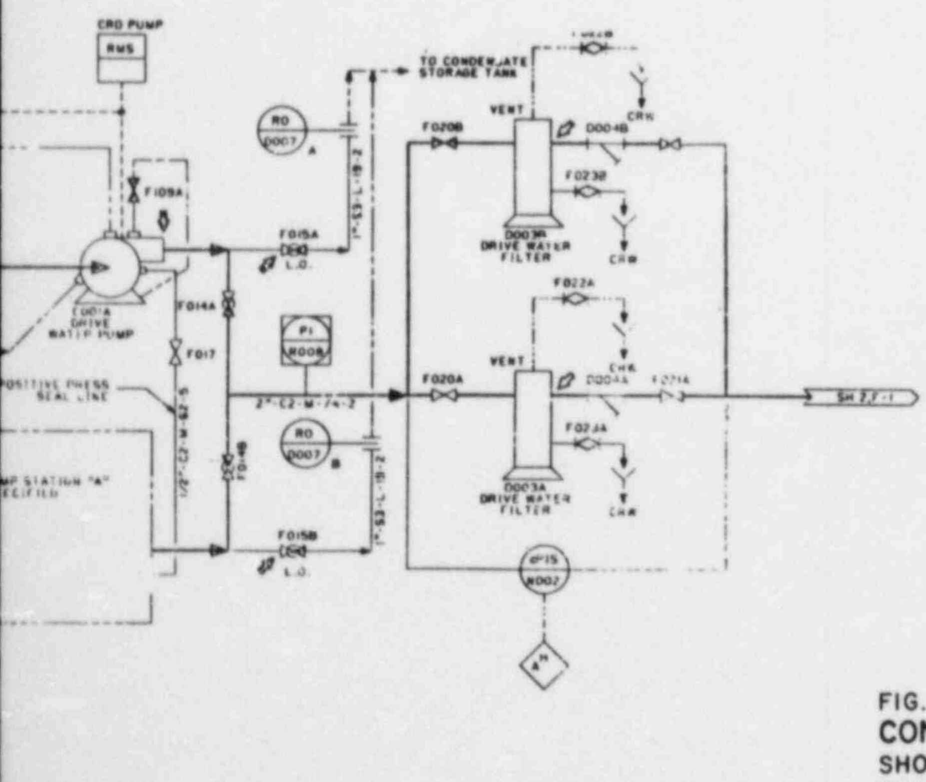
MPL NO. C11-1010

NOTES:

1. VALVE F007A-A CLOSES ON DRIVE INSERT SIGNAL. VALVE F007A-B CLOSES ON DRIVE WITHDRAWAL SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING. (SHEET 2, J-9)
2. REACTOR PRESSURE SENSING LINE, SH 2, K-13.
3. STABILIZING VALVE F007B IS AN ALTERNATE FOR STABILIZING VALVE F007A. (SHEET 2, J-9)
4. PROVIDE VENT VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM HIGH POINTS.
5. PROVIDE DRAIN VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM LOW POINTS.
6. PROVIDED FOR SYSTEM FLUSHING. (SHEET 2, H-9 & H-10)
7. AVAILABLE FOR TEMPORARY CONNECTION FOR INSTRUMENT FLUSHING. NO PERMANENT PIPING CONNECTIONS TO BE MADE TO THIS VALVE. (SHEET 2, E-8)
8. EXCEPT AT POINTS OF CONNECTION WITH NED SUPPLIED EQUIPMENT OR PIPING, THE PIPING DESIGNER SHALL SIZE PIPES IN CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.
9. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS, SEE SUPPL DOC 11.
10. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY SYSTEM NUMBER (C11) UNLESS OTHERWISE NOTED.
11. CRD NITROGEN AND AIR LINES SHALL BE OF NON-CORRODING MATERIAL.
12. THESE VALVES MUST BE OPEN FOR DRIVE MOVEMENT.
13. MULTIPLE ORIFICES CONNECTED IN SERIES AS SHOWN IN PURCHASE PART DRAWING OF ORIFICE D008. THE PRESSURE DROP ACROSS EACH ORIFICE IS 250 PSI AT PUMP RUN-OUT CONDITION. SEE MPL FOR THE QUANTITIES OF ORIFICES. VALVE F034 SUPPLEMENTS THE ORIFICES D008 FOR THE REQUIRED PRESSURE DROP.
14. THE SCRAM DISCHARGE VOLUME ARRANGEMENT SHOWN IS FOR REFERENCE ONLY. SEE CRD DESIGN SPEC FOR REQUIREMENT.
15. PIPING QUALITY CLASS EXTENDS TO CONNECTIONS WITH HCU. HCU DIAGRAM IS SHOWN FOR INFORMATION ONLY. FOR QUALITY CLASS OF THE HCU, SEE GROUP CLASSIFICATION DIAGRAM.
16. SOURCE OF CRD SYSTEM WATER SHALL BE NORMALLY FROM CONDENSATE TREATMENT SYSTEM. CONDENSATE STORAGE TANK IS THE ALTERNATE SOURCE IF CONDENSATE TREATMENT SYSTEM IS NOT IN OPERATION. FOR DETAILED DESIGN REQUIREMENTS FOR SOURCE AND QUALITY OF WATER, SEE SUPPL DOC 9.
17. CONTINUOUS FLOW TO REACTOR SAMPLE STATION SHALL BE APPROXIMATELY ONE (1) LITER PER MINUTE, MINIMUM AND SHALL BE TWO (2) LITERS PER MINUTE, MAXIMUM.
18. EXHAUST WATER FLOW FROM ONE MOVING DRIVE IS DISPERSED VIA THE HYDRAULIC CONTROL UNITS OF OTHER NON MOVING DRIVES TO THE REACTOR PRESSURE VESSEL.
19. SCRAM DISCHARGE DRAIN MUST BE SUBMERGED DUE TO POTENTIAL RELEASE OF STEAM AND RADIOACTIVE NONCONDENSIBLES.
20. THE F159B VALVE SHALL BE ADJUSTED SO THAT THE F180 VALVE AND THE F181 VALVE FULLY CLOSE AT LEAST FIVE (5) SECONDS AFTER THE F010 AND F011 VALVES, RESPECTIVELY, DURING A FULL CORE SCRAM.

NOTES (CONT)

21. THE F159A VALVE SHALL BE ADJUSTED SO THAT THE F010 AND F011 VALVES STARTS TO OPEN AT LEAST FIVE (5) SECONDS AFTER THE F180 AND F181 VALVES, RESPECTIVELY, UPON THE RESET OF A FULL CORE SCRAM.
22. THE SCRAM DISCHARGE VOLUME VENT LINE SHALL BE ROUTED VIA A DEDICATED LINE TO THE NON SUBMERGED DISCHARGE. THE VENT SYSTEM PIPING SHOULD NOT CONTAIN SIGNIFICANT FLOW RESTRICTIONS WHICH COULD INHIBIT VENTING OF THE SCRAM DISCHARGE VOLUME TO THE EXTENT OF LIMITING THE SCRAM DISCHARGE VOLUME DRAIN RATE. THE DESIGN OF THE NON-SUBMERGED VENT DISCHARGE MUST CONSIDER THE POTENTIAL FOR THE RELEASE OF RADIOACTIVITY DUE TO THE DISCHARGE OF NON-CONDENSIBLES, WATER AND STEAM WHICH MAY OCCUR DURING THE PERIOD AFTER SCRAM PRIOR TO VENT VALVE CLOSURE AND UPON SCRAM RESET WHEN THE VENT VALVES ARE REOPENED WITH THE SCRAM DISCHARGE VOLUME PRESSURIZED.
23. TO PREVENT LOOP SEALS FROM OCCURING IN THE RELATIVELY SMALL DIAMETER LINES OF THE SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN SYSTEM, A CONTINUOUS DOWNWARD PITCH AWAY FROM THE SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES MUST BE MAINTAINED. THERMAL EXPANSION EFFECTS SHOULD BE ADDRESSED IN THE DESIGN OF THE VENT AND DRAIN SYSTEMS.
24. THE DESIGN OF THE SCRAM DISCHARGE VOLUME (SDV), ASSOCIATED VENT AND DRAIN SYSTEM PIPING AND COMPONENTS MUST CONSIDER THE POTENTIAL PRESSURE, TEMPERATURE AND TRANSIENT LOADING WHICH MAY RESULT FROM: (A) THE DISCHARGE TO THE SCRAM DISCHARGE VOLUME AND DOWN THE VENT AND DRAIN LINE DURING A FULL SCRAM PRIOR TO VENT AND DRAIN VALVE CLOSURE; (B) THE PRESSURIZATION OF THE SCRAM DISCHARGE VOLUME TO REACTOR VESSEL PRESSURE FOLLOWING VENT AND DRAIN VALVE CLOSURE; AND (C) DEPRESSURIZATION OF THE SCRAM DISCHARGE VOLUME UPON SCRAM RESET AND OPENING OF THE SCRAM DISCHARGE VOLUME VENT AND DRAIN VALVES.
25. VACUUM BREAKER VALVE (F120) AND VENT VALVES (F010 AND F180) TO BE ON HIGH POINT OF VENT LINE. VACUUM BREAKER VALVE SHALL OPEN ON A DIFFERENTIAL PRESSURE SETTING OF NO GREATER THAN FIVE INCHES OF WATER. THE LOCATION OF THE VACUUM BREAKER SHOULD CONSIDER THE POTENTIAL RELEASE OF RADIOACTIVE NONCONDENSIBLES, WATER AND STEAM THAT COULD OCCUR IF THE VACUUM BREAKER VALVE WERE TO FAIL IN AN OPEN POSITION WHEN THE VENT SYSTEM PIPING IS PRESSURIZED.
26. INSTRUMENT LOW POINT PIPING TAP SHOULD BE ≤ 12 INCHES ABOVE THE SCRAM DISCHARGE INSTRUMENT VOLUME REFERENCE DATUM ELEVATION.
27. THE ELEVATION WITHIN THE SCRAM DISCHARGE INSTRUMENT VOLUME CORRESPONDING TO THE SCRAM INITIATION SET-POINT SHALL BE ≥ 10 INCHES BELOW THE LOWEST ELEVATION OF THE HORIZONTAL SCRAM DISCHARGE VOLUME PIPING.
28. SCRAM DISCHARGE VENT AND DRAIN VALVE CONTROL ROOM INDICATORS SHALL INDICATE OPEN WHEN BOTH VALVES ARE OPEN AND SHALL INDICATE CLOSED WHEN EITHER VALVE IS CLOSED.
29. THE PNEUMATIC PIPING FROM NEEDLE VALVE, F159B & SOLENOID VALVE, F009, TO THE VENT AND DRAIN VALVES, F181, F011, F010, F180, SHALL BE ROUTED AND SUPPORTED TO MINIMIZE THE POSSIBILITY OF PIPE CRIMP, AND ALSO IT SHOULD BE OF ADEQUATE PIPE OR TUBING THICKNESS AND MATERIAL TO MINIMIZE THE CONSEQUENCES OF PIPE CRIMP.



LEGEND

SOLENOID OPERATED VALVE W/BUILT-IN FLOW CONTROL

BLOCK TYPE NEEDLE VENT VALVE (ANGLE PATTERN)

LDS ------ LEAK DETECTOR SWITCH

RWM ------ ROD WORTH MINIMIZER

TS_C ------ TEMPERATURE SCANNER

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PRC APERTURE CARD

FIG. 4.2.3-8A
CONTROL ROD DRIVE HYDRAULIC SYSTEM
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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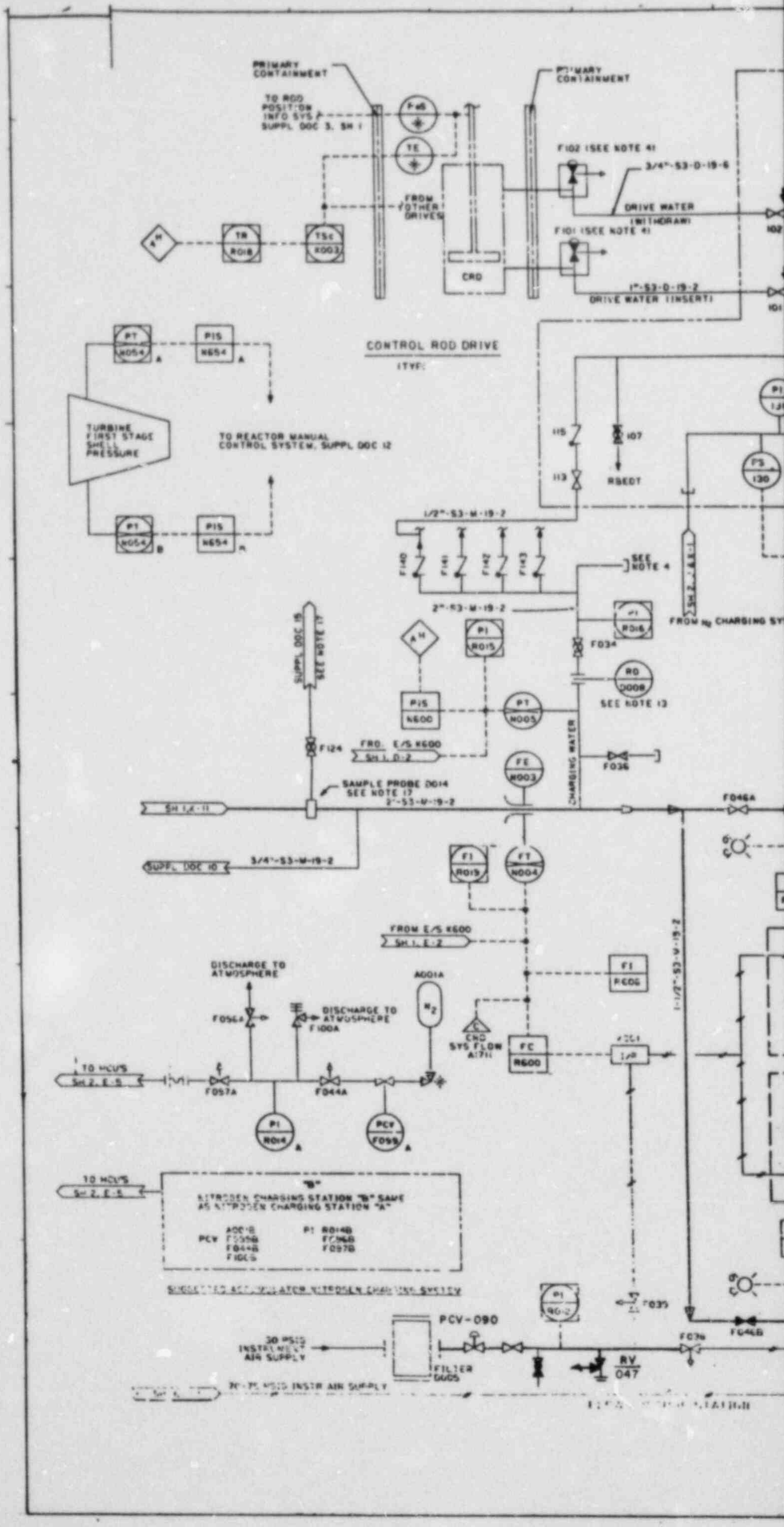
PRIMARY CONTAINMENT
TO ROD POSITION INFO SYS. SUPPL DOC 3, SH 1
PRIMARY CONTAINMENT
FROM OTHER DRIVES
CRD
F102 (SEE NOTE 4)
3/4"-S3-D-19-6
DRIVE WATER (WITHDRAW)
F101 (SEE NOTE 4)
1"-S3-D-19-2
DRIVE WATER (INSERT)
CONTROL ROD DRIVE
(TYP)
TR R018
TSC X003
PT N054 A
PIS N654 A
PT N054 B
PIS N654 B
TURBINE FIRST STAGE SHELL PRESSURE
TO REACTOR MANUAL CONTROL SYSTEM, SUPPL DOC 12
RBEDT
1/2"-S3-M-19-2
F140
F141
F142
F143
SEE NOTE 4
2"-S3-M-19-2
FROM N2 CHARGING SYS
PI R016
F034
RO D008
SEE NOTE 13
PI R015
PIS N600
PT N005
CHARGING WATER
SUPPL DOC 15
SEE NOTE 17
F124
FRO. E/S K600 SH 1, D-2
F036
FE N003
SAMPLE PROBE D014
SEE NOTE 17
2"-S3-M-19-2
SH 1, E-11
F046A
SUPPL DOC 10
3/4"-S3-M-19-2
FI R013
FT N004
FROM E/S K600 SH 1, E-2
FI R606
DISCHARGE TO ATMOSPHERE
DISCHARGE TO ATMOSPHERE
F056A
F100A
A001A
N_2
CRD SYS FLOW ALTN
FC R600
TO HCU'S SH 2, E-5
F057A
F044A
PI R014 A
PCV F059 A
1-1/2"-S3-V-19-2
TO HCU'S SH 2, E-5
"B"
NITROGEN CHARGING STATION "B" SAME AS NITROGEN CHARGING STATION "A"
A001B
PCV F059B
F044B
F100B
PI R014B
F056B
F057B
F039
PI
PCV-090
F046B
30 PSIG INSTRUMENT AIR SUPPLY
FILTER D605
RV 047
75 PSIG INSTR AIR SUPPLY

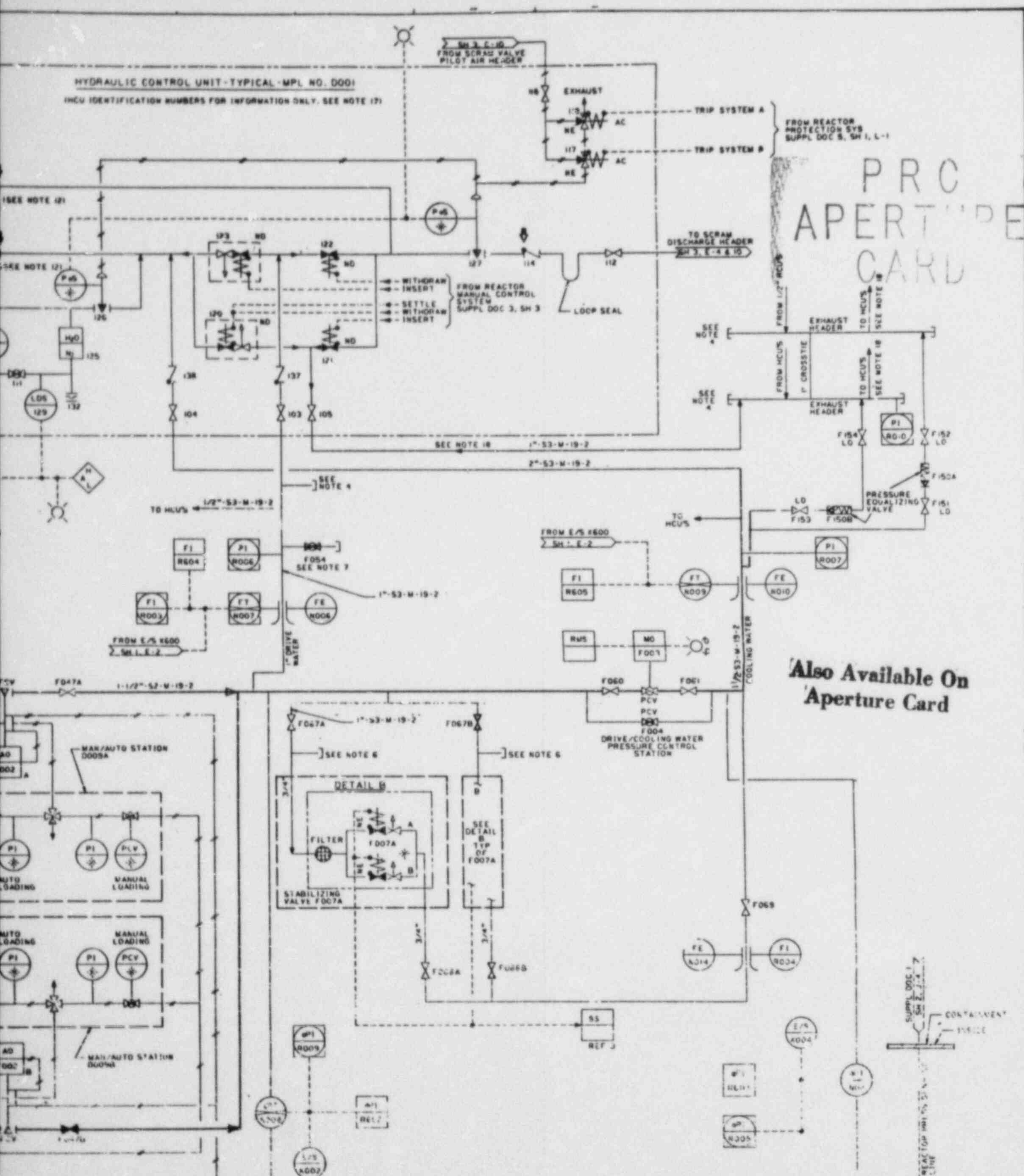

FIG. 4.2.3-8B

CONTROL ROD DRIVE HYD. SYS.

SHOREHAM NUCLEAR POWER STATION-UNIT I

FINAL SAFETY ANALYSIS REPORT

PRC APERTURE CARD

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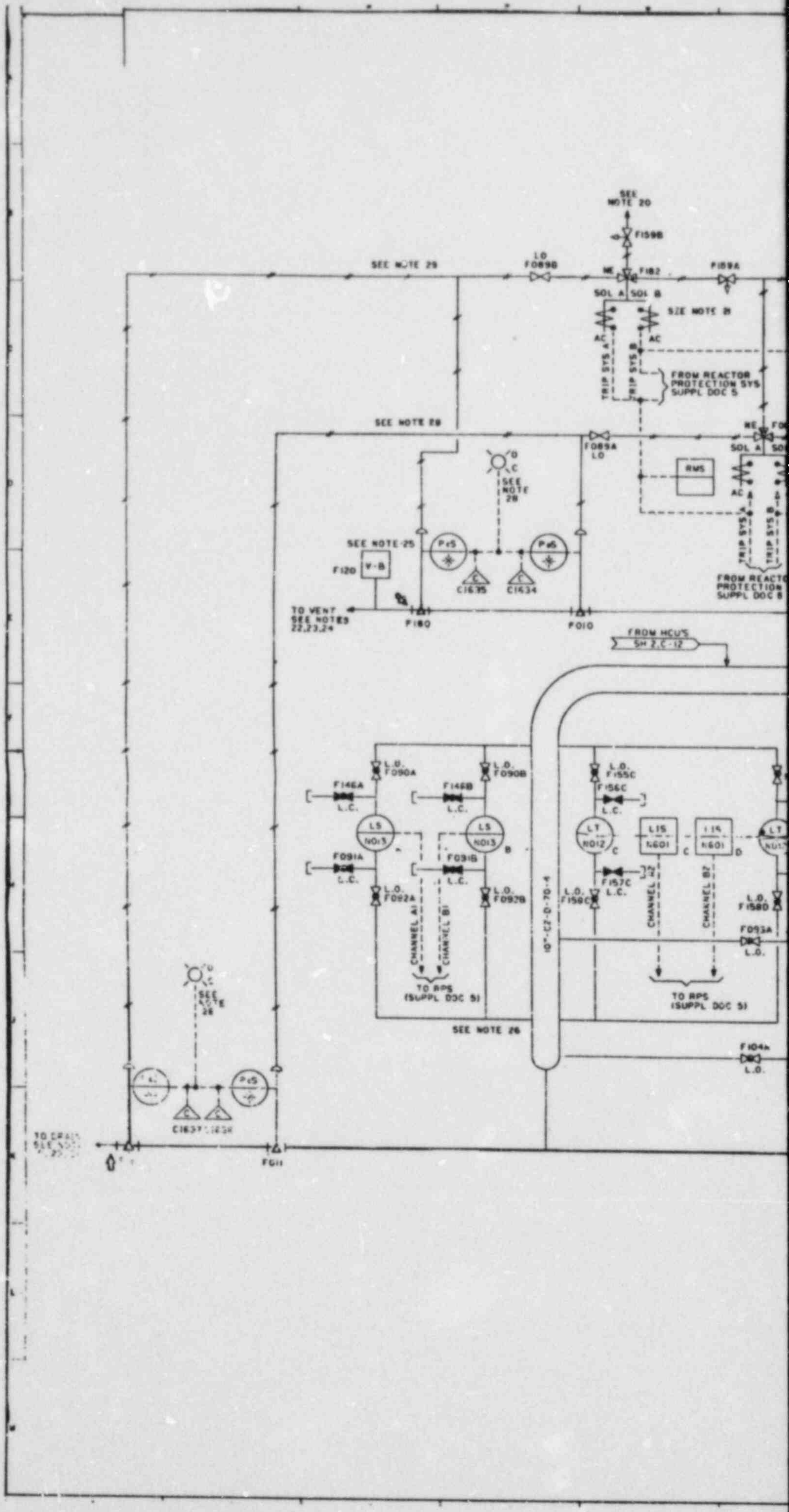
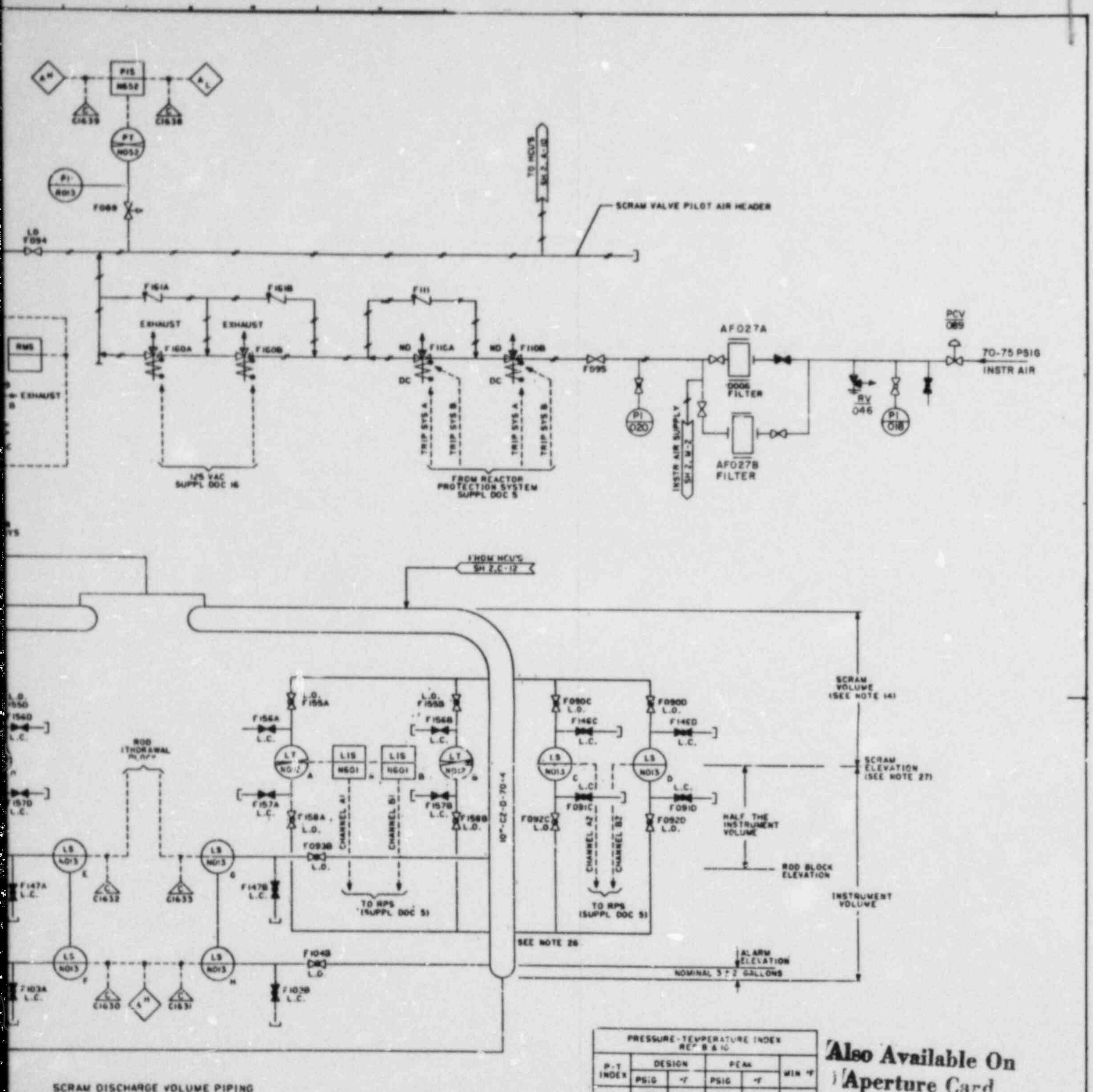

SEE NOTE 20
F159B
SEE NOTE 29
LO
F089B
NE
F182
F159A
SOL A
SOL B
SEE NOTE 21
AC
TRIP SYS A
TRIP SYS B
FROM REACTOR PROTECTION SYS SUPPL DOC 5
SEE NOTE 28
F089A
LO
RMS
SEE NOTE 28
SEE NOTE 25
F120
V-B
C1635
C1634
TO VENT SEE NOTES 22,23,24
F180
F010
FROM REACTOR PROTECTION SUPPL DOC 5
FROM HCU'S SH 2 C-12
L.O. F090A
L.O. F090B
F146A
F146B
L.C.
LS N013
F091A
F091B
L.O. F092A
L.O. F092B
CHANNEL A1
CHANNEL B1
TO RPS (SUPPL DOC 5)
SEE NOTE 26
L.O. F155C
F156C
LT N012
LIS N601
F157C
L.O. F158C
CHANNEL A2
CHANNEL B2
TO RPS (SUPPL DOC 5)
L.O. F158D
F093A
L.O.
F104A
L.O.
SEE NOTE 28
FO11



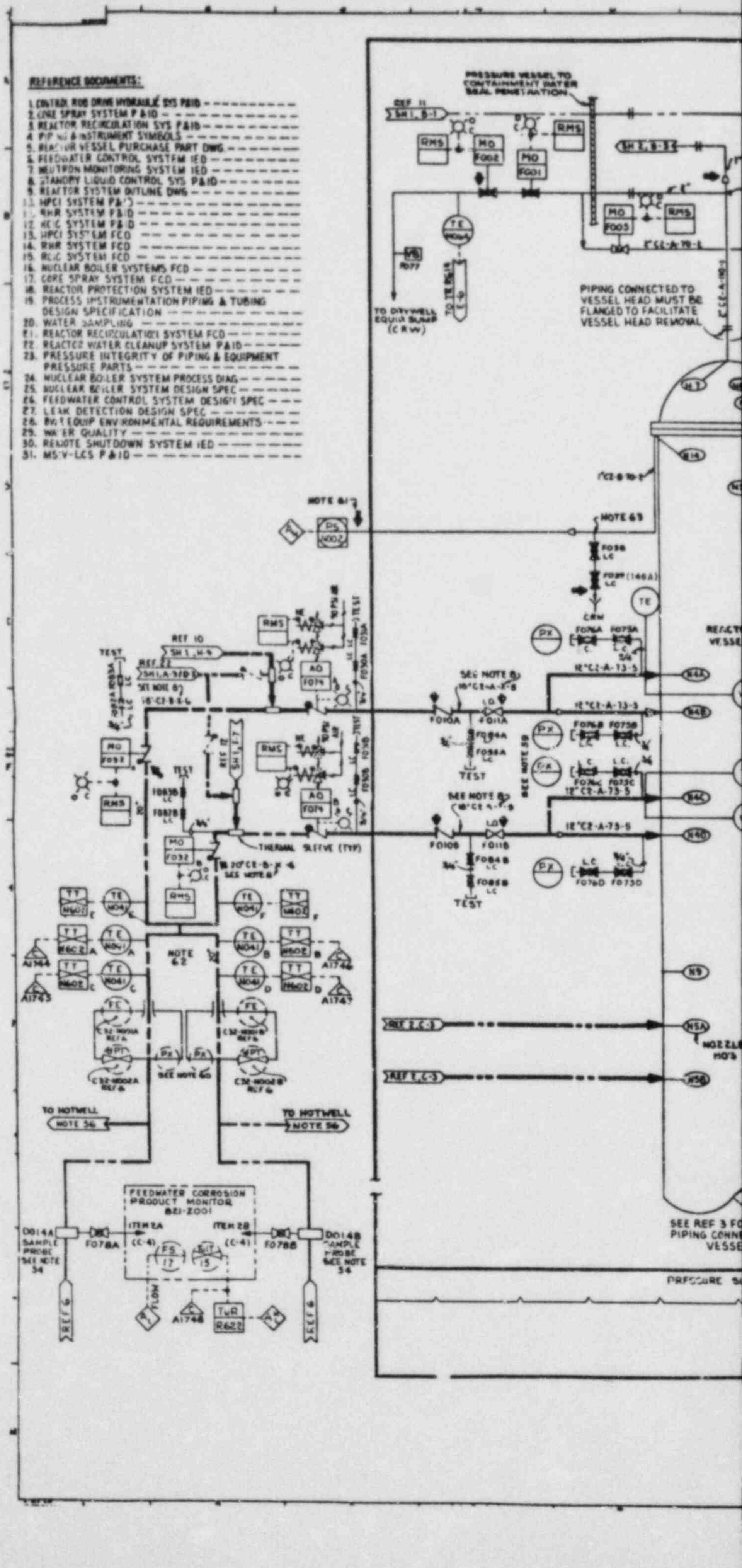
PRESSURE-TEMPERATURE INDEX REF 8 & 10					
P-T INDEX	DESIGN		PEAK		MIN °F
	PSIG	°F	PSIG	°F	
1	150	150	275	150	—
2	1750	150	1663	150	—
3	1135	580	[illegible]	540	50
4	1140	[illegible]	[illegible]	[illegible]	—
5	[illegible]	[illegible]	[illegible]	[illegible]	—
6	1720	500	[illegible]	535	—

Also Available On Aperture Card

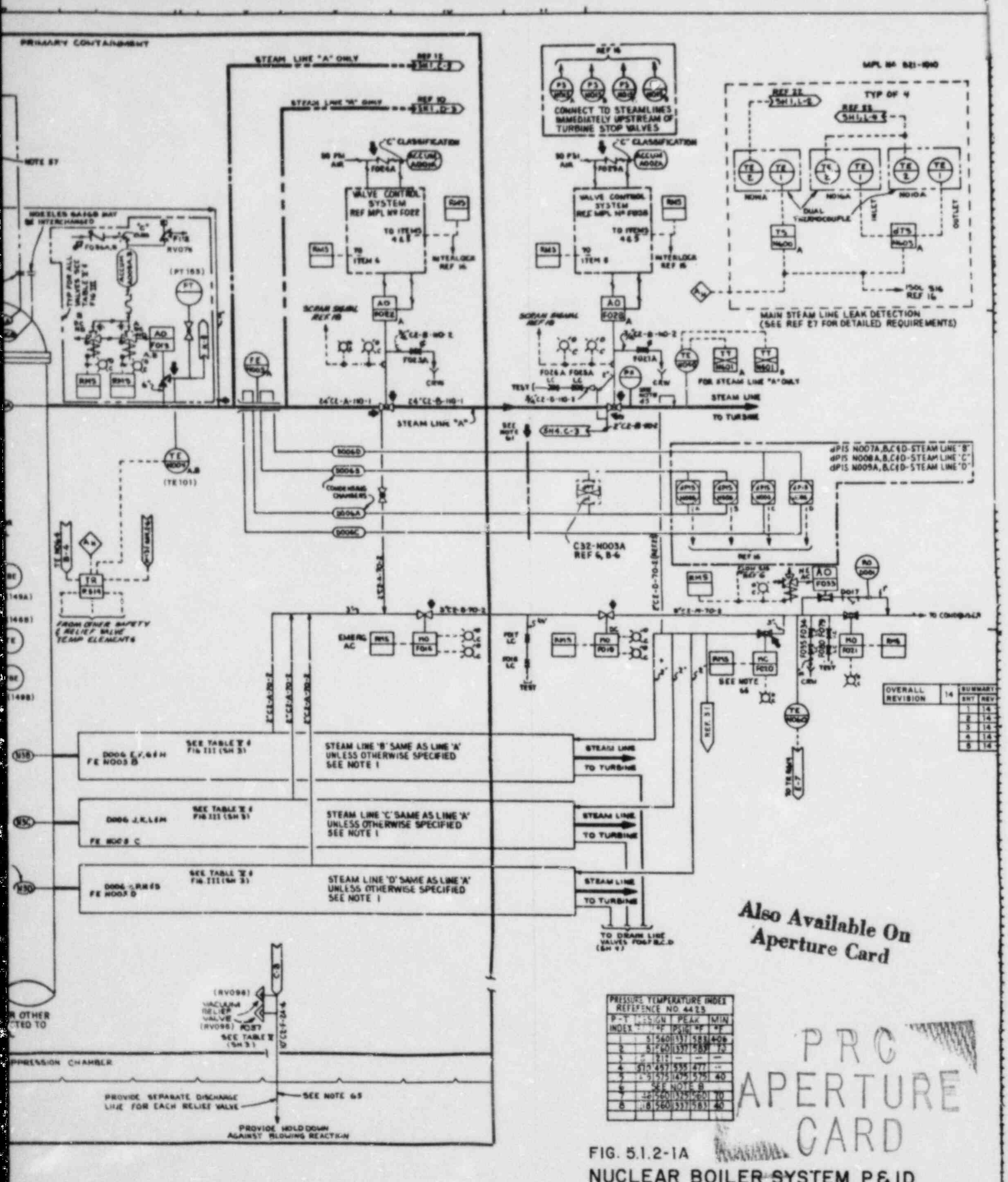
PRC APERTURE CARD

FIG. 4.2.3-8C
CONTROL ROD DRIVE
HYDRAULIC SYSTEM
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 -03



REFERENCE DOCUMENTS:
1. CONTROL ROD DRIVE HYDRAULIC SYS P&ID
2. CORE SPRAY SYSTEM P&ID
3. REACTOR RECIRCULATION SYS P&ID
4. PIPING & INSTRUMENT SYMBOLS
5. REACTOR VESSEL PURCHASE PART DWG.
6. FEEDWATER CONTROL SYSTEM IED
7. NEUTRON MONITORING SYSTEM IED
8. STANDBY LIQUID CONTROL SYS P&ID
9. REACTOR SYSTEM OUTLINE DWG
10. HPCI SYSTEM P&ID
11. RHR SYSTEM P&ID
12. RCIC SYSTEM P&ID
13. HPCI SYSTEM FCD
14. RHR SYSTEM FCD
15. RCIC SYSTEM FCD
16. NUCLEAR BOILER SYSTEMS FCD
17. CORE SPRAY SYSTEM FCD
18. REACTOR PROTECTION SYSTEM IED
19. PROCESS INSTRUMENTATION PIPING & TUBING DESIGN SPECIFICATION
20. WATER SAMPLING
21. REACTOR RECIRCULATION SYSTEM FCD
22. REACTOR WATER CLEANUP SYSTEM P&ID
23. PRESSURE INTEGRITY OF PIPING & EQUIPMENT PRESSURE PARTS
24. NUCLEAR BOILER SYSTEM PROCESS DIAG
25. NUCLEAR BOILER SYSTEM DESIGN SPEC
26. FEEDWATER CONTROL SYSTEM DESIGN SPEC
27. LEAK DETECTION DESIGN SPEC
28. BWR EQUIP ENVIRONMENTAL REQUIREMENTS
29. WATER QUALITY
30. REMOTE SHUTDOWN SYSTEM IED
31. MSIV-LCS P&ID
PRESSURE VESSEL TO CONTAINMENT WATER SEAL PENETRATION
PIPING CONNECTED TO VESSEL HEAD MUST BE FLANGED TO FACILITATE VESSEL HEAD REMOVAL
TO DRYWELL EQUIP SUMP (CRW)
NOTE 63
THERMAL SLEEVE (TYP)
TO HOTWELL
FEEDWATER CORROSION PRODUCT MONITOR B21-Z001



PRESSURE TEMPERATURE INDEX REFERENCE NO 4423					
P-T INDEX	DESIGN	°F	PEAK PSIG	°F	MIN °F
1	5	560	1337	583	406
2	5	560	1337	583	70
3	[illegible]	212	—	—	—
4	515	457	535	477	--
5	[illegible]5	575	1475	575	40
6	SEE NOTE 8				
7	[illegible]	560	1325	560	70
8	18	560	1337	583	40

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FIG. 5.1.2-1A

NUCLEAR BOILER SYSTEM P&ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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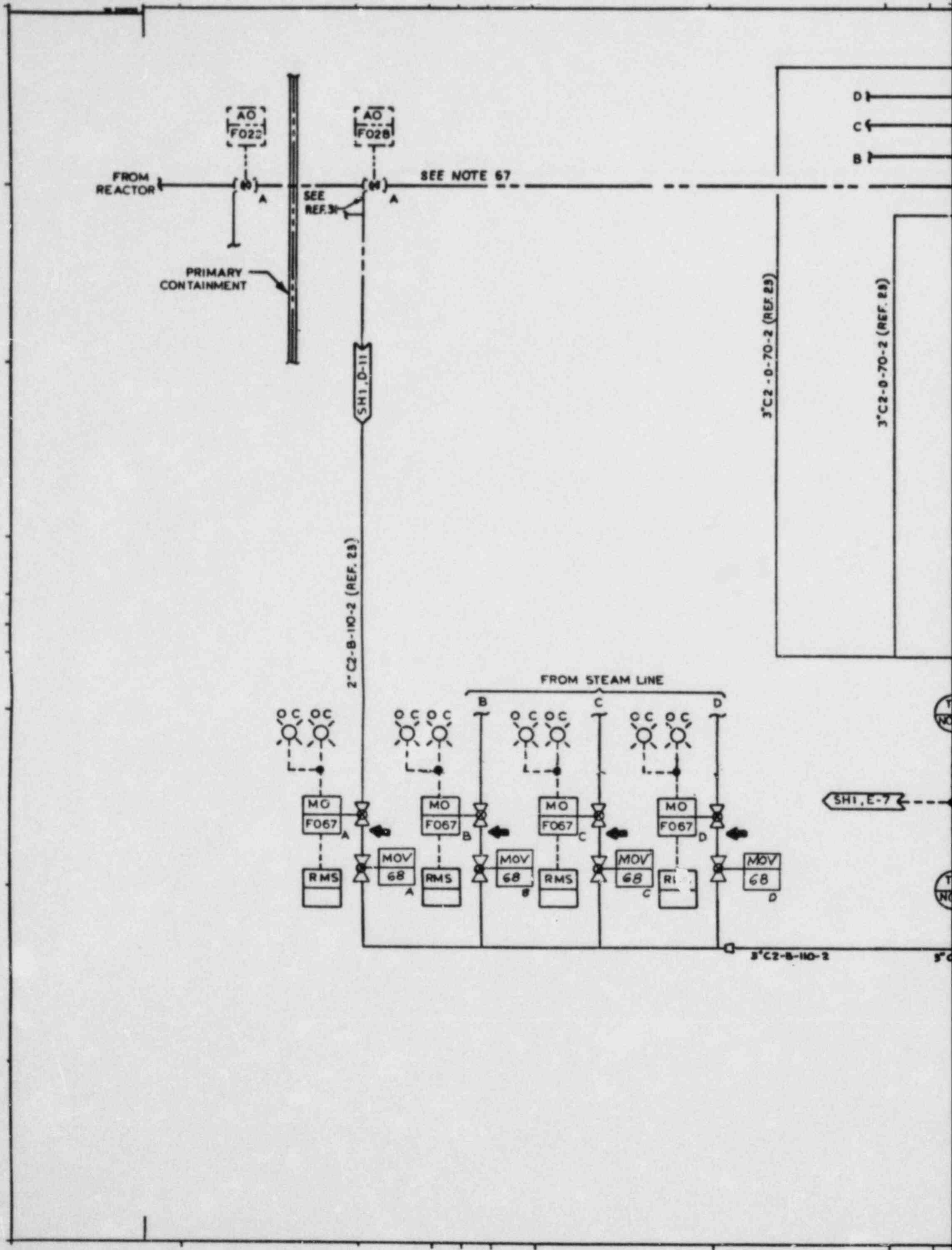
AO
F022
AO
F028
FROM REACTOR
A
SEE REF.31
A
SEE NOTE 67
PRIMARY CONTAINMENT
SH1, D-11
2" C2-B-110-2 (REF. 23)
3" C2-D-70-2 (REF. 23)
3" C2-D-70-2 (REF. 23)
D
C
B
FROM STEAM LINE
B
C
D
O C
MO
F067
MOV
68
RMS
SH1, E-7
3" C2-B-110-2

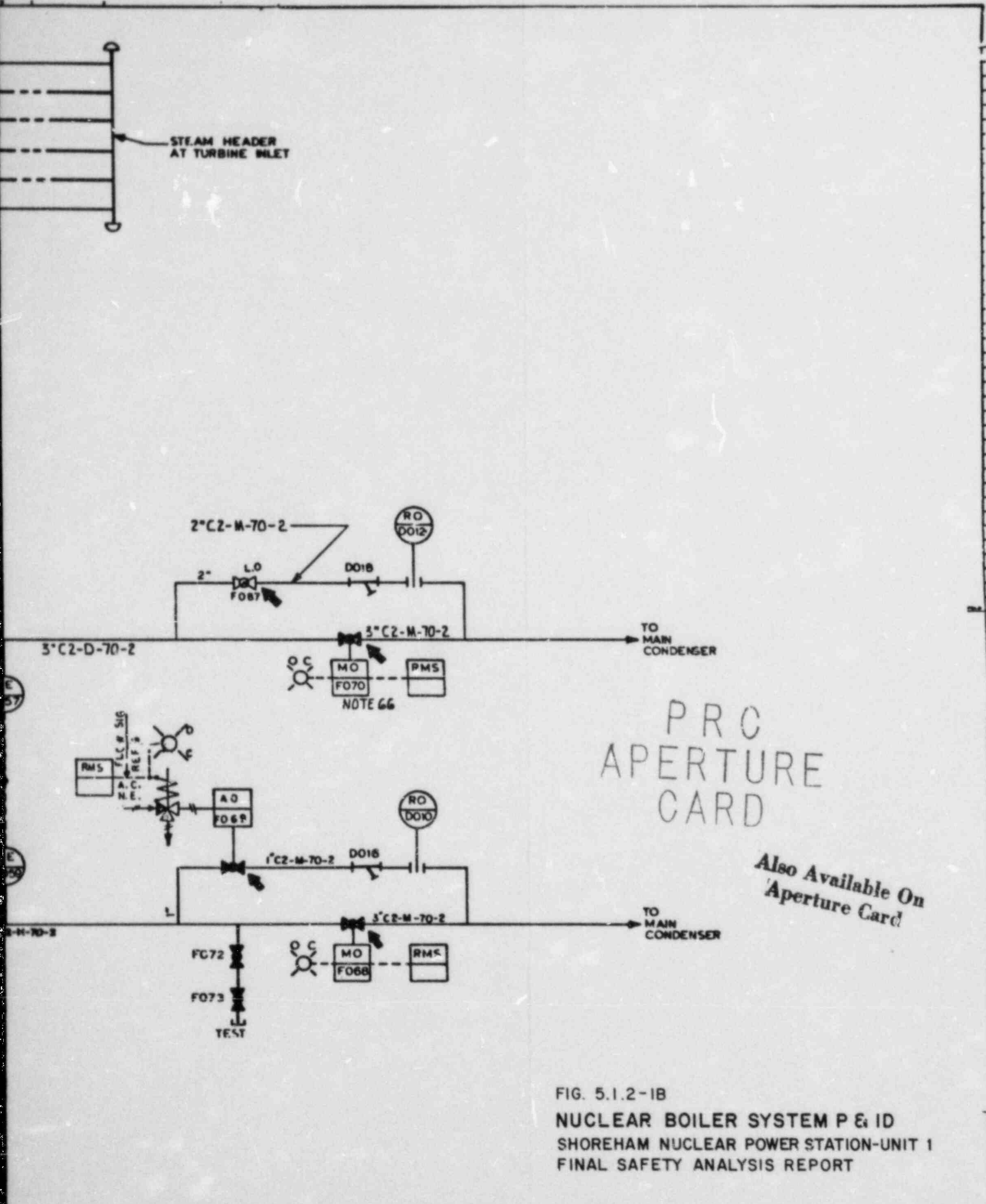

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FIG. 5.1.2-1B
NUCLEAR BOILER SYSTEM P & ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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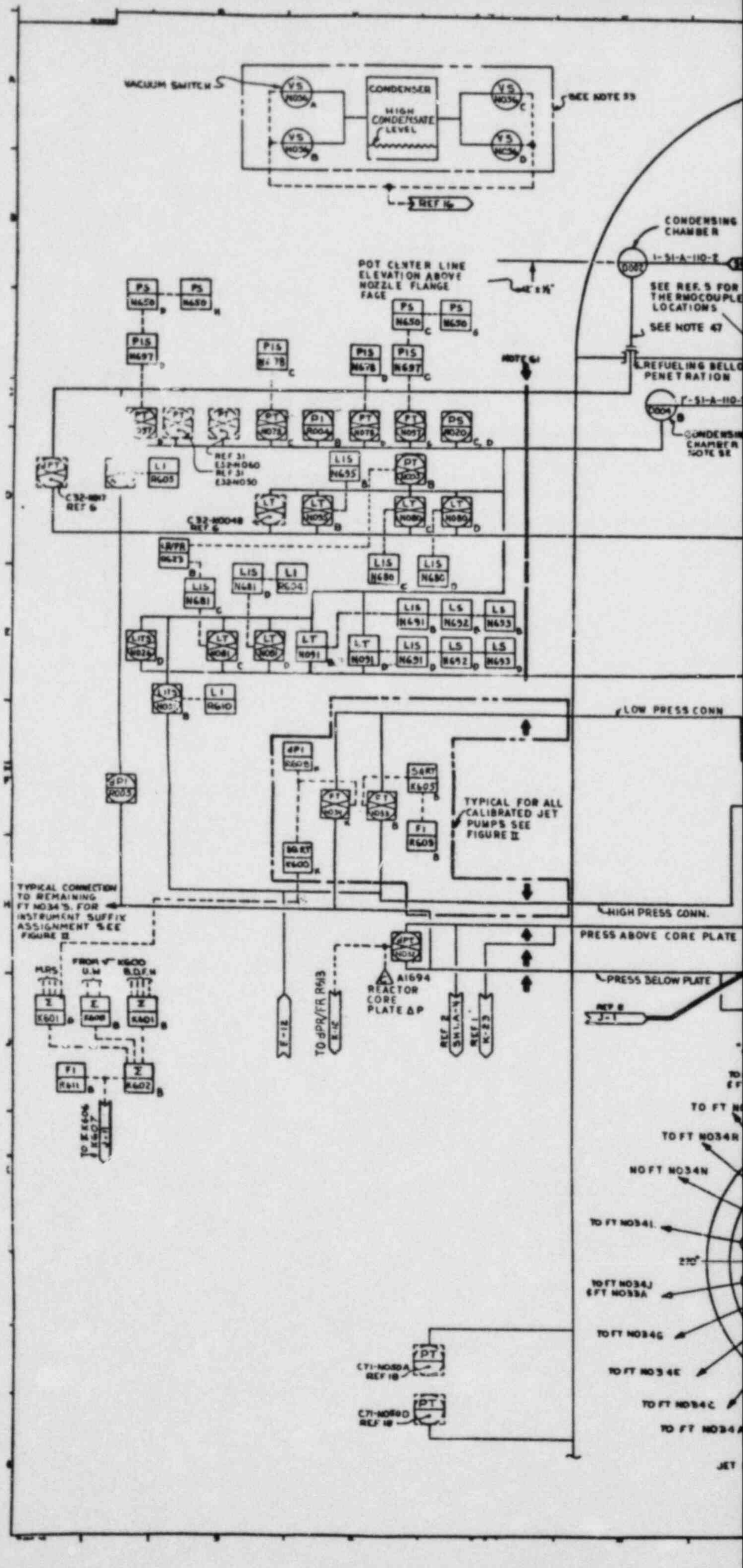
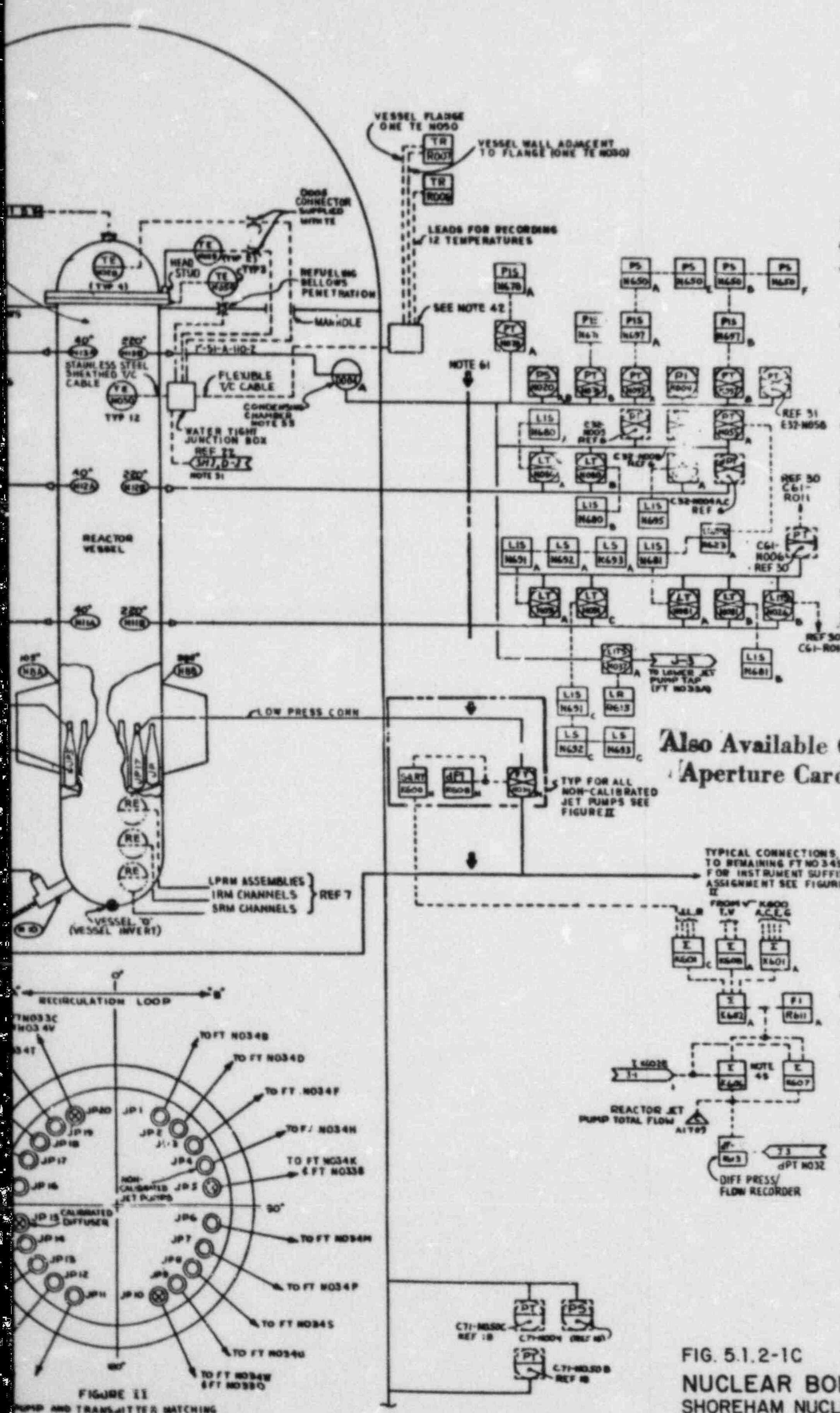


VACUUM SWITCH
CONDENSER
HIGH CONDENSATE LEVEL
SEE NOTE 33
REF 16
CONDENSING CHAMBER
SEE REF. 5 FOR THERMOCOUPLE LOCATIONS
SEE NOTE 47
REFUELING BELLOWS PENETRATION
POT CENTER LINE ELEVATION ABOVE NOZZLE FLANGE FACE
NOTE 61
CONDENSING CHAMBER NOTE 58
C32-N017 REF 6
C32-N004B REF 6
LOW PRESS CONN
TYPICAL FOR ALL CALIBRATED JET PUMPS SEE FIGURE II
TYPICAL CONNECTION TO REMAINING FT N034'S. FOR INSTRUMENT SUFFIX ASSIGNMENT SEE FIGURE II
HIGH PRESS CONN.
PRESS ABOVE CORE PLATE
PRESS BELOW PLATE
A1694 REACTOR CORE PLATE ΔP
TO dPR/FR R613
TO FT N034R
NO FT N034N
TO FT N034L
TO FT N034G
TO FT N034E
TO FT N034C
TO FT N034A
C71-N050A REF 18
C71-N050D REF 18



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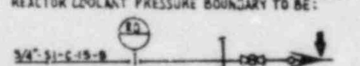
NOTES:

1. STEAM LINES ENCLOSED IN BOXES SHALL HAVE PART NOS. CORRESPONDING TO ITS RESPECTIVE LINE NO. UNLESS OTHERWISE NOTED.
EXAMPLE: XXXB IS ON LINE "B", XXXC IS ON LINE "C"
8. DESIGN PRESSURE AND TEMPERATURE TO BE ESTABLISHED BY DESIGNER BASED ON FEED PUMP SHUTOFF PRESSURE AND SYSTEM ARRANGEMENT F011 TO BE DESIGNED FOR MOST SEVERE UPSTREAM AND DOWNSTREAM SERVICE.
11. PIPE SIZES SHOWN ON THIS DRAWING ARE APPROXIMATE EXCEPT AT POINTS OF CONNECTION WITH BWRSD SUPPLIED EQUIPMENT OR PIPING. THE PIPING DESIGNER SHALL CHECK AND ADJUST PIPE SIZE IN ACCORDANCE WITH HIS PIPING LAYOUT FOR CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.
36. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NO. B21 UNLESS OTHERWISE NOTED.
42. T/C JUNCTION BOX LOCALLY MOUNTED (BY OTHERS) EACH T/C JUNCTION BOX TO HAVE OWN SET OF TERMINALS.
45. (PX) TO BE CONNECTED INTO THE STRAIGHT RUN OF PIPE IMMEDIATELY DOWNSTREAM OF F028 WITH UPSTREAM AND DOWNSTREAM STRAIGHT LENGTH FROM THE TAP TO GIVE AS ACCURATE A PRESSURE MEASUREMENT AS FEASIBLE. TAPS TO MEET ASME PTC 6 1964 "STEAM TURBINES" PARAGRAPH 4.74.
47. AN EXPANSION LEG SHALL BE PROVIDED IN INSTRUMENT SENSING LINE BETWEEN POT (PART D002) AND THE WATER-TIGHT PENETRATION SEAL THROUGH BOTTOM OF REACTOR WELL. THE EXPANSION LEG & PIPING INSTALLATION SHALL BE DESIGNED TO ALLOW FOR MAXIMUM CHANGE OF VESSEL LENGTH WITH TEMPERATURE TO AVOID OVERSTRESSING THE PIPING OR THE SEAL OR DAMAGE TO THE INSULATION AROUND THE VESSEL.
48. SEE TABLE I & II ON SHEET 3 FOR POWER SUPPLY ASSIGNMENT & CONTACT UTILIZATION FOR ALL LEVEL, PRESSURE & DIFFERENTIAL PRESSURE TRANSMITTERS AND SWITCHES.
49. SUMMERS K606 & K607 INPUTS SHALL BE INTERLOCKED WITH RECIRC PUMP & VALVES TO ADD INPUT WHEN BOTH PUMPS ARE RUNNING & THEIR DISCHARGE VALVES ARE OPEN OR SUBTRACT ONE INPUT WHEN THE CORRESPONDING PUMP IS STOPPED OR ITS DISCHARGE VALVE IS CLOSED.
50.
51. WIRE TO RECORDER R006.
52.
53. LOW CONDENSER VACUUM SWITCHES CONNECTED THROUGH SEPARATE CALIBRATION VALVES TO OPPOSITE SIDES OF THE CONDENSER ABOVE THE HIGH CONDENSATE LEVEL. THE VACUUM SWITCHES MUST BE ACCESSABLE DURING PLANT OPERATION.
54. SAMPLE PROBE(S) AND FEEDWATER SAMPLE STATION TO COMPLY WITH REFERENCE 20, WATER SAMPLING SECTION 8.
55. CENTERLINE ELEVATION OF CONDENSING CHAMBER TO BE +1 1/2" RELATIVE TO CENTERLINE OR VESSEL NOZZLE.
56. RECIRC LINES TO HOTWELL TO COMPLY WITH REF 29, WATER QUALITY SECTION 7.
57. LOCATE TEE AS CLOSE AS POSSIBLE TO RPV.
58. WATER LEVEL INSTRUMENTS FOR VARIOUS RANGES ARE CALIBRATED AS STATED BELOW ALL WATER LEVEL SWITCH SET POINTS ARE NOMINAL; IE; THE ANALYSES ARE PERFORMED WITH THE SWITCH TRIP UNCERTAINTY INCLUDED. REACTOR BUILDING TEMP ASSUMED TO BE 75°F.
A. FUEL ZONE: THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 0 PSIG IN THE VESSEL AND THE DRYWELL WITH NO JET PUMP FLOW. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ±6" OF WATER LEVEL AT CALIBRATION CONDITIONS.
B. WIDE RANGE: THE INSTRUMENTS ARE CALIBRATED FOR 1000 PSIG IN THE VESSEL, 135°F. IN THE DRYWELL AND 20 BTU/LB SUBCOOLING BELOW THE MIDDLE WATER LEVEL NOZZLE AND SATURATED CONDITIONS ABOVE THE MIDDLE WATER LEVEL NOZZLE WITH NO JET PUMP FLOW. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ±6" OF WATER LEVEL AT CALIBRATION CONDITIONS.
C. NARROW RANGE: (SAFEGUARDS AND FEEDWATER); THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 1000 PSIG IN THE VESSEL AND 135°F. IN THE DRYWELL. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ±1.5" OF WATER LEVEL AT CALIBRATION CONDITIONS
D. UPSET RANGE: THE INSTRUMENT IS CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 1000 PSIG IN THE VESSEL AND 135°F. IN THE DRYWELL.
E. SHUTDOWN: THE INSTRUMENT IS CALIBRATED FOR 120°F. WATER AT 0 PSIG IN VESSEL AND 80°F. IN THE DRYWELL.
59. (PX) TO BE CONNECTED IN A STRAIGHT RUN OF PIPE AS FAR AS POSSIBLE FROM ELBOWS, ETC. & LOCATED SO THAT ΔP'S FROM THE TAPS TO THE VESSEL NOZZLE ARE EQUAL. TAPS TO MEET ASME PTC 6 1964 "STEAM TURBINES" PARAGRAPH 4.7A. EIGHTEEN WIRES ARE TO BE PROVIDED PENETRATING THE DRYWELL FOR READOUT OF TEMPORARY PRESSURE TRANSMITTERS DURING START-UP.
60. (PX) ALTERNATE PRESSURE TAPS ARE TO BE TERMINATED INTO A SHUTOFF VALVE AND AN INSTRUMENT BALANCING VALVE.
61. PROVISION FOR ISOLATION OF INSTRUMENT LINE CONNECTING TO THE REACTOR COOLANT PRESSURE BOUNDARY TO BE:



NO MORE THAN ONE ORIFICE SHALL BE INSTALLED IN AN INSTRUMENT LINE. MINIMUM ORIFICE SIZE IS 1/4".
62. TEMPERATURE ELEMENTS MUST BE LOCATED BETWEEN 5 TO 6 PIPE DIAMETER FROM THE ENTRANCE OF THE FLOW NOZZLES PER ASME PTC 19.5 (1971 INTERIM ISSUE) "INSTRUMENT AND APPARATUS".
63. DRAIN VALVE CONNECTION SHOULD BE LOCATED AT LOW POINT.
64. DELETED
65. AVAILABLE CORROSION DATA INDICATES THAT FOR CARBON STEEL THE SAFETY/RELIEF VALVE DISCHARGE PIPING BELOW SUPPRESSION POOL WATER LEVEL SHOULD HAVE A CORROSION ALLOWANCE OF .250 INCHS FOR A 40 YEAR LIFE, SINCE THIS PIPING WILL CORRODE FROM BOTH SIDE.
66. VALVE F020 SHALL BE SUPPLIED WITH CLASS 1E POWER, AS SPECIFIED IN REFERENCE 31.
67. OUTBOARD MSIV-LCS SHALL BE CONNECTED INSIDE THE STEAM TUNNEL. THE PREFERRED CONNECTION SHALL BE TO A PROCESS OR DRAIN LINE THAT TIES INTO THE BOTTOM OF THE MAIN STEAM LINE.

FIG. 5.1.2-1C
NUCLEAR BOILER SYSTEM P&ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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FIGURE 1: ELEVATION CORRELATION CHART

REFERENCE	(COLD VESSEL) INCHES ABOVE VESSEL ZERO	DESCRIPTION OF TRIPS	INSTRUMENT PROVOKING TRIP	REACTOR VESSEL LEVEL IDENTITY (REF 25)	CONTROL ROOM WATER LEVEL IN... SEE NOTE S... — SAFEGUARDS — FUEL ZONE LR-R615 LI-R610	WIDE RANGE LR-SG-R623A,B LI-...-R04	NA... C...
TOP OF REACTOR HEAD FLANGE	867 MAX						
STEAM LINE NOZZLE N3A,B,C,D	640						
INSTRUMENT LINE NOZZLE N13A,B	586.25						
						+60"	+4
		TRIP RCIC(HPCI) TURBINE. CLOSE MAIN TURBINE STOP VALVES. TRIP FEED PUMPS.	TABLE I REF 6	8		+54.5"	+5
		HIGH LEVEL ALARM	REF 6	7			
		NORMAL WATER LEVEL	REF 6	5,6			
		LOW LEVEL ALARM	REF 6	4			
		SCRAM & CLOSE RHR SHUTDOWN COOLING ISOLATION VALVES. AUTO DEPRESSURIZATION PERMISSIVE.	TABLE I	3			+0
FEEDWATER SYSTEM & REACTOR PROTECTION SYSTEM. INSTRUMENT ZERO EXCEPT FUEL ZONE. BOTTOM OF DRYER SKIRT	516.75					0"	
INSTRUMENT LINE NOZZLE N12A,B	509	INITIATE RCIC & HPCI & TRIP RECIRC PUMPS. CLOSE PRIMARY SYS ISOL VALVES EXCEPT RHR SHUTDOWN COOLING ISOL VALVES.	TABLE I	2		-38"	
					+30"		
		INITIATE RHR & CORE SPRAY SYSTEMS, CONTRIBUTE TO AUTO DEPRESSURIZATION SYSTEM, START STANDBY DIESEL.	TABLE I	1		-126.4	
						-150"	
TOP OF ACTIVE FUEL AND FUEL ZONE WATER LEVEL ZERO	358 5/16				0"		
INSTRUMENT LINE NOZZLE N11 A,B	358						
	310 5/16						
					-150"		
JET PUMP DIFFUSER TAP NOZZLE N8A,B	143						
JET PUMP INSTRUMENT	122						

…CATION AND TRIP LEVELS			
	FEEDWATER		
…ROW RANGE		UPSET	SHUTDOWN
…2- LR-R608 …2- LI-R605A,B,C		C32- LR-R608	LI-R605
			400"
		180"	
…"	+60"		
…5.5"	+56.5"		
	+40.5"		
	+37.0		
	+33.5		
…5"	+12.5"		
…"	0"	0"	0"

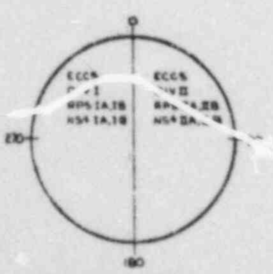
TABLE III: JET PUMP INSTRUMENTATION SUFFIX LETTER ASSIGNMENT

LOOP	A										B									
JP NO	11	12	13	14	15	16	17	18	19	20	1	2	3	4	5	6	7	8	9	10
FT N034 / E/S K603	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
dPI R608	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
"V" K600	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
Σ	K601A				K601C				K608A		K601B				K601D				K608B	
Σ	K602A										K602B									
FT N033	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D
"V" K603	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D
FI R609	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D

TABLE II: SAFETY/RELIEF VALVE SUFFIX ASSIGNMENT & ASSOCIATED EQUIPMENT

SAFETY/RELIEF VALVES	F013	A	B	C ※	D ※	E	F ※	G	H	J	K	L
ACCUMULATORS	A003	A	B	C	D	E	F	G	H	J	K	L
CHECK VALVES	F036	A	B	C	D	E	F	G	H	J	K	L
TEMPERATURE ELEMENTS	N004 (TE101)	A	B	C	D	E	F	G	H	J	K	L
VACUUM BREAKERS	F037 (RV095)	A	B	C	D	E	F	G	H	J	K	L
	(RV096)	A	B	C	D	E	F	G	H	J	K	L
AIR ACCUMULATOR RELIEF VALVE	F112 (RV076)	A	B	C	D	E	F	G	H	J	K	L
ADS FUNCTION		YES	YES	—	—	YES	—	—	YES	YES	YES	YES

※ CONTROL PROVIDED IN REMOTE SHUTDOWN SYS WHEN FURNISHED (REF 10)
(ADS) SAFETY/RELIEF VALVE FOR AUTO DEPRESSURIZATION



VESSEL PLAN VIEW
SEPARATION ASSIGNMENT WITH RESPECT TO VESSEL PLAN

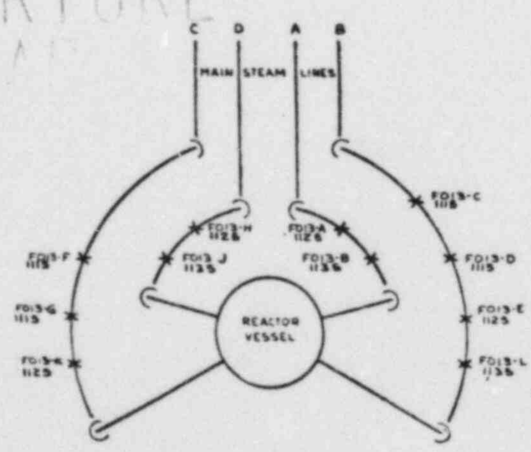


FIGURE III
SAFETY/RELIEF VALVE LOCATIONS & SPRING SET PRESSURE

FIG. 5.2.2-2A
NUCLEAR BOILER SYSTEM P& ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

TABLE I WATER LEVEL INSTRUMENT CONTACT UTILIZATION

INSTRUMENT NUMBER	TRIP UNIT	DIVISION	SYSTEM TRIP FUNCTION	TRIP LEVEL	POWER SUPPLY
LT-N080A, B	LIS-N680A, B	IA, IB	RPS	3	—
LT-N080C, D	LIS-N680C, D	IIA, IIB	RPS	3	—
LT-N081A, B	LIS-N681A, B	IA, IB	NS4	2	K613A C61-K005
LT-N081C, D	LIS-N681C, D	IIA, IIB	NS4	2	K613B K603
	LIS-N691A	I	ADS(A)/RHRA, CI/CS(A)	1	—
LT-N091A	LS-N692A	I	RPT(A)/RCIC/(HPCI)*	2	—
	LS-N693A	I	RCIC/(HPCI)*	8	—
	LIS-N691B	II	ADS(B)/RHRB, DI/CS(B)	1	—
LT-N091B	LS-N692B	II	RPT(B)/HPCI/(RCIC)*	2	—
	LS-N693B	II	HPCI/(RCIC)*	8	—
	LIS-N691C	I	ADS(A)/RHRA, CI/CS(A)	1	—
LT-N091C	LS-N692C	I	RPT(A)/RCIC/(HPCI)*	2	—
	LS-N693C	I	RCIC/(HPCI)*	8	—
	LIS-N691D	II	ADS(B)/RHRB, DI/CS(B)	1	—
LT-N091D	LS-N692D	II	RPT(B)/HPCI/(RCIC)*	2	—
	LS-N693D	II	HPCI/(RCIC)*	8	—
LT-N095A	LIS-N695A	I	ADS(A)	3	—
LT-N095B	LIS-N695B	II	ADS(B)	3	—
LT-N027	—	—	SHUTDOWN LEVEL	—	K603

* SYSTEM DESIGNATED IN PARENTHESES ( ) RECEIVES AN ISOLATED INPUT FROM THIS TRIP UNIT AS A BACKUP TO ITS PRIMARY SENSOR INPUT.

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TABLE 8 PRESSURE INSTRUMENT CONTACT UTILIZATION

INSTRUMENT NUMBER	TRIP UNIT	DIVISION	SYSTEM TRIP FUNCTION	POWER SUPPLY
PS-N020A, B	—	IA, IB	LOW PRESSURE INTERLOCK (NS4)	—
PS-N020C, D	—	IIA, IIB	LOW PRESSURE INTERLOCK (NS4)	—
	PIS-N697A, C	—	RECIRC PUMP TRIP RPT (A)	—
PT-N097A, C	PS-N650A, C	I	RHR	—
	PS-N650E, G	—	RECIRC DISCHARGE VALVE CLOSURE	—
	PIS-N697B, D	—	RECIRC PUMP TRIP RPT (B)	—
PT-N097B, D	PS-N650B, C	II	RHR	—
	PS-N650F, H	—	RECIRC DISCHARGE VALVE CLOSURE	—
PT-N078A, B	PIS-N678A, B	IA, IB	RPS	—
PT-N078C, D	PIS-N678C, D	IIA, IIB	RPS	—
PS-N015A, B	—	IA, IB	NS4	—
PS-N015C, D	—	IIA, IIB	NS4	—
dPIS-N006A, B	—	IA, IB	NS4	—
dPIS-N006C, D	—	IIA, IIB	NS4	—
dPIS-N007A, B	—	IA, IB	NS4	—
dPIS-N007C, D	—	IIA, IIB	NS4	—
dPIS-N008A, B	—	IA, IB	NS4	—
dPIS-N008C, D	—	IIA, IIB	NS4	—
dPIS-N009A, B	—	IA, IB	NS4	—
dPIS-N009C, D	—	IIA, IIB	NS4	—
VS-N056A,B	—	IA, IB	NS4	—
VS-N056C,D	—	IIA, IIB	NS4	—
PT-N055A,B	—	I, II	VESSEL PRESSURE	K613A, B
dPT-N032	—	—	CORE PLATE ΔP	K603

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FIG. 5.2.2-2B
NUCLEAR BOILER SYSTEM P & ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 - 08

REFERENCE DOCUMENTS:

1. CONTROL ROD DRIVE HYDRAULIC SYS P&ID
2. CORE SPRAY SYSTEM P&ID
3. REACTOR RECIRCULATION SYS P&ID
4. PIPING & INSTRUMENT SYMBOLS
5. REACTOR VESSEL PURCHASE PART DWGS.
6. FEEDWATER CONTROL SYSTEM IED
7. NEUTRON MONITORING SYSTEM IED
8. STANDBY LIQUID CONTROL SYS P&ID
9. REACTOR SYSTEM OUTLINE DWG
10. HPCI SYSTEM P&ID
11. RHR SYSTEM P&ID
12. RCIC SYSTEM P&ID
13. HPCI SYSTEM FCD
14. RHR SYSTEM FCD
15. RCIC SYSTEM FCD
16. NUCLEAR BOILER SYSTEMS FCD
17. CORE SPRAY SYSTEM FCD
18. REACTOR PROTECTION SYSTEM IED
19. PROCESS INSTRUMENTATION PIPING & TUBING DESIGN SPECIFICATION
20. WATER SAMPLING
21. REACTOR RECIRCULATION SYSTEM FCD
22. REACTOR WATER CLEANUP SYSTEM P&ID
23. PRESSURE INTEGRITY OF PIPING & EQUIPMENT PRESSURE PARTS
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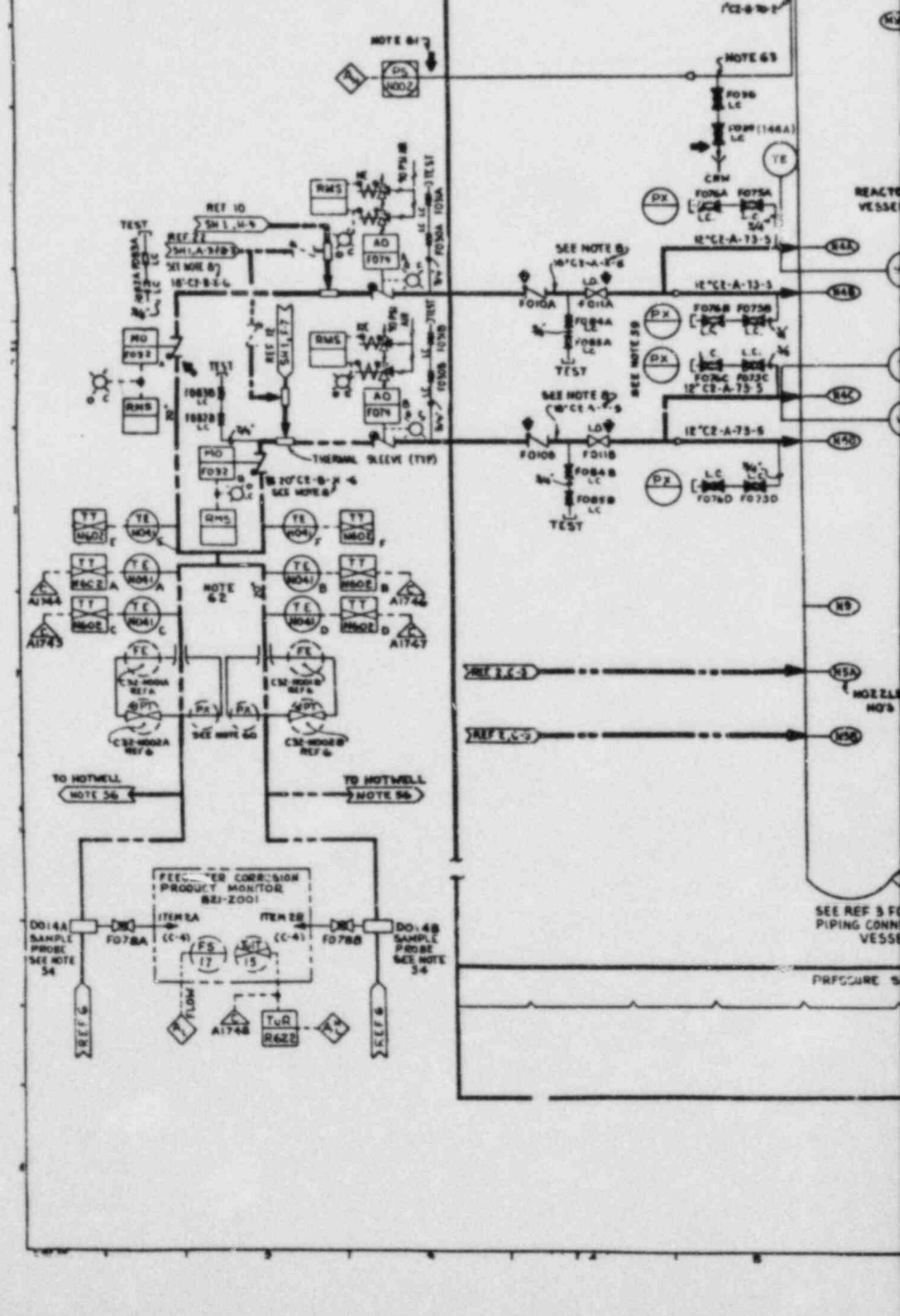
25. NUCLEAR BOILER SYSTEM DESIGN SPEC
26. FEEDWATER CONTROL SYSTEM DESIGN SPEC
27. LEAK DETECTION DESIGN SPEC
28. BWR EQUIP ENVIRONMENTAL REQUIREMENTS
29. WATER QUALITY
30. REMOTE SHUTDOWN SYSTEM IED
31. MSIV-LCS P&ID



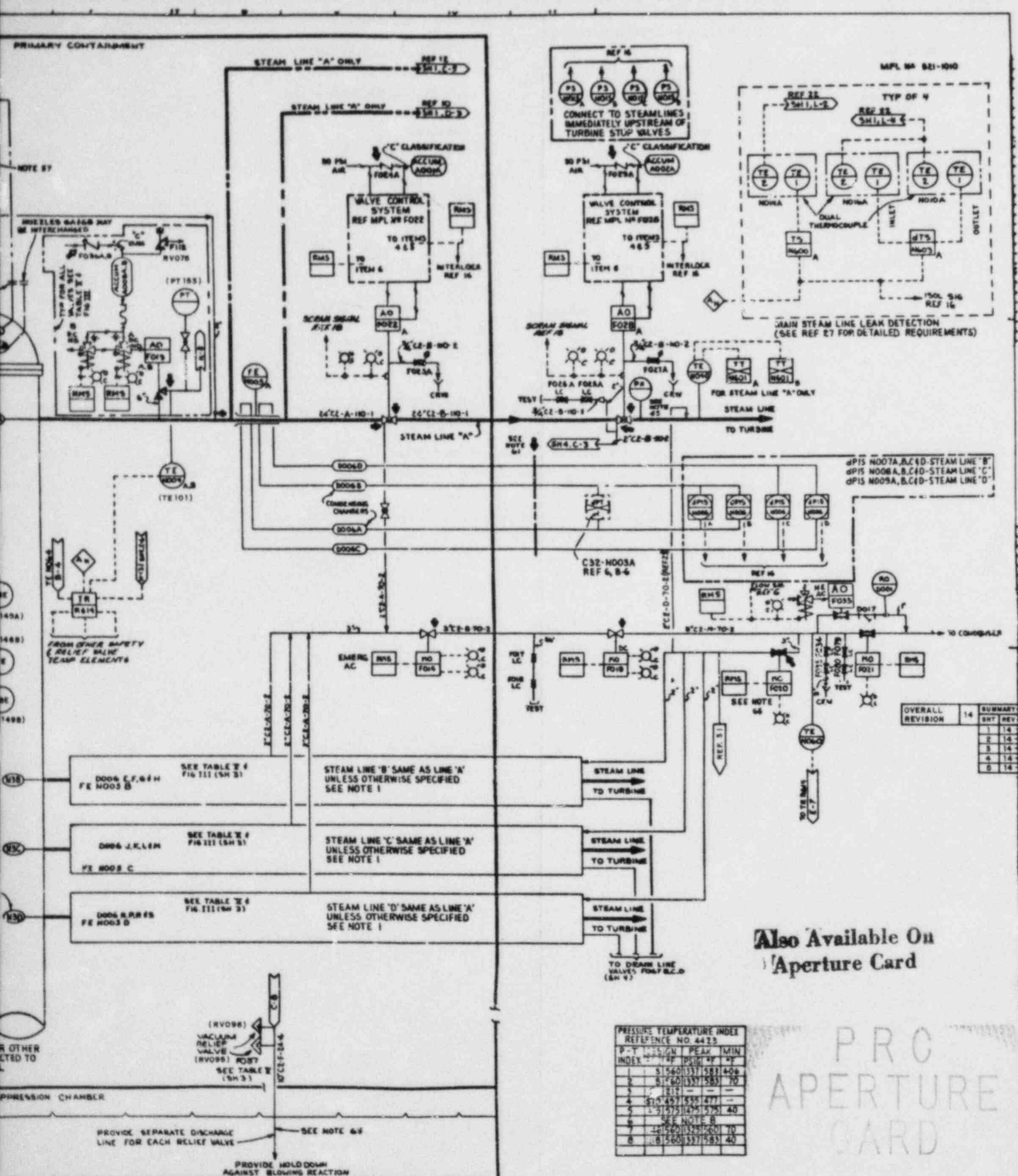


FIG. 7.3.1-12A
NUCLEAR BOILER SYSTEM P&ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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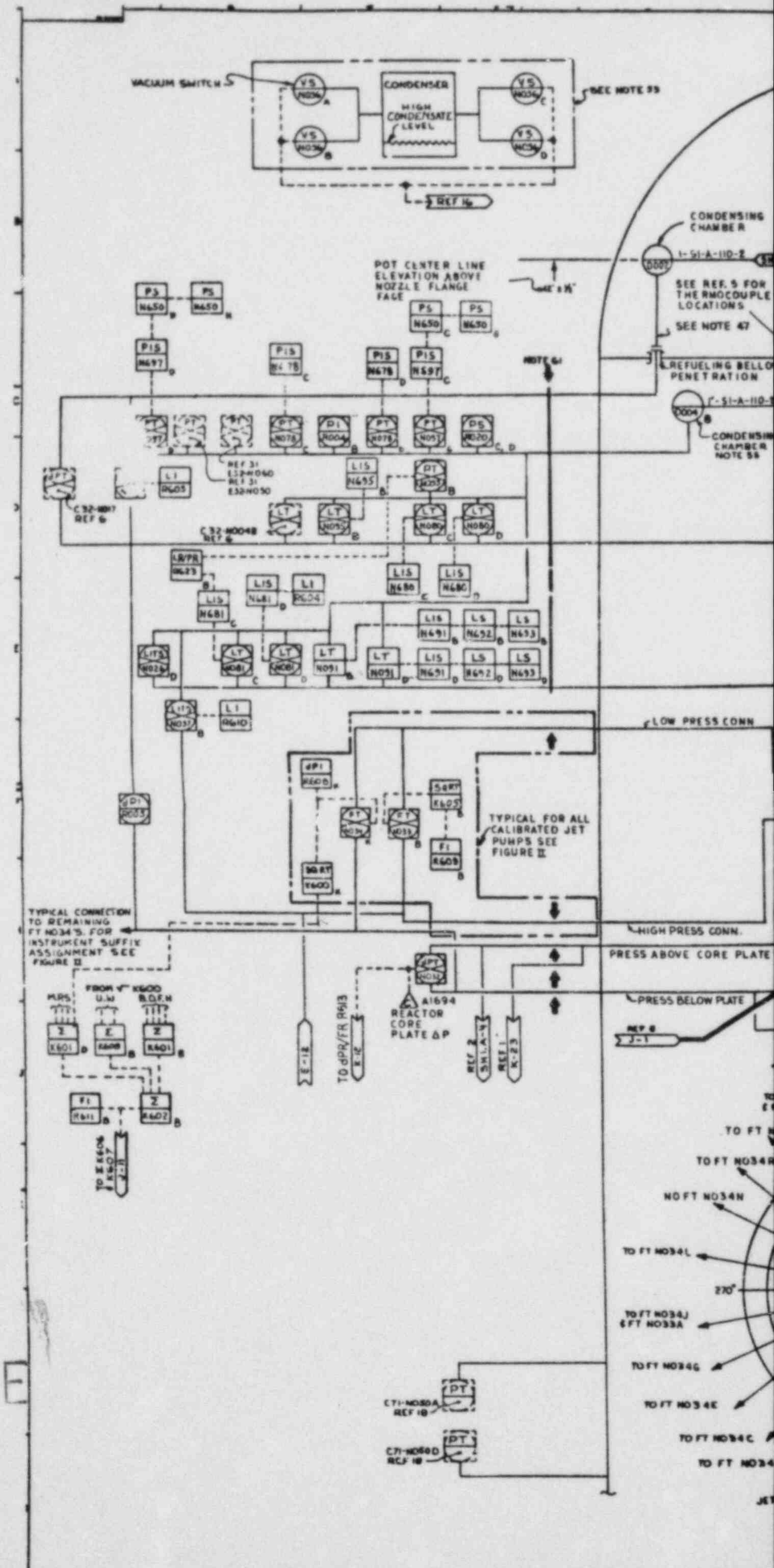
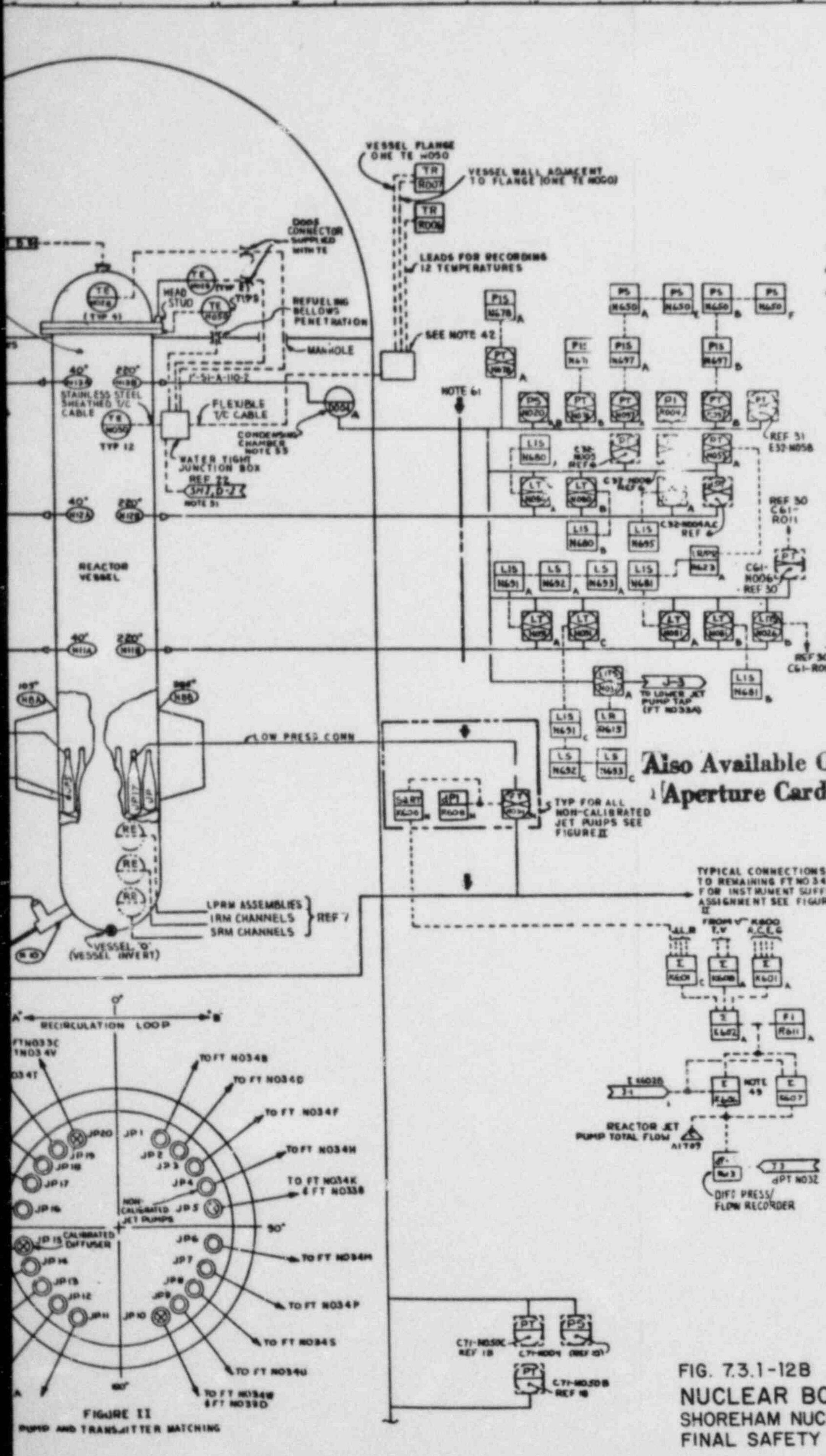


VACUUM SWITCH
CONDENSER
HIGH CONDENSATE LEVEL
SEE NOTE 39
REF 16
CONDENSING CHAMBER
POT CENTER LINE ELEVATION ABOVE NOZZLE FLANGE FACE
SEE REF. 5 FOR THERMOCOUPLE LOCATIONS
SEE NOTE 47
REFUELING BELLOWS PENETRATION
NOTE 61
CONDENSING CHAMBER NOTE 55
REF 31 E32-N060
REF 31 E32-N050
C32-N017 REF 6
C32-N004B REF 6
LOW PRESS CONN
TYPICAL FOR ALL CALIBRATED JET PUMPS SEE FIGURE II
TYPICAL CONNECTION TO REMAINING FT N034'S FOR INSTRUMENT SUFFIX ASSIGNMENT SEE FIGURE II
HIGH PRESS CONN.
PRESS ABOVE CORE PLATE
PRESS BELOW PLATE
REACTOR CORE PLATE ΔP
TO dPR/FR R613
C71-N050A REF 18
C71-N050D REF 18
TO FT N034R
TO FT N034N
TO FT N034L
270°
TO FT N034J & FT N033A
TO FT N034G
TO FT N034E
TO FT N034C



NOTES:

1. STEAM LINES ENCLOSED IN BOXES SHALL HAVE PART NOS. CORRESPONDING TO ITS RESPECTIVE LINE NO. UNLESS OTHERWISE NOTED. EXAMPLE: XXXB IS ON LINE "B", XXXC IS ON LINE "C".
8. DESIGN PRESSURE AND TEMPERATURE TO BE ESTABLISHED BY DESIGNER BASED ON FEED PUMP SHUTOFF PRESSURE AND SYSTEM ARRANGEMENT F011 TO BE DESIGNED FOR MOST SEVERE UPSTREAM AND DOWNSTREAM SERVICE.
11. PIPE SIZES SHOWN ON THIS DRAWING ARE APPROXIMATE EXCEPT AT POINTS OF CONNECTION WITH BWRSD SUPPLIED EQUIPMENT OR PIPING. THE PIPING DESIGNER SHALL CHECK AND ADJUST PIPE SIZE IN ACCORDANCE WITH HIS PIPING LAYOUT FOR CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.
36. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NO. B21 UNLESS OTHERWISE NOTED.
42. T/C JUNCTION BOX LOCALLY MOUNTED (BY OTHERS) EACH T/C JUNCTION BOX TO HAVE OWN SET OF TERMINALS.
45. PX TO BE CONNECTED INTO THE STRAIGHT RUN OF PIPE IMMEDIATELY DOWNSTREAM OF F028 WITH UPSTREAM AND DOWNSTREAM STRAIGHT LENGTH FROM THE TAP TO GIVE AS ACCURATE A PRESSURE MEASUREMENT AS FEASIBLE. TAPS TO MEET ASME PTC 6 1964 'STEAM TURBINES' PARAGRAPH 4.74.
47. AN EXPANSION LEG SHALL BE PROVIDED IN INSTRUMENT SENSING LINE BETWEEN POT (PART D002) AND THE WATER-TIGHT PENETRATION SEAL THROUGH BOTTOM OF REACTOR WELL. THE EXPANSION LEG & PIPING INSTALLATION SHALL BE DESIGNED TO ALLOW FOR MAXIMUM CHANGE OF VESSEL LENGTH WITH TEMPERATURE TO AVOID OVERSTRESSING THE PIPING OR THE SEAL OR DAMAGE TO THE INSULATION AROUND THE VESSEL.
48. SEE TABLE I & II ON SHEET 3 FOR POWER SUPPLY ASSIGNMENT & CONTACT UTILIZATION FOR ALL LEVEL, PRESSURE & DIFFERENTIAL PRESSURE TRANSMITTERS AND SWITCHES.
49. SUMMERS K606 & K607 INPUTS SHALL BE INTERLOCKED WITH RECIRC PUMP & VALVES TO ADD INPUT WHEN BOTH PUMPS ARE RUNNING & THEIR DISCHARGE VALVES ARE OPEN OR SUBTRACT ONE INPUT WHEN THE CORRESPONDING PUMP IS STOPPED OR ITS DISCHARGE VALVE IS CLOSED.
50.
51. WIRE TO RECORDER R006.
52.
53. LOW CONDENSER VACUUM SWITCHES CONNECTED THROUGH SEPARATE CALIBRATION VALVES TO OPPOSITE SIDES OF THE CONDENSER ABOVE THE HIGH CONDENSATE LEVEL. THE VACUUM SWITCHES MUST BE ACCESSABLE DURING PLANT OPERATION.
54. SAMPLE PROBE(S) AND FEEDWATER SAMPLE STATION TO COMPLY WITH REFERENCE 20, WATER SAMPLING SECTION 8.
55. CENTERLINE ELEVATION OF CONDENSING CHAMBER TO BE +1 1/2" RELATIVE TO CENTERLINE OR VESSEL NOZZLE.
56. RECIRC LINES TO HOTWELL TO COMPLY WITH REF 29, WATER QUALITY SECTION 7.
57. LOCATE TEE AS CLOSE AS POSSIBLE TO RPV.
58. WATER LEVEL INSTRUMENTS FOR VARIOUS RANGES ARE CALIBRATED AS STATED BELOW ALL WATER LEVEL SWITCH SET POINTS ARE NOMINAL; IE; THE ANALYSES ARE PERFORMED WITH THE SWITCH TRIP UNCERTAINTY INCLUDED. REACTOR BUILDING TEMP ASSUMED TO BE 75°F.
   - A. FUEL ZONE: THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 0 PSIG IN THE VESSEL AND THE DRYWELL WITH NO JET PUMP FLOW. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ± 6" OF WATER LEVEL AT CALIBRATION CONDITIONS.
   - B. WIDE RANGE: THE INSTRUMENTS ARE CALIBRATED FOR 1000 PSIG IN THE VESSEL, 135°F. IN THE DRYWELL AND 20 BTU/LB SUB-COOLING BELOW THE MIDDLE WATER LEVEL NOZZLE AND SATURATED CONDITIONS ABOVE THE MIDDLE WATER LEVEL NOZZLE WITH NO JET PUMP FLOW. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ± 6" OF WATER LEVEL AT CALIBRATION CONDITIONS.
   - C. NARROW RANGE: (SAFEGUARDS AND FEEDWATER): THE INSTRUMENTS ARE CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 1000 PSIG IN THE VESSEL AND 135°F. IN THE DRYWELL. WATER LEVEL SWITCH TRIP UNCERTAINTY IS ± 1.5" OF WATER LEVEL AT CALIBRATION CONDITIONS
   - D. UPSET RANGE: THE INSTRUMENT IS CALIBRATED FOR SATURATED WATER STEAM CONDITIONS AT 1000 PSIG IN THE VESSEL AND 135° F. IN THE DRYWELL.
   - E. SHUTDOWN: THE INSTRUMENT IS CALIBRATED FOR 120°F. WATER AT 0 PSIG IN VESSEL AND 80° F. IN THE DRYWELL.
59. PX TO BE CONNECTED IN A STRAIGHT RUN OF PIPE AS FAR AS POSSIBLE FROM ELBOWS, ETC. & LOCATED SO THAT ΔPS FROM THE TAPS TO THE VESSEL NOZZLE ARE EQUAL. TAPS TO MEET ASME PTC 6 1964 "STEAM TURBINES" PARAGRAPH 4.74. EIGHTEEN WIRES ARE TO BE PROVIDED PENETRATING THE DRYWELL FOR READOUT OF TEMPORARY PRESSURE TRANSMITTERS DURING START-UP.
60. PX ALTERNATE PRESSURE TAPS ARE TO BE TERMINATED INTO A SHUTOFF VALVE AND AN INSTRUMENT BALANCING VALVE.
61. PROVISION FOR ISOLATION OF INSTRUMENT LINE CONNECTING TO THE REACTOR COOLANT PRESSURE BOUNDARY TO BE:

   3/4"-51-C-15-B

   INSIDE CONTAINMENT

   NO MORE THAN ONE ORIFICE SHALL BE INSTALLED IN AN INSTRUMENT LINE. MINIMUM ORIFICE SIZE IS 1/4".
62. TEMPERATURE ELEMENTS MUST BE LOCATED BETWEEN 5 TO 6 PIPE DIAMETER FROM THE ENTRANCE OF THE FLOW NOZZLES PER ASME PTC 19.5 (1971 INTERIM ISSUE) "INSTRUMENT AND APPARATUS".
63. DRAIN VALVE CONNECTION SHOULD BE LOCATED AT LOW POINT.
64. DELETED
65. AVAILABLE CORROSION DATA INDICATES THAT FOR CARBON STEEL THE SAFETY/RELIEF VALVE DISCHARGE PIPING BELOW SUPPRESSION POOL WATER LEVEL SHOULD HAVE A CORROSION ALLOWANCE OF .250 INCHS FOR A 40 YEAR LIFE, SINCE THIS PIPING WILL CORRODE FROM BOTH SIDE.
66. VALVE F020 SHALL BE SUPPLIED WITH CLASS 1E POWER, AS SPECIFIED IN REFERENCE 31.
67. OUTBOARD MSIV-LCS SHALL BE CONNECTED INSIDE THE STEAM TUNNEL. THE PREFERRED CONNECTION SHALL BE TO A PROCESS OR DRAIN LINE THAT TIES INTO THE BOTTOM OF THE MAIN STEAM LINE.

FIG. 7.3.1-12B
NUCLEAR BOILER SYSTEM P&ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

FIGURE I ELEVATION CORRELATION CHART

REFERENCE	(COLD VESSEL) INCHES ABOVE VESSEL ZERO	DESCRIPTION OF TRIPS	INSTRUMENT(S) PROVIDING TRIP	REACTOR VESSEL LEVEL IDENTITY (REF 25)	CONTROL ROOM WATER LEVEL IN… SEE NOTE 5 — SAFEGUARDS — FUEL ZONE LR-R615 LI-R610	WIDE RANGE LR-R623A,B LI-…04	NA… C… C…
TOP OF REACTOR HEAD FLANGE	647 MAX						
STEAM LINE NOZZLE N3A,B,C,D	640						
INSTRUMENT LINE NOZZLE N12A,B	586.25						
						+60"	+6
		TRIP RCIC & HPCI TURBINE. CLOSE MAIN TURBINE STOP VALVES. TRIP FEED PUMPS	TABLE I REF 6	8		+56.5"	+5
		HIGH LEVEL ALARM	REF 6	7			
		NORMAL WATER LEVEL	REF 6	5,6			
		LOW LEVEL ALARM	REF 6	4			
		SCRAM & CLOSE RHR SHUTDOWN COOLING ISOLATION VALVES. AUTO DEPRESSURIZATION PERMISSIVE.	TABLE I	3			+5
FEEDWATER SYSTEM & REACTOR PROTECTION SYSTEM. INSTRUMENT ZERO EXCEPT FUEL ZONE. BOTTOM OF DRYER SKIRT	516.75					0"	
INSTRUMENT LINE NOZZLE N12A,B	509						
		INITIATE RCIC & HPCI & TRIP RECIRC PUMPS, CLOSE PRIMARY SYS ISOL VALVES EXCEPT RHR SHUTDOWN COOLING ISOL VALVES.	TABLE I	2		-38"	
					+30"		
		INITIATE RHR & CORE SPRAY SYSTEMS, CONTRIBUTE TO AUTO DEPRESSURIZATION SYSTEM, START STANDBY DIESEL.	TABLE I	1		-126.4	
						-150"	
TOP OF ACTIVE FUEL AND FUEL ZONE WATER LEVEL ZERO	358 9/16				0"		
INSTRUMENT LINE NOZZLE N11 A,B	358						
	310 3/16						
JET PUMP DIFFUSER TAP NOZZLE N8A,B	143				-150"		
JET PUMP INSTRUMENT	122						

OPERATION AND TRIP LEVELS			
	FEEDWATER		
NARROW RANGE		UPSET	SHUTDOWN
C32- LR-R608 C32- LI-R606A,B,C		C32- LR-R608	LI-R605
		180"	400"
+60"	+60"		
+56.5"	+56.5"		
	+40.5"		
	+37.0		
	+35.5		
+13.5"	+13.5"		
0"	0"	0"	0"

TABLE II: JET PUMP INSTRUMENTATION SUFFIX LETTER ASSIGNMENT

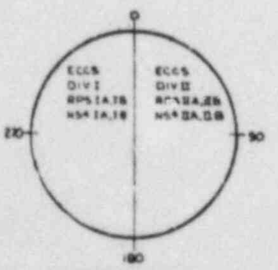
LOOP	A										B									
JP NO	11	12	13	14	15	16	17	18	19	20	1	2	3	4	5	6	7	8	9	10
FT N034 E/S X603	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
dPI R608	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
V K600	A	C	E	G	J	L	N	R	T	V	B	D	F	H	K	M	P	S	U	W
Σ	K601A				K601C				K608A		K601B				K601D				K608B	
Σ	K602A										K602B									
FT N033	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D
V R603	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D
FI R609	—	—	—	—	A	—	—	—	—	C	—	—	—	—	B	—	—	—	—	D

TABLE X: SAFETY/RELIEF VALVE SUFFIX ASSIGNMENT & ASSOCIATED EQUIPMENT

SAFETY/RELIEF VALVES	F013	A	B	C ※	D ※	E	F ※	G	H	J	K	L
ACCUMULATORS	A003	A	B	C	D	E	F	G	H	J	K	L
CHECK VALVES	F036	A	B	C	D	E	F	G	H	J	K	L
*TEMPERATURE ELEMENTS	N004 LTE(01)	A	B	C	D	E	F	G	H	J	K	L
VACUUM BREAKERS	F037 (RV095)	A	B	C	D	E	F	G	H	J	K	L
	(RV096)	A	B	C	D	E	F	G	H	J	K	L
AIR ACCUMULATOR RELIEF VALVE	F112 (RV076)	A	B	C	D	E	F	G	H	J	K	L
ADS FUNCTION		YES	YES	—	—	YES	—	—	YES	YES	YES	YES

※ CONTROL PROVIDED IN REMOTE SHUTDOWN SYS WHEN FURNISHED (REF 90)
(ADS) SAFETY/RELIEF VALVE FOR AUTO DEPRESSURIZATION

Also Available On Aperture Card



VESSEL PLAN VIEW
SEPARATION ASSIGNMENT WITH RESPECT TO VESSEL PLAN

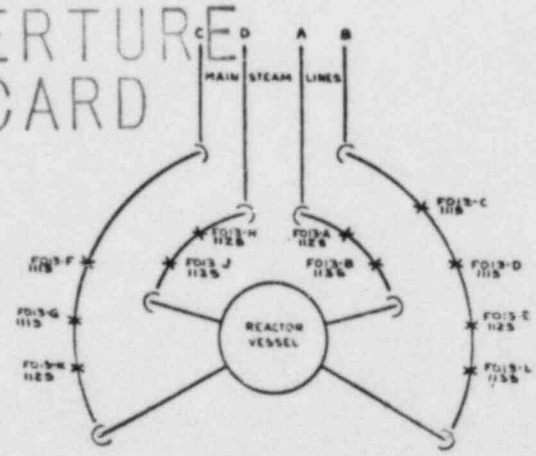


FIGURE III
SAFETY/RELIEF VALVE LOCATIONS & SPRING SET PRESSURE

FIG. 7.3.1-12C
NUCLEAR BOILER SYSTEM P& ID
SHOREHAM NUCLEAR POWER STATION-UNIT I
FINAL SAFETY ANALYSIS REPORT

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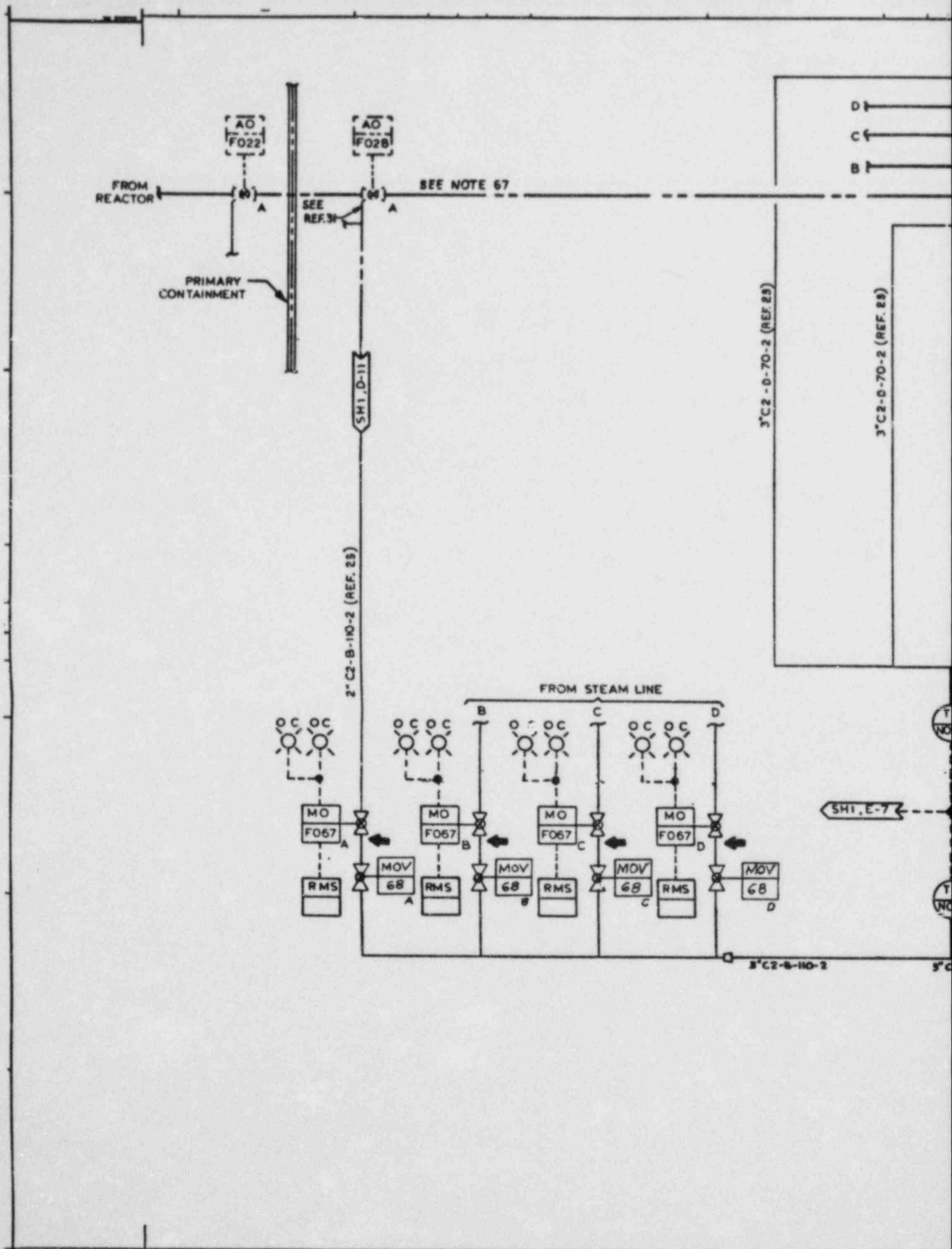

AO
F022
AO
F028
FROM REACTOR
A
SEE REF.31
A
SEE NOTE 67
PRIMARY CONTAINMENT
SH1, D-11
2" C2-B-110-2 (REF. 23)
3"C2-D-70-2 (REF. 23)
3"C2-D-70-2 (REF. 23)
D
C
B
FROM STEAM LINE
B
C
D
O C
O C
O C
O C
O C
O C
O C
O C
MO
F067
A
MO
F067
B
MO
F067
C
MO
F067
D
MOV
68
A
MOV
68
B
MOV
68
C
MOV
68
D
RMS
RMS
RMS
RMS
SH1, E-7
3"C2-B-110-2

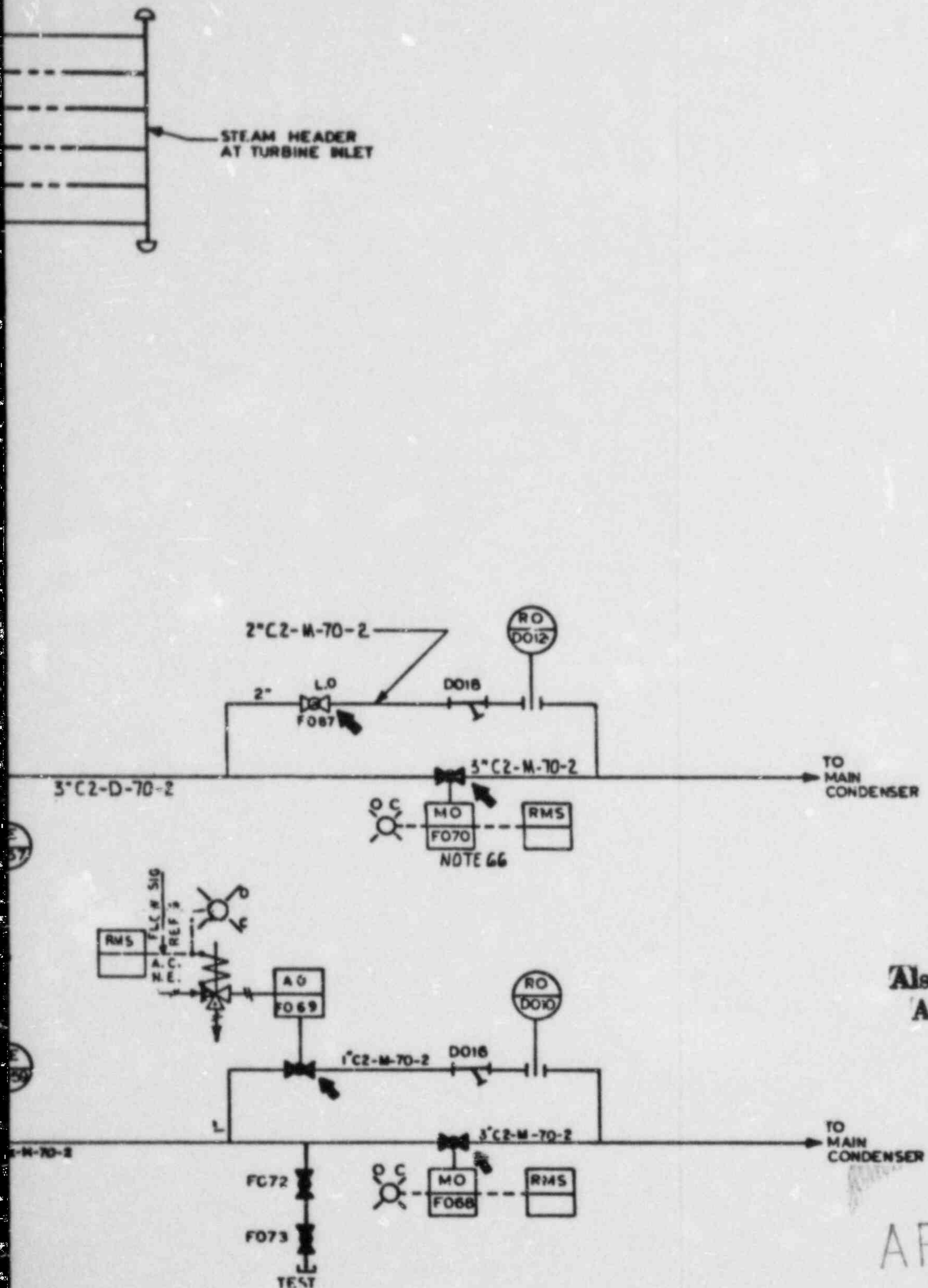

FIG. 7.3.1-12D
NUCLEAR BOILER SYSTEM P & ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

TABLE 1 WATER LEVEL INSTRUMENT CONTACT UTILIZATION

INSTRUMENT NUMBER	TRIP UNIT	DIVISION	SYSTEM TRIP FUNCTION	TRIP LEVEL	POWER SUPPLY
LT-N080A, B	LIS-N680A, B	IA, IB	RPS	3	—
LT-N080C, D	LIS-N680C, D	IIA, IIB	RPS	3	—
LT-N081A, B	LIS-N681A, B	IA, IB	NS4	2	H613A C61-K005
LT-N081C, D	LIS-N681C, D	IIA, IIB	NS4	2	H613B H603
	LIS-N691A	I	ADS(A)/RHRA, CI/CS(A)	1	—
LT-N091A	LS-N692A	I	RPT(A)/RCIC/(HPCI)*	2	—
	LS-N693A	I	RCIC/(HPCI)*	8	—
	LIS-N691B	II	ADS(B)/RHRB, DI/CS(B)	1	—
LT-N091B	LS-N692B	II	RPT(B)/HPCI/(RCIC)*	2	—
	LS-N693B	II	HPCI/(RCIC)*	8	—
	LIS-N691C	I	ADS(A)/RHRA, CI/CS(A)	1	—
LT-N091C	LS-N692C	I	RPT(A)/RCIC/(HPCI)*	2	—
	LS-N693C	I	RCIC/(HPCI)*	8	—
	LIS-N691D	II	ADS(B)/RHRB, DI/CS(B)	1	—
LT-N091D	LS-N692D	II	RPT(B)/HPCI/(RCIC)*	2	—
	LS-N693D	II	HPCI/(RCIC)*	8	—
LT-N095A	LIS-N695A	I	ADS(A)	3	—
LT-N095B	LIS-N695B	II	ADS(B)	3	—
LT-N027	—	—	SHUTDOWN LEVEL	—	H603

* SYSTEM DESIGNATED IN PARENTHESES ( ) RECEIVES AN ISOLATED INPUT FROM THIS TRIP UNIT AS A BACKUP TO ITS PRIMARY SENSOR INPUT.

TABLE II PRESSURE INSTRUMENT CONTACT UTILIZATION

INSTRUMENT NUMBER	TRIP UNIT	DIVISION	SYSTEM TRIP FUNCTION	POWER SUPPLY
PS-N020A, B	—	IA, IB	LOW PRESSURE INTERLOCK (NS4)	—
PS-N020C, D	—	IIA, IIB	LOW PRESSURE INTERLOCK (NS4)	—
	PS-N697A, C	—	RECIRC PUMP TRIP RPT (A)	—
PT-N097A, C	PS-N650A, C	I	RHR	—
	PS-N650E, G	—	RECIRC DISCHARGE VALVE CLOSURE	—
	PIS-N697B, D	—	RECIRC PUMP TRIP RPT (B)	—
PT-N097B, D	PS-N650B, D	II	RHR	—
	PS-N650F, H	—	RECIRC DISCHARGE VALVE CLOSURE	—
PT-N078A, B	PIS-N678A, B	IA, IB	RPS	—
PT-N078C, D	PIS-N678C, D	IIA, IIB	RPS	—
PS-N015A, B	—	IA, IB	NS4	—
PS-N015C, D	—	IIA, IIB	NS4	—
dPIS-N006A, C	—	IA, IB	NS4	—
dPIS-N006C, D	—	IIA, IIB	NS4	—
dPIS-N007A, B	—	IA, IB	NS4	—
dPIS-N007C, D	—	IIA, IIB	NS4	—
dPIS-N008A, B	—	IA, IB	NS4	—
dPIS-N008C, D	—	IIA, IIB	NS4	—
dPIS-N009A, B	—	IA, IB	NS4	—
dPIS-N009C, D	—	IIA, IIB	NS4	—
VS-N056A,B	—	IA, IB	NS4	—
VS-N056C,D	—	IIA, IIB	NS4	—
PT-N0[illegible]A,B	—	I, II	VESSEL PRESSURE	K613A, B
dPT-N032	—	—	CORE PLATE Δ P	K603

PRC
APERTURE
CARD

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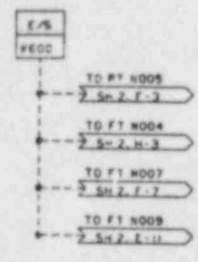
Aperture Card

FIG. 7.3.1-12E
NUCLEAR BOILER SYSTEM P & ID
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

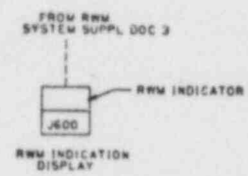
8307050091 -13

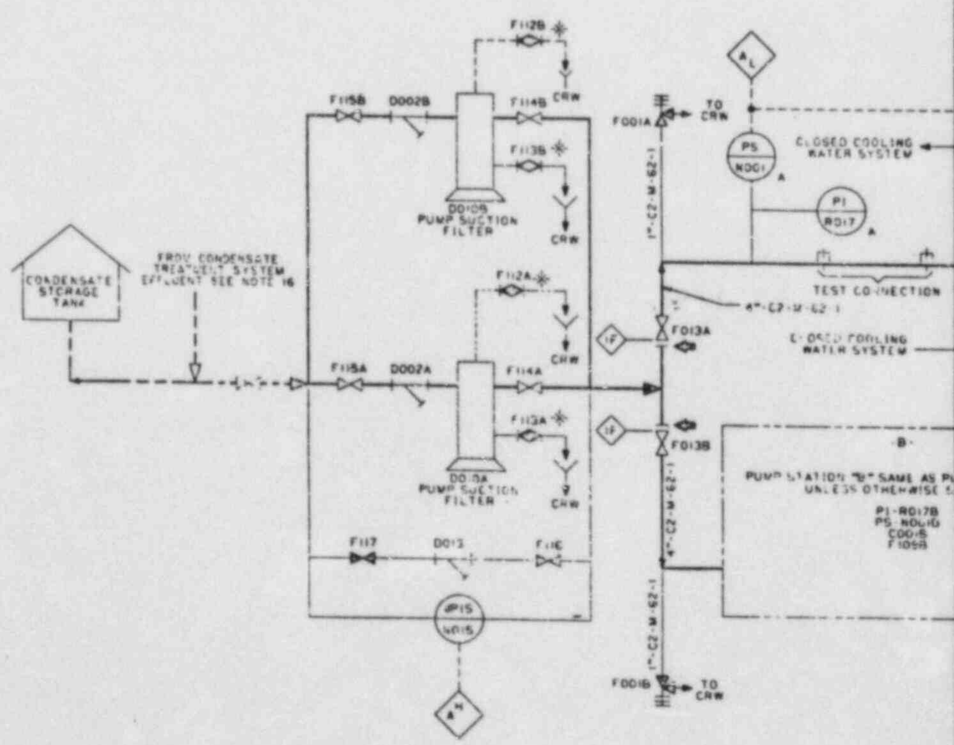
SUPPLEMENTAL DOCUMENTS

1. NUCLEAR BOILER SYS P&ID
2. CRD HYDRAULIC SYS PD
3. CRD HYDRAULIC SYS FCD
4. CRD HYDRAULIC SYS PIPING ARGMT
5. REACTOR PROTECTION SYS (RPS) IED
6. PIPING & INSTRUMENT SYMBOLS
7. PROCESS INSTRUMENT PIPING AND TUBING SPECIFICATION
8. PRESSURE INTEGRITY OF PIPING AND EQUIPMENT PRESSURE PARTS
9. CRD HYDRAULIC SYS DESIGN SPEC
10. REACTOR RECIRC SYS P&ID
11. CRD INSTRUMENT DATA SHEETS
12. REACTOR MANUAL CONTROL ELEM DIAG
13. WATER SAMPLING
14. WATER QUALITY
15. REACTOR WATER CLEANUP SYS P&ID
16. REACTOR RECIRC PUMP AND MG SET ELEM DIAG
17. CRD HYD SYS INSTR ELEM DIAG



POWER DISTRIBUTION





OVERALL REV. [illegible]

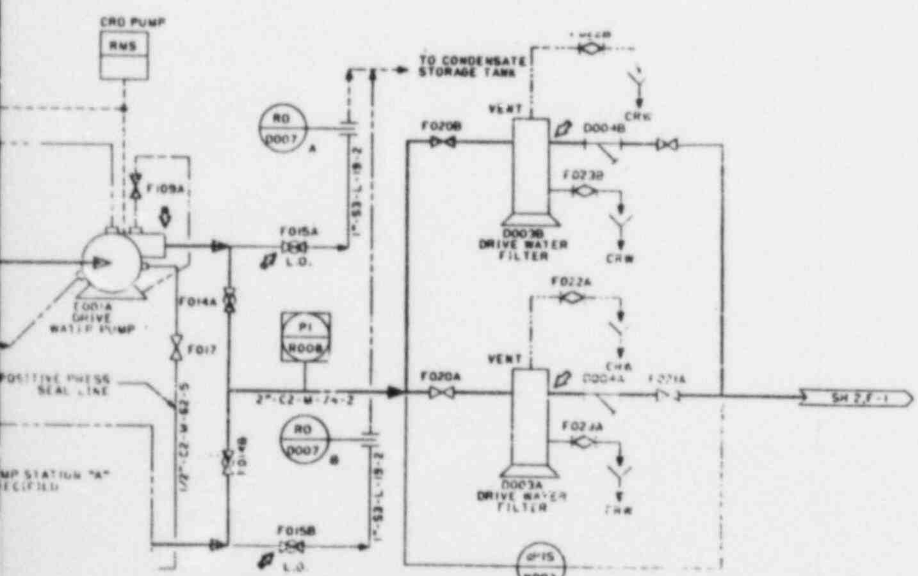
MPL NO. C11-[illegible]

NOTES:

1. VALVE F007A-A CLOSES ON DRIVE INSERT SIGNAL. VALVE F007A-B CLOSES ON DRIVE WITHDRAWAL SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING. (SHEET 2, J-9)
2. REACTOR PRESSURE SENSING LINE, SH 2, K-13.
3. STABILIZING VALVE F007B IS AN ALTERNATE FOR STABILIZING VALVE F007A. (SHEET 2, J-9)
4. PROVIDE VENT VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM HIGH POINTS.
5. PROVIDE DRAIN VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM LOW POINTS.
6. PROVIDED FOR SYSTEM FLUSHING. (SHEET 2, H-9 & H-10)
7. AVAILABLE FOR TEMPORARY CONNECTION FOR INSTRUMENT FLUSHING. NO PERMANENT PIPING CONNECTIONS TO BE MADE TO THIS VALVE. (SHEET 2, E-8)
8. EXCEPT AT POINTS OF CONNECTION WITH NED SUPPLIED EQUIPMENT OR PIPING, THE PIPING DESIGNER SHALL SIZE PIPES IN CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.
9. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS, SEE SUPPL DOC 11.
10. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY SYSTEM NUMBER (C11) UNLESS OTHERWISE NOTED.
11. CRD NITROGEN AND AIR LINES SHALL BE OF NON-CORRODING MATERIAL.
12. THESE VALVES MUST BE OPEN FOR DRIVE MOVEMENT.
13. MULTIPLE ORIFICES CONNECTED IN SERIES AS SHOWN IN PURCHASE PART DRAWING OF ORIFICE D008. THE PRESSURE DROP ACROSS EACH ORIFICE IS 250 PSI AT PUMP RUN-OUT CONDITION. SEE MPL FOR THE QUANTITIES OF ORIFICES. VALVE F034 SUPPLEMENTS THE ORIFICES D008 FOR THE REQUIRED PRESSURE DROP.
14. THE SCRAM DISCHARGE VOLUME ARRANGEMENT SHOWN IS FOR REFERENCE ONLY. SEE CRD DESIGN SPEC FOR REQUIREMENTS.
15. PIPING QUALITY CLASS EXTENDS TO CONNECTIONS WITH HCU. HCU DIAGRAM IS SHOWN FOR INFORMATION ONLY. FOR QUALITY CLASS OF THE HCU, SEE GROUP CLASSIFICATION DIAGRAM.
16. SOURCE OF CRD SYSTEM WATER SHALL BE NORMALLY FROM CONDENSATE TREATMENT SYSTEM. CONDENSATE STORAGE TANK IS THE ALTERNATE SOURCE IF CONDENSATE TREATMENT SYSTEM IS NOT IN OPERATION. FOR DETAILED DESIGN REQUIREMENTS FOR SOURCE AND QUALITY OF WATER, SEE SUPPL DOC 9.
17. CONTINUOUS FLOW TO REACTOR SAMPLE STATION SHALL BE APPROXIMATELY ONE (1) LITER PER MINUTE, MINIMUM AND SHALL BE TWO (2) LITERS PER MINUTE, MAXIMUM.
18. EXHAUST WATER FLOW FROM ONE MOVING DRIVE IS DISPERSED VIA THE HYDRAULIC CONTROL UNITS OF OTHER NON MOVING DRIVES TO THE REACTOR PRESSURE VESSEL.
19. SCRAM DISCHARGE DRAIN MUST BE SUBMERGED DUE TO POTENTIAL RELEASE OF STEAM AND RADIOACTIVE NONCONDENSIBLES.
20. THE F159B VALVE SHALL BE ADJUSTED SO THAT THE F180 VALVE AND THE F181 VALVE FULLY CLOSE AT LEAST FIVE (5) SECONDS AFTER THE F010 AND F011 VALVES, RESPECTIVELY, DURING A FULL CORE SCRAM.

NOTES (CONT)

21. THE F159A VALVE SHALL BE ADJUSTED SO THAT THE F010 AND F011 VALVES STARTS TO OPEN AT LEAST FIVE (5) SECONDS AFTER THE F180 AND F181 VALVES, RESPECTIVELY, UPON THE RESET OF A FULL CORE SCRAM.
22. THE SCRAM DISCHARGE VOLUME VENT LINE SHALL BE ROUTED VIA A DEDICATED LINE TO THE NON SUBMERGED DISCHARGE. THE VENT SYSTEM PIPING SHOULD NOT CONTAIN SIGNIFICANT FLOW RESTRICTIONS WHICH COULD INHIBIT VENTING OF THE SCRAM DISCHARGE VOLUME TO THE EXTENT OF LIMITING THE SCRAM DISCHARGE VOLUME DRAIN RATE. THE DESIGN OF THE NON SUBMERGED VENT DISCHARGE MUST CONSIDER THE POTENTIAL FOR THE RELEASE OF RADIOACTIVITY DUE TO THE DISCHARGE OF NON CONDENSIBLES, WATER AND STEAM WHICH MAY OCCUR DURING THE PERIOD AFTER SCRAM PRIOR TO VENT VALVE CLOSURE AND UPON SCRAM RESET WHEN THE VENT VALVES ARE REOPENED WITH THE SCRAM DISCHARGE VOLUME PRESSURIZED.
23. TO PREVENT LOOP SEALS FROM OCCURING IN THE RELATIVELY SMALL DIAMETER LINES OF THE SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN SYSTEM, A CONTINUOUS DOWNWARD PITCH AWAY FROM THE SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES MUST BE MAINTAINED. THERMAL EXPANSION EFFECTS SHOULD BE ADDRESSED IN THE DESIGN OF THE VENT AND DRAIN SYSTEMS.
24. THE DESIGN OF THE SCRAM DISCHARGE VOLUME (SDV), ASSOCIATED VENT AND DRAIN SYSTEM PIPING AND COMPONENTS MUST CONSIDER THE POTENTIAL PRESSURE, TEMPERATURE AND TRANSIENT LOADING WHICH MAY RESULT FROM: (A) THE DISCHARGE TO THE SCRAM DISCHARGE VOLUME AND DOWN THE VENT AND DRAIN LINE DURING A FULL SCRAM, PRIOR TO VENT AND DRAIN VALVE CLOSURE; (B) THE PRESSURIZATION OF THE SCRAM DISCHARGE VOLUME TO REACTOR VESSEL PRESSURE FOLLOWING VENT AND DRAIN VALVE CLOSURE; AND (C) DEPRESSURIZATION OF THE SCRAM DISCHARGE VOLUME UPON SCRAM RESET AND OPENING OF THE SCRAM DISCHARGE VOLUME VENT AND DRAIN VALVES.
25. VACUUM BREAKER VALVE (F120) AND VENT VALVES (F010 AND F180) TO BE ON HIGH POINT OF VENT LINE. VACUUM BREAKER VALVE SHALL OPEN ON A DIFFERENTIAL PRESSURE SETTING OF NO GREATER THAN FIVE INCHES OF WATER. THE LOCATION OF THE VACUUM BREAKER SHOULD CONSIDER THE POTENTIAL RELEASE OF RADIOACTIVE NONCONDENSIBLES, WATER AND STEAM THAT COULD OCCUR IF THE VACUUM BREAKER VALVE WERE TO FAIL IN AN OPEN POSITION WHEN THE VENT SYSTEM PIPING IS PRESSURIZED.
26. INSTRUMENT LOW POINT PIPING TAP SHOULD BE ± 12 INCHES ABOVE THE SCRAM DISCHARGE INSTRUMENT VOLUME REFERENCE DATUM ELEVATION.
27. THE ELEVATION WITHIN THE SCRAM DISCHARGE INSTRUMENT VOLUME CORRESPONDING TO THE SCRAM INITIATION SET-POINT SHALL BE ± 10 INCHES BELOW THE LOWEST ELEVATION OF THE HORIZONTAL SCRAM DISCHARGE VOLUME PIPING.
28. SCRAM DISCHARGE VENT AND DRAIN VALVE CONTROL ROOM INDICATORS SHALL INDICATE OPEN WHEN BOTH VALVES ARE OPEN AND SHALL INDICATE CLOSED WHEN EITHER VALVE IS CLOSED.
29. THE PNEUMATIC PIPING FROM NEEDLE VALVE, F159B & SOLENOID VALVE, F009, TO THE VENT AND DRAIN VALVES, F181, F011, F010, F180, SHALL BE ROUTED AND SUPPORTED TO MINIMIZE THE POSSIBILITY OF PIPE CRIMP, AND ALSO IT SHOULD BE OF ADEQUATE PIPE OR TUBING THICKNESS AND MATERIAL TO MINIMIZE THE CONSEQUENCES OF PIPE CRIMP.



LEGEND

SOLENOID OPERATED VALVE W/BUILT-IN FLOW CONTROL

BLOCK TYPE NEEDLE VENT VALVE (ANGLE PATTERN)

LDS ------ LEAK DETECTOR SWITCH

RWM ------ ROD WORTH MINIMIZER

TS_C ------ TEMPERATURE SCANNER

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FIG. 7.7.1-1A

CONTROL ROD DRIVE HYDRAULIC SYSTEM

SHOREHAM NUCLEAR POWER STATION - UNIT I

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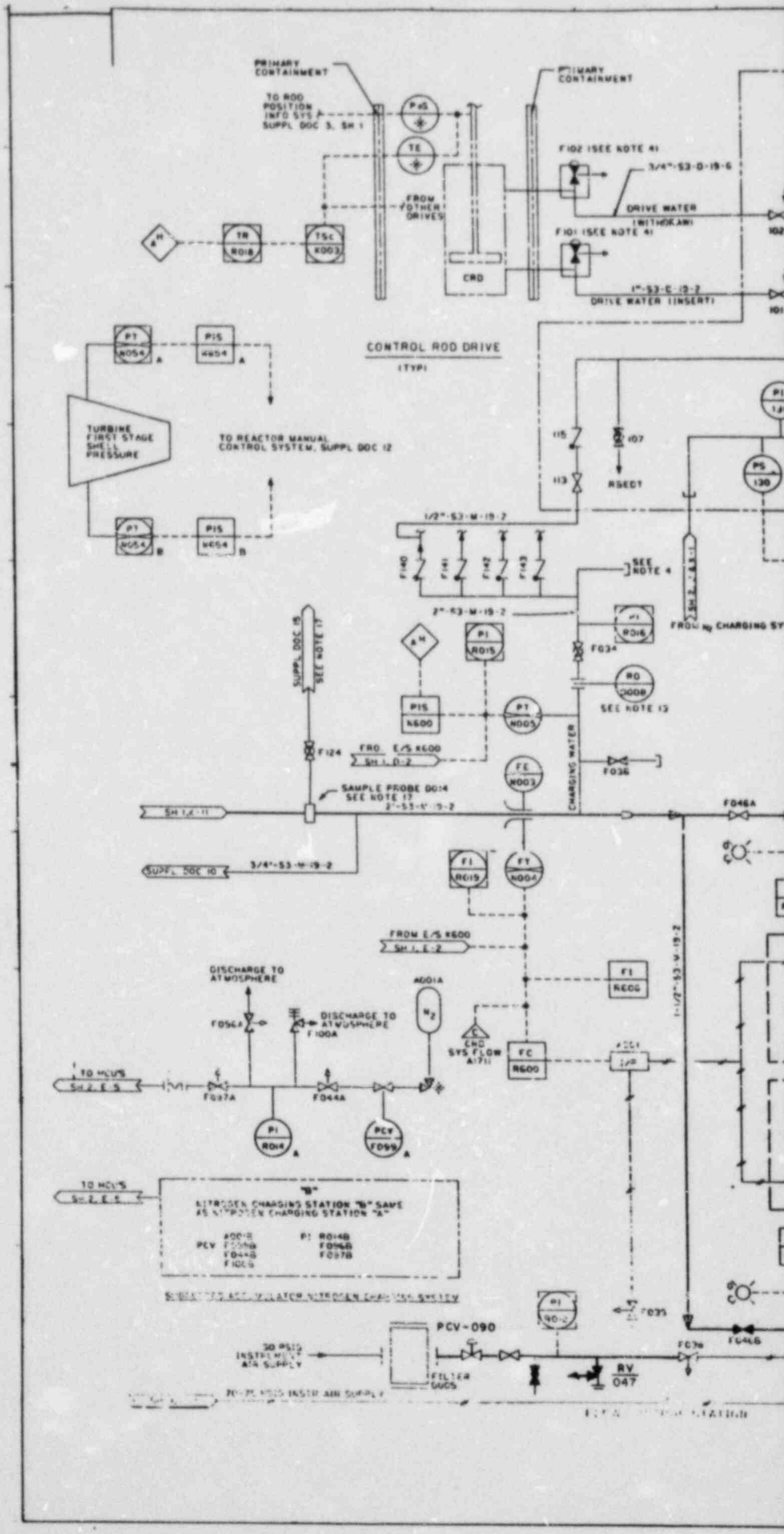

PRIMARY CONTAINMENT
TO ROD POSITION INFO SYS A SUPPL DOC 3, SH 1
FROM OTHER DRIVES
CRD
CONTROL ROD DRIVE (TYP)
F102 (SEE NOTE 4)
F101 (SEE NOTE 4)
DRIVE WATER (WITHDRAW)
DRIVE WATER (INSERT)
TURBINE FIRST STAGE SHELL PRESSURE
TO REACTOR MANUAL CONTROL SYSTEM, SUPPL DOC 12
1/2"-S3-M-19-2
2"-S3-M-19-2
SEE NOTE 4
SEE NOTE 13
CHARGING WATER
SAMPLE PROBE D014 SEE NOTE 17
SUPPL DOC 10
FROM E/S K600
DISCHARGE TO ATMOSPHERE
CRD SYS FLOW
"B"
NITROGEN CHARGING STATION "B" SAME AS NITROGEN CHARGING STATION "A"
PCV-090
30 PSIG INSTRUMENT AIR SUPPLY
FILTER

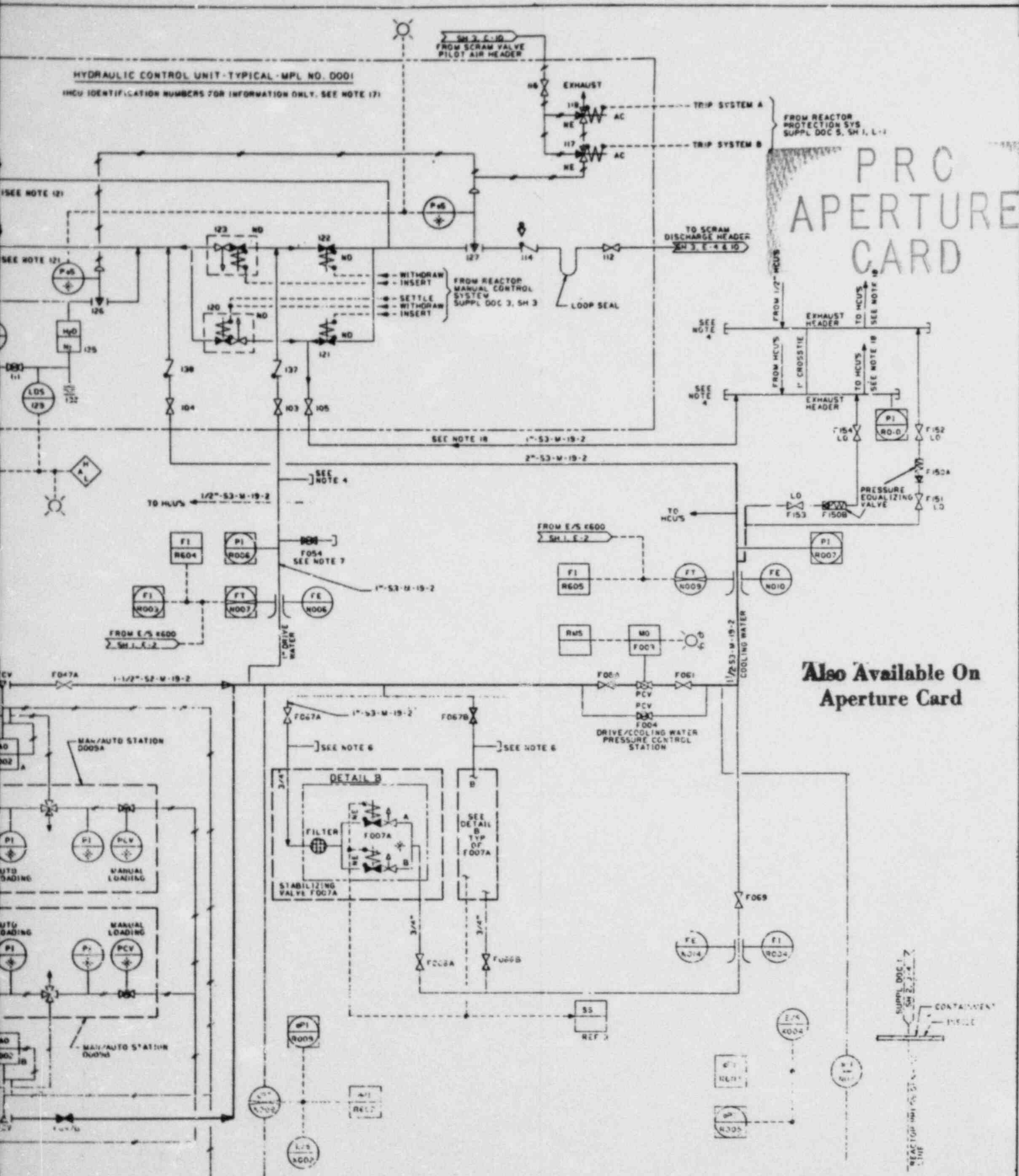

FIG. 7.7.1-1B

CONTROL ROD DRIVE HYD. SYS.

SHOREHAM NUCLEAR POWER STATION - UNIT I

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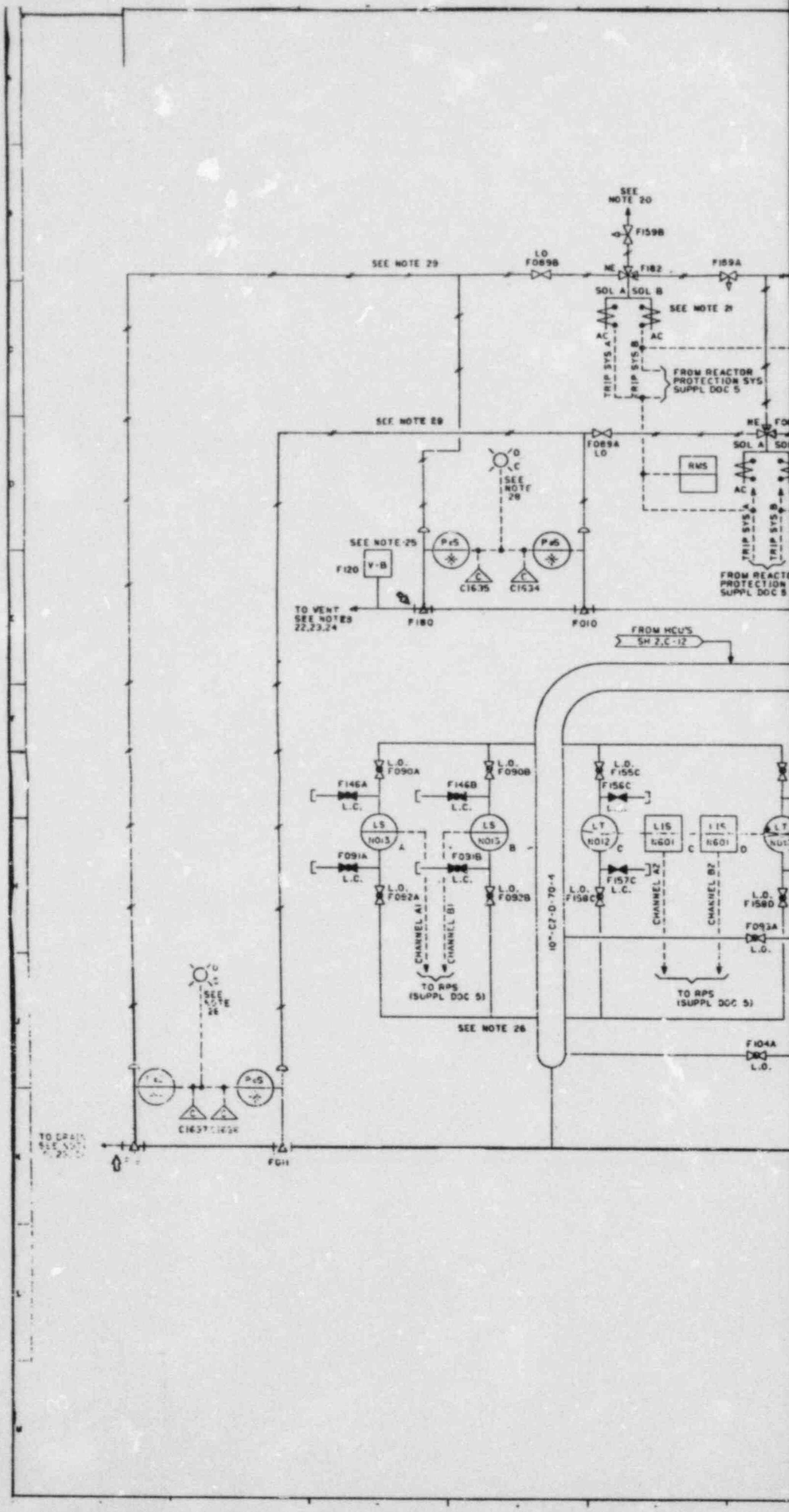
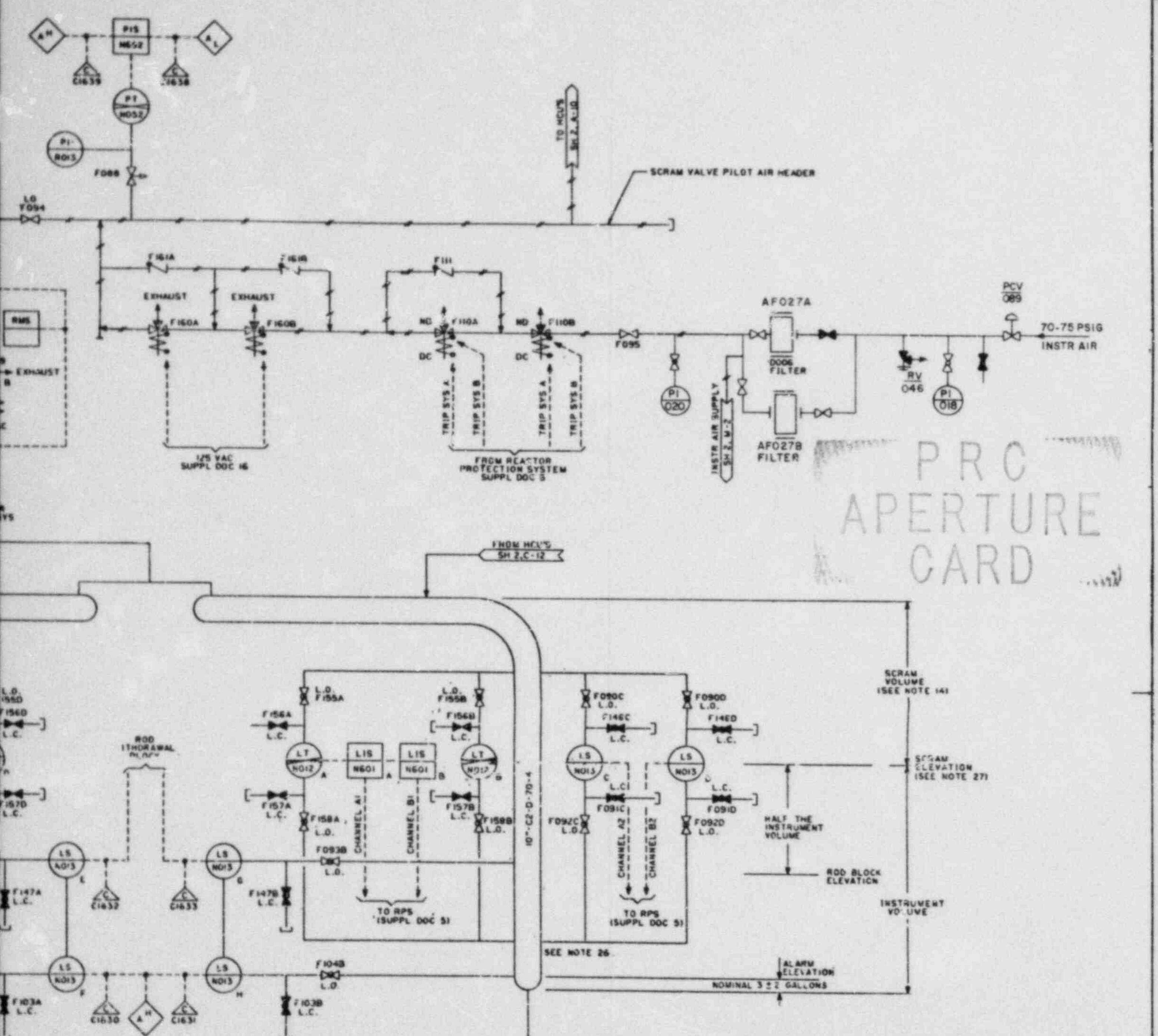

SEE NOTE 20
F159B
SEE NOTE 29
LO F089B
NE F182
F159A
SOL A SOL B
SEE NOTE 21
TRIP SYS A
TRIP SYS B
FROM REACTOR PROTECTION SYS SUPPL DOC 5
SEE NOTE 29
F089A LO
SEE NOTE 28
RMS
SEE NOTE 25
F120
V-B
C1635
C1634
TO VENT SEE NOTES 22,23,24
F180
F010
FROM HCU'S SH 2,C-12
L.O. F090A
L.O. F090B
F146A L.C.
F146B L.C.
L.O. F155C
F156C
LS N013
LT N012
LIS N601
F091A L.C.
F091B
F157C L.C.
L.O. F092A
L.O. F092B
L.O. F158C
CHANNEL A1
CHANNEL B1
CHANNEL A2
CHANNEL B2
10"-C2-D-70-4
L.O. F158D
F093A L.O.
TO RPS (SUPPL DOC 5)
SEE NOTE 26
SEE NOTE 28
F104A L.O.
F011



PRESSURE-TEMPERATURE INDEX REF 8 & 10					
P-T INDEX	DESIGN		PEAK		MIN °F
	PSIG	°F	PSIG	°F	
1	150	150	275	150	—
2	1750	150	1663	150	—
3	1135	560	1761	530	50
4	1140	[illegible]	[illegible]	[illegible]	—
5	[illegible]	[illegible]	[illegible]	[illegible]	—
6	1750	500	[illegible]	535	—

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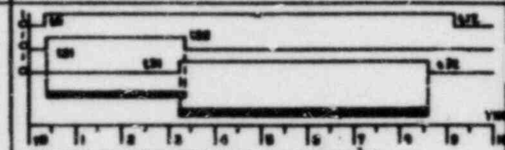
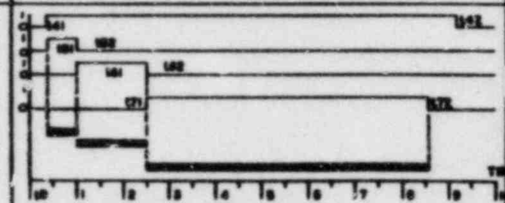
FIG. 7.7.1-1C
CONTROL ROD DRIVE
HYDRAULIC SYSTEM
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 -16

TABLE 1

TIME DEPENDENT INTERMEDIATE VARIABLES		
SYM.	DESCRIPTION	DEFINITION
T	INTERNALLY GENERATED TIME REFERENCE FUNCTION	
TIC TII TIS - -	INSERT CYCLE ACTIVE INSERT PERIOD OF INSERT CYCLE SETTLE PERIOD OF INSERT CYCLE ROD INSERT CONTROL SETTLE CONTROL	 ABSCISSA INDICATES TIME ELAPSED WHILE ADVANCING. THE INSERT CYCLE IS CONTROLLED BY IIC AND IIT AS FOLLOWS: 1. WHEN IIC IIT = 1, THE CYCLE IS ADVANCING WITH TIME. 2. WHEN IIC = 0, THE CYCLE IS RESET TO t0. 3. WHEN IIC $\overline{IIT}$ = 1, THE CYCLE STOPS, BUT DOES NOT RESET.
TWC TWI TWW TWS - - -	WITHDRAW CYCLE ACTIVE INSERT PERIOD OF WITHDRAW CYCLE WITHDRAW PERIOD OF WITHDRAW CYCLE SETTLE PERIOD OF WITHDRAW CYCLE UNLATCH CONTROL ROD WITHDRAW CONTROL SETTLE CONTROL	 ABSCISSA INDICATES TIME ELAPSED WHILE ADVANCING. THE WITHDRAW CYCLE IS CONTROLLED BY IWC AND IWT AS FOLLOWS: 1. WHEN IWC IWT = 1, THE CYCLE IS ADVANCING WITH TIME. 2. WHEN IWC = 0, THE CYCLE IS RESET TO t0. 3. WHEN IWC $\overline{IWT}$ = 1, THE CYCLE STOPS, BUT DOES NOT RESET.

SEE TABLE 2

TABLE 3

SYSTEM PERFORMANCE					
	INTERVAL	PARAMETER (SEE TABLE 2)	VALUE	MAX ALLOWED TIMES TO ASSURE PROPER DRIVE PERFORMANCE	UNITS
INSERT CYCLE	TIME DELAY TO ROD INSERT CONTROL	t21	0.42	0.0-0.8	SEC
	ROD INSERT CONTROL	t22-t21	2.00	2.8-3.1	SEC
	SWITCHING OVERLAP	t23-t22	0.10	0.0-1.0	SEC
	SETTLE CONTROL	t32-t22	5.20	4.2-6.3	SEC
WITHDRAW CYCLE	TIME DELAY TO UNLATCH CONTROL	[illegible]	0.42	0-0.6	SEC
	UNLATCH CONTROL	[illegible]	0.60	0.5-0.8	SEC
	INTERVAL BETWEEN UNLATCH AND WITHDRAW CONTROL	[illegible]	0.10	0.0-0.10	SEC
	ROD WITHDRAWAL CONTROL	[illegible]	1.50	1.3-1.7	SEC
	SWITCH OVERLAP FROM WITHDRAWAL TO SETTLE CONTROL	[illegible]	0.10	0.0-1.0	SEC
	SETTLE CONTROL	[illegible]	5.00	4.2-6.3	SEC

ROD XX-YY IS SELECTED (CONTACT 1)

ROD SELECTION IS INHIBITED OO-OO

ROD XX-YY IS SELECTED (CONTACT 2)

ROD SELECTION IS INHIBITED OO-OO

REF: [illegible] (C11-[illegible])

TABLE 2

SYSTEM PARAMETER VALUES				
	PARAMETER	VALUE	TOLERANCE	UNITS
INSERT CYCLE	[illegible]	[illegible]	+ 0.00 − [illegible]	SEC.
	[illegible]	[illegible]	± 0.00	
	[illegible]	0.42		
	[illegible]	[illegible]		
	[illegible]	[illegible]		
	[illegible]	[illegible]	± 0.00	SEC.
WITHDRAW CYCLE	141	[illegible]	+ 0.00 − [illegible]	SEC.
	142	[illegible]	± 0.00	
	[illegible]	0.42		
	[illegible]	1.02		
	[illegible]	1.12		
	[illegible]	2.02		
	171	[illegible]		
	172	[illegible]	± 0.00	SEC.

INSERT CYCLE

[illegible] = DELAY UNTIL ROD MOTION BEGINS

[illegible] = DRIVE IN TIME

[illegible] = SETTLE TIME

[illegible] = TIME WHEN CONTINUOUS INSERT CAN BE REQUESTED

[illegible] = CYCLE STOP POINT FOR CONTINUOUS INSERT

WITHDRAW CYCLE

[illegible] = DELAY UNTIL ROD MOTION BEGINS

[illegible] = DRIVE IN TIME (UNLATCH)

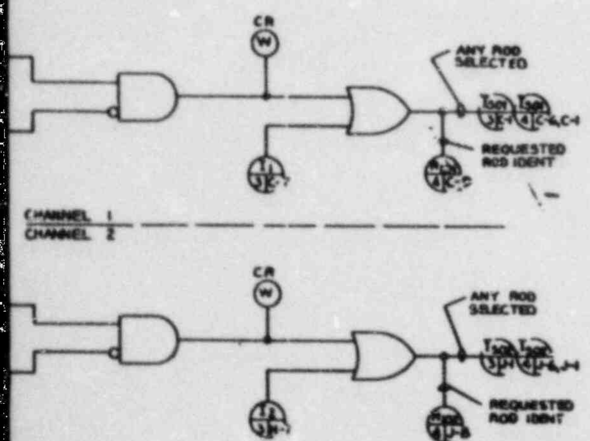
[illegible] = DELAY AFTER UNLATCH

[illegible] = DRIVE OUT TIME

[illegible] = SETTLE TIME

[illegible] = TIME WHEN CONTINUOUS WITHDRAW CAN BE REQUESTED

[illegible] = CYCLE STOP POINT FOR CONTINUOUS WITHDRAW



LEGEND:

[illegible] = SWITCHGEAR DEVICE FUNCTION NUMBER ANSI SPEC C37.2
RWM = ROD WORTH MINIMIZER
RPIS = ROD POSITION INFORMATION SYSTEM
NMS = NEUTRON MONITORING SYSTEM
PRM = POWER RANGE MONITOR
RBM = ROD BLOCK MONITOR
SRI = SELECT ROD INSERT
RMS = REMOTE MANUAL SWITCH

NOTES:

1. EACH CONTROL ROD, AS IT TRAVELS UP (INSERTED) OR DOWN (WITHDRAWN) PASSES A SERIES OF SWITCHES. THE TOP TWO POSITION SWITCHES ARE CALLED "OVERTRAVEL" AND THE BOTTOM TWO POSITION SWITCHES ARE CALLED "WITHDRAWN" (BACKSEAT & DISCONNECT). SWITCHES IN BETWEEN ARE DIVIDED INTO ODD (DRIFT) AND EVEN (LATCH) POSITIONS. AS THE ROD TRAVELS OVER ANY SWITCH AN INDICATION SIGNAL IS ACTIVATED. ANY EVEN SWITCH WILL INDICATE [illegible] POSITION [illegible] AND ANY ODD SWITCH WILL INDICATE "ODD".
2. WIRING FROM HCU (SCRAM VALVES AND ACCUMULATOR) TO CONTROL ROOM FOR ANNUNCIATION SHALL BE IN SERIAL CONNECTION FOR ALL HCU'S.
3. WIRING FROM HCU (ROD SCRAM TEST SWITCH IN TEST POSITION AND DISPLAY OF THOSE CONTROL RODS CHOSEN FOR "SELECT ROD INSERT FUNCTION") SHALL BE IN SERIAL CONNECTION FOR ALL HCU'S.
4. EACH ACCUMULATOR FAILURE WILL INITIATE AN ANNUNCIATION (ANNUNCIATOR HORN & FLASHING ANNUNCIATOR WINDOW) AND AN INDIVIDUAL [illegible] OPERATION OF THE "ACCUMULATOR TROUBLE ACKNOWLEDGE" SWITCH WILL CLEAR THE INPUT TO THE ANNUNCIATOR AND CHANGE THE INDIVIDUAL INDICATOR FROM FLASHING TO STEADY. CLEARING THE ACCUMULATOR TROUBLE WILL CLEAR THE INDIVIDUAL INDICATORS.
5. SEE REF 17 FOR DEFINITIONS OF VARIABLES APPEARING ON THIS FCD.
6. A LOGICAL "1" INDICATES A FAILED COMPARISON

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REFERENCE DOCUMENTS

1. CONTROL ROD DRIVE HYDRAULIC SYS P&ID
2. NEUTRON MONITORING SYS IED
3. FEEDWATER CONTROL SYS IED
4. FEEDWATER CONTROL SYS DESIGN SPEC
5. CONTROL ROD DRIVE HYDRAULIC SYS DESIGN SPEC
6. PROCESS COMPUTER SYS INPUT/OUTPUT REQUIREMENTS
7. POSITION INDICATOR PROBE CONN DIAG
8. REACTOR MANUAL CONTROL SYS ELEM DATA SHEET
9. NEUTRON MONITORING SYS FCD
10. REACTOR PROTECTION SYS IED
11. CONTROL ROD DRIVE HYDRAULIC INSTR SYS ELEM DIAG
12. CONTROL ROD DRIVE CONTROL SYS ELEM DIAG
13. REACTOR PROTECTION SYS ELEM DIAG
14. SERVICE PLATFORMS
15. REFUELING PLATFORM EQUIPMENT
16. ROD DRIVE CONTROL CABINET SCHEMATIC DIAGRAM
17. RWM OPERATOR PANEL SCHEMATIC DIAGRAM

SYMBOL DEFINITIONS:

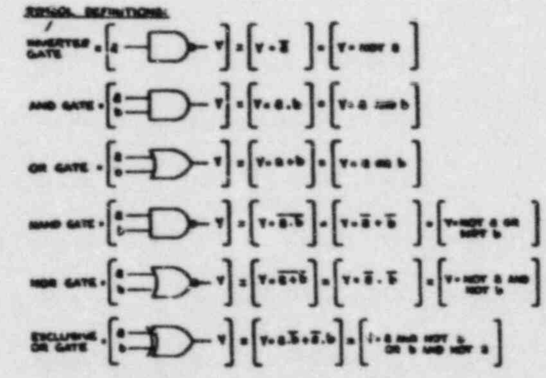
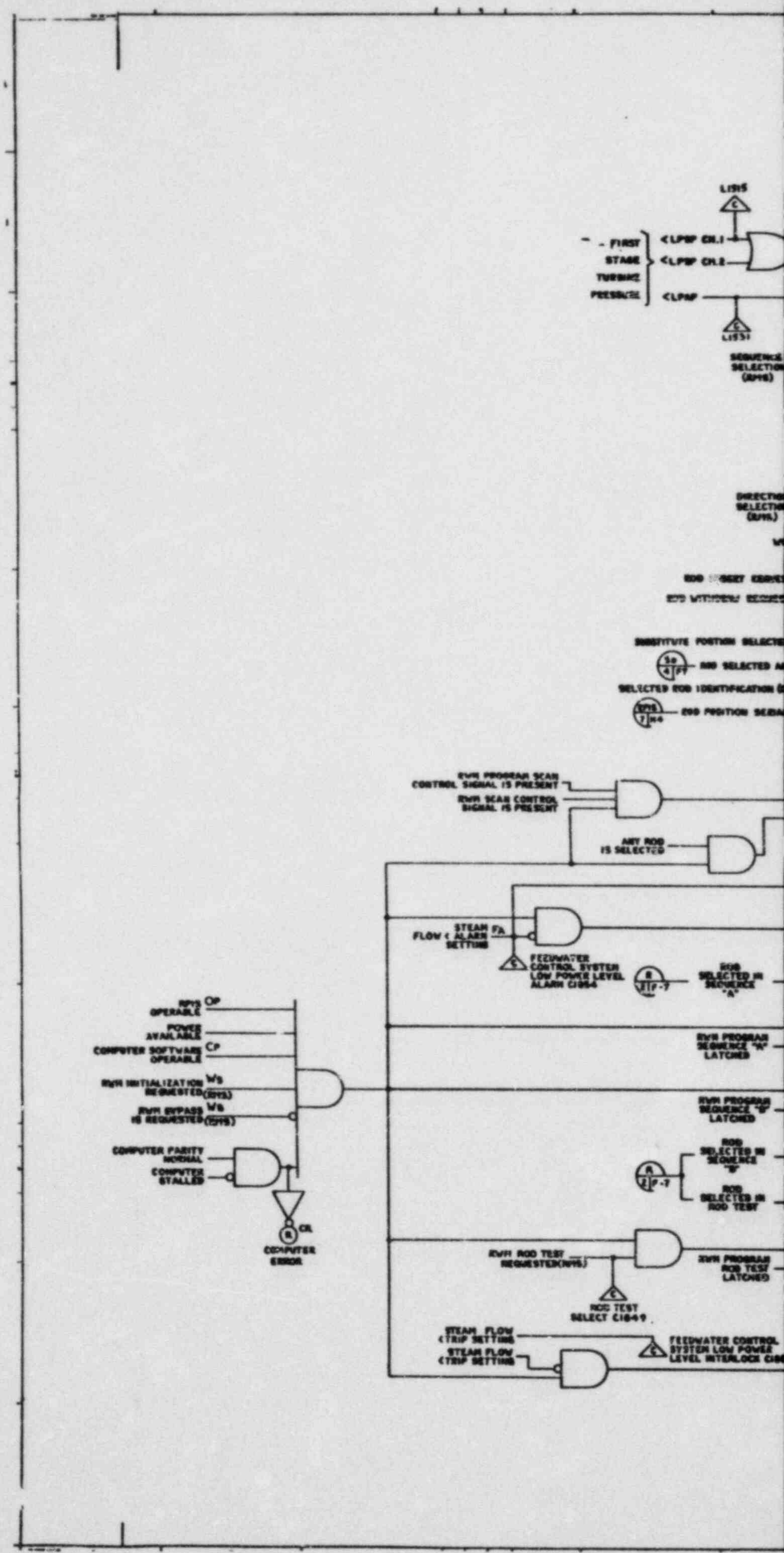


FIG. 7.7.1-2A
CONTROL ROD DRIVE HYD. SYS. FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 - 17



L1515
FIRST STAGE TURBINE PRESSURE
<LPSP CH.1
<LPSP CH.2
<LPAP
L1551
SEQUENCE SELECTION
DIRECTION SELECTION
RWM PROGRAM SCAN CONTROL SIGNAL IS PRESENT
RWM SCAN CONTROL SIGNAL IS PRESENT
ANY ROD IS SELECTED
STEAM FLOW < ALARM SETTING
FA
FEEDWATER CONTROL SYSTEM LOW POWER LEVEL ALARM C1664
ROD SELECTED IN SEQUENCE "A"
RWM PROGRAM SEQUENCE "A" LATCHED
RPIS OPERABLE
OP
POWER AVAILABLE
COMPUTER SOFTWARE OPERABLE
CP
RWM INITIALIZATION REQUESTED
WS
RWM BYPASS IS REQUESTED
WB
RWM PROGRAM SEQUENCE "B" LATCHED
COMPUTER PARITY NORMAL
COMPUTER STALLED
ROD SELECTED IN SEQUENCE "B"
ROD SELECTED IN ROD TEST
CR
COMPUTER ERROR
RWM ROD TEST REQUESTED
RWM PROGRAM ROD TEST LATCHED
ROD TEST SELECT C1649
STEAM FLOW <TRIP SETTING
STEAM FLOW <TRIP SETTING
FEEDWATER CONTROL SYSTEM LOW POWER LEVEL INTERLOCK

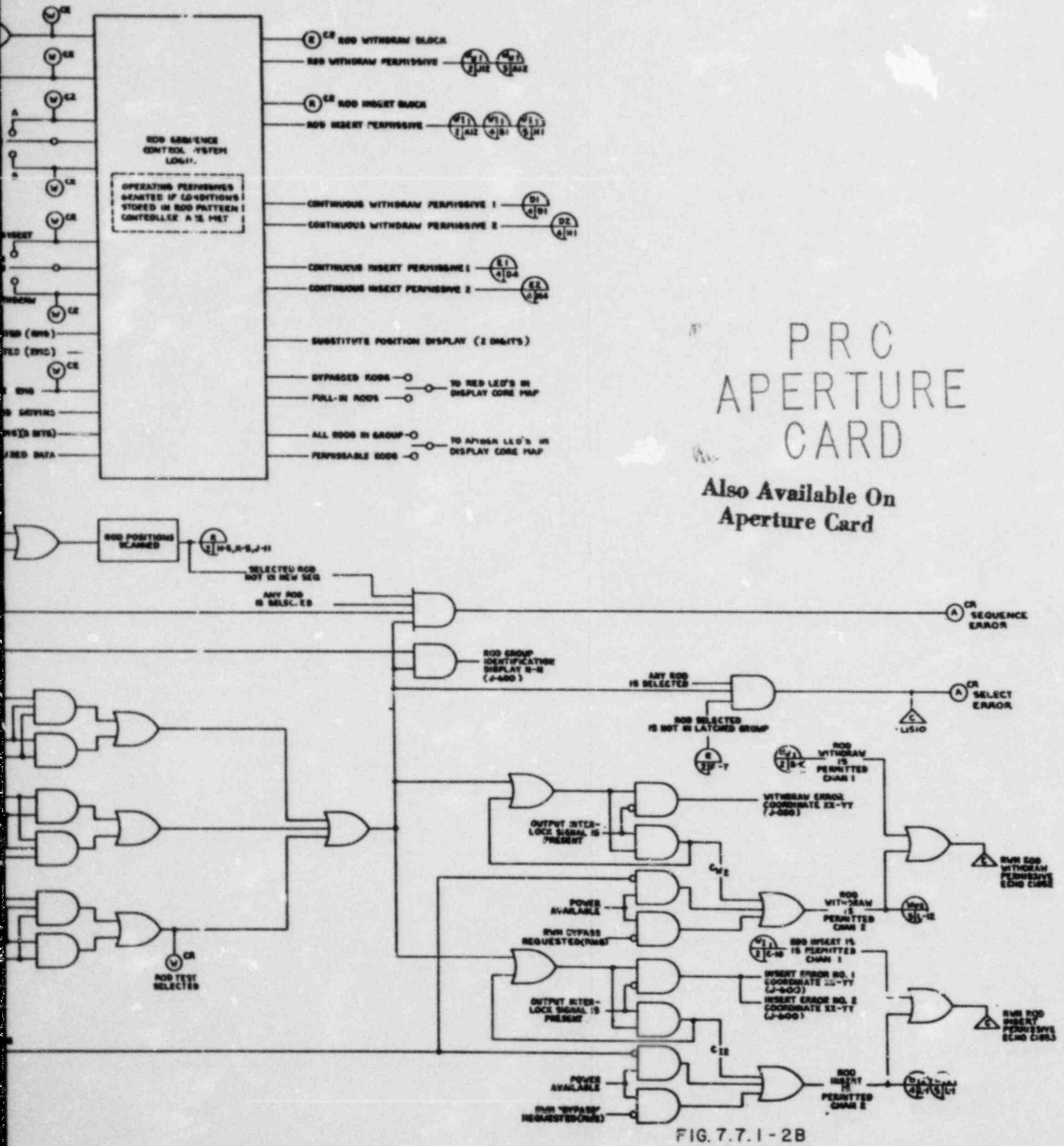


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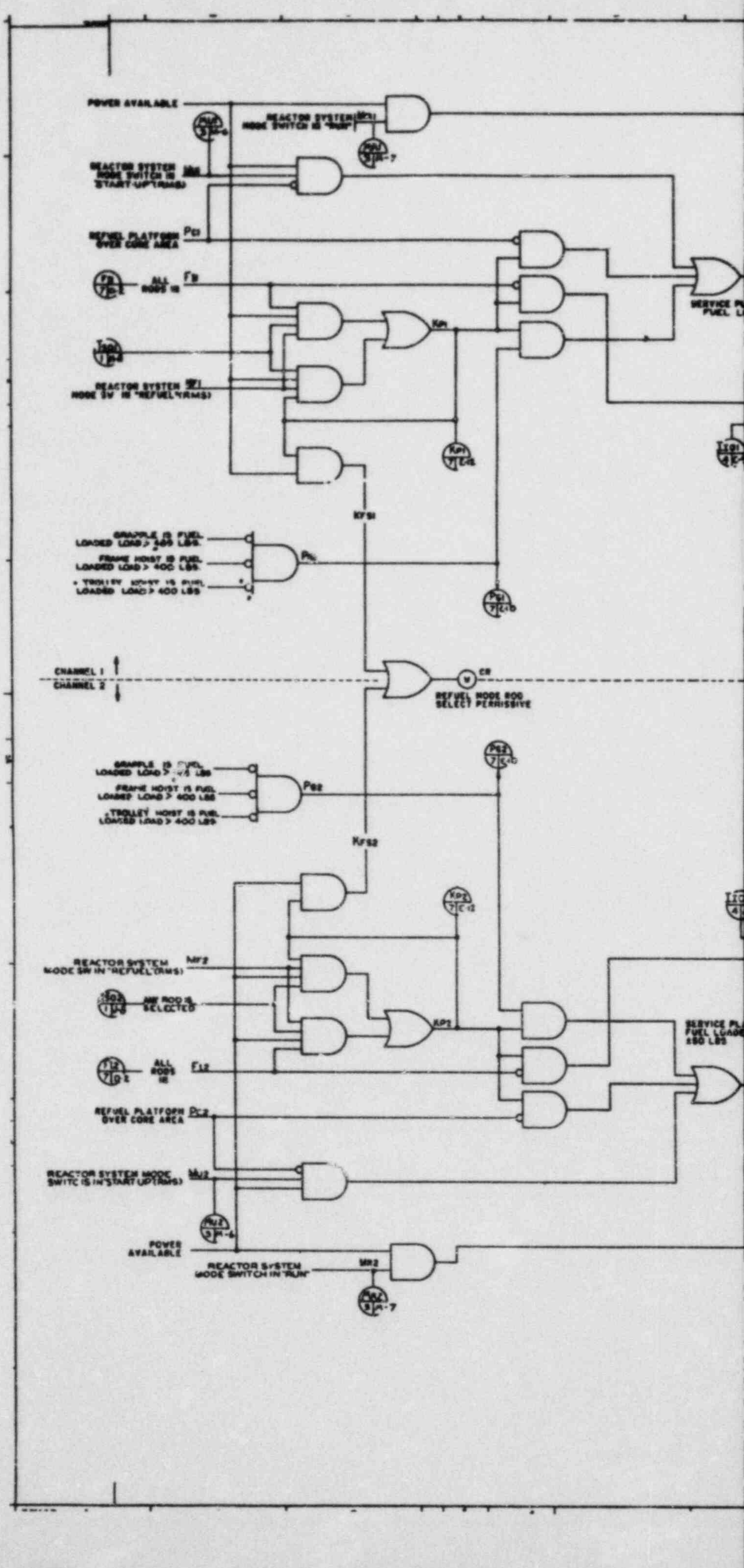
FIG. 7.7.1-2B
CONTROL ROD DRIVE HYD. SYS. FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 -18



POWER AVAILABLE
REACTOR SYSTEM MODE SWITCH IS "RUN"
REACTOR SYSTEM MODE SWITCH IS START-UP (RMS)
REFUEL PLATFORM OVER CORE AREA
ALL RODS IS
REACTOR SYSTEM MODE SW IN "REFUEL" (RMS)
GRAPPLE IS FUEL LOADED LOAD > 485 LBS
FRAME HOIST IS FUEL LOADED LOAD > 400 LBS
TROLLEY HOIST IS FUEL LOADED LOAD > 400 LBS
CHANNEL 1
CHANNEL 2
REFUEL MODE ROD SELECT PERMISSIVE
REACTOR SYSTEM MODE SW IN "REFUEL" (RMS)
ALL RODS IS
REFUEL PLATFORM OVER CORE AREA
REACTOR SYSTEM MODE SWITC IS IN START UP (RMS)
POWER AVAILABLE
REACTOR SYSTEM MODE SWITCH IN "RUN"

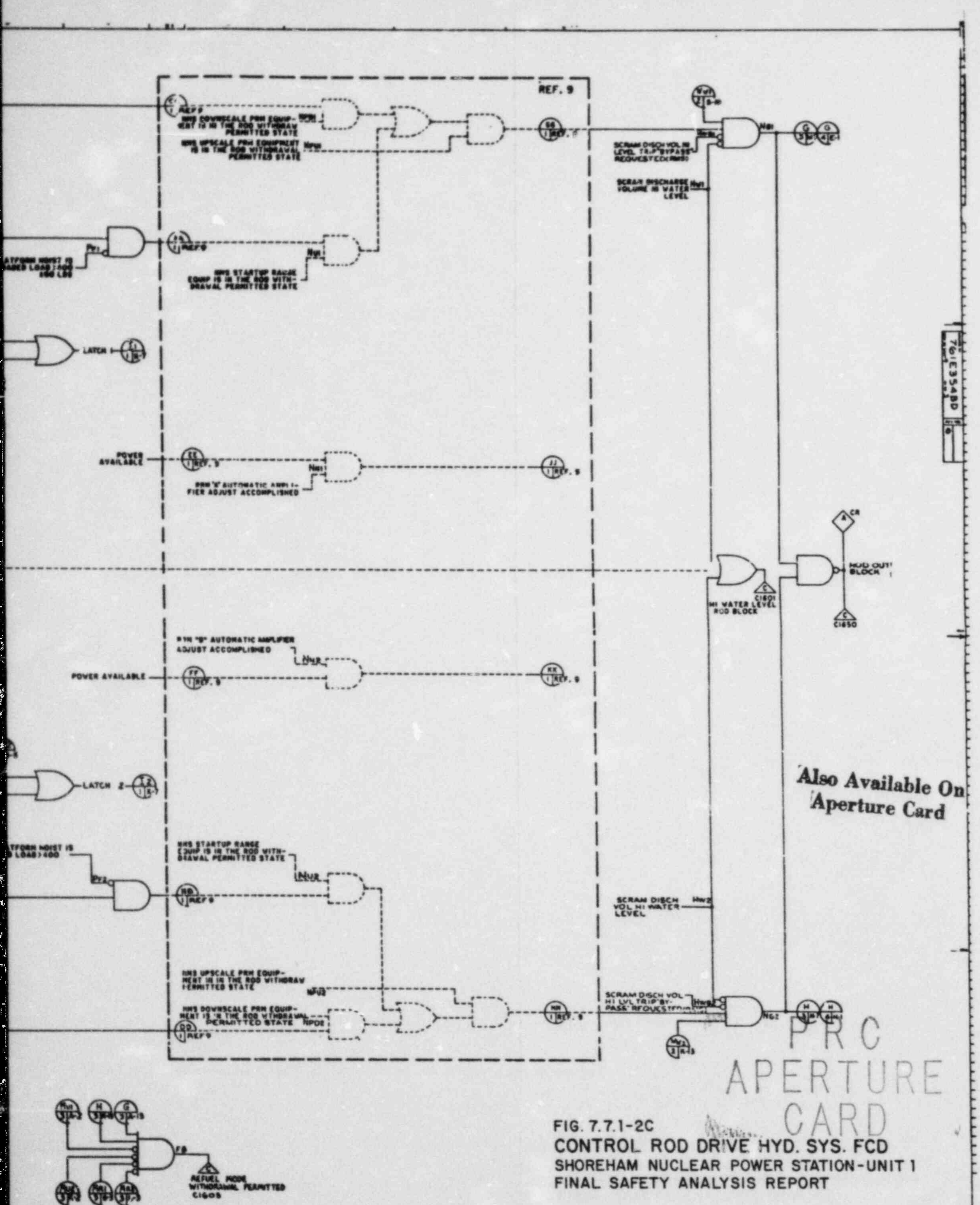
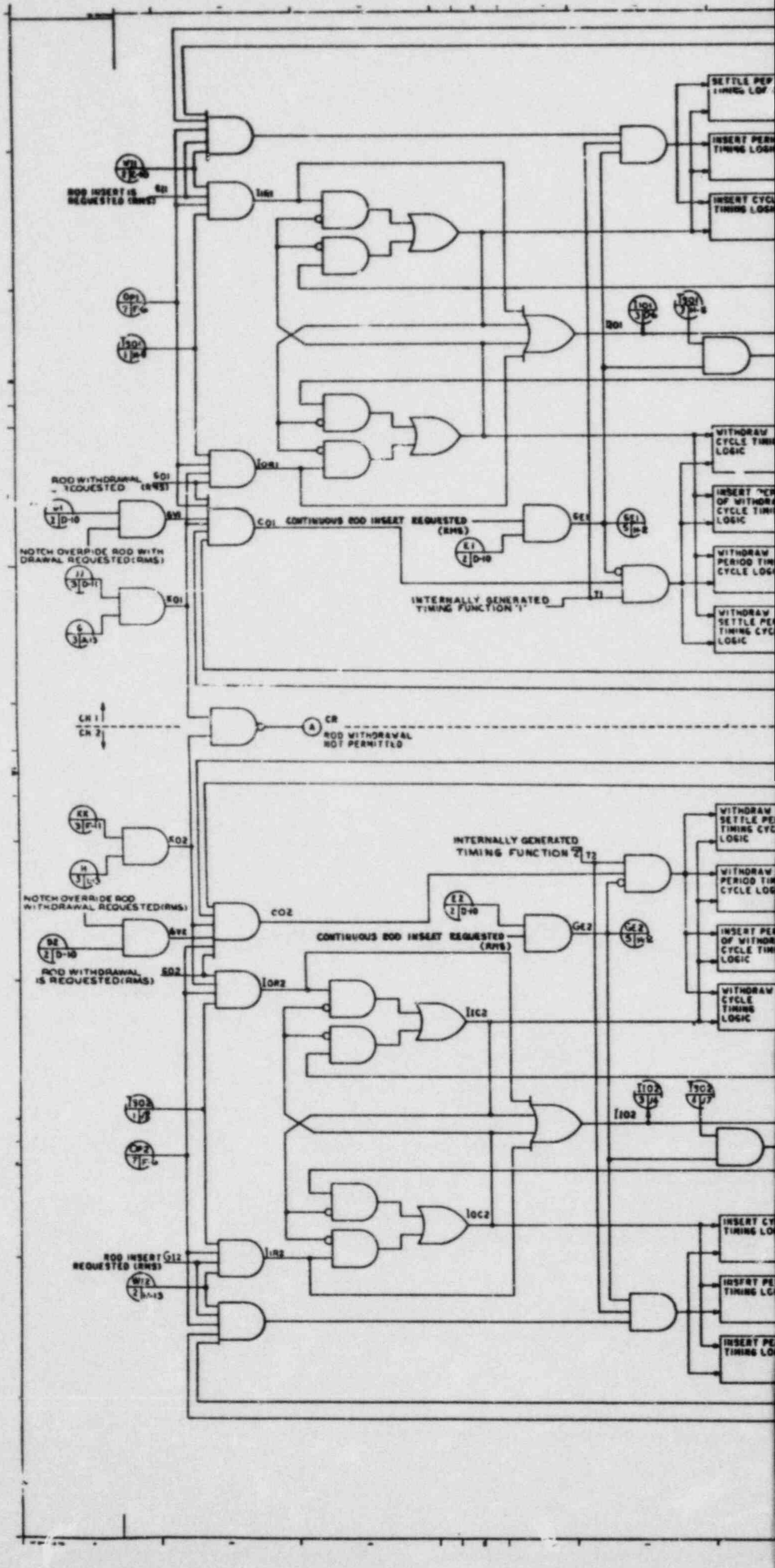


FIG. 7.7.1-2C
CONTROL ROD DRIVE HYD. SYS. FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT



ROD INSERT IS REQUESTED (RMS)
ROD WITHDRAWAL REQUESTED (RMS)
CONTINUOUS ROD INSERT REQUESTED (RMS)
NOTCH OVERRIDE ROD WITH DRAWAL REQUESTED (RMS)
INTERNALLY GENERATED TIMING FUNCTION '1'
CH 1
CH 2
CR
ROD WITHDRAWAL NOT PERMITTED
INTERNALLY GENERATED TIMING FUNCTION '2'
NOTCH OVERRIDE ROD WITHDRAWAL REQUESTED(RMS)
CONTINUOUS ROD INSERT REQUESTED (RMS)
ROD WITHDRAWAL IS REQUESTED(RMS)
ROD INSERT REQUESTED (RMS)

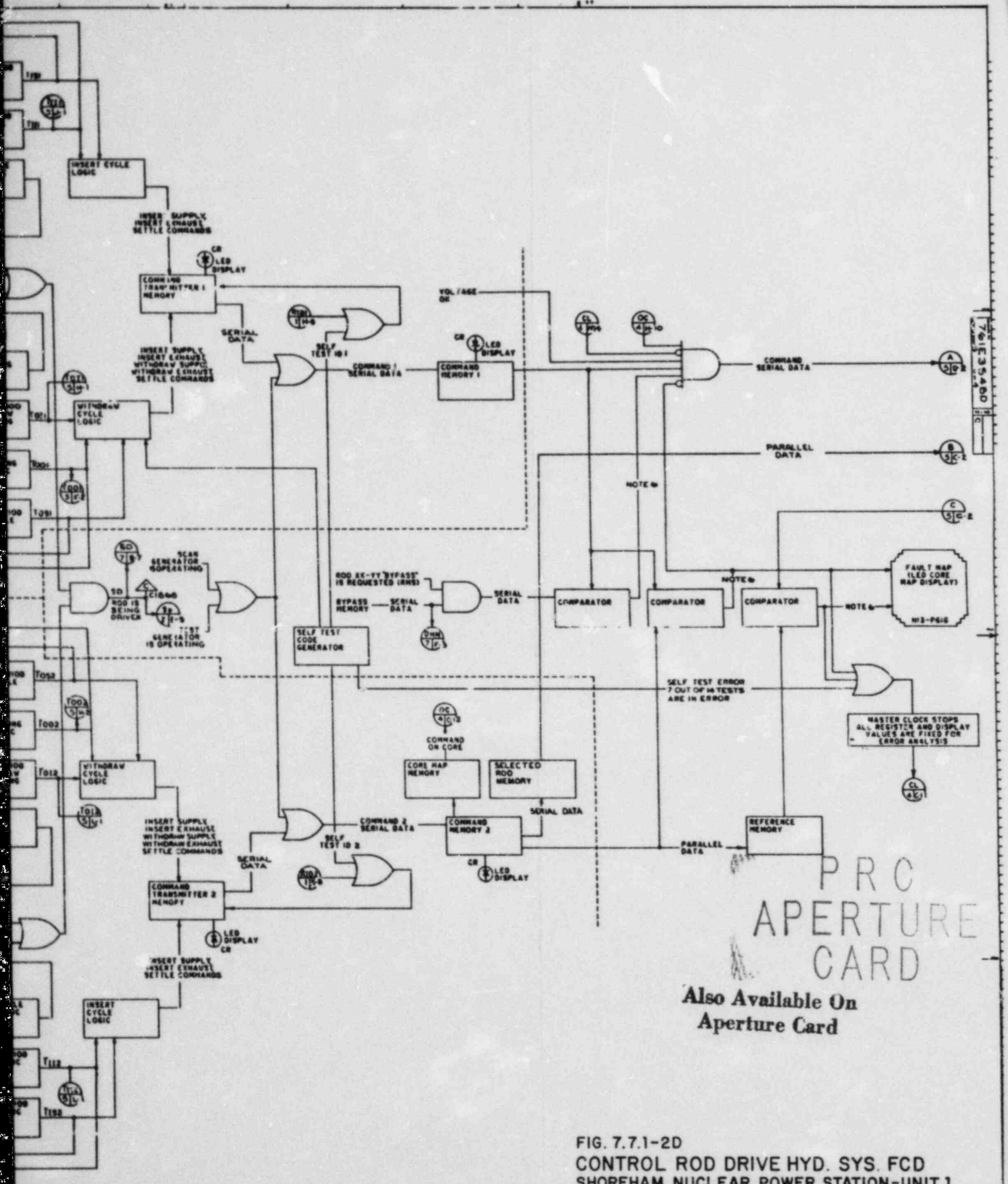


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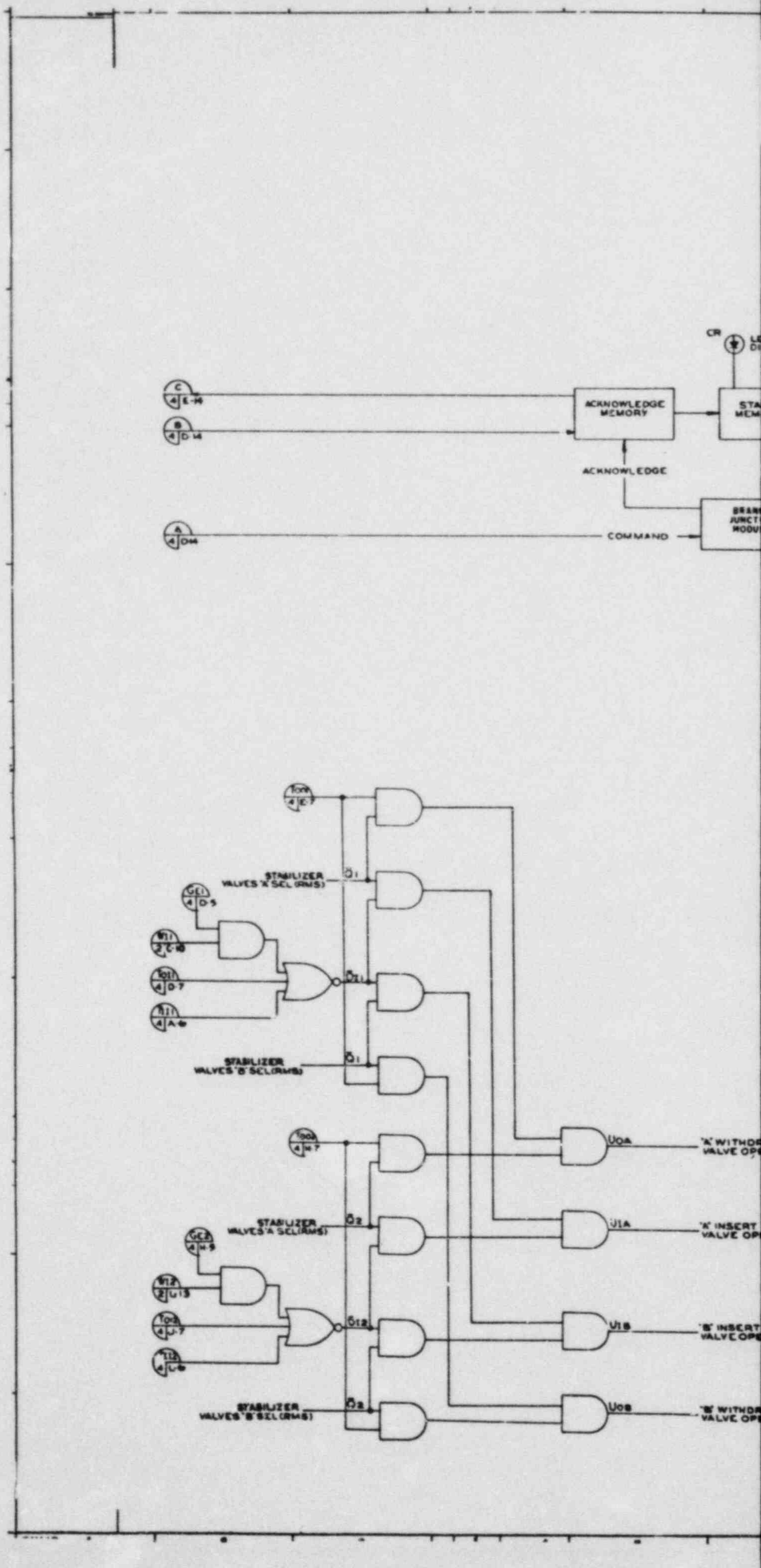
FIG. 7.7.1-2D
CONTROL ROD DRIVE HYD. SYS. FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 - 20



ACKNOWLEDGE MEMORY
ACKNOWLEDGE
COMMAND
STABILIZER VALVES 'A' SEL (RMS)
STABILIZER VALVES 'B' SEL (RMS)
'A' WITHDR
'A' INSERT
'B' INSERT
'B' WITHDR

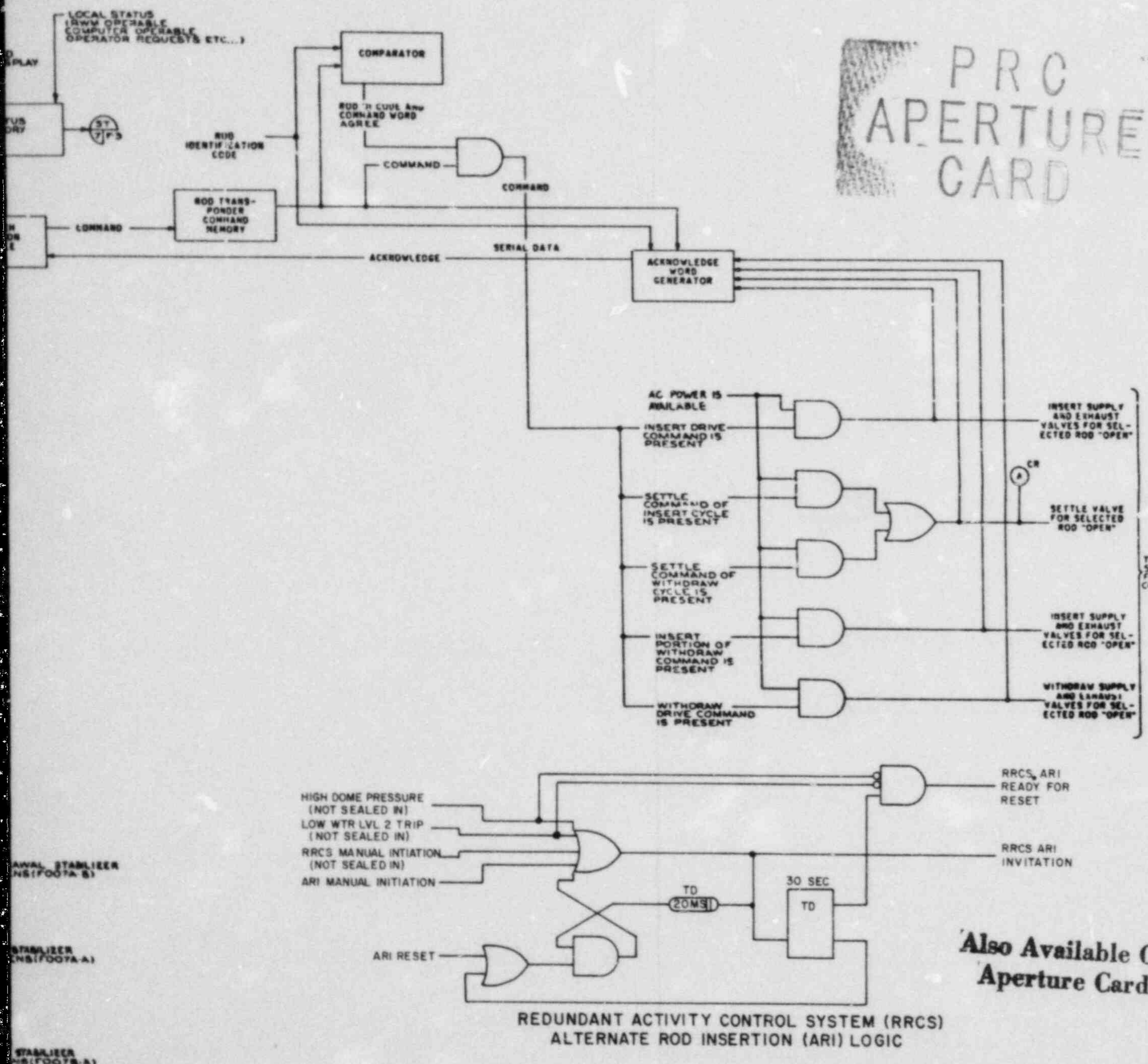


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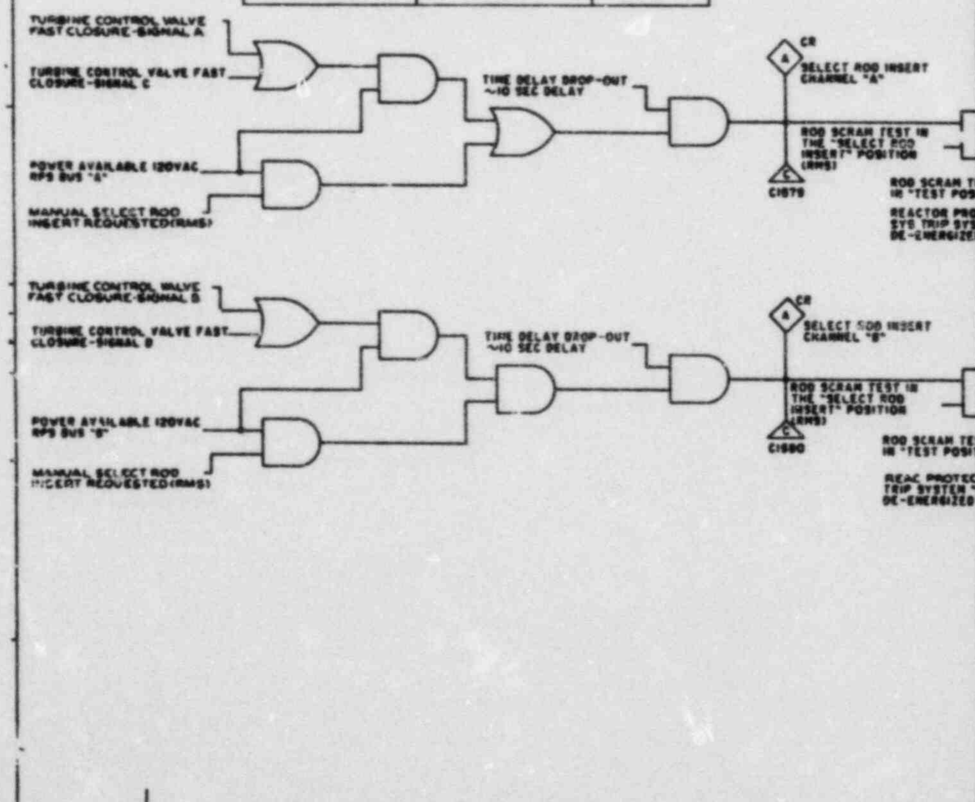
FIG. 7.7.1-2E
CONTROL ROD DRIVE HYD. SYS. FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 - 21

TABLE 4

FUNCTION	INITIATING DEVICE	TYPE
SCRAM DISCHARGE VOLUME NOT DRAINED	LEVEL SWITCH N013F, N013H	ANN.
DRIVE WATER FILTER HIGH DIFF PRESSURE	DIFF PRESS. IND SW N002	ANN.
CHARGING WATER HIGH PRESSURE	PRESS. IND SW N600	ANN.
DRIVE WATER PUMP "A" SUCTION LOW PRESS.	PRESS. SW N001A	ANN.
DRIVE WATER PUMP "B" SUCTION LOW PRESSURE	PRESS. SW N001B	ANN.
SCRAM VALVE PILOT AIR HEADER HIGH/LOW PRESS.	PRESS. SW N652	ANN.
CRD PUMP SUCTION FILTER HIGH DIFF PRESSURE	DIFF PRESS. IND SW N015	ANN.
STABILIZER VALVE SELECT SWITCH IN "A" POSITION	REMOTE MANUAL SW	IND LAMP W
STABILIZER VALVE SELECT SWITCH IN "B" POSITION	REMOTE MANUAL SW	IND LAMP W
VALVES AO F002A,B, F181, F182 & NOF005, ~~NOT FULLY CLOSED~~ NOT FULLY OPEN	LIMIT SWITCH ON VALVES	~~IND LAMP R~~ IND LAMP G
VALVES AO F010, F180 & AO F011, F181 ~~FULLY OPEN~~ FULLY CLOSED	LIMIT SWITCH ON VALVES	~~IND LAMP R~~ IND LAMP G
ACCUMULATOR LOW PRESSURE OR LEAK DETECTION (TYP FOR EACH ACCUMULATOR) (REF 1)(NOTE 2)	PRESSURE SW 130 OR LEAK DET SW 129	IND LAMP A (NOTE 4)
ANY ACCUMULATOR LOW PRESSURE OR ANY ACCUMULATOR LEAKAGE	PRESSURE SW 130 OR LEAK DET SW 129	ANN H/L (NOTE 4)
PUMPS C001A & B STOP START TRIP	AUX DEVICES ON PUMP BREAKER	IND LAMP GREEN RED WHITE



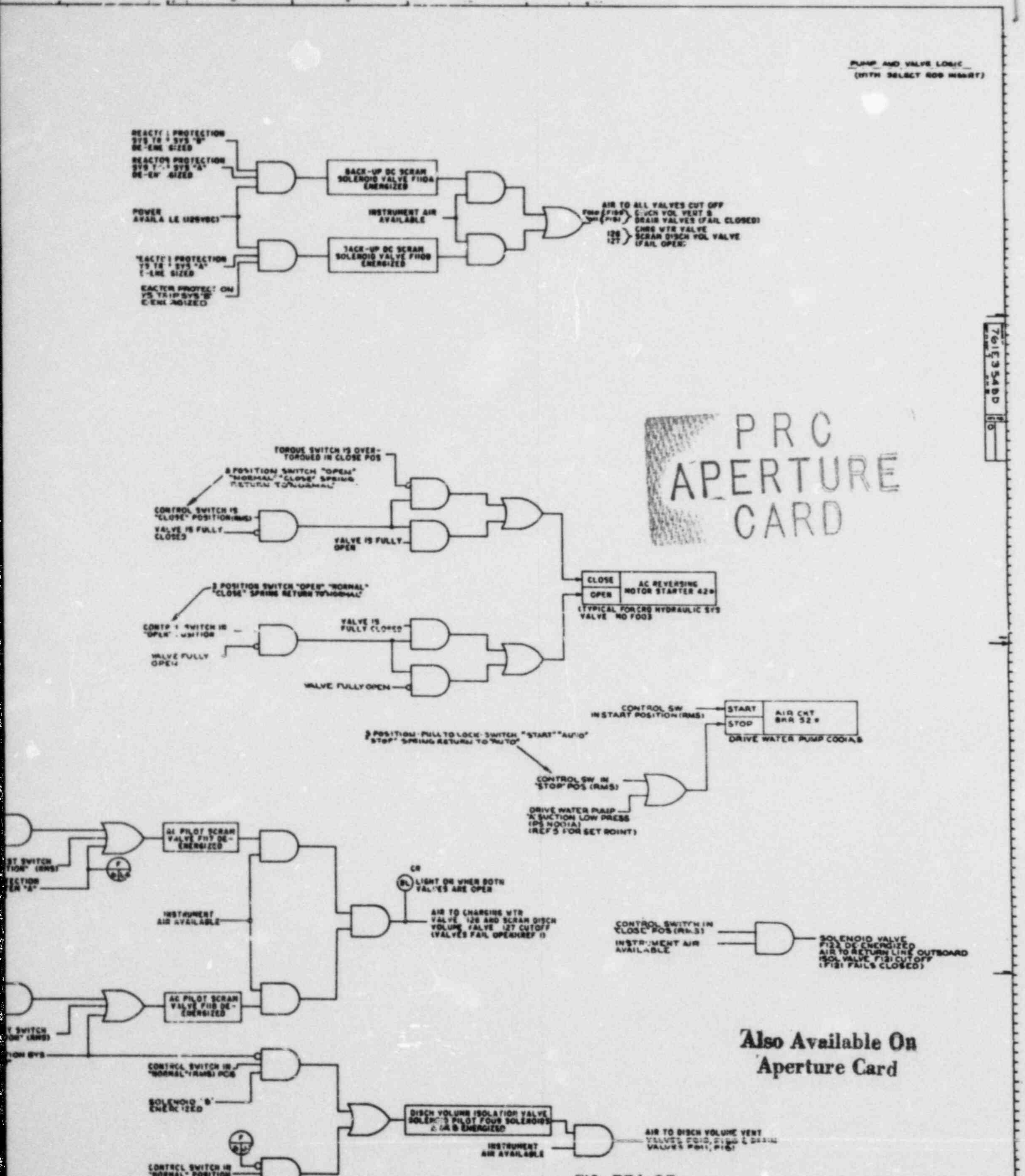


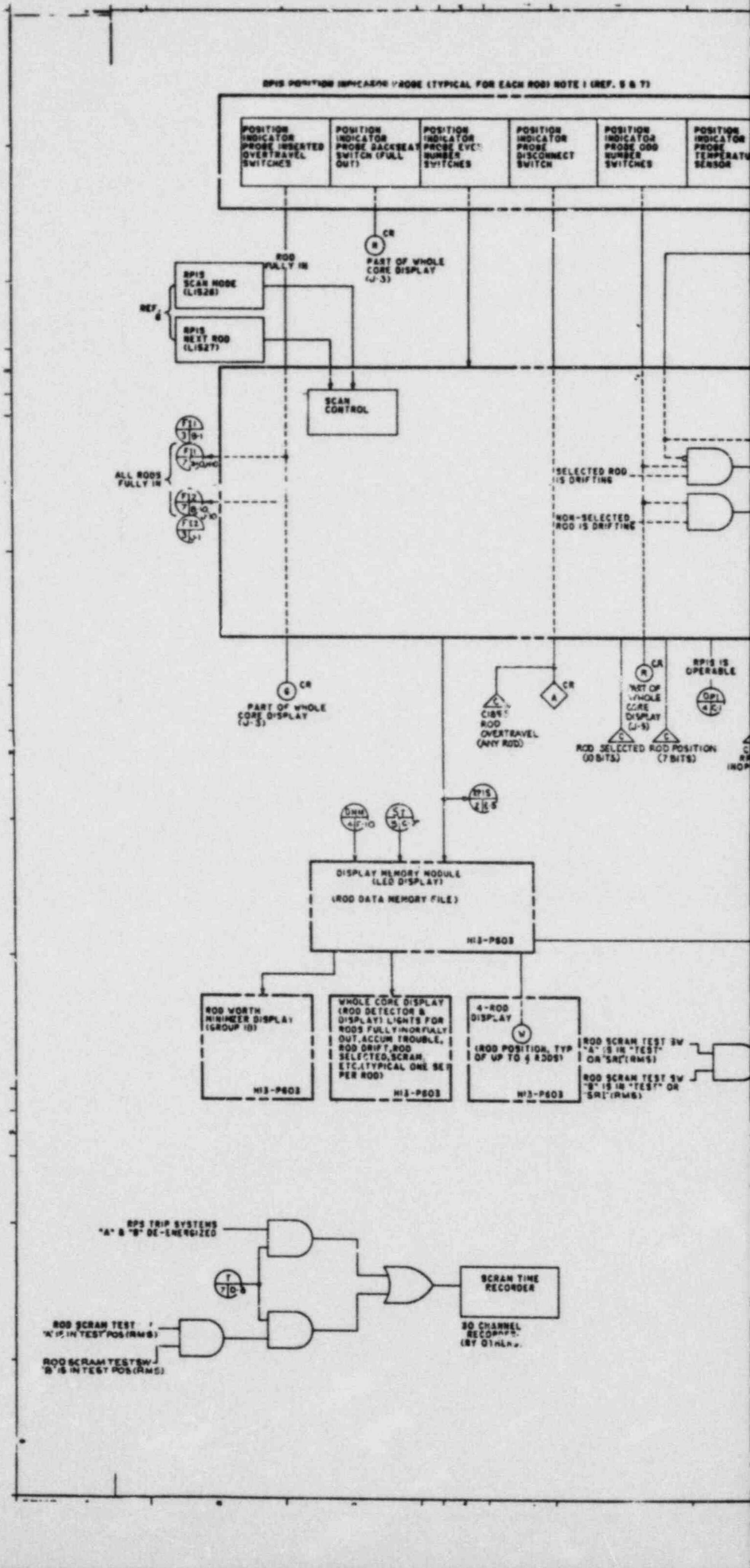
FIG. 7.7.1-2F

CONTROL ROD DRIVE HYD. SYS. FCD

SHOREHAM NUCLEAR POWER STATION-UNIT 1

FINAL SAFETY ANALYSIS REPORT

8307050091 - 22



POSITION INDICATOR PROBE INSERTED OVERTRAVEL SWITCHES
POSITION INDICATOR PROBE BACKSEAT SWITCH (FULL OUT)
POSITION INDICATOR PROBE DISCONNECT SWITCH
POSITION INDICATOR PROBE ODD NUMBER SWITCHES
ROD FULLY IN
PART OF WHOLE CORE DISPLAY (J-3)
RPIS SCAN MODE (L1526)
RPIS NEXT ROD (L1527)
SCAN CONTROL
ALL RODS FULLY IN
SELECTED ROD IS DRIFTING
NON-SELECTED ROD IS DRIFTING
RPIS IS OPERABLE
ROD OVERTRAVEL (ANY ROD)
ROD SELECTED (10 BITS)
ROD POSITION (7 BITS)
DISPLAY MEMORY MODULE (LED DISPLAY)
(ROD DATA MEMORY FILE)
H13-P603
ROD WORTH MINIMIZER DISPLAY (GROUP 10)
H13-P603
4-ROD DISPLAY
(ROD POSITION, TYP OF UP TO 4 RODS)
H13-P603
RPS TRIP SYSTEMS "A" & "B" DE-ENERGIZED
SCRAM TIME RECORDER

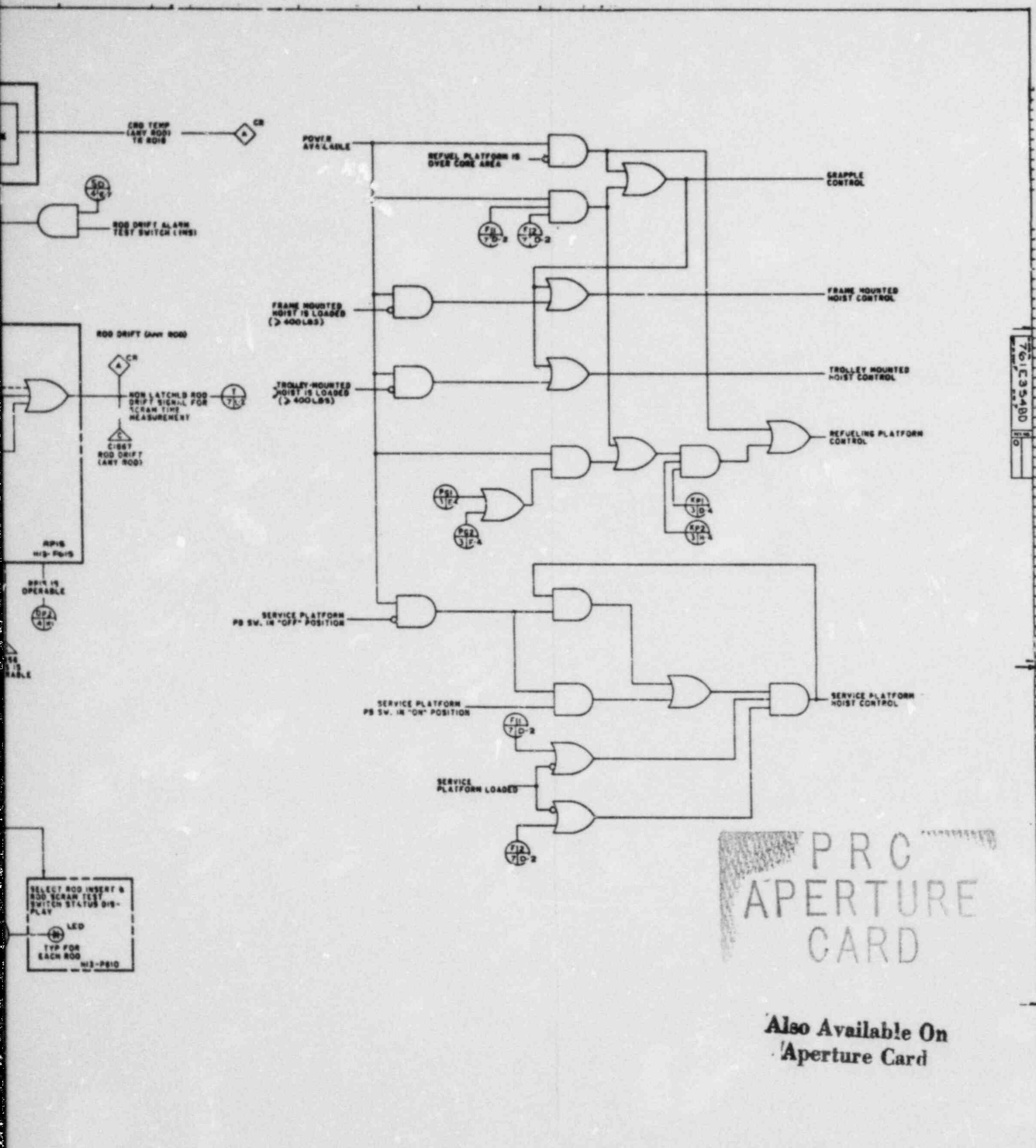
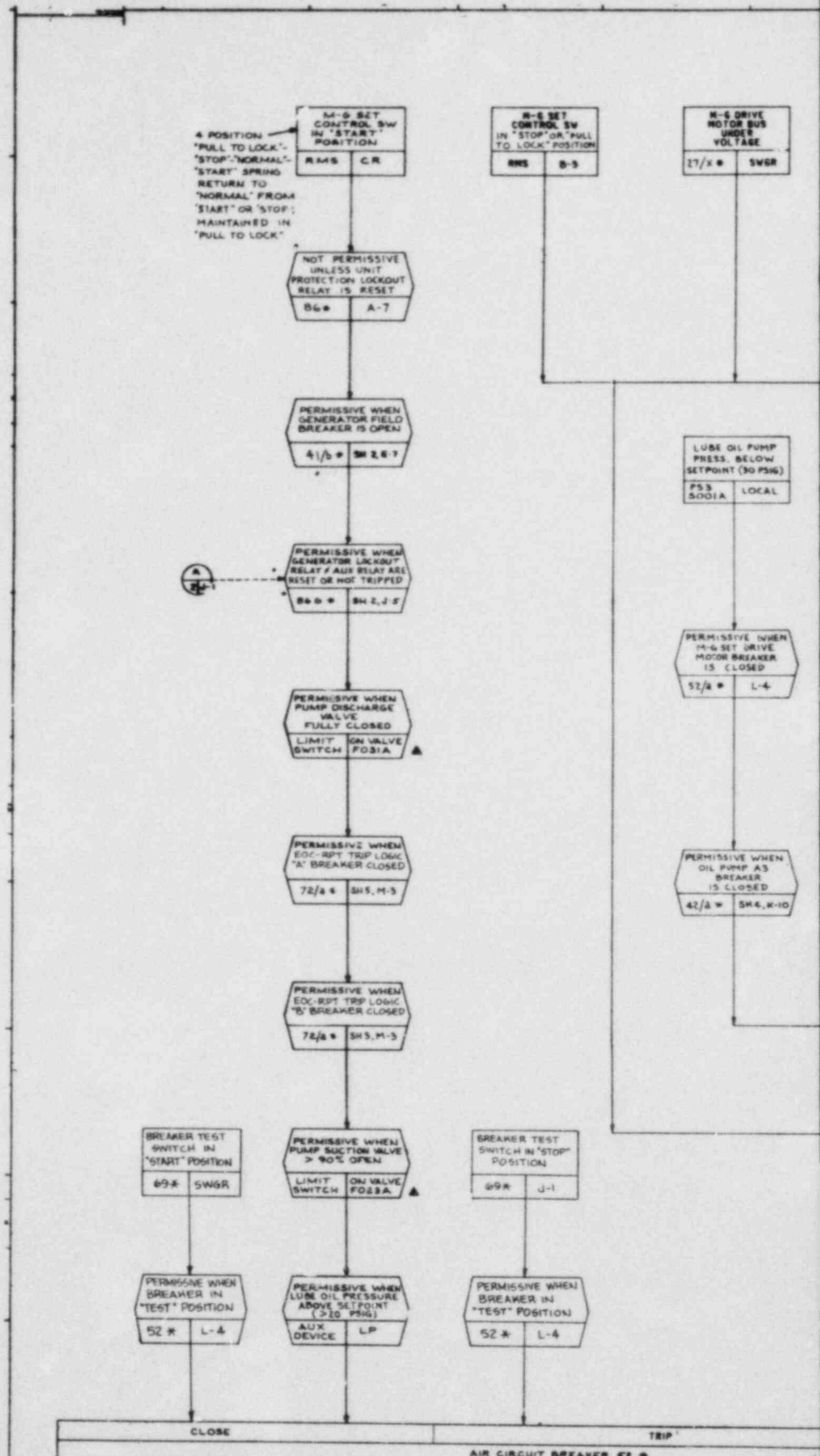


FIG. 7.7.1-2G
CONTROL ROD DRIVE HYD. SYS. FCD
SHOREHAM NUCLEAR POWER STATION-UNIT I
FINAL SAFETY ANALYSIS REPORT

8307050091 -23



M-G SET DRIVE MOTOR BREAKER
SEE NOTE 3

▲ SUFFIX LETTER 'A' IS REPLACED BY 'B' FOR SYSTEM B

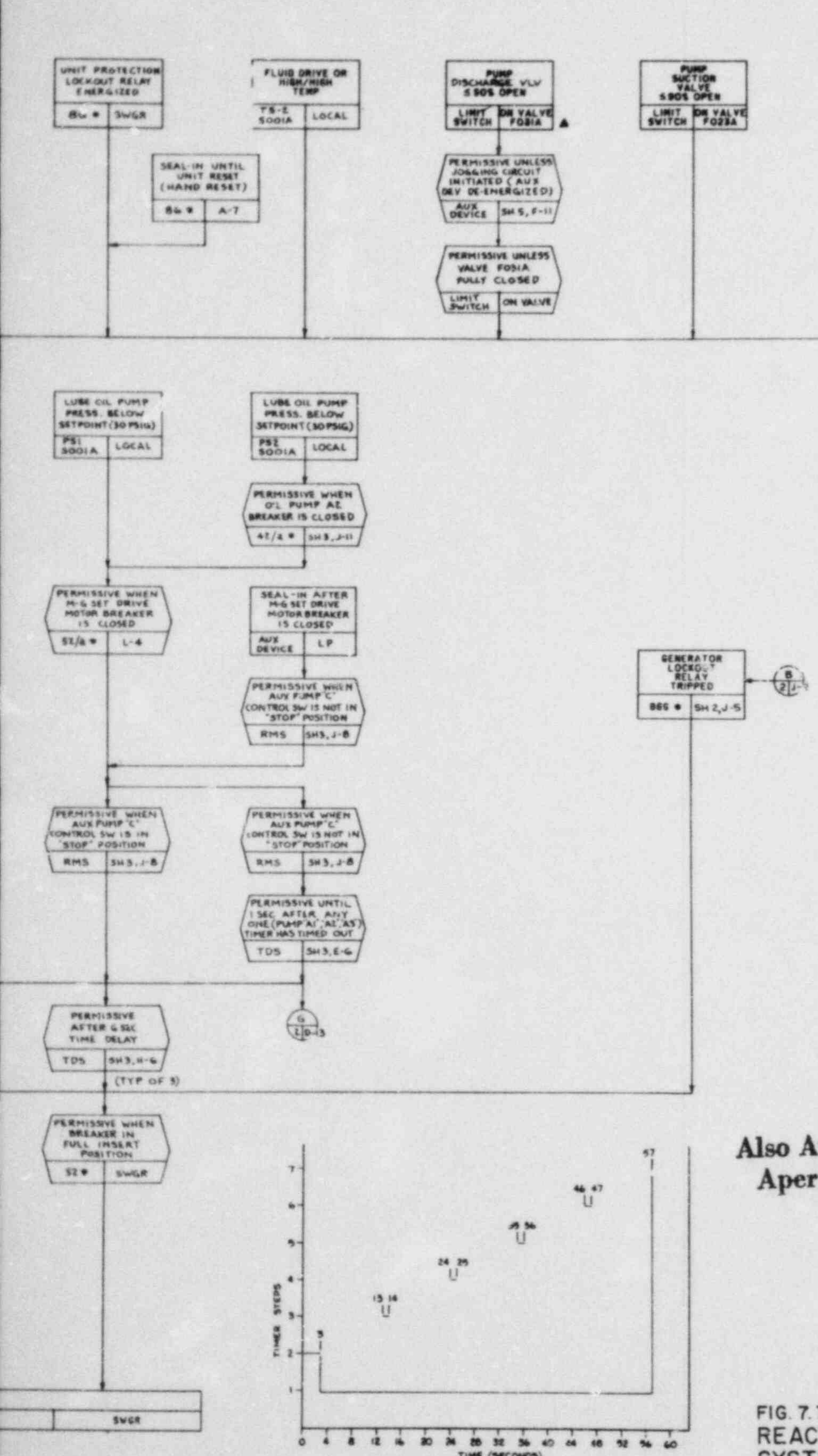


TABLE I
JOGGING SEQUENCE SCHEDULE

NOTES:

1. AUXILIARY RELAYS, SWITCHES, ETC. ARE NOT SHOWN EXCEPT WHERE NECESSARY TO CLARIFY THE FUNCTION
2. THIS DEVICE SENSES WHEN THE PUMP HAS STARTED AND FUNCTIONS AFTER 2 SEC TIME DELAY TO OPEN THE "SEAL-IN" TRANSFER VOLTAGE SUPPLY TO THE VOLTAGE REGULATOR AND INHIBIT THE INCOMPLETE SEQUENCE
3. FUNCTION IS SHOWN FOR RECIRCULATION SYSTEM "A" (EXCEPT AS NOTED) AND IS TYPICAL FOR RECIRCULATION SYSTEM "B"
4. DELETED
5. THE TRIP SIGNALS SHALL CAUSE THE DEVICE TO MAINTAIN SLIP AT A VALUE EQUAL TO THE SLIP WHICH IS PRESENT AT THE TIME THE TRIP SIGNAL WAS RECEIVED. LOSS OF MODE OF CONTROL (ELECTRICAL, HYDRAULIC, ETC.) SHALL ALSO CAUSE SLIP TO BE MAINTAINED AT THE SAME VALUE
6. VALVE HAS NO SEAL-IN FOR OPENING CIRCUIT; THEREFORE CONTROL SWITCH MUST BE HELD IN THE "OPEN" POSITION FOR THE VALVE TO REACH FULLY OPEN POSITION

LEGEND

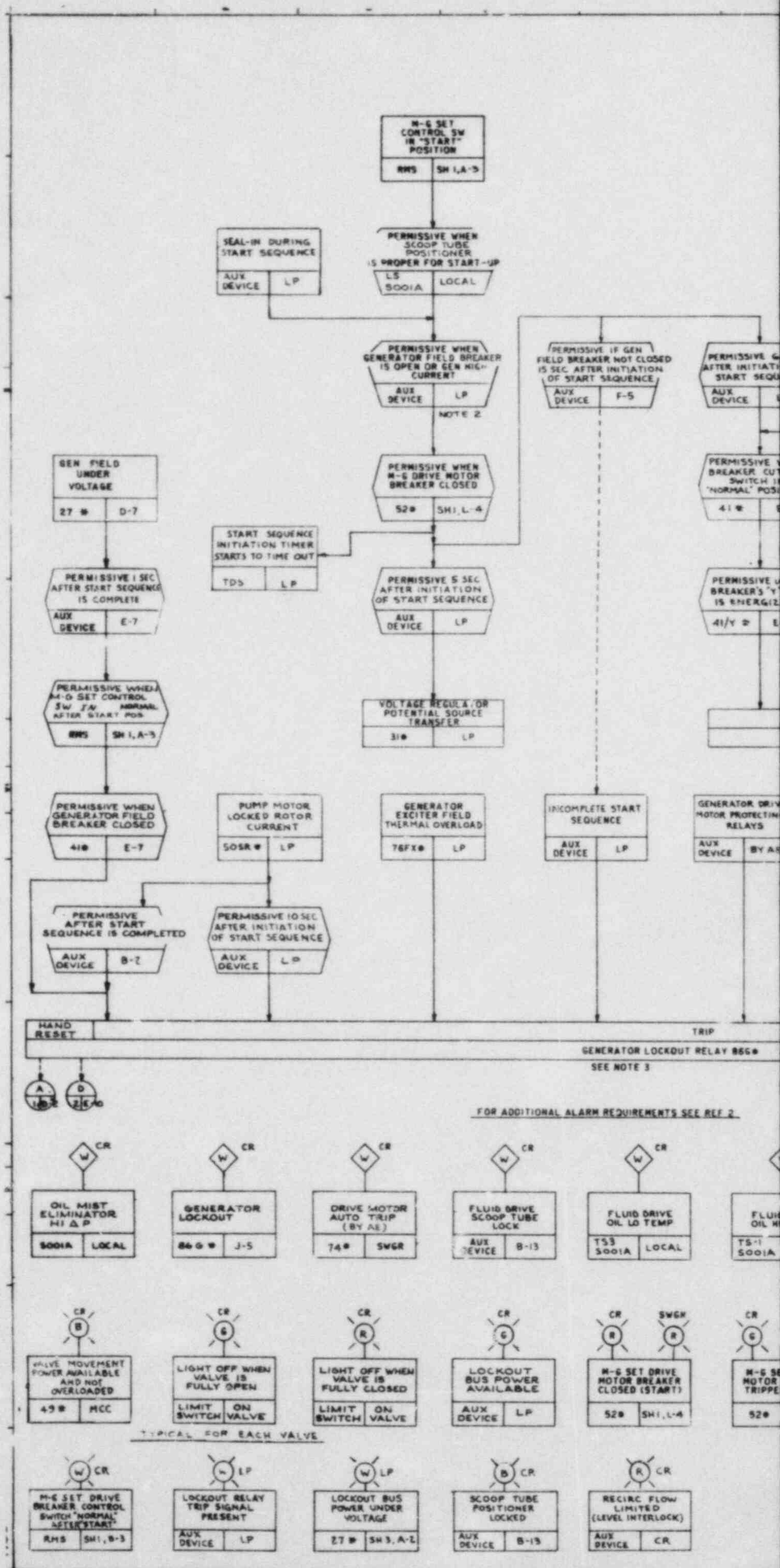
SWGR - M/G SET DRIVE MOTOR HIGH VOLTAGE SWITCHGEAR
* - SWITCHGEAR DEVICE FUNCTION NUMBER ANSI SPEC C37.2

Also Available On Aperture Card

REFERENCE DOCUMENTS

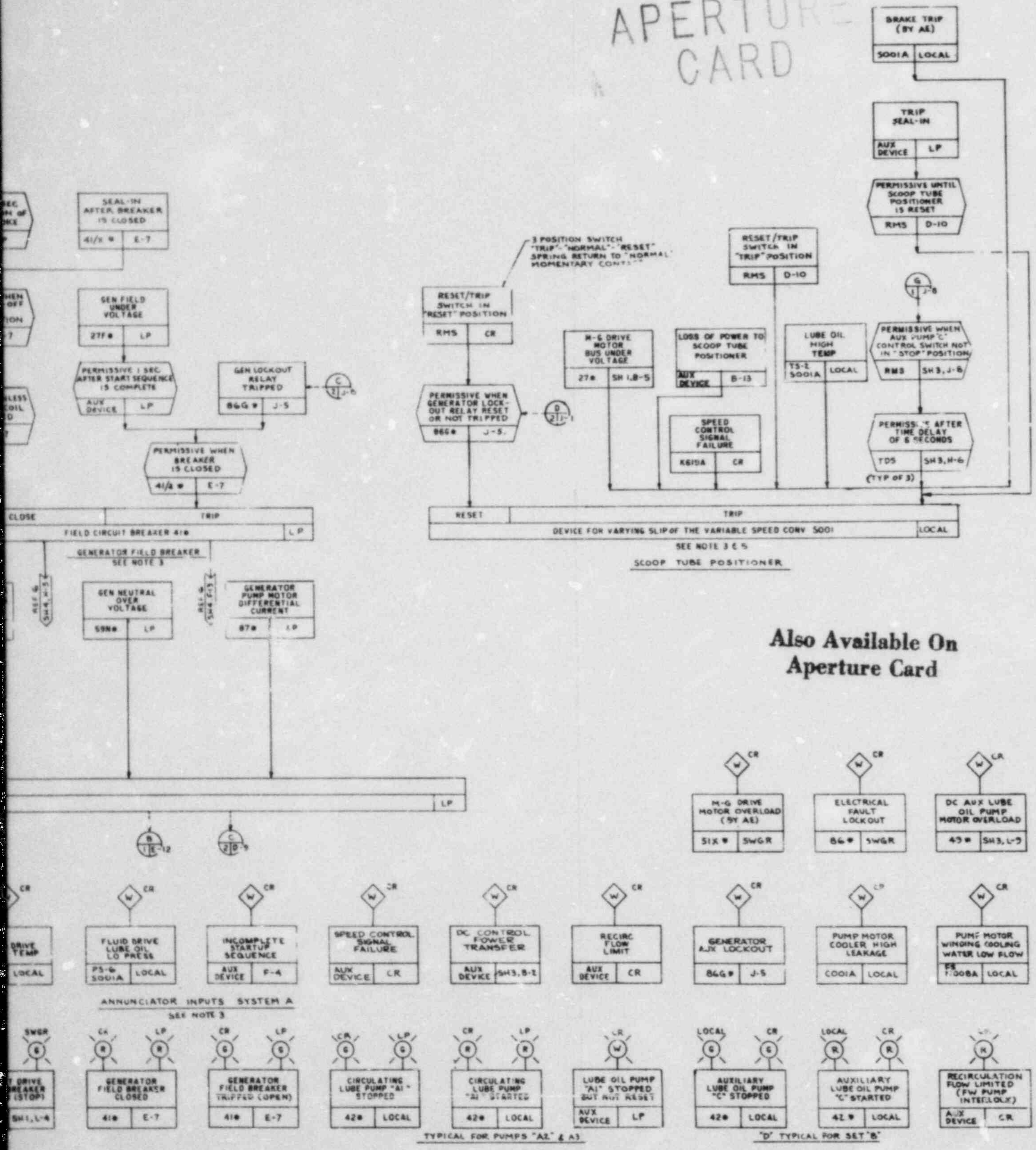
1. REACTOR RECIRC SYS DESN SPEC..........
2. REACTOR RECIRC SYS P&ID..............
3. NUCLEAR BOILER SYS P&ID..............
4. RESIDUAL HEAT REMOVAL SYS FCD..........
5. LOGIC SYMBOLS..........................
6. NUCLEAR BOILER SYS FCD...............
7. RPS SYSTEM IED........................
8. ANALOG TRIP SYS ELEM DIAG..........

FIG. 7.7.1-5A
REACTOR RECIRCULATION SYSTEM FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT



M-G SET CONTROL SW IN "START" POSITION
RMS
SH 1, A-3
SEAL-IN DURING START SEQUENCE
AUX DEVICE
LP
PERMISSIVE WHEN SCOOP TUBE POSITIONER IS PROPER FOR START-UP
LS 5001A
LOCAL
PERMISSIVE WHEN GENERATOR FIELD BREAKER IS OPEN OR GEN HIGH CURRENT
AUX DEVICE
LP
NOTE 2
PERMISSIVE IF GEN FIELD BREAKER NOT CLOSED 15 SEC AFTER INITIATION OF START SEQUENCE
AUX DEVICE
F-5
GEN FIELD UNDER VOLTAGE
27 #
D-7
PERMISSIVE WHEN M-G DRIVE MOTOR BREAKER CLOSED
52#
SH 1, L-4
START SEQUENCE INITIATION TIMER STARTS TO TIME OUT
TDS
LP
PERMISSIVE 1 SEC AFTER START SEQUENCE IS COMPLETE
AUX DEVICE
E-7
PERMISSIVE 5 SEC AFTER INITIATION OF START SEQUENCE
AUX DEVICE
LP
PERMISSIVE WHEN M-G SET CONTROL SW IN NORMAL AFTER START POS
RMS
SH 1, A-3
VOLTAGE REGULATOR POTENTIAL SOURCE TRANSFER
31#
LP
PERMISSIVE WHEN GENERATOR FIELD BREAKER CLOSED
41#
E-7
PUMP MOTOR LOCKED ROTOR CURRENT
50SR #
LP
GENERATOR EXCITER FIELD THERMAL OVERLOAD
76FX#
LP
INCOMPLETE START SEQUENCE
AUX DEVICE
LP
PERMISSIVE AFTER START SEQUENCE IS COMPLETED
AUX DEVICE
B-2
PERMISSIVE 10 SEC AFTER INITIATION OF START SEQUENCE
AUX DEVICE
LP
HAND RESET
TRIP
GENERATOR LOCKOUT RELAY 86G#
SEE NOTE 3
FOR ADDITIONAL ALARM REQUIREMENTS SEE REF 2
W
CR
OIL MIST ELIMINATOR HI ΔP
5001A
LOCAL
GENERATOR LOCKOUT
86 G #
J-5
DRIVE MOTOR AUTO TRIP (BY AE)
74#
SWGR
FLUID DRIVE SCOOP TUBE LOCK
AUX DEVICE
B-13
FLUID DRIVE OIL LO TEMP
TS3 5001A
LOCAL
B
G
R
VALVE MOVEMENT POWER AVAILABLE AND NOT OVERLOADED
49#
MCC
LIGHT OFF WHEN VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
LIGHT OFF WHEN VALVE IS FULLY CLOSED
LIMIT SWITCH
ON VALVE
LOCKOUT BUS POWER AVAILABLE
AUX DEVICE
LP
SWGR
M-G SET DRIVE MOTOR BREAKER CLOSED (START)
52#
SH 1, L-4
TYPICAL FOR EACH VALVE
M-G SET DRIVE BREAKER CONTROL SWITCH "NORMAL" AFTER "START"
RMS
SH 1, B-3
LOCKOUT RELAY TRIP SIGNAL PRESENT
AUX DEVICE
LP
LOCKOUT BUS POWER UNDER VOLTAGE
27 #
SH 3, A-2
SCOOP TUBE POSITIONER LOCKED
AUX DEVICE
B-13
RECIRC FLOW LIMITED (LEVEL INTERLOCK)
AUX DEVICE
CR

PRC APERTURE CARD



Also Available On Aperture Card

INDICATION LIGHT INPUTS SYSTEM A
SEE NOTE 3

FIG. 7.7.1-5B
REACTOR RECIRCULATING SYSTEM FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

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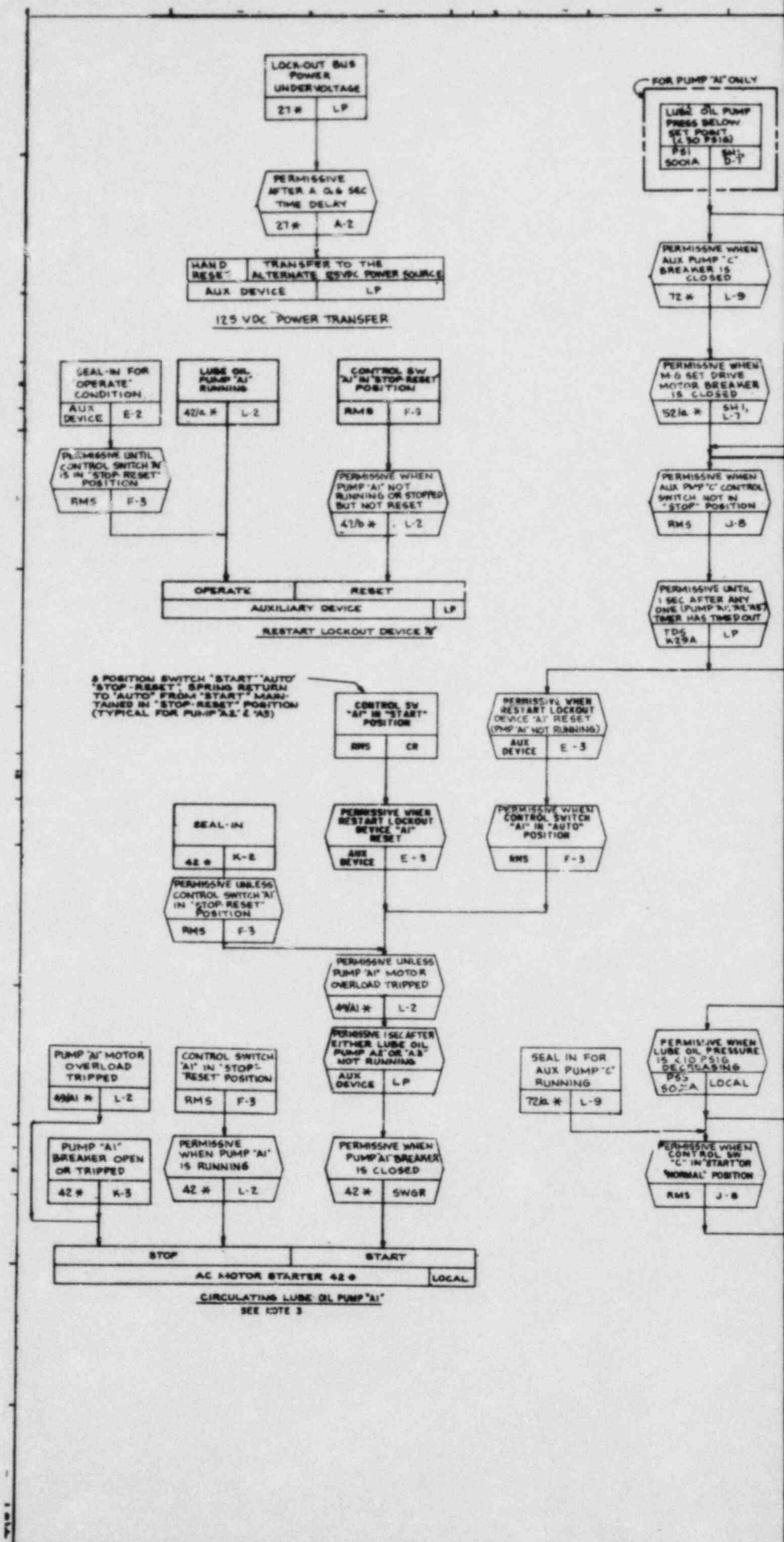
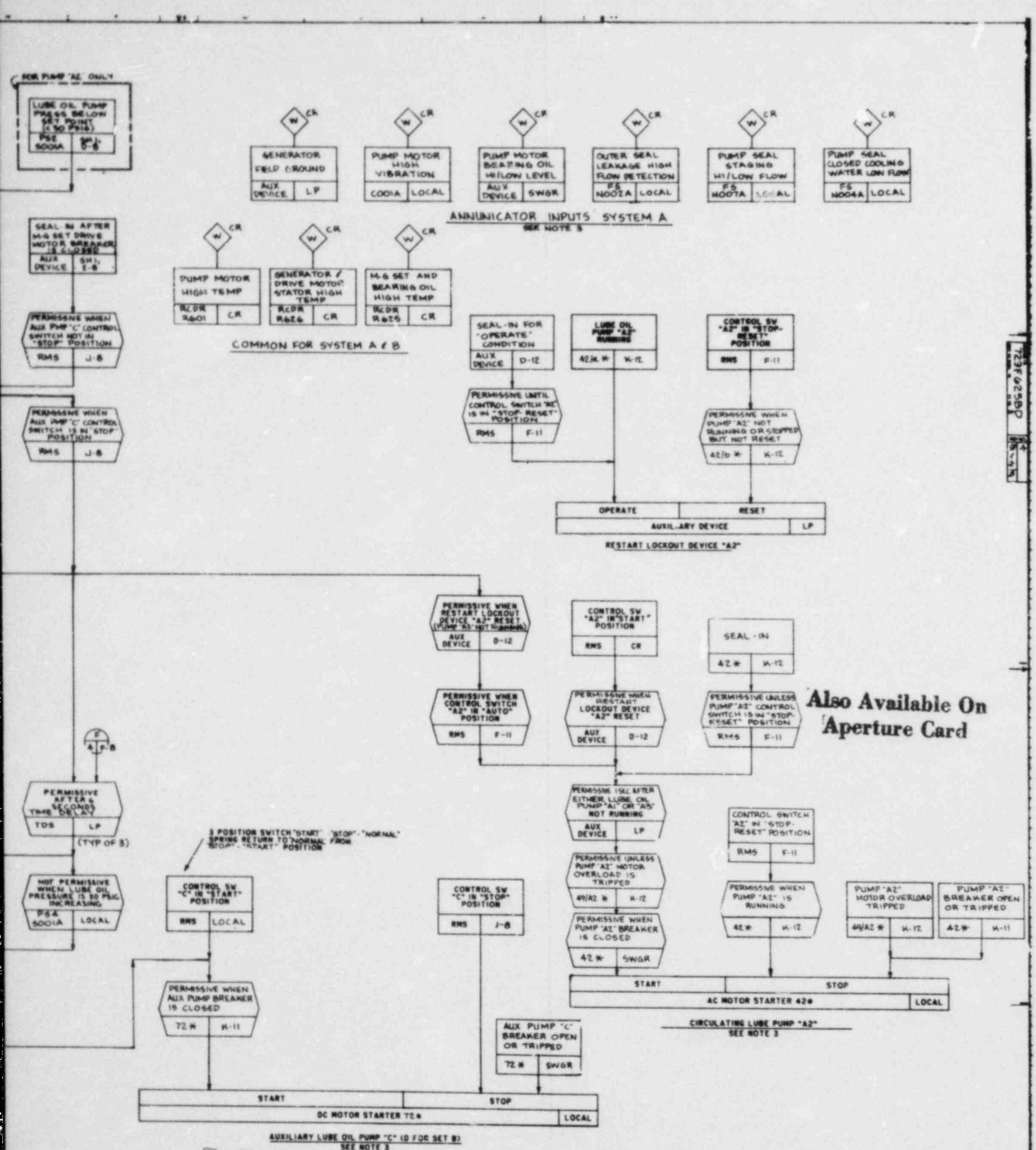

LOCK-OUT BUS POWER UNDERVOLTAGE
27 * | LP
PERMISSIVE AFTER A 0.6 SEC TIME DELAY
27 * | A-2
HAND RESET | TRANSFER TO THE ALTERNATE 125VDC POWER SOURCE
AUX DEVICE | LP
125 VDC POWER TRANSFER
SEAL-IN FOR "OPERATE" CONDITION
AUX DEVICE | E-2
LUBE OIL PUMP "A1" RUNNING
42/a * | L-2
CONTROL SW "A1" IN "STOP-RESET" POSITION
RMS | F-3
PERMISSIVE UNTIL CONTROL SWITCH "A" IS IN "STOP-RESET" POSITION
RMS | F-3
PERMISSIVE WHEN PUMP "A1" NOT RUNNING OR STOPPED BUT NOT RESET
42/b * | L-2
OPERATE | RESET
AUXILIARY DEVICE | LP
RESTART LOCKOUT DEVICE "A"
3 POSITION SWITCH "START" "AUTO" "STOP-RESET", SPRING RETURN TO "AUTO" FROM "START" MAINTAINED IN "STOP-RESET" POSITION (TYPICAL FOR PUMP "A2" & "A3")
CONTROL SW "A1" IN "START" POSITION
RMS | CR
PERMISSIVE WHEN RESTART LOCKOUT DEVICE "A1" RESET (PMP "A" NOT RUNNING)
AUX DEVICE | E-3
SEAL-IN
42 * | K-2
PERMISSIVE WHEN RESTART LOCKOUT DEVICE "A1" RESET
AUX DEVICE | E-3
PERMISSIVE WHEN CONTROL SWITCH "A1" IN "AUTO" POSITION
RMS | F-3
PERMISSIVE UNLESS CONTROL SWITCH "A1" IN "STOP RESET" POSITION
RMS | F-3
PERMISSIVE UNLESS PUMP "A1" MOTOR OVERLOAD TRIPPED
49/A1 * | L-2
PERMISSIVE 1 SEC AFTER EITHER LUBE OIL PUMP "A2" OR "A3" NOT RUNNING
AUX DEVICE | LP
PUMP "A1" MOTOR OVERLOAD TRIPPED
49/A1 * | L-2
CONTROL SWITCH "A1" IN "STOP-RESET" POSITION
RMS | F-3
PUMP "A1" BREAKER OPEN OR TRIPPED
42 * | K-3
PERMISSIVE WHEN PUMP "A1" IS RUNNING
42 * | L-2
PERMISSIVE WHEN PUMP "A1" BREAKER IS CLOSED
42 * | SWGR
STOP | START
AC MOTOR STARTER 42 * | LOCAL
CIRCULATING LUBE OIL PUMP "A1"
SEE NOTE 3
FOR PUMP "A1" ONLY
LUBE OIL PUMP PRESS BELOW SET POINT (≤ 30 PSIG)
PSI 5001A | SH1, D-7
PERMISSIVE WHEN AUX PUMP "C" BREAKER IS CLOSED
72 * | L-9
PERMISSIVE WHEN M-G SET DRIVE MOTOR BREAKER IS CLOSED
52/a * | SH1, L-7
PERMISSIVE WHEN AUX PMP "C" CONTROL SWITCH NOT IN "STOP" POSITION
RMS | J-8
PERMISSIVE UNTIL 1 SEC AFTER ANY ONE (PUMP "A1","A2","A3") TIMER HAS TIMED OUT
TDS K29A | LP
PERMISSIVE WHEN LUBE OIL PRESSURE IS ≤10 PSIG DECREASING
PSS 5001A | LOCAL
SEAL IN FOR AUX PUMP "C" RUNNING
72/a * | L-9
PERMISSIVE WHEN CONTROL SW "C" IN "START" OR "NORMAL" POSITION
RMS | J-8

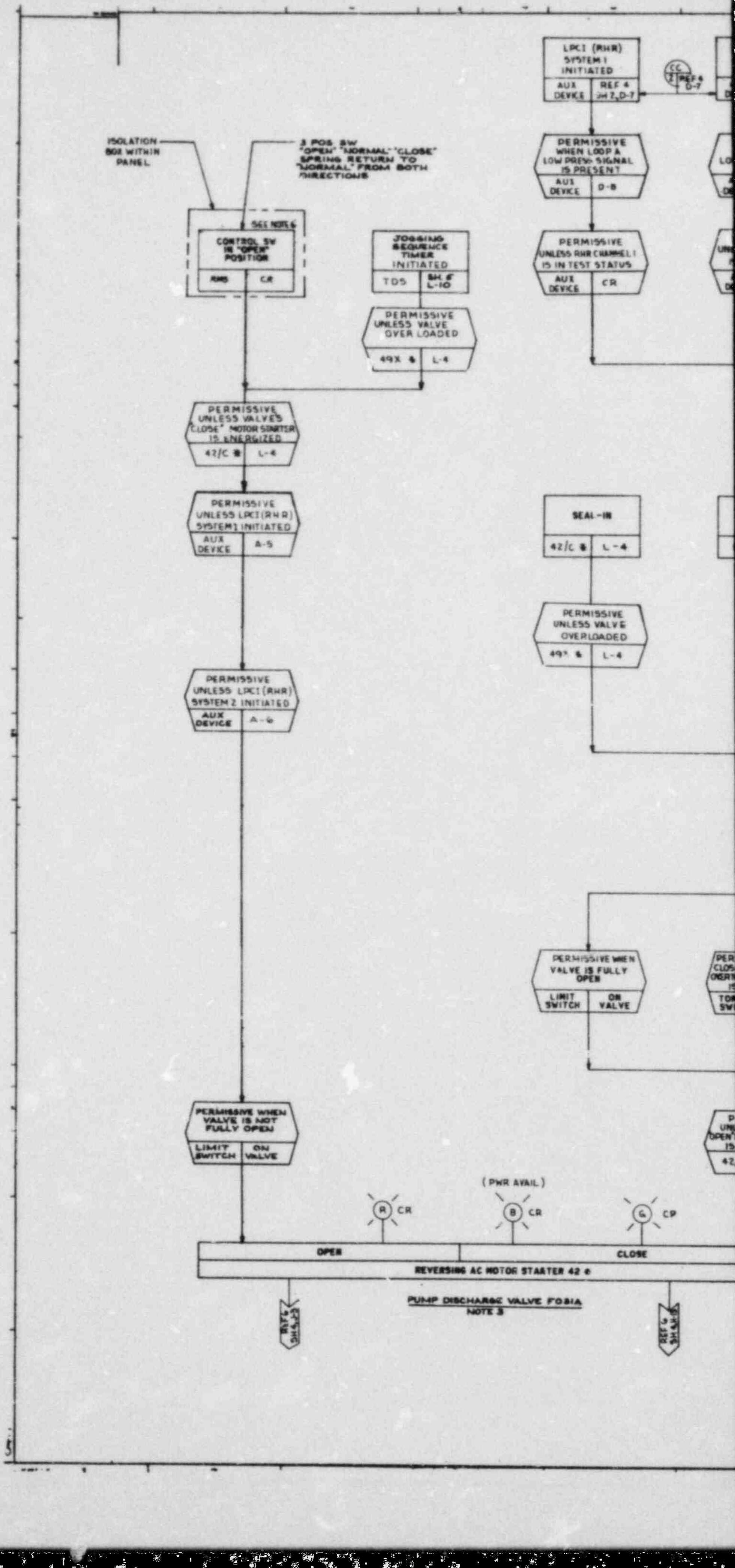


Also Available On Aperture Card

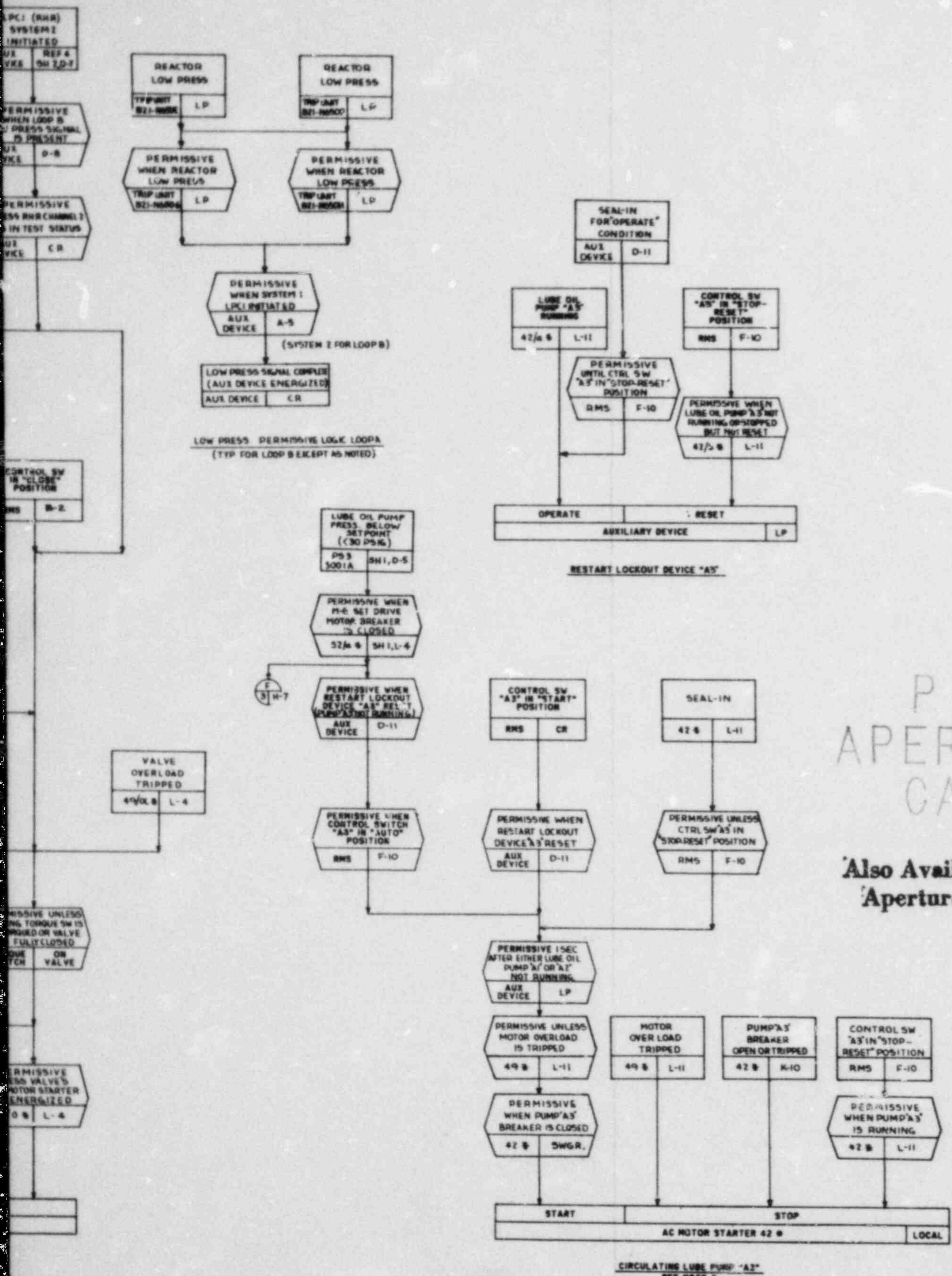
PRC APERTURE CARD

FIG. 7.7.1-5C
REACTOR RECIRCULATION SYS. FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

8307050091 - 26



LPCI (RHR) SYSTEM 1 INITIATED
AUX DEVICE
REF 4 SH 7, D-7
PERMISSIVE WHEN LOOP A LOW PRESS SIGNAL IS PRESENT
AUX DEVICE
D-8
PERMISSIVE UNLESS RHR CHANNEL 1 IS IN TEST STATUS
AUX DEVICE
CR
ISOLATION BOX WITHIN PANEL
3 POS SW "OPEN" "NORMAL" "CLOSE" SPRING RETURN TO "NORMAL" FROM BOTH DIRECTIONS
SEE NOTE 6
CONTROL SW IN "OPEN" POSITION
RMS
CR
JOGGING SEQUENCE TIMER INITIATED
TDS
SH 5 L-10
PERMISSIVE UNLESS VALVE OVER LOADED
49X
L-4
PERMISSIVE UNLESS VALVES "CLOSE" MOTOR STARTER IS ENERGIZED
42/C
L-4
PERMISSIVE UNLESS LPCI (RHR) SYSTEM 1 INITIATED
AUX DEVICE
A-5
SEAL-IN
42/C
L-4
PERMISSIVE UNLESS VALVE OVERLOADED
49X
L-4
PERMISSIVE UNLESS LPCI (RHR) SYSTEM 2 INITIATED
AUX DEVICE
A-6
PERMISSIVE WHEN VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE WHEN VALVE IS NOT FULLY OPEN
LIMIT SWITCH
ON VALVE
(PWR AVAIL)
R CR
B CR
G CR
OPEN
CLOSE
REVERSING AC MOTOR STARTER 42
PUMP DISCHARGE VALVE F051A
NOTE 3



PRC APERTURE CARD

Also Available On Aperture Card

FIG. 7.7.1-5D
REACTOR RECIRCULATION SYSTEM FCD
SHOREHAM NUCLEAR POWER STATION-UNIT I
FINAL SAFETY ANALYSIS REPORT

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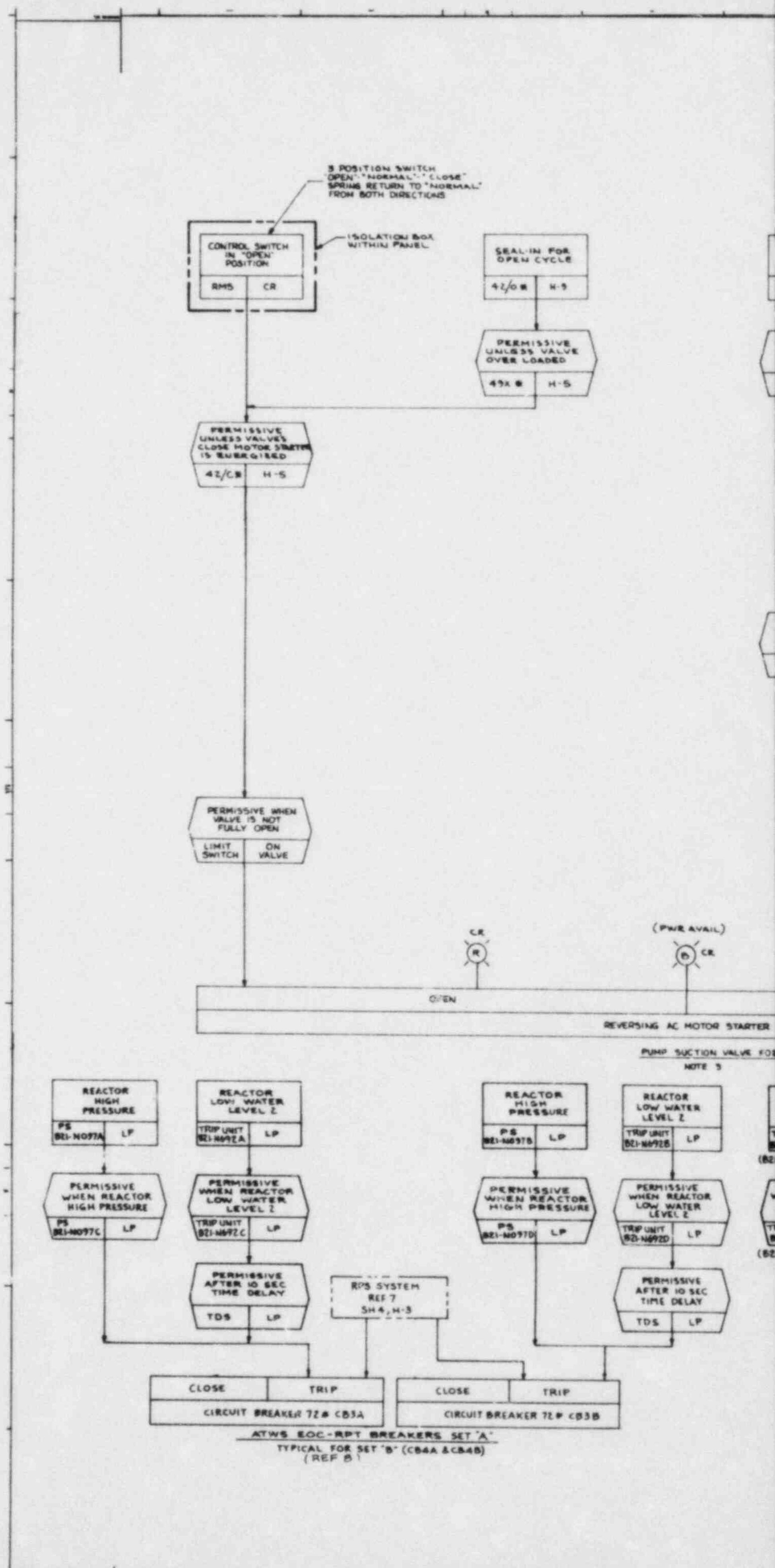
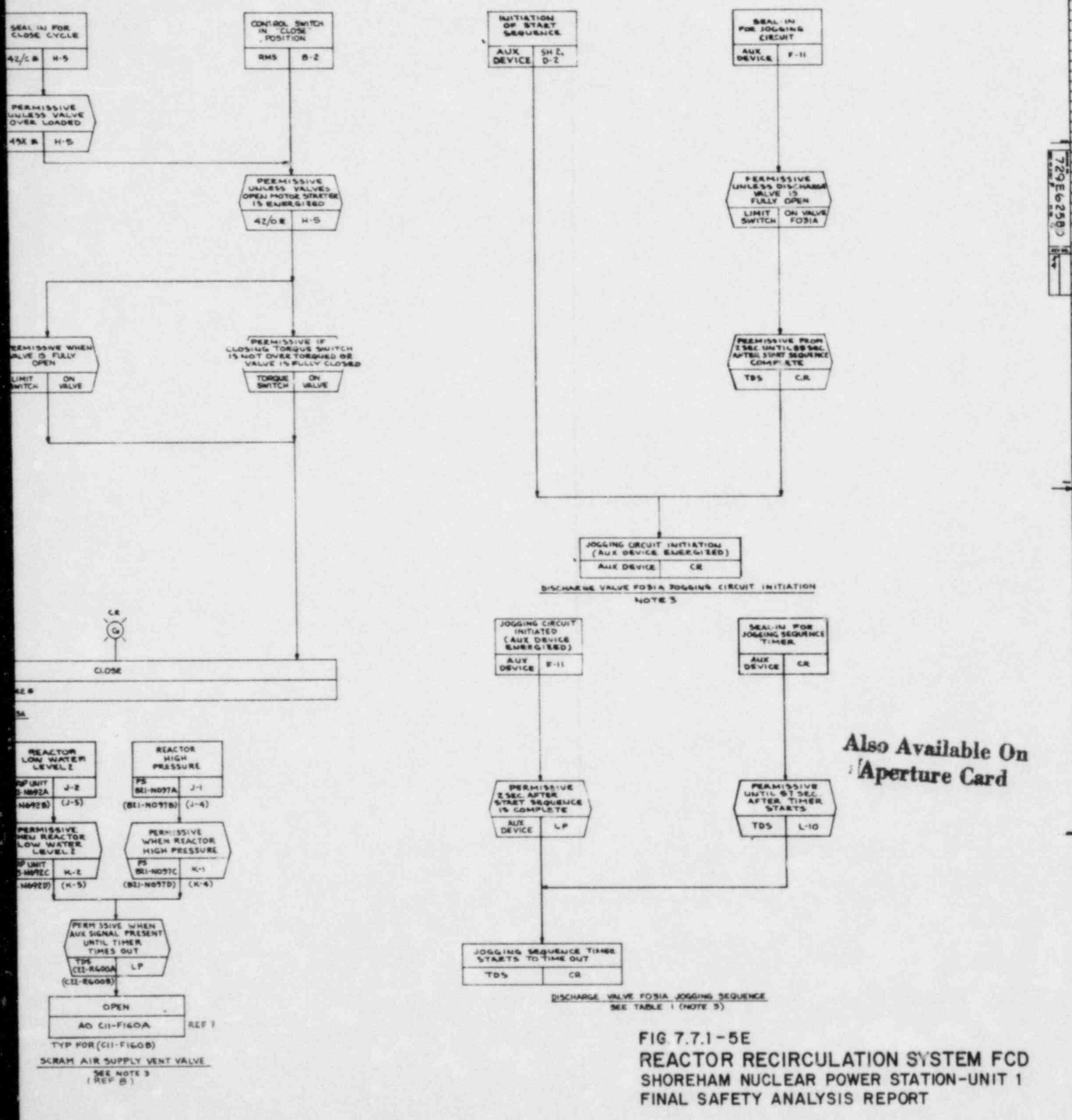


3 POSITION SWITCH
"OPEN"-"NORMAL"-"CLOSE"
SPRING RETURN TO "NORMAL"
FROM BOTH DIRECTIONS
ISOLATION BOX WITHIN PANEL
CONTROL SWITCH IN "OPEN" POSITION
RMS
CR
SEAL-IN FOR OPEN CYCLE
42/O ✱
H-5
PERMISSIVE UNLESS VALVE OVER LOADED
49X ✱
H-5
PERMISSIVE UNLESS VALVES CLOSE MOTOR STARTER IS ENERGISED
42/C ✱
H-5
PERMISSIVE WHEN VALVE IS NOT FULLY OPEN
LIMIT SWITCH
ON VALVE
CR
(PWR AVAIL)
CR
OPEN
REVERSING AC MOTOR STARTER
PUMP SUCTION VALVE FO
NOTE 5
REACTOR HIGH PRESSURE
PS B21-N097A
LP
REACTOR LOW WATER LEVEL 2
TRIP UNIT B21-N692A
LP
PERMISSIVE WHEN REACTOR HIGH PRESSURE
PS B21-N097C
LP
PERMISSIVE WHEN REACTOR LOW WATER LEVEL 2
TRIP UNIT B21-N692C
LP
PERMISSIVE AFTER 10 SEC TIME DELAY
TDS
LP
RPS SYSTEM
REF 7
SH 4, H-3
REACTOR HIGH PRESSURE
PS B21-N097B
LP
REACTOR LOW WATER LEVEL 2
TRIP UNIT B21-N692B
LP
PERMISSIVE WHEN REACTOR HIGH PRESSURE
PS B21-N097D
LP
PERMISSIVE WHEN REACTOR LOW WATER LEVEL 2
TRIP UNIT B21-N692D
LP
PERMISSIVE AFTER 10 SEC TIME DELAY
TDS
LP
CLOSE
TRIP
CIRCUIT BREAKER 72 ✱ CB3A
CLOSE
TRIP
CIRCUIT BREAKER 72 ✱ CB3B
ATWS EOC-RPT BREAKERS SET "A"
TYPICAL FOR SET "B" (CB4A & CB4B)
(REF 8)



Also Available On Aperture Card

FIG 7.7.1-5E
REACTOR RECIRCULATION SYSTEM FCD
SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

729E625BD

8307050091 - 28

Request 223.38 (7.0) (15.0)

The Rod Sequence Control System is assumed to function in Section 15.0 of the FSAR to mitigate or prevent several accidents and as such it must be designed in accordance with criteria equivalent to that of protection systems. Provide the design basis description, drawings and all other information for this system.

Response:

The rod sequence control system (RSCS) is described in Sections 7.7.1.12 and 7.7.2.12. The RSCS is not relied upon for the safety action required for the control rod drop accident or the continuous control rod withdrawal during startup accident. The RSCS is designed as a backup to procedural controls on the movement of control rods. There are no specific regulatory requirements for the RSCS. The RSCS is shown in the control rod drive hydraulic system FCD, Figures 7.7.1-2A through G.

The RSCS acts to prevent withdrawal of an out-of-sequence control rod, to prevent continuous control rod withdrawal errors during reactor startup, and to minimize the core reactivity transient during a rod drop accident. The consequences of a rod withdrawal error in the startup range were generically analyzed in NEDO-10527, demonstrating that the licensing basis criterion for fuel failure is still satisfied even when the RSCS fails to block rod withdrawal. The safety action required for the control rod withdrawal accident (a reactor scram) is provided by the safety-related intermediate range monitor (IRM) subsystem of the neutron monitoring systems (NMS). If the core flux scram trip setpoint is reached during a flux transient, the IRM or a second safety-related NMS scram trip, supplied by the average power range monitor (APRM), can initiate a scram to terminate the core power transient.

Section 15A.1.12 has been revised to reflect the ultimate safety actions provided by the IRM or APRM.

Request 223.72:

With respect to the rod sequence control system described in Revision 5 to the FSAR, provide the functional control diagrams and as-built drawings supported with explanations, as deemed necessary, to permit an independent evaluation of the system design depicted in the diagrams and drawings.

Response:

The rod sequence control system (RSCS) is a backup to procedural controls governing the movement of control rods. There are no specific regulatory requirements for the RSCS. The purpose and description of the RSCS are found in Sections 7.7.1.12 and 7.7.2.12.

The RSCS is depicted in the control rod drive hydraulic system FCD, Figures 7.7.1-2 A through G.

The RSCS acts to prevent withdrawal of an out-of-sequence control rod, to prevent continuous control rod withdrawl errors during reactor startup, and to minimize the core reactivity transient during a rod drop accident. The consequences of a rod withdrawal error in the startup range were generically analyzed in NEDO-10527, demonstrating that the licensing basis criterion for fuel failure is still satisfied even when the RSCS fails to block rod withdrawal accident (a reactor scram) is provided by the safety-related intermediate range monitor (IRM) subsystem of the neutron monitoring systems (NMS). If the core flux scram trip setpoint is reached during a flux transient, the IRM or a second safety-related NMS scram trip, supplied by the average power range monitor (APRM), can initiate a scram to terminate the core power transient.

hoists. The platform operator can immediately determine whether the platform and hoists are responding to his local instructions and can, in conjunction with the main control room operator, verify proper operation of each of the three categories of interlocks listed previously.

7.7.1.12 Rod Sequence Control System

7.7.1.12.1 System Identification

The rod sequence control system is a subsystem of the reactor manual control system.

The purpose of the rod sequence control system (RSCS) is to reduce the consequences of the postulated rod drop accident, to prevent withdrawal of an out-of-sequence control rod, to prevent continuous control rod withdrawal errors during startup. The RCCS accomplishes this by restricting the patterns of control rods that can be established to predetermined sets.

7.7.1.12.2 Power Sources

The RSCS will operate from the same instrument bus as the rod position information system (RPIS), the subsystem that is the primary data source for the RSCS. The RSCS is designed so that it will apply rod movement inhibits to the rod drive control system (RDCS) in the event of loss of input power.

7.7.1.12.3 Equipment Design

1. General Description

The RSCS is a subsystem of the reactor manual control system (RMCS). It receives inputs from the RPIS and the RDCS, both of which are also subsystems of the RMCS, and from the operator directly. The RSCS provides display outputs directly to the operator and rod movement interlocks to the RDCS.

The RSCS has five primary functional blocks plus buffering and interfacing hardware. The five are as follows:

a. rod pattern controller,

b. substitute position generator,

c. operator display,

d. tester, and

e. bypassed rod identifier.

Each of these is described in Section 7.7.1.12.3.4.

refueling interlocks are not required for any postulated design basis accident or for safe shutdown. Furthermore, the interlocks are required only for the refueling mode of plant operation. However, the system does conform to the specific industry standard and regulatory guide listed on Fig. 7.1.1-2, while the requirements of 10CFR50 Appendix B are met in the manner set forth in Chapter 17.

7.7.2.12 Rod Sequence Control System

7.7.2.12.1 Conformance to General Functional Requirements

The RSCS provides backup to procedural controls, by imposing restrictions on the movement of control rods to reduce the consequences of a postulated rod-drop accident, to prevent the withdrawal of an out-of-sequence control rod and to prevent continuous rod withdrawal errors during startup.

Both the RWM and the RSCS are compatible and redundant to each other.

7.7.2.12.2 Conformance to Specific Regulatory Requirements

There are no specific regulatory requirements for the RSCS. The quality level of design and hardware of the RSCS is equivalent to that of a single channel of a protection system. The RSCS neither interfaces with the RPS nor affects the ability of the reactor to respond to any requirement of the RPS.

7.7.2.13 Neutron Monitoring System Instrumentation and Controls

7.7.2.13.1 Source Range Monitor Subsystem

1. General Functional Requirements Conformance

   The arrangement of the neutron sources and startup chambers in the reactor is shown on Fig. 7.7.1-11. This arrangement produces at least three counts per sec in the SRM using the sensitivity noted in Section 7.7.1.13.1 and the design source strength at initial reactor start-up. If the discriminator setting is adjusted to produce the specified sensitivity, the signal-to-noise count ratio is well above the 2:1 design basis for cold start-up.

   If the multiplication of one section of the core increases to put that section of the reactor on a 20 sec period, the nearest SRM chamber shows an increase in count rate. In general, at least one detector indicates the change in multiplication (Figs. 7.7.2-1 through 7.7.2-6).

Normal startup procedures ensure that withdrawal of control rods is distributed about the core to prevent excessive multiplication in any one section of the core. Hence, each SRM chamber can respond in some degree.

This unlikely set of circumstances makes possible the rapid removal of a control rod. The dropping of the rod results in a high local K in a small region of the core. For large, loosely coupled cores, this would result in a highly peaked power distribution and subsequent shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion. Therefore, the method of analysis must be capable of accounting for any possible effects of the power distribution shifts.

In order to limit the worth of the rod which could be dropped, the rod sequence control system (RSCS) is installed. This system prevents the movement of an out-of-sequence rod in the 100 to 75 percent rod density range, and from the 75 percent rod density point to the preset power level the RSCS will only allow group notch mode rod withdrawals or insertions. The 75 percent rod density configuration corresponds to the condition in which 75 percent of the rods are fully withdrawn. With the condition that no out-of-sequence rod may be moved, the postulated rod drop accident cannot result in peak enthalpies in excess of 280 calories/g for any possible plant operation or core exposure conditions. Table 15.1.33-1 presents the parameters used in this analysis.

15.1.33.3 Accident Description

The accident is defined as:

1. The rod worth minimizer (RWM) does not function.

2. The highest rod worth that can be developed at any time in core life under any operating conditions, by one inadvertent operator error, drops from fully inserted position to fully withdrawn position.

3. The rod drops at the maximum speed of 2.79 ft/sec.

4. The scram time is 5 sec to the 90 percent insertion point.

5. The RSCS is assumed to function.

The sequence of events and the approximate times of occurrence are as follows:

Event	Approximate Elapsed Time, seconds
1. Reactor is operating at 75 percent control rod density pattern	-
2. RWM is not functioning	-
3. Maximum worth control blade becomes decoupled	-

15A.1.11.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

15A.1.12 Continuous Rod Withdrawal During Reactor Startup

15A.1.12.1 Identification of Causes

While operating in the power, source, and/or intermediate range of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod continuously.

15A.1.12.2 Sequence of Events and Systems Operation

Control rod withdrawal errors are highly improbable in the startup power range. The RSCS and the RWM prevent the operator from selecting and withdrawing an out-of-sequence control rod.

Continuous control rod withdrawal errors during reactor startup are precluded by the RSCS. The RSCS prevents the withdrawal of an out-of-sequence control rod in the 100 to 75 percent control rod density range and limits rod movement to the banked position mode of rod withdrawal from the 75 percent rod density to the desired power level. Since only in-sequence control rods can be withdrawn in the 100 to 75 percent control rod density and control rods are withdrawn in the banked position mode from the 75 percent control rod density point to the desired power level, there is no basis for the continuous control rod withdrawal error in the startup power range. See Section 15A.1.11 for description of continuous control rod withdrawal during power range operation. The bank position mode of the RSCS is described in Reference 4.

No operator actions are required to preclude this event since the plant design as discussed above prevents its occurrence.

For any operation involved in a possible initiating failure (or error), the necessary safety actions are taken (rod blocks) prior to any possible subsequent single failure.

The probability of initiating causes (or multiple errors) for this event alone is considered low enough to warrant its being categorized as an infrequent incident. The probability of further development of this event is extremely low because it is contingent upon the simultaneous failure of two redundant systems, the RSCS and RWM systems concurrent with high worth rod, out-of-sequence rod selection contrary to procedures, plus operator nonacknowledgment of continuous alarm annunciations prior to safety system actuation.

Not withstanding the design provisions to preclude rod withdrawal, a special analysis is included (Section 15A.1.12.4) to show that continuous withdrawal of an out-of-sequence rod in the startup range results in acceptable peak fuel enthalpies less than the licensing basis criterion.

15A.1.12.3 Core and System Performance

The performance of the RSCS and RWM prevent erroneous selection and withdrawal of an out-of-sequence control rod, as described in Section 15A.1.12.2. Thus, the core and system performance is not affected by such an operator error.

15A.1.12.4 Special Analysis

The continuous control rod withdrawal analysis in the startup range was performed to demonstrate that the licensing basis criterion for fuel failure will not be exceeded when an out-of-sequence control rod is withdrawn at the maximum allowable normal drive speed. The sequence and timing assumed in this special analysis is shown in Table 15A.1.12-1.

The rod sequence control system (RSCS) and the rod worth minimizer (RWM) constraints on rod sequences will prevent the continuous withdrawal of an out-of-sequence rod. This analysis was performed to demonstrate that, even for the unlikely event where the RWM and RSCS fail to block the continuous withdrawal of an out-of-sequence rod, the licensing basis criterion for fuel failure is still satisfied.

The methods and design basis used for performing the detailed analysis for this event, are similar to those previously approved for the control rod drop accident (CRDA) (References 10, 11, and 12). Additional simplied point model kinetics calculations were performed to evaluate the dependence of peak fuel enthalpy on the control blade worth. For the detailed calculation, the 50 percent control rod density pattern was selected as the initial starting condition which is consistent with the approved design basis for the CRDA (References 10, 11, and 12).

The licensing basis criterion for fuel failure is the contained energy of a fuel pellet located in the peak power region of the core shall not exceed 170cal/gm-UO_2.

15A.1.12.4.1 Methods of Analysis

Since the rod worth calculations using the approved design basis methods (References 10, 11, and 12) use three-dimensional geometry, it is not practical to do a detailed analysis of this event parameterizing control rod worths. Therefore, the methods of analysis employed were to perform a detailed evaluation of this event for a typical BWR and control rod worth (1 percent Δk)

and to use a point model calculation to evaluate the results over the expected ranges of out-of-sequence control rod worths. The detailed calculations are performed to demonstrate (1) the consequences of this event over the expected power operating range and (2) the validity of the approximate point model calculation. The point model calculation will demonstrate that the licensing criterion for fuel failure is easily satisfied over the range of expected out-of-sequence control rod worths. These methods are described in more detail below.

The methods used to perform the detailed calculation are identical to those used to perform the design basis control rod drop accident with the following exceptions:

a. The rod withdrawal rate is 3.6 fps rather than the blade drop velocity of 3.11 fps.

b. Scram is initiated either by the IRM or 15 percent APRM scram in the startup range. The IRM system is assumed to be in the worst bypass condition allowed by technical specifications.

c. The blade being withdrawn inserts along with remaining drives at technical specification insertion rates upon initiation of scram signal.

Examination of a number of rod withdrawal transients in the low power startup range, using an R-Z model, has shown clearly that higher fuel enthalpy addition would result from transients starting at the 1 percent power level rather than from lower power levels. The analysis further shows that for continuous rod withdrawal from these initial power levels (1 percent range) the APRM 15 percent power level scram is likely to be reached as soon as the degraded (worst bypass condition) IRM scram. Consequently, credit is taken for either the IRM or APRM 15 percent scram in meeting the consequences of this event. The transients for this response were initiated at 1 percent of power and were performed using the 15 APRM scram.

An initial point kinetics calculation was run to determine the line to scram based on an APRM scram setpoint of 15 percent power and an initial power level of 1 percent. From this time and the maximum allowable rod withdrawal speed, it is possible to show the degree of rod withdrawal before reinsertion due to the scram. From this information Figure 15A.1.12-1 showing the modified effective reactivity shape, was constructed.

The point model kinetics calculations use the same equations employed in the adiabatic approximation described on page 4-1 of Reference 10. The rod reactivity characteristics and scram reactivity functions are input identical to the adiabatic calculations, and the Doppler reactivity is input as a function

of core average fuel enthalpy. The Doppler reactivity feedback function input to the point model calculations was derived from the detailed analysis of the 1.6 percent rod worth case described above. This is a conservative assumption for higher rod worths since the power peaking and hence spatial Doppler feedback will be larger for higher rod worths. As will be seen in the results section, maximum enthalpies resulted from cases initiated at 1 percent of rated power. In this power range, the APRM will initiate scram at 15 percent of power; hence, the APRM 15 percent power scram was used for these calculations thereby eliminating the need to perform the spatial analysis required for the IRM scram. All other inputs are consistent with the detailed transient calculation.

The point model kinetics calculations results in core average enthalpies. The peak enthalpies were calculated using the following equations:

where

$\hat{h} = ho + (P/A)_T (\bar{h}_f - ho)$:

$\hat{h}$ = Final peak fuel enthalphy:

ho = Initial fuel enghalpy:

$\bar{h}_f$ = Total peaking factor (radial peaking) + (axial peaking) + (local fuel pin peaking).

For these calculations, the (radial x axial) peaking factors as a function of rod worth were obtained from the calculations performed in Section 3.6 of Reference 11 and are shown in Fig. 15A.1.12-2. It was conservatively assumed that no power flattening due to Doppler feedback occurred during the course of the transient.

15A.1.12.4.2 Results

The reactivity insertion resulting from moving the control rod is shown in Fig. 15A.1.12-1 for the point kinetics calculations. The core average power versus time and the global peaking factors from Section 3.6 of Reference 11 are shown in Figs. 15A.1.12-3 and 15A.1.12-2, respectively. The results of the point kinetics calculation are summarized in Table 15A.1.12-2 along with the results of the detailed analysis.

From Fig. 15A.1.12-3 and Table 15A.1.12-2, it is shown that the core average energy deposition is insensitive to control rod worth; therefore, the only change in peak enthalpy as a function of rod worth will result from differences in the global peaking which increases with rod worth. Comparison of the global peaking factors shown in Fig. 15A.1.12-2 with the values used in the

detailed calculations demonstrates that the Reference 11 values are reasonable for their application in this study. For all cases, the peak fuel enthalpy is well below the licensing design criteria of 170 cal/gm.

Cases 4 and 5 of Table 15A.1.12-2 show that the point kinetics calculations give conservative results relative to the detailed evaluations. The primary difference is that the global peaking will flatten during the transient due to Doppler feedback. This is accounted for in the detailed calculation but the point kinetics calculations conservatively assumed that the peaking remains constant at its initial value.

The differences in core average and peak enthalpy between Cases 1 and 5 are due to the fact that for Case 1, the scram was initiated by the 15 percent APRM scram setpoint, whereas, in Case 5, the scram was intiated by the IRM's. As seen in Fig. 15A.1.12-4 this occurred at a core average power of 21 percent. Since the APRM trip point will be reached first, it is reasonable to take credit for the APRM scram.

15A.1.12.4.3 Conclusions

From this study the following conclusions can be stated:

a. The resultant peak fuel enthalpies due to the continuous withdrawal of an out-of-sequence rod in the startup range results in peak fuel enthalpies which are significantly less than the licensing basis criteria of 170 cal/gm.

b. The point model calculations used to assess the sensitivity of peak enthalpy as a function of control rod worth are in good agreement with, and slightly conservative relative to the more detailed design basis model which is employed to evaluate the continuous rod withdrawal transient in the startup range.

15A.1.12.4 Barrier Performance

An evaluation of the barrier performance was not made for this event because resultant peak fuel enthalpies, due to continuous rod withdrawal, are significantly less than the licensing basis criterion.

15A.1.12.5 Radiological Consequences

An evaluation of the radiological consequences was not made for this event since no radioactive material is released from the fuel.

References in Appendix 15A

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5. R.B. Elkins, "Fuel Rod Prepressurization, Amendment 1," NEDO-23786-1, May 1978.

6. E.D. Fuller, letter to O.D. Parr, "NRC Request for Additional Information on Fuel Rod Prepressurization," June 8, 1978.

7. E.D. Fuller, letter to O.D. Parr, "NRC Request for Additional Information on Fuel Rod Pressurization," {sic} August 14, 1978.

8. R.B. Elkins, "Fuel Rod Prepressurization," NEDE-23786-P, March 1978.

9. F. Odar, "Safety Evaluation for General Electric Topical Report: Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors", NEDO-24154, 1980.

10. C.J. Paone et al, "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.

11. R.C. Stirn et al, "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, Supplement 1, July 1972.

12. R.C. Stirn, "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2, Exposed Cores," NEDO-10527, Supplement 2, January 1973.

TABLE 15A.1.12-1

SEQUENCE OF EVENTS FOR CONTINUOUS ROD WITHDRAWAL DURING REACTOR STARTUP

TIME (Sec)	EVENT
0	1. The reactor is critical and operating in the startup range.
>0	2. The operator selects and withdraws an out-of-sequence control rod at the maximum normal drive speed of 3.6 ips.
~4 sec	3. Both the RWM and the RSCS fail to block the selection (selection error) and continuous withdrawal (withdraw error) of the out-of-sequence rod.
4-8 sec	4. The reactor scram is initiated by the intermediate range monitor (IRM) system or the advantage power range monitor system (APRM).
5-9 sec	5. The prompt power burst is terminated by a combination of Doppler and/or scram feedback.
10 sec	6. The transient is finally terminated by the scram of all rods, including the control rod being withdrawn. Scram insertion times are assumed to be 5 seconds to 90 percent insertion.)

TABLE 15A.1.12-2

SUMMARY OF RESULTS FOR DETAILED AND POINT KINETICS EVALUATIONS OF CONTINUOUS ROD WITHDRAWAL IN THE STARTUP RANGE

Case	Control Rod Worth (%Δk)	$\bar{h}_f$(cal/gm)	P/A[1]	$\hat{h}$(cal/gm)
1	1.6	17.3	24.2	42.7
2	2.0	17.3	30.9	50.0
3	2.5	17.2	46.0	58.5
4	1.6 [2]	18.3	19.7 [3]	56.2
5	1.6 [4]	18.3	19.7	59.6

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[1] P/A = global peaking factor (Radial x Axial).

[2] Detailed transient calculation. All other data reported are for point kinetics calculations.

[3] The P/A = 19.7 is the initial value. For the detailed analysis this value will decrease during the course of the transient since the power shape will flatten due to Doppler feedback.

[4] Point kinetics calculation with IRM initiated scram and3-D simulator global peaking.

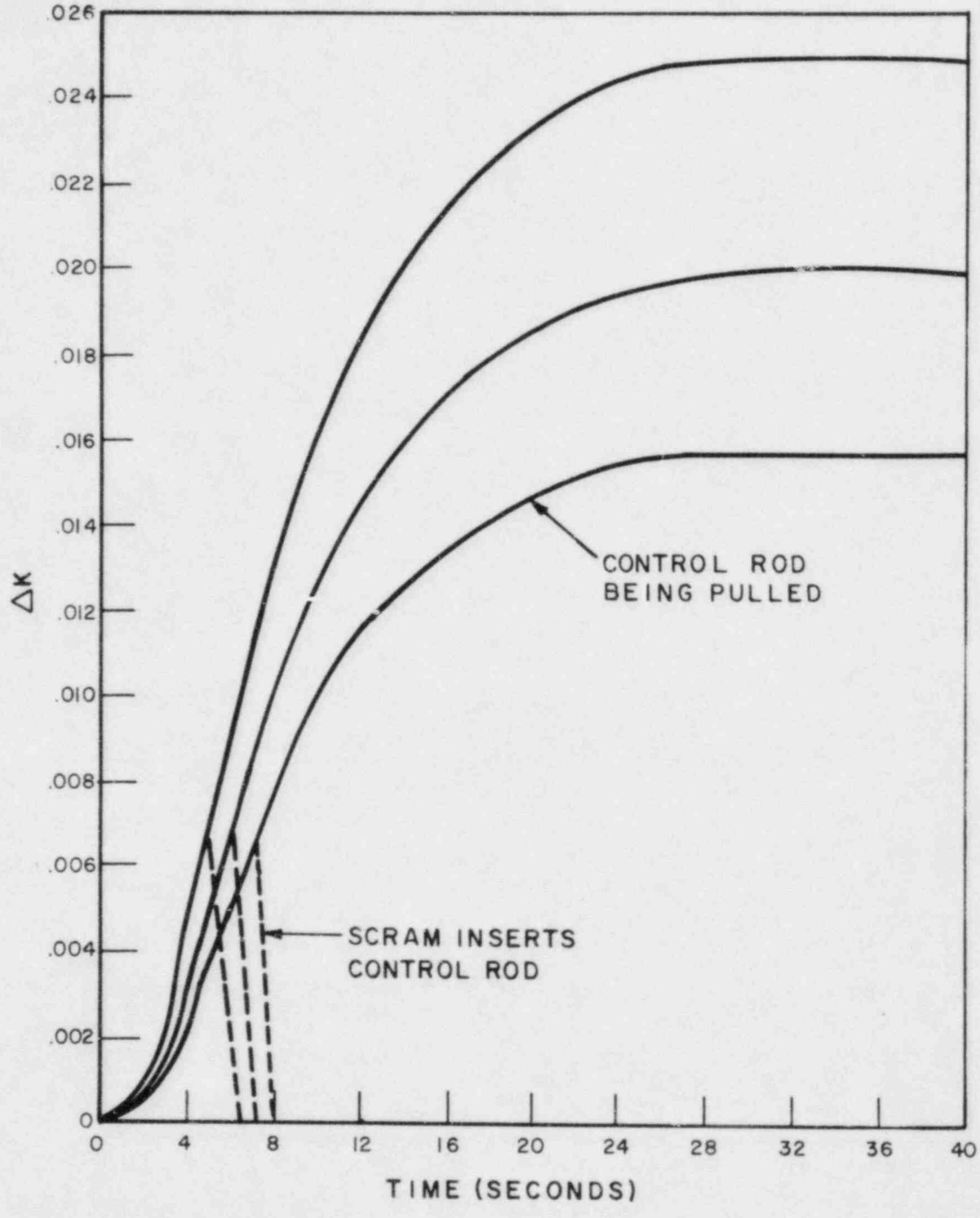

FIGURE 15A.1.12-1

POINT KINETICS CONTROL ROD REACTIVITY INSERTION

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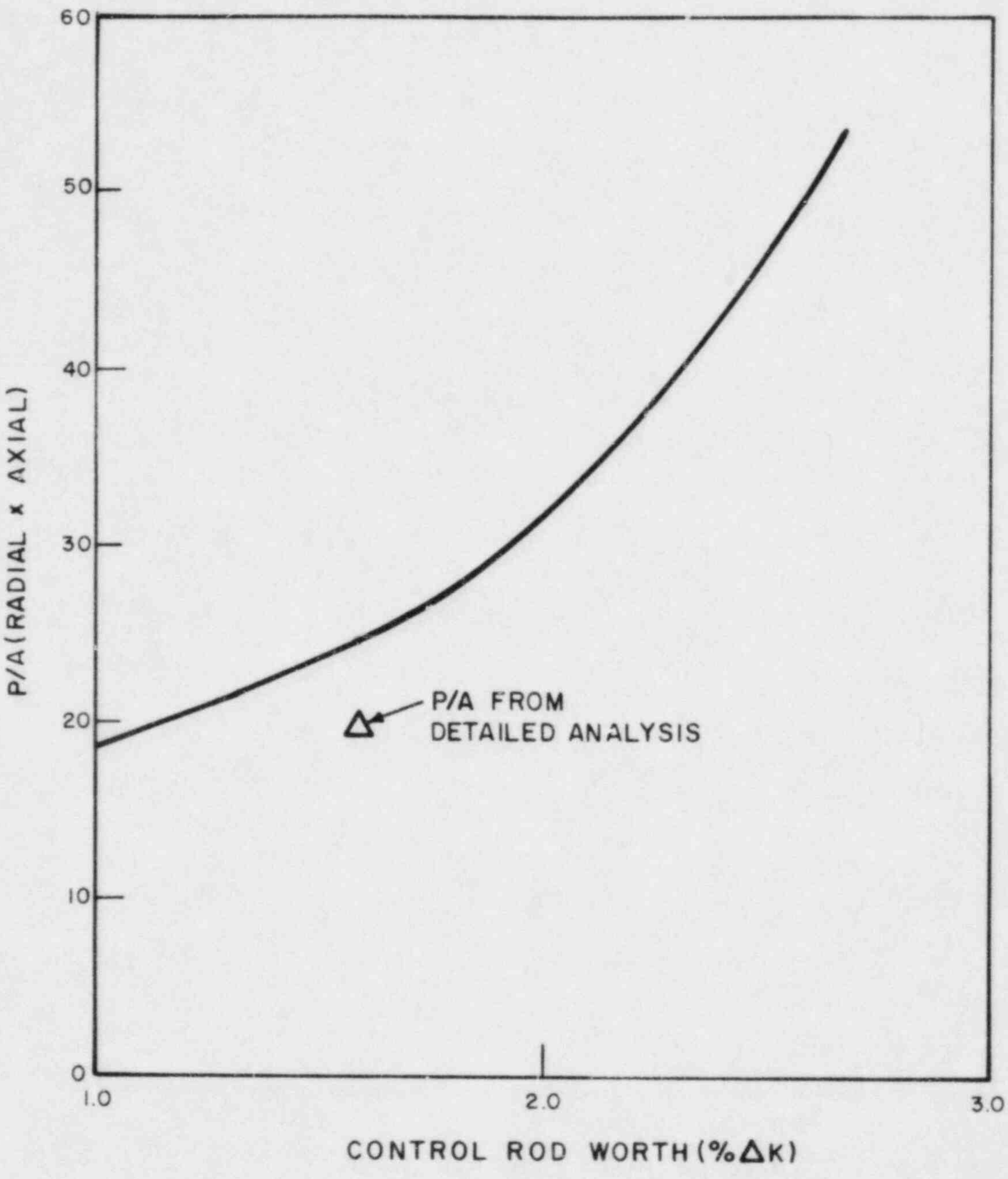


FIGURE 15A.1.12-2

P/A Vs. ROD WORTH NEDO-10527 SUPPLEMENT 1 [11] AND DETAILED ANALYSIS

SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT

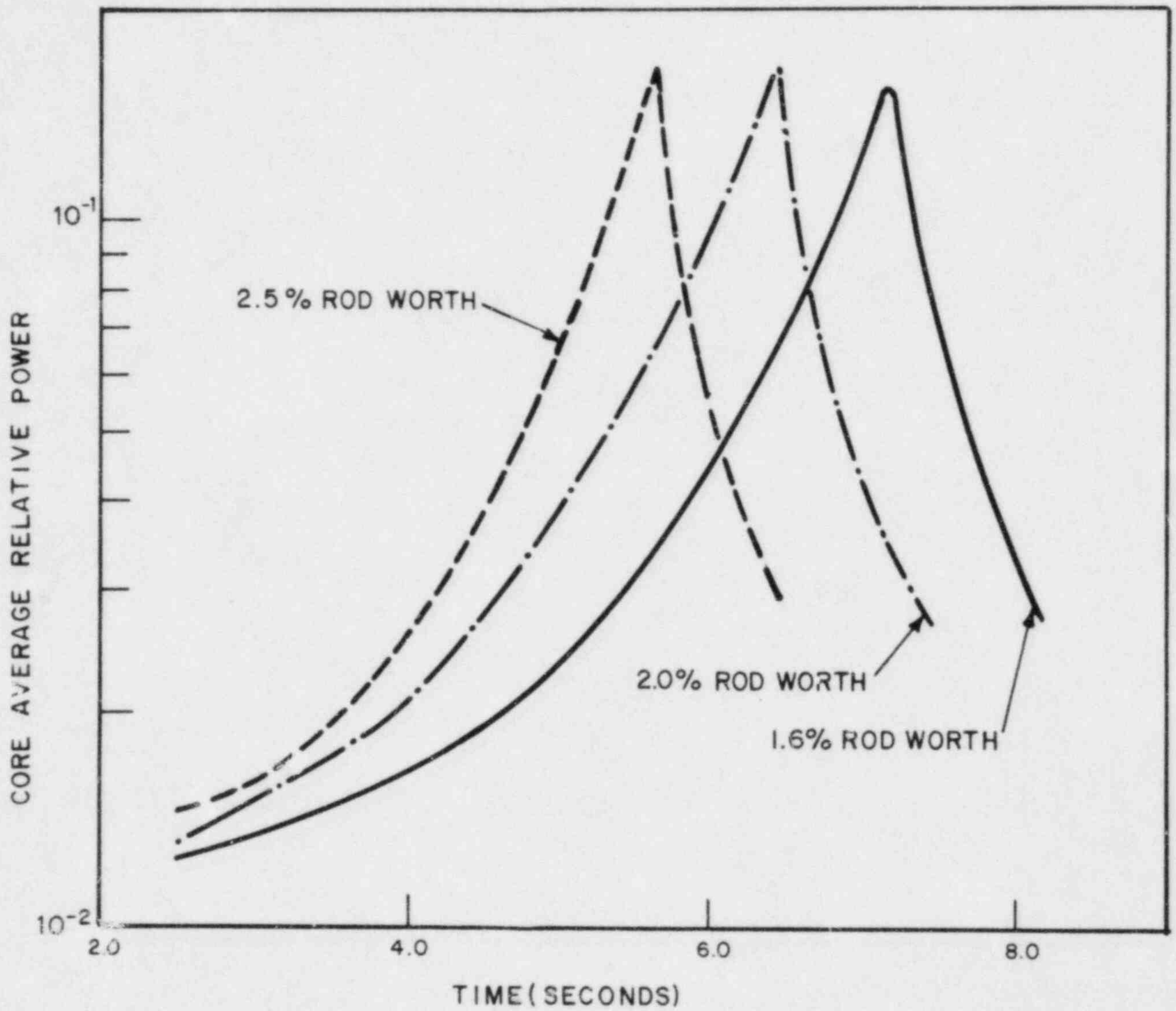
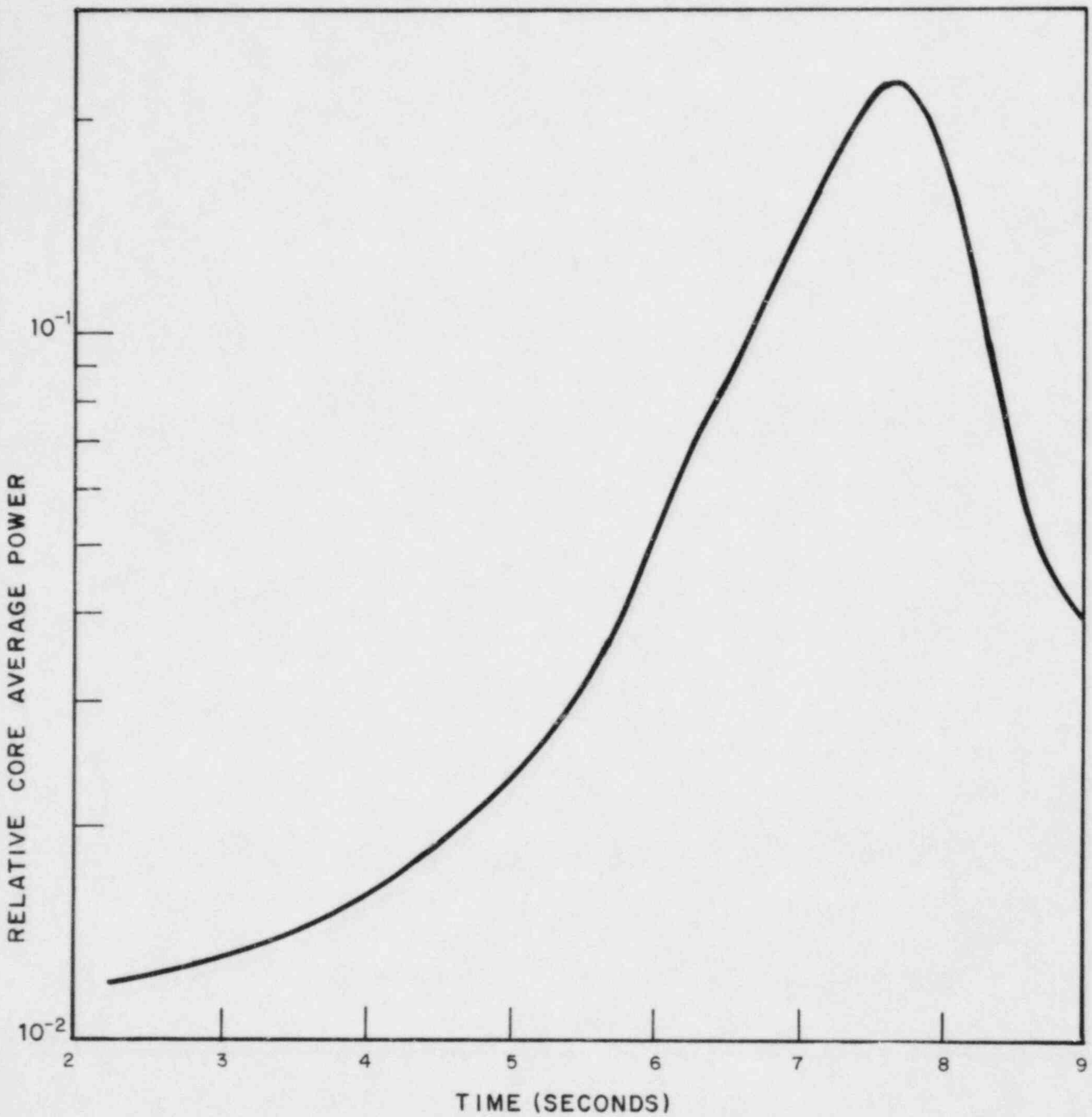


FIGURE 15A.1.12-3

CONTINUOUS RWE IN THE STARTUP RANGE CORE AVERAGE POWER Vs. TIME FOR 1.6%, 2.0% AND 2.5% ROD WORTH'S (POINT MODEL KINETICS)

SHOREHAM NUCLEAR POWER STATION-UNIT 1
FINAL SAFETY ANALYSIS REPORT



ASSUMPTIONS:

1. 1.6 % Δk ROD
2. 0.3 fps WITHDRAWAL VELOCITY
3. IRM SCRAM FOR WORST BYPASS CONDITION
4. $P_0 = 10^{-2}$ OF RATED
5. 1967 PRODUCT LINE TECH SPEC SCRAM RATE
6. EXPOSURE = 0.0 GWD/T

FIGURE 15A.1.12-4

CONTINUOUS CONTROL ROD WITHDRAWAL FROM HOT STARTUP

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