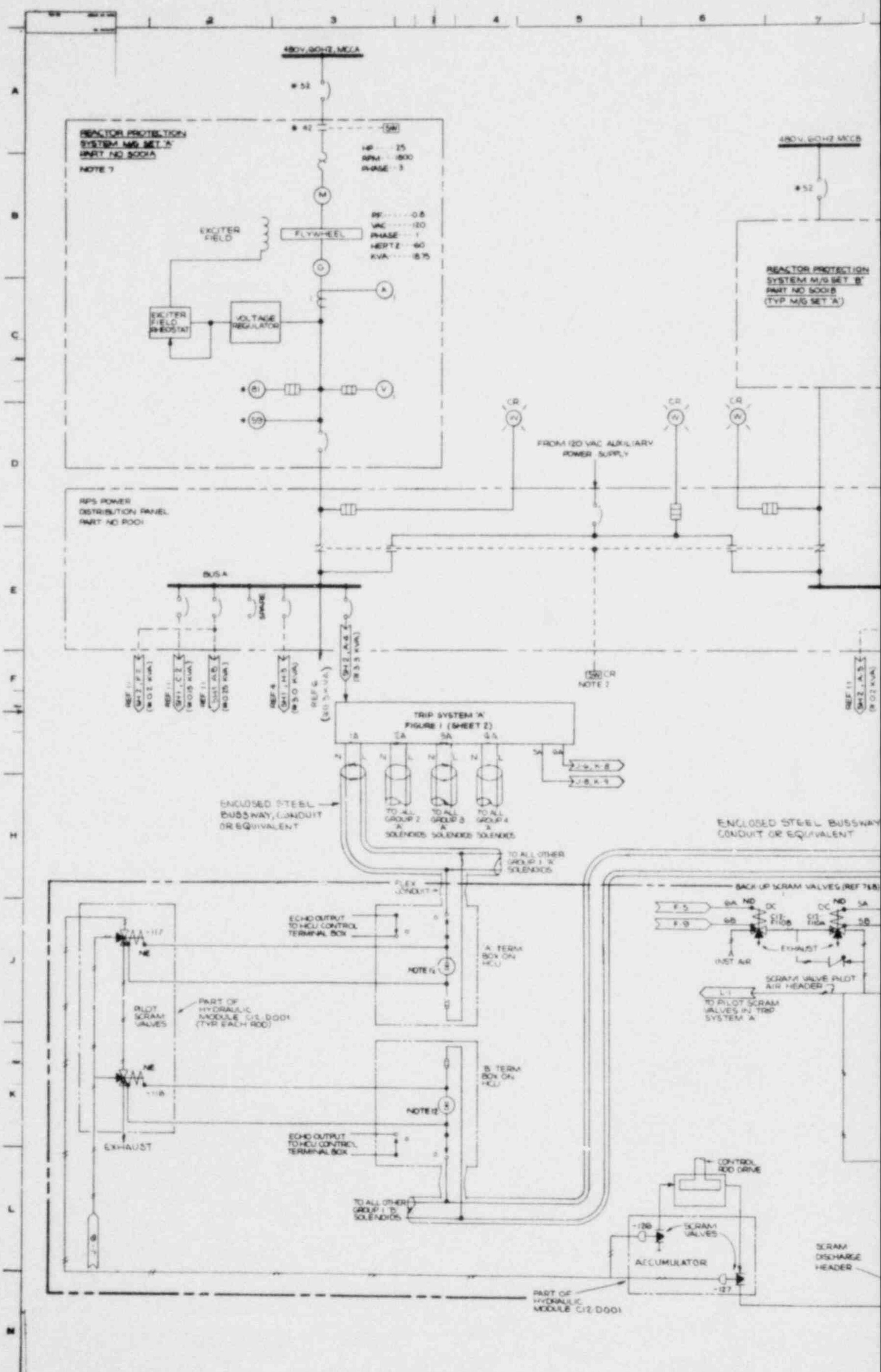
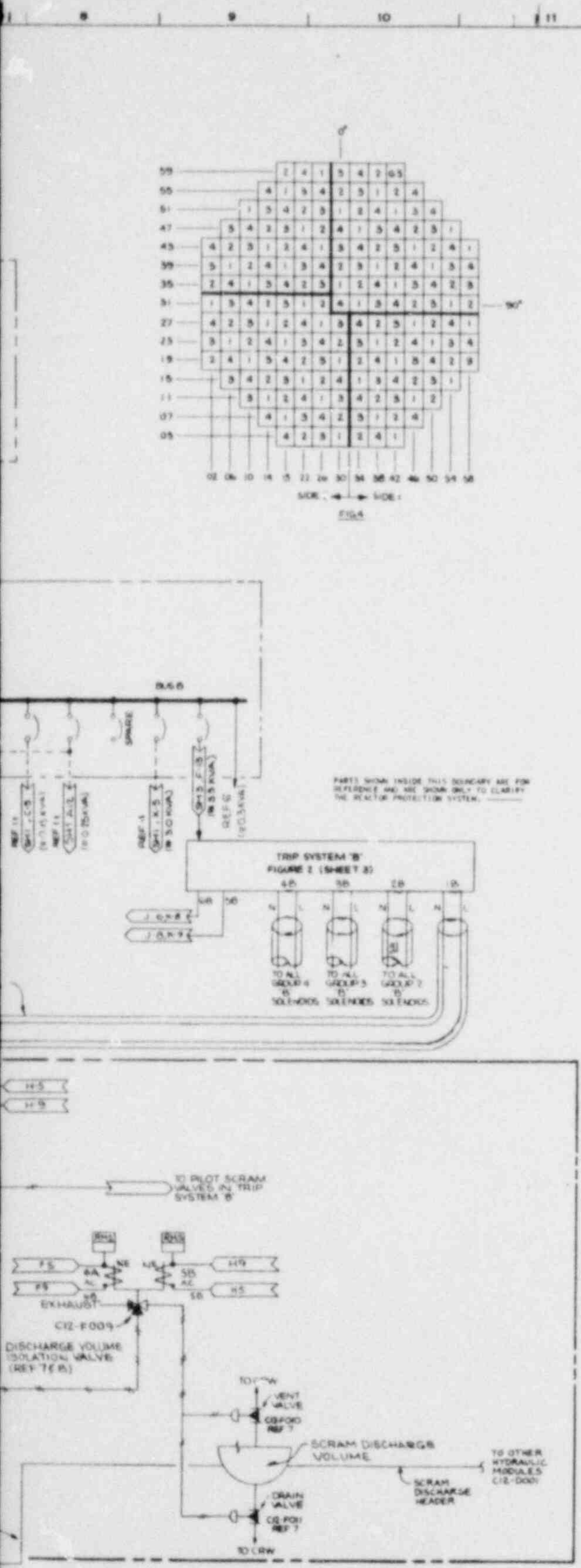

480V, 60HZ, MCC A
REACTOR PROTECTION SYSTEM M/G SET 'A' PART NO 5001A
NOTE 7
HP 25
RPM 1800
PHASE 3
FLYWHEEL
EXCITER FIELD
PF 0.8
VAC 120
PHASE 1
HERTZ 60
KVA 18.75
EXCITER FIELD RHEOSTAT
VOLTAGE REGULATOR
FROM 120 VAC AUXILIARY POWER SUPPLY
RPS POWER DISTRIBUTION PANEL PART NO P001
BUS A
SPARE
NOTE 2
480V, 60HZ MCC B
REACTOR PROTECTION SYSTEM M/G SET 'B' PART NO 5001B (TYP M/G SET 'A')
TRIP SYSTEM 'A' FIGURE 1 (SHEET 2)
ENCLOSED STEEL BUSSWAY, CONDUIT OR EQUIVALENT
TO ALL GROUP 2 A SOLENOIDS
TO ALL GROUP 3 A SOLENOIDS
TO ALL GROUP 4 A SOLENOIDS
TO ALL OTHER GROUP 1 'A' SOLENOIDS
FLEX CONDUIT
BACK-UP SCRAM VALVES
ECHO OUTPUT TO HCU CONTROL TERMINAL BOX
'A' TERM BOX ON HCU
NOTE 12
PILOT SCRAM VALVES
PART OF HYDRAULIC MODULE C12-D001 (TYP EACH ROD)
'B' TERM BOX ON HCU
EXHAUST
INST AIR
SCRAM VALVE PILOT AIR HEADER
TO PILOT SCRAM VALVES IN TRIP SYSTEM 'A'
TO ALL OTHER GROUP 1 'B' SOLENOIDS
CONTROL ROD DRIVE
SCRAM VALVES
ACCUMULATOR
SCRAM DISCHARGE HEADER
PART OF HYDRAULIC MODULE C12-D001

8005090322



NOTES:

1. DEVICES USED IN TRIP SYSTEM "A" ARE IDENTIFIED BY LETTERS A,C,E,G, ETC. THOSE USED IN TRIP SYSTEM "B" ARE B,D,F,H, ETC.
2. ACTUATION SHALL BE SUCH THAT ALTERNATE SOURCE CAN ONLY BE APPLIED TO EITHER BUS A OR B AT ONE TIME. SYSTEM SHALL BE ARRANGED SO THAT THE BUS CANNOT BE ENERGIZED FROM THE M-G SET AND ALTERNATE SOURCE SIMULTANEOUSLY.
3. ONE RESET SWITCH SHALL BE USED FOR BOTH TRIP SYSTEMS A AND B AND SO ARRANGED THAT ONE ACTUATION OF THE SWITCH WILL RESET ALL SCRAM PILOT VALVES (A AND B) IN GROUP 1 AND GROUP 4 AS WELL AS BACKUP SCRAM VALVE A. A DIFFERENT ACTUATION OF THE SWITCH WILL RESET ALL SCRAM PILOT VALVES (A AND B) IN GROUP 2 AND GROUP 3 AS WELL AS BACKUP SCRAM VALVE B. ACTUATION OF THE SWITCH TO BOTH GROUP 1-4 AND GROUP 2-3 POSITIONS SHALL BE REQUIRED BEFORE THE CRD DRAIN AND VENT VALVES RESET. THE RESET SWITCH SHALL BE SO CONSTRUCTED THAT RPS CHANNELS A2 AND B2 ARE PHYSICALLY SEPARATED FROM CHANNELS A1 AND B1.
4. MAIN STEAM LINE ISOLATION VALVE CLOSURE TRIP SHALL BE ARRANGED SO THAT ANY ONE STEAM LINE MAY BE ISOLATED (BY FULL CLOSURE OF ITS ISOLATION VALVES) AND THE ISOLATION VALVE FOR ANY ONE OTHER STEAM LINE CAN BE CLOSED (MORE THAN 10%) WITHOUT CAUSING A TRIP OF EITHER TRIP SYSTEM A OR B.
5. LOGIC FOR THE "TURBINE STOP VALVE CLOSURE" TRIP SHALL BE ARRANGED SO THAT CLOSURE OF 3 OUT OF 4 STOP VALVES WILL CAUSE A RPS TRIP. PROVISION SHALL BE MADE TO ALLOW CLOSURE OF ONE STOP VALVE (FOR TEST PURPOSES) WITHOUT CAUSING A TRIP OF EITHER TRIP SYSTEM A OR B.
6. TRIP CHANNELS FOR THE "TURBINE CONTROL VALVE FAST CLOSURE" TRIP SHALL BE DERIVED FROM THOSE EVENTS CAUSING FAST CLOSURE OF THE CONTROL VALVES.
7. EQUIPMENT RATINGS ARE ESTIMATED AND PRELIMINARY. ACTUAL VALUES TO BE DETERMINED AT TIME OF EQUIPMENT PROCUREMENT.
8. TRIP SIGNAL, WHEN ABSENT, DE-ENERGIZES THE SCRAM CONTACTOR.
9. EACH MAIN STEAM LINE RADIATION MONITOR MONITORS ALL FOUR MAIN STEAM LINES.
10. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NUMBER (C72) UNLESS OTHERWISE NOTED.
11. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET REF 15.
12. SELENIUM THYRECTOR SUPPRESSORS (OR EQUIVALENT) SHALL BE USED TO SUPPRESS ELECTRICAL ARCING OF SCRAM SOLENOIDS.
13. THE DISCHARGE VOLUME HIGH LEVEL BYPASS SWITCH SHALL BE SO CONSTRUCTED THAT RPS CHANNELS A1 AND B1 ARE PHYSICALLY SEPARATED FROM CHANNELS A2 & B2.
14. FOR ANY SINGLE ROD GROUP, (G1 ETC.), A AND B SOLENOID CABLES MAY BE RUN TOGETHER IN ONE CONDUIT.

REFERENCE DOCUMENTS:

		MPL NO.
1.	REACTOR PROTECTION SYS DESIGN SPEC.	C72-4010
2.	TURB. GEN. & STEAM BYPASS SYS DESIGN SPEC.	A62-4120
3.		
4.	NEUTRON MONITORING SYS IED	C51-1010
5.	NEUTRON MONITORING SYS FCD	C51-1030
6.	NUCLEAR BOILER SYS FCD	B21-1030
7.	CONTROL ROD DRIVE HYD SYS P&ID	C12-1010
8.	CONTROL ROD DRIVE HYD SYS FCD	C12-1030
9.	NUCLEAR BOILER P&ID	B21-1010
10.	RESIDUAL HEAT REMOVAL SYS P&ID	E12-1010
11.	PROCESS RAD MON SYS RECOMB/CARB BED IED	D17-1010
12.	LOGIC SYMBOLS	A42-1020
13.	PIPING & INSTRUMENT SYMBOLS	A42-1010
14.	PROCESS COMPUTER INPUT/OUTPUT REQUIREMENTS	C91-4010
15.	INSTRUMENT DATA SHEET	C72-3050
16.	RECIRCULATION PUMP TRIP DESIGN SPEC	B35-4020

NOTES (CONTINUED):

15. ALL ESSENTIAL 125 VDC BUS A WIRING IN CONTROL ROOM CABINETS SHALL BE SEPARATED FROM 125 VDC BUS B WIRING BY 6 INCHES OR BY METALLIC CONDUIT.
16. ALL LETTERS IN [ ] DESIGNATE RPT SYSTEM B.

LEGEND:

= SWITCHGEAR DEVICE FUNCTION NUMBER ANSI SPEC C37.2

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR PROTECTION SYSTEM IED SHEET 1	FIGURE 7.2-1a

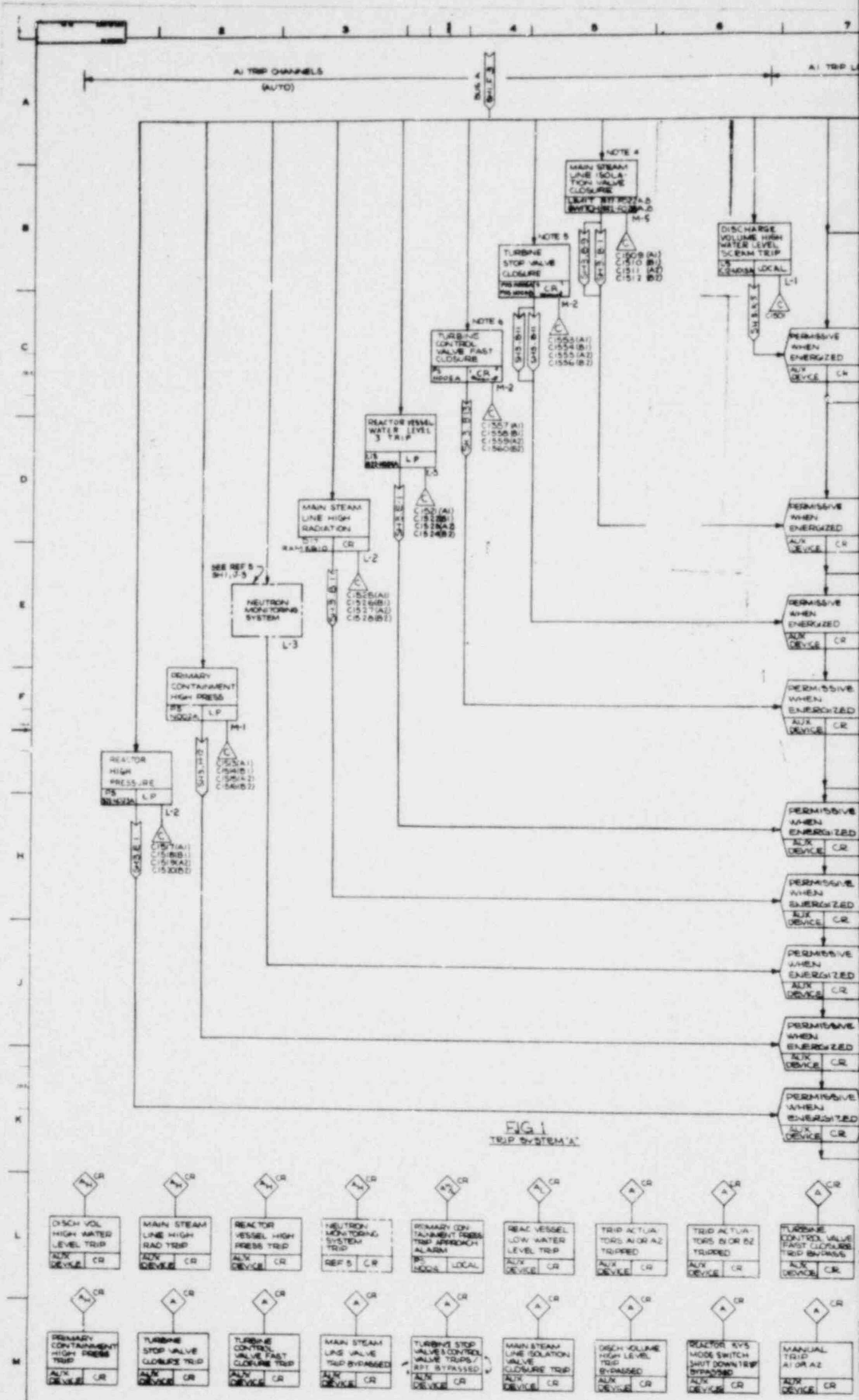
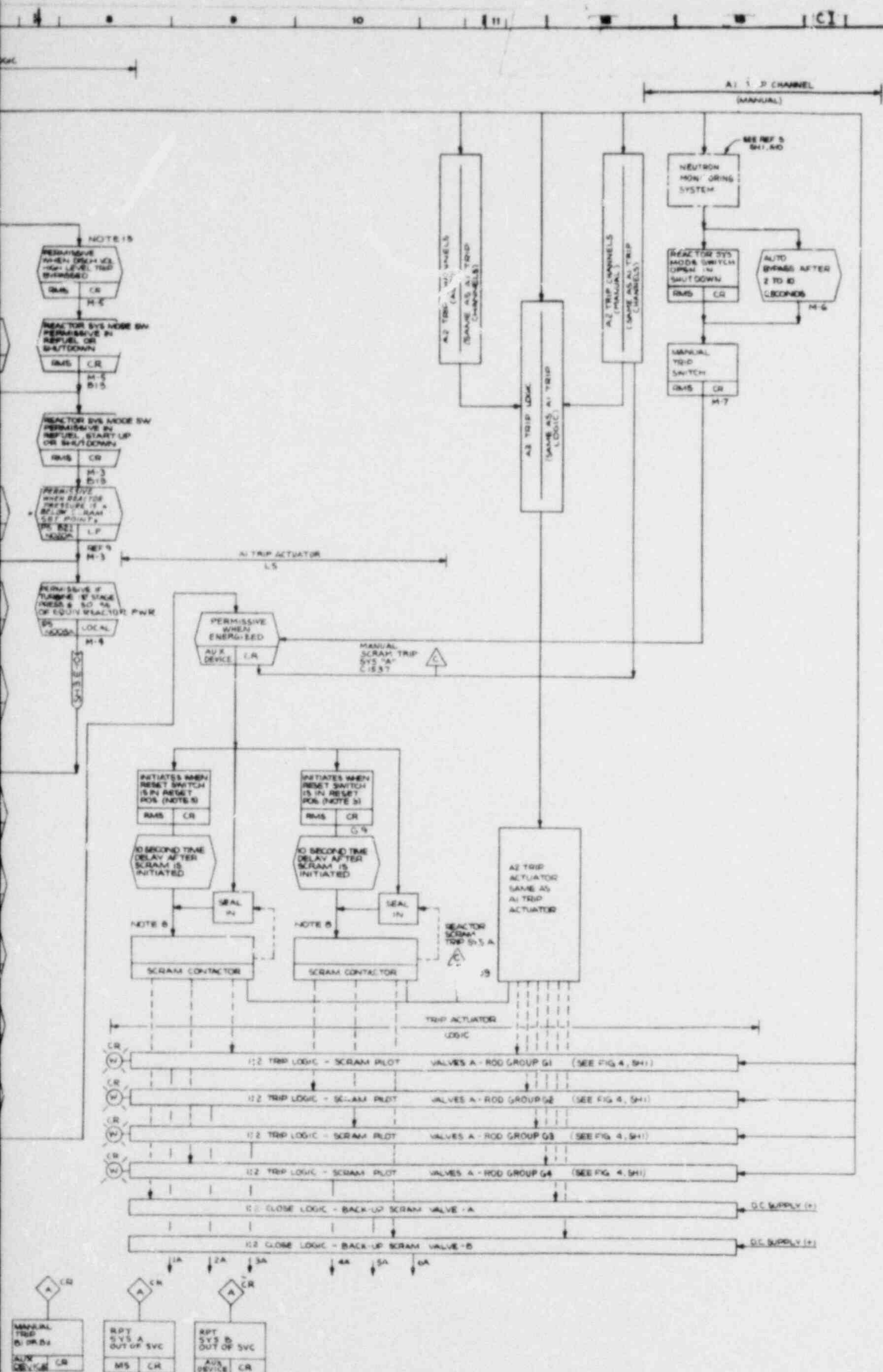


FIG 1
TRIP SYSTEM 'A'

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR PROTECTION SYSTEM IED SHEET 2	FIGURE 7.2-1b

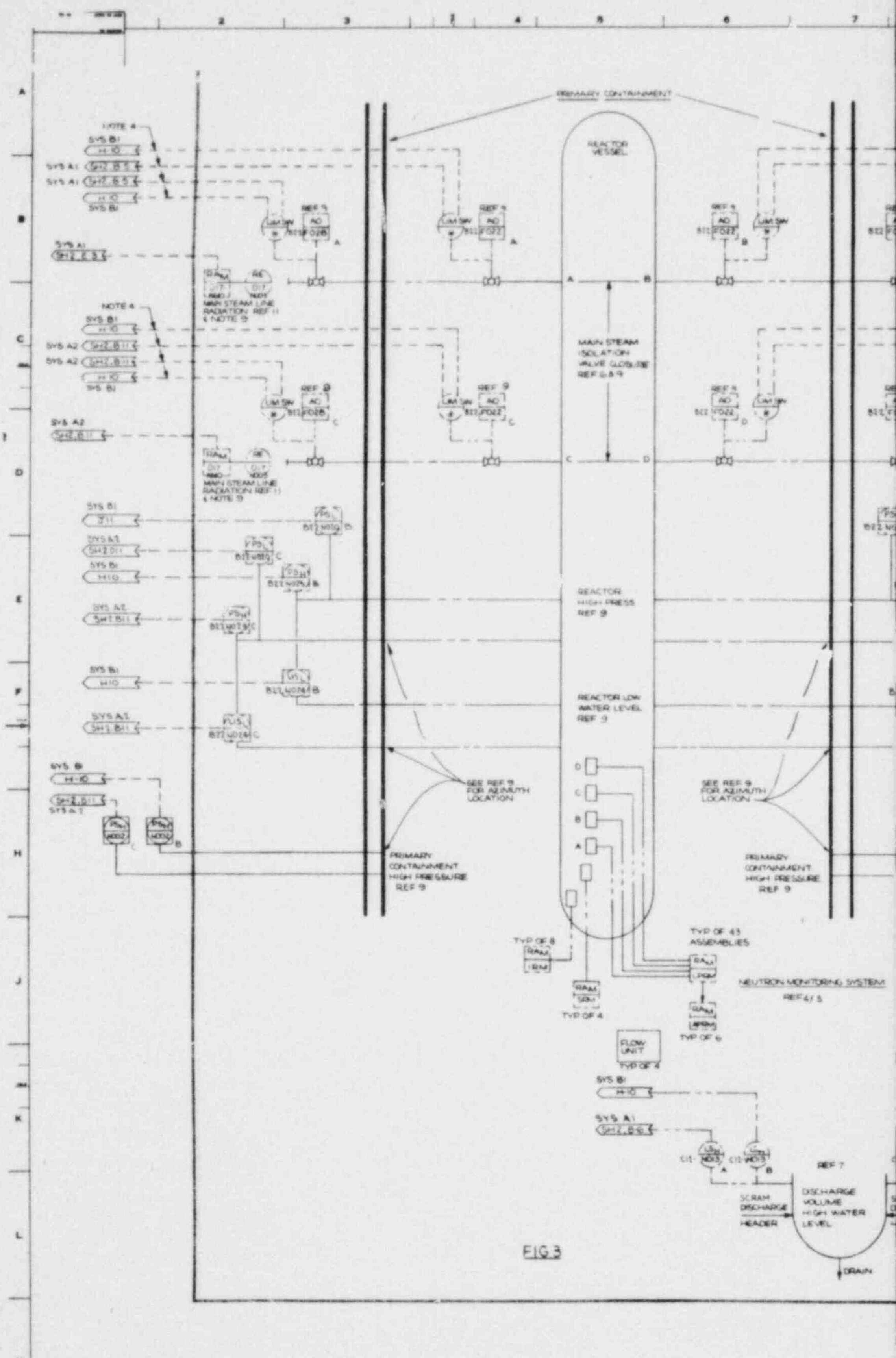

PRIMARY CONTAINMENT
REACTOR VESSEL
NOTE 4
MAIN STEAM LINE RADIATION REF 11 & NOTE 9
MAIN STEAM ISOLATION VALVE CLOSURE REF 6 & 9
REACTOR HIGH PRESS REF 9
REACTOR LOW WATER LEVEL REF 9
SEE REF 9 FOR AZIMUTH LOCATION
PRIMARY CONTAINMENT HIGH PRESSURE REF 9
TYP OF 8
TYP OF 43 ASSEMBLIES
TYP OF 4
TYP OF 6
NEUTRON MONITORING SYSTEM
REF 4/5
FLOW UNIT
TYP OF 4
REF 7
SCRAM DISCHARGE HEADER
DISCHARGE VOLUME HIGH WATER LEVEL
DRAIN

FIG 3

AMENDMENT NO. 10
July 1980

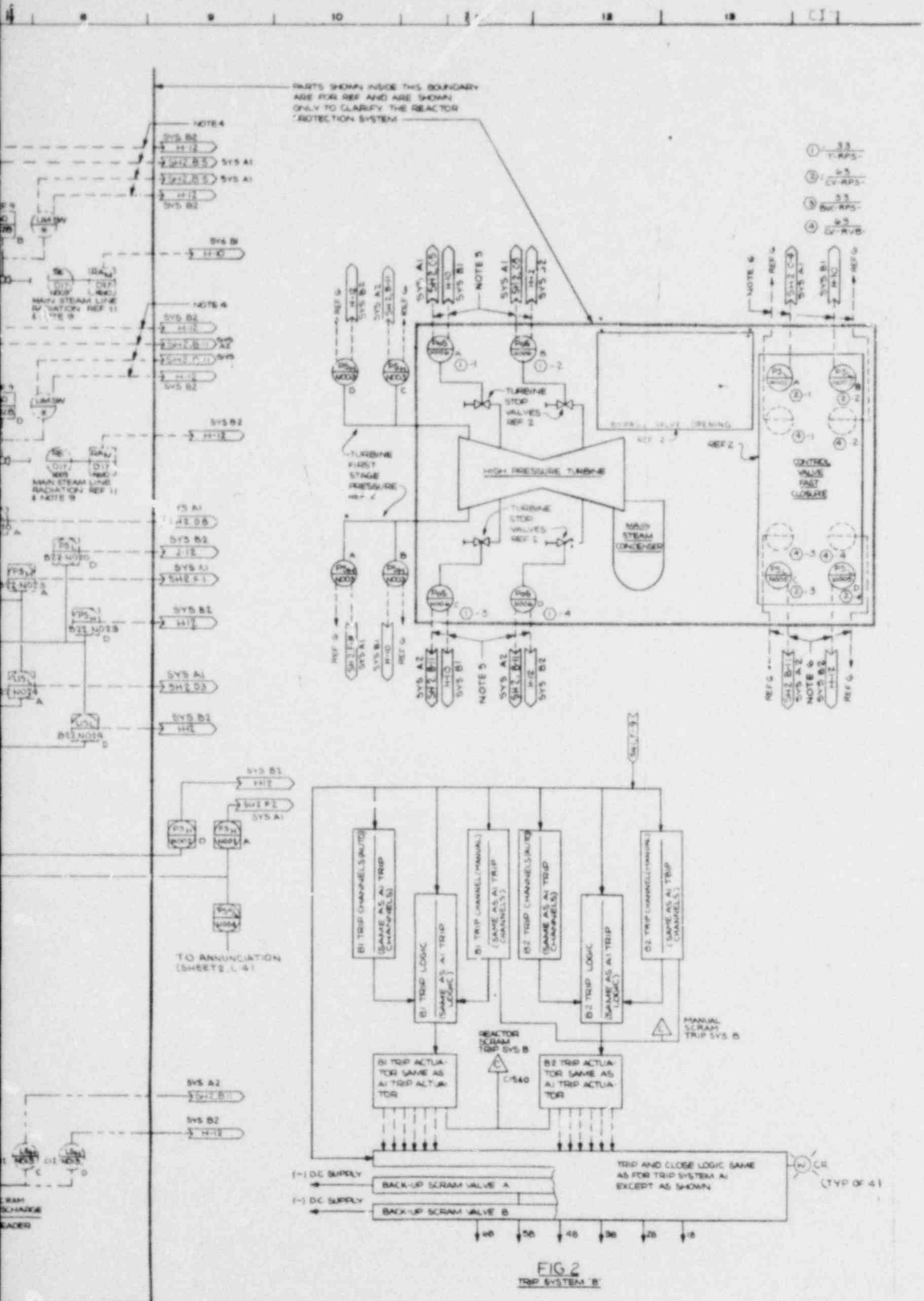
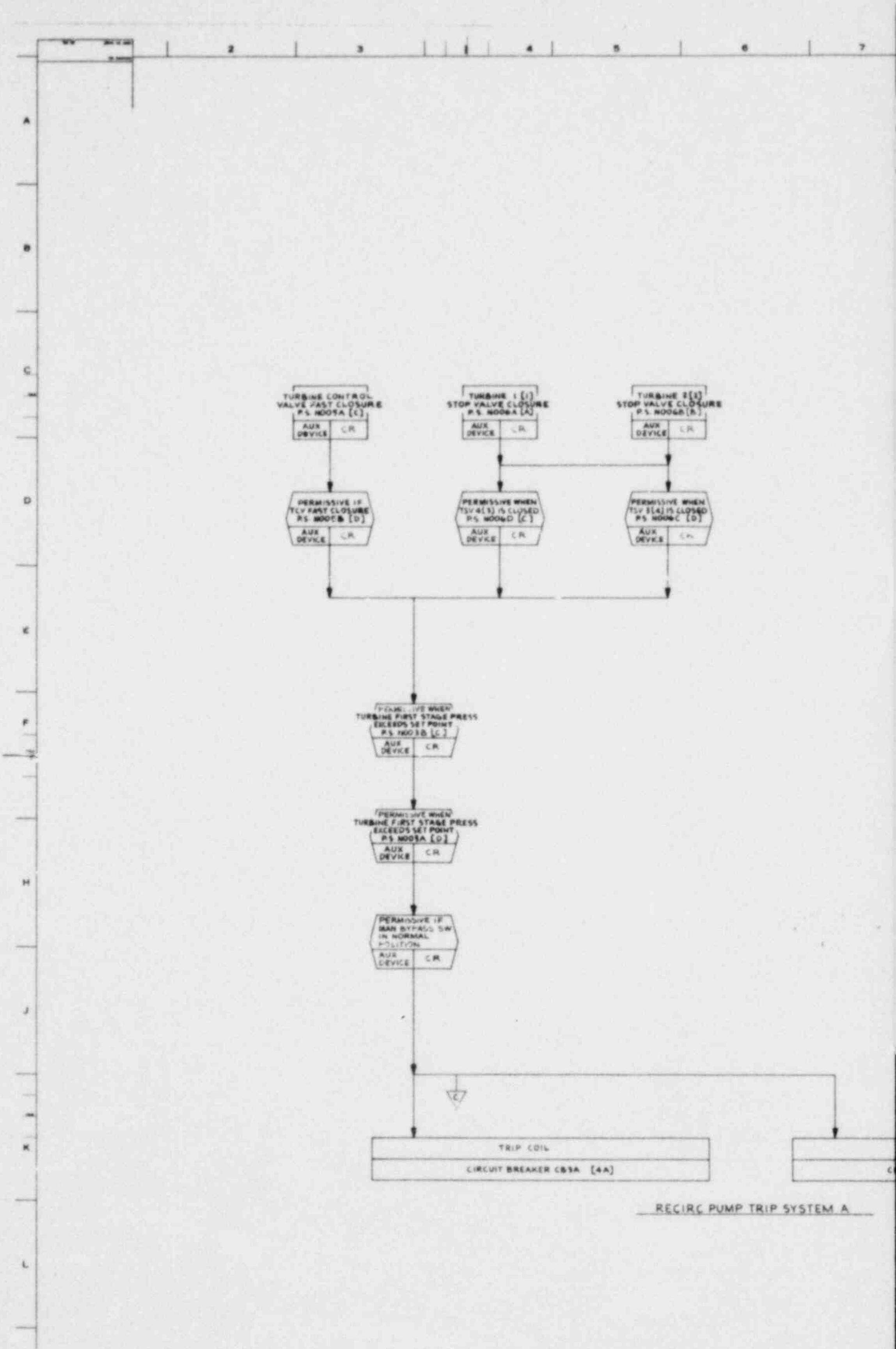


FIG 2
TRIP SYSTEM 'B'

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR PROTECTION SYSTEM IED SHEET 3	FIGURE 7.2-1c


TURBINE CONTROL VALVE FAST CLOSURE P.S. N005A [C]
AUX DEVICE
CR
TURBINE 1 [1] STOP VALVE CLOSURE P.S. N006A [A]
AUX DEVICE
CR
STOP VALVE CLOSURE
AUX DEVICE
CR
PERMISSIVE IF TCV FAST CLOSURE
AUX DEVICE
CR
PERMISSIVE WHEN TSV 4[3] IS CLOSED P.S. N006D [C]
AUX DEVICE
CR
PERMISSIVE WHEN
AUX DEVICE
TURBINE FIRST STAGE PRESS EXCEEDS SET POINT
AUX DEVICE
CR
PERMISSIVE WHEN TURBINE FIRST STAGE PRESS EXCEEDS SET POINT P.S. N003A [D]
AUX DEVICE
CR
PERMISSIVE IF MAN BYPASS SW IN NORMAL POSITION
AUX DEVICE
CR
TRIP COIL
CIRCUIT BREAKER CB3A [4A]
RECIRC PUMP TRIP SYSTEM A

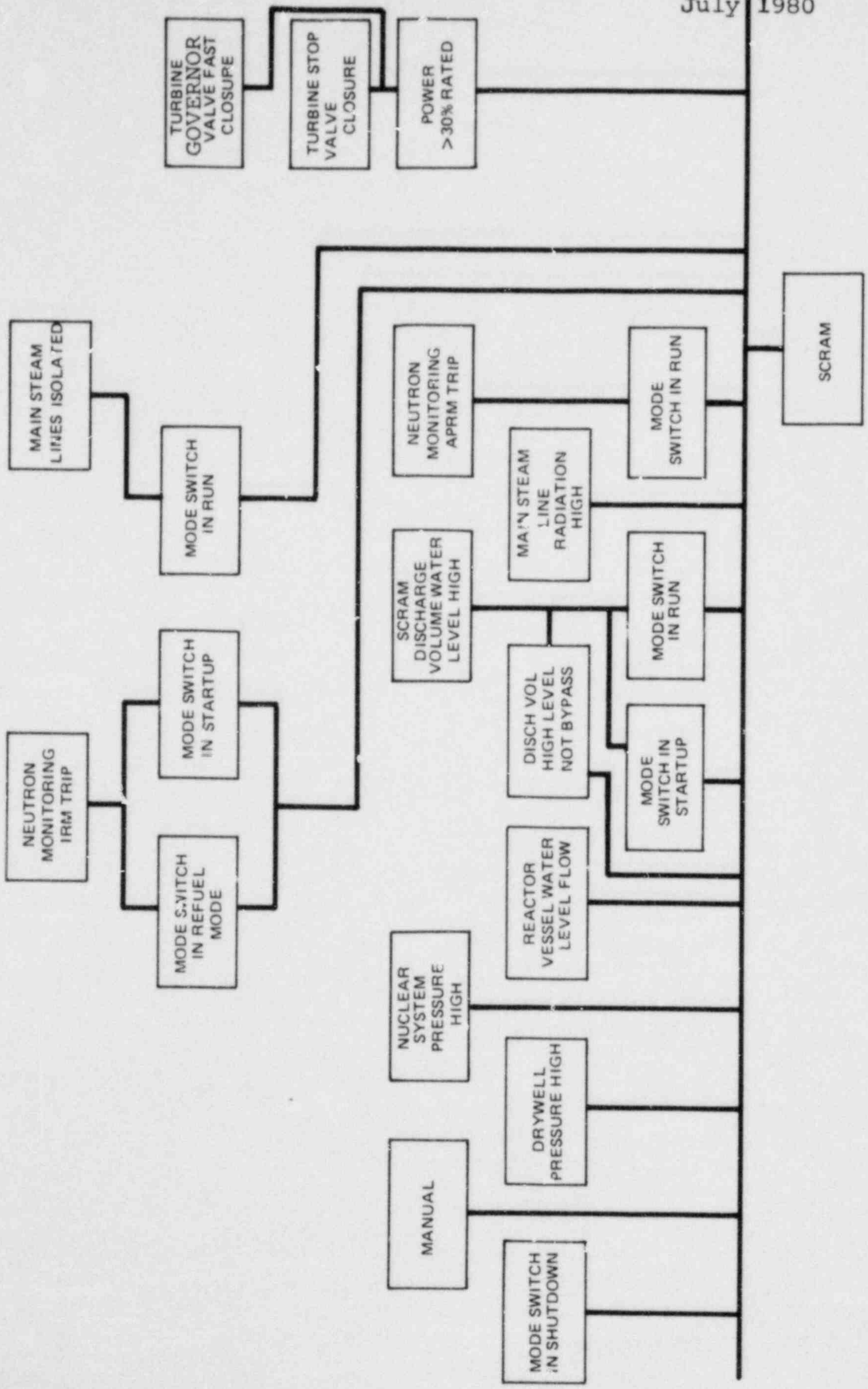
AMENDMENT NO. 10
July 1980

TRIP COIL

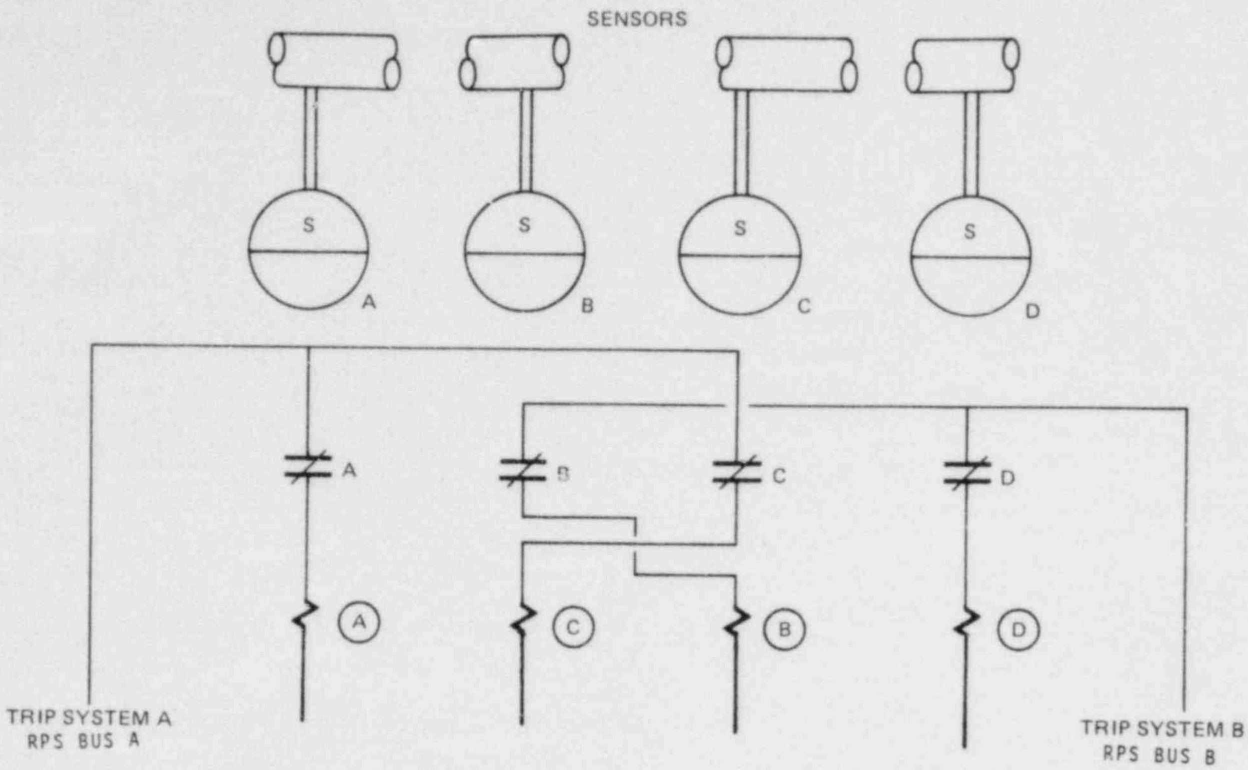
CIRCUIT BREAKER C85B [48]

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR PROTECTION SYSTEM IED SHEET 4	FIGURE 7.2-1d

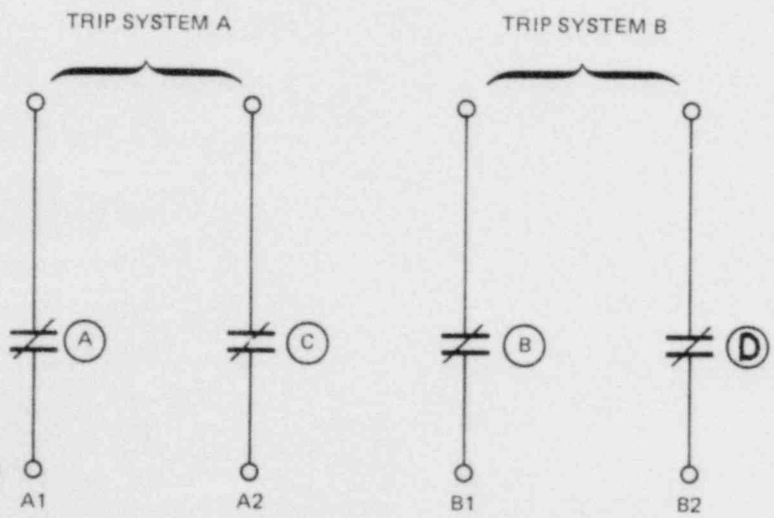
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR PROTECTION SYSTEM SCRAM FUNCTIONS	FIGURE 7.2-2



RPS SENSOR TRIP CHANNELS



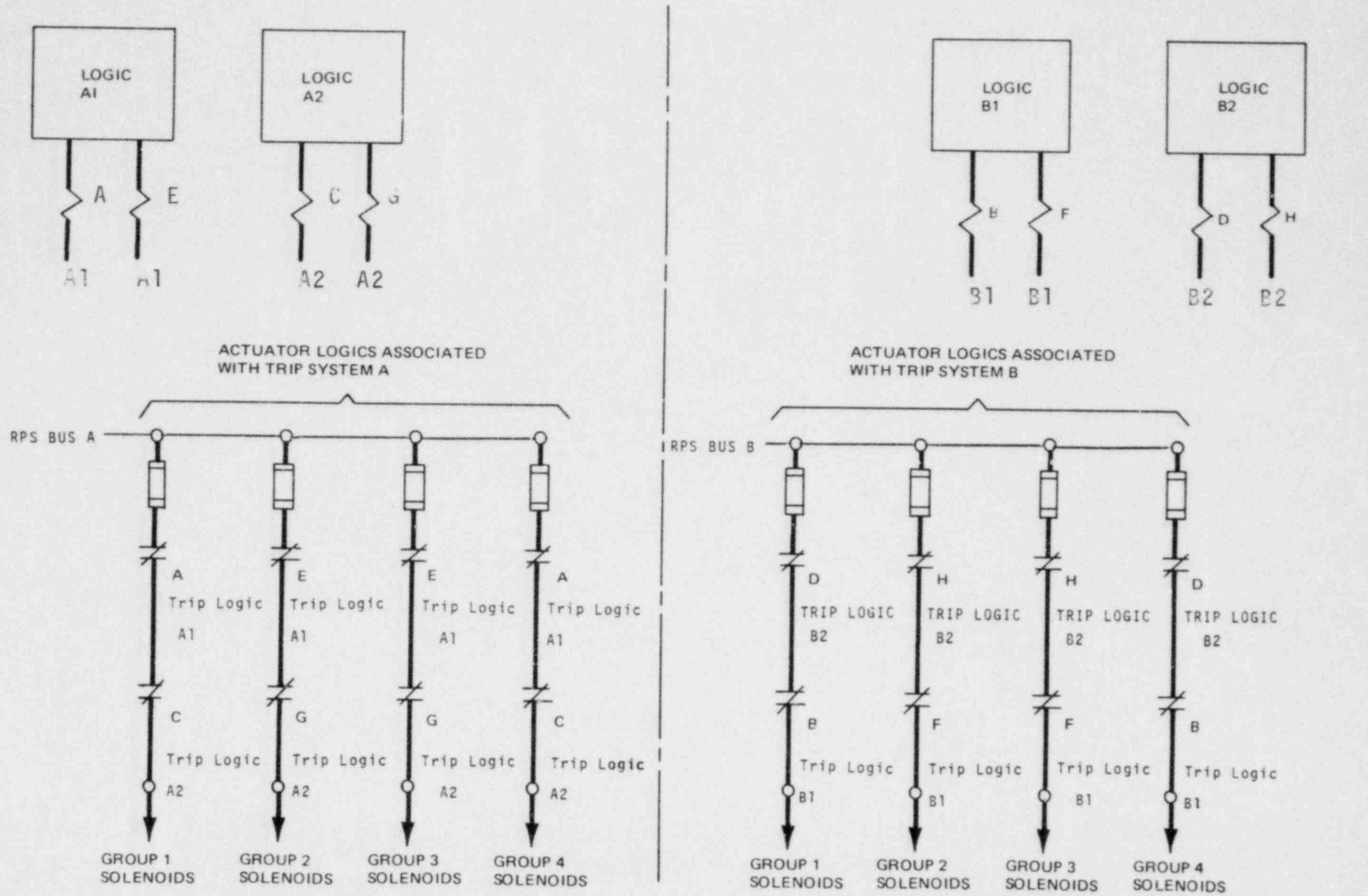
NOTE: CONTACTS SHOWN IN NORMAL (PLANT OPERATION) POSITION

RPS TRIP

CONFIGURATION FOR
SCRAM DISCHARGE VOLUME – HIGH WATER LEVEL –
TURBINE GOVERNOR VALVE FAST CLOSURE
REACTOR VESSEL LOW WATER LEVEL (LEVEL 3)

MAIN STEA' INE HIGH RADIATION
DRYWELL H H PRESSURE
REACTOR VE EL HIGH PRESSURE

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ARRANGEMENT OF RPS SENSOR TRIP CHANNELS AND TRIP LOGICS	FIGURE 7.2-3



NOTE: CONTACTS SHOWN IN NORMAL CONDITION (PLANT OPERATION) POSITION

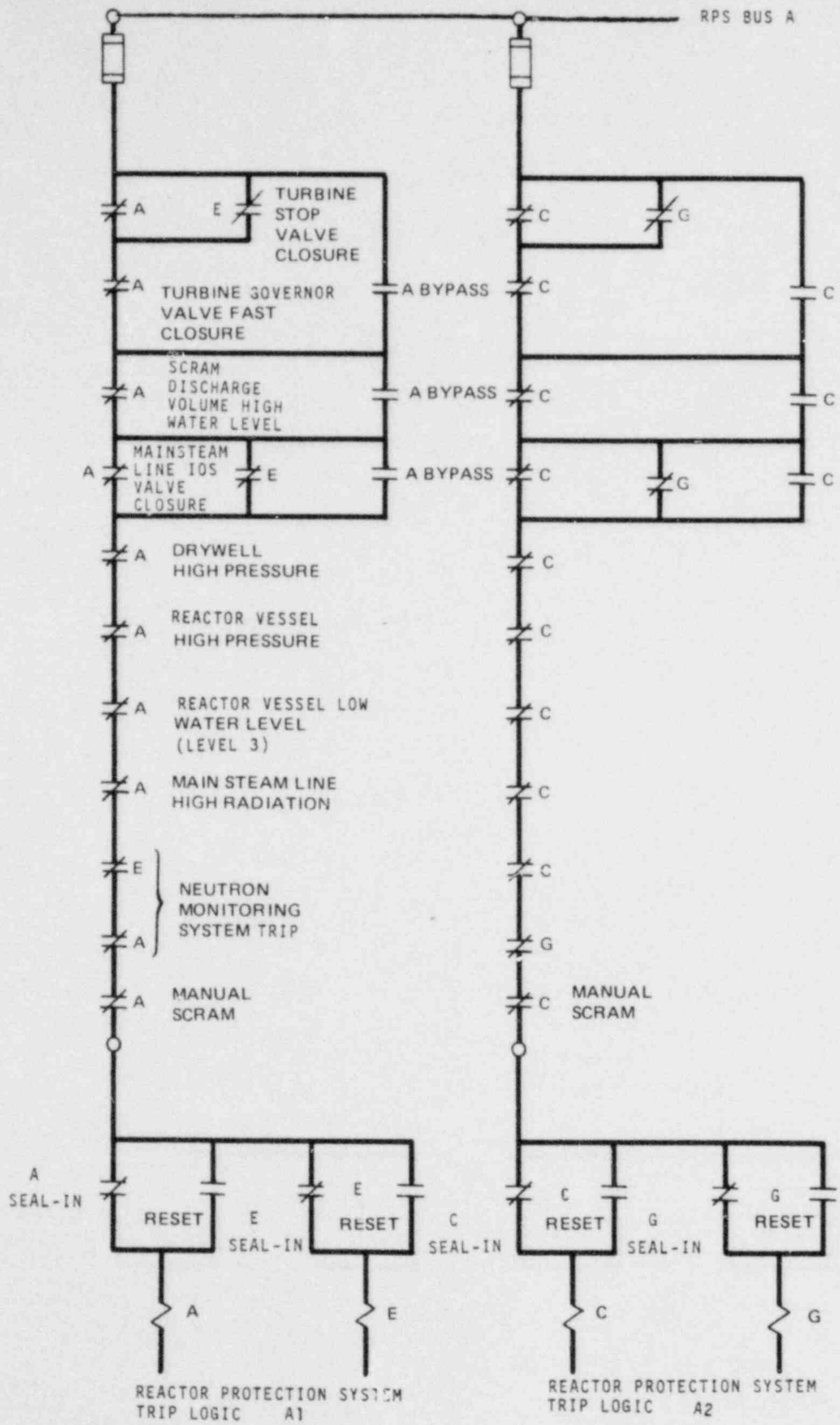
TRIP SYSTEM A

TRIP SYSTEM B

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

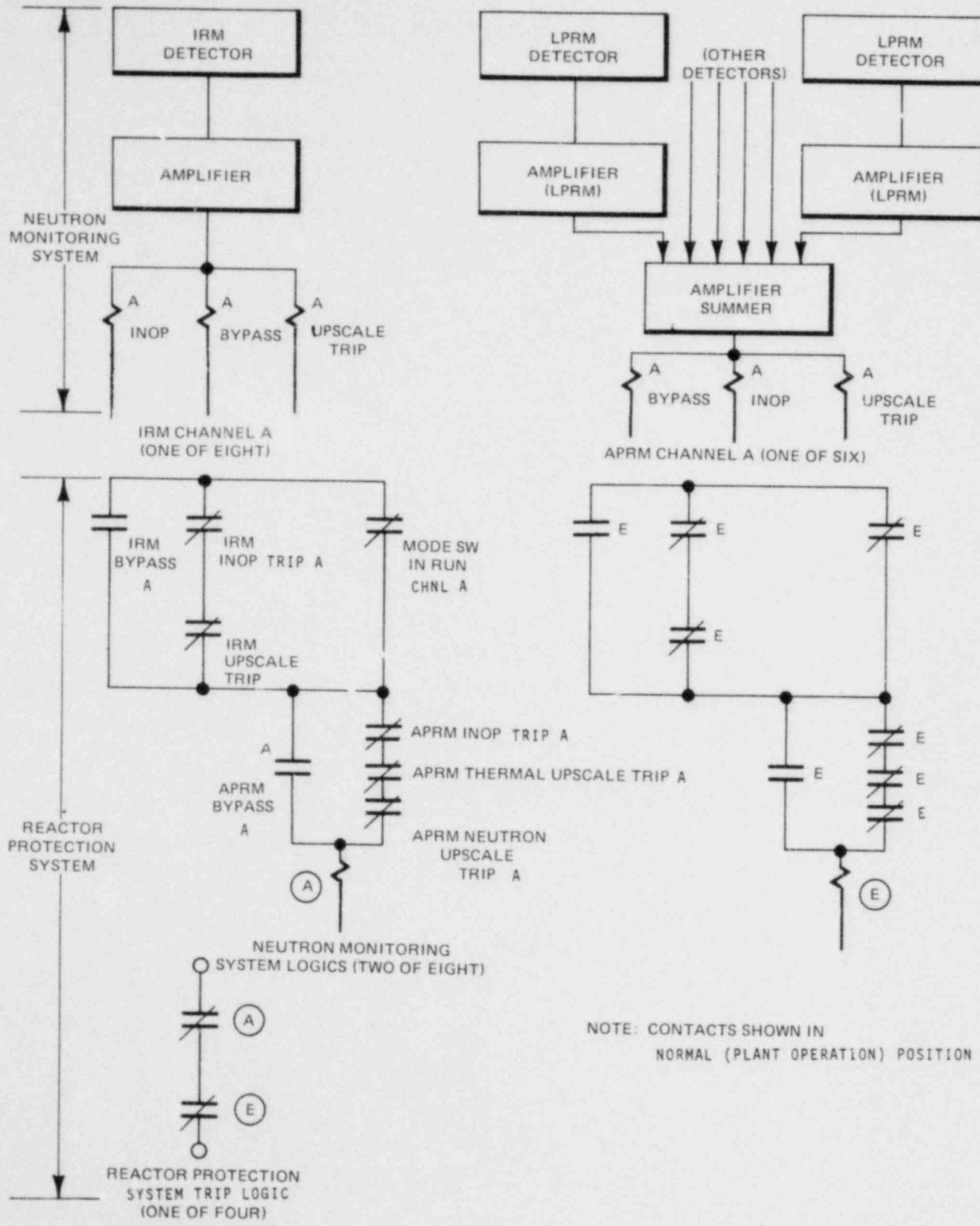
ARRANGEMENT OF ACTUATORS AND ACTUATOR LOGICS

FIGURE 7.2-4

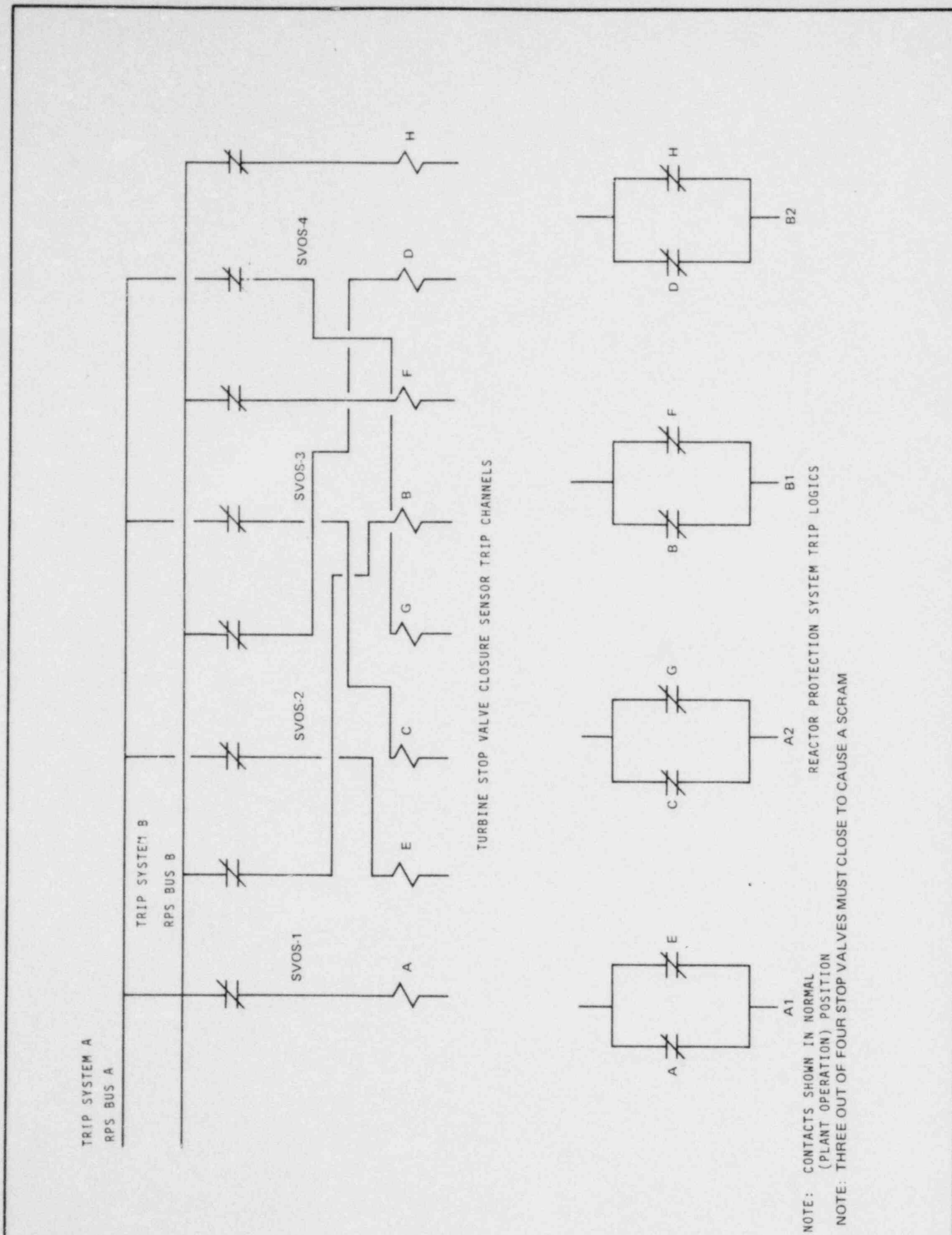


NOTE: CONTACTS SHOWN IN NORMAL (PLANT OPERATION) POSITION

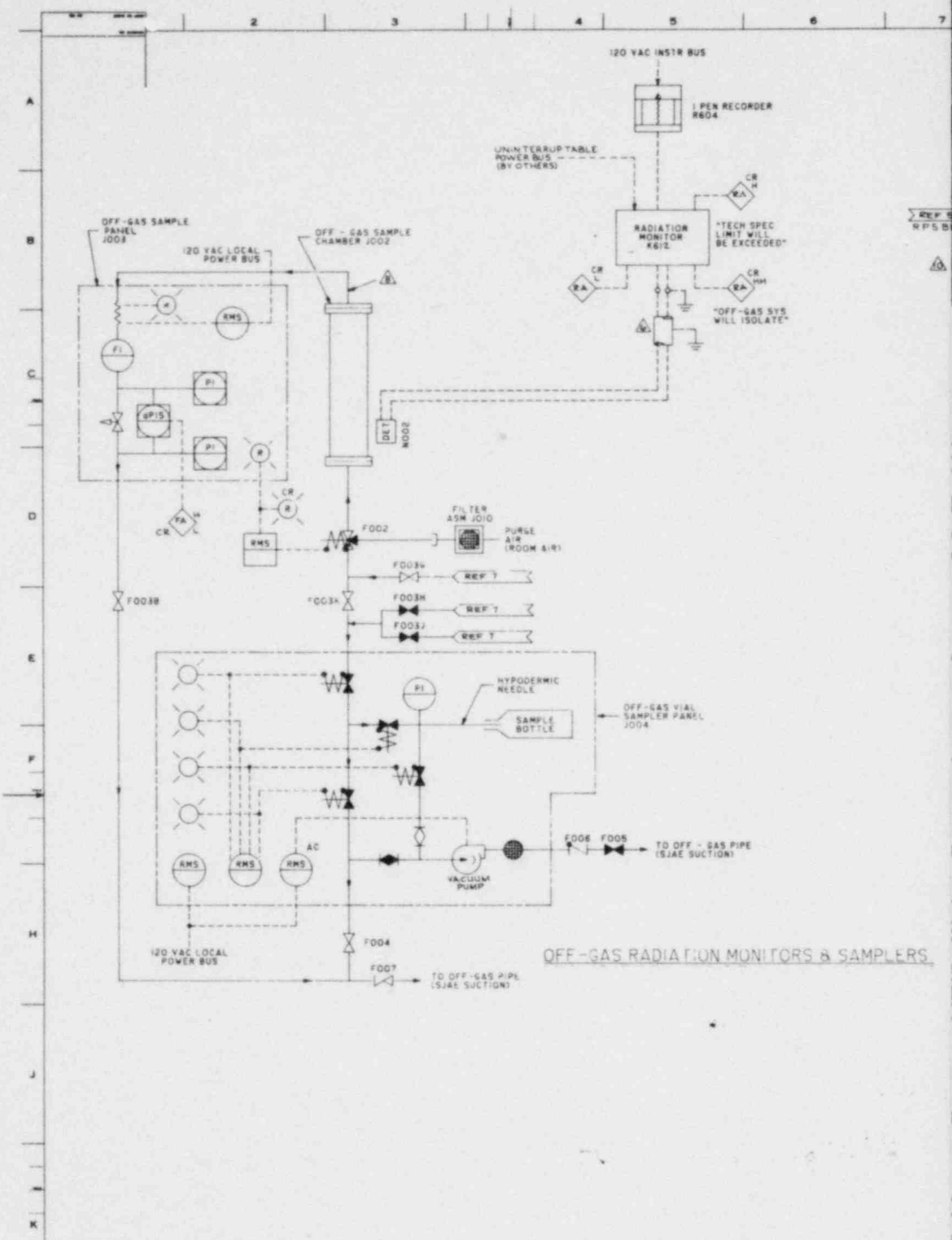
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	TRIP LOGICS IN ONE TRIP SYSTEM (SCHEMATIC)	FIGURE 7.2-5



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RELATIONSHIPS BETWEEN NEUTRON MONITORING SYSTEM AND REACTOR PROTECTION SYSTEM	FIGURE 7.2-6

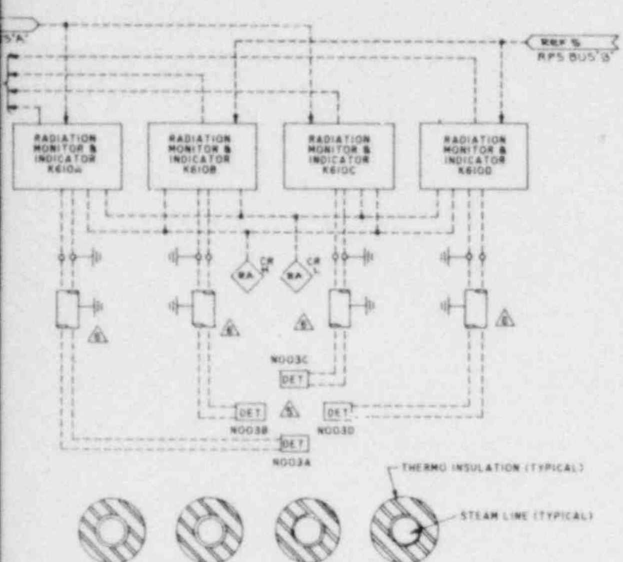


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONFIGURATION FOR TURBINE STOP VALVE CLOSURE REACTOR TRIP	FIGURE 7.2-7



OFF-GAS RADIATION MONITORS & SAMPLERS

AMENDMENT NO. 10
July 1980



MAIN STEAM LINE RADIATION MONITORS

NOTES:

1. THE OFF GAS VENT PIPE GAS SAMPLE LINE SHALL BE 1" X 0.058" WALL THICKNESS SEAMLESS STAINLESS STEEL TUBINE. THE TUBING MIN BEND RADIUS SHALL BE 20". THE TUBING LENGTH SHALL BE JOINED WITH SWAGELOK TYPE 1610-6-316 UNIONS. THE TUBING SHALL SLOPE SO THAT THE CONDENSATE WILL RUN TO DRAIN TEE.
2. A REMOVABLE SECTION SHALL BE PROVIDED NEAR THE ISOKINETIC PROBE FOR THE INSERTION OF A CHARCOAL FILTER HOLDER. THE FITTINGS ETC. SHALL PROVIDE SMOOTH TRANSITIONS WITHOUT DISCONTINUITIES OR REDUCING THE CROSS SECTIONAL AREA OF THE FLOW STREAM.
3. TEE SHALL BE UNION TEE SWAGELOK TYPE 1610-3-316.
4. ALARMS ARE ACTUATED BY RELAYS IN TRIP AUX. UNIT. DOWNSCALE ALARMS FOR LIQUID RADIATION MONITORS ARE ANNUNCIATED ON A SINGLE COMMON ANNUNCIATOR.
5. THE MAIN STEAM LINE RADIATION MONITOR DETECTORS (N003) SHALL BE LOCATED WITHIN THE STEAM LINE TUNNEL AS CLOSE AS PRACTICAL TO THE PRIMARY CONTAINMENT. THE DETECTORS SHALL BE ARRANGED SUCH THAT EACH DETECTOR WILL VIEW ALL STEAM LINES WITH APPROXIMATELY THE SAME RESPONSE. IT OS RECOMMENDED THAT THE DETECTOR OR DETECTOR ASSEMBLY BE FASTENED TO A ROD OR A PIPE AND INSERTED INTO SEALED PIPE WALLS FROM OUTSIDE THE STEAM TUNNEL. CAREFULLY ROUTE CABLES TO MINIMIZE HEAT EXPOSURE. NO LEAD SHIELDING IS REQUIRED.
6. ALL CABLES SHALL COMPLY WITH GE ENGR. SPEC REF 3.
7. ADDITIONAL ALARM IN RADWASTE BLDG (RAH) RADWASTE MONITOR ONLY.
8. DRAIN AT THE LOWER POINT OF OFF GAS SAMPLE LINE. SAMPLE LINE SHALL HAVE 2 MINUTE TRANSIT TIME TO ALLOW N[16] DECAY.
9. TWO OUT OF TWO HIGH HIGH RADIATION (RAHH) OR INOPERATIVE TRIP (CHANNELS A AND B) SHALL:
   -- SHUTDOWN AND ISOLATE (OUTBOARD VALVE) REACTOR BUILDING VENTILATION SYSTEM.
   -- INITIATE STANDBY GAS TREATMENT SYSTEM TRAIN B.
   -- CLOSE OUTBOARD PRIMARY CONTAINMENT PURGE AND VENT VALVES.
   TWO OUT OF TWO HIGH-HIGH RADIATION (RAHH) OR INOPERATIVE TRIP (CHANNELS C AND D) SHALL:
   -- SHUTDOWN AND ISOLATE (INBOARD VALVE) REACTOR BUILDING VENTILATION SYSTEM.
   -- INITIATE STANDBY GAS TREATMENT SYSTEM TRAIN A
   -- CLOSE INBOARD PRIMARY CONTAINMENT PURGE AND VENT VALVES.
   -- ANY ONE HIGH-HIGH RADIATION TRIP (RAHH) SHALL ALARM. SEE REF 7
10. ONE HIGH HIGH RADIATION (RAHH) OR INOPERATIVE TRIP OUT OF TWO ON TRIP SYSTEM "A" AND ONE HIGH-HIGH RADIATION (RAHH) OR INOPERATIVE TRIP OUT OF TWO ON TRIP SYSTEM "B" SHALL:
   -- CLOSE MAIN STEAM LINE ISOLATION VALVES.
   -- SCRAM REACTOR
   -- TURN OFF MECHANICAL VACUUM PUMP & CLOSE MECHANICAL VACUUM PUMP LINE VALVE.
   -- ANY ONE HIGH-HIGH RADIATION TRIP (RAHH) SHALL ALARM.
11. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET (SEE REF 9).
12. ANYONE UPSCALE TRIP (RAH) SHALL CLOSE BYPASS LINE VALVE, OPEN TREATMENT LINE VALVE AND ALARM.
13. ISOLATE OFF-GAS SYSTEM OUTLET AND DRAIN VALVES AND ALARM (REF 8).
14. *SUPPLIED AND MOUNTED BY OTHERS.*

SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NUMBERS

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. PIPING & INST. SYMBOLS	A42-1010
2. PROCESS RADIATION MONITORING DES. SPEC	D17-4010
3. SPECIAL WIRE & CABLE	A62-4010
4. RADWASTE SYSTEM P&ID	G11THRU G16-1010
5. REACTOR PROTECTION SYS IED	C71-C72-1010
6. NUCLEAR BOILER SYS FCD	B21-B22-1030
7. OFF-GAS SYSTEM P&ID	N62-N64-1010
8. OFF-GAS SYSTEM FCD	N62-1030
9. INST. DATA SH.	D17-3050

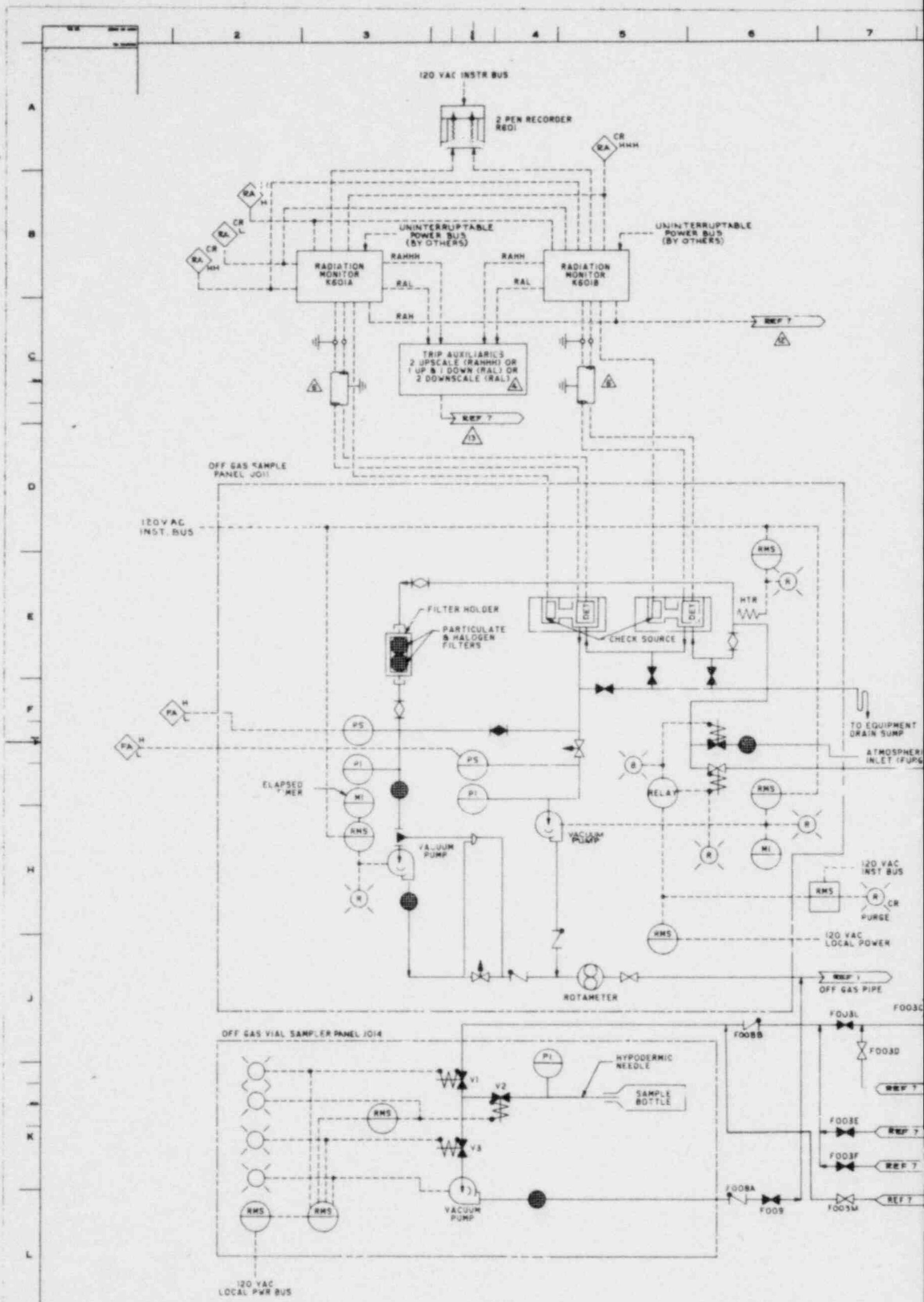
LEGEND:

SJAE	STEAM JET AIR EJECTOR
DET	DETECTOR
RAHHH	RADIATION ALARM HIGH HIGH HIGH
RAHH	RADIATION HIGH HIGH
RAH	RADIATION HIGH
RAL	DOWNSCALE OR INSTRUMENT TROUBLE
FAH/L	FLOW ALARM HIGH/LOW

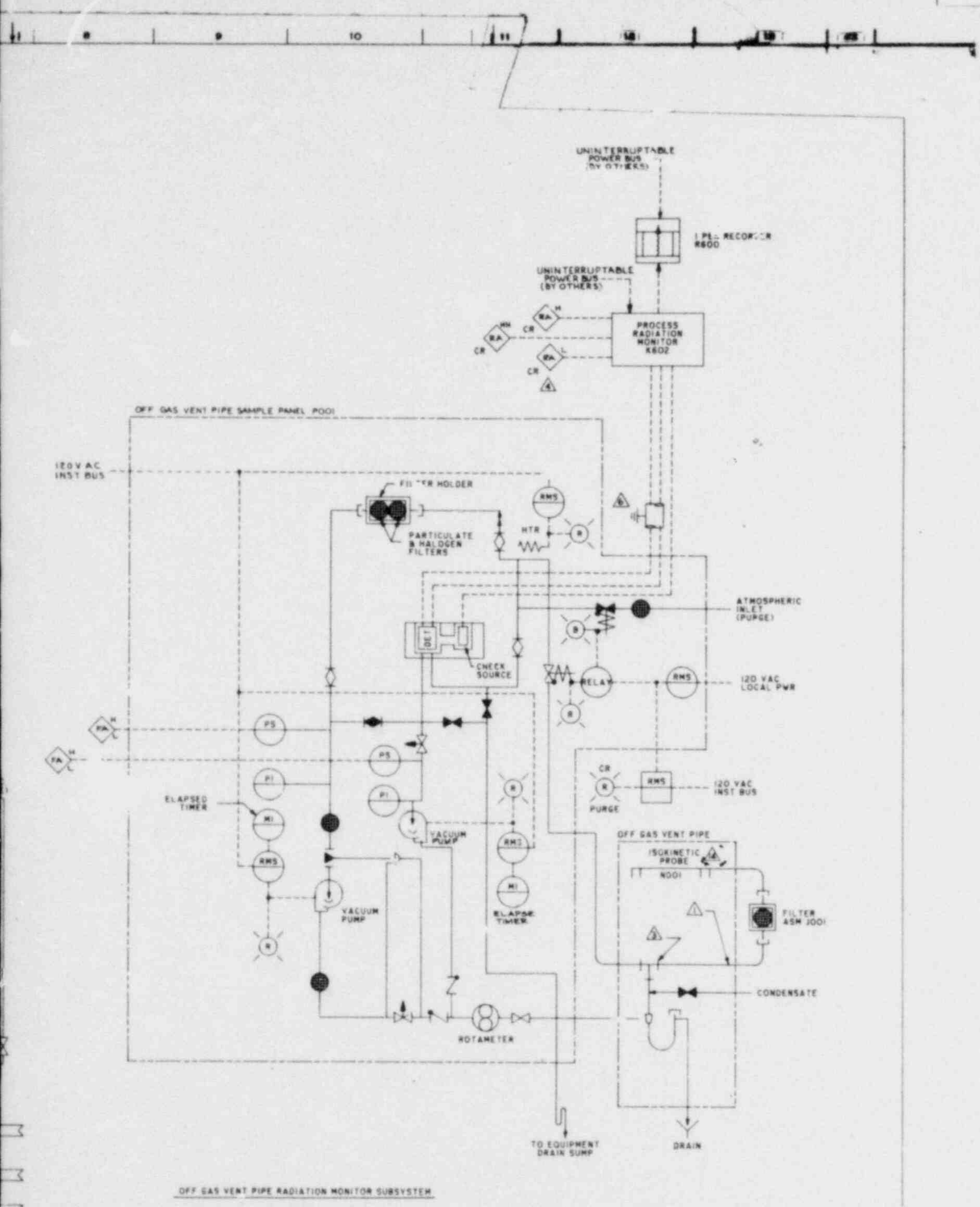
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 1

FIGURE 7.2-8a



OFF GAS RADIATION MONITORS & SAMPLERS SUBSYSTEM

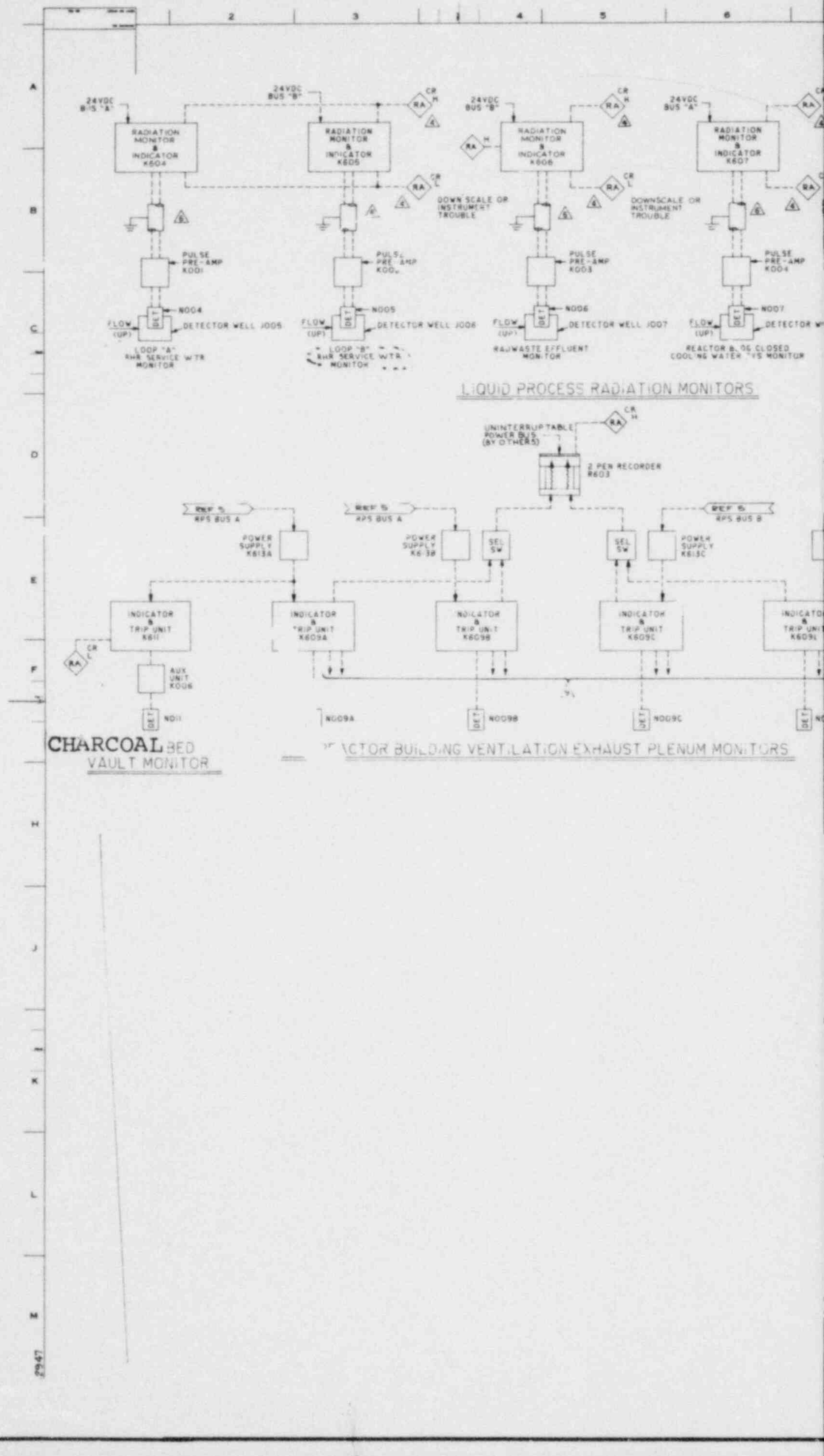


OFF GAS VENT PIPE RADIATION MONITOR SUBSYSTEM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

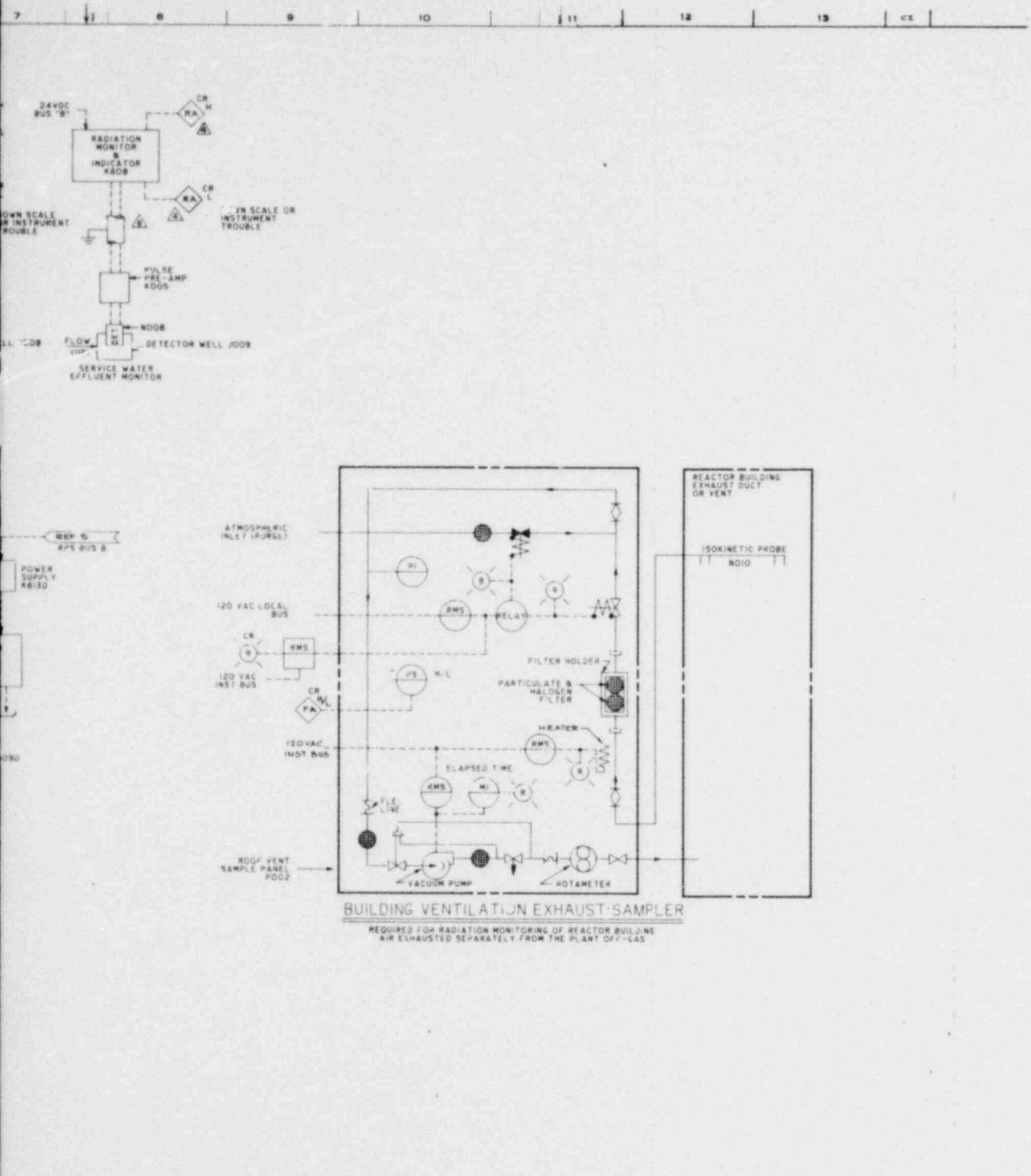
PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 2

FIGURE 7.2-8b

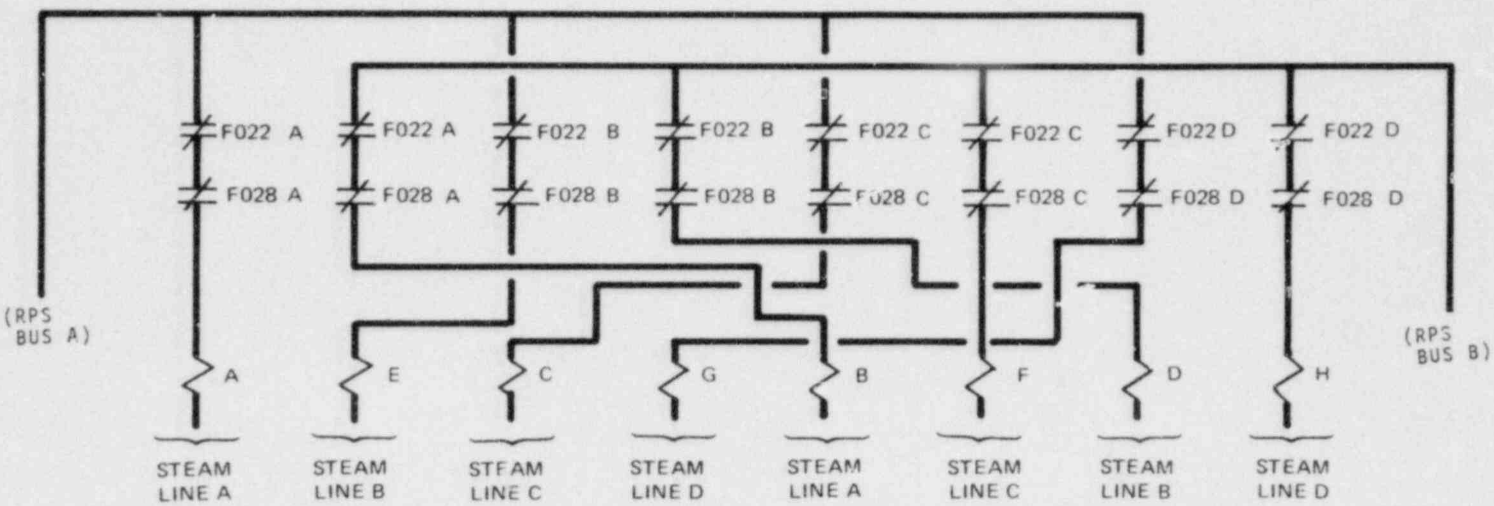


24VDC BUS "A"
24VDC BUS "B"
RADIATION MONITOR & INDICATOR K604
RADIATION MONITOR & INDICATOR K605
RADIATION MONITOR & INDICATOR K606
RADIATION MONITOR & INDICATOR K607
RA
CR H
CR L
DOWN SCALE OR INSTRUMENT TROUBLE
DOWNSCALE OR INSTRUMENT TROUBLE
PULSE PRE-AMP K001
PULSE PRE-AMP K003
PULSE PRE-AMP K004
DET
N004
N005
N006
N007
FLOW (UP)
DETECTOR WELL J005
DETECTOR WELL J006
DETECTOR WELL J007
LOOP "A" RHR SERVICE WTR MONITOR
LOOP "B" RHR SERVICE WTR MONITOR
RADWASTE EFFLUENT MONITOR
LIQUID PROCESS RADIATION MONITORS
UNINTERRUPTABLE POWER BUS (BY OTHERS)
2 PEN RECORDER R603
REF 5
RPS BUS A
RPS BUS B
POWER SUPPLY K613A
POWER SUPPLY K613C
SEL SW
INDICATOR & TRIP UNIT K611
INDICATOR & TRIP UNIT K609A
INDICATOR & TRIP UNIT K609B
INDICATOR & TRIP UNIT K609C
AUX UNIT K006
N011
N009A
N009B
N009C
CHARCOAL BED VAULT MONITOR
ACTOR BUILDING VENTILATION EXHAUST PLENUM MONITORS
A
B
C
D
E
F
H
J
K
L
M
2
3
4
5
6
2947

AMENDMENT NO. 10
July 1980

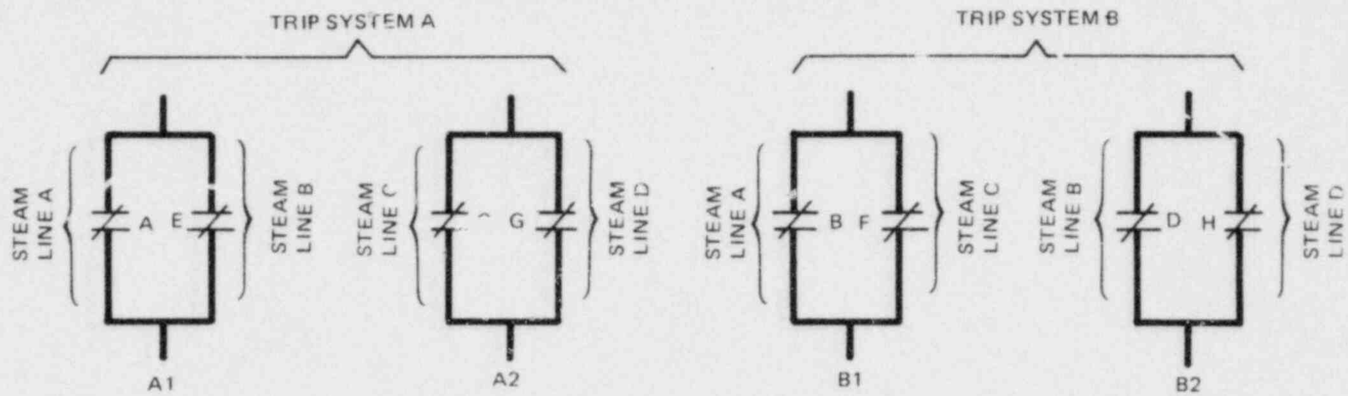


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 3	FIGURE 7.2- 8c



MAIN STEAM LINE ISOLATION TRIP SENSOR CHANNELS SEE NOTES BELOW

(SWITCH CONTACTS SHOWN IN POSITIONS
WHEN ISOLATION VALVES LESS THAN 10% CLOSED)



REACTOR PROTECTION SYSTEM TRIP LOGICS

(CONTACTS SHOWN IN NORMAL (PLANT OPERATION) POSITION

KEY

F022A · STEAM LINE A INBOARD VALVE	F022C · STEAM LINE C, INBOARD VALVE
F028A · STEAM LINE A OUTBOARD VALVE	F028C STEAM LINE C, OUTBOARD VALVE
F022B · STEAM LINE B INBOARD VALVE	F022D · STEAM LINE D, INBOARD VALVE
F020B · STEAM LINE B, OUTBOARD VALVE	F028D · STEAM LINE D, OUTBOARD VALVE

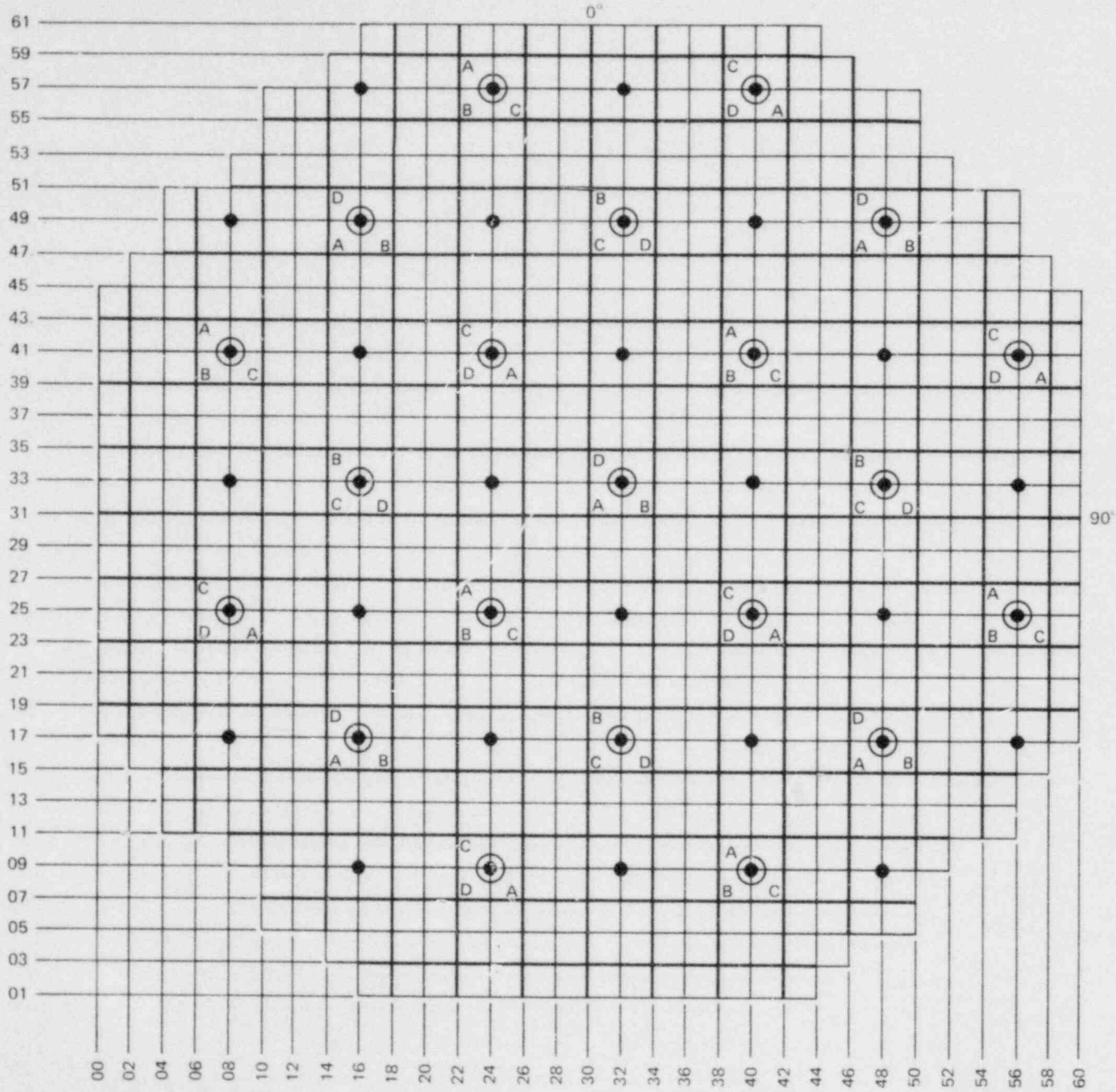
NOTE:
1. WIRING FOR THE TWO SWITCHES ON THE SAME VALVE IS PHYSICALLY SEPARATED.
2. ISOLATION OF THREE OR MORE STEAM LINES WILL CAUSE A SCRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONFIGURATION FOR MAIN STEAM LINE ISOLATION REACTOR TRIP

FIGURE 7.2-9

AMENDMENT NO. 10
July 1980



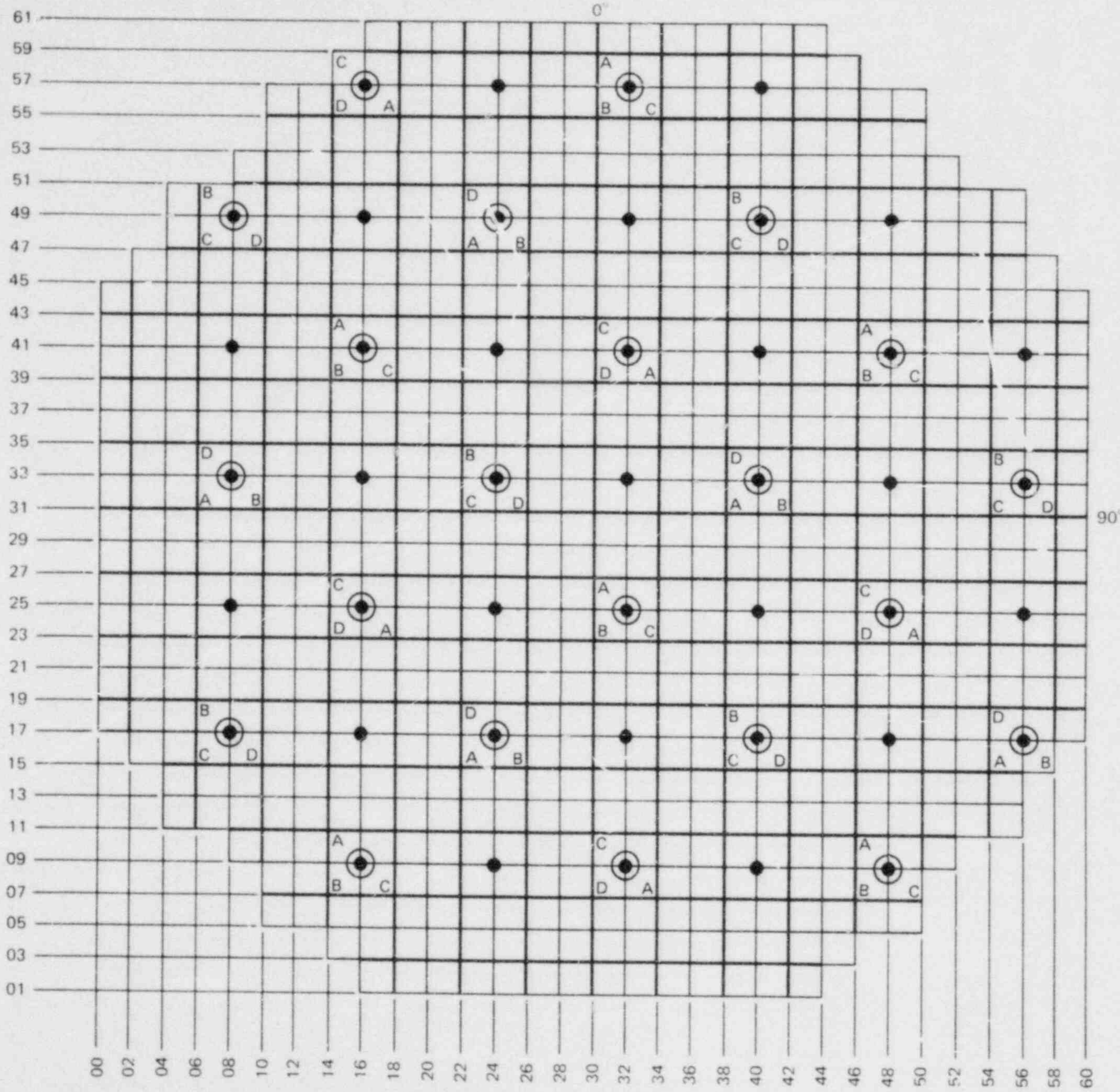
LPRM STRINGS PROVIDING INPUT TO APRM CHANNELS A, C, AND E
UPPER LEFT LETTER = INPUT FOR APRM CHANNEL A
LOWER LEFT LETTER = INPUT FOR APRM CHANNEL C
LOWER RIGHT LETTER = INPUT FOR APRM CHANNEL E

TRIP SYSTEM A

APRM SYSTEM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LPRM CHANNEL ARRANGEMENT IN THE CORE AND APRM CHANNEL ASSIGNMENTS (SHEET 1)	FIGURE 7.2-10a

AMENDMENT NO. 10
July 1980



LPRM STRINGS PROVIDING INPUT TO APRM CHANNELS B, D, AND F

UPPER LEFT LETTER = INPUT FOR APRM CHANNEL B
LOWER LEFT LETTER = INPUT FOR APRM CHANNEL D
LOWER RIGHT LETTER = INPUT FOR APRM CHANNEL F

TRIP SYSTEM B

APRM SYSTEM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LPRM CHANNEL ARRANGEMENT IN THE CORE AND APRM CHANNEL ASSIGNMENTS (SHEET 2)	FIGURE 7.2-10b

7.3 ENGINEERED SAFETY FEATURE SYSTEMS

7.3.1 DESCRIPTION

Section 7.3 describes the instrumentation and controls of the following plant Engineered Safety Features (ESF) systems:

a. Emergency Core Cooling Systems (ECCS)

b. Primary Containment and Reactor Vessel Isolation Control Systems (PCRVICS)

c. Main Steam Line Isolation Valve - Leakage Control System (MSLC)

d. RHRS-Containment Spray Cooling Mode (RCSCM)

e. RHRS-Suppression Pool Cooling Mode (RSPCM)

f. Standby Service Water System (SSW)

g. Main Control Room and Critical Switchgear Rooms HVAC System

h. Reactor Building Ventilation and Pressure Control System

i. Standby Gas Treatment System (SGTS)

j. Containment Instrument Air System (CIA)

k. Containment Atmosphere Control System (CAC)

The sources which supply power to the engineered safety feature systems originate from on-site AC and/or DC safety-related buses or, as in the case of the PCRVICS failsafe logic, from the non-safety-related RPS M-G sets. Refer to Chapter 8 for a complete discussion of the ESF systems power sources.

7.3.1.1 System Description

7.3.1.1.1 Emergency Core Cooling Systems (ECCS) - Instrumentation and Controls

The emergency core cooling systems are a network of the following subsystems. See 6.3.1 and 6.3.2.

a. High pressure core spray system (HPCS);

b. Automatic depressurization system (ADS);

c. Low pressure core spray system (LPCS);

d. Low pressure coolant injection (LPCI) mode of the residual heat removal system (RHRS).

The following plant variables are monitored and provide automatic initiation of the ECCS when these variables exceed predetermined limits:

1. Reactor Vessel Water Level

A low water level in the reactor vessel could indicate that reactor coolant is being lost through a breach in the reactor coolant pressure boundary and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes. Refer to Figure 7.3-9 (Nuclear Boiler P&ID) for a schematic arrangement of reactor vessel instrumentation.

2. Drywell Pressure

High pressure in the drywell could indicate a breach of the reactor coolant pressure boundary inside the drywell and that the core is in danger of becoming overheated as reactor coolant inventory diminishes.

7.3.1.1.1.1 High Pressure Core Spray (HPCS) System - Instrumentation and Controls

a. HPCS Function

The purpose of the HPCS is to provide high pressure reactor vessel core spray for small line breaks which do not depressurize the reactor vessel. In addition, HPCS is redundant to the RCIC system for mitigation of the consequences of various events listed in Appendix 15A. Refer also to 6.3.2.2.1.

b. HPCS Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 7.3-7 (HPCS P&ID). HPCS system component control logic is shown in Figure 7.3-8 (HPCS FCD) and Figure 7.3-4 (HPCS Power Supply FCD). Instrument specifications are listed in Tables 7.3-1 and 7.3-2. Plant Layout drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-7 (HPCS P&ID) and Figure 7.3-8 (HPCS and HPCS Power Supply FCD).

The HPCS is initiated automatically by either reactor vessel low water level (Trip Level 2) or drywell high pressure. The system is designed to operate automatically for at least 10 minutes without any actions required by the control room operator. Once initiated the HPCS logic seals-in and can be reset by the operator only when the initial conditions return to normal. Refer to Figure 7.3-8 for a schematic representation of the HPCS System Initiation logic.

Reactor vessel water level (Trip Level 2) is monitored by four redundant differential pressure switches. The switch contacts are arranged in a one-out-of-two twice logic arrangement to assure that no single event can prevent the initiation of the HPCS.

Initiation diversity is provided by drywell pressure which is monitored by four redundant pressure switches. The switches are electrically connected in a one-out-of-two twice logic arrangement to assure that no single instrument failure can prevent the initiation of the HPCS.

The HPCS components respond to an automatic initiation signal as follows (actions are simultaneous unless stated otherwise):

1. The HPCS Diesel Generator is signalled to start and its protective relays are bypassed. Once the Diesel is started it signals the start of its cooling water pump. See 6.3.1.1.8.2.

2. The HPCS pump motor is signalled to start.

3. The normally open pump suction from the condensate storage tank valve MO F001, is signalled to open.

4. The test return valves MO F010, MO F011 and MO F023 are signalled closed.

5. The HPCS injection valve MO F004 is signalled to open.

The HPCS pump discharge flow and pressure are monitored by pressure switches. If pump discharge pressure is normal but discharge flow is low enough that pump overheating may occur the minimum flow return line valve MO F012 is signalled open. The valve is automatically closed if flow is normal.

If the water level in the condensate storage tanks falls below a predetermined level, the suppression pool suction valve MO F015 automatically opens. When MO F015 is fully open the condensate storage tank suction valve MO F001 automatically closes. Two level switches are used to detect low water level in each of the condensate storage tanks. Either switch can cause automatic suction transfer. The suppression pool suction valve also automatically opens if high water level is detected in the suppression pool. Two level switches monitor suppression pool water level and either switch can initiate opening of the suppression pool suction valve. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other closes.

The HPCS provides makeup water to the reactor until the vessel water level reaches the high level trip (Trip Level 8) at which time the injection valve MOF004 is automatically closed. The pump will continue to run on minimum flow recirculation. The injection valve will automatically reopen if vessel level again drops to the low level (Trip Level 2) initiation point.

The HPCS pump motor and injection valve are provided with manual override controls. These controls permit the reactor operator to manually control the system following automatic initiation.

7.3.1.1.1.2 Automatic Depressurization System (ADS)-Instrumentation and Controls

a. ADS System Function

The automatic depressurization system is designed to provide automatic depressurization of the reactor vessel by activating seven safety/relief valves. These valves vent steam to the suppression pool in the event that the HPCS cannot maintain the reactor water level following a LOCA. ADS reduces the reactor pressure so that flow from the low pressure ECCS, LPCI system and LPCS, can inject into the reactor vessel in time to cool the core and limit fuel cladding temperature. Refer also to 6.3.2.2.2.

b. ADS Operation

Schematic arrangements of system mechanical equipment is shown in Figure 7.3-9 (Nuclear Boiler P&ID). ADS component control logic is shown in Figure 7.3-10 (Nuclear Boiler FCD). Instrumentation specifications are listed in Tables 7.3-3 and 7.3-4. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-9 (Nuclear Boiler P&ID) and Figure 7.3-10 (Nuclear Boiler FCD).

To prevent inadvertent actuation of the ADS two channels of logic for each ADS trip system (A & B) are used. Both channels must be activated to actuate an ADS trip system. Refer to Figure 7.3-7 for a schematic representation of the ADS initiation logic.

Each channel contains a single input from a drywell high pressure sensor. In addition, one channel includes two differential pressure sensor inputs monitoring reactor vessel low water level (Trip Level 3 and Trip Level 1). The second low water level trip (Trip Level 3) provides confirmation of a reactor vessel low water level condition. The other channel, in addition to drywell high pressure, includes a single reactor vessel low water level (Trip Level 1) input.

To assure that adequate makeup water is available after the vessel has been depressurized each logic channel includes a pump discharge pressure permissive signal indicating LPCI or LPCS system available for vessel water makeup. Any one of the three LPCI pumps or the LPCS pump is sufficient to permit automatic depressurization.

After receipt of the initiation signals and after a delay provided by timers, each of the two solenoid pilot air valves are energized. This allows pneumatic pressure from the accumulator to act on the air cylinder operator. Each ADS trip system timer can be reset manually to delay system initiation. If reactor vessel water level is restored by HPCS prior to the end of the time delay, ADS initiation will be prevented.

The ADS trip system A actuates the "A" solenoid pilot valve on each ADS relief valve. Similarly, the ADS trip system B actuates the "B" solenoid pilot valve on each ADS relief valve. Actuation of either solenoid pilot valve causes the ADS valve to open to provide depressurization.

Once initiated the ADS logic seals-in and can be reset by the control room operator only when either drywell pressure or vessel water level return to normal.

Two control switches (one for each trip system solenoid) are located in the main control room for each safety/relief valve associated with the ADS. Each switch controls one of the two solenoid pilot valves.

7.3.1.1.1.3 Low Pressure Core Spray (LPCS) - Instrumentation and Controls

a. LPCS Function

The purpose of the LPCS is to provide low pressure reactor vessel core spray following a loss-of-coolant accident when the vessel has been depressurized and vessel water level has not been restored by the HPCS. The LPCS is functionally diverse to the LPCI mode of the residual heat removal system. See 6.3.2.2.3.

b. LPCS Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 7.3-11 (LPCS P&ID). LPCS component control logic is shown in Figure 7.3-12 (LPCS FCD). Instrument specifications are listed in Tables 7.3-5 and 7.3-6. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-11 (LPCS P&ID) and Figure 7.3-12 (LPCS FCD).

The LPCS is initiated automatically by either reactor vessel low water level and/or drywell high pressure. The system is designed to operate automatically for at least 10 minutes without any actions required by the control room operator. Once initiated the LPCS logic seals-in and can be reset by the control room operator only when the initial conditions return to normal. Refer to Figure 7.3-12 for a schematic representation of the LPCS system initiation logic.

Reactor vessel water level (Trip Level 1) is monitored by two redundant differential pressure switches. To provide diversity drywell pressure is monitored by two redundant pressure switches. The vessel level switch contacts and the drywell pressure switch contacts are connected in a one-out-of-two twice logic arrangement so that no single instrument failure can prevent initiation of LPCS.

The LPCS components respond to an automatic initiation signal simultaneously (or sequentially as noted) as follows:

1. The Division 1 Diesel Generator is signalled to start,

2. The normally closed test return line to the suppression pool valve MO F012 is signalled closed,

3. When power (offsite or onsite) is available at the LPCS pump motor bus the LPCS pump is signalled to start,

4. Reactor pressure is monitored by a differential pressure switch which senses the pressure difference (vessel to LPCS) across the LPCS injection valve MO F005. When the differential pressure is low enough to protect the LPCS from overpressure and power is available to the pump motor bus, the injection valve is signalled to open.

The LPCS pump discharge flow is monitored by a differential pressure switch. When the pump is running and discharge flow is low enough that pump overheating may occur, the minimum flow return line valve MO F011 is opened. The valve is automatically closed if flow is normal.

The LPCS pump suction from the suppression pool valve MO F001 is normally open, the control switch is keylocked in the open position, and thus requires no automatic open signal for system initiation.

The LPCS pump and injection valve are provided with manual override controls. These controls permit the operator to manually control the system subsequent to automatic initiation.

7.3.1.1.1.4 RHRS - Low Pressure Coolant Injection (LPCI) Mode Instrumentation and Controls

a. LPCI Function

Low pressure coolant injection (LPCI) is an operating mode of the residual heat removal system (RHRS). The purpose of the LPCI system is to provide low pressure reactor vessel coolant makeup following a loss-of-coolant accident when the vessel has been depressurized and vessel water level is not restored by the HPCS. See 6.3.2.2.4.

b. LPCI Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 7.3-13 (RHR P&ID). LPCI component control logic is shown in Figure 7.3-14 (RHR FCD). Instrument specifications are listed in Tables 7.3-7 and 7.3-8. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-13 (RHR P&ID) and Figure 7.3-14 (RHR FCD).

The LPCI system is initiated automatically by either reactor vessel low water level and/or by drywell high pressure. The system is designed to operate automatically for at least 10 minutes without any actions required by the control room operator. Once initiated the LPCI logic seals-in and can be reset by the control room operator only when initial conditions return to normal. Refer to Figures 7.3-13 and 7.3-14 for a schematic representation of the LPCI A and the LPCS B/C initiation logic, respectively.

Reactor vessel water level (Trip Level 1) is monitored by two redundant differential pressure switches. To provide diversity drywell pressure is monitored by two redundant pressure switches.

To initiate the Division 2 LPCI (Loops B and C) the vessel level switch contacts and the two drywell pressure switch contacts are connected in a one-out-of-two-twice arrangement so that no single instrument failure can prevent initiation of LPCI.

The Division 1 LPCI (Loop A) receives its initiation signal from the LPCS logic.

The LPCI system components respond to an automatic initiation signal simultaneously (or sequentially as noted) as follows (the Loop A components are controlled from the Division 1 logic; the Loop B and C components are controlled from the Division 2 logic):

1. The Division 2 diesel generator is signalled to start from the Loop B and C initiation logic.

2. If normal auxiliary (offsite) power is available at the pump motor busses the LPCI Loop A, B, and C pumps are signalled to start. If offsite power is not available and the diesel generators are providing power to the pump motor busses sequential loading of the diesel generators is required. This is accomplished by delaying the start of the LPCI pumps A and B by 5 seconds while allowing the LPCS and LPCI C pumps to start immediately.

3. Reactor pressure is monitored by differential pressure switches which senses the pressure difference (vessel to LPCI system) across each LPCI injection valve MO F042 A, B, C. When the differential is low enough to protect the LPCI from overpressure and power is available at the associated pump motor bus, the injection valve is signalled to open,

4. The following normally closed valves are signalled closed to ensure proper system lineup:

   a) The RHR heat exchanger discharge to RCIC valves MO F026 A, B, and AO F065 AB,

   b) The RHR heat exchanger flush to suppression pool valves MO F011 A, B,

   c) The RHR heat exchanger steam pressure reducing valves AO F051 A, B,

   d) The RHR heat exchanger steam inlet isolation valves MO F052 A, B and MO F087 A, B,

   e) The test return line to the suppression pool valves MO F024 A, B and MO F021,

   f) The suppression pool spray valves MO F027 A, B.

5. The normally open heat exchanger bypass valves MO F048 A, B are signaled open. The open signal is automatically removed 10 minutes after system initiation to allow operator control of the valve for throttling purposes.

Each LPCI pump discharge flow is monitored by a differential pressure switch which, when the pump is running and following an 8-second time delay, opens the minimum flow return line valve MO F064 A, B, C if flow is low enough that pump overheating may occur. The valve is automatically closed if flow is normal. The 8-second time delay is provided to prevent reactor vessel inventory loss during the shutdown cooling mode of the RHRS (see 5.4.7.2.6(a)).

The three RHR pump suction from the suppression pool valves MO F004 A, B, C and the RHR heat exchanger inlet and outlet valves MO F047 A, B and MO F003 A, B have their control switches keylocked in the open position, and thus require no automatic open signal for system initiation.

The two series service water crosstie valves MO F093 and MO F094 have their control switches keylocked in the close position, and thus require no automatic close signal for system initiation.

The two series containment spray valves MO F016 A, B and MO F017 A, B, the two series RHR heat exchanger vent valves MO F073 A, B and F074 A, B and the RHR shutdown cooling mode suction valves MO F006 A, B are all normally closed and thus require no automatic close signal for system initiation.

The LPCI pump motors and injection valves are provided with manual override controls. These controls permit the operator to manually control the system subsequent to automatic initiation.

7.3.1.1.2 Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) - Instrumentation and Controls

a. PCRVICS Function

The PCRVICS includes the instrument channels, trip logics and actuation circuits that automatically initiate valve closure providing isolation of the primary containment and/or reactor vessel, and initiation of systems provided to limit the release of readioactive materials.

See 6.2.4 and Table 6.2-16 for a complete description of primary containment and reactor vessel process lines and isolation signals applied to each.

b. PCRVICS Operation

Schematic mechanical arrangements of containment isolation valves and other components initiated by PCRVICS are shown in Figures 7.3-13 (RHR P&ID), 7.3-9 (Nuclear Boiler P&ID), 3.2- 1 (RWCU P&ID), 3.2-3 (Reactor REURC P&ID), 11.2-2 (Equip. Drain Flow Diag.), 11.23 (Floor Drain Flow Diag.) and 3.2-16 (SGTS Flow Diag.). PCRVICS component control logic is shown in Figure 7.3-10 (Nuclear Boiler FCD), 7.3-14 (RHR FCD) and 7.3-1 (RWCU FCD). Instrument specification are listed in Tables 7.3-5 and 7.3-7. Plant Layout drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-10 (Nuclear Boiler FCD). Refer also to Figure 7.3-9 (Nuclear Boiler P&ID).

During normal plant operation, the isolation control system sensors and trip logic that are essential to safety are energized. When abnormal conditions are sensed, instrument contacts open and de-energize the trip logic and thereby initiate isolation. Once initiated the PCRVICS trip logics seal-in and may be reset by the operator only when the initial conditions return to normal.

The PCRVICS trip logic provides isolation signals to the Main Steam Line Isolation Valves AO F022 A, B, C, D and AO F028 A, B, C, D; to the Main Steam Line Drain Valves MO F016, F067 A, B, C, D, and MO F019; to the Reactor Water Sample Valves MO F019 and F020; to the RHR Shutdown Cooling System Valves MO F008, F009, F023, F040, F049, F053 A, B, F099 A, B; to the RHR Sample Line Valves Solenoid Operated Valve (SOV) F060 A, B and F075 A, B; to the Reactor Water Cleanup System Valves MO F001 and F004; to the Drywell Equipment Drain Valves AO F019 and F020; to the Drywell Floor Drain Valves AO F003 and F004; and to isolate the TIP system valves.

Each main steam line isolation valve (MSIV) has two control solenoids. Each solenoid receives inputs from two redundant ogics. A signal from either can deenergize the solenoid. For any one valve to close automatically, both of its solenoids must be de-energized.

The main steam line isolation valve logic has a minimum of four redundant instrument channels for each measured variable. One channel of each variable is connected to one trip logic. One group of redundant logics (A, C) is used to control one solenoid of both inboard and outboard valves of all four main steam lines, and the other group of redundant logics (B, D) is used to control the other solenoid of both inboard and outboard valves. The four PCRVICS trip logics are arranged in a one-out-of-two twice logic combination (Trip Logic A or C and B or D). Refer to Figure 7.3-2.

The main steam line drain valves, drywell equipment and floor drain valves, reactor water sample valves, the reactor water cleanup system, and residual heat removal system isolation valves also operate in pairs. The inboard valves close if both of the Division 1 isolation logics (A and B) are tripped, and the outboard valves close if both of the Division 2 logics (C and D) are tripped. Refer to Figure 7.3-3.

The PCRVICS also provides signals to start the standby gas treatment system; to remove nonessential loads from essential busses; and to isolate the reactor building ventilation system, and the primary containment purge and vent system.

The following variables provide inputs to the PCRVICS logics for initiation of reactor vessel and drywell isolation, as well as the initiation or trip of other plant functions when predetermined limits are exceeded. Combinations of these variables, as necessary, provide initiation of various isolating and initiating functions as described in Table 6.2-16 and below:

1. Reactor Vessel Low Water Level

A low water level in the reactor vessel could indicate that reactor coolant is being lost through a breach in the reactor coolant pressure boundary and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes.

Reactor vessel low water level initiates closure of various valves. The closure of these valves is intended to isolate a breach of the pipelines, conserve reactor coolant by closing off process lines, and limit the escape of radioactive materials from the primary containment through process lines that communicate with the primary coolant boundary or primary containment.

Two reactor vessel low water level isolation trip settings are used to complete the isolation of the primary containment and the reactor vessel. The first (and higher) reactor vessel low water level isolation trip (Trip Level 3) initiates closure of all RHR system isolation valves. The main steam lines are left open to allow the removal of heat from the reactor core. The second (and lower) reactor vessel low water level isolation trip (Trip Level 2) completes the isolation of the primary containment and reactor vessel by initiating closure of all other isolation valves and also provides inputs to logic which trips or initiates other plant equipment.

Reactor vessel water level is monitored by four redundant differential pressure switches. Each provides a low water level input to one of the four PCRVICS trip logics.

Diversity of trip initiation for pipe breaks inside of primary containment is provided by drywell high pressure.

2. Drywell High Pressure

High pressure in the drywell could indicate a breach of the reactor coolant pressure boundary inside the drywell and that the core is in danger of becoming overheated as reactor coolant inventory diminishes.

Drywell pressure is monitored by four redundant pressure switches. Each switch provides an input to one of the four trip logics.

3. Main Steam Line-High Radiation

The main steam line radiation monitoring senses the gross release of fission products from the fuel and initiates appropriate actions to limit fuel damage and contain the released fission products.

Four redundant detectors monitor the gross gamma radiation from the main steam lines. Each provides an input to one of the four PCRVICS trip logics.

Each monitoring channel consists of a gamma-sensitive ion chamber and a log radiation monitor. Each log radiation monitor has three trip circuits. One upscale trip circuit is used to initiate scram (see 7.2.1) and containment isolation, and alarm. The second circuit is used for an alarm and is set at a level below that of the upscale trip circuit used for scram and isolation. The third circuit is a downscale or inoperative trip that actuates an instrument trouble alarm in the control room and produces an isolation and scram trip signal.

When the main steam line radiation level exceeds a predetermined value, PCRVICS initiates closure of all main steam line isolation valves, main steam line drain valves and reactor water sample valves. The offgas system mechanical vacuum pump is tripped, and the mechanical vacuum pump lines are isolated.

4. Main Steam Line-Tunnel and Pipe Routing in Turbine Building High Ambient Temperature and Differential Temperature

High ambient temperature in the tunnel and pipe routing areas in the turbine building in which the main steam lines are located outside of the primary containment could indicate a leak in a main steam line. Such a leak may also be indicated by high differential temperature between the outlet and inlet ventilation air for these areas. The automatic closure of valves prevent the excessive loss of reactor coolant and the release of a significant amount of radioactive material from the reactor coolant pressure boundary.

Four redundant main steam line high ambient temperature sensors are provided in the main steam tunnel and four in the steam line area of the turbine building. Four redundant differential temperature sensors monitor the outlet and inlet ventilation air ducts of the main steam line tunnel. Each main steam line trip isolation logic is de-energized by high ambient temperature in the main steam tunnel on the turbine building and by high differential temperature in the tunnel inlet/outlet ventilation air.

When an increase in main steam line tunnel ambient or differential temperature is detected, trip signals initiate closure of all main steam line isolation and drain valves.

Diversity of trip initiation signals for main steam line tunnel ambient temperature and high differential temperature is provided by main steam line high flow, and steam line low pressure instrumentation.

5. Main Steam Line-High Flow

Main steam line high flow could indicate a breach in a main steam line. Automatic closure of isolation valves prevents excessive loss of reactor coolant and release of significant amounts of radioactive material from the reactor coolant pressure boundary.

Sixteen redundant differential pressure switches, four for each main steam line, monitor the main steam line flow. Four differential pressure switches for each main steam line provide inputs to each of the four trip logics.

When a significant increase in main steam line flow is detected, trip signals initiate closure of all main steam line isolation and drain valves.

6. Main Turbine Inlet - Low Steam Pressure

Low steam pressure at the turbine inlet while the reactor is operating could indicate a malfunction of the nuclear system pressure regulator in which the turbine governor valves or turbine bypass valves become fully open, and causes rapid depressurization of the reactor vessel. From reduced power, the rate of decrease of nuclear system saturation temperature could exceed the allowable rate of change of vessel temperature. A rapid depressurization of the reactor vessel while the reactor is near full power could result in undesirable differential pressures across the channels (around some fuel bundles) of sufficient magnitude to cause mechanical deformation of channel walls. Such depressurizations, without adequate preventive action, could require thorough vessel analysis or core inspection prior to returning the reactor to power operation.

Four redundant pressure sensors, one for each main steam line, monitor main steam line pressure and each provides an input to one of the four trip logics.

When a significant decrease in main steam line pressure is detected, the PCRVICS initiates closure of all main steam line isolation and drain valves.

The main steam line low pressure trip is bypassed by the reactor mode switch in the Shutdown, Refuel, and Startup modes of reactor operation. In the Run mode, the low pressure trip function is operative.

7. Reactor Building Ventilation Radiation Monitor - Instrumentation and Controls

The reactor building ventilation monitoring consists of four sensor and trip units. Each channel has two trips. The upscale trip indicates high radiation and the downscale trip indicates instrument trouble.

The reactor building ventilation radiation monitor senses reactor building exhaust to the elevated release point. In the event that radiation levels exceed predetermined limits the intake and exhaust dampers are closed.

8. Reactor Water Cleanup (RWCU) System-High Differential Flow

High differential flow in the reactor water cleanup system could indicate a breach of the reactor coolant pressure boundary of the cleanup system. The flow at the inlet to the system (suction from Recirc. lines) is compared with the flow at the outlets of the system (flow return to feedwater or flow to the main condenser and/or radwaste).

Two redundant differential flow sensors compare the reactor water cleanup system inlet-outlet flow. Each of the flow monitoring sensors provides an input to one of the two (inboard or outboard) logic trip channels.

When an increase in reactor water cleanup system differential flow is detected, the PCRVICS initiates closure of all reactor water cleanup system isolation valves.

Diversity of trip initiation signals for reactor water cleanup system line break is provided by instrumentation for reactor water level, differential flow, and ambient or differential temperature in RWCU equipment areas.

The reactor water cleanup system high differential flow trip is bypassed by an automatic timing circuit during normal reactor water cleanup system surges. This time delay bypass prevents inadvertent system isolations during system operational changes.

9. Reactor Water Cleanup (RWCU) System-Area High Ambient Temperature and Differential Temperature

High temperature in the equipment room areas of the reactor water cleanup system could indicate a breach in the reactor coolant pressure boundary in the cleanup system.

Six redundant ambient temperature and six differential temperature sensors monitor the reactor water cleanup system area temperatures. Three redundant ambient and three redundant differential temperature circuits are associated with each of the two trip logics. Six redundant ambient temperature elements are located in the following locations: Pump Room, Filter/Demin. Room, and Heat Exchanger Room. Six pairs of temperature elements are located in the ventilation supply and exhaust areas of the above locations.

When a significant increase in reactor water cleanup system area ambient or differential temperature is detected the PCRVICS initiates closure of all reactor water cleanup system isolation valves.

The output trip signal of each sensor initiates a logic trip and closure of either the inboard or outboard reactor water cleanup system isolation valve.

Diversity of trip initiation signals for high differential temperature is provided by two pair of differential temperature elements and associated differential temperature switches (DTS) for each reactor water cleanup system area. Each pair of temperature elements and its differential temperature switch are associated with one of two logic channels.

10. RHR System-Area High Ambient Temperature and Differential Temperature

High temperature in the equipment room areas of the RHR system could indicate a breach in the reactor coolant pressure boundary in the RHR system.

Four redundant ambient temperature and four redundant differential temperature sensors monitor the RHR system area temperatures. Two ambient and two differential temperature sensors are associated with one trip logic. The remaining temperature channels are associated with the other trip logic. The ambient temperature elements are located in each RHR equipment area. Four pairs of temperature elements are located in the ventilation supply and ventilation exhaust of each RHR equipment area.

When an increase in RHR system area ambient temperature or differential temperature is detected the PCRVICS initiates closure of all RHR system isolation valves.

The output trip signal of each sensor initiates a trip logic and closure of either the inboard or outboard RHR system isolation valve.

Diversity of trip initiation signals for RHR line break is provided by ambient temperature, differential temperature, and cooling flow instrumentation. An increase in space temperature, differential temperature, or flow will initiate RHR system isolation.

11. RHR System - Flow Rate Monitoring

High flow in the RHR system suction line from the reactor vessel could indicate a breach in the reactor coolant pressure boundary in the RHR system.

Two redundant differential pressure switches, one for each trip logic, monitor the RHR shutdown cooling mode suction line.

The output trip signal of each sensor initiates a logic trip and closure of either the inboard or outboard RHR system isolation valve.

12. Main Condenser Vacuum Trip

The main turbine condenser low vacuum signal could indicate a leak in the condenser. Initiation of automatic closure of various valves will prevent excessive loss of reactor coolant and the release of significant amounts of radioactive material.

Four redundant vacuum switches monitor the main condenser vacuum. Each switch provides an input to one of the four trip logics.

When a signficant decrease in main condenser vacuum is detected, the PCRVICS initiates closure of all main steam line isolation and drain valves.

Main condenser low vacuum trip can be bypassed manually when the turbine stop valve is less than 90% open.

7.3.1.1.3 Main Steamline Isolation Valve Leakage Control System (MSLCS) - Instrumentation and Control

a. MSLCS Function

The MSLCS is designed to minimize the release of fission products which could bypass the standby gas treatment system after the postulated LOCA. This is accomplished by directing the leakage through the closed main steamline isolation valves to bleed lines which pass the leakage flow into an area served by the standby gas treatment system. See 6.7.

b. MSLCS Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-25 (MSLC Flow Diag.). MSLC system component control logic is shown in Figure 7.3-15 (MSLC Control Logic Diag.). Instrument specifications are listed in Tables 7.3-27 and 7.3-28. Plant Layout drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-25 (MSLC Flow Diag.) and Figure 7.3-15 (MSLC Control Logic Diag.).

The MSLCS is manually actuated after a LOCA has occurred, provided that the reactor and steamline pressures are below the pressure permissive setpoints and the inboard MSIVs are fully closed. The outboard and inboard subsystems are provided with one remote manual initiating switch each.

When the inboard system is initiated, the exhaust blower FN-1 is actuated and the bleed valves V-2A,B,C,D and V-3A,B,C,D and the bypass valves V-1A,B,C,D are opened, heaters are actuated and timers are initiated. If the steamline pressure is greater than 5 psig after one minute, the bleed valves will close. If the pressure is not excessive, the bleed valves will remain open. After another minute, the bypass valve is closed. The flow is thus routed through the flow element. Within the next 30 seconds, a third timer allows flow to be monitored and the bleed valves to be closed, if necessary, for high flow.

When the outboard system is initiated depressurization valves V-9 and V-10 are opened and the exhaust blower FN-2 is activated. When the steam lines have depressurized to approximately atmospheric pressure, the depressurization branch valves V-4 and V-5 are closed and flow is diverted to the blower suction.

7.3.1.1.4 RHRS-Containment Spray Cooling Mode (RCSCM) - Instrumentation and Controls

a. Containment Spray Cooling Mode Function

The containment spray cooling mode is an operating mode of the Residual Heat Removal System. It is designed to condense steam in the suppression chamber air volume and/or the drywell atmosphere following a LOCA. See 6.5.2.

b. Containment Spray Cooling Mode Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 7.3-13 (RHR P&ID). RHR system component control logic is shown in Figure 7.3-14 (RHR FCD). Instrument specifications are listed in Tables 7.3-25 and 7.3-26. Plant Layout drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-13 (RHR P&ID) and Figure 7.3-14 (RHR FCD)

The containment spray cooling mode is initiated by the control room operator by diverting LPCI flow to either the suppression pool or the drywell by opening valves MO F027A,B or MO F016A,B, and MO F017A,B.

The following conditions must exist before the operator can initiate a containment spray cooling loop:

1. The LOCA signal which automatically initiated LPCI must still exist,

2. Drywell high pressure is monitored by two redundant pressure switches. One of the two switches must indicate high pressure,

3. The operator must close the LPCI injection valve MO F042 A, B.

7.3.1.1.5 RHRS-Suppression Pool Cooling Mode (RSPCM) Instrumentation and Controls

c. SPCM Function

The suppression pool cooling mode is an operating mode of the residual heat removal system. It is designed to prevent suppression pool temperature from exceeding predetermined limits following a reactor blowdown of the ADS or safety/relief valves.

d. SPCM Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 7.3-13 (RHR P&ID). Component control logic is shown in Figure 7.3-14 (RHR FCD). Instrument specifications are listed in Tables 7.3-23 and 7.3-24. Plant layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-13 (RHR P&ID) and Figure 7.3-14 (RHR FCD).

The suppression pool cooling mode is initiated by the Control Room operator either during normal plant operation or following a LOCA, when the suppression pool temperature monitoring system (see 7.6) indicates that pool temperature may exceed a predetermined limit.

During normal plant operation the operator initiates the SPCM as follows:

1. The RHR Pump (A or B) is started. The standby service water pump is started and the RHR heat exchanger service water discharge valve MO F068 A, B is opened automatically when the RHR pump starts.

2. The RHR test return line valve MO F024 A, B is opened.

3. The RHR heat exchanger inlet and outlet valves MO F047 A, B and MO F003A, B are keylocked open. The heat exchanger bypass valve MO F048 A, B and valve MO F024 A, B are throttled as necessary.

Subsequent to a LOCA the operator initiates the SPCM as follows:

1. Once reactor vessel water level has been restored, the LPCI flow must be terminated by closing the LPCI injection valve MO F042 A, B. Closing the injection valve causes the LOCA initiation logic to be overridden and allows operator control of the system.

2. The RHR test return line valve MO F024 A, B control logic also has LOCA signal override provisions. This allows the operator to open the valve.

3. The RHR heat exchanger inlet and outlet valves MO F047 A, B and MO F003 A, B are keylocked open. The heat exchanger bypass valve MO F041 A, B, after a time delay (a ten minute timer keeps this valve open following a LOCA) and valve MO F024 A, B are throttled as necessary.

7.3.1.1.6 Standby Service Water (SSW) System - Instrumentation and Controls

a. SSW Function

The standby service water system provides cooling water to the diesel generators, the RHR heat exchangers, the HPCS, RCIC, LPCI, and LPCS auxiliary equipment (room cooler, pump cooler) and the essential HVAC chillers. See 9.1.

b. SSW System Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 9.1-4 (SSW Flow Diag.). SSW component control logic is shown in Figure 7.3-17 (SSW Control Logic Diag.). Instrument specifications are listed in Tables 7.3-21 and 7.3-22. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 9.1-4 (SSW Flow Diag.) and Figure 7.3-17 (SSW Control Logic Diag.).

The SSW system is automatically initiated as follows:

1. The Division 1 SSW pump P-1A is started automatically if water level in the spray pond is sufficient and when either the RHR A pump, the LPCS pump, or the Division 1 diesel generator is started,

2. The Division 2 SSW pump P-1B is started automatically if water level in the spray pond is sufficient and when either the RHR B pump, RHR C pump, the Division 2 diesel generator, or the RCIC pump is started,

3. The HPCS service water pump C002 is automatically started when the HPCS pump is started.

Once the service water pumps are started the following occurs:

1. The RHR heat exchanger service water discharge valves MO F068A,B are signalled open,

2. After the SSW pumps discharge pressure exceeds a minimum value the pump discharge valves V-2A,2B and V-29 are signalled to open,

The SSW pumps are automatically tripped if spray pond water level becomes too low.

7.3.1.1.7 Main Control Room and Critical Switchgear Rooms HVAC System - Instrumentation and Controls

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-19 (HVAC Flow Diag.). Component control logic is shown in Figure 7.3-14 (HVAC Control Logic Diag.). Instrument specifications are listed in Tables 7.3-19 and 7.3-20. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-19

(HVAC Flow Diag.) and Figure 7.3-14 (HVAC Control Logic Diagram).

For a complete description of the Main Control Room and Critical Switchgear Rooms HVAC Instrumentation and Controls refer to 9.4.1.

7.3.1.1.8 Containment Atmosphere Control (CAC) System - Instrumentation and Controls

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-17 (CAC Flow Diag.). CAC component control ogic is shown in Figure 7.3-21 (CAC Control Logic Diag.). nstrument specifications are listed in Tables 7.3-17 and .3-18. Plant Layout Drawings and Electrical Schematics are dentified in 1.7. Operator Information Displays are shown in Figure 3.2-17 (CAC Flow Diag.) and Figure 7.3-21 (CAC Control Logic Diag.).

For a complete description of the CAC System Instrumentation and Controls refer to 6.2.5.

7.3.1.1.9 Standby Gas Treatment System (SGTS) - Instrumentation and Controls

For a complete description of the SGTS Instrumentation and Controls refer to 6.5.1.

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-16 (SGTS Flow Diag.). SGTS component control logic is shown in Figure 7.3-19 (SGTS Control Logic Diag.). Instrument specifications are listed in Tables 7.3-15 and 7.3-16. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-16 (SGTS Flow Diag.) and Figure 7.3-19 (SGTS Control Logic Diag.).

7.3.1.1.10 Reactor Building Ventilation and Pressure Control System - Instrumentation and Controls

a. System Function

The reactor building ventilation and pressure control system automatically maintains subatmospheric pressure of 1/4" water gage in the reactor building atmosphere. See 9.4.2.

b. System Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-18 (HVAC React. Bldg. Flow Diag.). System component control logic is shown in Figure 7.3-19 (SGTS Control

Logic Diag.). Instrument specifications are listed in Tables 7.3-15 and 7.3-16. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-18 (HVAC React. Bldg. Flow Diag.) and Figure 7.3-19 (SGTS Control Logic Diag.).

The differential pressure is monitored by eight redundant differential pressure transmitters, four in Division 1 and four in Division 2, which measure the differential pressure from the exterior of four sides of the reactor building to the fuel pool area. The signal indicating the least differential pressure from the four differential pressure transmitters in one division is selected and is used to control the position of a damper of that division in the normal reactor building exhaust fan units or upon the initiation of the standby gas treatment system by containment isolation signals high drywell pressure, low reactor water level, or reactor building exhaust high radiation, the reactor building ventilation and pressure control system then controls reactor building pressure by controlling the standby gas treatment system fan units. (See 6.5.1).

7.3.1.1.11 Containment Instrument Air (CIA) System

a. CIA System Function

The purpose of the containment instrument air system is to provide uninterruptable divisional instrument air to essential ADS valve accumulators inside primary containment and non-safety-related instrument air supplies to other valves inside containment as shown in Figure 3.2-21. The system consists of a safety-related portion, which is comprised of two divisions, and a non-safety-related portion. During normal operation, the non-safety-related portion of the system provides control air to ADS accumulators and other valves.

In the event of failure of the non-safety-related portions of the system, which is indicated by low header pressure and detected by two redundant pressure switches, the safety-related portion automatically maintains header pressure from two nitrogen bottle sources. The non-safety-related portion of the system is isolated from the safety-related portion upon detection of failure of the non-safety-related portion. See 9.3.1.2.2.

b. CIA System Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-21 (CIA Flow Diag.). CIA component control logic is shown in Figure 7.3-20 (CIA Control Logic Diag.). Instrument specifications are listed in Tables 7.3-13 and

7.3-14. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-21 (CIA Flow Diag.) and Figure 7.3-20 (CIA Control Logic Diag.).

The containment instrument air system is always in operation. The instrumentation and controls of the system perform the following functions:

1. Monitor CIA system header pressure
2. Monitor CIA system compressor operation
3. Isolate the non-safety-related portion of system in the event of failure in this portion
4. Maintain CIA system header pressure in the event of item 3 by sequentially opening nitrogen bottles.

7.3.1.2 Design Basis

The ESF systems are designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are presented in that chapter.

a. Variables Monitored to Provide Protective Action

The following variables are monitored in order to provide protective actions to the ESF systems:

1. HPCS
   a) Reactor Vessel Low Water Level (Trip Level 2)
   b) Drywell High Pressure
2. ADS
   a) Reactor Vessel Low Water Level (Trip Level 3)

b) Reactor Vessel Low Water Level (Trip Level 1)

c) Drywell High Pressure

3. LPCS and LPCI

a) Reactor Vessel Low Water Level (Trip Level 1)

b) Drywell High Pressure

4. PCRVICS

a) Reactor Vessel Low Water Level (Trip Level 3)

b) Reactor Vessel Low Water Level (Trip Level 2)

c) Main Steam Line High Radiation

d) Main Steam Line Area High Ambient and Differential Temperature

e) Main Steam Line High Flow

f) Turbine Inlet Low Steam Pressure

g) Elevated Release Point High Radiation

h) RWCU High Differential Flow

i) RWCU Area High Ambient Temperature and Differential Temperature

j) RHR Area High Ambient Temperature and Differential Temperature

k) RHR Suction High Flow

l) Main Condenser Low Vacuum

5. MSLCS

a) Reactor Vessel Low Pressure

6. RCSCM

a) Drywell High Pressure

7. RSPCM

   a) Suppression Pool Temperature

   b) Drywell High Pressure

   c) Reactor Vessel Low Water Level (Trip Level 1)

8. SSWS

   a) RHR, LPCS, RCIC, or Diesel Generator Start

9. Main Control Room and Critical Swgr. Room HVAC

   a) Duct Chlorine High Concentration

   b) Remote Air Intake High Radiation

   c) High Room Temperature

10. Reactor Bldg. Ventilation and Pressure Control

    a) Reactor Bldg. to Fuel Pool Area Differential Pressure

11. SGTS

    a) Reactor Vessel Low Water Level (Trip Level 2)

    b) Drywell High Pressure

    c) Reactor Building Ventilation High Radiation

12. CIAS

    a) Instrument Air Header Low Pressure

13. CACS

    a) Reactor Vessel Low Water Level (Trip Level 2)

    b) Drywell High Pressure

The plant conditions which require protective action involving the ESF systems are described in Chapter 15 and Appendix 15A.

b. Location and Minimum Number of Sensors

See Chapter 16 for the minimum number of sensors required to monitor safety-related variables. There are no sensors in the ESF systems which have a spatial dependence.

c. Prudent Operational Limits

Operational limits for each safety-related variable trip setting are selected with sufficient margin so that a spurious ESF system iniation is avoided. It is then verified by analysis that the release of radioactive materials, following postulated gross failures of the fuel or the nuclear system process barrier, is kept within acceptable bounds.

d. Margin

The margin between operational limits and the limiting conditions of operation of ESF systems are listed and the bases stated in Chapter 16, "Technical Specifications".

e. Levels

Levels requiring protective action are established in Chapter 16, "Technical Specifications".

f. Range of Transient, Steady State, and Environmental Conditions

Refer to Tables 3.11-1 through 3.11-5 and 3.1.2.1.4.1 for environmental conditions. Refer to 8.2.1 and 8.3.1 for the maximum and minimum range of energy supply to ESF instrumentation and controls. All ESF instrumentation and controls are specified and purchased to withstand the effects of energy supply extremes.

g. Malfunctions, Accidents, and Other Unusual Events Which Could Cause Damage to Safety System

Chapter 15, "Accident Analysis" describes the following credible accidents and events; floods, storms, tornados, earthquakes, fires, LOCA, pipe break outside containment. Each of these events is discussed below for the ESF systems.

1. Floods

The buildings containing ESF systems components have been designed to meet the PMF (Probable Maximum Flood) at the site location. This ensures that the buildings will remain watertight under PMF conditions including wind-generated wave action and wave runup. For a discussion of internal flooding protection refer to 3.4.1.4.1.2, 3.4.1.5.2, and 3.6.

2. Storms and Tornados

The buildings containing ESF systems components have been designed to withstand meteorological events described in 3.3.

3. Earthquakes

The structures containing ESF systems components have been seismically qualified as described in 3.7 and 3.8, and will remain functional during and following a safe shutdown earthquake (SSE). Seismic qualification of instrumentation and electrical equipment is discussed in 3.10.

4. Fires

To protect the ESF systems in the event of a postulated fire, the redundant portions of the systems are separated by fire barriers. If a fire were to occur within one of the sections or in the area of one of the panels, the ESF systems functions would not be prevented by the fire. The use of separation and fire barriers ensures that even though some portion of the systems may be affected, the ESF systems will continue to provide the required protective action. A fire detection system using heat detectors and product of combustion detectors is provided in PGCC floor sections and in panels containing ESF systems components mounted on these floor sections. A Halon fire suppression system is provided in the same areas.

5. LOCA

The ESF systems components located inside the drywell and functionally required during and/or following a LOCA have been environmentally qualified to remain functional as discussed in 3.11 and indicated in Table 3.11-1.

6. Pipe Break Outside Secondary Containment

This condition will not affect the ESF systems. Refer to 3.6.

7. Missiles

Protection for safety-related components is described in 3.5.

h. Minimum Performance Requirements

Minimum performance requirements for ESF instrumentation and controls are provided in Chapter 16, "Technical Specifications".

7.3.1.3 Final System Drawings

The final system drawings including:

1. Process and Instrumentation Diagrams (P&ID)/Flow Diagrams

2. Functional Control Diagrams (FCD)Control Logic Diagrams

have been provided for the ESF systems in this section.

ESF systems electrical interconnection and schematic diagrams are in 1.7.

Functional and architectural design difference between the PSAR and FSAR are listed in Table 1.3-8.

7.3.2 ANALYSIS

7.3.2.1 ESF Systems - Instrumentation and Controls

Chapter 15, "Accident Analysis," and Chapter 6, "Engineered Safety Feature Systems," evaluate the individual and combined capabilities of the ESF systems.

The ESF systems are designed such that a loss of instrument air, a plant load rejection, or a turbine trip will not prevent the completion of the safety function.

7.3.2.1.1 Conformance to 10 CFR 50 Appendix A

The following is a discussion of conformance to those General Design Criteria which apply specifically to the ESF systems. Refer to 7.1.2.2 for a discussion of General Design Criteria which apply equally to all safety-related systems.

a. Criterion 33

See 7.3.1.1.1 (HPCS).

b. Criterion 34

See 7.3.1.1.6 (SSW).

c. Criterion 35

See 7.3.1 (ECCS) and 7.3.1.1.6 (SSW).

d. Criterion 36, 37, 39, 40, 42, 43, 45, 46, and 61 - Fluid Systems

See 7.3.2, Regulatory Guide 1.22.

e. Criterion 38

See 7.3.1.1.4 (RCSCM), 7.3.1.1.5 (SPCM) and 7.3.1.1.6 (SSW)

f. Criterion 41

See 7.3.1.1.11 (CAC), and 7.3.1.1.9 (SGTS).

g. Criterion 44

See 7.3.1.1.6 (SSW).

h. Criterion 60 and 61

See 7.3.1.1.9 (SGTS) and 7.3.1.1.3 (MSLCS).

i. Criterion 64

See 7.3.1.1.2 (PCRVICS) and 7.3.1.1.8 (Reactor Bldg. Vent. Radiation).

7.3.2.1.2 Conformance to IEEE Standards

The following is a discussion of conformance to those IEEE standards which apply specifically to the ESF systems. Refer to 7.1.2.3 for a discussion of IEEE standards which apply equally to all safety-related systems.

IEEE 279-1971 Criteria for Protection Systems for Nuclear Power Generating Stations:

1. General Functional Requirement (IEEE 279-1971, Paragraph 4.1)

The ESF systems automatically initiate the appropriate protective actions, whenever the parameters described in 7.3.1.2.A reach predetermined limits, with precision and reliability assuming the full range of conditions and performance discussed in 7.3.1.2.

2. Single Failure Criterion (IEEE 297-1971, Paragraph 4.2)

ESF systems are not required to meet single failure criteria on an individual system (division) basis. However, on a network basis, the single failure criteria does apply to assure the completion of a protective function. Redundant sensors, wiring, logic, and actuated devices are physically and electrically separated such that a single failure will not prevent the protective function. Refer to 8.3.1.4 for a complete description of the WNP-2 separation criteria.

3. Quality Components (IEEE-279-1971, Paragraph 4.3)

For a discussion of the quality of ESF system components and modules refer to 3.11.

4. Equipment Qualification (IEEE 279-1971, Paragraph 4.4)

Vendor certification requires that the sensors associated with each of the RPS trip variables, manual switches, and trip logic components perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in conjunction with the existing field experience with these components in this application, will serve to qualify these components.

Qualification tests of the relay panels are conducted to confirm their adequacy for this service. In-situ operational testing of these sensors, channels, and the entire protection system will be performed during the preoperational test phase.

For a complete discussion of RPS Equipment Qualification refer to 3.5, 3.6, 3.10, and 3.11.

5. Channel Integrity (IEEE 279-1971, Paragraph 4.5)

For a discussion of ESF systems channel integrity under all extremes of conditions described in 7.3.1.2 refer to 3.10, 3.11, 8.2.1, and 8.3.1.

6. Channel Independence (IEEE 279-1971, Paragraph 4.6)

ESF systems channel independence is maintained through the application of the WNP-2 separation criteria as described in 8.3.1.4.

7. Control and Protection Interaction (IEEE 279-1971, Paragraph 4.7)

There are no ESF system and control system interactions.

8. Derivation of System Inputs (IEEE 279-1971, Paragraph 4.8)

The ESF variables are direct measures of the desired variables requiring protective actions. Refer to 7.3.1.1.1 thru 7.3.1.1.11.

9. Capability of Sensor Checks (IEEE 279-1971, Paragraph 4.9)

Refer to 7.3.2.1.3, Regulatory Guide 1.22.

10. Capability for Test and Calibration (IEEE 279-1971, Paragraph 4.10)

Refer to 7.3.2.1.3, Regulatory Guide 1.22.

11. Channel Bypass or Removal from Operation (IEEE 279-1971, Paragraph 4.11)

During periodic tests of any one ESF system channel, a sensor may be valved out of service and returned to service under the administrative control procedures. Since only one sensor is valved out of service at any given time during the test interval, protective action capability for ESF system automatic initiation is maintained through the remaining redundant instrument channels.

12. Operating Bypasses (IEEE 279-1971, Paragraph 4.12)

The ESF systems contain the following operating bypasses.

The PCRVICS has two bypasses. 1) main steam line low pressure operating bypass which is imposed by means of the mode switch. In all modes except run, the mode switch cannot be left in this position above 10% of rated power without inititating a scram. Therefore the bypass is removed by the normal reactor operating sequence, and 2) the low condenser vacuum bypass which is imposed by means of a manual bypass switch in conjunction with closure of the turbine stop valves, the reactor mode switch in any position other than "RUN", and reactor pressure below the low pressure setpoint. Bypass removal is accomplished automatically by the opening of the turbine stop valves or raising reactor pressure above the interlock pressure setpoint and manually by placing the bypass switch in normal position or by placing the mode switch in the "RUN" position.

13. Indication of Bypasses (IEEE 279-1971, Paragraph 4.13)

For a discussion of bypass and inoperability indication refer to 7.1.2.4, Regulatory Guide 1.47.

14. Access to Means for Bypassing (IEEE 279-1971, Paragraph 4.14)

Access to means of bypassing any safety action or function for the ESF systems is under the administrative control of the control room operator. The operator is alerted to bypasses as described in 7.1.2.4, Regulatory Guide 1.47.

Control switches which allow safety system bypasses are keylocked. All keylock switches in the control room are designed such that their key can only be removed when the switch is in the "accident" or "safe" position. All keys will normally be removed from their respective switches during operation and maintained under the control of the shift supervisor. Further, the key locker will be audited once per day by the shift supervisor. Should a key be required to change a valve position, it will be obtained from the shift supervisor via approved key control procedures.

15. Multiple Trip Settings (IEEE 279-1971, Paragraph 4.15)

There are no multiple set points within the ESF systems.

16. Completion of Protective Action Once Initiated (IEEE 279-1971, Paragraph 4.16)

Each of the automatically initiated ESF system control logics seal-in electrically and remain energized after initial conditions return to normal. Deliberate operator action is required to return (reset) an ESF system logic to normal.

17. Manual Initiation (IEEE 279-1971, Paragraph 4.17)

Refer to the discussion of Regulatory Guide 1.62 in 7.3.2.1.3.

18. Access to Setpoint Adjustments (IEEE 279-1971, Paragraph 4.18)

All access to ESF system set point adjustments, calibration controls, and test points are under the administrative control of the control room operator.

19. Identification of Protective Actions (IEEE 279-1971, Paragraph 4.19)

ESF protective actions are directly indicated and identified by annunciators located in the main control room and a typed record is available from the process computer.

20. Information Readout (IEEE 279-1971, Paragraph 4.20)

The ESF systems are designed to provide the operator with accurate and timely information pertinent to their status. They do not introduce signals that could cause anomalous indications confusing to the operator.

21. System Repair (IEEE 279-1971, Paragraph 4.21)

The ESF systems are designed to permit repair or replacement of components.

Recognition and location of a failed component will be accomplished during periodic testing or by annunciation in the main control room.

22. Identification (IEEE 279-1971, Paragraph 4.22)

The ESF panels are identified by colored nameplates. The nameplate shows the division to which each panel or rack is assigned, and also identifies the function in the system of each item of the control panel. The system to which each relay belongs is identified on the relay panels.

All wiring and cabling outside of panels are labeled to indicate its divisional assignment as well as its system assignment. See 8.3.1.3.

7.3.2.1.3 Conformance to Regulatory Guides

The following is a discussion of conformance to those Regulatory Guides which apply specifically to the ESF systems. Refer to 7.1.2.4 for a discussion of Regulatory Guides which apply equally to all safety-related systems.

a. Regulatory Guide 1.22-1972

The ESF systems instrumentation and controls are capable of being tested during normal plant operation unless that testing is detrimental to plant availability to verify the operability of each system component. Testing of safety-related sensors is accomplished by valving out each sensor, one at a time, and applying a test pressure source or in the case of the main steam line radiation sensors, the sensors may be removed and test sources applied. This verifies the operability of the sensor contacts, the sensor set point, and the associated logic components in the control room. Functional operability of temperature sensors may be verified by readout comparisons, applying a heat source to the locally mounted temperature sensing elements or by continuity testing.

For the HPCS, LPCS, and LPCI, testing for functional operability of the control logic relays can be accomplished by use of plug-in test jacks and switches in conjunction with single sensor tests.

Four test jacks are provided to allow ADS logic testing one for each logic channel. During testing, only one logic should be actuated at a time. However, when the test plug is plugged into one channel, the complement channel of that trip system is automatically rendered inoperative. Therefore, inadvertent ADS actuation cannot occur even if both channels are improperly placed in the test mode simultaneously. An alarm is provided if a test plug is inserted in both channels in a division at the same time. Operation of the test plug switch and the permissive contacts will close one of the two series relay contacts in the valve solenoid circuit. This will cause a panel light to come on indicating proper channel operation.

Annunciation is provided in the main control room whenever a test plug is inserted in a jack to indicate to the operator that an ECCS is in a test status.

Operability of air operated, solenoid operated, and motor operated valves is verified by actuating the valve control switches and monitoring the position change by position indicating lights at the control switch.

The ESF systems are provided with indications, status displays, annunciation, and computer printouts which aid the control room operator during periodic system tests to verify component operability.

b. Regulatory Guide 1.53-1973

Refer to IEEE 279 Para. 4.2, 7.3.2.1.2.

c. Regulatory Guide 1.62-1973 - Manual Initiation of Protective Actions

The HPCS, LPCS, and the Division 2 LPCI system are manually initiated at the system level from the Main Control Room by actuation of an armed pushbutton. The LPCS pushbutton also initiates the Division 1 LPCI system.

The ADS and the PCRVICS are manually initiated at the system (division) level by actuation of two armed pushbuttons (one for each logic channel).

The RHRS-Containment Spray Cooling Mode is manually initiated at the system (division) level by actuation of the RHR pump start control switch and by opening the Containment Spray or Suppression Chamber Spray valves.

The RHRS-Suppression Pool Cooling mode is manually initiated from the main control room by actuation of system pump and valve controls.

The MSLC system, the SGTS, and the CIA system are manually initiated at the system (division) level by actuation of the system start control switch.

The SSW system is manually initiated at the system (division) level by actuation of the pump start control switch.

The Main Control Room and Critical Switchgear HVAC is manually initiated at the system (division) level by actuation of individual fan start control switches.

The Containment Atmosphere Control system is manually initiated at the system (division) level by actuation of the Recombiner system start control switch.

The actuation of the system level manual initiation switches simulate all the actions of automatic or manual (individual equipment initiation) system actuation.

d. Regulatory Guide 1.96 - Design of Main Steam Isolation Valve Leakage Control System for Boiling Water Reactor Nuclear Power Plants.

The MSLCS is designed to remove leakage from Main steam line Isolation valves by drawing a positive suction on the main steam line downstream of the isolation valves, and transporting leakage to the Standby Gas Treatment System. See 5.5.5.4 for a description of valve stem packing leakage removal.

TABLE 7.3-1

HIGH PRESSURE CORE SPRAY SYSTEM INSTRUMENTATION SPECIFICATIONS

HPCS Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (4)	Required Accuracy (5)	Response Time (5)
Reactor Vessel Low Water Level (Level 2)	Level Switch (B22-N031A-D)	-150/0/+60" (1)	-38"	-	±7.5"	-
Drywell High Pressure	Pressure Switch (B22-N047A-D)	.25-12 psig	2 psig	-	±0.06 psi	0.6 sec.
Reactor Vessel High Water Level (Level 8)	Level Switch (B22-N024 A,C)	0-60"	+55.5"	-	±3"	-
Pump Discharge Pressure	Pressure Switch (E22-N012)	10-340 psig	120 psig	-	±10 psi	-
Pump Minimum Flow	Flow Switch (E22-N006)	0-1190 gpm	640 gpm	-	±160 gpm	0 milli sec
Suppression Pool High Water Level	Level Switch (E22-N002 A,B)	-	0" (5" above normal water level)	-	±0.5"	-
Condensate Storage Tanks Low Level	Level Switch (E22-N001 A-B)	-	0" (11,500 gal)	-	±0.5"	-
Diesel Fuel Day Tank Level Low	Level Switch (H) (DO-LS-21)	-	- El. 445'-0"	-	- ±0.5"	- -

NOTES FOR TABLE 7.3-1

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(3) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

(4) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(5) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-2

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION

Instrument Channel	Channels Provided	Minimum Channels Required
Reactor vessel low water level (Level 2)	4	2
Drywell high pressure	4	2
Condensate storage tanks low level	2	1
Suppression pool high level	2	1

TABLE 7.3-3

AUTOMATIC DEPRESSURIZATION SYSTEM INSTRUMENTATION SPECIFICATIONS

ADS Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (4)	Required Accuracy (5)	Response Time (5)
Reactor Vessel Low Water Level (Level 3)	Level Switch (B22-N038 A, B)	0-60" (1)	+12.5"	-	±3"	-
Reactor Vessel Low Water Level (Level 1)	Level Switch (B22-N037 (A-D)	-150/0/+60" (1)	-149"	-	±7.5"	-
Drywell High Pressure	Pressure Switch (B22-N048 A-D)	0.25-12 psig	2 psig	-	±0.06 psi	
LPCI Permissive	Pressure Switch (E12-N016 A-C, E12-N019 A-C)	10-240 psig	100 psig	-	±9 psi	-
LPCS Permissive	Pressure Switch (E21-N001, E21-N009)	10-340 psig	150 psig	-	±10 psi	
Automatic Depressurization Time Delay	Timer	0-180 sec.	105 sec.	-	±18 sec.	-

NOTES FOR TABLE 7.3-3

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(3) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

(4) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(5) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-4

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION OF THE ADS

Instrument Channel	Channels Provided	Minimum Channels Required
Reactor vessel low water level (Level 3)	2	1
Reactor vessel low water level (Level 1)	4	2
Drywell high pressure	4	2
LPCI permissive	6	2
Time delay	2	1
LPCS permissive	2	1

TABLE 7.3-5

LOW PRESSURE CORE SPRAY SYSTEM INSTRUMENTATION SPECIFICATIONS

LPCS FUNCTION	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (4)	Required Accuracy (5)	Response Time (5)
Reactor Vessel Low Water Level (Level 1)	Level Switch (B22-N037 A,C)	-150/0/+60" (1)	-149"	-	±7.5"	-
Drywell High Pressure	Pressure Switch (B22-N048 A,C)	.25-12 psig	2 psig	-	±0.06 psi	-
Injection Valve Differential Pressure	Differential Pressure Switch (E21-N006)	0-800 psid	747 psid	-	±12 psi	-
Pump Minimum Flow Bypass	Flow Switch (E21-N004)	0-1100 gpm	640 gpm	-	±170 gpm	-

NOTES FOR TABLE 7.3-5

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(3) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

(4) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(5) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-6

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION OF LPCS SYSTEM AND LPCI "A"

Instrument Channel	Channels Provided	Minimum Channels Required
Reactor vessel water level	2	1
Drywell high pressure	2	1
Pump discharge low flow	2	1
Valve differential pressure	2	1

TABLE 7.3-7

LOW PRESSURE COOLANT INJECTION INSTRUMENTATION SPECIFICATIONS

LPCS FUNCTION	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (4)	Required Accuracy (5)	Response (5) Time
Reactor Vessel Low Water Level (Level 1)	Level Switch (B22-N037 A-D)	-150/0/+60" (1)	-149"	-	±7.5"	-
Drywell High Pressure	Pressure Switch (B22-N048 A-D)	.25-12 psig	2 psig	-	±0.06 psi	-
LPCI Pump Delay (on Loss of Normal Power)	Timer	0-7.5 sec.	5 sec.	-	±0.75 sec.	-
Injection Valve Differential Pressure	Differential Pressure Switch (E12-N009 A-C)	0-1000 psid	700 psid	-	±20 psi	-
Pump Minimum Flow Bypass	Flow Switch (E12-N010 A-C)	0-15" H_2O	3" H_2O (700 GPM)	-	±120 gpm	-

NOTES FOR TABLE 7.3-7

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16, "Technical Specifications' for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(3) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

(4) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(5) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-8

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION OF LPCI "B" AND "C"

Instrument Channel	Channels Provided	Minimum Channels Required
Reactor vessel low water level	2	1
Drywell high pressure	2	1
LPCI pumps discharge low flow	2	2
Valve differential pressure	2	2

TABLE 7.3-9

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM
INSTRUMENT SPECIFICATIONS

PCRVICS Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (7)	Required Accuracy (8)	Response Time (8)
Reactor Vessel Low Water Level (Level 3)	Level Switch (B22-N024 A-D)	0-60" (1)	+12.5"	-	±3"	-
Reactor Vessel Low Water Level (Level 2)	Level Switch (B22-N026 A-D)	-150/0/+60" (1)	-38"	-	±7.5"	-
Main Steam Line High Radiation	Radiation Monitor (D17-K610 A-D)	$1 - 10^6$ CPS	3X normal		±30,000 CPS	1 sec.
Main Steam Line Area High Temp	Temperature and Differential Temperature	50-350°F 0-150°F	(4)	-	±7°F ±3°F	2.25 sec.
Main Steam Line Low Pressure	Pressure Switch (B22-N015 A-D)	50-1200 psig	840 psig	-	±15 psi at setpoint	1 sec.
Drywell High Pressure	Pressure Switch (C72-N002 A-D)	0.2-6 psig	2 psig	-	±0.05 psi	.6 sec.
Reactor Bldg. Ventilation Exhaust High Radiation	Radiation Monitor D17-K609 A-D)	$10^{-2} - 10^{2}$mR/hr	(later)	-	±9.5 mR/hr	.5 sec.
Main Condenser Low Vacuum	Pressure Switch (B22-N056 A-D)	0-30" hg ABS	23" hg ABS	-	±0.9" hg	-
Main Steam Line High Steam Flow	Differential Pressure Switch (E31-N008 A-D) (E31-N009 A-D) (E31-N010 A-D) (E31-N011 A-D)	-15/0/150 psi	132.5 psi	-	±3 psi	.1 sec.

TABLE 7.3-9 (Continued)

PCRVICS Function	Instrument	Instrument Range (2)	Trip Setting(3)	Margin(7)	Required Accuracy(8)	Response Time(8)
RCIC turbine steam line space high temperature	Temperature and Differential Temperature	50-350°F 0-150°F	(4) (4)	- -	±7°F ±3°F	2.25 sec. 2.25 sec.
RCIC turbine steam line high flow	Differential pressure switch (E31-N007 A,B) (E31-N013 A,B)	-200/0/+200 in H_2O	198 in. H_2O	-	±2%	.1 sec.
Reactor shutdown cooling system space high temperature	Temperature and Differential Temperature Switches	Thermocouple: 50-350°F 0-150°F 50-350°F	(4) (4)	- -	±7°F ±3°F	2.25 sec. 2.25 sec.
Reactor cleanup system space high temperature	Temperature and Differential Temperature Switches	50-350°F 0-150°F	(4) (4)	- -	±7°F ±3°F	2.25 sec. 2.25 sec.

* Setpoints will be established based upon operating data.

NOTES FOR TABLE 7.3-9

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(3) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

The initial values are the trip settings listed in the tables.

(4) Trip settings will be established during preoperational testing (normally 50°F above ambient).

(5) Setpoint will be determined from instrument calibration curve.

(6) Setpoint will be established after background readings are determined during startup.

(7) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(8) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-10

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION OF THE PCRVICS

Instrument Channel	Channels Provided	Minimum Channels Required
Reactor vessel low water level (Level 3)	4	3
Reactor vessel low water level (Level 2)	4	3
Reactor vessel high presure	4	3
Main steamline high radiation	4	3
Main steamline space hign temperature	4 temp 4 differential temp	3 temp 4 differential temp
Main steamline high flow	4	3
Main steamline low pressure	4	3
Drywell high pressure	4	3
Reactor Building ventilation exhaust high radiation	2	2
Main condenser low vacuum	4	3
Reactor water cleanup system differential flow	2	2

TABLE 7.3-11

REACT. BLDG. VENT. & PRESS. CONT. SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
React. Bldg. to Fuel Pool Area Differential Pressure	Differential Press. Trans. (REA DPT 1A1-1A4) (REA DPT 1B1-1B4)	-3 to + 7" H_2O	N/A	N/A	$\pm3\%$	-
	Differential Press. Controller (REA DPRC-1A, 1B)	-3 to + 7" H_2O	-1/4" H_2O Δp	N/A	$\pm3\%$	-

NOTES FOR TABLE 7.3-11

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-12

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR REACT. BLDG. VENT & PRESS CONTROL

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
React. Bldg. to Fuel Pool Area Differential Pressure	2	1

TABLE 7.3-13

CONTAINMENT INSTRUMENT AIR SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Reactor Vessel Low Water Level (Level 2)	Level Switch (B22-NO26A-D)	-150/0/160"(5)	-38"	-	±7.5"	-
Drywell High Press.	Press. Switch (C72-NOO2A-D)	0.2 - 6 psig	2 psig	-	±.05 psi	0.6 sec.
Header Press. Low	Press. Switch (C1A-PS21A,B)	-	150 psig	-	-	-

NOTES FOR TABLE 7.3-13

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

(5) Instrument zero equal to 527.5" above vessel zero.

TABLE 7.3-14

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR CONTAINMENT INSTRUMENT AIR SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Vessel Low Water Level	4	2
Drywell High Pressure	4	2
Header Pressure Low	2	1

TABLE 7.3-15

STANDBY GAS TREATMENT SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Reactor Vessel Low Water Level (Level 2)	Level Switch (B22-N026A-D)	-150/0/+60" (5)	-38"	-	±7.5"	-
Drywell High Press	Pressure Switch (C72-N002A-D)	0.2 - 6 psig	2 psig	-	±0.05 psi	0.6 sec
Reactor Bldg. Vent. Exhaust High Rad.	Radiation Monitor (D17-K609A-D)	10^{-2} - 10^{2} mR/hr	-	-	±9.5 mR/hr	0.5 sec

NOTES FOR TABLE 7.3-15

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

(5) Instrument zero equal to 527.5" above vessel zero.

TABLE 7.3-16

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR THE STANDBY GAS TREATMENT SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Vessel Water Level	4	2
Drywell High Pressure	4	2
Reactor Bldg. Vent High Rad	4	2

CONTAINMENT ATMOSPHERE CONTROL SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Recombiner Blwr. Discharge Temp. High	Temp. Switch (CAC TS 1A,B)	0 - 340°F	300°F	-	±1%	-
Preheater Discharge Temp. High	Temp. Switch (CAC TS 5A,B)	0 - 1500°F	1150°F	-	±1%	-
Moisture Separator Outlet Temp. High	Temp. Switch (CAC TS 6A,B)	0 - 340°F	200°F	-	±1%	-
Recycle Flow Low	Flow Switch (CAC FS 6A,B)	0 - 25 KSCFH	3 KSCFH	-	±1%	-
After Cooler Inlet Pressure High	Press. Switch (CAC PS 68A, B)	0 - 50 psig	45 psig	-	± 1%	-
Moisture Separator Level High	Level Switch (CAC LS 1A,B)	-	-	-	-	-

AMENDMENT NO. 10
July 1980

NOTES FOR TABLE 7.3-17

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-18

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR CONTAINMENT ATMOSPHERE CONTROL

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Recombiner Blwr. Disch. Temp. High	2	1
Preheater Discharge Temp. High	2	1
Moisture Separator Outlet Temp. High	2	1
Recycle Flow Low	2	1
After Cooler Inlet Pressure High	2	1
Moisture Seperator Level High	2	1

TABLE 7.3-19

MAIN CONTROL ROOM & CRITICAL SWGR. RM. HVAC SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Control Room Temp.	Temp. Controller (WMA T1C 12A, B)	-	78° DB	-	-	-
Control Room Humid	Humid Controller (WMA M1C 55A, B)	-	40% RH	-	-	-
Swgr. Rooms Temp.	Temp. Controllers (WMA T1C 52A, B) (WMA T1C 53A, B)	-	80° DB	-	$\pm$1%	-
Reactor Vessel Low Water Level (Level 2)	Level Switch (B22-NO26A-D)	-150/0/+60"(5)	-38"	-	$\pm$7.5"	-
Drywell High Press.	Pressure Switch (C72-NOO2A-D)	0.2 - 6 psig	2 psig	-	$\pm$0.05 psi	0.6 sec
Reactor Bldg. Vent. Exhaust High Rad.	Radiation Monitor (D17-K609A-D)	10^{-2} - 10^{2} mR/hr	-	-	$\pm$9.5 mR/hr	0.5 sec

NOTES FOR TABLE 7.3-19

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

(5) Instrument zero equal to 527.5" above vessel zero.

TABLE 7.3-20

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR CONT. RM. & SWGR. RM. HVAC

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Control Room Temperature	2	1
Control Room Humidity	2	1
Swgr. Rooms Temperatures	2	1
Reactor Vessel Water Level	4	2
Drywell High Pressure	4	2
Reactor Bldg. Vent High Rad.	4	2

TABLE 7.3-21

STANDBY SERVICE WATER SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Reactor Vessel Low Water Level (Level 2)	Level Switch (B22-NO26A-D)	-150/0/+60" (5)	-38"	-	±7.5"	-
Drywell High Press.	Pressure Switch (C72-NOO2A-D)	0.2 - 6 psig	2 psig	-	±0.05 psi	0.6 sec
Reactor Bldg. Vent. Exhaust High Rad.	Radiation Monitor (D17-K609A-D)	10^{-2} - 10^{2} mR/hr	-	-	±9.5 mR/hr	0.5 sec
Spray Pond Water Level Low	Level Switch (SW LS 1A3, 1B3)	-	El. 415'	-	±1%	-
Spray Pond Water Level Low	Level Switch (SW LS 1A2-1D2)	-	El. 433'-6"	-	±1%	-
Spray Pond Water Level Low	Level Switch (SW LS 1A4, 1B4)	-	El. 433'-9"	-	±1%	-
SW Discharge Pressure Low	Pressure Switch (SW PS 1A, 1B, 40B)	-	50 psig	-	±1%	-

NOTES FOR TABLE 7.3-21

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

(5) Instrument zero equal to 527.5" above vessel zero.

TABLE 7.3-22

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR THE STANDBY SERVICE WATER SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Vessel Water Level	4	2
Drywell High Pressure	4	2
Reactor Bldg. Vent High Rad.	4	2
Spray Pond Water Level Low (SW LS 1A3, 1B3)	2	1
Spray Pond Water Level Low (SW LS 1A2, 1B2)	2	1
Spray Pond Water Level Low (SW LS 1C2, 1D2)	2	1
Spray Pond Water Level Low (SW LS 1A4, 1B4)	2	1
SW Discharge Pressure Low	3	2

TABLE 7.3-23

RHRS - SUPPRESSION POOL COOLING MODE INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Reactor Vessel Low Water Level (Level 1)	Level Switch (B22-NO37A-D)	-150/0/+60"(5)	-149"	-	±7.5"	-
Drywell High Pressure	Pressure Switch (B22-NO48A-D)	0.25 - 12 psig	2 psig	-	±0.06 psi	0.6 sec
Suppression Pool Temperature High	Temperature Recorder	0 - 212°F	-	-	±2°F	15 sec

NOTES FOR TABLE 7.3-23

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

(5) Instrument zero equal to 527.5" above vessel zero.

TABLE 7.3-24

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR RHRS - SUPPRESSION POOL COOLING MODE

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Vessel Water Level	4	2
Drywell High Pressure	4	2
Suppression Pool Temp. High	2	1

TABLE 7.3-25

RHRS - CONTAINMENT SPRAY COOLING MODE SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Drywell High Press.	Pressure Switch (B22-N048A-D)	0.25 - 12 psig	2 psig	-	±0.06 psi	0.6 sec

NOTES FOR TABLE 7.3-25

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expect d range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specific s".

TABLE 7.3-26

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR RHRS - CONTAINMENT SPRAY COOLING MODE

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Drywell High Pressure	4	2

TABLE 7.3-27

MAIN STEAM LINE LEAKAGE CONTROL SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Reactor Pressure Low	Press. Switch (MSLC-PS20) (MSLC-PS24) (MSLC-PS8A-D) (MSLC-FS7A-D)	–	35 psig	–	–	–
MSLC Header Pressure Low	Press. Switch (MSLC-PS 25)	–	0 psig	–	–	–
MSLC Header Pressure Low	Press. Switch (MSLC-PS 70A-D)	–	5 psig	–	–	–
MSLC High Flow	Flow Switch (MSLC-FS 3A-D)	–	505 CFH	–	–	–

NOTES FOR TABLE 7.3-27

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.3-28

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR MAIN STEAM LINE LEAKAGE CONTROL SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Pressure Low (MSLC-PS 20) (MSLC-PS 8A-D)	5	1
MSLC Header Press. Low (MSLC-PS 25)	1	0
MSLC Header Press. Low (MSLC-PS 70A-D)	4	0
MSLC High Flow	4	0

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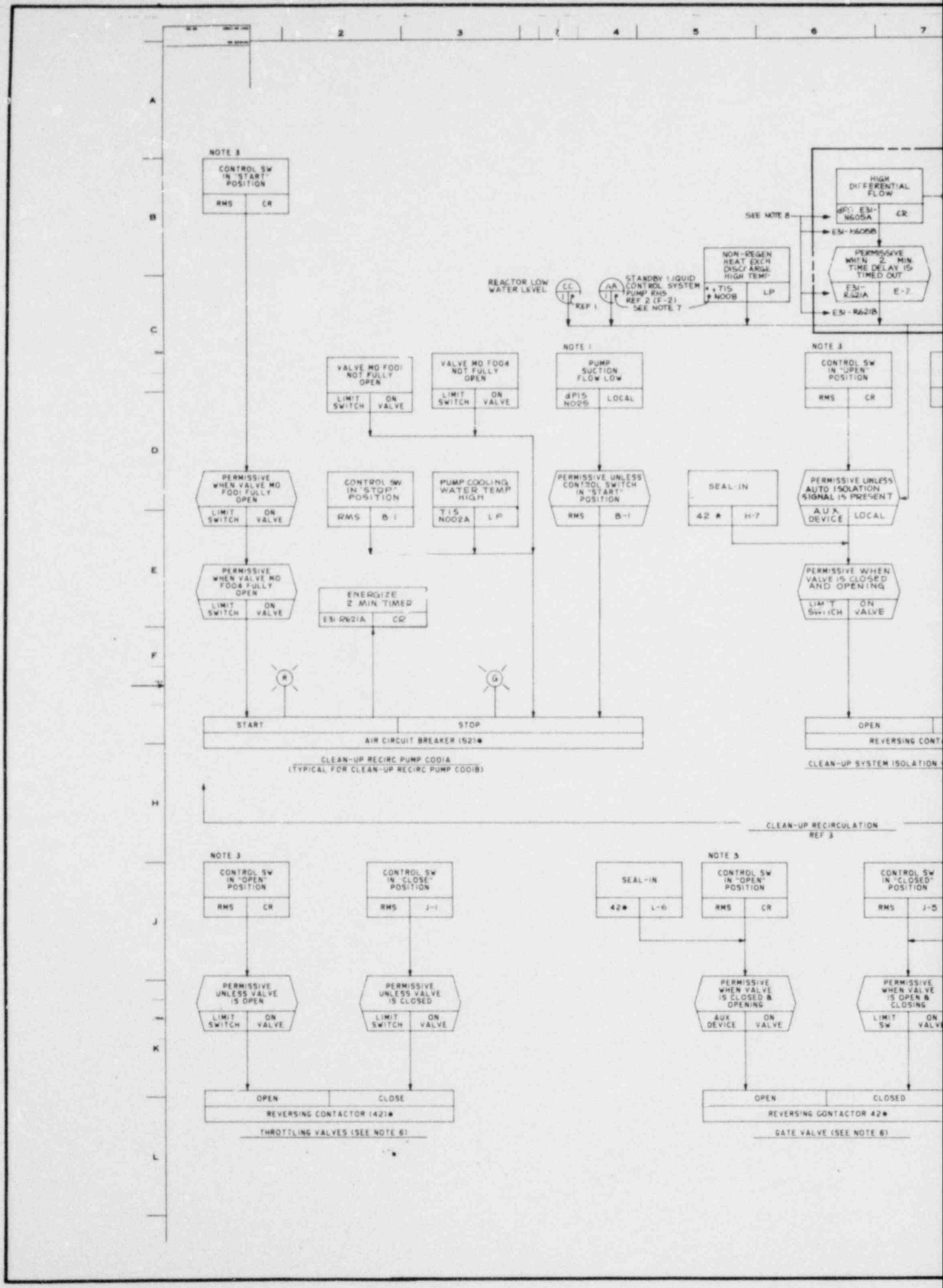
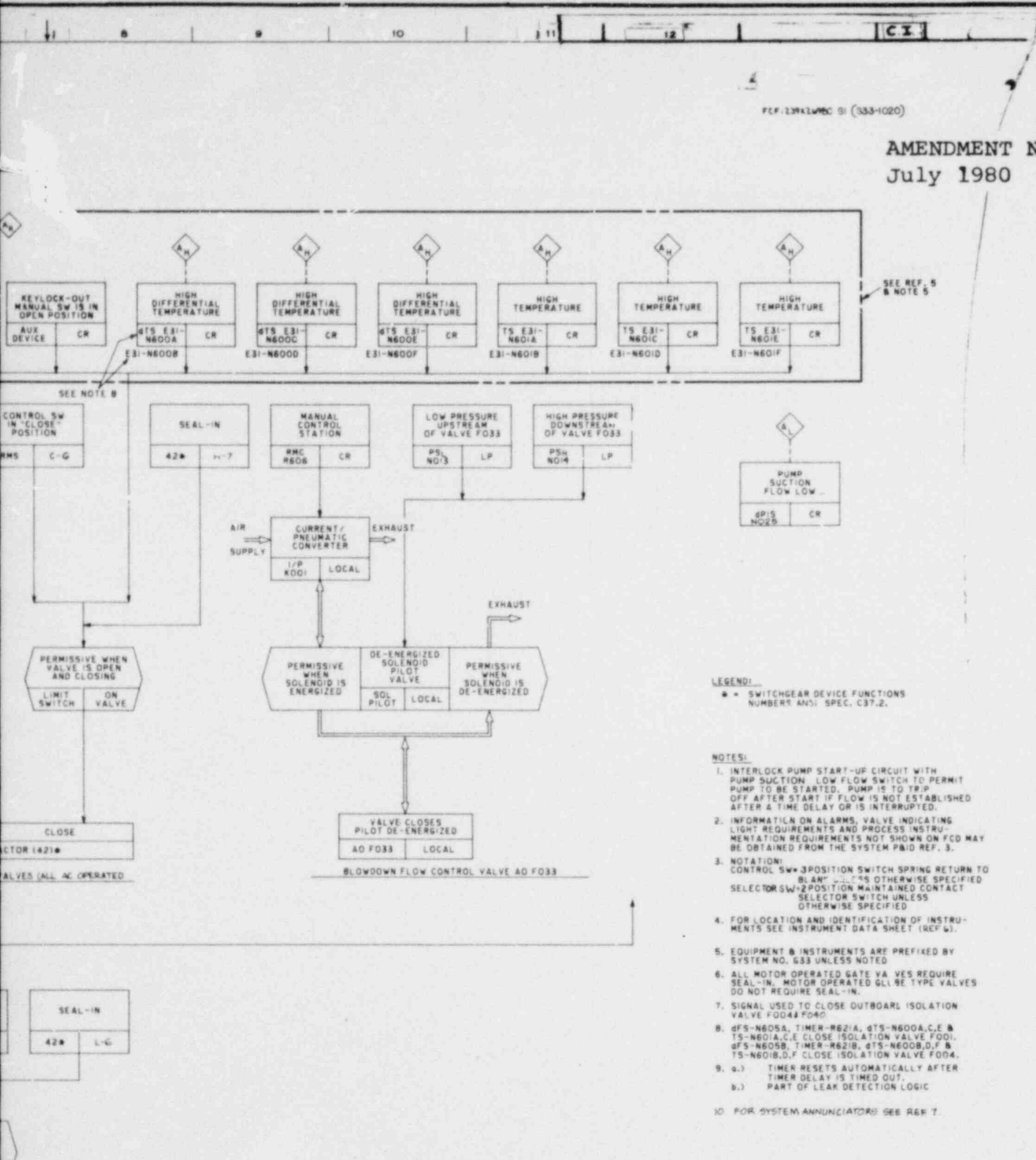


NOTE 3
CONTROL SW IN "START" POSITION
RMS CR
PERMISSIVE WHEN VALVE MO F001 FULLY OPEN
LIMIT SWITCH ON VALVE
PERMISSIVE WHEN VALVE MO F004 FULLY OPEN
LIMIT SWITCH ON VALVE
VALVE MO F001 NOT FULLY OPEN
LIMIT SWITCH ON VALVE
VALVE MO F004 NOT FULLY OPEN
LIMIT SWITCH ON VALVE
CONTROL SW IN "STOP" POSITION
RMS B-1
PUMP COOLING WATER TEMP HIGH
TIS N002A LP
ENERGIZE 2 MIN TIMER
E31-R621A CR
R
G
START
STOP
AIR CIRCUIT BREAKER (52)*
CLEAN-UP RECIRC PUMP C001A
(TYPICAL FOR CLEAN-UP RECIRC PUMP C001B)
REACTOR LOW WATER LEVEL
CC
REF 1
AA
STANDBY LIQUID CONTROL SYSTEM PUMP RHS REF 2 (F-2) SEE NOTE 7
NOTE 1
PUMP SUCTION FLOW LOW
dPIS N025 LOCAL
PERMISSIVE UNLESS CONTROL SWITCH IN "START" POSITION
RMS B-1
NON-REGEN HEAT EXCH DISCHARGE HIGH TEMP
TIS N008 LP
SEE NOTE 8
HIGH DIFFERENTIAL FLOW
dPIS E31-N605A CR
E31-N605B
PERMISSIVE WHEN 2 MIN TIME DELAY IS TIMED OUT
E31-R621A E-2
E31-R621B
NOTE 3
CONTROL SW IN "OPEN" POSITION
RMS CR
SEAL-IN
42* H-7
PERMISSIVE UNLESS AUTO ISOLATION SIGNAL IS PRESENT
AUX DEVICE LOCAL
PERMISSIVE WHEN VALVE IS CLOSED AND OPENING
LIMIT SWITCH ON VALVE
OPEN
REVERSING CONT
CLEAN-UP SYSTEM ISOLATION
CLEAN-UP RECIRCULATION
REF 3
NOTE 3
CONTROL SW IN "OPEN" POSITION
RMS CR
CONTROL SW IN "CLOSE" POSITION
RMS J-1
PERMISSIVE UNLESS VALVE IS OPEN
LIMIT SWITCH ON VALVE
PERMISSIVE UNLESS VALVE IS CLOSED
LIMIT SWITCH ON VALVE
OPEN CLOSE
REVERSING CONTACTOR (42)*
THROTTLING VALVES (SEE NOTE 6)
SEAL-IN
42* L-6
NOTE 3
CONTROL SW IN "OPEN" POSITION
RMS CR
CONTROL SW IN "CLOSED" POSITION
RMS J-5
PERMISSIVE WHEN VALVE IS CLOSED & OPENING
AUX DEVICE ON VALVE
PERMISSIVE WHEN VALVE IS OPEN & CLOSING
LIMIT SW ON VALVE
OPEN CLOSED
REVERSING CONTACTOR 42*
GATE VALVE (SEE NOTE 6)

AMENDMENT NO. 10
July 1980



LEGEND:

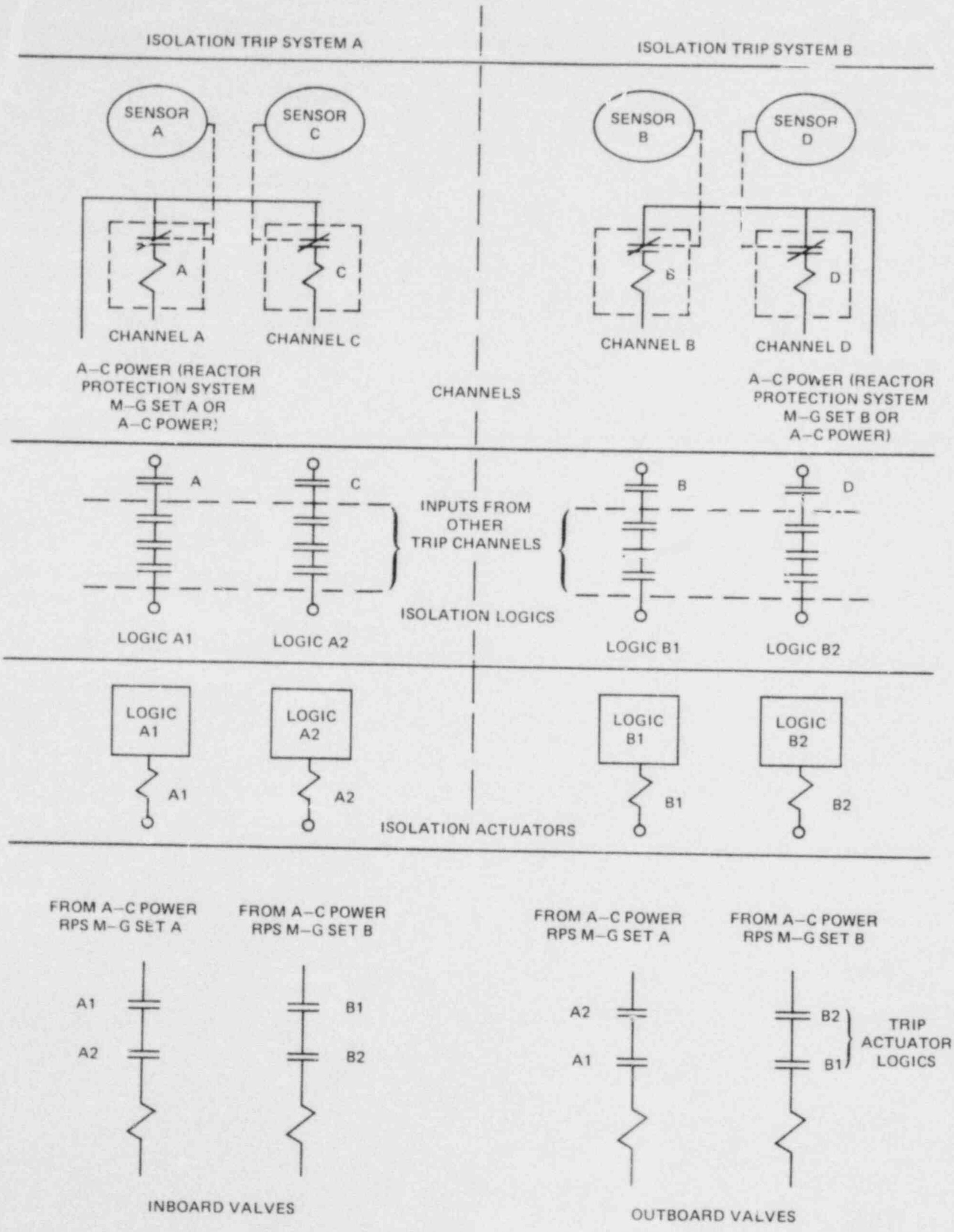
* = SWITCHGEAR DEVICE FUNCTIONS NUMBERS ANSI SPEC. C37.2.

NOTES:

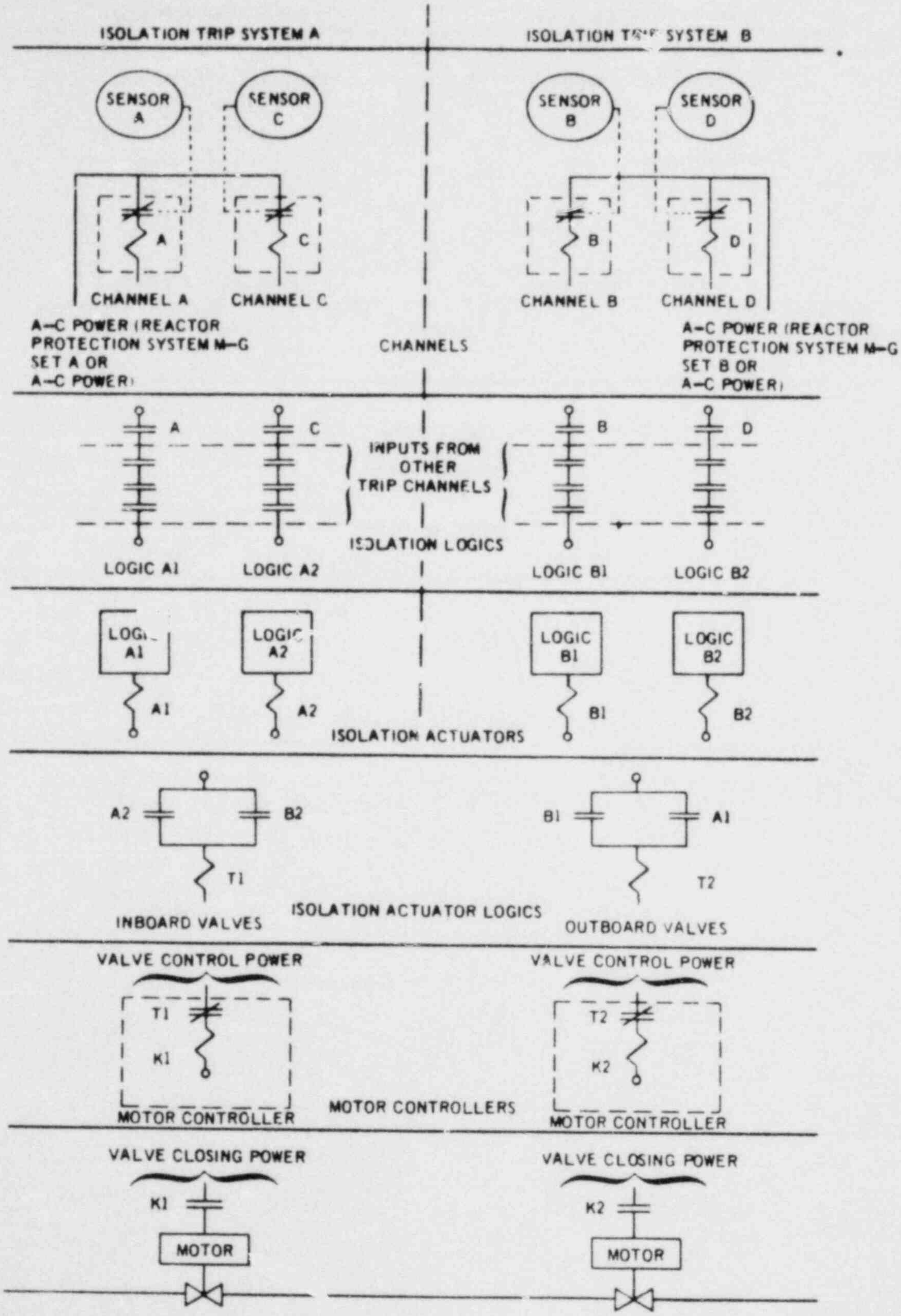
1. INTERLOCK PUMP START-UP CIRCUIT WITH PUMP SUCTION LOW FLOW SWITCH TO PERMIT PUMP TO BE STARTED. PUMP IS TO TRIP OFF AFTER START IF FLOW IS NOT ESTABLISHED AFTER A TIME DELAY OR IS INTERRUPTED.
2. INFORMATION ON ALARMS, VALVE INDICATING LIGHT REQUIREMENTS AND PROCESS INSTRUMENTATION REQUIREMENTS NOT SHOWN ON FCD MAY BE OBTAINED FROM THE SYSTEM P&ID REF. 3.
3. NOTATION: CONTROL SW=3POSITION SWITCH SPRING RETURN TO BLANK UNLESS OTHERWISE SPECIFIED SELECTOR SW=2POSITION MAINTAINED CONTACT SELECTOR SWITCH UNLESS OTHERWISE SPECIFIED
4. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET (REF 6).
5. EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NO. G33 UNLESS NOTED
6. ALL MOTOR OPERATED GATE VALVES REQUIRE SEAL-IN. MOTOR OPERATED GLOBE TYPE VALVES DO NOT REQUIRE SEAL-IN.
7. SIGNAL USED TO CLOSE OUTBOARD ISOLATION VALVE F004 & F040.
8. dFS-N605A, TIMER-R621A, dTS-N600A,C,E & TS-N601A,C,E CLOSE ISOLATION VALVE F001. dFS-N605B, TIMER-R621B, dTS-N600B,D,F & TS-N601B,D,F CLOSE ISOLATION VALVE F004.
9. a.) TIMER RESETS AUTOMATICALLY AFTER TIMER DELAY IS TIMED OUT.
   b.) PART OF LEAK DETECTION LOGIC
10. FOR SYSTEM ANNUNCIATORS SEE REF 7.

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. NUCLEAR BOILER SYS FCD	B21-1030
2. STANDBY LIQUID CONTROL SYS FCD	C41-1030
3. REACTOR WATER CLEAN-UP SYS P&ID	G33-1010
4. LOGIC SYMBOLS	A42-1030
5. LEAK DETECTION SYS	E31-1010
6. INSTRUMENT DATA SHEET	G33-3050
7. REACTOR WATER CLEANUP SYS	G33-1050

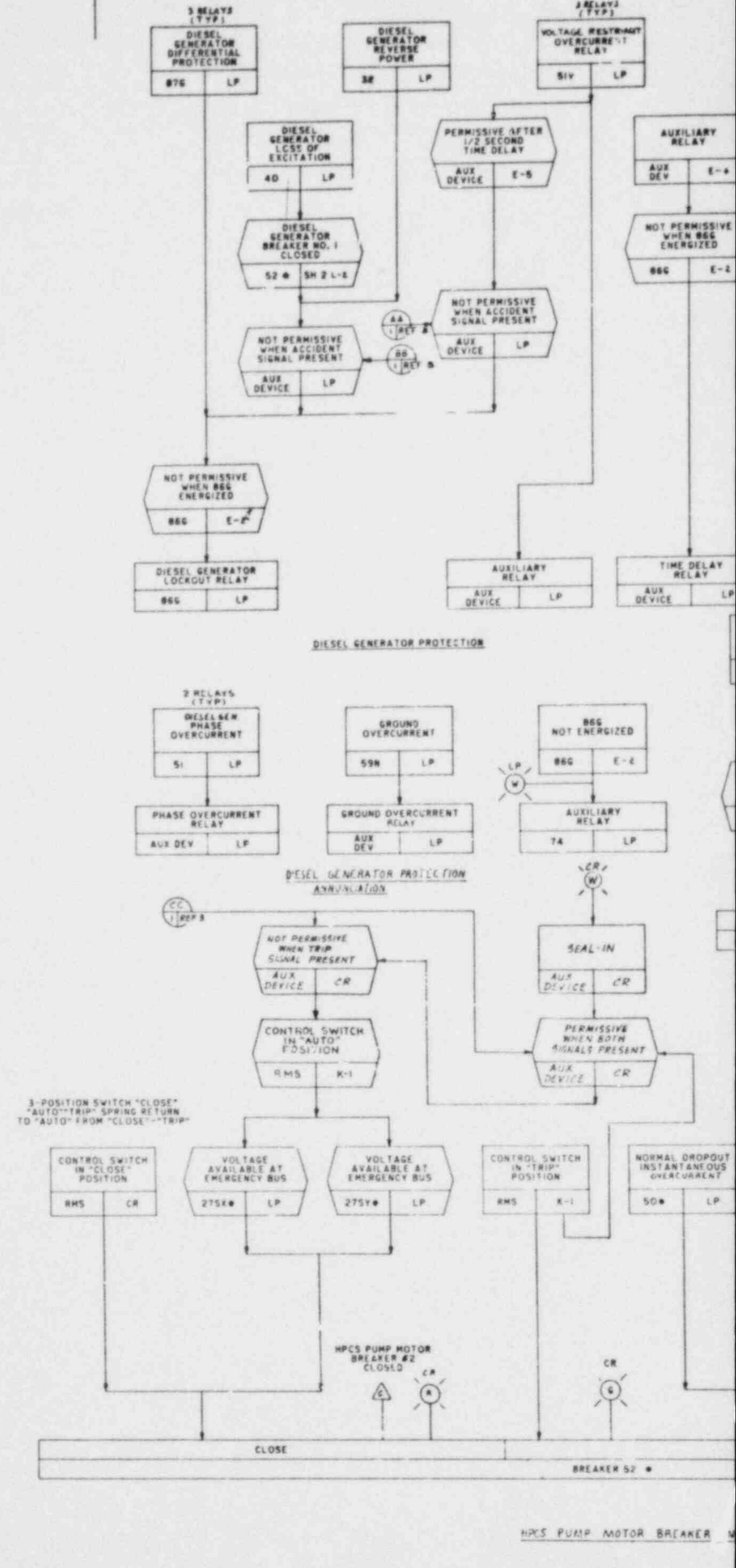
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR WATER CLEANUP SYSTEM FCD	FIGURE 7.3-1



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION CONTROL SYSTEM FOR MAIN STEAM LINE ISOLATION VALVES.	FIGURE 7.3-2

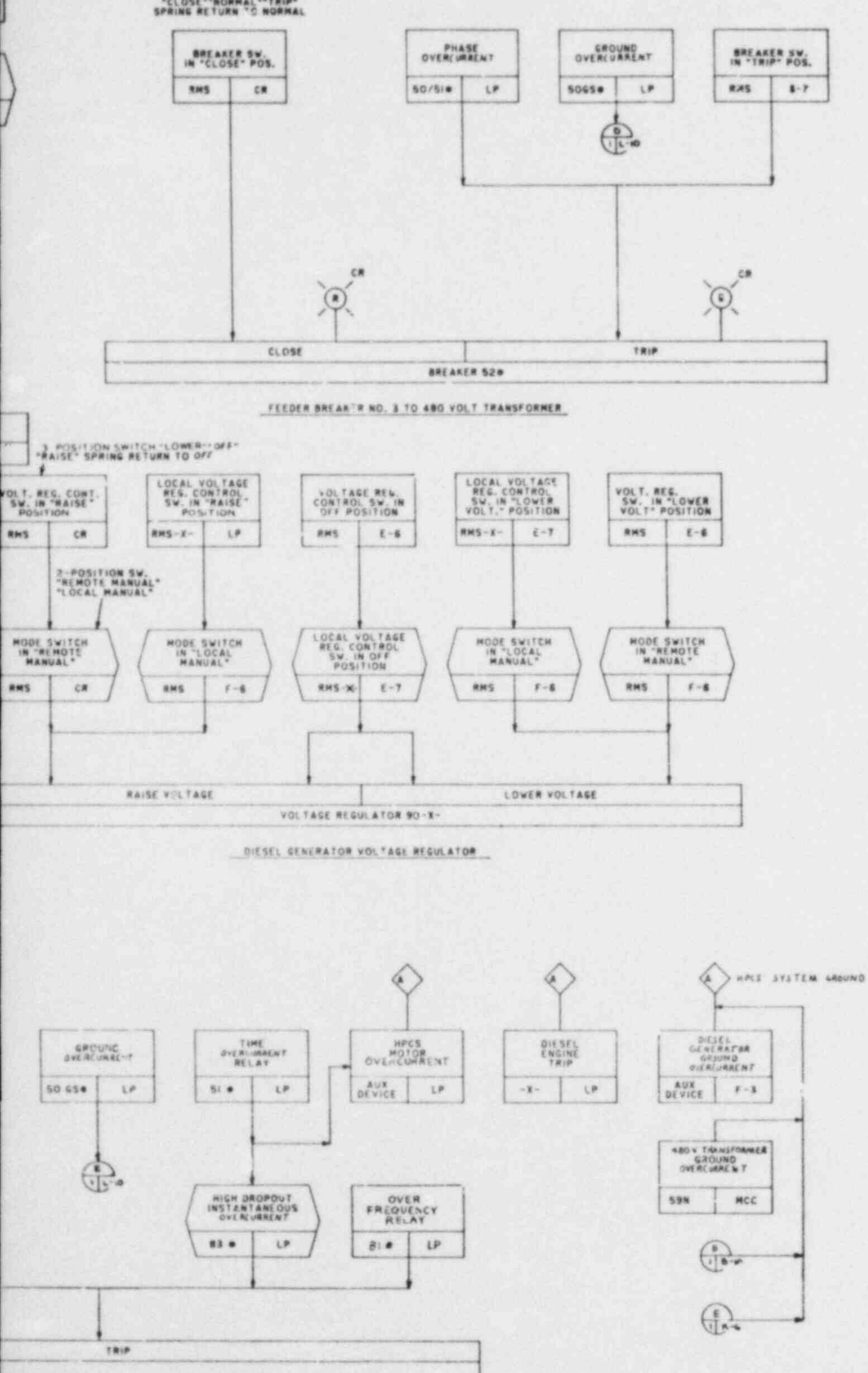


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ISOLATION CONTROL SYSTEM USING MOTOR - OPERATED VALVES	FIGURE 7.3-3



3 RELAYS (TYP)
DIESEL GENERATOR DIFFERENTIAL PROTECTION
87G LP
DIESEL GENERATOR REVERSE POWER
32 LP
3 RELAYS (TYP)
VOLTAGE RESTRAINT OVERCURRENT RELAY
51V LP
DIESEL GENERATOR LOSS OF EXCITATION
40 LP
PERMISSIVE AFTER 1/2 SECOND TIME DELAY
AUX DEVICE E-5
AUXILIARY RELAY
AUX DEV
NOT PERMISSIVE WHEN 86G ENERGIZED
86G
DIESEL GENERATOR BREAKER NO. 1 CLOSED
52 * SH 2 L-6
NOT PERMISSIVE WHEN ACCIDENT SIGNAL PRESENT
AUX DEVICE LP
AA
BB
NOT PERMISSIVE WHEN ACCIDENT SIGNAL PRESENT
AUX DEVICE LP
NOT PERMISSIVE WHEN 86G ENERGIZED
86G E-2
DIESEL GENERATOR LOCKOUT RELAY
86G LP
AUXILIARY RELAY
AUX DEVICE LP
TIME DELAY RELAY
AUX DEVICE
DIESEL GENERATOR PROTECTION
2 RELAYS (TYP)
DIESEL GEN PHASE OVERCURRENT
51 LP
GROUND OVERCURRENT
59N LP
86G NOT ENERGIZED
86G E-2
PHASE OVERCURRENT RELAY
AUX DEV LP
GROUND OVERCURRENT RELAY
AUX DEV LP
AUXILIARY RELAY
74 LP
DIESEL GENERATOR PROTECTION ANNUNCIATION
CC
NOT PERMISSIVE WHEN TRIP SIGNAL PRESENT
AUX DEVICE CR
SEAL-IN
AUX DEVICE CR
CONTROL SWITCH IN "AUTO" POSITION
RMS K-1
PERMISSIVE WHEN BOTH SIGNALS PRESENT
AUX DEVICE CR
3-POSITION SWITCH "CLOSE" "AUTO" "TRIP" SPRING RETURN TO "AUTO" FROM "CLOSE" "TRIP"
CONTROL SWITCH IN "CLOSE" POSITION
RMS CR
VOLTAGE AVAILABLE AT EMERGENCY BUS
27SX* LP
VOLTAGE AVAILABLE AT EMERGENCY BUS
27SY* LP
CONTROL SWITCH IN "TRIP" POSITION
RMS K-1
NORMAL DROPOUT INSTANTANEOUS OVERCURRENT
50* LP
HPCS PUMP MOTOR BREAKER #2 CLOSED
CR
CLOSE
BREAKER 52 *
HPCS PUMP MOTOR BREAKER

Amendment No. 10
July 1980



NOTES:

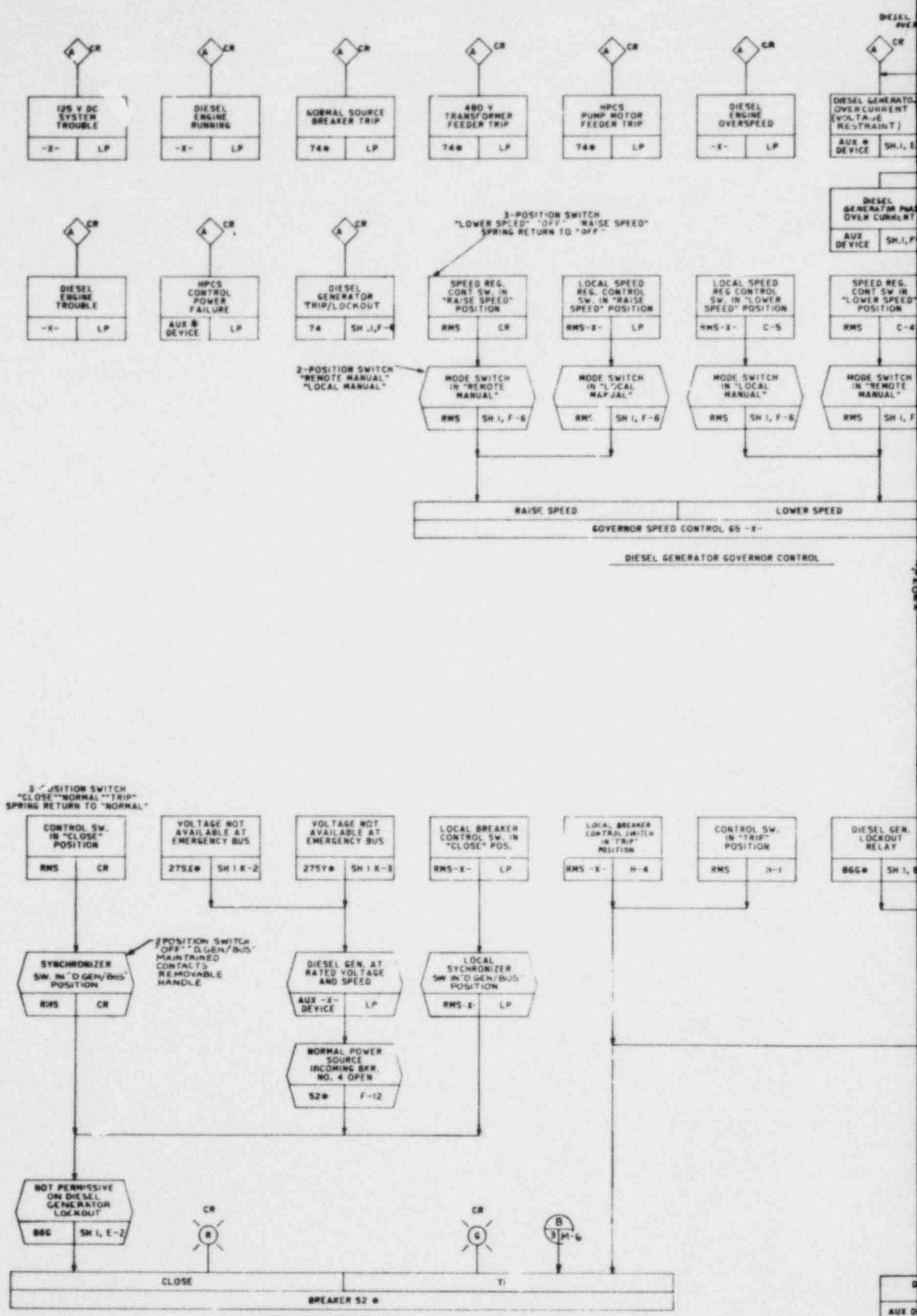
1. THIS DOCUMENT FORMS A PART OF REF. 5
2. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED WITH E22 UNLESS OTHERWISE NOTED.
3. ALL SWITCHGEAR DEVICE FUNCTION NUMBERS ARE PER ANSI SPEC C37.2. THOSE MTD IN SWGR ARE INDICATED WITH ●
4. -X- DEVICE LOCATED AT DIESEL GENERATOR PANEL AND FORMS A PART OF E22-S001. INTERNAL LOGICS ARE BY VENDOR OF THE EQUIPMENT.
5. FOR PROTECTIVE RELAY QUANTITY, SYNCHRONIZING SCHEME AND METERING SEE REF. 2.
6. THE HPCS SYSTEM POWER SUPPLY SHALL BE DESIGNED IN ACCORDANCE WITH REF 7 AS APPLICABLE

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. HIGH PRESSURE CORE SPRAY P&ID	E22-1010
2. HIGH PRESSURE CORE SPRAY ELECTRICAL ONELINE DIAGRAM	E22-1060
3. HIGH PRESSURE CORE SPRAY SYSTEM FCD	E22-1030
4. LOGIC SYMBOLS	A42-1030
5. ELECTRICAL EQUIP. SEPARATION FOR SAFEGUARD SYSTEMS	A62-4050

WAS' 'GTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

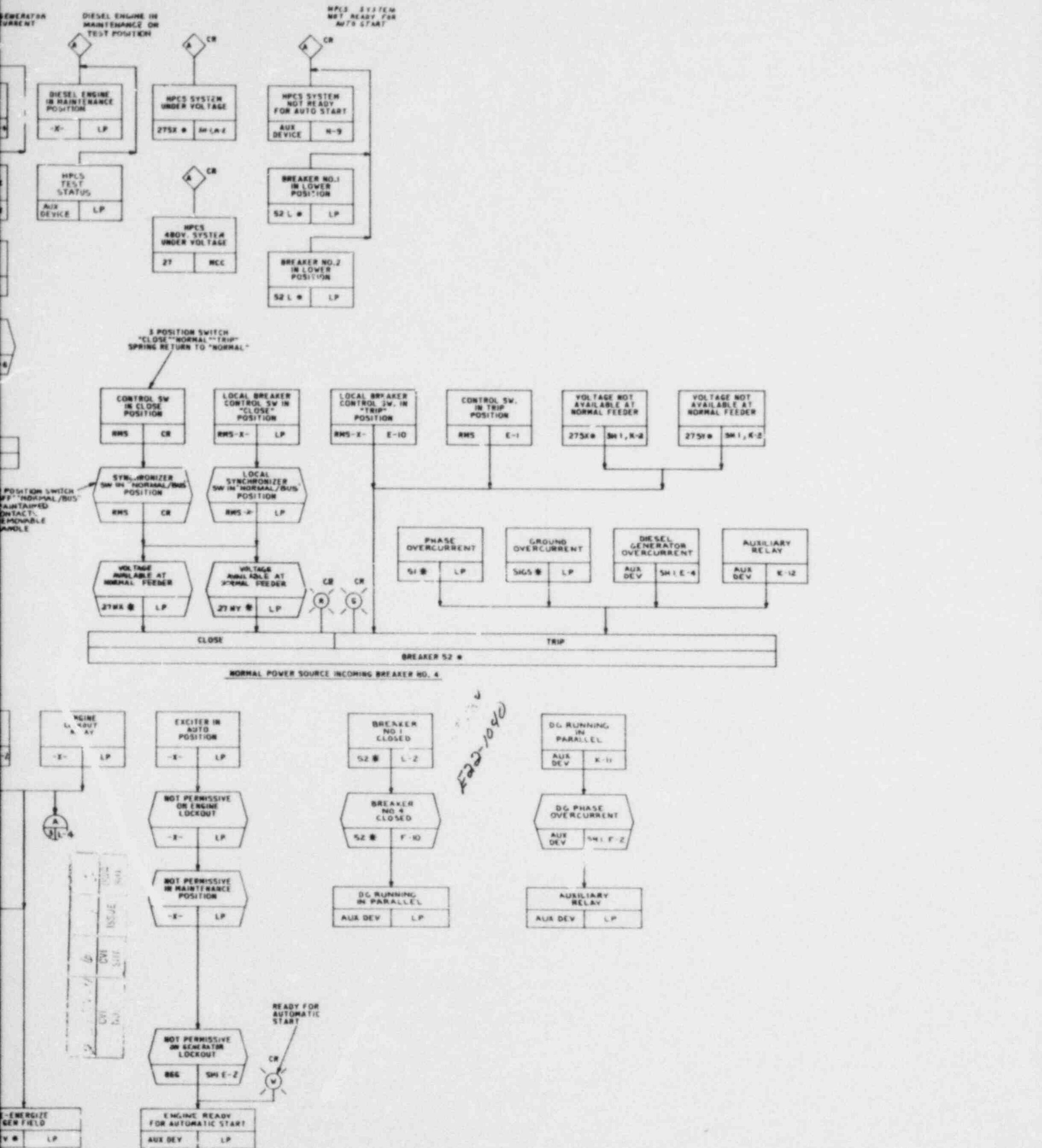
HPCS POWER SUPPLY SYSTEM
FUNCT. CONT. DGRM.

FIGURE 7.3-4a

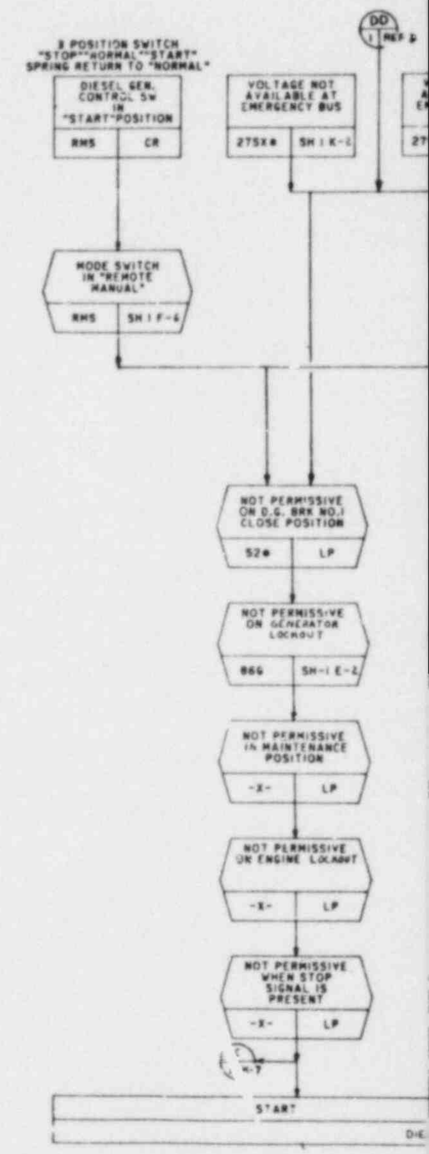


125 V DC SYSTEM TROUBLE
-X- LP
DIESEL ENGINE RUNNING
-X- LP
NORMAL SOURCE BREAKER TRIP
74* LP
480 V TRANSFORMER FEEDER TRIP
74* LP
HPCS PUMP MOTOR FEEDER TRIP
74* LP
DIESEL ENGINE OVERSPEED
-X- LP
DIESEL GENERATOR OVERCURRENT (VOLTAGE RESTRAINT)
AUX * DEVICE
DIESEL GENERATOR PHASE OVER CURRENT
AUX DEVICE
DIESEL ENGINE TROUBLE
-X- LP
HPCS CONTROL POWER FAILURE
AUX & DEVICE LP
DIESEL GENERATOR TRIP/LOCKOUT
74
3-POSITION SWITCH "LOWER SPEED" "OFF" "RAISE SPEED" SPRING RETURN TO "OFF"
SPEED REG. CONT SW. IN "RAISE SPEED" POSITION
RMS CR
LOCAL SPEED REG. CONTROL SW. IN "RAISE SPEED" POSITION
RMS-X- LP
LOCAL SPEED REG CONTROL SW. IN "LOWER SPEED" POSITION
RMS-X- C-5
SPEED REG. CONT SW IN "LOWER SPEED" POSITION
RMS C-4
2-POSITION SWITCH "REMOTE MANUAL" "LOCAL MANUAL"
MODE SWITCH IN "REMOTE MANUAL"
RMS SH I, F-6
MODE SWITCH IN "LOCAL MANUAL"
RMS SH I, F-6
MODE SWITCH IN "LOCAL MANUAL"
RMS SH I, F-6
MODE SWITCH IN "REMOTE MANUAL"
RMS
RAISE SPEED
LOWER SPEED
GOVERNOR SPEED CONTROL 65 -X-
DIESEL GENERATOR GOVERNOR CONTROL
3-POSITION SWITCH "CLOSE" "NORMAL" "TRIP" SPRING RETURN TO "NORMAL"
CONTROL SW. IN "CLOSE" POSITION
RMS CR
VOLTAGE NOT AVAILABLE AT EMERGENCY BUS
27SX* SH I K-2
VOLTAGE NOT AVAILABLE AT EMERGENCY BUS
27SY* SH I K-3
LOCAL BREAKER CONTROL SW. IN "CLOSE" POS.
RMS-X- LP
LOCAL BREAKER CONTROL SWITCH IN "TRIP" POSITION
RMS -X- H-4
CONTROL SW. IN "TRIP" POSITION
RMS
DIESEL GEN. LOCKOUT RELAY
86G*
2 POSITION SWITCH "OFF" "D.GEN/BUS" MAINTAINED CONTACTS REMOVABLE HANDLE
SYNCHRONIZER SW. IN "D GEN/BUS" POSITION
RMS CR
DIESEL GEN. AT RATED VOLTAGE AND SPEED
AUX -X- DEVICE LP
LOCAL SYCHRONIZER SW IN "D GEN/BUS" POSITION
RMS-X- LP
NORMAL POWER SOURCE INCOMING BKR. NO. 4 OPEN
52* F-12
NOT PERMISSIVE ON DIESEL GENERATOR LOCKOUT
86G SH I, E-2
CR
R
CR
G
B
3 M-6
CLOSE
BREAKER 52 *
DIESEL GENERATOR BREAKER NO. 1

Amendment No. 10
July 1980

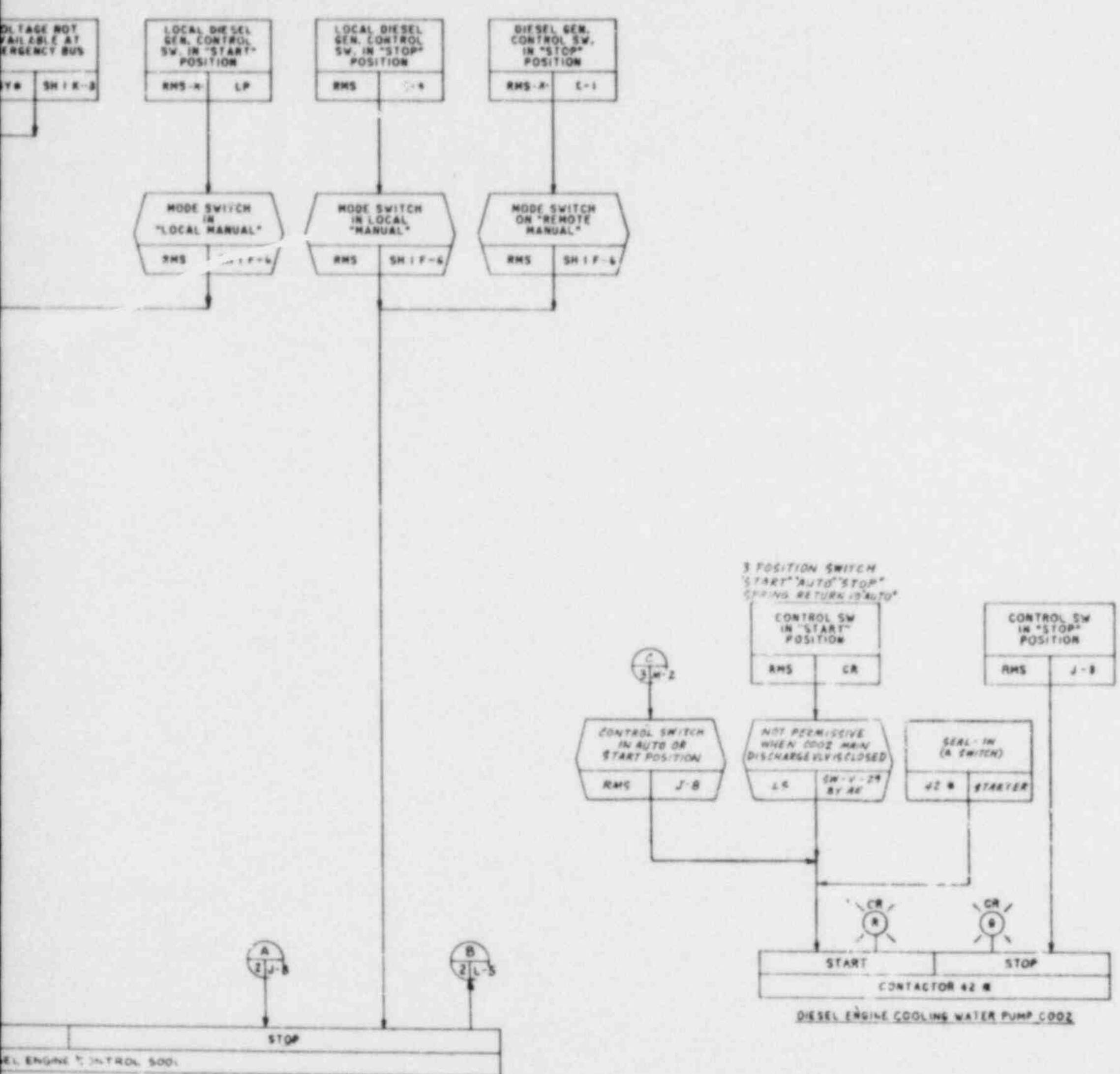


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HPCS POWER SUPPLY SYSTEM FUNCT. CONTR. DGRM.	FIGURE 7.3-4b

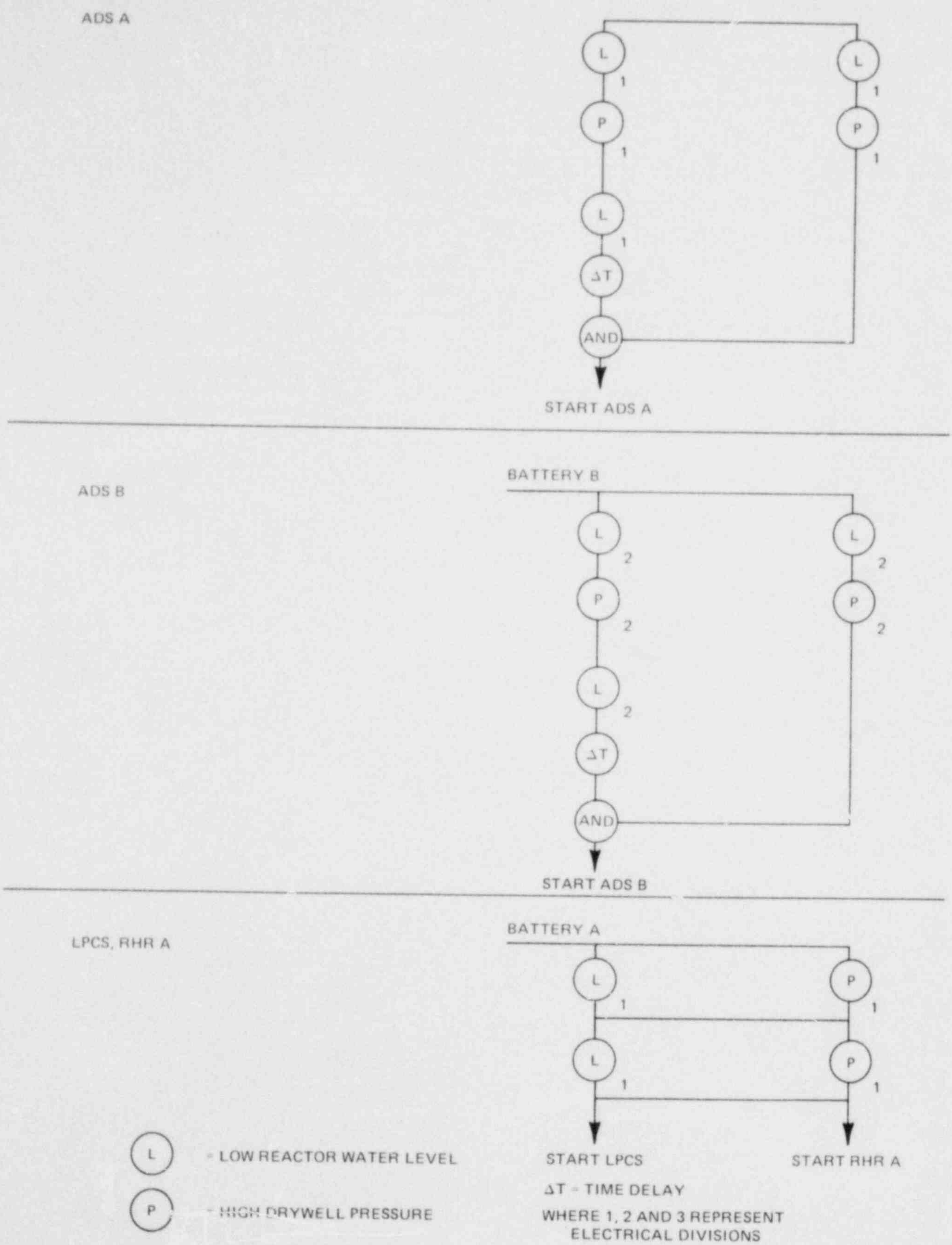


3 POSITION SWITCH
"STOP" "NORMAL" "START"
SPRING RETURN TO "NORMAL"
DIESEL GEN. CONTROL SW IN "START" POSITION
RMS
CR
DD
1
REF 2
VOLTAGE NOT AVAILABLE AT EMERGENCY BUS
275X*
SH 1 K-2
MODE SWITCH IN "REMOTE MANUAL"
RMS
SH 1 F-6
NOT PERMISSIVE ON D.G. BRK NO.1 CLOSE POSITION
52*
LP
NOT PERMISSIVE ON GENERATOR LOCKOUT
866
SH-1 E-2
NOT PERMISSIVE IN MAINTENANCE POSITION
-X-
LP
NOT PERMISSIVE ON ENGINE LOCKOUT
-X-
LP
NOT PERMISSIVE WHEN STOP SIGNAL IS PRESENT
-X-
LP
START

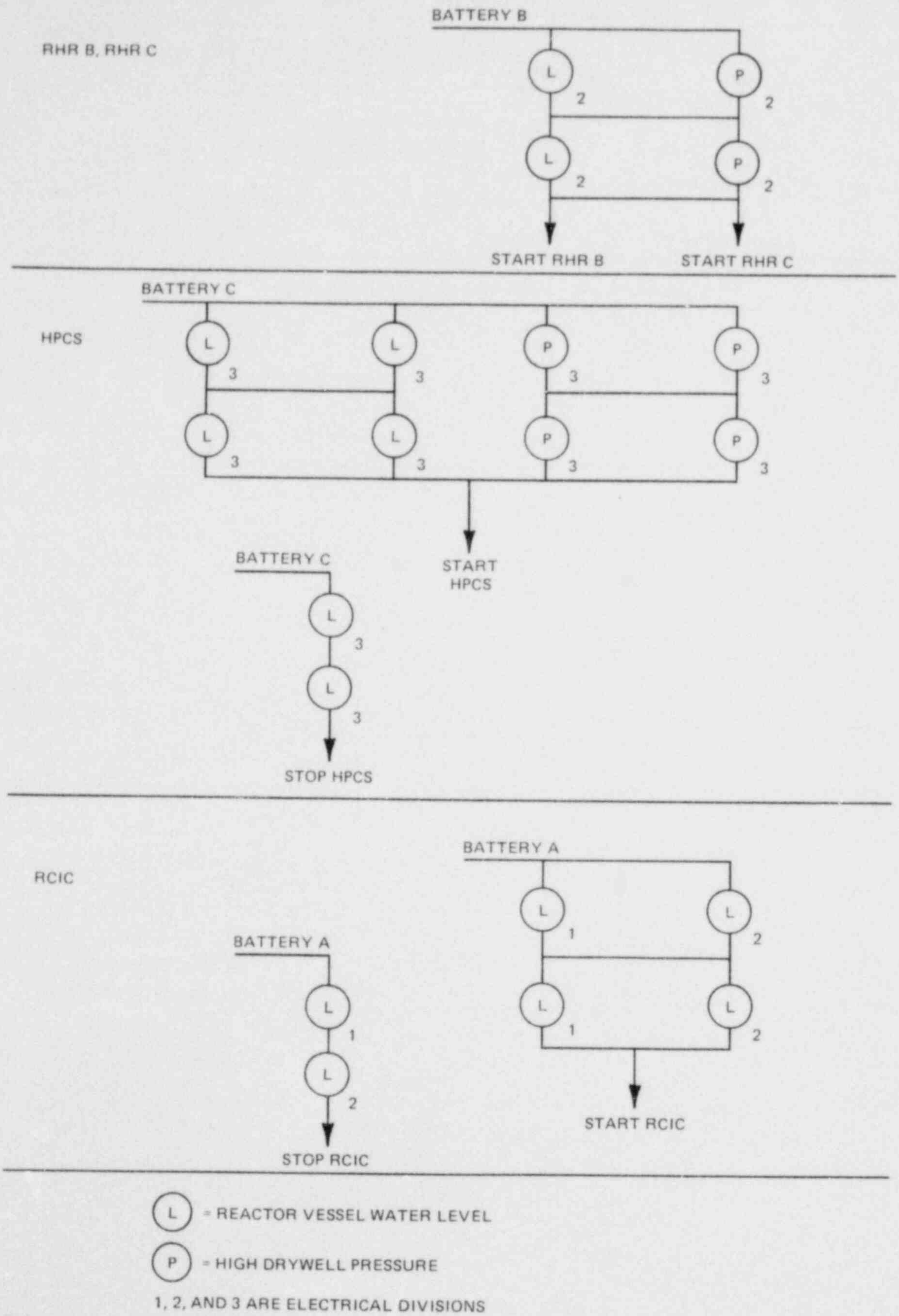
Amendment No. 10
July 1980



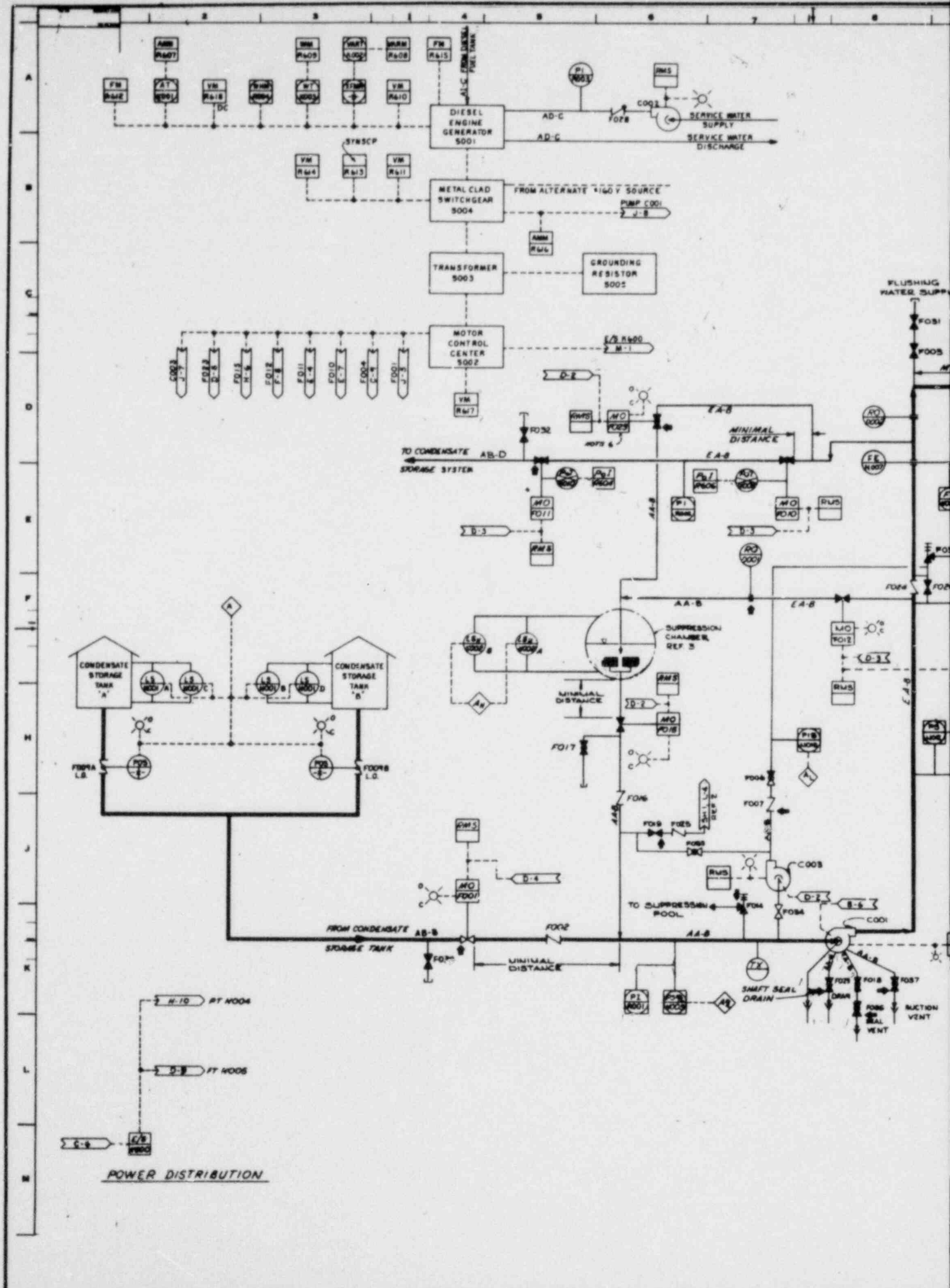
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HPCS POWER SUPPLY SYSTEM FUNCT. CONTR. DGRM.	FIGURE 7.3-4c



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	INITIATION LOGIC- ADS, LPCS, LPCI A.	FIGURE 7.3-5



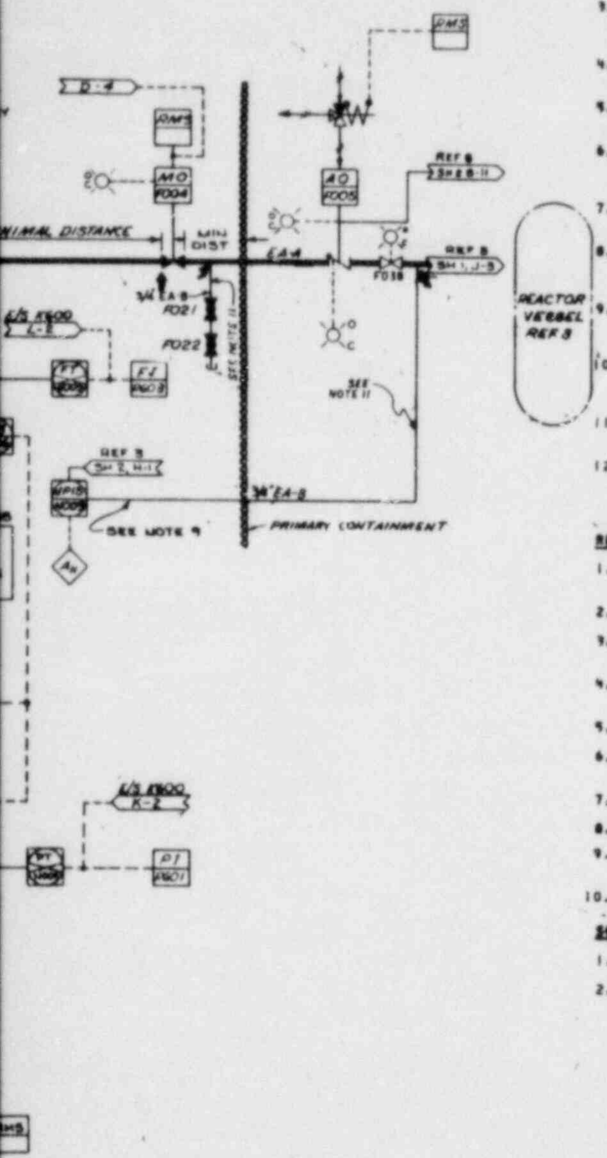
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	INITIATION LOGIC LPCI B AND C, HPCS, RCIC.	FIGURE 7.3-6



POWER DISTRIBUTION

AMENDMENT NO. 10
July 1980

FCF 238X185AD E22-1010



NOTES:

1. EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY SYSTEM NUMBER E22 UNLESS OTHERWISE NOTED.
2. PIPING HIGH POINT VENTS AND LOW POINT DRAINS ARE TO BE ADDED AS NECESSARY.
3. CHEMICAL CLEANING CONNECTIONS, VALVES, ETC., IF REQUIRED, ARE TO BE PROVIDED AS NECESSARY.
4. INSTRUMENT LINE DESIGN AND VALVING MUST COMPLY WITH INSTRUMENTATION SPEC, REF 7.
5. THE METHOD OF MOUNTING LOCAL INSTR IS TO BE DETERMINED BY OTHERS.
6. VALVE F029 SHALL BE INSTALLED WITH THE PACKING GLAND ON THE UPSTREAM SIDE OF THE VALVE DISC.
7. VALVE F010 SHALL BE LOCATED AS CLOSE AS PRACTICAL TO VALVE F011.
8. FOR ADDITIONAL CONTROL ROOM LIGHTS, SYS ALARMS AND REMOTE MANUAL SWITCHES, SEE REF 1 AND REF 4.
9. PROVISION FOR CONTAINMENT ISOLATION BY OTHERS TO BE IN ACCORDANCE WITH CURRENT LICENSING REQUIREMENTS.
10. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET REF. 9.
11. THIS LINE MAY BE CLASS "B" IF IT IS 3/4 INCH OR LESS IN DIAMETER.
12. FLUSHING CONNECTIONS SHALL BE PROVIDED IN ACCORDANCE WITH REF. 10. TEMPORARY STRAINER SCREENS SHALL BE PROVIDED ON THE SUCTION SIDE OF ALL PUMPS IN ACCORDANCE WITH REF. 10.

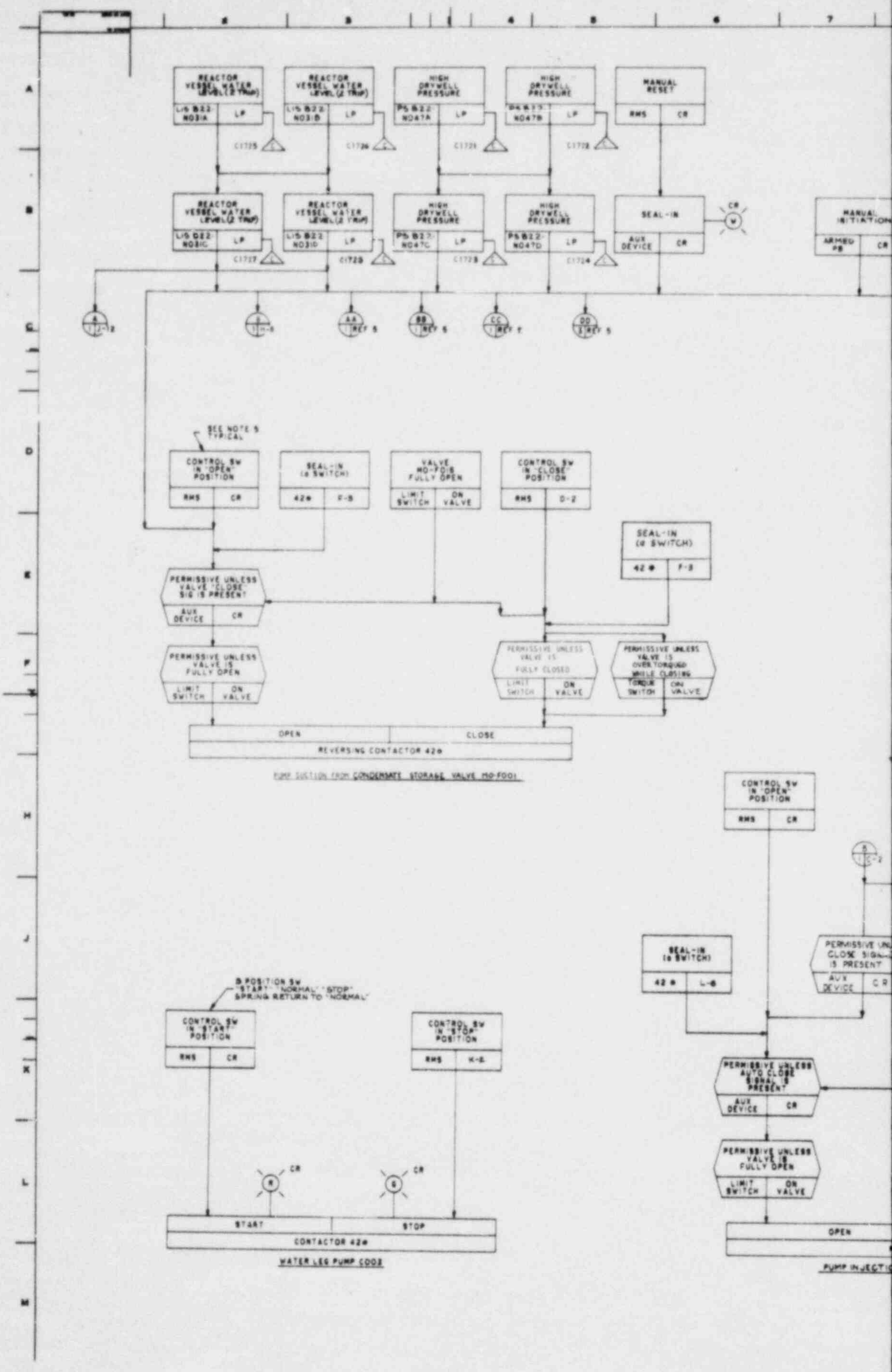
REFERENCE DOCUMENTS:	MPL ITEM NO.
1. HIGH PRESSURE CORE SPRAY FCD	E22-1030
2. RHR SYSTEM P&ID	E12-1010
3. NUCLEAR BOILER SYSTEM P&ID	B22-1010
4. HIGH PRESSURE CORE SPRAY ELECT ONE LINE DIA	E22-1060
5. HIGH PRESSURE CORE SPRAY P.D.	E22-1020
6. HIGH PRESSURE CORE SPRAY DESIGN SPECIFICATION	E22-4010
7. PROCESS INSTRUMENTATION	A62-4070
8. LEAK DETECTION SYS P&ID	E31-1010
9. HIGH PRESSURE CORE SPRAY IDS	E22-3050
10. CLEANING OF PIPING & EQUIPMENT	A62-4140

SUPPORTING DOCUMENTS:	MPL ITEM NO.
1. PIPING & INSTR SYMBOLS	A42-1010
2. PRESSURE INTEGRITY OF PIPING & EQUIP PRESSURE PARTS	A62-4030

LEGEND

WHM -------WATTHOUR METER
WM -------WATTMETER
WT -------WATT TRANSDUCER
VARM-------VARMETER
VART-------VARTRANSDUCER
AMM -------AMMETER
VM -------VOLTMETER
FM -------FREQUENCY METER
SYNSCP ----SYNCHROSCOPE
XFMR -----TRANSFORMER
AT-------CURRENT TRANSDUCER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HPCS P&ID	FIGURE 7.3-7



REACTOR VESSEL WATER LEVEL(2 TRIP)
LIS B22-N031A
LP
REACTOR VESSEL WATER LEVEL(2 TRIP)
LIS B22-N031B
HIGH DRYWELL PRESSURE
PS B22-N047A
HIGH DRYWELL PRESSURE
PS B22-N047B
MANUAL RESET
RMS
CR
REACTOR VESSEL WATER LEVEL(2 TRIP)
LIS B22-N031C
REACTOR VESSEL WATER LEVEL(2 TRIP)
LIS B22-N031D
HIGH DRYWELL PRESSURE
PS B22-N047C
HIGH DRYWELL PRESSURE
PS B22-N047D
SEAL-IN
AUX DEVICE
MANUAL INITIATION
ARMED PB
SEE NOTE 5 TYPICAL
CONTROL SW IN "OPEN" POSITION
SEAL-IN (a SWITCH)
42a
VALVE MO-F018 FULLY OPEN
LIMIT SWITCH
ON VALVE
CONTROL SW IN "CLOSE" POSITION
SEAL-IN (a SWITCH)
PERMISSIVE UNLESS VALVE "CLOSE" SIG IS PRESENT
PERMISSIVE UNLESS VALVE IS FULLY OPEN
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE CLOSING
TORQUE SWITCH
OPEN
CLOSE
REVERSING CONTACTOR 42a
PUMP SUCTION FROM CONDENSATE STORAGE VALVE MO-F001
CONTROL SW IN "OPEN" POSITION
SEAL-IN (a SWITCH)
PERMISSIVE UNLESS CLOSE SIGNAL IS PRESENT
PERMISSIVE UNLESS AUTO CLOSE SIGNAL IS PRESENT
PERMISSIVE UNLESS VALVE IS FULLY OPEN
3 POSITION SW "START" "NORMAL" "STOP" SPRING RETURN TO "NORMAL"
CONTROL SW IN "START" POSITION
CONTROL SW IN "STOP" POSITION
START
STOP
CONTACTOR 42a
WATER LEG PUMP C002
PUMP INJECTIO

AMENDMENT NO. 10
July 1980

REF 258X185AD G1 (E22-1080)

NOTES:

1. A) PUMP-MOTOR COMBINATION STARTERS SHALL BE PROVIDED WITH THERMAL OVERLOADS WHICH TRIP ON OVERLOAD. BREAKERS SHALL PROVIDE SHORT CIRCUIT PROTECTION. TRIPPING OF EITHER TYPE OF [illegible] IS ANNUNCIATED VIA AN ALARM RELAY.

   B) VALVE MOTORS ARE TO BE PROVIDED WITH THERMAL OVERLOAD TRIPS & LOSS OF POWER ANNUNCIATOR. OVERLOAD TRIPS TO BE BYPASSED UNLESS VALVE UNDER TEST. IN ADDITION VALVE MOTOR CIRCUITS ARE TO BE PROVIDED WITH SHORT CIRCUIT CURRENT PROTECTION.

2. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY [illegible] NUMBER E22 UNLESS OTHERWISE NOTED.

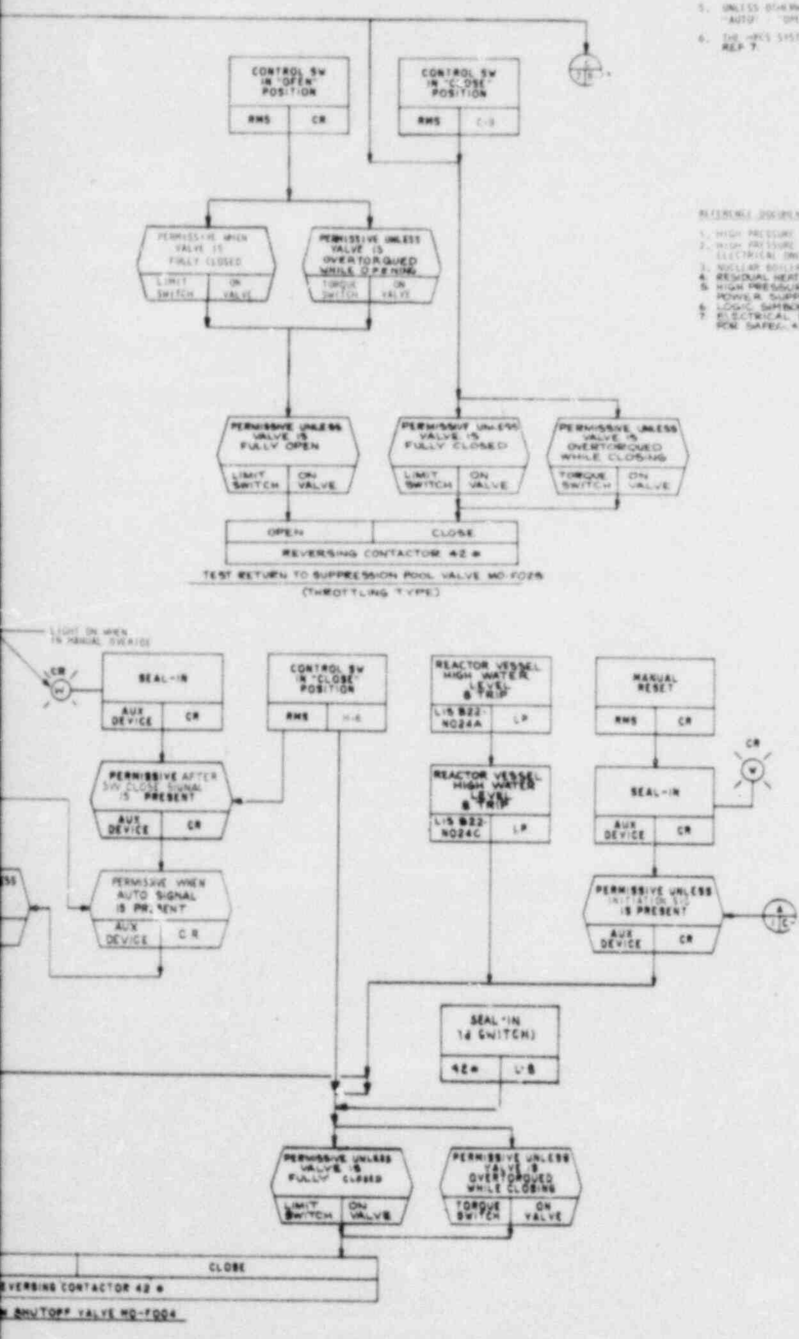
3. FOR ADDITIONAL ALARMS & PROCESS INSTRUMENTATION NOT SHOWN SEE REF. 1.

4. * SWITCHGEAR DEVICE FUNCTION NUMBERS ANSI-IEEE C37.2.

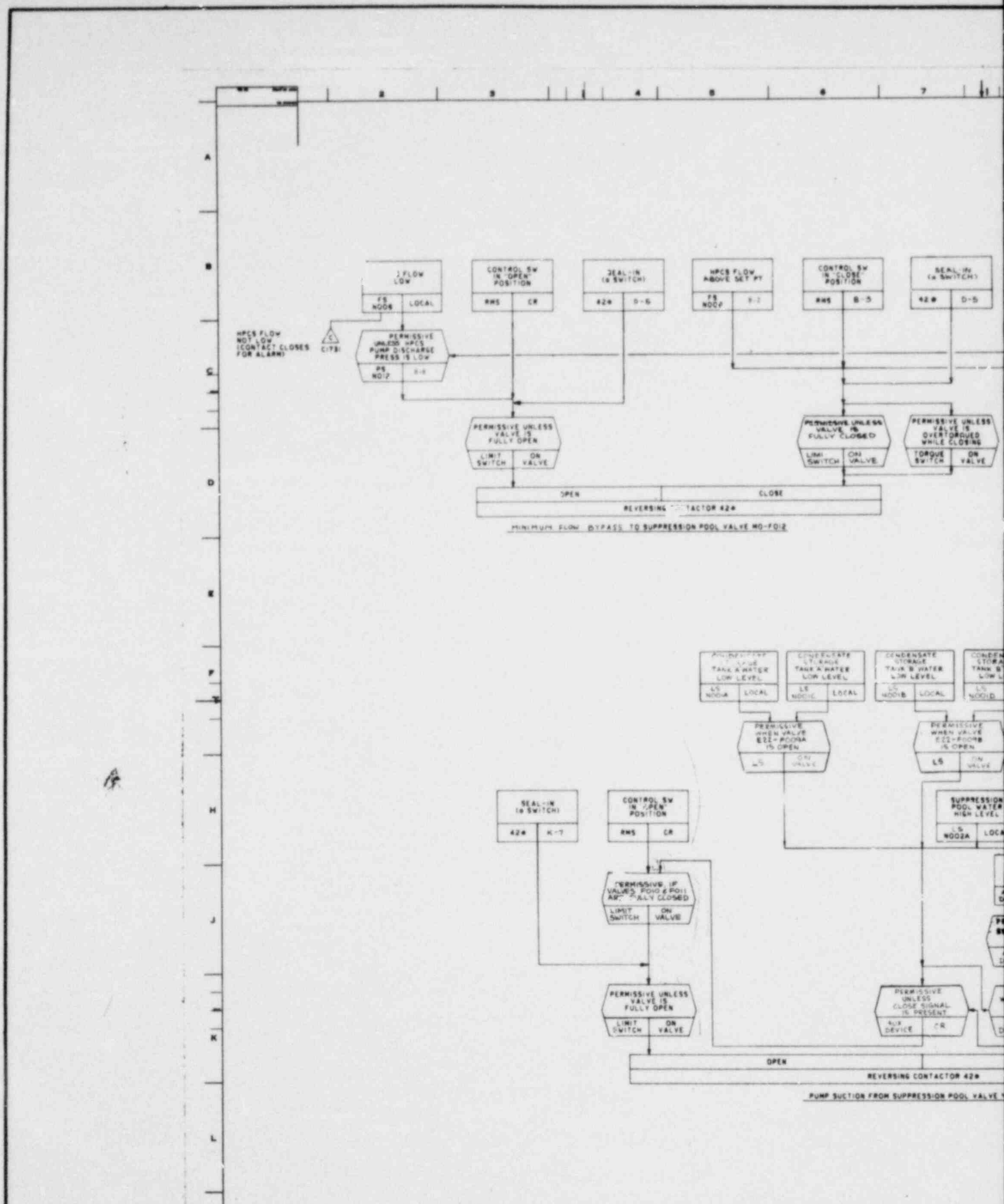
5. UNLESS OTHERWISE NOTED ALL RMS SHALL BE 3 POSITION SWITCHES, "CLOSE" - "AUTO" - "OPEN" SPRING RETURN TO "AUTO", FROM "CLOSE" - "OPEN".

6. THE HPCS SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH REF 7.

REFERENCE DOCUMENTS	MPL ITEM NO.
1. HIGH PRESSURE CORE SPRAY P&ID	E22-1010
2. HIGH PRESSURE CORE SPRAY ELECTRICAL ONE LINE DIAGRAM	E22-1060
3. NUCLEAR BOILER SYSTEM P&ID	B22-1010
4. RESIDUAL HEAT REMOVAL SYS P&ID	E12-1010
5. HIGH PRESSURE CORE SPRAY POWER SUPPLY FCD	E22-1040
6. LOGIC SYMBOLS	A42-1050
7. ELECTRICAL EQUIP. SEPARATION FOR SAFEGUARD SYSTEMS	A42-4050

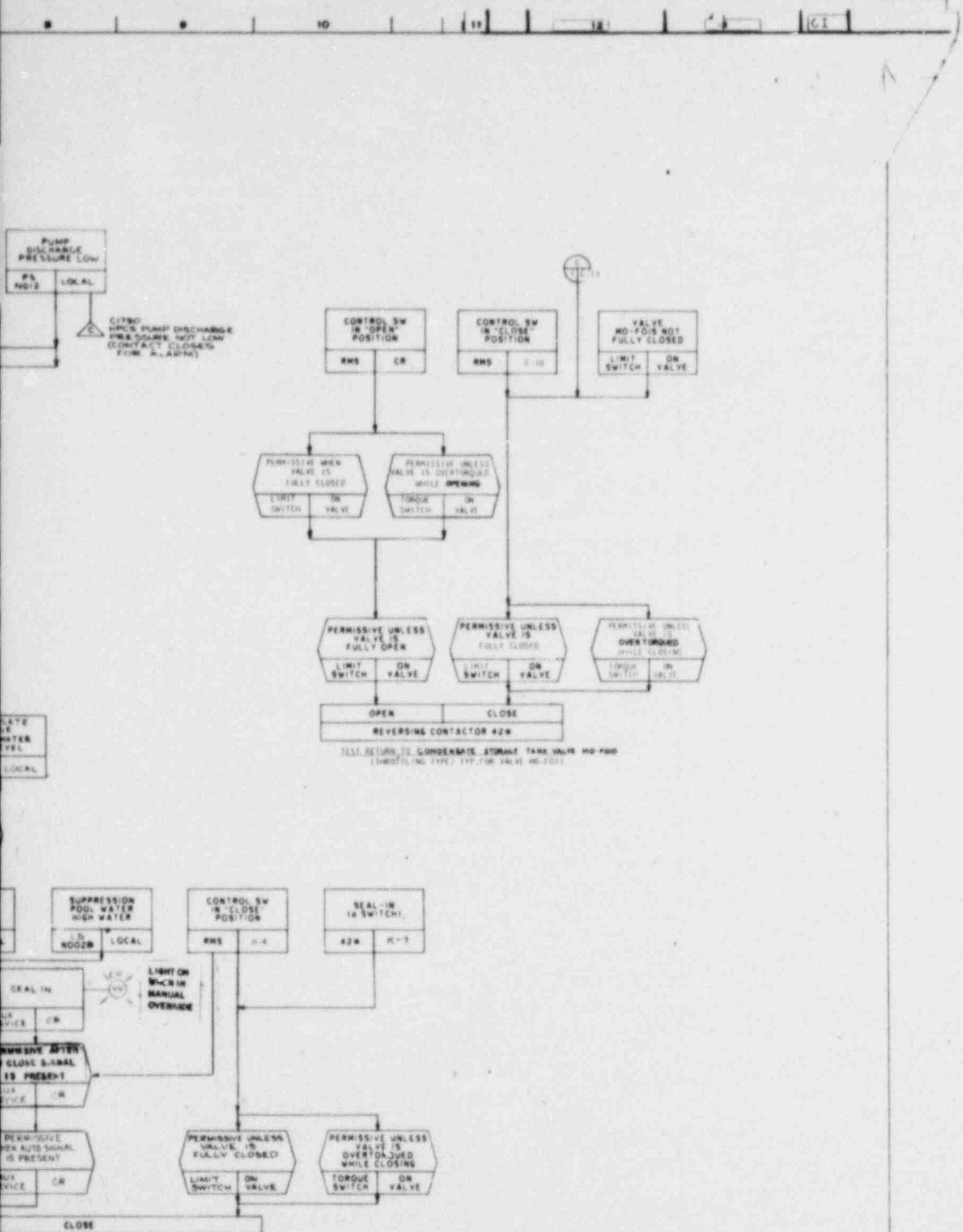


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY SYSTEM FCD SHEET 1	FIGURE 7.3-8a

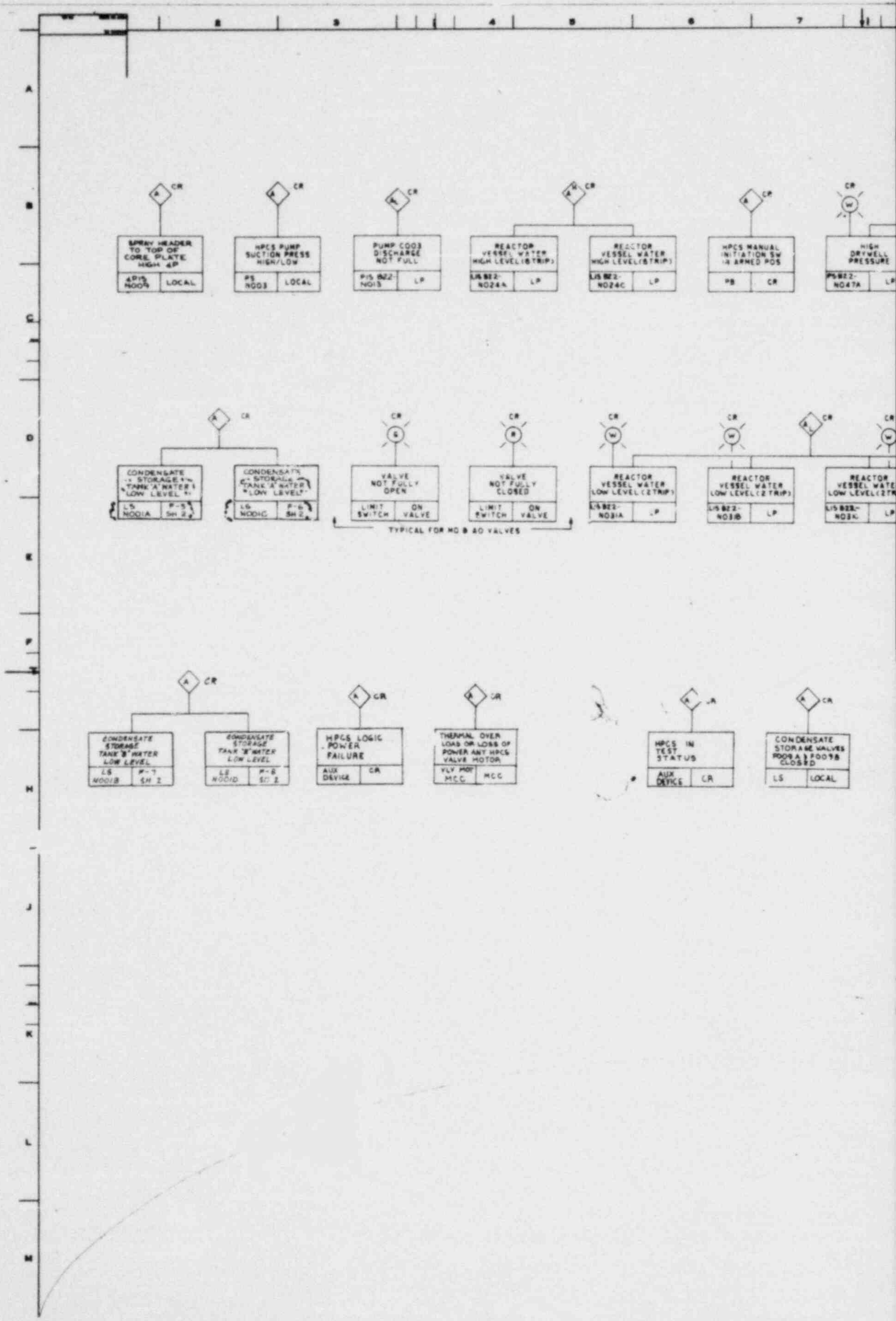


FLOW LOW
FS N006
LOCAL
CONTROL SW IN "OPEN" POSITION
RMS
CR
SEAL-IN (a SWITCH)
42#
D-6
HPCS FLOW ABOVE SET PT
FS N007
CONTROL SW IN "CLOSE" POSITION
RMS
B-5
SEAL-IN (a SWITCH)
42#
D-5
HPCS FLOW NOT LOW (CONTACT CLOSES FOR ALARM)
C(TS)
PERMISSIVE UNLESS HPCS PUMP DISCHARGE PRESS IS LOW
PS N012
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
LIMI SWITCH
ON VALVE
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE CLOSING
TORQUE SWITCH
ON VALVE
OPEN
CLOSE
REVERSING TACTOR 42#
MINIMUM FLOW BYPASS TO SUPPRESSION POOL VALVE MO-F012
CONDENSATE STORAGE TANK A WATER LOW LEVEL
LS N001A
LOCAL
LS N001C
LOCAL
CONDENSATE STORAGE TANK B WATER LOW LEVEL
LS N001B
LOCAL
LS N001D
PERMISSIVE WHEN VALVE E22-F009A IS OPEN
LS
ON VALVE
PERMISSIVE WHEN VALVE E22-F009B IS OPEN
LS
ON VALVE
SEAL-IN (a SWITCH)
42#
K-7
CONTROL SW IN "OPEN" POSITION
RMS
CR
SUPPRESSION POOL WATER HIGH LEVEL
LS N002A
PERMISSIVE IF VALVES F010 & F011 ARE FULLY CLOSED
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS CLOSE SIGNAL IS PRESENT
AUX DEVICE
CR
OPEN
REVERSING CONTACTOR 42#
PUMP SUCTION FROM SUPPRESSION POOL VALVE

AMENDMENT NO. 10
July 1980

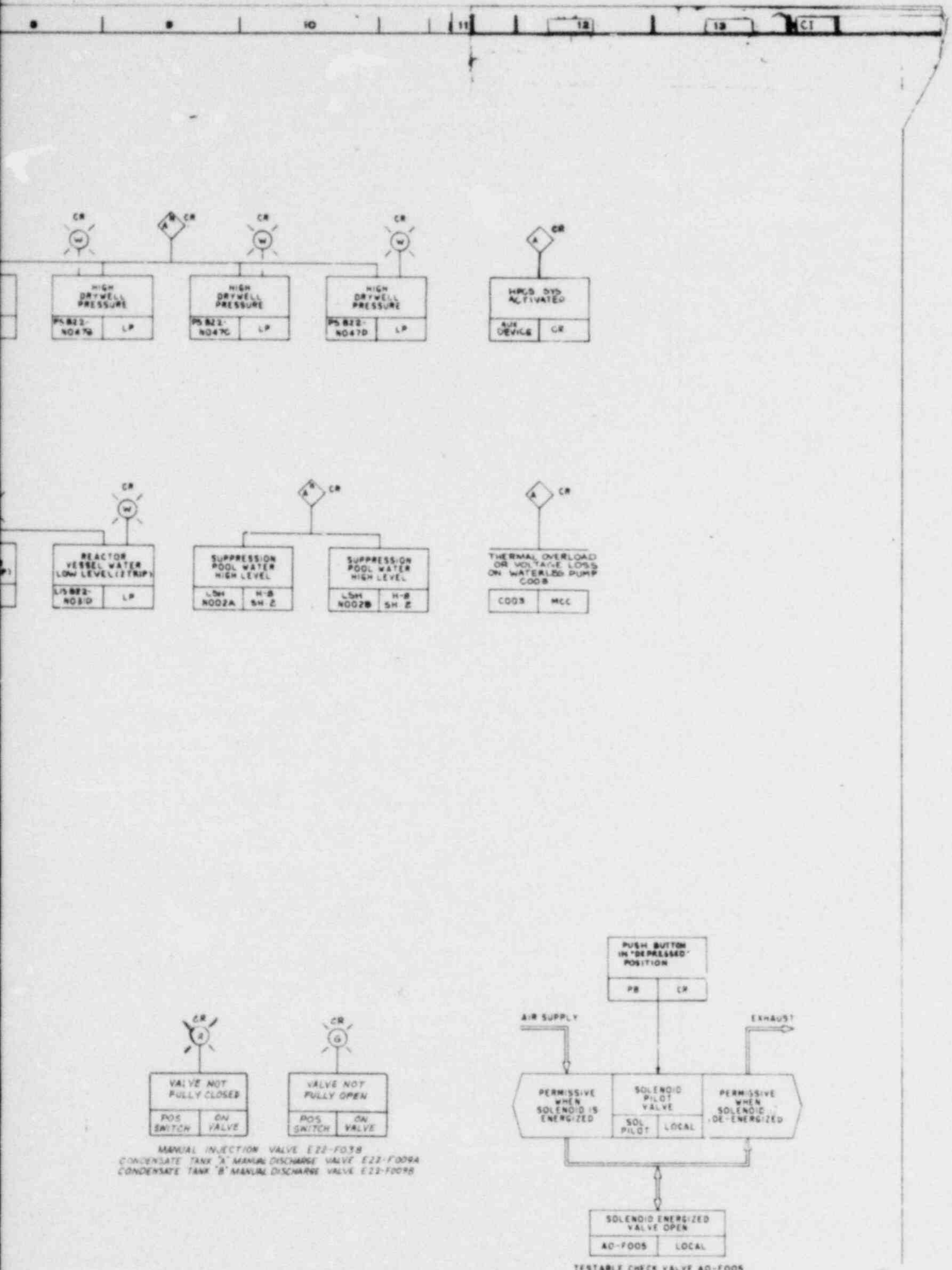


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY SYSTEM FCD SHEET 2	FIGURE 7.3-8b



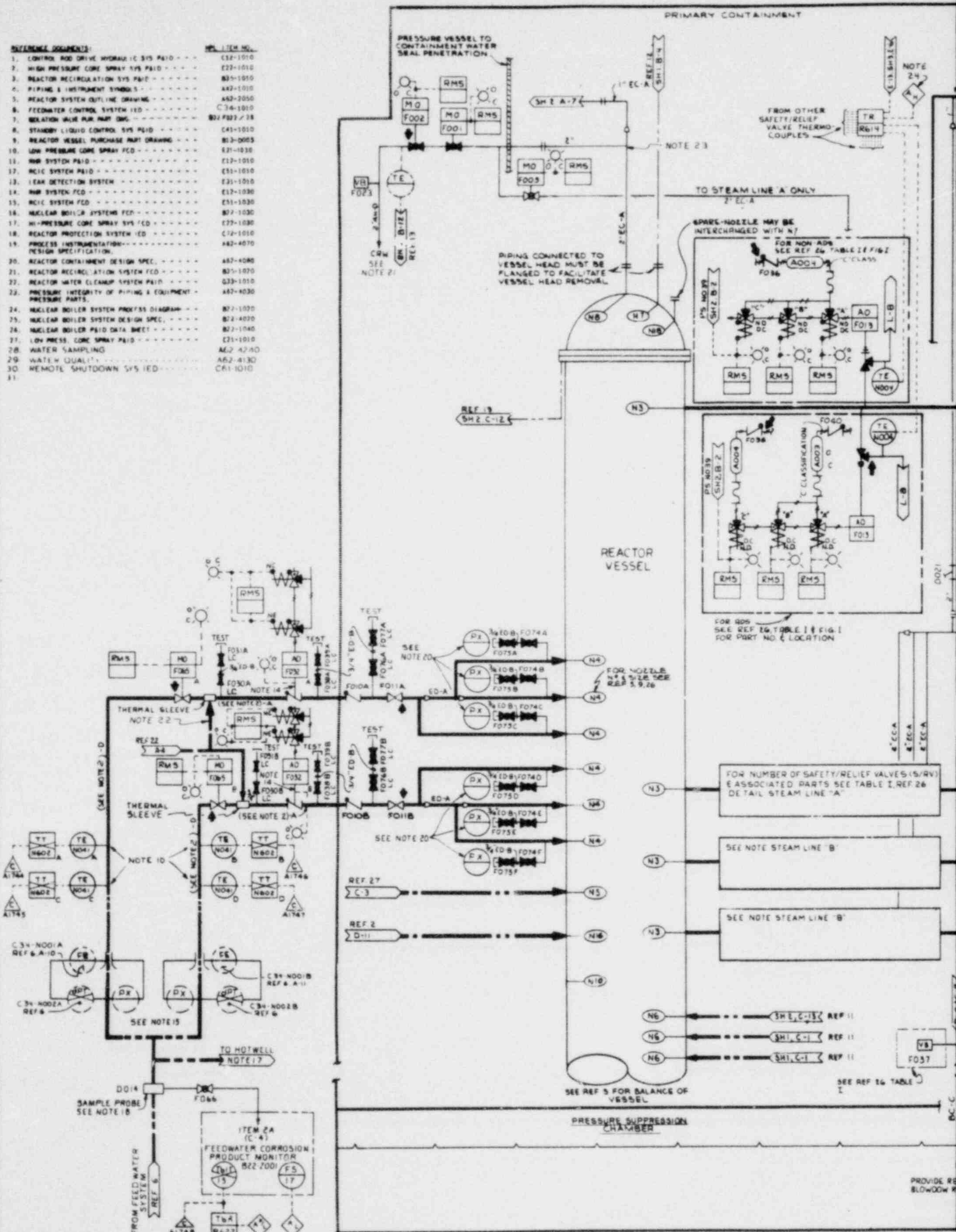
SPRAY HEADER TO TOP OF CORE PLATE HIGH ΔP
ΔPIS N004
LOCAL
HPCS PUMP SUCTION PRESS HIGH/LOW
PS N003
LOCAL
PUMP C003 DISCHARGE NOT FULL
PIS B22-N013
LP
REACTOR VESSEL WATER HIGH LEVEL (8 TRIP)
LIS B22-N024A
LP
REACTOR VESSEL WATER HIGH LEVEL (8 TRIP)
LIS B22-N024C
LP
HPCS MANUAL INITIATION SW IN ARMED POS
PB
CR
HIGH DRYWELL PRESSURE
PS B22-N047A
CONDENSATE STORAGE TANK 'A' WATER LOW LEVEL
LS N001A
P-5 SH 2
CONDENSATE STORAGE TANK 'A' WATER LOW LEVEL
LS N001C
P-6 SH 2
VALVE NOT FULLY OPEN
LIMIT SWITCH
ON VALVE
VALVE NOT FULLY CLOSED
LIMIT SWITCH
ON VALVE
TYPICAL FOR MO & AO VALVES
REACTOR VESSEL WATER LOW LEVEL (2 TRIP)
LIS B22-N031A
LP
REACTOR VESSEL WATER LOW LEVEL (2 TRIP)
LIS B22-N031B
LP
CONDENSATE STORAGE TANK 'B' WATER LOW LEVEL
LS N001B
P-7 SH 2
CONDENSATE STORAGE TANK 'B' WATER LOW LEVEL
LS N001D
P-8 SH 2
HPCS LOGIC POWER FAILURE
AUX DEVICE
CR
THERMAL OVER LOAD OR LOSS OF POWER ANY HPCS VALVE MOTOR
VLV MOT MCC
MCC
HPCS IN TEST STATUS
AUX DEVICE
CR
CONDENSATE STORAGE VALVES F009A & F009B CLOSED
LS
LOCAL

AMENDMENT NO. 10
July 1980

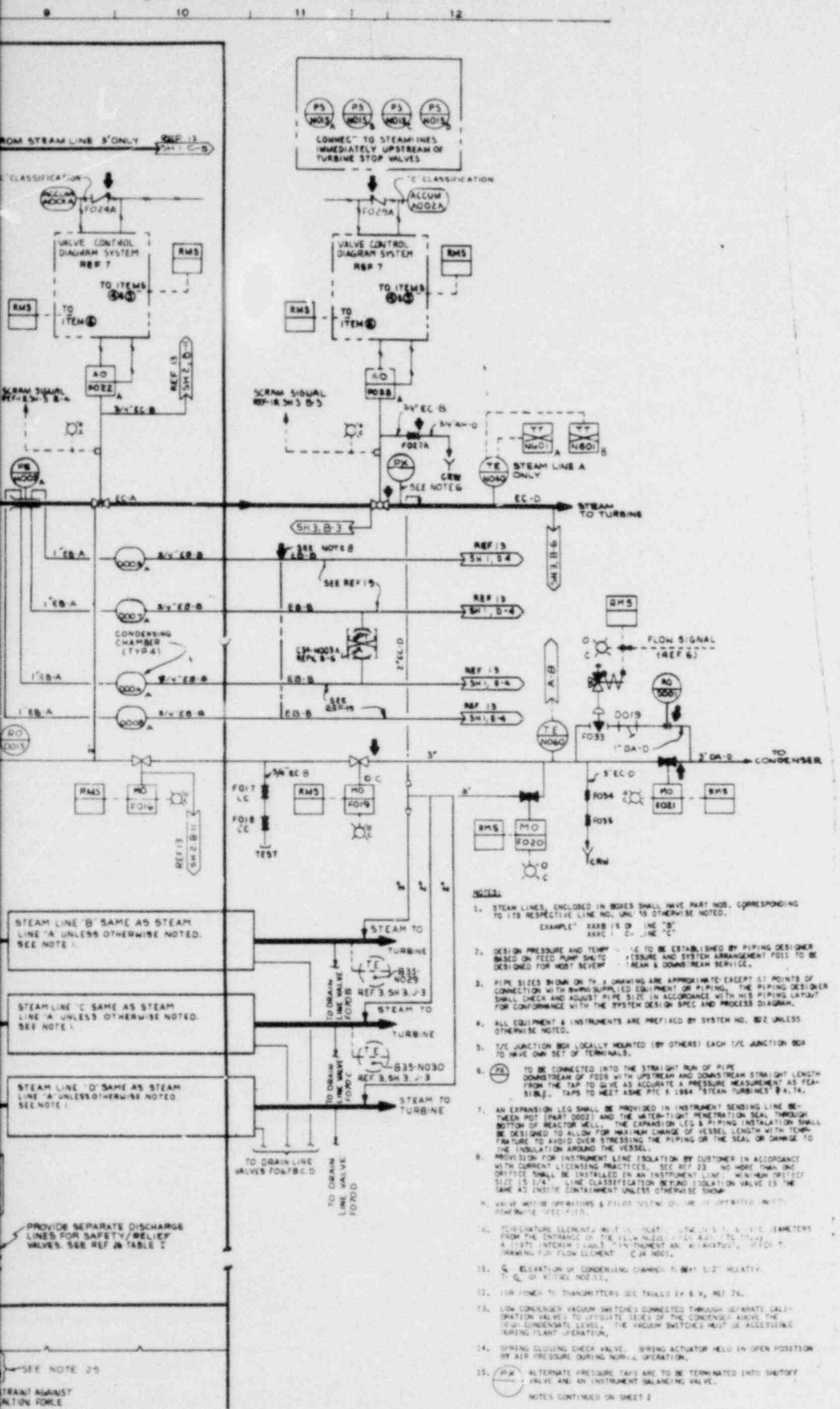


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY SYSTEM FCD SHEET 3	FIGURE 7.3-8c

REFERENCE DOCUMENTS:		NPL ITEM NO.
1.	CONTROL ROD DRIVE HYDRAULIC SYS P&ID	C12-1010
2.	HIGH PRESSURE CORE SPRAY SYS P&ID	E22-1010
3.	REACTOR RECIRCULATION SYS P&ID	B33-1010
4.	PIPING & INSTRUMENT SYMBOLS	A42-1010
5.	REACTOR SYSTEM OUTLINE DRAWING	A62-2050
6.	FEEDWATER CONTROL SYSTEM IED	C34-1010
7.	ISOLATION VALVE PUR. PART DWG.	B21-F022/28
8.	STANDBY LIQUID CONTROL SYS P&ID	C41-1010
9.	REACTOR VESSEL PURCHASE PART DRAWING	B13-D003
10.	LOW PRESSURE CORE SPRAY FCD	E21-1030
11.	RHR SYSTEM P&ID	E12-1010
12.	RCIC SYSTEM P&ID	E51-1010
13.	LEAK DETECTION SYSTEM	E31-1010
14.	RHR SYSTEM FCD	E12-1030
15.	RCIC SYSTEM FCD	E51-1030
16.	NUCLEAR BOILER SYSTEMS FCD	B22-1030
17.	HI-PRESSURE CORE SPRAY SYS FCD	E22-1030
18.	REACTOR PROTECTION SYSTEM IED	C72-1010
19.	PROCESS INSTRUMENTATION DESIGN SPECIFICATION.	A62-4070
20.	REACTOR CONTAINMENT DESIGN SPEC.	A62-4080
21.	REACTOR RECIRCULATION SYSTEM FCD	B33-1070
22.	REACTOR WATER CLEANUP SYSTEM P&ID	G33-1010
23.	PRESSURE INTEGRITY OF PIPING & EQUIPMENT PRESSURE PARTS.	A62-4030
24.	NUCLEAR BOILER SYSTEM PROCESS DIAGRAM	B22-1070
25.	NUCLEAR BOILER SYSTEM DESIGN SPEC.	B22-4020
26.	NUCLEAR BOILER P&ID DATA SHEET	B22-1040
27.	LOW PRESS. CORE SPRAY P&ID	E21-1010
28.	WATER SAMPLING	A62-4240
29.	WATER QUALITY	A62-4130
30.	REMOTE SHUTDOWN SYS IED	C61-1010
31.		



AMENDMENT NO. 10
July 1980



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"

2. DESIGN PRESSURE AND TEMP [illegible] TO BE ESTABLISHED BY PIPING DESIGNER BASED ON FEED PUMP SHUTO[illegible] [illegible]SSURE AND SYSTEM ARRANGEMENT F011 TO BE DESIGNED FOR MOST SEVER[illegible] [illegible]TREAM & DOWNSTREAM SERVICE.

3. PIPE SIZES SHOWN ON THIS DRAWING ARE APPROXIMATE EXCEPT AT POINTS OF CONNECTION WITH BWR-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.

4. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NO. B22 UNLESS OTHERWISE NOTED.

5. T/C JUNCTION BOX LOCALLY MOUNTED (BY OTHERS) EACH T/C JUNCTION BOX TO HAVE OWN SET OF TERMINALS.

6. PX TO BE CONNECTED INTO THE STRAIGHT RUN OF PIPE 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" ¶4.74.

7. 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 INSTALATION SHALL BE DESIGNED TO ALLOW FOR MAXIMUM CHANGE OF VESSEL LENGTH WITH TEMPERATURE TO AVOID OVER STRESSING THE PIPING OR THE SEAL OR DAMAGE TO THE INSULATION AROUND THE VESSEL.

8. PROVISION FOR INSTRUMENT LINE ISOLATION BY CUSTOMER IN ACCORDANCE WITH CURRENT LICENSING PRACTICES. SEE REF 23. NO MORE THAN ONE ORIFICE SHALL BE INSTALLED IN AN INSTRUMENT LINE. MINIMUM ORIFICE SIZE IS 1/4". LINE CLASSIFICATION BEYOND ISOLATION VALVE IS THE SAME AS INSIDE CONTAINMENT UNLESS OTHERWISE SHOWN.

9. VALVE MOTOR OPERATORS & PILOT [illegible] ARE [illegible] OPERATED UNLESS OTHERWISE SPECIFIED.

10. TEMPERATURE ELEMENTS MUST [illegible] DIAMETERS FROM THE ENTRANCE OF THE FLOW NOZZLE [illegible] 4 (1971 INTERIM [illegible]) "INSTRUMENT AND APPARATUS". [illegible] DRAWING FOR FLOW ELEMENT C34 N001.

11. ℄ ELEVATION OF CONDENSING CHAMBER TO BE ± 1/2" RELATIVE TO ℄ OF VESSEL NOZZLE.

12. FOR POWER TO TRANSMITTERS SEE TABLES IV & V, REF 76.

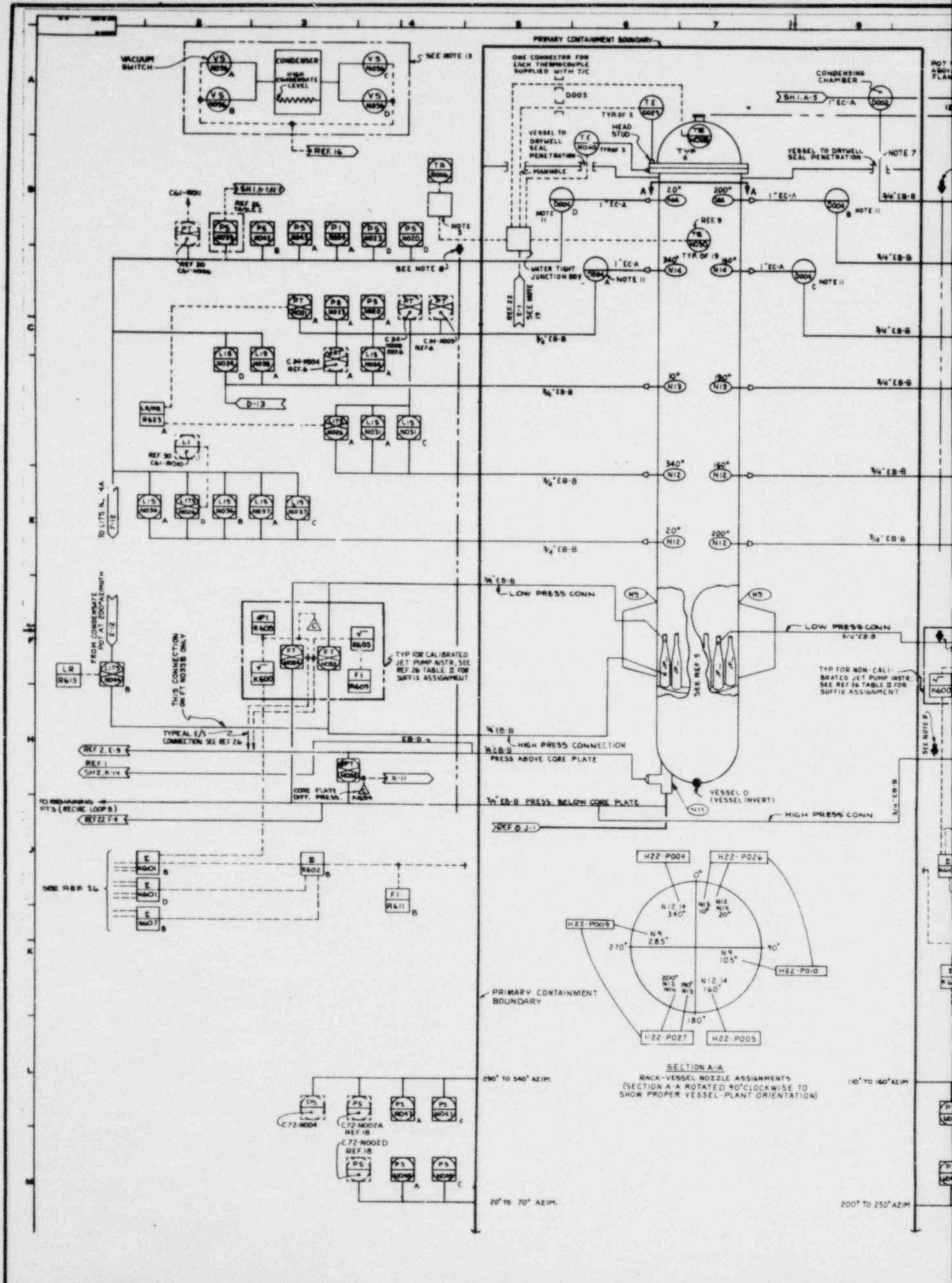
13. LOW CONDENSER VACUUM SWITCHES CONNECTED THROUGH SEPARATE CALIBRATION VALVES TO OPPOSITE SIDES OF THE CONDENSER ABOVE THE HIGH CONDENSATE LEVEL. THE VACUUM SWITCHES MUST BE ACCESSIBLE DURING PLANT OPERATION.

14. SPRING CLOSING CHECK VALVE. SPRING ACTUATOR HELD IN OPEN POSITION BY AIR PRESSURE DURING NORMAL OPERATION.

15. PX ALTERNATE PRESSURE TAPS ARE TO BE TERMINATED INTO SHUTOFF VALVE AND AN INSTRUMENT BALANCING VALVE.

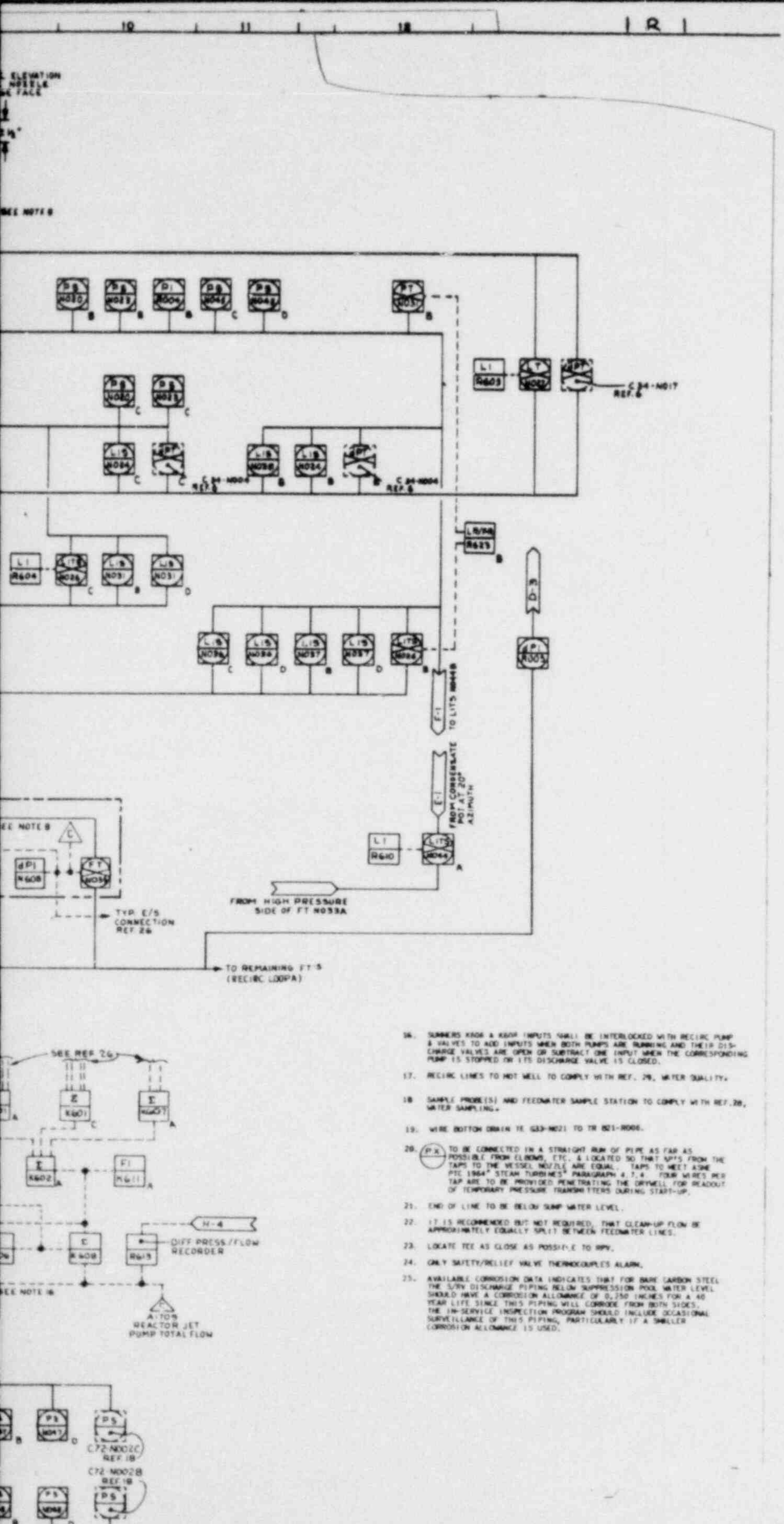
NOTES CONTINUED ON SHEET 2

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM P&ID SHEET 1	FIGURE 7.3-9a



PRIMARY CONTAINMENT BOUNDARY
VACUUM SWITCH
CONDENSER
HIGH CONDENSATE LEVEL
SEE NOTE 13
REF 16
ONE CONNECTOR FOR EACH THERMOCOUPLE SUPPLIED WITH T/C
D005
TYP OF 2
VESSEL TO DRYWELL SEAL PENETRATION
MANHOLE
HEAD STUD
CONDENSING CHAMBER
SH 1, A-5
1" EC-A
VESSEL TO DRYWELL SEAL PENETRATION
NOTE 7
NOTE 11
SEE NOTE 8
WATER TIGHT JUNCTION BOX
TYP OF 19
3/4" EB-B
LOW PRESS CONN
HIGH PRESS CONNECTION
PRESS ABOVE CORE PLATE
3/4" EB-B PRESS BELOW CORE PLATE
CORE PLATE DIFF. PRESS.
VESSEL D (VESSEL INVERT)
HIGH PRESS CONN
TYP FOR CALIBRATED JET PUMP INSTR, SEE REF 26 TABLE II FOR SUFFIX ASSIGNMENT
TYP FOR NON-CALIBRATED JET PUMP INSTR. SEE REF 26 TABLE II FOR SUFFIX ASSIGNMENT
THIS CONNECTION ON FT N003B ONLY
FROM CONDENSATE POT AT 200° AZIMUTH
TYPICAL E/S CONNECTION SEE REF 26
REF 2, E-9
REF 1 SH 2, K-14
TO REMAINING FTS (RECIRC LOOP B)
REF 22 F-4
REF 8, J-1
SEE REF 26
TO LITS N…4A
H22-P004
H22-P026
H22-P009
H22-P010
H22-P027
H22-P005
SECTION A-A
RACK-VESSEL NOZZLE ASSIGNMENTS
(SECTION A-A ROTATED 90° CLOCKWISE TO SHOW PROPER VESSEL-PLANT ORIENTATION)
PRIMARY CONTAINMENT BOUNDARY
290° TO 340° AZIM.
110° TO 160° AZIM
20° TO 70° AZIM.
200° TO 250° AZIM
C72-N004
C72-N002A REF 18
C72-N002D REF 18

AMENDMENT NO. 10
July 1980



16. SUMMERS K606 & K607 INPUTS SHALL BE INTERLOCKED WITH RECIRC PUMP & VALVES TO ADD INPUTS WHEN BOTH PUMPS ARE RUNNING AND THEIR DISCHARGE VALVES ARE OPEN OR SUBTRACT ONE INPUT WHEN THE CORRESPONDING PUMP IS STOPPED OR ITS DISCHARGE VALVE IS CLOSED.

17. RECIRC LINES TO HOT WELL TO COMPLY WITH REF. 28, WATER QUALITY.

18. SAMPLE PROBE(S) AND FEEDWATER SAMPLE STATION TO COMPLY WITH REF.28, WATER SAMPLING.

19. WIRE BOTTOM DRAIN TE G33-N021 TO TR B21-R006.

20. (PX) TO BE CONNECTED IN A STRAIGHT RUN OF PIPE AS FAR AS POSSIBLE FROM ELBOWS, ETC. & LOCATED SO THAT SP'S FROM THE TAPS TO THE VESSEL NOZZLE ARE EQUAL. TAPS TO MEET ASME PTC 1964" STEAM TURBINES" PARAGRAPH 4.7.4 FOUR WIRES PER TAP ARE TO BE PROVIDED PENETRATING THE DRYWELL FOR READOUT OF TEMPORARY PRESSURE TRANSMITTERS DURING START-UP.

21. END OF LINE TO BE BELOW SUMP WATER LEVEL.

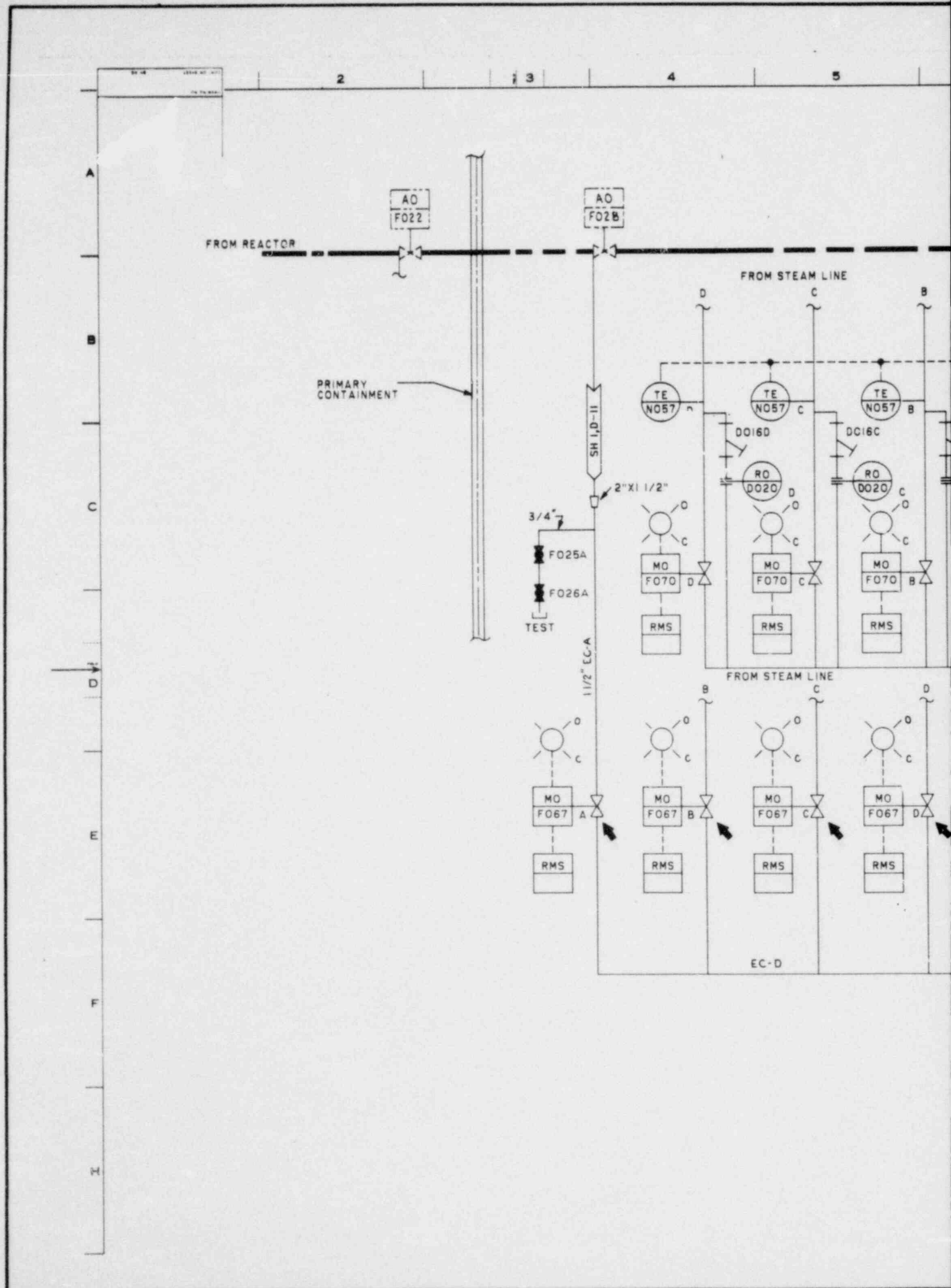
22. IT IS RECOMMENDED BUT NOT REQUIRED, THAT CLEAN-UP FLOW BE APPROXIMATELY EQUALLY SPLIT BETWEEN FEEDWATER LINES.

23. LOCATE TEE AS CLOSE AS POSSIBLE TO RPV.

24. ONLY SAFETY/RELIEF VALVE THERMOCOUPLES ALARM.

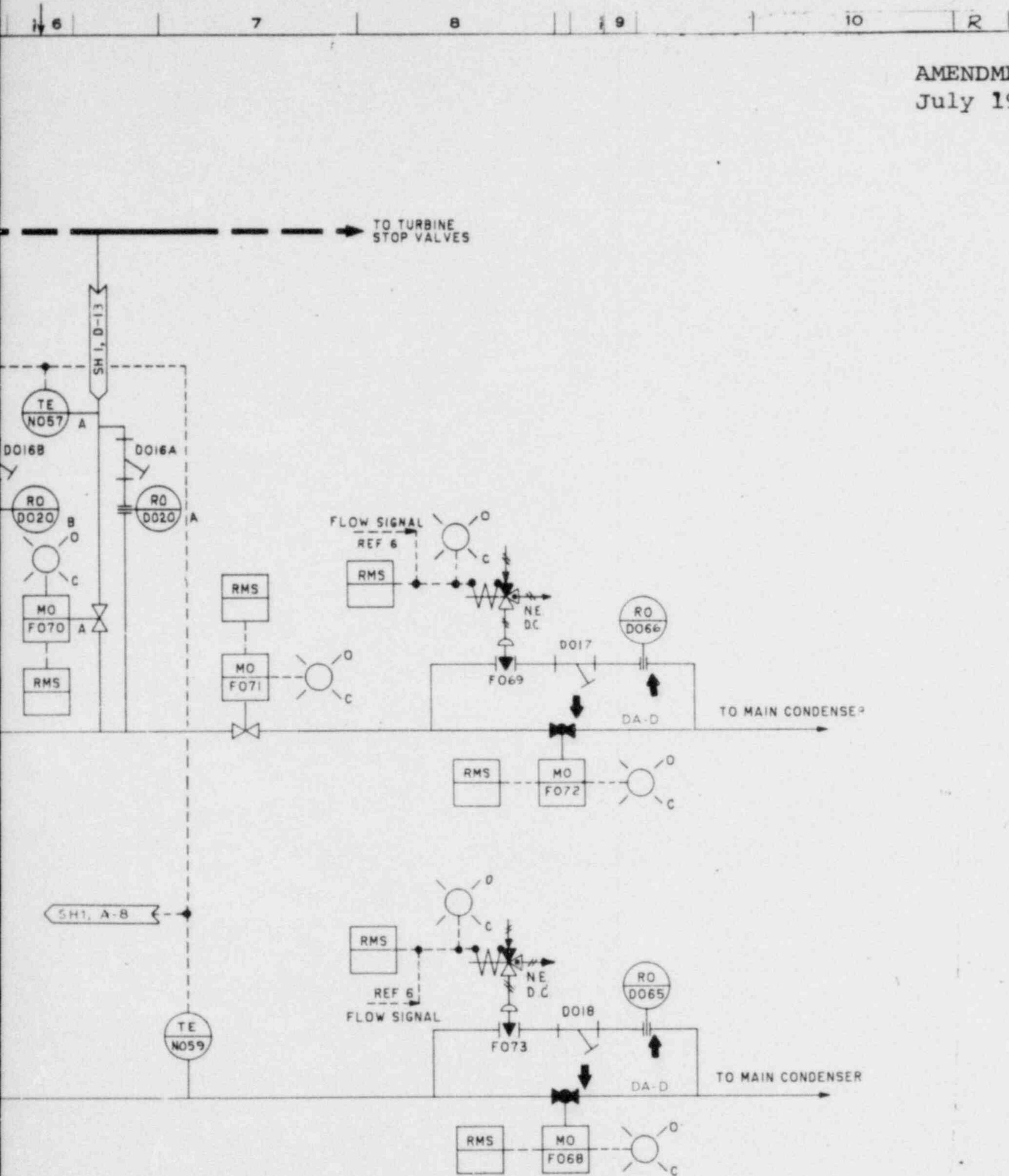
25. AVAILABLE CORROSION DATA INDICATES THAT FOR BARE CARBON STEEL THE S/RV DISCHARGE PIPING BELOW SUPPRESSION POOL WATER LEVEL SHOULD HAVE A CORROSION ALLOWANCE OF 0.250 INCHES FOR A 40 YEAR LIFE SINCE THIS PIPING WILL CORRODE FROM BOTH SIDES. THE IN-SERVICE INSPECTION PROGRAM SHOULD INCLUDE OCCASIONAL SURVEILLANCE OF THIS PIPING, PARTICULARLY IF A SMALLER CORROSION ALLOWANCE IS USED.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM P&ID SHEET 2	FIGURE 7.3-9b

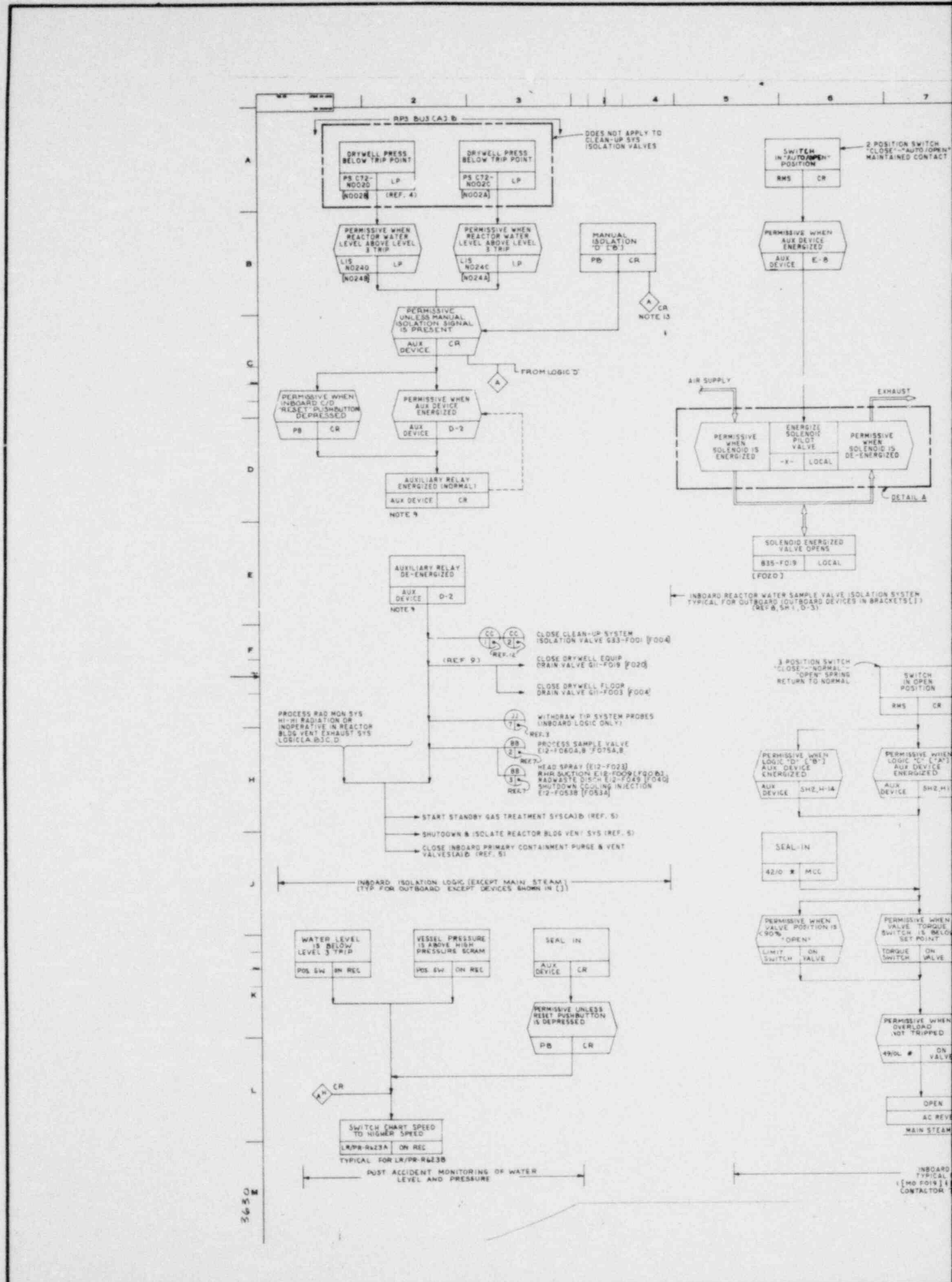


2
3
4
5
A
B
C
D
E
F
H
FROM REACTOR
AO
F022
AO
F028
FROM STEAM LINE
D
C
B
PRIMARY CONTAINMENT
SH 1,D-11
2"X1 1/2"
3/4"
F025A
F026A
TEST
1 1/2" EC-A
TE
N057
DO16D
DO16C
RO
DO20
MO
F070
RMS
FROM STEAM LINE
MO
F067
RMS
EC-D

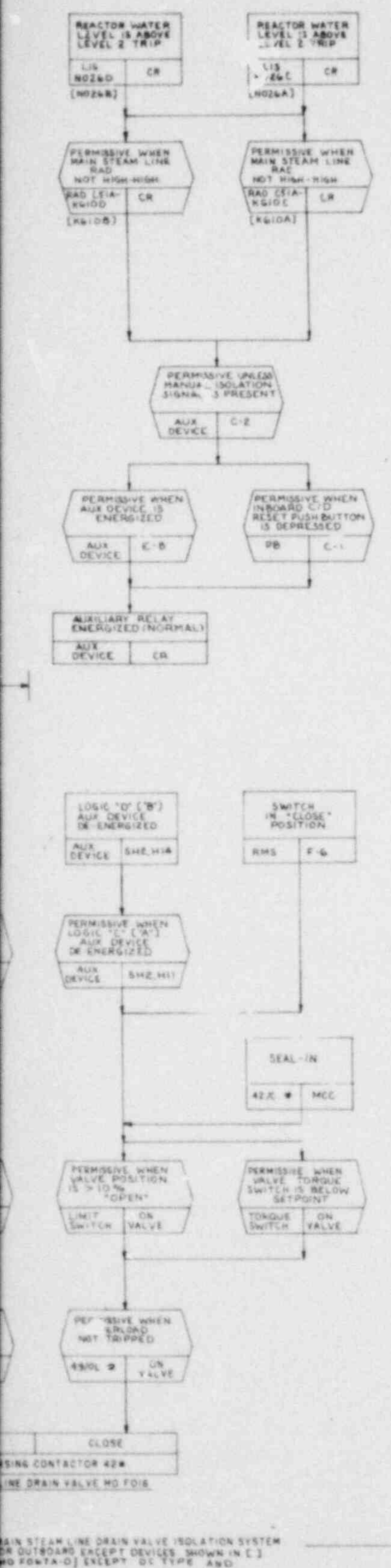
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM P&ID SHEET 3	FIGURE 7.3-9c


RPS BUS (A) B
DOES NOT APPLY TO CLEAN-UP SYS ISOLATION VALVES
DRYWELL PRESS. BELOW TRIP POINT
PS C72-N002D [N002B] LP (REF. 4)
DRYWELL PRESS. BELOW TRIP POINT
PS C72-N002C [N002A] LP
PERMISSIVE WHEN REACTOR WATER LEVEL ABOVE LEVEL 3 TRIP
LIS N024D [N024B] LP
PERMISSIVE WHEN REACTOR WATER LEVEL ABOVE LEVEL 3 TRIP
LIS N024C [N024A] LP
MANUAL ISOLATION "D" ["B"]
PB CR
NOTE 13
PERMISSIVE UNLESS MANUAL ISOLATION SIGNAL IS PRESENT
AUX DEVICE CR
FROM LOGIC "D"
PERMISSIVE WHEN INBOARD C/D "RESET" PUSHBUTTON DEPRESSED
PB CR
PERMISSIVE WHEN AUX DEVICE ENERGIZED
AUX DEVICE D-2
AUXILIARY RELAY ENERGIZED (NORMAL)
AUX DEVICE CR
NOTE 9
AUXILIARY RELAY DE-ENERGIZED
AUX DEVICE D-2
NOTE 9
CLOSE CLEAN-UP SYSTEM ISOLATION VALVE G33-F001 [F004]
CLOSE DRYWELL EQUIP DRAIN VALVE G11-F019 [F020]
CLOSE DRYWELL FLOOR DRAIN VALVE G11-F003 [F004]
(REF 9)
WITHDRAW TIP SYSTEM PROBES (INBOARD LOGIC ONLY)
PROCESS SAMPLE VALVE E12-F060A,B [F075A,B]
HEAD SPRAY (E12-F023)
RHR SUCTION E12-F009 [F008]
RADWASTE DISCH E12-F049 [F040]
SHUTDOWN COOLING INJECTION E12-F053B [F053A]
PROCESS RAD MON SYS HI-HI RADIATION OR INOPERATIVE IN REACTOR BLDG VENT EXHAUST SYS LOGIC(A,B)C,D
START STANDBY GAS TREATMENT SYS(A)B (REF. 5)
SHUTDOWN & ISOLATE REACTOR BLDG VENT SYS (REF. 5)
CLOSE INBOARD PRIMARY CONTAINMENT PURGE & VENT VALVES(A)B (REF. 5)
INBOARD ISOLATION LOGIC (EXCEPT MAIN STEAM)
(TYP FOR OUTBOARD EXCEPT DEVICES SHOWN IN [ ])
WATER LEVEL IS BELOW LEVEL 3 TRIP
POS. SW ON REC
VESSEL PRESSURE IS ABOVE HIGH PRESSURE SCRAM
POS. SW ON REC
SEAL IN
AUX DEVICE CR
PERMISSIVE UNLESS RESET PUSHBUTTON IS DEPRESSED
PB CR
SWITCH CHART SPEED TO HIGHER SPEED
LR/PR-R623A ON REC
TYPICAL FOR LR/PR-R623B
POST ACCIDENT MONITORING OF WATER LEVEL AND PRESSURE
SWITCH IN "AUTO/OPEN" POSITION
RMS CR
2 POSITION SWITCH "CLOSE"-"AUTO/OPEN" MAINTAINED CONTACT
PERMISSIVE WHEN AUX DEVICE ENERGIZED
AUX DEVICE E-8
AIR SUPPLY
EXHAUST
PERMISSIVE WHEN SOLENOID IS ENERGIZED
ENERGIZE SOLENOID PILOT VALVE
-X- LOCAL
PERMISSIVE WHEN SOLENOID IS DE-ENERGIZED
DETAIL A
SOLENOID ENERGIZED VALVE OPENS
B35-F019 LOCAL
[F020]
INBOARD REACTOR WATER SAMPLE VALVE ISOLATION SYSTEM
TYPICAL FOR OUTBOARD (OUTBOARD DEVICES IN BRACKETS [ ])
(REF 8, SH 1, D-3)
3 POSITION SWITCH "CLOSE"-"NORMAL"-"OPEN" SPRING RETURN TO NORMAL
SWITCH IN OPEN POSITION
RMS CR
PERMISSIVE WHEN LOGIC "D" ["B"] AUX DEVICE ENERGIZED
AUX DEVICE SH2,H-14
PERMISSIVE WHEN LOGIC "C" ["A"] AUX DEVICE ENERGIZED
SEAL-IN
42/O MCC
PERMISSIVE WHEN VALVE POSITION IS <90% "OPEN"
LIMIT SWITCH ON VALVE
PERMISSIVE WHEN VALVE TORQUE SWITCH IS BELOW SET POINT
TORQUE SWITCH ON VALVE
PERMISSIVE WHEN OVERLOAD NOT TRIPPED
49/OL ON VALVE
OPEN
AC REV
MAIN STEAM
INBOARD
TYPICAL
CONTACTOR
3630

AMENDMENT NO. 10
July 1980



NOTES:

1. WHEN TEST SOLENOID PILOT IS ENERGIZED, THE MAIN STEAM ISOLATION VALVE OPERATOR IS SLOWLY EXHAUSTED (60 SEC CLOSURE TIME) WHILE VALVE IS CLOSED BY ACTION OF THE VALVE SPRING WITHOUT AID OF AIR PRESSURE.
2. THE ALARMS AND VALVE INDICATING LIGHTS SHOWN ON THE FCD ARE SYSTEM REQUIREMENTS IN ADDITION TO THOSE SHOWN ON THE SYSTEM P&ID. ADDITIONAL INFORMATION ON ALARMS, VALVE POSITION INDICATING LIGHTS, AND PROCESS INSTRUMENTATION NOT SHOWN ON THIS FCD MAY BE OBTAINED FROM REF. 1. AUXILIARY RELAYS & DEVICES ARE NOT SHOWN ON FCD EXCEPT WHERE REQUIRED TO CLARIFY FUNCTION.
4. MAIN STEAM LINE ISOLATION LOGIC A & C, RPS "A" SOLENOID LOGIC AND THE VALVE TEST SOLENOID SHALL BE POWERED FROM THE REACTOR PROTECTION SYSTEM M-G SET A; LOGIC B & D AND RPS "B" SOLENOID LOGIC FROM RPS M-G SET B.
5. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY SYSTEM NO. B22 UNLESS OTHERWISE NOTED.
6. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET LISTED IN MPL (REF. 22)
7. AUTO DEPRESSURIZATION CONTROL LOGIC CIRCUIT SHALL HAVE REDUNDANT POWER SUPPLIES SO THAT A SINGLE FAILURE WILL NOT DISABLE THE AUTO DEPRESSURIZATION FUNCTION.
8. THE NUCLEAR BOILER SYS SHALL BE DESIGNED IN ACCORDANCE WITH REF. 18 AND WITH "PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEMS IEEE 279" AS APPLICABLE TO THE CONTROL CIRCUITRY.
9. ISOLATION LOGIC SHALL BE "FAILSAFE", I.E., LOGIC SHALL BE DESIGNED TO INITIATE ISOLATION FUNCTIONS WHEN DE-ENERGIZED.
10. REMOVED
11. REMOVED
12. THE DEVICES IN THIS AREA, AS WELL AS OTHER DEVICES IN THIS LOGIC ARE ESSENTIAL AND MUST MEET THE REQUIREMENTS OF IEEE 279.
13. ALARM FROM ROTARY CONTACT
14. AE TO SUPPLY DEVICE TYPE, NUMBERS AND QUANTITY.
15. SEE REF. 4 FOR INITIATION OF "SELECT ROD INSERT".
16. REMOVED
17. REMOVED
18. % REACTOR POWER DERIVED FROM FIRST STAGE TURBINE PRESSURE DETERMINE GROUP OF RELIEF VALVE TO BE OPEN.
19. FOR ADDITIONAL RECIRC PUMP A&B BREAKER CLOSE & TRIP LOGIC SEE REACTOR RECIRC SYS FCD (REF. 2)

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. NUCLEAR BOILER SYS P&ID	B22-1010
2. REACTOR RECIRC SYS FCD	B35-1070
3. NEUTRON MONITORING SYS FCD	C51-1020
4. REACTOR PROTECTION SYS IED	C72-1010
5. PROCESS RADIATION SYS IED	D13, D14 C17 & D18-1010
6. RHR SYS P&ID	E12-1010
7. RHR SYS FCD	E12-1030
8. REACTOR RECIRC SYS P&ID	B35-1010
9. RADWASTE SYS P&ID *	G11 THRU G16-1010
10. RADWASTE SYS FCD *	G11 THRU G16-1030
11. REACTOR WATER CLEANUP SYS P&ID	G33-1010
12. REACTOR WATER CLEANUP SYS. FCD	G33-1020
13. SCHEMATIC CONTROL DIAGRAM	B22-F022 & F028
14. LOGIC SYMBOLS	A42-1030
15. PIPING & INSTRUMENT SYMBOLS	A42-1010
16. LOW PRESSURE CORE SPRAY P&ID	E21-1010
17. LOW PRESSURE CORE SPRAY FCD	E21-1030
18. ELEC EQUIP. SEPARATION FOR SAFEGUARD SYS	A62-4050
19. LEAK DETECTION SYS P&ID	E31-1010
20. NUCLEAR BOILER SYS P&ID DATA	B22-1040
21. TURB GEN. & STM BYPASS SYS	A62-4120
22. INSTRUMENT DATA SHEET	B22-3050
23. LEAK DETECTION SYS DES SPEC	E31-4010
24. REACTOR PROTECTION SYS. ELEM DIAG (C72A)	C72-1050
25. REMOVED	
26. REMOVED	
27. REMOVED	

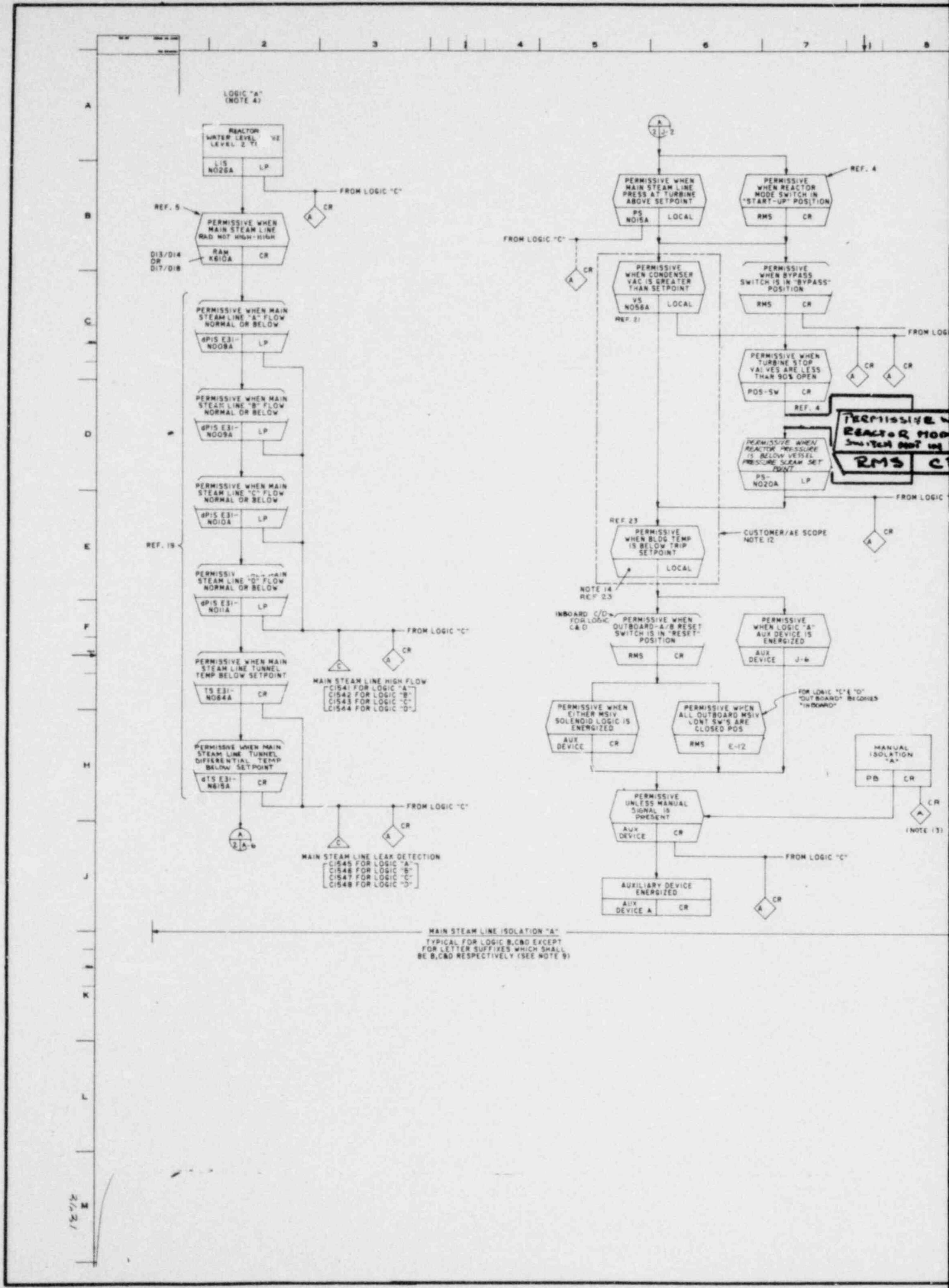
LEGEND:

IEEE	INSTITUTE OF ELECTRICAL & ELECTRONICS ENGINEERS
ADS	AUTO DEPRESSURIZATION SYSTEM
RPS	REACTOR PROTECTION SYSTEM
*	APPLIES IF IN GE SCOPE OF SUPPLY
#	SWITCHGEAR DEVICE FUNCTION NO'S ASA SPEC C37.2
RAM	RADIATION MONITOR

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

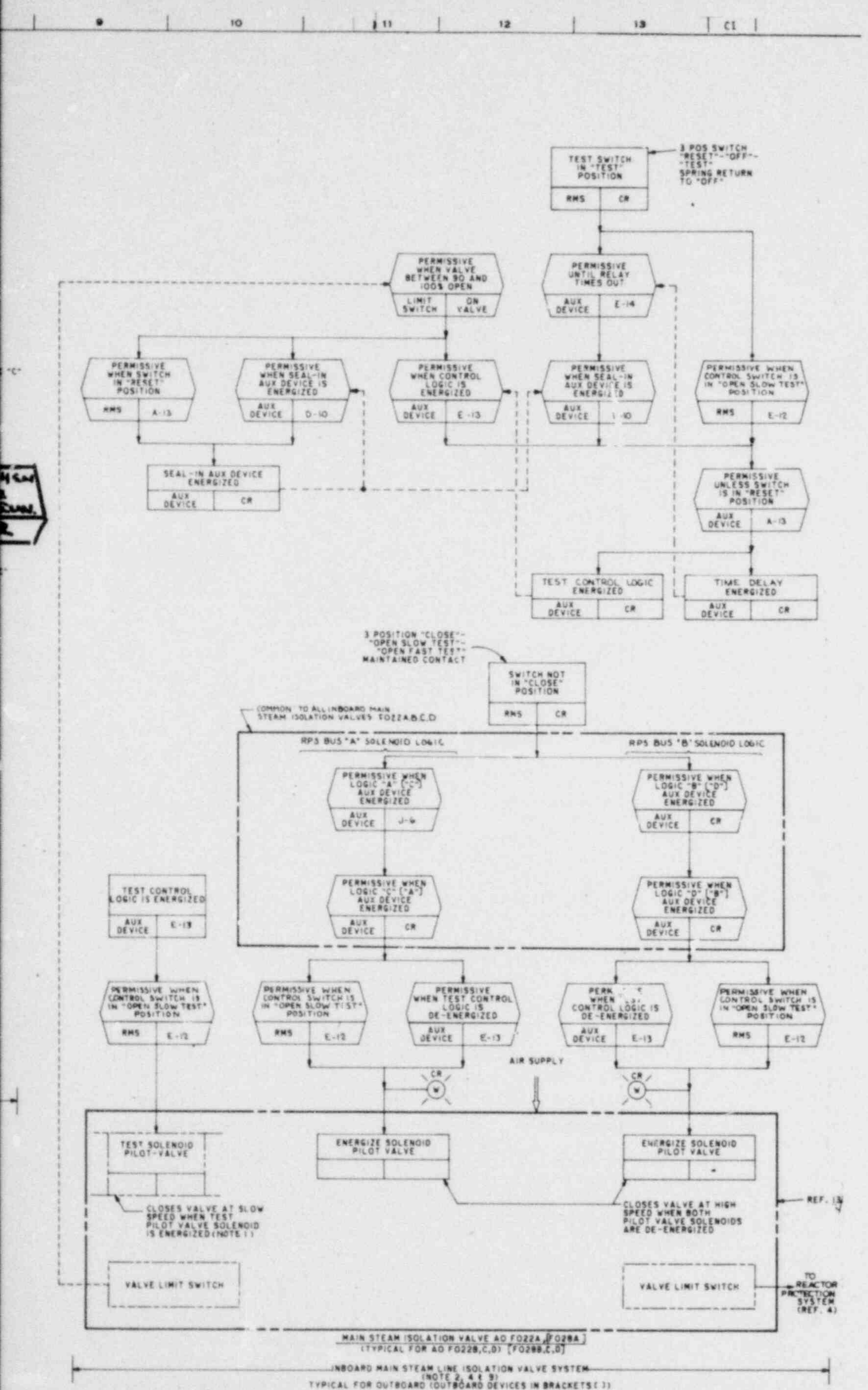
NUCLEAR BOILER SYSTEM FCD
SHEET 1

FIGURE 7.3-10a

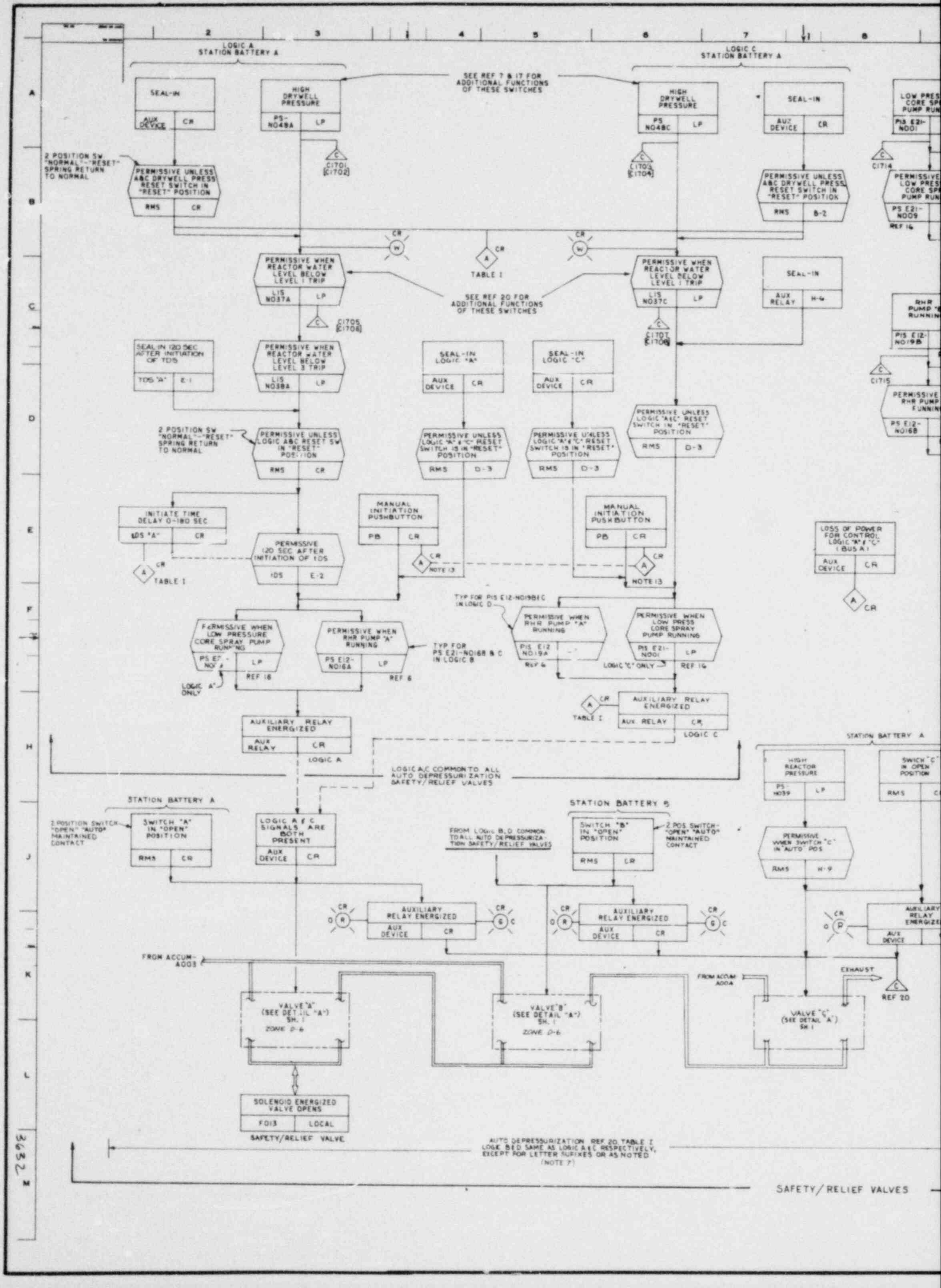


LOGIC "A" (NOTE 4)
REACTOR WATER LEVEL LEVEL 2
LIS N026A
LP
FROM LOGIC "C"
CR
REF. 5
PERMISSIVE WHEN MAIN STEAM LINE RAD NOT HIGH-HIGH
RAM K610A
D13/D14 OR D17/D18
PERMISSIVE WHEN MAIN STEAM LINE "A" FLOW NORMAL OR BELOW
dPIS E31-N008A
PERMISSIVE WHEN MAIN STEAM LINE "B" FLOW NORMAL OR BELOW
dPIS E31-N009A
PERMISSIVE WHEN MAIN STEAM LINE "C" FLOW NORMAL OR BELOW
dPIS E31-N010A
REF. 19
STEAM LINE "D" FLOW NORMAL OR BELOW
dPIS E31-N011A
PERMISSIVE WHEN MAIN STEAM LINE TUNNEL TEMP BELOW SETPOINT
TS E31-N064A
PERMISSIVE WHEN MAIN STEAM LINE TUNNEL DIFFERENTIAL TEMP BELOW SETPOINT
dTS E31-N615A
MAIN STEAM LINE HIGH FLOW
C1541 FOR LOGIC "A"
C1542 FOR LOGIC "B"
C1543 FOR LOGIC "C"
C1544 FOR LOGIC "D"
MAIN STEAM LINE LEAK DETECTION
C1545 FOR LOGIC "A"
C1546 FOR LOGIC "B"
C1547 FOR LOGIC "C"
C1548 FOR LOGIC "D"
PERMISSIVE WHEN MAIN STEAM LINE PRESS AT TURBINE ABOVE SETPOINT
PS N015A
LOCAL
REF. 4
PERMISSIVE WHEN REACTOR MODE SWITCH IN "START-UP" POSITION
RMS
PERMISSIVE WHEN CONDENSER VAC IS GREATER THAN SETPOINT
VS N056A
REF. 21
PERMISSIVE WHEN BYPASS SWITCH IS IN "BYPASS" POSITION
PERMISSIVE WHEN TURBINE STOP VALVES ARE LESS THAN 90% OPEN
POS-SW
PERMISSIVE WHEN REACTOR MODE SWITCH NOT IN
PERMISSIVE WHEN REACTOR PRESSURE IS BELOW VESSEL PRESSURE SCRAM SET POINT
PS-N020A
REF. 23
PERMISSIVE WHEN BLDG TEMP IS BELOW TRIP SETPOINT
CUSTOMER/AE SCOPE NOTE 12
NOTE 14 REF 23
INBOARD C/D FOR LOGIC C&D
PERMISSIVE WHEN OUTBOARD-A/B RESET SWITCH IS IN "RESET" POSITION
PERMISSIVE WHEN LOGIC "A" AUX DEVICE IS ENERGIZED
AUX DEVICE
J-6
FOR LOGIC "C"&"D" "OUTBOARD" BECOMES "INBOARD"
PERMISSIVE WHEN EITHER MSIV SOLENOID LOGIC IS ENERGIZED
PERMISSIVE WHEN ALL OUTBOARD MSIV LONT SW'S ARE CLOSED POS
E-12
MANUAL ISOLATION "A"
PB
(NOTE 13)
PERMISSIVE UNLESS MANUAL SIGNAL IS PRESENT
AUXILIARY DEVICE ENERGIZED
AUX DEVICE A
MAIN STEAM LINE ISOLATION "A"
TYPICAL FOR LOGIC B,C&D EXCEPT FOR LETTER SUFFIXES WHICH SHALL BE B,C&D RESPECTIVELY (SEE NOTE 9)

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM FCD SHEET 2	FIGURE 7.3-10b



LOGIC A
STATION BATTERY A
SEAL-IN
AUX DEVICE | CR
HIGH DRYWELL PRESSURE
PS-N048A | LP
SEE REF 7 & 17 FOR ADDITIONAL FUNCTIONS OF THESE SWITCHES
LOGIC C
STATION BATTERY A
HIGH DRYWELL PRESSURE
PS N048C | LP
SEAL-IN
AUX DEVICE | CR
LOW PRESS CORE SPRAY PUMP RUNNING
PIS E21-N001
2 POSITION SW "NORMAL"-"RESET" SPRING RETURN TO NORMAL
PERMISSIVE UNLESS A&C DRYWELL PRESS RESET SWITCH IN "RESET" POSITION
RMS | CR
C1701 [C1702]
C1703 [C1704]
PERMISSIVE UNLESS A&C DRYWELL PRESS RESET SWITCH IN "RESET" POSITION
RMS | B-2
C1714
PS E21-N009
REF 14
TABLE I
PERMISSIVE WHEN REACTOR WATER LEVEL BELOW LEVEL 1 TRIP
LIS N037A | LP
SEE REF 20 FOR ADDITIONAL FUNCTIONS OF THESE SWITCHES
PERMISSIVE WHEN REACTOR WATER LEVEL BELOW LEVEL 1 TRIP
LIS N037C | LP
SEAL-IN
AUX RELAY | H-4
C1705 [C1706]
C1707 [C1708]
RHR PUMP "B" RUNNING
PIS E12-N019B
SEAL-IN 120 SEC AFTER INITIATION OF TDS
TDS "A" | E-1
PERMISSIVE WHEN REACTOR WATER LEVEL BELOW LEVEL 3 TRIP
LIS N038A | LP
SEAL-IN LOGIC "A"
AUX DEVICE | CR
SEAL-IN LOGIC "C"
AUX DEVICE | CR
C1715
PERMISSIVE WHEN RHR PUMP "B" RUNNING
PS E12-N016B
PERMISSIVE UNLESS LOGIC "A&C" RESET SWITCH IN "RESET" POSITION
RMS | D-3
2 POSITION SW "NORMAL"-"RESET" SPRING RETURN TO NORMAL
PERMISSIVE UNLESS LOGIC A&C RESET SW IN "RESET" POSITION
RMS | CR
PERMISSIVE UNLESS LOGIC "A" & "C" RESET SWITCH IS IN "RESET" POSITION
RMS | D-3
PERMISSIVE UNLESS LOGIC "A" & "C" RESET SWITCH IS IN "RESET" POSITION
RMS | D-3
MANUAL INITIATION PUSHBUTTON
PB | CR
MANUAL INITIATION PUSHBUTTON
PB | CR
INITIATE TIME DELAY 0-180 SEC
TDS "A" | CR
LOSS OF POWER FOR CONTROL LOGIC "A" & "C" (BUS A)
AUX DEVICE | CR
PERMISSIVE 120 SEC AFTER INITIATION OF TDS
TDS | E-2
NOTE 13
NOTE 13
TYP FOR PIS E12-N019B & C IN LOGIC D
PERMISSIVE WHEN RHR PUMP "A" RUNNING
PIS E12 N019A
REF 4
PERMISSIVE WHEN LOW PRESS CORE SPRAY PUMP RUNNING
PIS E21-N001 | LP
LOGIC "C" ONLY
REF 14
PERMISSIVE WHEN LOW PRESSURE CORE SPRAY PUMP RUNNING
PS E2 | LP
LOGIC "A" ONLY
REF 18
PERMISSIVE WHEN RHR PUMP "A" RUNNING
PS E12-N016A | LP
REF 6
TYP FOR PS E21-N016B & C IN LOGIC B
AUXILIARY RELAY ENERGIZED
AUX RELAY | CR
LOGIC C
AUXILIARY RELAY ENERGIZED
AUX RELAY | CR
LOGIC A
LOGIC A/C COMMON TO ALL AUTO DEPRESSURIZATION SAFETY/RELIEF VALVES
STATION BATTERY A
HIGH REACTOR PRESSURE
PS-N039 | LP
SWITCH "C" IN OPEN POSITION
RMS
STATION BATTERY A
2 POSITION SWITCH "OPEN" "AUTO" MAINTAINED CONTACT
SWITCH "A" IN "OPEN" POSITION
RMS | CR
LOGIC A & C SIGNALS ARE BOTH PRESENT
AUX DEVICE | CR
STATION BATTERY B
FROM LOGIC B,D COMMON TO ALL AUTO DEPRESSURIZATION SAFETY/RELIEF VALVES
SWITCH "B" IN "OPEN" POSITION
RMS | CR
2 POS SWITCH "OPEN" "AUTO" MAINTAINED CONTACT
PERMISSIVE WHEN SWITCH "C" IN "AUTO" POS
RMS | H-9
AUXILIARY RELAY ENERGIZED
AUX DEVICE | CR
AUXILIARY RELAY ENERGIZED
AUX DEVICE | CR
AUXILIARY RELAY ENERGIZED
AUX DEVICE
FROM ACCUM-A003
FROM ACCUM A004
EXHAUST
REF 20
VALVE "A" (SEE DETAIL "A") SH. 1 ZONE D-6
VALVE "B" (SEE DETAIL "A") SH. 1 ZONE D-6
VALVE "C" (SEE DETAIL "A") SH 1
SOLENOID ENERGIZED VALVE OPENS
F013 | LOCAL
SAFETY/RELIEF VALVE
AUTO DEPRESSURIZATION REF 20, TABLE I LOGIC B&D SAME AS LOGIC A&C RESPECTIVELY, EXCEPT FOR LETTER SUFFIXES OR AS NOTED (NOTE 7)
SAFETY/RELIEF VALVES
3632

AMENDMENT NO. 10
July 1980

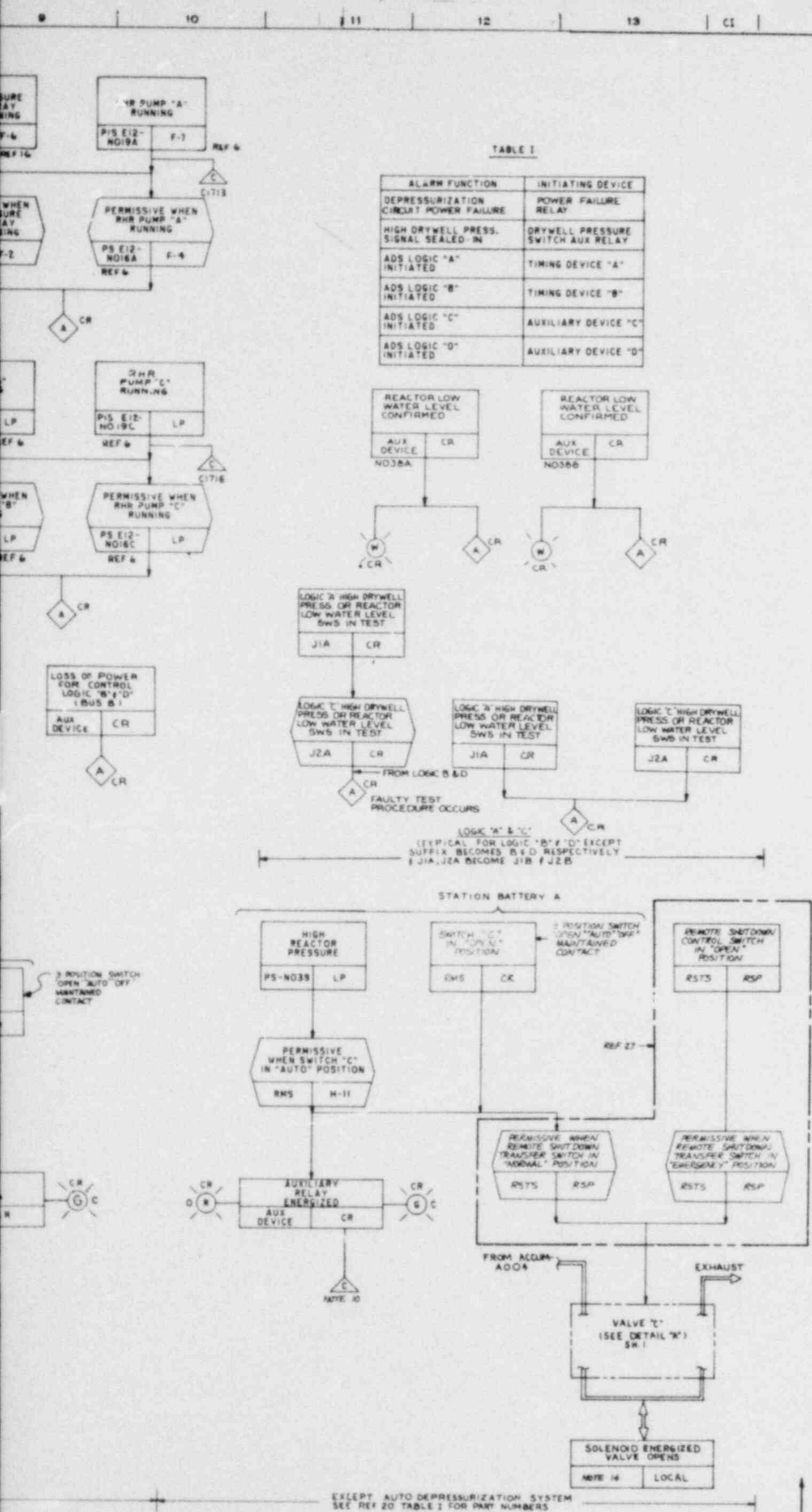


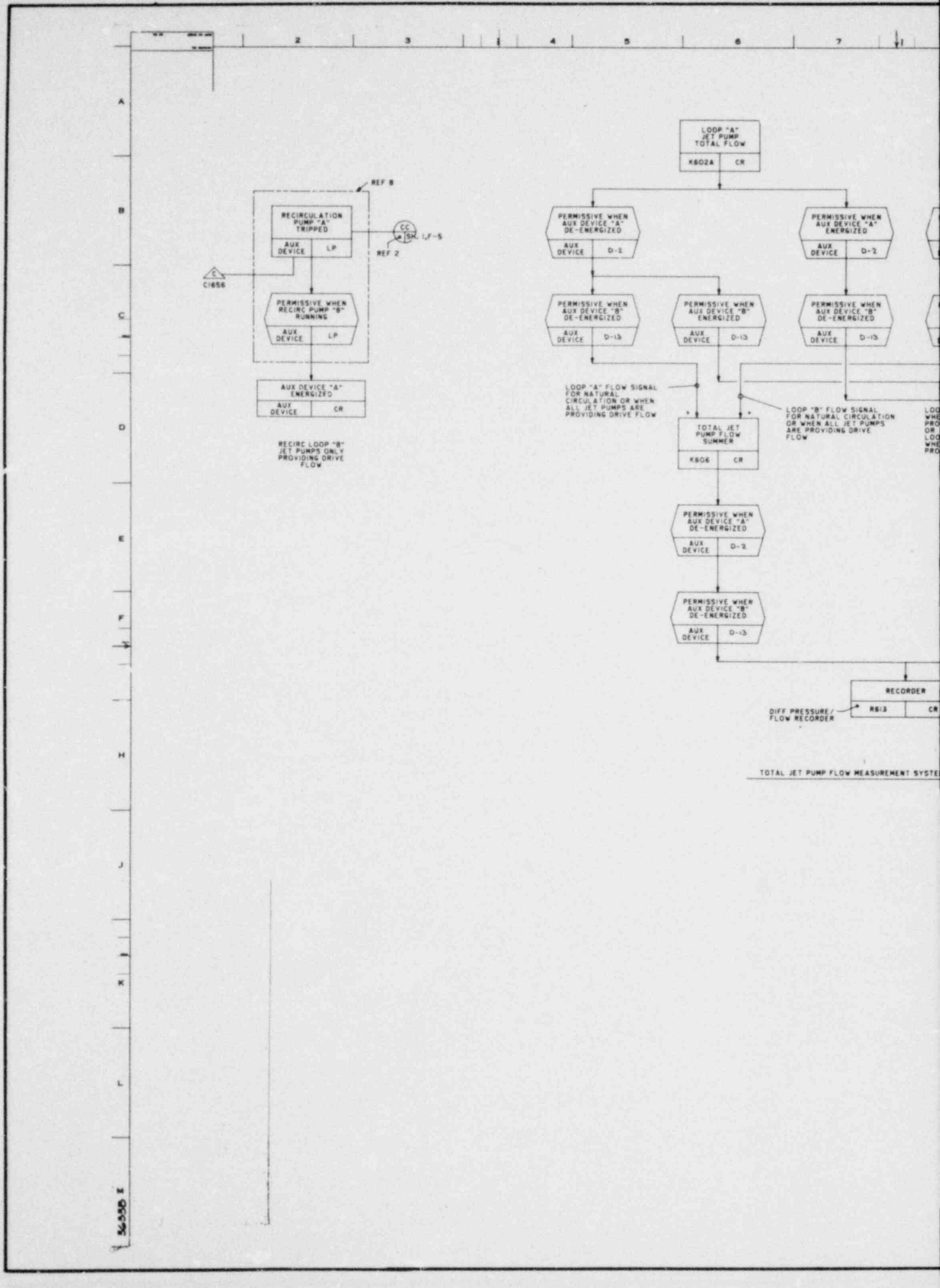
TABLE I

ALARM FUNCTION	INITIATING DEVICE
DEPRESSURIZATION CIRCUIT POWER FAILURE	POWER FAILURE RELAY
HIGH DRYWELL PRESS. SIGNAL SEALED-IN	DRYWELL PRESSURE SWITCH AUX RELAY
ADS LOGIC "A" INITIATED	TIMING DEVICE "A"
ADS LOGIC "B" INITIATED	TIMING DEVICE "B"
ADS LOGIC "C" INITIATED	AUXILIARY DEVICE "C"
ADS LOGIC "D" INITIATED	AUXILIARY DEVICE "D"

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

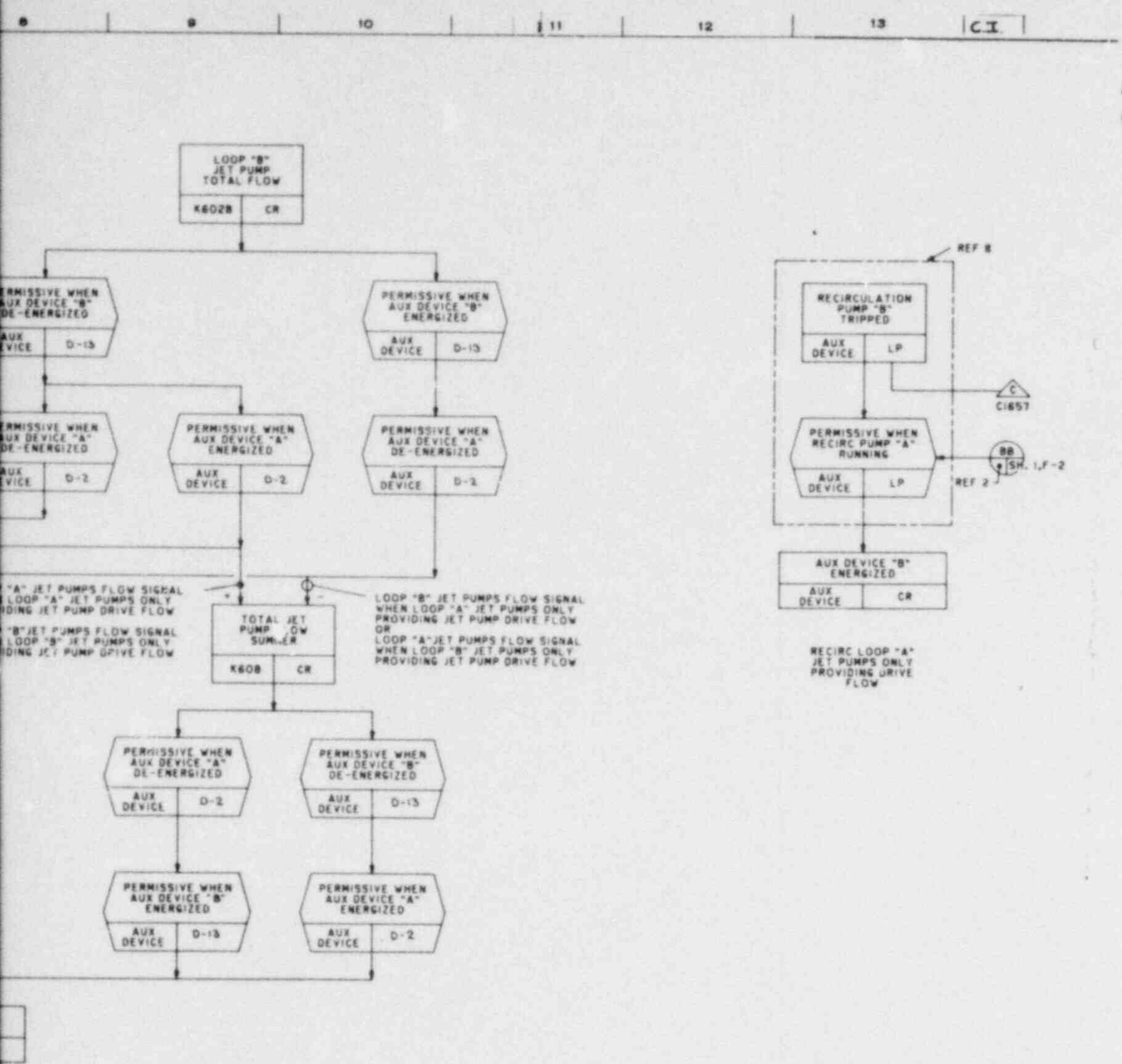
NUCLEAR BOILER SYSTEM FCD
SHEET 3

FIGURE 7.3-10c

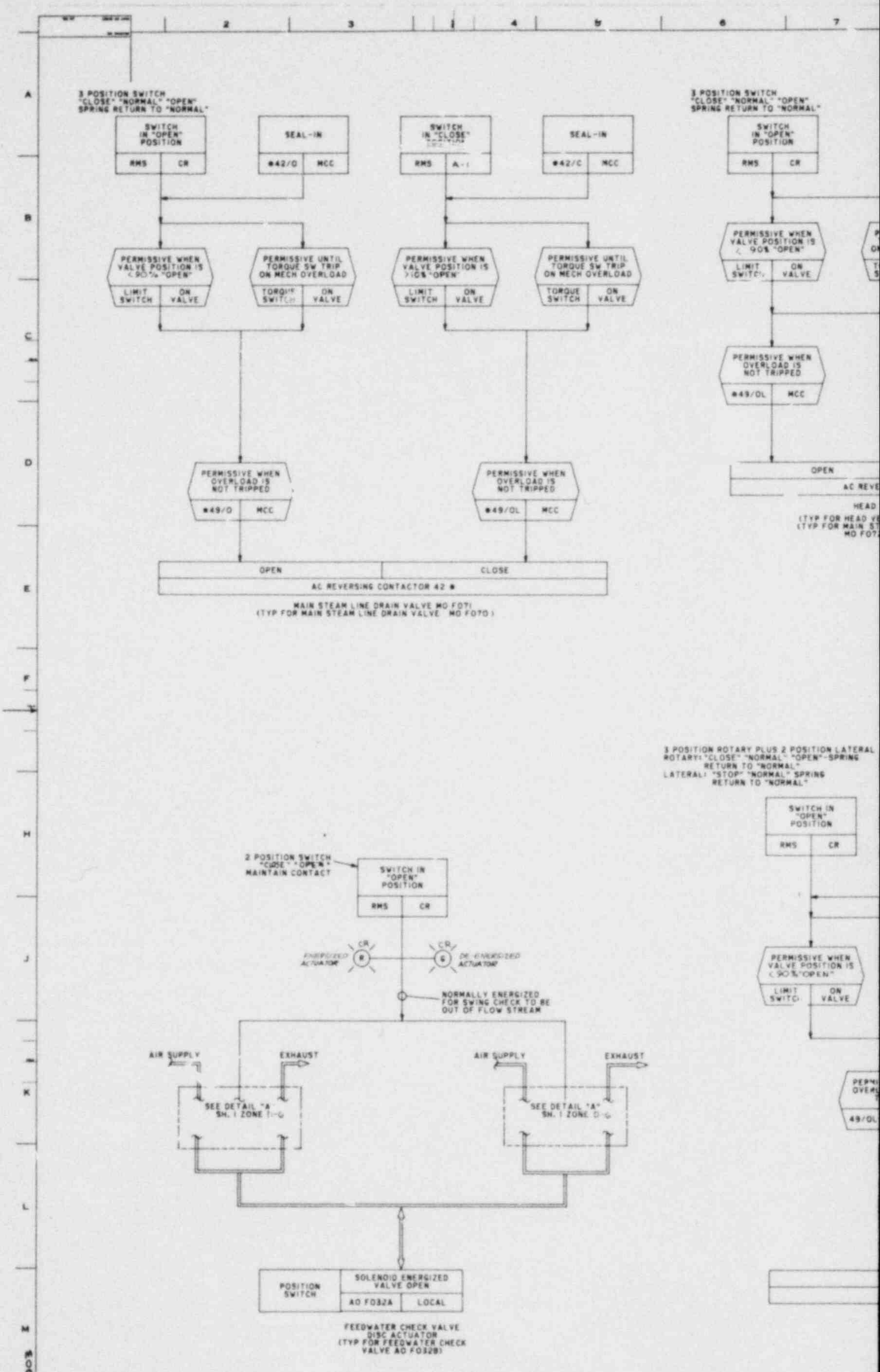


LOOP "A" JET PUMP TOTAL FLOW
K602A
CR
REF B
RECIRCULATION PUMP "A" TRIPPED
AUX DEVICE
LP
CC
SH. 1,F-5
REF 2
C
C1656
PERMISSIVE WHEN RECIRC PUMP "B" RUNNING
AUX DEVICE
LP
AUX DEVICE "A" ENERGIZED
AUX DEVICE
CR
RECIRC LOOP "B" JET PUMPS ONLY PROVIDING DRIVE FLOW
PERMISSIVE WHEN AUX DEVICE "A" DE-ENERGIZED
AUX DEVICE
D-2
PERMISSIVE WHEN AUX DEVICE "A" ENERGIZED
AUX DEVICE
D-2
PERMISSIVE WHEN AUX DEVICE "B" DE-ENERGIZED
AUX DEVICE
D-13
PERMISSIVE WHEN AUX DEVICE "B" ENERGIZED
AUX DEVICE
D-13
PERMISSIVE WHEN AUX DEVICE "B" DE-ENERGIZED
AUX DEVICE
D-13
LOOP "A" FLOW SIGNAL FOR NATURAL CIRCULATION OR WHEN ALL JET PUMPS ARE PROVIDING DRIVE FLOW
LOOP "B" FLOW SIGNAL FOR NATURAL CIRCULATION OR WHEN ALL JET PUMPS ARE PROVIDING DRIVE FLOW
TOTAL JET PUMP FLOW SUMMER
K606
CR
PERMISSIVE WHEN AUX DEVICE "A" DE-ENERGIZED
AUX DEVICE
D-2
PERMISSIVE WHEN AUX DEVICE "B" DE-ENERGIZED
AUX DEVICE
D-13
RECORDER
K613
CR
DIFF PRESSURE/ FLOW RECORDER
TOTAL JET PUMP FLOW MEASUREMENT SYSTE
A
B
C
D
E
F
H
J
K
L
2
3
4
5
6
7

AMENDMENT NO. 10
July 1980

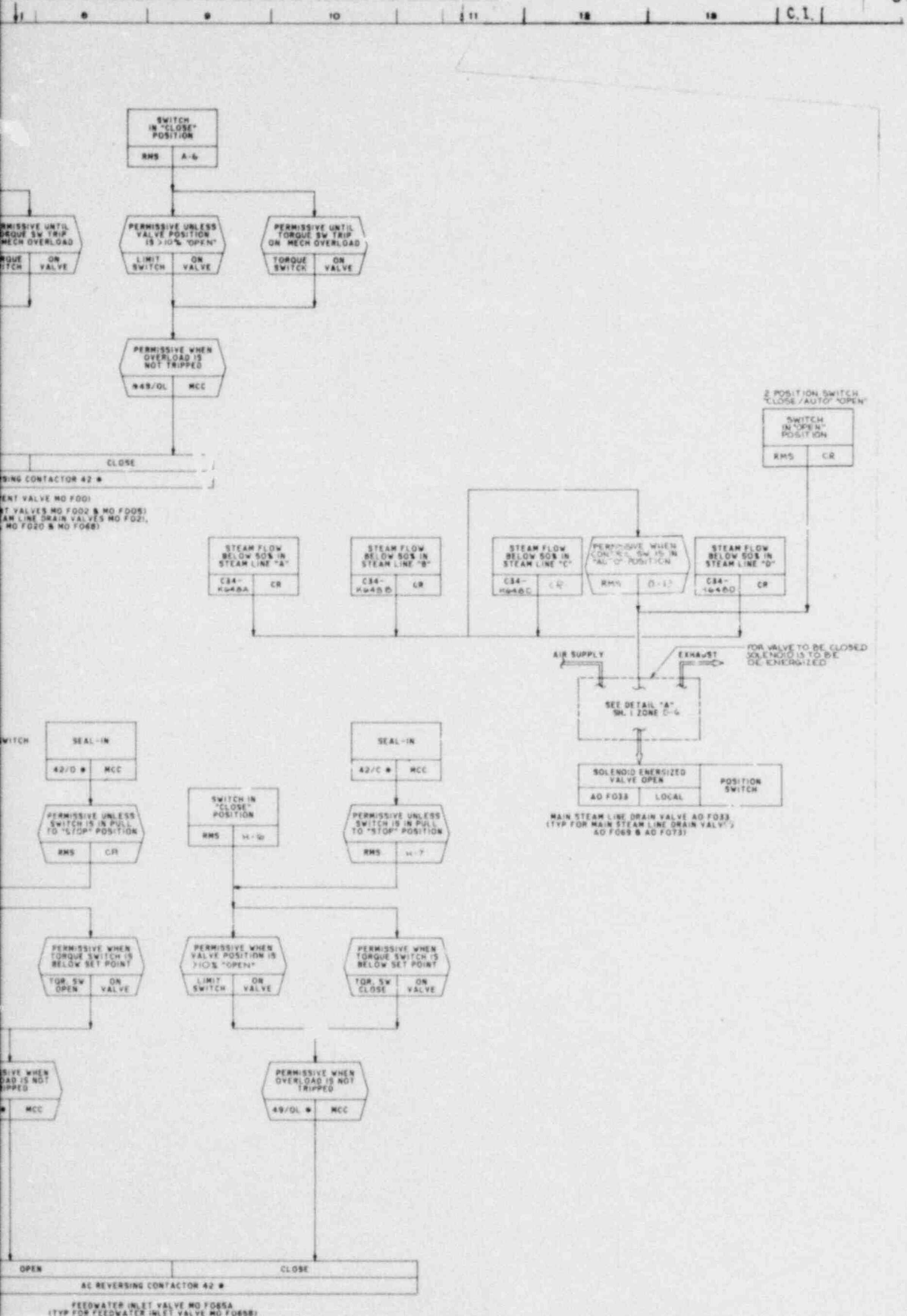


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM FCD SHEET 4	FIGURE 7.3-10d

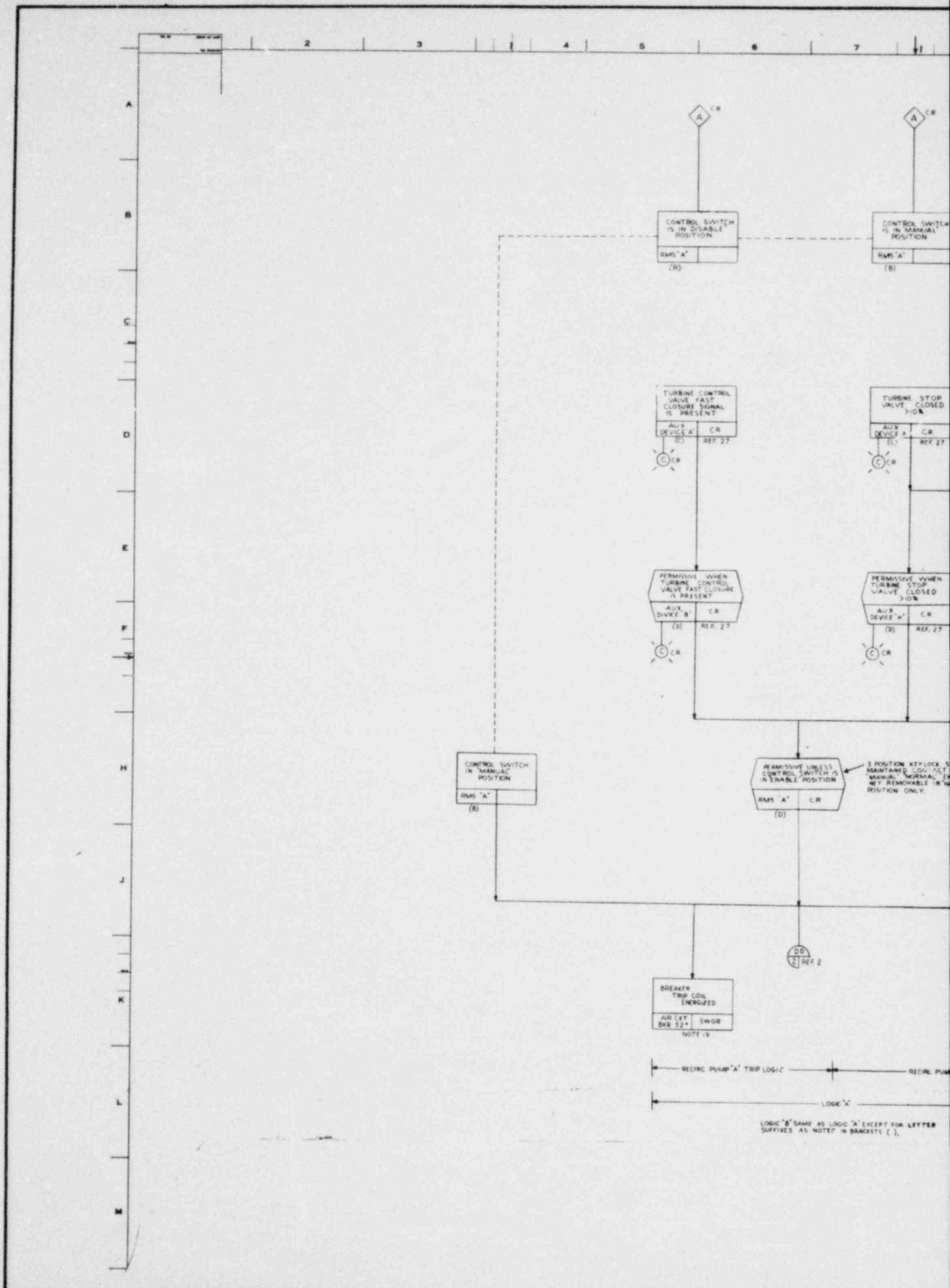


2
3
4
5
6
7
A
B
C
D
E
F
H
J
K
L
M
3 POSITION SWITCH "CLOSE" "NORMAL" "OPEN" SPRING RETURN TO "NORMAL"
SWITCH IN "OPEN" POSITION
RMS
CR
SEAL-IN
#42/O
MCC
SWITCH IN "CLOSE"
RMS
A-1
SEAL-IN
#42/C
MCC
PERMISSIVE WHEN VALVE POSITION IS <90% "OPEN"
LIMIT SWITCH
ON VALVE
PERMISSIVE UNTIL TORQUE SW TRIP ON MECH OVERLOAD
TORQUE SWITCH
ON VALVE
PERMISSIVE WHEN VALVE POSITION IS >10% "OPEN"
LIMIT SWITCH
ON VALVE
PERMISSIVE UNTIL TORQUE SW TRIP ON MECH OVERLOAD
TORQUE SWITCH
ON VALVE
PERMISSIVE WHEN OVERLOAD IS NOT TRIPPED
#49/O
MCC
PERMISSIVE WHEN OVERLOAD IS NOT TRIPPED
#49/OL
MCC
OPEN
CLOSE
AC REVERSING CONTACTOR 42 #
MAIN STEAM LINE DRAIN VALVE MO F071
(TYP FOR MAIN STEAM LINE DRAIN VALVE MO F070)
3 POSITION SWITCH "CLOSE" "NORMAL" "OPEN" SPRING RETURN TO "NORMAL"
SWITCH IN "OPEN" POSITION
RMS
CR
PERMISSIVE WHEN VALVE POSITION IS < 90% "OPEN"
LIMIT SWITCH
ON VALVE
PERMISSIVE WHEN OVERLOAD IS NOT TRIPPED
#49/OL
MCC
OPEN
AC REVE
HEAD
(TYP FOR HEAD VE
(TYP FOR MAIN ST
MO F07
3 POSITION ROTARY PLUS 2 POSITION LATERAL
ROTARY: "CLOSE" "NORMAL" "OPEN"-SPRING RETURN TO "NORMAL"
LATERAL: "STOP" "NORMAL" SPRING RETURN TO "NORMAL"
SWITCH IN "OPEN" POSITION
RMS
CR
PERMISSIVE WHEN VALVE POSITION IS <90% "OPEN"
LIMIT SWITCH
ON VALVE
49/OL
2 POSITION SWITCH "CLOSE" "OPEN" MAINTAIN CONTACT
SWITCH IN "OPEN" POSITION
RMS
CR
CR
R
ENERGIZED ACTUATOR
CR
G
DE-ENERGIZED ACTUATOR
NORMALLY ENERGIZED FOR SWING CHECK TO BE OUT OF FLOW STREAM
AIR SUPPLY
EXHAUST
SEE DETAIL "A" SH. 1 ZONE D-G
AIR SUPPLY
EXHAUST
SEE DETAIL "A" SH. 1 ZONE D-G
POSITION SWITCH
SOLENOID ENERGIZED VALVE OPEN
AO F032A
LOCAL
FEEDWATER CHECK VALVE DISC ACTUATOR
(TYP FOR FEEDWATER CHECK VALVE AO F032B)
4003

AMENDMENT NO. 10
July 1980

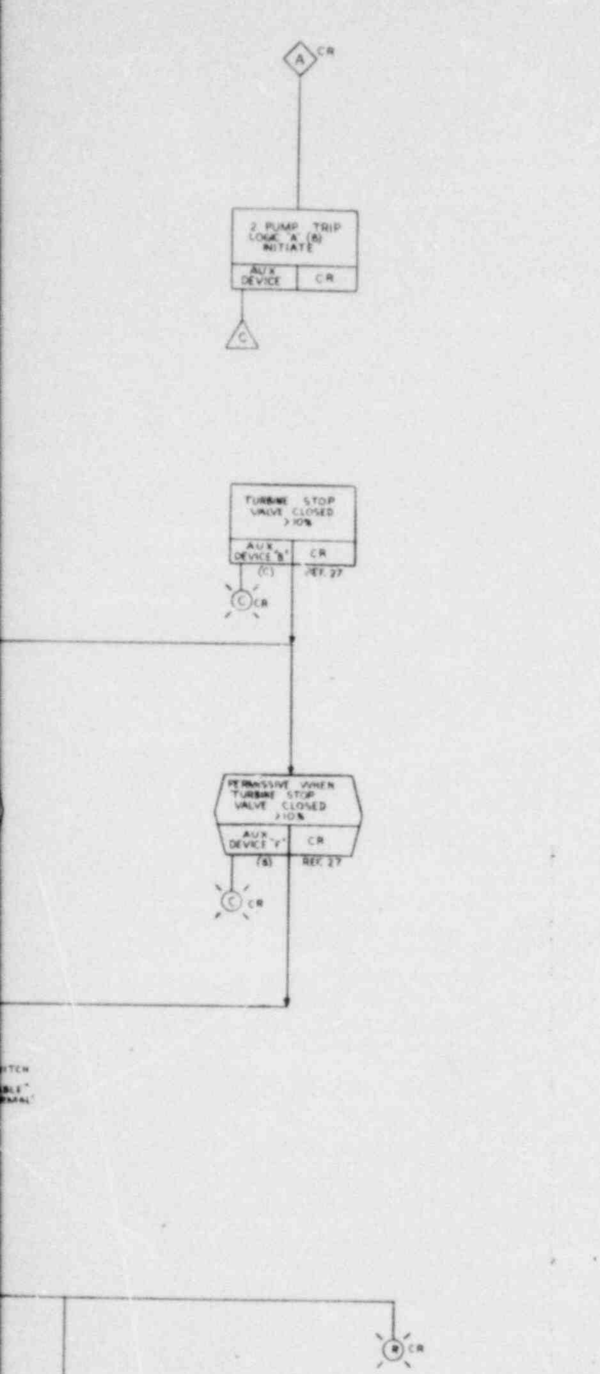


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM FCD SHEET 5	FIGURE 7.3-10e

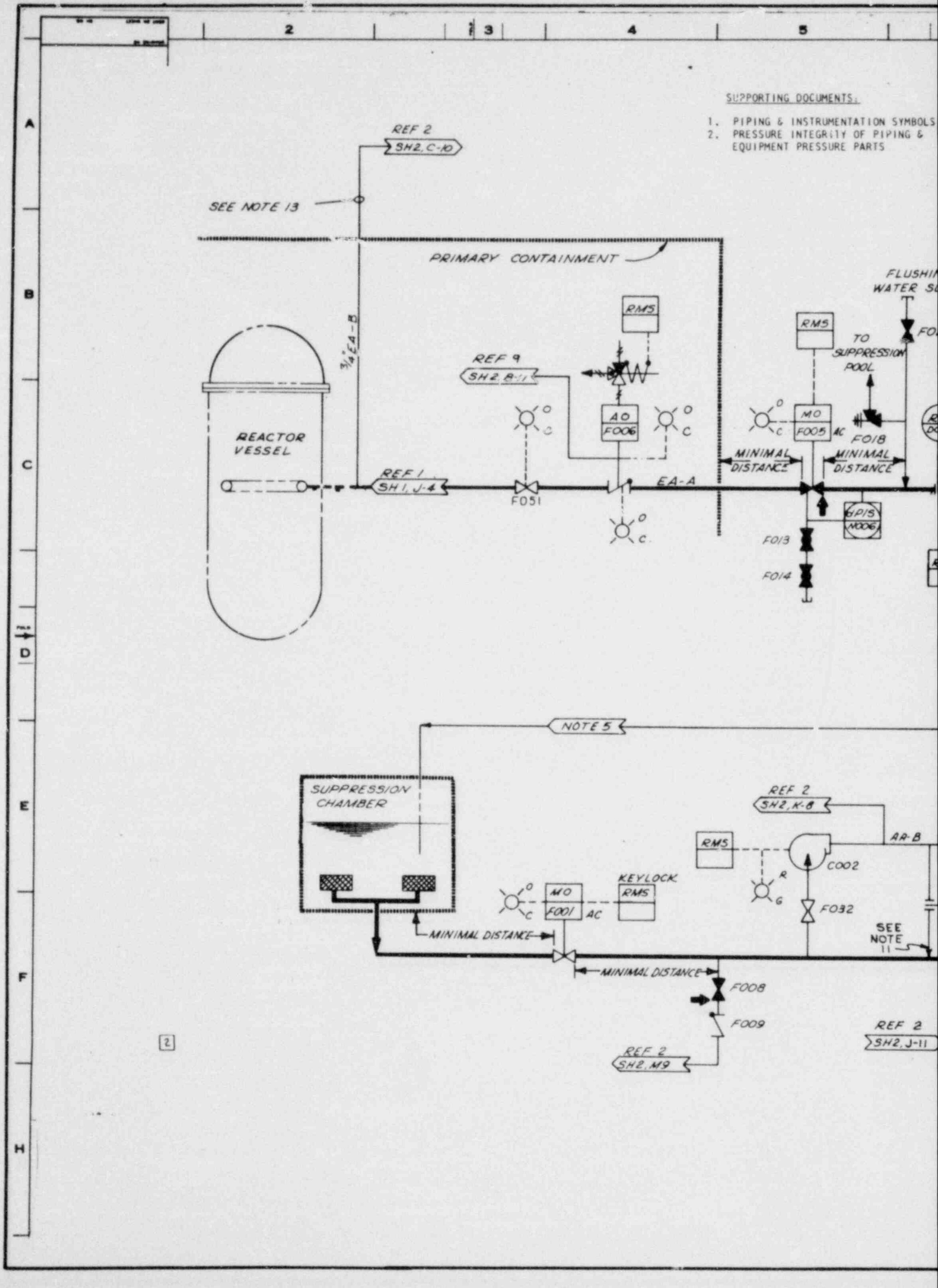


2
3
4
5
6
7
A
B
C
D
E
F
H
J
K
L
M
CR
CONTROL SWITCH IS IN DISABLE POSITION
RMS "A"
(R)
CONTROL SWITCH IS IN MANUAL POSITION
RMS "A"
(B)
TURBINE CONTROL VALVE FAST CLOSURE SIGNAL IS PRESENT
AUX DEVICE "A"
CR
(C)
REF 27
TURBINE STOP VALVE CLOSED >10%
AUX DEVICE A
CR
REF 27
PERMISSIVE WHEN TURBINE CONTROL VALVE FAST CLOSURE IS PRESENT
AUX DIVICE "B"
CR
(D)
REF 27
PERMISSIVE WHEN TURBINE STOP VALVE CLOSED >10%
AUX DEVICE "H"
CR
(D)
REF. 27
CONTROL SWITCH IN "MANUAL" POSITION
RMS "A"
(B)
PERMISSIVE UNLESS CONTROL SWITCH IS IN "ENABLE" POSITION
RMS "A"
CR
(D)
POSITION ONLY.
DD
2
REF 2
BREAKER TRIP COIL ENERGIZED
AIR CKT BKR 52*
SWGR
NOTE 19
RECIRC PUMP "A" TRIP LOGIC
LOGIC "A"
LOGIC "B" SAME AS LOGIC "A" EXCEPT FOR LETTER SUFFIXES AS NOTED IN BRACKETS ( ).

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM FCD	FIGURE 7.3-10f


SUPPORTING DOCUMENTS:
1. PIPING & INSTRUMENTATION SYMBOLS
2. PRESSURE INTEGRITY OF PIPING & EQUIPMENT PRESSURE PARTS
REF 2
SH2, C-10
SEE NOTE 13
PRIMARY CONTAINMENT
3/4" EA-B
REACTOR VESSEL
REF 9
SH2, B-11
RMS
AO
F006
REF 1
SH1, J-4
F051
EA-A
MO
F005
AC
F018
TO SUPPRESSION POOL
FLUSHING WATER S
MINIMAL DISTANCE
GPIS
N006
F013
F014
NOTE 5
SUPPRESSION CHAMBER
REF 2
SH2, K-8
AR-B
C002
F032
KEYLOCK
RMS
F001
MINIMAL DISTANCE
SEE NOTE 11
F008
F009
REF 2
SH2, M9
REF 2
SH2, J-11

6 | 7 | 8 | 9 | 10 IRT

AMENDMENT NO. 10
July 1980

FCF 239X287AD (E21-1010)

MPL ITEM NO.
A42-1010
A62-4030

REFERENCE DOCUMENTS:

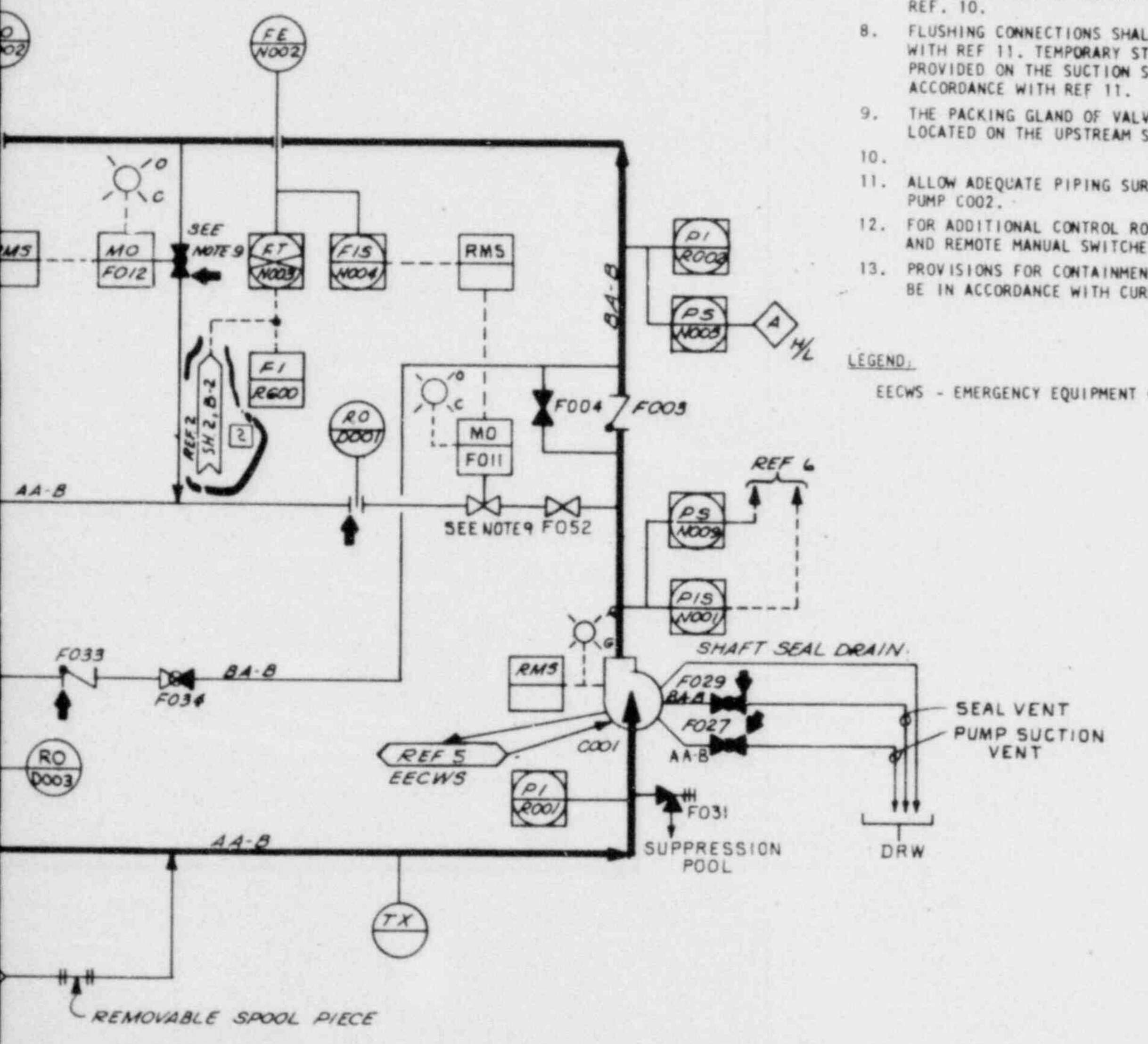
	REFERENCE DOCUMENTS	MPL ITEM NO.
1.	NUCLEAR BOILER SYSTEM P&ID	B22-1010
2.	RHR SYSTEM P&ID	E12-1010
3.	LPCS SYSTEM FCD	E21-1030
4.	PROCESS INSTRUMENTATION	A62-4070
5.	EMERG. EQUIP. COOLING WATER	A62-4230
6.	NUCLEAR BOILER SYSTEM FCD	B22-1030
7.	LPCS SYSTEM PROC DIAG	E21-1020
8.	LPCS SYSTEM DESIGN SPEC	E21-4010
9.	LEAK DETECTION SYS. IED	E31-1010
10.	LPCS SYSTEM INST. DATA SH	E21-3050
11.	CLEANING OF PIPING & EQUIP	A62-4140

NOTES:

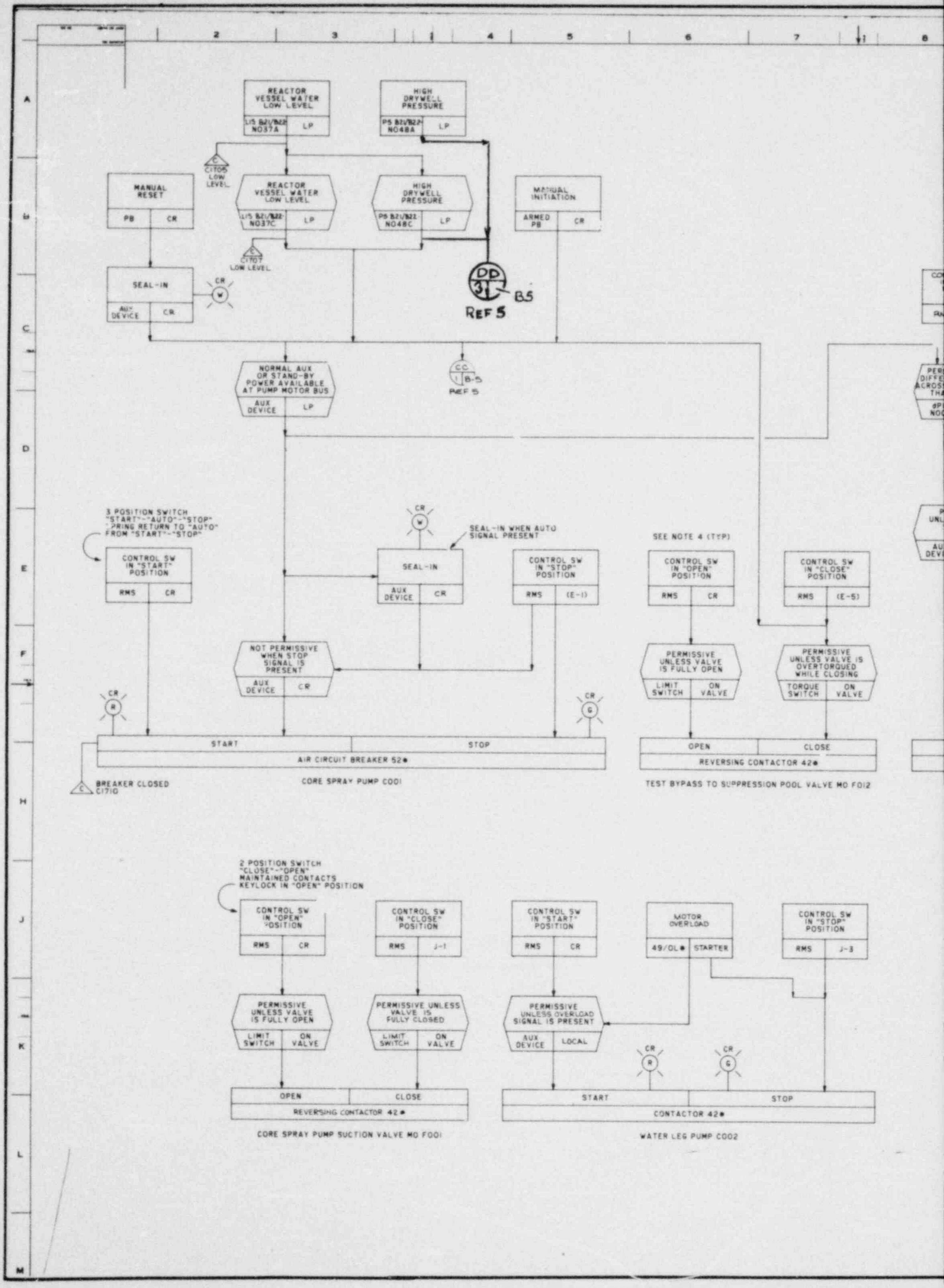
1. PIPING HIGH POINT VENTS & LOW POINT DRAINS ARE TO BE ADDED AS NECESSARY.
2. CHEMICAL CLEANING CONNECTIONS, VALVES, ETC., IF REQUIRED, ARE PROVIDED AS NECESSARY.
3. INSTRUMENT LINE DESIGN AND VALVING MUST COMPLY WITH INSTRUMENT PIPING SPECIFICATION, SEE REF. 4.
4. THE METHOD OF MOUNTING LOCAL INSTRUMENTS IS TO BE DETERMINED BY OTHERS.
5. THIS LINE MAY BE RETURNED TO SUPPRESSION CHAMBER THROUGH A SEPARATE LINE OR ATTACHED TO THE RHR SYSTEM TEST RETURN LINES. IF RETURNED THROUGH THE RHR LINES, A CHECK VALVE MUST BE PROVIDED DOWNSTREAM OF RO D001 TO PREVENT BACK FLOW. SEE REF. 2, SH 2, ZONE (F-7).
6. VENT, DRAIN, & RELIEF VALVE DISCHARGE SYSTEMS TO CRW AND DRW OR SUPPRESSION POOL ARE BY PIPING DESIGNER.
7. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE REF. 10.
8. FLUSHING CONNECTIONS SHALL BE PROVIDED IN ACCORDANCE WITH REF 11. TEMPORARY STRAINER SCREENS SHALL BE PROVIDED ON THE SUCTION SIDE OF ALL PUMPS IN ACCORDANCE WITH REF 11.
9. THE PACKING GLAND OF VALVE F011, & F012 SHALL BE LOCATED ON THE UPSTREAM SIDE OF VALVE DISK.
10.
11. ALLOW ADEQUATE PIPING SURFACE AREA FOR COOLING OF PUMP C002.
12. FOR ADDITIONAL CONTROL ROOM LIGHTS, SYSTEM ALARMS, AND REMOTE MANUAL SWITCHES SEE REF. 3.
13. PROVISIONS FOR CONTAINMENT ISOLATION BY OTHERS TO BE IN ACCORDANCE WITH CURRENT LICENSING REQUIREMENTS.

LEGEND:

EECWS - EMERGENCY EQUIPMENT COOLING WATER SYSTEM



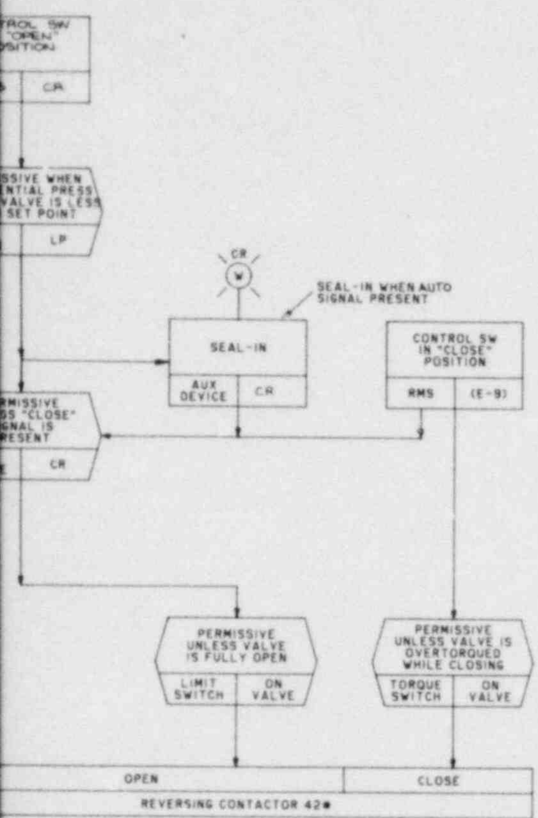
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LPCS P&ID	FIGURE 7.3-11



REACTOR VESSEL WATER LOW LEVEL
L/S B21/B22-N037A LP
HIGH DRYWELL PRESSURE
PS B21/B22-N048A LP
C C1705 LOW LEVEL
MANUAL RESET
PB CR
REACTOR VESSEL WATER LOW LEVEL
L/S B21/B22-N037C LP
HIGH DRYWELL PRESSURE
PS B21/B22-N048C LP
MANUAL INITIATION
ARMED PB CR
C C1707 LOW LEVEL
SEAL-IN
AUX DEVICE CR
CR W
DD 31 B5
REF 5
C.C. 1 B-5
REF 5
NORMAL AUX OR STAND-BY POWER AVAILABLE AT PUMP MOTOR BUS
AUX DEVICE LP
3 POSITION SWITCH "START"-"AUTO"-"STOP" SPRING RETURN TO "AUTO" FROM "START"-"STOP"
CONTROL SW IN "START" POSITION
RMS CR
SEAL-IN WHEN AUTO SIGNAL PRESENT
SEAL-IN
AUX DEVICE CR
CONTROL SW IN "STOP" POSITION
RMS (E-1)
SEE NOTE 4 (TYP)
CONTROL SW IN "OPEN" POSITION
RMS CR
CONTROL SW IN "CLOSE" POSITION
RMS (E-5)
NOT PERMISSIVE WHEN STOP SIGNAL IS PRESENT
AUX DEVICE CR
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH ON VALVE
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE CLOSING
TORQUE SWITCH ON VALVE
START STOP
AIR CIRCUIT BREAKER 52●
OPEN CLOSE
REVERSING CONTACTOR 42●
BREAKER CLOSED C C1710
CORE SPRAY PUMP C001
TEST BYPASS TO SUPPRESSION POOL VALVE MO F012
2 POSITION SWITCH "CLOSE"-"OPEN" MAINTAINED CONTACTS KEYLOCK IN "OPEN" POSITION
CONTROL SW IN "OPEN" POSITION
RMS CR
CONTROL SW IN "CLOSE" POSITION
RMS J-1
CONTROL SW IN "START" POSITION
RMS CR
MOTOR OVERLOAD
49/OL● STARTER
CONTROL SW IN "STOP" POSITION
RMS J-3
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH ON VALVE
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
LIMIT SWITCH ON VALVE
PERMISSIVE UNLESS OVERLOAD SIGNAL IS PRESENT
AUX DEVICE LOCAL
OPEN CLOSE
REVERSING CONTACTOR 42●
START STOP
CONTACTOR 42●
CORE SPRAY PUMP SUCTION VALVE MO F001
WATER LEG PUMP C002

9	10	11	12	13	C I

FCF 239X287 G2,G3,G8 & G10 (E21-1030)



CORE SPRAY INJECTION VALVE MO F005

NOTES:

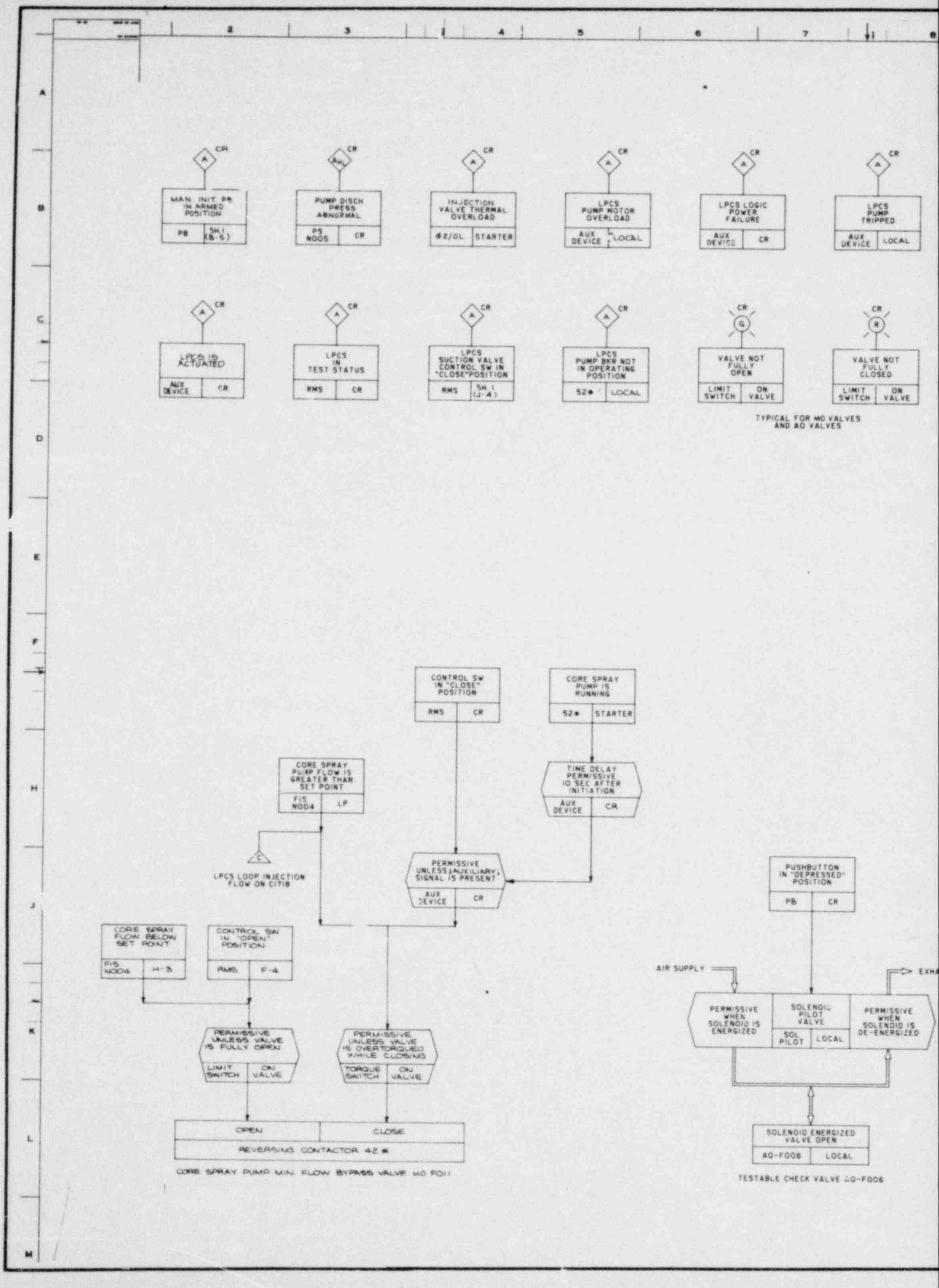
1. PUMP MOTOR SHALL BE PROTECTED WITH OVERLOAD PROTECTION. PROTECTION RELAYS ARE TO BE APPLIED SO AS TO MAINTAIN POWER ON THE MOTOR AS LONG AS POSSIBLE WITHOUT IMMEDIATE DAMAGE TO EMERGENCY POWER SYSTEM.
2. VALVE MOTORS ARE TO BE PROVIDED WITH THERMAL OVERLOAD TRIPS AND ANNUNCIATION. THE PHASE 1 & 3 OVERLOAD DEVICES SHALL BE IN THE MOTOR CIRCUIT AND THE PHASE 2 OVERLOAD DEVICE IN THE ANNUNCIATOR CIRCUIT. IN ADDITION, VALVE MOTOR CIRCUITS ARE TO BE PROVIDED WITH SHORT CIRCUIT CURRENT PROTECTIVE TRIPS.
3. FOR ADDITIONAL ALARMS AND PROCESS INSTRUMENTATION NOT SHOWN, SEE LPCS SYSTEM P&ID.
4. UNLESS OTHERWISE NOTED, ALL RMS SHALL BE 3 POSITION SWITCHES, "CLOSE" "AUTO" "OPEN", SPRING RETURN TO "AUTO" FROM "CLOSE" OR "OPEN".
5. CONTROL AND MOTIVE POWER FOR LPCS SHALL BE FROM SAME SOURCE AS THE RHR LOOP "B" EQUIPMENT (REF. 5).
6. THE LPCS SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH IEEE 279-1971 AND REF. DOC. 8.

REFERENCE DRAWINGS:	MPL ITEM NO.
1. HIGH PRESSURE CORE SPRAY SYSTEM P&ID	E22-1010
2. HIGH PRESSURE CORE SPRAY SYSTEM ELECTRICAL ONE-LINE DIAGRAM	E22-1060
3. NUCLEAR BOILER SYSTEM P&ID	B21/B22-1010
4. RESIDUAL HEAT REMOVAL SYSTEM P&ID	E12-1010
5. RESIDUAL HEAT REMOVAL SYSTEM FCD	E12-1030
6. NUCLEAR BOILER SYS FCD	B21/B22-1030
7. LOGIC SYMBOLS	A42-1030
8. ELECTRICAL EQUIPMENT SEPARATION FOR SAFEGUARD SYSTEMS	A62-4050

LEGEND:

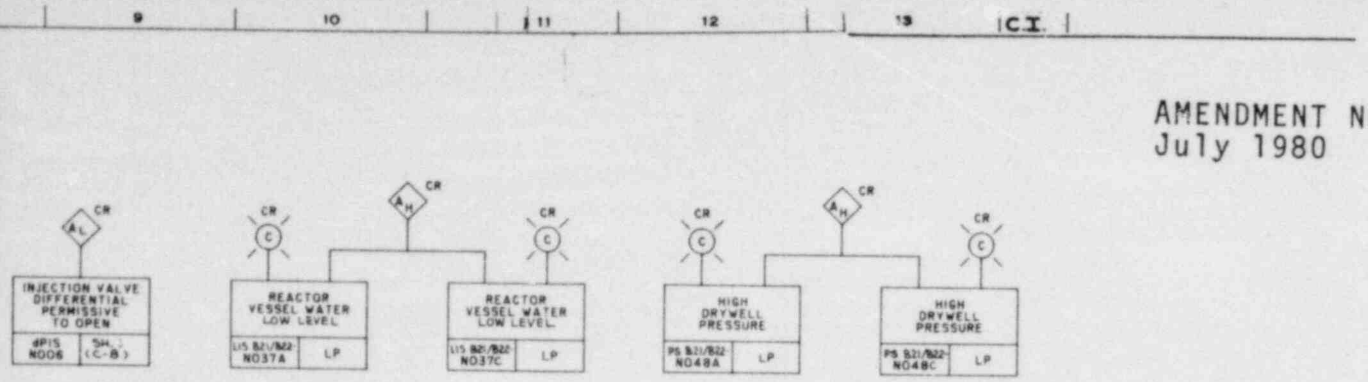
* - SWITCHGEAR DEVICE FUNCTION NO. ANSI SPEC. C37.2
IEEE - INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LPCS FCD SHEET 1	FIGURE 7.3-12a

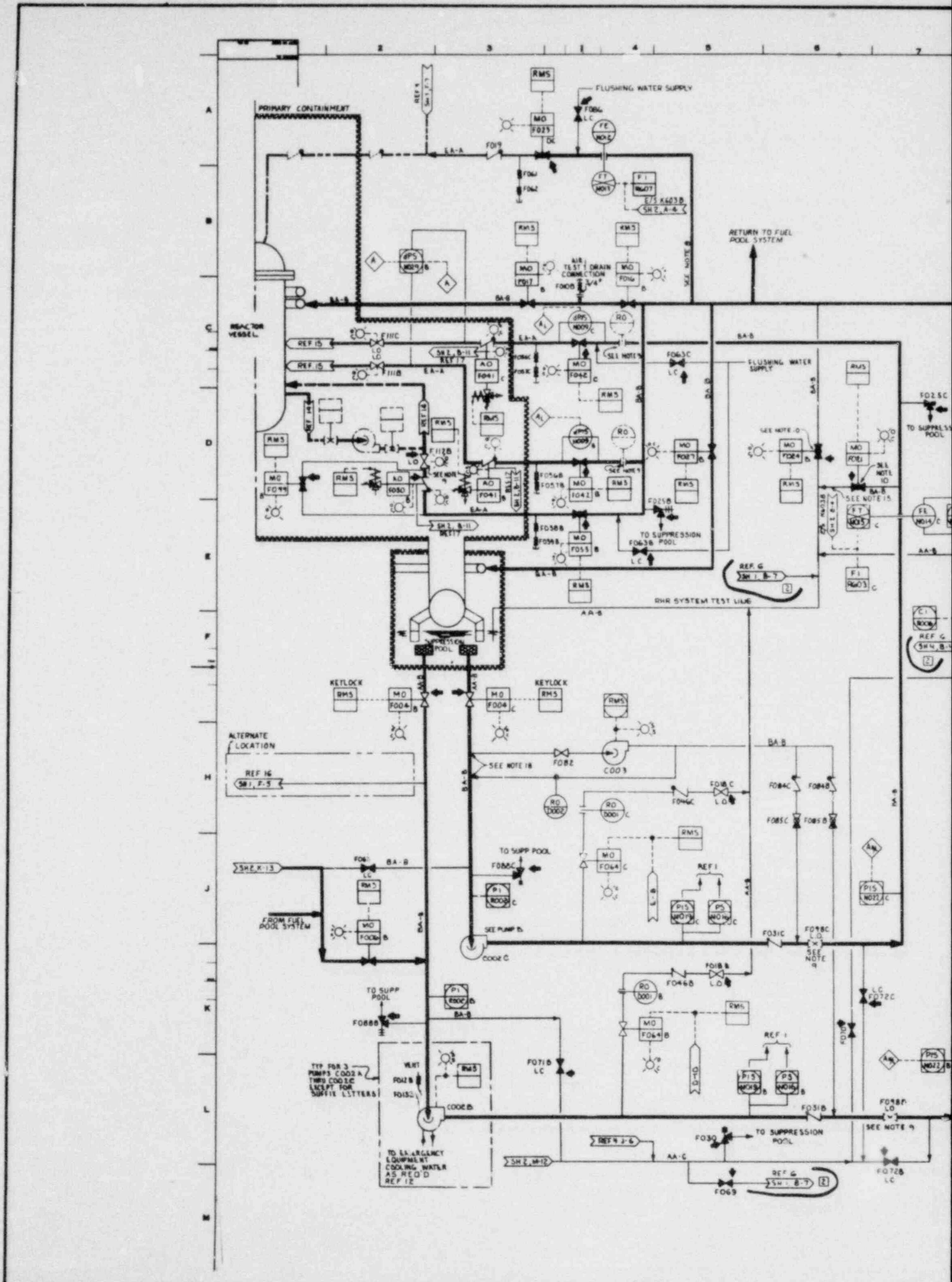


MAN. INIT. PB IN ARMED POSITION
PB
SH.1 (B-5)
PUMP DISCH PRESS ABNORMAL
PS N005
CR
INJECTION VALVE THERMAL OVERLOAD
#2/OL
STARTER
LPCS PUMP MOTOR OVERLOAD
AUX DEVICE
LOCAL
LPCS LOGIC POWER FAILURE
AUX DEVICE
CR
LPCS PUMP TRIPPED
AUX DEVICE
LOCAL
LPCS IS ACTUATED
AUX DEVICE
CR
LPCS IN TEST STATUS
RMS
CR
LPCS SUCTION VALVE CONTROL SW IN "CLOSE" POSITION
RMS
SH.1 (J-4)
LPCS PUMP BKR NOT IN OPERATING POSITION
52
LOCAL
VALVE NOT FULLY OPEN
LIMIT SWITCH
ON VALVE
VALVE NOT FULLY CLOSED
LIMIT SWITCH
ON VALVE
TYPICAL FOR MO VALVES AND AO VALVES
CONTROL SW IN "CLOSE" POSITION
RMS
CR
CORE SPRAY PUMP IS RUNNING
52
STARTER
CORE SPRAY PUMP FLOW IS GREATER THAN SET POINT
FIS N004
LP
TIME DELAY PERMISSIVE 10 SEC AFTER INITIATION
AUX DEVICE
CR
LPCS LOOP INJECTION FLOW ON C1718
PERMISSIVE UNLESS (AUXILIARY) SIGNAL IS PRESENT
AUX DEVICE
CR
PUSHBUTTON IN "DEPRESSED" POSITION
PB
CR
CORE SPRAY FLOW BELOW SET POINT
FIS N004
H-3
CONTROL SW IN "OPEN" POSITION
RMS
F-4
AIR SUPPLY
EXHA
PERMISSIVE WHEN SOLENOID IS ENERGIZED
SOLENOID PILOT VALVE
SOL PILOT
LOCAL
PERMISSIVE WHEN SOLENOID IS DE-ENERGIZED
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE CLOSING
TORQUE SWITCH
ON VALVE
OPEN
CLOSE
REVERSING CONTACTOR 42
CORE SPRAY PUMP MIN. FLOW BYPASS VALVE MO F011
SOLENOID ENERGIZED VALVE OPEN
AO-F006
LOCAL
TESTABLE CHECK VALVE AO-F006

AMENDMENT NO. 10
July 1980

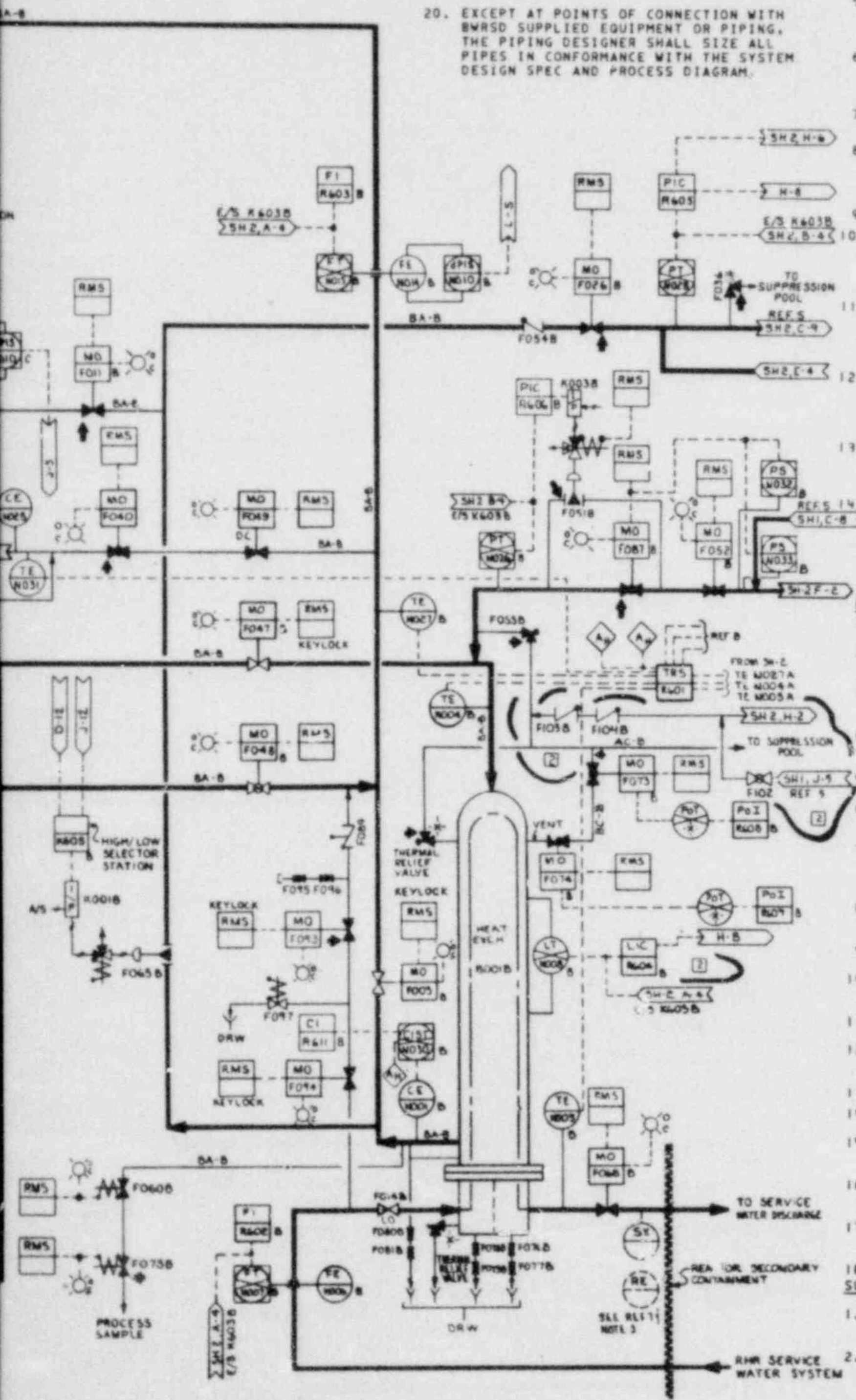


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LPCS FCD SHEET 2	FIGURE 7.3-12b


PRIMARY CONTAINMENT
FLUSHING WATER SUPPLY
RETURN TO FUEL POOL SYSTEM
REACTOR VESSEL
SUPPRESSION POOL
RHR SYSTEM TEST LINE
KEYLOCK
ALTERNATE LOCATION
SEE NOTE 18
FROM FUEL POOL SYSTEM
TO SUPP POOL
TO SUPPRESSION POOL
TYP FOR 3 PUMPS C002A THRU C002C EXCEPT FOR SUFFIX LETTERS
TO EMERGENCY EQUIPMENT COOLING WATER AS REQ'D REF 12
RMS
MO
AO
F004
F006
F023
F041
F042
F055
F064
F069
C002C
C002B
C003
REF 15
REF 16
REF 1
REF 6

NOTES:

15. A SEPARATE SUPPRESSION POOL PENETRATION (IF USED) FOR 'C' LOOP SHALL TERMINATE UNDER WATER.
16. LINE FROM RELIEF VALVES TO SLOPE DOWNWARD CONTINUOUSLY TO SUPPRESSION POOL.
17.
18. ALLOW ADEQUATE PIPING SURFACE AREA FOR COOLING OF PUMP C003.
19. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET REF. 18.
20. EXCEPT AT POINTS OF CONNECTION WITH BWRSD SUPPLIED EQUIPMENT OR PIPING, THE PIPING DESIGNER SHALL SIZE ALL PIPES IN CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.



MPL ITEM Nº E12-1010

NOTES:

1. INSTRUMENT LINE DESIGN & VALVING MUST COMPLY WITH INSTRUMENT PIPING SPECIFICATION REFERENCE 9.
2. THE METHOD OF MOUNTING LOCAL INSTRUMENTS IS TO BE DETERMINED BY OTHERS.
3. PIPING HIGH POINT VENTS & LOW POINT DRAINS ARE TO BE ADDED AS NECESSARY.
4. FLUSHING CONNECTIONS SHALL BE PROVIDED IN ACCORDANCE WITH REF 2. TEMPORARY STRAINER SCREENS SHALL BE PROVIDED ON THE SUCTION SIDE OF ALL PUMPS IN ACCORDANCE WITH REF 2.
5. DISCHARGE LINES FOR COOLING WATER TO BE ROUTED UPSTREAM OF SERVICE WATER RADIATION MONITORS.
6. EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NO. (E12) UNLESS OTHERWISE NOTED.
7. ALL MOTOR OPERATED VALVES ARE AC OPERATED UNLESS OTHERWISE NOTED.
8. REACTOR HEAD SPRAY LINE CONNECTION TO MAIN LINE SHALL BE AS CLOSE AS PRACTICIBLE TO VALVE F016.
9. RECOMMENDED, BUT NOT REQUIRED.
10. VALVE SHOULD BE INSTALLED WITH PACKING GLAND(S), UPSTREAM SIDE OF DISK.
11. VALVES F005, F025ABC, F030 & F088 SHALL BE 1" RELIEF VALVES.
12. BETWEEN VALVES MO F008 & MO F009 CONSIDERATION SHOULD BE GIVEN TO THERMAL EXPANSION OF THE CONTAINED WATER.
13. FOR ADDITIONAL CONTROL ROOM LIGHTS, SYSTEM ALARMS & REMOTE MANUAL SWITCHES SEE REF 3.
14. PROVISIONS FOR CONTAINMENT ISOLATION BY OTHERS TO BE IN ACCORDANCE WITH CURRENT LICENSING REQUIREMENTS.

SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NO.

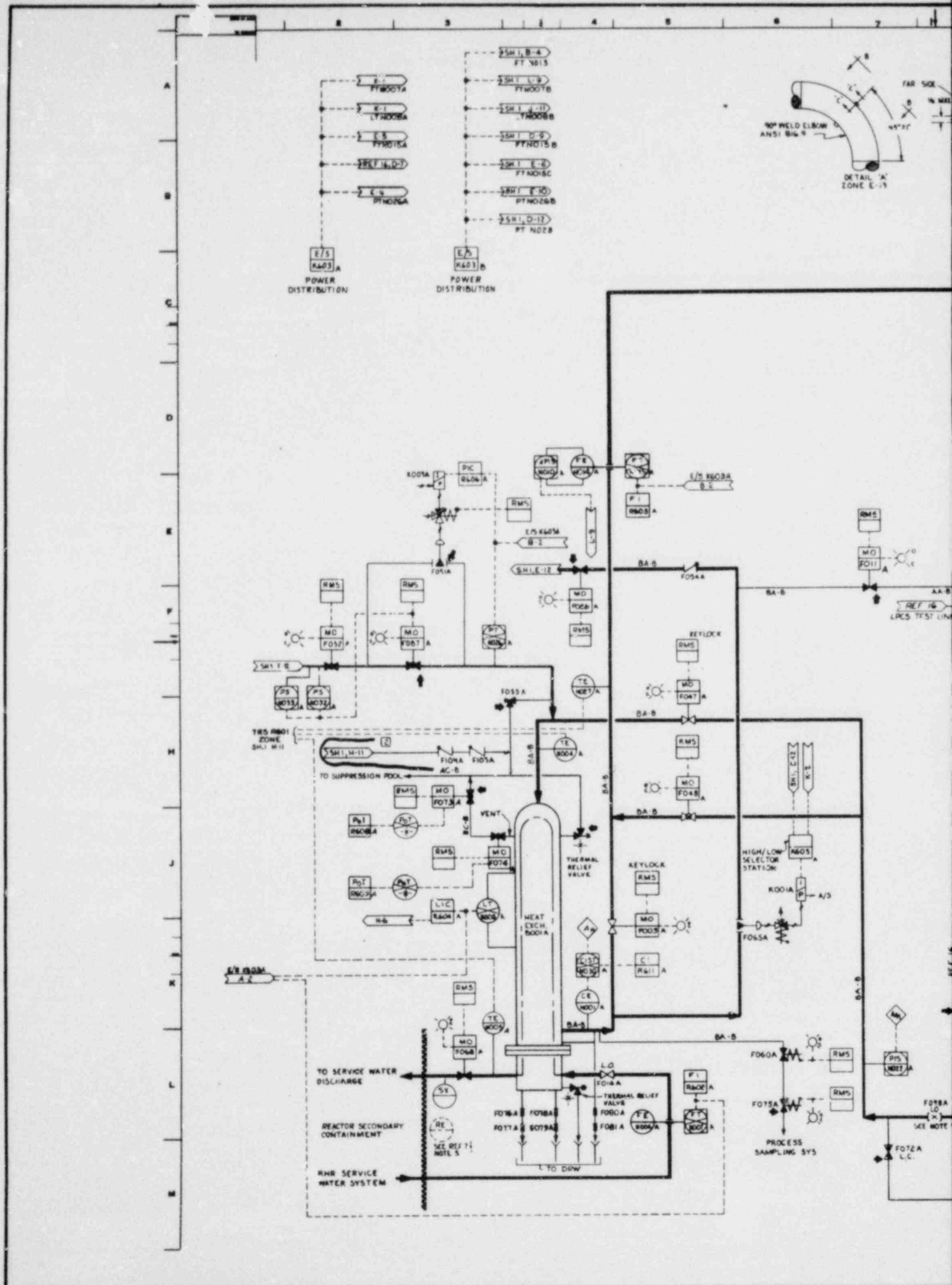
REFERENCE DOCUMENTS	MPL ITEM NO.
1. NUCLEAR BOILER SYS FCD	B22-1030
2. CLEANING OF PIPING & EQUIPMENT	A62-4140
3. RESIDUAL HEAT REMOVAL FCD	E12-1030
4. HIGH PRESSURE CORE SPRAY P&ID	E22-1010
5. RCIC SYSTEM P&ID	E51-1010
6. RADWASTE SYSTEM P&ID	G11-1010
7. PROCESS RADIATION MONITORING SYSTEM	D17-1010
8.	
9. PROCESS INSTR	A62-4070
10. RHR SYSTEM PROCESS DATA	E12-1020
11. RHR SYS DESIGN SPEC	E12-4010
12. EMERGENCY EQUIPMENT COOLING WATER	A62-4230
13. RCIC SYS FCD	E51-1030
14. REAC RECIRC SYS P&ID	B35-1010
15. NUCLEAR BOILER SYS P&ID	B22-1010
16. LOW PRESSURE CORE SPRAY P&ID	E21-1010
17. LEAK DETECTION SYSTEM IED	E31-1010
18. RHR SYS IDS	E12-1050

SUPPORTING DOCUMENTS	MPL ITEM NO.
1. PIPING & INSTRUMENT SYMBOLS	A42-1010
2. PRESSURE INTEGRITY OF PIPING & EQUIPMENT PRESSURE PARTS	A62-4430

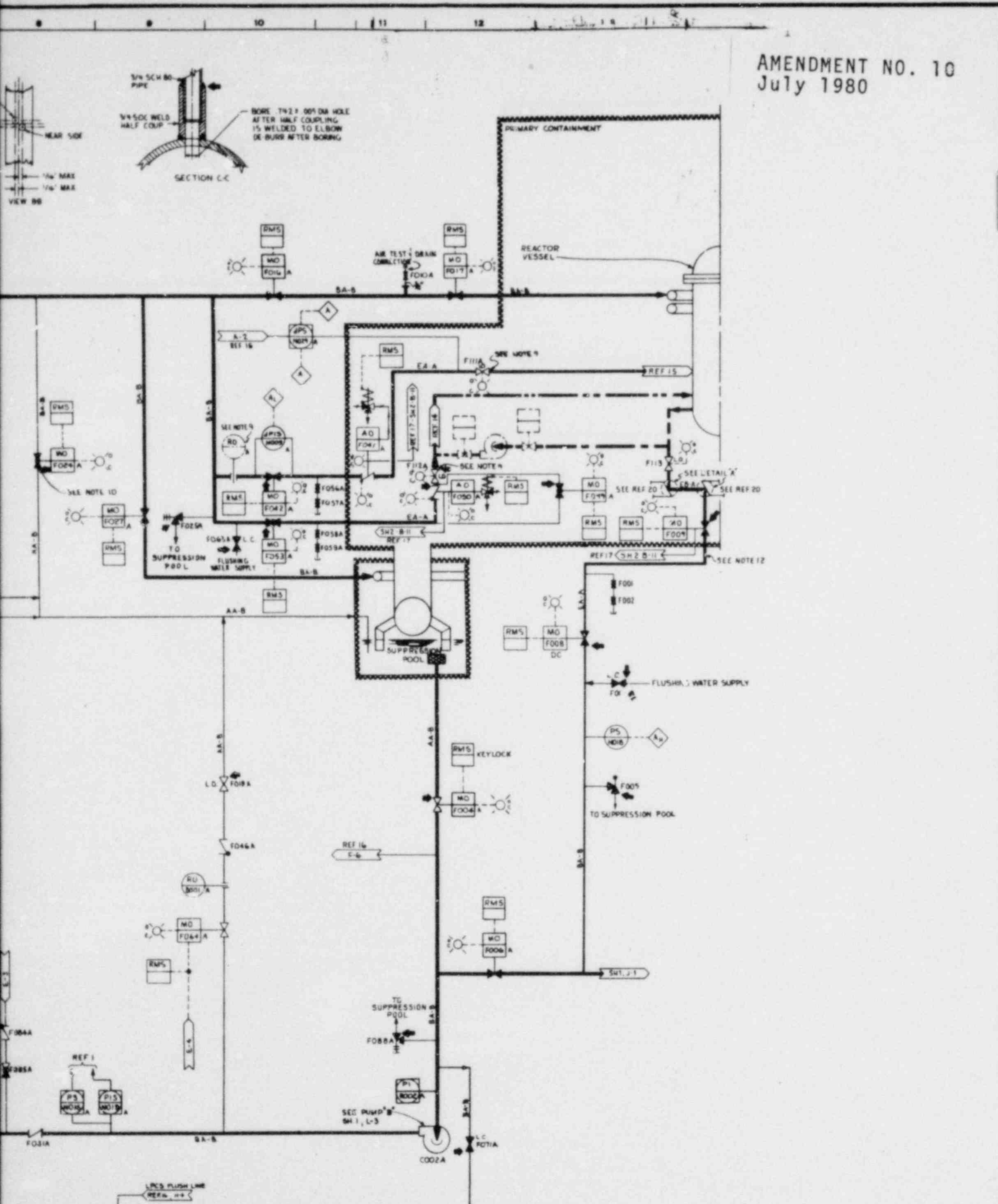
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RHR P&ID SHEET 1

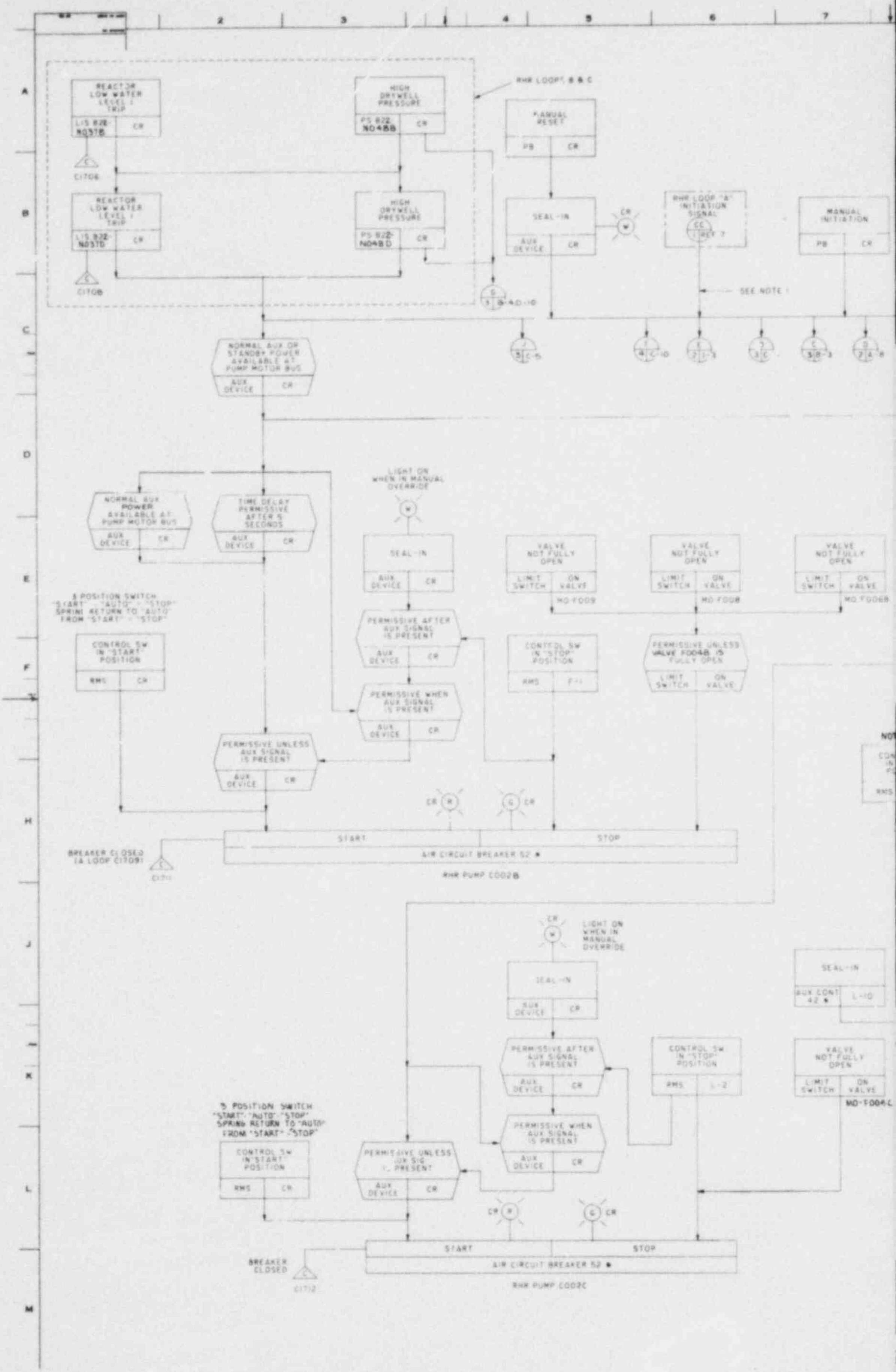
FIGURE 7.3-13a


POWER DISTRIBUTION
POWER DISTRIBUTION
90° WELD ELBOW ANSI B16.9
DETAIL 'A' ZONE E-13
FAR SIDE
KEYLOCK
KEYLOCK
TO SUPPRESSION POOL
VENT
THERMAL RELIEF VALVE
HEAT EXCH. B001A
HIGH/LOW SELECTOR STATION
TO SERVICE WATER DISCHARGE
REACTOR SECONDARY CONTAINMENT
RHR SERVICE WATER SYSTEM
TO DRW
PROCESS SAMPLING SYS
LPCS TEST LINE
SEE NOTE

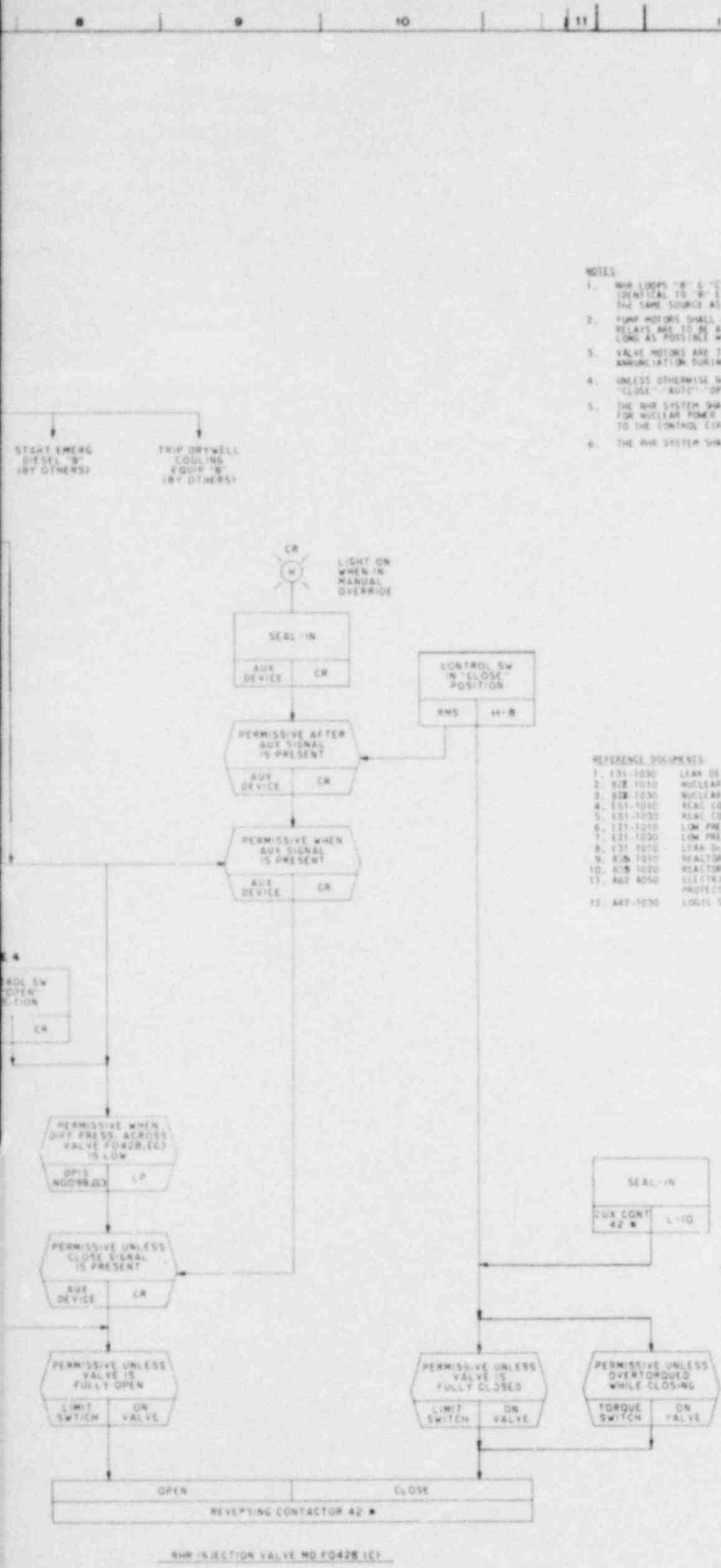
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RHR P&ID SHEET 2	FIGURE 7.3-13b



REACTOR LOW WATER LEVEL 1 TRIP
LIS B22-N037B
CR
C1706
REACTOR LOW WATER LEVEL 1 TRIP
LIS B22-N037D
C1708
HIGH DRYWELL PRESSURE
PS B22-N048B
HIGH DRYWELL PRESSURE
PS B22-N048D
RHR LOOPS B & C
MANUAL RESET
PB
SEAL-IN
AUX DEVICE
RHR LOOP "A" INITIATION SIGNAL
SEE NOTE 1
MANUAL INITIATION
NORMAL AUX OR STANDBY POWER AVAILABLE AT PUMP MOTOR BUS
NORMAL AUX POWER AVAILABLE AT PUMP MOTOR BUS
TIME DELAY PERMISSIVE AFTER 5 SECONDS
LIGHT ON WHEN IN MANUAL OVERRIDE
VALVE NOT FULLY OPEN
LIMIT SWITCH
ON VALVE
MO-F009
MO-F008
MO-F006B
3 POSITION SWITCH "START" - "AUTO" - "STOP" SPRING RETURN TO "AUTO" FROM "START" - "STOP"
CONTROL SW IN "START" POSITION
RMS
PERMISSIVE AFTER AUX SIGNAL IS PRESENT
PERMISSIVE WHEN AUX SIGNAL IS PRESENT
CONTROL SW IN "STOP" POSITION
PERMISSIVE UNLESS VALVE F004B IS FULLY OPEN
PERMISSIVE UNLESS AUX SIGNAL IS PRESENT
START
STOP
AIR CIRCUIT BREAKER 52
BREAKER CLOSED (A LOOP C1709)
C1711
RHR PUMP C002B
LIGHT ON WHEN IN MANUAL OVERRIDE
SEAL-IN
AUX CONT 42
L-10
L-2
MO-F006C
PERMISSIVE UNLESS AUX SIG IS PRESENT
BREAKER CLOSED
C1712
RHR PUMP C002C



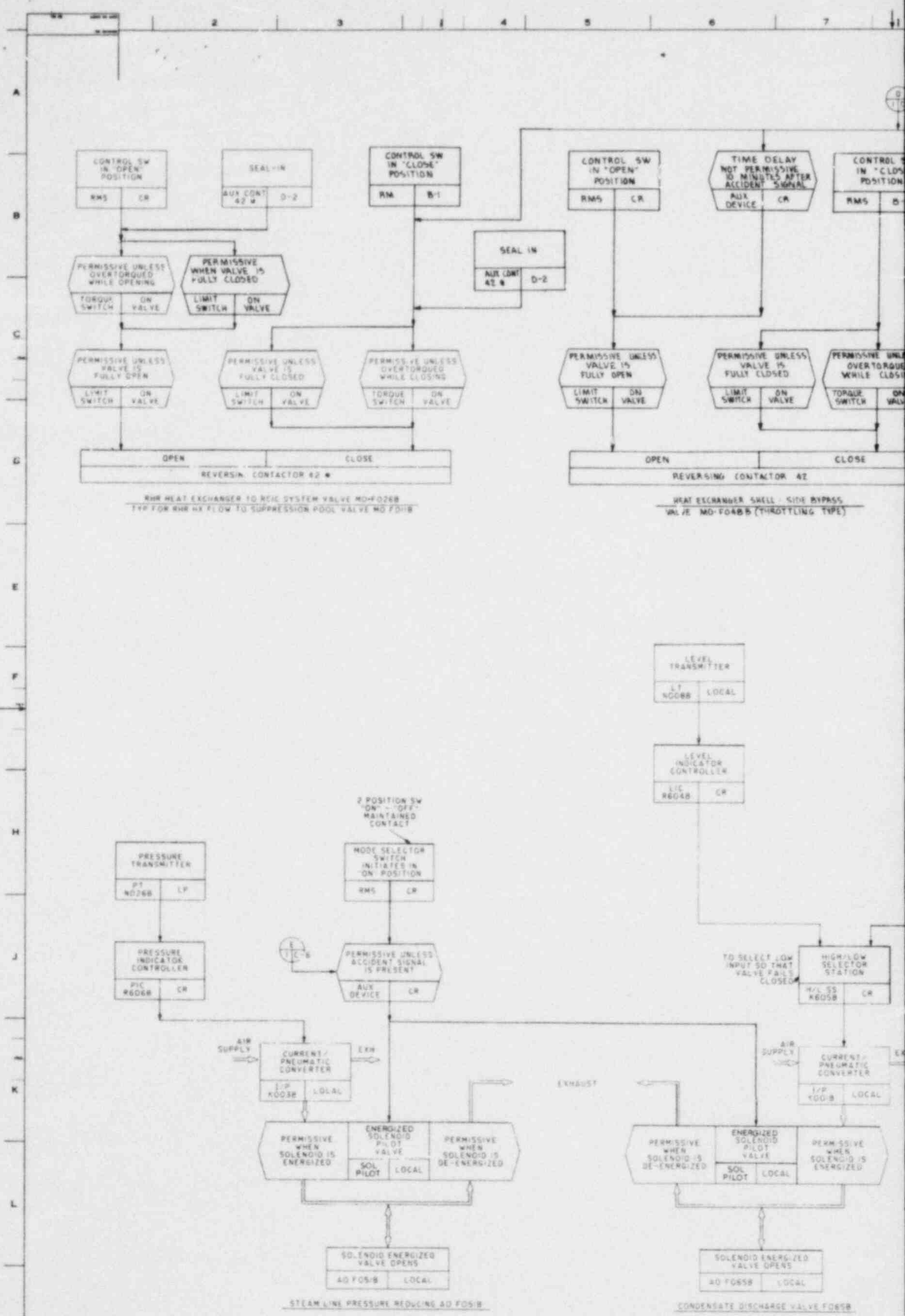
NOTES:

1. RHR LOOPS "B" & "C" LOGIC AND EQUIPMENT ARE SHOWN. RHR LOOP "A" IS IDENTICAL TO "B" EXCEPT CONTROL & MOTIVE POWER SHALL BE FROM THE SAME SOURCE AS THE LPCS SYSTEM (REF 7), AND AS NOTED.
2. PUMP MOTORS SHALL BE PROTECTED WITH OVERLOAD PROTECTION. PROTECTIVE RELAYS ARE TO BE APPLIED SO AS TO MAINTAIN POWER ON THE MOTOR AS LONG AS POSSIBLE WITHOUT IMMEDIATE DAMAGE TO EMERGENCY POWER SYSTEM.
3. VALVE MOTORS ARE TO BE PROVIDED WITH THERMAL OVERLOAD TRIPS AND ANNUNCIATION DURING TESTING AND LOSS OF POWER ANNUNCIATION.
4. UNLESS OTHERWISE NOTED, ALL RMS SHALL BE 3 POSITION SWITCHES "CLOSE"-"AUTO"-"OPEN", SPRING RETURN TO "AUTO" FROM "CLOSE" - "OPEN".
5. THE RHR SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH "PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEMS (IEEE 279)", AS APPLICABLE TO THE CONTROL CIRCUITRY.
6. THE RHR SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH REF 11.

REFERENCE DOCUMENTS:

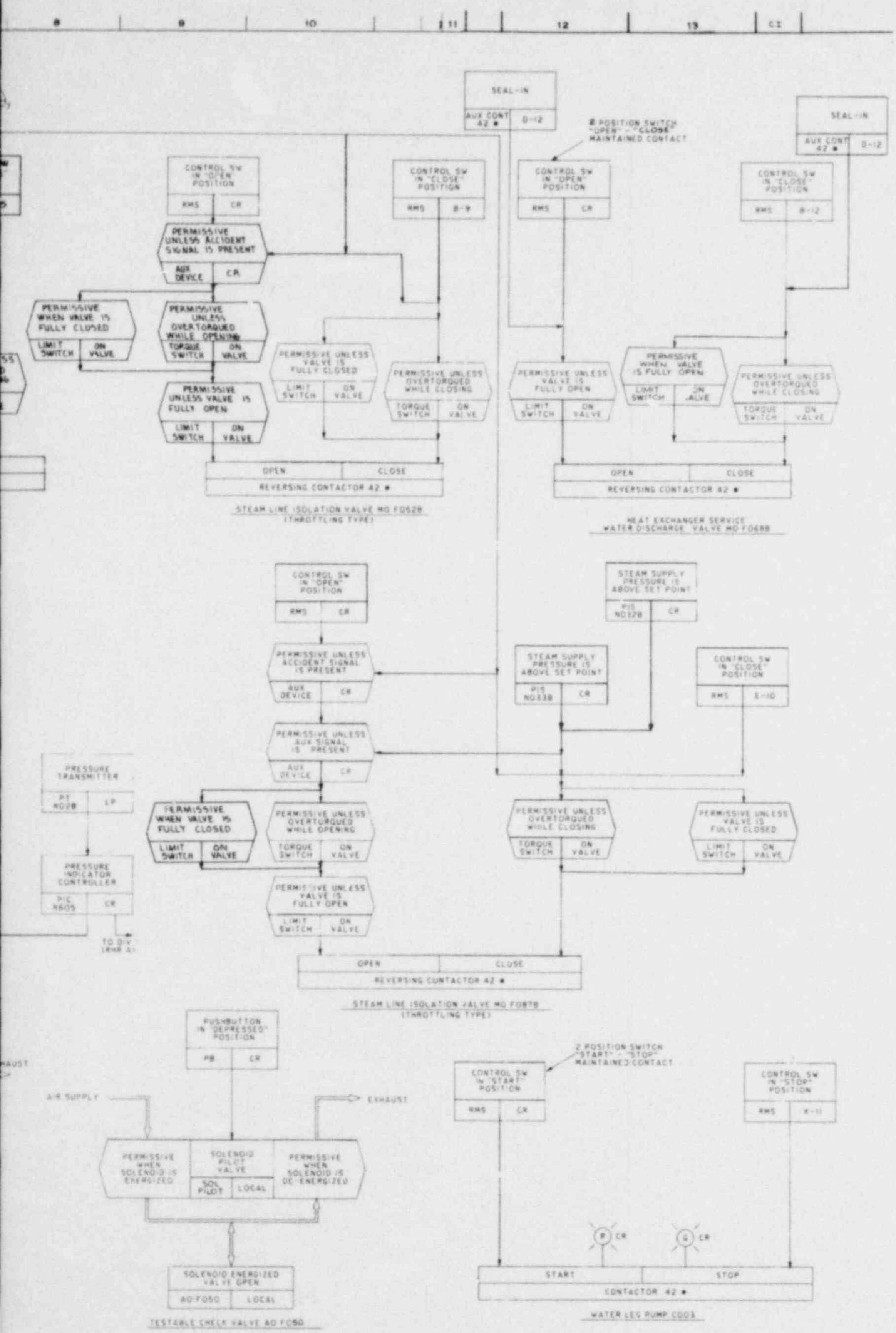
1.	E31-1030	LEAK DETECTION SYSTEM (FCD)
2.	B22-1010	NUCLEAR BOILER SYSTEM (P&ID)
3.	B22-1030	NUCLEAR BOILER SYSTEM (FCD)
4.	E51-1010	REAC CORE ISLN CLG SYSTEM (P&ID)
5.	E51-1030	REAC CORE ISLN CLG SYSTEM (FCD)
6.	E21-1010	LOW PRESS. CORE SPRAY SYS (P&ID)
7.	E21-1030	LOW PRESS. CORE SPRAY SYS (FCD)
8.	E31-1010	LEAK DETECTION SYSTEM (IED)
9.	B35-1010	REACTOR RECIRCULATION SYS (P&ID)
10.	B35-1020	REACTOR RECIRCULATION SYS (FCD)
11.	A62-4050	ELECTRICAL EQUIPMENT SEPARATION FOR PROTECTION SYSTEMS
12.	A42-1030	LOGIC SYMBOLS

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 1	FIGURE 7.3-14a

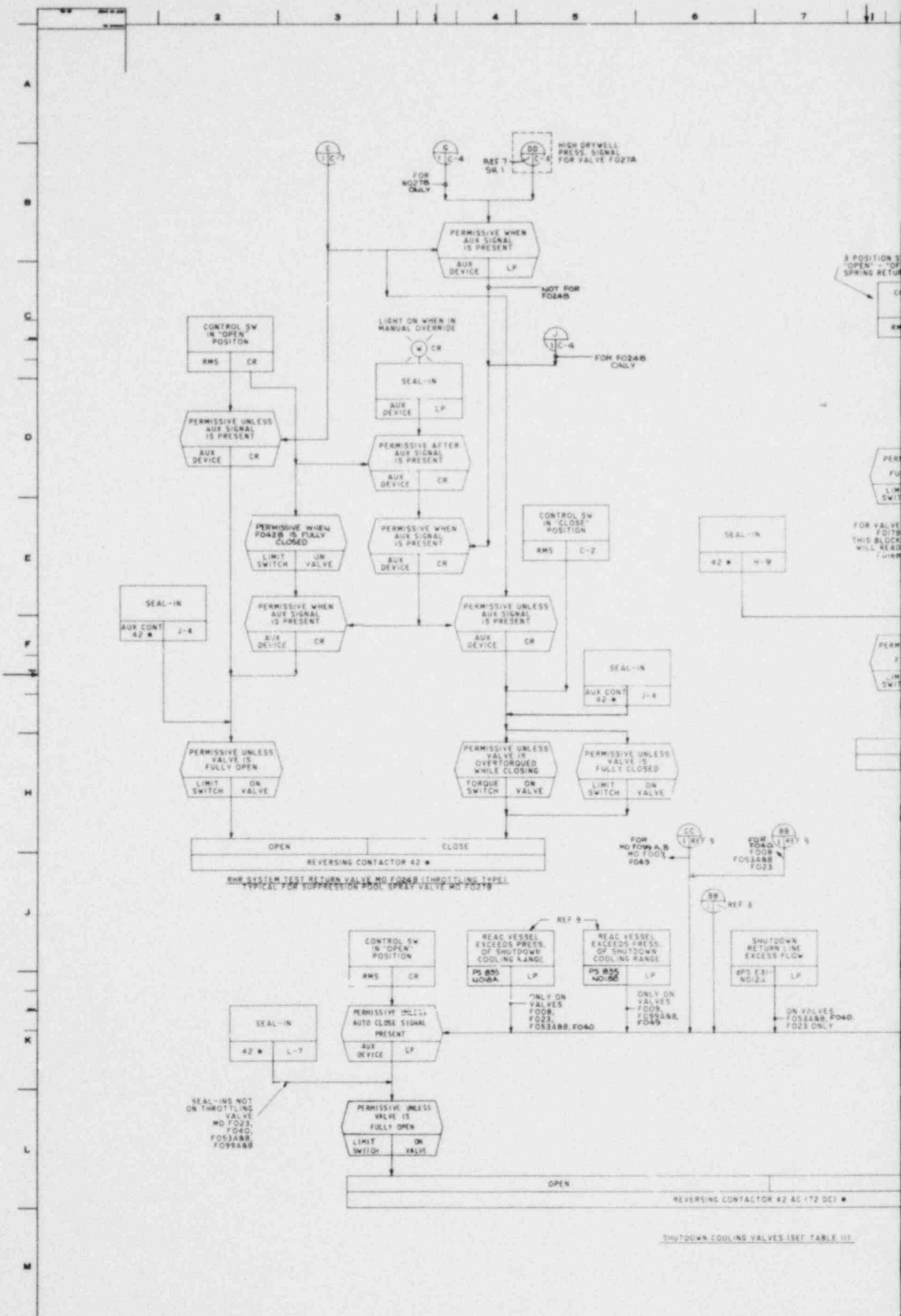


CONTROL SW IN "OPEN" POSITION
RMS
CR
SEAL-IN
AUX CONT 42 ●
D-2
CONTROL SW IN "CLOSE" POSITION
RM
B-1
PERMISSIVE UNLESS OVERTORQUED WHILE OPENING
TORQUE SWITCH
ON VALVE
PERMISSIVE WHEN VALVE IS FULLY CLOSED
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS OVERTORQUED WHILE CLOSING
TORQUE SWITCH
ON VALVE
OPEN
CLOSE
REVERSIN CONTACTOR 42 ●
RHR HEAT EXCHANGER TO RCIC SYSTEM VALVE MO-F026B
TYP FOR RHR HX FLOW TO SUPPRESSION POOL VALVE MO F011B
SEAL IN
AUX CONT 42 ●
D-2
CONTROL SW IN "OPEN" POSITION
RMS
CR
TIME DELAY NOT PERMISSIVE 10 MINUTES AFTER ACCIDENT SIGNAL
AUX DEVICE
CR
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
LIMIT SWITCH
ON VALVE
OPEN
CLOSE
REVERSING CONTACTOR 42
HEAT EXCHANGER SHELL - SIDE BYPASS
VALVE MO-F048B (THROTTLING TYPE)
LEVEL TRANSMITTER
LT N008B
LOCAL
LEVEL INDICATOR CONTROLLER
LIC R604B
CR
2 POSITION SW "ON" - "OFF" MAINTAINED CONTACT
PRESSURE TRANSMITTER
PT N026B
LP
MODE SELECTOR SWITCH INITIATES IN "ON" POSITION
RMS
CR
PRESSURE INDICATOR CONTROLLER
PIC R606B
CR
PERMISSIVE UNLESS ACCIDENT SIGNAL IS PRESENT
AUX DEVICE
CR
TO SELECT LOW INPUT SO THAT VALVE FAILS CLOSED
HIGH/LOW SELECTOR STATION
H/L SS K605B
CR
AIR SUPPLY
CURRENT/ PNEUMATIC CONVERTER
I/P K003B
LOCAL
EXH
EXHAUST
AIR SUPPLY
CURRENT/ PNEUMATIC CONVERTER
I/P Y00-B
LOCAL
PERMISSIVE WHEN SOLENOID IS ENERGIZED
ENERGIZED SOLENOID PILOT VALVE
SOL PILOT
LOCAL
PERMISSIVE WHEN SOLENOID IS DE-ENERGIZED
PERMISSIVE WHEN SOLENOID IS DE-ENERGIZED
ENERGIZED SOLENOID PILOT VALVE
SOL PILOT
LOCAL
PERMISSIVE WHEN SOLENOID IS ENERGIZED
SOLENOID ENERGIZED VALVE OPENS
AO F051B
LOCAL
STEAM LINE PRESSURE REDUCING AO F051B
SOLENOID ENERGIZED VALVE OPENS
AO F065B
LOCAL
CONDENSATE DISCHARGE VALVE F065B

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 2	FIGURE 7.3-14b



HIGH DRYWELL PRESS. SIGNAL FOR VALVE F027A
REF 7 SH.1
FOR N027B ONLY
PERMISSIVE WHEN AUX SIGNAL IS PRESENT
AUX DEVICE
LP
NOT FOR F024B
3 POSITION S "OPEN" - "OF SPRING RETUR
CONTROL SW IN "OPEN" POSITON
RMS
CR
LIGHT ON WHEN IN MANUAL OVERRIDE
FOR F024B ONLY
SEAL-IN
PERMISSIVE UNLESS AUX SIGNAL IS PRESENT
PERMISSIVE AFTER AUX SIGNAL IS PRESENT
PERMISSIVE WHEN F042B IS FULLY CLOSED
LIMIT SWITCH
ON VALVE
PERMISSIVE WHEN AUX SIGNAL IS PRESENT
CONTROL SW IN "CLOSE" POSITION
C-2
SEAL-IN
42 ✱
H-9
FOR VALVE
AUX CONT 42 ✱
J-4
PERMISSIVE WHEN AUX SIGNAL IS PRESENT
PERMISSIVE UNLESS AUX SIGNAL IS PRESENT
PERMISSIVE UNLESS VALVE IS FULLY OPEN
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE CLOSING
TORQUE SWITCH
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
OPEN
CLOSE
REVERSING CONTACTOR 42 ✱
RHR SYSTEM TEST RETURN VALVE MO F024B (THROTTLING TYPE)
TYPICAL FOR SUPPRESSION POOL SPRAY VALVE MO F027B
REF 5
REF 8
REF 9
CONTROL SW IN "OPEN" POSITION
REAC VESSEL EXCEEDS PRESS. OF SHUTDOWN COOLING RANGE
ONLY ON VALVES F008, F023, F053A&B, F040
ONLY ON VALVES F009, F099A&B, F049
SHUTDOWN RETURN LINE EXCESS FLOW
ON VALVES F053A&B, F040, F023 ONLY
SEAL-IN
42 ✱
L-7
PERMISSIVE UNLESS AUTO CLOSE SIGNAL PRESENT
SEAL-INS NOT ON THROTTLING VALVE MO F023, F040, F053A&B, F099A&B
PERMISSIVE UNLESS VALVE IS FULLY OPEN
OPEN
REVERSING CONTACTOR 42 AC (72 DC) ✱
SHUTDOWN COOLING VALVES (SEE TABLE III)

AMENDMENT NO. 10
July 1980

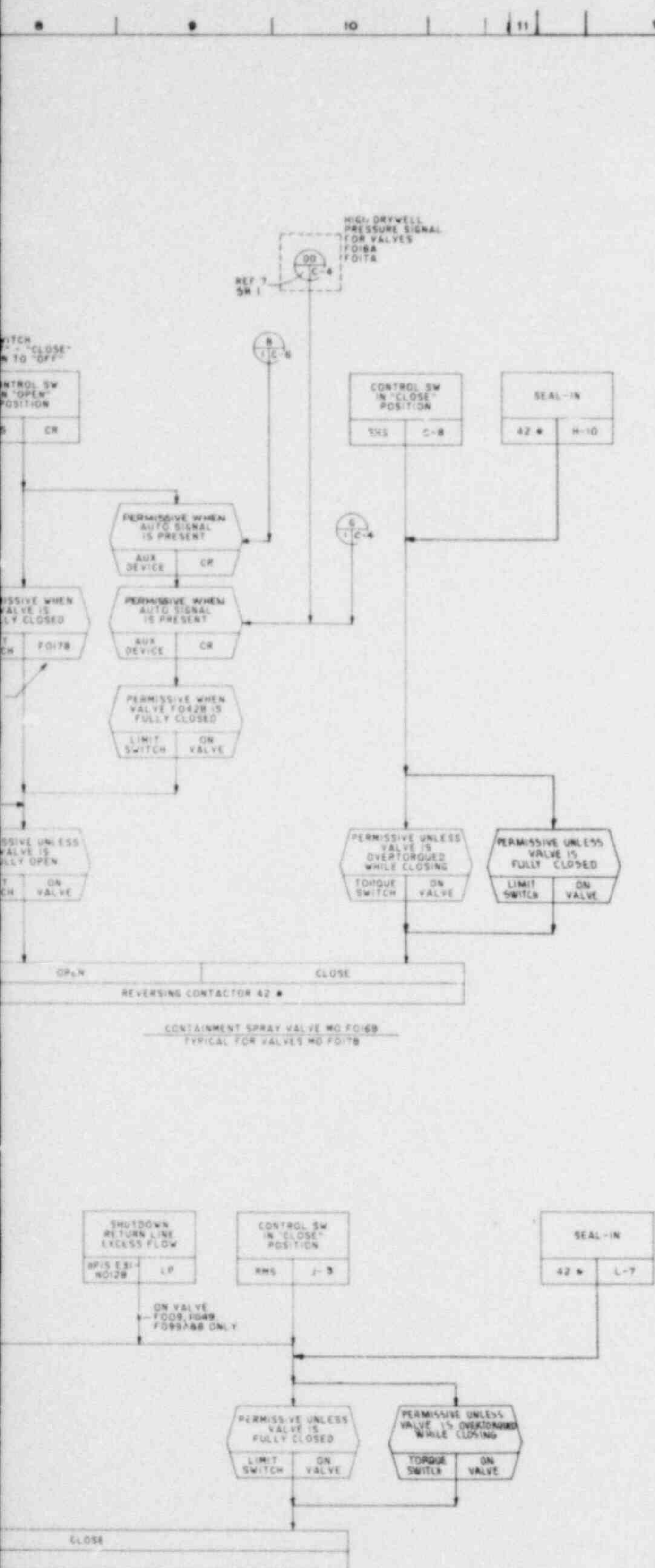
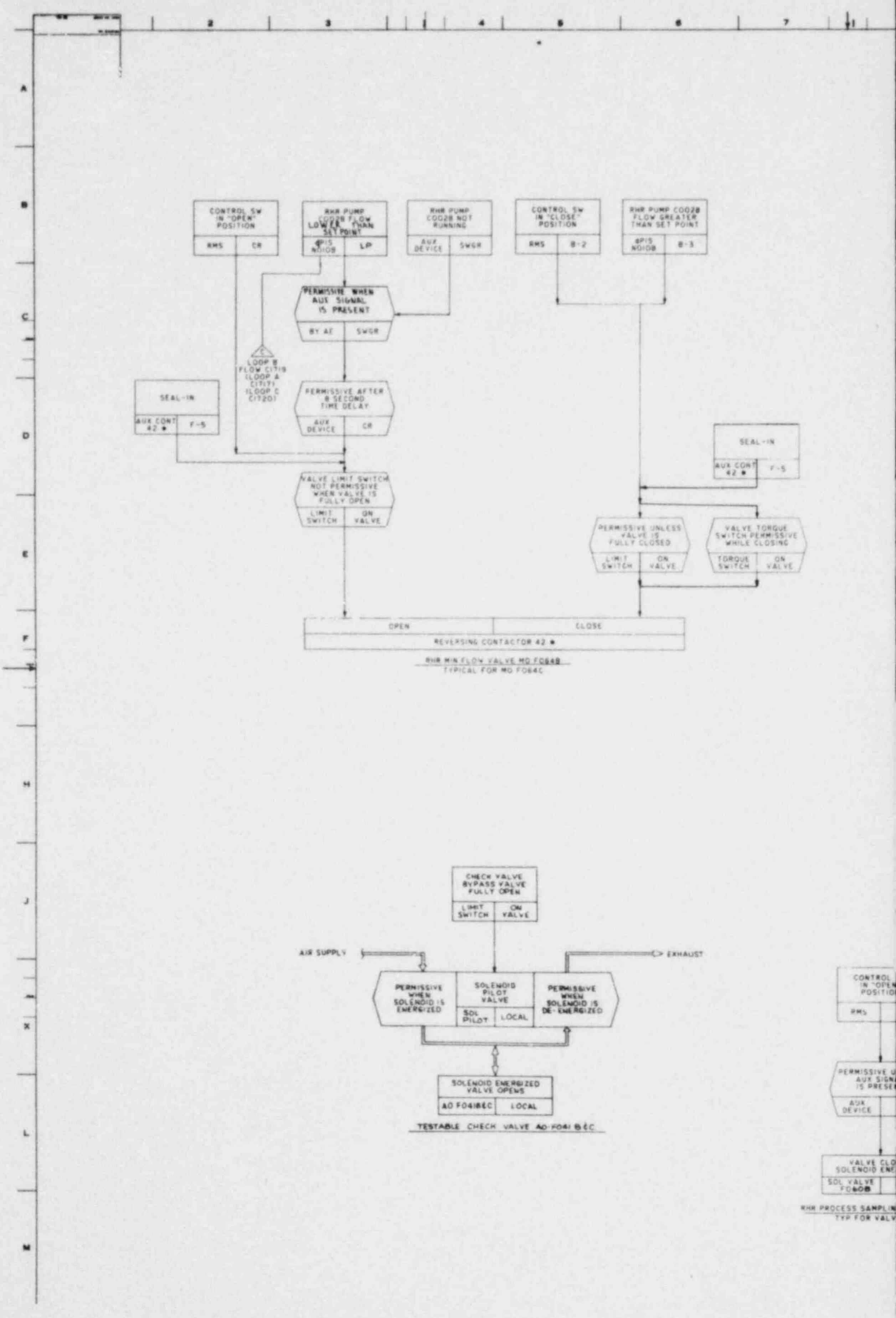
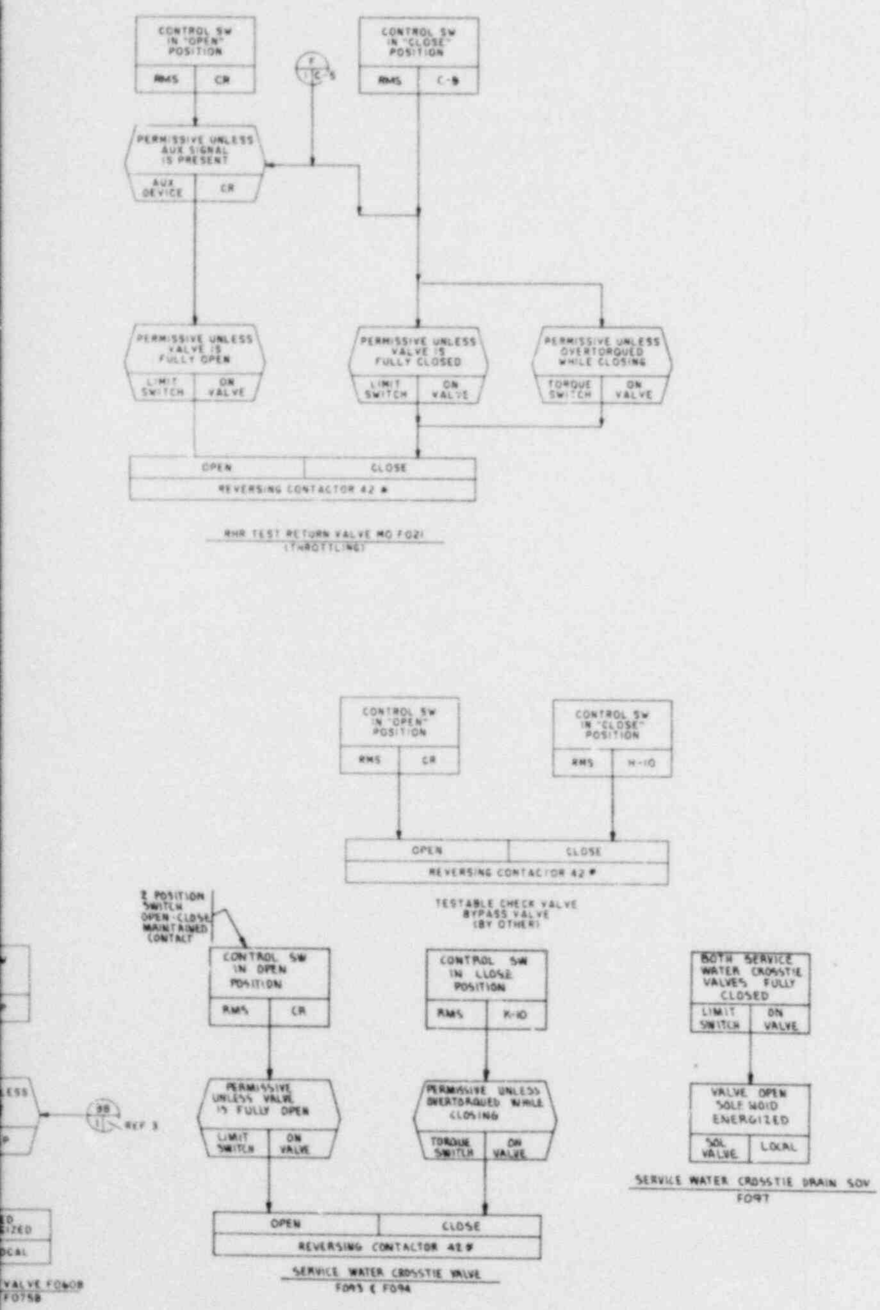


TABLE II

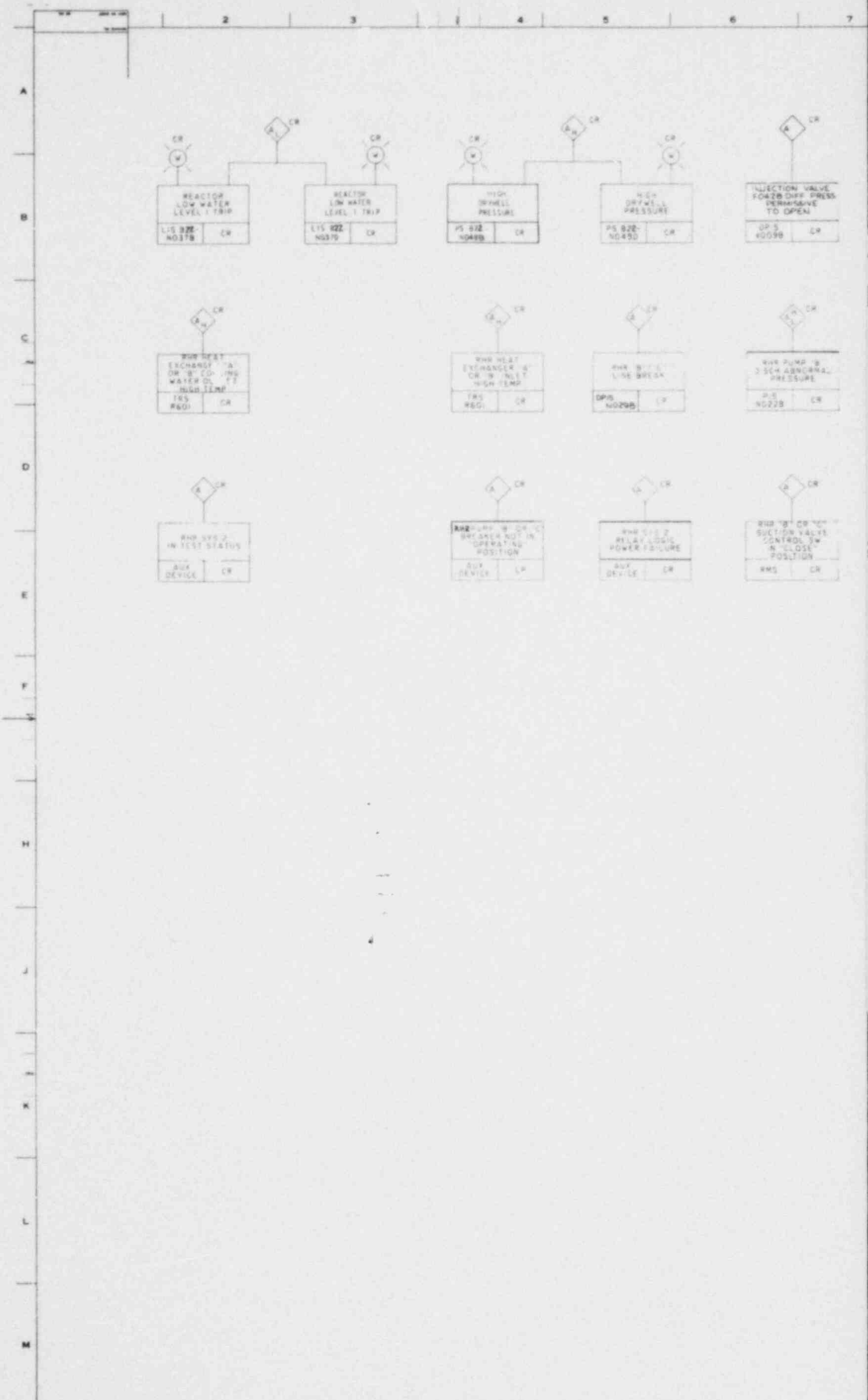
	VALVE NO.
SUCTION ISOLATION	MO F009
RADWASTE DISCHARGE	MO F040
INJECTION VALVE	MO F053A,B
RADWASTE DISCHARGE	MO F049
TEST CV BYPASS VALVE	MO F099A & B
HEAD SPRAY ISOLATION	MO F023
SUCTION ISOLATION	MO F008

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 3	FIGURE 7.3-14c


CONTROL SW IN "OPEN" POSITION
RMS
CR
RHR PUMP C002B FLOW LOWER THAN SETPOINT
LP
RHR PUMP C002B NOT RUNNING
AUX DEVICE
SWGR
CONTROL SW IN "CLOSE" POSITION
RMS
B-2
RHR PUMP C002B FLOW GREATER THAN SET POINT
B-3
PERMISSIVE WHEN AUX SIGNAL IS PRESENT
BY AE
SWGR
SEAL-IN
AUX CONT 42 *
F-5
PERMISSIVE AFTER 8 SECOND TIME DELAY
AUX DEVICE
CR
VALVE LIMIT SWITCH NOT PERMISSIVE WHEN VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
SEAL-IN
AUX CONT 42 *
F-5
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
LIMIT SWITCH
ON VALVE
VALVE TORQUE SWITCH PERMISSIVE WHILE CLOSING
TORQUE SWITCH
ON VALVE
OPEN
CLOSE
REVERSING CONTACTOR 42 *
RHR MIN FLOW VALVE MO F064B
TYPICAL FOR MO F064C
CHECK VALVE BYPASS VALVE FULLY OPEN
LIMIT SWITCH
ON VALVE
AIR SUPPLY
EXHAUST
PERMISSIVE WHEN SOLENOID IS ENERGIZED
SOLENOID PILOT VALVE
SOL PILOT
LOCAL
PERMISSIVE WHEN SOLENOID IS DE-ENERGIZED
SOLENOID ENERGIZED VALVE OPENS
AO F041B&C
LOCAL
TESTABLE CHECK VALVE AO-F041 B&C

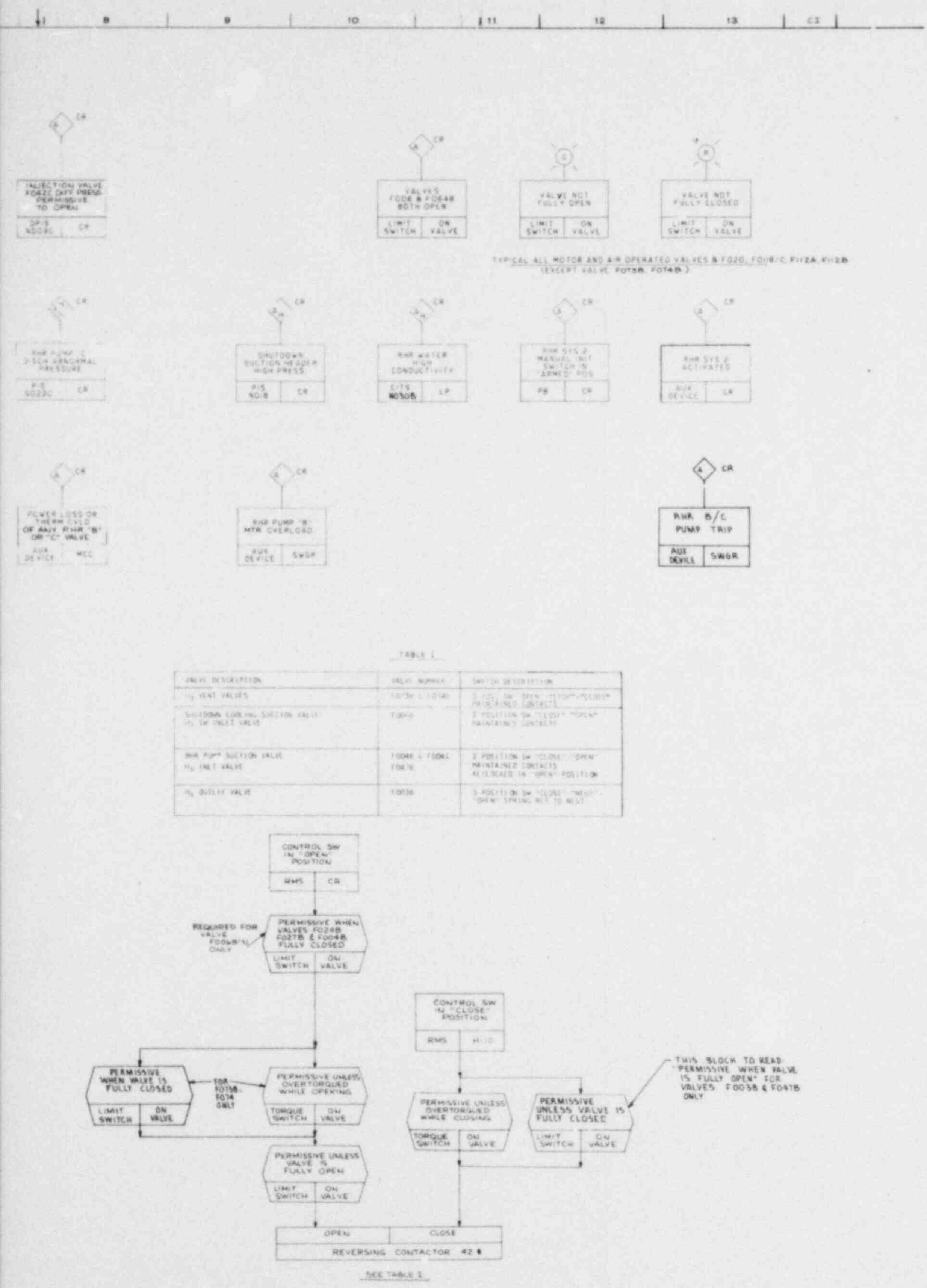


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 4	FIGURE 7.3-14d



2
3
4
5
6
7
A
B
C
D
E
F
H
J
K
L
M
CR
REACTOR LOW WATER LEVEL 1 TRIP
LIS B21E-N037B
CR
REACTOR LOW WATER LEVEL 1 TRIP
LIS B22-N037D
HIGH DRYWELL PRESSURE
PS B22-N048B
HIGH DRYWELL PRESSURE
PS B22-N045D
INJECTION VALVE F042B DIFF PRESS PERMISSIVE TO OPEN
RHR HEAT EXCHANGER "B" OR "D" INLET HIGH TEMP
TRS R601
RHR PUMP "B" DISCH ABNORMAL PRESSURE
RHR SYS 2 IN TEST STATUS
AUX DEVICE
RHR PUMP "B" OR "C" BREAKER NOT IN OPERATING POSITION
RHR SYS 2 RELAY LOGIC POWER FAILURE
RHR "B" OR "C" SUCTION VALVE CONTROL SW IN "CLOSE" POSITION
RMS

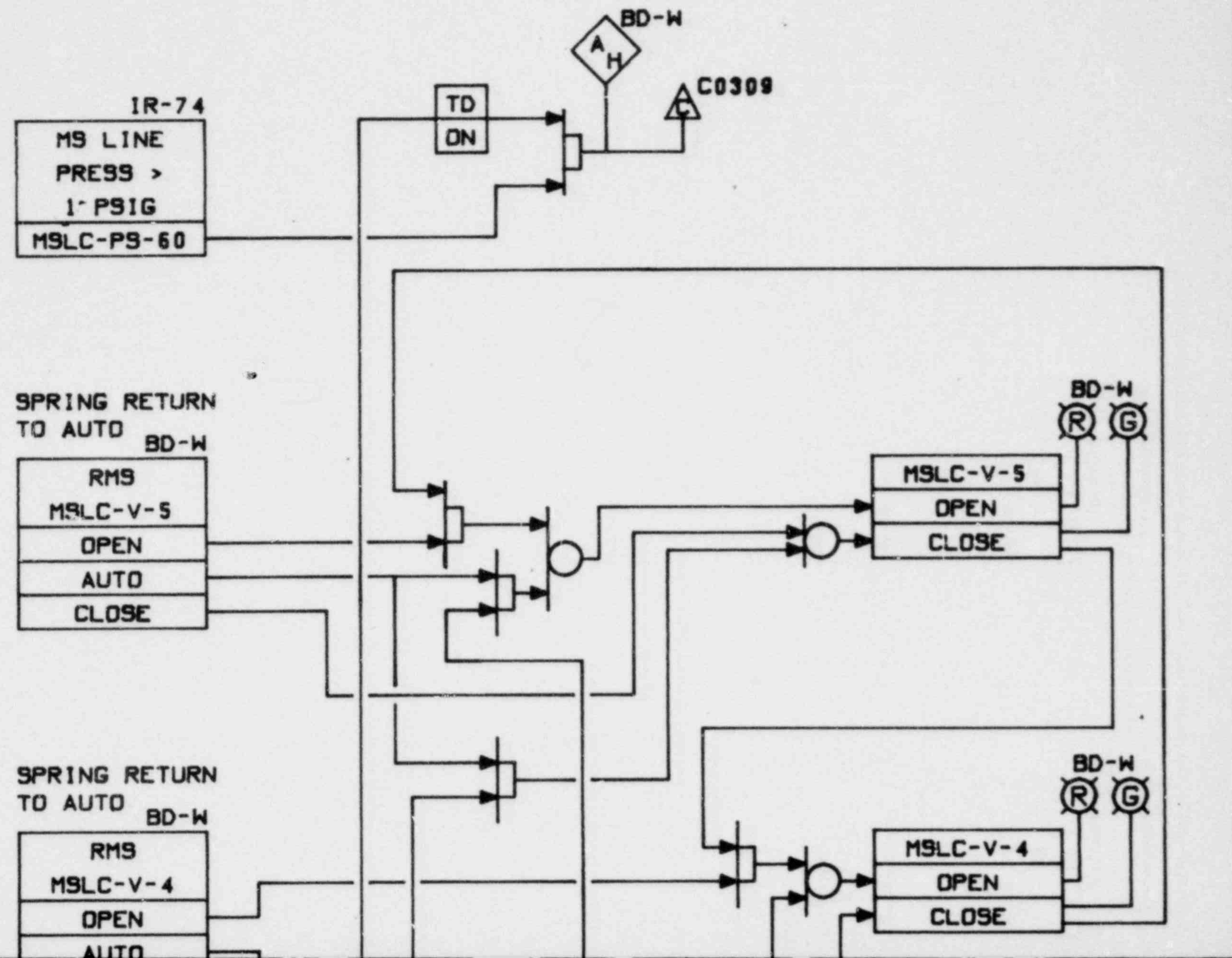
AMENDMENT NO. 10
July 1980



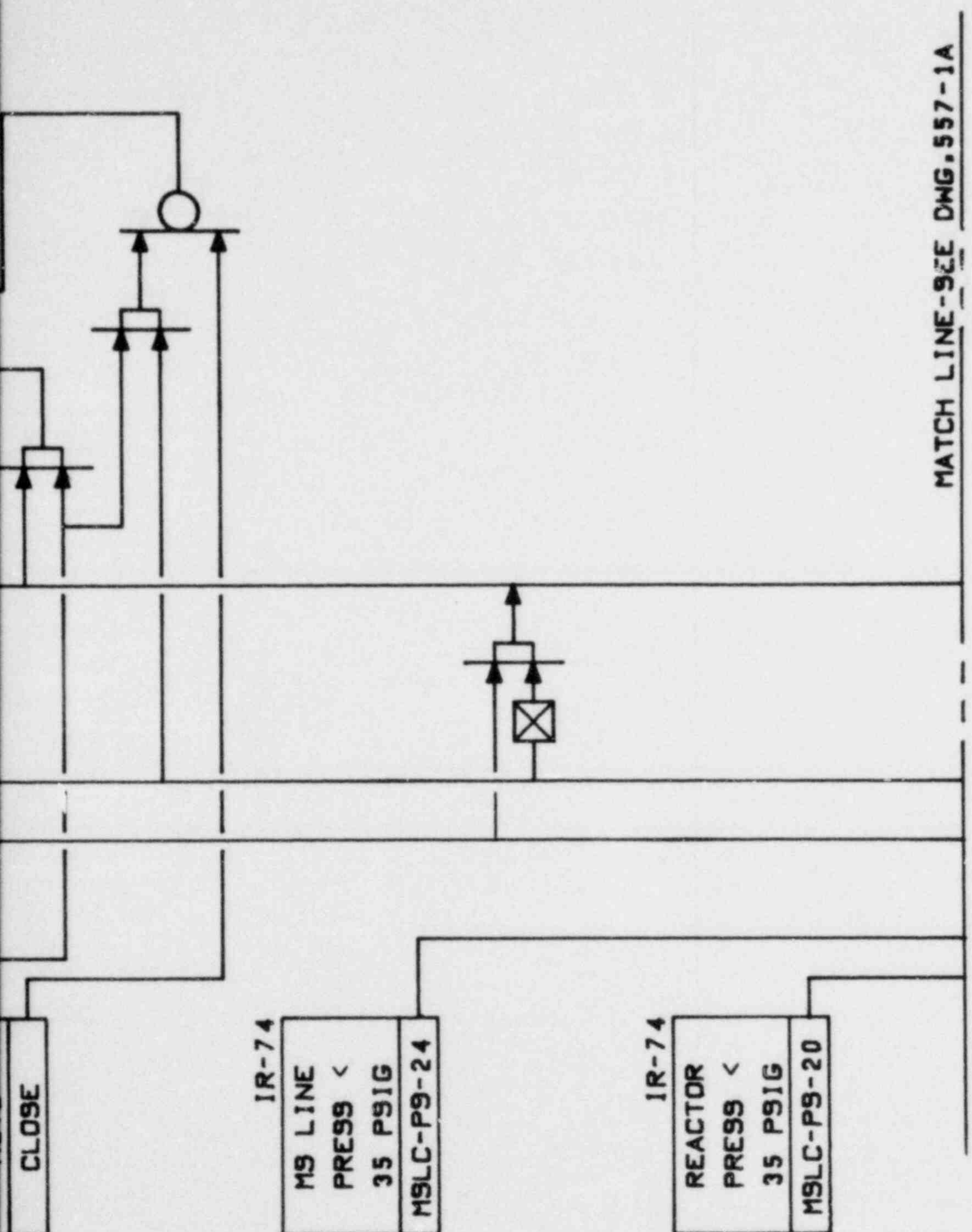
VALVE DESCRIPTION	VALVE NUMBER	SWITCH DESCRIPTION
H_2 VENT VALVES	[illegible]	3 POS. SW "OPEN"-"STOP"-"CLOSE" MAINTAINED CONTACTS
SHUTDOWN COOLING SUCTION VALVE H_2 SW INLET VALVE	[illegible]	2 POSITION SW "CLOSE"-"OPEN" MAINTAINED CONTACTS
RHR PUMP SUCTION VALVE H_2 INLET VALVE	F004B & F004C [illegible]	2 POSITION SW "CLOSE"-"OPEN" MAINTAINED CONTACTS [illegible] IN "OPEN" POSITION
H_2 OUTLET VALVE	[illegible]	3 POSITION SW "CLOSE"-"NEUT"-"OPEN" SPRING RET TO NEUT

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RESIDUAL HEAT REMOVAL SYSTEM FCD SHEET 5	FIGURE 7.3-14e

OUTBOARD SYSTEM DIV.II



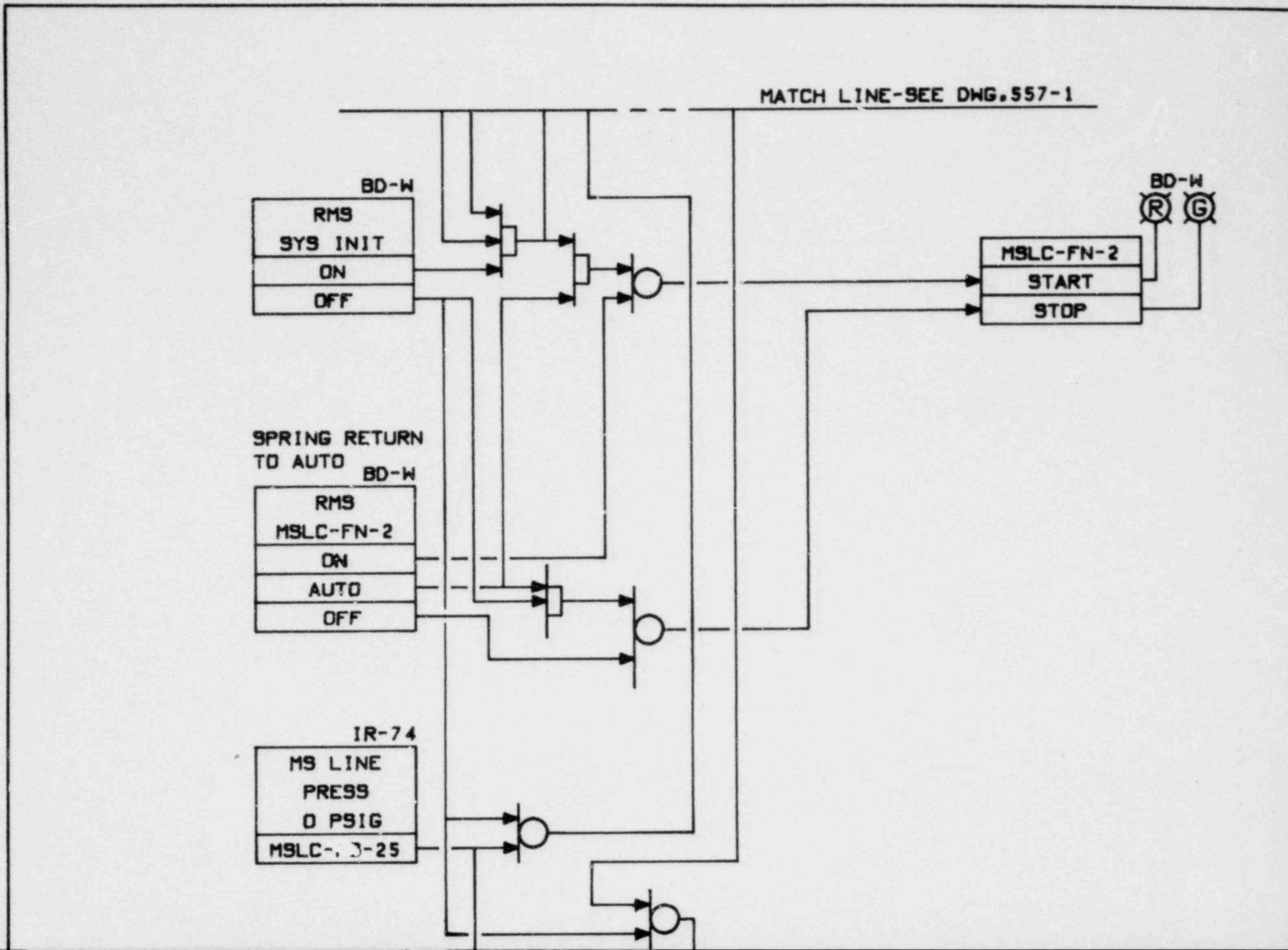
AMENDMENT NO. 10
July 1980



I & C DWG. NO. M-620

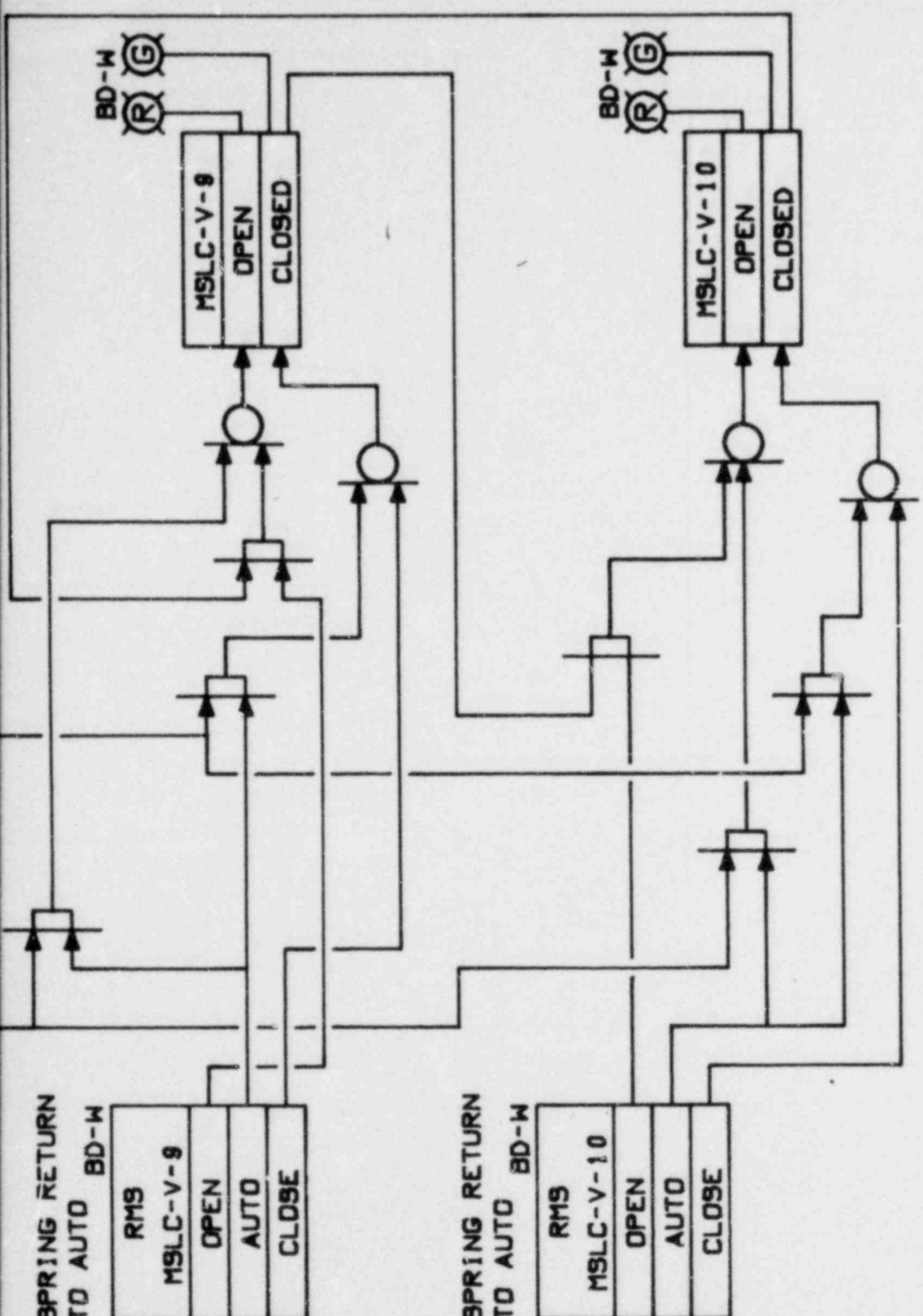
SHEET 557-1

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - MSIV-LCS	FIGURE 7.3-15a



MATCH LINE-SEE DWG.557-1
BD-W
RMS
SYS INIT
ON
OFF
BD-W
R
G
MSLC-FN-2
START
STOP
SPRING RETURN
TO AUTO
BD-W
RMS
MSLC-FN-2
ON
AUTO
OFF
IR-74
MS LINE
PRESS
0 PSIG

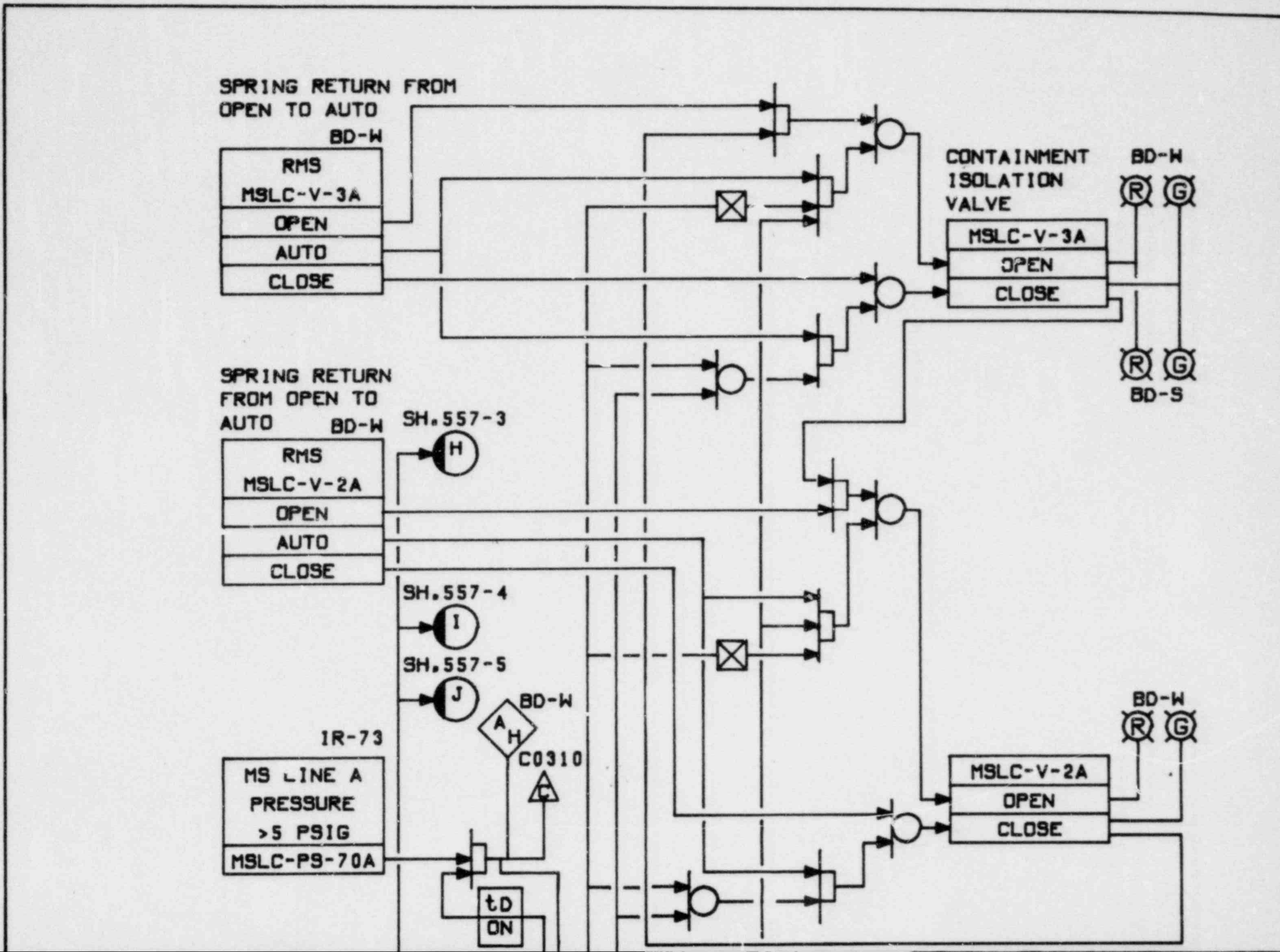
AMENDMENT NO. 10
July 1980



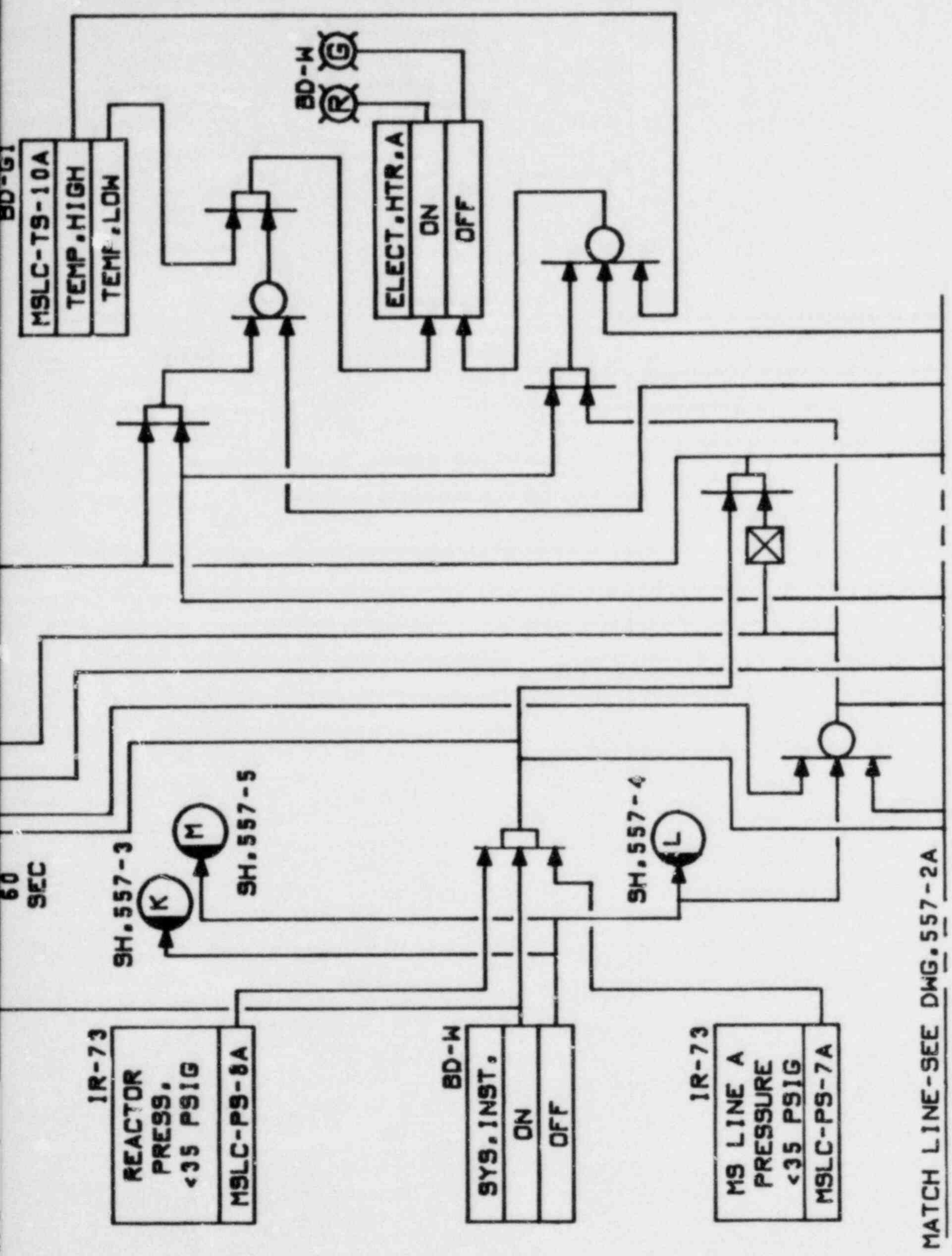
I & C DWG. NO. M-620

SHEET 557-1A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - MSIV-LCS	FIGURE 7.3-15b


SPRING RETURN FROM
OPEN TO AUTO
BD-W
RMS
MSLC-V-3A
OPEN
AUTO
CLOSE
CONTAINMENT
ISOLATION
VALVE
MSLC-V-3A
OPEN
CLOSE
BD-W
R
G
R
G
BD-S
SPRING RETURN
FROM OPEN TO
AUTO
BD-W
RMS
MSLC-V-2A
OPEN
AUTO
CLOSE
SH.557-3
H
SH.557-4
I
SH.557-5
J
BD-W
A
H
C0310
C
IR-73
MS LINE A
PRESSURE
>5 PSIG
MSLC-PS-70A
tD
ON
BD-W
R
G
MSLC-V-2A
OPEN
CLOSE

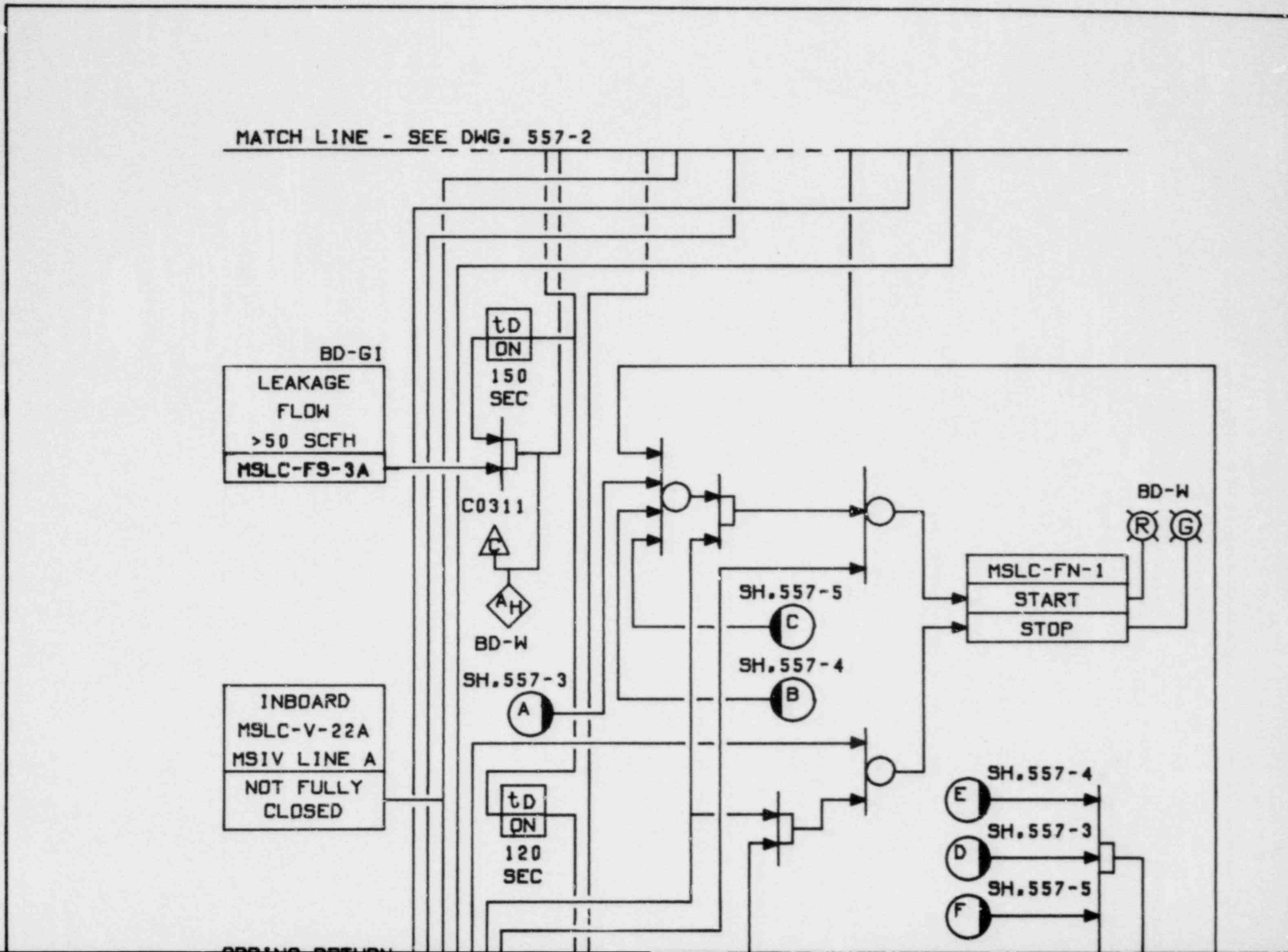
AMENDMENT NO. 10
July 1980



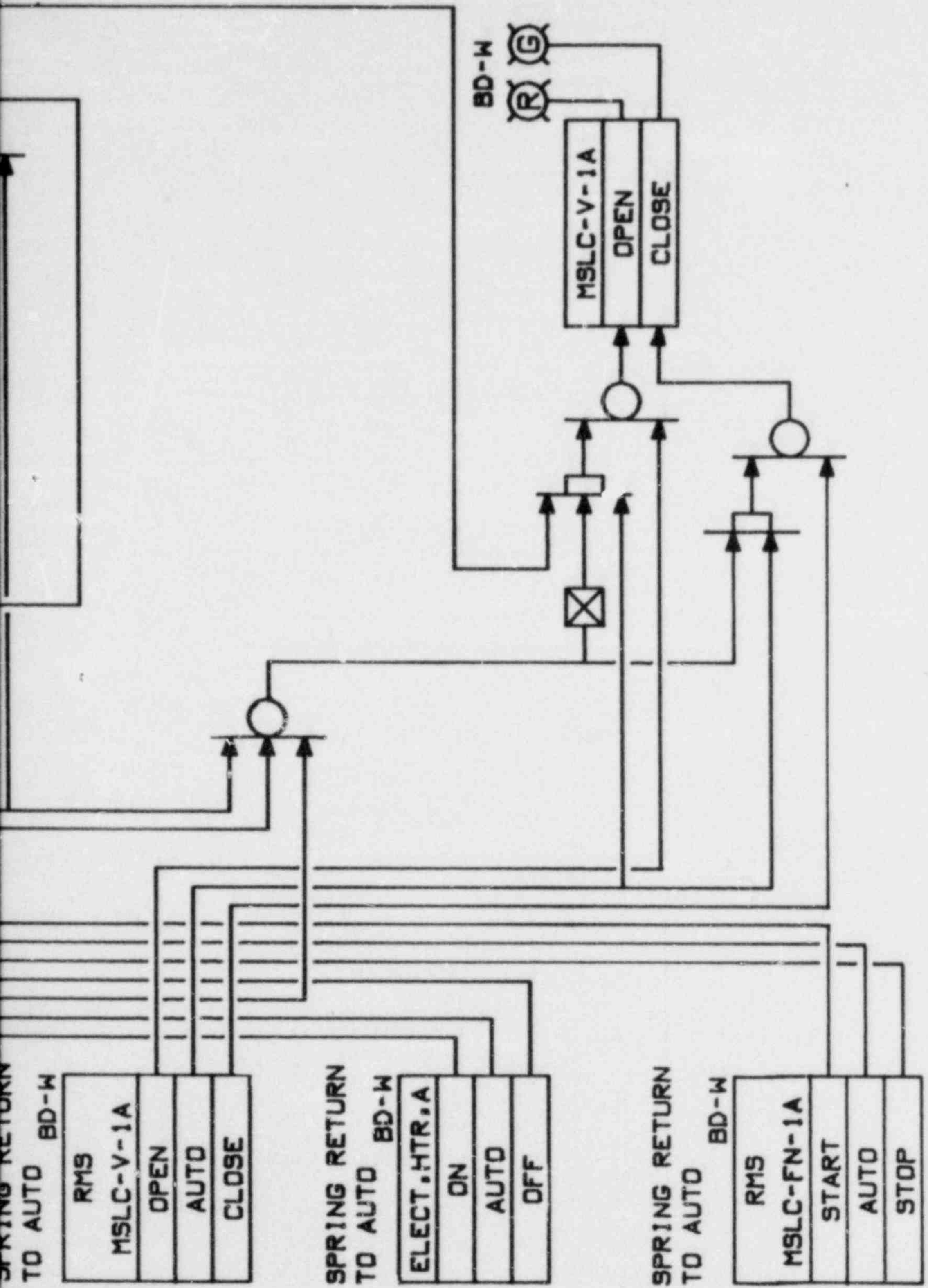
I & C DWG. NO. M-620

SHEET 557-2

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - MSIV-LCS	FIGURE 7.3- 15c


MATCH LINE - SEE DWG. 557-2
BD-GI
LEAKAGE
FLOW
>50 SCFH
MSLC-FS-3A
tD
ON
150
SEC
C0311
C
AH
BD-W
SH.557-3
A
INBOARD
MSLC-V-22A
MSIV LINE A
NOT FULLY
CLOSED
tD
ON
120
SEC
SH.557-5
C
SH.557-4
B
BD-W
R
G
MSLC-FN-1
START
STOP
SH.557-4
E
SH.557-3
D
SH.557-5
F

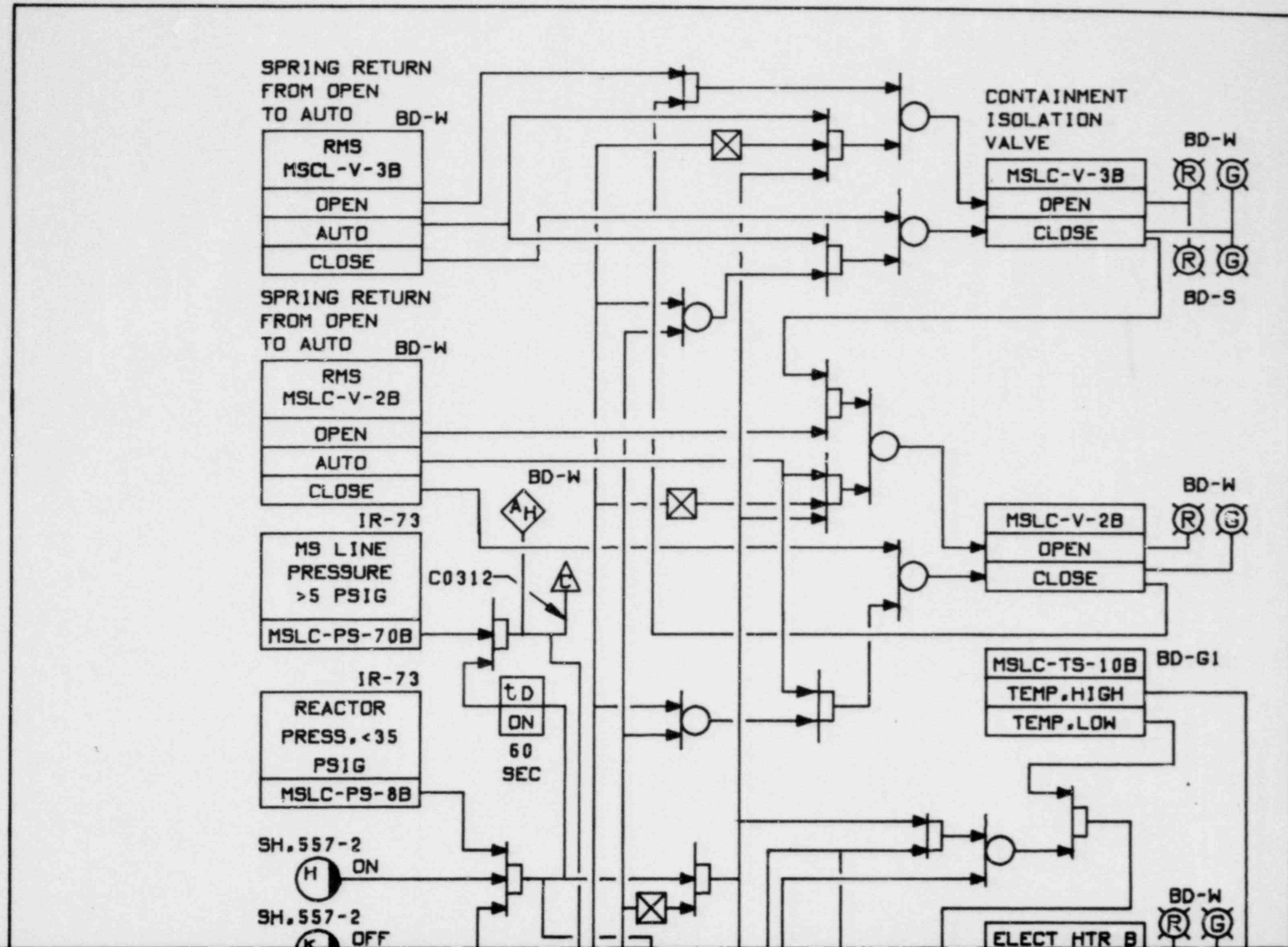
AMENDMENT NO. 10
July 1980



I & C DWG. NO. M-620

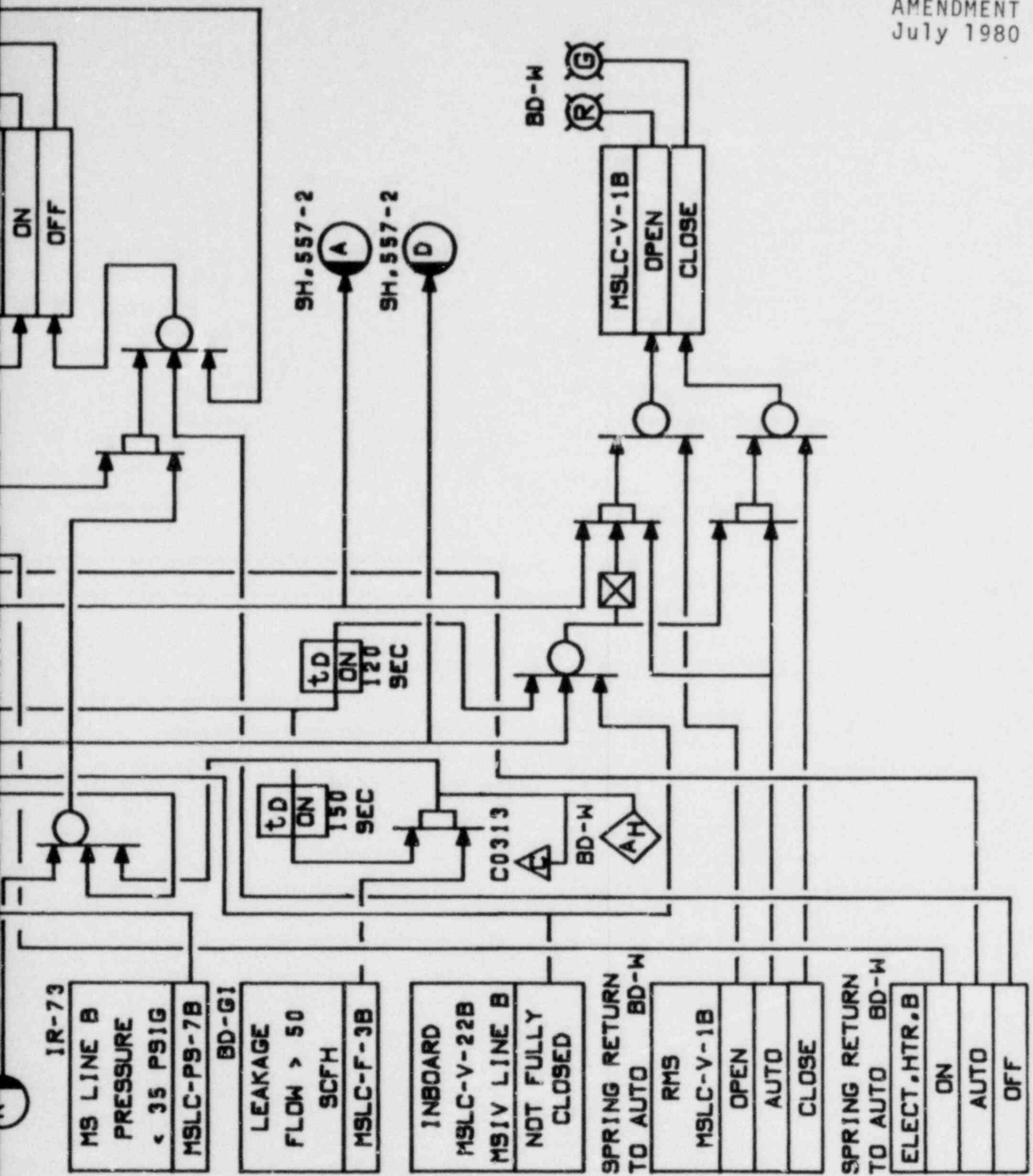
SHEET 557-2A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - MSIV-LCS	FIGURE 7.3-15d



SPRING RETURN FROM OPEN TO AUTO
BD-W
RMS MSCL-V-3B
OPEN
AUTO
CLOSE
CONTAINMENT ISOLATION VALVE
BD-W
MSLC-V-3B
OPEN
CLOSE
BD-S
SPRING RETURN FROM OPEN TO AUTO
BD-W
RMS MSLC-V-2B
OPEN
AUTO
CLOSE
BD-W
BD-W
MSLC-V-2B
OPEN
CLOSE
IR-73
MS LINE PRESSURE >5 PSIG
MSLC-PS-70B
C0312
IR-73
REACTOR PRESS.<35 PSIG
MSLC-PS-8B
tD
ON
60 SEC
MSLC-TS-10B
TEMP.HIGH
TEMP.LOW
BD-G1
SH.557-2
ON
SH.557-2
OFF
BD-W
ELECT HTR B

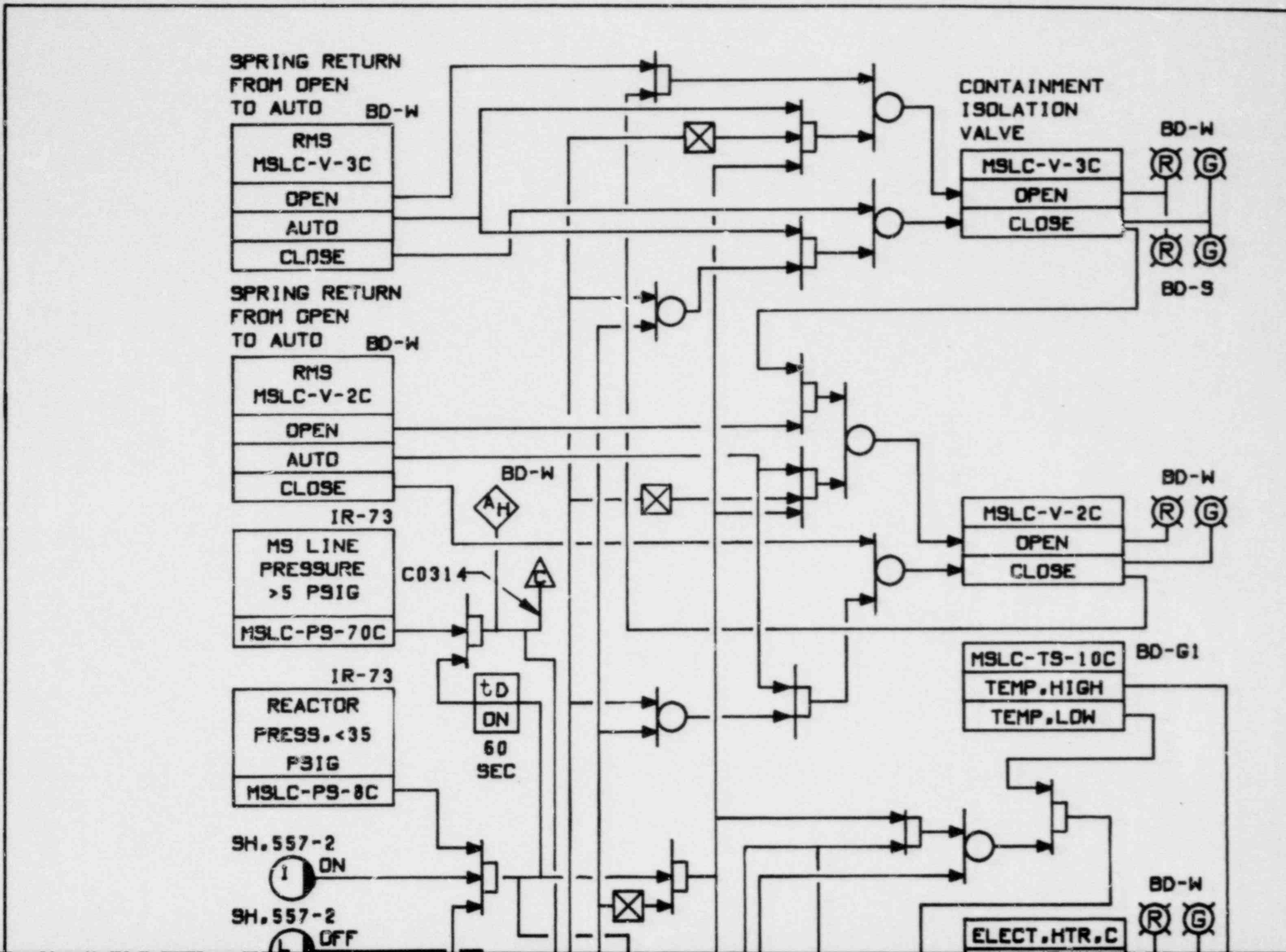
AMENDMENT NO. 10
July 1980



I & C DWG. NO. M-620

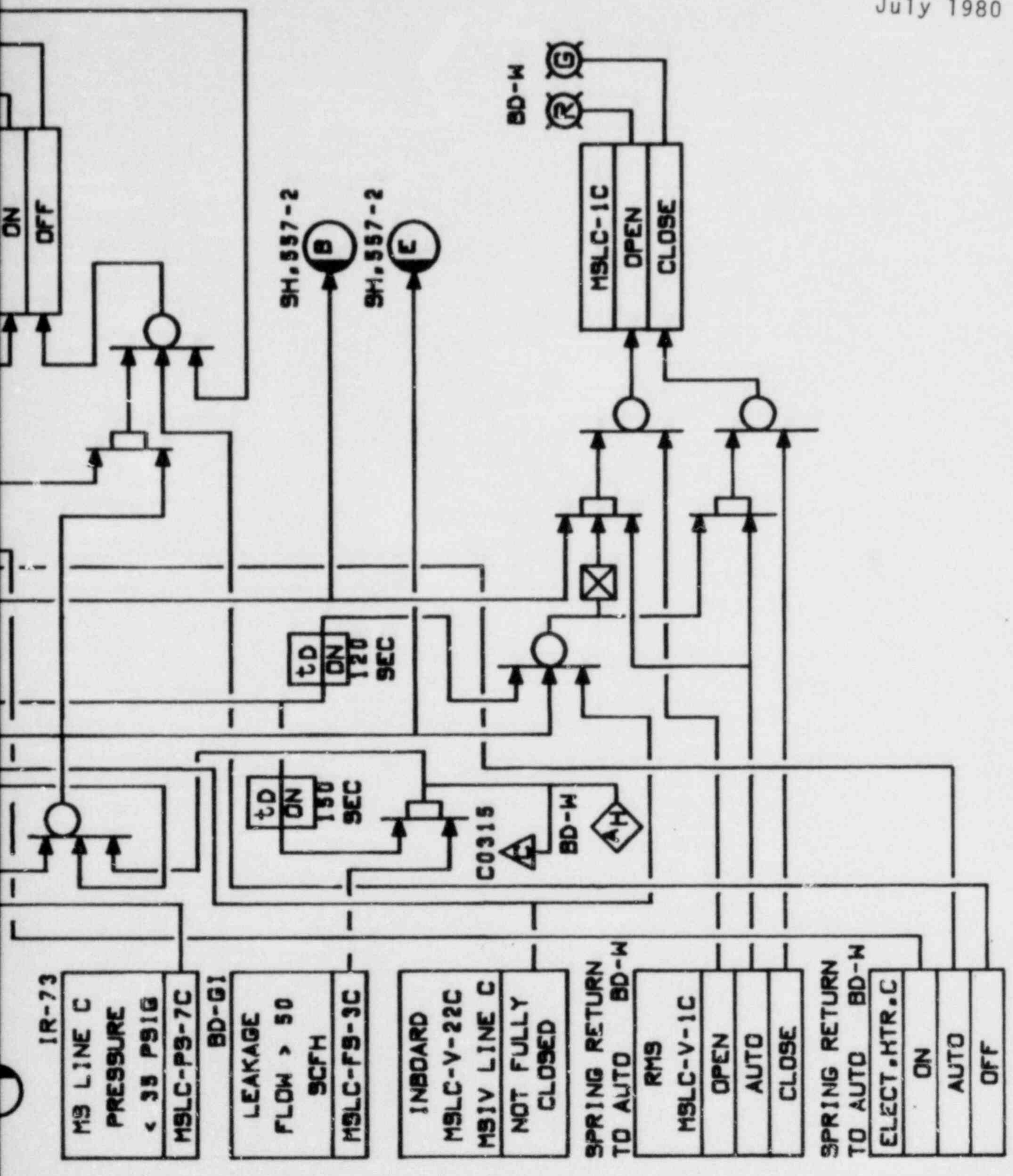
SHEET 557-3

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - MSIV-LCS	FIGURE 7.3-15e



SPRING RETURN
FROM OPEN
TO AUTO
BD-W
RMS
MSLC-V-3C
OPEN
AUTO
CLOSE
SPRING RETURN
FROM OPEN
TO AUTO
BD-W
RMS
MSLC-V-2C
OPEN
AUTO
CLOSE
BD-W
IR-73
MS LINE
PRESSURE
>5 PSIG
MSLC-PS-70C
C0314
IR-73
REACTOR
PRESS.<35
PSIG
MSLC-PS-8C
tD
ON
60
SEC
SH.557-2
ON
SH.557-2
OFF
CONTAINMENT
ISOLATION
VALVE
MSLC-V-3C
OPEN
CLOSE
BD-W
BD-S
MSLC-V-2C
OPEN
CLOSE
BD-W
MSLC-TS-10C
TEMP.HIGH
TEMP.LOW
BD-G1
ELECT.HTR.C
BD-W

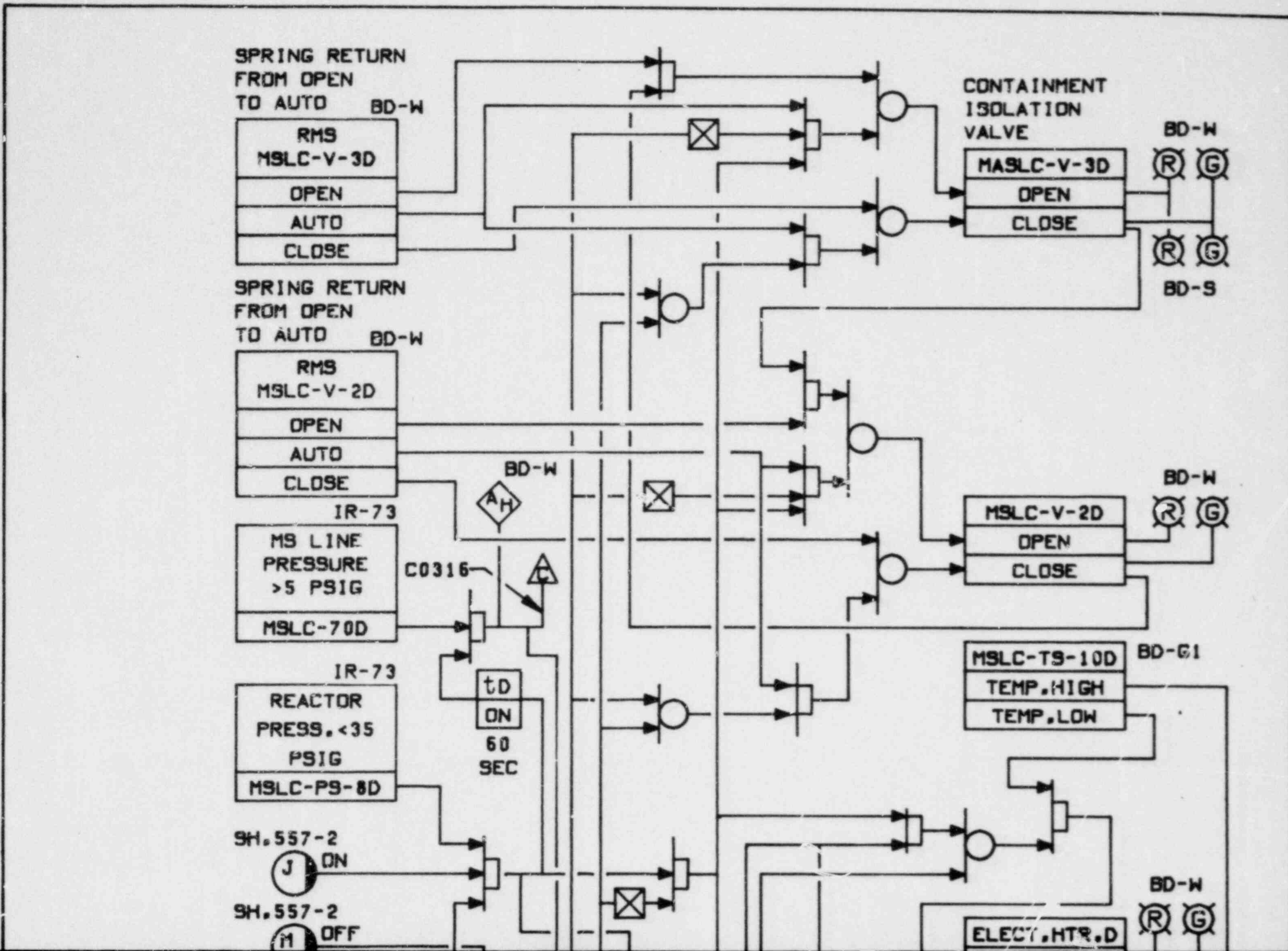
AMENDMENT NO. 10
July 1980

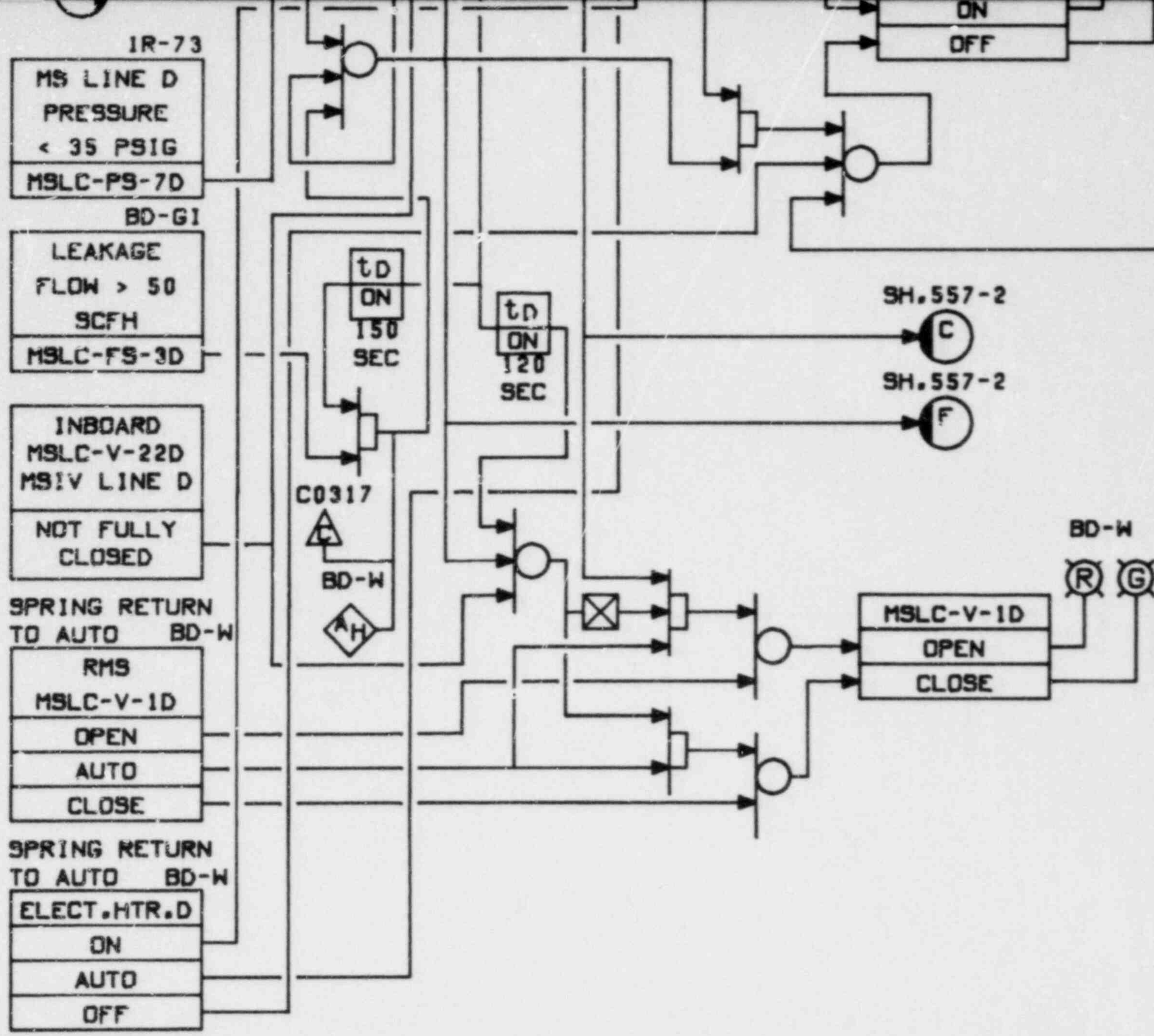


I & C DWG. NO. M-620

SHEET 557-4

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - MSIV-LCS	FIGURE 7.3- 15f


SPRING RETURN FROM OPEN TO AUTO
BD-W
RMS MSLC-V-3D
OPEN
AUTO
CLOSE
CONTAINMENT ISOLATION VALVE
MASLC-V-3D
OPEN
CLOSE
BD-W
BD-S
SPRING RETURN FROM OPEN TO AUTO
BD-W
RMS MSLC-V-2D
OPEN
AUTO
CLOSE
BD-W
IR-73
MS LINE PRESSURE >5 PSIG
MSLC-70D
C0316
MSLC-V-2D
OPEN
CLOSE
BD-W
MSLC-TS-10D
BD-G1
TEMP.HIGH
TEMP.LOW
IR-73
REACTOR PRESS.<35 PSIG
MSLC-PS-8D
tD
ON
60 SEC
SH.557-2
ON
SH.557-2
OFF
BD-W
ELECT.HTR.D



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

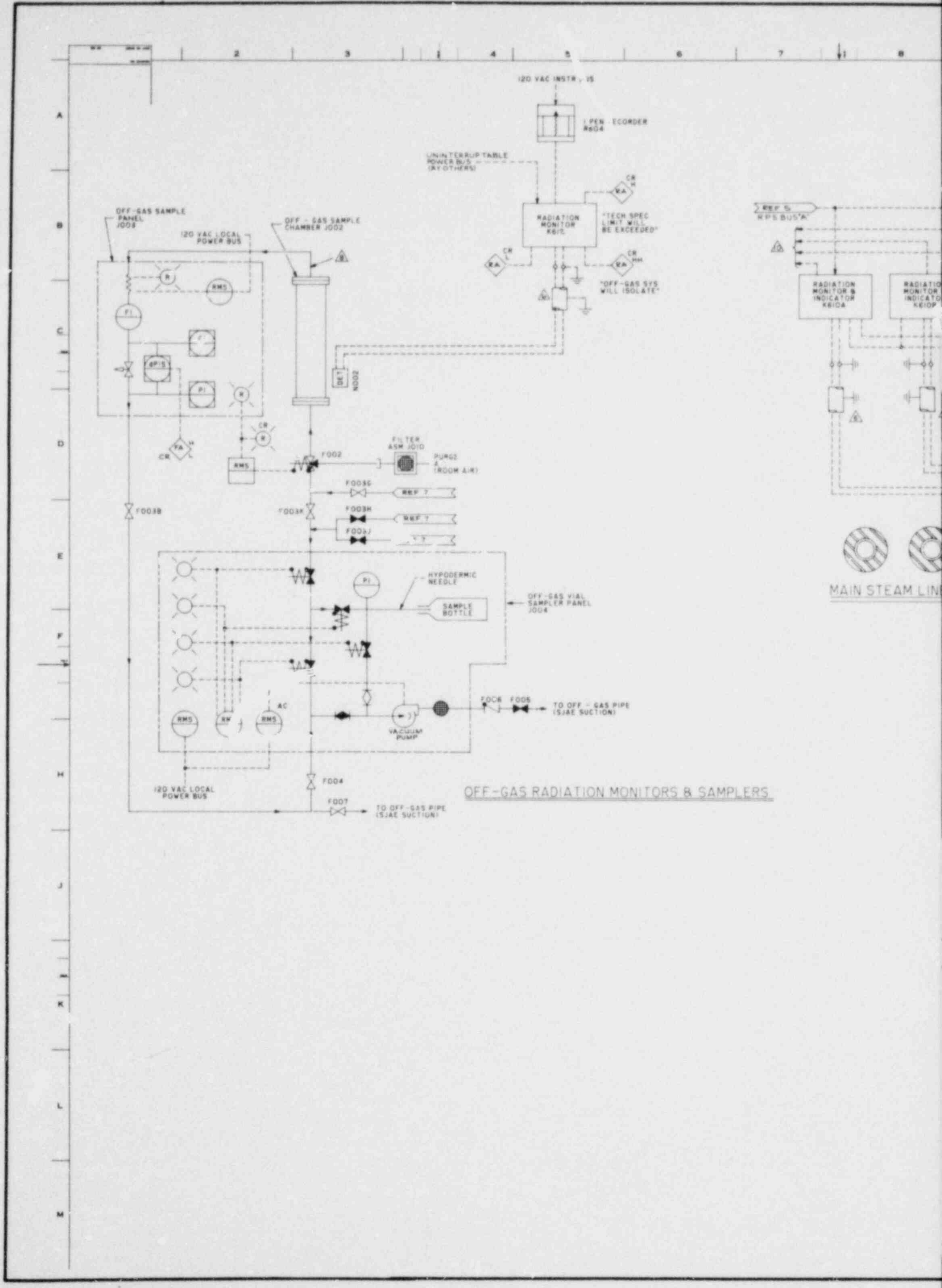
CONTROL LOGIC DIAGRAM - MSIV-LCS

FIGURE 7.3-15g

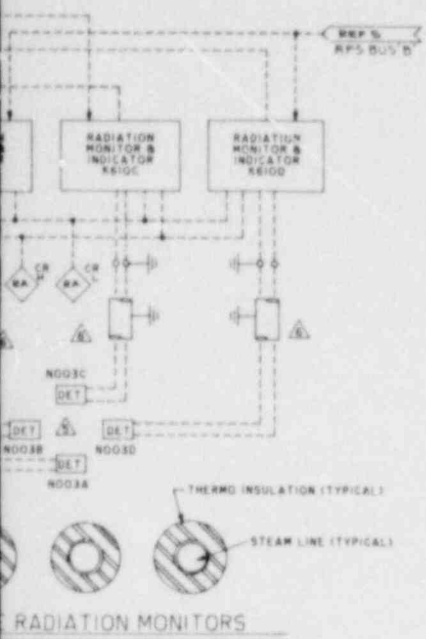
SHEET 557-5

I & C DWG. NO. M-620

AMENDMENT NO. 10
July 1980


120 VAC INSTR. BUS
1 PEN RECORDER R604
UNINTERRUPTABLE POWER BUS (BY OTHERS)
RADIATION MONITOR K615
"TECH SPEC LIMIT WILL BE EXCEEDED"
"OFF-GAS SYS WILL ISOLATE"
OFF-GAS SAMPLE PANEL J003
120 VAC LOCAL POWER BUS
OFF - GAS SAMPLE CHAMBER J002
DET N002
FILTER ASM J010
PURGE A (ROOM AIR)
F002
F003G
F003H
F003J
F003K
F003B
REF 7
HYPODERMIC NEEDLE
SAMPLE BOTTLE
OFF-GAS VIAL SAMPLER PANEL J004
VACUUM PUMP
F006
F005
TO OFF - GAS PIPE (SJAE SUCTION)
F004
F007
REF 5
RPS BUS 'A'
RADIATION MONITOR & INDICATOR K610A
RADIATION MONITOR & INDICATOR K610P
MAIN STEAM LIN
OFF-GAS RADIATION MONITORS & SAMPLERS

FCF 259X571 02,05,08,010 (D17-1010)



NOTES:

1. THE OFF GAS VENT PIPE GAS SAMPLE LINE SHALL BE 1" X 0.058" WALL THICKNESS SEAMLESS STAINLESS STEEL TUBING. THE TUBING MIN BEND RADIUS SHALL BE 20". THE TUBING LENGTH SHALL BE JOINED WITH SWAGELOK TYPE 1610-6-316 UNIONS. THE TUBING SHALL SLOPE SO THAT THE CONDENSATE WILL RUN TO DRAIN TEE.
2. A REMOVABLE SECTION SHALL BE PROVIDED NEAR THE ISOKINETIC PROBE FOR THE INSERTION OF A CHARCOAL FILTER HOLDER. THE FITTINGS ETC. SHALL PROVIDE SMOOTH TRANSITIONS WITHOUT DISCONTINUITIES OR REDUCING THE CROSS SECTIONAL AREA OF THE FLOW STREAM.
3. TEE SHALL BE UNION TEE SWAGELOK TYPE 1610-3-316.
4. ALARMS ARE ACTUATED BY RELAYS IN TRIP AUX. UNIT. DOWNSCALE ALARMS FOR LIQUID RADIATION MONITORS ARE ANNUNCIATED ON A SINGLE COMMON ANNUNCIATOR.
5. THE MAIN STEAM LINE RADIATION MONITOR DETECTORS (N003) SHALL BE LOCATED WITHIN THE STEAM LINE TUNNEL AS CLOSE AS PRACTICAL TO THE PRIMARY CONTAINMENT. THE DETECTORS SHALL BE ARRANGED SUCH THAT EACH DETECTOR WILL VIEW ALL STEAM LINES WITH APPROXIMATELY THE SAME RESPONSE. IT OS RECOMMENDED THAT THE DETECTOR OR DETECTOR ASSEMBLY BE FASTENED TO A ROD OR A PIPE AND INSERTED INTO SEALED PIPE WALLS FROM OUTSIDE THE STEAM TUNNEL. CAREFULLY ROUTE CABLES TO MINIMIZE HEAT EXPOSURE. NO LEAD SHIELDING IS REQUIRED.
6. ALL CABLES SHALL COMPLY WITH GE ENGR. SPEC REF 3.
7. ADDITIONAL ALARM IN RADWASTE BLDG (RAH) RADWASTE MONITOR ONLY.
8. DRAIN AT THE LOWTR POINT OF OFF GAS SAMPLE LINE. SAMPLE LINE SHALL HAVE 2 MINUTE TRANSIY TIME TO ALLOW N16 DECAY.
9. TWO OUT OF TWO HIGH HIGH RADIATION (RAHH) OR INOPERATIVE TRIP (CHANNELS A AND B) SHALL:
   -- SHUTDOWN AND ISOLATE (OUTBOARD VALVE) REACTOR BUILDING VENTILATION SYSTEM.
   -- INITIATE STANDBY GAS TREATMENT SYSTEM TRAIN B.
   -- CLOSE OUTBOARD PRIMARY CONTAINMENT PURGE AND VENT VALVES.
   TWO OUT OF TWO HIGH-HIGH RADIATION (RAHH) OR INOPERATIVE TRIP (CHANNELS C AND D) SHALL:
   -- SHUTDOWN AND ISOLATE (INBOARD VALVE) REACTOR BUILDING VENTILATION SYSTEM.
   -- INITIATE STANDBY GAS TREATMENT SYSTEM TRAIN A
   -- CLOSE INBOARD PRIMARY CONTAINMENT PURGE AND VENT VALVES.
   -- ANY ONE HIGH-HIGH RADIATION TRIP (RAHH) SHALL ALARM. SEE REF 7
10. ONE HIGH HIGH RADIATION (RAHH OR INOPERATIVE TRIP OUT OF TWO ON TRIP SYSTEM "A" AND ONE HIGH-HIGH RADIATION (RAHH) OR INOPERATIVE TRIP OUT OF TWO ON TRIP SYSTEM "B" SHALL:
    -- CLOSE MAIN STEAM LINE ISOLATION VALVES.
    -- SCRAM REACTOR
    -- TURN OFF MECHANICAL VACUUM PUMP & CLOSE MECHANICAL VACUUM PUMP LINE VALVE.
    -- ANY ONE HIGH-HIGH RADIATION TRIP (RAHH) SHALL ALARM.
11. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET (SEE REF 9)
12. ANYONE UPSCALE TRIP (RAH) SHALL CLOSE BYPASS LINE VALVE, OPEN TREATMENT LINE VALVE AND ALARM.
13. ISOLATE OFF-GAS SYSTEM OUTLET AND DRAIN VALVES AND ALARM (REF 8).
14. *SUPPLIED AND MOUNTED BY OTHERS.*

SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NUMBERS

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. PIPING & INST. SYMBOLS	A42-1010
2. PROCESS RADIATION MONITORING DES. SPEC	D17-4010
3. SPECIAL WIRE & CABLE	A62-4010
4. RADWASTE SYSTEM P&ID	G11THRU G16-1010
5. REACTOR PROTECTION SYS IED	C71-C72-1010
6. NUCLEAR BOILER SYS FCD	B21-B22-1030
7. OFF-GAS SYSTEM P&ID	N62-N64-1010
8. OFF-GAS SYSTEM FCD	N62-1030
9. INST DATA SH.	D17-3050

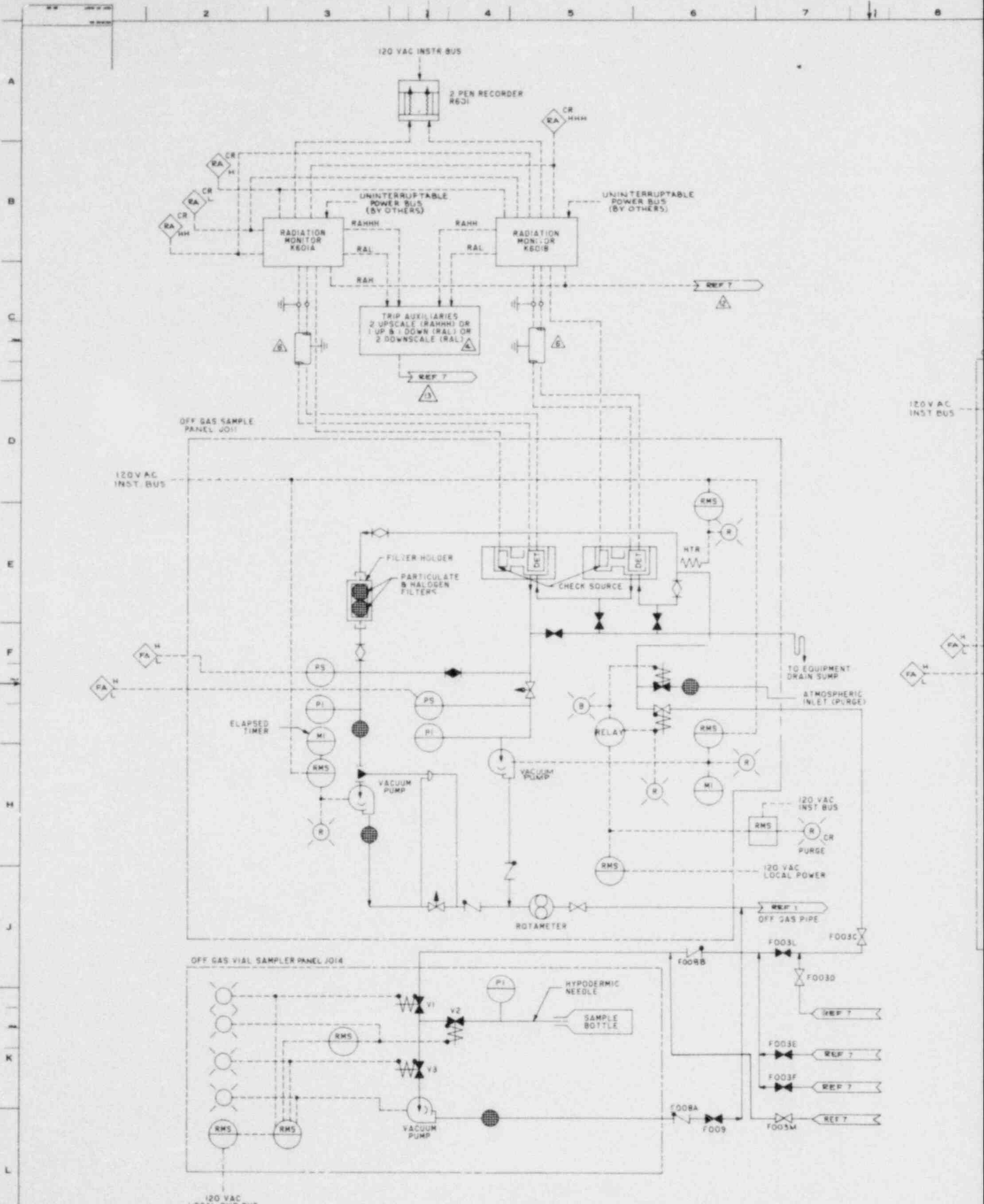
LEGEND:

SJAE	STEAM JET AIR EJECTOR
DET	DETECTOR
RAHHH	RADIATION ALARM HIGH HIGH HIGH
RAHH	RADIATION HIGH HIGH
RAH	RADIATION HIGH
RAL	DOWNSCALE OR INSTRUMENT TROUBLE
FAH/L	FLOW ALARM HIGH/LOW

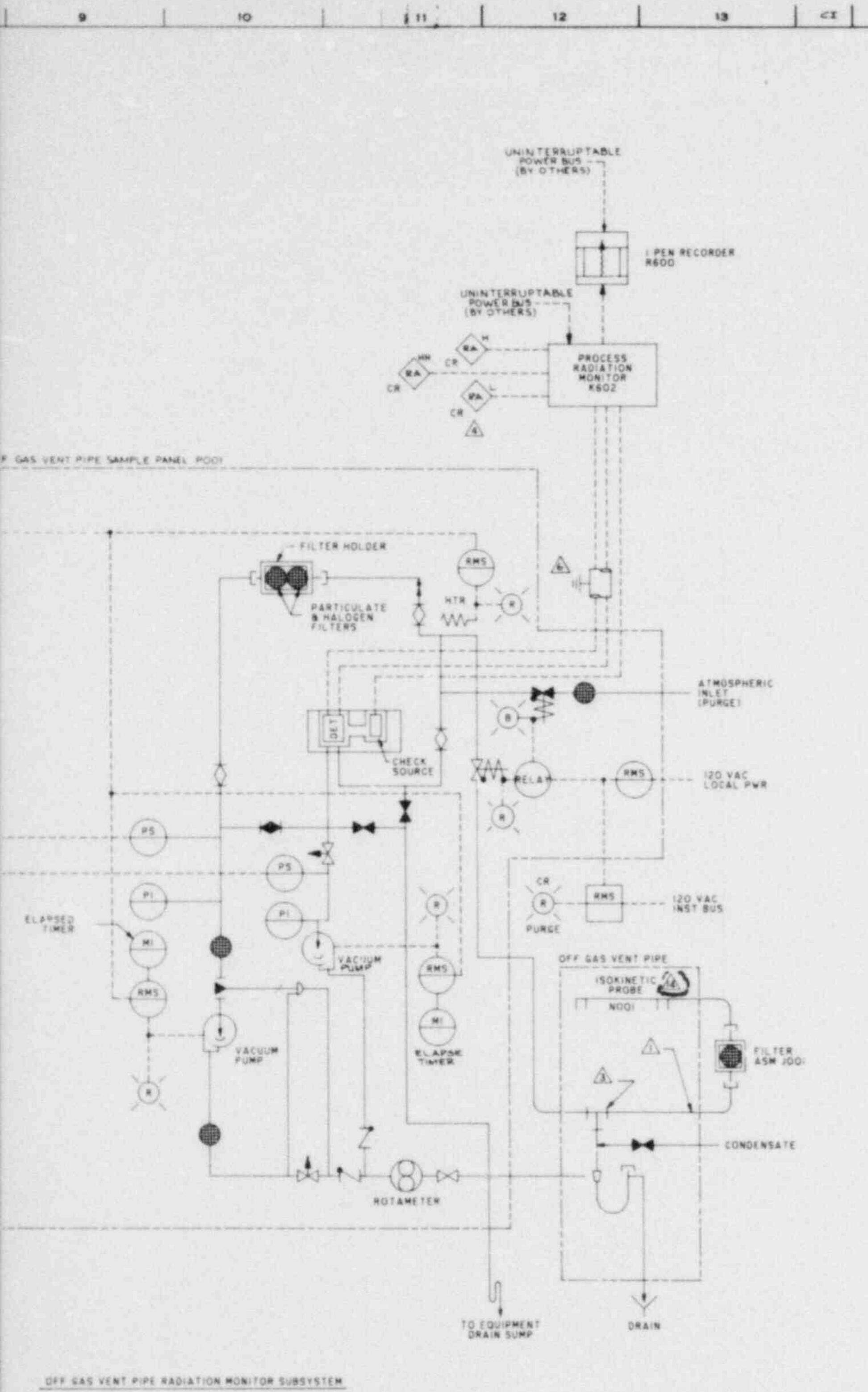
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 1

FIGURE 7.3-16a



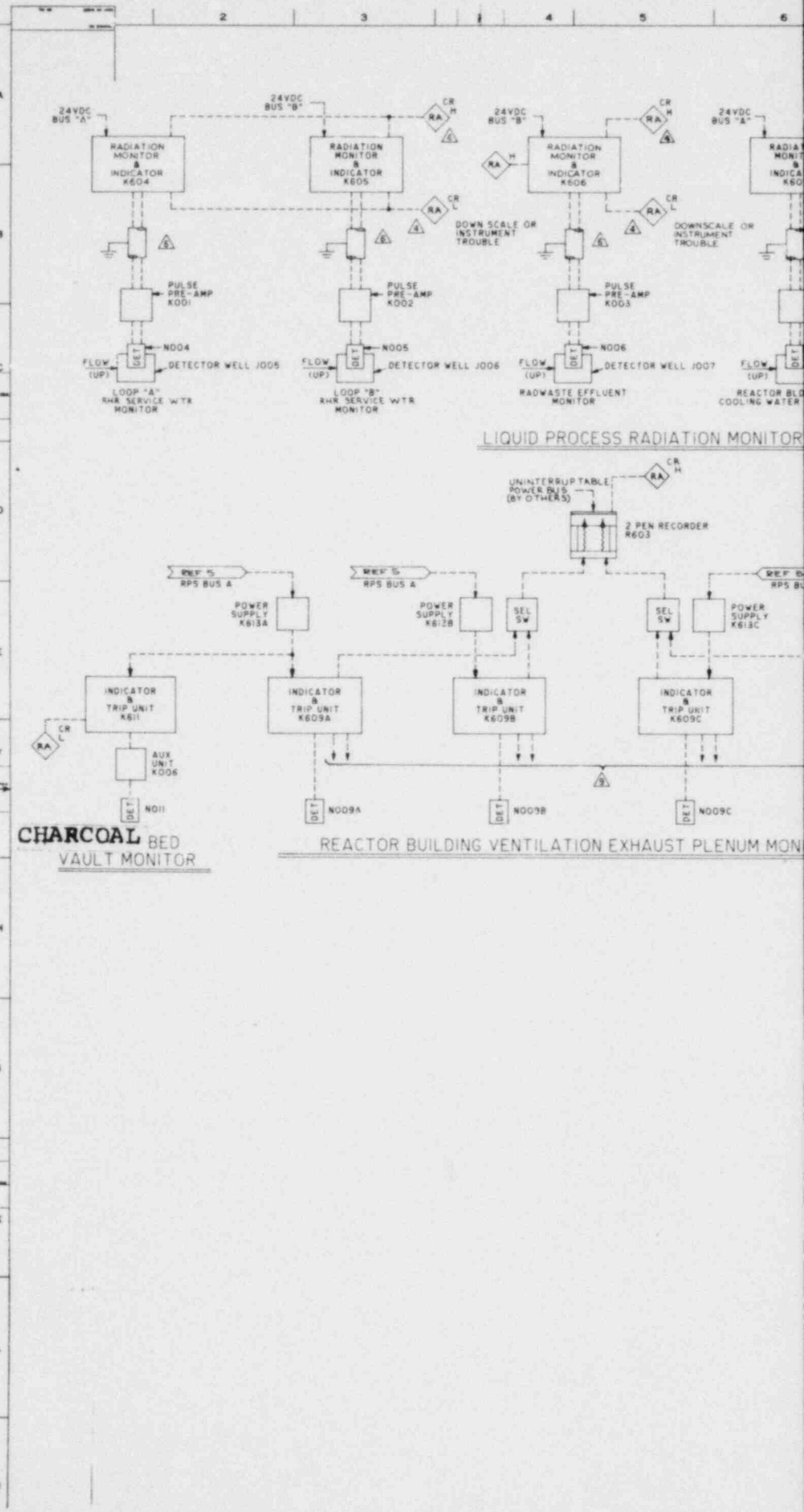
OFF GAS RADIATION MONITORS & SAMPLERS SUBSYSTEM



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

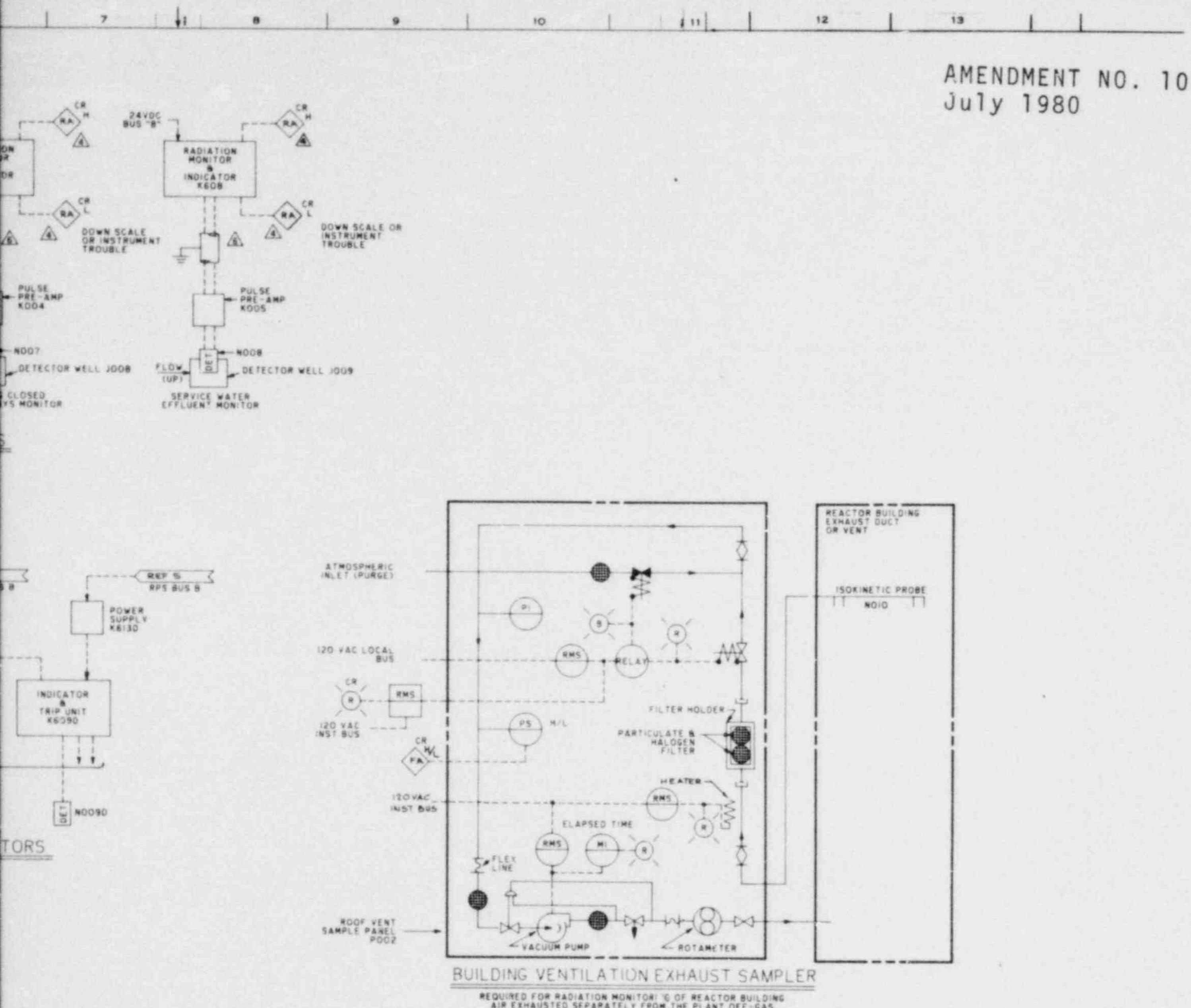
PROCESS RADIATION MONITORING SYSTEM
(RECOMB/CHARCOAL BED) IED SHEET 2

FIGURE 7.3-16b

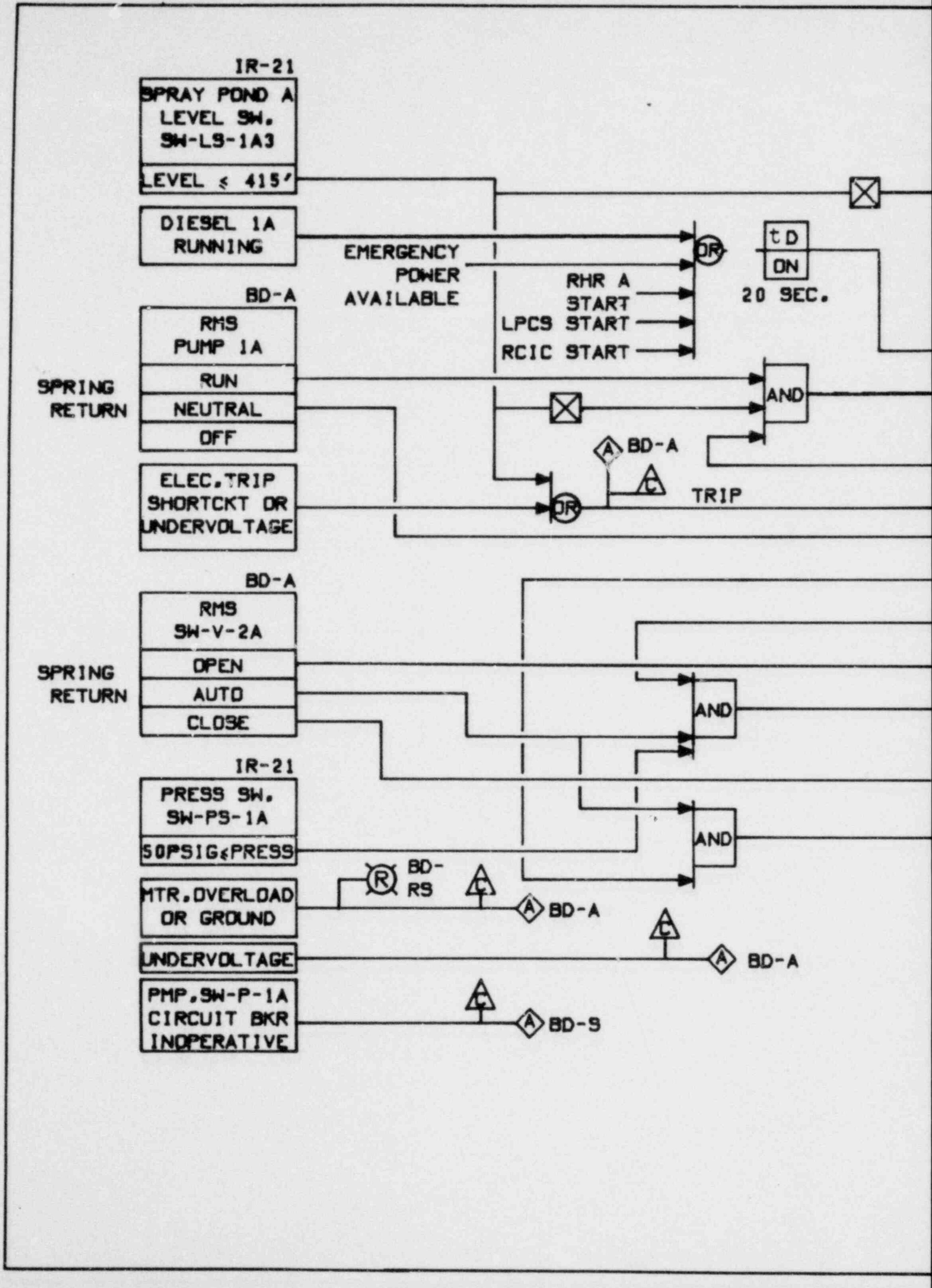


2
3
4
5
6
A
B
C
D
E
F
H
J
K
L
M
24VDC BUS "A"
24VDC BUS "B"
RADIATION MONITOR & INDICATOR K604
RADIATION MONITOR & INDICATOR K605
RADIATION MONITOR & INDICATOR K606
RA
CR H
CR L
DOWN SCALE OR INSTRUMENT TROUBLE
DOWNSCALE OR INSTRUMENT TROUBLE
PULSE PRE-AMP K001
PULSE PRE-AMP K002
PULSE PRE-AMP K003
DET
N004
N005
N006
FLOW (UP)
DETECTOR WELL J005
DETECTOR WELL J006
DETECTOR WELL J007
LOOP "A" RHR SERVICE WTR MONITOR
LOOP "B" RHR SERVICE WTR MONITOR
RADWASTE EFFLUENT MONITOR
LIQUID PROCESS RADIATION MONITOR
UNINTERRUPTABLE POWER BUS (BY OTHERS)
2 PEN RECORDER R603
REF 5 RPS BUS A
POWER SUPPLY K613A
POWER SUPPLY K612B
POWER SUPPLY K613C
SEL SW
INDICATOR & TRIP UNIT K611
INDICATOR & TRIP UNIT K609A
INDICATOR & TRIP UNIT K609B
INDICATOR & TRIP UNIT K609C
AUX UNIT K006
N011
N009A
N009B
N009C
CHARCOAL BED VAULT MONITOR
REACTOR BUILDING VENTILATION EXHAUST PLENUM MON

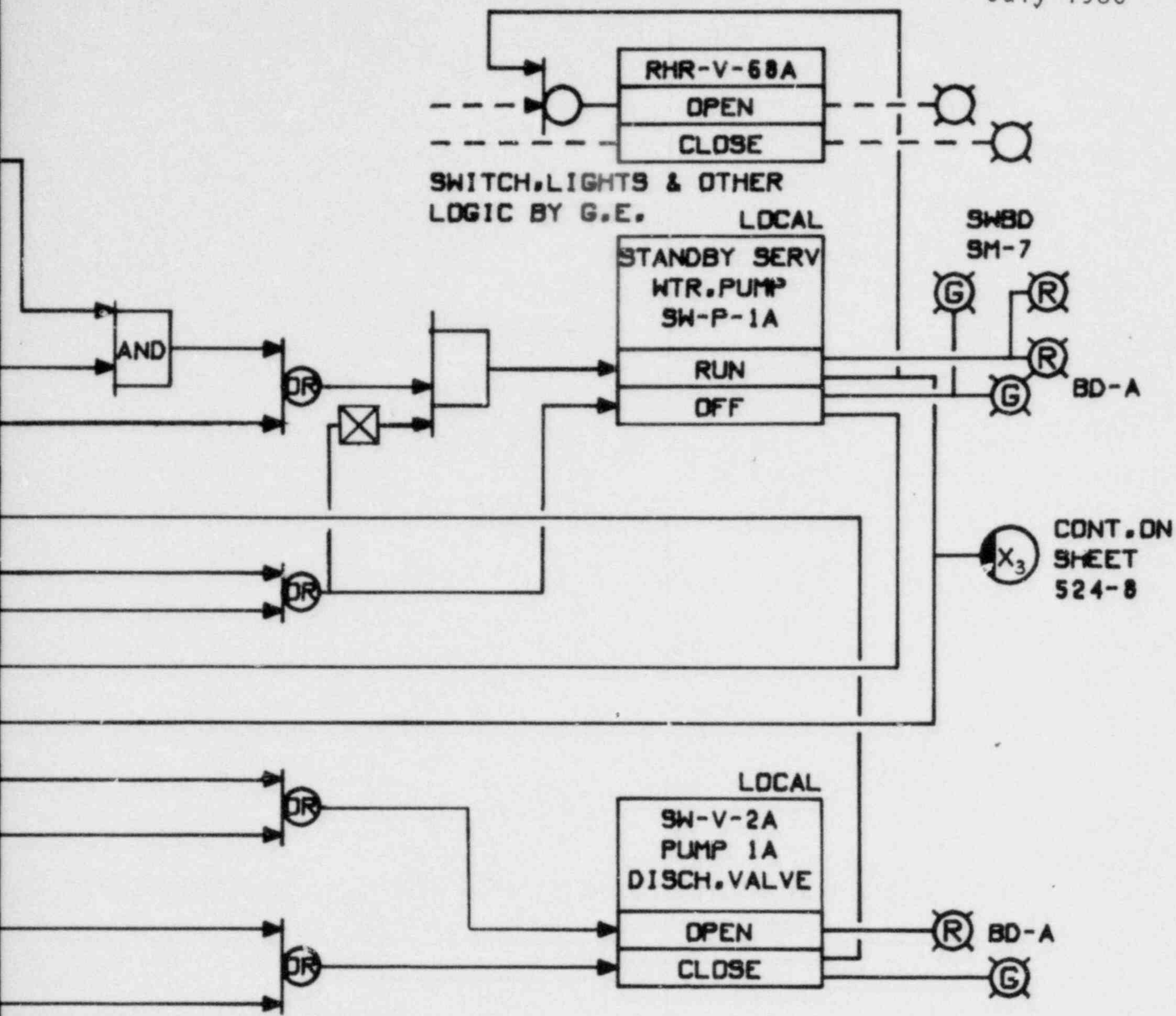
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 3	FIGURE 7.3-16c


IR-21
SPRAY POND A LEVEL SW. SW-LS-1A3
LEVEL ≤ 415'
DIESEL 1A RUNNING
EMERGENCY POWER AVAILABLE
RHR A START
LPCS START
RCIC START
OR
tD ON
20 SEC.
BD-A
RMS PUMP 1A
SPRING RETURN
RUN
NEUTRAL
OFF
AND
A BD-A
C
ELEC.TRIP SHORTCKT OR UNDERVOLTAGE
OR
TRIP
BD-A
RMS SW-V-2A
SPRING RETURN
OPEN
AUTO
CLOSE
AND
IR-21
PRESS SW. SW-PS-1A
50PSIG≤PRESS
AND
R BD-RS
MTR.OVERLOAD OR GROUND
C
A BD-A
UNDERVOLTAGE
C
A BD-A
PMP.SW-P-1A CIRCUIT BKR INOPERATIVE
C
A BD-S

AMENDMENT NO. 10
July 1980

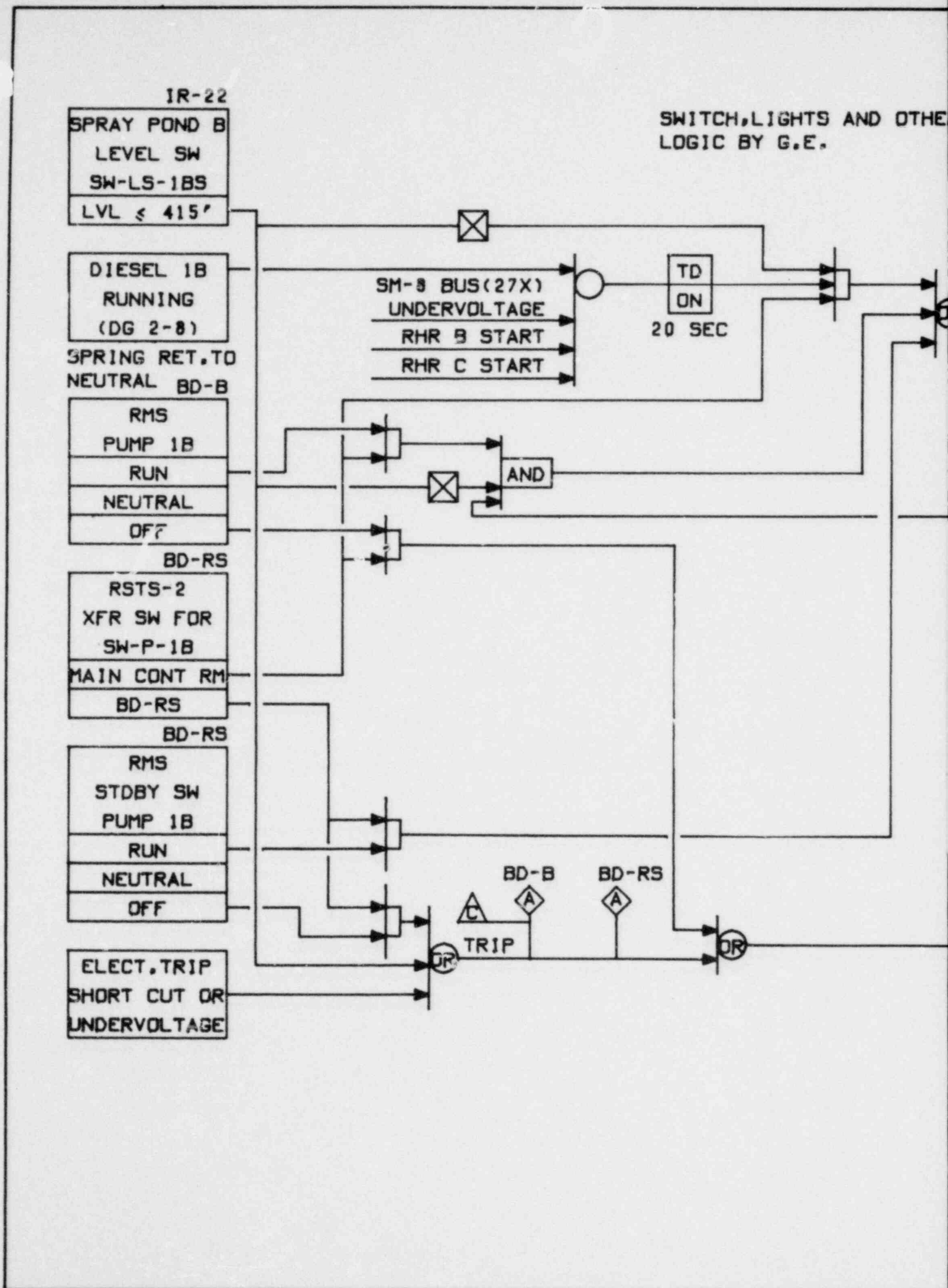


I & C DWG. NO. M-620

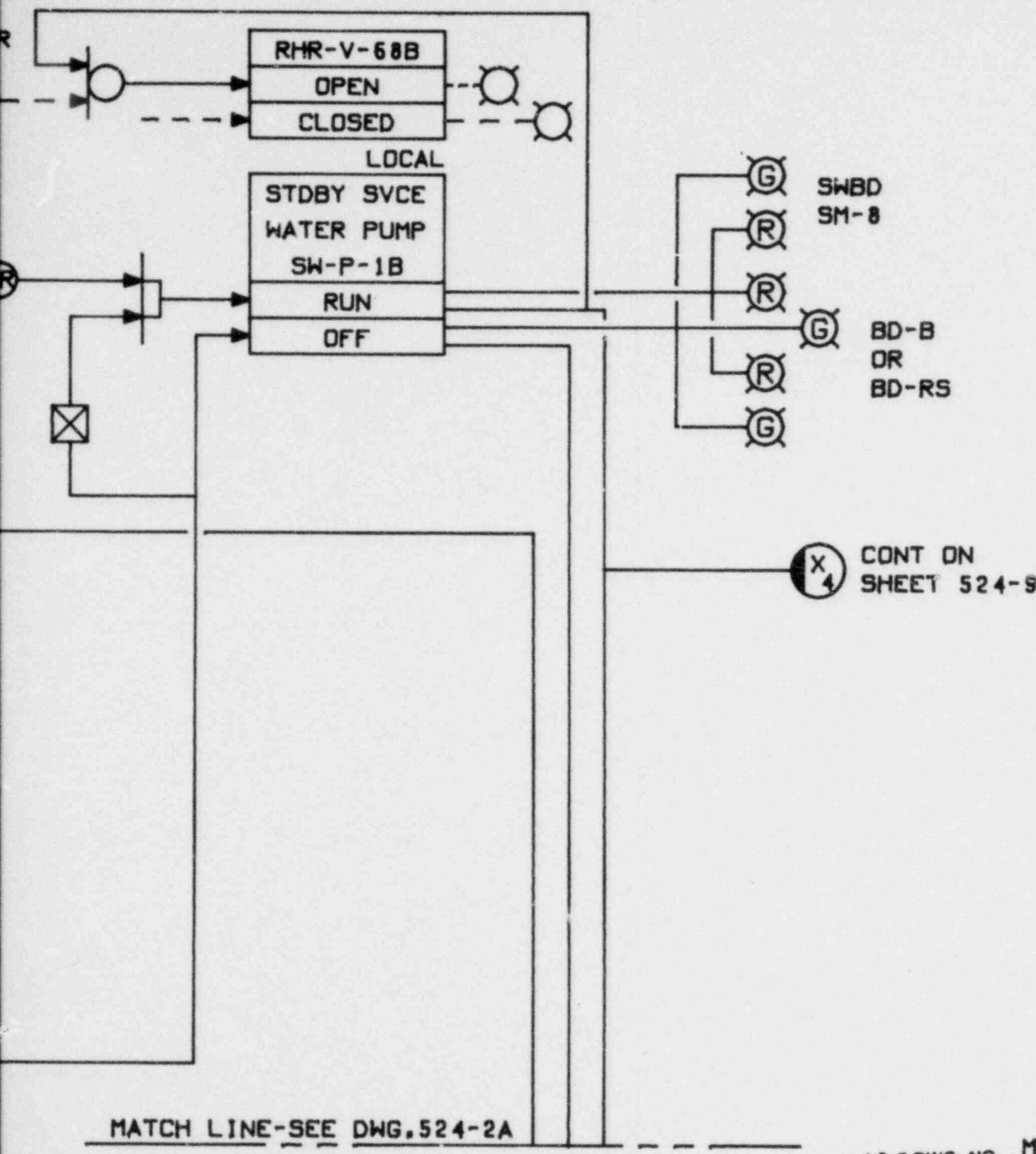
SHEET 524-1

STANDBY SERVICE WATER PUMP 1A SHT. 1
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17a


IR-22
SPRAY POND B
LEVEL SW
SW-LS-1BS
LVL ≤ 415'
SWITCH,LIGHTS AND OTHE
LOGIC BY G.E.
DIESEL 1B
RUNNING
(DG 2-8)
SM-8 BUS(27X)
UNDERVOLTAGE
RHR B START
RHR C START
TD
ON
20 SEC
SPRING RET.TO
NEUTRAL
BD-B
RMS
PUMP 1B
RUN
NEUTRAL
OFF
AND
BD-RS
RSTS-2
XFR SW FOR
SW-P-1B
MAIN CONT RM
BD-RS
BD-RS
RMS
STDBY SW
PUMP 1B
RUN
NEUTRAL
OFF
BD-B
BD-RS
C
A
A
TRIP
OR
OR
ELECT.TRIP
SHORT CUT OR
UNDERVOLTAGE

AMENDMENT NO. 10
July 1980

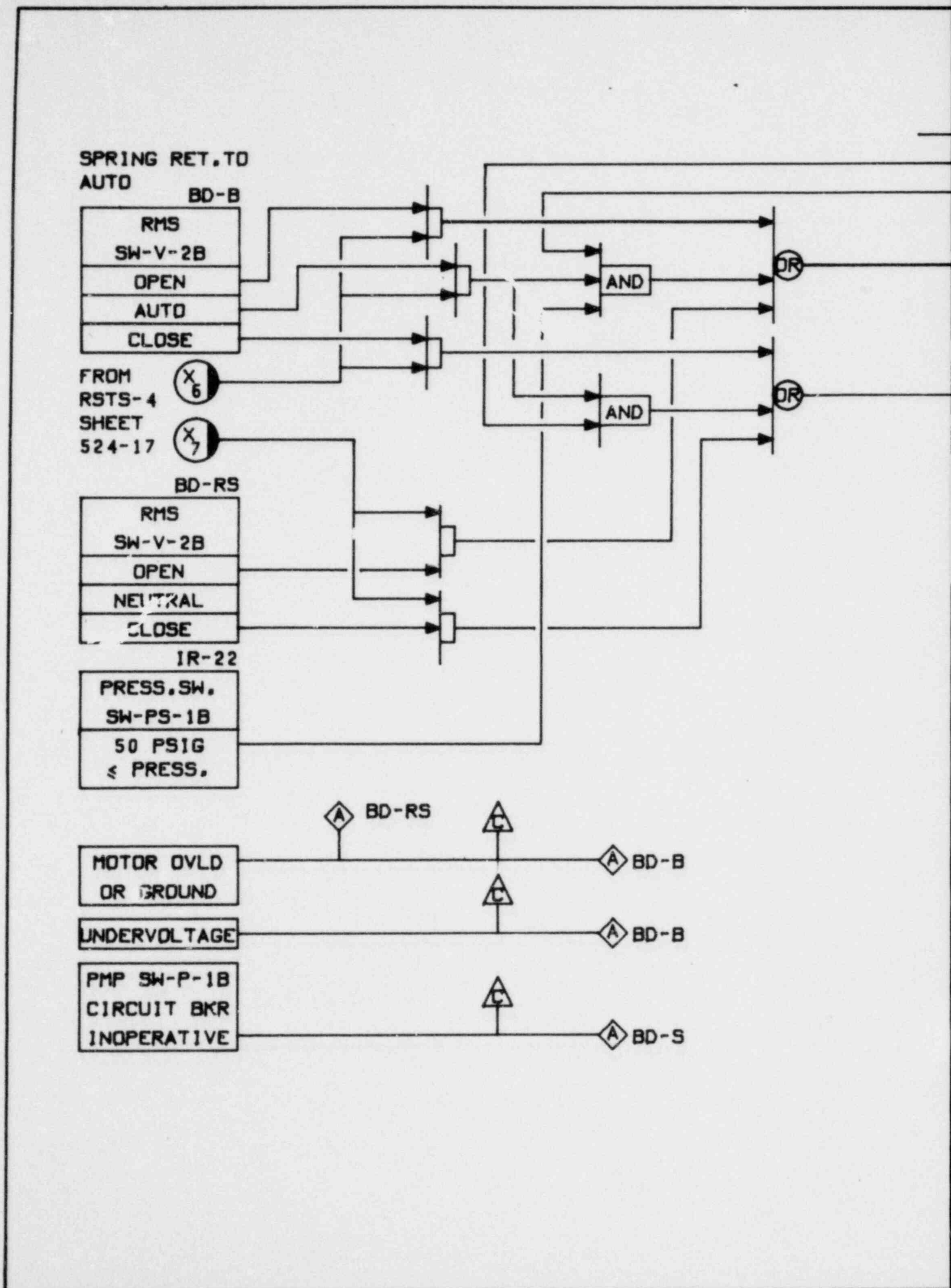


I & C DWG. NO. M-620

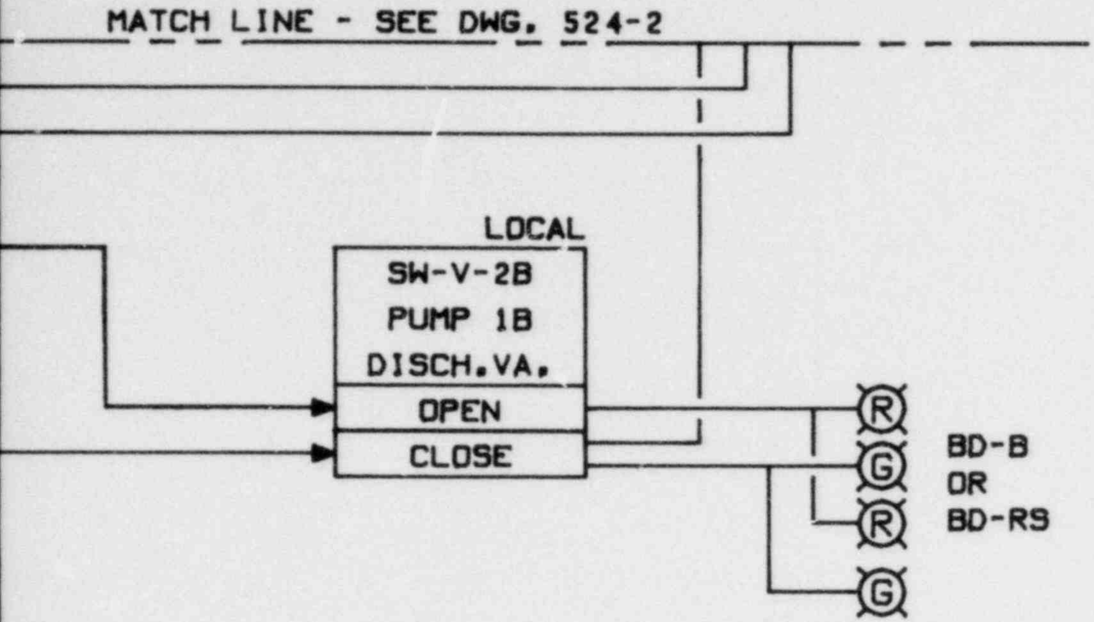
SHEET 524-2

STANDBY SERVICE WATER PUMP 1B
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17b



SPRING RET. TO AUTO
BD-B
RMS SW-V-2B
OPEN
AUTO
CLOSE
FROM RSTS-4 SHEET 524-17
X 6
X 7
AND
AND
OR
OR
BD-RS
RMS SW-V-2B
OPEN
NEUTRAL
CLOSE
IR-22
PRESS. SW. SW-PS-1B
50 PSIG ≤ PRESS.
A BD-RS
C
MOTOR OVLD OR GROUND
A BD-B
UNDERVOLTAGE
A BD-B
PMP SW-P-1B CIRCUIT BKR INOPERATIVE
A BD-S



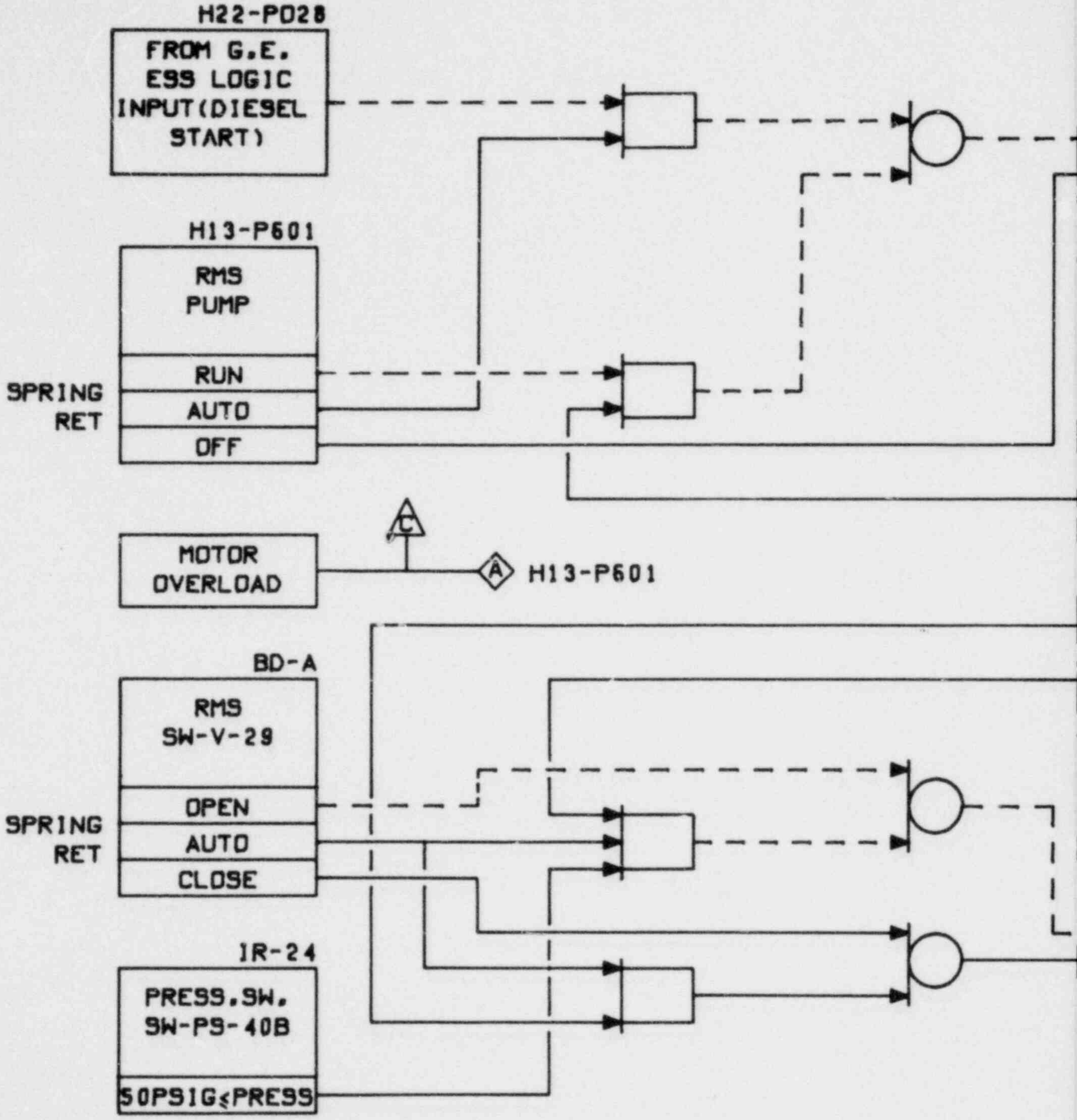
REF.DWGS.
STANDBY SERVICE WATER SYS.
FLOW DIAG.M-524

I & C DWG. NO. M-620

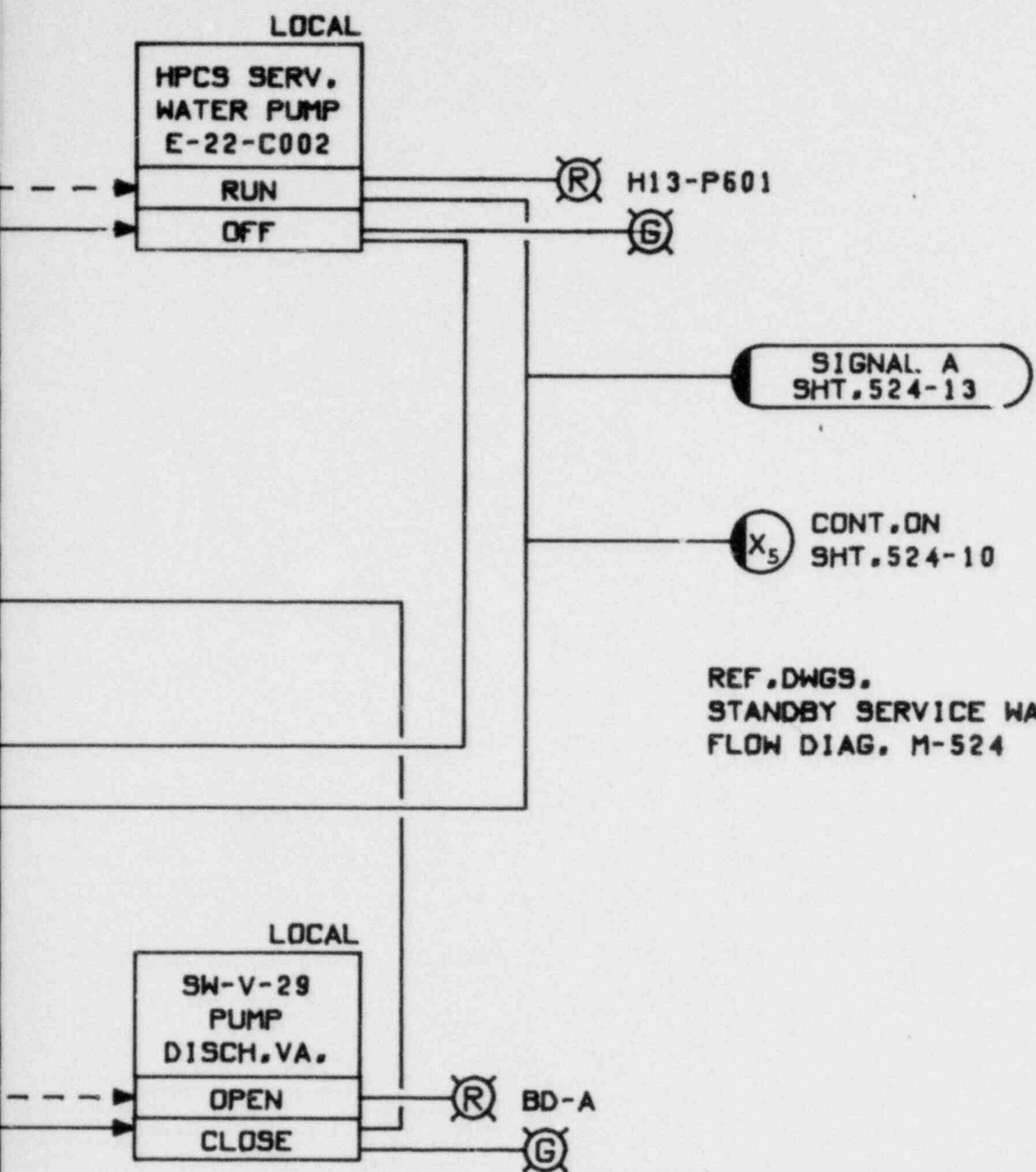
SHEET 524-2A

STANDBY SERVICE WATER PUMP 1B
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3- 17c


H22-P028
FROM G.E.
ESS LOGIC
INPUT(DIESEL
START)
H13-P601
RMS
PUMP
RUN
AUTO
OFF
SPRING
RET
C
MOTOR
OVERLOAD
A
H13-P601
BD-A
RMS
SW-V-29
OPEN
AUTO
CLOSE
SPRING
RET
IR-24
PRESS.SW.
SW-PS-40B
50PSIG≤PRESS

AMENDMENT NO. 10
July 1980



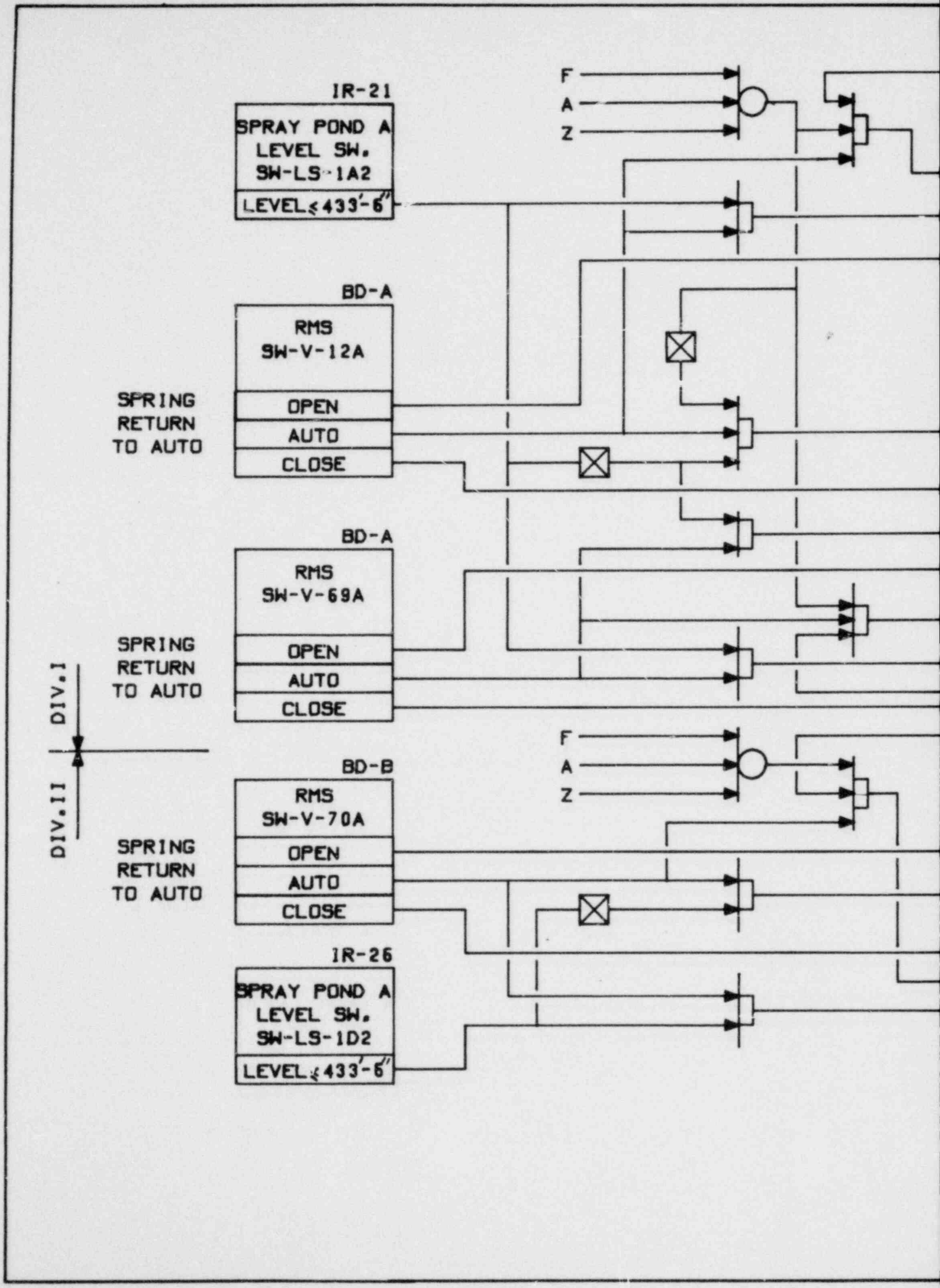
REF.DWGS.
STANDBY SERVICE WATER SYSTEM
FLOW DIAG. M-524

I & C DWG. NO. M-620

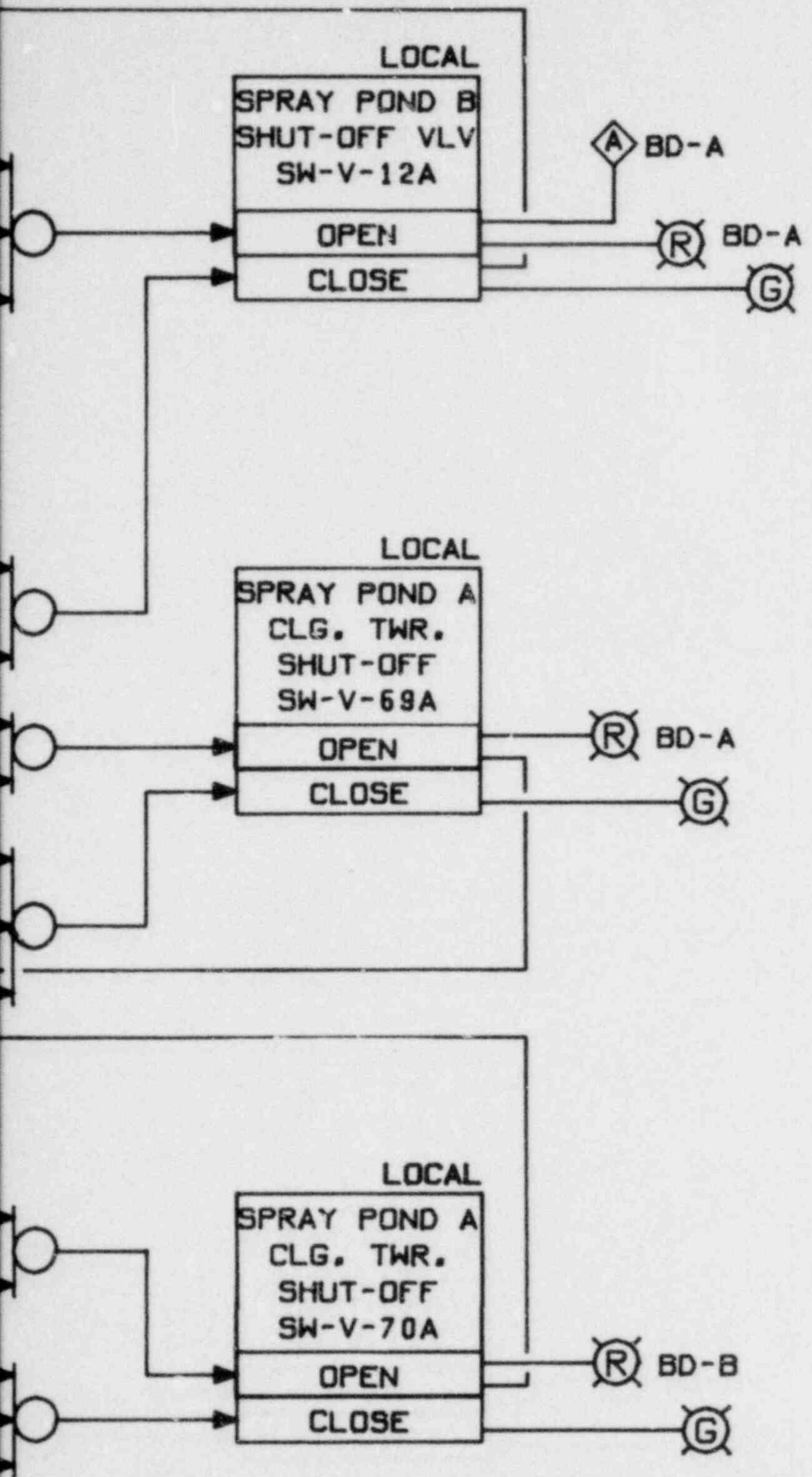
SHEET 524-3

DIV. III
HPCS SERVICE WATER PUMP E-22-C002
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3- 17d


IR-21
SPRAY POND A
LEVEL SW.
SW-LS-1A2
LEVEL ≤ 433'-6"
F
A
Z
BD-A
RMS
SW-V-12A
SPRING
RETURN
TO AUTO
OPEN
AUTO
CLOSE
BD-A
RMS
SW-V-69A
SPRING
RETURN
TO AUTO
OPEN
AUTO
CLOSE
DIV.I
DIV.II
F
A
Z
BD-B
RMS
SW-V-70A
SPRING
RETURN
TO AUTO
OPEN
AUTO
CLOSE
IR-26
SPRAY POND A
LEVEL SW.
SW-LS-1D2
LEVEL ≤ 433'-6"

AMENDMENT NO. 10
July 1980



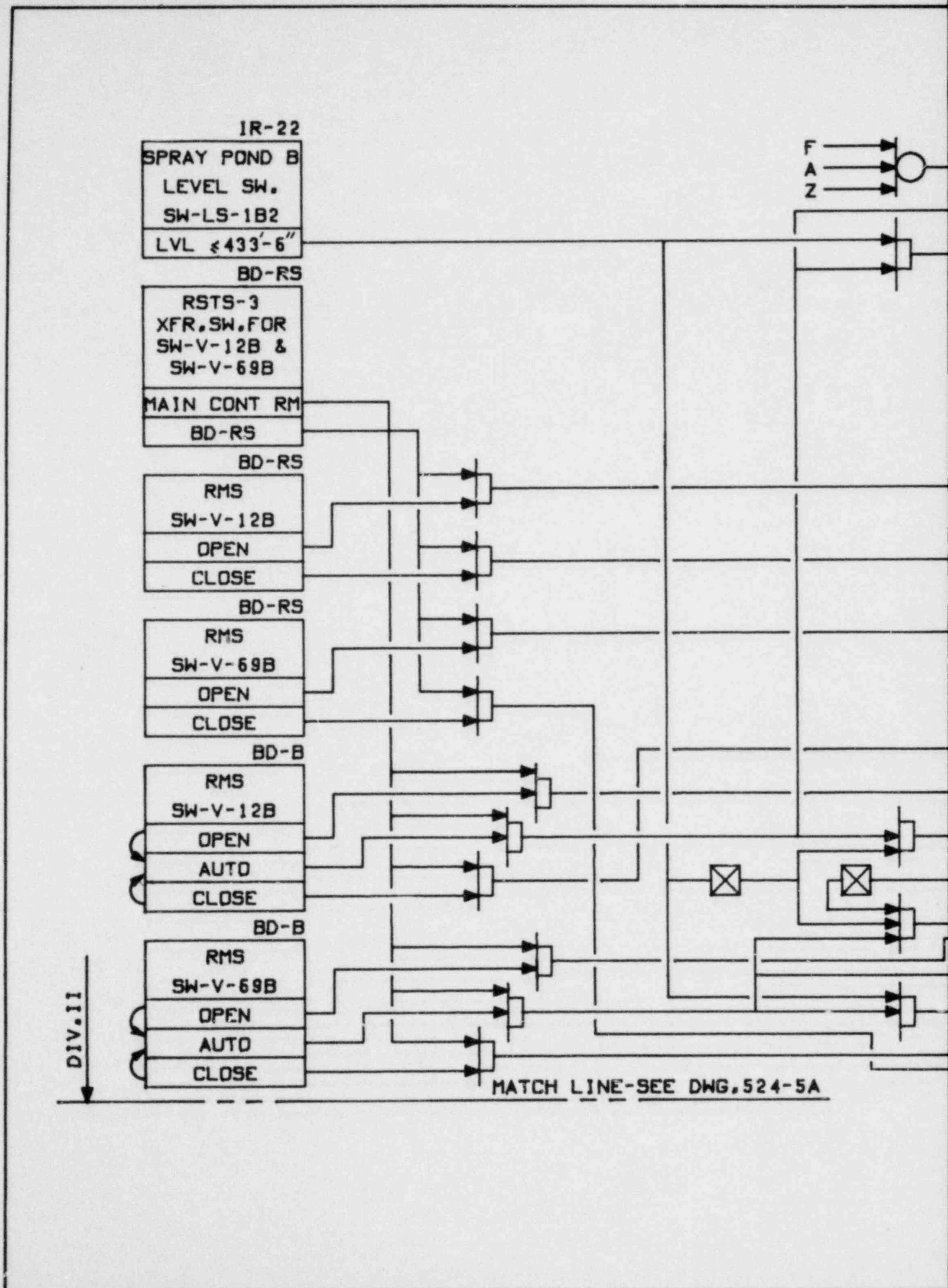
REF.DWG.
STANDBY SERV.WTR.FL.D. M-524

I & C DWG. NO. M-620

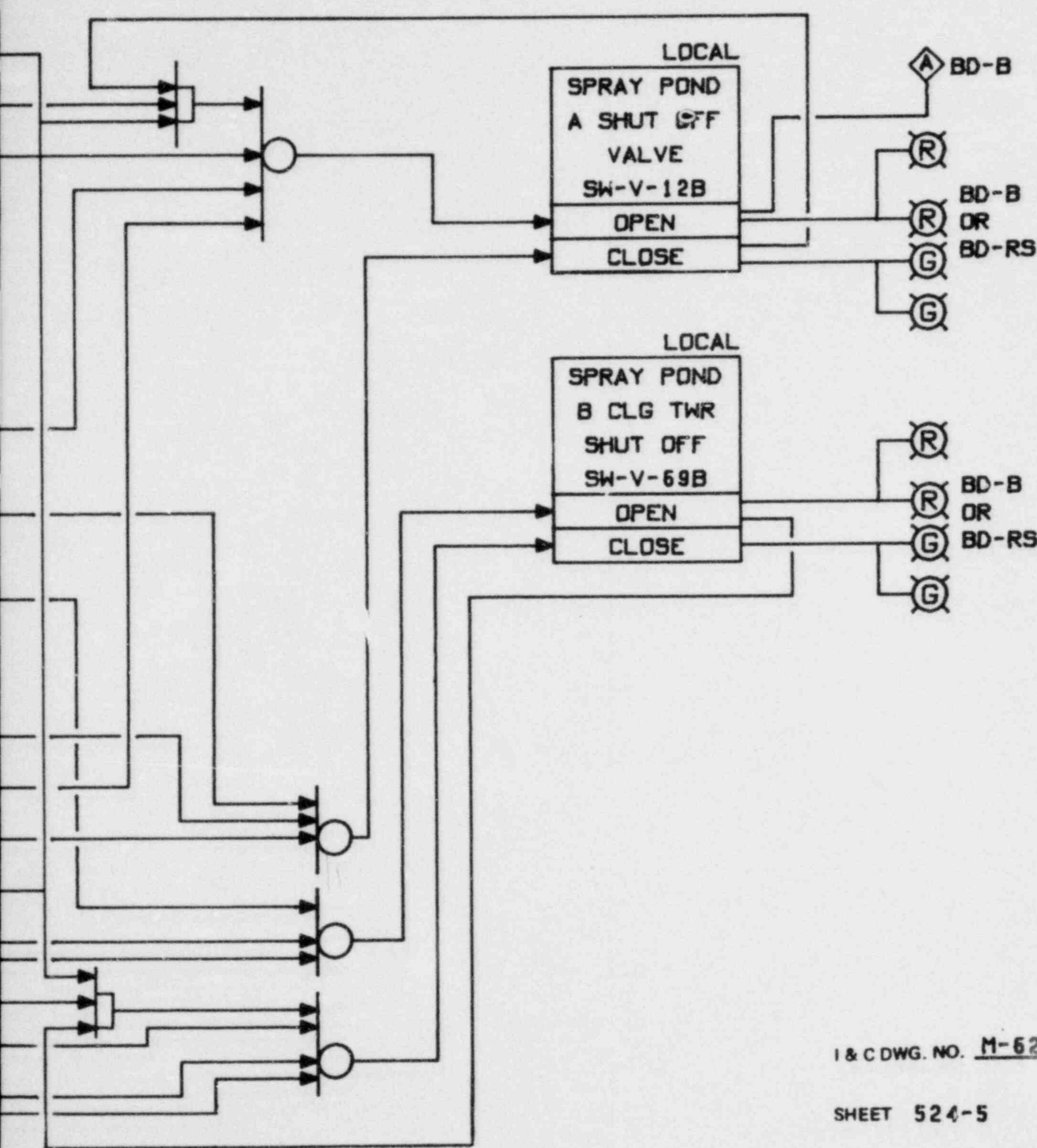
SHEET 524-4

SPRAY POND A LEVEL
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17e


IR-22
SPRAY POND B
LEVEL SW.
SW-LS-1B2
LVL ≤433'-6"
F
A
Z
BD-RS
RSTS-3
XFR.SW.FOR
SW-V-12B &
SW-V-69B
MAIN CONT RM
BD-RS
BD-RS
RMS
SW-V-12B
OPEN
CLOSE
BD-RS
RMS
SW-V-69B
OPEN
CLOSE
BD-B
RMS
SW-V-12B
OPEN
AUTO
CLOSE
BD-B
RMS
SW-V-69B
OPEN
AUTO
CLOSE
DIV.II
MATCH LINE-SEE DWG.524-5A

AMENDMENT NO. 10
July 1980

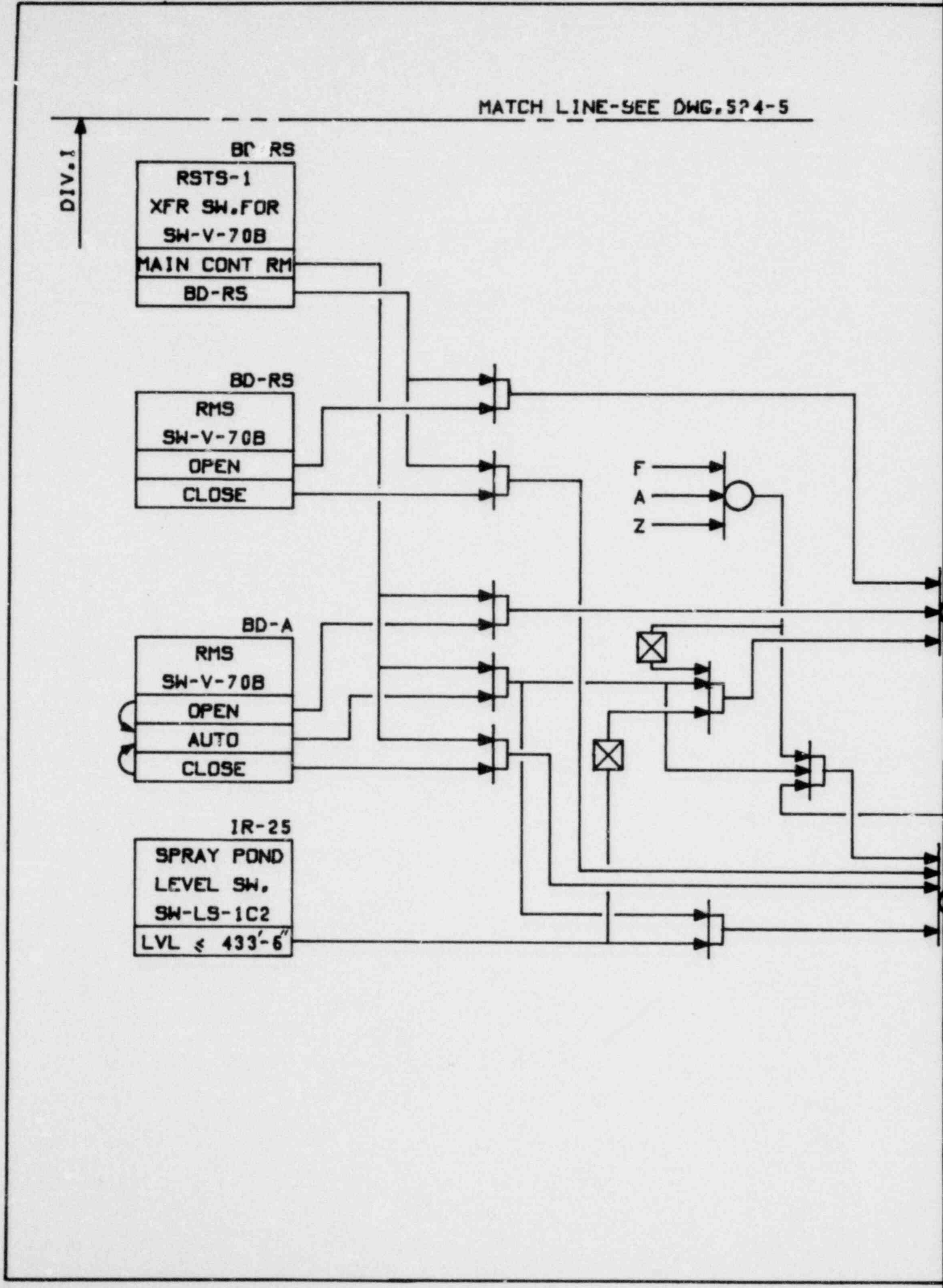


I & C DWG. NO. M-620

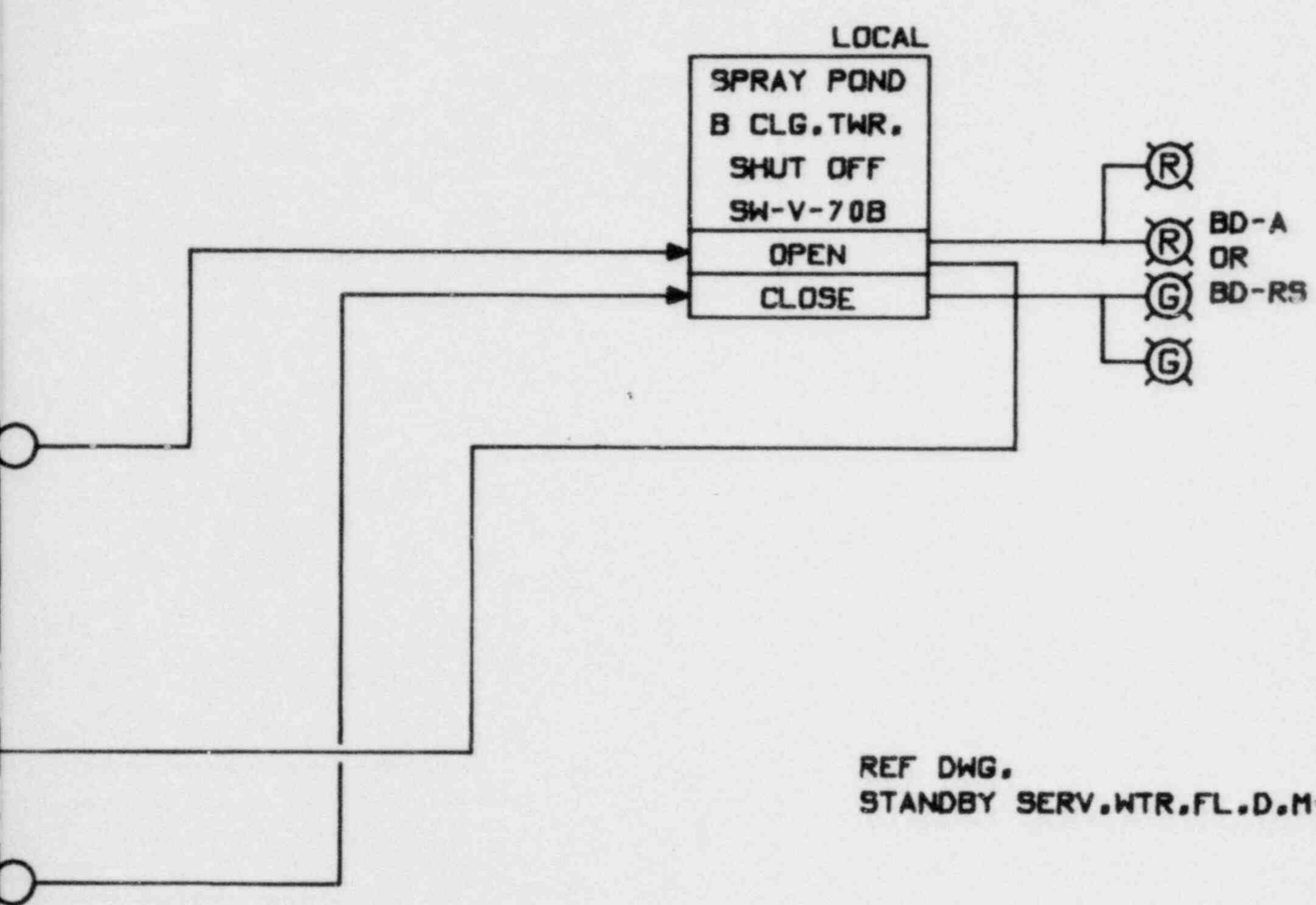
SHEET 524-5

SPRAY POND B LEVEL
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3- 17f



MATCH LINE-SEE DWG.5?4-5
DIV.I
BC RS
RSTS-1
XFR SW.FOR
SW-V-70B
MAIN CONT RM
BD-RS
BD-RS
RMS
SW-V-70B
OPEN
CLOSE
F
A
Z
BD-A
RMS
SW-V-70B
OPEN
AUTO
CLOSE
IR-25
SPRAY POND
LEVEL SW.
SW-LS-1C2
LVL ≤ 433'-6"



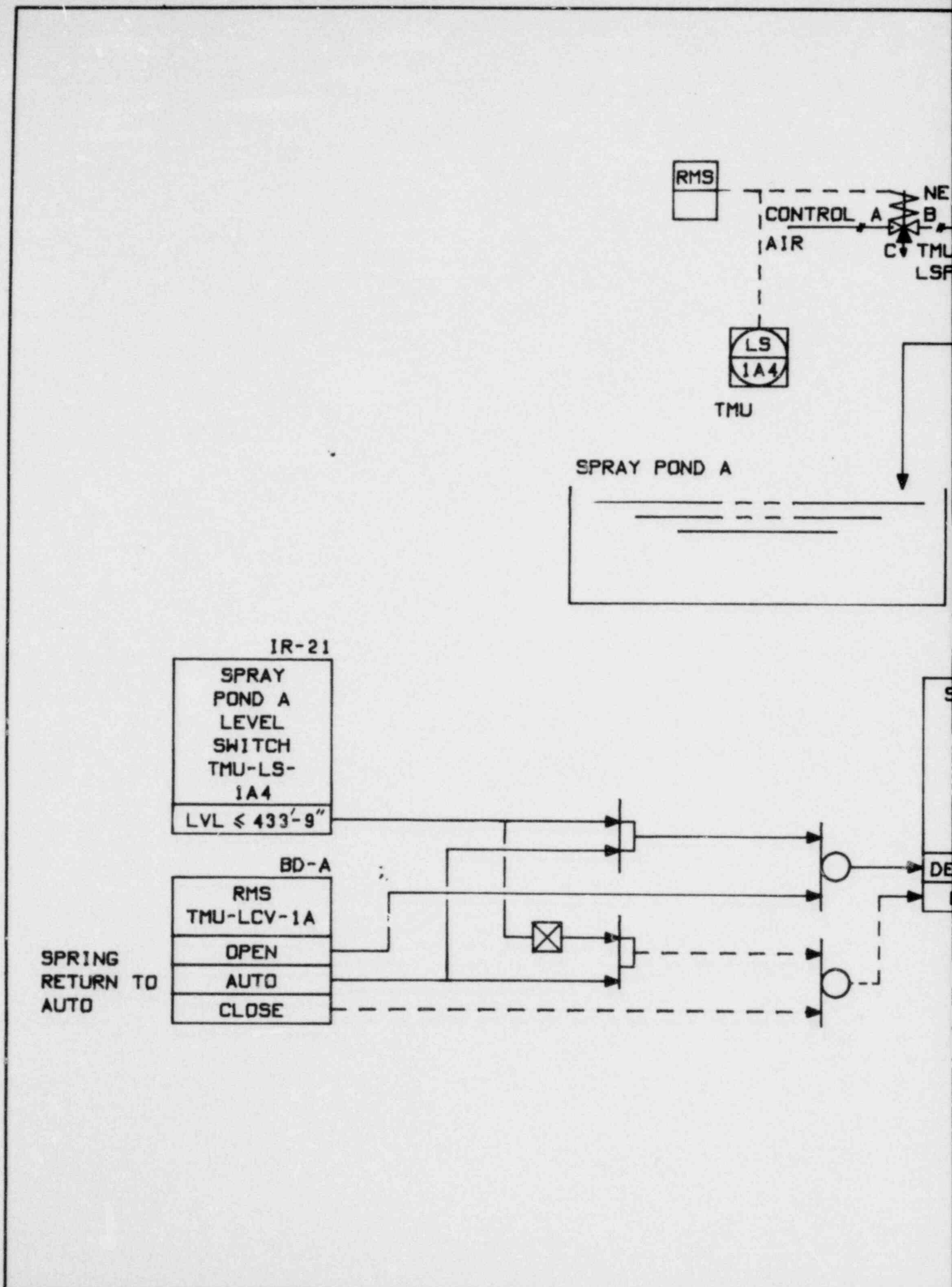
REF DWG.
STANDBY SERV.WTR.FL.D.M-524

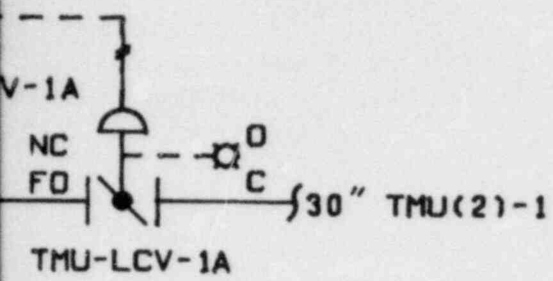
I & C DWG. NO. M-620

SHEET 524-5A

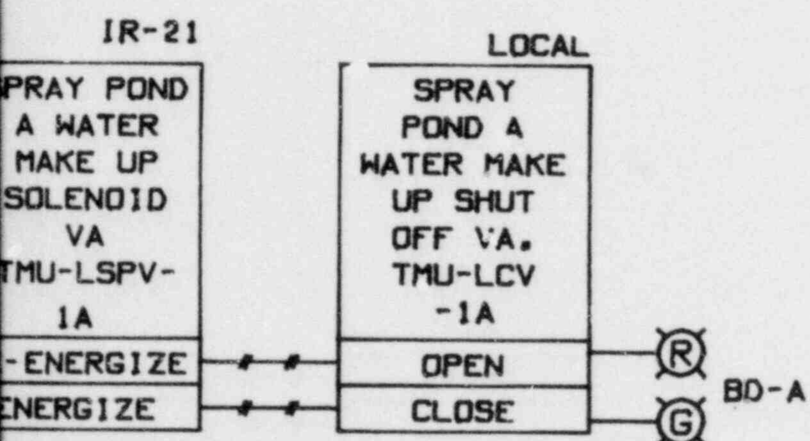
SPRAY POND B LEVEL
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3- 17g


RMS
CONTROL
AIR
A
B
C
NE
TMU
LSF
LS
1A4
TMU
SPRAY POND A
IR-21
SPRAY
POND A
LEVEL
SWITCH
TMU-LS-
1A4
LVL ≤ 433'-9"
BD-A
RMS
TMU-LCV-1A
OPEN
AUTO
CLOSE
SPRING
RETURN TO
AUTO
DE



REF DWG.
STANDBY SERVICE WATER FL.-D M-524



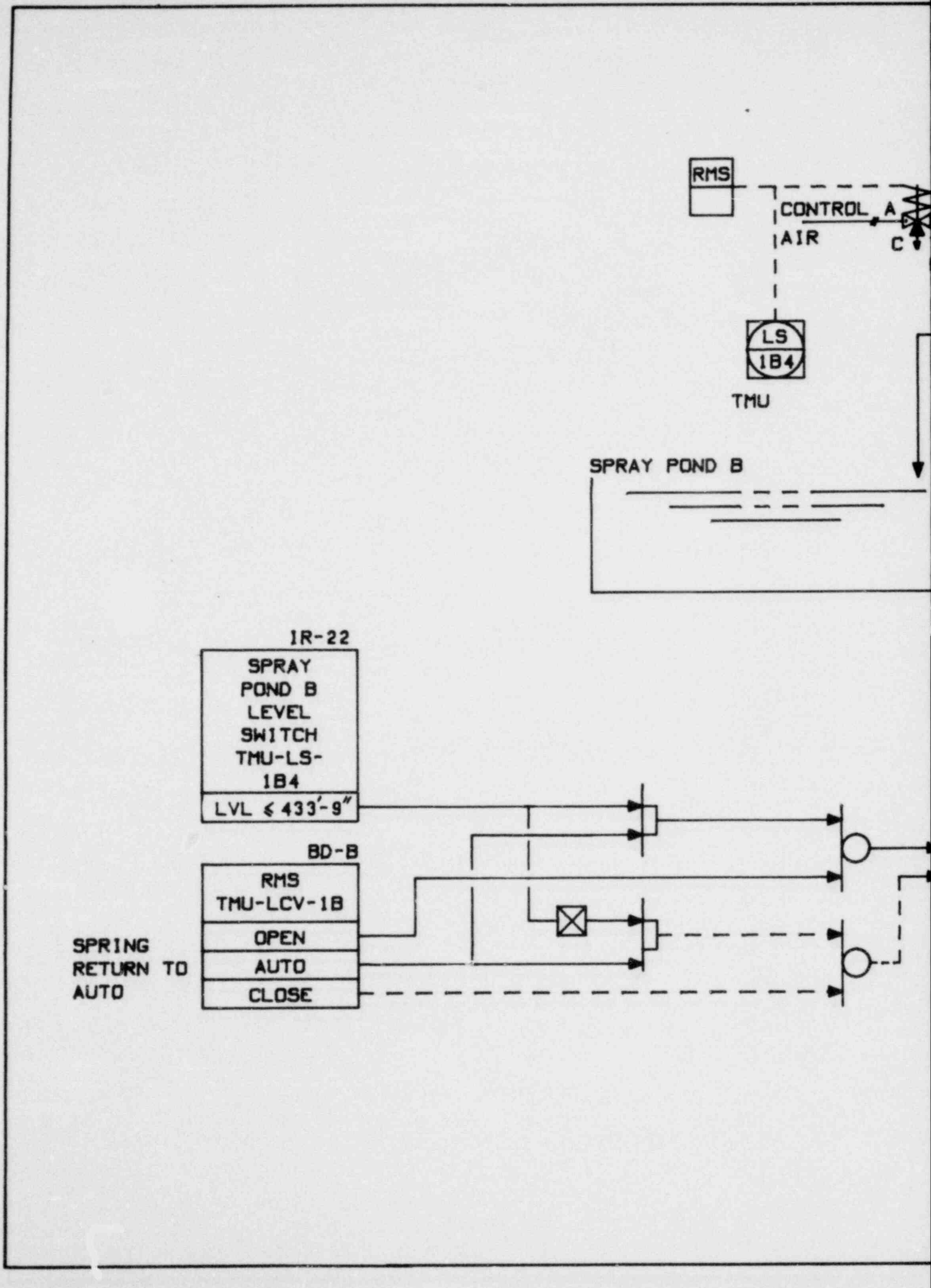
DIV. I

I & C DWG. NO. M-620

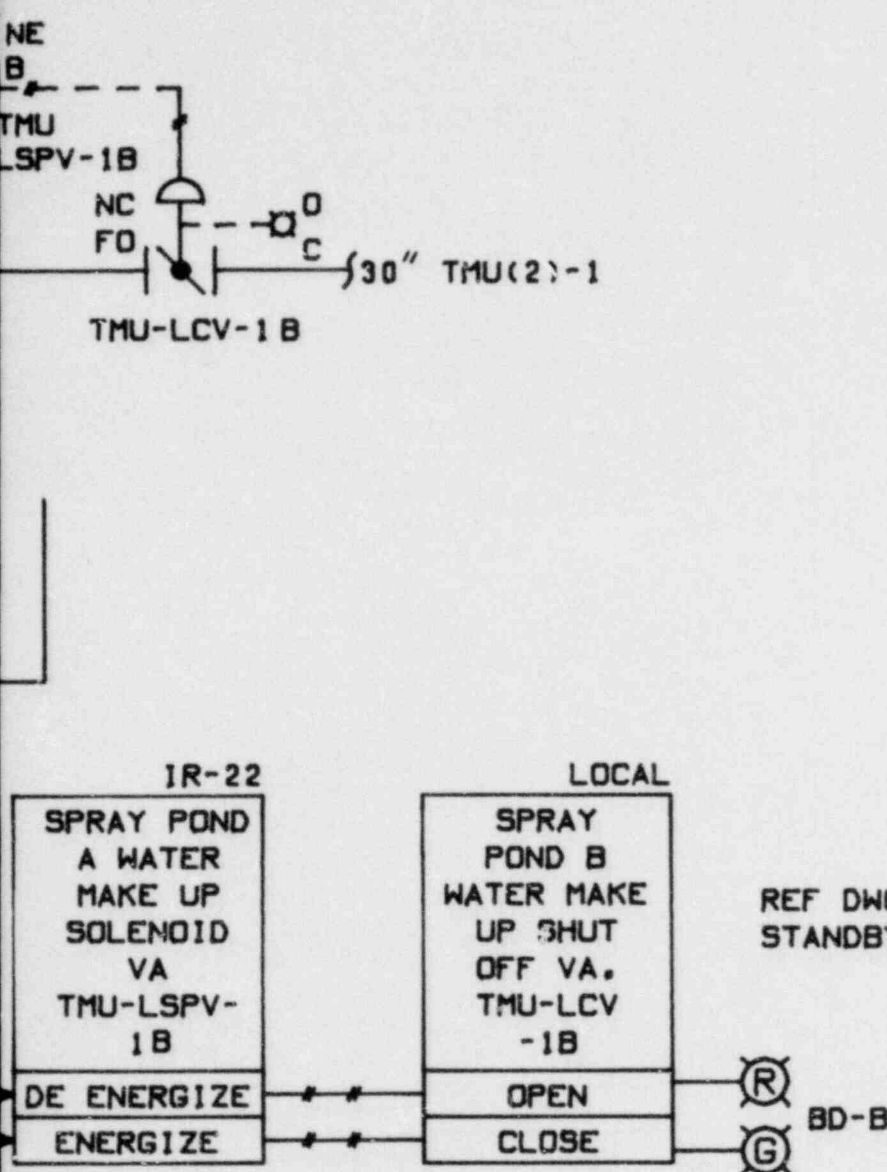
SHEET 524-6

SPRAY POND A WATER MAKE-UP
CONTROL SCHEMATIC AND LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17h



RMS
CONTROL
AIR
A
C
LS
1B4
TMU
SPRAY POND B
1R-22
SPRAY POND B LEVEL SWITCH TMU-LS-1B4
LVL ≤ 433'-9"
BD-B
RMS TMU-LCV-1B
OPEN
AUTO
CLOSE
SPRING RETURN TO AUTO



REF DWG.
STANDBY SERVICE WATER FL.D. M-524

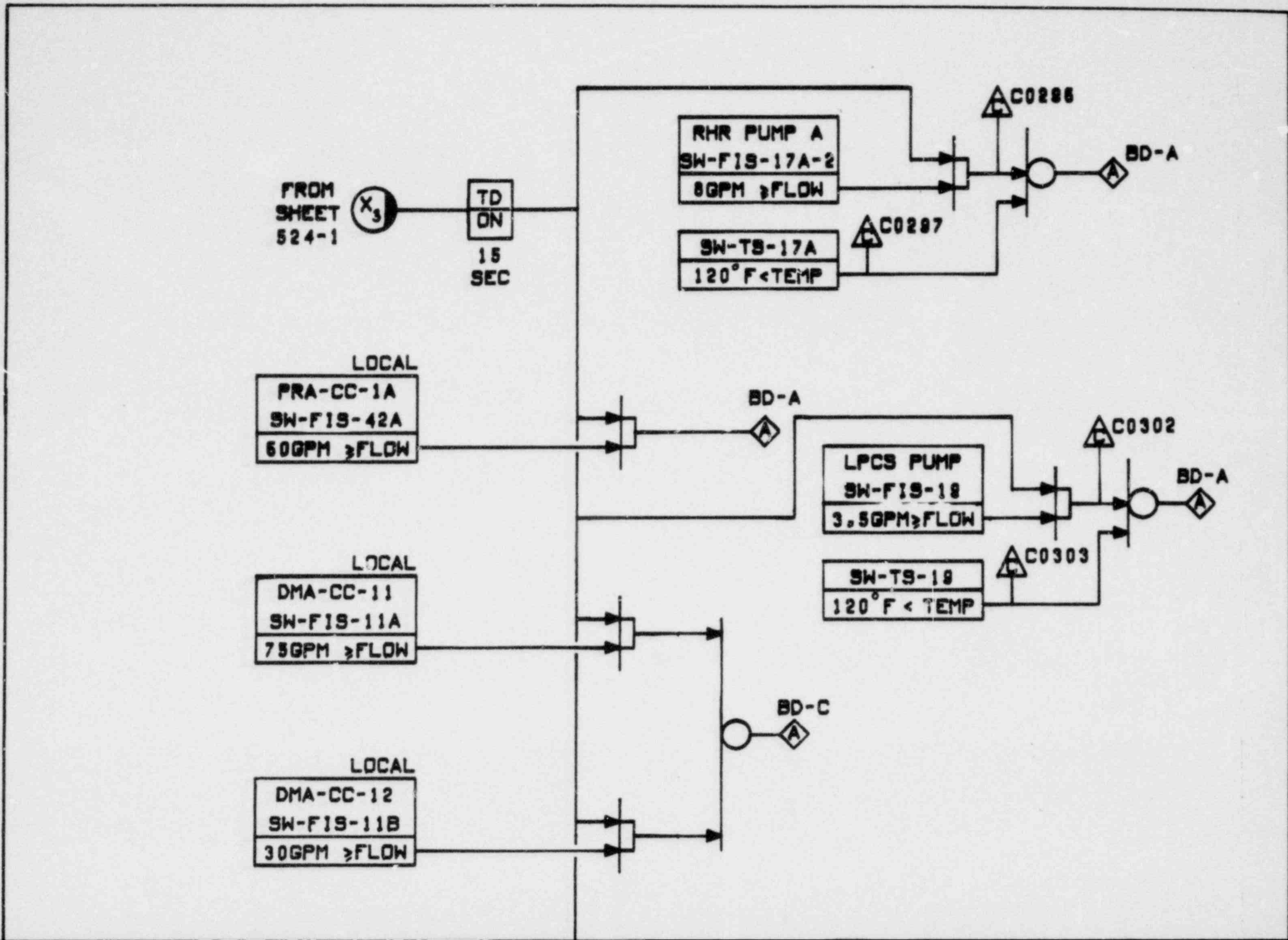
DIV. II

I & C DWG. NO. M-620

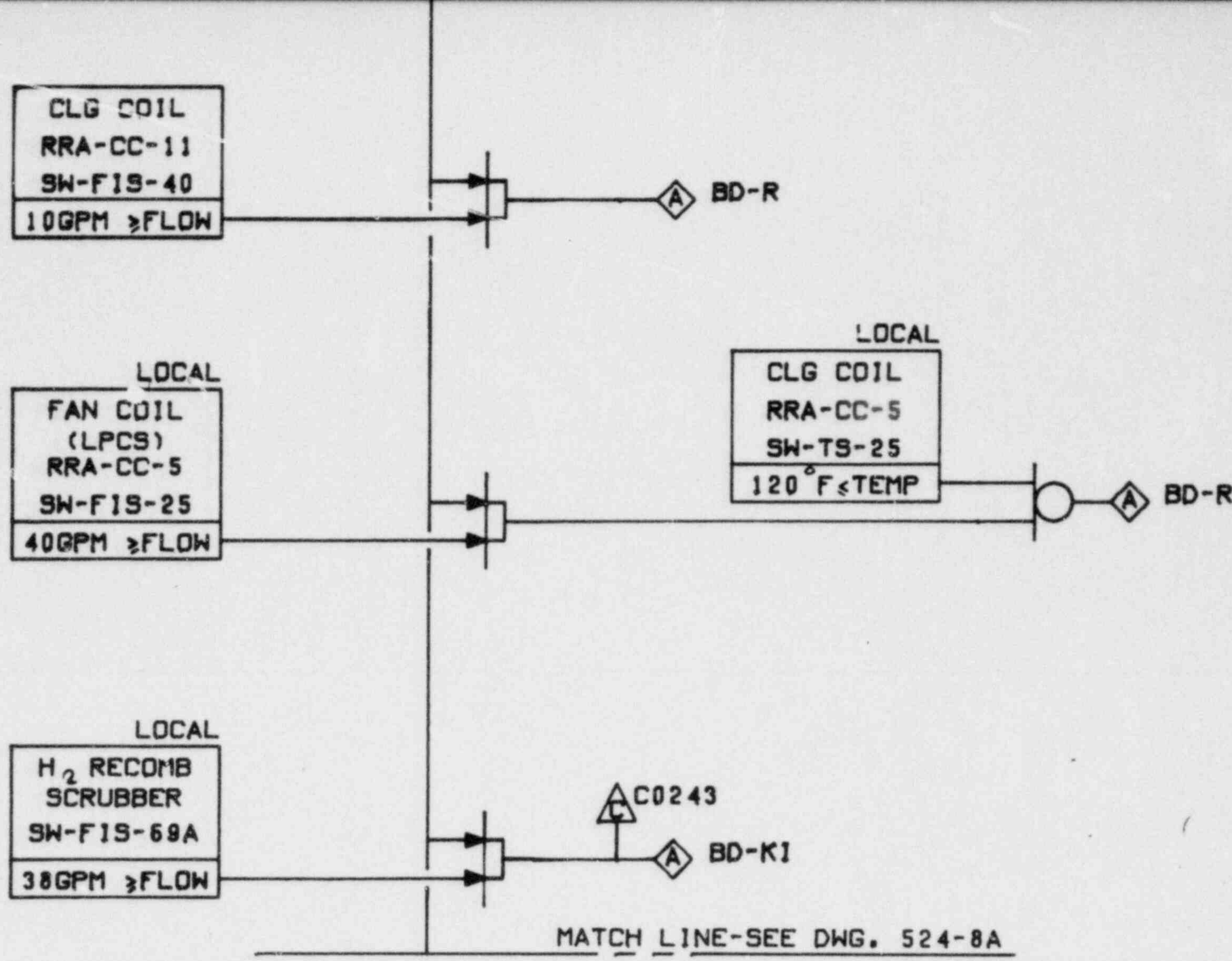
SHEET 524-7

SPRAY POND B WATER MAKE-UP
CONTROL SCHEMATIC AND LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17i



FROM SHEET 524-1
X_3
TD ON
15 SEC
RHR PUMP A
SW-FIS-17A-2
8GPM ≥FLOW
C0286
BD-A
SW-TS-17A
120°F<TEMP
C0287
LOCAL
PRA-CC-1A
SW-FIS-42A
60GPM ≥FLOW
BD-A
LPCS PUMP
SW-FIS-18
3.5GPM≥FLOW
C0302
BD-A
SW-TS-18
120°F < TEMP
C0303
LOCAL
DMA-CC-11
SW-FIS-11A
75GPM ≥FLOW
BD-C
LOCAL
DMA-CC-12
SW-FIS-11B
30GPM ≥FLOW



SHEET 524-8

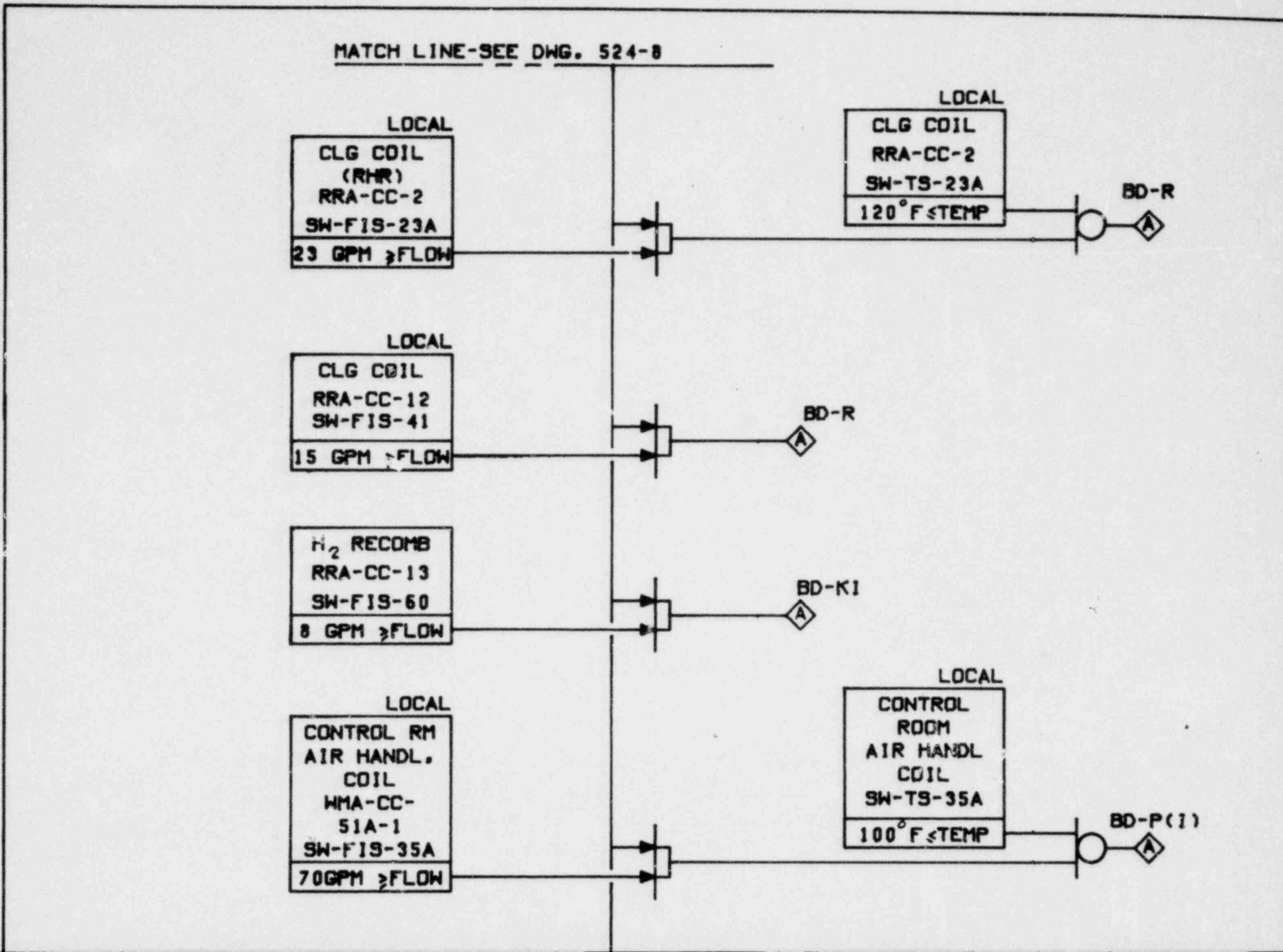
I&C DWG. NO. M-620

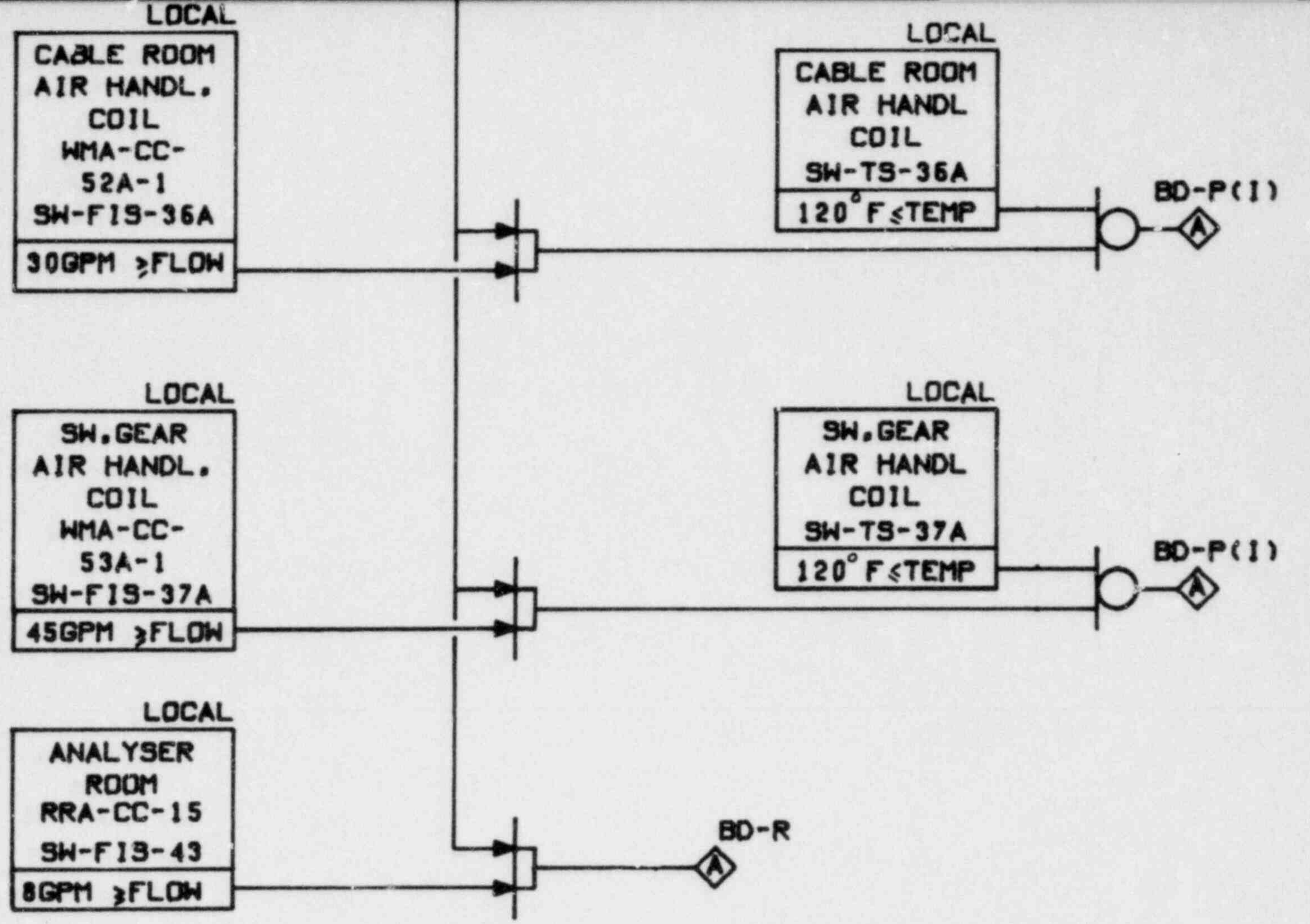
AMENDMENT NO. 10
July 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL LOGIC DIAGRAM - SSW SYSTEM

FIGURE 7.3-17j


MATCH LINE-SEE DWG. 524-8
LOCAL
CLG COIL (RHR) RRA-CC-2 SW-FIS-23A
23 GPM ≥FLOW
LOCAL
CLG COIL RRA-CC-2 SW-TS-23A
120°F ≤TEMP
BD-R
LOCAL
CLG COIL RRA-CC-12 SW-FIS-41
15 GPM ≥FLOW
BD-R
H2 RECOMB RRA-CC-13 SW-FIS-60
8 GPM ≥FLOW
BD-KI
LOCAL
CONTROL RM AIR HANDL. COIL WMA-CC-51A-1 SW-FIS-35A
70GPM ≥FLOW
LOCAL
CONTROL ROOM AIR HANDL COIL SW-TS-35A
100°F ≤TEMP
BD-P(I)
A



REF DWG.
STANDBY SERVICE WATER FL.D. M-524

STANDBY SERVICE WATER A LOW FLOW ALARM
CONTROL LOGIC DIAGRAM

I&C DWG. NO. M-620

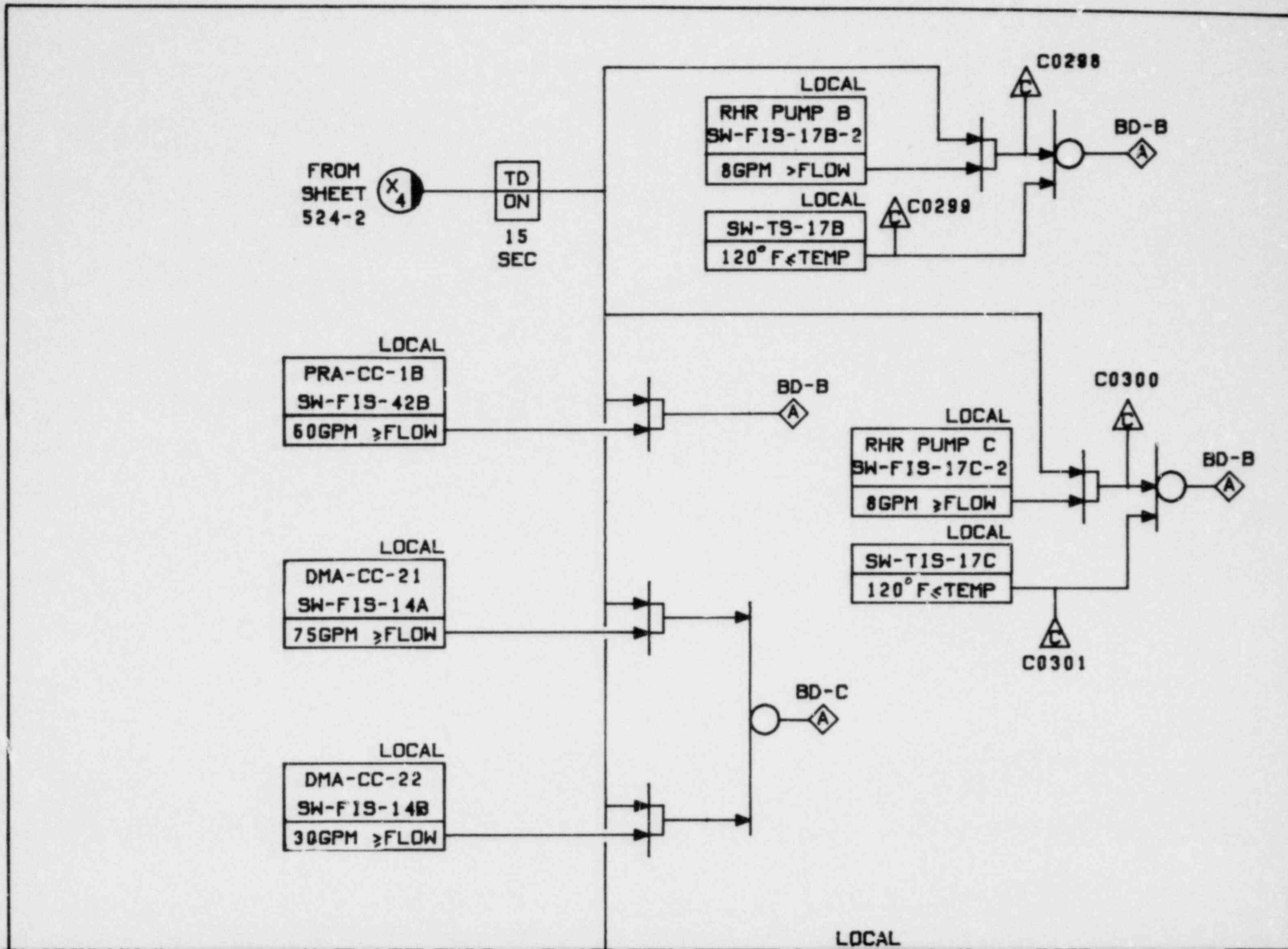
SHEET 524-8A

AMENDMENT NO. 10
July 1980

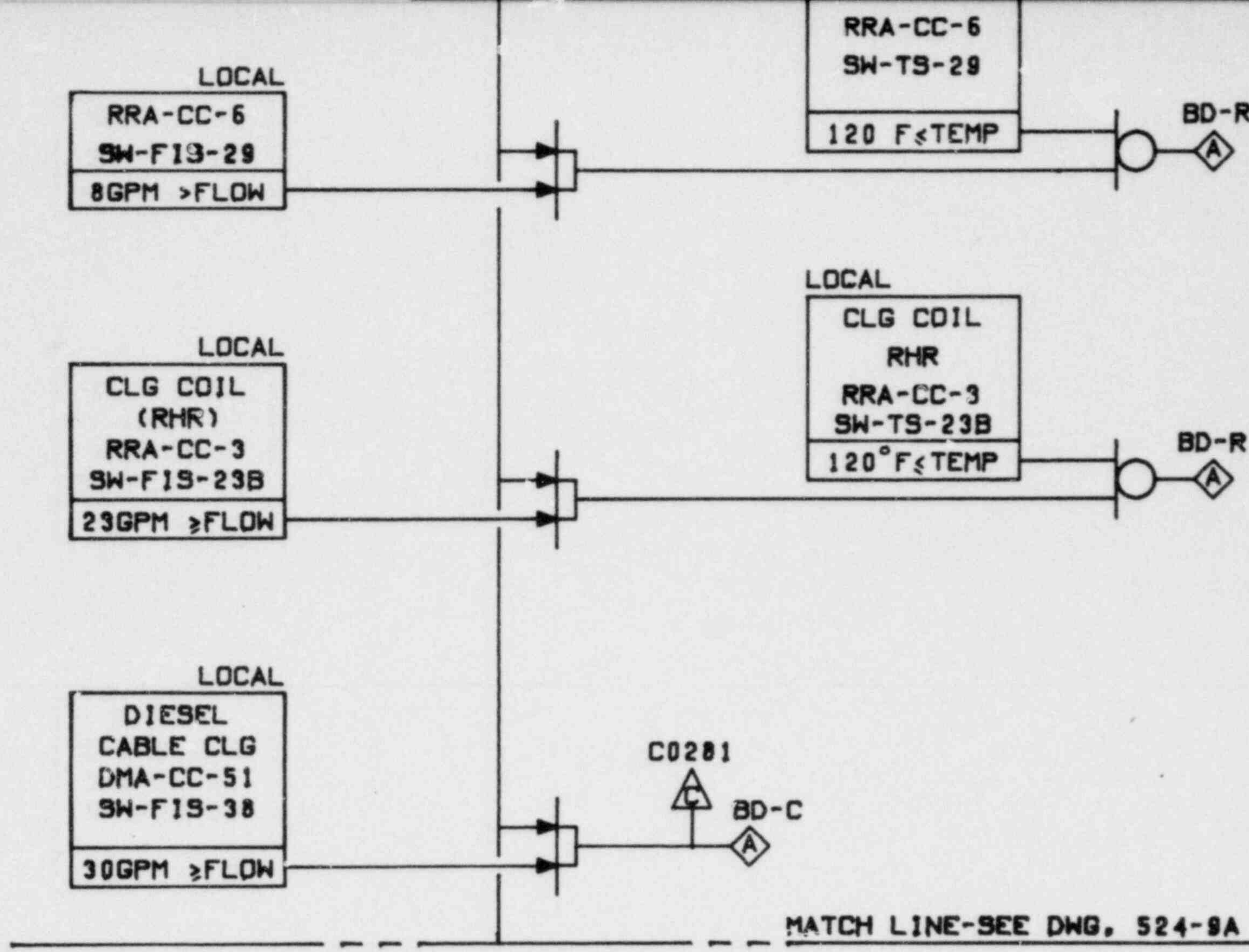
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL LOGIC DIAGRAM - SSW SYSTEM

FIGURE 7.3-17k



FROM SHEET 524-2
X 4
TD ON
15 SEC
LOCAL
RHR PUMP B
SW-FIS-17B-2
8GPM >FLOW
C0298
BD-B
LOCAL
SW-TS-17B
120° F≤TEMP
C0299
LOCAL
PRA-CC-1B
SW-FIS-42B
60GPM ≥FLOW
BD-B
C0300
LOCAL
RHR PUMP C
SW-FIS-17C-2
8GPM ≥FLOW
BD-B
LOCAL
SW-TIS-17C
120° F≤TEMP
C0301
LOCAL
DMA-CC-21
SW-FIS-14A
75GPM ≥FLOW
BD-C
LOCAL
DMA-CC-22
SW-FIS-14B
30GPM ≥FLOW
LOCAL



SHEET 524-8

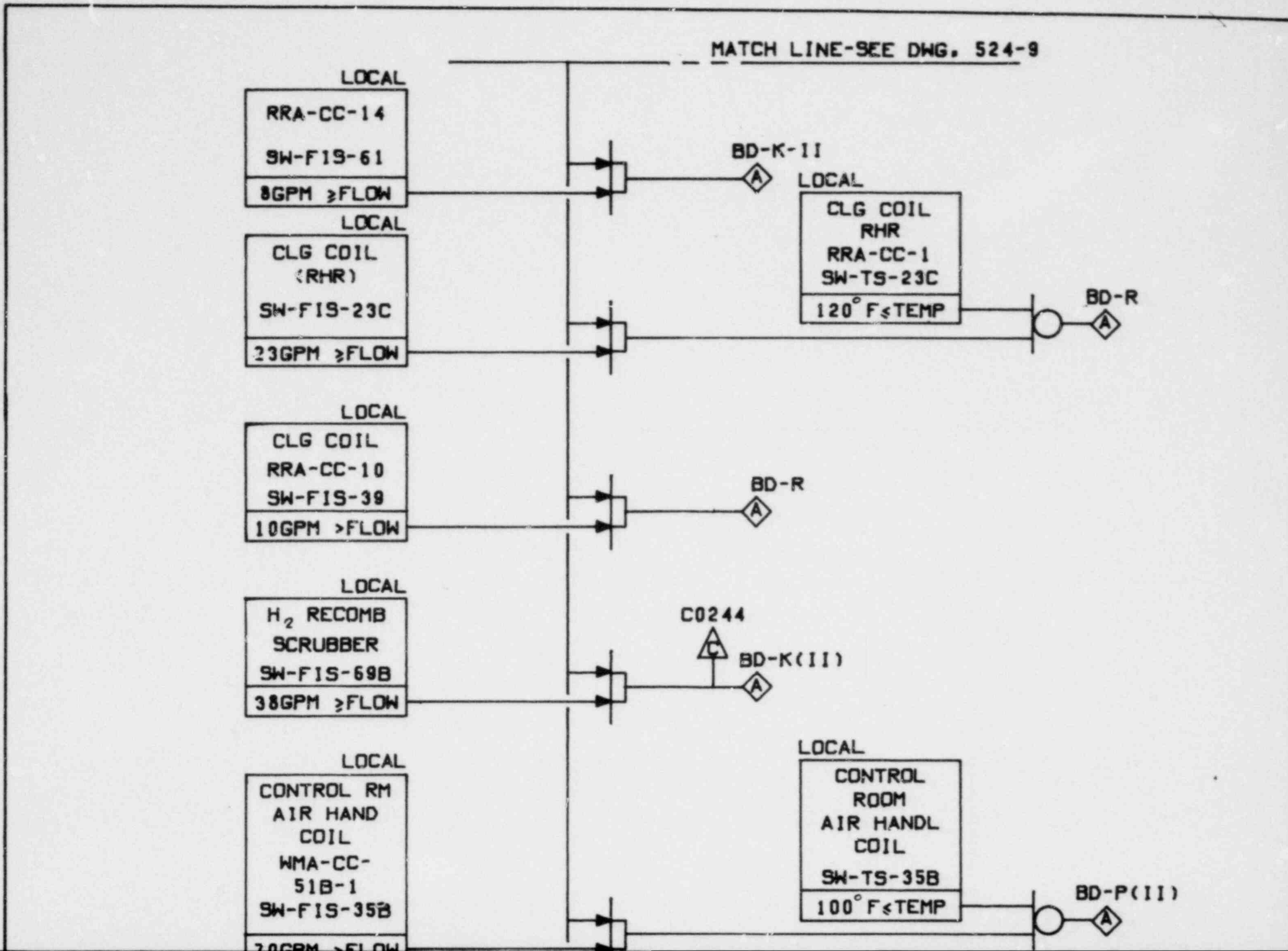
I&C DWG. NO. M-620

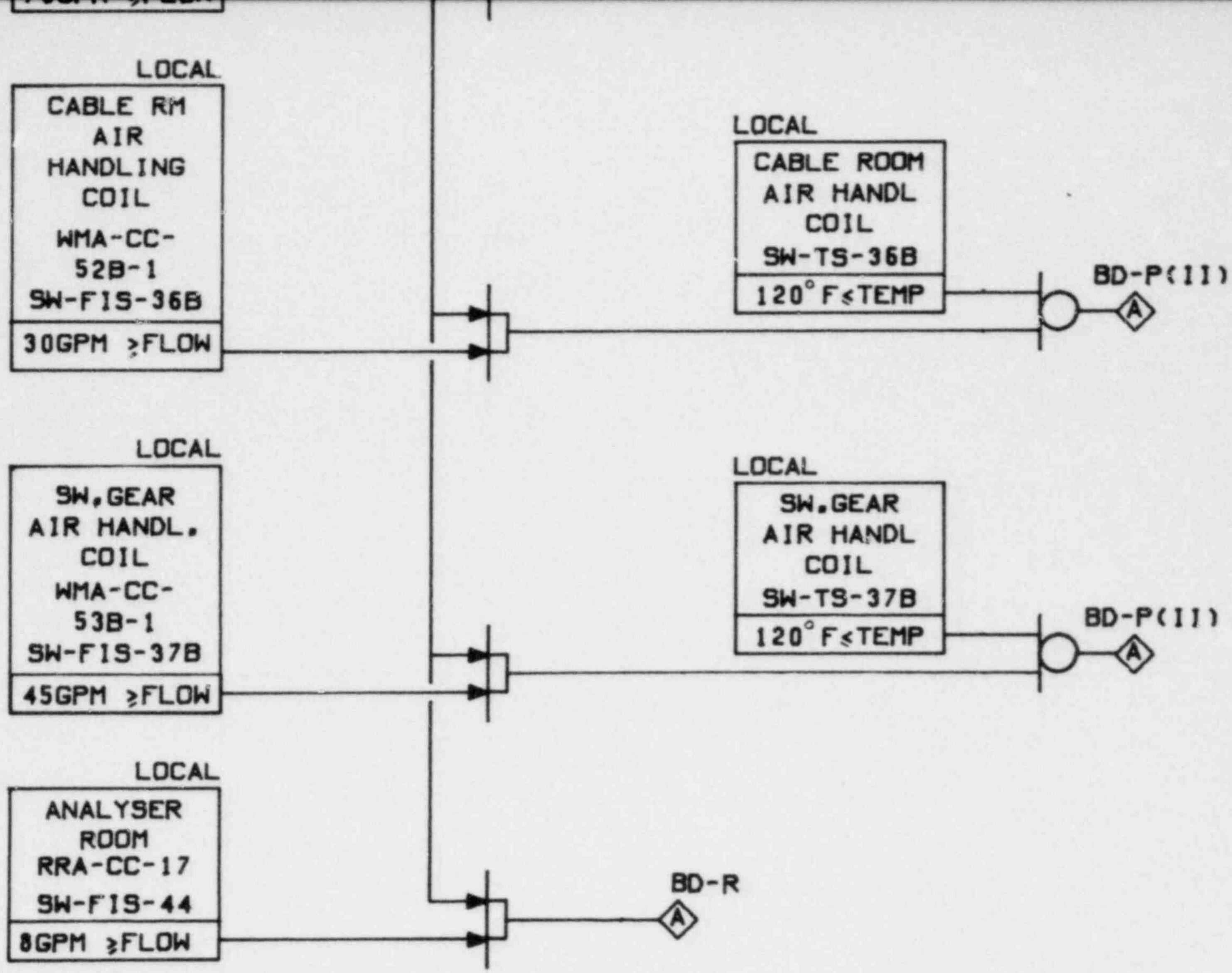
AMENDMENT NO. 10
July 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL LOGIC DIAGRAM - SSW SYSTEM

FIGURE 7.3-17L


MATCH LINE-SEE DWG. 524-9
LOCAL
RRA-CC-14
SW-FIS-61
8GPM ≥FLOW
BD-K-II
LOCAL
CLG COIL
RHR
RRA-CC-1
SW-TS-23C
120° F≤TEMP
BD-R
LOCAL
CLG COIL
(RHR)
SW-FIS-23C
23GPM ≥FLOW
LOCAL
CLG COIL
RRA-CC-10
SW-FIS-39
10GPM >FLOW
BD-R
LOCAL
H2 RECOMB
SCRUBBER
SW-FIS-69B
38GPM ≥FLOW
C0244
BD-K(II)
LOCAL
CONTROL RM
AIR HAND
COIL
WMA-CC-
51B-1
SW-FIS-35B
LOCAL
CONTROL
ROOM
AIR HANDL
COIL
SW-TS-35B
100° F≤TEMP
BD-P(II)



STANDBY SERVICE WATER B LOW FLOW ALARM
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

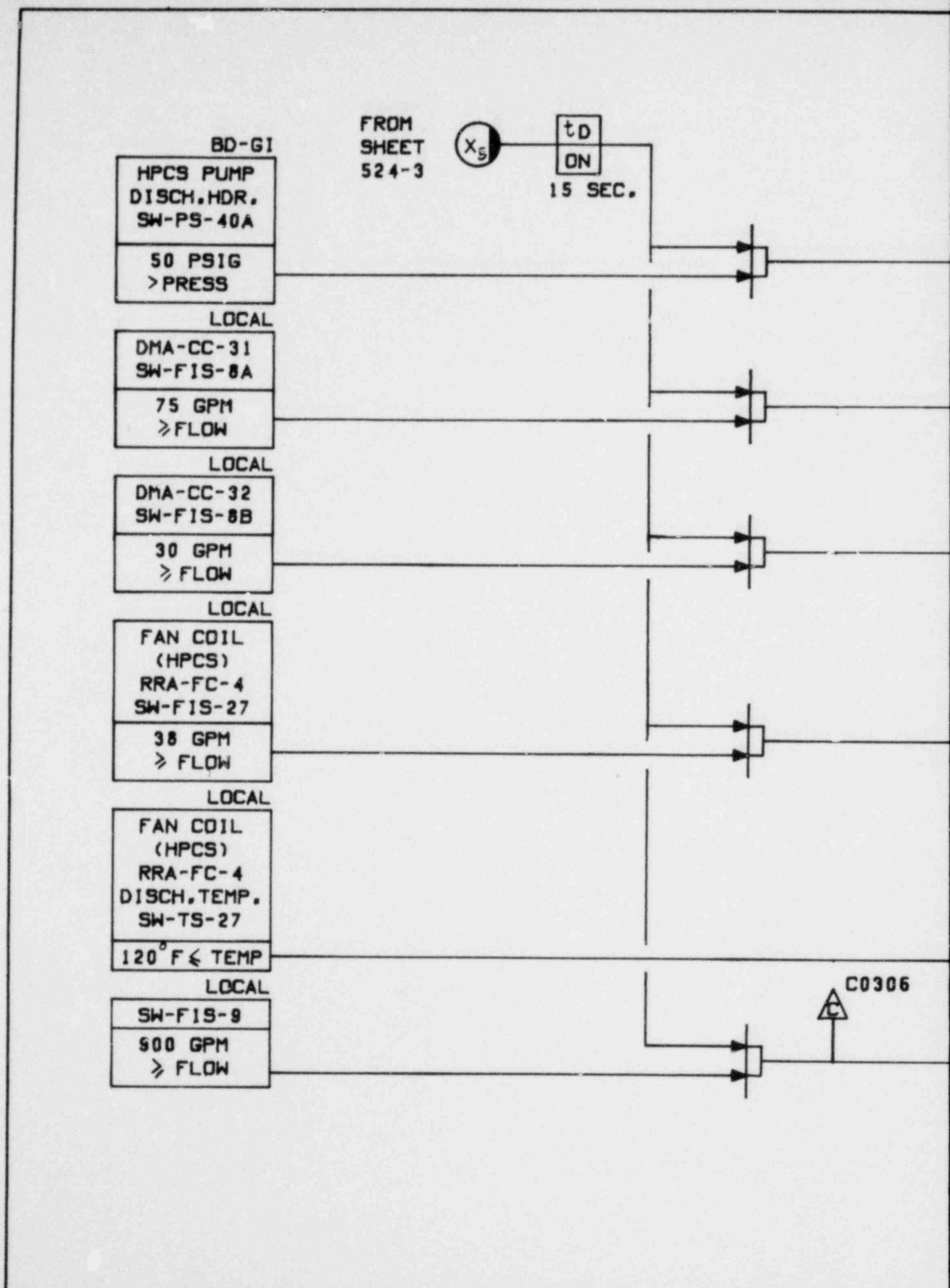
CONTROL LOGIC DIAGRAM - SSW SYSTEM

FIGURE 7.3-17m

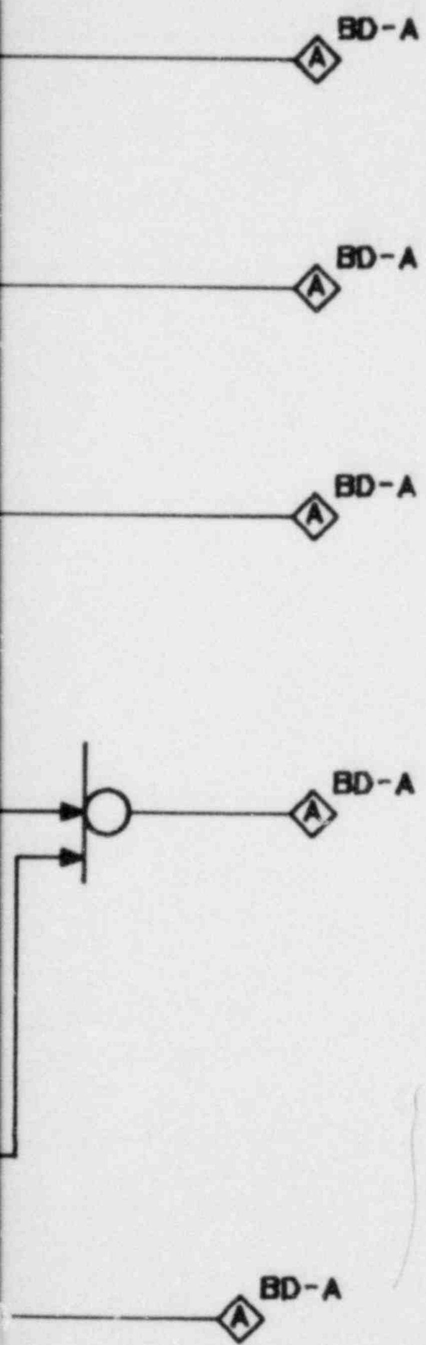
I&C DWG. NO. M-620

SHEET 524-9A

AMENDMENT NO. 10
July 1980


FROM
SHEET
524-3
X_S
t_D
ON
15 SEC.
BD-GI
HPCS PUMP
DISCH.HDR.
SW-PS-40A
50 PSIG
>PRESS
LOCAL
DMA-CC-31
SW-FIS-8A
75 GPM
≥FLOW
LOCAL
DMA-CC-32
SW-FIS-8B
30 GPM
≥ FLOW
LOCAL
FAN COIL
(HPCS)
RRA-FC-4
SW-FIS-27
38 GPM
≥ FLOW
LOCAL
FAN COIL
(HPCS)
RRA-FC-4
DISCH.TEMP.
SW-TS-27
120°F ≤ TEMP
LOCAL
SW-FIS-9
900 GPM
≥ FLOW
C
C0306

AMENDMENT NO. 10
July 1980



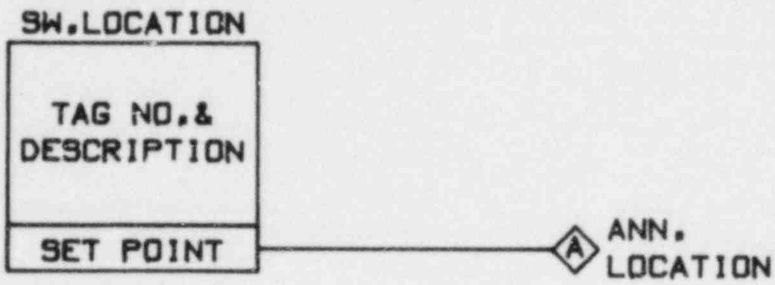
REF.DWG.
STANDBY SERVICE WATER FL. D. M-524

I & C DWG. NO. M-620

SHEET 524-10

HPCS SERVICE WATER 'LOW FLOW' ALARM
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17n



TAG NO.	S LOCA
SW-LS-1A1	BD-
SW-TS-1A	BD-
SW-LS-1C1	BD-
SW-TS-1C	BD-
SW-LS-1B1	BD-
SW-LS-1D1	BD-
SW-TS-1B	BD-
SW-TS-1D	BD-

I. TION	DESCRIPTION	SET POINT	ANN. LOCATION	REMARK
GI	SPRAY POND "A" HI/LO WTR.LEV.	434'-6" / 433'-6"	BD-A	
GI	SPRAY POND "A" HI/LO WTR.TEMP.	85°F/35°F	BD-A	
GI	SPRAY POND "B" HI/LO WTR.LEV.	434'-6" / 433'-6"	BD-A	
GI	SPRAY POND "B" HI/LO WTR.TEMP.	85°F/35°F	BD-A	
GII	SPRAY POND "B" HI/LO WTR.LEV.	434'-6" / 433'-6"	BD-B	
GII	SPRAY POND "A" HI/LO WTR.LEV.	434'-6" / 433'-6"	BD-B	
GII	SPRAY POND "B" HI/LO WTR. TEMP.	85°F/35°F	BD-B	
GII	SPRAY POND "A" HI/LO WTR.TEMP.	85°F/35°F	BD-B	

I & C DWG. NO. M-620

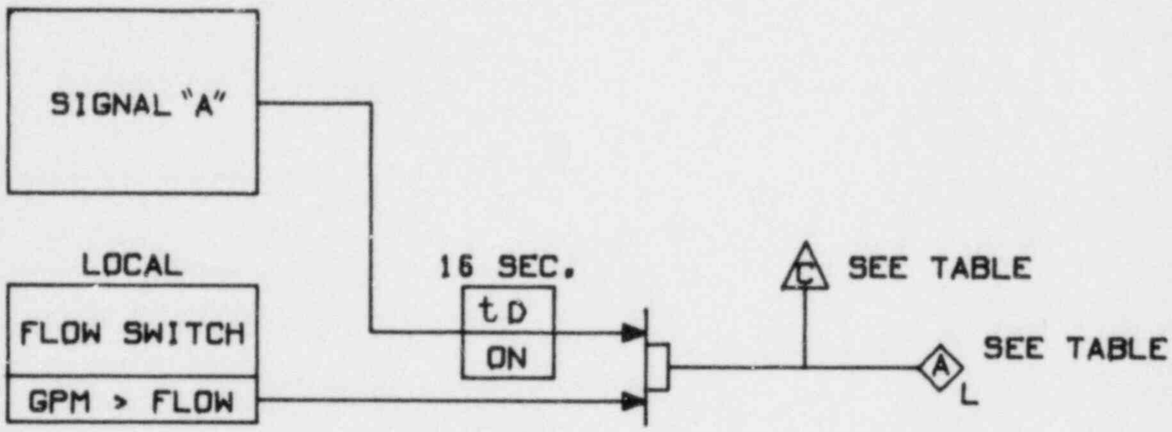
SHEET 524-11

ALARM ANNUNCIATOR & COMPUTER INPUT
CONTROL LOGIC DIAGRAM

AMENDMENT NO. 10
July 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17o

CON
F
CO
CO



SIGNAL "A"
LOCAL
FLOW SWITCH
GPM > FLOW
16 SEC.
tD
ON
C
SEE TABLE
A
L
SEE TABLE

P. INPUT PTS		ANN. LOCATION	SIGNAL "A"	FLOW SWITCH NUMBER	SET POINT
IS					
304		BD-C	DIESEL 1A START	SW-FIS-12	FLOW < 1400 GPM
305		BD-C	DIESEL 1B START	SW-FIS-15	FLOW < 1400 GPM

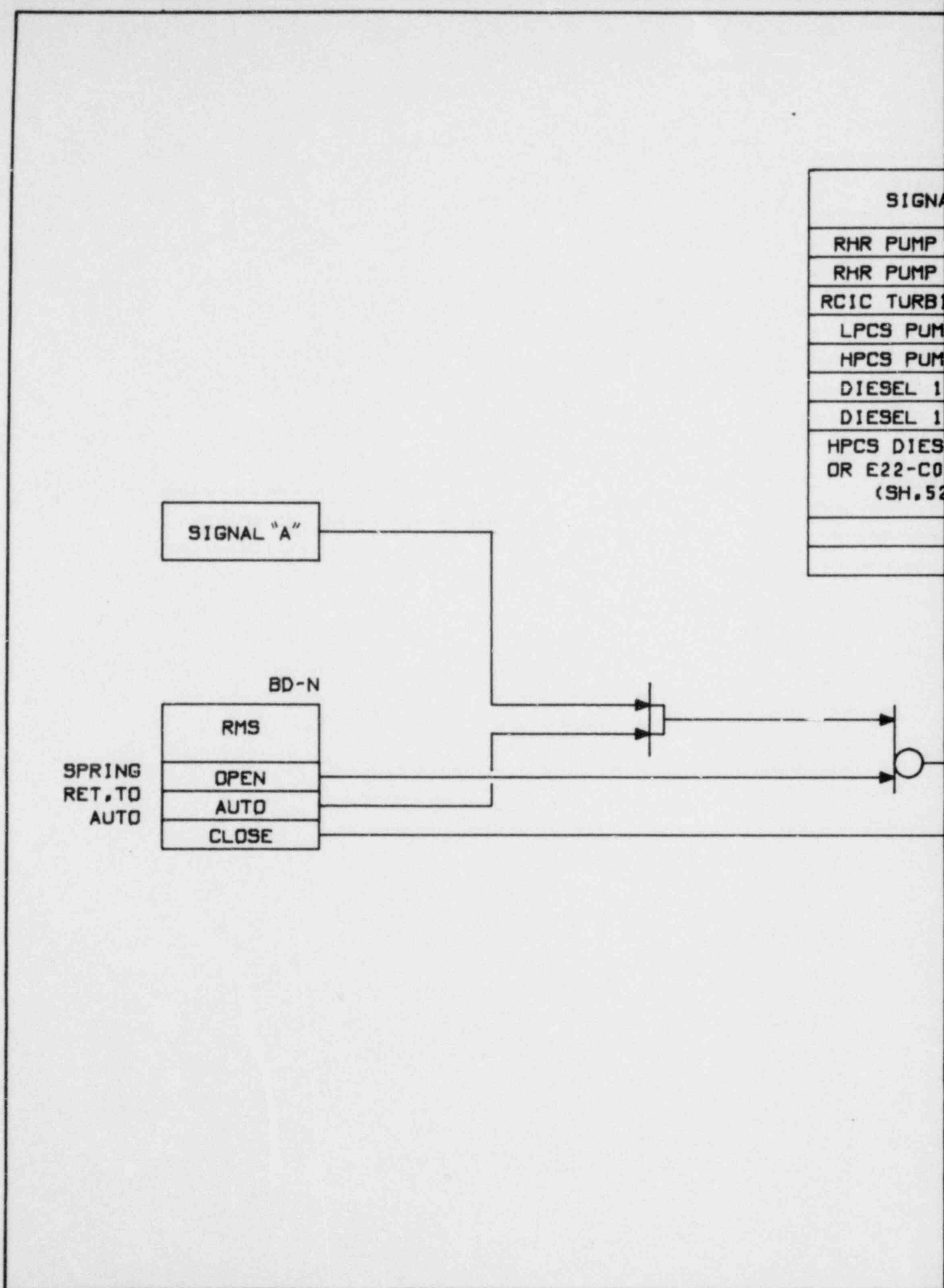
I & C DWG. NO. M-620

SHEET 524-12

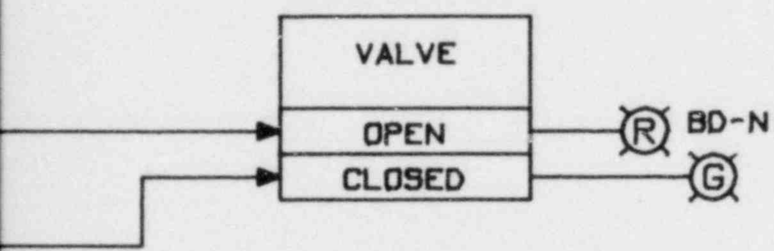
STANDBY SERVICE WATER "LOW FLOW ALARM"
CONTROL LOGIC DIAGRAM

AMENDMENT NO. 10
July 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17p


SIGNA
RHR PUMP
RHR PUMP
RCIC TURBI
LPCS PUM
HPCS PUM
DIESEL 1
DIESEL 1
HPCS DIES
OR E22-C0
(SH.52
SIGNAL "A"
BD-N
RMS
SPRING
RET,TO
AUTO
OPEN
AUTO
CLOSE

L A	RMS FOR VALVE	RMS & LIGHT LOCATION	SERVICE
A START	SW-V-24A	BD-N	RHR A BRG.& ROOM CLG.WTR.
C START	SW-V-24C		C
NE START	SW-V-34		RCIC PUMP ROOM CLG.WTR.
P START	SW-V-44		MOTOR BRG.& ROOM CLG.WTR.
P START	SW-V-54		PUMP ROOM CLG.WTR.
A START	SW-V-4A		DIESEL 1A CLG.WTR.
B START	SW-V-4B		1B
EL START 02 START 4-3)	SW-V-4C		HPCS DIESEL

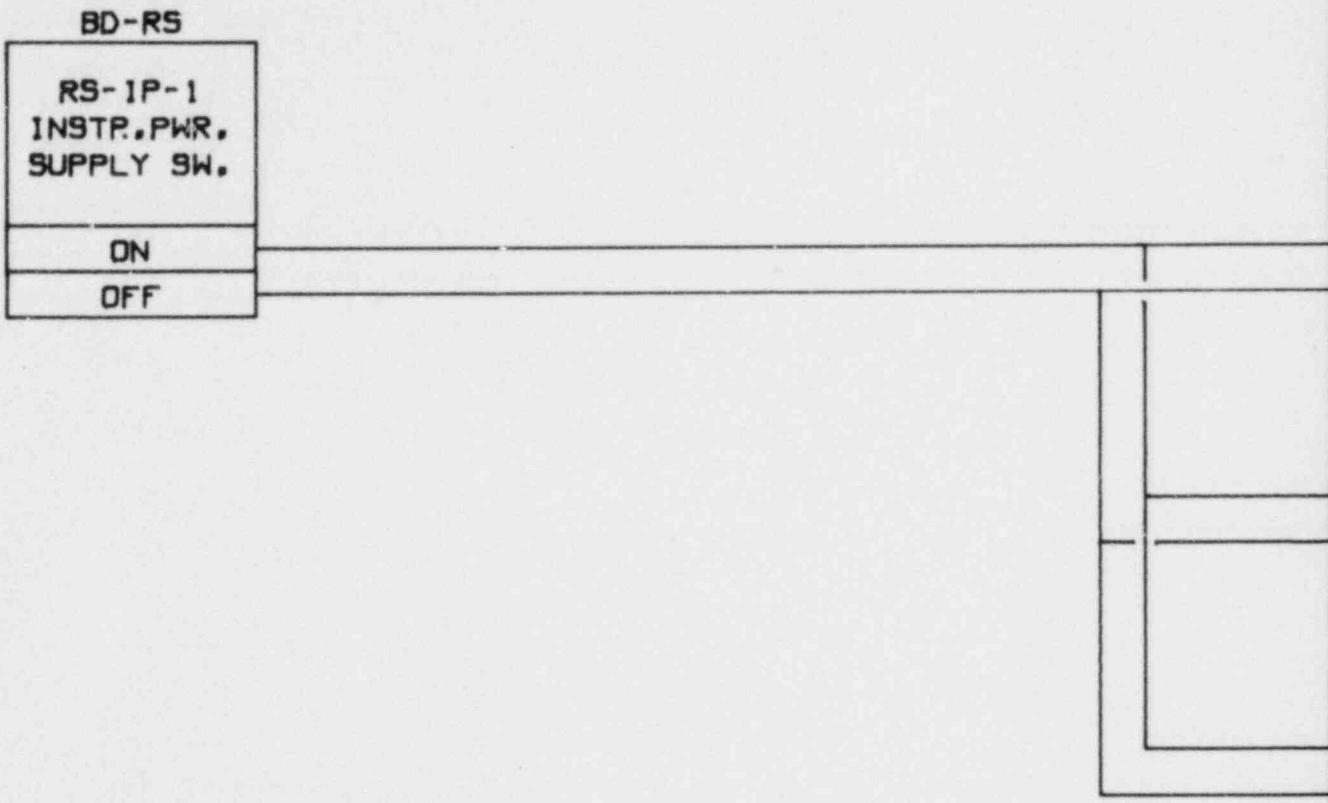


I & C DWG. NO. M-620

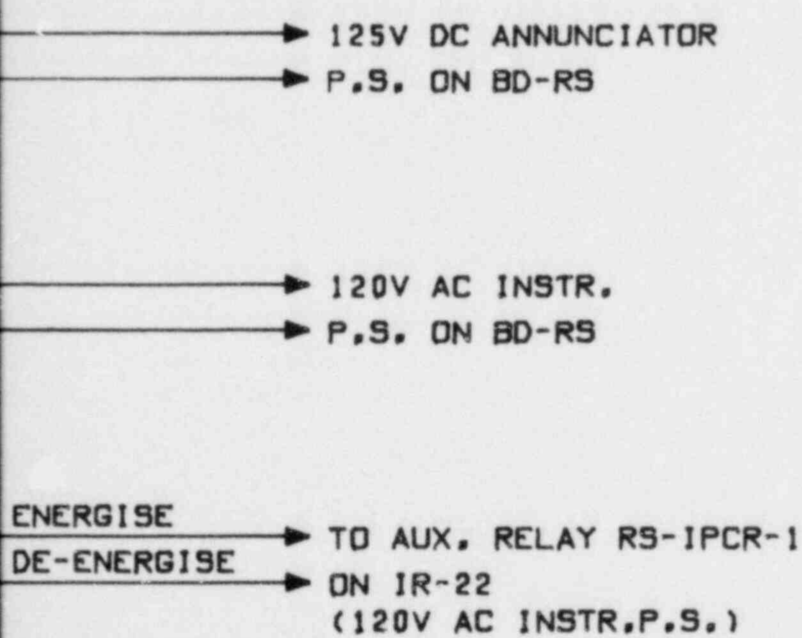
SHEET 524-13

STANDBY SERVICE WATER SYSTEM SHEET 13
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17q


BD-RS
RS-1P-1
INSTR.PWR.
SUPPLY SW.
ON
OFF

AMENDMENT No. 10
July 1980

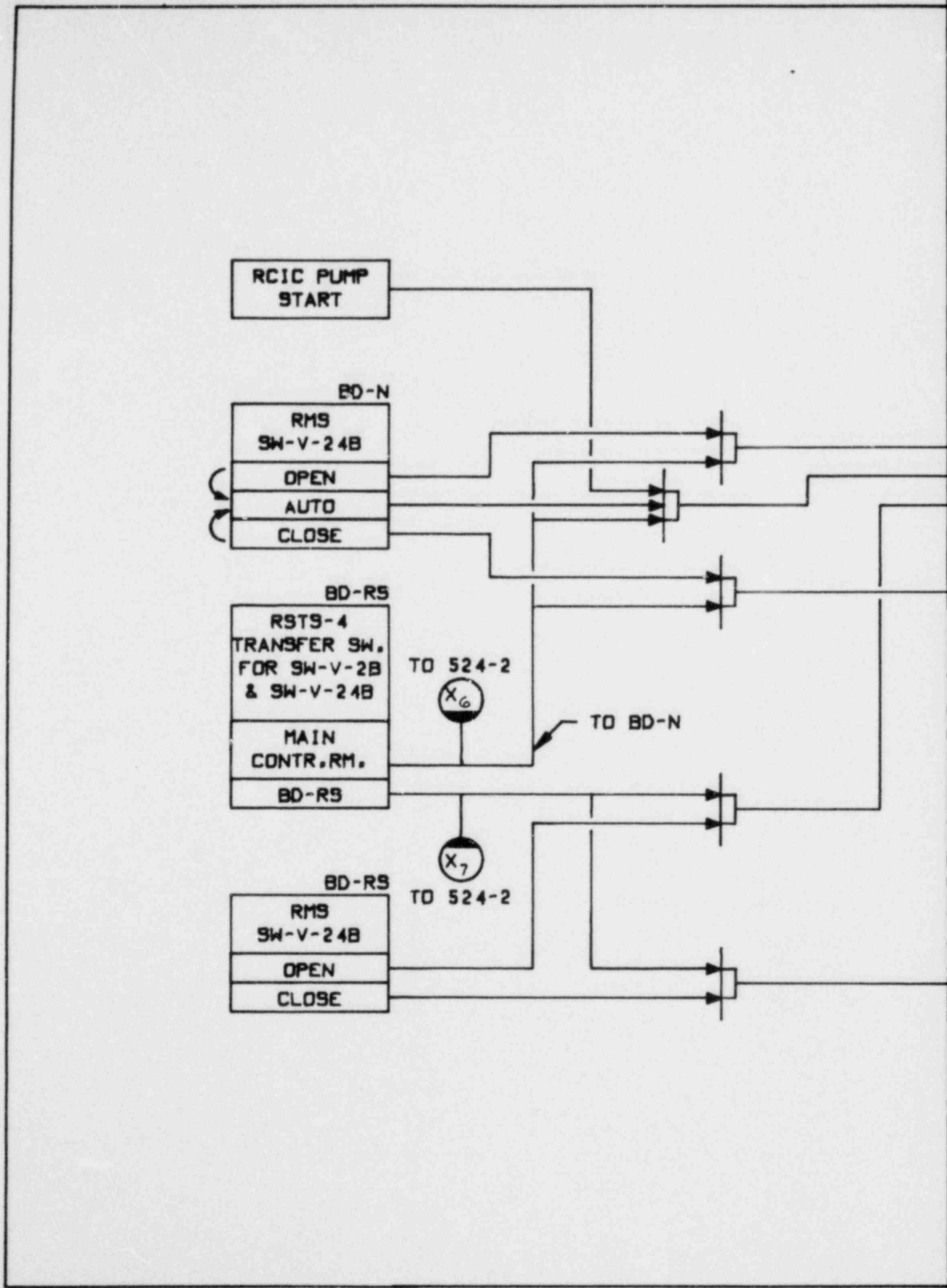


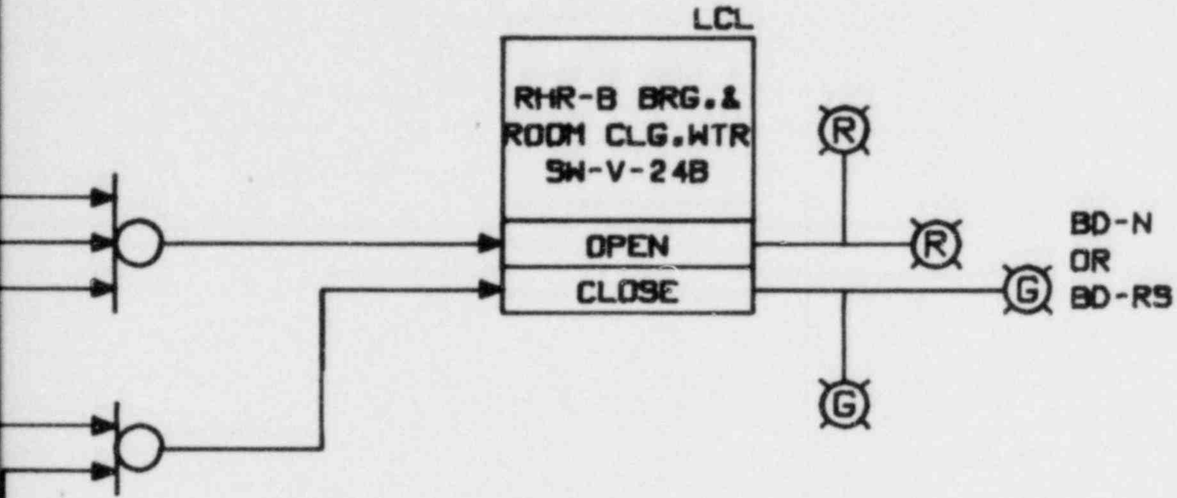
I & C DWG. NO. M-620

SHEET 524-14

POWER SUPPLY LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17r


RCIC PUMP START
BD-N
RMS SW-V-24B
OPEN
AUTO
CLOSE
BD-RS
RSTS-4 TRANSFER SW. FOR SW-V-2B & SW-V-24B
MAIN CONTR.RM.
BD-RS
TO 524-2
X6
TO BD-N
X7
TO 524-2
BD-RS
RMS SW-V-24B
OPEN
CLOSE



I & C DWG. NO. M-620

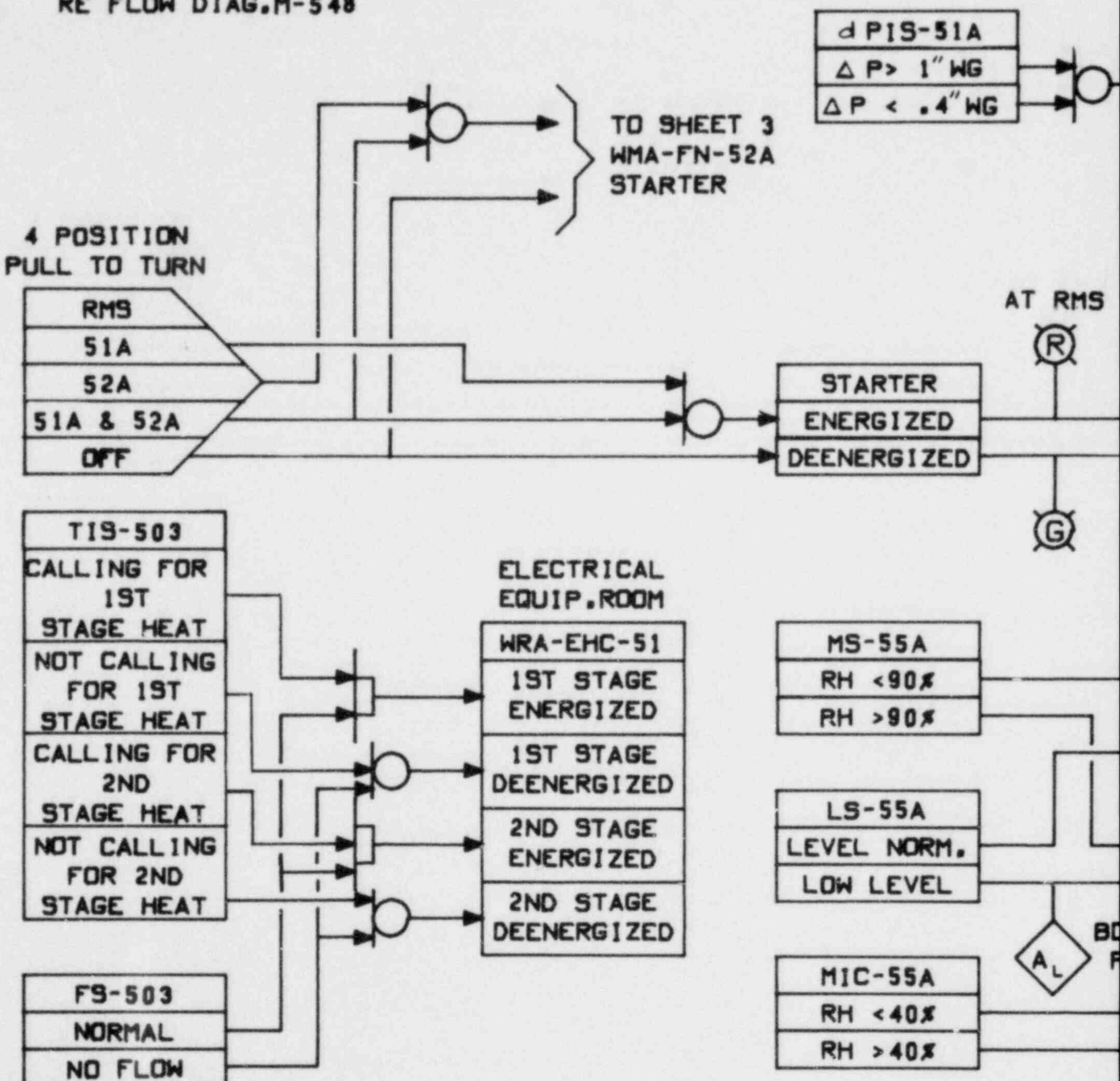
SHEET 524-17

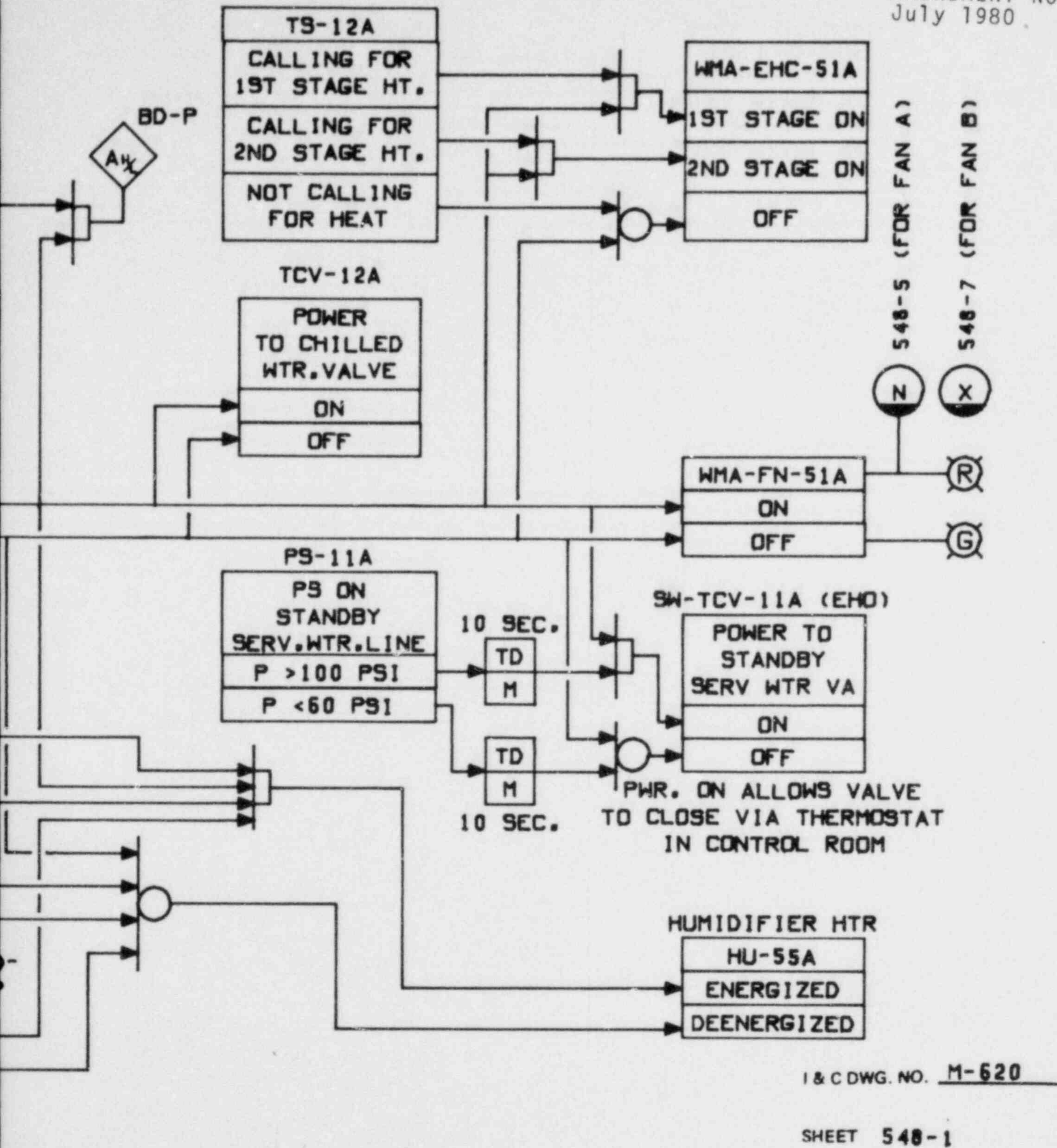
RHR-B BEARING & PUMP ROOM CLG.WTR.
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SSW SYSTEM	FIGURE 7.3-17s

NOTE
LOGIC FOR DIVISION II (B SUFFIX) UNITS SIMILAR, EXCEPT AS NOTED.

RE FLOW DIAG. M-548





WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3-18a

LOCATION

INSTR TAG NO.

CONDITION

ALARM LOCATION

INSTR. TA
WMA-TS-206
-207
-208
-209
-210
-211
-212
-213
-215
-216
-224
-503
-510
WMA-SMD-51
WMA-SMD-52A
WMA-SMD-53A
WMA-SMD-53B
WEA-SMD-51
WEA-SMD-53A
WEA-SMD-53B
WEA-dPS-73A
WEA-dPS-73B
CL_2 ANALYSE
CL_2 ANALYSE

AMENDMENT NO. 10
July 1980

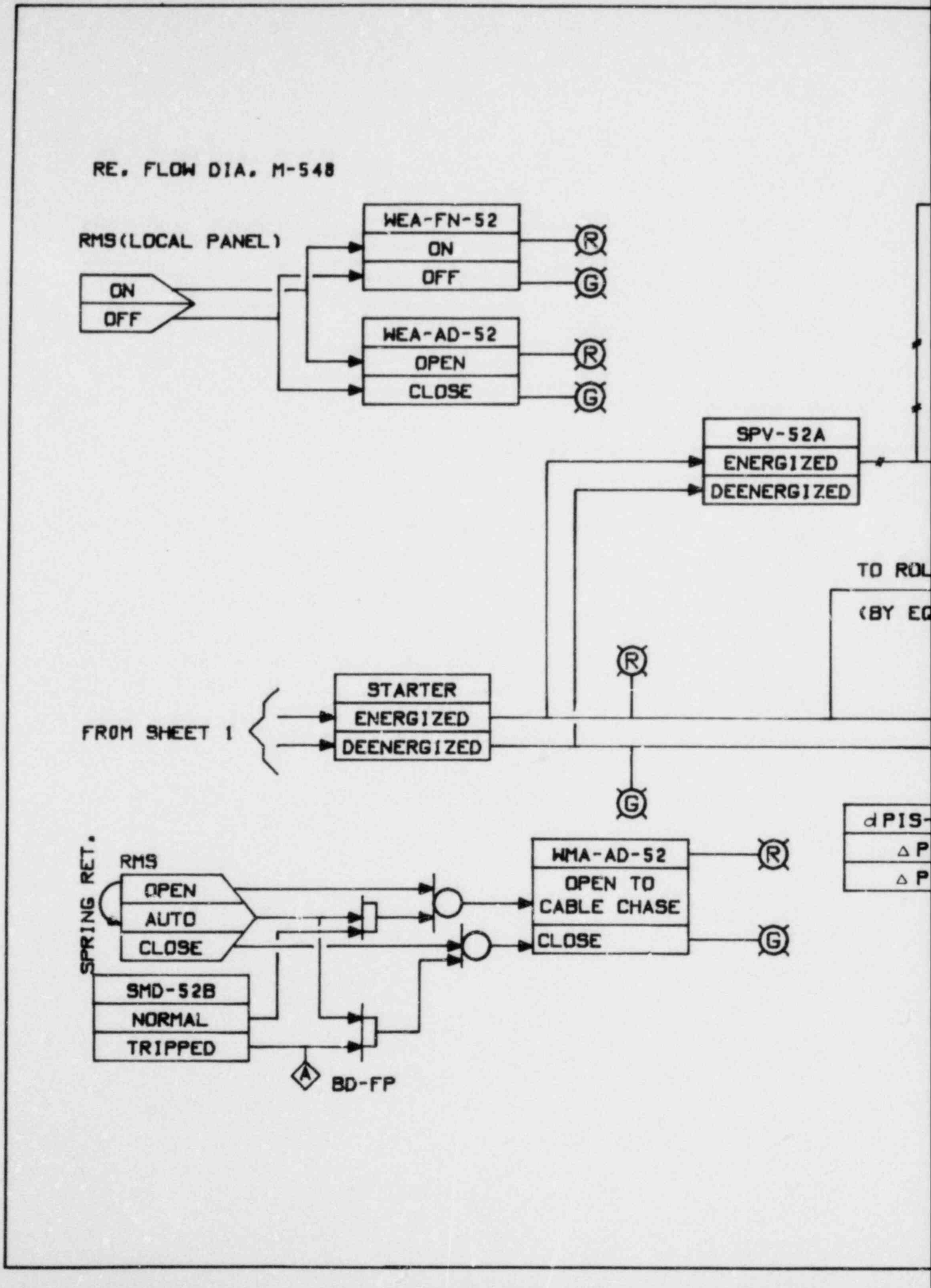
NO.	INSTR. LOCATION	CONDITION	ALARM LOCATION
	LOCAL	TEMP > 125°F	FIRE PNL
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		TEMP >	
		DP< 0.5" W.G.	BD-P1
		DP< 0.5" W.G.	BD-P2
SR-15			BD-P1
SR-16			BD-P2

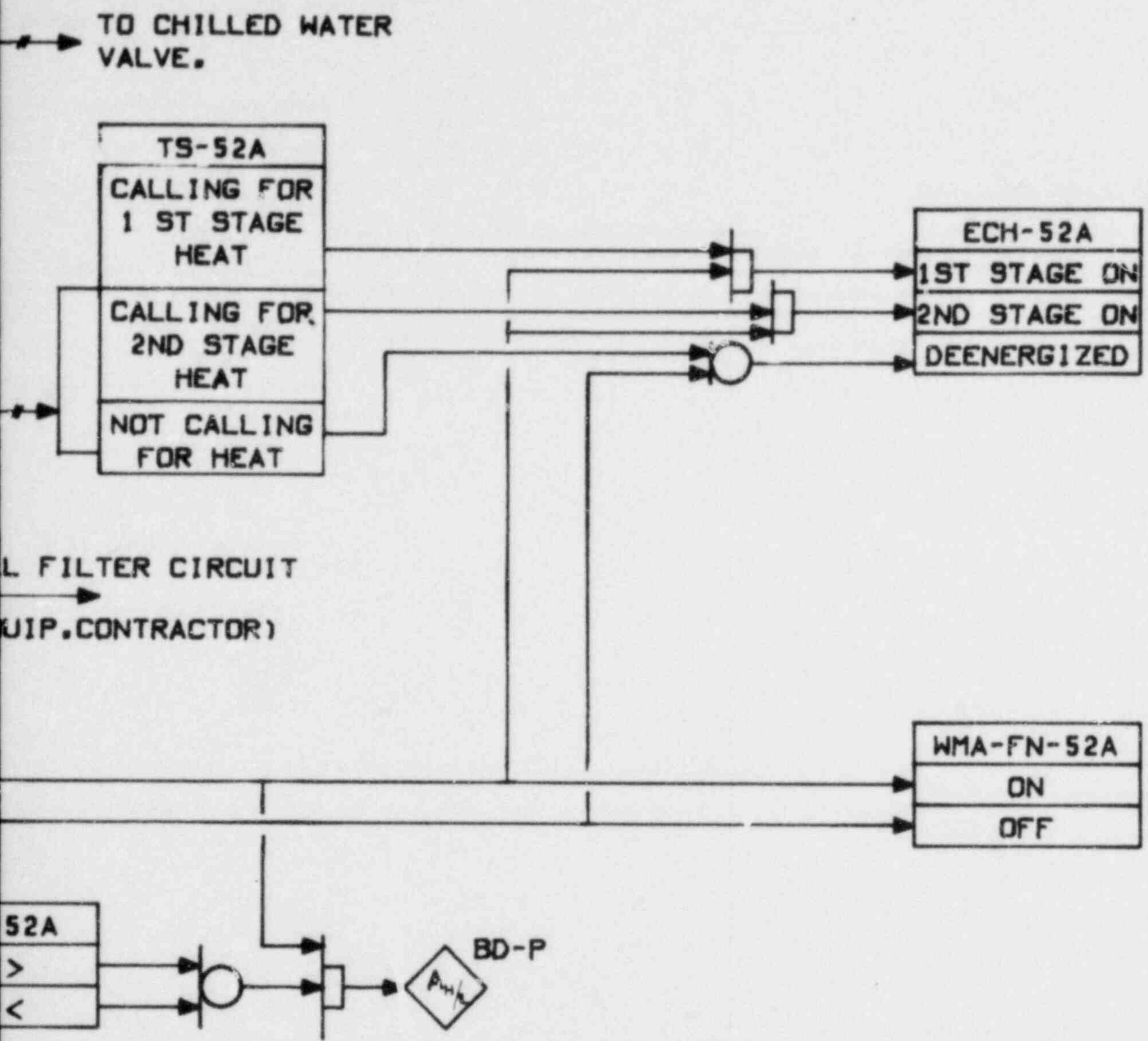
I & C DWG. NO. M-620

SHEET 548-2

HVAC ALARMS
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3- 18b


RE. FLOW DIA. M-548
RMS(LOCAL PANEL)
ON
OFF
WEA-FN-52
ON
OFF
WEA-AD-52
OPEN
CLOSE
R
G
SPV-52A
ENERGIZED
DEENERGIZED
TO ROU
(BY EQ
STARTER
ENERGIZED
DEENERGIZED
FROM SHEET 1
dPIS-
ΔP
ΔP
SPRING RET.
RMS
OPEN
AUTO
CLOSE
WMA-AD-52
OPEN TO
CABLE CHASE
CLOSE
SMD-52B
NORMAL
TRIPPED
A
BD-FP

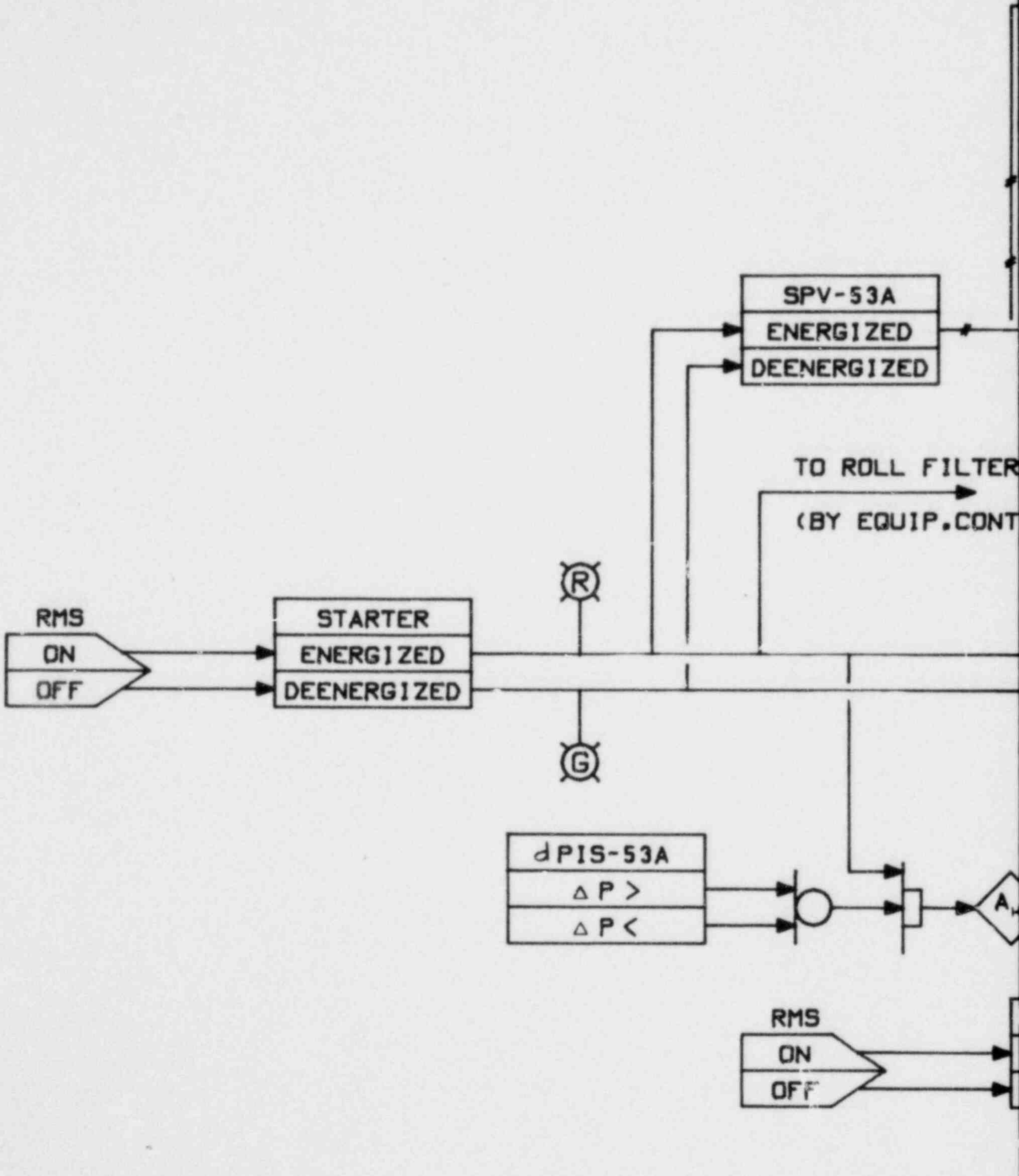


I & C DWG. NO. M-620

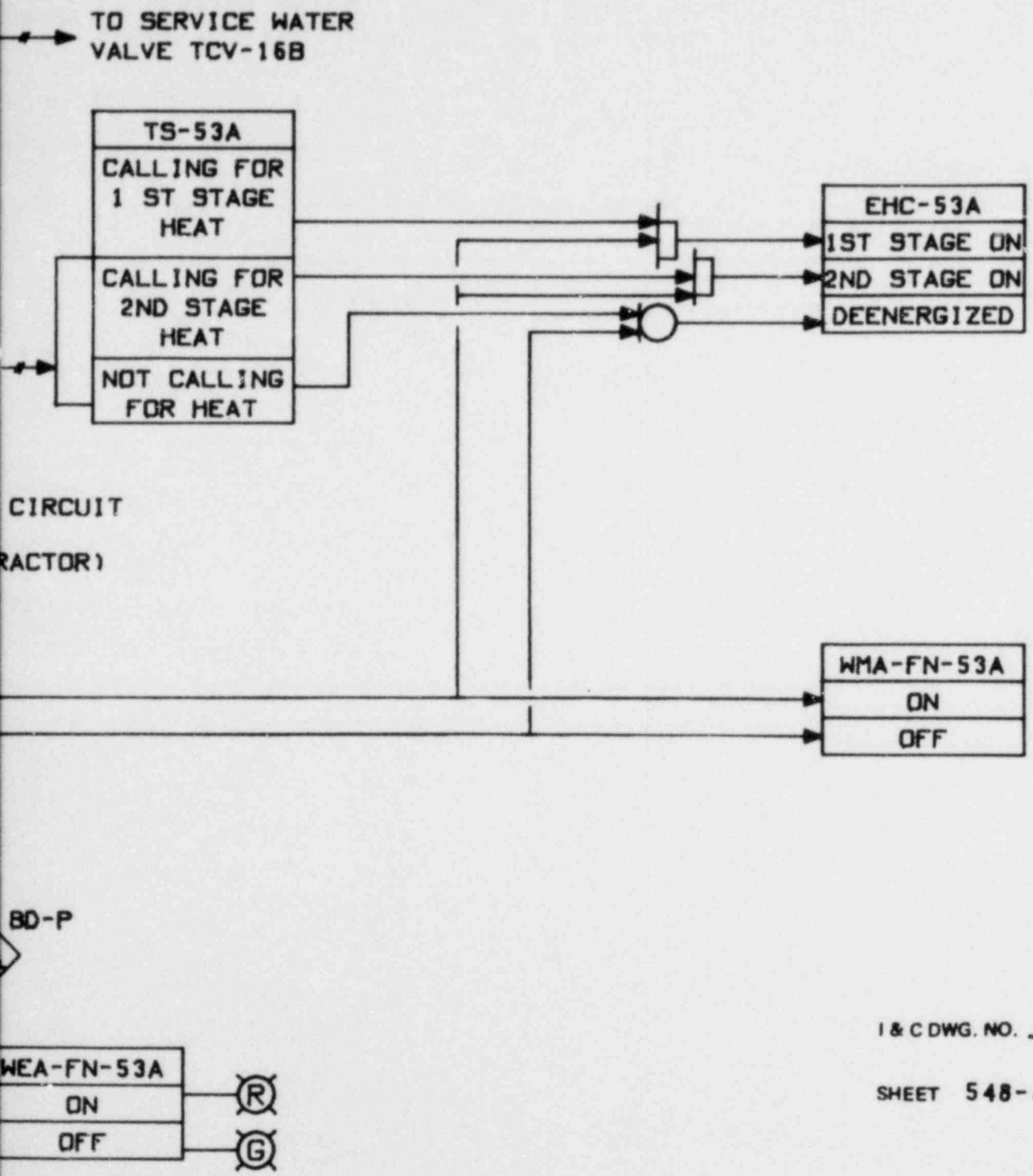
SHEET 548-3

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3- 18c

RE. FLOW DIA. M-548



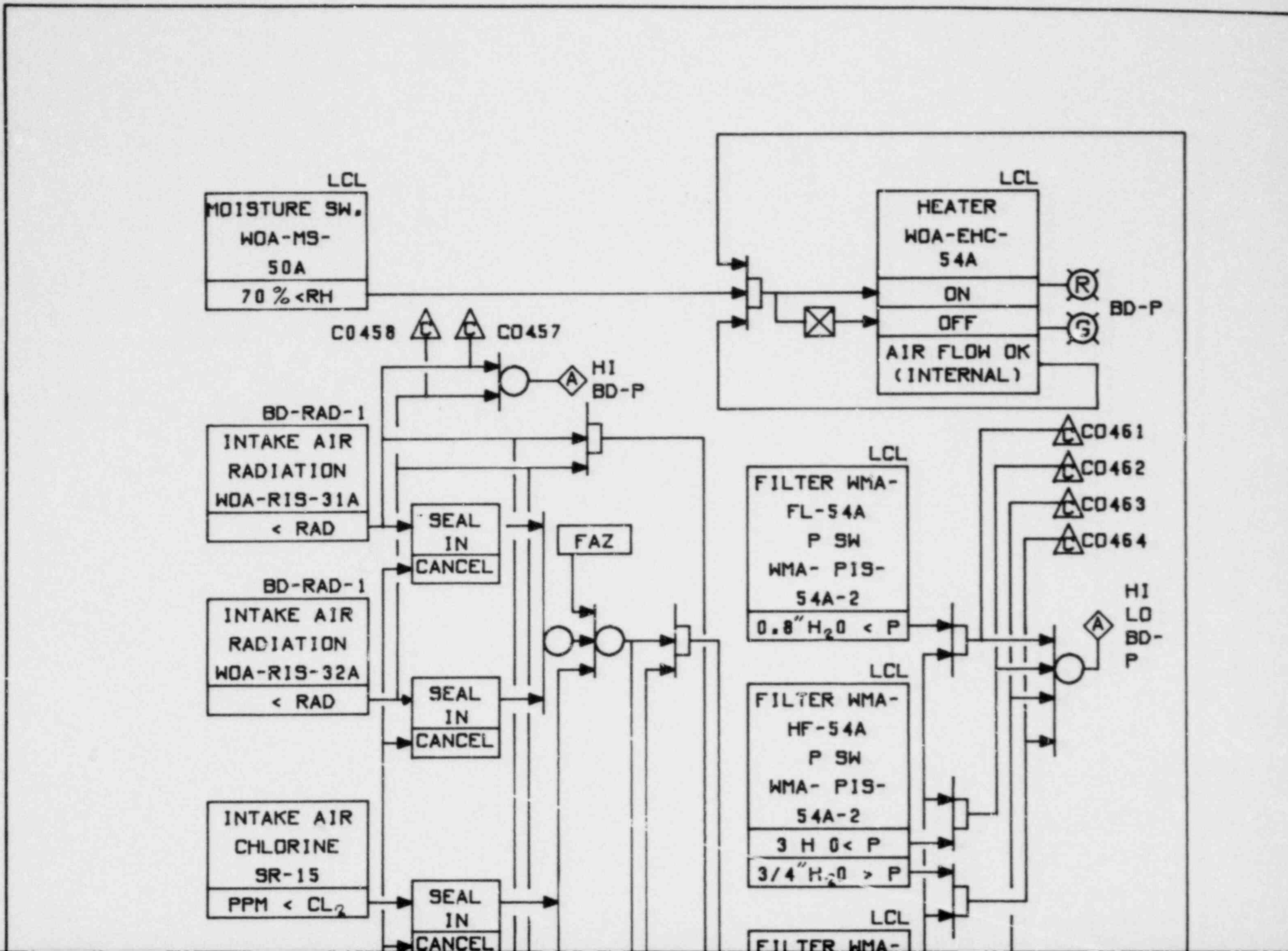
AMENDMENT NO. 10
July 1980



I & C DWG. NO. M-620

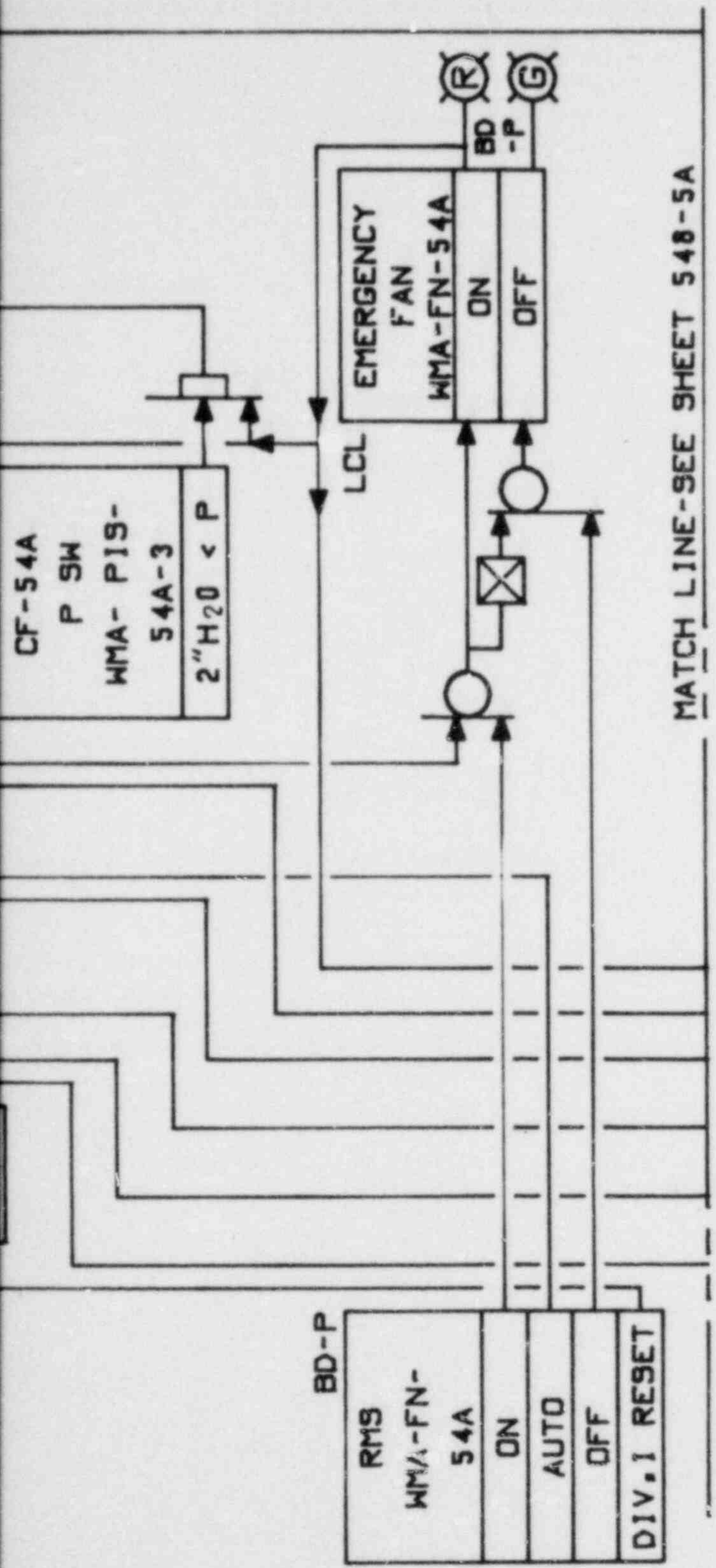
SHEET 548-4

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3-18d



LCL
MOISTURE SW.
WOA-MS-
50A
70 % <RH
LCL
HEATER
WOA-EHC-
54A
ON
OFF
AIR FLOW OK
(INTERNAL)
R
G
BD-P
C0458
C0457
A
HI
BD-P
BD-RAD-1
INTAKE AIR
RADIATION
WOA-RIS-31A
< RAD
SEAL
IN
CANCEL
FAZ
BD-RAD-1
INTAKE AIR
RADIATION
WOA-RIS-32A
< RAD
SEAL
IN
CANCEL
INTAKE AIR
CHLORINE
SR-15
PPM < CL2
SEAL
IN
CANCEL
LCL
FILTER WMA-
FL-54A
P SW
WMA- PIS-
54A-2
0.8" H2O < P
C0461
C0462
C0463
C0464
HI
LO
BD-
P
LCL
FILTER WMA-
HF-54A
P SW
WMA- PIS-
54A-2
3 H O< P
3/4" H2O > P
LCL
FILTER WMA-

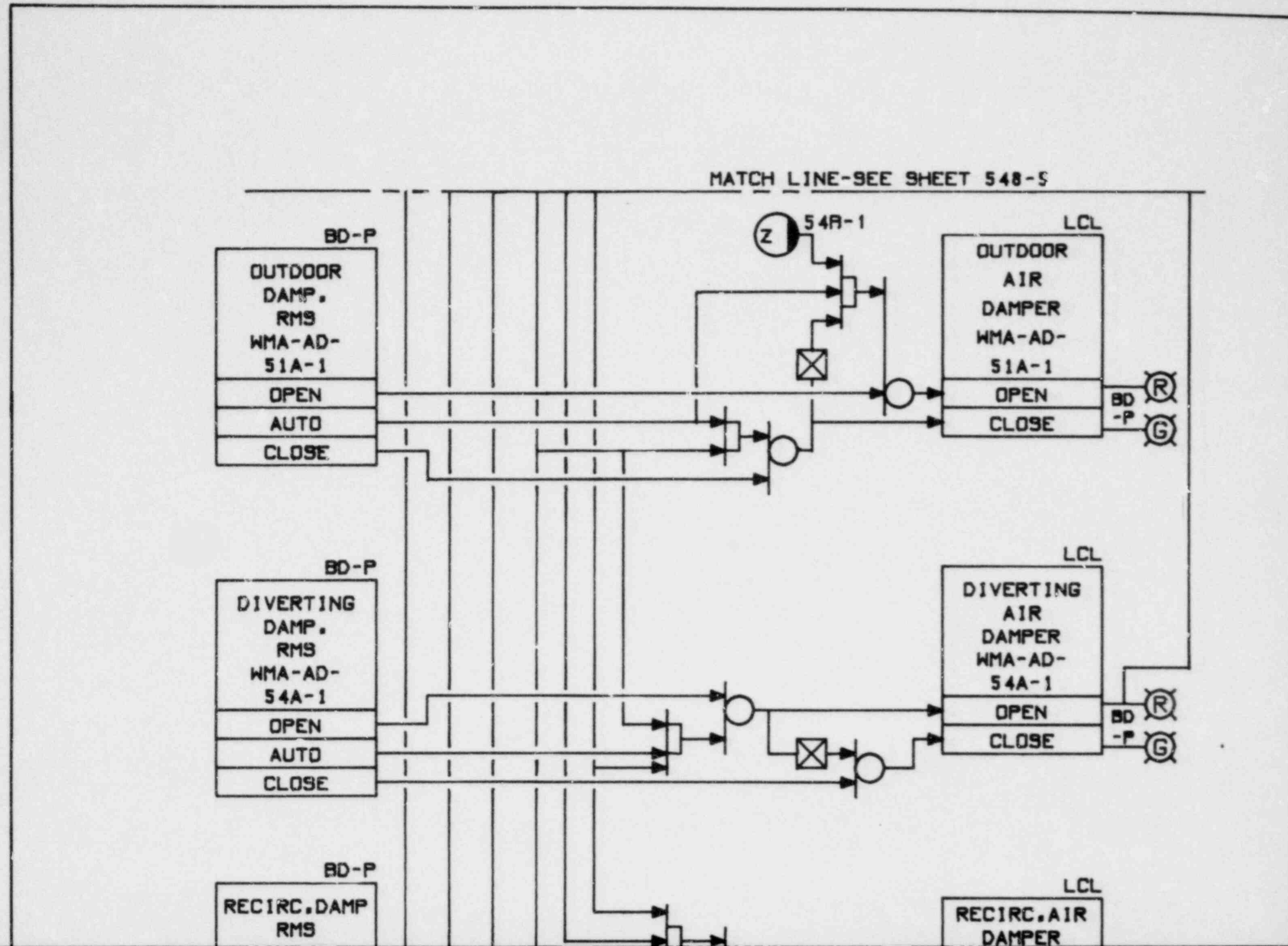
AMENDMENT NO. 10
July 1980

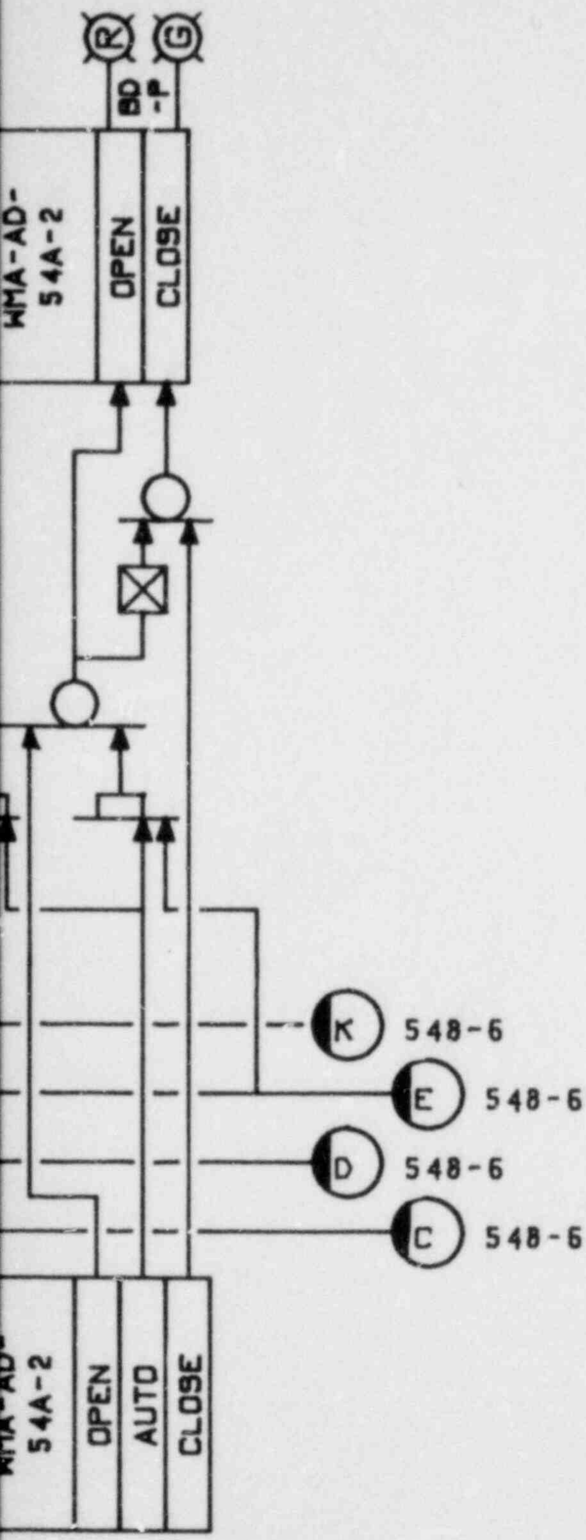


I & C DWG. NO. M-620

SHEET 548-5

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3- 18e

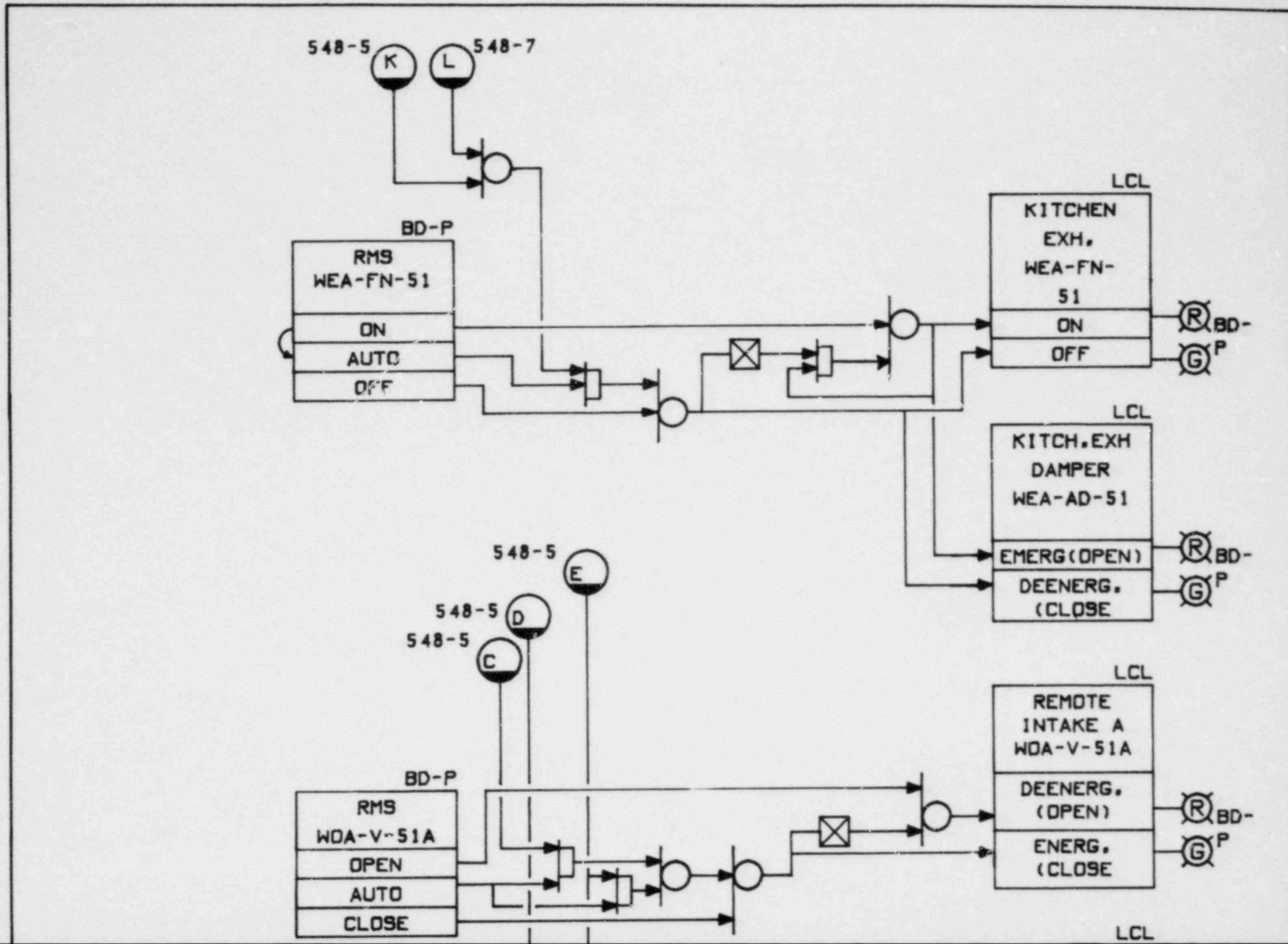

MATCH LINE-SEE SHEET 548-S
54A-1
Z
BD-P
OUTDOOR DAMP. RMS WMA-AD-51A-1
OPEN
AUTO
CLOSE
LCL
OUTDOOR AIR DAMPER WMA-AD-51A-1
OPEN
CLOSE
BD -P
R
G
BD-P
DIVERTING DAMP. RMS WMA-AD-54A-1
OPEN
AUTO
CLOSE
LCL
DIVERTING AIR DAMPER WMA-AD-54A-1
OPEN
CLOSE
BD -P
R
G
BD-P
RECIRC.DAMP RMS
LCL
RECIRC.AIR DAMPER



I & C DWG. NO. M-620

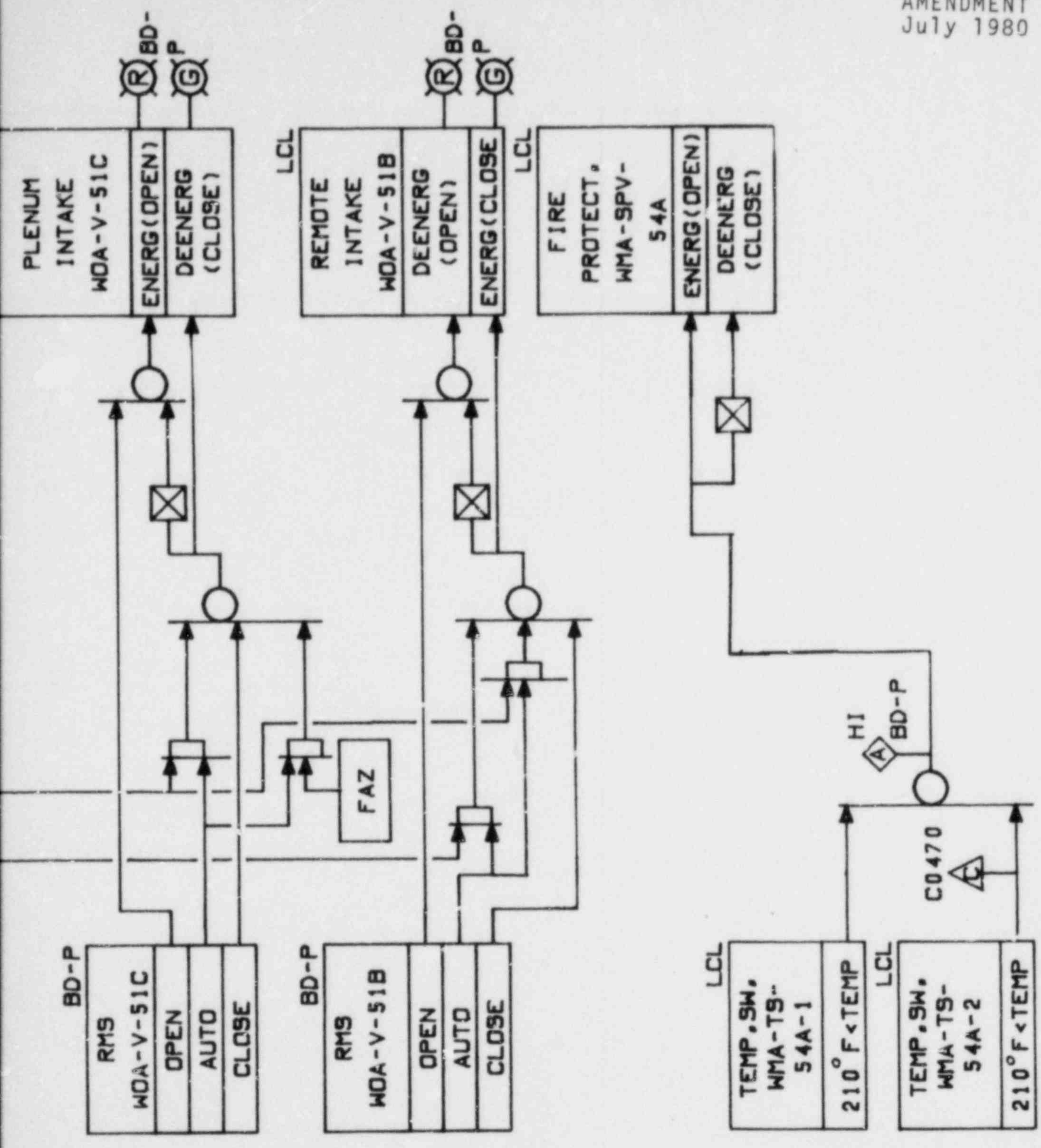
SHEET 548-5A

WASHINGTON PUBLIC POWER SUPPLY S' STEM NUCLEAR PR.JJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3- 18f



548-5
K
L
548-7
BD-P
RMS
WEA-FN-51
ON
AUTO
OFF
LCL
KITCHEN
EXH,
WEA-FN-
51
ON
OFF
R
G
BD-
P
LCL
KITCH.EXH
DAMPER
WEA-AD-51
EMERG(OPEN)
DEENERG.
(CLOSE
R
G
BD-
P
548-5
E
548-5
D
548-5
C
BD-P
RMS
WOA-V-51A
OPEN
AUTO
CLOSE
LCL
REMOTE
INTAKE A
WOA-V-51A
DEENERG.
(OPEN)
ENERG.
(CLOSE
R
G
BD-
P
LCL

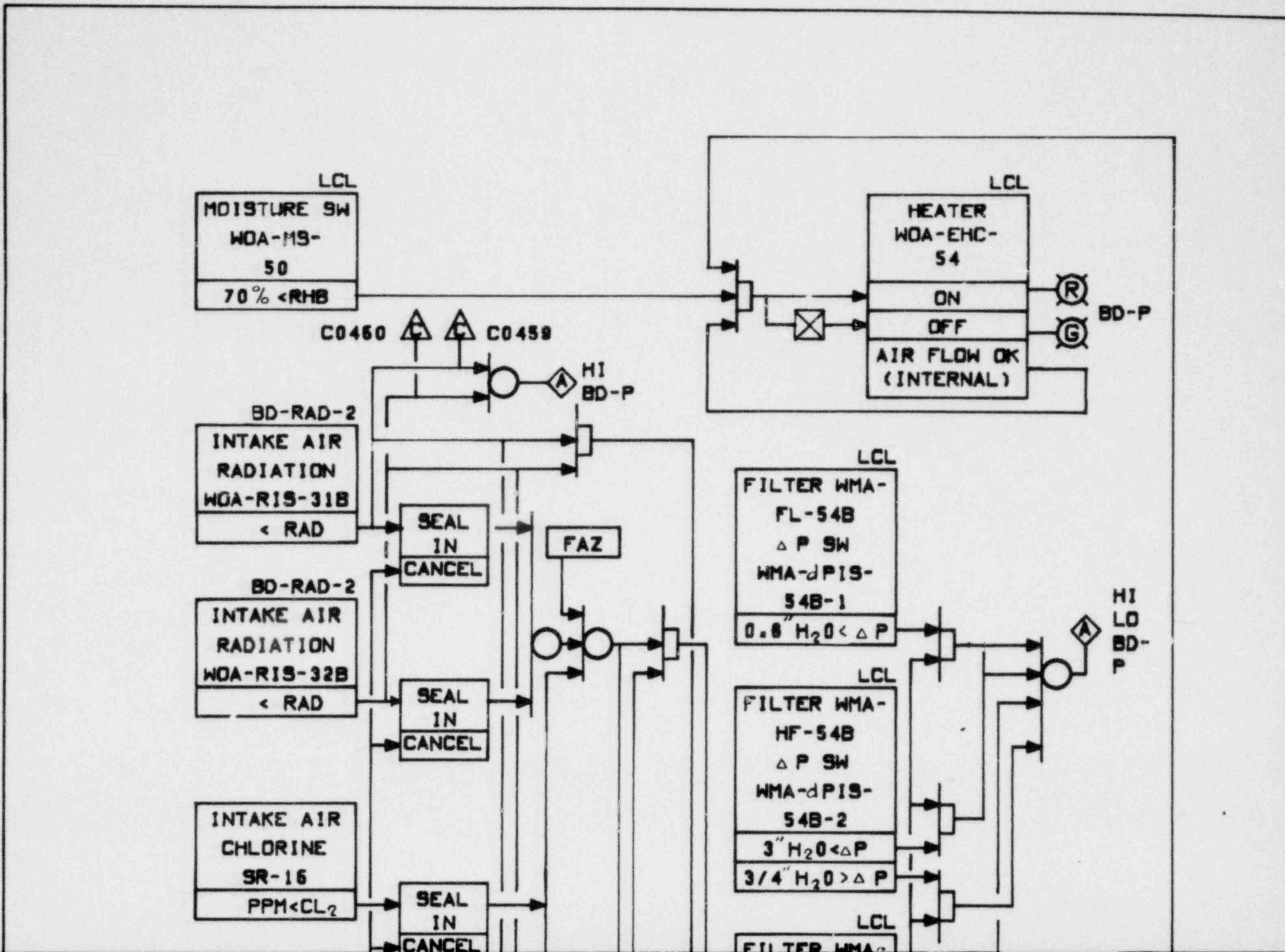
AMENDMENT NO. 10
July 1980



I & C DWG. NO. M-620

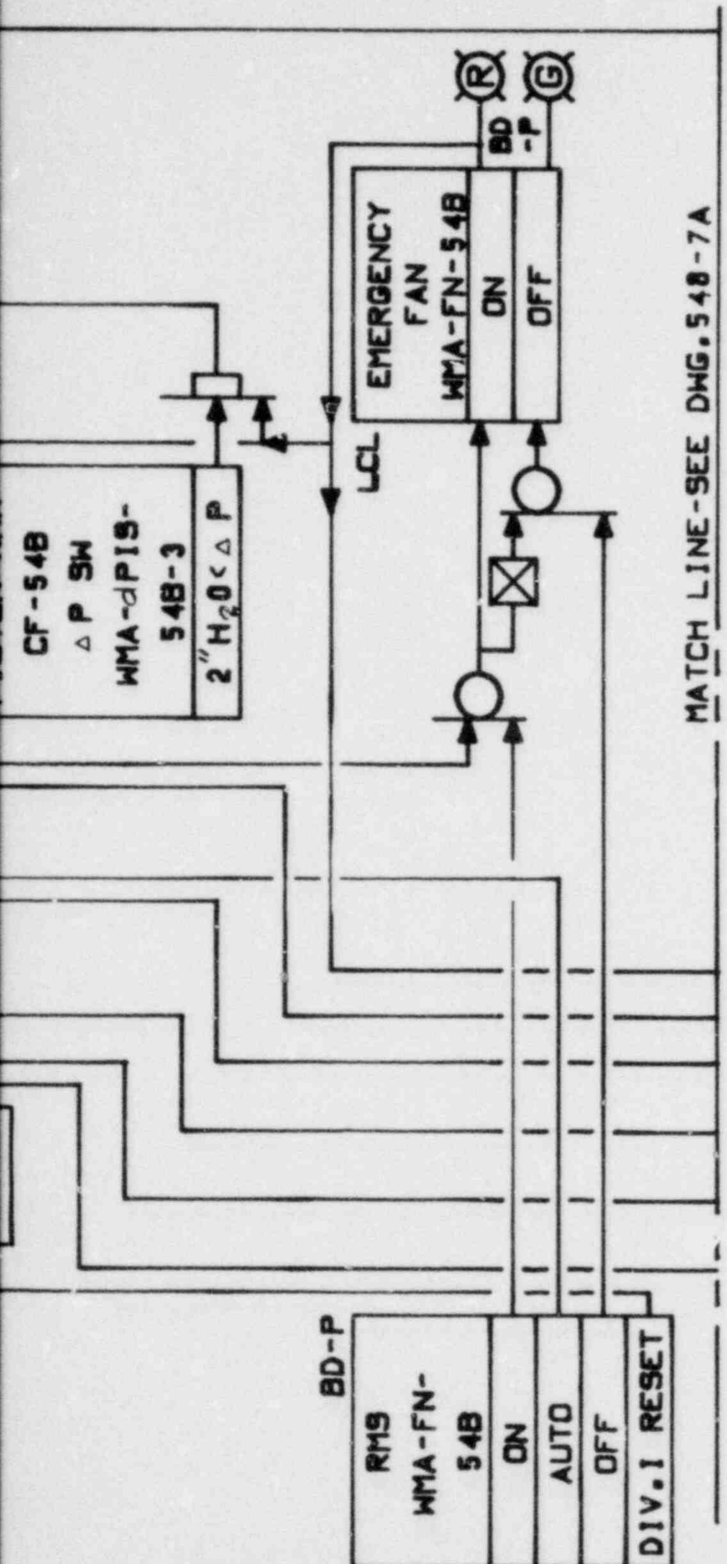
SHEET 548-6

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3- 18g

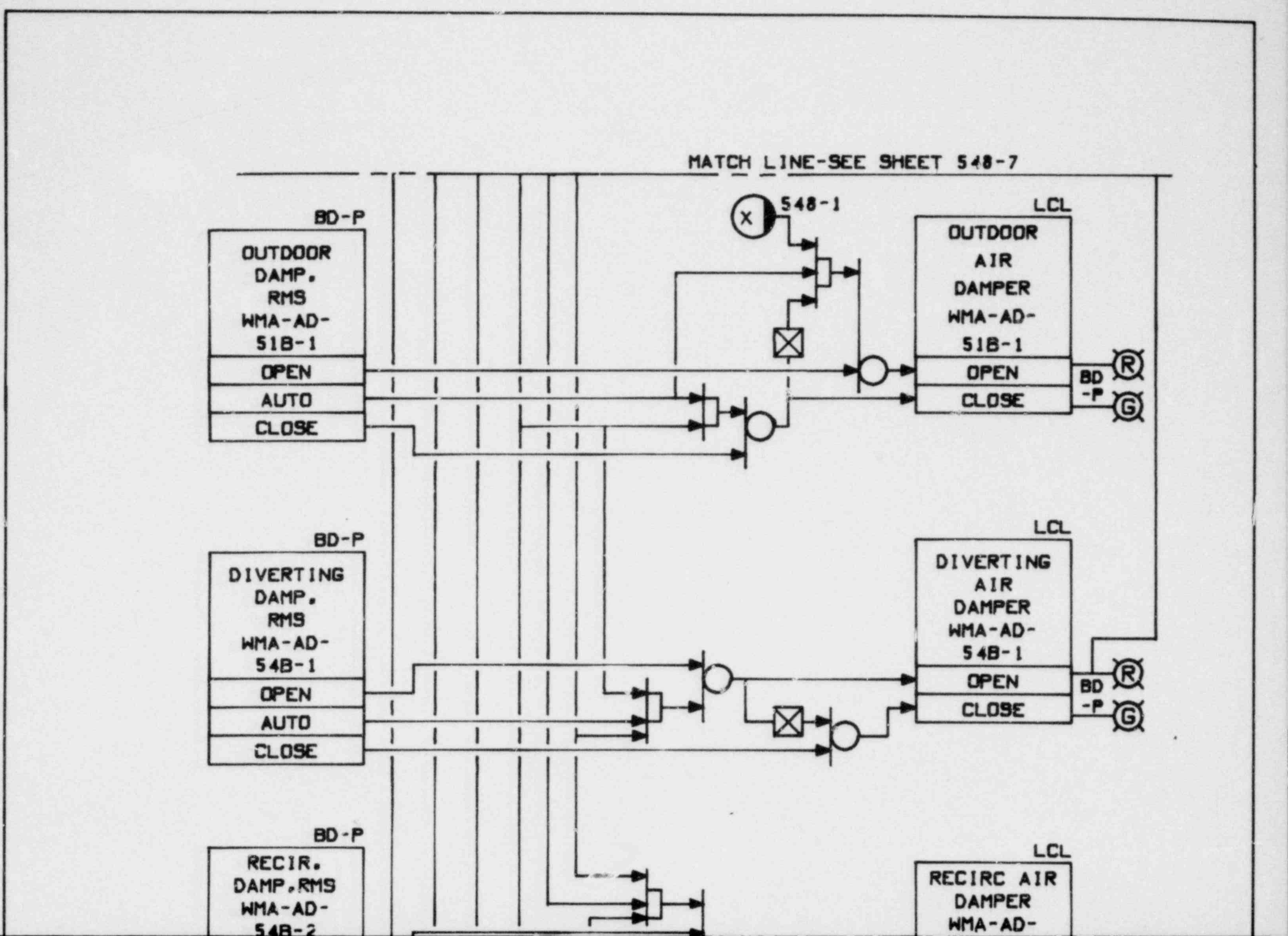


LCL
MOISTURE SW
WOA-MS-
50
70% <RHB
LCL
HEATER
WOA-EHC-
54
ON
OFF
AIR FLOW OK
(INTERNAL)
BD-P
C0460
C0459
HI
BD-P
BD-RAD-2
INTAKE AIR
RADIATION
WOA-RIS-31B
< RAD
SEAL
IN
CANCEL
FAZ
BD-RAD-2
INTAKE AIR
RADIATION
WOA-RIS-32B
< RAD
SEAL
IN
CANCEL
INTAKE AIR
CHLORINE
SR-16
PPM<CL2
SEAL
IN
CANCEL
LCL
FILTER WMA-
FL-54B
ΔP SW
WMA-dPIS-
54B-1
0.8" H2O < ΔP
HI
LO
BD-
P
LCL
FILTER WMA-
HF-54B
ΔP SW
WMA-dPIS-
54B-2
3" H2O<ΔP
3/4" H2O > ΔP
LCL

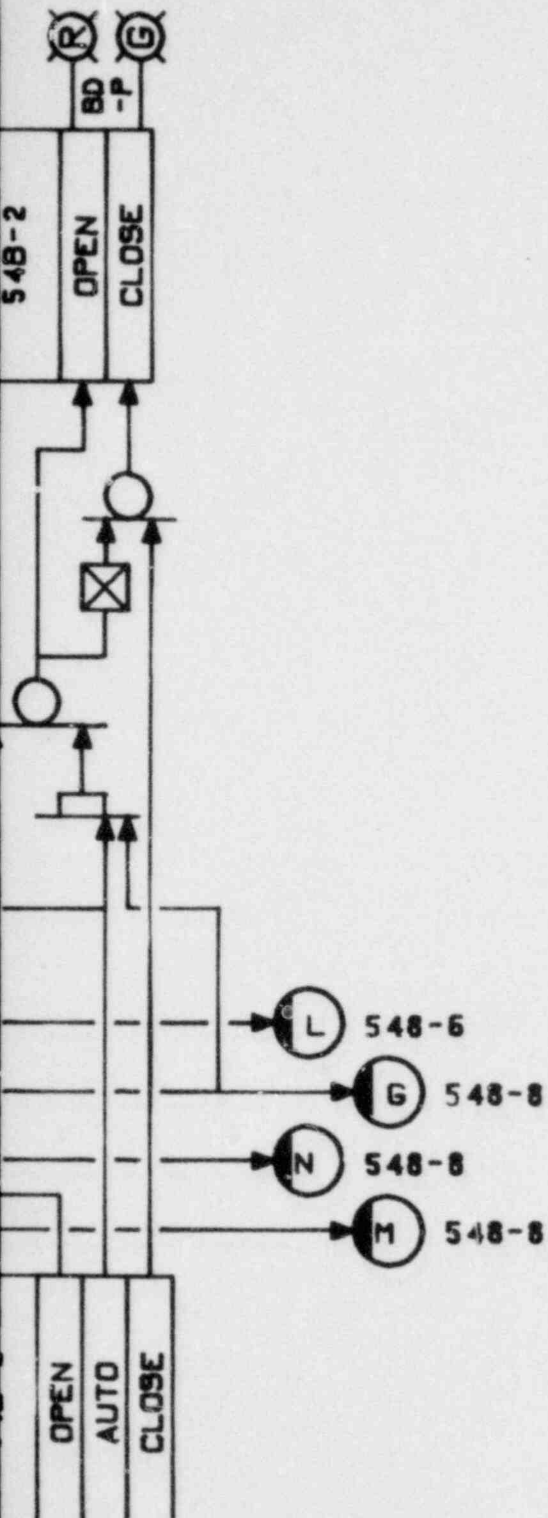
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3-18h


MATCH LINE-SEE SHEET 548-7
548-1
BD-P
OUTDOOR DAMP. RMS WMA-AD-51B-1
OPEN
AUTO
CLOSE
LCL
OUTDOOR AIR DAMPER WMA-AD-51B-1
OPEN
CLOSE
BD -P
R
G
X
BD-P
DIVERTING DAMP. RMS WMA-AD-54B-1
OPEN
AUTO
CLOSE
LCL
DIVERTING AIR DAMPER WMA-AD-54B-1
OPEN
CLOSE
BD -P
R
G
BD-P
RECIR. DAMP.,RMS WMA-AD-54B-2
LCL
RECIRC AIR DAMPER WMA-AD-

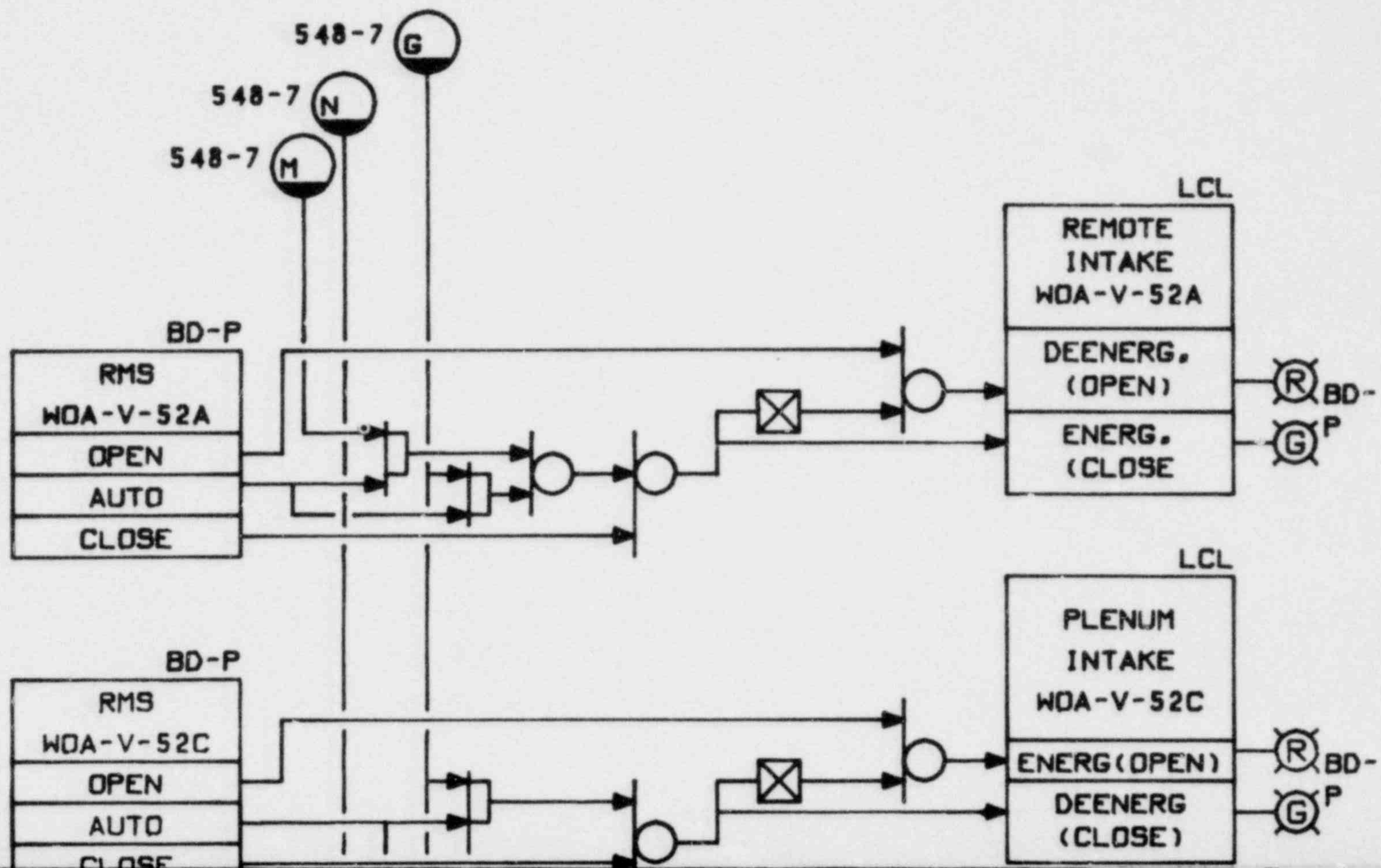
AMENDMENT NO. 10
July 1980



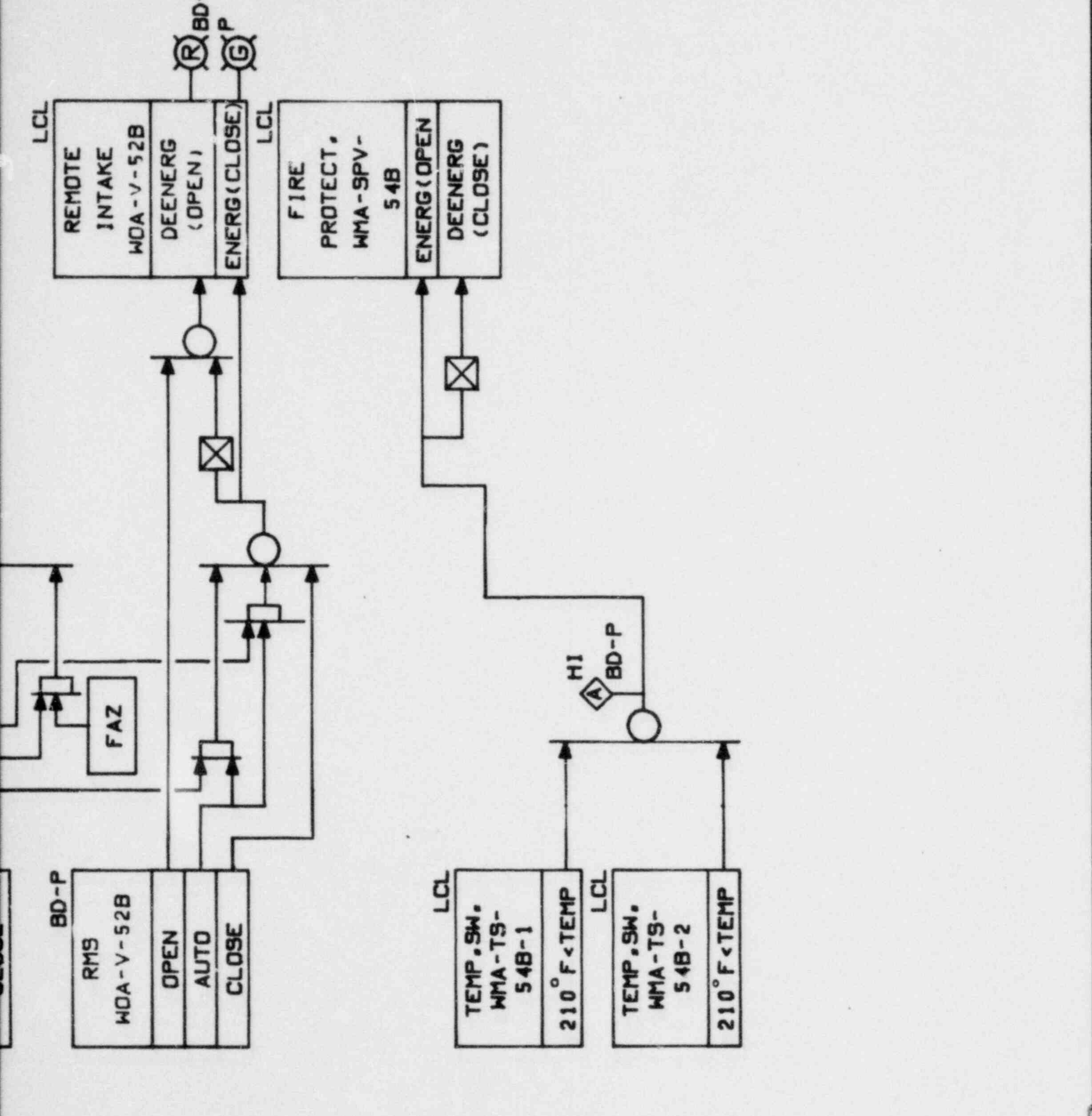
I & C DWG. NO. M-620

SHEET 548-7A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3-18i


548-7 G
548-7 N
548-7 M
BD-P
RMS
WOA-V-52A
OPEN
AUTO
CLOSE
LCL
REMOTE
INTAKE
WOA-V-52A
DEENERG.
(OPEN)
ENERG.
(CLOSE
R
G
BD-
P
BD-P
RMS
WOA-V-52C
OPEN
AUTO
LCL
PLENUM
INTAKE
WOA-V-52C
ENERG(OPEN)
DEENERG
(CLOSE)
R
G
BD-
P

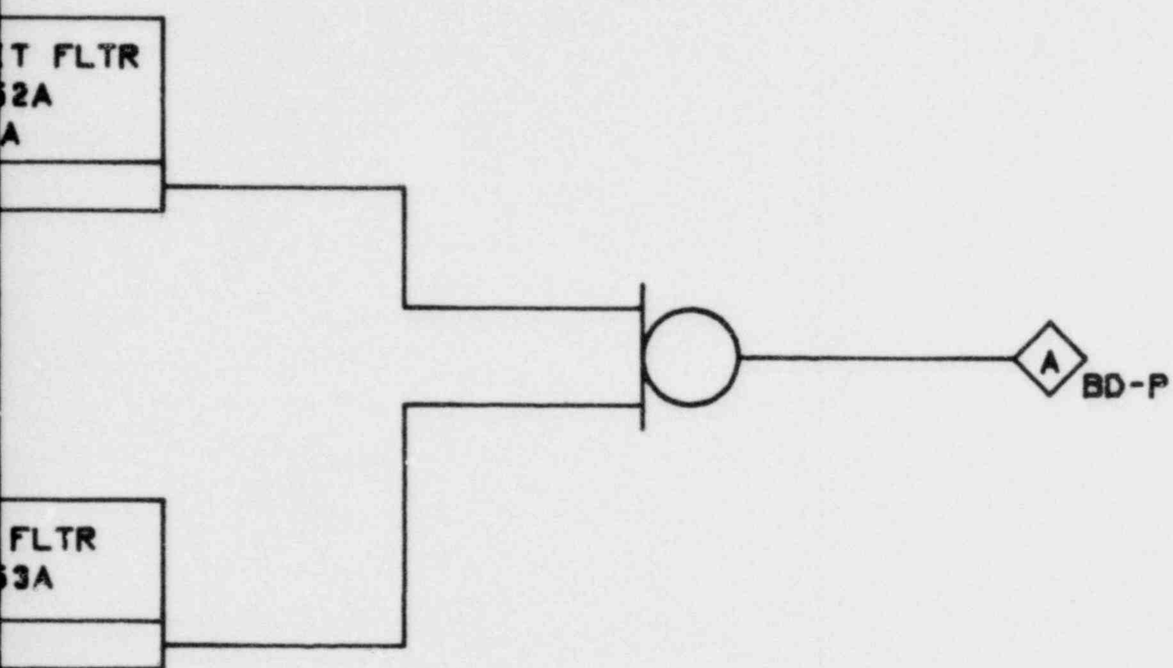
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - HVAC MAIN CONTROL ROOM & CRITICAL SWITCHGEAR	FIGURE 7.3- 18j

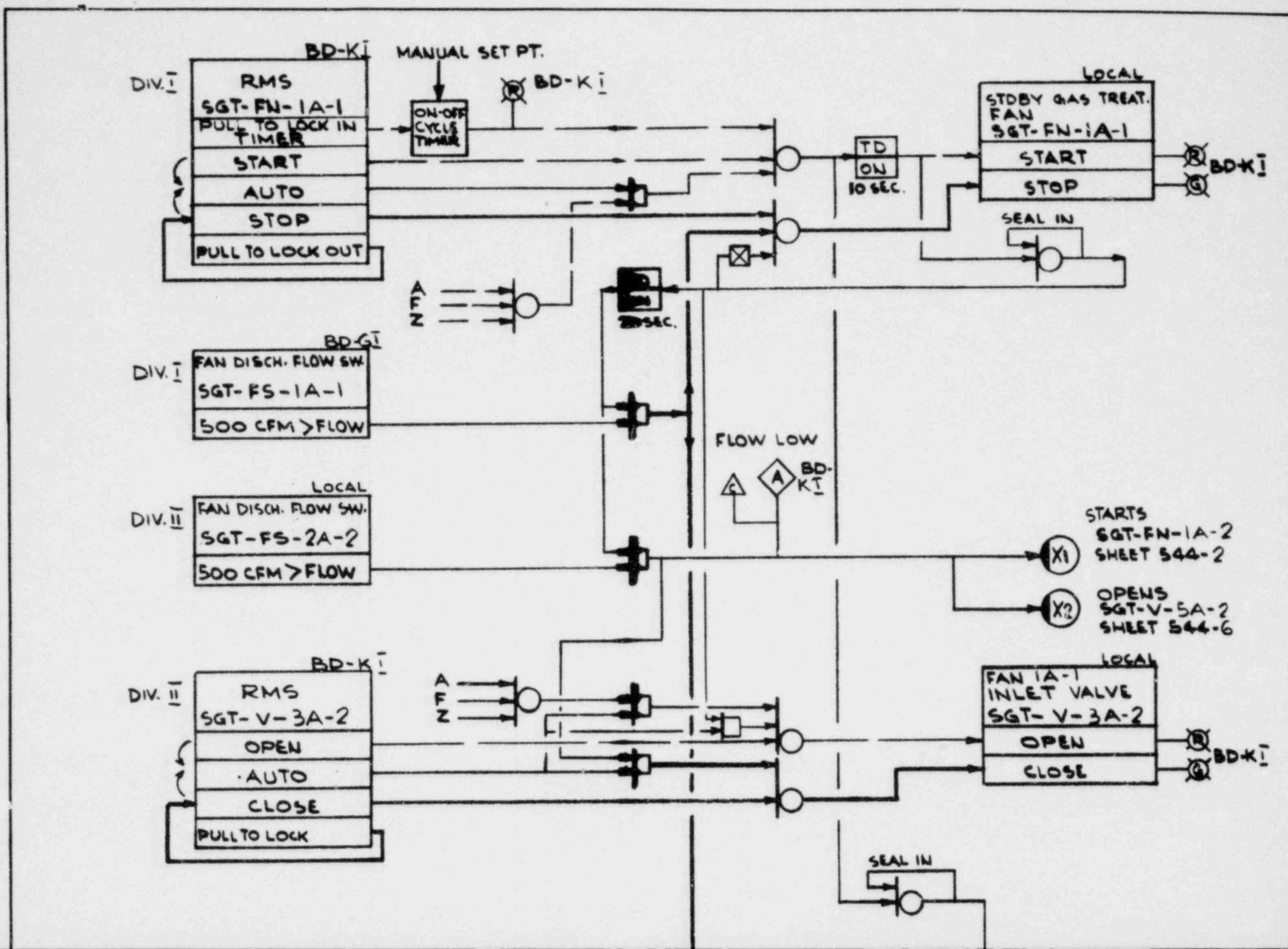
CABLE RM UNI
WMA-FL-5
POS-62
FLTR POS

SWGR UNIT
WMA-FL-5
FLTR POS



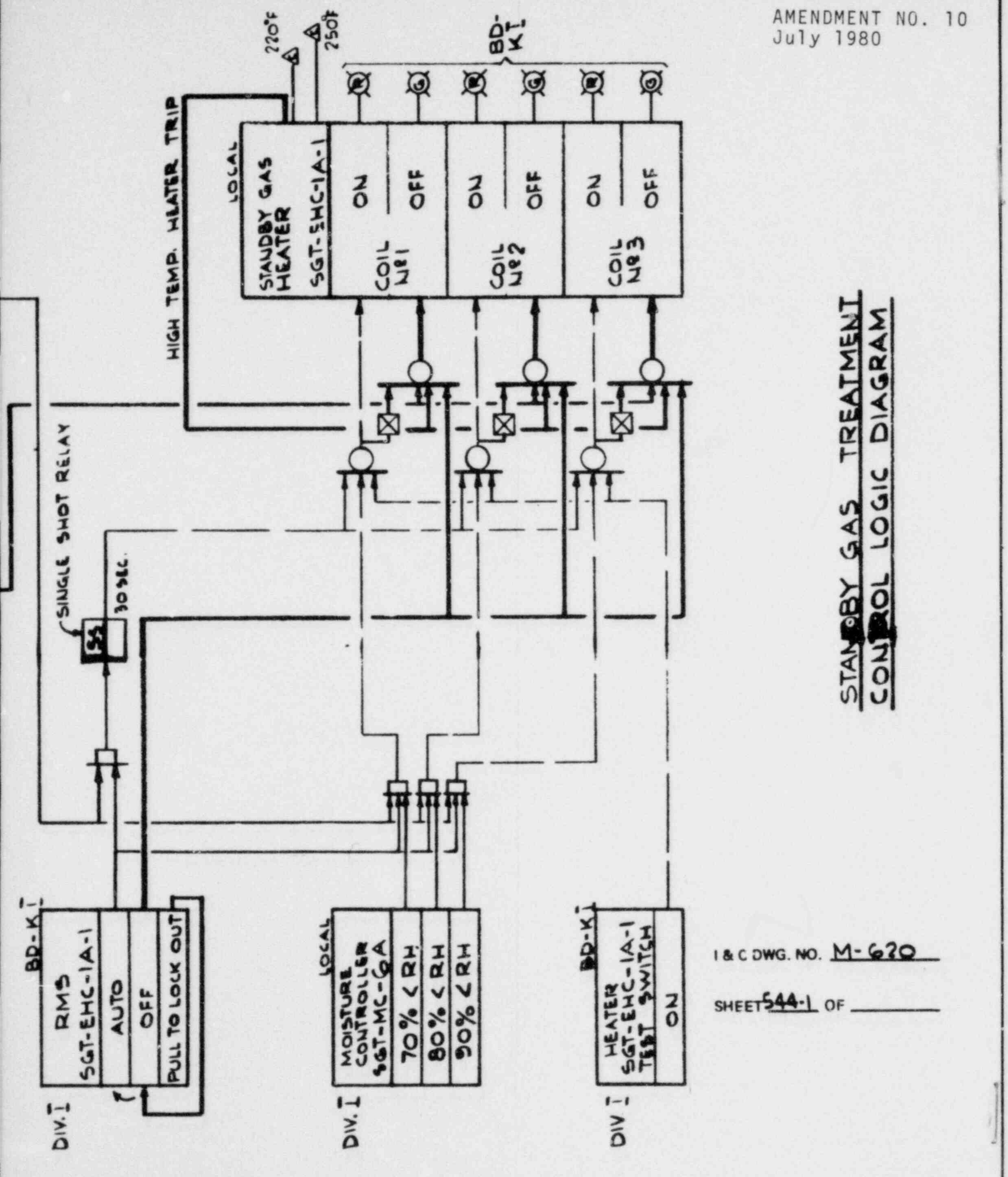
IMILAR LOGIC FOR DIV-II W/SUFFIX-'B'

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FILTER RUNOUT ALARM CONTROL LOGIC DIAGRAM	FIGURE 7.3- 18k

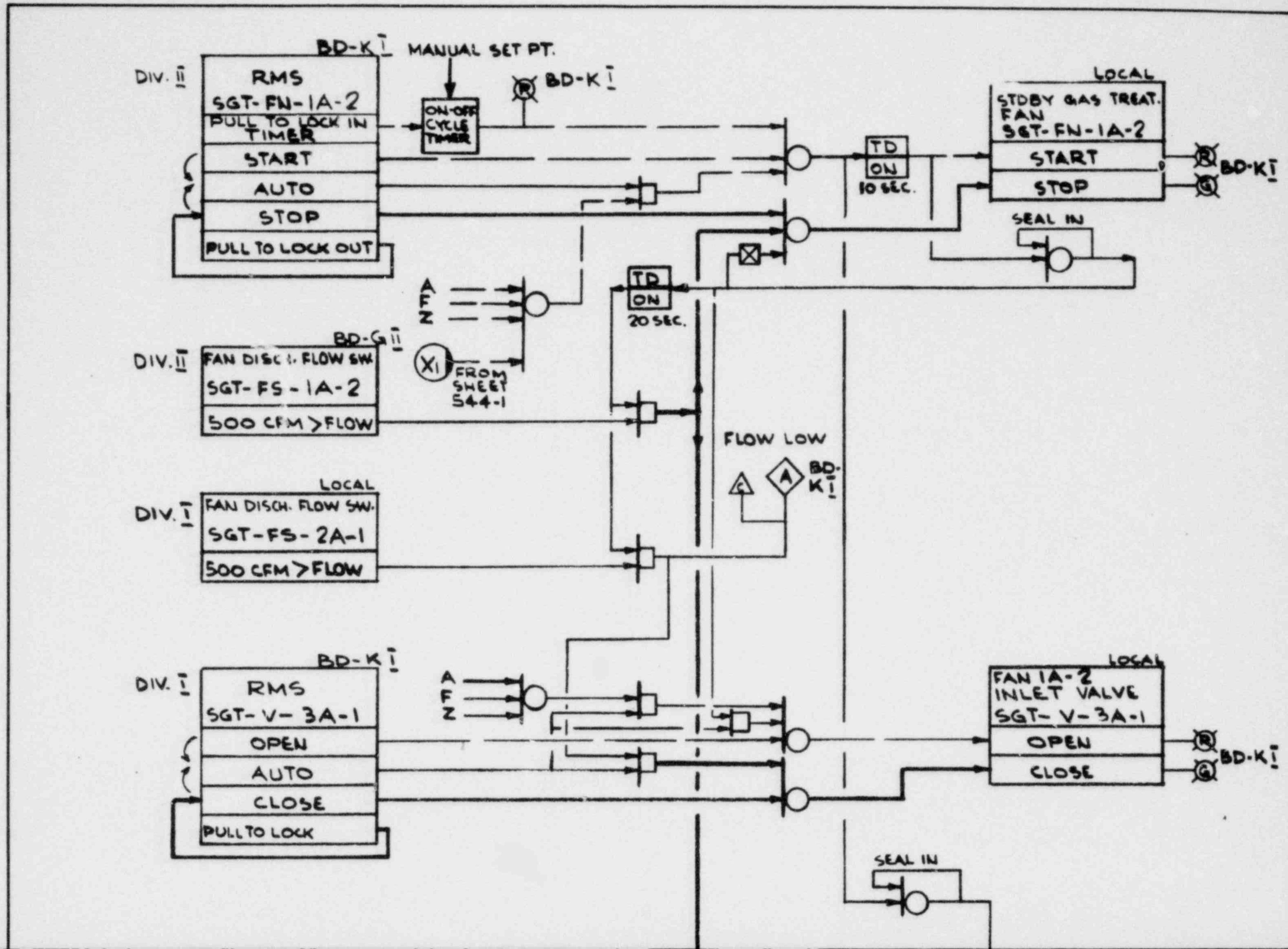


DIV. I
BD-K I
RMS
SGT-FN-1A-1
PULL TO LOCK IN TIMER
START
AUTO
STOP
PULL TO LOCK OUT
MANUAL SET PT.
ON-OFF CYCLE TIMER
BD-K I
TD ON
10 SEC.
LOCAL
STDBY GAS TREAT. FAN
SGT-FN-1A-1
START
STOP
BD-K I
SEAL IN
A
F
Z
DIV. I
BD-G I
FAN DISCH. FLOW SW.
SGT-FS-1A-1
500 CFM > FLOW
FLOW LOW
BD-K I
DIV. II
LOCAL
FAN DISCH. FLOW SW.
SGT-FS-2A-2
500 CFM > FLOW
X1
STARTS
SGT-FN-1A-2
SHEET 544-2
X2
OPENS
SGT-V-5A-2
SHEET 544-6
DIV. II
BD-K I
RMS
SGT-V-3A-2
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
FAN 1A-1
INLET VALVE
SGT-V-3A-2
OPEN
CLOSE
BD-K I
SEAL IN

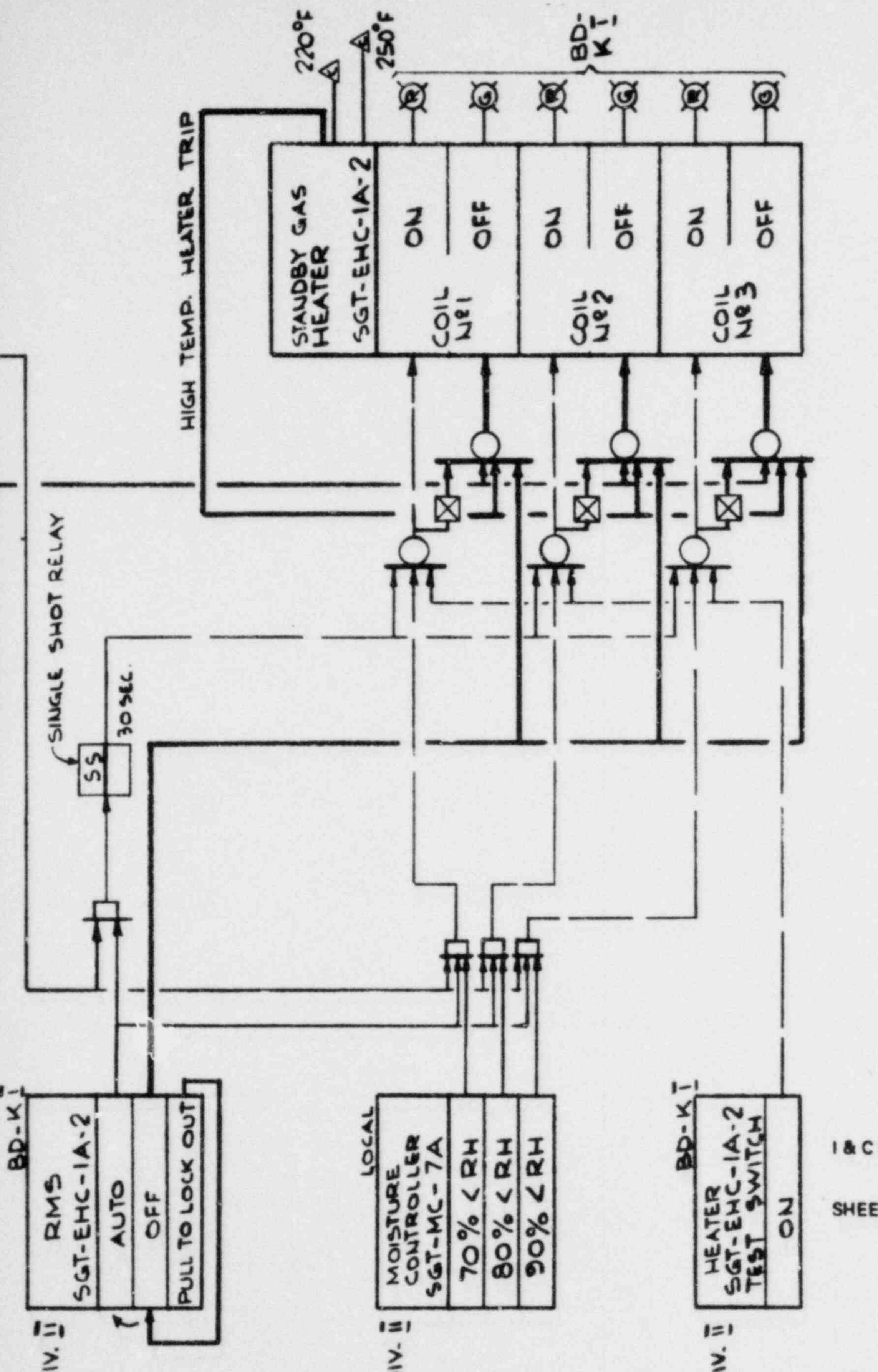
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SGTS	FIGURE 7.3-19a


DIV. II
BD-K I
RMS
SGT-FN-1A-2
PULL TO LOCK IN TIMER
START
AUTO
STOP
PULL TO LOCK OUT
MANUAL SET PT.
ON-OFF CYCLE TIMER
BD-K I
TD ON 10 SEC.
LOCAL
STDBY GAS TREAT. FAN SGT-FN-1A-2
START
STOP
BD-K I
SEAL IN
A
F
Z
TD ON 20 SEC.
DIV. II
BD-G II
FAN DISC. FLOW SW.
SGT-FS-1A-2
500 CFM > FLOW
X1
FROM SHEET 544-1
FLOW LOW
A
BD-K I
DIV. I
LOCAL
FAN DISCH. FLOW SW.
SGT-FS-2A-1
500 CFM > FLOW
DIV. I
BD-K I
RMS
SGT-V-3A-1
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
FAN 1A-2 INLET VALVE
SGT-V-3A-1
OPEN
CLOSE
BD-K I
SEAL IN

AMENDMENT NO. 10
July 1980

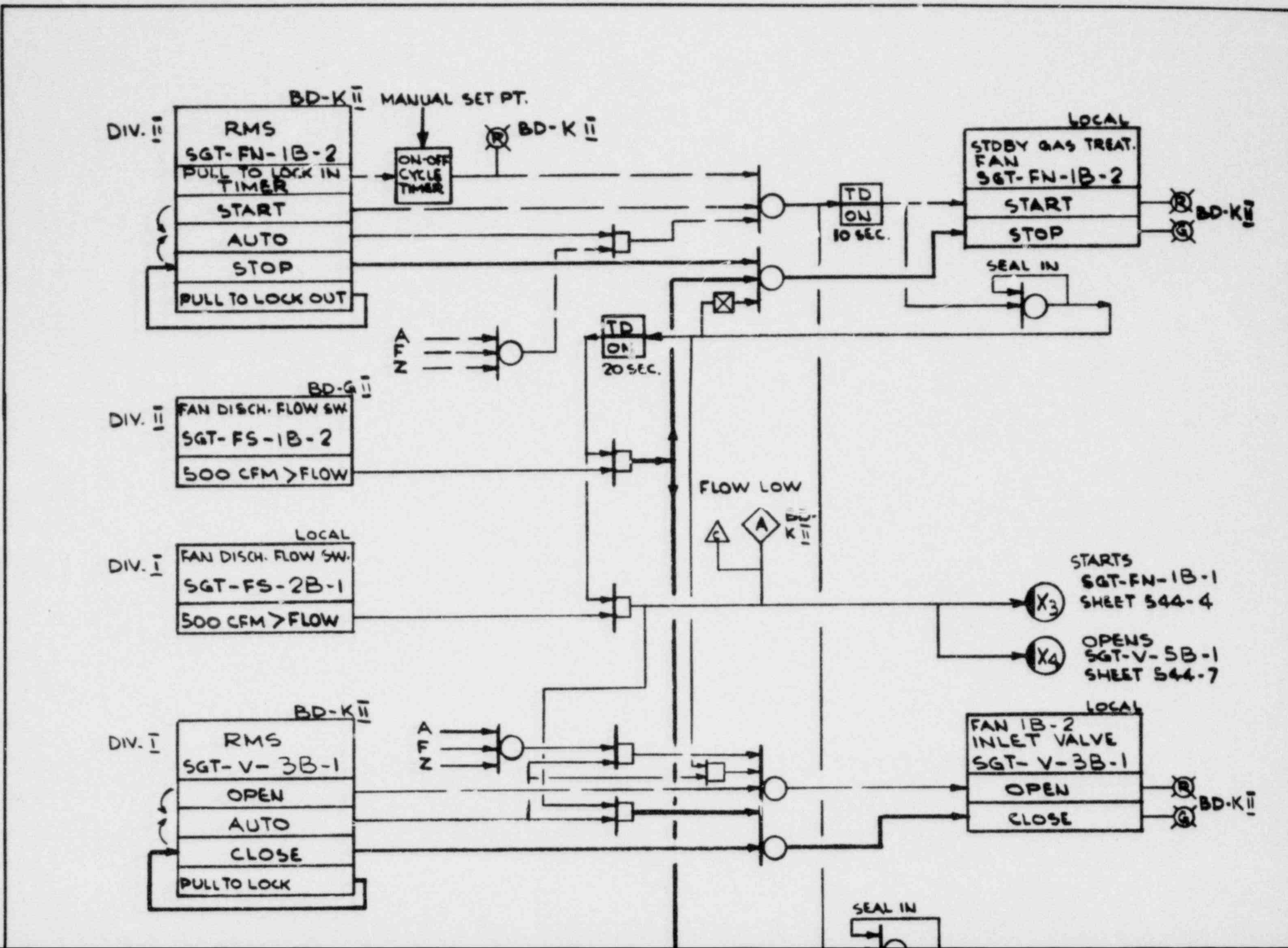


STANDBY GAS TREATMENT
CONTROL LOGIC DIAGRAM

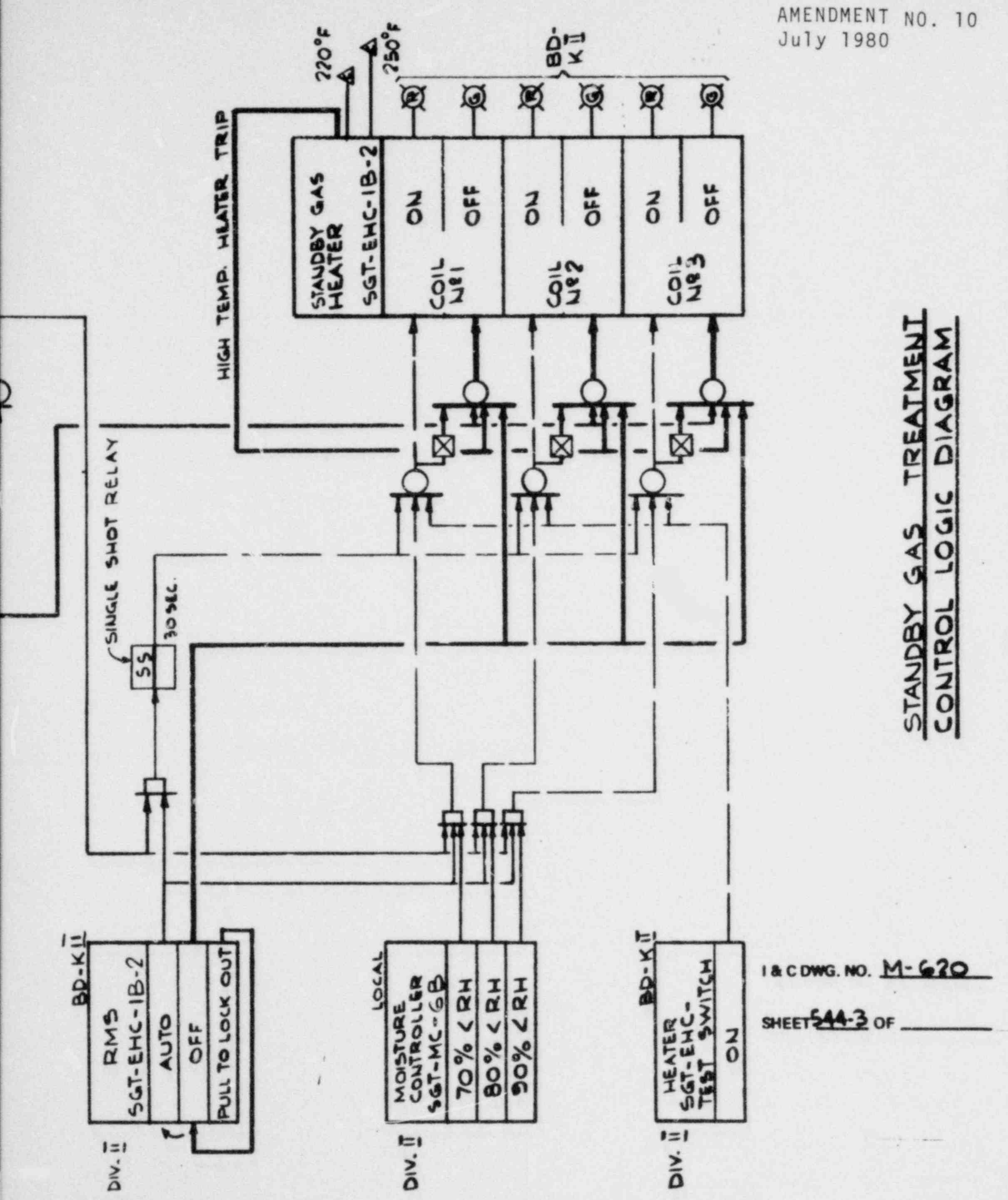
I & C DWG. NO. M-620

SHEET 544-2 OF

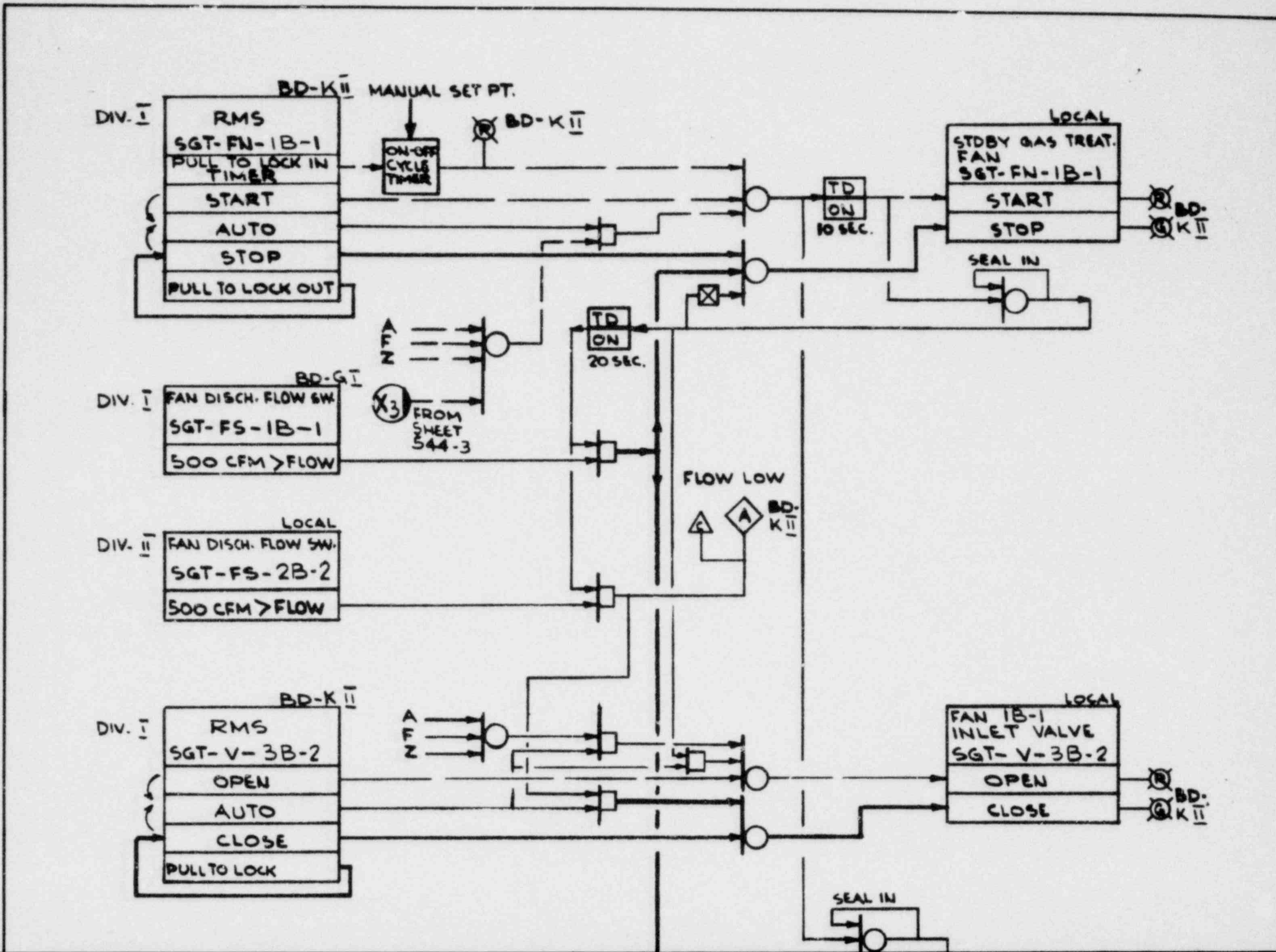
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SGTS	FIGURE 7.3-19b


DIV. II
RMS
SGT-FN-1B-2
PULL TO LOCK IN TIMER
START
AUTO
STOP
PULL TO LOCK OUT
BD-K II
MANUAL SET PT.
ON-OFF CYCLE TIMER
BD-K II
TD
ON
10 SEC.
LOCAL
STDBY GAS TREAT. FAN
SGT-FN-1B-2
START
STOP
BD-K II
SEAL IN
A
F
Z
TD
ON
20 SEC.
DIV. II
BD-G II
FAN DISCH. FLOW SW.
SGT-FS-1B-2
500 CFM > FLOW
FLOW LOW
A
BD-K II
DIV. I
LOCAL
FAN DISCH. FLOW SW.
SGT-FS-2B-1
500 CFM > FLOW
STARTS SGT-FN-1B-1 SHEET 544-4
OPENS SGT-V-5B-1 SHEET 544-7
DIV. I
BD-K II
RMS
SGT-V-3B-1
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
FAN 1B-2 INLET VALVE
SGT-V-3B-1
OPEN
CLOSE
BD-K II
SEAL IN

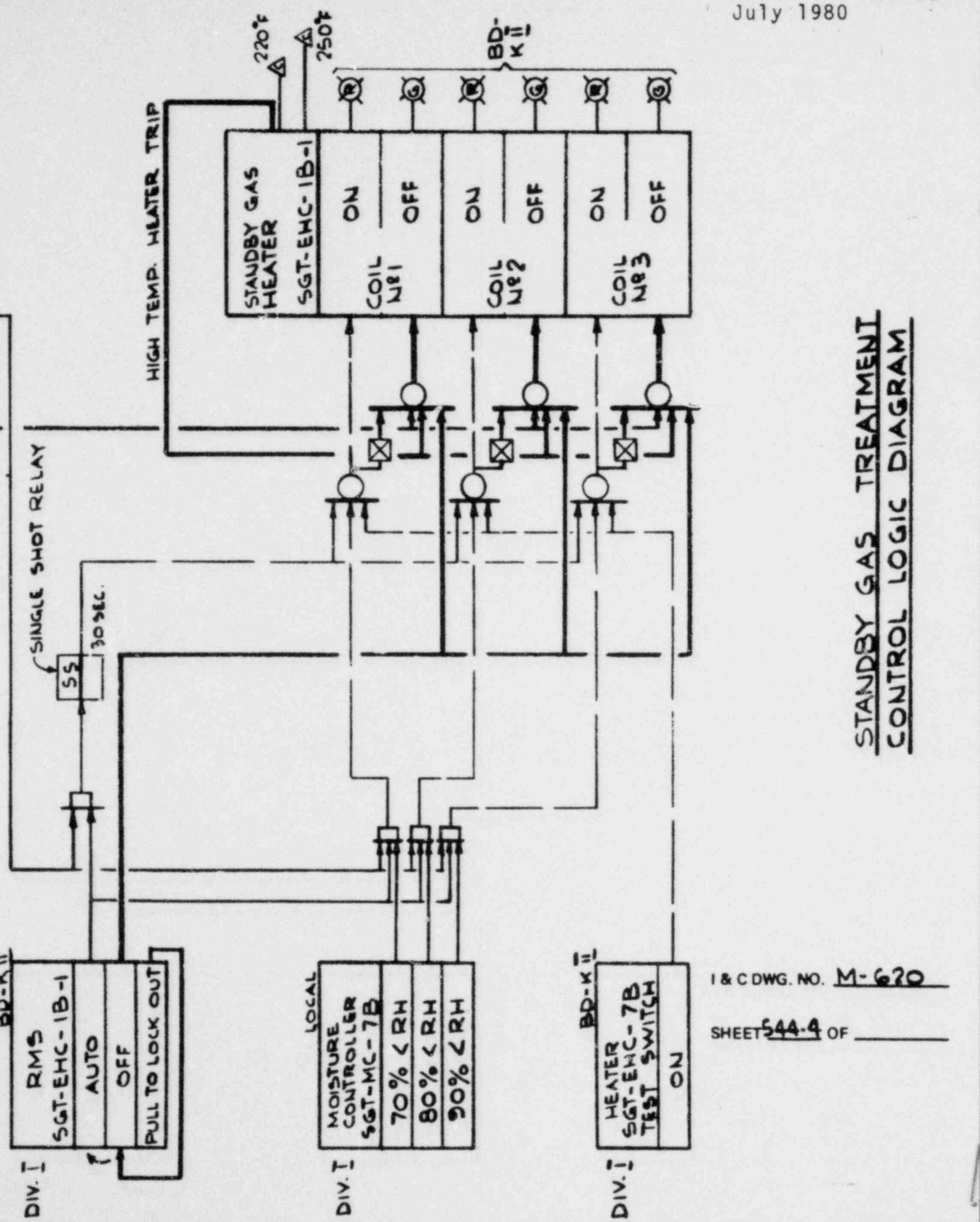
AMENDMENT NO. 10
July 1980



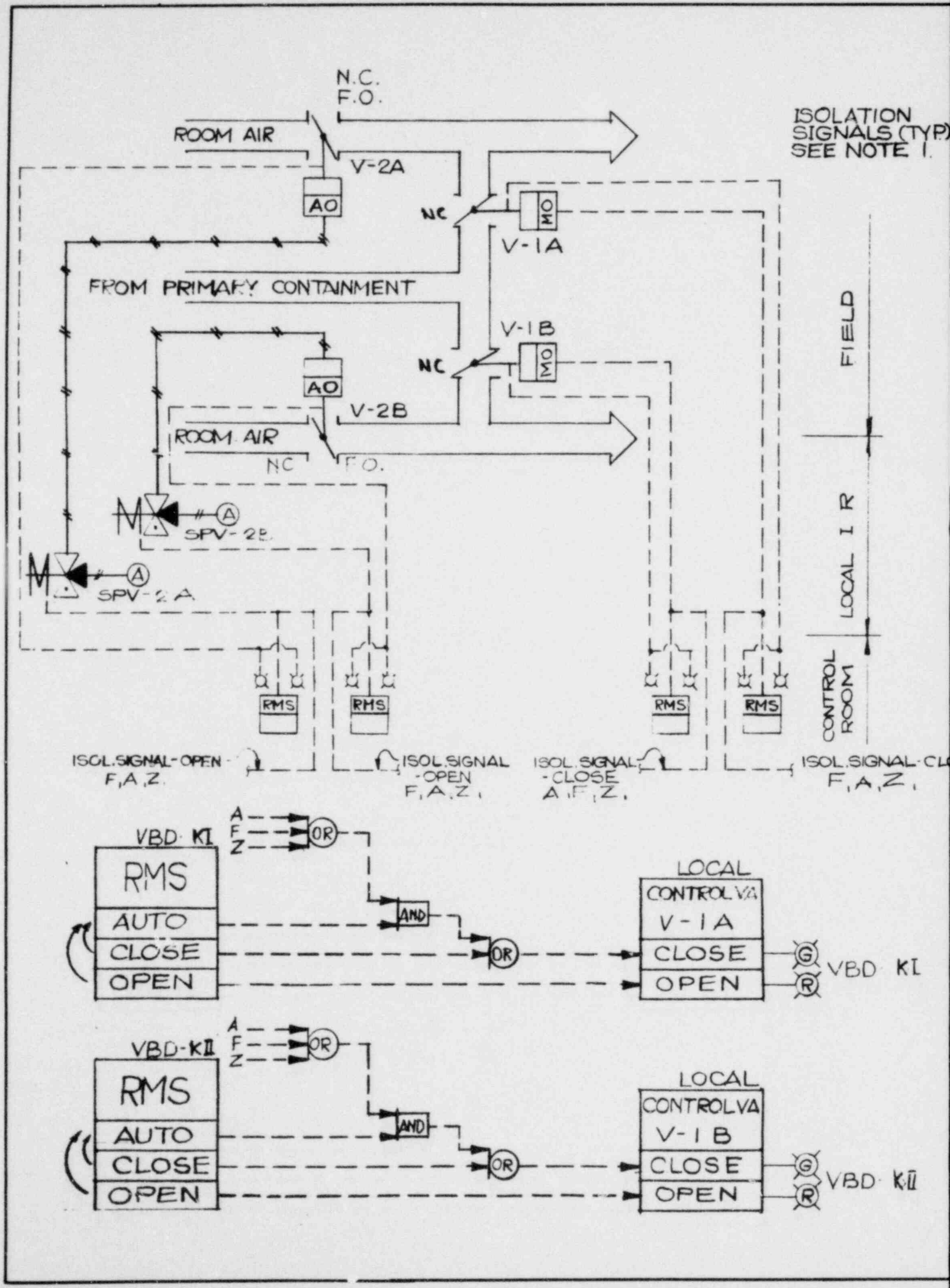
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SGTS	FIGURE 7.3-19c


DIV. I
BD-K II
RMS
SGT-FN-1B-1
PULL TO LOCK IN TIMER
START
AUTO
STOP
PULL TO LOCK OUT
MANUAL SET PT.
ON-OFF CYCLE TIMER
BD-K II
TD
ON
10 SEC.
LOCAL
STDBY GAS TREAT. FAN
SGT-FN-1B-1
START
STOP
BD-K II
SEAL IN
A
F
Z
TD
ON
20 SEC.
DIV. I
BD-G I
FAN DISCH. FLOW SW.
SGT-FS-1B-1
500 CFM > FLOW
X3
FROM SHEET 544-3
FLOW LOW
BD-K II
DIV. II
LOCAL
FAN DISCH. FLOW SW.
SGT-FS-2B-2
500 CFM > FLOW
DIV. I
BD-K II
RMS
SGT-V-3B-2
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
FAN 1B-1
INLET VALVE
SGT-V-3B-2
OPEN
CLOSE
BD-K II
SEAL IN

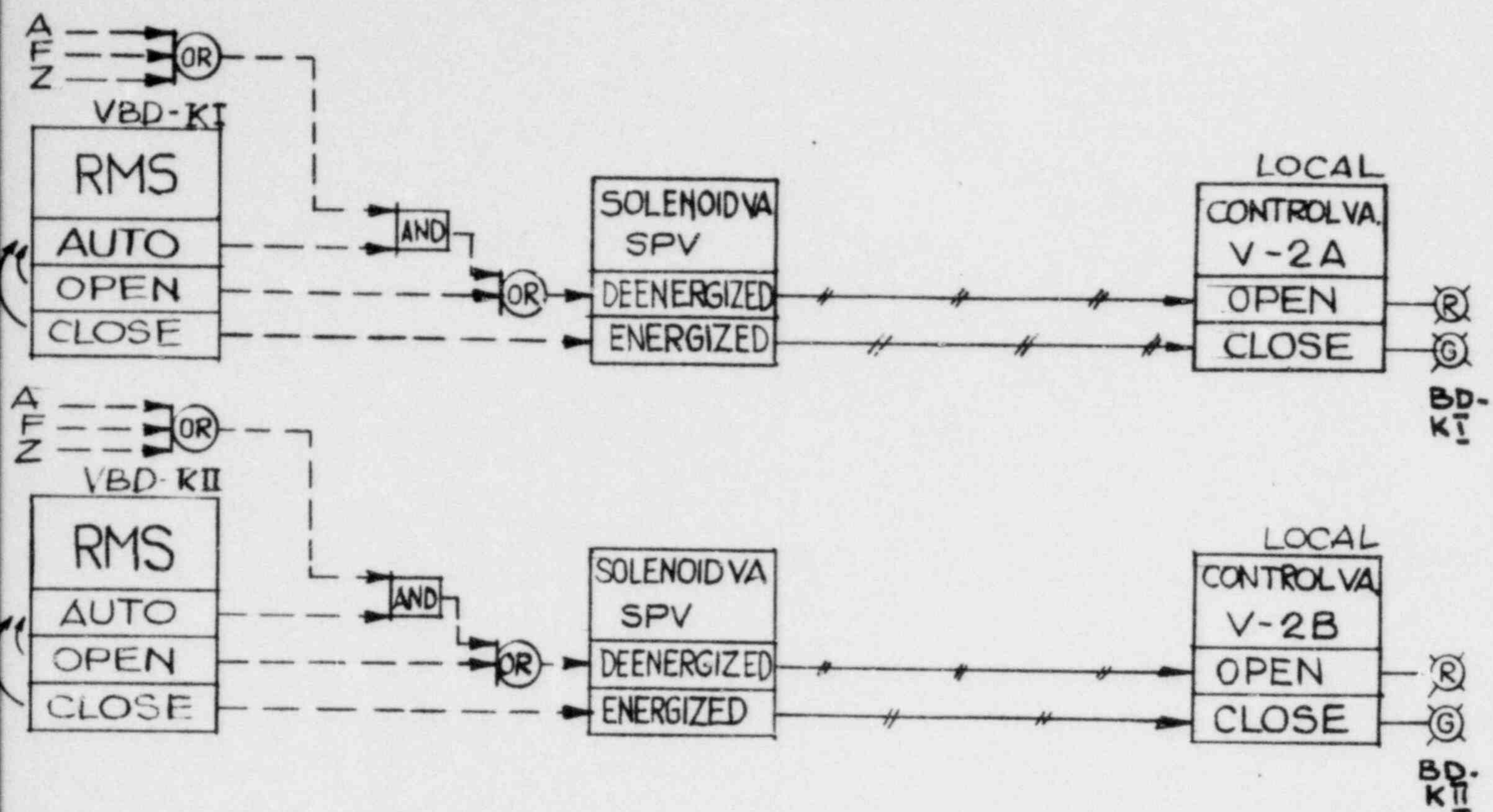
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SGTS	FIGURE 7.3-19d


N.C.
F.O.
ROOM AIR
V-2A
AO
NC
MO
V-1A
ISOLATION
SIGNALS (TYP)
SEE NOTE 1.
FROM PRIMARY CONTAINMENT
V-1B
NC
MO
AO
V-2B
ROOM AIR
NC
F.O.
SPV-2B
SPV-2A
FIELD
LOCAL I R
CONTROL ROOM
RMS
RMS
RMS
RMS
ISOL.SIGNAL-OPEN
F,A,Z.
ISOL.SIGNAL
-OPEN
F,A,Z.
ISOL.SIGNAL
-CLOSE
A,F,Z.
ISOL.SIGNAL-CL
F,A,Z.
A
F
Z
OR
VBD KI
RMS
AUTO
CLOSE
OPEN
AND
OR
LOCAL
CONTROL VA
V-1A
CLOSE
OPEN
G
R
VBD KI
A
F
Z
OR
VBD-KII
RMS
AUTO
CLOSE
OPEN
AND
OR
LOCAL
CONTROL VA
V-1B
CLOSE
OPEN
G
R
VBD KII

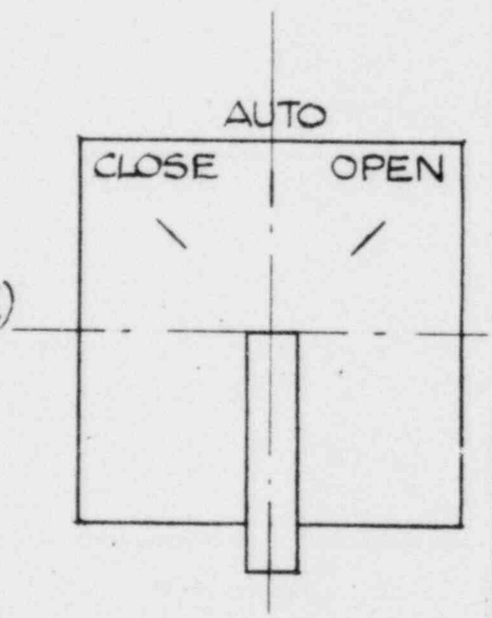
AMENDMENT NO. 10
July 1980



OSE

NOTES:

1. ISOLATION SIGNAL CODES:
   - A- REACTOR VESSEL LOW WATER LEVEL (DIV. I & II)
   - F- HIGH DRYWELL PRESSURE (DIV. I & II)
   - Z- HIGH RADIATION, REACTOR BUILDING VENTILATION SYSTEM (DIV. I & II)
2. ALL RMS ARE SPRING RETURN TO AUTO POSITION (SEE FIG. 1)
3. FOR REFERENCES SEE DWG. # M 544

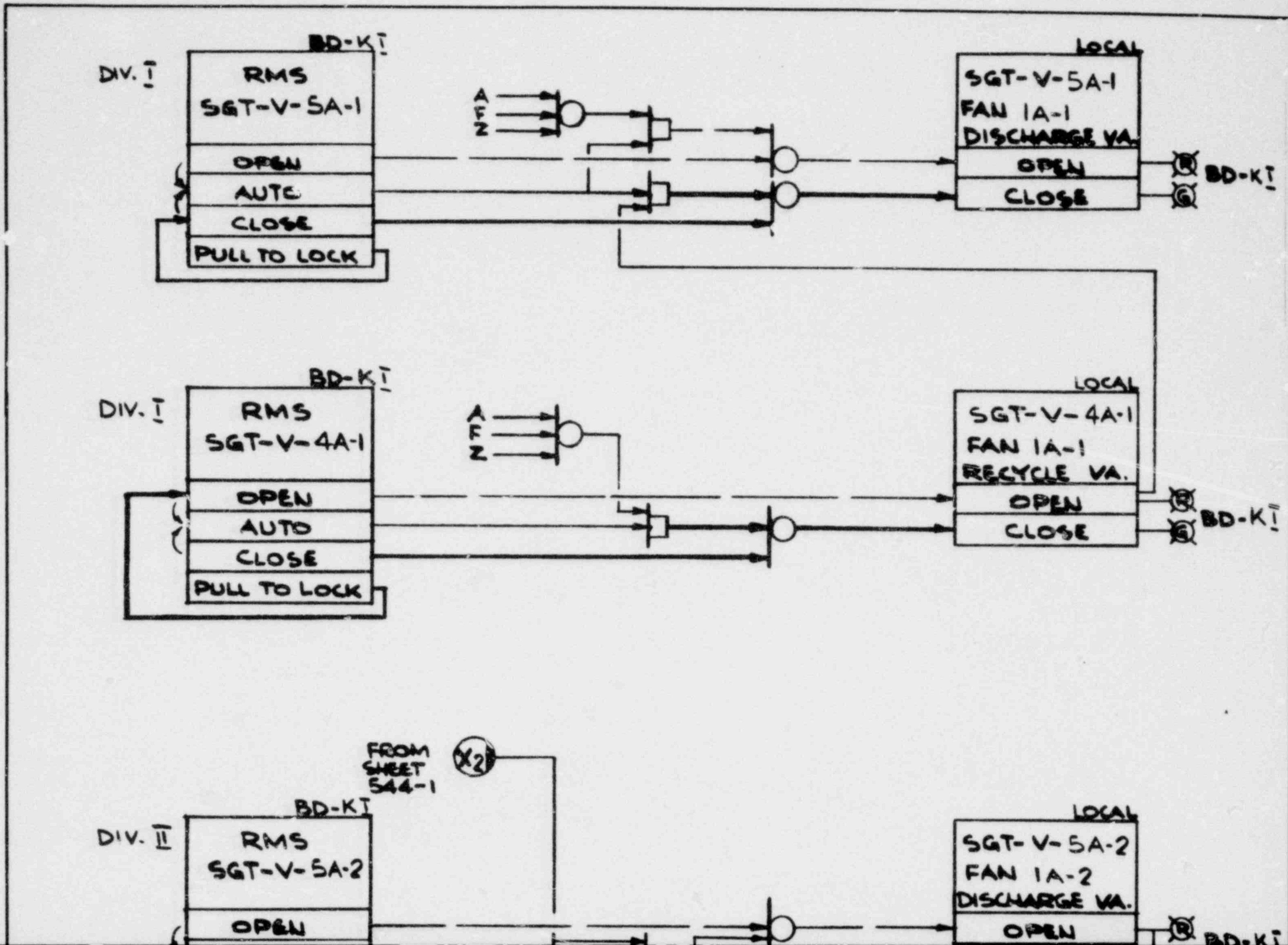


HVAC SGT INLETS

I & C DWG. NO. M-620

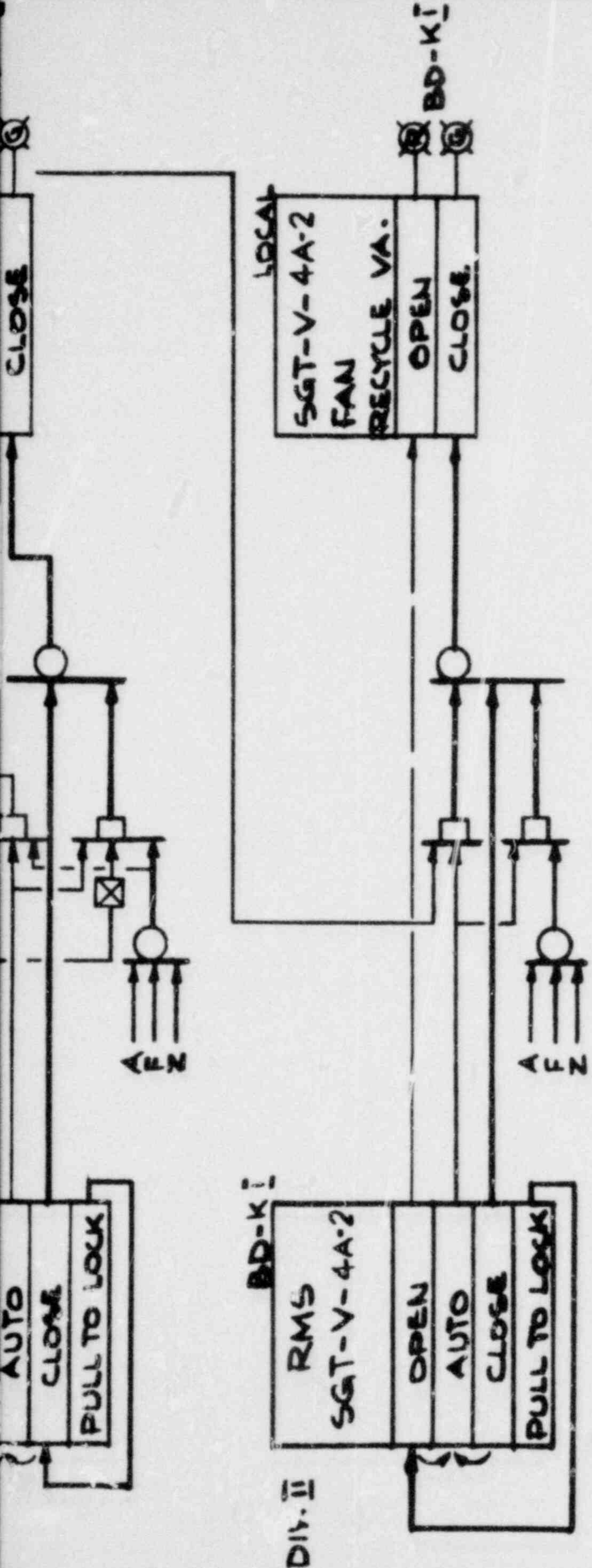
SHEET ~~544~~ 5 OF

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SGTS	FIGURE 7.3-19e



DIV. I
BD-K I
RMS
SGT-V-5A-1
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
SGT-V-5A-1
FAN 1A-1
DISCHARGE VA.
OPEN
CLOSE
R
G
BD-K I
DIV. I
BD-K I
RMS
SGT-V-4A-1
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
SGT-V-4A-1
FAN 1A-1
RECYCLE VA.
OPEN
CLOSE
R
G
BD-K I
FROM SHEET 544-1
X2
DIV. II
BD-K I
RMS
SGT-V-5A-2
OPEN
LOCAL
SGT-V-5A-2
FAN 1A-2
DISCHARGE VA.
OPEN
R
BD-K I

AMENDMENT NO. 10
July 1980

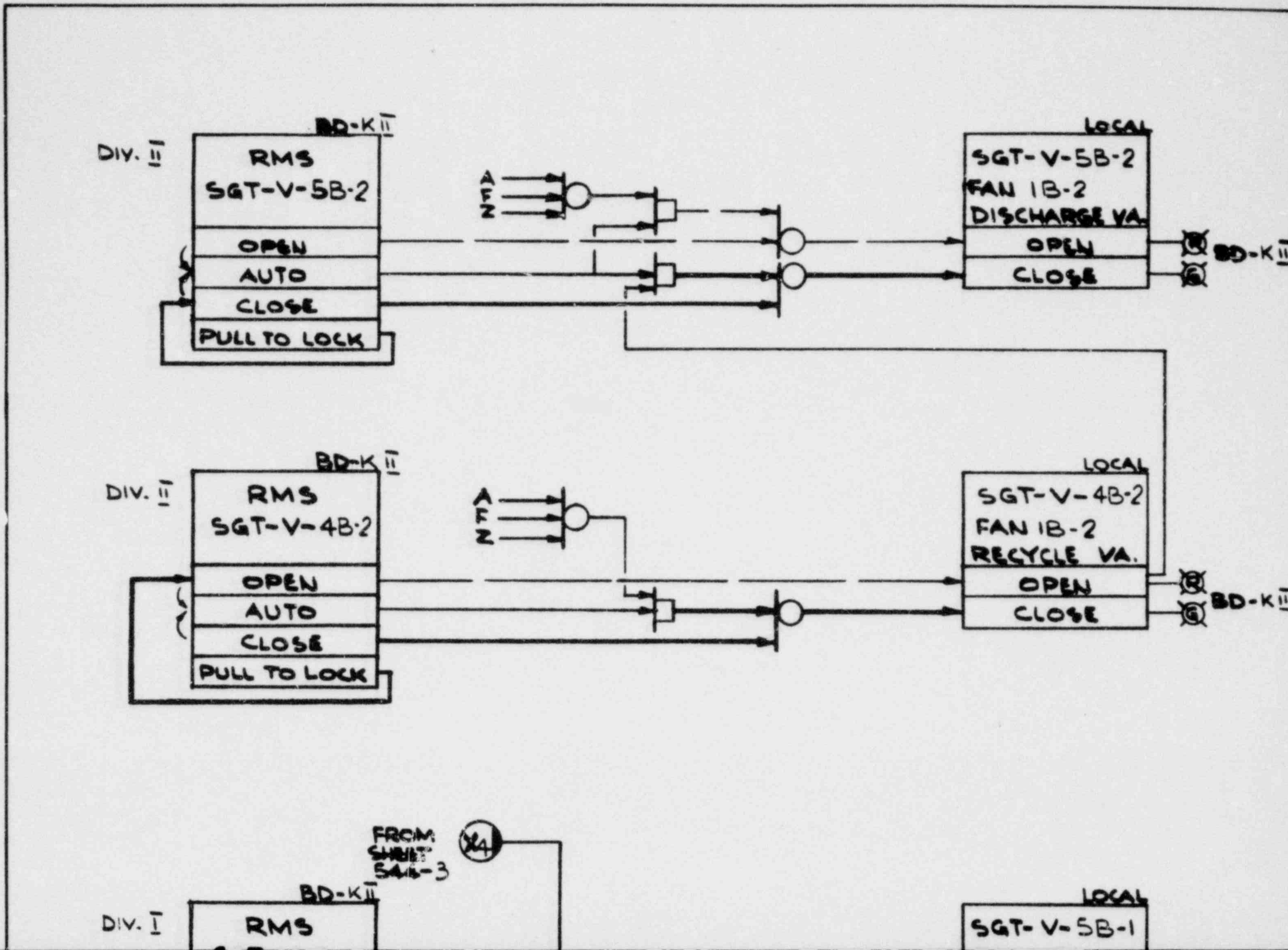


STANDBY GAS TREATMENT
CONTROL LOGIC DIAGRAM.

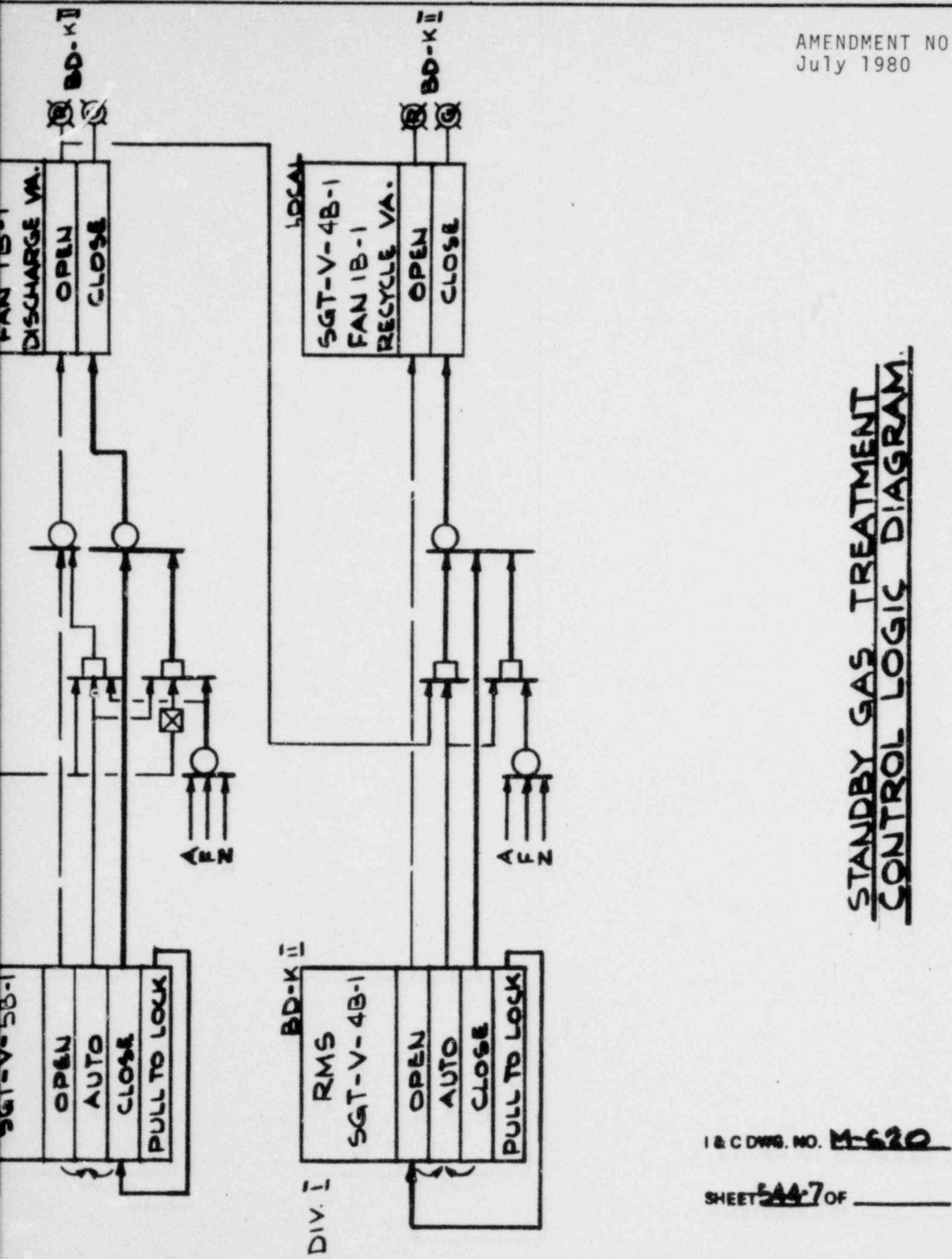
I & C DWG. NO. M-620

SHEET 5A4-6 OF ______

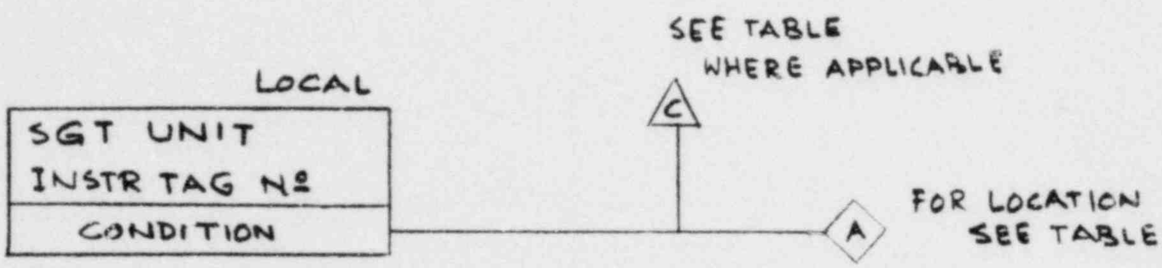
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SGTS	FIGURE 7.3-19f


DIV. II
BD-K II
RMS
SGT-V-5B-2
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
SGT-V-5B-2
FAN 1B-2
DISCHARGE VA.
OPEN
CLOSE
R
G
BD-K II
DIV. II
BD-K II
RMS
SGT-V-4B-2
OPEN
AUTO
CLOSE
PULL TO LOCK
A
F
Z
LOCAL
SGT-V-4B-2
FAN 1B-2
RECYCLE VA.
OPEN
CLOSE
R
G
BD-K II
FROM SHEET 546-3
X4
DIV. I
BD-K II
RMS
LOCAL
SGT-V-5B-1

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - SGTS	FIGURE 7.3-19g



INSTR. TA
SGT-dPIS
-
-
-
SGT-TS-
-
-
-
-
-
-
-
-
-
-
-
SGT-MS-
SGT-MS
TS
TS
TS
TS-
TS
TS
TS
TS
FS
FS

AMENDMENT NO. 10
July 1980

G Nº	ANNUN LOCATION	COMP. INP POINT	CONDITION
-1A	BD-KI	C0211	DP> 1.25" wg
2A	↓	C0212	DP> 0.8" wg
3A		C0214	DP> 3" wg
4A		C0218	DP> 2" wg
5A		C0220	DP> 2" wg
6A	↓	C0222	DP> 3" wg
1B	BD-KII	C0210	DP> 1.25" wg
2B	↓	C0213	DP> 0.8" wg
3B		C0215	DP> 3" wg
4B		C0219	DP> 2" wg
5B		C0221	DP> 2" wg
6B	↓	C0223	DP> 3" wg
1A	BD-KI	—	TEMP> 210°F
1B	BD-KII	—	TEMP> 210°F
8A-1	BD-KI	—	TEMP> 210°F
8B-1	BD-KII	—	TEMP> 210°F
A-1	BD-KI	—	TEMP< 80°F
B-1	BD-KII	—	TEMP< 80°F
A-2	BD-KI	—	TEMP> 120°F
B-2	BD-KII	—	TEMP> 120°F
2A-4	BD-KI	—	TEMP> 120°F
2B-4	BD-KII	—	TEMP> 120°F
A-3	BD-KI	—	TEMP< 80°F
B-3	BD-KII	—	TEMP< 80°F
4A	BD-KI	—	MOIST> 70%RH
-4B	BD-KII	—	MOIST> 70%RH
6A	BD-KI	—	TEMP> 250°F
6B	BD-KII	—	TEMP> 250°F
7A	BD-KI	—	TEMP> 250°F
7B	BD-KII	—	TEMP> 250°F
-6A-1	BD-KI	—	TEMP> 210°F
6B-1	BD-KII	—	TEMP> 210°F
7A-1	BD-KI	—	TEMP> 210°F
-7B-1	BD-KII	—	TEMP> 210°F
2A-1	BD-KI		FLOW< 500 cfm
2B-2	BD-KII		FLOW< 500 cfm

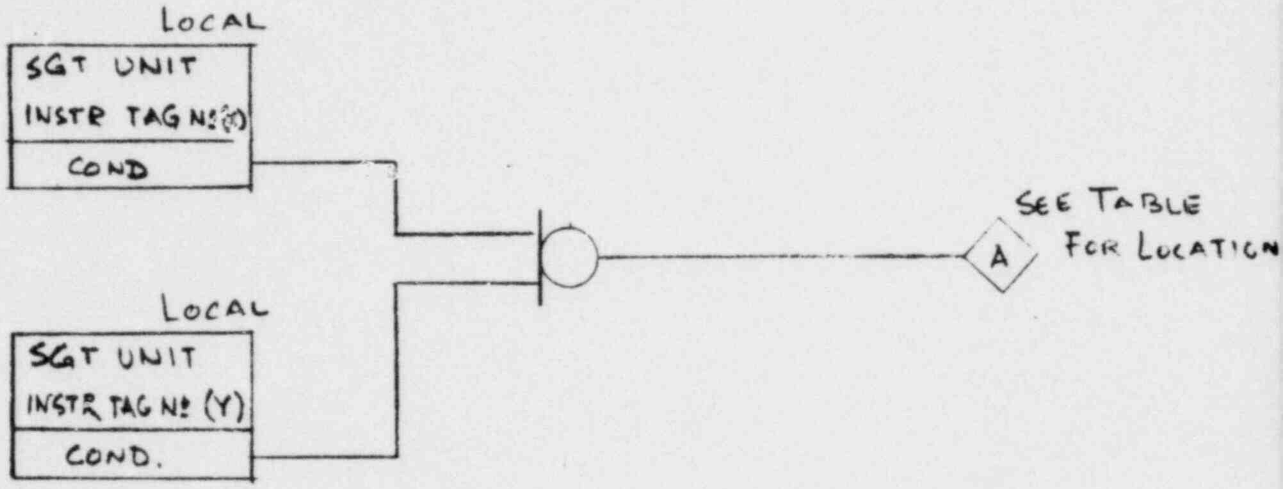
I & C DWG. NO. M620

SHEET 544-8 OF ______

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL LOGIC DIAGRAM - SGTS

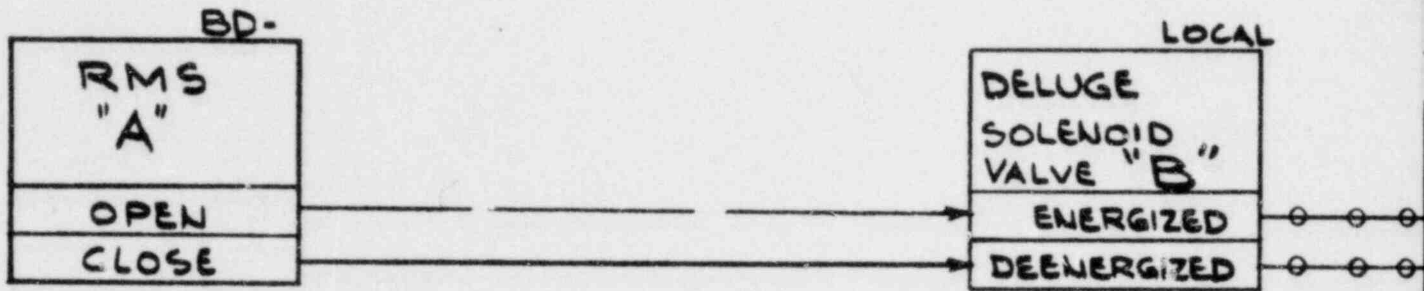
FIGURE 7.3-19h


LOCAL
SGT UNIT
INSTR TAG No (X)
COND
LOCAL
SGT UNIT
INSTR TAG No (Y)
COND.
A
SEE TABLE
FOR LOCATION

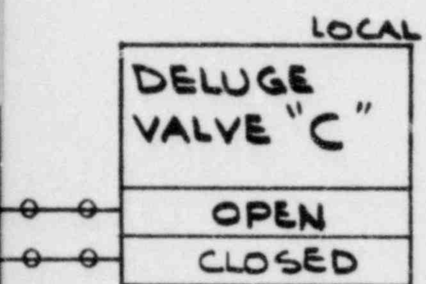
INSTR. TAG Nº (X)	CONDITION	INSTR TAG Nº (Y)	CONDITION	ANNUNCIATOR LOCATION
SGT-TS-1A-3	TEMP < 80 °F	SGT-TS-2A-1	TEMP < 80 °F	BD-K I
SGT-TS-1B-3	TEMP < 80 °F	SGT-TS-2B-1	TEMP < 80 °F	BD-K II
SGT-TS-1A-4	TEMP > 120 °F	SGT-TS-2A-2	TEMP > 120 °F	BD-K I
SGT-TS-1B-4	TEMP > 120 °F	SGT-TS-2B-2	TEMP > 120 °F	BD-K II

I & C DWG. NO. M620

SHEET 544-9 OF ________


BD-
RMS "A"
OPEN
CLOSE
LOCAL
DELUGE SOLENOID VALVE "B"
ENERGIZED
DEENERGIZED

RMS "A"	BD-	DELUGE SOL. VA. "B"	ELECT. DIV.	DELUGE VALVE "C"
SGT-SPV-F1	K I	SGT-SPV-F1	A	SGT-PCV-F1
F2	K I	F2	A	F2
F3	K I	F3	A	F3
F4	K II	F4	B	F4
F5	K II	F5	B	F5
F6	K II	F6	B	F6

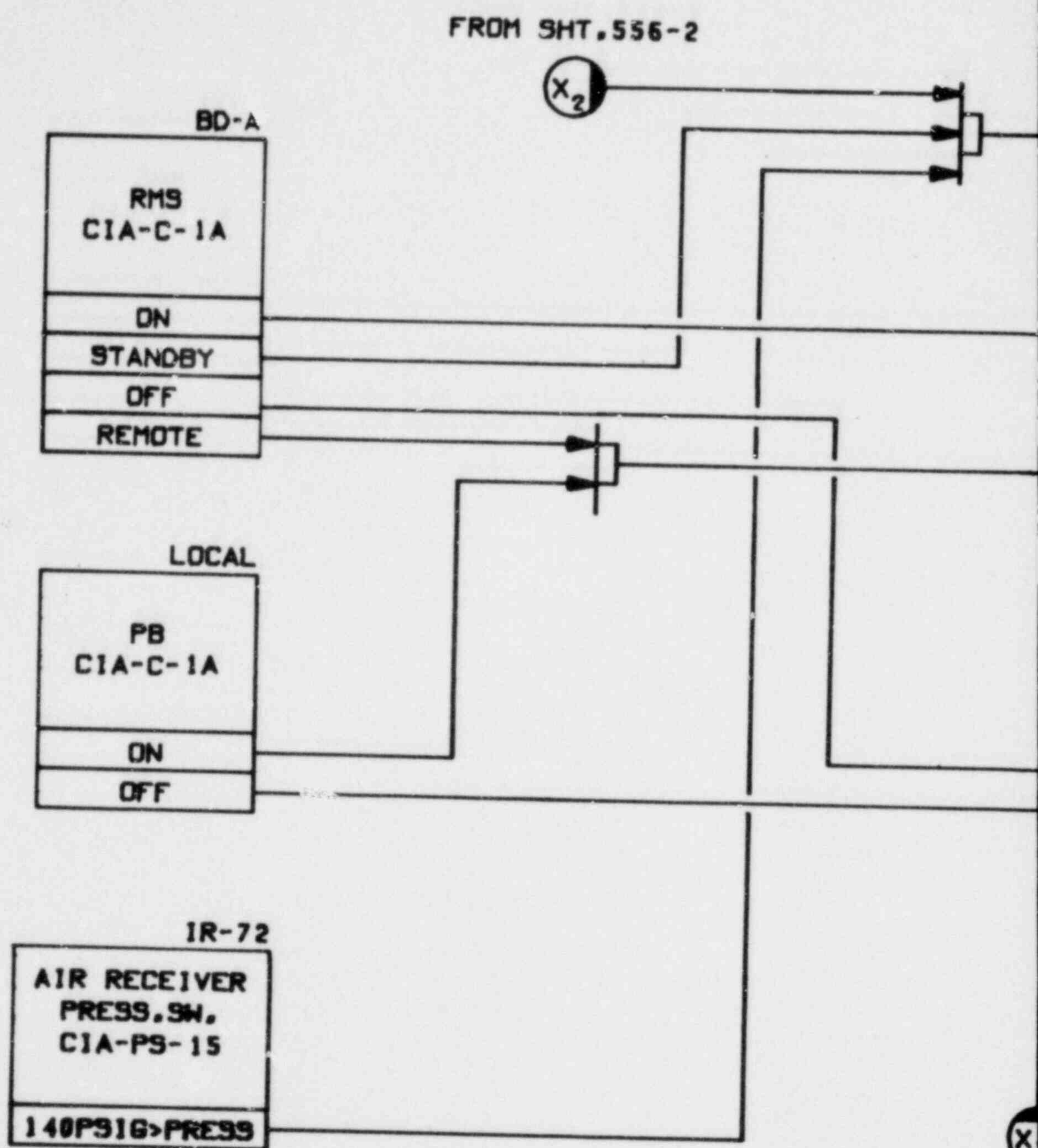


I & C DWG. NO. M-620

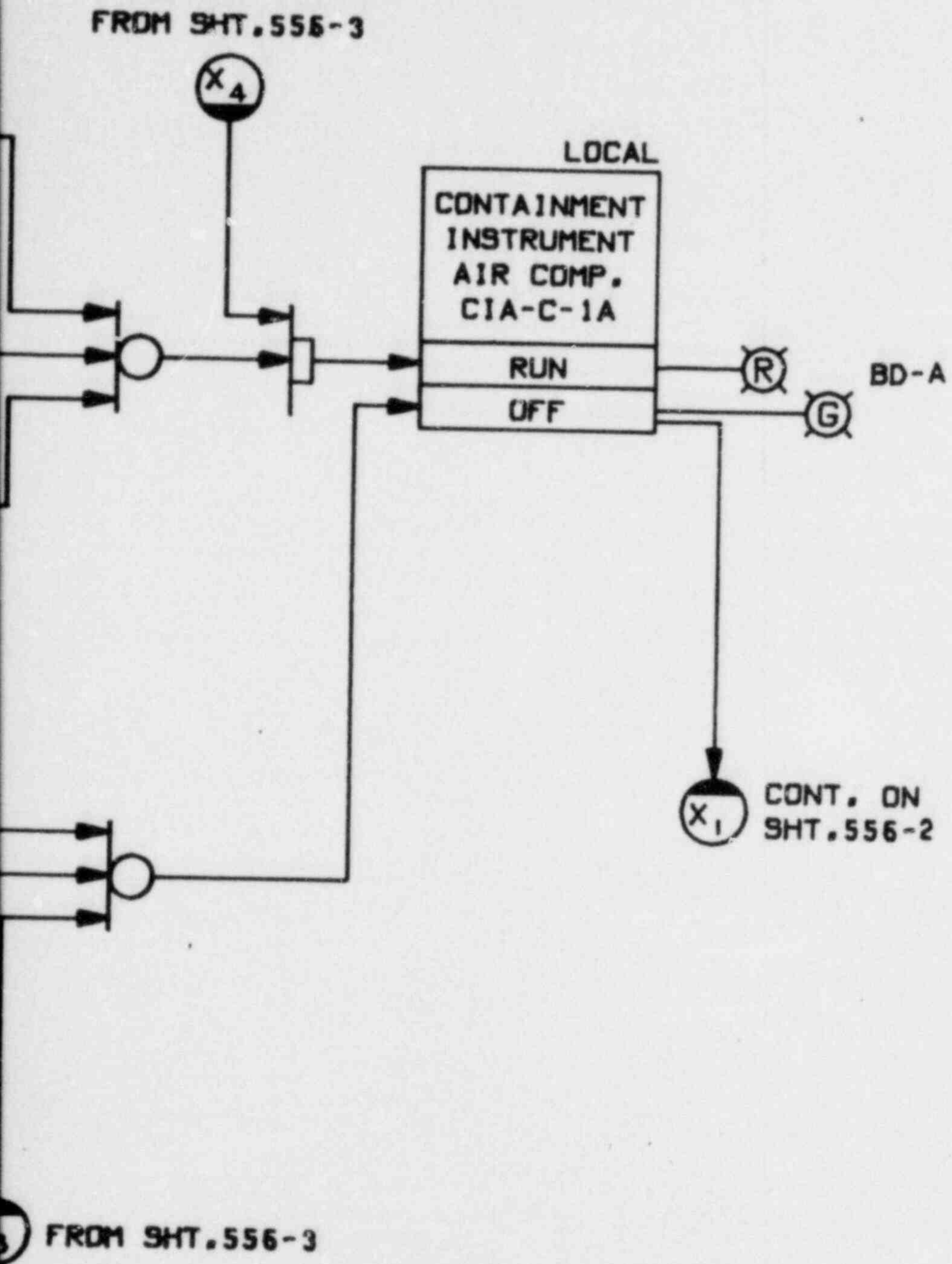
SHEET 544-12 OF

STANDBY GAS TREATMENT DELUGE VALVE CONTROL LOGIC DIAGRAM.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM -SGTS	FIGURE 7.3-19j



FROM SHT.556-2
X2
BD-A
RMS
CIA-C-1A
ON
STANDBY
OFF
REMOTE
LOCAL
PB
CIA-C-1A
ON
OFF
IR-72
AIR RECEIVER
PRESS.SW.
CIA-PS-15
140PSIG>PRESS

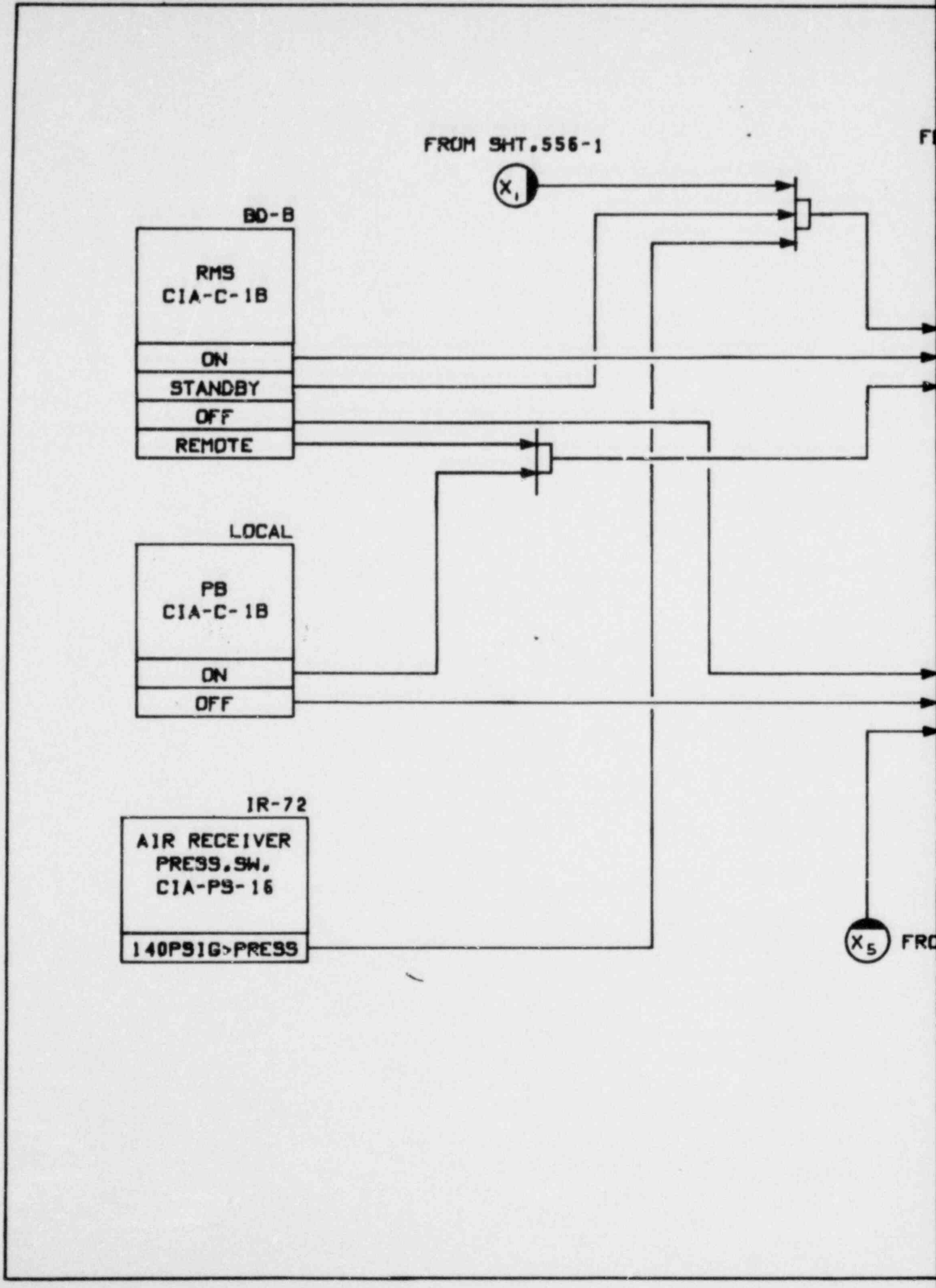


I & C DWG. NO. M-620

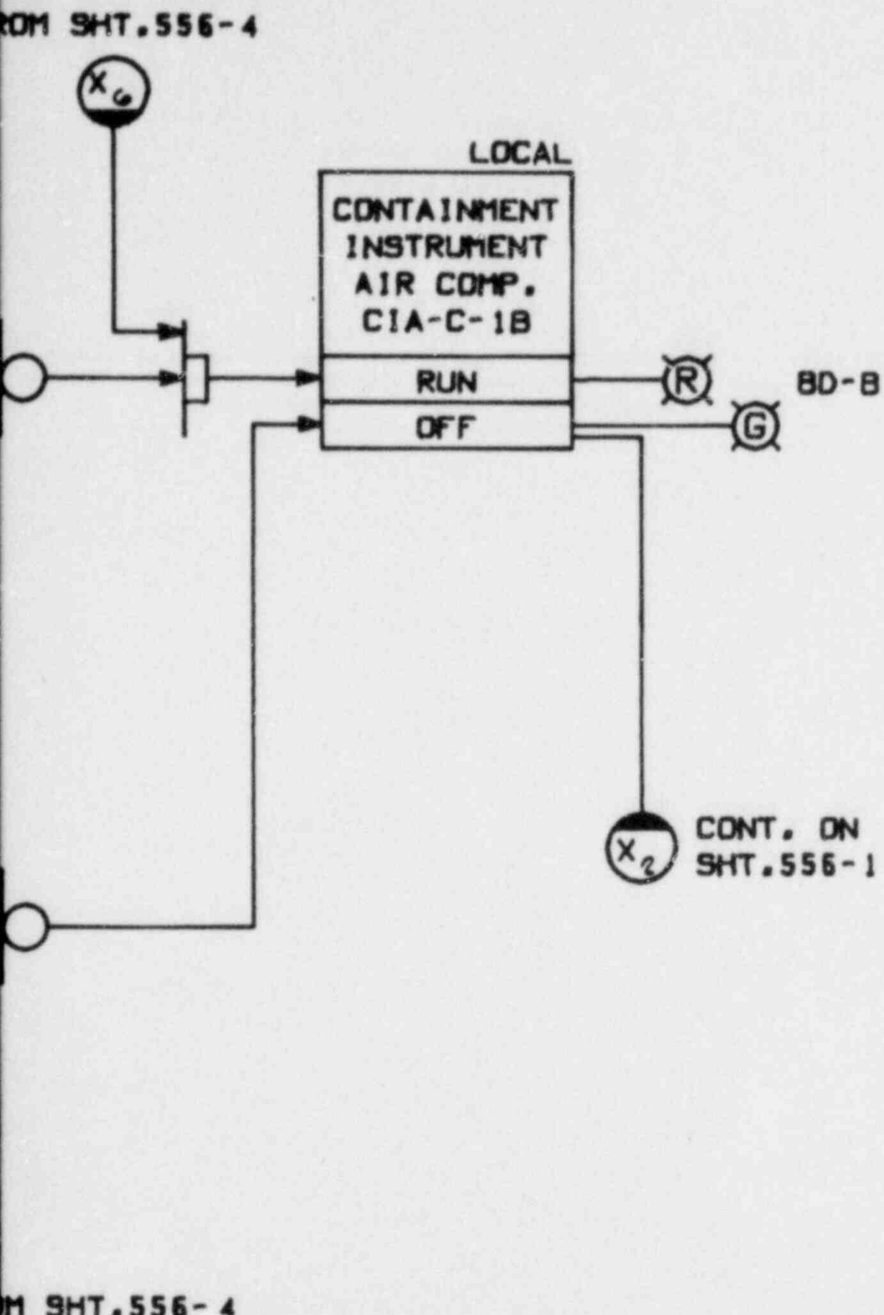
SHEET 556-1

CONTAINMENT INSTRUMENT AIR COMPRESSOR CIA-C-1A
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20a


FROM SHT.556-1
X_1
BD-B
RMS
CIA-C-1B
ON
STANDBY
OFF
REMOTE
LOCAL
PB
CIA-C-1B
ON
OFF
IR-72
AIR RECEIVER
PRESS.SW.
CIA-PS-16
140PSIG>PRESS
X_5

AMENDMENT NO. 10
July 1980

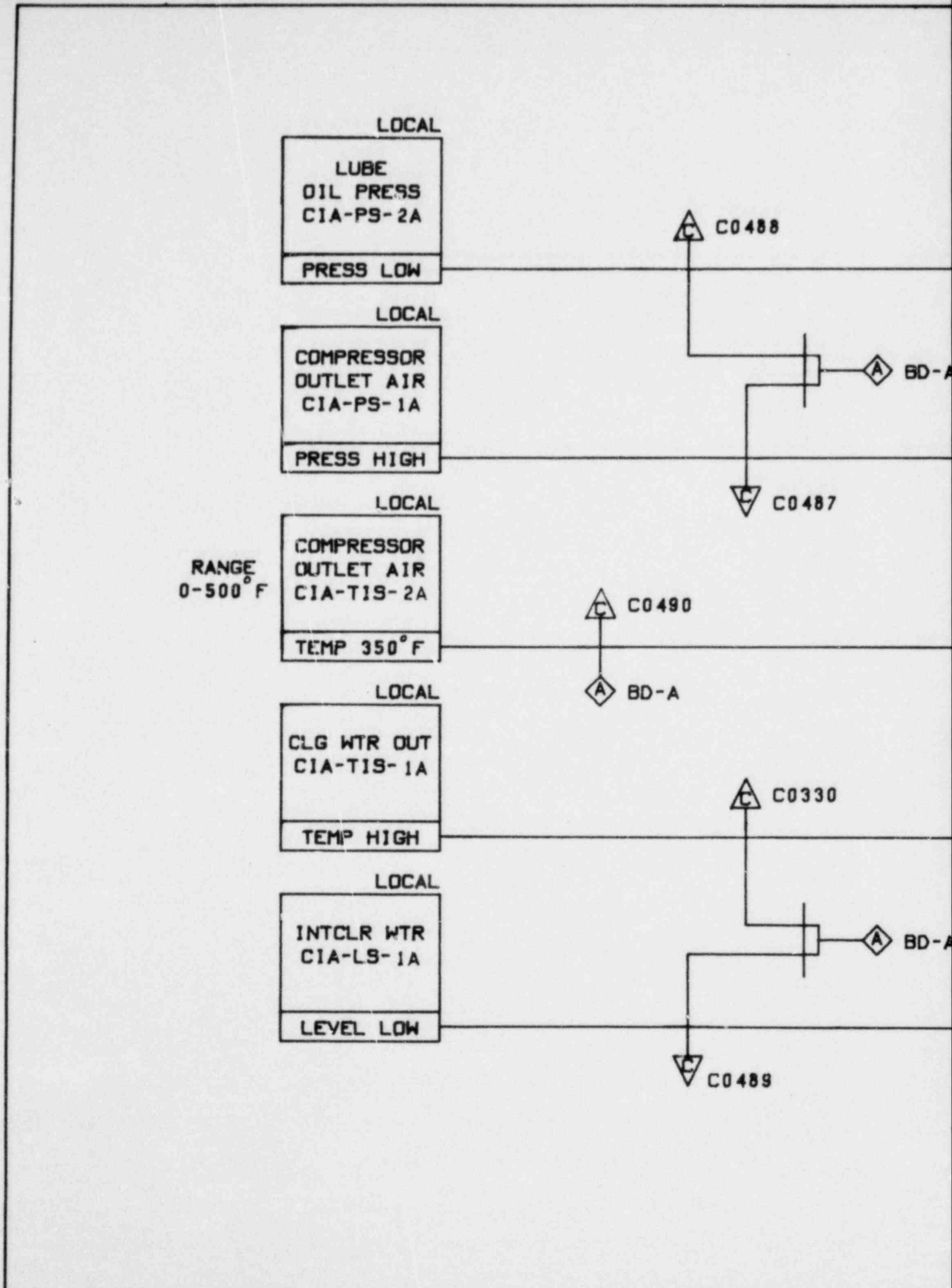


I & C DWG. NO. M-620

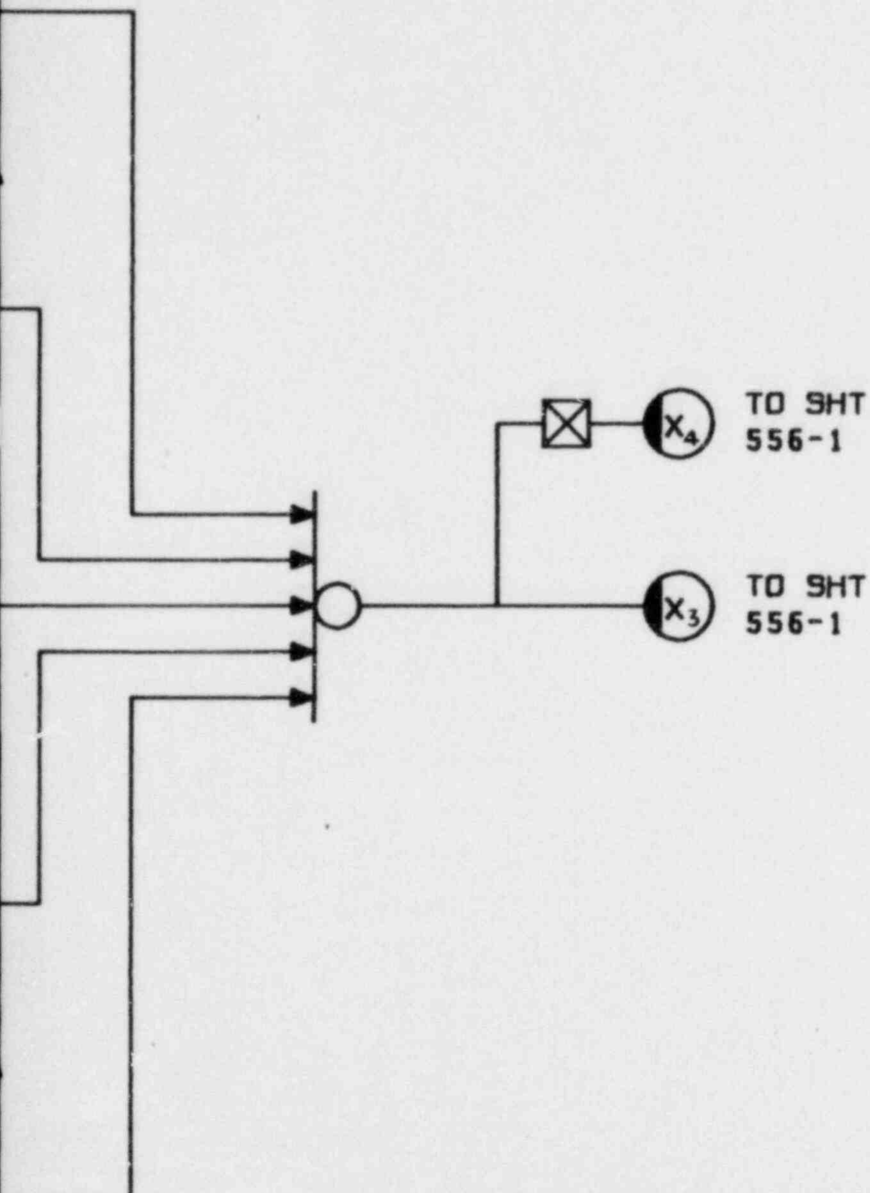
SHEET 556-2

CONTAINMENT INSTRUMENT AIR COMPRESSOR CIA-C-1B
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20b



LOCAL
LUBE
OIL PRESS
CIA-PS-2A
PRESS LOW
C0488
LOCAL
COMPRESSOR
OUTLET AIR
CIA-PS-1A
PRESS HIGH
BD-A
C0487
LOCAL
RANGE
0-500°F
COMPRESSOR
OUTLET AIR
CIA-TIS-2A
TEMP 350°F
C0490
BD-A
LOCAL
CLG WTR OUT
CIA-TIS-1A
TEMP HIGH
C0330
LOCAL
INTCLR WTR
CIA-LS-1A
LEVEL LOW
BD-A
C0489

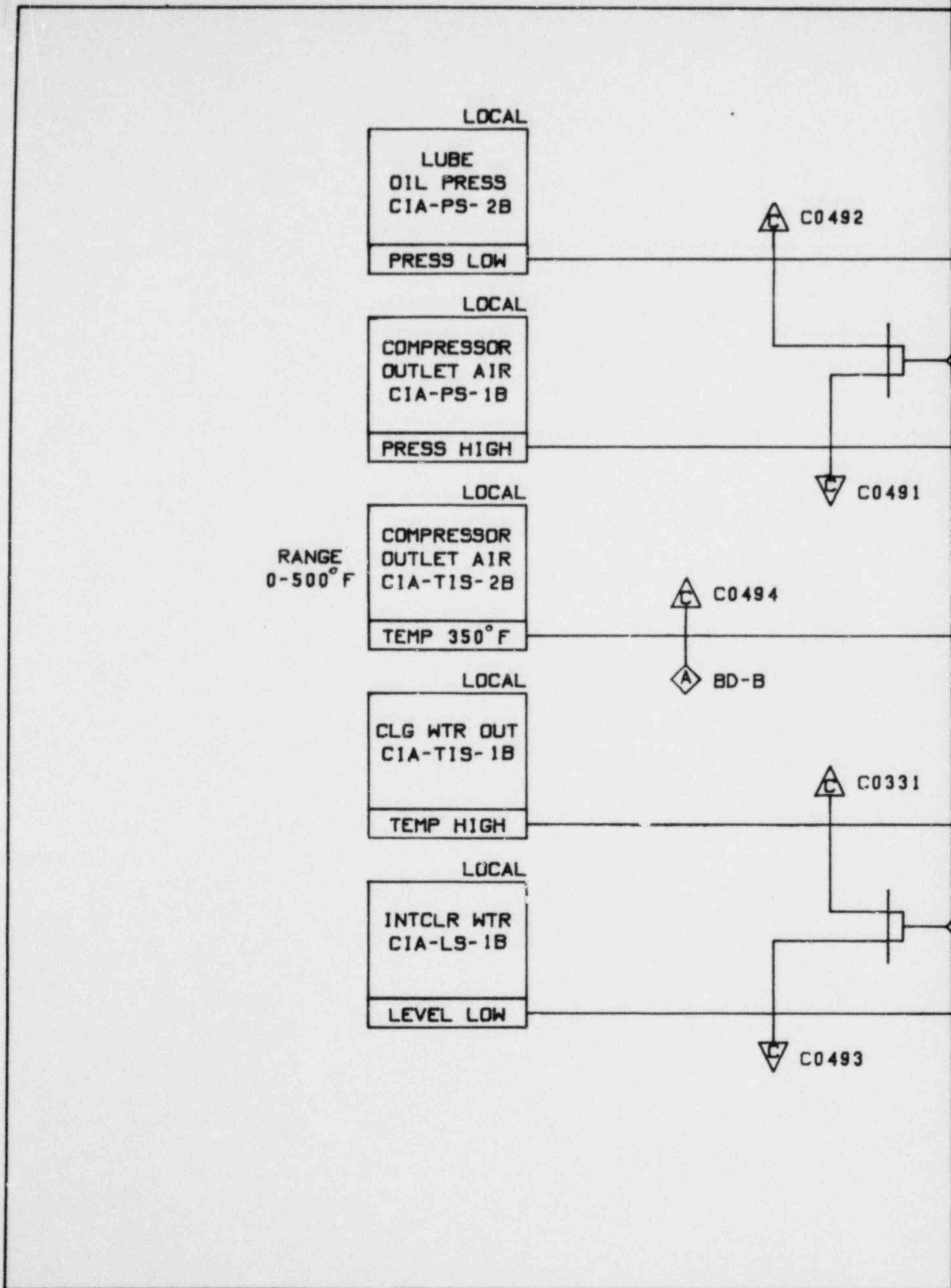


I & C DWG. NO. M-620

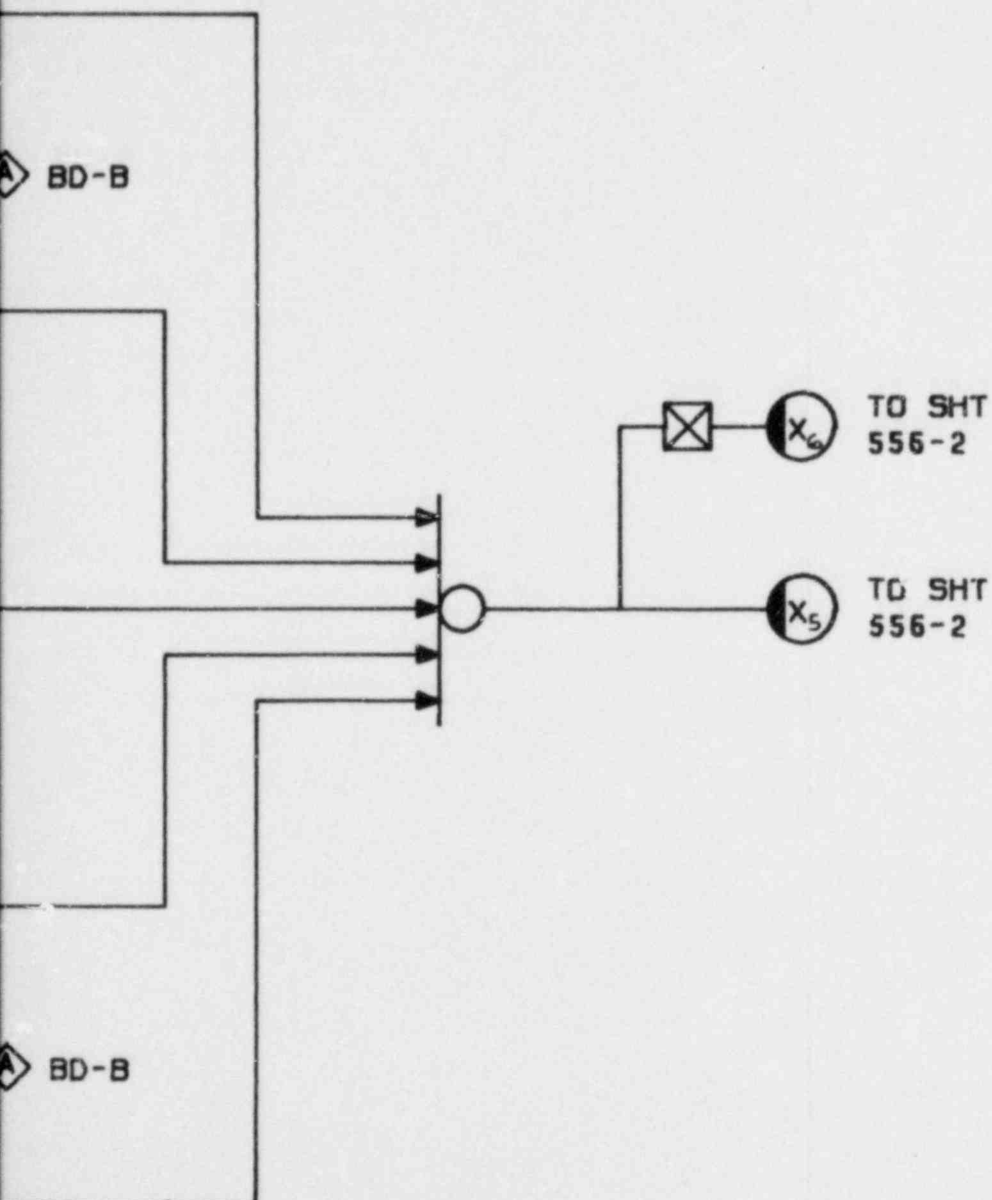
SHEET 556-3

CIAS LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20c


LOCAL
LUBE
OIL PRESS
CIA-PS- 2B
PRESS LOW
C0492
LOCAL
COMPRESSOR
OUTLET AIR
CIA-PS-1B
PRESS HIGH
C0491
LOCAL
RANGE
0-500°F
COMPRESSOR
OUTLET AIR
CIA-TIS-2B
TEMP 350°F
C0494
BD-B
LOCAL
CLG WTR OUT
CIA-TIS-1B
TEMP HIGH
C0331
LOCAL
INTCLR WTR
CIA-LS-1B
LEVEL LOW
C0493

AMENDMENT NO. 10
July 1980



I & C DWG. NO. M-620

SHEET 556-4

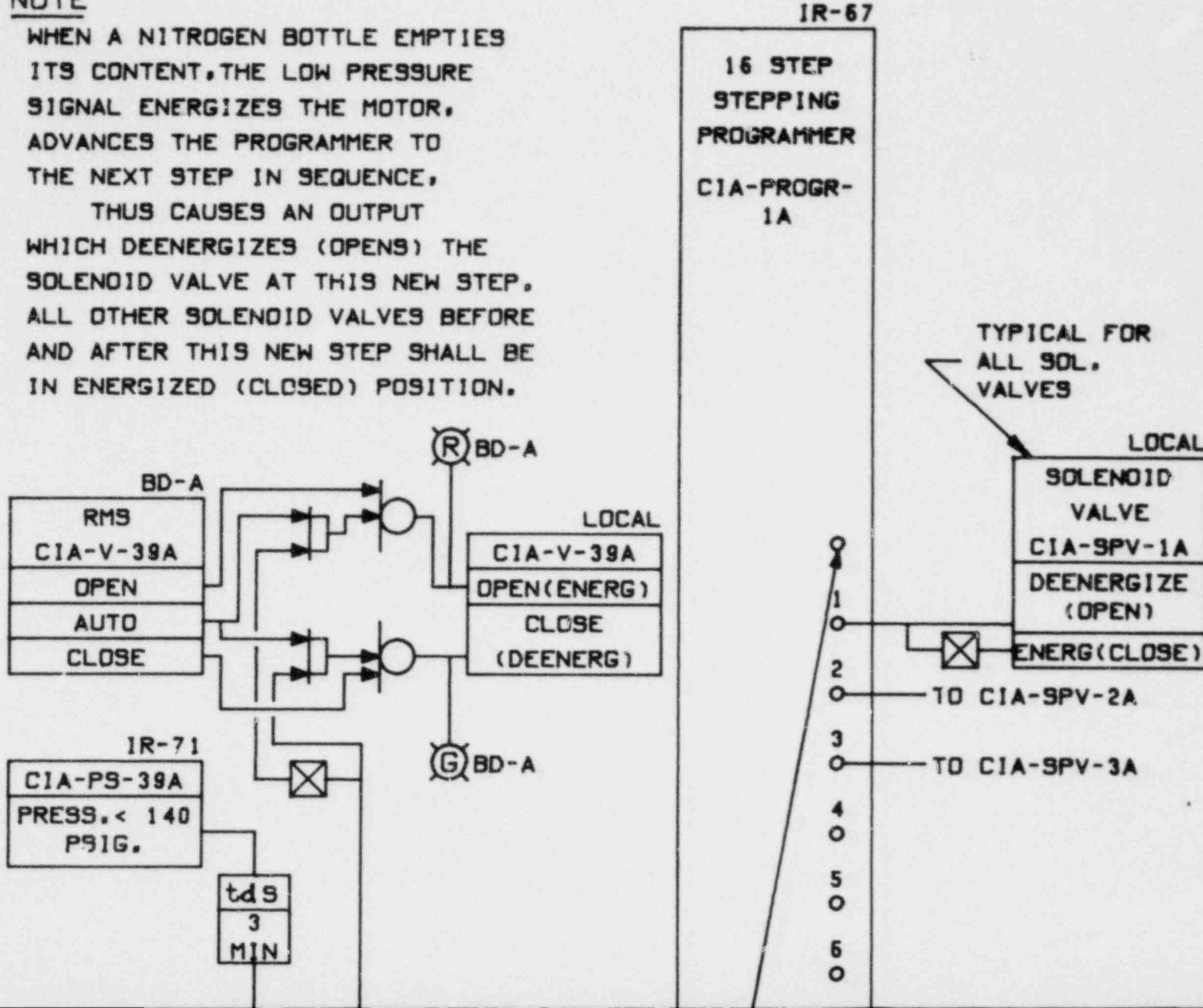
CIAS LOGIC DIAGRAM

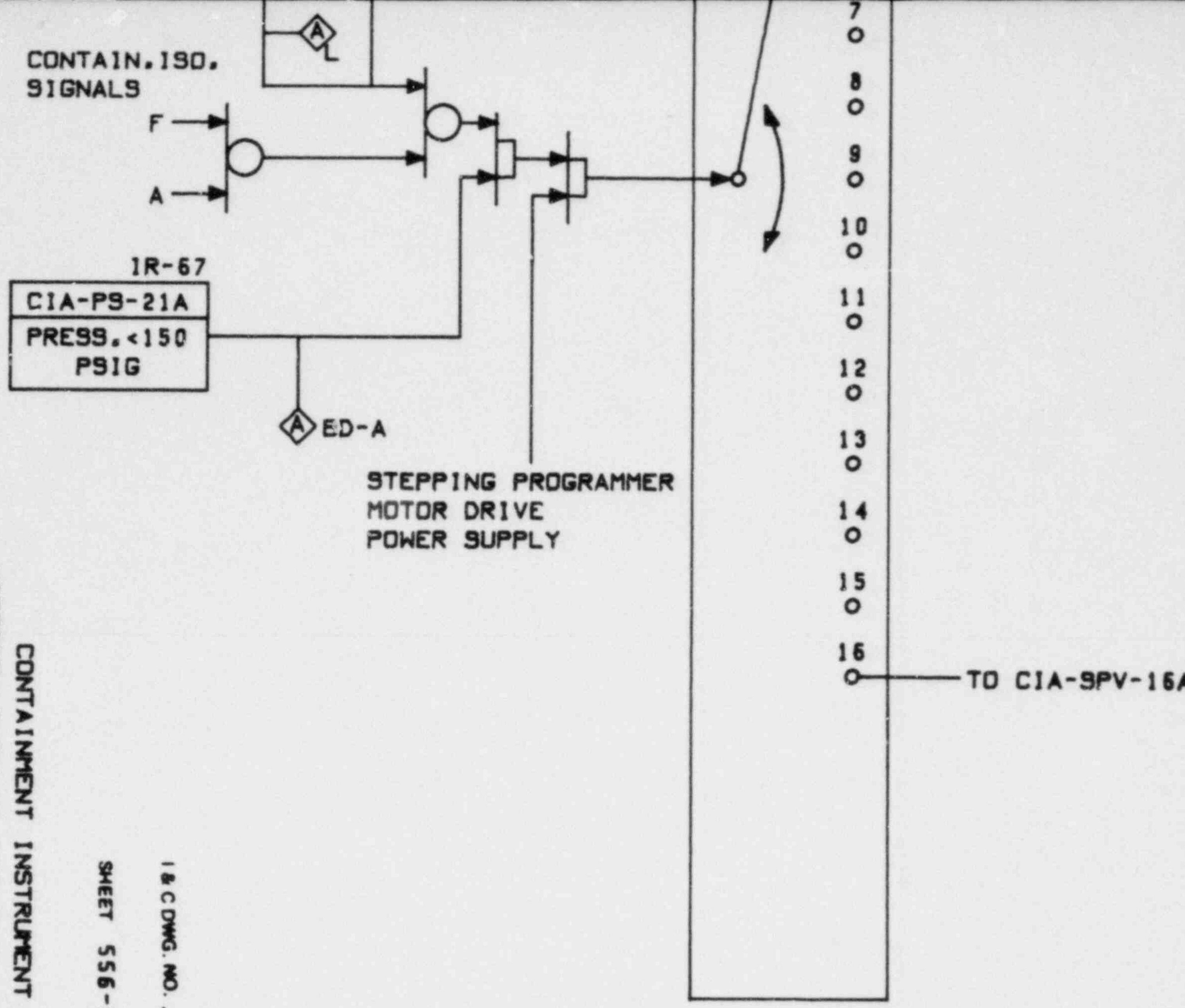
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20d

NOTE

WHEN A NITROGEN BOTTLE EMPTIES ITS CONTENT, THE LOW PRESSURE SIGNAL ENERGIZES THE MOTOR, ADVANCES THE PROGRAMMER TO THE NEXT STEP IN SEQUENCE,

THUS CAUSES AN OUTPUT WHICH DEENERGIZES (OPENS) THE SOLENOID VALVE AT THIS NEW STEP. ALL OTHER SOLENOID VALVES BEFORE AND AFTER THIS NEW STEP SHALL BE IN ENERGIZED (CLOSED) POSITION.





CONTAINMENT INSTRUMENT AIR
NITROGEN BACK-UP CIS-TK-2A DISCHARGE
CONTROL LOGIC DIAGRAM

I&C DWG. NO. M-620

SHEET 556-5

AMENDMENT NO. 10
July 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

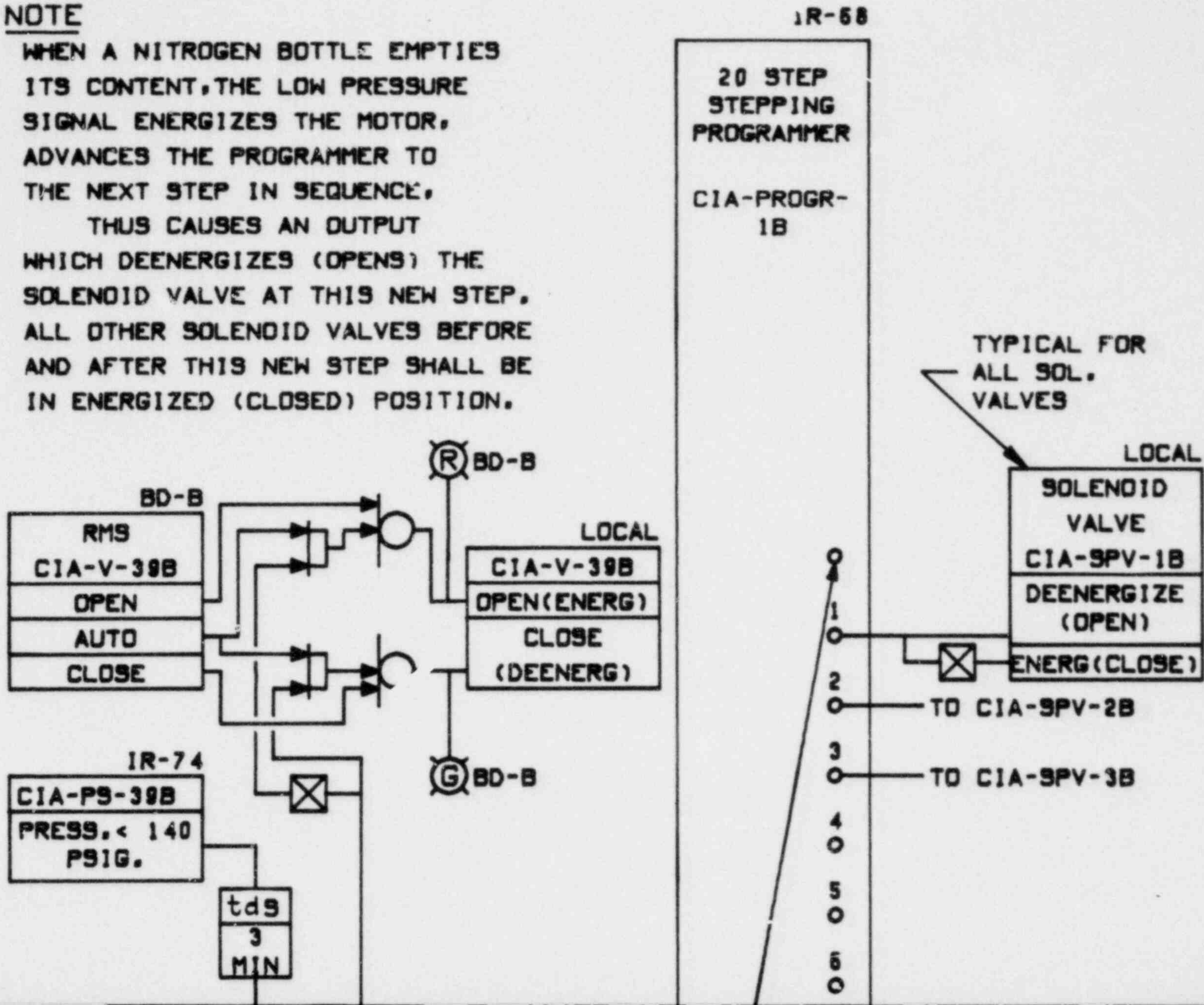
CONTROL LOGIC DIAGRAM - CIAS

FIGURE 7.3-20e

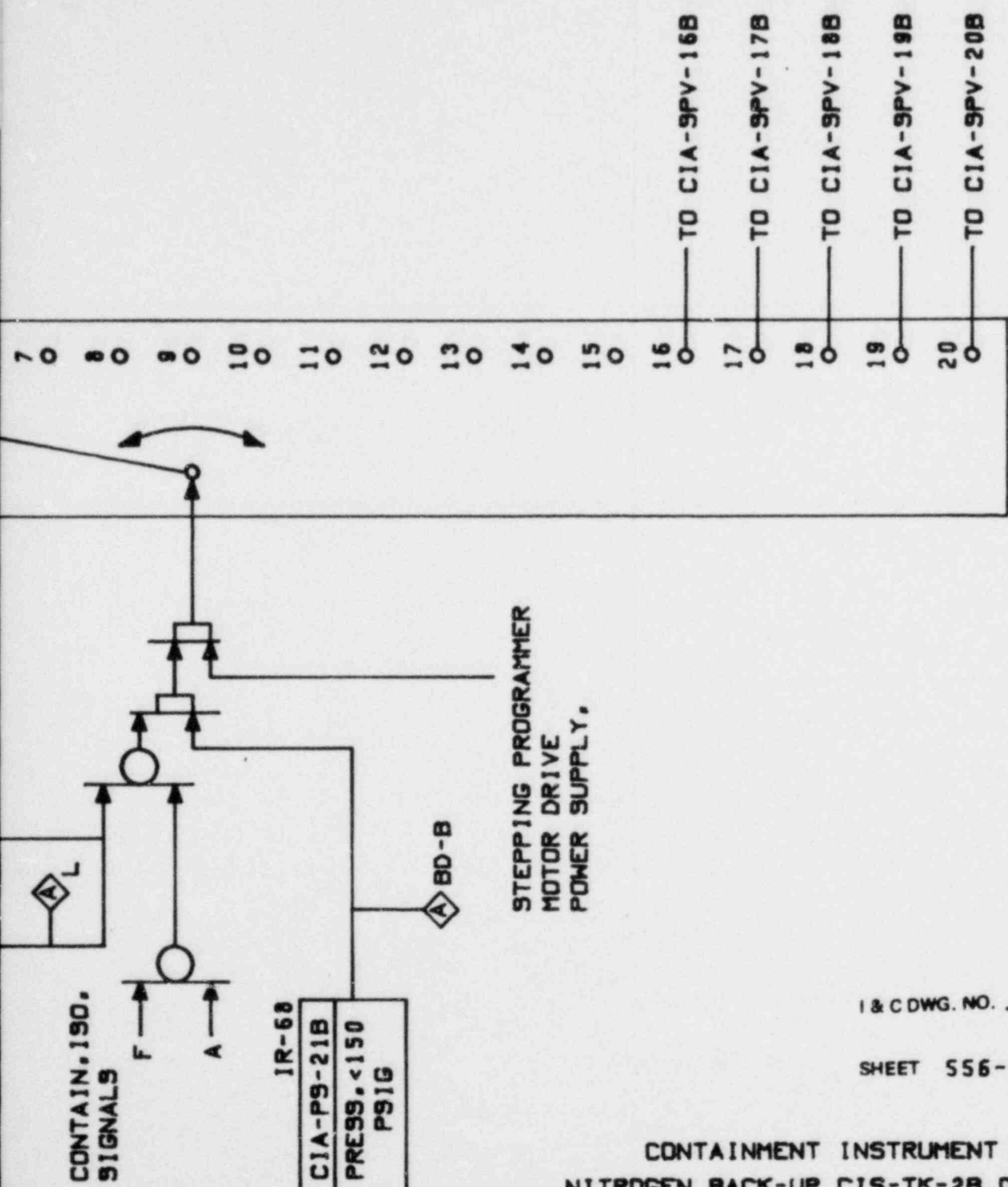
NOTE

WHEN A NITROGEN BOTTLE EMPTIES ITS CONTENT, THE LOW PRESSURE SIGNAL ENERGIZES THE MOTOR, ADVANCES THE PROGRAMMER TO THE NEXT STEP IN SEQUENCE.

THUS CAUSES AN OUTPUT WHICH DEENERGIZES (OPENS) THE SOLENOID VALVE AT THIS NEW STEP. ALL OTHER SOLENOID VALVES BEFORE AND AFTER THIS NEW STEP SHALL BE IN ENERGIZED (CLOSED) POSITION.



AMENDMENT NO. 10
July 1980

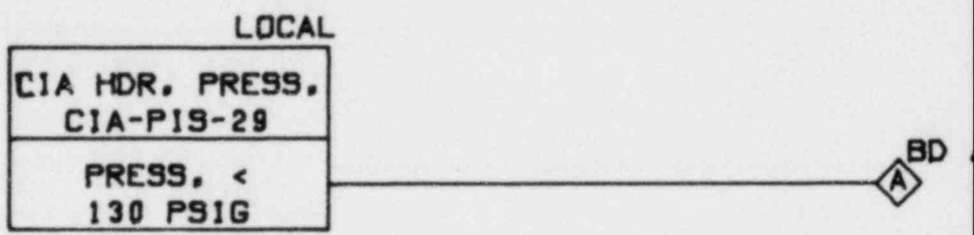


I & C DWG. NO. M-620

SHEET 556-6

CONTAINMENT INSTRUMENT AIR
NITROGEN BACK-UP CIS-TK-2B DISCHARGE
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20f

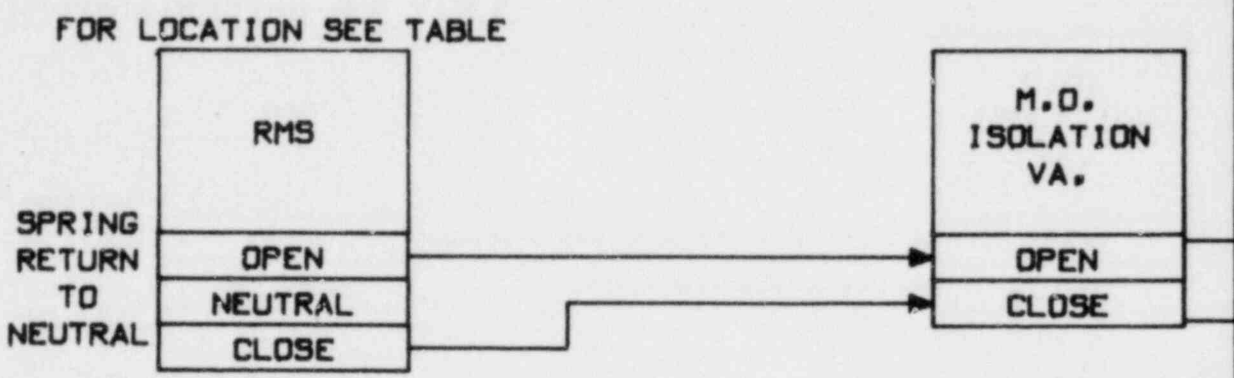

LOCAL
CIA HDR. PRESS.
CIA-PIS-29
PRESS. <
130 PSIG
BD
A

I & C DWG. NO. M-620

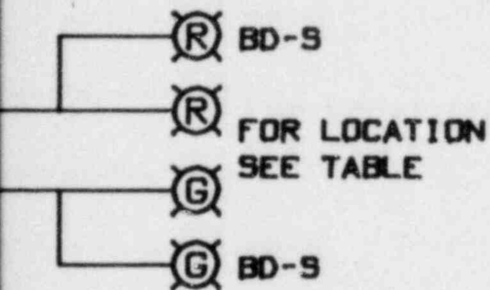
SHEET 556-7

CIAS LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20g


FOR LOCATION SEE TABLE
RMS
SPRING RETURN TO NEUTRAL
OPEN
NEUTRAL
CLOSE
M.O. ISOLATION VA.
OPEN
CLOSE

RMS	RMS & IND.LT. LOC.	VALVE NO.	REMARK
	BD-A	CIA-V-20	DIV I
	BD-A	CIA-V-30A	DIV I
	BD-B	CIA-V-30B	DIV II



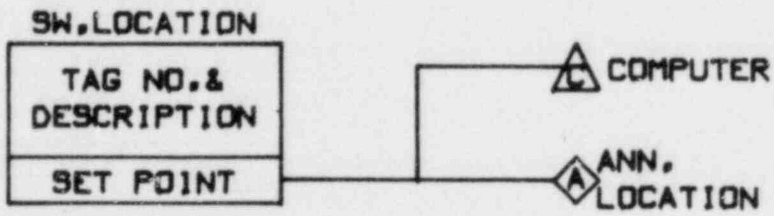
I & C DWG. NO. M-620

SHEET 556-8

CONTAINMENT ISOLATION VALVE
CONTROL LOGIC DIAGRAM

AMENDMENT NO. 10
July 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20h



TAG NO.	SW. LOCATION	
CIA-dPIS-1A	LCL	FILTE
CIA-dPIS-1B	LCL	FILTE
CIA-dPIS-2A	LCL	FILTE
CIA-dPIS-2B	LCL	FILTE

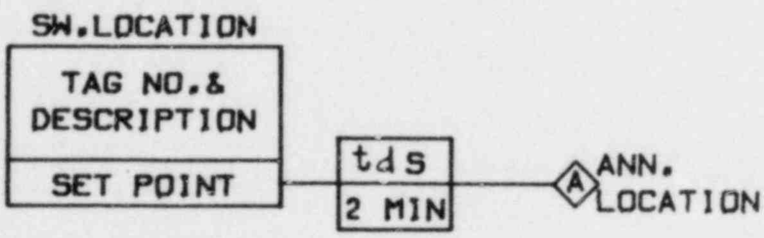
DESCRIPTION	SET POINT	ANN. LOCATION	REMARK
R CIA-F-1A DIFF.PRESS.HI	120" H_2O INCR.	BD-A	C0322
R CIA-F-1B DIFF.PRESS.HI	120" H_2O INCR.	BD-B	C0323
RSCIA-F-2A DIFF.PRESS.HI	120" H_2O INCR,	BD-A	C0320
RS CIA-F-3A DIFF.PRESS.HI	120" H_2O INCR.	BD-B	C0321

I & C DWG. NO. M-620

SHEET 556-9

ALARM ANNUNCIATOR &COMPUTER INPUT
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20i



TAG NO.	SW. LOCATION	
CIA-PS-3	IR-71	CIA

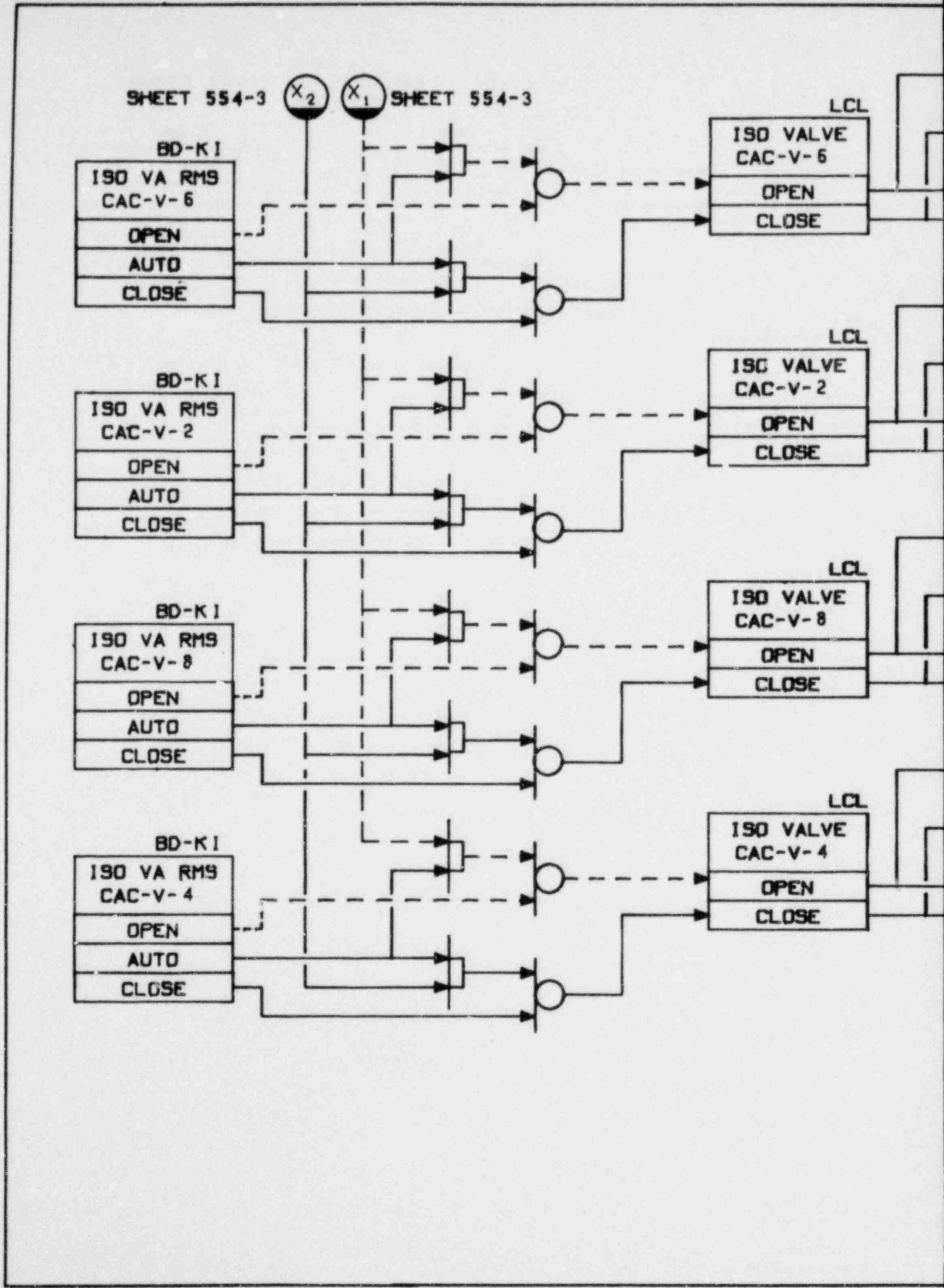
DESCRIPTION	SET POINT	ANN. LOCATION	REMARK
AIR RECEIVER PRESSURE	140 PSIG DECR.	BD-A	

I & C DWG. NO. M-620

SHEET 556-10

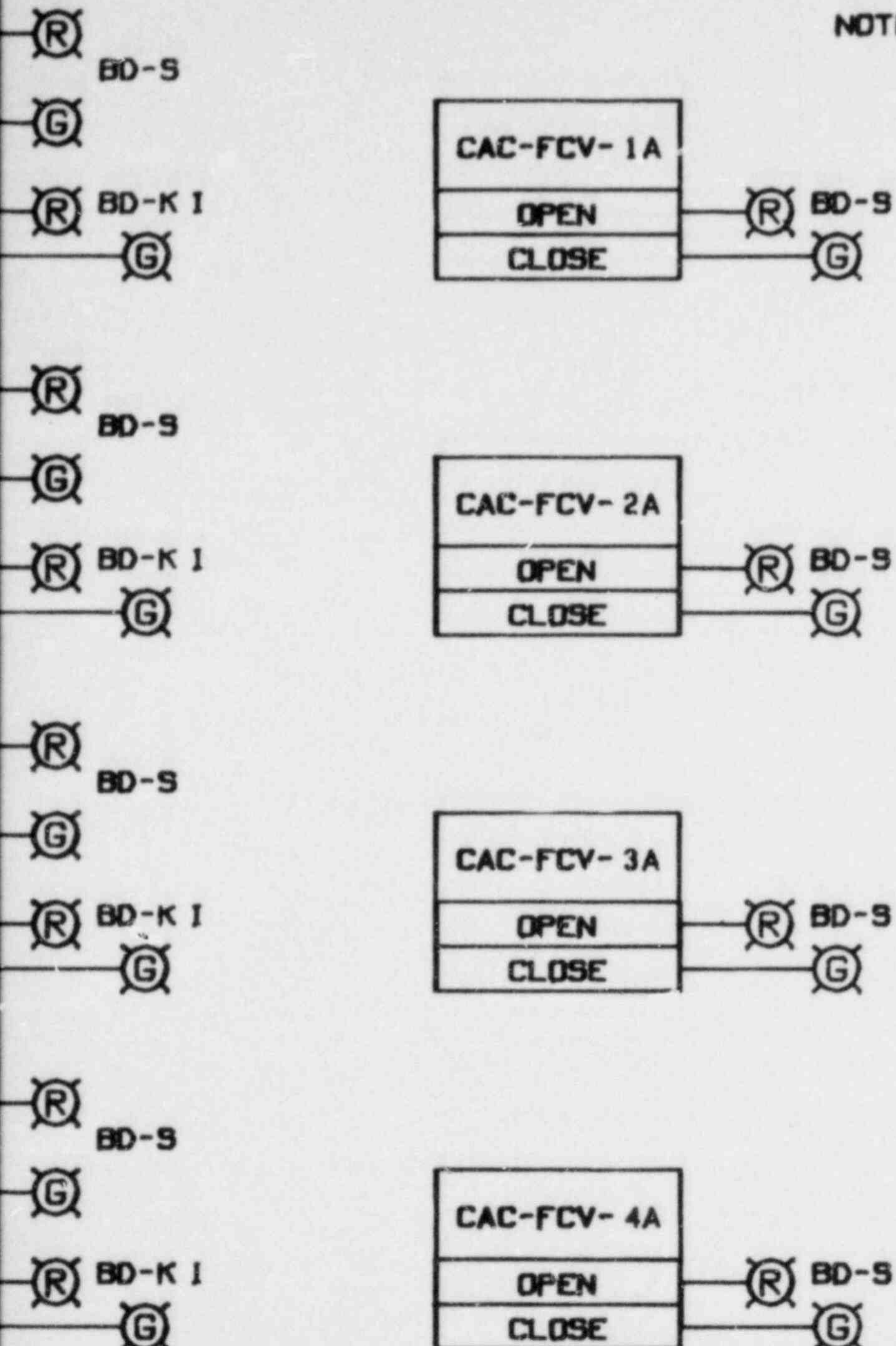
ALARM ANNUNCIATOR INPUT
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM - CIAS	FIGURE 7.3-20j


SHEET 554-3
X2
X1
SHEET 554-3
BD-KI
ISO VA RMS
CAC-V-6
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-6
OPEN
CLOSE
BD-KI
ISO VA RMS
CAC-V-2
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-2
OPEN
CLOSE
BD-KI
ISO VA RMS
CAC-V-8
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-8
OPEN
CLOSE
BD-KI
ISO VA RMS
CAC-V-4
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-4
OPEN
CLOSE

NOTE - ISO VALVES ARE
125 VDC OPERATED

AMENDMENT NO. 10
July 1980

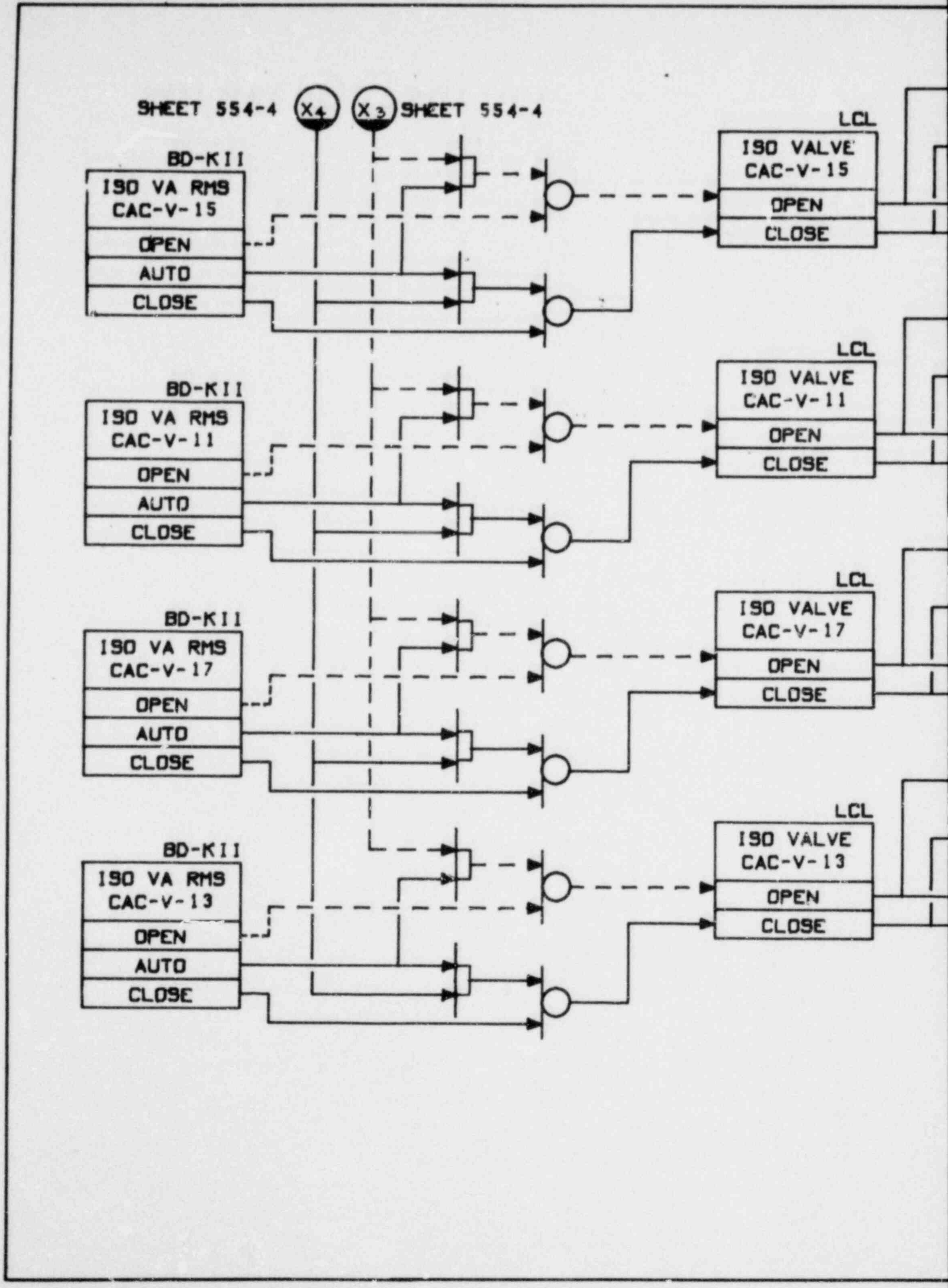


I & C DWG. NO. M-620

SHEET 554-1

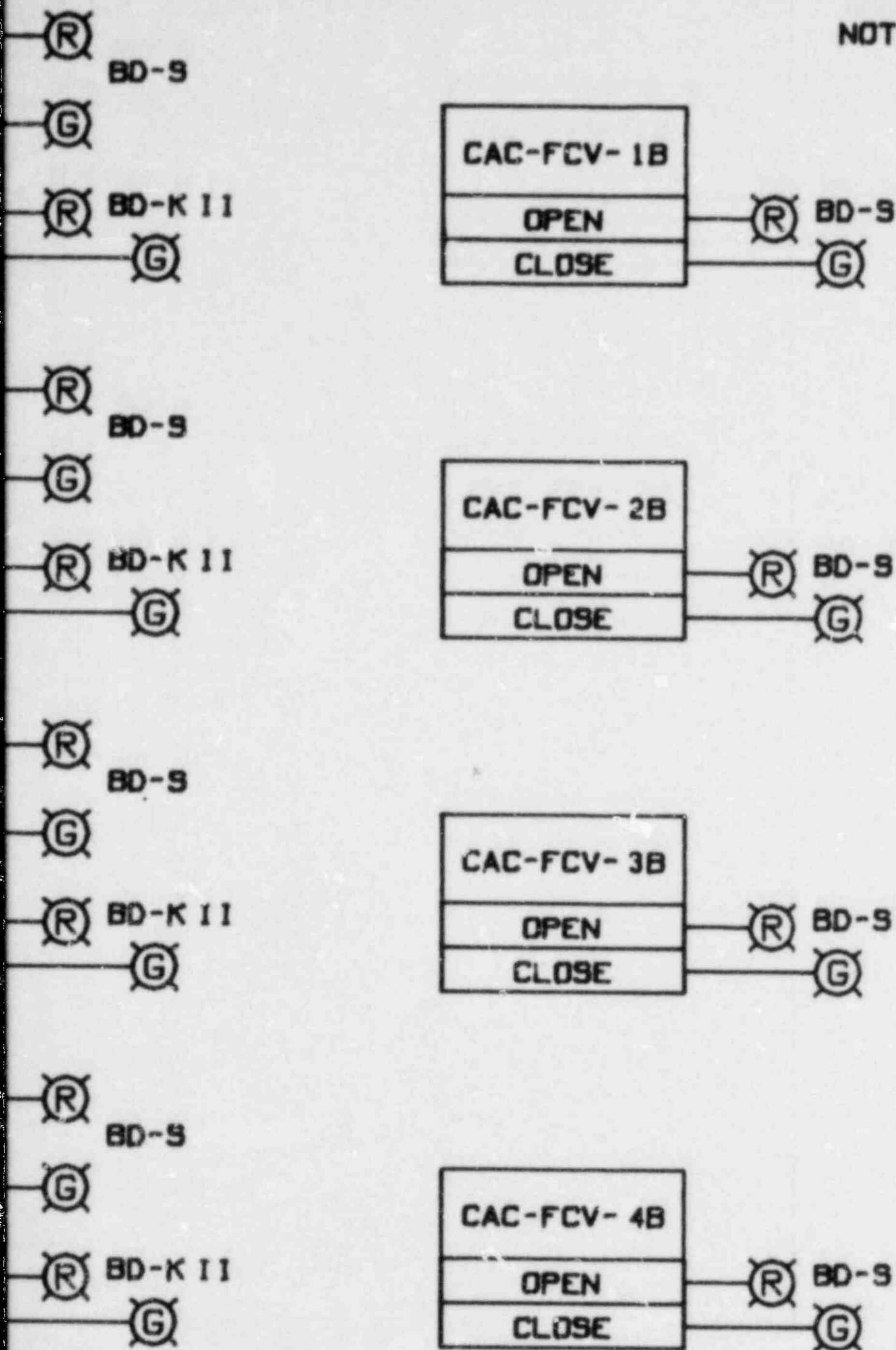
H_2 RECOMBINER CAC-HR-1A
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM CONTAINMENT ATMOSPHERE CONTROL SYSTEM	FIGURE 7.3- 21a


SHEET 554-4
X4
X3
SHEET 554-4
BD-K11
ISO VA RMS
CAC-V-15
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-15
OPEN
CLOSE
BD-K11
ISO VA RMS
CAC-V-11
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-11
OPEN
CLOSE
BD-K11
ISO VA RMS
CAC-V-17
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-17
OPEN
CLOSE
BD-K11
ISO VA RMS
CAC-V-13
OPEN
AUTO
CLOSE
LCL
ISO VALVE
CAC-V-13
OPEN
CLOSE

AMENDMENT NO. 10
July 1980

NOTE - ISO VALVES ARE
125 VDC OPERATED

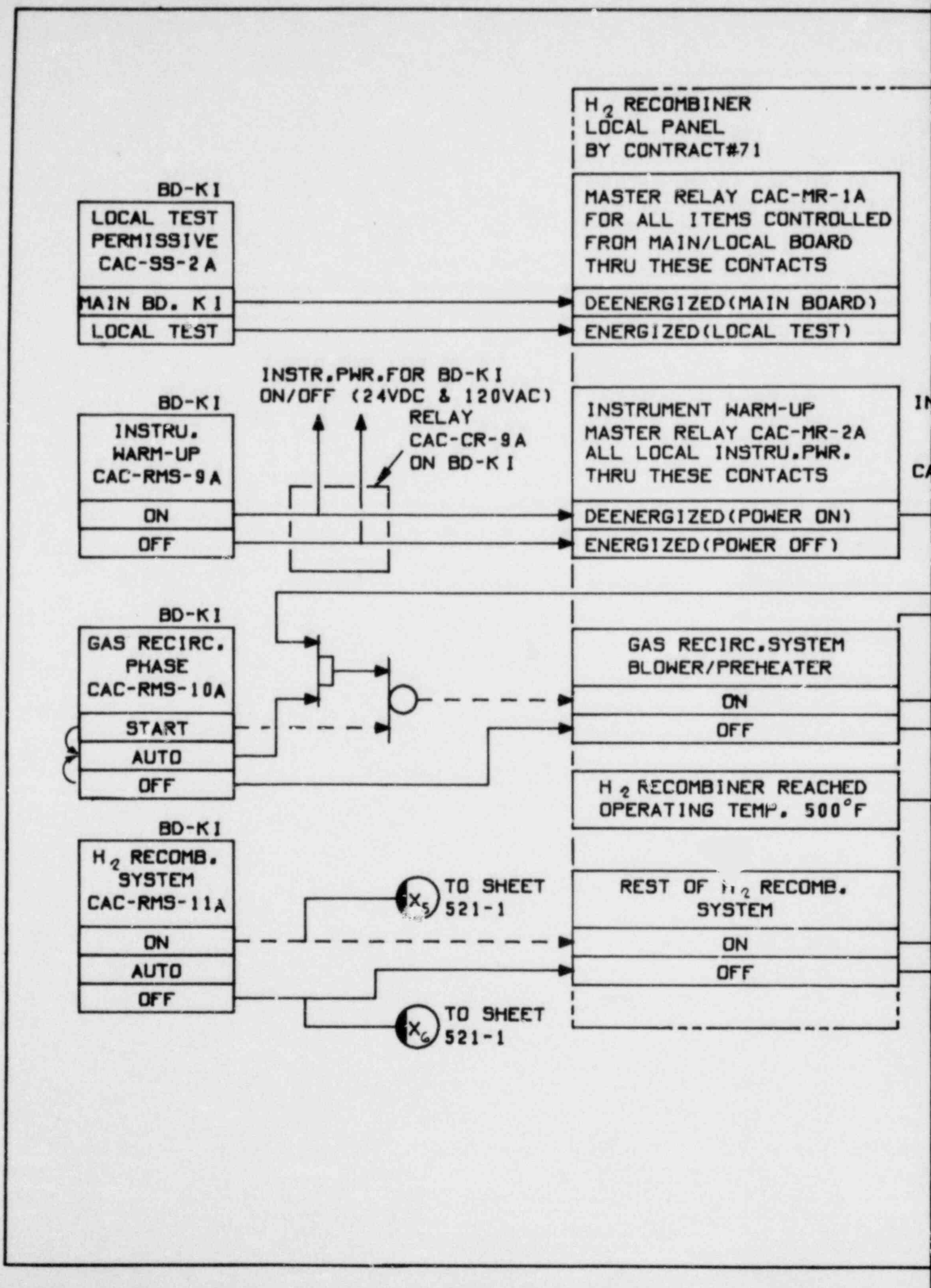


I & C DWG. NO. M-620

SHEET 554-2

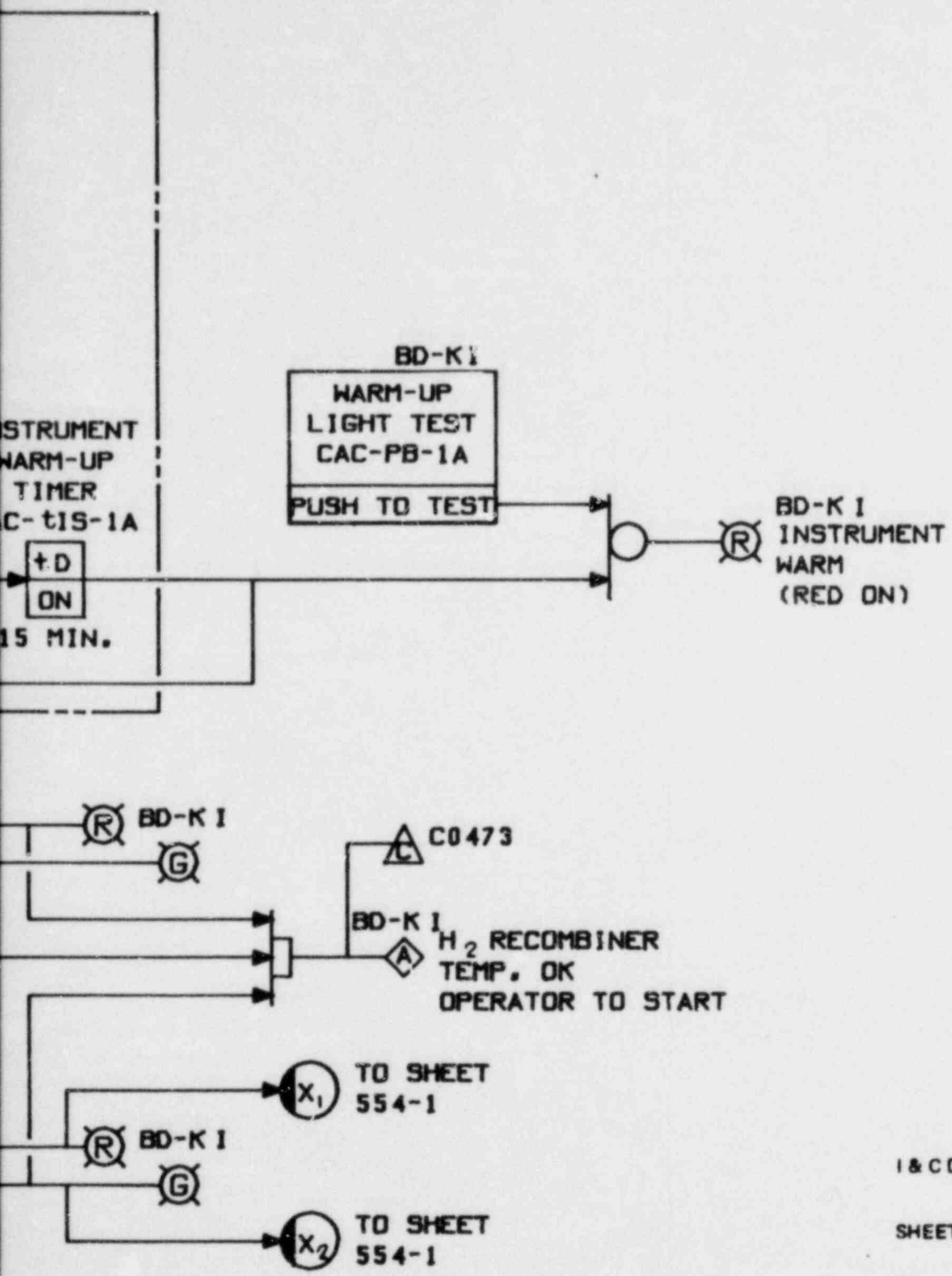
H_2 RECOMBINER CAC-HR-1B
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM CONTAINMENT ATMOSPHERE CONTROL SYSTEM	FIGURE 7.3- 21b



H2 RECOMBINER
LOCAL PANEL
BY CONTRACT#71
BD-KI
LOCAL TEST PERMISSIVE CAC-SS-2A
MAIN BD. KI
LOCAL TEST
MASTER RELAY CAC-MR-1A FOR ALL ITEMS CONTROLLED FROM MAIN/LOCAL BOARD THRU THESE CONTACTS
DEENERGIZED(MAIN BOARD)
ENERGIZED(LOCAL TEST)
INSTR.PWR.FOR BD-KI ON/OFF (24VDC & 120VAC)
RELAY CAC-CR-9A ON BD-KI
BD-KI
INSTRU. WARM-UP CAC-RMS-9A
ON
OFF
INSTRUMENT WARM-UP MASTER RELAY CAC-MR-2A ALL LOCAL INSTRU.PWR. THRU THESE CONTACTS
DEENERGIZED(POWER ON)
ENERGIZED(POWER OFF)
BD-KI
GAS RECIRC. PHASE CAC-RMS-10A
START
AUTO
OFF
GAS RECIRC.SYSTEM BLOWER/PREHEATER
ON
OFF
H2 RECOMBINER REACHED OPERATING TEMP. 500°F
BD-KI
H2 RECOMB. SYSTEM CAC-RMS-11A
ON
AUTO
OFF
X5 TO SHEET 521-1
REST OF H2 RECOMB. SYSTEM
ON
OFF
X6 TO SHEET 521-1

AMENDMENT NO. 10
July 1980

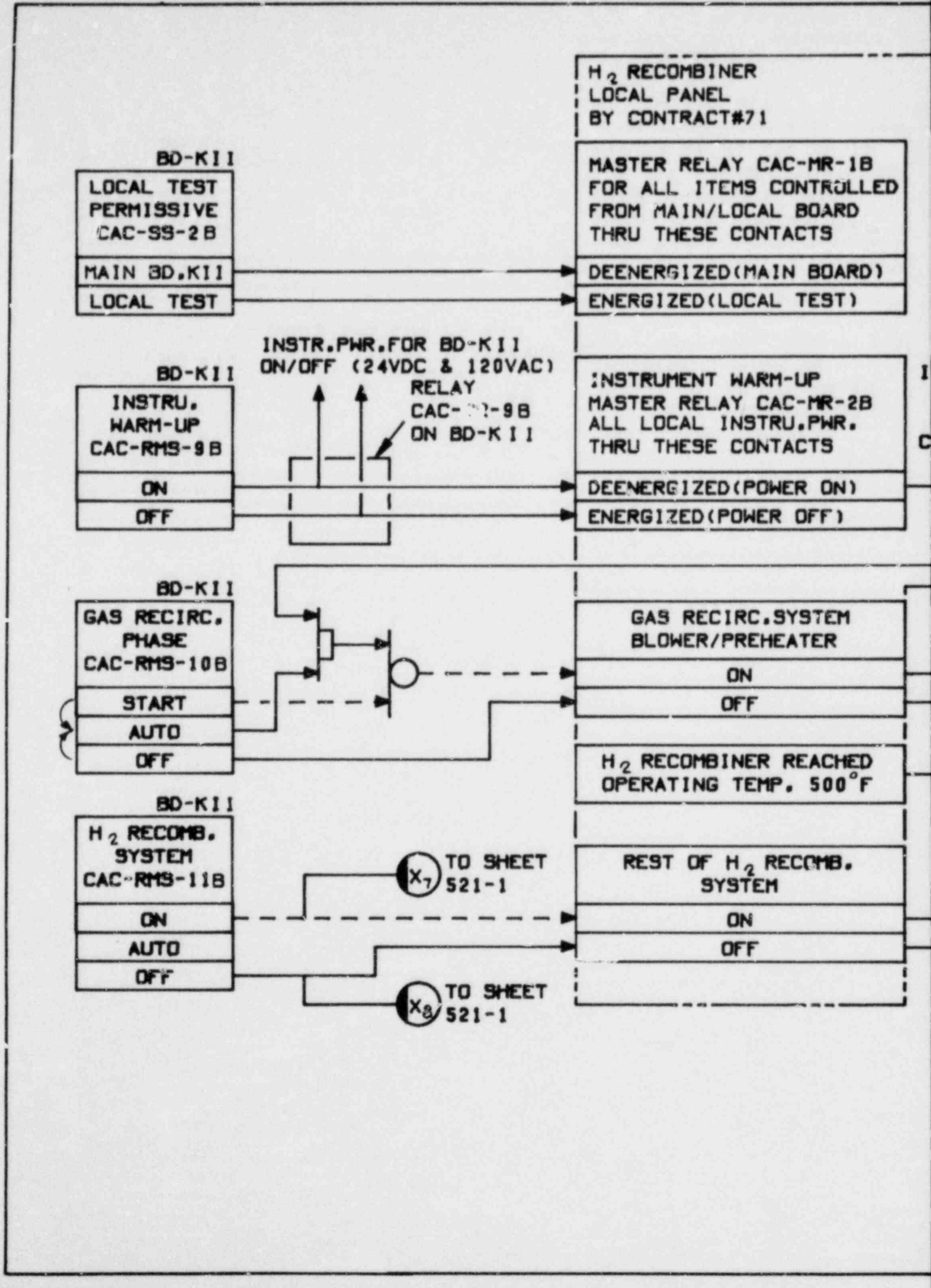


I & C DWG. NO. M-620

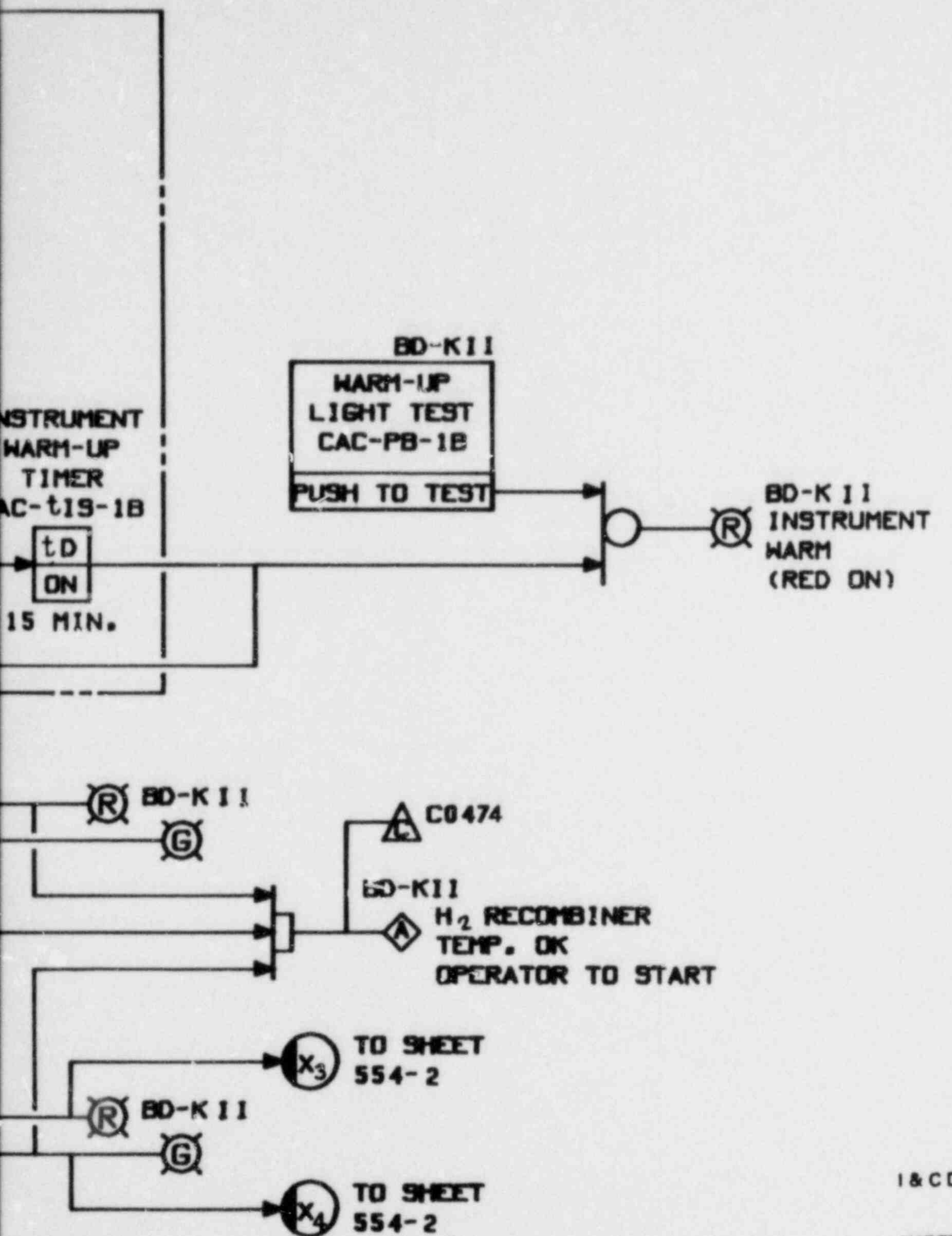
SHEET 554-3

H_2 RECOMBINER CAC-HR-1A
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM CONTAINMENT ATMOSPHERE CONTROL SYSTEM	FIGURE 7.3-21c


H2 RECOMBINER
LOCAL PANEL
BY CONTRACT#71
BD-KII
LOCAL TEST
PERMISSIVE
CAC-SS-2B
MAIN BD.KII
LOCAL TEST
MASTER RELAY CAC-MR-1B
FOR ALL ITEMS CONTROLLED
FROM MAIN/LOCAL BOARD
THRU THESE CONTACTS
DEENERGIZED(MAIN BOARD)
ENERGIZED(LOCAL TEST)
INSTR.PWR.FOR BD-KII
ON/OFF (24VDC & 120VAC)
BD-KII
INSTRU.
WARM-UP
CAC-RMS-9B
ON
OFF
RELAY
ON BD-KII
INSTRUMENT WARM-UP
MASTER RELAY CAC-MR-2B
ALL LOCAL INSTRU.PWR.
THRU THESE CONTACTS
DEENERGIZED(POWER ON)
ENERGIZED(POWER OFF)
BD-KII
GAS RECIRC.
PHASE
CAC-RMS-10B
START
AUTO
OFF
GAS RECIRC.SYSTEM
BLOWER/PREHEATER
ON
OFF
H2 RECOMBINER REACHED
OPERATING TEMP. 500°F
BD-KII
H2 RECOMB.
SYSTEM
CAC-RMS-11B
ON
AUTO
OFF
TO SHEET
521-1
REST OF H2 RECOMB.
SYSTEM
ON
OFF
TO SHEET
521-1

AMENDMENT NO. 10
July 1980



I & C DWG. NO. M-620

SHEET 554-4

H_2 RECOMBINER CAC-HR-1B
CONTROL LOGIC DIAGRAM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL LOGIC DIAGRAM CONTAINMENT ATMOSPHERE CONTROL SYSTEM	FIGURE 7.3-21d

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

7.4.1 DESCRIPTION

This section discusses the instrumentation and controls of the following systems required for safe plant shutdown:

a. Reactor Core Isolation Cooling (RCIC) System

b. Standby Liquid Control System (SLCS)

c. RHR Shutdown Cooling Mode (RSCM)

d. Remote Shutdown System (RSS)

The sources which supply power to the safe shutdown systems originate from on-site AC and/or DC safety-related busses. Refer to Chapter 8 for a complete discussion of the safety-related power sources.

7.4.1.1 Reactor Core Isolation Cooling (RCIC) System

a. RCIC System Function

The reactor core isolation cooling system (see 5.4.6.2) instrumentation is designed to maintain or supplement reactor vessel water inventory during the following conditions:

1. Normal Operation. When the reactor vessel is isolated from its primary heat sink (the main condenser) and maintained in the hot standby condition;

2. Normal Operation. When the reactor vessel is isolated and accompanied by a loss of normal coolant flow from the reactor feedwater system;

3. When the plant is being shutdown and normal coolant flow from the feedwater system is started before the reactor is depressurized to a level where the reactor shutdown cooling mode of the RHR system can be placed into operation.

4. When required as a backup to the High Pressure Core Spray System to mitigate the consequences of the rod drop accident by automatically supplying cooling water to the reactor if vessel low water level is sensed.

b. RCIC System Operation

Schematic arrangements of system mechanical equipment is shown in Figure 7.4-1 (RCIC P&ID). RCIC system component control logic is shown in Figure 7.4-2 (RCIC FCD). Instrumentation Specifications are listed in Tables 7.4-1 and 7.4-2. Plant layout drawings, and electrical schematics are identified in 1.7. Operator Information displays are shown in Figure 7.4-1 (RCIC P&ID) and Figure 7.4-2 (RCIC FCD).

The RCIC System can be initiated either manually or automatically. The control room operator can initiate RCIC by operating the manual initiation pushbutton which simulates an automatic initiation or by activating each piece of equipment sequentially as required.

RCIC is automatically initiated by four redundant differential pressure switches, arranged in a one-out-of-two-twice logic configuration, which sense reactor vessel low water level (Trip Level 2).

The RCIC steam line isolation and the turbine steam exhaust motor-operated (MO) valves are keylocked in the open position, the turbine trip and throttle valve is normally open and require no change of position for automatic system initiation.

The RCIC system responds to an automatic initiation signal as follows (actions are simultaneous unless stated otherwise):

1. The pump suction from the condensate storage tanks valve MO F010 is signaled open,

2. To ensure pump discharge flow is directed to the reactor vessel only, the test return line to the condensate storage tanks valves MO F022 and MO F059 are signaled closed.

3. The turbine steam inlet and the turbine lube oil cooler cooling water supply valves MO F045 and MO F046 are signaled to open,

4. When the turbine steam inlet valve MO F045 is open, the RCIC pump discharge to reactor vessel valve MO F013 is signaled open, valve MO F013 is prohibited from opening or if open, automatically closes when MO F045 or the turbine trip and throttle valve is closed.

5. The barometric condenser vacuum tank vacuum pump is signaled to start,

6. When valve MO F045 leaves the closed position the RCIC turbine is accelerated in speed until the automatic flow controller setpoint is reached and the system discharge flow is controlled by the turbine electronic governor mechanism.

During system operation if the barometric condenser vacuum tank water level becomes high the condenser condensate discharge pump is automatically started and the condensate returned to the RCIC pump suction. When the system is not operating excess tank water is discharged through isolation valves AO F004 and AO F005 to the clean condensate system.

In the event that the water level in the condensate storage tanks should become low the RCIC pump suction is automatically transferred from the condensate storage tank to the suppression pool by opening valve MO F031. Once valve F031 is fully open the condensate storage tank valve MO F010 is automatically closed.

The RCIC system includes design feaures which provide system equipment protection or accomplish primary containment isolation if certain types of abnormal events occur. The turbine is automatically shut down by closing the turbine trip and throttle valve if any of the following conditions are detected:

1. Turbine overspeed
2. High turbine exhaust pressure
3. RCIC isolation signal
4. Low pump suction pressure
5. Reactor vessel high water level (Trip Level B)
6. Manual trip actuated by the control room operator.

To protect the RCIC pump from overheating during low flow conditions the pump discharge flow and pressure is monitored. If the pump discharge pressure switch indicates the pump is running and the pump discharge flow switch indicates low flow, the minimum flow return line valve MO F019 is automatically opened. The minimum flow valve is automatically closed when flow is normal, or when either the turbine trip and throttle valve or the steam inlet valve MO F045 is closed.

Air operated (AO) valves F025, F026, and F054 and a condensate drain pot are provided in a drain pipeline arrangement just upstream of the turbine supply valve. Upon receipt of an RCIC initiation signal, the drainage path is isolated by closing AO F025 and AO F026. The water level in the steam line drain condensate pot is controlled by a level switch and a valve AO F054 which energizes to allow condensate to flow out of the drain pot by bypassing the steam trap.

RCIC system turbine exhaust line vacuum breaker valves MO F080 and MO F086 are normally open but close automatically following system trip on low steam line pressure if drywell pressure exceeds the setpoint.

The leak detection portion of RCIC system will automatically signal the steamline warmup valve MO F076 closed, override the keylock control switch position and signal the steamline inboard isolation valve MO F063 and the outboard steamline isolation valve MO F008 closed if any of the following abnormal conditions exist. For a complete description of the RCIC system leak detection isolation signals, see 7.6.1.3.1.

1. Redundant differential temperature and ambient temperature switches sense RCIC and RHR equipment area ventilation air inlet and outlet high temperature or high ambient temperature. RHR heat exchanger steam line valve MO F064 is signaled closed by the RHR temperature switches.

2. Redundant differential temperature and ambient temperature switches sense RCIC pipe routing area ventilation air inlet and outlet high temperature or high ambient temperature.

3. Redundant differential pressure switches sense RCIC or RHR/RCIC steam line high flow or instrument line break.

4. Redundant pressure switches sense RCIC turbine exhaust diaphragm high pressure. Both switches must actuate to cause isolation.

5. A pressure switch senses RCIC low steam supply pressure.

The RCIC system may be isolated after initiation by the control room operator by actuation of a pushbutton which causes the outboard steamline isolation valve to close.

7.4.1.2 Standby Liquid Control System (SLCS)

a. SLCS Function

The Standby Liquid Control System (see 9.3.5) instrumentation is designed to initiate injection of a liquid neutron absorber into the reactor. Other instrumentation is provided to maintain this liquid chemical solution well above saturation temperature in readiness for injection.

The SLCS is a redundant method of manually shutting down the reactor to cold shutdown conditions from normal operation or from anticipated transient conditions when manual control rod insertion capability is lost.

b. SLCS Operation

Schematic arrangements of system mechanical equipment is shown in Figure 7.4-3 (SLCS P&ID). SLCS component control logic is shown in Figure 7.4-4 (SLCS FCD) and 1.7. Operator informaion displays are shown in Figures 7.4-3 and 7.4-4.

The SLCS is initiated by the control room operator by turning a keylocked switch to the system A or system B "RUN" position. The key is removable in the center "STOP" position. Should the selected pump fail to start, the key switch may be used to select the alternate pump loop.

When the SLCS is initiated, both the explosive-operated valves fire and both tank discharge valves MO F001 A, B, start to open immediately. The pump that has been selected for injection will not start until one of the tank discharge valves is at least 90% open.

Pumps are interlocked so that either the storage tank discharge valve or the test tank discharge valve must be open for the pump to run. When the SLCS is initiated the outboard isolation valve of the reactor water cleanup system is automatically closed.

7.4.1.3 RHRS/Reactor Shutdown Cooling Mode (RSCM)

a. RSCM Function

The shutdown cooling mode (see 5.4.7.1) of the RHR System is used during a normal reactor shutdown or for long term cooling after vessel water level has been restored following accident conditions.

The RSCM consists of instrumentation designed to provide decay heat removal capability for the core by accomplishing the following:

1. Reactor cooling during shutdown operation after the vessel pressure is reduced to approximately 50 psig.

2. Cooling the reactor water to a temperature at which reactor refueling and servicing can be accomplished.

3. Diverting part of the shutdown flow to the reactor vessel head to condense the steam generated from the hot walls of the vessel while it is being flooded.

b. RSCM Operation

See 5.4.7.2.6(a) for a complete description of the RSCM operation.

7.4.1.4 Remote Shutdown System (RSS)

a. RSS Function

The RSS is designed to achieve a cold reactor shutdown from outside the main control room following these postulated conditions:

1. The plant is at normal operating conditions and all plant personnel have been evacuated from the main control room and it is inaccessible.

2. The initial event that causes the main control room to become inaccessible is assumed to be such that the reactor operator can manually scram the reactor before leaving the main control room.

3. The main turbine pressure regulators may be controlling reactor pressure via the bypass valves. However, in the interest of demonstrating that the plant can accommodate even loss of the turbine controls, it is assumed that this turbine generator control panel function is also lost. Therefore, main steam line isolation is assumed to occur at a specified low turbine inlet pressure and

reactor pressure is relieved through the relief valves to the suppression pool.

4. The reactor feedwater system which is normally available is also assumed to be inoperable. Reactor vessel water inventory is provided by the RCIC system.

5. Division 1 Emergency DC power is assumed to be available.

The RSS is required only during times of Main Control Room inaccessibility when normal plant operating conditions exist, i.e., no transients or accidents are occurring. For this reason the RSS function is not single failure proof and only the equipment which interfaces directly with safety-related equipment (RHR, RCIC, etc.) is required to be of a safety-related quality.

b. Remote Shutdown System Operation

Some of the existing systems used for normal reactor shutdown operation are also utilized in the remote shutdown capability to shutdown the reactor from outside the main control room. The remote shutdown capability is designed to control the required shutdown systems from outside the main control room irrespective of shorts, opens, or grounds in the control circuit in the main control room that may have resulted from an event causing an evacuation. The functions needed for remote shutdown control are provided with manual transfer switches which override controls from the main control room and transfer the controls to the remote shutdown control. Remote shutdown control is not possible without actuation of the transfer devices. All necessary power supplies and control logic are also transferred. Operation of the transfer devices causes an alarm in the main control room. Access to the remote shutdown panel is administratively and procedurally controlled. All system equipment (i.e., valves and pumps) necessary for proper system lineup and complete system control are located on the remote shutdown panel.

Manual activation of safety/relief valves and the initiation of Reactor Core Isolation Cooling (RCIC) system will maintain reactor water inventory and bring the reactor to a hot shutdown condition after scram. During this phase of shutdown, the suppression pool will be cooled by operating the Residual Heat Removal (RHR) system in the suppression pool cooling mode. Reactor pressure will be controlled and core decay and sensible heat rejected to the suppression pool by relieving steam pressure through the relief valves.

Manual operation of the relief valves will cool the reactor and reduce its pressure at a controlled rate until reactor pressure becomes so low that the RCIC system is unable to sustain operation. The RHR system will then be operated in the shutdown cooling mode using the RHR system heat exchanger to cool reactor water and bring the reactor to the cold low pressure condition.

The following RCIC System equipment/functions have transfer and control switches located on the remote shutdown control panel:

F008 - Motor-operated valve (steam supply line isolation)
F045 - Motor-operated valve (steam to turbine)
F010 - Motor-operated valve (pump suction from condensate storage)
F013 - Motor-operated valve (pump discharge to reactor)
F019 - Motor-operated valve (pump bypass to suppression pool)
C001 - Motor-operated valve (trip throttle valve)
F046 - Motor-operated valve (cooling water supply valve)
F064 - Motor-operated valve (RHR Cond. heat exch. steam line isolation valve)
F063 - Motor-operated valve (steam supply isolation valve)
F031 - Motor-operated valve (pump suction from suppression pool)
F022 - Motor-operated valve (test bypass to condensate storage)
P1 - Condensate pump from RCIC vacuum tank
P2 - RCIC vacuum pump
F069 - Motor-operated valve (vacuum pump discharge to suppression pool)
F068 - Motor-operated valve (turbine exhaust to suppression pool)
F059 - Motor-operated valve (test bypass to condensate storage)
F080 - Motor-operated valve (vacuum breaker isolation)
F086 - Motor-operated valve (vacuum breaker isolation)
See Figure 7.4-1 (RCIC P&ID).

The following RCIC System instrumentation is provided on the remote shutdown control panel:

1. RCIC Flow Controller and indicator

2. RCIC Turbine Speed

3. Indicating lights are provided for:

   a) Turbine tripped

b) Turbine bearing oil low pressure

c) Turbine governor bearing oil temperature high

d) Turbine coupling end bearing oil temperature high

The following RHR System equipment/functions have transfer and control switches located at the remote shutdown control panel:

F004B - Motor-operated valve (suppression pool pump suction)
F006B - Motor-operated valve (shutdown cooling pump suction)
C002B - Residual heat removal pump
F047B - Motor-operated valve (heat exchanger inlet)
F048B - Motor-operated valve (heat exchanger bypass)
F003B - Motor-operated valve (heat exchanger outlet)
F026B - Motor-operated valve (condensate discharge to RCIC suction line)
F087B - Motor-operated valve (steam reducing bypass)
F068B - Motor-operated valve (heat exchanger cooling water outlet)
F006A - Motor-operated valve (shutdown cooling pump suction)
F009 - Motor-operated valve (inboard shutdown isolation)
F008 - Motor-operated valve (outboard shutdown isolation)
F016B - Motor-operated valve (containment spray)
F027B - Motor-operated valve (suppression pool spray)
F042B - Motor-operated valve (LPCI injection)
F053B - Motor-operated valve (shutdown cooling injection)
F052B - Motor-operated valve (steam reducing isolation)
F024B - Motor-operated valve (test line)
F011B - Motor-operated valve (condensate discharge to suppression pool)
F049B - Motor-operated valve (discharge to radwaste)
F023 - Motor-operated valve (reactor head spray)
See Figure 7.3-13 (RHR P&ID).

The following RHR instrumentation is located on the remote shutdown control panel:

1. RHR flow indicator

The following nuclear boiler system equipment have transfer and control switches located on the remote shutdown control panel:

Three air operated relief valves, (non-ADS)

The following nuclear boiler instrumentation is provided on the remote shutdown control panel:

1. Level indicator

2. Reactor pressure indicator

The following recirculation system valve has transfer and control switches located on the remote shutdown control panel:

F023A - Motor-operated valve (recirculation pump suction)

The following standby service water system (SSW) equipment/ functions have transfer and control switches located at the remote shutdown control panel:

1. SSW pump 1B

2. SSW return to cooling tower, shut off valve SW-V-70B.

3. SSW return to cooling tower, shut off valve SW-V-69B.

4. SSW return to spray pond, valve SW-V-12B.

5. SSW pump 1B discharge, valve SW-V-2B.

6. RHR pump E12-C002B cooling water shut off, valve SW-V-24B.

The following primary containment environmental parameters are indicated on the remote shutdown panel:

1. Drywell air temperature at several locations.

2. Drywell pressure.

3. Suppression chamber air temperature.

4. Suppression pool water temperature.

5. Suppression pool water level.

7.4.1.5 Design Basis

The safe shutdown systems are designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are presented in that chapter.

a. Variables monitored to provide protective actions

The following variables are monitored in order to provide protective actions to the safe shutdown systems:

1. RCIC - Reactor vessel low water level (Trip Level 2)

All other safe shutdown systems are initiated by operator actions.

The plant conditions which require protective action involving safe shutdown are described in Chapter 15 and Appendix 15A.

b. Location and Minimum Number of Sensors

See Chapter 16 for the minimum number of sensors required to monitor safety-related variables. There are no sensors in the safe shutdown systems which have a spatial dependence.

c. Prudent Operational Limits

Prudent operational limits for each safety-related variable trip setting are selected with sufficient margin so that a spurious safe shutdown system initiation is avoided. It is then verified by analysis that the release of radioactive materials, following postulated gross failures of the fuel or the nuclear system process barrier, is kept within acceptable bounds.

d. Margin

The margin between operational limits and the limiting conditions of operation of safe shutdown systems are those parameters as listed in Chapter 16, "Technical Specifications".

e. Levels

Levels requiring protective action are established in Chapter 16, "Technical Specifications".

f. Range of Transient, Steady State, and Environmental Conditions

Refer to Tables 3.11-1 through 3.11-5 and 3.1.2.1.4.1 for environmental conditions. Refer to 8.2.1 and 8.3.1 for the maximum and minimum range of energy supply to the Safe Shutdown Systems instrumentation and controls. All safety-related instrumentation and controls are specified and purchased to withstand the effects of energy supply extremes.

g. Malfunctions, Accidents, and Other Unusual Events Which Could Cause Damage to Safety System

Chapter 15, "Accident Analysis" describes the following credible accidents and events: floods, storms, tornados, earthquakes, fires, LOCA, pipe break outside containment, and feedwater line break. Each of these events is discussed below for the safe shutdown systems.

1. Floods

The buildings containing safe shutdown system components have been designed to meet the PMF (Probable Maximum Flood) at the site location. This ensures that the buildings will remain water-tight under PMF conditions including wind generated wave action and wave runup. For a discussion of internal flooding protection refer to 3.4.1.4.1.2, 3.4.1.5.2, and 3.6.

2. Storms and Tornados

The buildings containing safe shutdown system components have been designed to withstand meteorological events described in 3.3.

3. Earthquakes

The structures containing safe shutdown system components have been seismically qualified as described in 3.7 and 3.8, and will remain functional during and following a safe shutdown earthquake (SSE). Seismic qualification of instrumentation and electrical equipment is discussed in 3.10.

4. Fires

To protect the safe shutdown systems in the event of a postulated fire, the redundant portions of the systems are separated by fire barriers or physical distance. The use of separation and fire barriers ensures that even though some portion of the systems may be affected, the safe shutdown systems will continue to provide the required protective action. A fire detection system using heat detectors and product of combustion detectors is provided in PGCC floor sections and in panels containing safe shutdown system components mounted on these floor sections. A Halon fire suppression system is provided in the same areas.

5. LOCA

The safe shutdown systems components located inside the drywell which are functionally required following a LOCA have been environmentally qualified to remain functional as discussed in 3.11 and indicated in Table 3.11-1.

6. Pipe Break Outside Secondary Containment

This condition will not affect the Safe Shutdown Systems. Refer to 3.6.

7. Missiles

Protection for safe shutdown systems is described in 3.5.

h. Minimum Performance Requirements

Minimum performance requirements for safe shutdown systems instrumentation and controls are provided in Chapter 16, "Technical Specifications".

7.4.1.6 Final System Drawings

The final system drawings including:

1. Process and Instrumentation Diagrams (P&ID)
2. Functional Control Diagrams (FCD)

have been provided for the safe shutdown systems.

Functional and architectural design difference between the PSAR and FSAR are listed in Table 1.3-8.

7.4.2 ANALYSIS

The safe shutdown systems are designed such that loss of instrument air, a plant load rejection, or a turbine trip will not prevent the completion of the safety function.

7.4.2.1 Conformance To 10 CFR 50 Appendix A - General Design Criteria

The following is a discussion of conformance to those general design criteria which apply specifically to the safe shutdown systems. Refer to 7.1.2.2 for a discussion of General Design Criteria which apply equally to all safety-related systems.

a. General Design Criteria 19 - Control Room

The remote shutdown system consists of equipment located outside the control room which is sufficient to provide and assure prompt hot shutdown of the reactor and to maintain safe conditions during hot shutdown. The equipment also provides capability for subsequent cold shutdown of the reactor.

b. General Design Criterion 26 - Reactivity Control System Redundancy and Capability

SLCS provides an independent reactivity control system redundant to manual control rod movement.

c. General Design Criterion 29 - Protection Against Anticipated Operational Occurrences

The SLCS maintains the reactor sub-critical by introducing poison into the reactor in the event the manual insertion of control rods cannot achieve subcriticality in the reactor.

d. General Design Criterion 34 - Residual Heat Removal

The reactor shutdown cooling mode of the residual heat removal system removes residual heat from the reactor when it is shutdown and the main steamlines are isolated, to maintain the fuel and reactor coolant pressure boundary within design limits. Redundant cooling routes are provided to meet the single failure criteria.

7.4.2.2 Conformance To IEEE Standards

The following is a discussion of conformance to those IEEE Standards which apply specifically to the Safe Shutdown Systems. Refer to 7.1.2.3 for a discussion of IEEE Standards which apply equally to all safety-related systems.

a. IEEE 279-1971

The reactor shutdown cooling mode of the residual heat removal system utilizes the same equipment used by the LPCI mode. Therefore, refer to 7.3.2 for the RSCM standards and regulatory compliance.

Conformance of the remote shutdown system to IEEE Standards is provided in the analyses section for each system whose instrumentation and controls are part of the remote shutdown system (Refer to 7.3, 7.4, 7.5, and 7.6) and as noted below.

1. General Functional Requirement (IEEE 279-1971, Paragraph 4.1)

RCIC is automatically initiated when reactor vessel water level is determined to be below a predetermined limit.

SLCS is initiated by the control room operator. Display instrumentation in the control room provides the operator with information on reactor vessel water level, pressure, neutron flux level, control rod position, and scram valve status allowing assessment of the need for initiation of the SLCS.

2. Single-Failure Criterion (IEEE 279-1971, Paragraph 4.2)

The RCIC system is not required to meet the single-failure criterion. The RCIC initiation sensors and associated logic do, however, meet the single-failure criterion for automatic system initiation. The single-failure criteria is met through physical and electrical separation of equipment as described in 8.3.1.4.

SLCS serves as backup to the control rod drive (CRD) system for controlling reactivity if the CRD fails. It is not necessary for SLCS to meet the single failure criterion. However, the pumps and motors, the explosive valves, and the storage tank outlet valves are redundant so that no single failure in these components will prevent initiation of SLCS.

3. Quality of Components and Modules (IEEE 279-1971, Paragraph 4.3)

Refer to 3.11 for RCIC, SLCS and RSS conformance.

4. Equipment Qualification (IEEE 279-1971, Paragraph 4.4)

Vendor certification requires that the safety-related RCIC, SLCS and RSS sensors, manual switches and logic components perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in conjunction with the existing field experience with these components in this application, will serve to qualify these components.

Qualification tests of the relay panels are conducted to confirm their adequacy for this service. In-situ operational testing of these sensors, channels, and the entire protection system will be performed during the preoperational test phase.

For a complete discussion of RCIC and SLCS equipment qualification refer to 3.5, 3.6, 3.10, and 3.11.

5. Channel Integrity (IEEE 279-1971, Paragraph 4.5)

For a discussion of RCIC, SLCS and RSS Channel Integrity under all extremes of conditions described in 7.4.1.2, refer to 3.11.

6. Channel Independence (IEEE 279-1971, Paragraph 4.6)

Channel independence is maintained through application of the WNP-2 separation criteria as described in 8.3.1.4.

7. Control and Protection Interaction (IEEE 279-1971, Paragraph 4.7)

The RCIC and SLCS systems have no interaction with plant control systems.

8. Derivation of System Inputs (IEEE 279-1971, Paragraph 4.8)

All inputs to the RCIC system that are essential to its operation are direct measures of appropriate variables.

SLCS display instrumentation in the control room provides the operator with directly measured information on reactor vessel water level, pressure, neutron flux level, control rod position and valve status. Based on this information the operator can assess the need for SLCS.

9. Capability for Sensor Checks (IEEE 279-1971, Paragraph 4.9)

Refer to 7.4.2.3, Regulatory Guide 1.22.

10. Capability for Test and Calibration (IEEE 279-1971, Paragraph 4.10)

Refer to 7.4.2.3, Regulatory Guide 1.22.

11. Channel Bypass or Removal from Operation (IEEE 279-1971, Paragraph 4.11)

Calibration of a sensor which introduces a single instrument channel trip will not cause a protective action without the coincident trip of a second channel. Removal of a sensor from operation during calibration does not prevent the redundant instrument channel from functioning.

The discharge pump motors for SLCS are redundant, so that one motor may be removed from service during normal plant operation.

12. Operating Bypasses (IEEE 279-1971, Paragraph 4.12)

There are no operating bypasses within the RCIC system or the SLCS.

13. Indication of Bypasses (IEEE 279-1971, Paragraph 4.13)

For a discussion of bypass and inoperability indication refer to 7.1, Regulatory Guide 1.47.

14. Access to Means for Bypassing (IEEE 279-1971, Paragraph 4.14)

Access to means of bypassing any safety action or function for RCIC and SLCS is under the administrative control of the control room operator. The operator is alerted to bypasses as described in 7.1, Regulatory Guide 1.47.

Control switches which allow safety system bypasses are keylocked. All keylock emergency switches in the control room are designed such that their key can only be removed when the switch is in the "accident" or "safe" position. All keys will normally be removed from their respective switches during operation and maintained under the control of the shift supervisor. Further, the key locker will be audited once per day by the shift supervisor. Should a key be required to change a valve position, it will be obtained from the shift supervisior via approved key control procedures.

15. Multiple Set Points (IEEE 279-1971, Paragraph 4.15)

There are no multiple setpoints within the RCIC or SLCS systems.

16. Completion of Protective Action Once it is Initiated (IEEE 279-1971, Paragraph 4.16)

Once RCIC is initiated by reactor vessel low water level the logic seals-in and system operation may be terminated by the operator only if the water level returns to normal. The system is automatically stopped on high vessel water level, system malfunction trip signals or if steam supply pressure drops below that necessary to sustain turbine operation.

The SLCS explosive valves remain open once fired. The injection valves will not close and discharge pump motors will continue to run unless terminated by operator action.

17. Manual Initiation (IEEE 279-1971, Paragraph 4.17)

Refer to 7.4.2, Regulatory Guide 1.62, for a discussion of the manual initiation of RCIC and SLCS.

18. Access to Set Point Adjustment (IEEE 279-1971, Paragraph 4.18)

All access to setpoint adjustments for RCIC are under administrative control of the control room operator.

The operation of SLCS is not dependent on or affected by any set point adjustment or calibration.

19. Identification of Protective Actions (IEEE 279-1971, Paragraph 4.19)

Automatic initiation of the RCIC system is annunciated in the control room.

The explosive valve status of SLCS, once fired, is indicated in the control room.

20. Information Readout (IEEE 279-1971, Paragraph 4.20)

The RCIC system is designed to provide the operator with accurate and timely information pertinent to its status. It does not give anomalous indications confusing to the operator.

The SLCS discharge pressure of sodium pentaborate pumps and storage tank level for the SLCS is indicated in the control room.

21. System Repair (IEEE 279-1971, Paragraph 4.21)

The RCIC and SLCS systems are designed to permit repair or replacement of components during normal plant operation.

Recognition and location of a failed component will be accomplished during periodic testing or by annunciation in the control room.

22. Identification (IEEE 279-1971, Paragraph 4.22)

All controls and instruments for RCIC and SLCS are located in separate sections of the main control room panel and clearly identified by nameplates. Relays are located in separate panels for RCIC and SLCS use only. Relays and panels are identified by nameplates. All wiring and cabling is labeled to indicate its divisional assignment as well as its system assignment (see 8.3.1.3).

7.4.2.3 NRC Regulatory Guides Conformance

Regulatory Guide conformance for remote shutdown control and instrumentation is provided in the analysis sections of Chapter 7 for each system whose instrumentation and controls are part of the remote shutdown system.

Conformance to Regulatory Guides for the RHR Shutdown Cooling Mode is discussed in 7.3.2.

The following is a discussion of conformance to those Regulatory Guides which apply specifically to the RCIC system and/or the SLCS. Refer to 7.1.2.4 for a discussion of Regulatory Guides which apply equally to all safety-related systems.

a. Regulatory Guide 1.22 - Periodic Testing of Protection System Actuation Functions

The RCIC system is capable of being completely tested during normal plant operation to verify that each element of the system, is capable of performing its intended safety function.

All sensors for RCIC are installed with calibration taps and instrument valves to permit testing during normal plant operation by valving out the sensors and supplying a test pressure source.

The explosive valve control circuits continuity is continuously monitored and is annunciated in the control room.

The explosive valves may be tested during plant shutdown. The explosive valve control circuits are continuously monitored and annunciated in the control room. The remainder of the SLCS may be tested during normal plant operation to verify that each element is capable of performing its intended function.

Testing of RCIC system and SLCS sensors during normal plant operation is accomplished by valving out each sensor from its process line and applying a test pressure source. This verifies the operability of the sensor, its calibration range, and the operability of associated control room logic components.

b. RG 1.53 - Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

See IEEE 279, Para 4.2, in 7.4.2 for RCIC and SLCS.

c. RG 1.62 - Manual Initiation of Protective Actions

The RCIC system is initiated at the system level manually from the main control room by actuation of an armed pushbutton which simulates an automatic initiation.

The SLCS is initiated manually at the system level from the main control room by actuation of the system pump select/start switch which starts the selected pump and fires the associated squib valve.

TABLE 7.4-1

REACTOR CORE ISOLATION COOLING SYSTEM INSTRUMENT SPECIFICATIONS

RCIC Function	Instrument	Instrument Range (2)	Trip Setting (3)	Margin (4)	Required Accuracy (5)	Response Time (5)
Reactor Vessel High Water Level (Level 8)	Level Switch (B22-N024 B,D)	0-60" (1)	+55.5"	-	±3"	-
RCIC System Pump Low Suction Pressure	Pressure Switch (E51-N006)	30" Hg Vac. 10 psig	15" Hg Vac.	-	2 psi	-
Reactor Vessel Low Water Level (Level 2)	Level Switch (B22-N037 A-D)	-150/0/+60" (1)	-38"	-	±7.5"	-
Drywell High Pressure	Pressure Switch (B22-N048 A-D)	0.25 - 12 psig	2 psig		.06 psi	0.6 sec.
Condensate Storage Tanks Low Water Level	Level Switch (E51-N015 A,B)	-	0" (11,500 GAL.)		0.5	

NOTES FOR TABLE 7.4-1

(1) Instrument zero equal to 527.5" above Vessel zero.

(2) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(3) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary.

(4) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(5) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.4-2

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR THE REACTOR CORE ISOLATION COOLING SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Vessel Water Level High	2	2
Turbine Exhaust High Pressure	4	2
RCIC Pump Low Suction Pressure	1	1
Reactor Vessel Low Water Level	4	2
RCIC Steam Supply Low Pressure	4	2
RCIC Steam Supply High Flow	4	2
Drywell High Pressure	4	2
Condensate Storage Tanks Low Water Level	2	1

BLANK

7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

7.5.1 DESCRIPTION

7.5.1.1 General

Section 7.5 describes the instrumentation which provides information to the operator to enable him to assess the status of safety-related systems, and the need to perform required safety functions.

The Safety-Related Display Instrumentation is listed in Table 7.5-1. It tabulates equipment illustrated on the various system P&IDs, IEDs, and FCDs located in 7.2, 7.3, 7.4, and 7.6.

The instrumentation and ranges shown in Table 7.5-1 are selected on the basis of giving the reactor operator the necessary information to perform normal plant operations and yet the capability to track process variables pertinent to safety following design basis accidents.

The following information is provided to the control room operator to monitor reactor conditions and allow assessment of safety system status following a design basis accident.

The power sources to the instrumentation described in this section originate from either the Division 1, Division 2, or Division 3 safety-related emergency AC and/or DC busses unless indicated otherwise.

7.5.1.1.1 Reactor Water Level

Wide range water level is sensed by two redundant differential pressure transmitters and is recorded in the control room. The range of the recorded level is from the top of the feedwater control range (just above the high level turbine trip point) down to a point near the top of the active fuel.

7.5.1.1.2 Reactor Pressure

Reactor pressure is sensed by redundent pressure transmitters and recorded in the control room.

7.5.1.2 Reactor Shutdown Indication

The following information is provided to the control room operator to monitor reactor shutdown.

1. Control rod status lamps indicating each rod fully inserted.

   a) Control rod scram pilot valve position status lamps indicating open valves.

2. Neutron monitoring power range channels and recorders downscale. The power sources are from RPS MG sets.

3. Annunciators for RPS variables and trip logic in the tripped state.

4. The process computer provides logging of trips and control rod position log and provides thermal hydraulic information to the operator which he uses to keep the plant operating within technical specification limits.

7.5.1.3 Primary Containment and Reactor Vessel Isolation Indication

The following information is provided to the control room operator to monitor the integrity of the primary containment.

1. Isolation valve position lamps indicating valve closure.

2. Main steam line flow indication.

3. Annunciators for the primary containment and reactor vessel isolation system variables and trip logic in the tripped state.

4. Process computer logging of trips.

7.5.1.4 ECCS and RCIC Indication

The following information is provided to the control room operator to monitor ECCS and RCIC system status.

1. Annunciators for HPCS, LPCS, RHR, ADS and RCIC sensor initiation logic trips.

2. Flow and/or pressure indications for each ECCS and RCIC are provided.

3. ECCS and RCIC valve position indication.

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

4. Process computer logging of trips in the ECCS and RCIC.

5. Relief valve discharge pipe temperature monitors.

7.5.1.5 Containment Indictations

The following information is provided to the control room operator to monitor primary containment status.

1. Primary Containment Pressure Monitoring

There are two redundant drywell pressure monitoring sensors. Both cover a pressure range during reactor normal operation and following a loss-of-coolant accident. The instrumentation consists of two redundant transmitters and recorders.

2. Primary Containment Temperature

Containment temperature is monitored continuously by redundant indicators and recorders in the main control room. Points of measurement are as follows:

No. of Points	Description	Range	Type of Readout
4	Air inlet vicinity recirculation pump motors	50-1701°F	Recorders
5	Fan coil inlets	50-1701°F	Recorders
5	Fan coil outlets	50-1701°F	Recorders
3	Sacrificial shield space (lower area)	50-1701°F	Indicators
3	Sacrificial shield space (upper area)	50-4001°F	Indicators & Computer
3	Control drive area	50-4001°F	Indicators
3	Reactor pressure vessel head flange area	50-4001°F	Indicators
5	Upper drywell area	50-4001°F	Recorders
2	Return duct from head area	50-400°F	Indicators

No. of Points	Description	Range	Type of Readout
5	Upper return ring header	50-4001°F	Indicators
5	Safety and relief valve area	50-4001°F	Indicators

3. Primary Containment Moisture

Containment moisture level is monitored by five (5) dew point sensors located immediately outside of the primary containment.

A containment air sample is circulated through the sensor units and returned to containment.

Readout is on strip chart recorders in the main control room with a range of 0-150% humidity.

4. Primary Containment Radiation

The atmosphere of the primary containment is monitored for lowlevel (leak detection) and high-level (LOCA) radioactivity and recorded in the control room on two redundant recorders. Offline monitors consist of local racks in the reactor building sample room, which contain shielded radiation detectors, and panels in the main control room, which contain count ratemeters, recorders, and controls.

The leak detection monitor is a three-channel unit containing particulate, and noble gas scintillation detectors. The detectors are of high sensitivity to detect small leaks in the reactor coolant pressure boundary.

Sample gas piped from the containment to the local leak detector racks and vent gas is pumped back to the containment. The control room operator has complete control of the operation and checking of the monitor system from the main control room.

The LOCA detector is an ionization chamber type detector used to detect a rupture of the pressure boundary which will release large amounts of radioactive material into the primary containment. The leak detection channels are isolated when a LOCA occurs because they would rapidly be saturated by the high levels of radioactivity.

5. Primary Containment Hydrogen and Oxygen Concentration

Atmosphere samples from three locations inside the primary containment and one location in the suppression chamber are sequentially monitored for hydrogen and oxygen percentage levels by each of two redundant analyzers. For precise locations of each sample point see Table 7.5-2.

Each gas analyzer cabinet contains a hydrogen and an oxygen analyzer with sample conditioning and sample programming means. The programmer also admits standardizing gases periodically to calibrate the analyzers. Vent gases are pumped back to the primary containment at all times.

The analyzers are single range, i.e., 0-10% hydrogen and 0-25% oxygen. System accuracy is $\pm$2% of full scale. The output signal from each analyzer is sent to two redundant recorders in the main control room. Each analyzer has two adjustable alarm contacts which annunciate abnormal conditions in the main control room.

6. Suppression Chamber Pressure

Suppression chamber pressure is recorded in the main control room from two separate pressure transmitter systems. Range of recording is from 0-100 psig.

7. Suppression Pool Temperature Monitoring.

Suppression pool temperature is monitored by 26 redundant sensors. Two sensors monitor the area above the pool and 24 sensors monitor the suppression pool water. Temperature above the pool surface is recorded on a two-pen recorder with an alarm. Refer also to 7.6.1.7.

8. Suppression Pool Water Level Monitoring.

Suppression pool water level is monitored by two redundant sensors. Each sensor consists of one level transmitter which is recorded in the main control room.

7.5.1.6 Other Safety-Related and Auxiliary Support System Indications

The following information is provided to the control room operator to monitor the status of other safety-related and auxiliary supporting systems.

1. Main Control Room HVAC

   Redundant temperature indications are provided in the control room to monitor control room temperature.

2. Standby Gas Treatment System

   Each division of SGTS provides loop flow indication.

3. Containment Instrument Air

   Each division of CIA provides system line pressure indication.

4. Main Steam Line Leakage Control System

   Each division of MSLC provides system pressure indication. The inboard system also provides flow indication.

5. Standby Service Water System

   Each division of SSW provides pump discharge pressure and flow indications.

7.5.2 ANALYSIS

The safety-related display instrumentation provides adequate information to allow the reactor operator to perform the necessary manual safety functions during normal operation, transients, and accident conditions.

Normal Operation

The information channel ranges and indicators were selected on the basis of giving the reactor operator the necessary information to perform all the normal plant startup, steady state maneuvers, and to be able to track all the process variables pertinent to safety.

Abnormal Transient Occurrences

The ranges of indicators and recorders provided are capable of covering the extremes of process variables and provide adequate information for all abnormal transient events.

Accident Conditions

Information readouts are designed to accommodate all credible accidents for operator actions, information, and event tracking requirements, and cover all other design basis events or incident requirements.

Regulatory Guide 1.97, Rev. 2 is scheduled to be issued in August 1980. The Guide will provide direction to near-term OL plants (WNP-2) for implementation of specific requirements (quality, single failure, power source, etc.) as they apply to safety-related display instrumentation.

WNP-2 has responded to Regulatory Guide 1.97, draft 2 in letter G02-80-29, D. L. Renberger (WPPSS) to H. Denton (NRC) dated Feb. 1, 1980. The response included comments on the guide as well as a detailed description of the WNP-2 design.

A complete analysis will be provided for safety-related display instrumentation following issuance of Regulatory Guide 1.97, Rev. 2.

TABLE 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

Design Criteria	Type Readout	Number of Channels	Range	Accuracy	Location
Reactor Vessel Pressure	Recorder	2	0-1,500 psig	± 2% FS	CR
Reactor Vessel Water Level	Recorder	2	-150"/0/+60"	± 2% FS	CR
RCIC Flow	Meter	1	0-700 GPM	± 1% FS	CR
RCIC Discharge Pressure	Meter	1	0-1500 psig	± 2% FS	CR
HPCS Flow	Meter	1	0-10,000 GPM	± 2% FS	CR
HPCS Discharge Pressure	Meter	1	0-1,500 psig	± 2% FS	CR
LPCS Flow	Meter	1	0-10,000 GPM	± 2% FS	CR
RHR Flow (LPCI and Shut-down Cooling)	Meter	3	0-10,000 GPM	± 2% FS	CR
RHR Service Water Flow	Meter	2	0-10,000 GPM	± 2% FS	CR
Drywell Pressure	Recorder	2 2	0-2 psig LOW 2 0-100 psig HIGH	± 2% FS	CR
MSIV-LCS Outboard Steam Line Header Pressure	Meter	1	0-1200 psig	± 2% FS	CR
MSIV-LCS Steam Line Pressure Between MSIV's	Meter	4	0-1200 psig	± 2% FS	CR
MSIV/LCS Leakage Flow	Meter	4	0-0.5 CFM	+ 2% FS	CR

TABLE 7.5-1 (Continued)

Design Criteria	Type Readout	Number of Channels	Range	Accuracy	Location
SSW System Pump Discharge Line Pressure	Meter	2 1	0-300 psig 0-100 psig	± 1% FS ± 1% FS	CR CR
Suppression Pool Water Level	Recorder	2	-25"-0-+25"	± 1%	CR
Main Control Room Temperature	Meter	2	50-100°F	± 2%	CR
SGTS Flow Rate	Meter	1	0-6000 CFM	± 3%	CR
CAC System Flow Rate	Meter	1	0-300 CFM	± 3%	CR
Primary Containment Hydrogen	Recorder	2	0-10% H_2	± 1%	CR
Suppression Pool Water Temperature	Recorder	2	50-400°F	± 1%	CR

TABLE 7.5-2

CONTAINMENT HYDROGEN AND OXYGEN MONITORING SYSTEM SAMPLE POINT LOCATIONS

SP	PENETRATION #	SAMPLE POINT AZIMUTH	SAMPLE POINT ELEVATION
74	72c	188°-24'	560'-0"
75	72d	191°-36'	560'-0"
76	72e	193°	531'-0"
77	82c	230°	479'-4"
78	85d	18°-12'	545'-2-1/4"
79	85e	13°-44'	545'-1-1/2"
80	73d	45°	531'-0"
81	84b	40°	479°-4"

7.6 ALL OTHER INSTRUMENTATION SYSTEMS REQUIRED FOR SAFETY

7.6.1 DESCRIPTION

Section 7.6 describes the instrumentation and control systems required for safety not discussed in other sections. The systems include:

1. Process Radiation Monitoring System
2. High Pressure/Low Pressure Systems Interlocks
3. Leak Detection System (LDS)
4. Neutron Monitoring System (NMS)
5. Recirculation Pump Trip System (RPT)
6. Spent Fuel Pool Cooling and Cleanup System
7. Suppression Pool Temperature Monitoring System

The sources which supply power to the safety-related systems described in this section originate from on-site ac and/or dc safety-related busses or as in the case of the fail-safe logic NMS and portions of the LDS, from the non-safety-related RPS MG-sets. Refer to Chapter 8 for a complete description of the safety-related systems power sources.

7.6.1.1 Process Radiation Monitoring System - Instrumentation and Controls

The safety-related portions of the process radiation monitoring system are described in 7.2.1.1.b and 7.3.1.1.2.b.

7.6.1.2 High Pressure/Low Pressure Systems Interlocks

a. System Function

Instrumentation and controls are provided to prevent overpressurization of low pressure systems which interface with the reactor coolant pressure boundary.

b. System Operation

Schematic arrangement of mechanical equipment for the systems involved is shown in Figures 7.3-13 (RHR P&ID) and 7.3-11 (LPCS P&ID). Component control logic for the systems involved is shown in Figures 7.3-14 (RHR FCD) and 7.3-12 (LPCS FCD). Electrical schematics are identified in 1.7. Instrument specifications are listed in Tables 7.6-5 and 7.6-6.

The following high pressure/low pressure interlock equipment is provided:

Interlocked Process Line	Type	Valve	Parameter Sensed	Purpose
RHRS Shut-down Supply	MO MO	F009 F008	Reactor pressure	Prevents valve opening until reactor pressure is below system design pressure
RHRS Shut-down Return	MO	F053	Reactor Pressure	Prevents valve opening until reactor pressure is below system design pressure
RHRS Head Spray	MO	FO23	Reactor Pressure	Prevents valve opening until reactor pressure is below system design pressure
LPCI	MO	FO42	Differential pressure across valve	Does not allow valve to open until differential pressure is low
LPCS	MO	F005	Differential pressure across valve	Does not allow valve to open until differential pressure is low

The shutdown cooling suction valves, head spray valve, and discharge valve have redundant and diverse interlocks to prevent the valves from being opened when the primary system pressure is above the subsystem design pressure. These valves also receive a signal to close when reactor pressure is above system pressure.

The LPCI and LPCS discharge valves MO F042 and MO F005 are prevented from opening until differential pressure across the valves is low enough to prevent system overpressurization.

7.6.1.3 Leak Detection System - Instrumentation and Controls

The safety-related portions of the Leak Detection System are as follows:

1. Main Steam Line Leak Detection

2. RCIC System Leak Detection

3. RHR System Leak Detection

4. Reactor Water Cleanup System Leak Detection

5. Drywell/Reactor Bldg. Leak Detection

a. Leak Detection System Function

The leak detection system instrumentation and controls is designed to monitor leakage from the reactor coolant pressure boundary and initiate alarms and/or isolation when predetermined limits are exceeded. Refer to 5.2.5.

b. Leak Detection System Operation

Schematic Arrangements of system mechanical equipment and operator information displays are shown in Figure 7.6-1 (LDS IED). LDS component control logic is shown in Figures 7.3-14 (RHR FCD), 7.4-2 (RCIC FCD), 7.3-10 (NUC. BLR. FCD). Instrument specifications are listed in Tables 7.6-7 and 7.6-8. Plant Layout Drawings and Electrical Schematics are identified in 1.7.

Systems or parts of systems which contain water or steam and which are in direct communication with the reactor vessel, are provided with leakage detection systems.

Each of the required leakage detection systems inside the primary containment is designed with a capability to detect leakage less than established leakage rate limits. Refer to Chapter 16, "Technical Specifications".

Major components within the primary containment that by nature of their design are sources of leakage (e.g., pump seals, valve stem packing, equipment warming drains), are collected ultimately in an equipment drain sump located in the reactor building and thereby identified.

Equipment associated with systems within the primary containment (e.g., vessels, piping, fittings) share a common volume. Steam or water leaks from such equipment are collected ultimately in the floor drain sumps located in the reactor building and identified.

Each of the sumps is protected against overflowing to prevent leaks of an identified source from masking those from unidentified sources.

Outside the primary containment, the piping within each system monitored for leakage is in compartments or rooms separate from other systems wherever feasible so that leakage may be detected by sump level, ambient or differential area temperature or high process flow.

Sensors, wiring, and associated equipment of the leak detection system which are associated with the isolation valve logic are designed to withstand the conditions that follow a design basis loss-of-coolant accident. (See Tables 3.11-1, 3.11-2, and 3.11-3.)

The operator is kept aware of the status of the leak detection system variables through meters and recorders which indicate the measured variables in the control room. If a trip occurs, the condition is annunicated in the control room.

. RCIC System Leak Detection

The steam lines of the RCIC system are monitored for leaks by the leak detection system. Leaks from the RCIC will cause a change in at least one of the following monitored parameters: sensed equipment and pipe routing area temperatures, steam flow rate, or steam pressure. If the monitored variables indicate that a leak may exist, the detection system initiates an RCIC isolation signal.

The RCIC leak detection system consists of the following:

a) equipment area and pipe routing area high ambient and differential temperature,

b) high flow rate (differential pressure) through the steam line,

c) the turbine exhaust diaphragm high pressure,

d) low steam line inlet pressure.

Outputs from all four monitoring circuits are used to generate the RCIC auto-isolation signals (one for each division) to isolate the inboard and outboard isolation valves.

The following is a description of each RCIC leak detection method:

a. RCIC Area Temperature Monitoring

The RCIC area ambient and differential temperature monitoring circuits are similar to those described for the main steam line tunnel temperature monitoring system. (See 7.3.1.1.2.b).

Two redundant temperature monitoring channels are provided. Each redundant instrument provides input to one of two logic channels (Division 1 or 2).

Using 1 out of 2 logic, any RCIC equipment area or pipe routing high area ambient or high differential temperature initiates an isolation of the RCIC system.

A bypass/test switch is provided in each logic channel for the purpose of testing the temperature monitor without initiating RCIC system isolation.

Diversity is provided by RCIC steam line flow and pressure monitoring.

b. RCIC Flow Rate Monitoring

The steam line flow rate from the reactor vessel leading to the RCIC turbine is monitored by two redundant differential pressure switches. In the presence of a leak, the flow rate monitor responds by generating the auto-isolation signal. See 7.4.1.1.b.

High flow in the steam line initiates isolation of the RCIC system.

Diversity is provided by ambient temperature, differential temperature and RCIC steam line pressure monitoring.

c. RCIC Turbine Exhaust Diaphragm Pressure Monitoring

The RCIC turbine exhaust diaphragm pressure is monitored by four redundant pressure switches. In the presence of a leak, the RCIC system responds by generating the isolation signal. See 7.4.1.1.b.

Using 2 out of 2 logic high turbine exhaust diaphragm pressure initiates isolation of the RCIC system.

Diversity is provided by ambient temperature and differential temperature.

d. RCIC Pressure Monitoring

The steam line pressure from the reactor vessel leading to the RCIC turbine is monitored by four redundant pressure switches. In the presence of a leak, the RCIC system responds by generating the auto-isolation signal. See 7.4.1.1.b.

Using 2 out of 2 logic low pressure in the steam line initiates isolation of the RCIC system.

Diversity is provided by ambient temperature, differential temperature and RCIC steam line flow monitoring.

2. RHR System Leak Detection

The steam line to the RHR heat exchangers is monitored for leaks by the leak detection system. Leaks from the RHR system are detected by equipment area ambient and differential temperature monitoring, shutdown cooling suction flow rate, and by steam line flow rate in the common RHR/RCIC steam line. If the monitored parameters indicate that a leak exists, the LDS initiates an RHR isolation signal.

The RHR leak detection system consists of the following:

a) equipment area high ambient and differential temperature,

b) high flow rate through the common RCIC/RHR steam line, (refer to discussion in 7.6.1.3.B.1.b)

c) shutdown cooling suction line high flow rate.

Outputs from all three circuits are used to generate the RHR auto-isolation signal (one for each division) to isolate the inboard and outboard isolation valves.

The following is a description of each RHR leak detection method:

a. RHR Area Temperature Monitoring

The RHR area temperature monitoring circuit is similar to the one described for the main steam line tunnel temperature monitoring system (See 7.3.1.1.2.b).

Two redundant temperature monitoring channels are provided. Each redundant instrument provides input to one of two logic channels (Division 1 or 2).

Using 1 out of 2 logic, high RHR area ambient or differential temperature initiates an RHR isolation signal closing the RHR inboard and outboard isolation valves.

A bypass/test switch is provided in each logic channel for the purpose of testing the temperature monitor without initiating RHR system isolation.

Diversity is provided by RHR/RCIC steam line and RHR shutdown cooling suction line high flow rate monitoring.

b. RHR Flow Rate Monitoring

Flow rate monitoring is provided on the RHR shutdown cooling suction line and on the RHR steam line to the RHR condensing heat exchanger by redundant differential pressure switches.

Flow rates in excess of predetermined limits indicate a line leak or break.

Two redundant differential pressure switches monitor flow and each provides an input to one of the two logic channels (Division 1 or 2).

Using 1 out of 2 logic, the high flow rate initiates an isolation of the RHR inboard and outboard isolation valves.

Diversity is provided by ambient differential temperature monitoring.

3. Reactor Water Clean-Up System Leak Detection

The RWCU leak detection system monitors high differential flow. Automatic isolation of the RWCU system isolation valves is initiated when high differential flow exists.

The RWCU leak detection system consists of the following:

a) Leakage monitoring by the flow comparison of RWCU system water inlet and outlet flow rate, and

b) Ambient and differential temperature monitoring.

The following is a description of each RWCU leak detection method:

a. RWCU Differential Flow Monitoring

RWCU system inlet flow is compared to RWCU outlet flow to the feedwater lines or to the main condenser. A flow element, flow transmitter and square root converter for each of these three lines provides signals to a common flow summer which trips two differential flow alarm units on a high differential flow condition. The high differential flow rate initiates a 45 sec time delay which bypa_ s the alarm and isolation signal during the normal operational system surges, i.e., pump startup or valving changes. If the high differential flow conditions still exists after the time delay, then alarm and isolation are initiated. Flow and differential flow indications are provided in the main control room.

Using 1 out of 2 logic in each logic channel (Division 1 or 2), the RWCU flow comparison monitoring initiates RWCU isolation signal. The signal closes the inboard and outboard isolation valves, after a time delay, when the flow rate difference exceeds a preset limit.

Diversity is provided by high differential flow and ambient and differential temperature.

b. RWCU Area Temperature Monitoring

Refer to 7.3.1.1.2.b.

4. Drywell Floor Drain Leak Detection Sump

The Drywell Floor Drain Sump collects unidentified leakage within the drywell. The leakage is gravity fed to the HPCS pump room sump. The flow is monitored and leakage in excess of the acceptable limit is annunciated in the control room.

5. Drywell Equipment Drain Leak Detection

The drywell equipment drain sump collects identified leakage within the drywell. The leakage is gravity fed to the reactor building equipment drain sump.

6. Reactor Building Floor Drain and Equipment Drain Sumps Leak Detection

The floor drain sumps leak detection instrumentation is designed to detect leakage from unidentified sources. The equipment drain sump leak detection instrumentation is designed to detect leakage from identified sources.

A level switch is mounted in each sump and controls the associated sump pump. The sump pump starts when the upper sump level is exceeded and turns off when the level in the sump reaches the lower setpoint. A timer is started and stopped by the upper and lower sump level switches. If the timer exceeds a predetermined setpoint, high flow is annunciated. The timer resets to read when the sump pump stops. A second timer starts timing when the sump pump stops. If the sump pump starts before the second timer reaches a predetermined setpoint, high flow rate is annunciated. The timer is reset to read when the sump pump starts.

In addition, wall-mounted level switches are provided in each sump room to detect ECCS passive failures and provide annunciation in the main control room.

7.6.1.4 Neutron Monitoring System (NMS) - Instrumentation and Controls

The safety-related portions of the neutron monitoring system are as follows:

1. Intermediate Range Monitor (IRM)

2. Local Power Range Monitor (LPRM)

3. Average Power Range Monitor (APRM)

a. Neutron Monitoring System Function

The neutron monitoring system instrumentation and controls are designed to monitor reactor power (neutron flux) from startup through full power operation.

b. Neutron Monitoring System Operation

The neutron monitoring system uses incore detectors, either fixed (LPRM) or removable (IRM), to determine neutron flux levels.

NMS will initiate a scram when predetermined limits are exceeded and provide operator information during and after accident conditions.

The NMS component control logic is shown in Figure 7.6-6.

7.6.1.4.1 Intermediate Range Monitor (IRM)

a. IRM Function

The IRM monitors neutron flux from the upper portion of the SRM range to the lower portion of the power range (APRM) as shown in Figure 7.6-22.

b. IRM Operation

The IRM has eight channels, each of which includes one detector that can be positioned in the core by remote control. Refer to Figures 7.6-5 and 7.6-2. The detectors are inserted into the core for a reactor startup and are withdrawn after the reactor mode selector switch is placed in the RUN position.

Each detector assembly consists of a fission chamber attached to a low-loss, quartz-fiber-insulated transmission cable. When coupled to the signal conditioning equipment, the detector produces a reading of full scale on the most sensitive range. The detector cable is connected underneath the reactor vessel to a triple-shielded cable that is connected to the preamplifier.

The preamplifier converts current pulses to voltage pulses, modifies the voltage signal, and provides impedance matching. The preamplifier output signal is then sent to the IRM signal conditioning electronics (see Figure 7.6-8).

Each IRM channel input signal from the preamplifier can be amplified and attenuated. IRM preamplification is selected by a remote range switch that provides 10 ranges of increasing attenuation (The first 6 called low range and the last 4 called high range). As the neutron flux of the reactor core increases the signal from the fission chamber is attenuated to keep the input signal to the inverter in the same range. The output signal, which is proportional to neutron flux at the detector, is amplified and supplied to a locally mounted meter, a remote meter and recorder.

The IRM Scram Trip Functions are discussed in 7.2.1.1.B. The IRM trips are shown in Table 7.6-1.

The IRM range switches must be up-ranged or down-ranged to follow increases and decreases in power within the range of the IRM to prevent either a scram or a rod block. The IRM detectors must be inserted into the core whenever these channels are needed, and withdrawn from the core, when permitted, to prevent unnecessary burnup.

7.6.1.4.2 Local Power Range Monitor (LPRM)

a. LPRM Function

The LPRM's provide localized neutron flux detection over the full power range for input to the APRM.

b. LPRM Operation

The LPRM includes 43 detector strings having detectors located at different axial heights in the core; each detector string contains four fission chambers. Figure 7.6-3 shows the LPRM detector radial layout scheme.

The LRPM assembly consists of four neutron detectors permanently installed in a housing (see Fig. 7.6-9).

Each LPRM detector assembly contains four ion chambers with an associated solid sheath cable. The chambers are vertically spaced in the LPRM detector assemblies in a way that gives adequate axial coverage of the core, complementing the radial coverage given by the horizontal arrangement of the LPRM detector assemblies.

Each chamber consists of two concentric cyclinders, which act as electrodes. The inner cylinder (the collector) is mounted on insulators and is separated from the outer cylinder by a small gap. The gas between the electrodes is ionized by the charged particles produced as a result of neutron fissioning of the uranium-coated outer electrode. The chamber is operated at a polarizing potential of approximately 100 Vdc. The negative ions produced in the gas are accelerated to the collector by the potential difference maintained between the electrodes. In a given neutron flux, all the ions produced in the ion chamber can be collected if the polarizing voltage is high enough. When this situation exists, the ion chamber is considered to be saturated. Output current is then independent of operating voltage, (Reference A).

Each assembly also contains a calibration tube for a traversing in-core probe. The enclosing tube around the entire assembly contains holes that allow circulation of the reactor coolant water to cool the ion chambers. Numerous tests have been performed on the chamber assemblies including tests of linearity, lifetime, gamma sensitivity, and cable effects, (Reference A).

The current signals from the LPRM detectors are transmitted to the LPRM amplifiers in the control room. The current signal from a chamber is transmitted directly to its amplifier through coaxial cable. The amplifier is a linear current amplifier whose voltage output is proportional to the current input and therefore proportional to the magnitude of the neutron flux. Low level output signals are provided that are suitable as an input to the computer, recorders, etc. The output of each LPRM amplifier is isolated to prevent interference of the signal by inadvertent grounding or application of stray voltage at the signal terminal point.

When a central control rod is selected for movement, the output signals from the amplifiers associated with the nearest 16 LPRM detectors are displayed on reactor control panel meters. The four LPRM detector signals from each of the four LPRM assemblies are displayed on 16 separate meters. The operator can readily obtain readings of all the LPRM amplifiers by selecting the control rods in order.

The trip circuits for the LPRM provide trip signals to activate lights, instrument inoperative signals, and annunciators. These trip circuits use the 24Vdc power supply and are set to trip on loss of power. They also trip when power is not available for the LPRM amplifiers. Table 7.6-2 indicates the trips.

The trip levels can be adjusted to within ±5% of full-scale deflection and are accurate to ±1% of full-scale deflection in the normal operating environment.

Each LPRM channel may be individually bypassed. When the maximum number of bypassed LPRMs associated with any APRM channel has been exceeded, an inoperative trip is generated by that APRM.

Each individual chamber of the assembly is a moisture-proof, pressure-sealed unit. The chambers are designed to operate up to 600°F and 1250 psig. The wiring, cables, and connectors located within the drywell are designed for continuous duty up to 270°F, 100% relative humidity and a 4-hour single exposure rating of 482°F at 100% relative humidity.

Power for the LPRM is supplied by the two RPS buses. Approximately half of the LPRMs are supplied from each bus. Each LPRM amplifier has a separate power supply (ICPS) in the control room, which furnishes the detector polarizing potential. This power supply is adjustable from 75 to 200 Vdc. The maximum current output is three milliamps. This ensures that the chambers can be operated in the saturated region at the maximum specified neutron fluxes. For maximum variation in the input voltage or line frequency, and over extended ranges of temperature and humidity, the output voltage varies no more than two volts. Each page of amplifiers is supplied operating voltages from a separate low voltage power supply.

7.6.1.4.3 Average Power Range Monitor (APRM)

a. APRM Function

The function of the APRM is to average signals from the LPRM's and provide a flow reference reactor scram when neutron flux exceeds predetermined limits.

b. APRM Operation

The APRM has six redundant channels. Each channel uses input signals from a number of LPRM channels. Three APRM channels are associated with each trip system of the RPS.

The APRM channel uses electronic equipment that averages the output signals from a selected set of LPRMs, trip units that actuate automatic devices, and signal readout equipment. Each APRM channel can average the output signals from as many as 24 LPRMs. Assignment of LPRMs to an APRM follows the pattern shown in Figure 7.6-4. Position A is the bottom position, Positions B and C are above Position A, and Position D is the topmost LPRM detector position. The pattern provides LPRM signals from all four core axial LPRM detector positions.

The APRM amplifier gain can be adjusted by combining fixed resistors and potentiometers to allow calibration. The averaging circuit automatically corrects for the number of unbypassed LPRM amplifiers providing inputs to the APRM.

Refer to 7.2.1.1.b for a description of the APRM inputs to the RPS.

APRM system trips are summarized in Table 7.6-3. The APRM circuit arrangment for RPS trip input is shown in Figure 7.6-10.

One of the two recirculation flow signals in each trip system may be bypassed at any time. One of the three APRMs in each trip system may be bypassed at any time. An interlock circuit provides an inoperative trip output from an APRM whenever the minimum number of LPRM inputs to it is not met.

The APRM channels receive power from the 120 Vac RPS MG sets. Power for each APRM trip unit is supplied from the same power supply as the APRM it services. APRM Channels A, C, and E are powered from the bus used for Trip System A of the RPS; APRM Channels B, D, and F are powered from the bus used for RPS Trip System B. The ac bus used for a given APRM channel also supplies power to its associated LPRMs.

7.6.1.5 Recirculation Pump Trip (RPT) System - Instrumentation and Controls

See Appendix H and 5.4 for a description of the RPT system.

7.6.1.6 Spent Fuel Pool Cooling and Cleanup System (FPC) - Instrumentation and Control

a. FPC System Function

The function of the FPC system is to remove decay heat from the spent fuel storage pool to insure adequate cooling of irradiated stored fuel assemblies. The FPC system also purifies the storage pool water, maintains water clairty for fuel handling operations, and fills and drains the fuel transfer canal. Refer to 9.1.3.

b. FPC System Operation

Schematic arrangement of the FPC system mechanical equipment is shown in Figure 3.2-12 (FPC P&ID). FPC system component control logic is shown in Figure 7.6-11 (FPC Logic Diag.). Instrument Specifications are listed in Tables 7.6-11 and 7.6-12. Plant layout drawings and electrical schematics are shown in 1.7. Operator information displays are shown in Figure 3.2-12 (FPC P&ID) and Figure 7.6-11 (FPC Logic Diag.)

The FPC System consists of two redundant cooling loops. The system is manually initiated and one loop runs continuously when the pool contains spent fuel.

Instrumentation is provided to monitor the pool temperature, pump suction and discharge pressures, and water conductivity to allow the control room operator to assess system operation.

7.6.1.7 Suppression Pool Temperature Monitoring System - Instrumentation and Controls

a. System Function

The suppression pool temperature monitoring (SPTM) system is designed to monitor suppression pool water temperature and alert the plant operator to the potentially hazardous condition of elevated pool water temperature.

The instrumentation for the SPTM system is shown in Figure 3.2-8.

b. System Operation

The suppression pool temperature monitoring system consists of 24 dual element thermocouples. Sixteen thermocouples are arranged near the surface of the pool whereas the remaining 8 are located midlevel in the pool. This arrangement was chosen to track pool stratification. The sensors are separated into two redundant divisions and maintained throughout the system.

The time constant for the thermocouples is no greater than 15 seconds. The time from signal output of sensor to initiation of alarm is no greater than 0.5 seconds. The difference between measurement reading and actual temperature is within $\pm 2°F$.

Each division of the Suppression Pool Temperature Monitoring System is provided with temperature readout devices in the main control room consisting of four indicators and a four channel recorder. Each division is provided with an audible and visual annunciator in the control rool.

7.6.1.8 Design Basis

The safety-related systems described in 7.6 are designed to provide timely protective action inputs to other safety systems to protect against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, Appendix 15A, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are also presented in Chapter 15.

The station conditions which require protective actions are described in Chapter 15 and Appendix 15A.

a. Variables Monitored to Provide Protective Actions

The following variables are monitored in order to provide protective action inputs:

1. High Pressure/Low Pressure System Interlocks
    a) Reactor pressure
    b) Differential pressure across the LPCS and LPCI injection valves
2. Leak Detection System
    a) RCIC area temperatures - differential and ambient
    b) RCIC steam line flow rate
    c) RCIC turbine exhaust diaphragm pressure

d) RCIC steam line pressure

e) RHR area temperatures - differential and ambient

f) RHR shutdown cooling suction flow

g) RWCU area temperatures - differential and ambient

h) RWCU differential flow

i) Identified and Unidentified leakage from the drywell floor and equipment drain sumps.

3. Neutron Monitoring System

a) IRM neutron flux

b) APRM neutron flux

4. Spent Fuel Pool Cooling and Cleanup System

a) Fuel Pool Temperature

5. Suppression Pool and Chamber Temperature Monitoring System

a) Suppression Pool and Chamber Temperature

The plant conditions which require protective action involving the safety-related systems discussed in 7.6 are described in Chapter 15 and Appendix 15A.

b. Location and Minimum Number of Sensors

See Chapter 16 for the minimum number of sensors required to monitor safety-related variables. The IRM and LPRM detectors are the only sensors which have spatial dependence.

c. Prudent Operational Limits

Prudent operational limits for each safety-related variable trip setting are selected to be far enough above or below normal operating levels so that a spurious safety system initiation is avoided. It is then verified by analysis that the release of radioactive materials, following postulated gross failures of the fuel or nuclear system process barrier, is kept within acceptable bounds.

d. Margin

The margin between operational limits and the limiting conditions of operation of the safety-related systems are those parameters as listed in Chapter 16, "Technical Specifications".

e. Levels

Levels requiring protective action are established in Chapter 16, "Technical Specifications".

f. Range of Transient, Steady State, and Environmental Conditions

Refer to Tables 3.11-1 through 3.11-5 and 3.1.2.1.4.1 for environmental conditions. Refer to 8.2.1 and 8.3.1 for the maximum and minimum range of energy supply to the safety-related instrumentation and controls of the systems described in 7.6. All safety-related instrumentation and controls are specified and purchased to withstand the effects of energy supply ranges.

Environmental conditions for proper operation of the systems described in 7.6 are discussed in 3.10 and 3.11.

g. Malfunctions, Accidents, and Other Unusual Events Which Could Cause Damage to Safety Systems

Chapter 15 and Appendix 15A, "Accident Analysis" describes the following credible accidents and events; floods, storms, tornados, earthquakes, fires, LOCA, pipe break outside containment, and missiles.

1. Floods

The buildings containing safety-related components have been designed to meet the PMF (Probable Maximum Flood) at the site location. This ensures that the buildings will remain water tight under PMF including wind generated wave action and wave runup. Therefore, none of the functions are affected by flooding. For a discussion of internal flooding protection refer to 3.4.1.4.1.2, 3.4.1.5.2, and 3.6.

2. Storms and Tornados

The buildings containing safety-related components have been designed to withstand all credible meteorological events and tornados as described in subsection 3.3.

3. Earthquakes

The structures containing safety-related system components have been seismically qualified as described in 3.7 and 3.8, and will remain functional during and following a safe shutdown earthquake (SSE).

4. Fires

To protect the safety systems in the event of a postulated fire, the components have been separated by distance or fire barriers. The use of separation and fire barriers ensures that, even though some portion of the system may be affected, the safety function will not be prevented.

Within the control room PGCC (underfloor cable routing ducts) heat detectors and products of combustion detectors are provided to initiate a halon fire suppression system.

5. LOCA

The safety-related systems components described in 7.6 located inside the drywell and functionally required during and/or following a LOCA have been environmentally qualified to remain functional as discussed in 3.11 and as indicated in Table 3.11-1.

6. Pipe Break Outside Containment

Protection for these components is described in 3.6.

7. Missiles

Protection for safety-related components is described in 3.5.

h. Minimum Performance Requirements

Minimum performance requirements for safety-related systems instrumentation and controls are provided in Chapter 16, "Technical Specifications".

7.6.1.9 Final System Drawings

The final system drawings including:

1. Process and Instrumentation Diagrams (P&ID)/Flow Diagrams

2. Functional Control Diagrams (FCD)/Control Logic Diagrams

3. Instrument and Electrical Drawing (IED)

have been provided for the safety-related systems in this section.

Electrical interconnection and schematic diagrams are in 1.7.

Functional and architectural design difference between the PSAR and FSAR are listed in Table 1.3-8.

7.6.2 ANALYSIS

7.6.2.1 Safety-Related Systems - Instrumentation and Controls

Chapter 15, "Accident Analysis," evaluates the individual and combined capabilities of the safety-related systems described in 7.6.

The safety-related systems described in 7.6 are designed such that a loss of instrument air, a plant load rejection, or a turbine trip will not prevent the completion of the safety function.

7.6.2.2 Conformance to 10 CFR 50, Appendix A - General Design Criteria (GDC)

The following is a discussion of conformance to those General Design Criteria which apply specifically to the safety-related systems described in 7.6. Refer to 7.1.2.2 for a discussion of General Design Criteria which apply equally to all safety-related systems.

GDC's for the NMS and process radiation monitoring system are discussed in 7.2.2.1 and 7.3.2.1.1.

a. Criterion 12

The NMS provides protective actions to the RPS to assure that fuel design limits are not exceeded.

b. General Design Criterion 13 - Instrumentation and Control

The safety-related instrumentation and controls monitor variables over their anticipated ranges for normal operation, anticipated occurrences and accident conditions and initiate protective actions to limit or prevent fuel damage and maintain the integrity of the reactor coolant pressure boundary and the primary containment.

c. General Design Criterion 15 - Reactor Coolant System Design

The safety-related systems provide sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. If the monitored variables exceed their predetermined settings, automatic safety actions are provided.

d. Criterion 30, 34, 35, 38, 40

The leak detection system provides means for detecting the source of reactor coolant leakage.

e. Criterion 61, 62 and 63

The spent fuel pool cooling and cleanup system provides reliable fuel pool residual heat removal capability.

7.6.2.3 Conformance to IEEE Standards

The following is a discussion of conformance to those IEEE standards which apply specifically to the safety-related systems described in 7.6. Refer to 7.1.2.3 for a discussion of IEEE standards which apply equally to all safety-related systems.

a. IEEE 279-(1971) - Criteria for Protection Systems for Nuclear Power Generating Stations

1. General Functional Requirement (IEEE 279-1971, Paragraph 4.1)

The safety-related systems described in 7.6 automatically initiate protective actions when a condition monitored reaches a preset level for all conditions described in the design bases 7.6.1. For example, the leak detection system initiates containment isolation by closure of containment isolation valves when area temperatures exceed preset limits.

2. Single Failure Criterion (IEEE 279-1971, Paragraph 4.2)

The safety-related systems described in 7.6 are not required to meet single failure criteria on an individual system basis. However, on a network basis, the single failure criteria does apply to assure the completion of a protective function. Redundant sensors, wiring, logic, and actuated devices are physically and electrically separated such that a single failure will not prevent the protective function. Refer to 8.3.1.4 for a complete description of the WNP-2 separation criteria.

3. Quality of Components and Modules (IEEE 279-1971, Paragraph 4.3)

Refer to 3.11 for a discussion of safety system component quality.

4. Equipment Qualification (IEEE 279-1971, Paragraph 4.4)

Vendor certification requires that the sensors associated with each of the systems required for safety trip variables, manual switches, and trip logic components perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in conjunction with the existing field experience with these components in this application, will serve to qualify these components.

Qualification tests of the relay panels are conducted to confirm their adequacy for this service. In-situ operational testing of these sensors, channels, and the entire protection system will be performed during the preoperational test phase.

For a complete discussion of equipment qualification for the safety-related systems described in 7.6 refer to 3.5, 3.6, 3.10, and 3.11.

5. Channel Integrity (IEEE 279-1971, Paragraph 4.5)

For a discussion of channel integrity for the safety-related systems described in 7.6 under all extremes of conditions described in 7.6.1.9 refer to 3.10, 3.11, 8.2.1, and 8.3.1.

6. Channel Independence (IEEE 279-1971, Paragraph 4.6)

System channel independence is maintained by application of the WNP-2 separation criteria as described in 8.3.1.4.

7. Control and Protection System Interaction (IEEE 279-1971, Paragraph 4.7)

There are no control and protection system interactions for the systems described in 7.6.

8. Derivation of System Inputs (IEEE 279-1971, Paragraph 4.8)

The variables discussed in 7.6 are direct measures of the desired variables indicating the need for protective action.

9. Capability for Sensor Checks (IEEE 279-1971, Paragraph 4.9)

For a discussion of sensor checks for the safety-related systems described in 7.6, refer to Regulatory Guide 1.22 in 7.6.2.4.

10. Capability for Test and Calibration (IEEE 2791971, Paragraph 4.10)

For a discussion of the test and calibration capability of the safety-related systems described in 7.6, refer to Regulatory Guide 1.22 in 7.6.2.4.

11. Channel Bypass or Removal from Operation (IEEE 279-1971, Para 4.11)

During periodic testing, any one sensor of the safety-related systems described in 7.6 may be valved-out-of-service and returned-to-service under administrative control procedures. Since only one sensor is valved-out-of-service at any given time during the test interval, protective capability for the safety-related variables is maintained through the remaining redundant instrument channels.

A sufficient number of IRM channels has been provided to permit any one IRM channel in a given trip system to be manually bypassed and still ensure that the remaining operable IRM channels comply with the single failure criterion.

One IRM manual bypass switch has been provided for each RPS trip system. The mechanical characteristics of this switch permit only one of the four IRM channels of that trip system to be bypassed at any time. In order to accommodate a single failure of this bypass switch, electrical interlocks have also been incorporated into the bypass logic to prevent bypassing of more than one IRM in that trip system at any time. Consequently, with any IRM bypassed in a given trip system, three IRM channels remain in operation to satisfy the protection system requirements.

In a similar manner, one APRM manual bypass switch has been provided for each RPS trip system to permit one of the three APRMs to be bypassed at any time. Mechanical interlocks have been provided with the bypass switch and elecrical interlocks have been provided in the bypass circuitry to accommodate the possibility of switch failure. With the maximum number of APRMs bypassed by the switches, sufficient APRM channels remain in operation to provide the necessary protection for the reactor.

The leak detection system logic is provided with a bypass/test switch for the purpose of testing temperature sensors without initiating associated system isolation. Operation of one switch at a time will not prevent the remaining redundant isolation logic from providing system isolation if required.

12. Operating Bypasses (IEEE 279-1971, Paragraph 4.12)

There are no operating bypasses for any of the safety-related systems described in 7.6.

13. Indication of Bypasses (IEEE 279-1971, Paragraph 4.13)

For a discussion of automatic bypass indication for the safety-related systems described in 7.6 refer to 7.1.2.4, Regulatory Guide 1.47.

14. Access to Means for Bypassing (IEEE 279-1971, Paragraph 4.14)

Access to bypassing any safety action or function is under administrative control of the control room operator. The operator is alerted to bypasses as described in 7.1.2.4, Regulatory Guide 1.47.

Control switches which allow safety system bypasses are keylocked. All keylock emergency switches in the control room are designed such that their key can only be removed when the switch is in the "accident" or "safe" position. All keys will normally be removed from their respective switches during operation and maintained under the control of the shift supervisor. Further, the key locker will be audited once per day by the shift supervisor. Should a key be required to change a valve position, it will be obtained from the shift supervisor via approved key control procedures.

15. Multiple Set Points (IEEE 279-1971, Paragraph 4.15)

There are no multiple set points within the safety-related systems described in 7.6.

16. Completion of Protective Action Once it is Initiated (IEEE 279-1971, Paragraph 4.16)

Each control logic for the safety-related systems described in 7.6 seals-in electrically and remain energized or de-energized. After initial conditions return to normal deliberate operator action is required to return (reset) the safety system logic to normal.

The spent fuel pool cooling and cleanup system is initiated manually for continuous pool cooling when the pool contains spent fuel.

17. Manual Initiation (IEEE 279-1971, Paragraph 4.17)

For a discussion of the manual initiation capability for the safety-related systems described in 7.6, refer to Regulatory Guide 1.62 in 7.6.2.4.

18. Access to Set Point Adjustments, Calibration, and Test Points (IEEE 279-1971, Paragraph 4.18)

During reactor operation access to setpoint adjustments, calibration controls, and test points for the safety-related systems variables described in 7.6 is under administrative control of the control room operator.

19. Identification of Protective Actions (IEEE 2791971, Paragraph 4.19)

When any sensor of the safety-related systems described in 7.6 exceeds its predetermined setpoint a control room annunciator is initiated to identify that variable and a typed record is available from the process computer.

20. Information Readout (IEEE 279-1971, Paragraph 4.20)

The safety-related systems described in 7.6 are designed to provide the operator with accurate and timely information pertinent to their status. This information does not give anomalous indications confusing to the operator.

21. System Repair (IEEE 279-1971, paragraph 4.21)

During periodic testing of the safety-related systems described in 7.6 (except as noted) the operator can determine any defective component and replace it during plant operation.

Replacement of IRM and LPRM detectors must be accomplished during plant shutdown. Repair of the remaining portions of the neutron monitoring system may be accomplished during plant operation by appropriate bypassing of the defective instrument channel. The design of the system facilitates rapid diagnosis and repair.

22. Identification of Protection Systems (IEEE 2791971, Paragraph 4.22)

Each cabinet containing safety system components is labeled with the system designation and the particular redundant portion is listed on a distictively colored marker plate. Cabling outside the cabinets is identified specifically as belonging to a particular safety system. See 8.3.1.3. Redundant racks are identified by the identification marker plates.

7.6.2.4 Conformance to NRC Regulatory Guides

The following is a disscussion of conformance to those Regulatory Guides which apply specifically to the safety-related systems discussed in 7.6. Refer to 7.1.2.3 for a discussion of Regulatory Guides which apply equally to all safety-related systems.

a. Regulatory Guide 1.22 (2/72) - Periodic Testing of Protection System Actuation Functions

The APRMS are calibrated to reactor power by using reactor heat balance and the (TIP) system to establish the relative local flux profile. LPRM gain settings are determined from the local flux profiles measured by the TIP system once the total reactor heat balance has been determined.

The gain-adjustment-factors for the LPRMs are produced as a result of the process computer nuclear calculations involving the reactor heat balance and the TIP flux distributions. These adjustments, when incorporated into the LPRMs permit the nuclear calculations to be completed for the next operating interval and establish the APRM calibration relative to reactor power.

The IRMs are calibrated by comparison with the APRMs.

The proper operation of the sensors and the logics associated with the leak detection systems is verified during the leak detection system preoperational test and during inspection tests that are provided for the various components during plant operation. Each temperature switch, both ambient and differential types which provide isolation signals, is connected to one element of a dual thermocouple element.

Each temperature switch contains a trip light which illuminates when the temperature exceeds the set point. To verify the thermocouple (sensor) input, a comparison of the redundant sensor readings, one from each trip channel, and the recorded channel is made. The recorded channel monitors the second of the dual thermocouples. The first element is part of the division one trip channel. To test the temperature trips a simulated trip level signal is input to the device from an external source. In addition, keylock test switches are provided so that instrument and logic channels can be tested without sending an isolation signal to the system involved. Thus, a complete system check can be confirmed by checking actuation of the trip logic relay associated with each temperature switch.

RWCU differential flow leak detection alarm units are tested by inputting an electrical signal to simulate a high differential flow. Alarm and indicator lights monitor the status of the trip circuit.

All other system instrumentation is tested and calibrated during normal reactor operation by valving out the instrumentation and supplying a test pressure source.

b. Regulatory Guide 1.45 (5/73)

The leakage to the primary reactor containment from identified sources such as valve stem packing, recirculation pump seal, fuel storage pool, head seal, etc. is separated so that flow rates are monitored separately from unidentified leakage and total flow rate can be established and monitored. The leakage from the main steam line safety/relief valves is identified leakage because of the location of the sensors which detect this leakage, but the leakage is not completely separated from unidentified sources. Separation of this leakage is not required since any leak from the main steam line safety/relief valves would not be from a crack or break in the line so there would be no identified leakage from the S/R valve lines during plant operation which necessitates separation from unidentified leakage. The leakage to the reactor containment from unidentified sources is collected and this flow rate is monitored with an accuracy of better than one gallon per minute.

The following required detection methods are used to monitor unidentified leakage:

1. Sump level and flow monitoring;

2. Airborne particulate radioactivity monitoring;

3. Airborne gaseous radioactivity monitoring.

Provisions are made to monitor systems connected to the RCPB for signs of intersystem leakage, including radioactivity monitoring of process fluids (Process Rad System) and reactor vessel water level monitoring (NSSS).

The sensitivity and response time of each system for detection of unidentified leakage is one gallon per minute in less than one hour, except for the airborne particulate radioactivity and airborne gaseous activity monitoring channels, which have sensitivities of 10^{-9} Ci/cm^3 and 10^{-6} Ci/cm^3 respectively, which are the sensitivities suggested for these channels by Regulatory Guide 1.45.

The leakage detection system is qualified for operation following an OBE. The particulate radioactivity monitoring channel is qualified for operation following an SSE.

Indicators and alarms for each leakage detection subsystem are provided in the main control room. At the site, procedures for converting various indications e.g. temp, Δt, and pressure, to a flow rate measurement will be provided by means of conversion curves wherever meaningful.

Major components within the drywell that by nature of their design are sources of leakage (e.g., sump seals, valve stem packing), are contained and piped to an equipment drain sump and thereby identified.

Equipment associated with systems within the drywell (e.g., vessels, piping, fittings) share a common free volume, therefore, their leakage detection systems are common. Steam or water leaks from such equipment are collected ultimately in an area drain sump.

Each of the sumps are protected against overflowing leaks from one source masking those from another.

As added back-up to the unidentified leakage drain system, the main steam lines within the steam tunnel inside the containment are monitored by temperature detectors within the tunnel.

c. Regulatory Guide 1.53 (6/73) - Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

See IEEE 279-1971, Paragraph 4.2, 7.6.2.3.

d. Regulatory Guide 1.62 (10/73) - Manual Initiation of Protective Actions

The FPC system is manually initiated from the main control room by actuation of system pump and valve controls.

7.6.3 REFERENCES

7.6-1 Morgan, W.R., "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors", APED-5706, November, 1968 (Rev. April, 1969).

7.6-2 Hatch Amendment 7, June 24, 1969, pages 7-3.0-1 and 7-5.0-1.

TABLE 7.6-1

IRM TRIPS

TRIP FUNCTION	NORMAL SETPOINT	TRIP ACTION
IRM upscale (high-high)	120/125 Full Scale	Scram, annunciator, red light display
or IRM inoperative	(See Note)	Scram, annunciator, red light display
IRM upscale (high)	108/125 Full Scale	Rod block, annunciator amber light display.
IRM downscale	3/125 Full Scale	Rod block (exception on most sensitive scale) annunciator, white light display
IRM bypassed		White light display

Note: IRM is inoperative if module interlock chain is broken, OPERATE-CALIBRATE switch is not in OPERATE position, or detector polarizing voltage is below 80 V.

TABLE 7.6-2

LPRM SYSTEM TRIPS

TRIP FUNCTION	TRIP ACTION	TRIP SETPOINT	TRIP ACTION
LPRM downscale	2% to full scale	3%	White light and annunciator
LPRM upscale	2% to full scale	100%	Amber light and annunciator
LPRM bypass	Manual switch		White light and APRM averaging compensation

TABLE 7.6-3

APRM SYSTEM TRIPS

TRIP FUNCTION	TRIP POINT RANGE	NOMINAL SETPOINT	ACTION
APRM downscale	2% to full scale	3%	Rod block, annunciator, white light display
APRM upscale (high)	Setpoint varied with flow, slope adjustable, intercepts separately adjustable	R(0.66 Flow + 42%) in run mode; fixed 12% in startup mode (R=Design Peaking Factor / Operating Peaking Factor)	Rod block, annunciator, amber light display
**APRM upscale (thermal power)	Setpoint varied with flow, slope adjustable, intercepts separately adjustable	R(0.66 flow + 54%)	Scram, annunciator, red light display
APRM upscale (neutron)	2% to full scale	120% in run mode 15% in startup mode	Scram, annunciator, red light display
APRM inoperative	Calibrate switch or too few inputs	Not in operate mode or module interlock chain broken or less than - 14 of 27 inputs or 14 of 22 inputs	Scram, rod block, annunciator, red light display
APRM Bypass	Manual Switch		white light

** APRM signal passes through a 6 second time constant circuit to simulate heat flux.

TABLE 7.6-4

NOT USED

TABLE 7.6-5

HIGH TO LOW PRESSURE SYSTEM INTERLOCKS INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
RHR Shutdown Cooling Isolation Pressure High	Press. Switch (B35-N018A,B)	50 - 1173 psig	135 psig	-	$\pm 2\%$	-
LPCI Injection Valve Differential Pressure High	Diff. Press Switch (E12-N009A, B, C)	0 - 1000 psid	696 psid	-	$\pm 2\%$	-
LPCS Injection Valve Differential Pressure High	Diff. Press. Switch (E21-N006)	0 - 1000 psid	696 psid	-	$\pm 2\%$	-

NOTES FOR TABLE 7.6-5

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.6-6

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR HIGH TO LOW PRESS. SYSTEM INTERLOCKS

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
RHR Shutdown Cooling Isol. Press. High	2	1
LPCI Injection Valve Diff. Press. High	3	3
LPCS Injection Valve Diff. Press. High	1	1

TABLE 7.6-7

LEAK DETECTION SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
RCIC Steam Supply Press. Low	Press. Switch (E31-NO22A-D)	3 - 78 psig	50 psig	–	–	–
RCIC Steam Supply High Flow or Instrument Line Break	Diff. Press. Switch (E31-N007A,B) (E31-NO13A,B)	200-0-200" H_2O	105" H_2O 181" H_2O	–	–	–
RCIC Turbine Exhaust Press. High	Press. Switch (E51-N012A-D)	0.5 - 80 psig	25 psig	–	–	–
RCIC Equipment Area High Diff. Temp.	Diff. Temp (E31-N603A,B)	0 - 150°F	–	–	–	–
RCIC Pipe Routing Area High Diff. Temp.	Diff. Temp. Switch (E31-N613A,B)	0 - 150°F	–	–	–	–
RWCU Equipment Area High Diff. Temp.	Diff. Temp. Switch (E31-N600A-F)	0 - 150°F	–	–	–	–
RWCU Equipment Area Ambient Temp. High	Temp. Switch (E31-N601A-F)	50 - 350°F	–	–	–	–

AMENDMENT NO. 10
July 1980
Page 1 of 4

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
RCIC Equipment Area Ambient Temp. High	Temp. Switch (E31-N602A,B)	50 - 350°F	-	-	-	-
RCIC Pipe Routing Area Ambient Temp. High	Temp. Switch (E31-N612A,B)	50 - 350°F	-	-	-	-
Main Steam Line Tunnel Temp. High	Temp. Switch (E31-N604A-D)	50 - 350°F	-	-	-	-
Main Steam Line Tunnel Diff. Temp. High	Diff. Temp. Switch (E31-N615A-D)	0 - 150°F	-	-	-	-
RHR Area Ambient Temp. High	Temp Switch (E31-N608A,B)	50 - 350°F	-	-	-	-
RHR Equipment Area Diff. Temp. High	Diff. Temp. Switch (E31-N614A,B)	0 - 150°F	-	-	-	-
Main Steam Line Pipe Routing Area in Turb. Bldg. Ambient Temp. High	Temp. Switch (E31-N616A-D Thru E31-N627A-D)	50 - 350°F	-	-	-	-

TABLE 7.6-7 (Continued)

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Main Steam Line Steam High Flow	Diff. Press. Switch (E31-N008A-D thru E31-N011A-D)	-15 to 0 to 150 psid	104 psid	-	-	-
RWCU Diff. Flow High	Diff. Flow Switch (E31-N605A,B)	-	57 gpm	-	-	-
React. Bldg. Equip. Draw Sump Level High	Level Switch (EDR-LS-N014)	-	-	-	-	-
React. Bldg. Floor Drain Sumps Level High	Level Switch (FDR-LS-N006A, B) (FDR-LS-N005A, B)	-	-	-	-	-

NOTES FOR TABLE 7.6-7

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.6-8

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR THE LEAK DETECTION SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
RCIC Steam Supply Press. Low	4	1
RCIC Steam Supply High Flow or Instrument Line Break	4	1
RCIC Turbine Exhaust Press. High	4	1
RCIC Equip. Area High Diff. Temp.	2	1
RCIC Pipe Routing Area High Diff. Temp.	2	1
RWCU Equip. Area High Diff. Temp.	6	1
RWCU Equip. Area Ambient Temp. High	6	1
RCIC Equip. Area Ambient Temp. High	2	1
RCIC Pipe Routing Area Diff. Temp.	2	1
Main Steam Line Tunnel Temp. High	4	1
Main Steam Line Tunnel Diff. Temp. High	4	1
RHR Area Ambient Temp. High	2	1
RHR Equip. Area Diff. Temp. High	2	1

TABLE 7.6-8 (continued)

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Main Steam Line Pipe Routing Area in Turbine Gen. Bldg. Ambient Temp. High	48	2
Main Steam Line Steam High Flow	16	2
RWCU Diff. Flow High	2	1
Reactor Bldg. Equip. Drain Sump Level High	1	1
Reactor Bldg. Floor Drain Sumps Level High	4	1

TABLE 7.6-9

NEUTRON MONITORING SYSTEM INSTRUMENTATION SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
IRM High High	-	-	120/125 of F.S.	-	-	-
IRM INOP	-	-	(5)	-	-	-
APRM UPSCALE (Thermal Power)	-	-	R(0.66 Flow + 54%) (6)	-	-	-
APRM UPSCALE (Neutron)	-	2% to F.S.	120% in Run 15% in Startup	-	-	-
APRM INOP	-	-	(7)	-	-	-

NOTES FOR TABLE 7.6-9

(1) See Chapter 16,"Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

(5) INOP trip is activated when the trip unit is removed from its position, the unit is in "Calibration", or the detector polarizing voltage is below 80V.

(6) $R = \frac{\text{Design Peaking Factor}}{\text{Operating Peaking Factor}}$

(7) INOP trip is activated when the trip unit is removed from its position, the unit is in "Calibration", or 14 of 27 inputs or 14 or 22 inputs exist.

TABLE 7.6-10

CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR THE NEUTRON MONITORING SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
IRM High High	8	2
IRM INOP	8	2
APRM UPSCALE (Thermal Power)	8	2
APRM UPSCALE (Neutron)	8	2
APRM INOP	8	2

TABLE 7.6-11

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM INSTRUMENTATIONN SPECIFICATIONS

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)

LATER

NOTES FOR TABLE 7.6-11

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

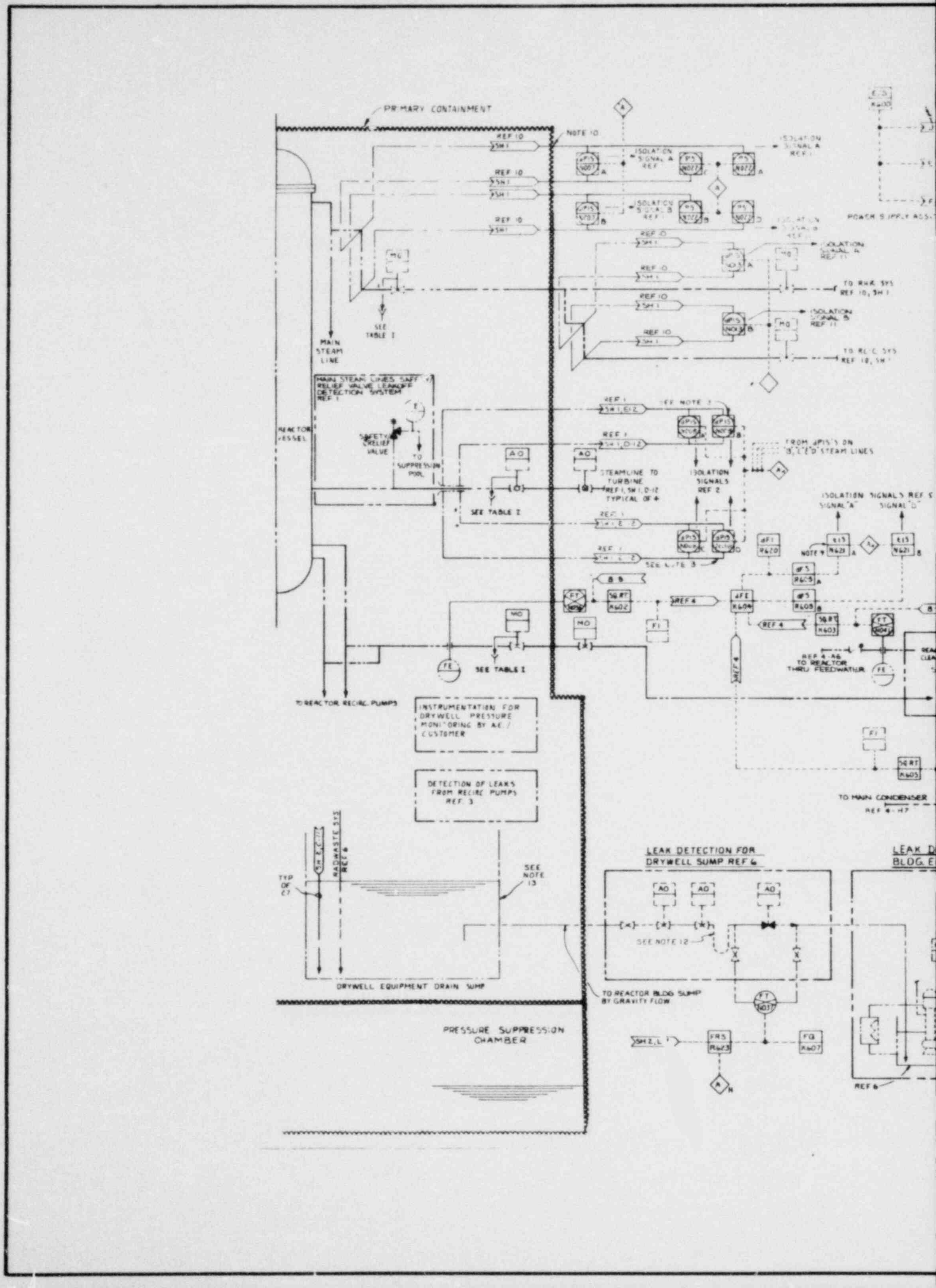
(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.6-12

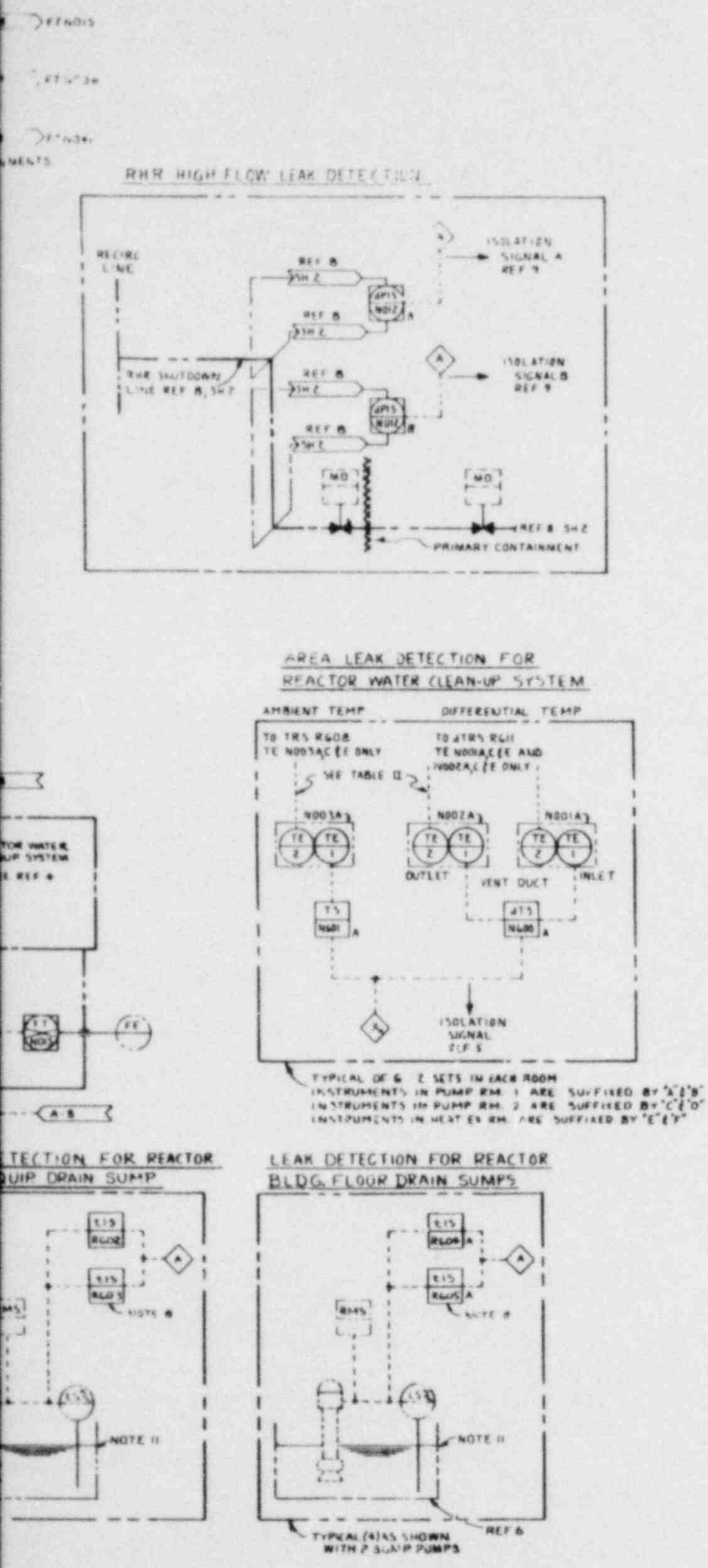
CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR THE SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)

LATER


PRIMARY CONTAINMENT
NOTE 10
REF 10
ISOLATION SIGNAL A
ISOLATION SIGNAL B
TO RHR SYS REF 10, SH 1
TO RCIC SYS REF 10, SH 1
SEE TABLE I
MAIN STEAM LINE
MAIN STEAM LINES SAFETY RELIEF VALVE LEAKOFF DETECTION SYSTEM REF 1
REACTOR VESSEL
SAFETY RELIEF VALVE
TO SUPPRESSION POOL
STEAMLINE TO TURBINE REF 1, SH 1, D-12 TYPICAL OF 4
ISOLATION SIGNALS REF 2
SEE NOTE 3
ISOLATION SIGNALS REF 5
SIGNAL "A"
SIGNAL "D"
TO REACTOR RECIRC PUMPS
REF 4-A6 TO REACTOR THRU FEEDWATER
INSTRUMENTATION FOR DRYWELL PRESSURE MONITORING BY A.E./ CUSTOMER
DETECTION OF LEAKS FROM RECIRC PUMPS REF. 3
TO MAIN CONDENSER
LEAK DETECTION FOR DRYWELL SUMP REF 6
RADWASTE SYS REF 6
SEE NOTE 13
SEE NOTE 12
DRYWELL EQUIPMENT DRAIN SUMP
TO REACTOR BLDG SUMP BY GRAVITY FLOW
PRESSURE SUPPRESSION CHAMBER
POWER SUPPLY ASSY
REF 6

Amendment No. 10
July 1980



NOTES:

1. REMOVED
2. REMOVED

5. INSTRUMENT LINE VALVING MUST COMPLY WITH INSTRUMENT PIPING STANDARD REF. 14.

7. ONE ΔTS IS USED TO MEASURE SUMP FILLUP TIME AND THE SECOND ΔTS IS MONITORING SUMP PUMPOUT TIME
8. BYPASS TIMER SHALL BE PROVIDED TO OVERRIDE THE ISOLATION SIGNAL DURING THE REACTOR WATER CLEANUP SYSTEM START UP

15. TEMPERATURE ELEMENTS SHALL BE PLACED ALONG THE MAIN STEAM LINE PIPING APPROXIMATELY 36 INCHES ABOVE THE STEAM LINE. TEMPERATURE DETECTORS SHALL BE LOCATED OR SHIELDED SO THAT THE DETECTOR IS SENSITIVE TO THE AIR TEMPERATURE AND NOT THE RADIATED HEAT FROM OF EQUIPMENT.

12. HIGH PRESS. CORE SPRAY SYSTEM P&ID - - - - - - E22-1010
13. LEAK DETECTION SYSTEM DESIGN SPEC. - - - - - - E31-4010
14. LOW PRESS. CORE SPRAY SYSTEM P&ID - - - - - - E21-1010
15. LEAK DETECTION SYSTEM IDS - - - - - - E31-3010

SUPPORTING DOCUMENTS:

1. DRYWELL COOLING SYSTEM DESIGN SPEC - - - - - - A62-4200

RECEIVED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LEAK DETECTION SYSTEM IED - SHT. 1	FIGURE 7.6-1a

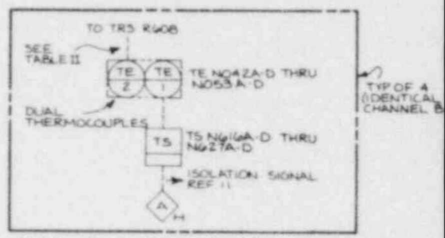
TABLE II
EQUIPMENT AREA TEMPERATURE MONITORING

AREA MONITORED	AMBIENT TEMPERATURE: TE	TR POINT NO.	ALARM GROUPING	DIFFERENTIAL TEMPERATURE: TE INLET	TE OUTLET	dTR POINT NO.	ALARM GROUPING
STEAM PIPE TUNNEL	N031A	R608-1	SW-1	N029A	N030A	R611-1	SW-1
PRIMARY CONTAINMENT " " " " " "	N017A B C D	R608-2 3 4 5	SW-2	N020	N019	R611-2	SW-2
RHR EQUIP. ROOM 1 RHR EQUIP. ROOM 2	N018A N018B	R608-6 R608-7	SW-3	N027A N027B	N028A N028B	R611-6 R611-7	SW-3
RCIC PIPE ROUTING RCIC EQUIP AREA	N024A N004A	R608-8 R608-9	SW-4 SW-5	N025A N005A	N026A N006A	R611-8 R611-9	SW-4 SW-5
RWCU PUMP ROOM 1 RWCU PUMP ROOM 2 RWCU HX ROOM	N003A N003C N003E	R608-10 R608-11 R608-12	SW-6	N001A N001C N001E	N002A N002C N002E	R611-10 R611-11 R611-12	SW-6
TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG TURBINE BLDG	N042A N043A N044A N045A N046A N047A N048A N049A N050A N051A N052A N053A	R608-13 R608-14 R608-15 R608-16 R608-17 R608-18 R608-19 R608-20 R608-21 R608-22 R608-23 R608-24	SW-1				

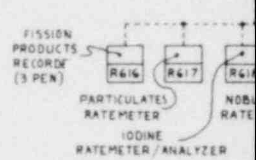
TABLE I

THE FOLLOWING VALVES INSIDE THE DRYWELL ARE EQUIPPED WITH VALVE STEM LEAK-OFF. TYPICAL ARRANGEMENT SHOWN.

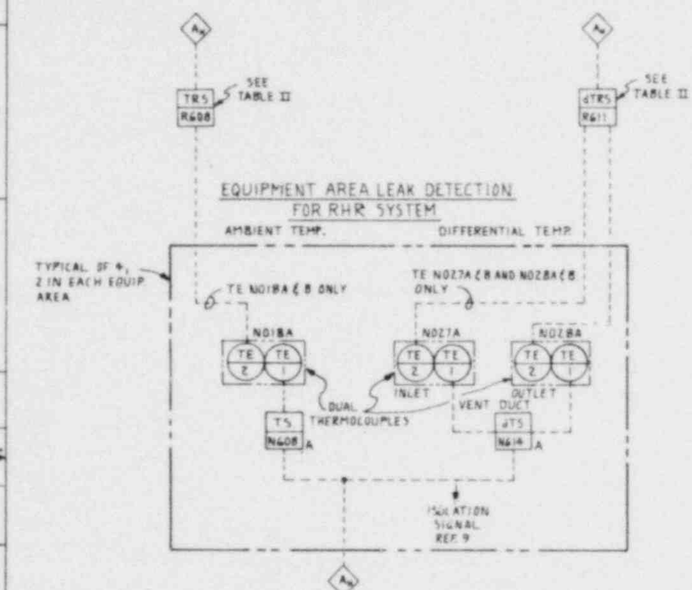
VALVES WITH LEAK-OFF	LOCATION	LEAK-OFF EQUIP SUFFIX	TR POINT NO
B22-F022A,B,C,D	REF 1, SHT 1, C-10	A1,A2,A3,A4	R622-1-4
B35-F023A & B B35-F060A & B B35-F061A & B B35-F067A & B	REF 3, SHT 1, REF 3, SHT 1, B-3 REF 3, SHT 1, K-9 REF 3, SHT 1, B-3	B1,B2 B3,B4 B5,B6 B7,B8	R622-5-12
E12-F009 E12-F041A E12-F041B E12-F041C E12-F050A E12-F050B	REF 8, SHT 2, E-12 REF 8, SHT 2, D-11 REF 8, SHT 1, E-3 REF 8, SHT 1, G-3 REF 8, SHT 2, E-12 REF 8, SHT 1, C-3	C1 C3 C4 C5 C6 C7	R612-1-6
E21-F006	REF 14, C-4	G1	R612-7
E22-F005	REF 12, C-11	D1	R612-8
E51-F063 E51-F066	REF 10, SH1, D-5 REF 10, SH1, B-4	E1 E2	R612-9-10
G33-F001 G33-F100 G33-F101 G33-F102 G33-F106	REF 4, C-2 REF 4, F-1 REF 4, D-2 REF 4, D-1 REF 4, F-1	F1 F2 F3 F4 F5	R612-11 THRU 15
B22-F016	REF 1, SH1, F10	A5	R612-16



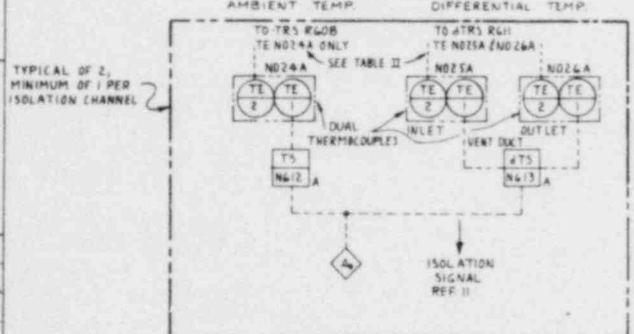
AREA LEAK DETECTION FOR MAIN STEAM LINE TURBINE BLDG NOTE 15
AMBIENT TEMP. CHANNEL A



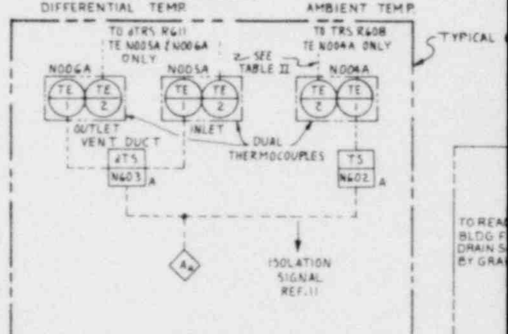
EQUIPMENT AREA LEAK DETECTION FOR RHR SYSTEM
AMBIENT TEMP. DIFFERENTIAL TEMP.



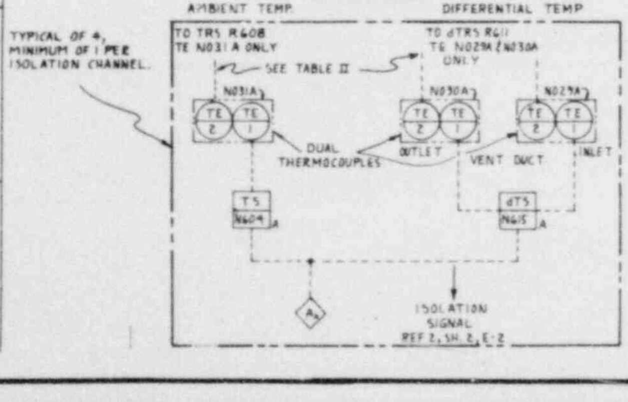
AREA LEAK DETECTION FOR RCIC PIPE ROUTING AREA
AMBIENT TEMP. DIFFERENTIAL TEMP.



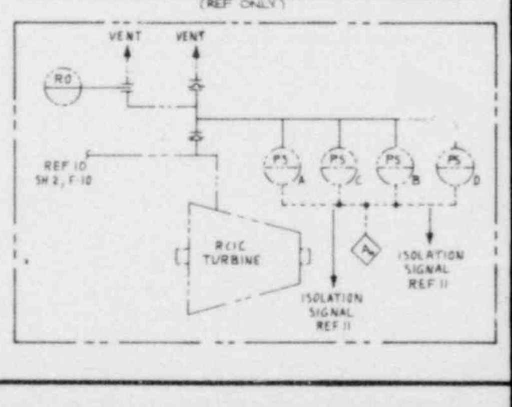
EQUIPMENT AREA LEAK DETECTION FOR RCIC SYSTEM
DIFFERENTIAL TEMP. AMBIENT TEMP.



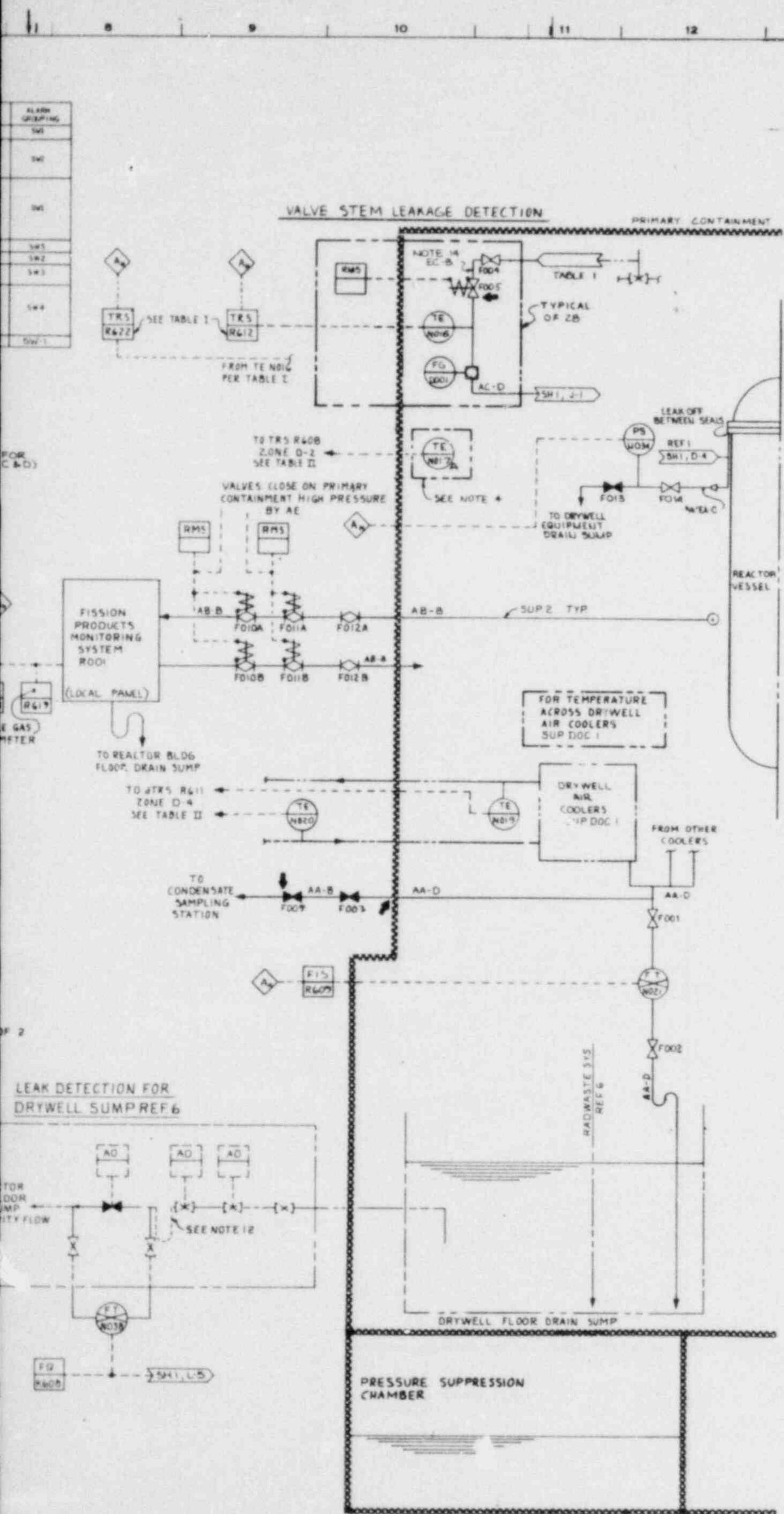
AREA LEAK DETECTION FOR MAIN STEAM LINE PIPE TUNNEL
AMBIENT TEMP. DIFFERENTIAL TEMP.



SAFETY SYSTEM FOR RCIC TURBINE EXHAUST
(REF ONLY)

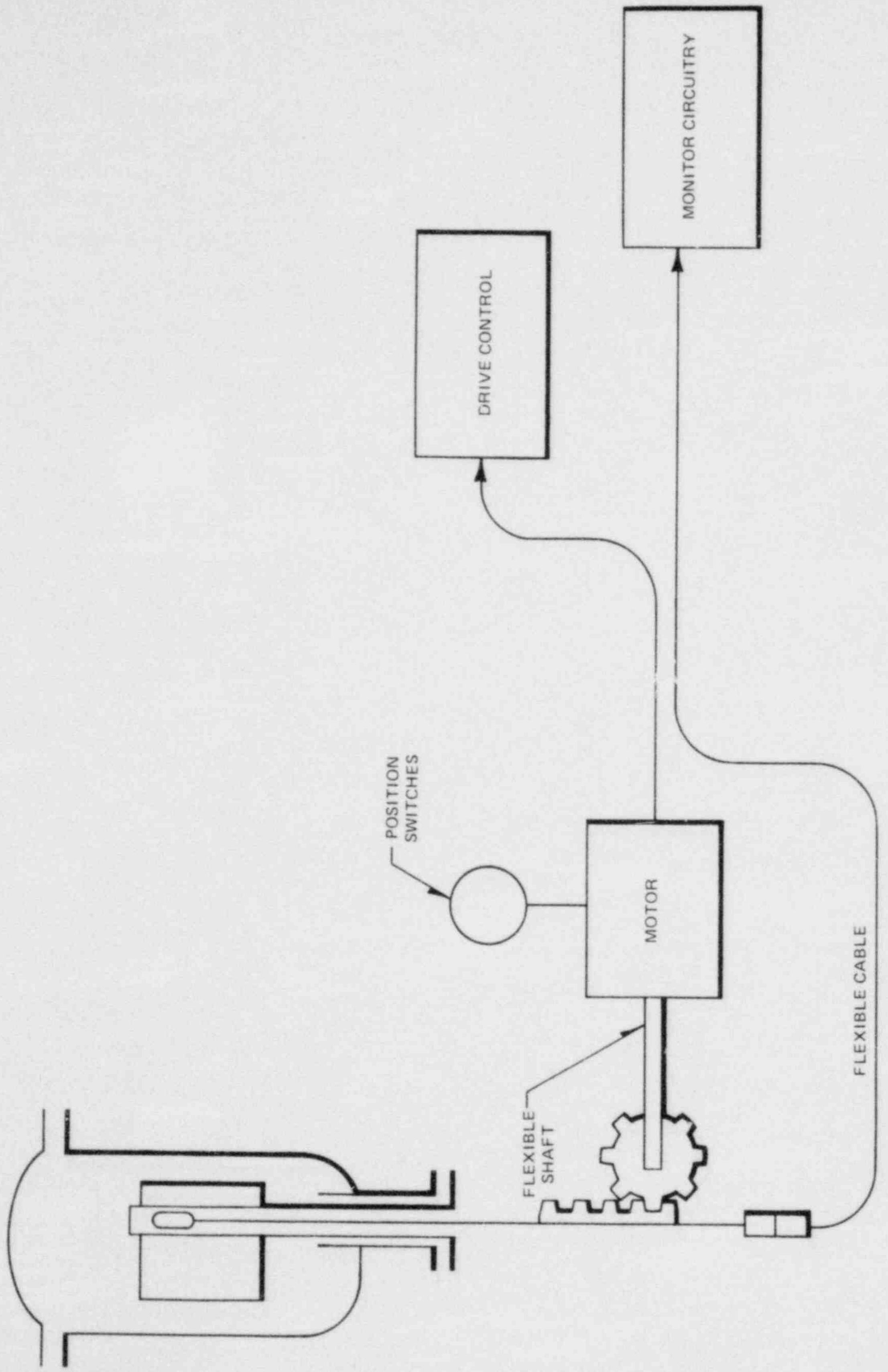


AMENDMENT NO. 10
July 1980



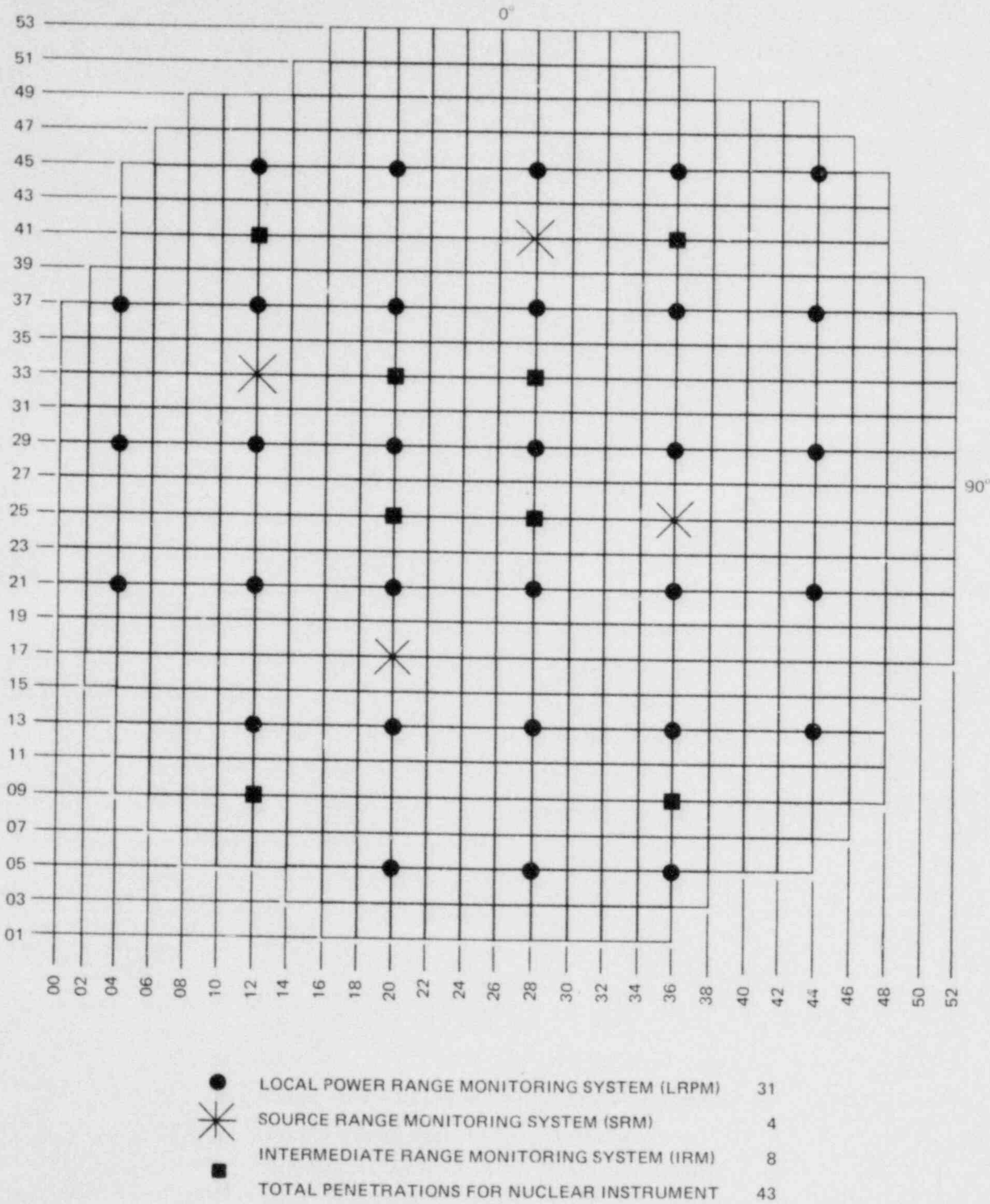
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LEAK DETECTION SYSTEM IED - SHEET 2	FIGURE 7.6-1b

AMENDMENT NO. 10
July 1980

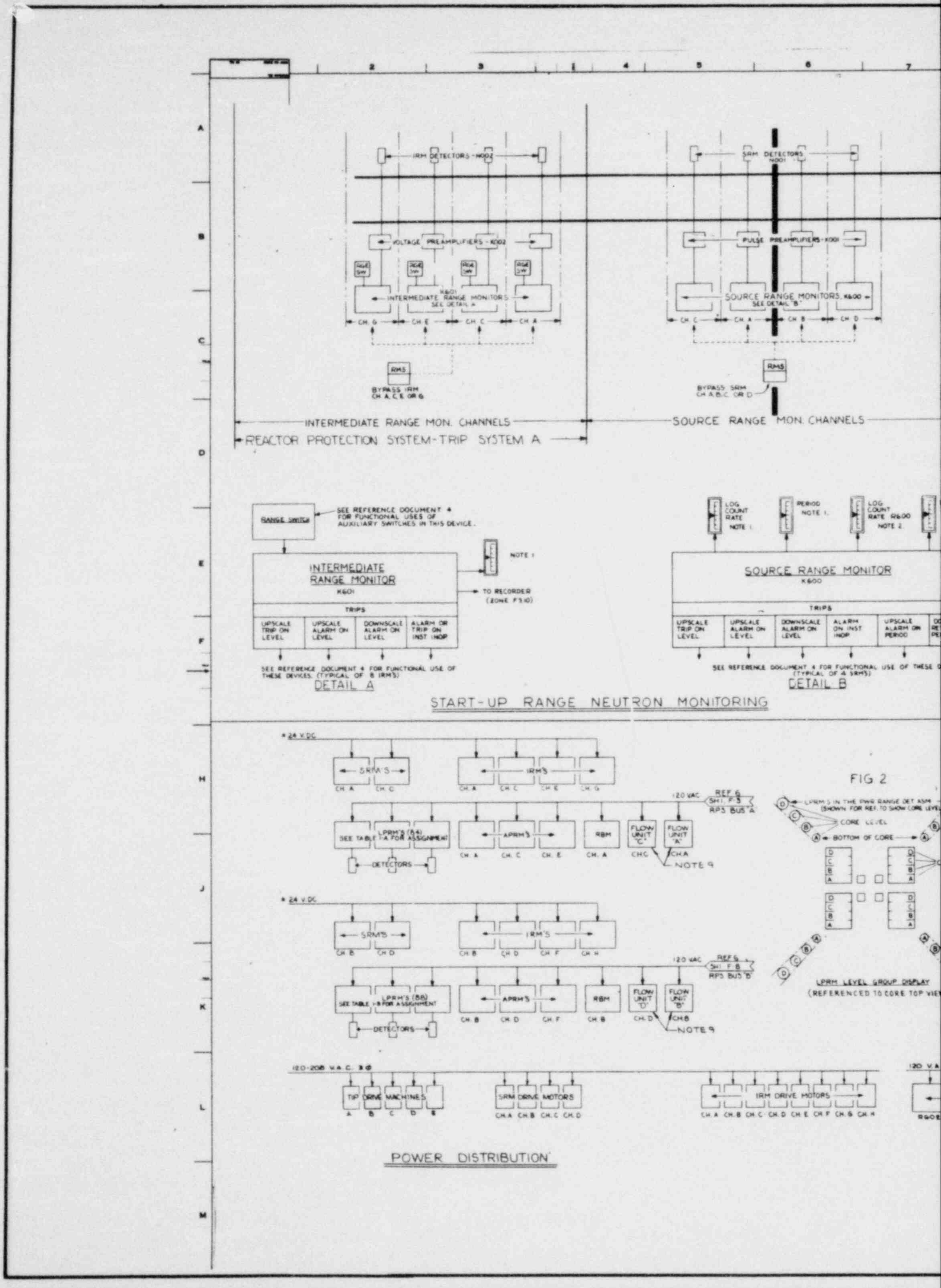


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	DETECTOR DRIVE SYSTEM SCHEMATIC.	FIGURE 7.6-2

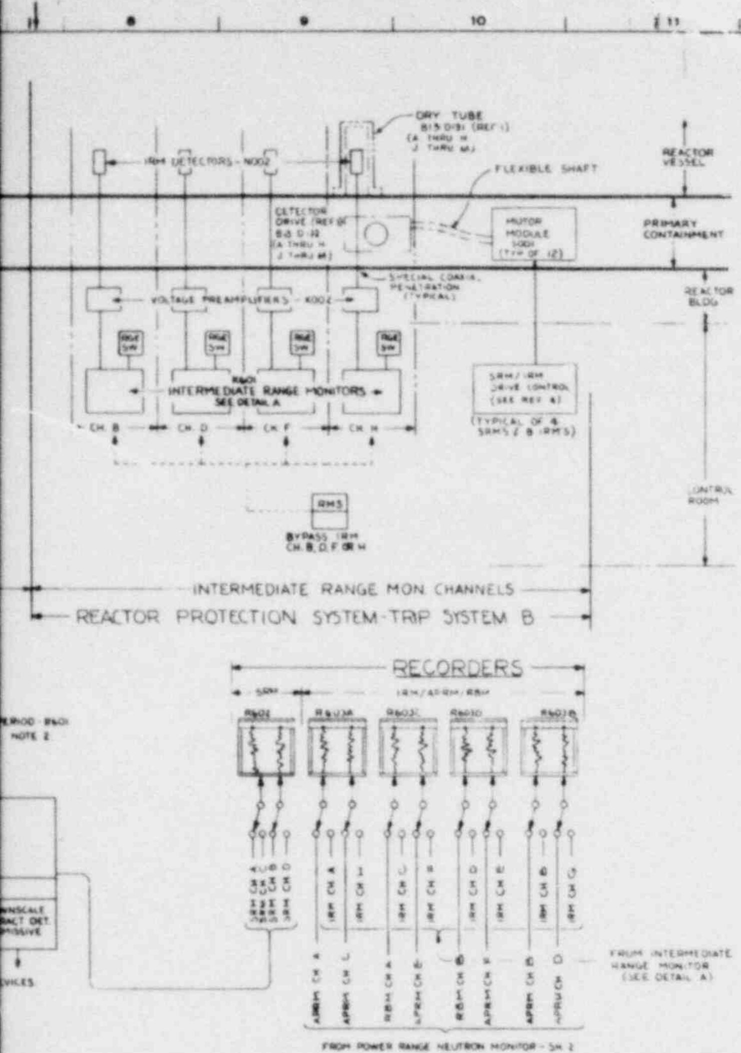
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	VESSEL PENETRATIONS FOR NUCLEAR INSTRUMENTATION.	FIGURE 7.6-3


IRM DETECTORS - N002
VOLTAGE PREAMPLIFIERS - K002
K601 INTERMEDIATE RANGE MONITORS SEE DETAIL A
CH. G CH. E CH. C CH. A
RMS
BYPASS IRM CH A,C,E OR G
SRM DETECTORS N001
PULSE PREAMPLIFIERS - K001
SOURCE RANGE MONITORS, K600 SEE DETAIL "B"
CH. C CH. A CH. B CH. D
RMS
BYPASS SRM CH A,B,C OR D
INTERMEDIATE RANGE MON. CHANNELS
SOURCE RANGE MON. CHANNELS
REACTOR PROTECTION SYSTEM-TRIP SYSTEM A
RANGE SWITCH
SEE REFERENCE DOCUMENT 4 FOR FUNCTIONAL USES OF AUXILIARY SWITCHES IN THIS DEVICE.
INTERMEDIATE RANGE MONITOR K601
NOTE 1
TO RECORDER (ZONE F3.10)
TRIPS
UPSCALE TRIP ON LEVEL
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ALARM OR TRIP ON INST INOP
SEE REFERENCE DOCUMENT 4 FOR FUNCTIONAL USE OF THESE DEVICES. (TYPICAL OF 8 IRM'S)
DETAIL A
LOG COUNT RATE NOTE 1
PERIOD NOTE 1
LOG COUNT RATE R600 NOTE 2
SOURCE RANGE MONITOR K600
TRIPS
UPSCALE TRIP ON LEVEL
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ALARM ON INST INOP
UPSCALE ALARM ON PERIOD
SEE REFERENCE DOCUMENT 4 FOR FUNCTIONAL USE OF THESE DEVICES (TYPICAL OF 4 SRMS)
DETAIL B
START-UP RANGE NEUTRON MONITORING
+24 V.DC
SRM'S
CH. A CH. C
IRM'S
CH. A CH. C CH. E CH. G
120 VAC
REF 6 SH1 F3
RPS BUS 'A'
SEE TABLE 1-A FOR ASSIGNMENT
LPRM'S (84)
APRM'S
CH. A CH. C CH. E
RBM
CH. A
FLOW UNIT 'C'
FLOW UNIT 'A'
CH.C CH.A
NOTE 9
DETECTORS
+24 V.DC
SRM'S
CH. B CH. D
IRM'S
CH. B CH. D CH. F CH. H
120 VAC
REF 6 SH1 F8
RPS BUS 'B'
SEE TABLE 1-B FOR ASSIGNMENT
LPRM'S (88)
APRM'S
CH. B CH. D CH. F
RBM
CH. B
FLOW UNIT 'D'
FLOW UNIT 'B'
CH. D CH. B
NOTE 9
DETECTORS
FIG 2
CORE LEVEL
BOTTOM OF CORE
LPRM LEVEL GROUP DISPLAY
(REFERENCED TO CORE TOP VIEW)
120-208 V.A.C. 3Ø
TIP DRIVE MACHINES
A B C D E
SRM DRIVE MOTORS
CH.A CH.B CH.C CH.D
IRM DRIVE MOTORS
CH.A CH.B CH.C CH.D CH.E CH.F CH.G CH.H
POWER DISTRIBUTION

AMENDMENT NO. 10
July 1980

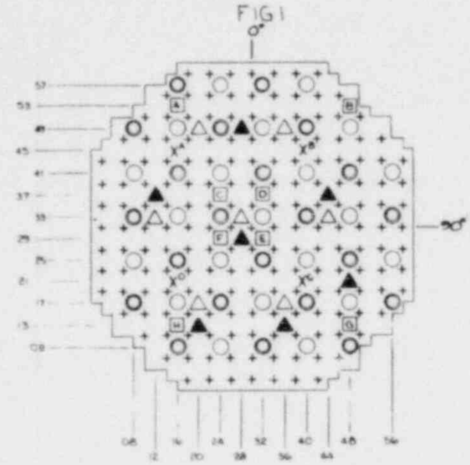


FCF 239X275G3 (C51-1010)

ABBREVIATIONS

SRM	SOURCE RANGE MONITOR
IRM	INTERMEDIATE RANGE MONITOR
RBM	ROD BLOCK MONITOR
LPRM	LOCAL POWER RANGE MONITOR
APRM	AVERAGE POWER RANGE MONITOR
TIP	TRAVERSING IN-CORE PROBE
RGE SW	RANGE SWITCH
CH	CHANNEL

FIG 1

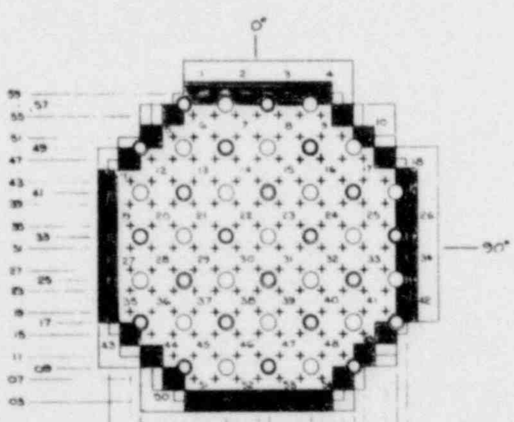


DETECTOR & CONTROL ELEMENT ARRANGEMENT
(TOP VIEW OF CORE)

LEGEND

+	CONTROL RODS (185)
X	SRM DETECTORS (4)
▲	SRM EMITTING SOURCES (7)
△	SRM SPARE EMITTING SOURCE POSITION (7)
○	LPRM DETECTOR ASM IN THE PWR RANGE DETECTOR ASM - TRIP SYS 'A' (21)
○	LPRM DETECTOR ASM IN THE PWR RANGE DETECTOR ASM - TRIP SYS 'B' (22)
□	IRM DETECTORS (8)

FIG 3



(TOP VIEW OF CORE)

WHEN A ROD IS SELECTED IN ANY GROUP, THE LPRM DETECTOR ASSEMBLIES ASSIGNED TO THAT GROUP (SEE FIG. 3) ARE ROUTED VIA THE LPRM TO THE RBM'S SUCH THAT THE "A" & "C" LEVEL DETECTOR SIGNALS GO TO RBM "A" AND THE "B" & "D" LEVEL DETECTOR SIGNALS GO TO RBM "B". WHEN A PERIPHERAL ROD (SHADED IN FIG. 3) IS SELECTED, THE RBM'S ARE AUTOMATICALLY BYPASSED. THE LPRM SIGNALS ARE ROUTED TO THE LPRM LEVEL GROUP DISPLAY AS SHOWN IN FIG. 3. WHEN A DETECTOR ASSEMBLY IS NOT PRESENT IN A GROUP THE CORRESPONDING READOUTS IN THE LPRM LEVEL GROUP DISPLAY WILL BE ZERO.

NOTES:

1. PARTS ARE LOCATED ADJACENT TO OR ON THE SIGNAL CONDITIONING EQUIPMENT PERFORMING THE FUNCTION INDICATED.
2. PART IS LOCATED ON THE MAIN CONTROL ROOM PANEL.
3. POSITION INFORMATION IS INPUT EVERY 1 INCH. FLUX LEVEL INFORMATION IS INPUT EVERY 3 INCHES ON WITHDRAWAL.
4. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY NO. C51 UNLESS OTHERWISE NOTED.
5. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET LISTED IN MPL FOR EACH INSTRUMENT.
6.
7. EXCEPT FOR PORT NO. 10, THE EXACT ASSIGNMENT OF TIP GUIDE TUBES FROM SPECIFIC INDEXING MECHANISMS TO SPECIFIC POWER RANGE DETECTOR ASSEMBLIES IN RESPECTIVE GROUPS SHOWN IS DETERMINED BY OTHERS AND WILL BE DELINEATED LATER.
8. APRM CHANNEL "C" OUTPUT SIGNAL SHALL GO TO THE RECIRC SYSTEM EXCEPT WHEN CHANNEL "C" IS BY-PASSED APRM "E" SIGNAL SHALL AUTOMATICALLY GO TO THE RECIRC SYSTEM.
9. FLOW UNIT INCLUDES FLOW SUMMER, POWER SUPPLY & SQUARE ROOT FUNCTIONS AS SHOWN ON REFERENCE DWG. 2.

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. REACTOR ASSEMBLY ARRANGEMENT	B13-2010
2. REACTOR RECIRCULATION SYS.-P&ID	B35/B05-1010
3. CONTROL ROD HYDRAULIC SYS.-FCD	C11/C12-1010
4. NEUTRON MONITORING SYS.-FCD	C51-1020
5. DESIGN SPECIFICATION	C51-4010
6. REACTOR PROTECTION SYS IED.	C71/72-1010

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM IED - SHEET 1	FIGURE 7.6-4a

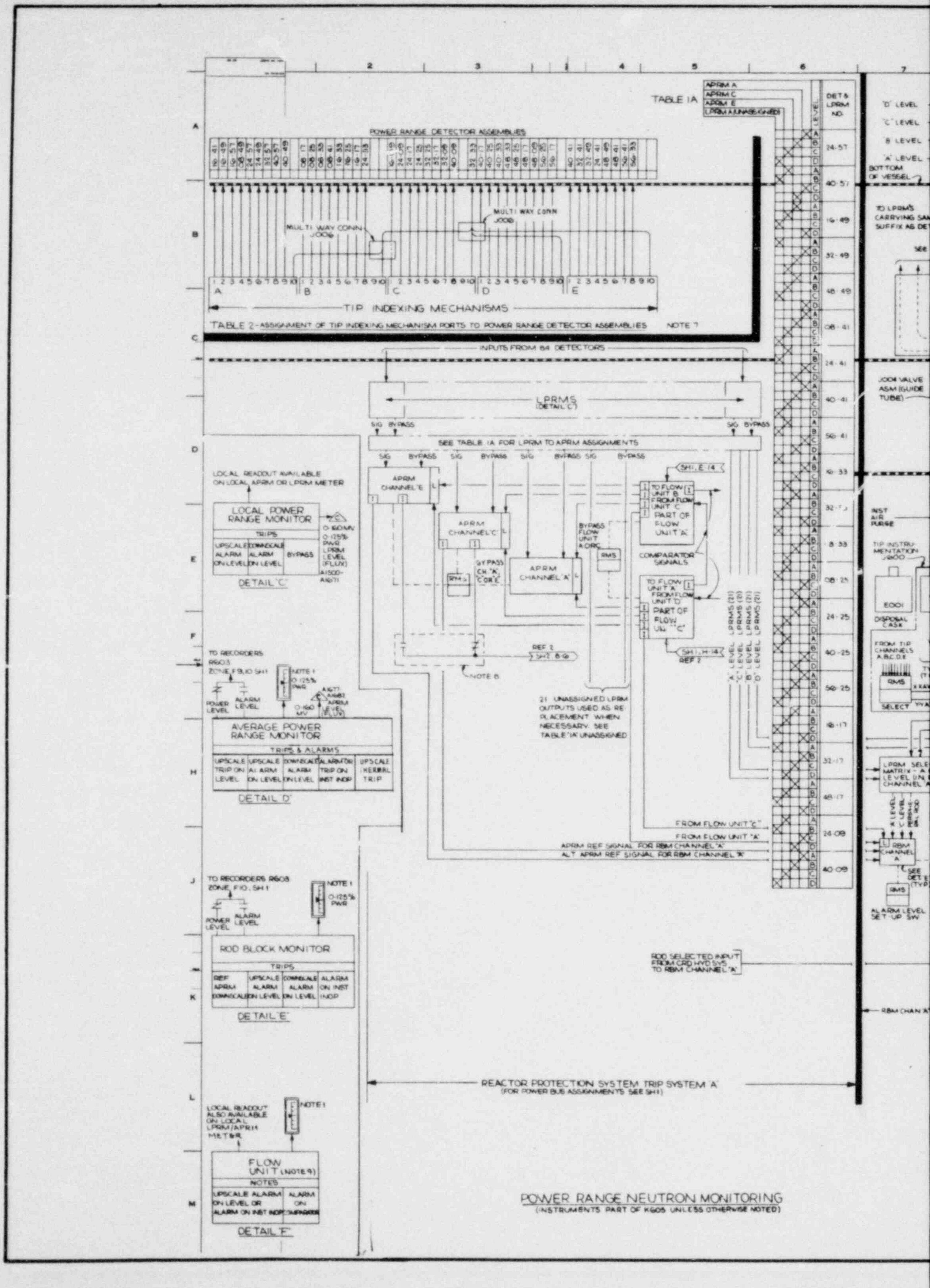
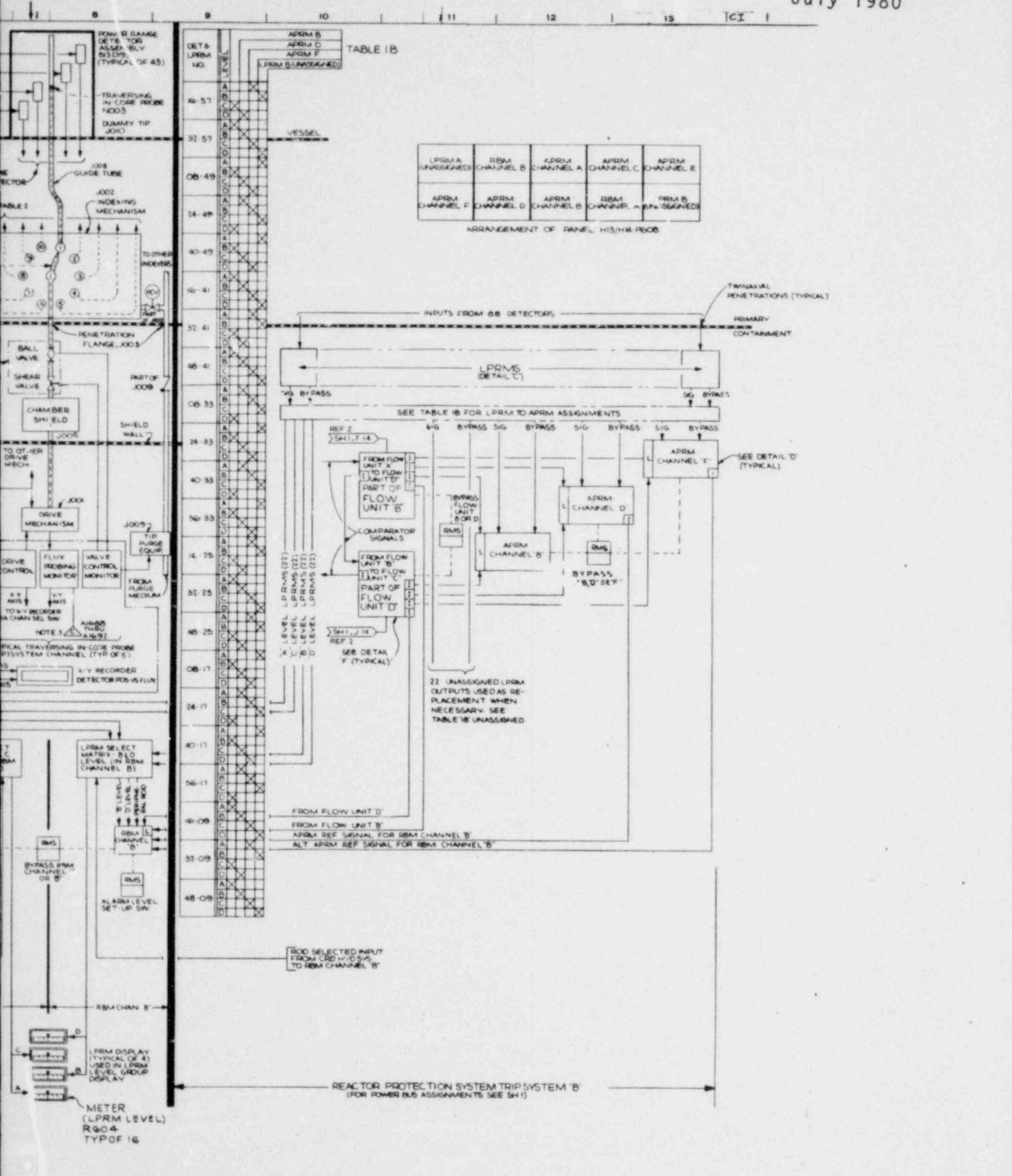


TABLE IA
APRM A
APRM C
APRM E
LPRM A (UNASSIGNED)
LEVEL
DETS LPRM NO.
POWER RANGE DETECTOR ASSEMBLIES
MULTI WAY CONN J006
MULTI WAY CONN J008
TIP INDEXING MECHANISMS
TABLE 2 - ASSIGNMENT OF TIP INDEXING MECHANISM PORTS TO POWER RANGE DETECTOR ASSEMBLIES
NOTE 7
INPUTS FROM 84 DETECTORS
LPRMS (DETAIL 'C')
SIG
BYPASS
SEE TABLE IA FOR LPRM TO APRM ASSIGNMENTS
APRM CHANNEL 'E'
APRM CHANNEL 'C'
APRM CHANNEL 'A'
TO FLOW UNIT B
FROM FLOW UNIT 'C'
PART OF FLOW UNIT 'A'
TO FLOW UNIT 'A'
FROM FLOW UNIT 'D'
PART OF FLOW UNIT 'C'
COMPARATOR SIGNALS
BYPASS FLOW UNIT A OR C
RMS
NOTE 8
21 UNASSIGNED LPRM OUTPUTS USED AS REPLACEMENT WHEN NECESSARY. SEE TABLE 'IA' UNASSIGNED
LOCAL READOUT AVAILABLE ON LOCAL APRM OR LPRM METER
LOCAL POWER RANGE MONITOR
TRIPS
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
BYPASS
DETAIL 'C'
TO RECORDERS R603
AVERAGE POWER RANGE MONITOR
TRIPS & ALARMS
UPSCALE TRIP ON LEVEL
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ALARM OR TRIP ON INST INOP
UPSCALE THERMAL TRIP
DETAIL 'D'
TO RECORDERS R608
ROD BLOCK MONITOR
REF APRM DOWNSCALE
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ALARM ON INST INOP
DETAIL 'E'
FROM FLOW UNIT 'C'
FROM FLOW UNIT 'A'
APRM REF SIGNAL FOR RBM CHANNEL 'A'
ALT APRM REF SIGNAL FOR RBM CHANNEL 'A'
ROD SELECTED INPUT FROM CRD HYD SYS TO RBM CHANNEL 'A'
REACTOR PROTECTION SYSTEM TRIP SYSTEM 'A'
(FOR POWER BUS ASSIGNMENTS SEE SH1)
LOCAL READOUT ALSO AVAILABLE ON LOCAL LPRM/APRM METER
FLOW UNIT (NOTE 9)
NOTES
UPSCALE ALARM ON LEVEL OR ALARM ON INST INOP
ALARM ON COMPARATOR
DETAIL 'F'
'D' LEVEL
'C' LEVEL
'B' LEVEL
'A' LEVEL
BOTTOM OF VESSEL
J004 VALVE ASM (GUIDE TUBE)
INST AIR PURGE
E001
DISPOSAL CASK
RBM CHANNEL 'A'
ALARM LEVEL SET-UP SW
POWER RANGE NEUTRON MONITORING
(INSTRUMENTS PART OF K605 UNLESS OTHERWISE NOTED)

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM IED - SHEET 2	FIGURE 7.6-4b

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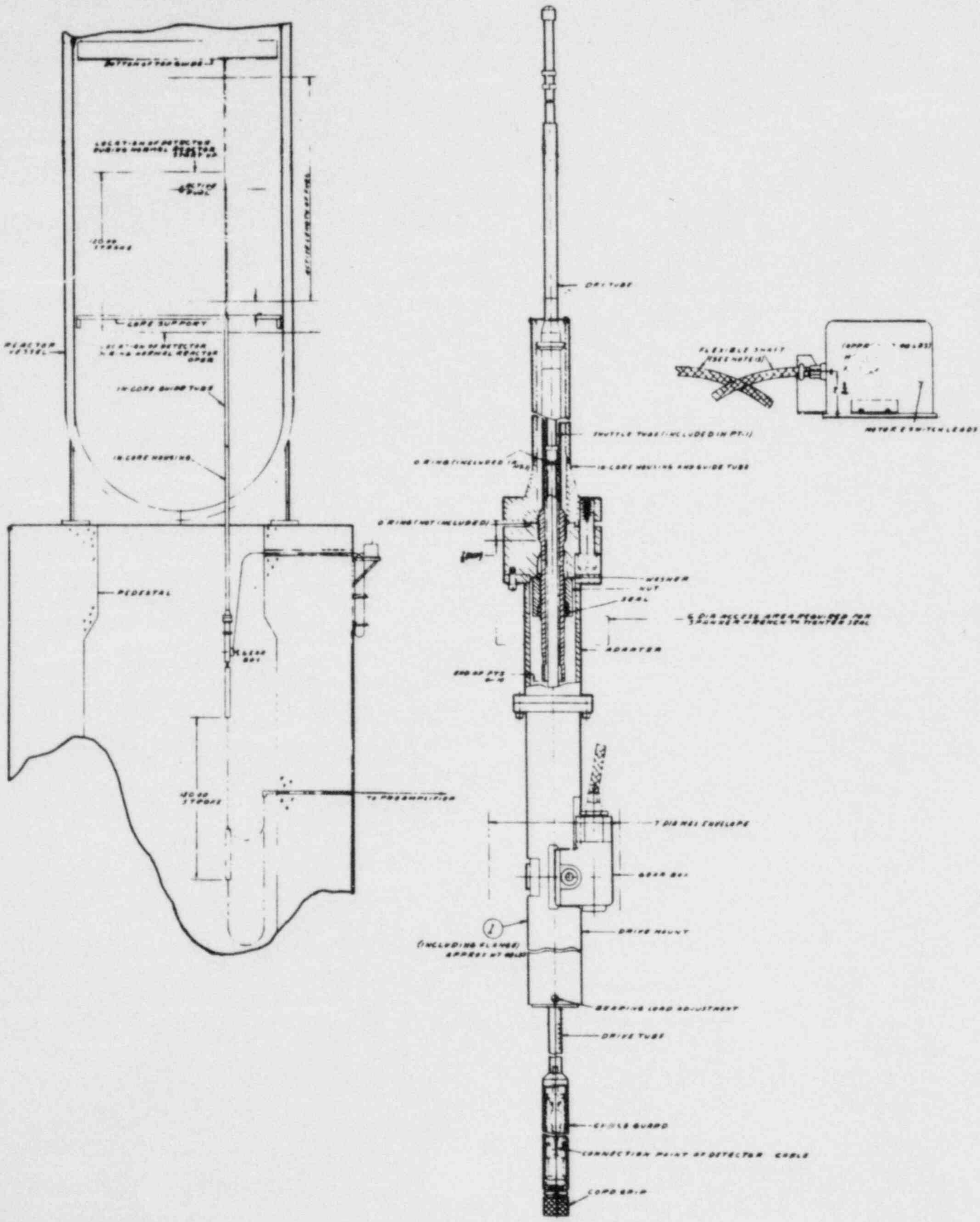


WASHINGTON PUBLIC POWER JPPLY SYSTEM NUCLEAR PROJECT NO. 2	SRM/IRM NEUTRON MONITORING UNIT	FIGURE 7.6-5

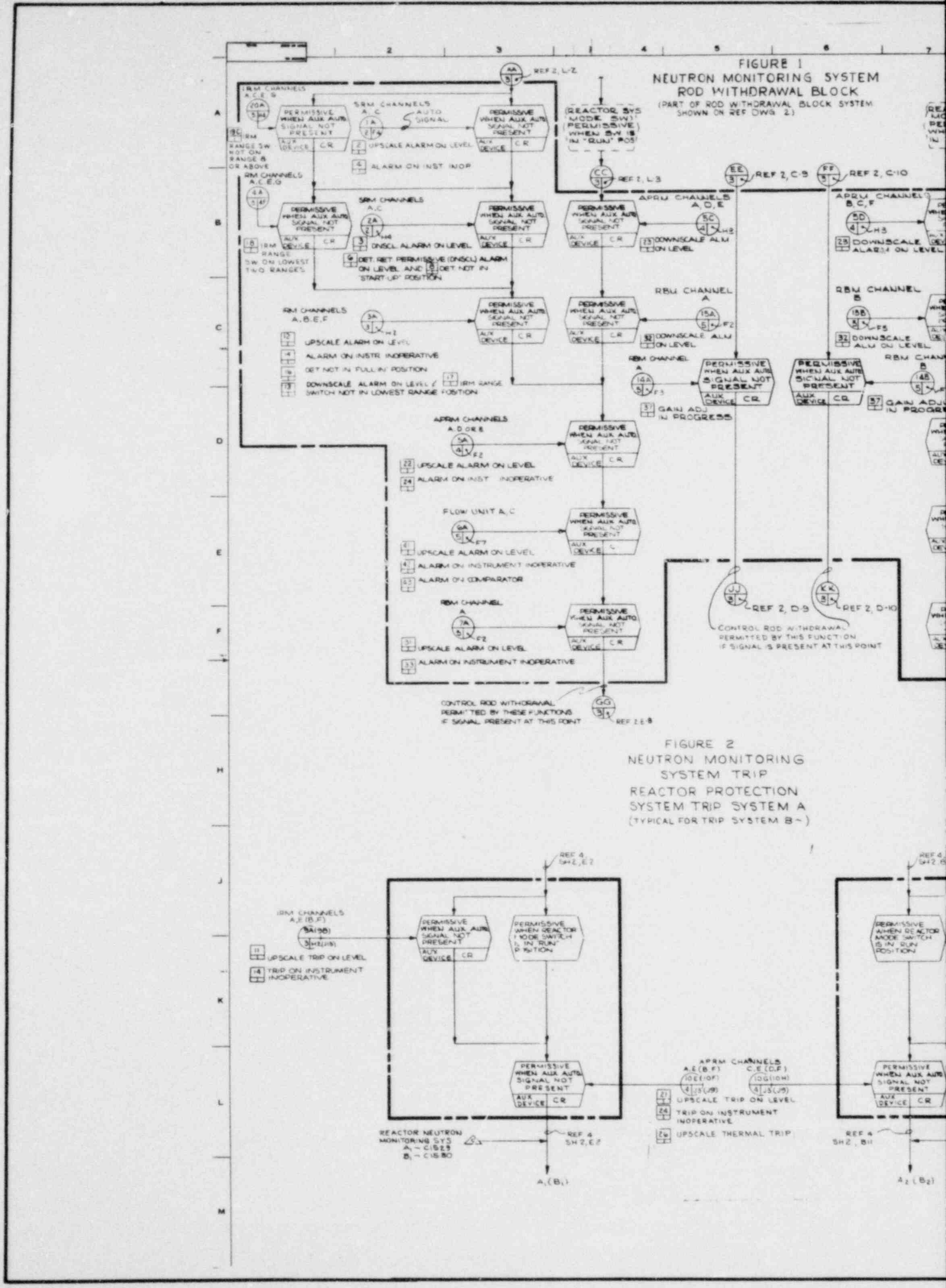
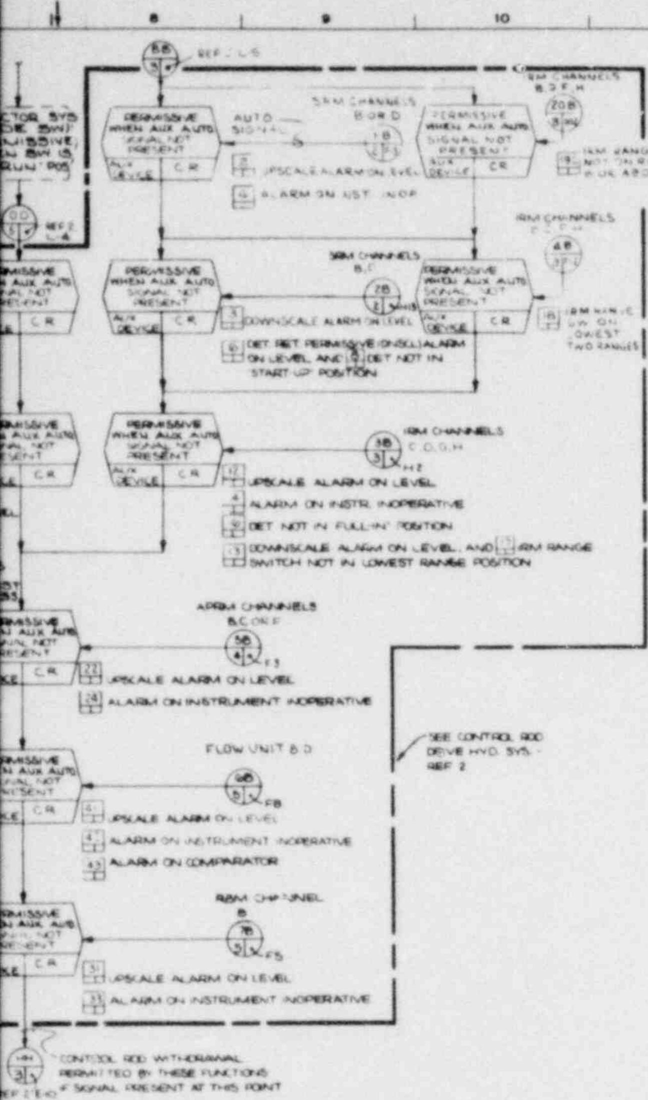


FIGURE 1
NEUTRON MONITORING SYSTEM
ROD WITHDRAWAL BLOCK
(PART OF ROD WITHDRAWAL BLOCK SYSTEM SHOWN ON REF DWG 2)
IRM CHANNELS A,C,E,G
SRM CHANNELS A,C
AUTO SIGNAL
PERMISSIVE WHEN AUX AUTO SIGNAL NOT PRESENT
AUX DEVICE CR
RANGE SW NOT ON RANGE 8 OR ABOVE
UPSCALE ALARM ON LEVEL
ALARM ON INST. INOP
REACTOR SYS MODE SW PERMISSIVE WHEN SW IS IN "RUN" POS
REF 2, L-2
REF 2, L-3
REF 2, C-9
REF 2, C-10
APRM CHANNELS A,D,E
APRM CHANNELS B,C,F
DOWNSCALE ALM ON LEVEL
DOWNSCALE ALARM ON LEVEL
IRM RANGE SW ON LOWEST TWO RANGES
DNSCL ALARM ON LEVEL
DET. RET PERMISSIVE (DNSCL) ALARM ON LEVEL AND DET NOT IN "START-UP" POSITION
RBM CHANNEL A
RBM CHANNEL B
RM CHANNELS A,B,E,F
UPSCALE ALARM ON LEVEL
ALARM ON INSTR INOPERATIVE
DET NOT IN 'FULL IN' POSITION
DOWNSCALE ALARM ON LEVEL & IRM RANGE SWITCH NOT IN LOWEST RANGE POSITION
GAIN ADJ IN PROGRESS
APRM CHANNELS A,D OR B
UPSCALE ALARM ON LEVEL
ALARM ON INST INOPERATIVE
FLOW UNIT A,C
UPSCALE ALARM ON LEVEL
ALARM ON INSTRUMENT INOPERATIVE
ALARM ON COMPARATOR
RBM CHANNEL A
UPSCALE ALARM ON LEVEL
ALARM ON INSTRUMENT INOPERATIVE
REF 2, D-9
REF 2, D-10
CONTROL ROD WITHDRAWAL PERMITTED BY THIS FUNCTION IF SIGNAL IS PRESENT AT THIS POINT
CONTROL ROD WITHDRAWAL PERMITTED BY THESE FUNCTIONS IF SIGNAL PRESENT AT THIS POINT
REF 2, E-8
FIGURE 2
NEUTRON MONITORING SYSTEM TRIP
REACTOR PROTECTION SYSTEM TRIP SYSTEM A
(TYPICAL FOR TRIP SYSTEM B ~)
REF 4 SH 2, E7
IRM CHANNELS A,E (B,F)
UPSCALE TRIP ON LEVEL
TRIP ON INSTRUMENT INOPERATIVE
PERMISSIVE WHEN REACTOR MODE SWITCH IS IN "RUN" POSITION
APRM CHANNELS A,E (B,F) C,E (D,F)
UPSCALE TRIP ON LEVEL
TRIP ON INSTRUMENT INOPERATIVE
UPSCALE THERMAL TRIP
REACTOR NEUTRON MONITORING SYS A1 ~ C1523 B1 ~ C1580
A1 (B1)
A2 (B2)
REF 4 SH 2, B11

AMENDMENT NO. 10
July 1980



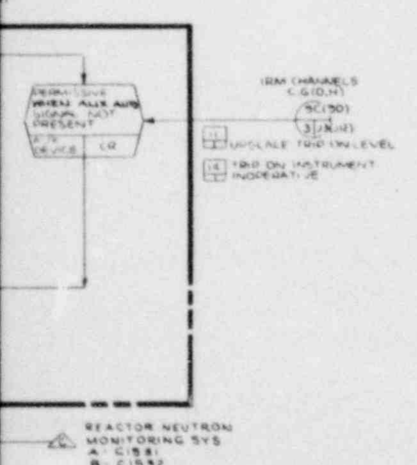
NOTES:
1. INPUTS TO COMPUTER ARE ISOLATED CLOSE TO ALARM CONTACTS.
2. SEE SHEET 7.
3. SEE SHEET 7.
4. THE ENTIRE NEUTRON MONITORING SYSTEM IS A FULLY AUTOMATIC SYSTEM EXCEPT FOR MANUAL OPERATED SWITCHES.
5. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY C51 UNLESS OTHERWISE NOTED.
6. CHANNELS A,C,E & G ARE FOR TRIP SYSTEM A. CHANNELS B,D,F & H ARE FOR TRIP SYSTEM B.

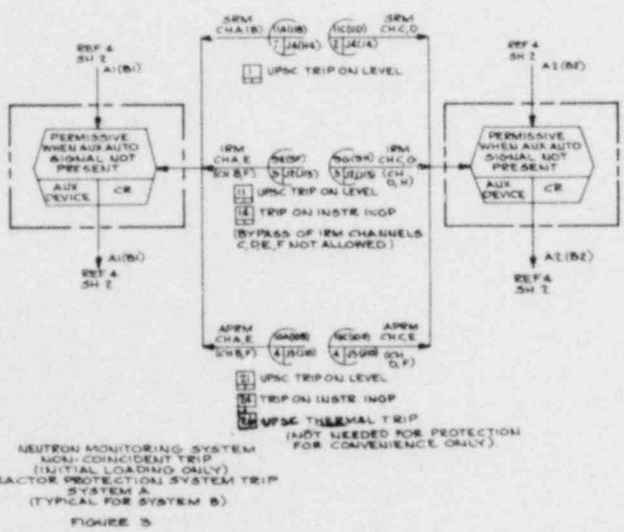
LEGEND
IRM — INTERMEDIATE RANGE MONITOR
RBM — ROD BLOCK MONITOR
APRM — AVERAGE POWER RANGE MONITOR
SRM — SOURCE RANGE MONITOR
LPRM — LOCAL POWER RANGE MONITOR
TIP — TRAVERSING IN CORE PROBE
MOC — MULTIPLE OUTPUT CONTROLLER

REFERENCE DOCUMENTS:

NOTE: SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NUMBERS

	MPL ITEM NO
1 NEUTRON MONITORING SYSTEM IED	C51-1010
2 CONTROL ROD DRIVE HYD SYS FCD	C11 & C12-1030
3 NUCLEAR BOILER SYS FCD	B21-1030
4 REACTOR PROTECTION SYS IED	C71, C72-1010
5 PROCESS COMPUTER SYSTEM INPUT/OUTPUT REQUIREMENTS DESIGN SPEC	C91 & C92-4010
6 LOGIC SYMBOLS	A41-1010
7 NEUTRON MONITORING SYS ARRANGEMENT	C51-2010

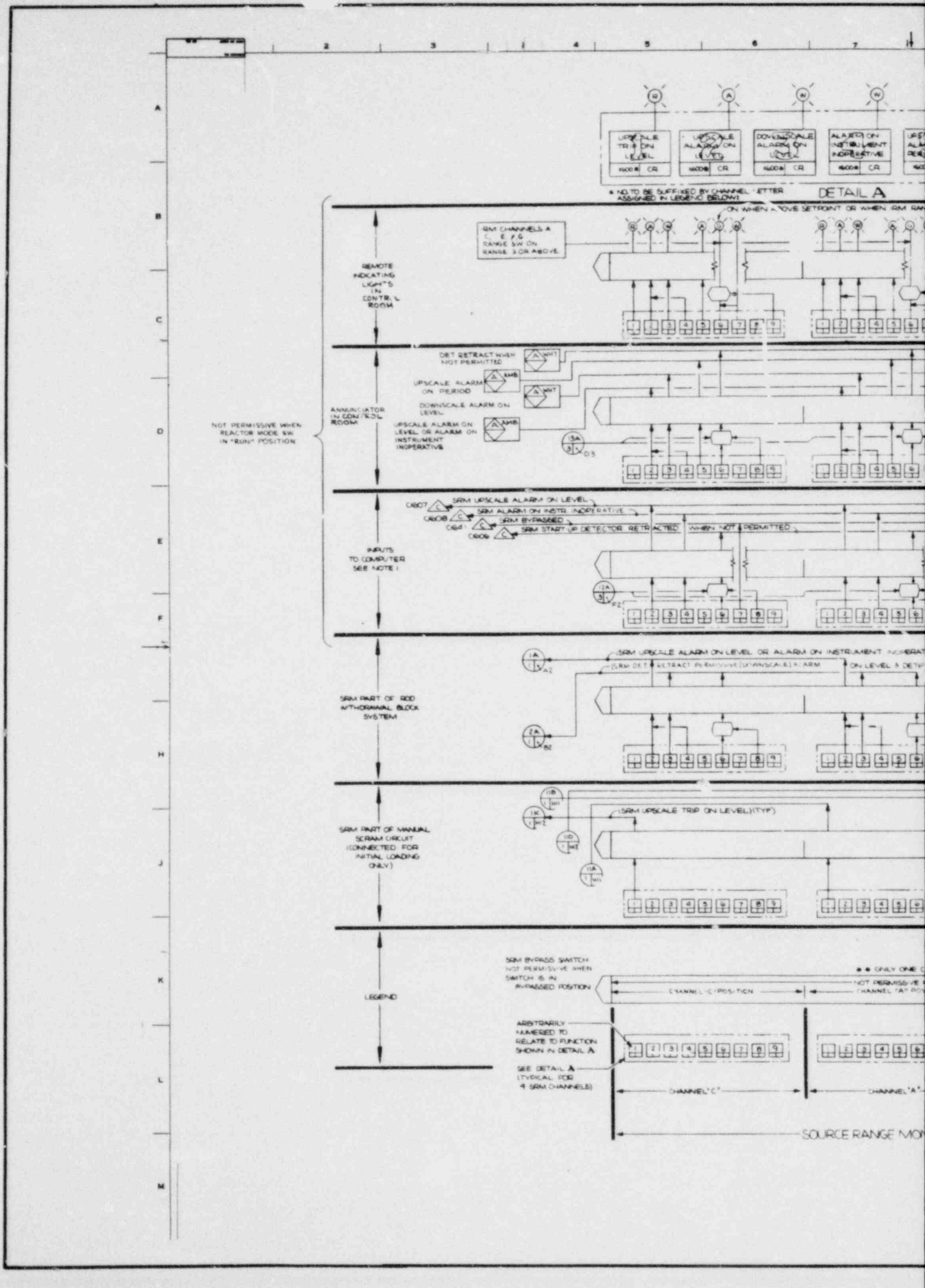




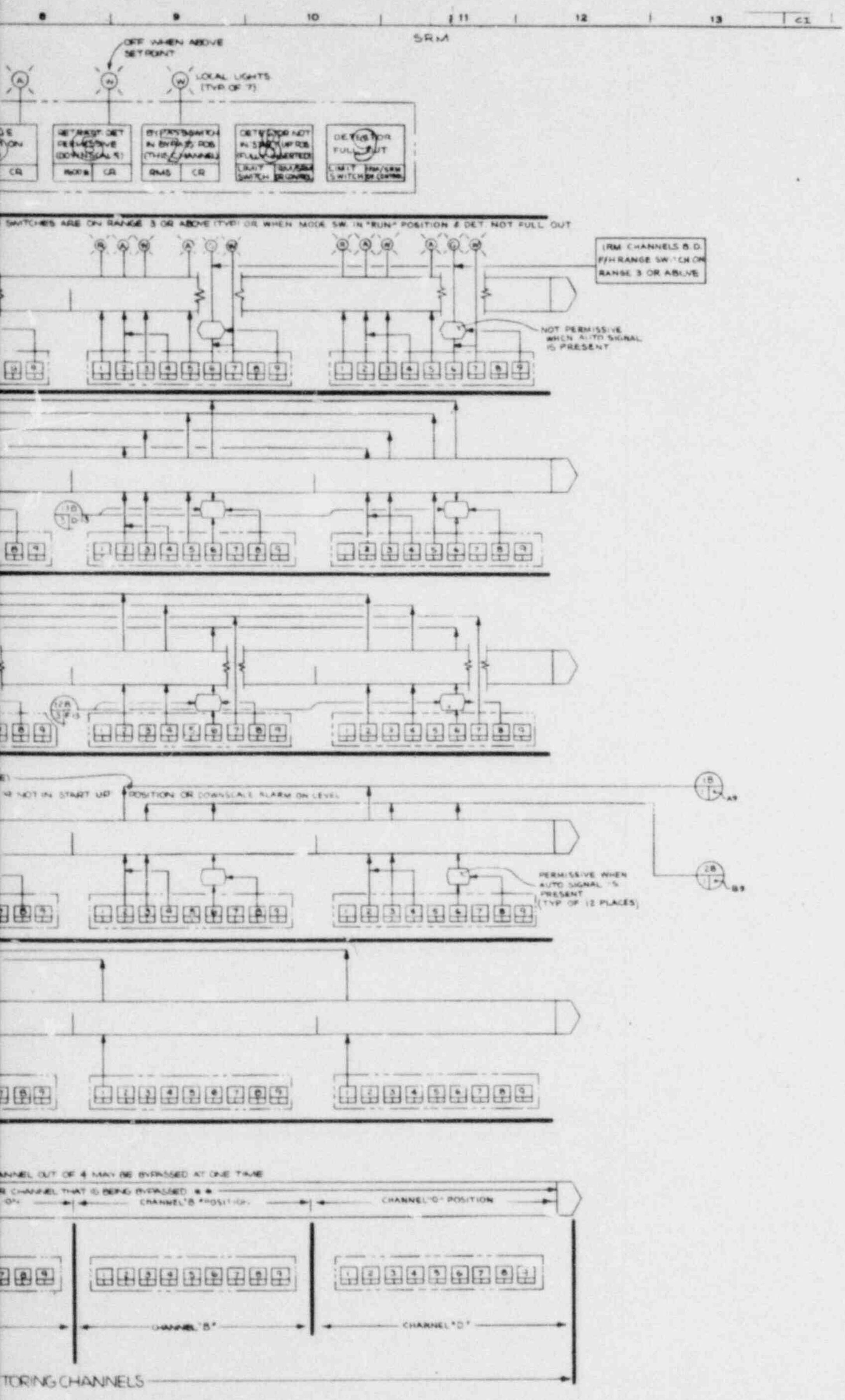
NEUTRON MONITORING SYSTEM
NON-COINCIDENT TRIP
(INITIAL LOADING ONLY)
REACTOR PROTECTION SYSTEM TRIP
SYSTEM A
(TYPICAL FOR SYSTEM B)

FIGURE 3

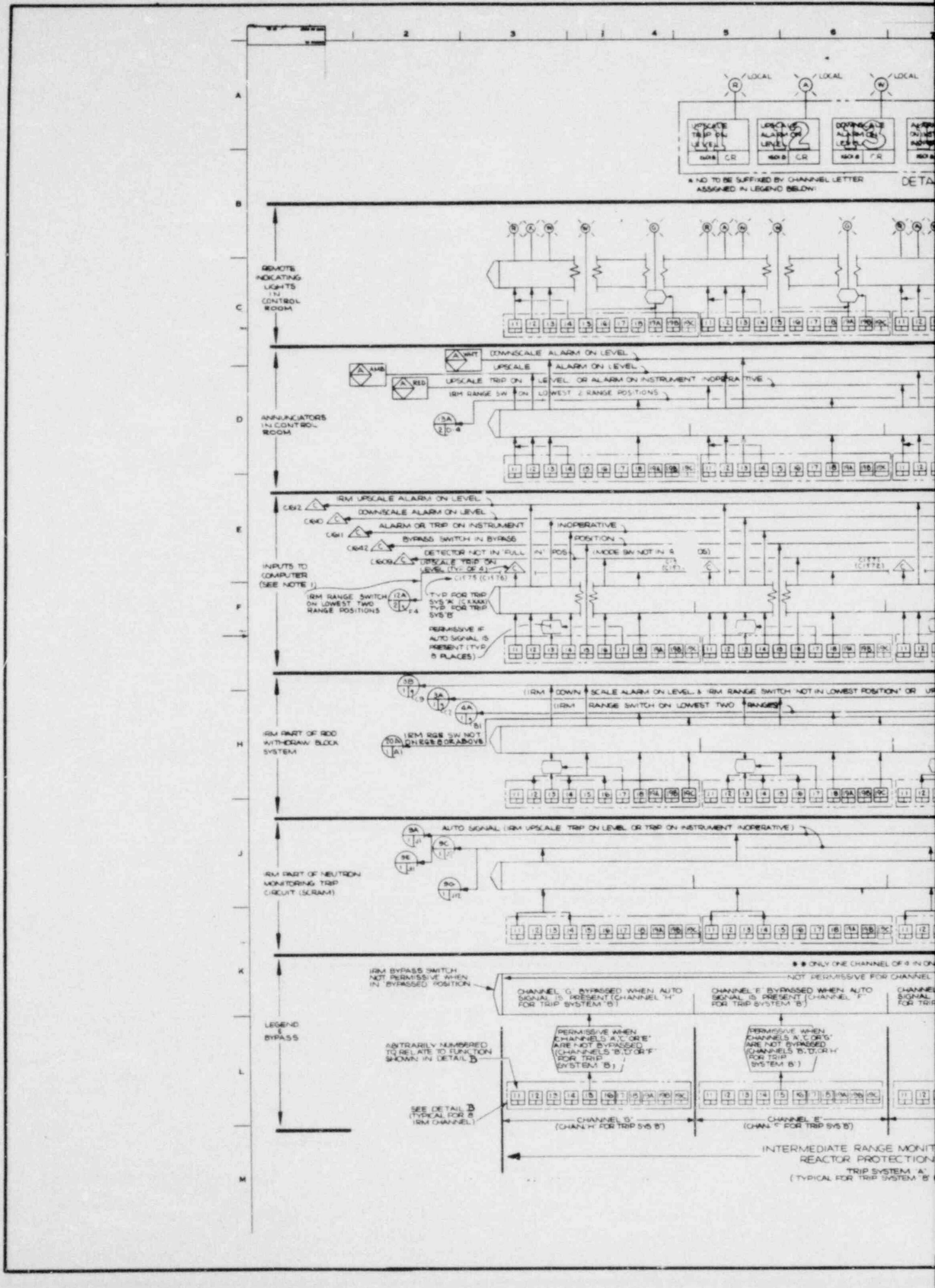
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6a


UPSCALE TRIP ON LEVEL
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ALARM ON INSTRUMENT INOPERATIVE
* NO. TO BE SUFFIXED BY CHANNEL LETTER ASSIGNED IN LEGEND BELOW
DETAIL A
ON WHEN ABOVE SETPOINT OR WHEN IRM RANGE
IRM CHANNELS A, C, E & G RANGE SW ON RANGE 3 OR ABOVE
REMOTE INDICATING LIGHTS IN CONTROL ROOM
DET RETRACT WHEN NOT PERMITTED
UPSCALE ALARM ON PERIOD
DOWNSCALE ALARM ON LEVEL
UPSCALE ALARM ON LEVEL OR ALARM ON INSTRUMENT INOPERATIVE
ANNUNCIATOR IN CONTROL ROOM
NOT PERMISSIVE WHEN REACTOR MODE SW IN "RUN" POSITION
SRM UPSCALE ALARM ON LEVEL
SRM ALARM ON INSTR. INOPERATIVE
SRM BYPASSED
SRM START UP DETECTOR RETRACTED WHEN NOT PERMITTED
INPUTS TO COMPUTER SEE NOTE 1
(SRM UPSCALE ALARM ON LEVEL OR ALARM ON INSTRUMENT INOPERAT
(SRM DET RETRACT PERMISSIVE)(DOWNSCALE) ALARM
ON LEVEL & DETE
SRM PART OF ROD WITHDRAWAL BLOCK SYSTEM
(SRM UPSCALE TRIP ON LEVEL)(TYP)
SRM PART OF MANUAL SCRAM CIRCUIT (CONNECTED FOR INITIAL LOADING ONLY)
SRM BYPASS SWITCH NOT PERMISSIVE WHEN SWITCH IS IN BYPASSED POSITION
CHANNEL 'C' POSITION
** ONLY ONE
NOT PERMISSIVE
CHANNEL "A" POS
LEGEND
ARBITRARILY NUMBERED TO RELATE TO FUNCTION SHOWN IN DETAIL A
SEE DETAIL A (TYPICAL FOR 4 SRM CHANNELS)
CHANNEL 'C'
CHANNEL 'A'
SOURCE RANGE MON

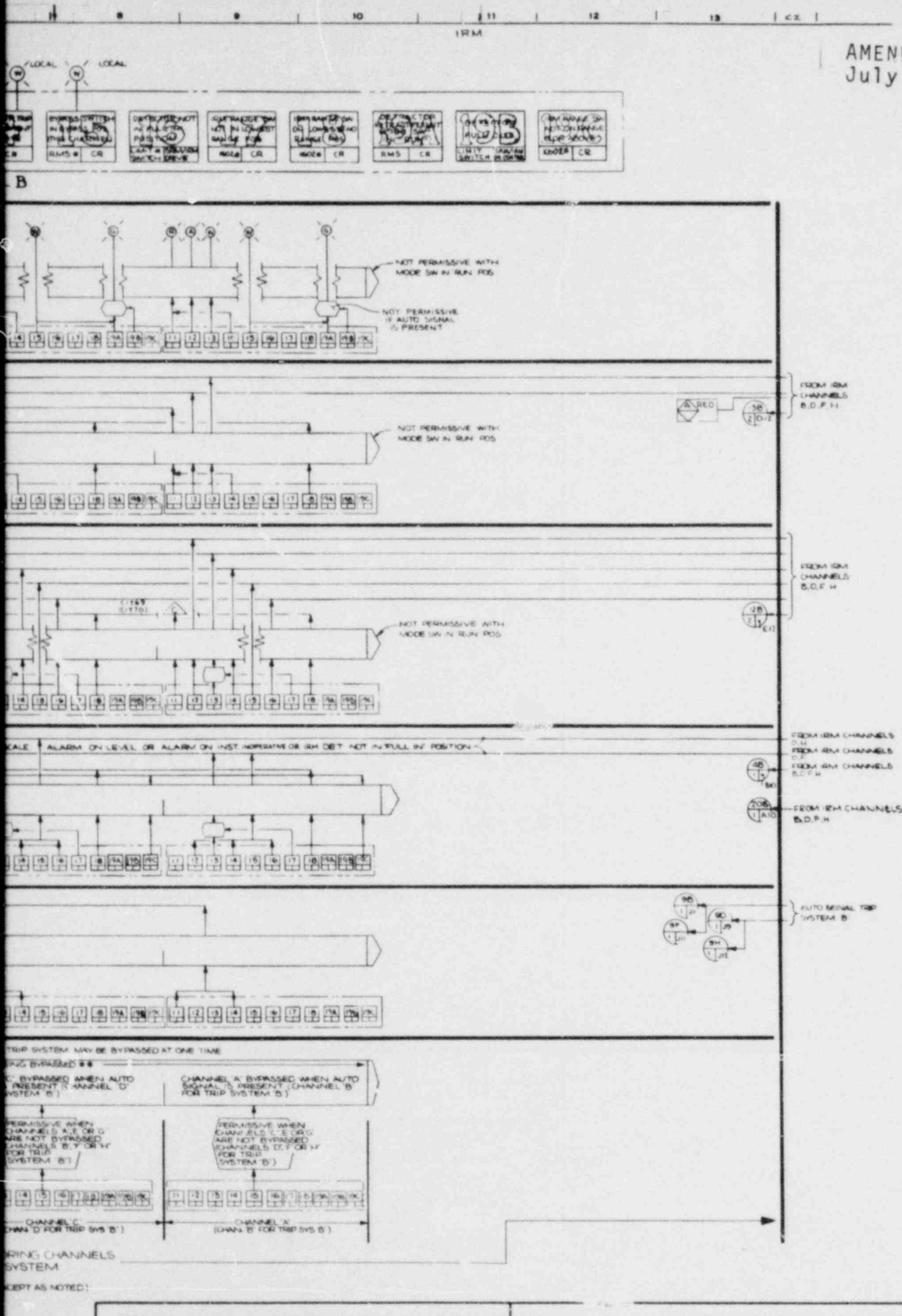
AMENDMENT NO. 10
July 1980



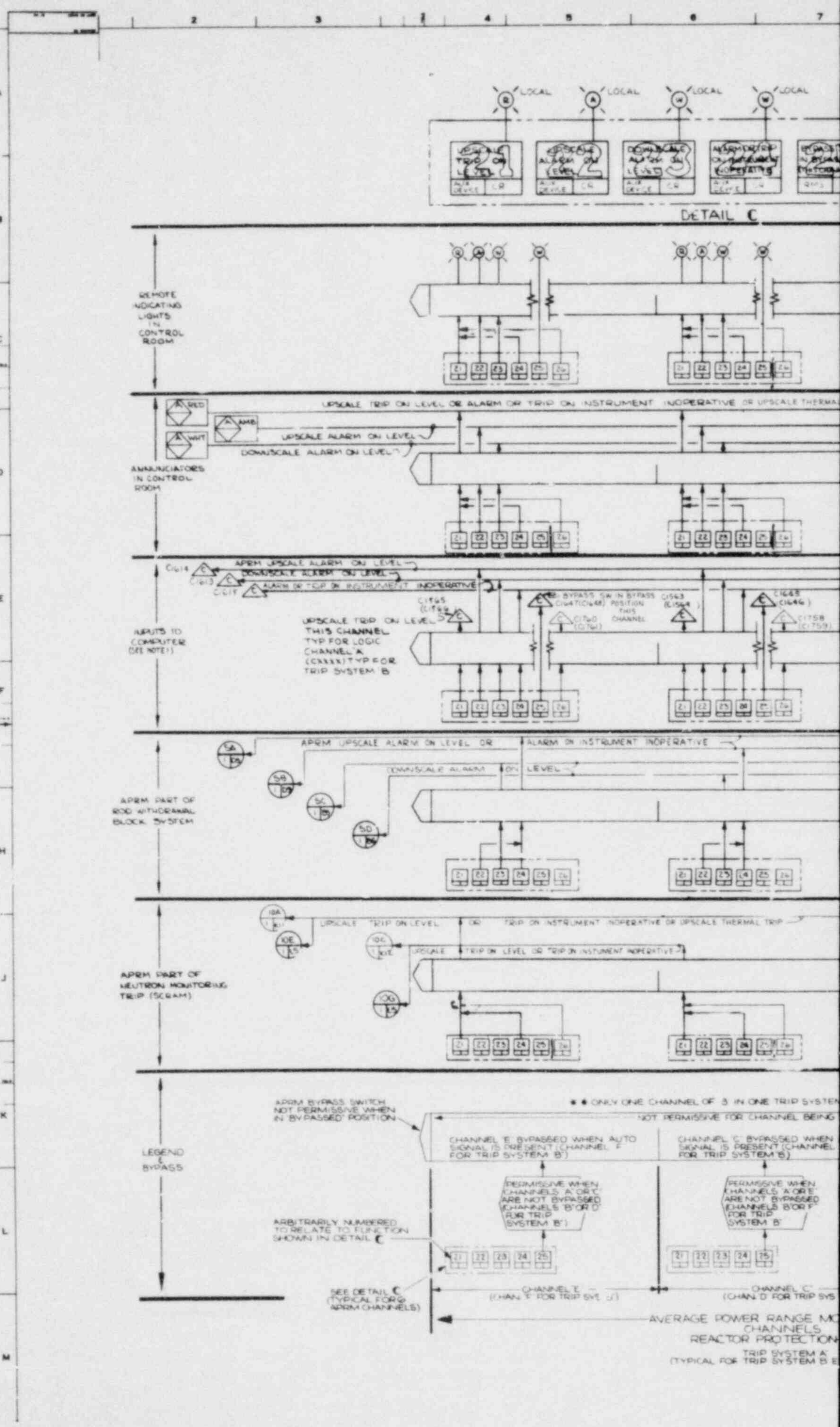
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6b


LOCAL
UPSCALE TRIP ON LEVEL
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
CR
* NO TO BE SUFFIXED BY CHANNEL LETTER ASSIGNED IN LEGEND BELOW
DETA
REMOTE INDICATING LIGHTS IN CONTROL ROOM
DOWNSCALE ALARM ON LEVEL
UPSCALE ALARM ON LEVEL
UPSCALE TRIP ON LEVEL OR ALARM ON INSTRUMENT INOPERATIVE
IRM RANGE SW ON LOWEST 2 RANGE POSITIONS
ANNUNCIATORS IN CONTROL ROOM
IRM UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ALARM OR TRIP ON INSTRUMENT INOPERATIVE
BYPASS SWITCH IN BYPASS POSITION
DETECTOR NOT IN 'FULL IN' POS (MODE SW NOT IN R OS)
UPSCALE TRIP ON LEVEL (TYP OF 4)
INPUTS TO COMPUTER (SEE NOTE 1)
IRM RANGE SWITCH ON LOWEST TWO RANGE POSITIONS
PERMISSIVE IF AUTO SIGNAL IS PRESENT (TYP 8 PLACES)
IRM DOWN SCALE ALARM ON LEVEL & 'IRM RANGE SWITCH NOT IN LOWEST POSITION' OR
IRM RANGE SWITCH ON LOWEST TWO RANGES
IRM RGE SW NOT ON RGE 8 OR ABOVE
IRM PART OF ROD WITHDRAW BLOCK SYSTEM
AUTO SIGNAL (IRM UPSCALE TRIP ON LEVEL OR TRIP ON INSTRUMENT INOPERATIVE)
IRM PART OF NEUTRON MONITORING TRIP CIRCUIT (SCRAM)
** ONLY ONE CHANNEL OF 4 IN ON
NOT PERMISSIVE FOR CHANNEL
IRM BYPASS SWITCH NOT PERMISSIVE WHEN IN 'BYPASSED' POSITION
CHANNEL 'G' BYPASSED WHEN AUTO SIGNAL IS PRESENT (CHANNEL 'H' FOR TRIP SYSTEM 'B')
CHANNEL 'E' BYPASSED WHEN AUTO SIGNAL IS PRESENT (CHANNEL 'F' FOR TRIP SYSTEM 'B')
LEGEND & BYPASS
PERMISSIVE WHEN CHANNELS A,'C' OR 'E' ARE NOT BYPASSED (CHANNELS 'B','D' OR 'F' FOR TRIP SYSTEM 'B')
PERMISSIVE WHEN CHANNELS 'A','C' OR 'G' ARE NOT BYPASSED (CHANNELS 'B','D' OR 'H' FOR TRIP SYSTEM 'B')
ARBITRARILY NUMBERED TO RELATE TO FUNCTION SHOWN IN DETAIL B
SEE DETAIL B (TYPICAL FOR 8 IRM CHANNEL)
CHANNEL 'G' (CHAN 'H' FOR TRIP SYS 'B')
CHANNEL 'E' (CHAN 'F' FOR TRIP SYS 'B')
INTERMEDIATE RANGE MONIT
REACTOR PROTECTION
TRIP SYSTEM 'A'
(TYPICAL FOR TRIP SYSTEM 'B'

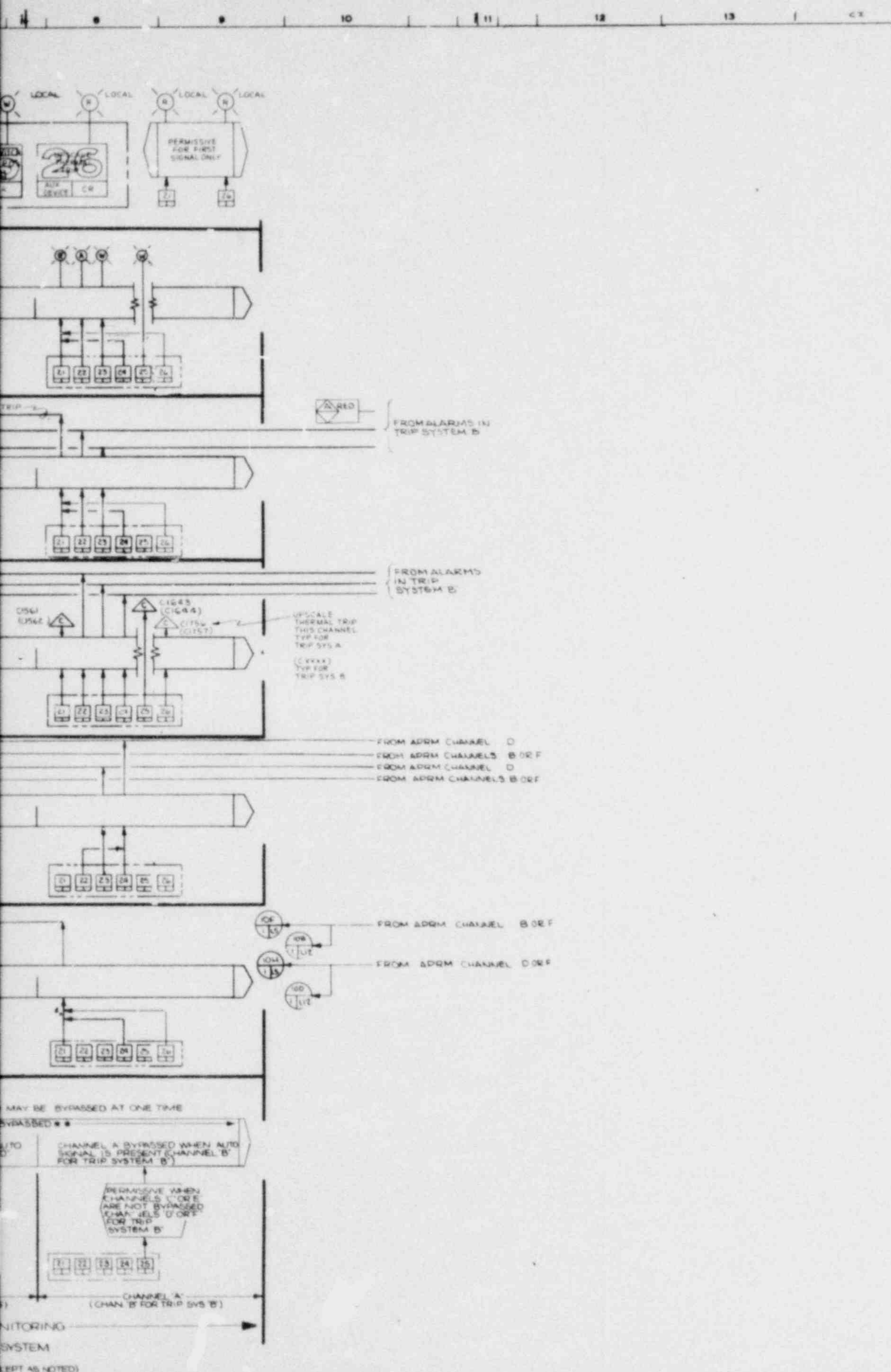
AMENDMENT NO. 10
July 1980



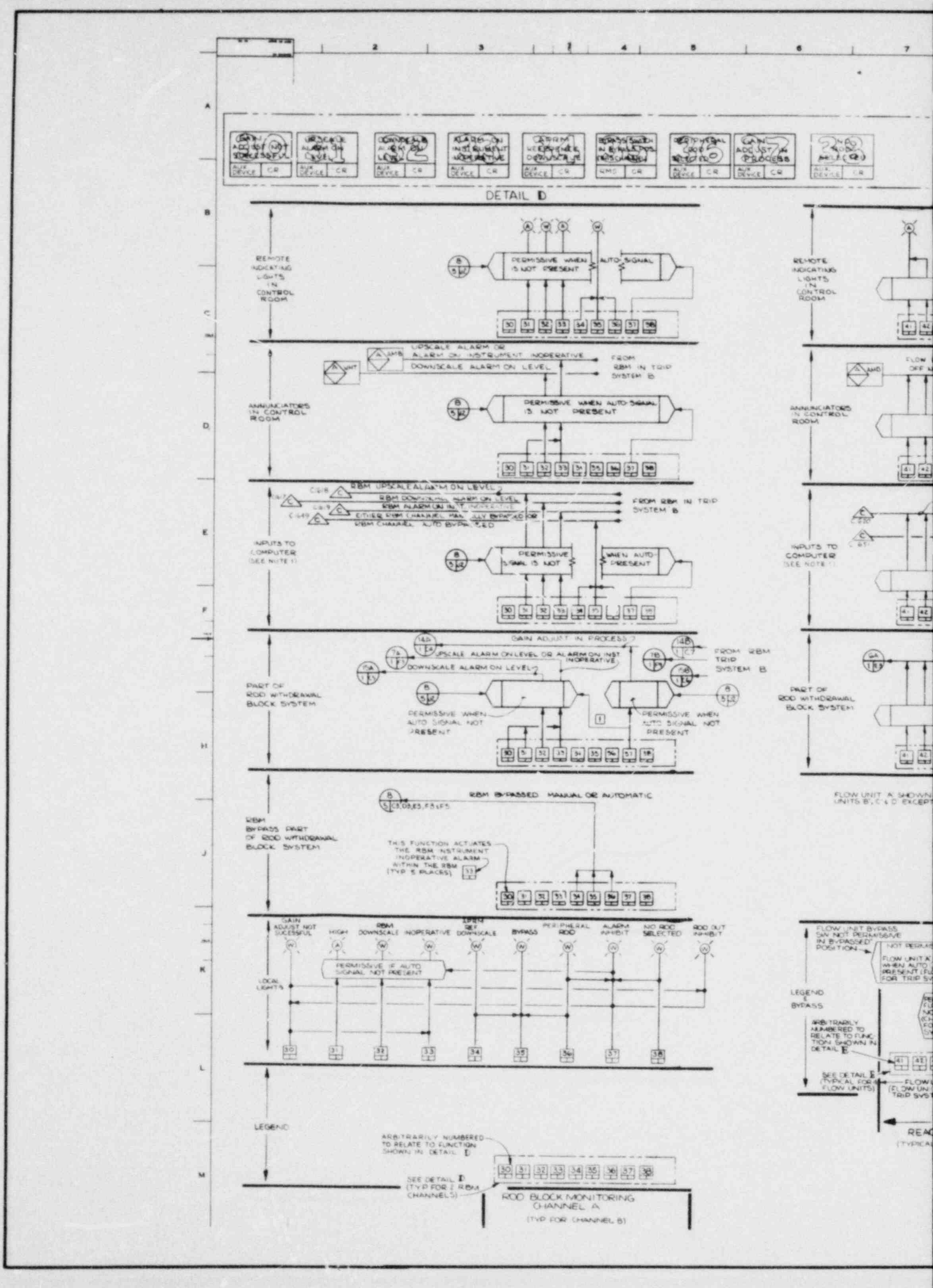
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6c


DETAIL C
REMOTE INDICATING LIGHTS IN CONTROL ROOM
UPSCALE TRIP ON LEVEL OR ALARM OR TRIP ON INSTRUMENT INOPERATIVE OR UPSCALE THERMAL
UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ANNUNCIATORS IN CONTROL ROOM
APRM UPSCALE ALARM ON LEVEL
DOWNSCALE ALARM ON LEVEL
ALARM OR TRIP ON INSTRUMENT INOPERATIVE
UPSCALE TRIP ON LEVEL THIS CHANNEL TYP FOR LOGIC CHANNEL 'A' (CXXXX) TYP FOR TRIP SYSTEM B
INPUTS TO COMPUTER (SEE NOTE 1)
APRM UPSCALE ALARM ON LEVEL OR ALARM ON INSTRUMENT INOPERATIVE
DOWNSCALE ALARM ON LEVEL
APRM PART OF ROD WITHDRAWAL BLOCK SYSTEM
UPSCALE TRIP ON LEVEL OR TRIP ON INSTRUMENT INOPERATIVE OR UPSCALE THERMAL TRIP
UPSCALE TRIP ON LEVEL OR TRIP ON INSTUMENT INOPERATIVE
APRM PART OF NEUTRON MONITORING TRIP (SCRAM)
LEGEND & BYPASS
APRM BYPASS SWITCH NOT PERMISSIVE WHEN IN 'BYPASSED' POSITION
ONLY ONE CHANNEL OF 3 IN ONE TRIP SYSTEM
NOT PERMISSIVE FOR CHANNEL BEING
CHANNEL 'E' BYPASSED WHEN AUTO SIGNAL IS PRESENT (CHANNEL 'F' FOR TRIP SYSTEM 'B')
CHANNEL 'C' BYPASSED WHEN SIGNAL IS PRESENT (CHANNEL FOR TRIP SYSTEM 'B')
PERMISSIVE WHEN CHANNELS 'A' OR 'C' ARE NOT BYPASSED (CHANNELS 'B' OR 'D' FOR TRIP SYSTEM 'B')
PERMISSIVE WHEN CHANNELS 'A' OR 'E' ARE NOT BYPASSED (CHANNELS 'B' OR F FOR TRIP SYSTEM 'B')
ARBITRARILY NUMBERED TO RELATE TO FUNCTION SHOWN IN DETAIL C
SEE DETAIL C (TYPICAL FOR 6 APRM CHANNELS)
CHANNEL 'E' (CHAN. 'F' FOR TRIP SYS. 'B')
CHANNEL 'C' (CHAN. 'D' FOR TRIP SYS
AVERAGE POWER RANGE MO
CHANNELS
REACTOR PROTECTION
TRIP SYSTEM 'A'
(TYPICAL FOR TRIP SYSTEM 'B'

AMENDMENT NO. 10
July 1980

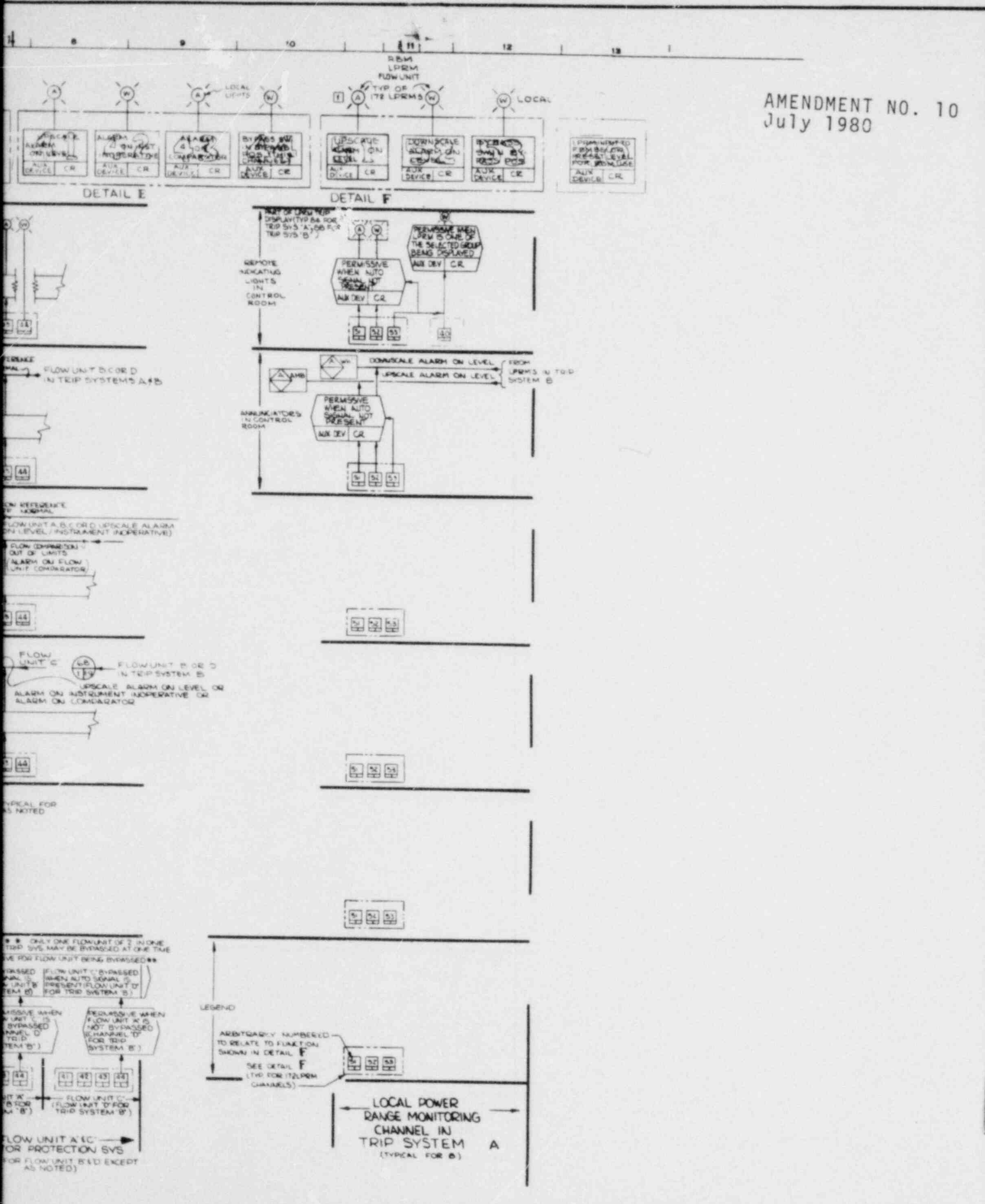


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6d

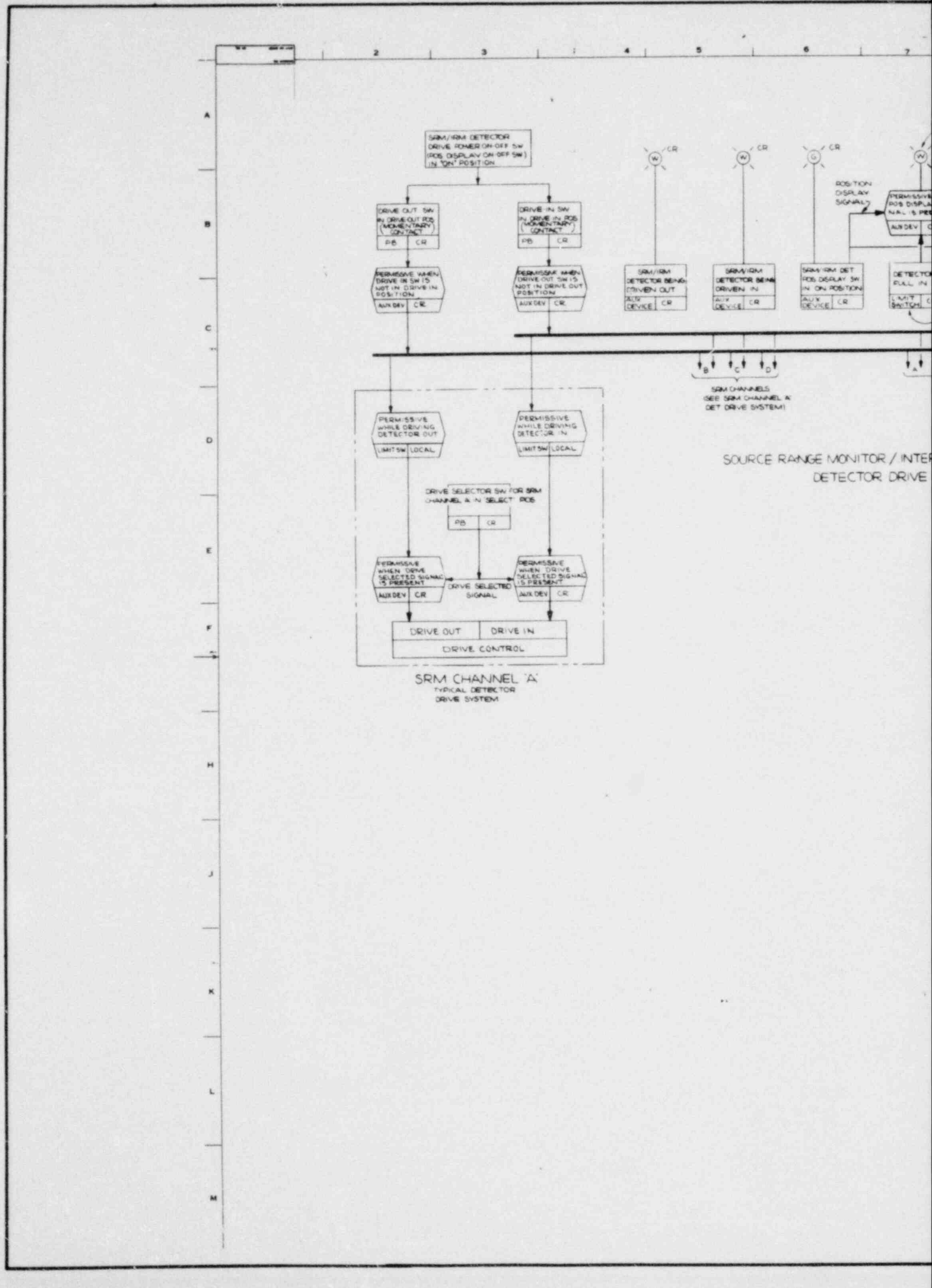


DETAIL D
REMOTE INDICATING LIGHTS IN CONTROL ROOM
PERMISSIVE WHEN AUTO SIGNAL IS NOT PRESENT
UPSCALE ALARM OR ALARM ON INSTRUMENT INOPERATIVE
DOWNSCALE ALARM ON LEVEL
FROM RBM IN TRIP SYSTEM B
ANNUNCIATORS IN CONTROL ROOM
RBM UPSCALE ALARM ON LEVEL
RBM DOWNSCALE ALARM ON LEVEL
EITHER RBM CHANNEL MANUALLY BYPASSED OR RBM CHANNEL AUTO BYPASSED
INPUTS TO COMPUTER (SEE NOTE 1)
GAIN ADJUST IN PROCESS
UPSCALE ALARM ON LEVEL OR ALARM ON INST. INOPERATIVE
DOWNSCALE ALARM ON LEVEL
PART OF ROD WITHDRAWAL BLOCK SYSTEM
RBM BYPASSED MANUAL OR AUTOMATIC
RBM BYPASS PART OF ROD WITHDRAWAL BLOCK SYSTEM
THIS FUNCTION ACTUATES THE RBM INSTRUMENT INOPERATIVE ALARM WITHIN THE RBM (TYP 5 PLACES)
GAIN ADJUST NOT SUCESSFUL
HIGH
RBM DOWNSCALE
INOPERATIVE
REF DOWNSCALE
BYPASS
PERIPHERAL ROD
ALARM INHIBIT
NO ROD SELECTED
ROD OUT INHIBIT
PERMISSIVE IF AUTO SIGNAL NOT PRESENT
LOCAL LIGHTS
LEGEND
ARBITRARILY NUMBERED TO RELATE TO FUNCTION SHOWN IN DETAIL D
SEE DETAIL D (TYP FOR 2 RBM CHANNELS)
ROD BLOCK MONITORING CHANNEL A
(TYP FOR CHANNEL B)
FLOW UNIT A SHOWN UNITS B, C & D EXCEPT
LEGEND & BYPASS
ARBITRARILY NUMBERED TO RELATE TO FUNCTION SHOWN IN DETAIL E
SEE DETAIL E (TYPICAL FOR 4 FLOW UNITS)

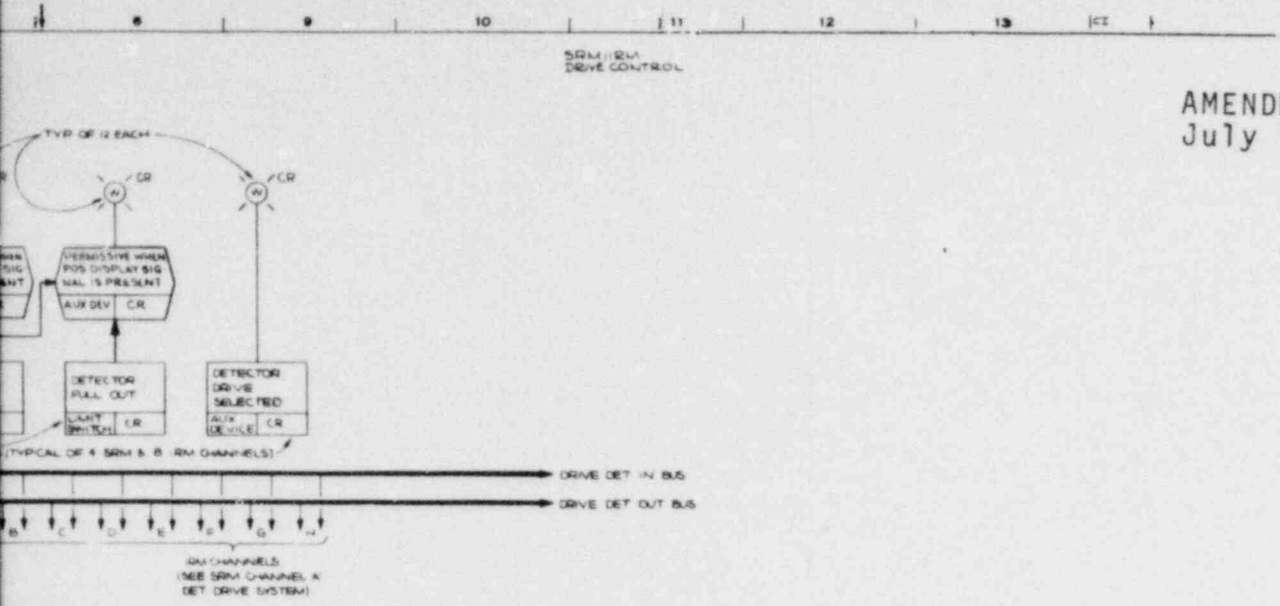
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONING CONTROL DIAGRAM	FIGURE 7.6-6e

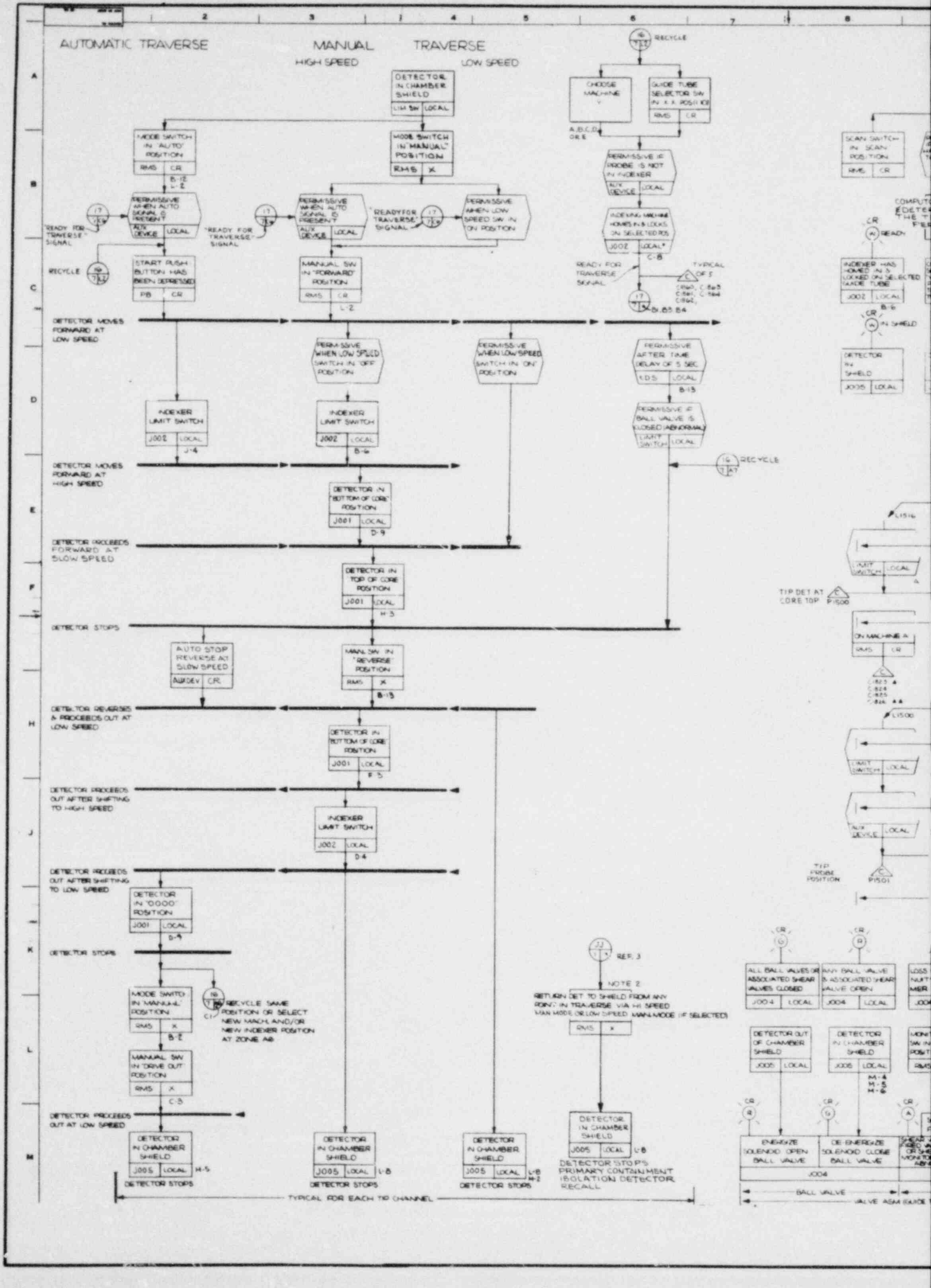

SRM/IRM DETECTOR DRIVE POWER ON-OFF SW (POS DISPLAY ON-OFF SW) IN "ON" POSITION
DRIVE OUT SW IN DRIVE OUT POS (MOMENTARY CONTACT)
PB
CR
DRIVE IN SW IN DRIVE IN POS (MOMENTARY CONTACT)
PERMISSIVE WHEN DRIVE IN SW IS NOT IN DRIVE IN POSITION
PERMISSIVE WHEN DRIVE OUT SW IS NOT IN DRIVE OUT POSITION
AUX DEV
SRM/IRM DETECTOR BEING DRIVEN OUT
SRM/IRM DETECTOR BEING DRIVEN IN
SRM/IRM DET POS DISPLAY SW IN ON POSITION
AUX DEVICE
POSITION DISPLAY SIGNAL
LIMIT SWITCH
SRM CHANNELS (SEE SRM CHANNEL 'A' DET DRIVE SYSTEM)
PERMISSIVE WHILE DRIVING DETECTOR OUT
PERMISSIVE WHILE DRIVING DETECTOR IN
LIMIT SW
LOCAL
DRIVE SELECTOR SW FOR SRM CHANNEL A IN 'SELECT' POS
PERMISSIVE WHEN 'DRIVE SELECTED SIGNAL' IS PRESENT
DRIVE SELECTED SIGNAL
DRIVE OUT
DRIVE IN
DRIVE CONTROL
SRM CHANNEL 'A'
TYPICAL DETECTOR DRIVE SYSTEM
SOURCE RANGE MONITOR / INTER
DETECTOR DRIVE

AMENDMENT NO. 10
July 1980

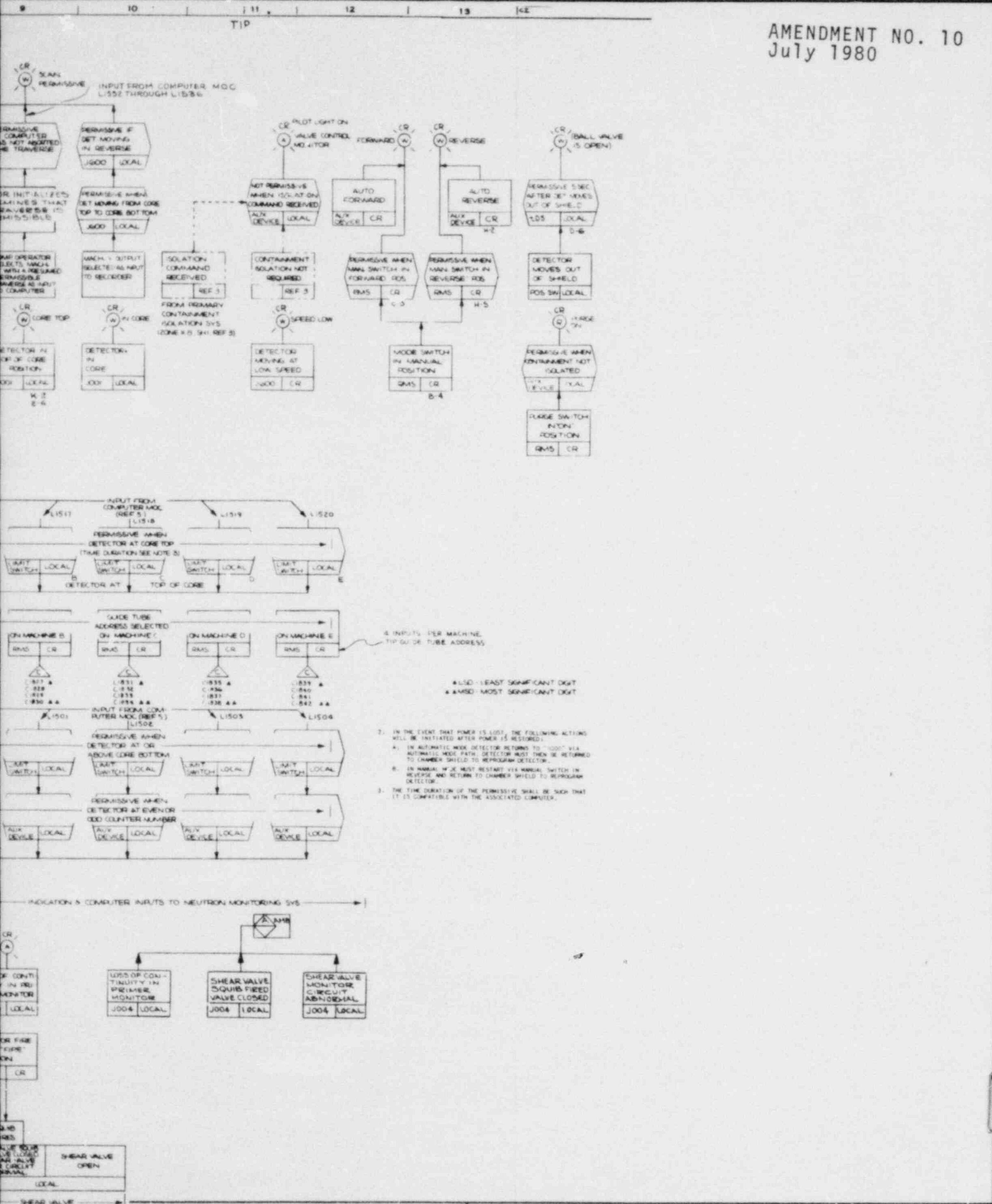


MEDIATE RANGE MONITOR
SYSTEM

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6f


AUTOMATIC TRAVERSE
MANUAL TRAVERSE
HIGH SPEED
LOW SPEED
RECYCLE
DETECTOR IN CHAMBER SHIELD
LIM SW LOCAL
CHOOSE MACHINE
GUIDE TUBE SELECTOR SW
RMS CR
A,B,C,D, OR E
MODE SWITCH IN "AUTO" POSITION
MODE SWITCH IN "MANUAL" POSITION
RMS X
PERMISSIVE IF PROBE IS NOT IN INDEXER
AUX DEVICE LOCAL
SCAN SWITCH IN SCAN POSITION
PERMISSIVE WHEN AUTO SIGNAL IS PRESENT
"READY FOR TRAVERSE" SIGNAL
PERMISSIVE WHEN LOW SPEED SW IN ON POSITION
INDEXING MACHINE HOMES IN & LOCKS ON SELECTED POS
J002 LOCAL
READY FOR TRAVERSE SIGNAL
TYPICAL OF 5
START PUSH BUTTON HAS BEEN DEPRESSED
PB CR
MANUAL SW IN "FORWARD" POSITION
INDEXER HAS HOMED IN & LOCKED ON SELECTED GUIDE TUBE
DETECTOR MOVES FORWARD AT LOW SPEED
PERMISSIVE WHEN LOW SPEED SWITCH IN "OFF" POSITION
PERMISSIVE WHEN LOW SPEED SWITCH IN "ON" POSITION
PERMISSIVE AFTER TIME DELAY OF 5 SEC
DETECTOR IN SHIELD
J005 LOCAL
INDEXER LIMIT SWITCH
PERMISSIVE IF BALL VALVE IS CLOSED (ABNORMAL)
LIMIT SWITCH LOCAL
DETECTOR MOVES FORWARD AT HIGH SPEED
DETECTOR IN "BOTTOM OF CORE" POSITION
J001 LOCAL
DETECTOR PROCEEDS FORWARD AT SLOW SPEED
DETECTOR IN TOP OF CORE POSITION
TIP DET AT CORE TOP
DETECTOR STOPS
AUTO STOP REVERSE AT SLOW SPEED
AUX DEV CR
MAN. SW IN "REVERSE" POSITION
ON MACHINE A
DETECTOR REVERSES & PROCEEDS OUT AT LOW SPEED
DETECTOR IN BOTTOM OF CORE POSITION
DETECTOR PROCEEDS OUT AFTER SHIFTING TO HIGH SPEED
DETECTOR PROCEEDS OUT AFTER SHIFTING TO LOW SPEED
TIP PROBE POSITION
DETECTOR IN "0000" POSITION
DETECTOR STOPS
REF. 3
NOTE 2
MODE SWITCH IN "MANUAL" POSITION
RECYCLE SAME POSITION OR SELECT NEW MACH. AND/OR NEW INDEXER POSITION AT ZONE A8
RETURN DET TO SHIELD FROM ANY POINT IN TRAVERSE VIA HI SPEED MAN MODE OR LOW SPEED MAN MODE (IF SELECTED)
ALL BALL VALVES OR ASSOCIATED SHEAR VALVES CLOSED
ANY BALL VALVE & ASSOCIATED SHEAR VALVE OPEN
J004 LOCAL
MANUAL SW IN "DRIVE OUT" POSITION
DETECTOR OUT OF CHAMBER SHIELD
DETECTOR IN CHAMBER SHIELD
DETECTOR PROCEEDS OUT AT LOW SPEED
ENERGIZE SOLENOID OPEN BALL VALVE
DE-ENERGIZE SOLENOID CLOSE BALL VALVE
DETECTOR STOPS
DETECTOR STOPS PRIMARY CONTAINMENT ISOLATION DETECTOR RECALL
BALL VALVE
TYPICAL FOR EACH TIP CHANNEL

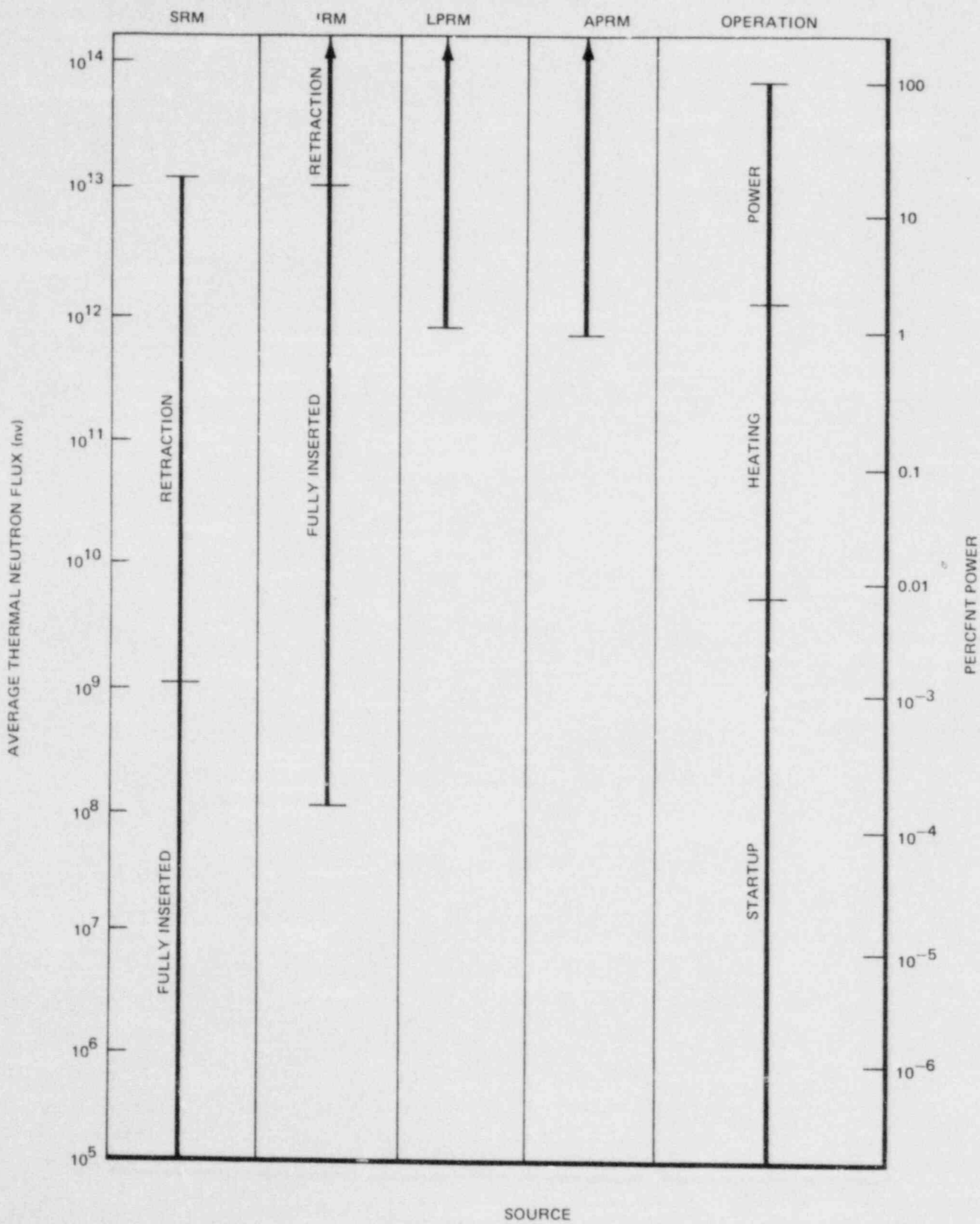
AMENDMENT NO. 10
July 1980



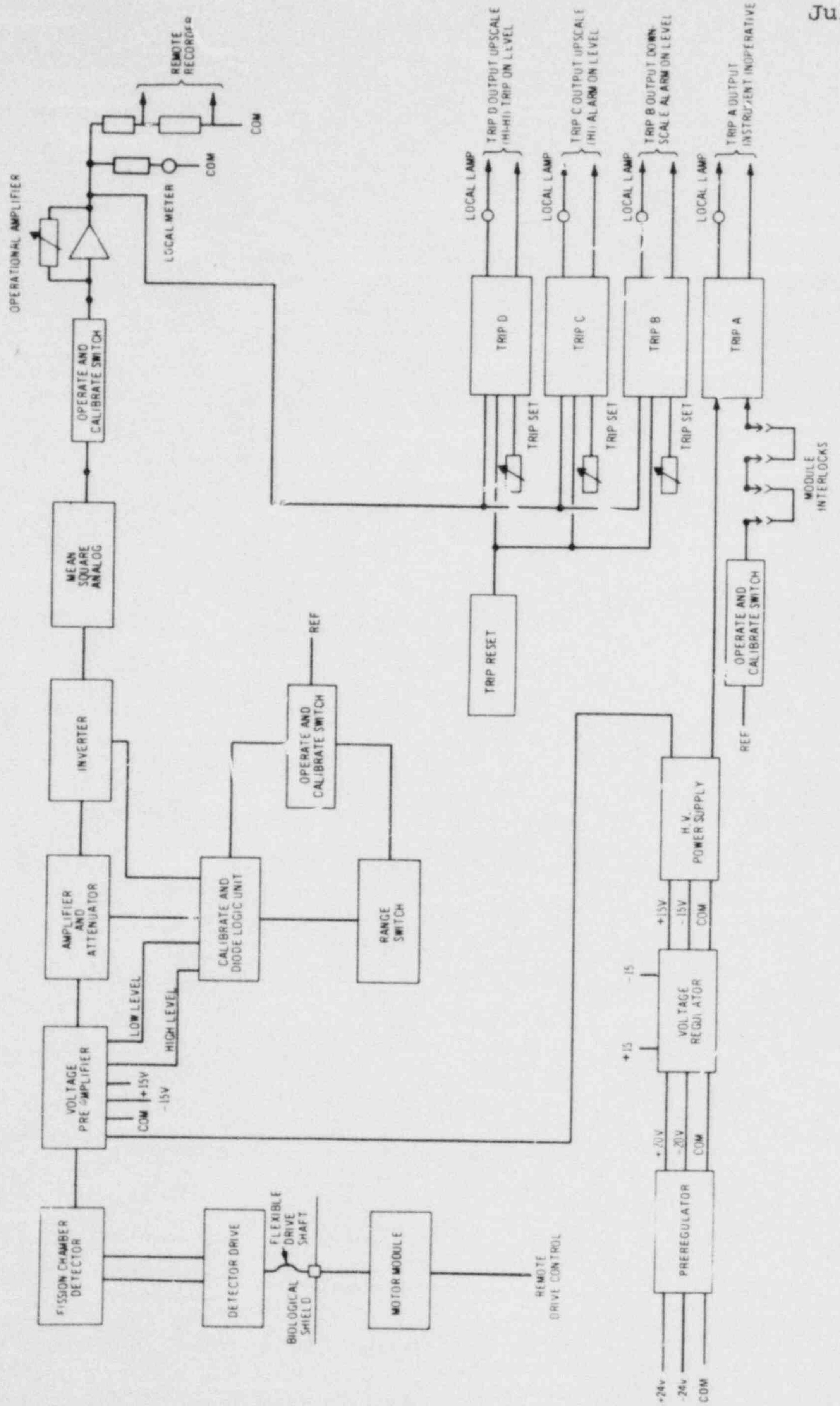
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM

FIGURE 7.6-6g

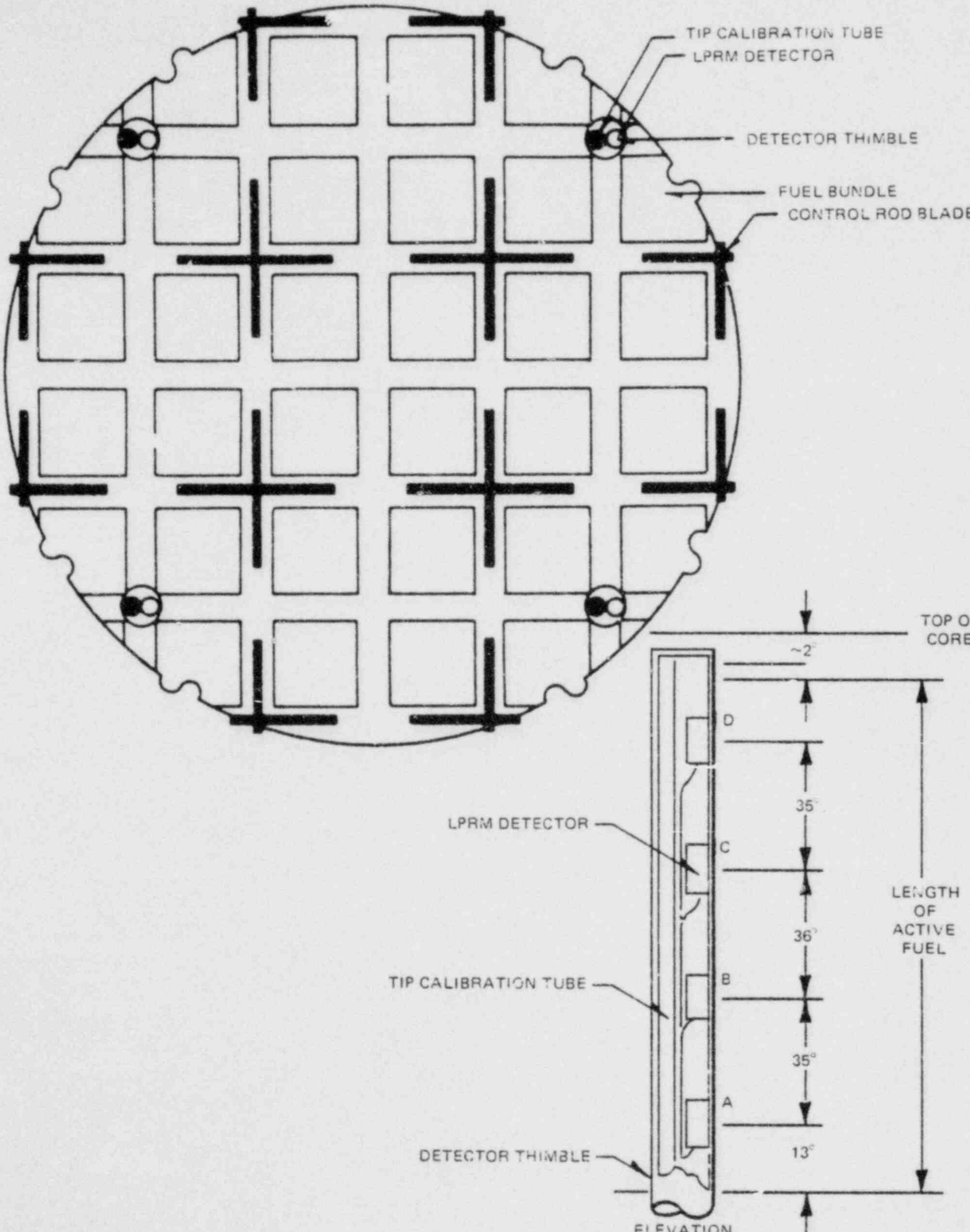


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RANGES OF NEUTRON MONITORING SYSTEM	FIGURE 7.6-7



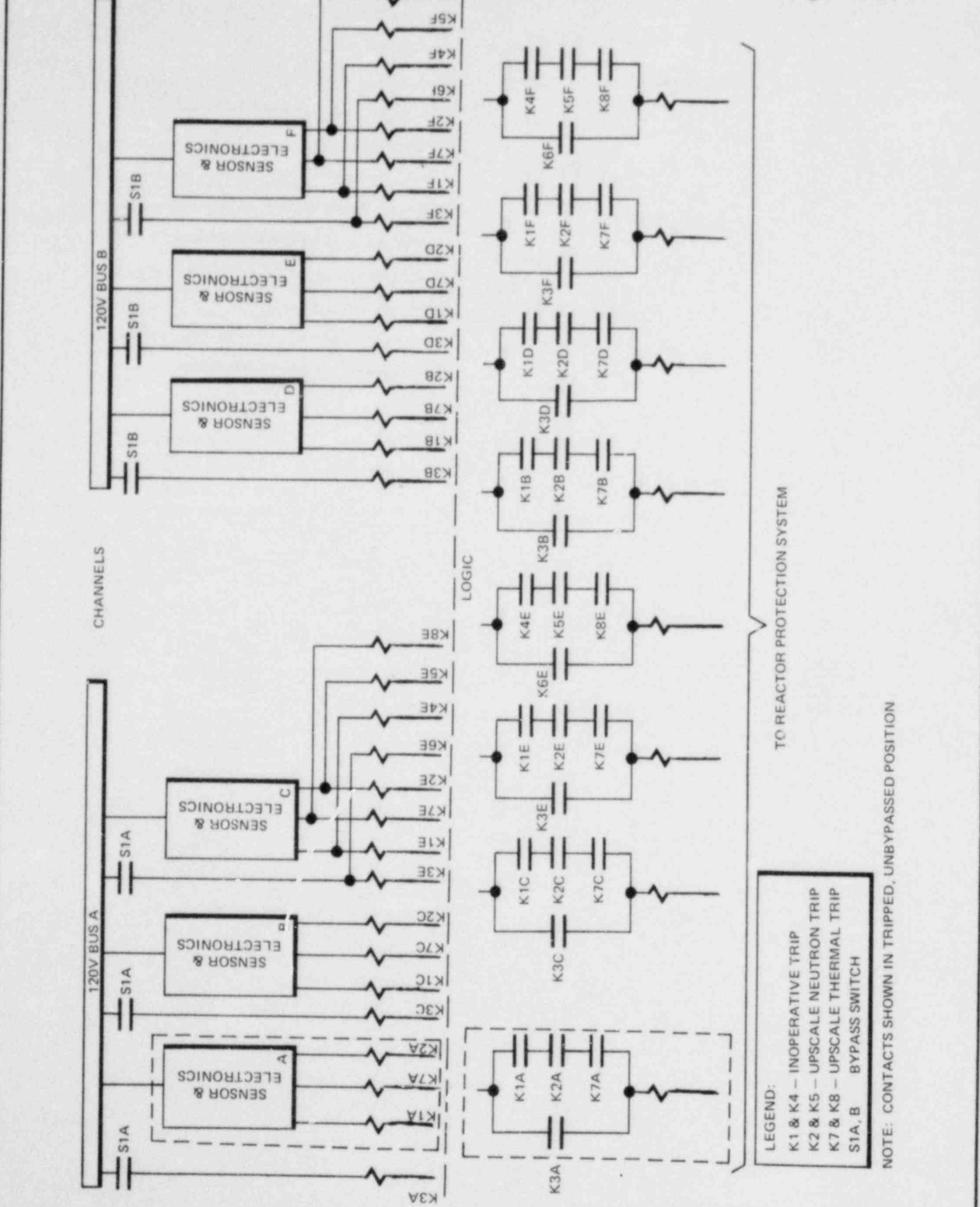
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FUNCTIONAL BLOCK DIAGRAM OF IRM CHANNEL.	FIGURE 7.6-8

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	POWER RANGE MONITOR DETECTOR ASSEMBLY LOCATION	FIGURE 7.6-9

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	APRM CIRCUT ARRANGEMENT FOR REACTOR PROTECTION SYSTEM INPUT.	FIGURE 7.6-10

Figure 7.6-11 later

7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

7.7.1 DESCRIPTION

Section 7.7 describes instrumentation and controls of major plant control systems whose functions are not essential for the safety of the plant. This section also describes instrumentation and controls, not essential for the safety of the plant, which are not discussed in any other FSAR section. The systems include:

1. Reactor Vessel Instrumentation
2. Reactor Manual Control System (RMCS)
3. Recirculation Flow Control System
4. Feedwater Control System
5. Pressure Regulator and Turbine - Generator System
6. Neutron Monitoring System (TIP, SRM, RBM)
7. Process Computer System and Rod Worth Minimizer Function (RWM)
8. Rod Sequence Control System (RSCS)
9. Loose Parts Detection System (LPDS)
10. Refueling Interlocks

Refer to Tables 7.7-1 and 7.7-2 for system design and supply responsibility and similarity to licensed reactors, respectively.

7.7.1.1 Reactor Vessel - Instrumentation

Figure 7.3-9 (Nuclear Boiler System P&ID) shows the arrangements of the sensors, and sensing equipment used to monitor reactor vessel conditions.

a. System Function

The purpose of the reactor vessel instrumentation is to monitor key reactor vessel variables to provide the operator with information during normal plant operation, startup and shutdown.

b. System Operation

The following is a discussion of each reactor vessel variable monitored:

1. Reactor Vessel Temperature

The reactor vessel temperature is determined on the basis of reactor coolant temperature. Temperatures needed for operation and for compliance with the Technical Specification operating limits are obtained from one of several sources, depending on the operating condition. During normal operation, either reactor pressure and/or the inlet temperature of the coolant in the recirculation loops can be used to determine the vessel temperature. Below the operating span of the temperature sensors (RTD's) in the recirculation loop and above 212°F, the vessel pressure is used for determining the temperature. Below 212°F, the vessel coolant, and thus the vessel temperature, is determined by the reactor water cleanup system inlet temperature. During normal operation, vessel thermal transients are limited via operational constraints on parameters other than temperature.

2. Reactor Vessel Water Level

Figure 7.7-1 shows the water level range and the reactor vessel tap location for each water level range. The instruments that sense the water level are differential pressure devices with a condensate reference leg, calibrated to be accurate at a specific vessel pressure and liquid temperature condition. The following is a description of each water level range shown on Figure 7.7-1:

a) Shutdown Water Level Range: This range is used to monitor the reactor water level during reactor shutdown conditions when the reactor system is flooded for maintenance and head removal. The vessel temperature and pressure condition that is used for the calibration is 0 psig and 120°F water in the vessel. The two vessel instrument tap elevations used for this water level measurement are located at the top of the reactor vessel head and the instrument tap just below the bottom of the dryer skirt.

b) Upset Water Level Range: This range is used to monitor the reactor water level above the narrow range scale (see c.

below). The design and vessel taps are the same as outlined above. The vessel pressure and temperature condition for accurate indication is at the normal power operating point. The upset water level is continuously indicated by a recorder in the control room. The upset range upper limit is higher than the narrow range upper limit. Therefore when the indication is upscale on the narrow range recorder, water level indication may be read immediately from the upset range recorder. See the Feedwater Control System discussion in 7.7.1.4.

c) Narrow Water Level Range: This range uses reactor vessel taps at the elevation near the top of the dryer skirt and the taps at an elevation near the bottom of the dryer skirt. The zero of the instrument is the bottom of the dryer skirt and the instruments are calibrated to be accurate at the normal power operating point. The Feedwater Control System uses this range for its water level control and indication inputs. See the discussion on the Feedwater Control System, 7.7.1.4.

d) Wide Water Level Range: This range uses reactor vessel taps at the elevation near the top of the dryer skirt and the taps at an elevation near the top of the active fuel. The zero of the instrument is the bottom of the dryer skirt and the instruments are calibrated to be accurate at the normal power operating point.

e) Fuel Zone Water Level Range: This range uses reactor vessel taps at the elevation near the top of the dryer skirt and the taps at the jet pump diffuser skirt. The zero of the instrument is the top of the active fuel and the instruments are calibrated to be accurate at 0 psig and saturated condition.

In order to decouple the change in measured water level with changes in drywell temperature, the elevation drop from reactor vessel penetration to the drywell penetration remains uniform for the narrow range and wide range water level instrument lines.

3. Reactor Core Hydraulics

A differential pressure transmitter indicates core plate pressure drop by measuring the pressure difference between the core inlet plenum and the space just above the core support assembly. The instrument sensing line used to determine the pressure below the core support assembly attaches to the same reactor vessel tap that is used for the injection of the liquid from the Standby Liquid Control System. An instrument sensing line is provided for measuring pressure above the core support assembly. The differential pressure of the core plate is recorded in the main control room.

Another differential pressure device indicates the jet pump developed head by measuring the pressure difference between the pressure above the core and the pressure below the core plate. This is indicated locally and in the main control room.

4. Reactor Vessel Pressure

Pressure switches, indicators, and transmitters detect reactor vessel internal pressure from the same instrument lines used for measuring reactor vessel water level.

7.7.1.2 Reactor Manual Control System (RMCS) - Instrumentation and Controls

a. RMCS Function

The Reactor Manual Control System provides the operator with the means to make changes in nuclear reactivity by the operator manipulating control rods so that reactor power level and power distribution can be controlled.

This system includes the interlocks that inhibit rod movement (rod block) under certain conditions. The Reactor Manual Control System does not include any of the circuitry or devices used to automatically or manually scram the reactor; these devices are discussed in 7.2, "Reactor Protection System." In addition, the mechanical devices of the control rod drives and the control rod drive hydraulic system are not included in the Reactor Manual Control System. The latter mechanical components are described in 4.1.3, "Reactivity Control System."

b. RMCS Operation

The RMCS includes the following:

1. Control Rod Drive - Control System

2. Rod Block Trip System

3. Rod Position Probes

4. Position Indication Electronics

Figure 7.7-2 (CRD P&ID) shows the layout of the control rod drive-hydraulic system. Figure 7.7-3 (CRD FCD) shows the functional arrangement of devices for the control of components in the control rod drive hydraulic system. The logic diagram for the Reactor Manual Control System is shown in Figure 7.7-4. Although the figures also show the arrangement of scram devices, these devices are not part of the Reactor Manual Control System. Control rods are moved by water pressure, from a control rod drive pump, on the appropriate end of the control rod drive cylinder. The pressurized water moves a piston, attached by a connecting rod to the control rod. Three modes of control rod operation are used: insert, withdraw, and settle. Four solenoid-operated valves are associated with each control rod to accomplish these actions.

1. Control Rod Drive Control System

When the operator selects a control rod for motion and operates the rod insert or withdraw control switch, independent messages are formulated in the A and B portions of the rod drive control system. A comparison test is made of these two messages, and identical results confirmed; then a serial message in the form of electrical pulses is transmitted to all hydraulic control units (HCU). The message contains two portions, (1) the identity or "address" of the selected HCU, and (2) operation data on the action to be executed. Only one HCU responds to this message and it proceeds to execute the rod movement commands.

On receipt of the transmitted signal the responding HCU transmits three portions of a message back to the control room for comparison with the original message:

a) its own hard-wire identity "address",

b) its own operations currently being executed, and

c) status indications of valve positions, accumulator conditions, and test switch positions.

In a similar manner, rod withdrawal is accomplished by formulating a message containing a different operation code. The responding HCU decodes the message and proceeds to execute the withdrawal command by operation of HCU valves shown in Figure 7.7-2.

In either rod motion direction, the A and B messages are formulated and compared each millisecond and, if they agree, the A message is transmitted to the HCU selected by the operator. Continued rod motion depends on receipt of a train of sequential messages because the HCU insert, withdraw, and settle valve control circuits are ac-coupled. The system must operate in a dynamic manner to effect rod motion.

Any disagreement between the A and B formulated messages or the responding echo message will prevent rod motion. Electrical noise disruptions will have only a momentary effect on system operation. Correct operation of the system will resume when the noise source ceases.

In Figure 7.7-5, the three action loops of the solid-state reactor manual control system are depicted:

a) Operator Follow Mode: This high speed loop (0.0002-sec duration) services the control rod selected by the operator to transmit action commands and receive status indications, i.e., presence of rod blocks.

b) Scan Mode: This medium speed loop (0.045-sec duration) continously monitors the other control rods in the reactor, one at a time, to update their status display.

c) Self Test Mode: This low speed loop (on the order of 20 to 100-sec duration) automatically exercises one HCU at a time to ensure correct execution of actions commanded. This provides for a continuous, periodic self-test of the entire reactor manual control system.

In the event that any discrepancy is detected in one of these three modes of operation, a rod motion inhibit is applied. This situation is alarmed and annunciated on the reactor control console as an "activity disagree" condition. The control rod drive control system is also designed to produce a rod motion inhibit condition should any failure of the system occur.

The cause of the discrepancy or failure must be corrected before rod movement can proceed. Note, however, that this system cannot affect normal shutdown capability via the reactor protection system.

The rod selection circuitry is arranged so that a rod selection is sustained until either another rod is selected or separate action is taken to revert the selection circuitry to a no-rod selection condition. Initiating movement of the selected rod prevents the selection of any other rod until the

BLANK

movement cycle of the selected rod has been completed. Reversion to the no-rod selected condition is not possible (except for loss of control circuit power) until any moving rod has completed the movement cycle.

The direction in which the selected rod moves is determined by the position of four switches located on the reactor control panel. These four switches, "insert", "withdraw", "continuous insert" and "continuous withdraw", are pushbuttons which return by spring action to an off position.

The following is a description of the operation of the reactor manual control system during an insert cycle. The cycle is described in terms of the insert, withdraw, and settle commands from the reactor manual control system.

> With a control rod selected for movement, depressing the "insert" switch and then releasing the switch energizes the insert command for a limited time. Just as the insert command is removed, the settle command is automatically energized and remains energized for a limited time. The insert command time setting and the rate of drive water flow provided by the control rod drive hydraulic system determine the distance traveled by a rod. The time setting results in a one-notch (6-in.) insertion of the selected rod for each momentary application of a rod-in signal from the rod movement switch. Continuous insertion of a selected control rod is possible by holding the "insert" switch.

> A second switch can be used to affect insertion of a selected control rod. This switch is the "continuous insert" switch. By holding this switch "in," the unit maintains the insert command in a continuous, energized state to cause continuous insertion of the selected control rod. When released, the timers are no longer bypassed and normal insert and settle cycles are initiated to stop the drive.

The following is a description of the operation of the reactor manual control system during a withdraw cycle. The cycle is described in terms of the insert, withdraw, and settle commands.

> With a control rod selected for movement, depressing the "withdrawal" switch energizes the insert valves at the beginning of the withdrawal cycle to allow the collet fingers to disengage the index tube. When the insert valves are de-energized, the withdraw and

settle valves are energized for a controlled period of time. The withdraw valve is de-energized before motion is complete; the drive then settles until the collet fingers engage. The settle valve is then de-energized, completing the withdraw cycle. This withdraw cycle is the same whether the withdraw switch is held continuously or momentarily depressed position. The timers that control the withdraw cycle are set so that the rod travels one notch (6-in.) per cycle. Provisions are included to prevent further control rod motion in the event of timer failure.

A selected control rod can be continuously withdrawn if the "withdraw" switch is held in the depressed position at the same time that the "continuous withdraw" switch is held in the depressed position. With both switches held in these positions, the withdraw and settle commands are continuously energized.

2. Rod Block Trip System

This portion of the reactor manual control system, upon receipt of input signals from other systems, inhibits movement or selection of control rods.

a) Grouping of Channels

The same grouping of neutron monitoring equipment (SRM, IRM, APRM, and RBM) that is used in the Reactor Protection System is also used in the rod block circuitry.

Half of the total monitors (SRM, IRM, APRM, and RBM) provide inputs to one of the RMCS rod block logic circuits and the remaining half provide inputs to the other RMCS rod block logic circuit. Two recirculation flow units provide a rod block signal to one logic circuit; the remaining units provide an input to the other logic circuit. The flow unit comparator provides trip signals to each flow unit trip circuit. Scram discharge volume high water level signals are provided as inputs into both of the two rod block logic circuits. Both rod block logic circuits sense when the high water level scram trip for the scram discharge volume is bypassed.

The rod withdrawal block from the Rod Worth Minimizer trip affects both rod block logic circuits. The rod insert block from the Rod Worth Minimizer and rod sequence control function prevent both notch insertion and continuous insertion.

The APRM and RBM (see 7.7.1.8) rod block settings are varied as a function of recirculation flow. Analyses show that the selected settings are sufficient to avoid both reactor protection system action and local fuel damage as a result of a single control rod withdrawal error. Mechanical switches in the SRM and IRM detector drive systems provide the position signals used to indicate that a detector is not fully inserted. Additional detail on the Neutron Monitoring System is available in 7.7.1.6, 7.7.1.7, and 7.7.1.8. The rod block from scram discharge volume high water level utilizes one non-indicating float switch installed on the scram discharge volume. A second float switch provides a control room annunciation of increasing level below the level at which a rod block occurs.

b) Rod Block Functions

The following discussion describes the various rod block functions and explains the intent of each function. The instruments used to sense the conditions for which a rod block is provided are discussed in the following sections. Figure 7.7-4 shows all the rod block functions on a logic diagram. The rod block functions provided specifically for refueling situations are described in 7.7.1.13, "Refueling Interlocks."

(1) With the mode switch in the SHUTDOWN position, no control rod can be withdrawn. This enforces compliance with the intent of the shutdown mode.

(2) The circuitry is arranged to initiate a rod block regardless of the position of the mode switch for the following conditions:

(a) Any Average Power Range Monitor (APRM) upscale rod block alarm. The purpose of this rod block function is to avoid conditions that would require reactor protection system action if allowed to proceed. The APRM upscale rod block alarm setting is selected to initiate a rod block before the APRM high neutron flux scram setting is reached.

(b) Any APRM inoperative alarm. This assures that no control rod is withdrawn unless the average power range neutron monitoring channels are either in service or correctly bypassed.

(c) Scram discharge volume high water level. This assures that no control rod is withdrawn unless enough capacity is available in the scram discharge volume to accommodate a scram. The setting is selected to initiate a rod block earlier than the scram that is initiated on scram discharge volume high water level.

(d) Scram discharge volume high water level scram trip bypassed. This assures that no control rod is withdrawn while the scram discharge volume high water level scram function is out of service.

(e) The Rod Worth Minimizer (RWM) function of the Process Computer System can initiate a rod insert block and a rod withdrawal block. The Rod Sequence Control System can initiate a rod insert block and rod withdrawal block. The purpose of these functions is to reinforce procedural controls that limit the reactivity worth of control rods under lower power conditions. The rod block trip settings are based on the allowable control rod worth limits established for the design basis rod drop accident. Adherence to prescribed control rod patterns is the normal method by which this reactivity restriction is observed. Additional information on the Rod Worth Minimizer function is available in 7.7.1.10, "Rod Worth Minimizer Function", and on the Rod Sequence Control System in 7.7.1.11, "Rod Sequence Control System."

(f) Rod Position Information System malfunction. This assures that no control rod can be withdrawn unless the Rod Position Information System is in service.

(g) Either Rod Block Monitor (RBM) upscale alarm. This function is provided to stop the erroneous withdrawal of a control rod so that local fuel damage does not result. Although local fuel damage poses no significant threat in terms of radioactive material released from the nuclear system, the trip setting is selected so that no local fuel damage results from a single control rod withdrawal error during power range operation.

(h) Either RBM inoperative alarm. This assures that no control rod is withdrawn unless the RBM channels are in service or correctly bypassed.

(3) With the Reactor mode switch in the RUN position, any of the following conditions initiates a rod block.

(a) Any APRM downscale alarm. This assures that no control rod will be withdrawn during power range operation unless the average power range neutron monitoring channels are operating correctly or are correctly bypassed. All unbypassed APRMs must be on scale during reactor operations in the RUN mode.

(b) Either RBM downscale alarm. This assures that no control rod is withdrawn during power range operation unless the RBM channels are operating correctly or are correctly bypassed. Unbypassed RBMs must be on scale during reactor operations in the RUN mode.

(c) Any recirculation flow unit upscale, inoperative, or comparator alarm. This assures that no control rod is withdrawn unless the flow channels are operable, the difference between flow units is within limits, and the flow ra is not unusually high.

(4) With the mode switch in the STARTUP or REFUEL position, any of the following conditions initiates a rod block:

(a) Any Source Range Monitor (SRM) detector not fully inserted into the core when the SRM count level is below the retract permit level and any IRM range switch on either of the two lowest ranges. This assures that no control rod is withdrawn unless all SRM detectors are correctly inserted when they must be relied on to provide the operator with neutron flux level information.

(b) Any SRM upscale level alarm. This assures that no control rod is withdrawn unless the SRM detectors are correctly retracted during a reactor startup. The rod block setting is selected at the upper end of the range over which the SRM is designed to detect and measure neutron flux.

(c) Any SRM downscale alarm. This assures that no control rod is withdrawn unless the SRM count rate is above the minimum prescribed for low neutron flux level monitoring.

(d) Any SRM inoperative alarm. This assures that no control rod is withdrawn during low neutron flux level operations unless neutron monitoring capability is available.

(e) Any Intermediate Range Monitor (IRM) detector not fully inserted into the core. This assures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available.

(f) Any IRM upscale alarm. This assures that no control rod is withdrawn unless the intermediate range neutron monitoring equipment is correctly upranged during a reactor startup. This rod block also provides a means to stop rod withdrawal in time to avoid conditions requiring reactor protection system action (scram) in the event that a rod withdrawal error is made during low neutron flux level operations.

(g) Any IRM downscale alarm except when range switch is on the lowest range. This assures that no control rod is withdrawn during low neutron flux level operations unless the neutron flux is being correctly monitored. This rod block prevents the continuation of a reactor startup if the operator upranges the IRM too far for the existing flux level. Thus, the rod block ensures that the Intermediate Range Monitor is on scale if control rods are to be withdrawn.

(h) Any IRM inoperative alarm. This assures that no control rod is withdrawn during low neutron flux level operations unless neutron monitoring capability is available.

c) Rod Block Bypasses

To permit continued power operation during repair or calibration of equipment for selected functions that provide rod block interlocks, a limited number of manual bypasses are permitted as follows:

1 SRM channel

2 IRM channels (1 on either RPS Bus A or Bus B)

2 APRM channels (1 on either RPS Bus A or Bus B)

1 RBM channel

The permissible IRM and APRM bypasses are arranged in the same way as in the reactor protection system. The IRMs are arranged as two groups of equal numbers of channels. One manual bypass is allowed in each group. The groups are chosen so that adequate monitoring of the core is maintained with one channel bypassed in each group. The same type of grouping and bypass arrangement is used for the APRMs. The arrangement allows the bypassing of one IRM and one APRM in each rod block logic circuit.

These bypasses are affected by positioning switches in the control room. A light in the control room indicates the bypassed condition.

An automatic bypass of the SRM detector position rod block is effected as the neutron flux increases beyond a preset low level on the SRM instrumentation. The bypass allows the detectors to be partially or completely withdrawn as a reactor startup is continued.

An automatic bypass of the RBM rod block occurs when the power level is below a preselected level or when a peripheral control rod is selected. Either condition indicates that local fuel damage is not threatened and that RBM action is not required.

The rod worth minimizer and rod sequence control system rod block function is automatically bypassed when reactor power increases above a preselected value in the power range. The rod worth minimizer can be manually bypassed for maintenance at any time.

Figures 7.7-4 and 7.6-6 show the rod block interlocks used in the reactor manual control system. Figure 7.7-4 shows the general functional arrangement of the interlocks, and Figure 7.6-6 shows in greater detail the rod blocking functions that originate in the neutron monitoring system.

3. Rod Position Probes

The position probe is a long cylindrical assembly that fits inside the control rod drive. It includes fifty-three magnetically operated reed switches, located along the length of the probe and operated by a permanent magnet fixed to the moving part of the hydraulic drive mechanism. As the drive, and with it the control rod blade, moves along its length, the magnet causes reed switches to close as it passes over the switch locations. The particular switch closed then indicates where the control rod drive, and hence the rod itself is positioned.

The switches are located as follows: one at each of twenty-five notch (even) positions; one at each of twenty-four mid-notch (odd) positions; two at the fully inserted position (approximately the same location as the "00" notch); one at the fully withdrawn position (approximately the same location as the "48" notch position); and, one at the overtravel or decoupled position.

All of the mid-notch or odd switches are wired in parallel and treated as one switch (for purposes of external connections), and the two full-in switches are wired in parallel and treated as one switch. These and the remaining switches are wired in a 5 x 6 array (the switches short the intersections) and routed out in an 11-wire cable to the processing electronics (the probe also includes a thermocouple which is wired out separate from the 5 x 6 array). See Figure 7.7-6.

4. Position Indication Electronics

The electronics consists of a set of probe multiplexer cards (one per 4-rod group where the 4-rod group is the same as the display grouping described above), a set of file control cards (one per 11 multiplexer cards), and one set of master control and processing cards serving the whole system. All probe multiplexer cards are the same except that each has a pair of plug-in daughter cards containing the identity code of one 4-rod group (the probes for the corresponding 4 rods are connected to the probe multiplexer card). The system operates on a continuous scanning basis with a complete cycle every 45 milliseconds.

The operation is as follows: The control logic generates the identity code of one rod in the set, and transmits it using time multiplexing to all of the file control cards. These in turn transmit the identity with timing signals to all of the probe multiplexer cards. The one multiplexer card with the matching rod identity will respond and transmit its identity (locally generated) plus the raw probe data for that rod back through the file control card to the master control and processing logic. The processing logic does several checks on the returning data. First, a check is made to verify that an answer was received. Next, the identity of the answering data is checked against that which was sent. Finally, the format of the data is checked for legitimacy. Only a single even position or, full-in plus position "00", or full-out plus position "48," or odd, or overtravel, or blank (no switch closed) are legitimate. Any other combination of switches is flagged as a fault.

If the data passes all of these tests, it is (a) decoded and transmitted in multiplexed form to the displays in the main control panel, and (b) loaded into a memory to be read by the computer as required.

As soon as one rod's data is processed, the next rod's identity is generated and processed and so on for all of the rods. When data for all rods has been gathered, the cycle repeats. The Reactor Manual Control System is totally operable from the main control room. Manual operation of individual control rods is possible with a jog switch to effect control rod insertion, withdrawal, or settle. Rod position indicators, described below, provide the necessary information to ascertain the operating state and position of all control rods. Conditions which prohibit control rod insertion are alarmed with the rod block annunciator.

A rod information display on the reactor control panel is patterned after a top view of the reactor core. The display allows the operator to acquire information rapidly by scanning.

Colored windows provide an overall indication of rod pattern and allow the operator to quickly identify an abnormal indication. The following information for each control rod is presented in the display:

Rod fully inserted (green)
Rod fully withdrawn (red)
Selected rod identification (coordinate position, white)
Accumulator trouble (amber)
Rod scram (blue)
Rod drift (red)

Also dispersed throughout the display, in locations representative of the physical location of LPRM strings in the core, are LPRM lights as follows:

LPRM low flux level (white)
LPRM high flux level (amber)

Another display shows the positions of the control rod selected for movement and the other rods in the rod group. For display purposes the control rods are considered in groups of four adjacent rods (a four-rod group) centered around a common core volume monitored by four LPRM strings. Rod groups at the periphery of the core may have less than four rods. The four-rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A lighted background on the digital display indicates which of the four rods is selected for movement. On either side of the four-rod position display are indicated the readings of the 16 LPRM channels (four LPRM strings) surrounding the core volume common to the four rods of the group.

The four-rod display allows the operator to focus attention on the portion of the core where rod motion is occurring. A full core rod position display would tend to be confusing and difficult to read. The sixteen associated LPRM displays permit the operator to monitor the core reactivity during rod motion. In addition, on demand by the operator, the process computer will provide a print out of all rod positions.

During startup or shutdown, all rods of a given sequence are either fully withdrawn or fully inserted. These patterns are indicated on the full core display with the full-in or full-out lights. In addition to the whole core display, a drifting rod is indicated by an alarm and red light in the control room. The rod drift condition is also monitored by the process computer.

An indication is also provided for rod trend beyond the limits of normal rod movement. If the rod drive piston moves to the overtravel position, an alarm is sounded in the control room. The overtravel alarm provides a means to verify that the driveto-rod coupling is intact because, with the coupling in its normal condition, the drive cannot be physically withdrawn to the overtravel position. Coupling integrity can be checked by attempting to withdraw the drive to the overtravel position.

For the displays above the selected rod identification, accumulator trouble and rod scram indicators are provided to the displays, by the rod drive control system. The LPRM high and low flux levels and the sixteen LPRM readings are provided by the power range monitor system. The remaining information to the displays and the position information for the process computer are provided by the rod position information subsystem.

The following main control room lights are provided to allow the operator to know the conditions of the control rod drive hydraulic system and the control circuitry:

Stabilizer valve selector switch position
Insert command energized
Withdraw command energized
Settle command energized
Withdrawal not permissive
Continuous withdrawal
Pressure control valve position

Flow control valve position
Drive water pump low suction pressure (alarm and pump trip)
Drive water filter high differential pressure (alarm only)
Charging water (to accumulator) high pressure (alarm only)
Control rod drive temperature (alarm only)
Scram discharge volume not drained (alarm only)
Scram valve pilot air header high/low pressure (alarm only)

7.7.1.3 Recirculation Flow Control System - Instrumentation and Controls

For a complete description see Appendix H.

7.7.1.4 Feedwater Control System - Instrumentation and Controls

a. System Function

The feedwater control system controls the flow of feedwater into the reactor vessel to maintain the vessel water level within predetermined limits during all normal plant operating modes. The range of water level is based upon the requirements of the steam separators (this includes limiting carryover, which affects turbine performance, and carryunder, which affects recirculation pump operation). The feedwater control system utilizes vessel water level, steam flow, and feedwater flow as a three-element control.

Single-element control is also available based on water level only. Normally, the signal from the feedwater flow is equal to the steam flow signal; thus, if a change in the steam flow occurs, the feedwater flow follows. The steam flow signal provides anticipation of the change in water level that will result from change in load. The level signal provides a correction for any mismatch between the steam and feedwater flow which causes the level of the water in the reactor vessel to rise or fall accordingly.

b. System Operation

During normal plant operation, the feedwater control system automatically regulates feedwater flow into the reactor vessel. The system can be manually operated. (See Figure 7.7-7.)

The feedwater flow control instrumentation measures the water level in the reactor vessel, the feedwater flow rate into the reactor vessel, and the steam flow rate from the reactor vessel. During automatic operation, these three measurements are used for controlling feedwater flow.

The optimum reactor vessel water level is determined by the requirements of the steam separators. The separators limit water carry-over in the steam going to the turbines and limit steam carry-under in water returning to the core. The water level in the reactor vessel is maintained within ± 2 in. of the set point value during normal operation and within the high and low level trip set points during normal plant maneuvering transients. This control capability is achieved during plant load changes by balancing the mass flow rate of feedwater to the reactor vessel with the steam flow from the reactor vessel. The feedwater flow is regulated by controlling the speed of the turbine-driven feedwater pumps to deliver the required flow to the reactor vessel.

The following is a discussion of the variables sensed for system operation:

1. Reactor Vessel Water Level

Reactor vessel narrow range water level is measured by three identical, independent sensing systems. For each channel, a differential pressure transmitter senses the difference between the pressure caused by a constant reference column of water and the pressure caused by the variable height of water in the reactor vessel. The differential pressure transmitter is installed on lines that serve other systems. (See 7.7.1.1, "Reactor Vessel Instrumentation".) Two of the differential pressure signals are used for indication and control and the third for indication only. The narrow range level signal from one of the two control channels can be selected by the operator as the signal to be used for feedwater flow control. A third narrow range level sensing channel is used in conjunction with the two control channels to provide failure tolerant trips of the main turbine and feed pump prime movers. All three narrow range reactor level signals and reactor pressure are indicated in the main control room. A fourth level sensing system (upset range) provides level information beyond the span of the narrow range devices. The selected narrow range water level and upset range water level signals are continually recorded in the main control room.

2. Main Steam Line Steam Flow

Steam flow is sensed at each main steam line flow restrictor by a differential pressure transmitter. A signal proportional to the true mass steam flow rate is linearized and indicated in the main control room. The signals are summed to produce a total steam flow signal for indication and feedwater flow control. The total steam flow signal is recorded in the main control room.

3. Feedwater Flow

Feedwater flow is sensed at a flow element in each feedwater line by differential pressure transmitters. Each feedwater signal is linearized and then summed to provide a total mass flow signal which is recorded in the control room. In addition, feedwater flow through each pump is sensed at a point downstream of the feed pump discharge. After being linearized, the flow is compared to the demand flow for that pump. Resulting error is used to adjust the actuator in the direction necessary to zero the flow error. Valve position control or turbine speed change are the flow adjustment techniques involved.

Three modes of feedwater flow control and thus level control are provided.

a) Startup up automatic level control

b) Run mode automatic flow control

c) Manual control

Separate level controllers are provided for each automatic mode. Each level controller contains set point deviation meters, an output indicator, a manual output control, manual automatic switching capability and a manually operated set point adjustment. In the start up level control mode, measured level is compared to level set point within the controller. The resulting signal is conditioned by the proportional plus integral controller circuits and transmitted to the startup level control valve.

During normal operation three element automatic control is provided. The total steam flow signal, modified by the conditioned level error signal, provides a flow demand signal to the feedwater flow control loop. The demanded flow is compared to actual flow in each active pump. The resulting flow error signal after conditioning by the proportional plus integral flow controller changes the turbine speed, zeroing the error signal.

Manual control is available by selecting manual on the controller manual-automatic stations. Flow change is accomplished by depressing the raise button or lower button depending on the desired flow change. Automatic inventory control is available with any single pump or any combination of two pumps.

The level control system also provides interlocks and control functions to other systems. When one of the reactor feed pumps is lost and coincident or subsequent low water level exists, recirculation flow is reduced to within the power capabilities of the remaining reactor feed pumps. This reduction aids in avoiding a low level scram by reducing the steaming rate. Reactor recirculation flow is also reduced on sustained low feedwater flow coincident with low recirculation flow control valve position to ensure that adequate NPSH will be provided for the recirculation system.

Alarms on steam flow are provided for use in the Rod Worth Minimizer logic. Interlocks from steam flow and feedwater flow are used to initiate insertion of the Rod Worth Minimizer block. An alarm on low steam flow indicates that the above rod worth minimizer insertion interlock set point is being approached. Alarms are also provided for (1) high and low water level and (2) reactor high pressure. Interlocks will trip the plant turbine and feedwater pumps in event of reactor high water level.

Feedwater is delivered to the reactor vessel through turbine-driven feedwater pumps, which are arranged in parallel. The turbines are driven by steam from the reactor vessel. During planned operation, the feedwater control signal from the level controller is fed to the turbine speed control systems, which adjust the speed of their associated turbines so that feedwater flow is proportional to the feedwater demand signal. Each turbine can be controlled by its manual/automatic transfer station. If the feedwater control to the turbine signal is lost, an alarm unit in the feedwater control circuit initiates an alarm in the control room and locks the turbine or startup valve at its speed or position just prior to losing the signal. The feedwater controller, and the manual/auto transfer stations associated with each turbine speed controller, are the bumpless transfer types.

7.7.1.5 Pressure Regulator and Turbine-Generator System - Instrumentation and Controls

a. System Function

As a direct cycle boiling water reactor the turbine is slaved to the reactor in that all (except steam to the moisture separator reheaters) steam generated by the reactor is normally accepted by the turbine. The operation of the reactor requires pressure regulation be employed to maintain a constant (within the range of the regulator controller proportional band setting) turbine inlet pressure with load following ability accomplished by variation of the reactor recirculation flow.

The turbine pressure regulator normally controls the turbine governor valves to maintain constant (within the range of the regulator controller proportional band setting) turbine inlet pressure at a particular value. In addition, the pressure regulator also operates the steam bypass valves such that a portion of nuclear boiler rated flow can be bypassed when operating at steam flow loads above that which can be accepted by the turbine as well as during the startup and shutdown phase.

The overall turbine-generator and pressure control system accomplishes the following:

1. Control turbine speed and turbine acceleration.

2. Control the steam bypass system to keep reactor pressure within limits, and avoid large power transients.

3. Control main turbine inlet pressure within the proportional band setting of the pressure regulator.

4. Match nuclear steam supply to turbine steam requirements by the following functions:

    a) Adjust recirculation system flow to the turbine load demands when the recirculation control is in the automatic load following mode.

    b) Adjust the pressure setpoint of the pressure control unit in order to improve the load response of the plant, when the recirculation control is in the automatic load following mode.

b. System Operation

Pressure control is accomplished by controlling main steam pressure immediately upstream of the main turbine stop and governor valves through modulation of the turbine-governor or steam-bypass valves. Command signals to these valves are generated by redundant control elements using the sensed turbine inlet pressure signals as the feedback, as shown in Figure 7.7-9. For normal operation, the turbine governor valves regulate steam pressure; however, whenever the total steamflow demand from the pressure regulator exceeds the capacity of the turbine governor valves, the pressure control system sends the excess steamflow directly to the main condenser, through the steam bypass valves. The plant ability to follow grid-system load demands is enabled by adjusting reactor power level, by varying reactor recirculation flow (manually or automatically), or by manually moving control rods. In response to the resulting steam production changes, the pressure control system adjusts the turbine governor valve to accept the steam output change, thereby regulating steam pressure. In addition, when the reactor is automatically following turbine speed/load demands, the pressure control system permits an immediate steamflow response to fast changes in load demand, thus utilizing part of the stored energy in the vessel.

1. Steam Pressure Control

During normal plant operation, steam pressure is controlled by the main turbine governor valves, positioned in response to the pressure regulation demand signal. (See Figure 7.7-9.) The steam bypass valves are normally closed. When the reactor is in the automatic load-following mode, fast-load demand changes require an early initial response in turbine steamflow preceding the actual change in steam production. This is accomplished by temporary, automatic adjustments to the pressure setpoint. Because the fast response in turbine steamflow (equivalent load) acts to reduce the load-demand error, as seen by the reactor recirculation controls, as well as tending to reduce neutron flux, a compensatory feed-forward loop is added to load demand error in accordance with pressure set point changes.

As an example of the use of the pressure regulator setpoint adjustment, suppose an increase in plant load is demanded by an increased turbine-generator requirement. The turbine-generator demands more steam to maintain its constant speed. The pressure control system, by means of the fast response feedback loop, adjusts the pressure regulator setpoint to a temporary lower value. This temporary decrease in setpoint permits the turbine governor valves to open to meet the turbine-generator steam demand while the reactor recirculation flow change is being made.

Two essentially identical regulators are provided so that the one with the greatest steam flow demand is the controlling regulator. Separate pressure taps for each regulator are provided at the turbine inlet.

The turbine governor valve (steam flow) demand signal is limited, after passage through the low value gate in Figure 7.7-9, to that required for full opening of the turbine governor valves. Thus, if the pressure control system requests additional steam flow from the reactor when the governor valves reach wide open, the control signal error to the bypass valves will increase and cause bypass actuation.

Control for the turbine governor valve is designed so that the valves will close upon loss of control system electric power or loss of hydraulic system pressure.

2. Steam Bypass System

The steam bypass equipment is designed to control steam pressure when reactor steam generation exceeds turbine requirements such as during startup (pressure, speed ramping and synchronizing), sudden load reduction, and cooldown.

The bypass capacity of the system is 25% of NSSS rated steam flow; sudden load reductions of up to the capacity of the steam bypass can be accommodated without reactor scram.

Normally, the bypass valves are held closed and the pressure regulator controls the turbine governor valves, directing all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the regulator controls system pressure by opening the bypass valves. If the capacity of the bypass valves is exceeded while the turbine cannot accept an increase in steam flow, the system pressure will rise and reactor protection system action will cause shutdown of the reactor.

The bypass valves are an automatically-operated, regulating type which are proportionally controlled by the turbine pressure regulator and control system.

The turbine control system provides a signal to the bypass valves corresponding to the error between the turbine governor valve opening required by the controlling pressure regulator and the turbine governor valve position demanded by the output of the low value gate circuit. (see Figure 7.7-8 and 7.7-9.) An adjustable bias signal is provided to maintain the bypass valves closed for momentary differences during normal operational transients.

3. Turbine Speed/Load Control System

The control signals supplied by the pressure regulator to the turbine control system and the signals which the pressure regulator requires from the turbine control system are shown in

Figures 7.7-9 and 7.7-10. The turbine control system is designed to receive and supply the following signals;

a) Signal 1 - The load demand signal varies from no load to rated load.

b) Signal 2 - The pressure control signal varies from no load to rated load.

c) Signal 3 - The flow limit signal range varies from 50 percent flow to 140 percent flow.

d) Signal 4 - The governor valve position (flow) demand signal varies to close or open the valve. The turbine flow limiter limits the governor valve position demand signal so that it does not exceed the value corresponding to valves fully open. Signal 4 is used by the pressure regulator to operate the bypass valves when high steam pressure causes the pressure control signal, Signal 2, to be higher than Signal 4.

4. Turbine Speed-Load Control Interfaces

a) Normal Operation

During base-load plant operation, the turbine load reference is held above the desired load, such that the pressure regulation demand governs the turbine governor valves. At this time, the system is not designed with an interface between the system dispatcher and the WNP-2 turbine control system. Accordingly, the operator must select the load for automatic load following operation. During automatic load-following operation, turbine speed-load demand fluctuations cause the reactor recirculation flow to vary the core flow and therefore reactor steam generation and turbine power output. When the turbine load demand increase exceeds the limits of the reactor recirculation system automatic-flow-control range, further increases in turbine output are prevented by the pressure regulator maintaining steam pressure.

b) Behavior of Turbine Outside of Normal Operation

(1) Turbine Startup.

Prior to turbine startup, sufficient reactor steam flow is generated to permit the steam bypass valves to maintain reactor pressure control while the turbine is brought up to speed and synchronized under its speed-load control.

(2) Partial Load Rejection.

During partial-load rejection transients, which are apparent to the reactor as a reduction in turbine load demand resulting from an increase in generator (or grid) frequency above rated, the turbine-pressure control scheme allows the reduced turbine speedload demand to bias the pressure regulation demand and thereby directly regulate the turbine governor valves.

(3) Isolated Grid Operation.

After grid disturbances that result in one or several plants feeding into a remaining partial (isolated) grid, grid stability can be maximized by removing the reactor recirculation flow to thermal power response from the primary plant load response loop.

This requires the turbine governor valves to be under speed control with pressure control transferred to the steam bypass valves, the control system design must cause a small net bypass steamflow demand at steady-state.

(4) Turbine Shutdown or Turbine-Generator Trip.

During turbine shutdown or turbine-generator trip conditions, the main turbine stop valves and governor valves are, or will be, closed. Reactor steamflow will then be passed through the steam bypass valves under steam pressure control, and through the reactor safety/relief valves, as needed.

(5) Steam Bypass Operation.

Fast opening of the steam bypass valves during turbine trips or generator load rejections requires coordinated action with the turbine control system. When the turbine governor valves are under pressure control, no bypass steamflow is demanded; conversely, when the turbine speed-load demand falls below the pressure regulation demand, a net bypass flow demand is computed. During turbine or generator trip events resulting in fast-closure of the turbine stop or governor valves, the turbine governor valve demand is immediately tripped to zero as an anticipatory response, causing the bypass steamflow demand to equal the initial pressure regulation demand.

(6) Loss of Turbine Control System Power.

Turbine controls and valves are designed so that the turbine stop and governor valves will close upon loss of control system power or hydraulic pressure.

c) Operator Information

Process variables which are controlled by the pressure regulator, speed/load control system are displayed on the turbine-generator section of the main control board. Manual and automatic control modes for the various turbine-generator operational modes (such as startup, normal operation, and shutdown), are available to the operator from the main control board. Auto display lights are provided to inform the operator as to the operating mode of the turbine-generator unit.

In the event of control malfunction during an automatic control mode, control is transferred to the manual mode and annunciated to alert the operator of the condition.

At least two pressure control channels, operating redundantly, receive inputs from independent pressure transducers in the main steam line upstream of the main steam stop valves and from the pressure reference unit. Main steam pressure indications and pressure setpoint adjustments/indications are located on the turbine control panel. Pressure set point adjustment is limited to about one psi per second by motor speed. In the event of failure of either regulator, an alarm is provided in the main control room.

The pressure regulator has the following controls and information displayed in the main control room:

a) Main steam pressure transducer output regulator A.

b) Main steam pressure transducer output regulator B.

c) Main steam pressure regulator set point A.

d) Main steam pressure regulator set point B.

e) Individual bypass valve position indicator.

f) Individual bypass valve demand control signal.

g) Bypass valve test controls.

h) Pressure regulator selection control.

7.7.1.6 Neutron Monitoring System - Traversing In-core Probe (TIP) Subsystem - Instrumentation and Controls

a. System Function

Flux readings along the axial length of the core are obtained by fully inserting the traversing ion chamber into one of the calibration guide tubes, then taking data as the chamber is withdrawn. The analog data is available for driving a recorder and for use by the process computer. One traversing ion chamber and its associated drive mechanism is provided for each group of seven to nine fixed in-core assemblies.

b. TIP System Operation

The number of TIP machines is indicated in Figure 7.7-14. The TIP machines have the following components:

1. One Traversing In-core Probe (TIP),

2. One drive mechanism,

3. One indexing mechanism, and

4. Up to 10 in-core guide tubes.

The subsystem allows calibration of LPRM signals by correlating TIP signals to LPRM signals as the TIP is positioned in various radial and axial locations in the core. The guide tubes inside the reactor are divided into groups. Each group has its own associated TIP machine.

A TIP drive mechanism uses a fission chamber attached to a flexible drive cable (Figure 7.7-10). The cable is driven from outside the drywell by a gearbox assembly. The flexible cable is contained by guide tubes that penetrate the reactor core. The guide tubes are a part of the LPRM detector assembly. The indexing mechanism allows the use of a single detector in any one of ten different tube paths. The 10th tube is used for TIP cross calibration with the other TIP machines. The control system provides for both manual and semi-automatic operation. Electronics of the TIP panel amplify and display the TIP signal. Core position versus neutron flux is recorded on an X-Y recorder in the main control room and is provided to the computer. A block diagram of the drive system is shown in Figure 7.6-2. Actual operating experience has shown the system to reproduce within 1.0% of full scale in a sequence of tests (Reference 1).

A valve system is provided with a valve on each guide tube entering the drywell. A ball valve and a cable shearing valve are mounted in the guide tubing just outside the drywell. The ball valves are closed except when the TIP is in operation. They maintain the leak tightness integrity of the drywell. A valve is also provided for a nitrogen gas purge line to the indexing mechanisms. A guide tube ball valve opens only when the TIP is being inserted. The shear valve is used only if containment isolation is required when the TIP is beyond the ball valve and power to the TIP fails. The shear valve, which is controlled by a manually operated keylock switch, can cut the cable and close off the guide tube. The shear valves are actuated by explosive squibs.

The continuity of the squib circuits is monitored by indicator lights in the main control room. Upon receipt of containment isolation command from the NSSS, all machines are put in automatic full speed withdraw condition, removing the TIP detector from the containment and allowing the ball valves to close. The purge valve is also closed at this time.

7.7.1.7 Neutron Monitoring System - Source Range Monitor (SRM)

a. SRM Function

The SRM provides neutron flux information during reactor start-up and low flux level operations.

b. SRM Operation

There are 4 SRM channels. Each includes one detector that can be physically positioned in the core from the control room (see Figures 7.6-3 and 7.7-14).

The detectors are inserted into the core for a reactor startup. They can be withdrawn if the indicated count rate is between preset limits or if the IRM is on the third range or above (see Figure 7.6-4).

During initial fuel load neutron flux is monitored by source range neutron monitoring channels, providing a scram signal when the preset flux level of any channel has been reached. This logic is removed from the scram circuitry after completion of initial fueling.

Each detector assembly consists of a miniature fission chamber and a low-noise, quartz-fiber-insulated transmission cable. The sensitivity of the detector is 1.2 x 10-3 cps/nv nominal, 5.0 x 10-4 cps/nv minimum, and 2.5 x 10-3 cps/nv maximum. The detector cable is connected underneath the reactor vessel to the multipleshielded coaxial cable. This shielded cable carries the pulses to a pulse current preamplifier located outside the drywell.

The detector and cable are located inside the reactor vessel in a dry tube sealed against reactor vessel pressure. A remotecontrolled detector drive system moves the detector along the dry tube. Vertical positioning of the chamber is possible from above the centerline of the active length of fuel to 30 inches below the reactor fuel region (see Figures 7.6-2 and 7.6-5. When a detector arrives at a travel end point, detector motion is automatically stopped. SRM/IRM drive control arrangement and logic are presented in Figures 7.6-2 and 7.6-6. The electronics for the source range monitors, their trips, and their bypasses are located in four cabinets. Source range signal conditioning equipment is designed so that it can also be used for open vessel experiments.

A current pulse preamplifier provides amplification and impedance matching for the signal conditioning electronics (Figure 7.7-14).

The signal conditioning equipment converts the current pulses to analog dc currents that correspond to the logarithm of the count rate (LCR). The equipment also derives the period. The output is displayed on front panel meters and is provided to remote meters and recorders. The LCR meter displays the rate of occurrence of the input current pulses. The period meter displays the time in seconds for the count rate to change by a factor of 2.7. In addition, the equipment contains integral test and calibration circuits, trip circuits, power supplies, and selector circuits.

The trip outputs of the SRM operate in the fail-safe mode. Loss of power to the SRM causes the associated outputs to become tripped.

The SRM provides signals indicating SRM upscale, downscale, inoperative, and incorrect detector position to the Reactor Manual Control System to block rod withdrawal under certain conditions (see Table 7.7-4). Any SRM channel can initiate a rod block. These rod blocking functions are discussed in 7.7.1.2.2, Rod Block Trip System.

One of the four SRM channels can be bypassed at any one time by the operation of a switch on the operator's control panel.

Inspection and testing are performed as required on the SRM detector drive mechanism; the mechanism can be checked for full insertion and retraction capability. The various combinations of SRM trips can be introduced to ensure the operability of the rod blocking functions.

7.7.1.8 Neutron Monitoring System - Rod Block Monitor (RBM)

a. RBM Function

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during operator control rod manipulations.

b. RBM Operation

The RBM has two channels. Each channel uses input signals from a number of LPRM channels. A trip signal from either RBM channel initiates a rod block. One RBM channel can be bypassed without loss of subsystem function. The minimum number of LPRM inputs required for each RBM channel to prevent an instrument inoperative alarm is four when using four LPRM assemblies, three when using three LPRM assemblies, and two when using two LPRM assemblies. See Figures 7.7-13 and 7.6-6 (NMS FCD).

The RBM signal is generated by averaging a set of LPRM signals. One RBM channel averages the signals from LPRM detectors at the A and C positions in the assigned LPRM assemblies. The second RBM channel averages the signals from the LPRM detectors at the B and D positions. Assignment of LPRM assemblies to be used in RBM averaging is controlled by the selection of control rods. Note that the RBM is automatically bypassed and the output set to zero if a peripheral rod is selected. If any LPRM detector assigned to an RBM is bypassed, the computed average signal is adjusted automatically to compensate for the number of LPRM input signals.

When a control rod is selected, the gain of each RBM channel output is normalized to an assigned APRM channel. The assigned APRM channel is on the same RPS trip system as the RBM channel. This gain setting is held constant during the movement of that particular control rod to provide an indication of the change in the relative local power level. If the APRM used to normalize the RBM reading is indicating less than 3[illegible] power, the RBM is zeroed and the RBM outputs are bypassed. See Figures 7.7-11 and 7.7-12.

If the normalizing APRM is bypassed, the normalizing signal is automatically provided by a second APRM. In the operating range, the RBM signal is accurate to approximately 1% of full scale.

The RBM supplies a trip signal to the reactor manual control system to inhibit control rod withdrawal. The trip is initiated when RBM output exceeds the rod block set point (see Table 7.7-5). There are three parallel rod block set point lines that have an adjustable slope. These lines provide a set point that is a function of the recirculation driving loop flow. The intercepts of these set point lines with rated flow are adjustable. The normal settings are approximately 107% for the upper line, 99% for the intermediate line, and 91% for the lower line. Lights indicate which rod block set point line is active. Two percent below the intermediate and lower rod set point lines are the set-up-permissive (and set-down) lines. On increasing power, an indicator will light so the operator can evaluate the conditions and manually change to the next higher rod block set point line. On decreasing power, these lines will provide automatic set down. Either RBM can inhibit control rod withdrawal.

The operator can bypass one of the two RBMs at any time.

When increasing power the set-up permissive lamp will light requiring the operator to evaluate conditions before manually changing to the next higher rod block set point line.

7.7.1.9 Process Computer System - Instrumentation

a. System Function

The function of the process computer system is to provide a quick and accurate determination of core thermal performance; to improve data reduction, accounting, and logging functions; and to supplement procedural requirements for control rod manipulation during reactor startup and shutdown.

b. System Operation

The central processor performs various calculations, makes necessary interpretations, and provides for general input/output device control and buffered transmission between I/O devices and memory.

An automatic priority interrupt (API) module provides processor capability to respond rapidly to important process functions and to operate at optimum speed.

Core memory is a random access type utilizing a 24-bit word and operating at a 800 nanosecond cycle time. A processor parity check feature is capable of stopping computer operation subsequent to completing an instruction in which a parity error is detected. The core memory has suitable shutdown protection to prevent information destruction in the event of loss of power or incorrect operating voltage. Capability is provided to maintain real time by utilizing necessary calendar type programs to compute year, month, day, hour, minute, second, and cycle. This is done automatically except in the event of a processor shutdown. In this case the operator is required to update the computer with the correct time when restarting the system.

Bulk memory consists of serial access magnetic drum and disc, and is used for storing all programs and data. Capability is provided to protect selectable portions of bulk memory against information destruction caused by an inadvertent attempt to write over the programs or by a system power failure.

The peripheral equipment is divided into two classifications; I/O equipment used to read data into and out of the computer and, output equipment which is used only for data output, storage, display and alarm.

The peripheral I/O equipment consists of one operator console, one I/O typewriter, one magnetic tape unit, and one card reader, all located in the main control room.

The output only peripheral devices include; two periodic typewriters, one turbine diagnostic typewriter, one on-demand typewriter, one alarm typewriter and two black-and-white CRT displays, also located in the control room. Additional output equipment consists of a card punch, and a line printer.

One remote terminal will be available for use by the reactor engineering staff.

The process I/O hardware consists of an analog input scanner, a digital I/O controller, corresponding I/O terminations and signal conditioners. The analog scanner accepts analog signals from plant instrumentation and converts them to digital representation for use in the computer. The digital I/O controller senses plant contact actuations by groups and is used to read status information from plant instrumentation, including alarms and binary code signals. Intermittent signals and pulse type inputs are sensed by automatic program interrupt change detection hardware in the central processor and allow immediate processing of information that might otherwise be lost if digital scanning were used. The digital controller also provides latched digital outputs to operate displays.

During routine operation the operator uses a console located in the main control room to enter information into the computer and for requesting various special functions from it. Diagnostic alarm lights, digital display, and associated selection switches on the operator console, together with CRT displays or typewriter outputs, permit the operator to communicate with the processor.

The programming and maintenance console is used by programmers and maintenance personnel to permit necessary control of the system for troubleshooting and maintenance functions. This console is a part of the central processor located in the computer equipment area of the control room.

The process computer system has self-checking provisions. It performs diagnostic checks to determine the operability of certain portions of the system hardware and performs internal programming checks to verify that input signals and selected program computations are either within specific limits or within reasonable bounds.

All the computer equipment, except for peripherals, is designed for continuous duty from 0°C to 50°C, and 5% to 95% relative humidity ambient. The peripherals are designed to operate under more restrictive environmental conditions. All components are installed in an air conditioned room.

Total core thermal power is calculated from a reactor heat balance. Total power is then distributed to every six-inch segment of each fuel assembly by calculation. Using plant inputs of pressure, temperature, flow, LPRM levels, control rod positions, and the calculated fuel exposure. Interactive computational methods are used to establish a compatible relationship between the core coolant flow and core power distribution. The results subsequently are interpreted as local power at specified axial segments for each fuel bundle in the core.

After calculating the power distribution within the core, the computer uses appropriate reactor operating limit criteria to establish alarm trip settings (ATS) for each LPRM channel. These settings are expressed as maximum acceptable LPRM values to which the actual scanned LPRM readings are compared. The scanned LPRM, when exceeding the ATS, will sound an alarm and thereby assist the operator to maintain core operation within permissible thermal limits established by prescribed maximum fuel rod power density and minimum critical power ratio criteria.

The core power distribution calculation sequence is completed periodically and on demand. The sequence requires 10 to 20 minutes to execute. Subsequent to executing the program the computer prints a periodic log for record purposes.

Each LPRM reading is ordinarily scanned once per minute. During power level changes, as sensed by a rod withdrawal or by an APRM channel, the scan rate is increased to once every 5 seconds. This fast core monitoring during power level changes is initiated automatically by the processor and, together with appropriate computational methods, provides nearly continuous re-evaluation of core thermal limits with subsequent modification to the LPRM ATS based on the new reactor operating level. Execution of these rapid computations does not exceed 3 minutes and yields ATS values that are conservative with respect to the more accurate periodic power distribution calculation, which requires up to 20 minutes to execute. This range of surveillance and the rapidity with which the computer responds to reactor changes permit more rapid power maneuvering with the assurance that thermal operating limits will not be exceeded.

Flux level and position data from the Traversing In-core Probe (TIP) equipment are read into the computer. The computer evaluates the data and determines gain adjustment factors by which the LPRM amplifier gains can be altered to compensate for exposure-induced sensitivity loss. The LPRM amplifier gains are not to be physically altered except immediately prior to a whole core calibration using the TIP system. The gain adjustment factor computations help to indicate to the operator when such a calibration procedure is necessary.

Using the power distribution data, a distribution of fuel exposure increments from the time of previous power distribution calculation is determined and is used to update the distribution of cumulative fuel exposure. Each fuel bundle is identified by batch and location, and its exposure is stored for each of the axial segments used in the power distribution calculation. These data are printed out on operator demand.

Exposure increments are determined periodically for each quarterlength section of each control rod. The corresponding cumulative exposure totals are periodically updated and printed out on operator demand.

The exposure increment of each local power range monitor is determined periodicallly and is used to update both the cumulative ion chamber exposures and the correction factors for exposure-dependent LPRM sensitivity loss. These data are printed out on operator demand.

The computer provides on-line capability to determine monthly and on-demand isotopic composition for each one-quarter-length section of each fuel bundle in the core. This evaluation consists of computing the weight of one neptunium, three uranium, and five plutonium isotopes as well as the total uranium and total plutonium content. The isotopic composition is calculated for each one-quarter length of each fuel bundle and summed accordingly by bundles and batches. The method of analysis consists of relating the computed fuel exposure and average void fraction for the fuel to computer stored isotopic characteristics applicable to the specific fuel type. The output is on punched cards which can be used off-line in combination with the card reader and the line printer to obtain a printed record. The cards also permit flexibility in transmitting the data to other offline devices for additional data processing.

The processor is capable of checking each analog input variable against two types of limits for alarm purposes:

1. Process alarm limits are determined by the computer during computation or as preprogrammed at some fixed value by the operator, and

2. A reasonableness limit of the analog input signal level programmed.

The alarming sequence consists of an audible buzzer, a console alarm light, a typewriter message and CRT message for the variables that exceed process alarm limits. An acknowledge pushbutton is provided to reset the buzzer to normal. A variable that is returning to normal is signified by a typewritten message.

The processor provides the capability to alarm the main control room annunciator system in the event of abnormal PCS operation. Abnormal conditions for alarm include, but are not limited to, loss of power, over temperature, parity alarm, stall alarm, and selected program-driven PCS contacts.

The processor measures and stores the values of ten analog variables at 5-second intervals to provide a 10-minute past history of data. An on-demand request permits the operator to initiate typing of this 10-minute segment of data and terminate the log printout when desired.

The processor automatically prints the values of these 10 analog variables for the 5-minute periods before and after a reactor scram.

A digital trend capability is provided for logging the values of as many as 10 nuclear boiler system operator-selected analog inputs and calculations variables. The periodicity of the log is limited to a normal selection of intervals, which can be adjusted as desired by program control.

A status alarm function scans digital inputs once each second and provides a printed record of system alarms. The record includes point description and time of occurrence.

The alarm logs required by the asssociated process programs are typed by the alarm typewriter. Alarm printouts inform the operator of computer system malfunctions, system operation exceeding acceptable limits, and potentially unreasonable, offnormal, or failed input sensors.

7.7.1.10 Rod Worth Minimizer Function

a. System Function

The rod worth minimizer (RWM) functions to assist and supplement the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low power level control rod procedures. The Rod Worth Minimizer portion of the computer prevents the operator from establishing control rod patterns that are not consistent with pre-stored RWM sequences by initiating appropriate rod withdrawal block, and rod insert block interlock signals to the reactor manual control system rod block circuitry (see Figure 7.7-4). The RWM sequences stored in the computer memory are based on control rod withdrawal procedures designed to limit (and thereby minimize) individual control rod worths to acceptable levels as determined by the design basis rod drop accident.

b. System Operation

The RWM function does not interfere with normal reactor operation, and in the event of a failure does not itself cause rod patterns to be established. The RWM will not function upon loss of offsite power. The RWM function can be bypassed and its block function can be disabled only by specific procedural control initiated by the operator.

The following operator and sensor inputs are utilized by the RWM:

1. Rod Test. In selecting this input, the operator is permitted to withdraw and reinsert one control rod at a time while all

other control rods are maintained in the fully inserted position.

2. Normal/Bypass Mode. A keylock switch permits the operator to apply permissives to RWM rod block functions at any time during plant operation.

3. System Start/Reset. This input is initiated by the operator to start or restart the RWM programs and system at any time during plant operation.

4. Control Rod Select. Binary coded identification of the control rod selected by the operator.

5. Control Rod Position. Binary coded identification of the selected control rod position.

6. Control Rod Drive Selected and Driving. The RWM program utilizes this input as a logic diagnostic verification of the integrity of the rod select input data.

7. Control Rod Drift. The RWM program recognizes a position change of any control rod using the control rod drift signal input.

8. Reactor Power Level. Flow signals are used to implement two digital inputs to permit program control of the RWM function. These two inputs, the low power set point and the low power alarm set point, are used to disable the RWM function at power levels above the intended service range of the RWM function.

9. Permissive Echoes. Rod withdraw and rod insert permissive echo inputs are utilized by the RWM as a verification "echo" feedback to the system hardware to assure proper response of an RWM output.

10. Diagnostic Inputs. The RWM utilizes selected diagnostic inputs, to verify the integrity and performance of the processor.

Isolated contact outputs to plant instrumentation provide RWM block functions to the Reactor Manual Control System to permit or inhibit withdrawal or insertion of a control rod. These actions do not affect any normal instrumentation displays associated with the selection of a control rod.

The RWM control panel provides the following indication:

1. Insert Error. Control rod coordinate identification for as many as two insert errors.

2. Withdrawal Error. Control rod coordinate identification for one withdrawal error.

3. Latch Group Identification of the RWM sequence group number currently enforced by the computer.

4. Rod Test Select Indicates that the rod test function test selected by the operator was honored by the RWM Program.

5. RWM Bypass-Indicates that the RWM is manually bypassed.

6. Select Error-Indicates a control rod selection error.

7. Rod Blocks-Indication that a withdrawal block, or insertion block is in effect for all control rods.

8. Out of Sequence Indication that the actual rod patterns is out of sequence while the RWM system is in process of determining which group to latch. This is done for restarts, power decreasing below the low power alarm point (LPAP), rod drift, and system diagnosis.

    a) Below LPSP. Indication that the reactor core power is below the low power set point (LPSP).

    b) Below LPAP. Indication that the reactor core power is below the low power alarm point (LPAP).

    c) Rod Drift. Indication that a rod drift condition is detected.

7.7.1.11 Rod Sequence Control System (RSCS) - Instrumentation and Controls

a. System Function

The purpose of the rod sequence control system (RSCS) is to reduce the consequences of the postulated rod drop accident to an acceptable level. This is accomplished by constraining control rod movement to predetermined patterns and sequences.

b. System Operation

The rod sequence control system (RSCS) is a subsystem of the reactor manual control system (RMCS). 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 as follows:

1. rod pattern controller,

2. substitute position generator,

3. operator display,

4. tester, and

5. bypassed rod identifier.

The RSCS receives inputs from RPIS, from RDCS, from the plant and from the operator. From the RPIS, the RSCS receives rod position data including numerical position, full-in, full-out, and indication of faulty position probe information. The RDCS provides the identification of the selected rod, rod motion commands (insert or withdraw), and a signal indicating when a rod is being driven. The RSCS also receives first-stage turbine pressure, which is converted to identify the operating power level of the plant. Display selections from the operator (which determine what information is displayed), rod sequence selection, request to enter substitute position data for a rod, test initiate commands, and the identity of a bypassed rod (this is under keylock control with strict technical specification control on the number and combinations of rods that can be bypassed).

The RSCS provides outputs to the RDCS and to the plant operator. To the RDCS it provides three interlocks, one for rod insert permission, one for rod withdrawal permission, and one for continuous rod movement (more than one notch permission. For the plant operator, the RSCS indicates for rod insert block, rod withdraw block, sequence selected, and power level (below or above the low-power setpoint). In addition, a core map shaped display with two indicators per rod is provided. One of these indicators will show, at the operator's option, either the rods in the active group that can be inserted or those that can be withdrawn. The second indicator, independently controlled, can show the rods that are bypassed, and which rods belong to the active group or which rods are full-in (the "active" group is the group within the selected sequence that contains the selected rod). In addition to the outputs described, the RSCS has numerous displays on the hardware for maintenance and surveillance assistance.

The following is a discussion of the system equipment:

1. Rod Pattern Controller (RPC)

The RPC is a hardwired controller that contains all of the logic and pattern data necessary to determine permitted rod motion, either insert of withdrawal, for all rods. The RPC maintains a complete set of rod positions which is updated continuously from the rod position information system. Based on this input data and the wired-in logic rules, the RPC will continuously generate insert and withdrawal permission data. From the selected rod identification input it will generate insert and withdrawal inhibit outputs for the selected rod.

In the event of missing position data for a given rod (identified by a fault code at the input), the RPC will apply rod movement inhibits for all rods. This may be cleared in one of two ways. First, if it corresponds to a one-notch position only, a substitute position may be entered (from the substitute position generator). The rod with the fault may then be driven one additional notch. If the fault remains, the rod may then be bypassed, provided all of the technical specification criteria are met, by setting up bypass controls behind a keylocked cover. A bypassed rod may be left where it is or driven in. It may not be driven out.

2. Substitute Position Generator

The substitute position generator (SPG) is a set of electronics that uses the position data from RPIS, the identity of the selected rod, the rod driving signals to generate the next expected position of the rod being driven.

If the rod has valid data at the next position, those data are read and the projected data are dumped. If the position data are missing at the next rod position, the projected data can be inserted by the operator into the RPC to be used in place of the missing rod position data. At this point, the rod movement is restricted to the notch drive mode. If data are still missing after the next rod movement, a rod block will be applied and that rod must be bypassed in order to make further rod movements.

3. Operator Display

The operator's display is a small front panel mounted unit that includes readouts and controls necessary for normal operation of the RSCS. It includes individual indicators for rod inhibits, modes, and a numeric display of the position that will be substituted when a substitution is called for. In addition to these system status indicators, individual rod status indicators are available as described above.

The operator's display also includes the control switches to select rod displays RSCS sequence, and to enable substitute position (projected data) entry to the RPC.

4. Tester

The tester is a built-in test fixture which, when manually started, will generate input test signals and monitor the rod inhibit outputs. The tester will go through a sufficient set of input combinations and sequences to exercise all of the critical logic elements in the RSCS. This will be an off-line test, but will not require any disconnecting of the hardware.

5. Bypassed Rod Identifier

The bypassed rod identifier is a set of logic hardware with input switches on a circuit card which can be set up to identify a specific rod (one set of switches per bypassed rod) to be bypassed with respect to the RSCS. There will be 20 cards maximum capacity meaning 20 rods maximum limited by hardware. The actual limit under any given operating condition is controlled by the technical specifications.

Refer to 4.3.2.6, "Control Rod Patterns and Reactivity Worths," for a description of the permissible control rod withdrawal sequences.

All signals leaving or entering the RSCS and/or RPIS will be buffered to minimize the chance of an event outside these subsystems feeding back into and disabling the RSCS capability to apply rod blocks.

This system is designed to meet the environmental conditions described in Table 3.11-4 for the control room.

The RSCS is designed primarily to mitigate the consequences of the postulated rod drop accident, which analysis shows to be of no concern at power levels in excess of 20% rated thermal power. Mitigation is achieved by constraining control rod movements by the operator to pre-determined patterns and sequences which ensure that control rods of high worth are not obtained below the 20% power level. The design criterion is that any potential rod drop accident should not result in fuel rod enthalpies in excess of 280 cal/g. Over the operating ranges of power level and fuel exposure, the resultant fuel rod enthalpy is a function of several parameters, of which control rod worth is the most significant and controllable.

To meet the 20% rated power level requirement, the RSCS is required to be in operation during reactor startup and shutdown between 0 and a nominal 30% rated thermal power. The RSCS function is supported by the redundant action of the Rod Worth Minimizer (RWM), which is programmed to permit only the same rod patterns and sequences as prescribed for the RSCS. While the startup or shutdown of the reactor may continue without the RWM when a second licensed operator is present to check rod movements, the RSCS must be operable at all times over the prescribed power range (0-30%). For example, the reactor may not be started up (rods pulled) if the RSCS is inoperable, and if the RSCS fails during startup or shutdown (0-30%) power, rods may be moved only by scramming the reactor.

Control rod drives may have to be declared inoperable for various reasons as defined in the technical specifications, and this in turn leads to restrictions on the position and use of such rods. The RSCS is designed to be compatible with these restrictions and has facilities for the bypassing of inoperable control rods.

7.7.1.12 Loose Parts Detection System (LPDS) - Instrumentation and Controls

a. LPDS Function

The loose parts detection system monitors the reactor vessel for the presence of internal loose parts. Internal movement of components or core vibration of internals is not required to be monitored by this equipment.

b. LPDS Operation

The loose parts detection sensors are mounted on the exterior of the primary coolant system and located at natural collection points where loose parts will most likely impact. During startup and normal plant operation the LPDS is on line to provide visual and audio information to the operator. Also, this data is recorded on magnetic tape for detailed analysis at a later date and to provide a startup base line signature of various pumps and valves to be used for comparative analysis. LPDS schematic arrangements are shown in Figure 7.7-15 (LPDS P&ID).

7.7.1.13 Refueling Interlocks - Instrumentation and Controls

a. Refueling Interlocks Function

The purpose of the refueling interlocks is to restrict the movement of control rods and the operation of refueling equipment. This rein-forces operational procedures that prevent the reactor from becoming critical during refueling operations.

b. Refueling Interlocks Operation

The refueling interlocks circuitry senses the condition of the refueling equipment and the control rods to prevent the movement of the refueling equipment or withdrawal of control rods (rod block). Redundant circuitry is provided to sense the following conditions:

1. All rods inserted
2. Refueling platform positioned near or over the core
3. Refueling platform hoists fuel-loaded (fuel grapple, frame-mounted hoist, trolley-mounted hoist)
4. Service platform hoist fuel-loaded, and
5. Reactor Mode Switch in "Refuel" position.

The indicated conditions are combined in logic circuits to satisfy all restrictions on refueling equipment operations and control rod movement (Figure 7.7-3). A two-channel circuit indicates that all rods are in. The rod-in condition for each rod is established by the closure of a magnetically operated reed switch in the rod position indicator probe. The rod-in switch must be closed for each rod before the all-rods-in signal is generated. Both channels must indicate "all-rods-in" to allow refueling equipment to be used.

During refueling operations, no more than one control rod is permitted to be withdrawn; this is enforced by a redundant logic circuit that uses the all-rods-in signal and a rod selection signal from the reactor manual control system to prevent the selection of a second rod for movement with any other rod not fully inserted. Control rod withdrawal is prevented by comparison between the A and B portions of the RMCS for rod position with a subsequent rod withdrawal block if necessary. The simultaneous selection of two control rods is prevented by the interconnection arrangement of the select pushbuttons. With the mode switch in the REFUEL position, the circuitry prevents the withdrawal of more than one control rod and the movement of the loaded refueling platform over the core with any control rod withdrawn.

Operation of refueling equipment is prevented by interrupting the power supply to the equipment. The refueling platform is provided with two mechanical switches attached to the platform, which are tripped open by a long, stationary ramp mounted adjacent to the platform rail. The switches open before the platform or any of its hoists are physically located over the reactor vessel to indicate the approach of the platform toward its position over the core.

Load cell readout is provided for all hoists. Indicators display given hoist loads directly to the operator. Load sensing is by hydraulic load cells that use demineralized water as the operating fluid. Associated interlock and load functions are performed by pressure switches that sense the pressure generated by the hydraulic load cells.

The three hoists on the refueling platform and the hoist on the service platform are provided with switches that open when the hoists are fuel loaded. The switches open at a load weight that is lighter than that of a single fuel assembly. This indicates when fuel is loaded on any hoist.

A bypass for the service platform hoist load interlock is provided. When the service platform is no longer needed, its power plug is removed. This de-energizes the power supply to the hoist. The platform can then be moved away from the core.

De-energizing the hoist power supply opens the hoist load switches and gives a false indication that the hoist is loaded. This indication prevents control rod withdrawal with the mode switch in the STARTUP or REFUEL positions. A bypass plug allows control rod movement in this situation. The bypass plug is physically arranged to prevent the connection of the service platform power plug unless the bypass plug is removed.

The rod block interlocks and refueling platform interlocks provide two independent levels of interlock action. The interlocks which restrict operation of the platform hoist and grapple provide a third level of interlock action since they would be required only after a failure of a rod block and refueling platform interlock.

In the refueling mode, the control room operator has an indicator light for "Refueling Mode Select Permissive" whenever all control rods are fully inserted. He can compare this indication with control rod position data from the computer as well as control rod in-out status on the full core status display. Whenever a control rod withdrawal block situtation occurs, the operator receives annunciation and computer logs of the rod block. The operator can compare these outputs with the status of the variable providing the rod block condition. Both channels of the control rod withdrawal interlocks must agree that permissive conditions exist in order to move control rods; otherwise, a control rod withdrawal block occurs. Failure of one channel may initiate a rod withdrawal block, and will not prevent application of a valid control rod withdrawal block from the remaining operable channel (see Table 7.7-3).

In terms of refueling platform interlocks, the platform operator has analog type readout indicators for the platform x-y position relative to the reactor core.

The position of the grapple is shown in a digital indicator immediately below the platform position indicators. Analog load cell indications of hoist loads are given for each hoist by locally mounted indicators. Individual pushbutton and rotary control switches are provided for local control of the platform and its hoists. The platform operator can immediately determine whether the platform and hoists are responding to his local instructions, and can, in conjunction with the control room operator, verify proper operation of each of the three categories of interlocks listed previously.

7.7.1.14 Design Differences

Refer to Table 7.7-2 for a list of Instrumentation and Control system designs and their similarity to designs of other nuclear power plants.

7.7.2 ANALYSIS

Refer to the safety evaluations in Chapter 15 and Appendix A of Chapter 15. Chapter 15 shows that the systems described in 7.7 are not utilized to provide any design basis accident safety function. Safety functions are provided by other systems.

Chapter 15 also evaluates all credible control system failure modes, the effects of those failures on plant functions, and the response of various safety-related systems to those failures.

The major plant control systems described above have no direct interface with any safety-related systems and, thus, control system failures, other than those described in Chapter 15, have no effect on the safety-related systems.

TABLE 7.7-1

DESIGN AND SUPPLY RESPONSIBILITY OF PLANT CONTROL SYSTEMS

	GE DESIGN	GE SUPPLY	B&R DESIGN	OTHERS SUPPLY
Reactor Vessel Instrumentation	X	X		
Reactor Manual Control System	X	X		
Recirculation Flow Control System	X	X		
Feedwater Control System	X	X	X	X
Press. Regulator & Turbine Generator System			X	X
Neutron Monitoring System				
SRM	X	X		
RBM	X	X		
TIP	X	X		
Process Comp. & RWM	X	X		
Rod Sequence Control System	X	X		
Loose Parts Detection System			X	X
Refueling Interlocks	X	X		

TABLE 7.7-2

SIMILARITY TO LICENSED REACTORS

Instrumentation and Controls (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design
(1) Neutron Monitoring System (TIP, SRM, RBM)	LaSalle	Identical
(2) Refueling Interlocks	LaSalle	Identical
(3) Reactor Manual Control System	Zimmer-1	Identical
(4) Reactor Vessel - Instrumentation	Zimmer-1	Identical
(5) Recirculation Flow-Control System	Zimmer-1	Identical
(6) Feedwater Control System	Zimmer-1	Identical
(7) Pressure Regulator and Turbine-Generator System	Zimmer-1	Identical
(8) Rod Sequence Control System	Zimmer-1	Identical
(9) Refueling Interlocks	Zimmer-1	Identical
(10) Process Computer (RWM)	Vermont Yankee & subsequent plants	See Note 1
(11) Loose Parts Detection System	None	--

Note 1:

A General Electric Model 4010 process computer is used on this plant instead of a model 4020, as used on Vermont Yankee. This difference in computer equipment is insignificant.

TABLE 7.7-3

REFUELING INTERLOCK EFFECTIVENESS

SITUATION	REFUELING PLATFORM POSITION	REFUELING TMH*	PLATFORM FMH*	HOISTS FG*	SERVICE PLATFORM HOIST	CONTROL RODS	MODE SWITCH	ATTEMPTS	RESULT
1.	Not near core	UL*	UL*	UL*	UL*	All rods in	Refuel	Move refueling platform over core	No restrictions
2.	Not near core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod
3.	Not near core	UL	UL	UL	UL	One rod withdrawn	Refuel	Move refueling platform over core	No restrictions
4.	Not near core	Any hoist loaded or FG not fully up			UL	One rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core
5.	Not near core	UL	UL	UL	UL	One rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core
6.	Over core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod
7.	Over core	Any hoist loaded or FG not fully up				All rods in	Refuel	Withdraw rods	Rod block
8.	Not near core	UL	UL	UL	L*	All rods in	Refuel	Withdraw rods	Rod block
9.	Not near core	UL	UL	UL	L	All rods in	Refuel	Operate serviced platform hoist	No restrictions
10.	Not near core	UL	UL	UL	L	One rod withdrawn	Refuel	Operate service platform hoist	Hoist operation prevented
11.	Not near core	UL	UL	UL	UL	All rods in	Startup	Move refueling platform over core	Platform stopped before over core
12.	Not near core	UL	UL	UL	L	All rods in	Startup	Operate service platform hoist	No restictions
13.	Not near core	UL	UL	UL	L	One rod withdrawn	Startup	Operate serviced platform hoist	Hoist operation prevented
14.	Not near core	UL	UL	UL	L	All rods in	Startup	Withdraw rods	Rod block
15.	Not near core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	No restrictions
16.	Over core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	Rod block
17.	Any		Any condition		Any condition	Any condition, reactor not at power	Startup	Turn mode switch to RUN	Scram

* TMH - Trolley Mounted Hoist   FMH - Frame Mounted Hoist   FG - Fuel Grapple   UL - Unloaded   L - Fuel Loaded

TABLE 7.7-4

SRM SYSTEM TRIPS

TRIP FUNCTION	NORMAL SETPOINT	TRIP ACTION
SRM upscale (high) or	10^5 c/s	Rod block, amber light display annunciator.
SRM instrument inoperative	(See Note)	Rod block, amber light display annunciator.
Detector Retraction	100 c/s	Bypass detector full-in limit switch when above present limit, annunciator, green light display, green light display, rod block when below preset limit with IRM range switch on first two ranges.
SRM period	50 sec.	Annunciator, amber light display.
SRM downscale	3 c/s	Rod block, annunciator, white light display.
SRM bypassed		White light display.

Note: SRM is inoperative if module interlock chain is broken. OPERATE-CALIBRATE switch is not in OPERATE position or detector polarizing voltage is below 300V.

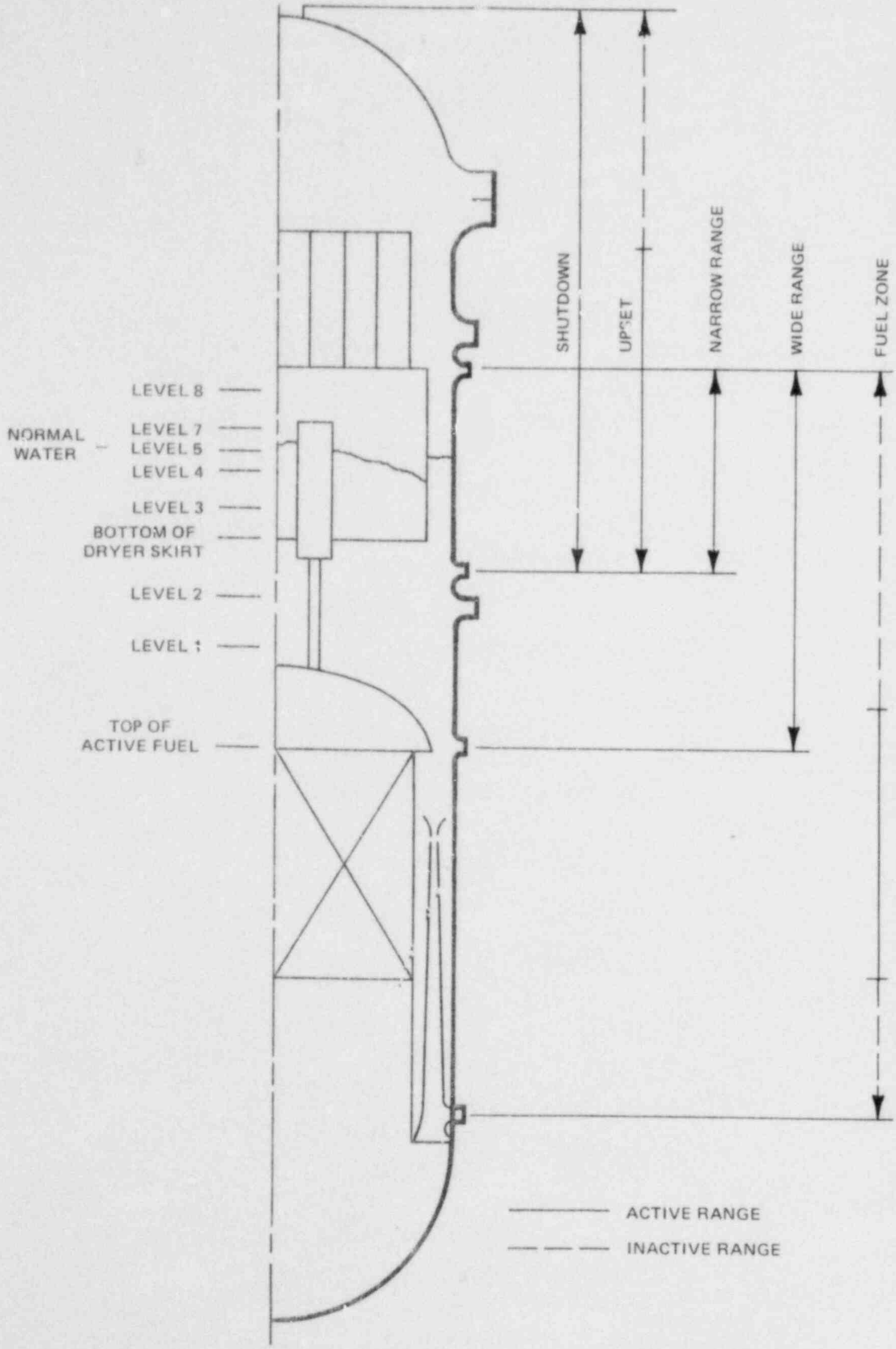
TABLE 7.7-5

RBM SYSTEM TRIPS

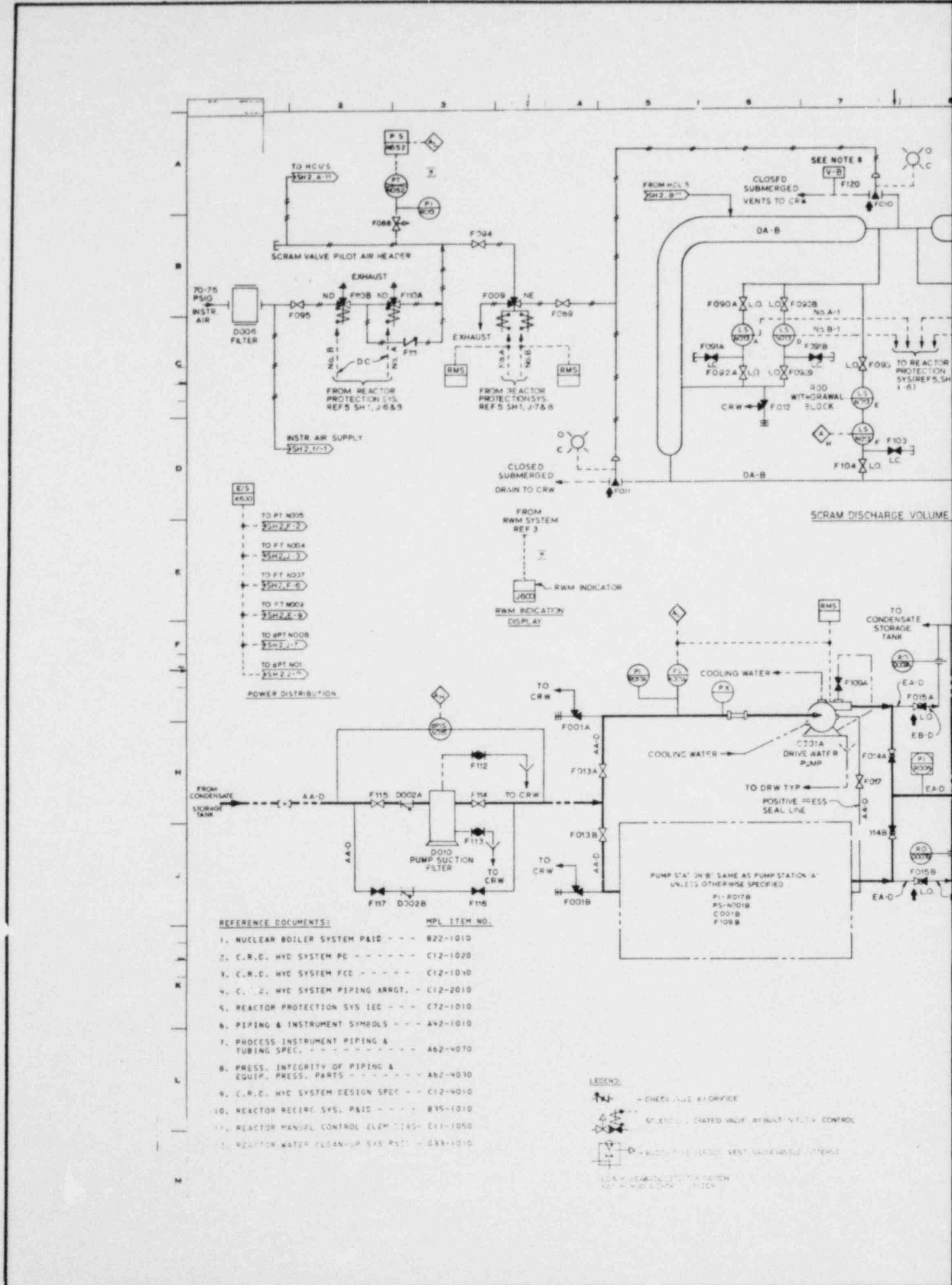
TRIP FUNCTION	NOMINAL SETPOINT	TRIP ACTION
RBM upscale (high)	R(.66 Flow + 41%) normal R(.66 Flow + 33%) intermediate R(.66 Flow + 25%) low	Rod block, annunciator, amber light display
RBM inoperative	(See note)	Rod block, annunciator, amber light display
RBM downscale	5/125 Full Scale	Rod block, annunciator, white light display
RMB bypassed	Manual switch or peripheral rod selected or APRM reference below 30%	White light display

Note: RBM is inoperative if module interlock chain is broken, OPERATE-CALIBRATE switch is not in OPERATE position, less than 50% of available LPRM signals are above 3% threshold, or internal logic self-test circuits indicate trouble.

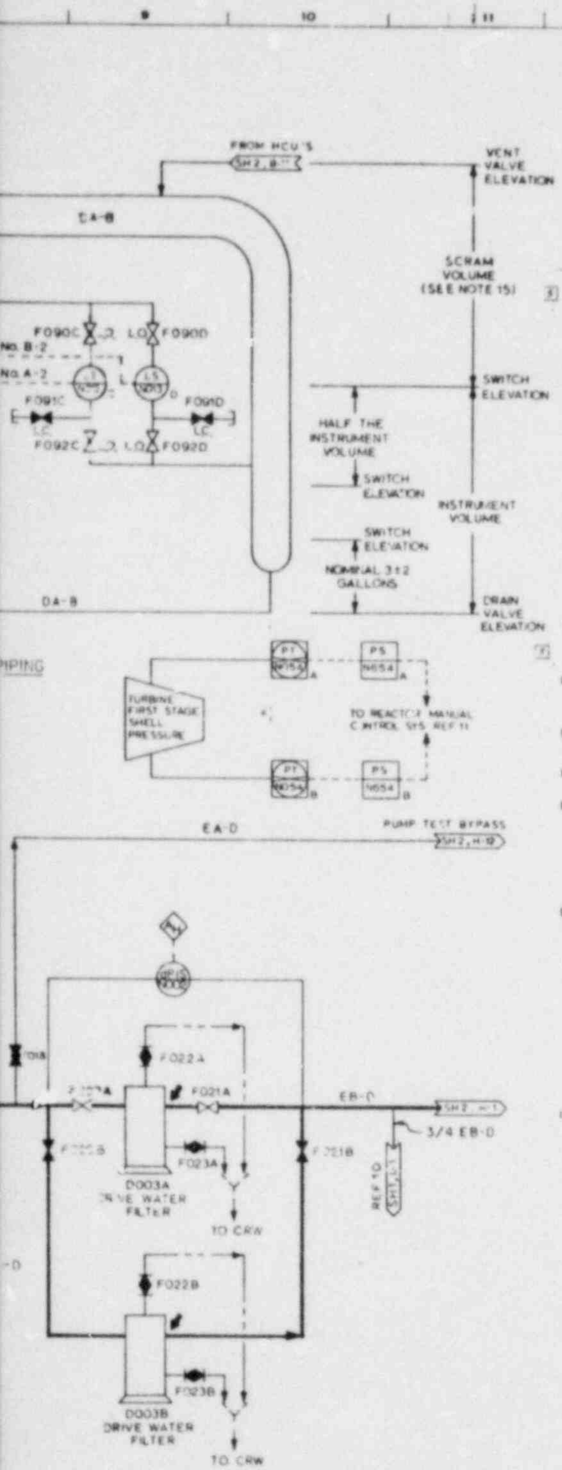
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	WATER LEVEL RANGE DEFINITION.	FIGURE 7.7-1



TO HCU'S
SCRAM VALVE PILOT AIR HEADER
EXHAUST
70-75 PSIG INSTR. AIR
D006 FILTER
F095
F088
F094
F009
F089
F110A
F011
EXHAUST
RMS
FROM REACTOR PROTECTION SYS. REF 5 SH 1, J-6&9
FROM REACTOR PROTECTION SYS. REF 5 SH1, J-7&8
INSTR. AIR SUPPLY
FROM HCU'S
SEE NOTE 8
CLOSED SUBMERGED VENTS TO CRW
F120
F010
DA-B
F090A
F090B
F091A
F091B
F092A
F092B
F095
F012
CRW
TO REACTOR PROTECTION SYS(REF 5,SH 1-6)
ROD WITHDRAWAL BLOCK
F103
F104
CLOSED SUBMERGED DRAIN TO CRW
SCRAM DISCHARGE VOLUME
FROM RWM SYSTEM REF 3
RWM INDICATOR
RWM INDICATION DISPLAY
TO PT N005
TO PT N004
TO PT N007
TO PT N009
TO dPT N008
TO dPT N01
POWER DISTRIBUTION
TO CRW
F001A
COOLING WATER
TO CONDENSATE STORAGE TANK
F109A
EA-D
F015A
EB-D
F014A
F017
TO DRW TYP
POSITIVE PRESS SEAL LINE
F013A
F013B
F001B
AA-D
F112
F114
F115
D002A
TO CRW
FROM CONDENSATE STORAGE TANK
F113
D010 PUMP SUCTION FILTER
F117
D002B
F116
C001A DRIVE WATER PUMP
PUMP STATION 'B' SAME AS PUMP STATION 'A' UNLESS OTHERWISE SPECIFIED
PI-R017B
PS-N001B
C001B
F109B
F015B
REFERENCE DOCUMENTS:
MPL ITEM NO.
1. NUCLEAR BOILER SYSTEM P&ID - - - B22-1010
2. C.R.D. HYD SYSTEM PC - - - - - - C12-1020
3. C.R.D. HYD SYSTEM FCD - - - - - C12-1030
5. REACTOR PROTECTION SYS IED - - - C72-1010
6. PIPING & INSTRUMENT SYMBOLS - - - A42-1010
7. PROCESS INSTRUMENT PIPING & TUBING SPEC. - - - - - - - - - - A62-4070
8. PRESS. INTEGRITY OF PIPING & EQUIP. PRESS. PARTS - - - - - - A62-4030
9. C.R.D. HYD SYSTEM DESIGN SPEC - - C12-4010
10. REACTOR RECIRC SYS. P&ID - - - - B35-1010
LEGEND:



(C12 1010)

NOTES:

1. VALVE F007A-A CLOSES ON DRIVE INSERT SIGNAL. VALVE F007A-B CLOSES ON DRIVE WITHDRAWL SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING. (SH 2,K-9)

2.

3. STABILIZING VALVE F007B IS AN ALTERNATE FOR STABILIZING VALVE F007A. (SH2,L & M-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. (SH2,K & L-8)

7. AVAILABLE FOR TEMPORARY CONNECTION FOR INSTRUMENT FLUSHING. NO PERMANENT PIPING CONNECTIONS TO BE MADE TO THIS VALVE. (SH2,E-9)

8. VACUUM BREAKER TO . ON HIGH POINT OF VENT LINE.

9. EXCEPT AT POINTS OF CONNECTION WITH BWRSD SUPPLIED EQUIPMENT OR PIPING, THE PIPING DESIGNER SHALL SIZE PIPES IN CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.

10. FOR LOCATION & IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET (C12-3050).

11. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NUMBER (C12) UNLESS OTHERWISE NOTED.

12. CRD NITROGEN & AIR LINES SHALL BE OF A NON-CORRODING MATERIAL.

13. THESE VALVES MUST BE OPEN FOR DRIVE MOVEMENT.

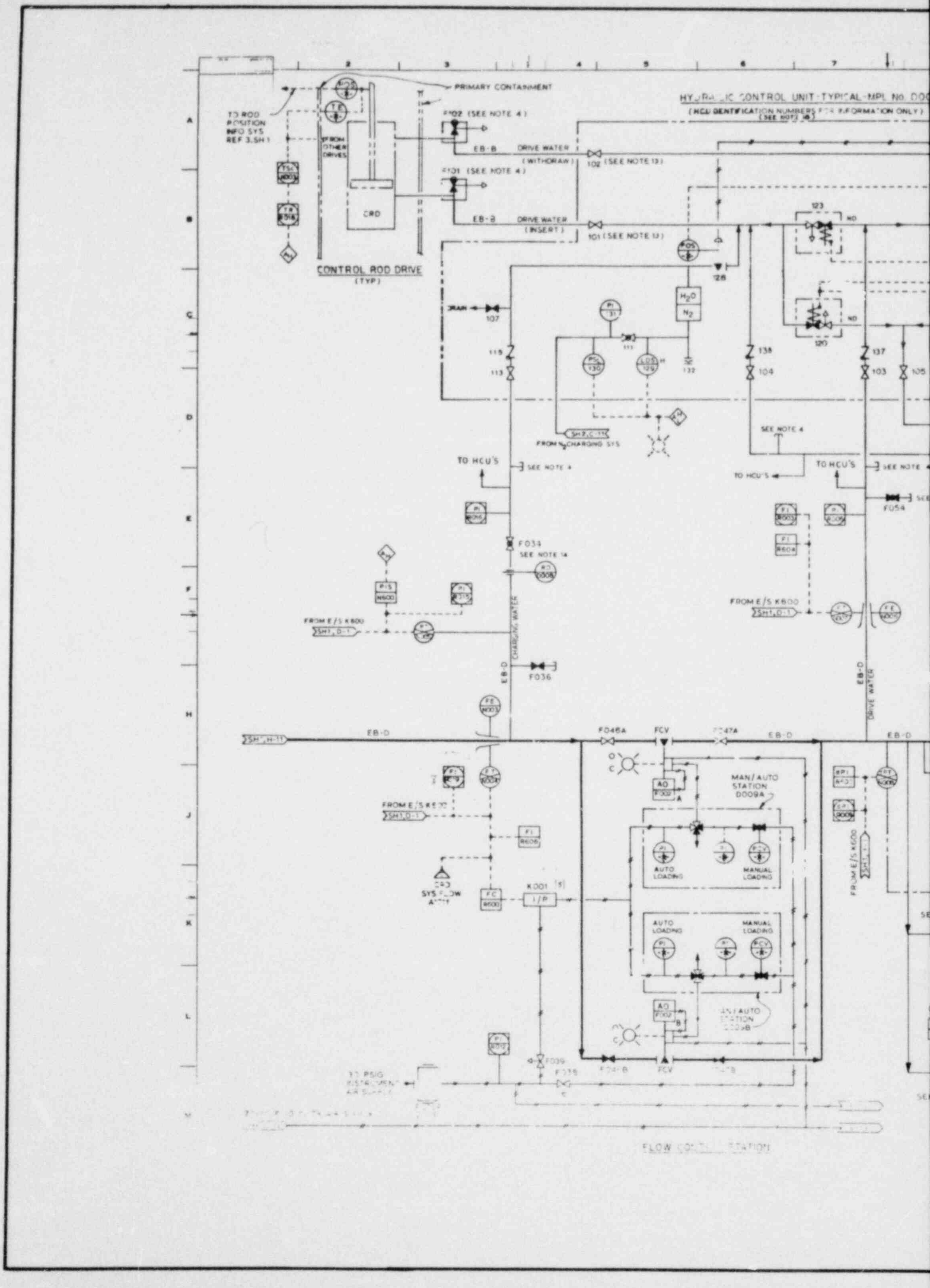
14. MULTIPLE ORIFICES CONNECTED IN SERIES AS SHOWN IN PURCHASE PART DRAWING OF ORIFICE D008. THE PRESSURE DROP ACROSS EACH ORIFICE IS 250 PSI AT PUMP RUN-OUT CONDITION. SEE MPL FOR THE QUANTITIES OF ORIFICES. VALVE F034 SUPPLEMENTS THE ORIFICES D008 FOR THE REQUIRED PRESSURE DROP.

15. THE SCRAM DISCHARGE VOLUME ARRANGEMENT SHOWN IS FOR REFERENCE ONLY. SEE CRD DESIGN SPECIFICATION FOR REQUIREMENT.

16. SPRING LOADED PISTON ACTUATED VALVE HELD OPEN BY AIR PRESSURE DURING NORMAL OPERATION.

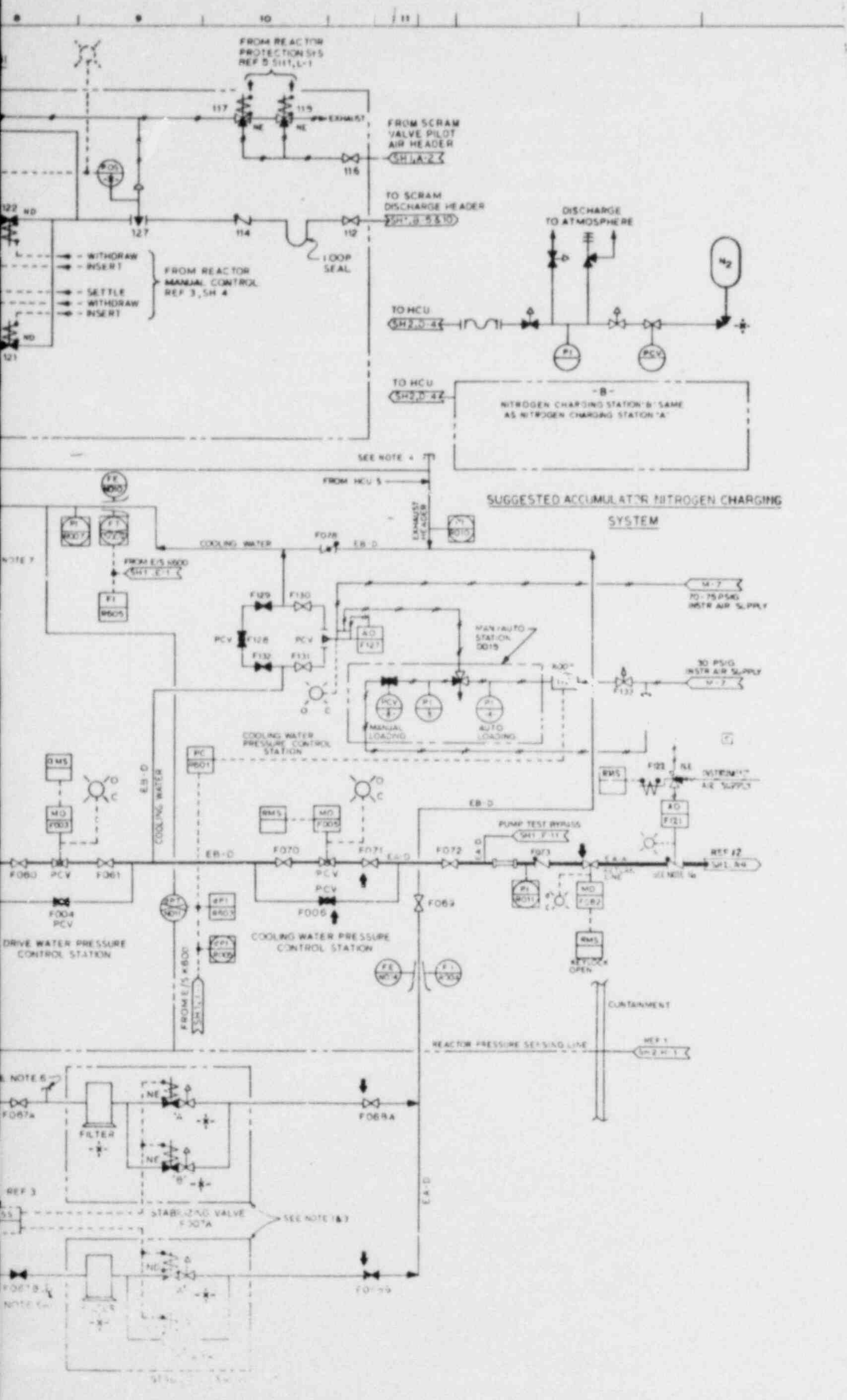
18. PIPING QUALITY CLASS EXTENDS TO CONNECTIONS WITH HCU. HCU DIAGRAM IS SHOWN FOR INFORMATION ONLY. FOR QUALITY CLASS OF THE HCU SEE GROUP CLASSIFICATION DIAGRAM.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID	FIGURE 7.7-2a



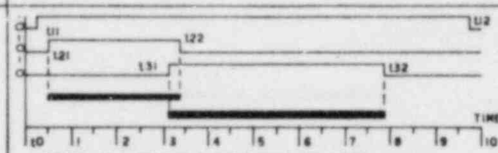
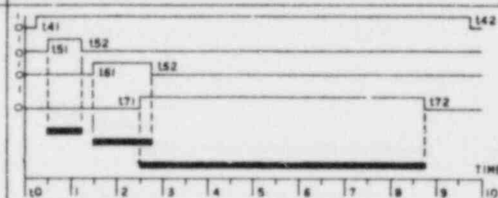
PRIMARY CONTAINMENT
TO ROD POSITION INFO SYS REF 3.SH 1
FROM OTHER DRIVES
CRD
CONTROL ROD DRIVE (TYP)
HYDRAULIC CONTROL UNIT-TYPICAL-MPL No. DO
(HCU IDENTIFICATION NUMBERS FOR INFORMATION ONLY)
F102 (SEE NOTE 4)
F101 (SEE NOTE 4)
EB-B
DRIVE WATER (WITHDRAW)
DRIVE WATER (INSERT)
102 (SEE NOTE 13)
101 (SEE NOTE 13)
DRAIN
107
115
113
FROM N2 CHARGING SYS
TO HCU'S
SEE NOTE 4
F034
SEE NOTE 14
FROM E/S K600
CHARGING WATER
EB-D
F036
DRIVE WATER
F046A
FCV
F047A
MAN/AUTO STATION D009A
AUTO LOADING
MANUAL LOADING
K001
I/P
FC
R600
F039
F038
FCV
FLOW CONTROL STATION

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID	FIGURE 7.7- 2b

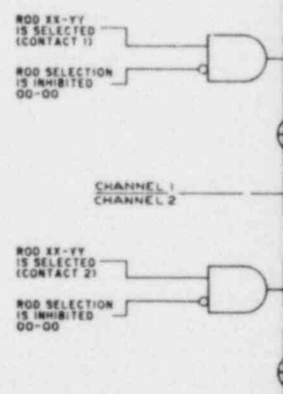
TABLE 1

TIME DEPENDENT INTERMEDIATE VARIABLES		
SYM.	DESCRIPTION	DEFINITION
T	INTERNALLY GENERATED TIME REFERENCE FUNCTION	
T_{IC} T_{II} T_{IS} - -	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 I_{IC} AND I_{IT} AS FOLLOWS: 1. WHEN I_{IC} I_{IT} = 1, THE CYCLE IS ADVANCING WITH TIME. 2. WHEN I_{IC} = 0, THE CYCLE IS RESET TO t_0 3. WHEN I_{IC} $\bar{I}_{IT}$ = 1, THE CYCLE STOPS, BUT DOES NOT RESET.
T_{OC} T_{OI} T_{OO} T_{OS} - - -	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 I_{OC} AND I_{OT} AS FOLLOWS: 1. WHEN I_{OC} I_{OT} = 1, THE CYCLE IS ADVANCING WITH TIME. 2. WHEN I_{OC} = 0, THE CYCLE IS RESET TO t_0. 3. WHEN I_{OC} $\bar{I}_{OT}$ = 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	$t_{21}-t_{11}$	0.10	0.0-0.6	SEC
	ROD INSERT CONTROL	$t_{22}-t_{21}$	2.90	2.5-3.1	SEC
	SWITCHING OVERLAP	$t_{22}-t_{31}$	[illegible]	0.0-1.5	SEC
	SETTLE CONTROL	$t_{32}-t_{22}$	[illegible]	4.2-6.3	SEC
WITHDRAW CYCLE	TIME DELAY TO UNLATCH CONTROL	$t_{51}-t_{41}$	0.10	0-0.4	SEC
	UNLATCH CONTROL	$t_{52}-t_{51}$	0.60	0.5-0.8	SEC
	INTERVAL BETWEEN UNLATCH AND SETTLE CONTROL	$t_{61}-t_{52}$	0.10	0.0-0.15	SEC
	ROD WITHDRAWAL CONTROL	$t_{62}-t_{61}$	1.50	1.3-1.7	SEC
	SWITCH OVERLAP FROM WITHDRAWAL TO SETTLE CONTROL	$t_{62}-t_{71}$	0.10	0.0-1.5	SEC
	SETTLE CONTROL	$t_{72}-t_{62}$	[illegible].00	4.2-6.3	SEC



4822

7 | 8 | 9 | 10 | 11 | 12 | 13 | C.I.

FCF	MPL	APPLICABLE SHEETS 1 2 3 4 5 6 6A 7
239X238 G1 THRU G4,G8,G10	C11-1030	X X X X X X X
239X239 G1 THRU G4,G8,G10	C12-1030	X X X X X X X

TABLE 2

SYSTEM PARAMETER VALUES				
	PARAMETER	VALUE	TOLERANCE	UNITS
	t11	0.30	+ 0.05 / - 0.25	SEC.
	t12	9.10	± 0.02	
INSERT CYCLE	t21	0.40		
	t22	3.30		
	t31	3.20		
	t32	7.80	± 0.02	SEC.
	t41	0.30	+ 0.05 / - 0.25	SEC.
	t42	9.10	± 0.02	
	t51	0.40		
THDRAW CYCLE	t52	1.00		
	t61	1.10		
	t62	2.80		
	t71	2.50		
	t72	8.60	± 0.02	SEC.

NSERT CYCLE

t11 = DELAY UNTIL ROD MOTION BEGINS

t21 = DRIVE IN TIME

t22 = SETTLE TIME

K t22, t = TIME WHEN CONTINUOUS INSERT CAN BE REQUESTED

K t22, t = CYCLE STOP POINT FOR CONTINUOUS INSERT

WITHDRAW CYCLE

t41 = DELAY UNTIL ROD MOTION BEGINS

t51 = DRIVE IN TIME (UNLATCH)

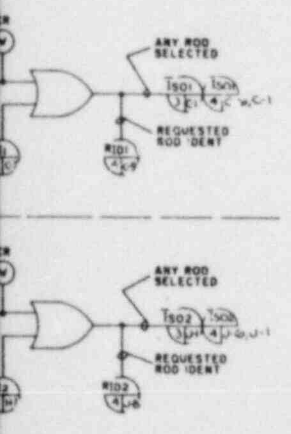
t52 = DELAY AFTER UNLATCH

t61 = DRIVE OUT TIME

t62 = SETTLE TIME

K t62, t = TIME WHEN CONTINUOUS WITHDRAW CAN BE REQUESTED

K t62, t = CYCLE STOP POINT FOR CONTINUOUS WITHDRAW



LEGEND:

- \# = 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:

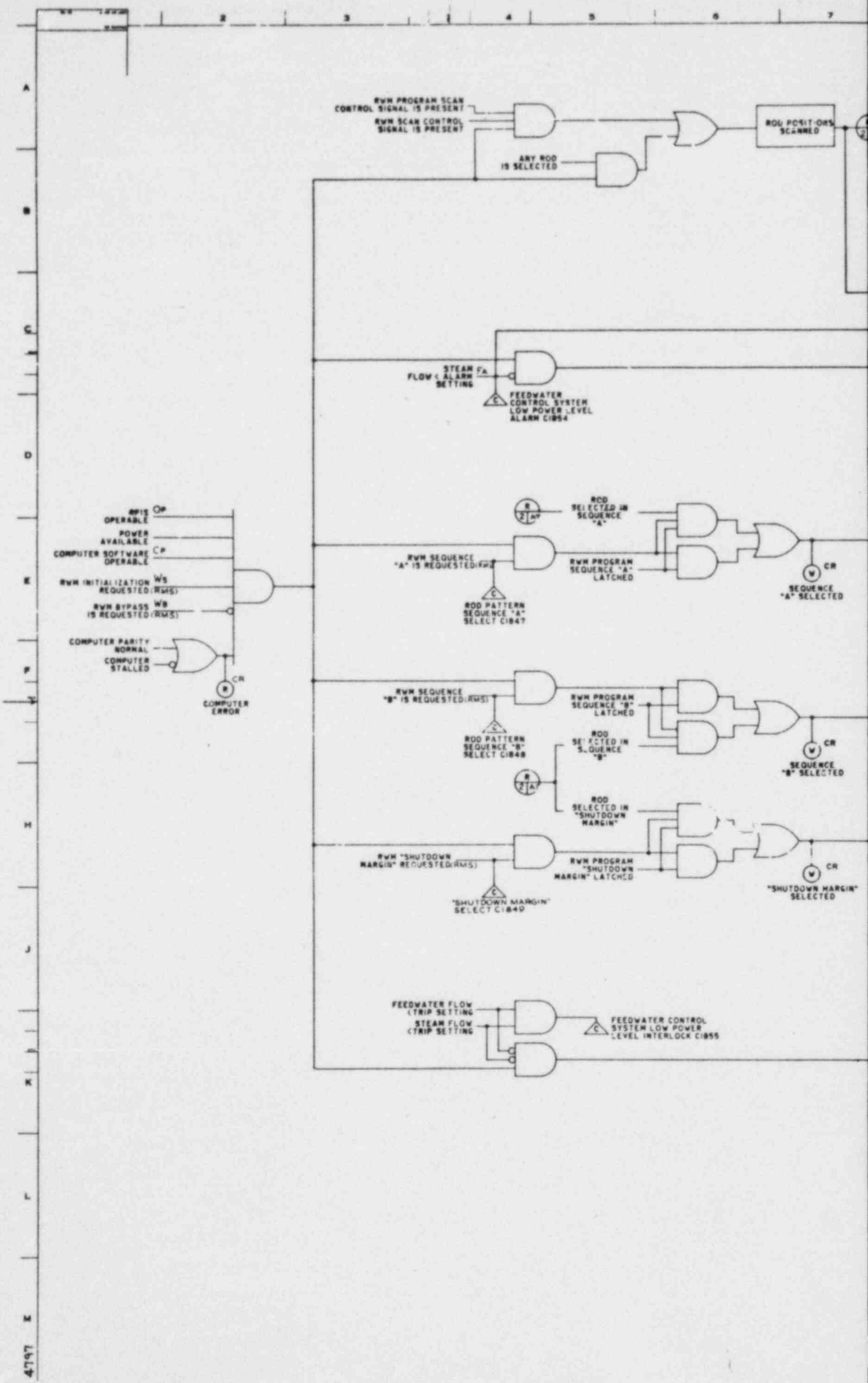
1. EACH CONTROL ROD, AS IT TRAVELS UP (INSERTED) OR DOWN (WITHDRAWN) PASSES A NUMBER OF SWITCHES. THE TOP TWO POSITION SWITCHES ARE CALLED "OVERTRAVEL" AND THE BOTTOM TWO POSITIONS ARE CALLED "WITHDRAWN" (BACKSEAT & DISCONNECT). SWITCHES IN BETWEEN ARE DIVIDED INTO ODD (DRIFT) AND EVEN (LATCH) POSITIONS. AS THE ROD TRAVELS OVER ANY SWITCH AN INDICATING SIGNAL IS ACTUATED. ANY EVEN SWITCH WILL INDICATE NUMERIC POSITION (e.g., 00, 02....48) AND ANY ODD SWITCH WILL INDICATE "ODD".
2. WIRING FROM HCU (SCRAM VALVES AND ACCUMULATOR) TO CONTROL ROOM FOR ANNUNCIATION SHALL BE IN SERIES 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 SERIES CONNECTION FOR ALL HCU'S.
4. EACH ACCUMULATOR FAILURE WILL INITIATE AN ANNUNCIATION (ANNUNCIATOR HORN & FLASHING ANNUNCIATOR WINDOW) AND AN INDIVIDUAL FLASHING INDICATOR (PART OF THE WHOLE CORE DISPLAY). 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 12 FOR DEFINITIONS OF VARIABLES APPEARING ON THIS FCD.
6. A LOGICAL '1' INDICATES A FAILED COMPARISON

SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE ITEM NUMBERS

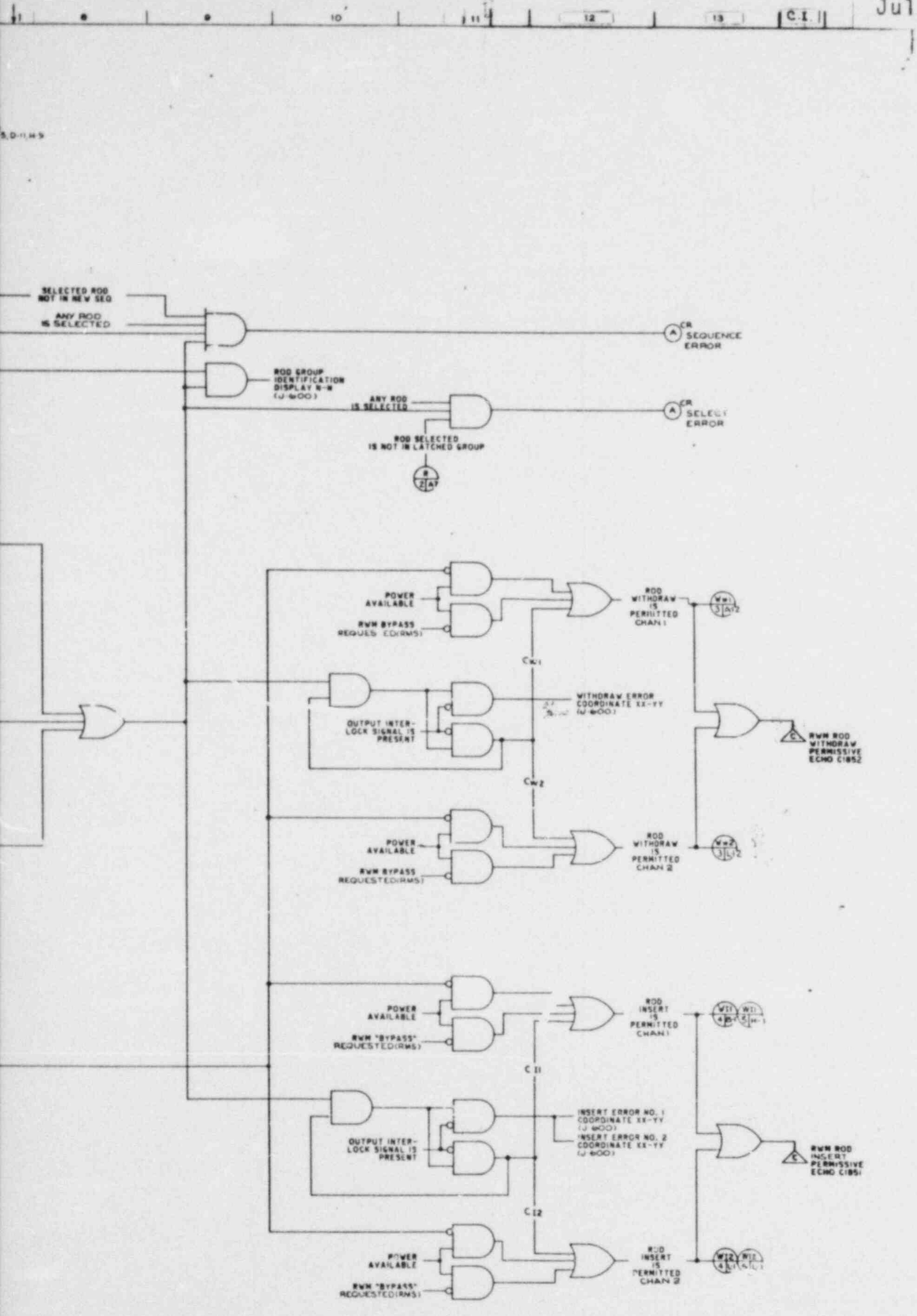
REFERENCE DOCUMENTS:	MPL ITEM NUMBERS
1. CONTROL ROD DRIVE HYDRAULIC SYS. P&ID	C11/C12-1010
2. NEUTRON MONITORING SYS. IED	C51-1010
3. FEEDWATER CONTROL SYS. IED	C33/C34-1010
4. FEEDWATER CONTROL SYS. DESIGN SPEC.	C33/C34-4010
5. CONTROL ROD DRIVE HYDRAULIC SYS. DESIGN SPEC.	C11/C12-4010
6. PROCESS COMPUTER SYS. INPUT/OUTPUT REQUIREMENTS	C91/C92-4010
7. POSITION INDICATOR PROBE CONNECTION DIAG.	10482506
8. REMOVED	
9. NEUTRON MONITORING SYS. FCD	C51-1020
10. REACTOR PROTECTION SYS. IED	C71/C72-1010
11. REMOVED	
12. CONTROL ROD DRIVE CONTROL SYS. ELEM DIAG	C11/C12-1050

AMENDMENT NO. 10
July 1980

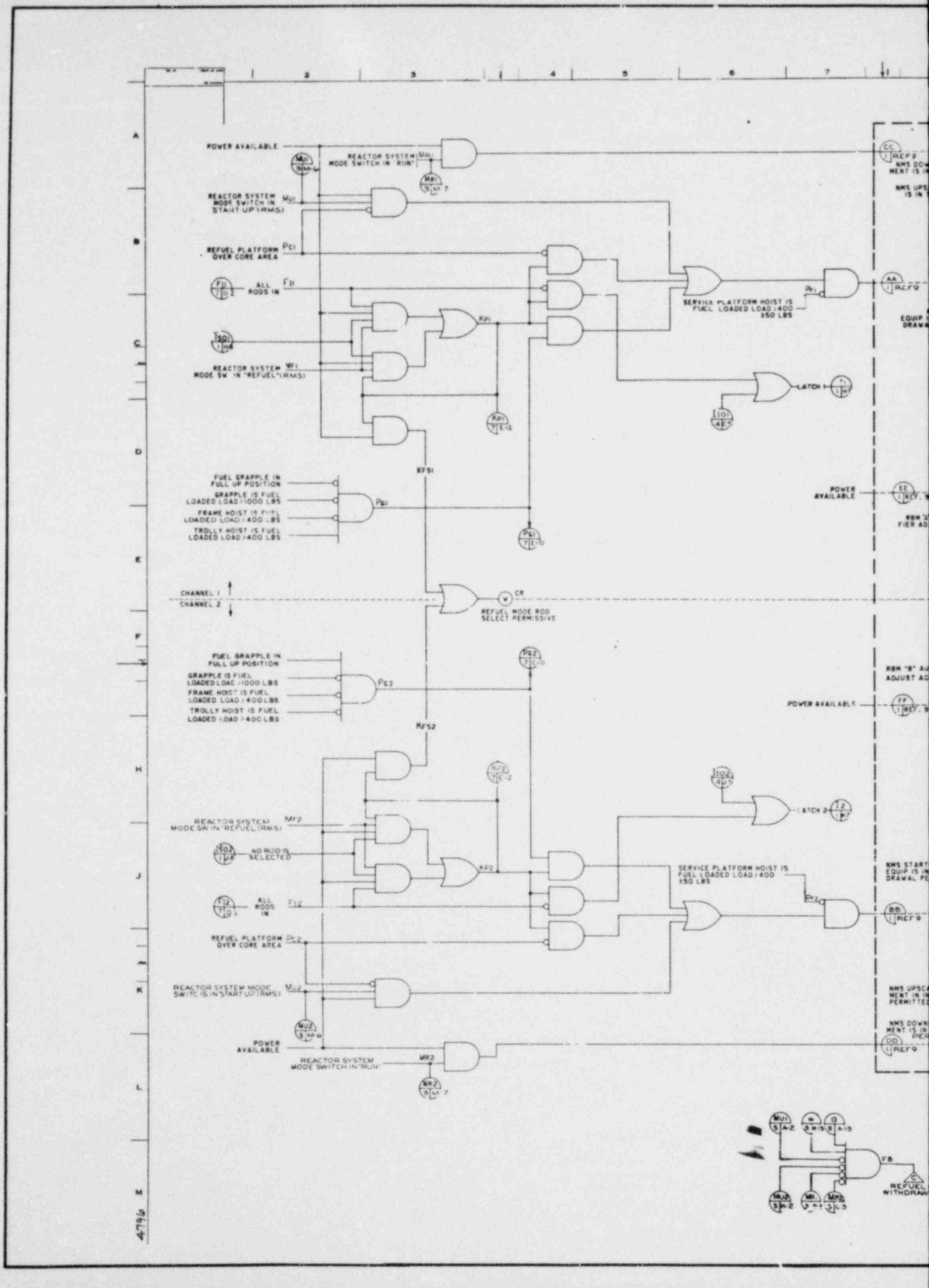
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD	FIGURE 7.7-3a



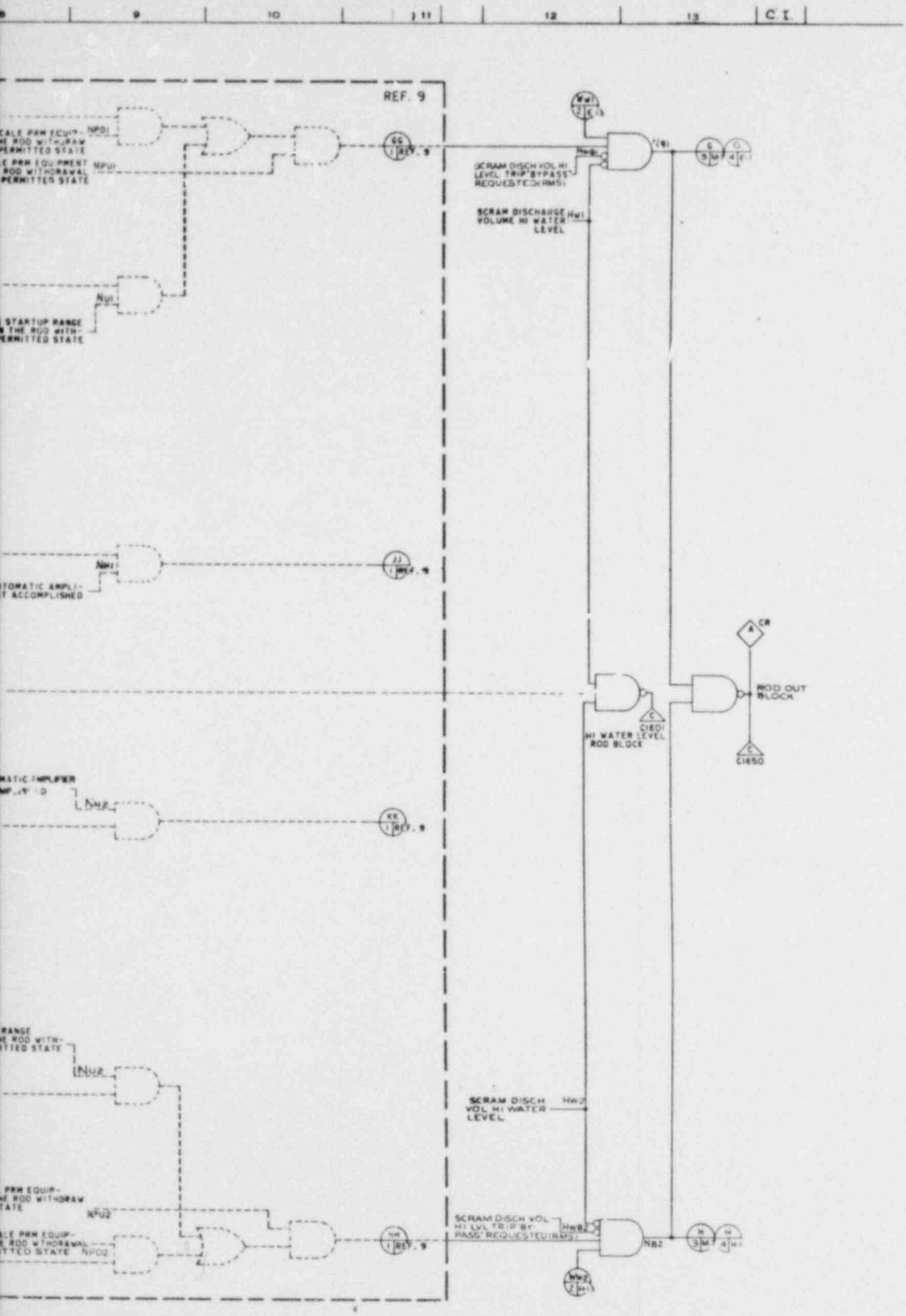
RWM PROGRAM SCAN CONTROL SIGNAL IS PRESENT
RWM SCAN CONTROL SIGNAL IS PRESENT
ANY ROD IS SELECTED
ROD POSITIONS SCANNED
STEAM FLOW < ALARM SETTING
FEEDWATER CONTROL SYSTEM LOW POWER LEVEL ALARM C1854
RPIS OPERABLE
POWER AVAILABLE
COMPUTER SOFTWARE OPERABLE
RWM INITIALIZATION REQUESTED (RMS)
RWM BYPASS IS REQUESTED (RMS)
COMPUTER PARITY NORMAL
COMPUTER STALLED
COMPUTER ERROR
ROD SELECTED IN SEQUENCE "A"
RWM SEQUENCE "A" IS REQUESTED (RMS)
RWM PROGRAM SEQUENCE "A" LATCHED
ROD PATTERN SEQUENCE "A" SELECT C1847
SEQUENCE "A" SELECTED
RWM SEQUENCE "B" IS REQUESTED (RMS)
RWM PROGRAM SEQUENCE "B" LATCHED
ROD PATTERN SEQUENCE "B" SELECT C1848
ROD SELECTED IN SEQUENCE "B"
SEQUENCE "B" SELECTED
ROD SELECTED IN "SHUTDOWN MARGIN"
RWM "SHUTDOWN MARGIN" REQUESTED (RMS)
RWM PROGRAM "SHUTDOWN MARGIN" LATCHED
"SHUTDOWN MARGIN" SELECT C1849
"SHUTDOWN MARGIN" SELECTED
FEEDWATER FLOW < TRIP SETTING
STEAM FLOW < TRIP SETTING
FEEDWATER CONTROL SYSTEM LOW POWER LEVEL INTERLOCK C1855
4797



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD	FIGURE 7.7-3b



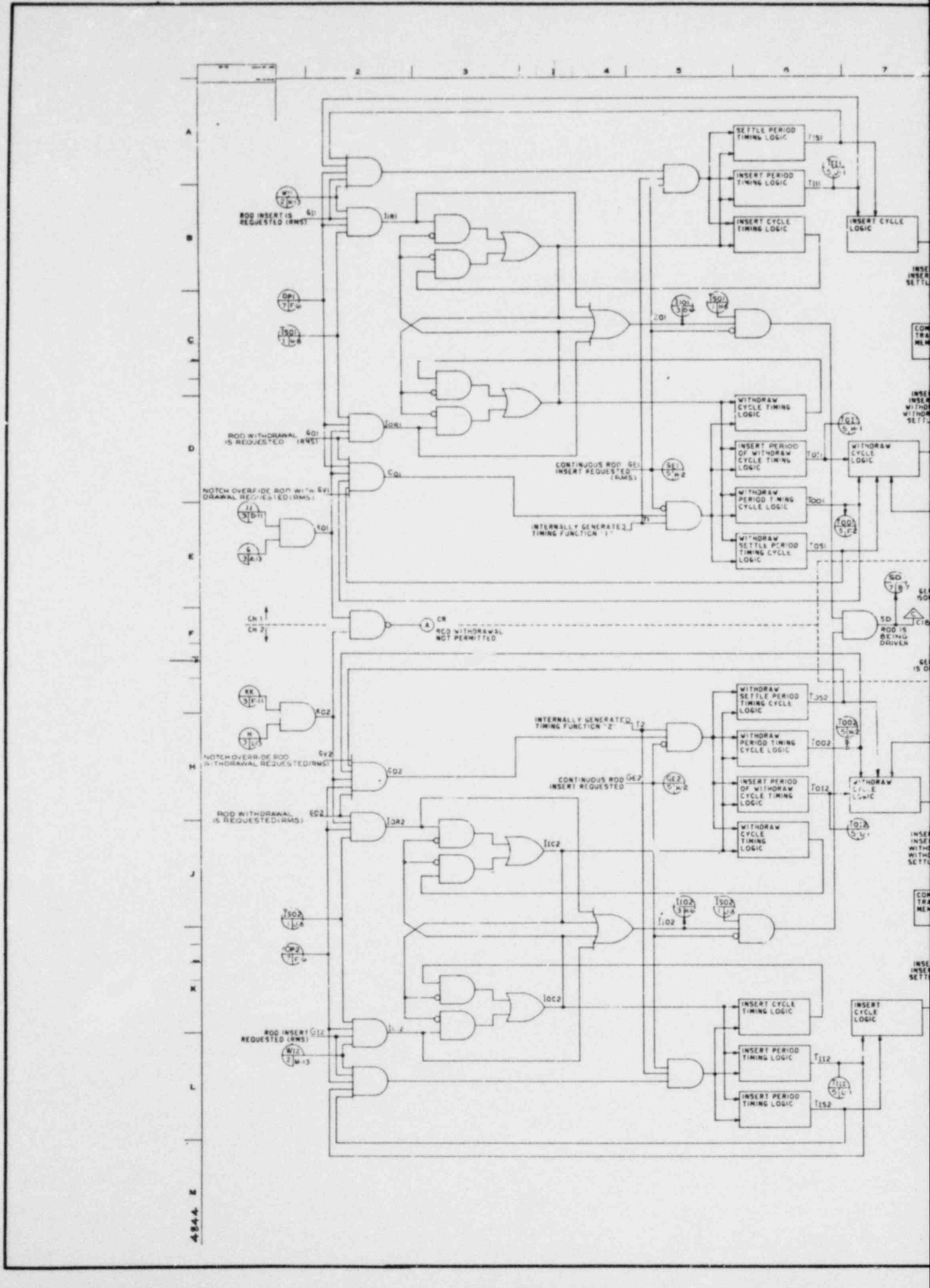
POWER AVAILABLE
REACTOR SYSTEM MODE SWITCH IN "RUN"
REACTOR SYSTEM MODE SWITCH IN START UP (RMS)
REFUEL PLATFORM OVER CORE AREA
ALL RODS IN
REACTOR SYSTEM MODE SW IN "REFUEL" (RMS)
SERVICE PLATFORM HOIST IS FUEL LOADED LOAD > 400 ±50 LBS
LATCH
FUEL GRAPPLE IN FULL UP POSITION
GRAPPLE IS FUEL LOADED LOAD > 1000 LBS
FRAME HOIST IS FUEL LOADED LOAD > 400 LBS
TROLLY HOIST IS FUEL LOADED LOAD > 400 LBS
CHANNEL 1
CHANNEL 2
REFUEL MODE ROD SELECT PERMISSIVE
KFS1
KFS2
REACTOR SYSTEM MODE SW IN REFUEL (RMS)
NO ROD IS SELECTED
LATCH 2
REACTOR SYSTEM MODE SWITC IS IN START UP (RMS)
REACTOR SYSTEM MODE SWITCH IN "RUN"
NMS DOWNSCALE EQUIP... MENT IS IN ...
NMS UPSCALE ... IS IN ...
EQUIP ... DRAWAL ...
RBM "A" ... FIER ADJ...
RBM "B" AU... ADJUST AD...
NMS START... EQUIP IS IN... DRAWAL PO...
NMS UPSCA... MENT IN IN... PERMITTED
NMS DOWN... MENT IS IN ... PER...
REFUEL ... WITHDRAW...
4796



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

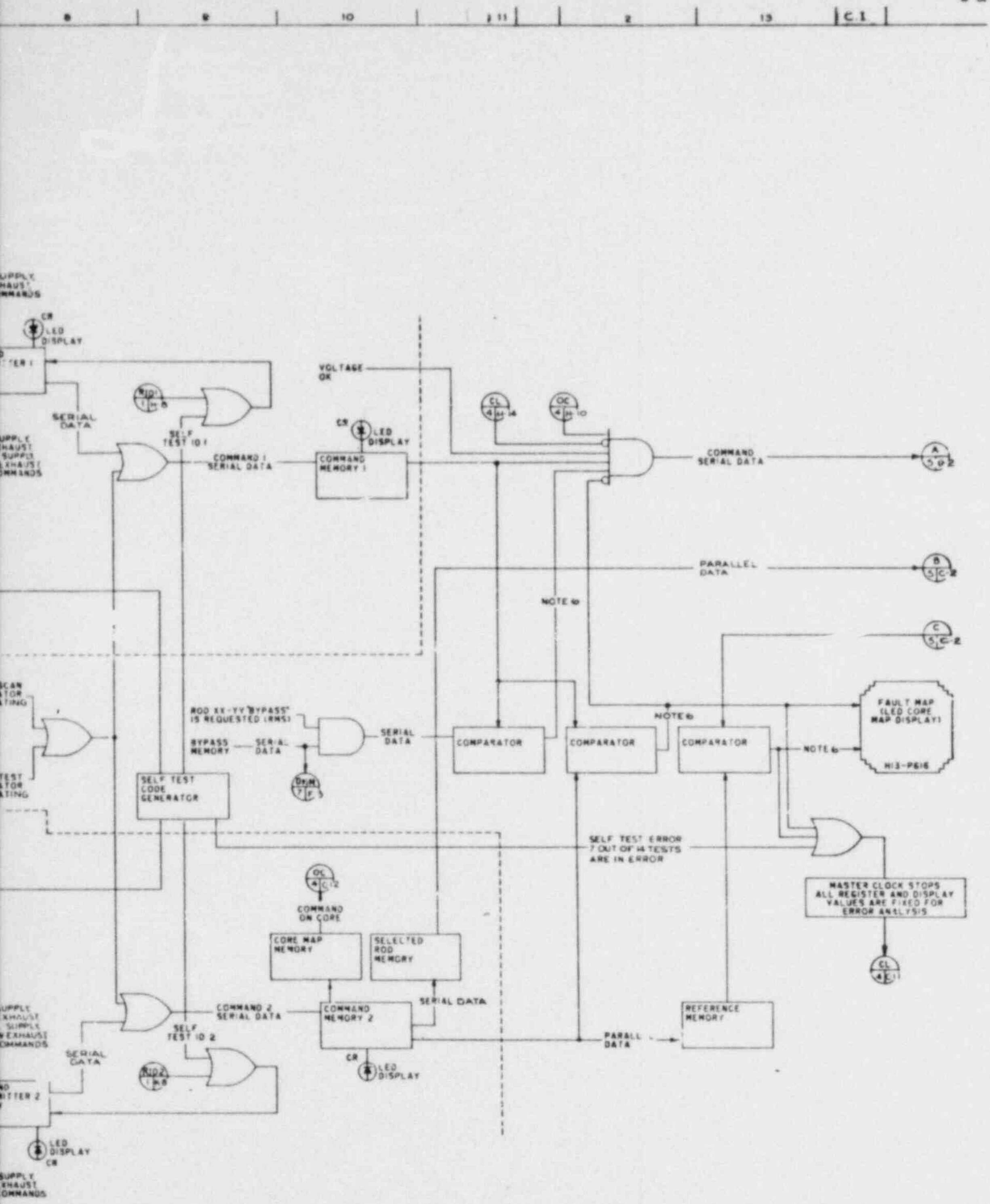
CONTROL ROD DRIVE HYDRAULIC SYSTEM
FCD

FIGURE
7.7-3c

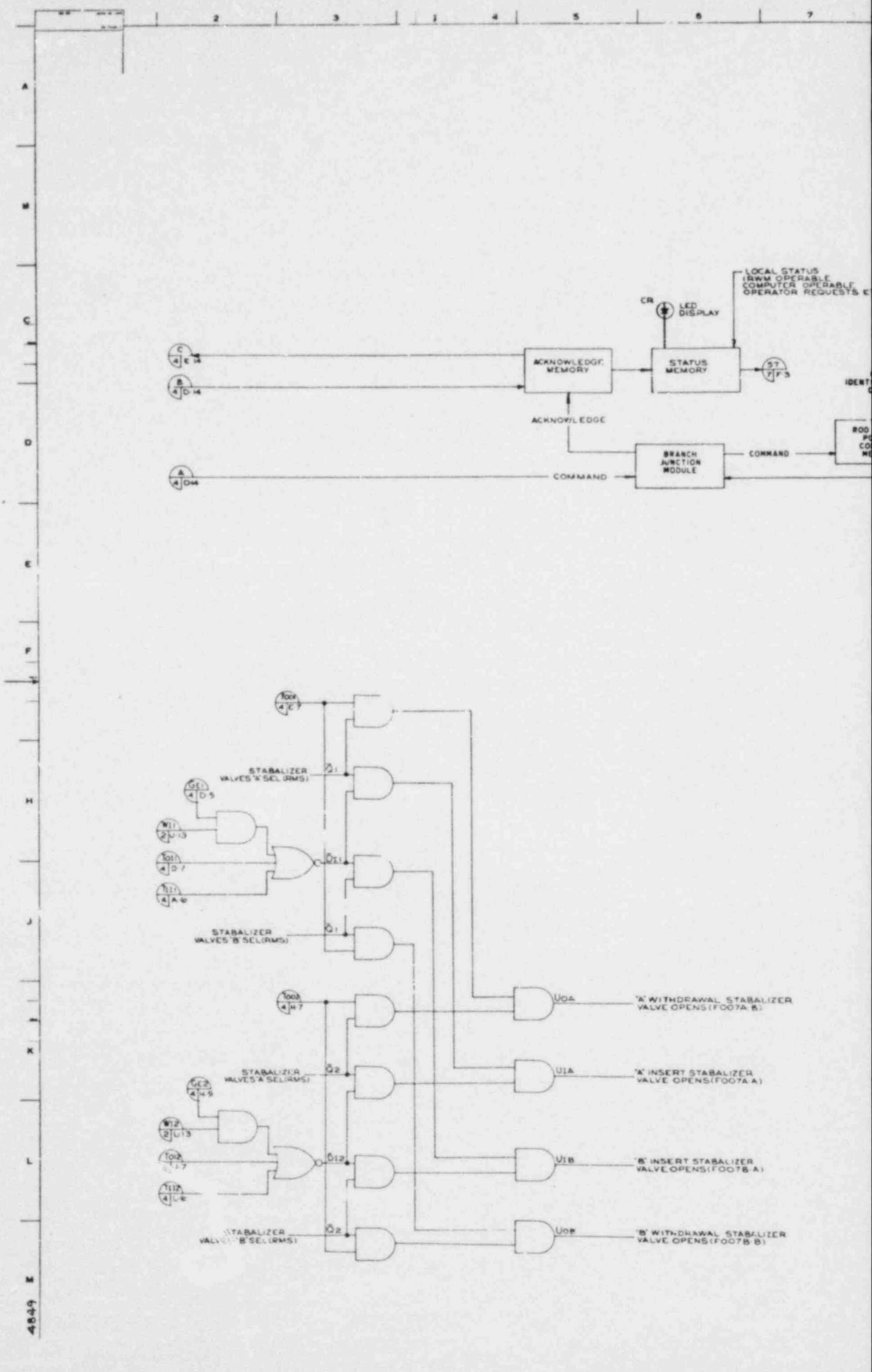


SETTLE PERIOD TIMING LOGIC
INSERT PERIOD TIMING LOGIC
INSERT CYCLE TIMING LOGIC
INSERT CYCLE LOGIC
ROD INSERT IS REQUESTED (RMS)
WITHDRAW CYCLE TIMING LOGIC
INSERT PERIOD OF WITHDRAW CYCLE TIMING LOGIC
WITHDRAW CYCLE LOGIC
ROD WITHDRAWAL IS REQUESTED (RMS)
CONTINUOUS ROD INSERT REQUESTED (RMS)
NOTCH OVERRIDE ROD WITHDRAWAL REQUESTED (RMS)
WITHDRAW PERIOD TIMING CYCLE LOGIC
INTERNALLY GENERATED TIMING FUNCTION "1"
WITHDRAW SETTLE PERIOD TIMING CYCLE LOGIC
CH 1
CH 2
ROD WITHDRAWAL NOT PERMITTED
ROD IS BEING DRIVEN
INTERNALLY GENERATED TIMING FUNCTION "2"
CONTINUOUS ROD INSERT REQUESTED
ROD INSERT REQUESTED (RMS)
4844

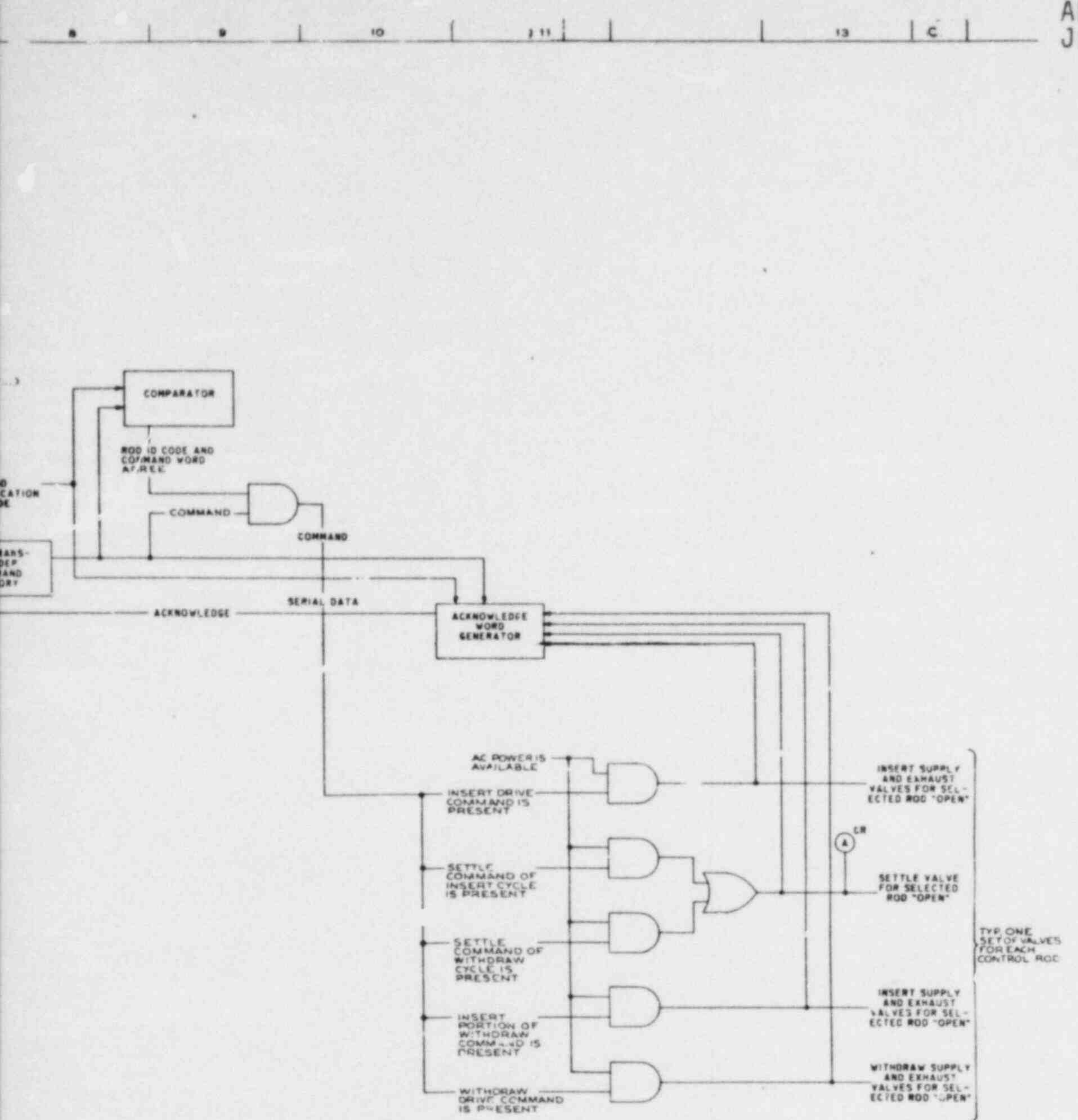
AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD	FIGURE 7.7-3d



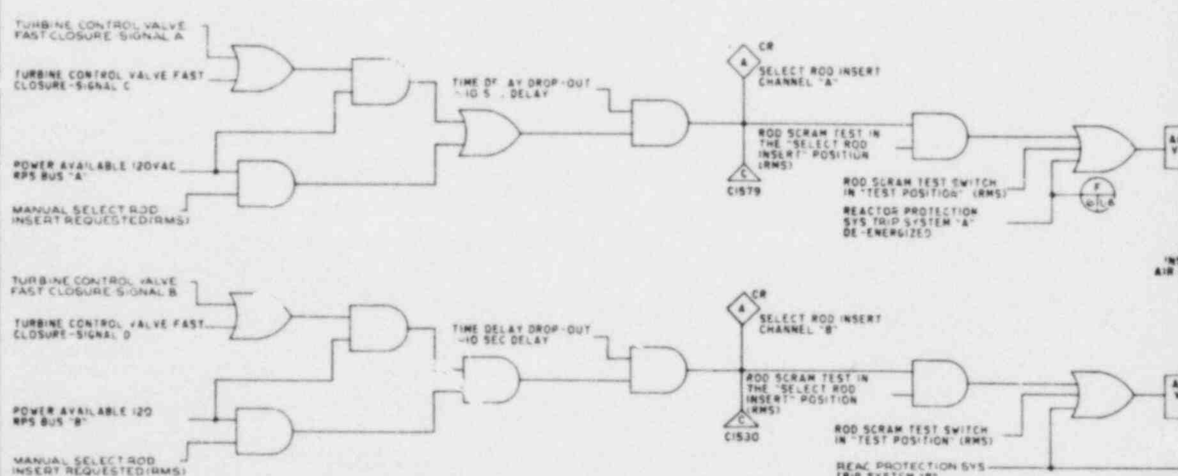
LOCAL STATUS
(RWM OPERABLE
COMPUTER OPERABLE
OPERATOR REQUESTS
CR
LED DISPLAY
ACKNOWLEDGE MEMORY
STATUS MEMORY
ACKNOWLEDGE
BRANCH JUNCTION MODULE
COMMAND
COMMAND
STABALIZER VALVES 'A' SEL (RMS)
STABALIZER VALVES 'B' SEL (RMS)
STABALIZER VALVES 'A' SEL (RMS)
STABALIZER VALVES 'B' SEL (RMS)
U0A
'A' WITHDRAWAL STABALIZER VALVE OPENS (F007A B)
U1A
'A' INSERT STABALIZER VALVE OPENS (F007A A)
U1B
'B' INSERT STABALIZER VALVE OPENS (F007B A)
U0B
'B' WITHDRAWAL STABALIZER VALVE OPENS (F007B B)
4849



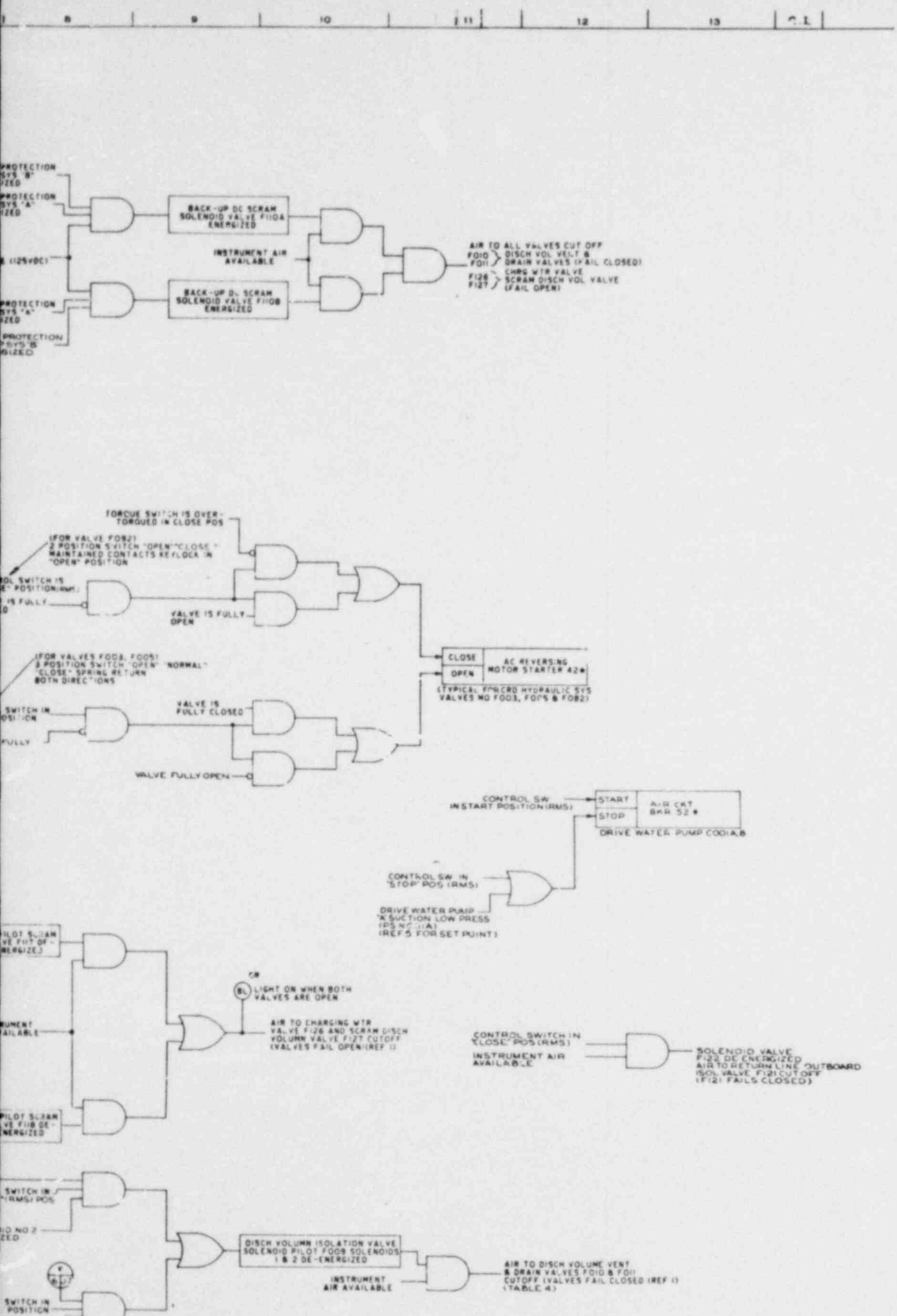
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD	FIGURE 7.7-3e

TABLE 4

FUNCTION	INITIATING DEVICE	TYPE
SCRAM DISCHARGE VOLUME NOT DRAINED	LEVEL SWITCH N013F	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 N012	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 F002AB, F121 & MO F003, F005 IS NOT FULLY CLOSED NOT FULLY OPEN	LIMIT SWITCH ON VALVES	IND LAMP R IND LAMP G
VALVES AO F010, & AO F011 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)



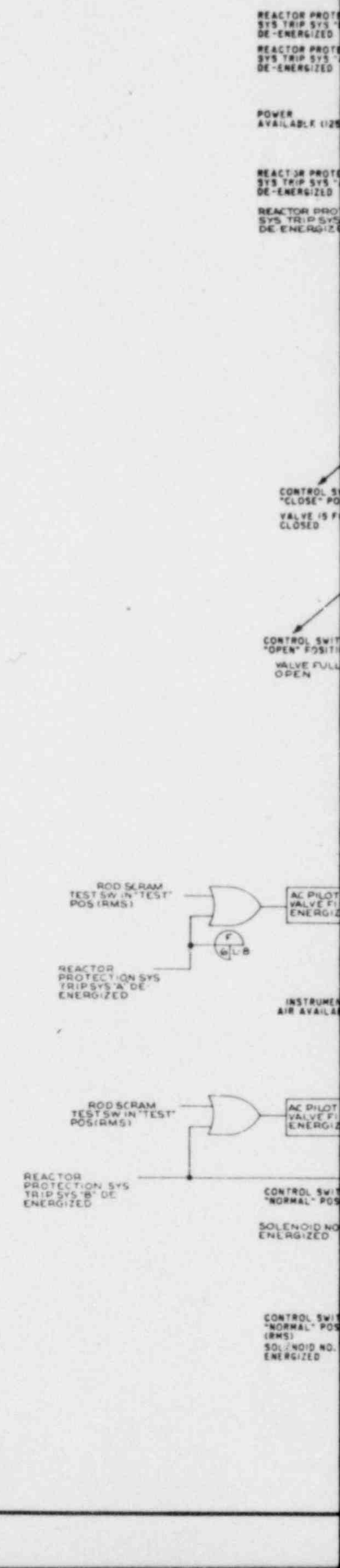
4830



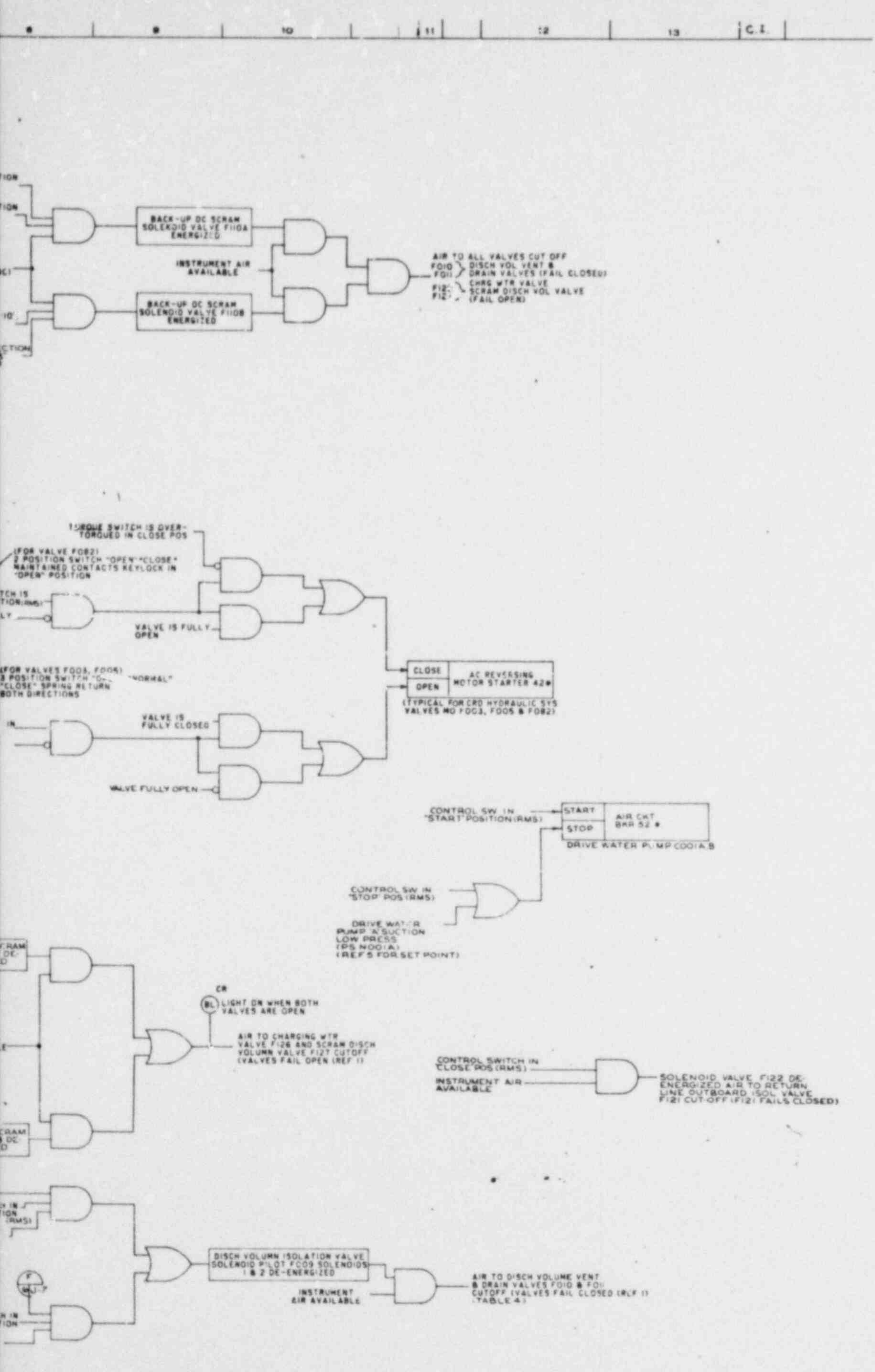
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD	FIGURE 7.7-3f

TABLE 4

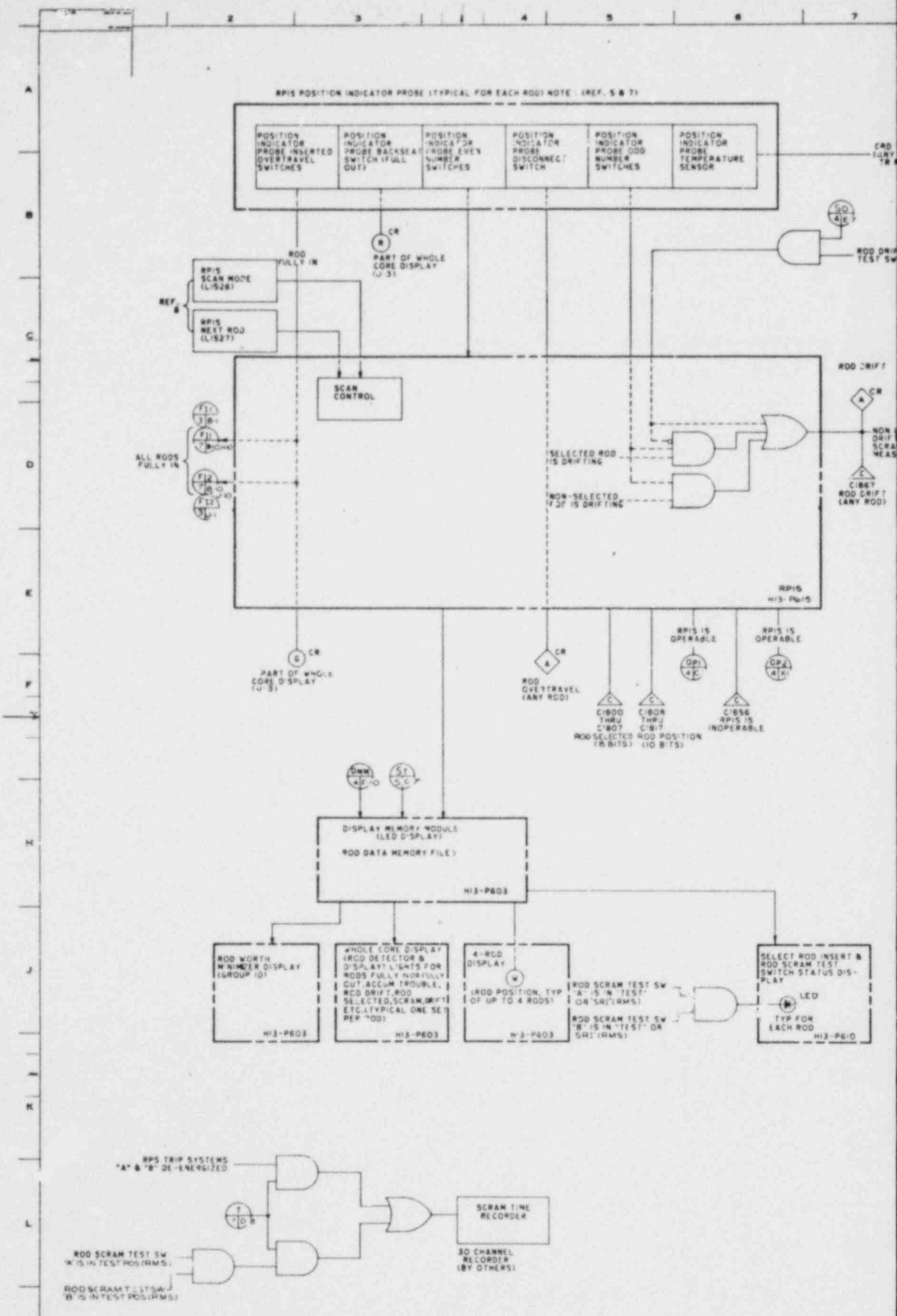
FUNCTION	INITIATING DEVICE	TYPE
SCRAM DISCHARGE VOLUME NO. DRAINED	LEVEL SWITCH N013F	ANN.
DRIVE WATER FILTER HIGH DIFF PRESSURE	DIFF PRESS. IND SW N002	ANN.
CHARGING WATER HIGH PRESSURE	PRESS. IND SW N600	ANN.
DRIVE WATER PUMP "A" SUCTION LOW PRESS.	PRESS. SW N001A	ANN.
DRIVE WATER PUMP "B" SUCTION LOW PRESSURE	PRESS. SW N001B	ANN.
SCRAM VALVE PILOT AIR HEADER HIGH/LOW PRESS.	PRESS. SW N012	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 F002AB, F121 & MO F003, F005 IS NOT FULLY CLOSED NOT FULLY OPEN	LIMIT SWITCH ON VALVES	IND LAMP R IND LAMP G
VALVES AO F010, & AO F011 FULLY OPEN FULLY CLOSED	LIMIT SWITCH ON VALVES	IND LAMP R IND LAMP G
ACCUMULATOR LOW PRESS OR LEAK DETECTION (TYP FOR EACH ACCUMULATOR) (REF 1) (NOTE 2)	PRESS SW 130 OR LEAK DET SW 129	IND LAMP A (NOTE 4)
ANY ACCUMULATOR LOW PRESS OR ANY ACCUMULATOR LEAKAGE	PRESS SW 130 OR LEAK DET SW 129	ANN H/L (NOTE 4)



4830



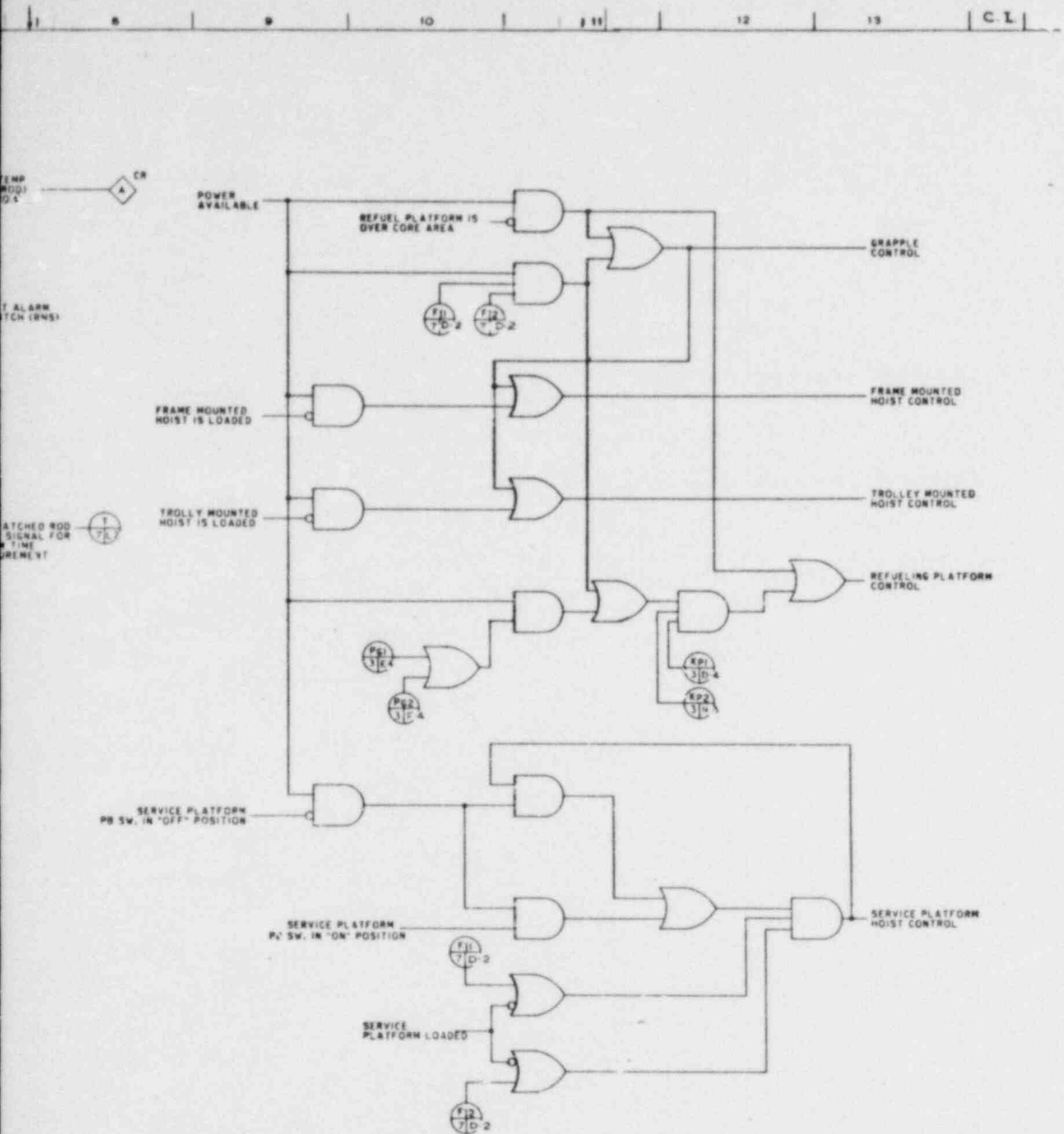
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD	FIGURE 7.7-3g



RPIS POSITION INDICATOR PROBE (TYPICAL FOR EACH ROD) NOTE: (REF. 5 & 7)
POSITION INDICATOR PROBE INSERTED OVERTRAVEL SWITCHES
POSITION INDICATOR PROBE BACKSEAT SWITCH (FULL OUT)
POSITION INDICATOR PROBE EVEN NUMBER SWITCHES
POSITION INDICATOR PROBE DISCONNECT SWITCH
POSITION INDICATOR PROBE ODD NUMBER SWITCHES
POSITION INDICATOR PROBE TEMPERATURE SENSOR
ROD FULLY IN
PART OF WHOLE CORE DISPLAY (J-3)
RPIS SCAN MODE (L1526)
RPIS NEXT ROD (L1527)
REF. 6
SCAN CONTROL
ALL RODS FULLY IN
SELECTED ROD IS DRIFTING
NON-SELECTED ROD IS DRIFTING
ROD DRIFT
C1867 ROD DRIFT (ANY ROD)
RPIS H13-P615
RPIS IS OPERABLE
PART OF WHOLE CORE DISPLAY (J-3)
ROD OVERTRAVEL (ANY ROD)
C1800 THRU C1807 ROD SELECTED (8 BITS)
C1808 THRU C1817 ROD POSITION (10 BITS)
C1856 RPIS IS INOPERABLE
DISPLAY MEMORY MODULE (LED DISPLAY)
ROD DATA MEMORY FILE)
H13-P603
ROD WORTH MINIMIZER DISPLAY (GROUP 10)
WHOLE CORE DISPLAY (RCD DETECTOR & DISPLAY) LIGHTS FOR RODS FULLY IN OR FULLY OUT, ACCUM TROUBLE, RCD DRIFT, ROD SELECTED, SCRAM, DRIFT ETC.(TYPICAL ONE SET PER ROD)
4-ROD DISPLAY
(ROD POSITION, TYP OF UP TO 4 RODS)
ROD SCRAM TEST SW "A" IS IN "TEST" OR "SRI"(RMS)
ROD SCRAM TEST SW "B" IS IN "TEST" OR "SRI"(RMS)
SELECT ROD INSERT & ROD SCRAM TEST SWITCH STATUS DISPLAY
LED
TYP FOR EACH ROD
H13-P610
RPS TRIP SYSTEMS "A" & "B" DE-ENERGIZED
ROD SCRAM TEST SW "A" IS IN TEST POS(RMS)
ROD SCRAM TEST SW "B" IS IN TEST POS(RMS)
SCRAM TIME RECORDER
30 CHANNEL RECORDER (BY OTHERS)

4850

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD	FIGURE 7.7-3h

INPUTS

FROM		SYM	DEFINITION
OPERATOR CONTROLS	REQUESTS	$B_1 \cdots B_n$	THE nth ROD IS REQUESTED
		B_I	ROD INSERTION IS REQUESTED
		B_O	NOTCH ROD WITHDRAWAL IS REQUESTED
		B_E	EMERGENCY ROD INSERTION IS REQUESTED
		B_V	NOTCH OVERRIDE ROD WITHDRAWAL IS REQUESTED
	STATUS	R_O	ROD REQUESTS ARE INHIBITED
		Q	CRD HYDRAULIC STABILIZER SOLENOID 'A' VALVES ARE SELECTED.
		$\bar{Q}$	CRD HYDRAULIC STABILIZER SOLENOID 'B' VALVES ARE SELECTED.
	MODE SELECTION	M_{R1}	REACTOR IS IN THE RUN MODE, INPUT 1
		M_{R2}	REACTOR IS IN THE RUN MODE, INPUT 2
		M_{F1}	REACTOR IS IN THE REFUEL MODE, INPUT 1
		M_{F2}	REACTOR IS IN THE REFUEL MODE, INPUT 2
		M_{U1}	REACTOR IS IN THE STARTUP MODE, INPUT 1
		M_{U2}	REACTOR IS IN THE STARTUP MODE, INPUT 2
NEUTRON MONITORING SYSTEM	STATUS	N_{U1}	NMS STARTUP RANGE EQUIPMENT IS IN THE ROD WITHDRAWAL PERMITTED STATE, INPUT 1.
		N_{U2}	NMS STARTUP RANGE EQUIPMENT IS IN THE ROD WITHDRAWAL PERMITTED STATE, INPUT 2.
			THE STARTUP RANGE EQUIPMENT USUALLY CONSISTS OF THE SRM, IRM, APRM AND RBM HI INOP CIRCUITRY, AND THE FLOW UNIT COMPARATOR, UPSCALE AND INOP CIRCUITRY.
		N_{P1}	NMS POWER RANGE EQUIPMENT IS IN THE ROD WITHDRAWAL PERMITTED STATE, INPUT 1.
		N_{P2}	NMS POWER RANGE EQUIPMENT IS IN THE ROD WITHDRAWAL PERMITTED STATE, INPUT 2.
			THE POWER RANGE EQUIPMENT USUALLY CONSISTS OF APRM, HI INOP AND DOWNSCALE CIRCUITRY, AND THE FLOW UNIT COMPARATOR, UPSCALE AND INOP CIRCUITRY.
RBM	STATUS	N_{N1}	RBM IS NULLED, INPUT 1
		N_{N2}	RBM IS NULLED, INPUT 2
RWM	STATUS	W_O	RWM WILL PERMIT ROD WITHDRAWAL
		W_I	RWM WILL PERMIT ROD INSERTION
RPIS	STATUS	O_P	RPIS IS OPERATIVE
		F_{I1}	ALL RODS ARE FULL-IN, INPUT 1
		F_{I2}	ALL RODS ARE FULL-IN, INPUT 2
DIS. VOLUME	STATUS	H_{W1}	SCRAM DISCHARGE VOLUME LEVEL IS HIGH, INPUT 1
		H_{W2}	SCRAM DISCHARGE VOLUME LEVEL IS HIGH, INPUT 2
		H_{WB1}	SCRAM DISCHARGE VOLUME HIGH LEVEL IS BYPASSED, INPUT 1
		H_{WB2}	SCRAM DISCHARGE VOLUME HIGH LEVEL IS BYPASSED, INPUT 2
SERVICE & REFUEL PLATFORM	STATUS	P_{F1}	SERVICE PLATFORM JIB CRANE HOIST LOAD IS GREATER THAN 400# & A FUEL BUNDLE IS PICKED UP, INPUT 1
		P_{F2}	SERVICE PLATFORM JIB CRANE HOIST LOAD IS GREATER THAN 400# & A FUEL BUNDLE IS PICKED UP, INPUT 2
		P_{G1}	GRAPPLE IS DOWN OR FRAME HOIST IS FUEL LOADED OR TROLLEY HOIST IS FUEL LOADED, OR GRAPPLE IS FUEL LOADED, INPUT 1
		P_{G2}	GRAPPLE IS DOWN OR FRAME HOIST IS FUEL LOADED OR TROLLEY HOIST IS FUEL LOADED, OR GRAPPLE IS FUEL LOADED, INPUT 2
		P_{C1}	REFUEL PLATFORM IS OVER THE CORE, INPUT 1
		P_{C2}	REFUEL PLATFORM IS OVER THE CORE, INPUT 2
		T	SEE SHEET 1, REF 1

INTERMEDIATE COMBINATIONS OF REDUNDANT INPUTS

FROM		SYM	DESCRIPTION AND EQUATION	
OPERATOR CONTROLS	MODE SELECTION	M_R	REACTOR IS IN RUN MODE	$M_R = M_{R1} M_{R2}$
		M_F	REACTOR IS IN REFUEL MODE	$M_F = M_{F1} M_{F2}$
		M_U	REACTOR IS IN STARTUP MODE	$M_U = M_{U1} M_{U2}$
NEUTRON MONITORING SYS.	STATUS	N_U	NMS STARTUP RANGE EQUIPMENT IN ROD WITHDRAWAL PERMITTED STATE	$N_U = N_{U1} N_{U2}$
		N_P	NMS POWER RANGE EQUIPMENT IN ROD WITHDRAWAL PERMITTED STATE	$N_P = N_{P1} N_{P2}$
RBM	STATUS	N_N	RBM IS NULLED	$N_N = N_{N1} N_{N2}$
DISCHARGE VOLUME	STATUS	H_{WB}	SCRAM DISCHARGE VOLUME HI LEVEL BYPASSED	$H_{WB} = H_{WB1} + H_{WB2}$
		H_W	SCRAM DISCHARGE VOLUME LEVEL IS HIGH	$H_W = H_{W1} + H_{W2}$
RPIS	STATUS	F_I	ALL RODS ARE FULL-IN	$F_I = F_{I1} F_{I2}$
SERVICE & REFUEL PLATFORM	STATUS	P_F	SERVICE PLATFORM JIB CRANE HOIST IS GREATER THAN 400# & A FUEL BUNDLE IS PICKED UP	$P_F = P_{F1} + P_{F2}$
		P_G	GRAPPLE IS DOWN OR FRAME HOIST IS FUEL LOADED OR TROLLEY HOIST IS FUEL [illegible]	$P_G = P_{G1} + P_{G2}$
		P_C	REFUEL PLATFORM IS OVER CORE	$P_C = P_{C1} + P_{C2}$

INTERMEDIATE VARIABLES

TO	SYM	DEFINITION AND EQUATION	
INTERNAL CONTROL SIGNALS	I_{IT}	INSERT CYCLE ADVANCE	$I_{IT} = T \, I_{IC}$
	I_{OT}	WITHDRAW CYCLE ADVANCE	$I_{OT} = T \, I_{OC}$
	I_R	REFUEL MODE ONE ROD CONTROL.	$I_R = [illegible]$
	I_{SR}	SETTLE CONTROL.	$I_{SR} = [illegible]$
	I_{GI}	[illegible] CONTROL.	$I_{GI} = [illegible]$
	I_{IO}	ROD IN MOTION.	$I_{IO} = I_{IC} + I_{OC}$
	I_{OR}	ACTIVATE OUT-CLOCK	$I_{OR} = [illegible]$
	I_{OC}	OUT-CLOCK ACTIVE	$I_{OC} = [illegible]$
	I_{IR}	ACTIVATE IN-CLOCK	$I_{IR} = [illegible]$
	I_{IC}	IN-CLOCK ACTIVE	$I_{IC} = [illegible]$
	I_{SO}	NO ROD IS SELECTED	$I_{SO} = [illegible]$

AMENDMENT NO. 10
July 1980

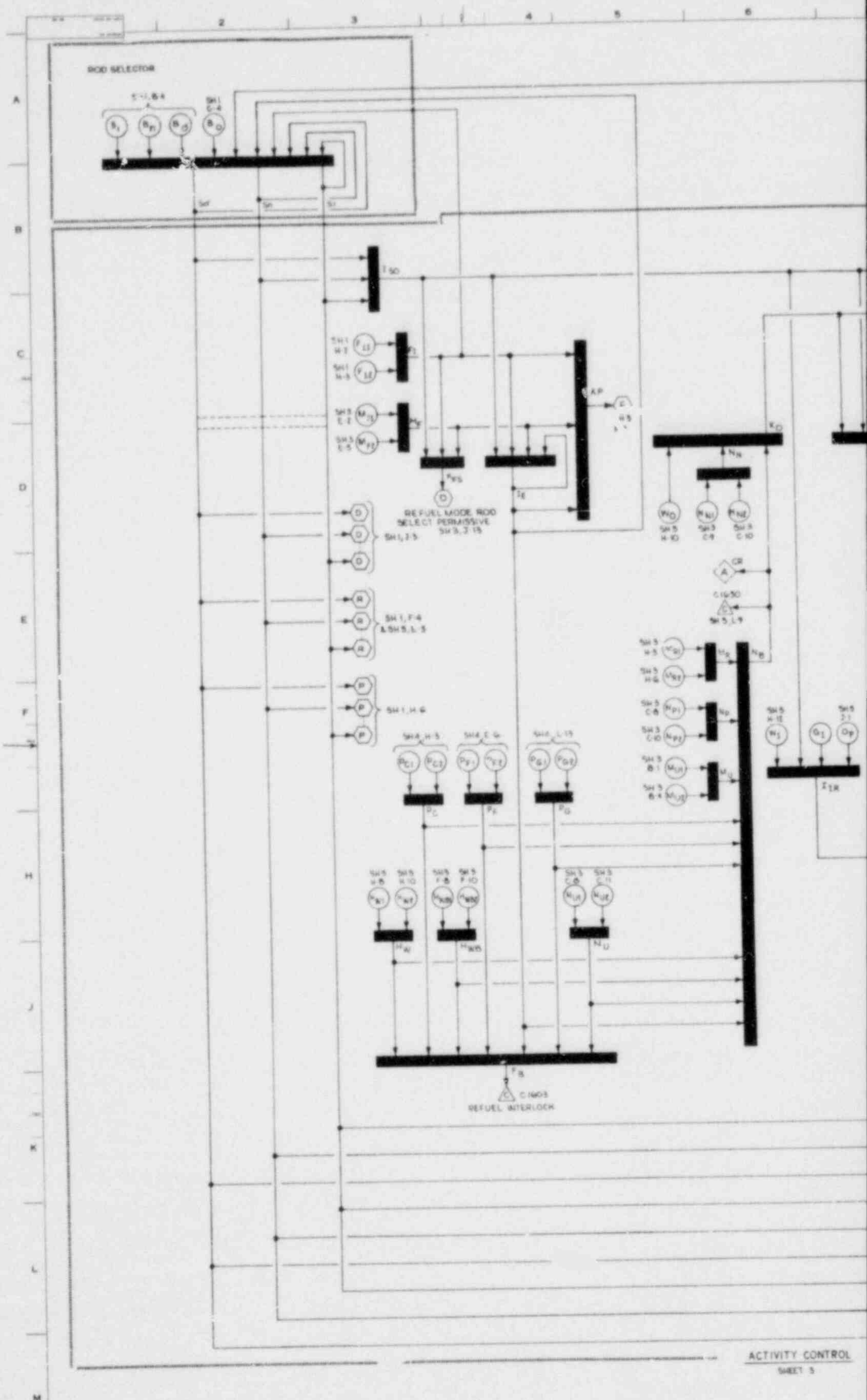
OUTPUTS

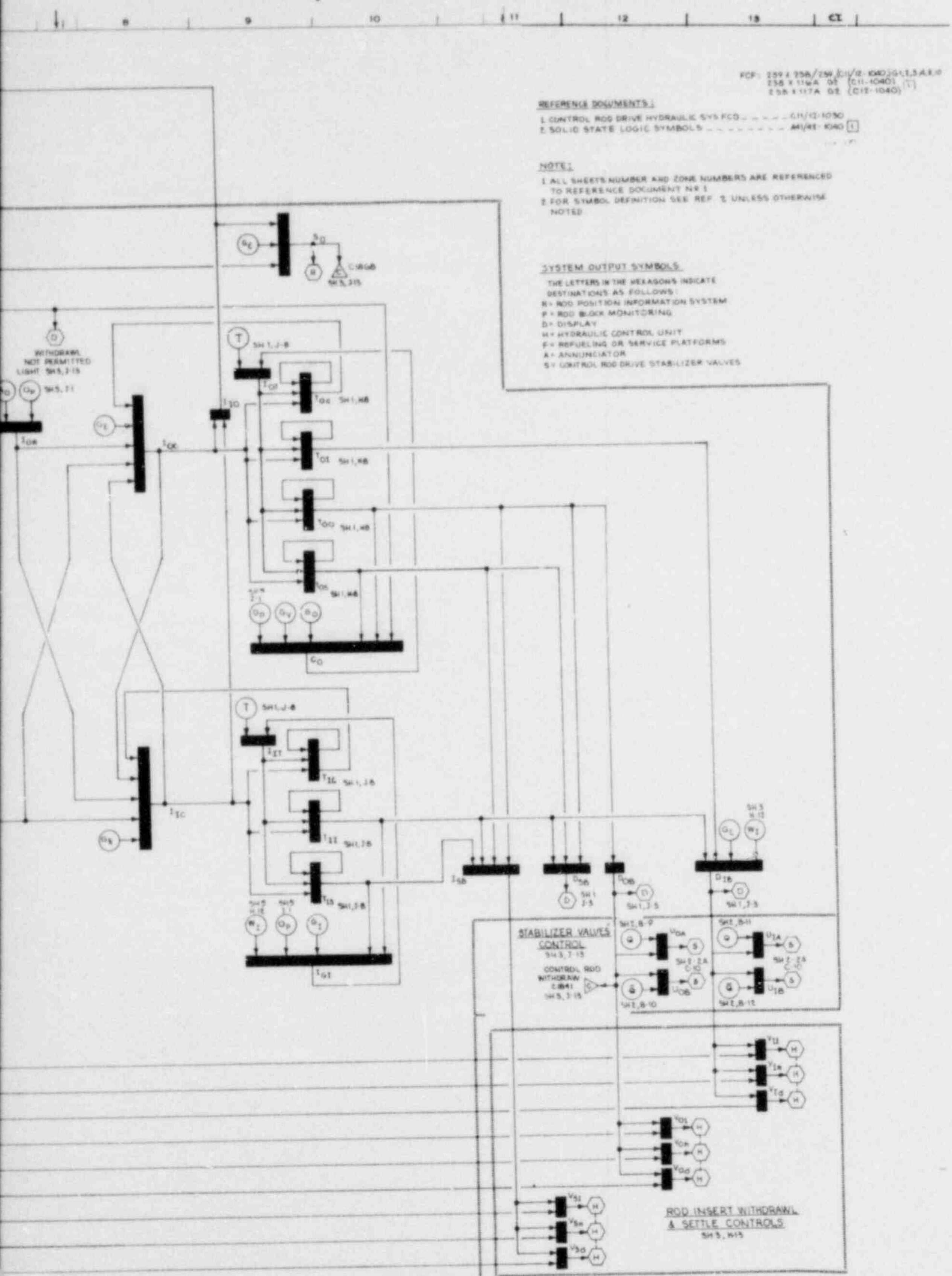
SYM	MEANING (STATUS OR COMMAND) AND DEFINING EQUATION	
[illegible]	THE ith ROD IS SELECTED.	[illegible]
[illegible]	ROD WITHDRAWAL IS PERMITTED.	[illegible]
[illegible]	A ROD IS BEING CONTINUOUSLY WITHDRAWN (NOTCH OVERRIDE).	[illegible]
[illegible]	A ROD IS BEING INSERTED.	[illegible]
[illegible]	A ROD IS BEING WITHDRAWN.	[illegible]
[illegible]	A ROD IS SETTLING.	[illegible]
[illegible]	REFUEL MODE ROD SELECTION IS PERMITTED.	[illegible]
[illegible]	(SEE OPERATOR STATUS SELECTION INPUTS)	
[illegible]	[illegible]	[illegible]
[illegible]	OPEN "A" INSERT STABILIZER VALVE.	[illegible]
[illegible]	OPEN "A" WITHDRAW STABILIZER VALVE.	[illegible]
[illegible]	OPEN "B" INSERT STABILIZER VALVE.	[illegible]
[illegible]	OPEN "B" WITHDRAW STABILIZER VALVE.	[illegible]
[illegible]	(SEE OUTPUT TO DISPLAYS)	
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	(SEE OUTPUT TO DISPLAYS)	
[illegible]	[illegible]	[illegible]
[illegible]	THE REACTOR IS IN THE REFUEL MODE [illegible] WITHDRAWN.	[illegible]
[illegible]	(SEE OUTPUT TO ANNUNCIATORS)	
[illegible]	REFUEL MODE ROD WITHDRAWAL IS PERMITTED.	[illegible]
[illegible]	(SEE OUTPUT TO DISPLAYS)	
[illegible]	(SEE OUTPUT TO [illegible])	

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

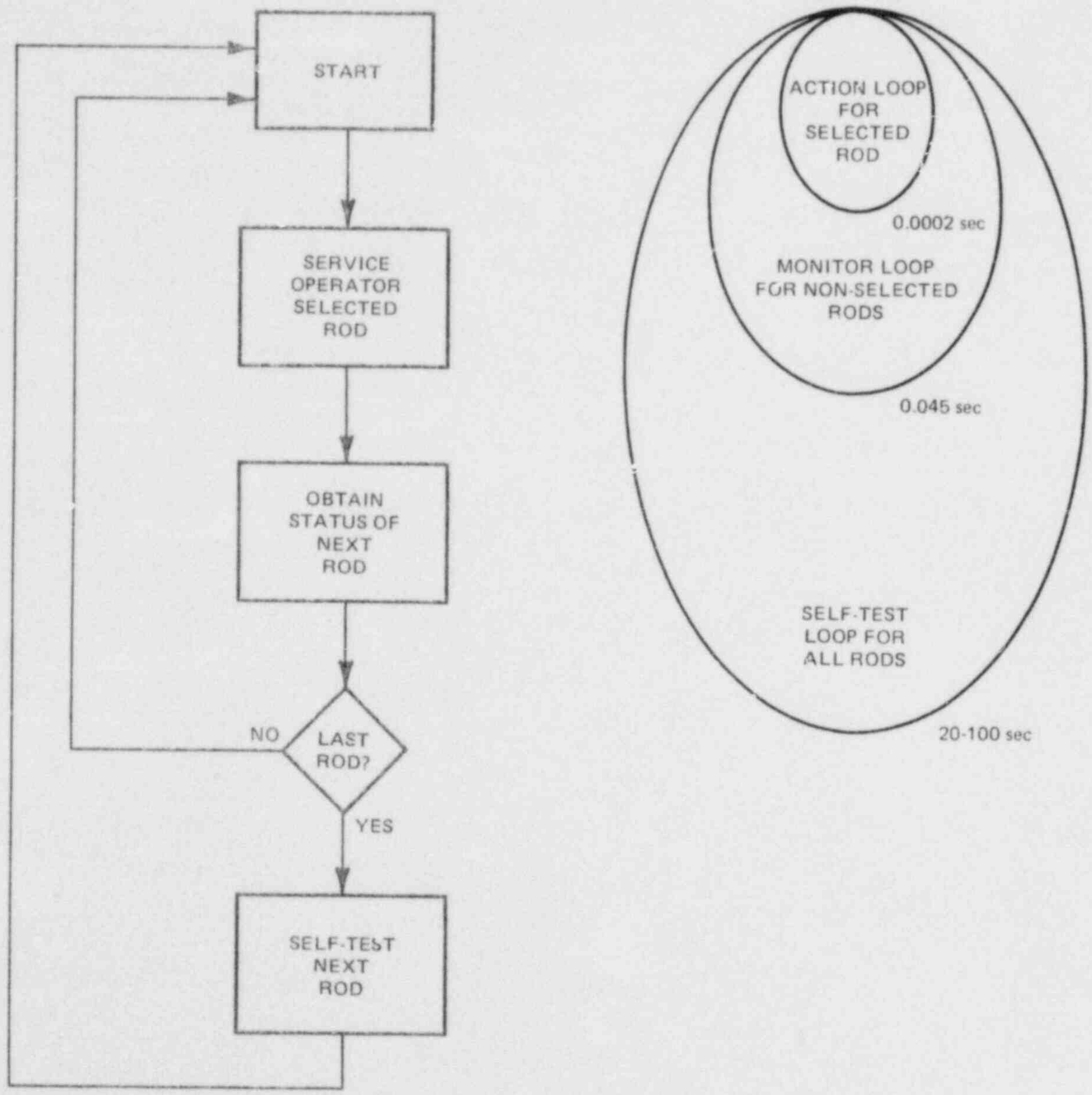
REACTOR MANUAL CONTROL SYSTEM

FIGURE 7.7-4a

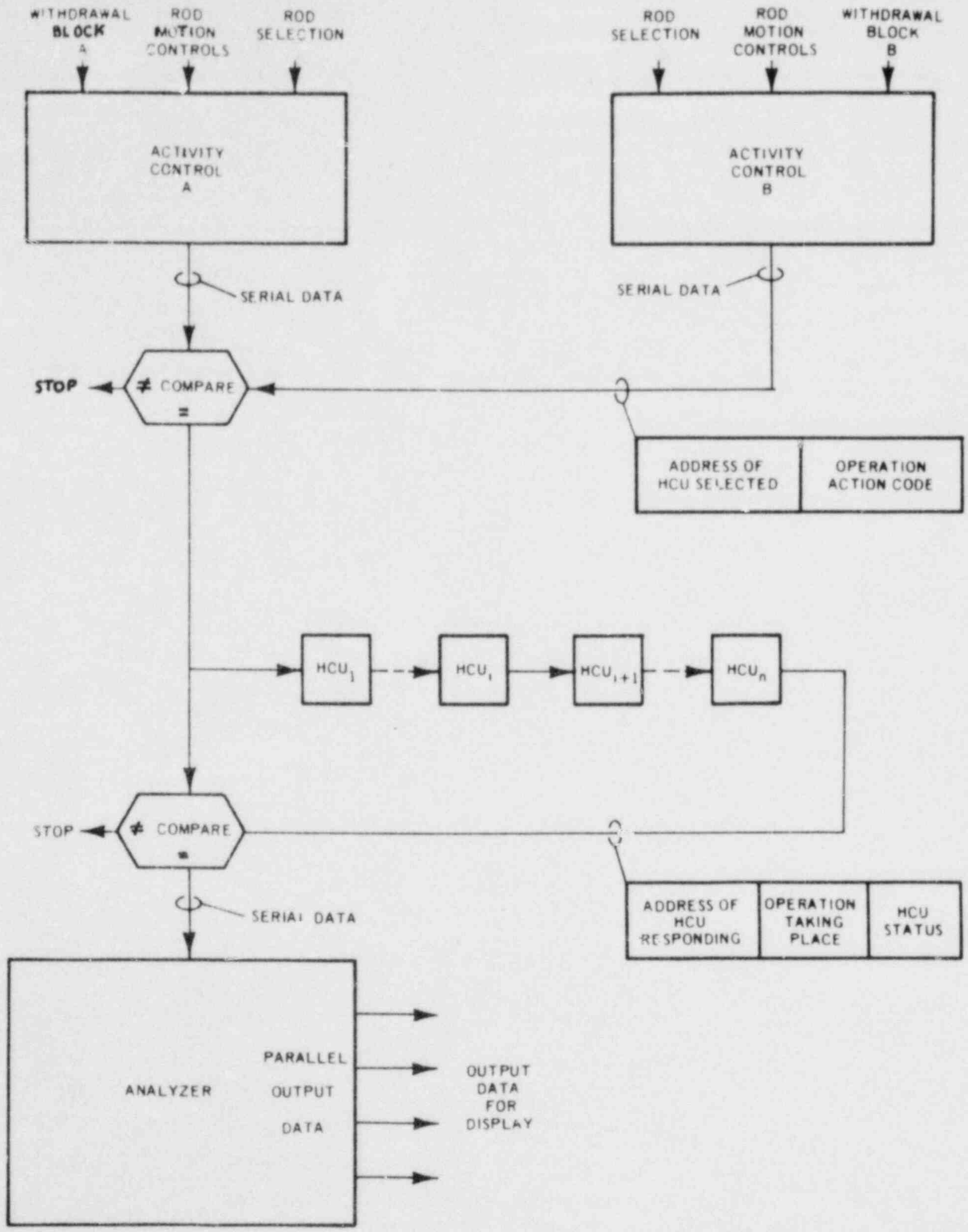

ROD SELECTOR
REFUEL MODE ROD SELECT PERMISSIVE
REFUEL INTERLOCK
ACTIVITY CONTROL
SHEET 5



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR MANUAL CONTROL SYSTEM	FIGURE 7.7-4b



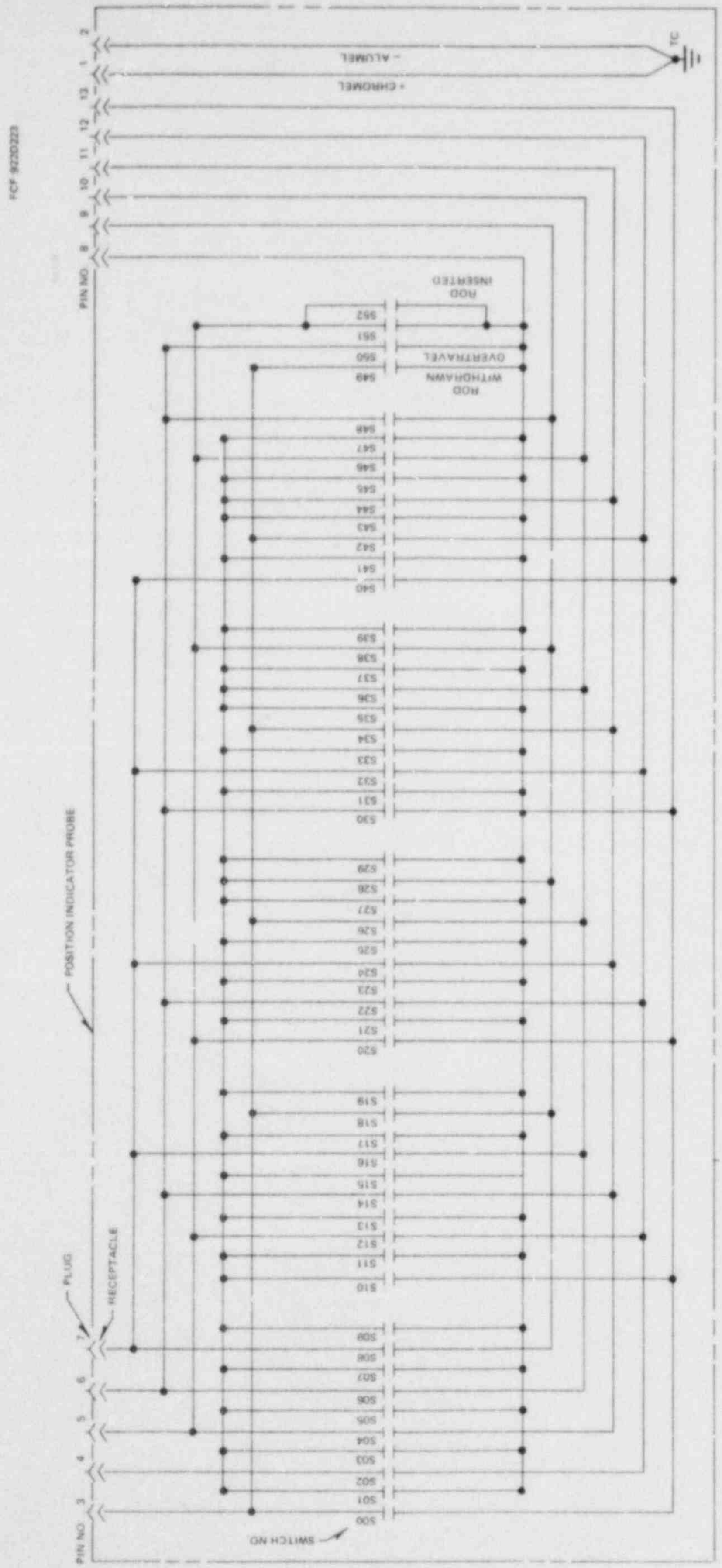
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR MANUAL CONTROL SELF-TEST PROVISIONS.	FIGURE 7.7-5a



AMENDMENT NO. 10
July 1980

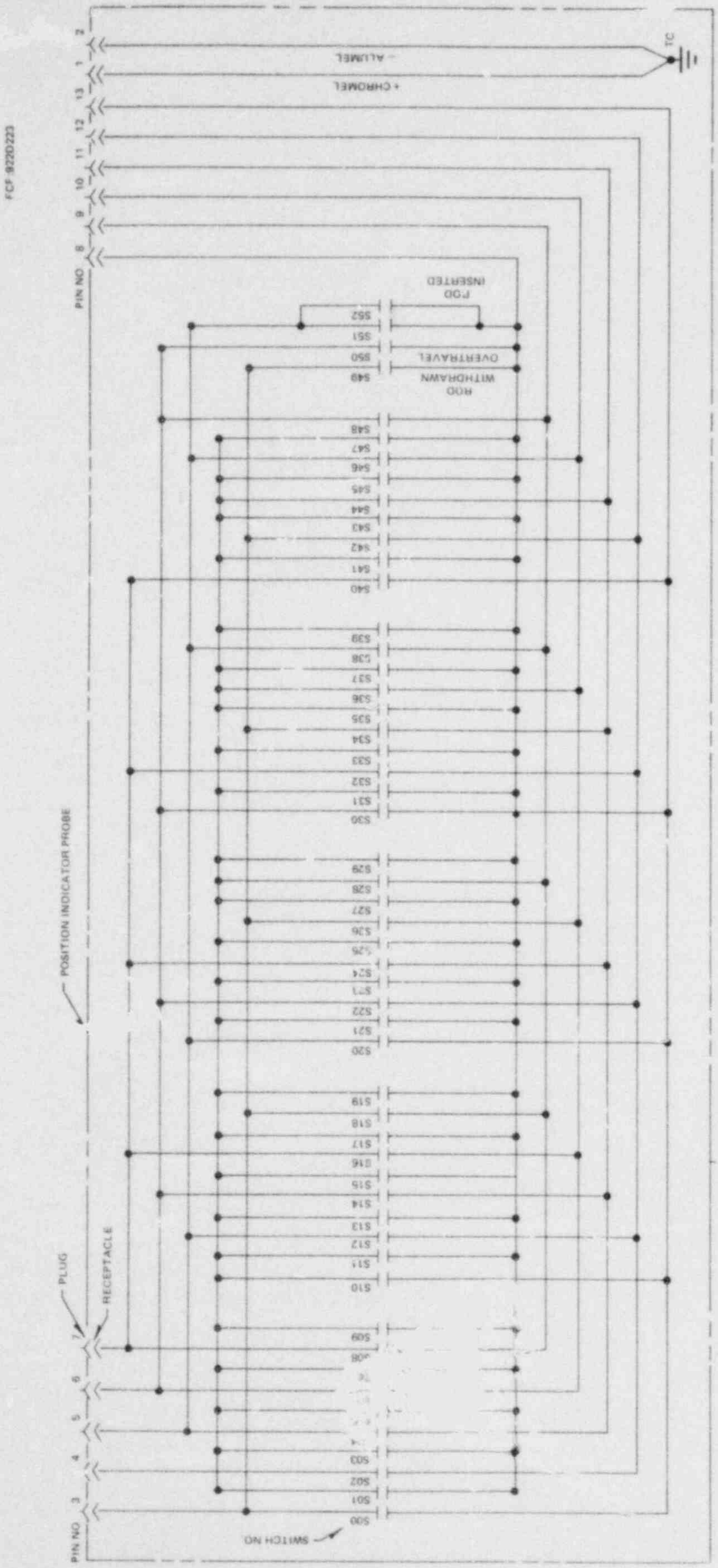
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR MANUAL CONTROL SYSTEM OPERATION	FIGURE 7.7-5b

AMENDMENT NO. 10
July 1980

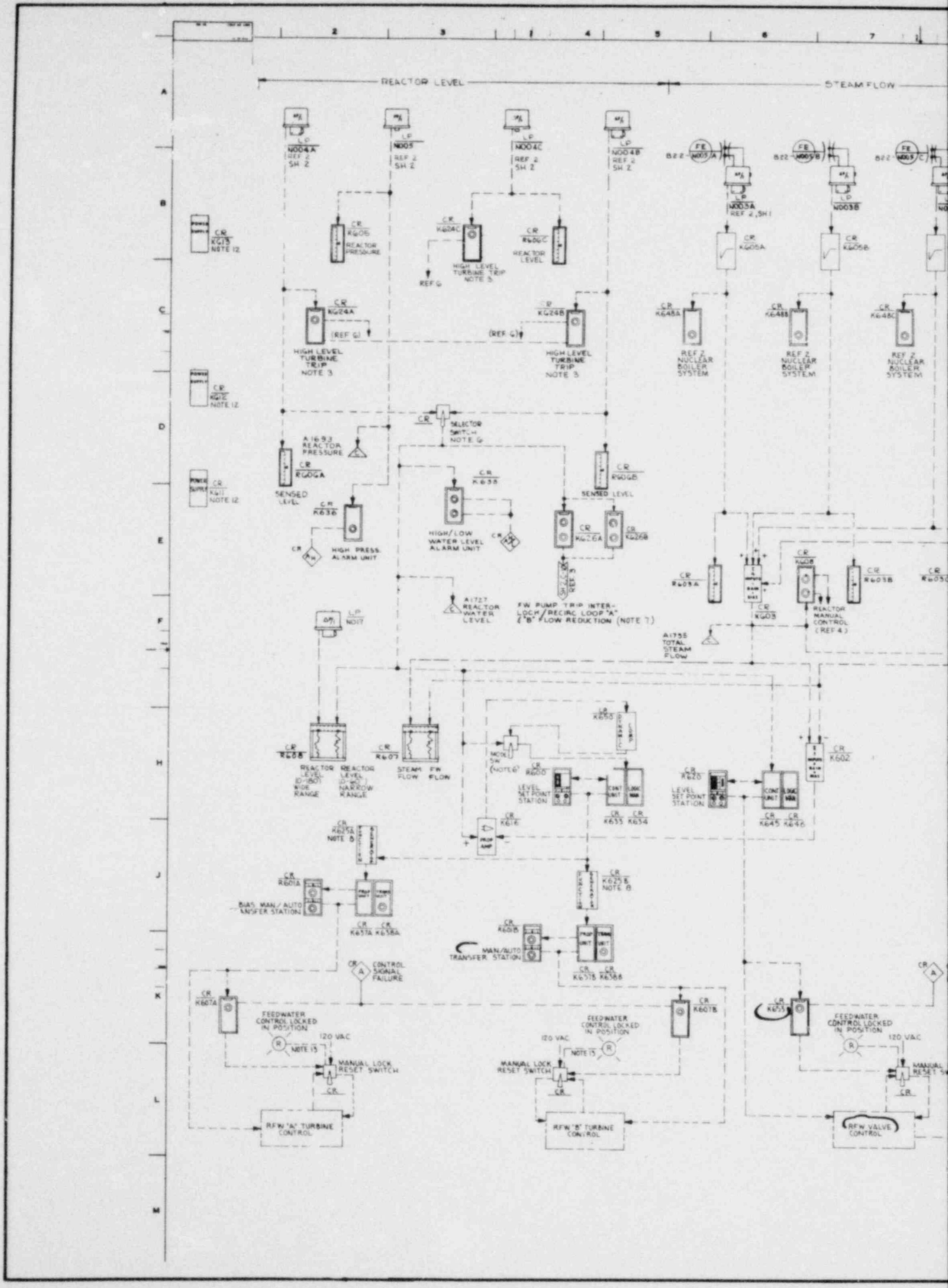


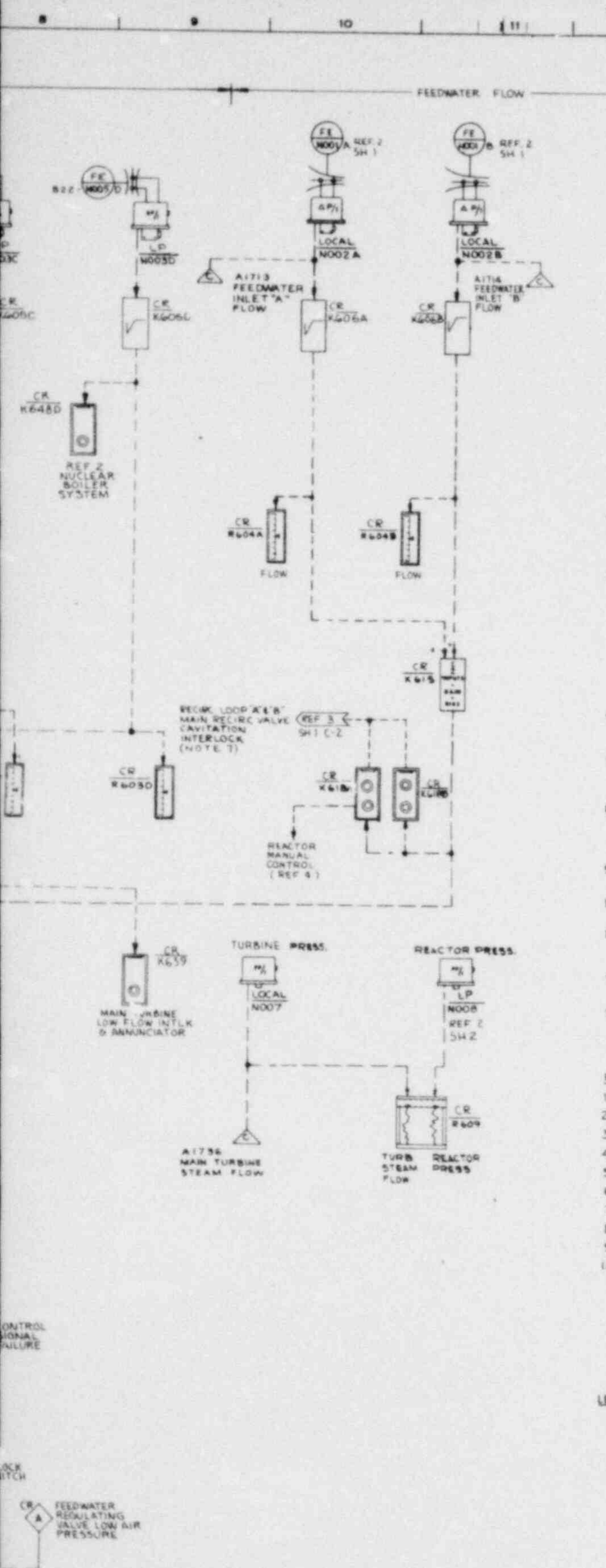
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ELEVEN-WIRE POSITION PROBE	FIGURE 7.7-6

AMENDMENT NO. 10
July 1980

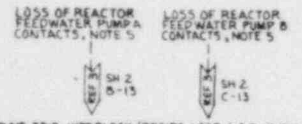


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ELEVEN-WIRE POSITION PROBE	FIGURE 7.7-6


REACTOR LEVEL
STEAM FLOW
LP N004A REF 2 SH 2
LP N005 REF 2 SH 2
LP N004C REF 2 SH 2
LP N004B REF 2 SH 2
B22-FE-N005A
B22-FE-N005B
LP N003A REF 2,SH 1
LP N003B
CR K613 NOTE 12
CR R605 REACTOR PRESSURE
CR K624C HIGH LEVEL TURBINE TRIP NOTE 3
REF 6
CR R606C REACTOR LEVEL
CR K605A
CR K605B
CR K624A HIGH LEVEL TURBINE TRIP NOTE 3
(REF 6)
CR K624B HIGH LEVEL TURBINE TRIP NOTE 3
CR K648A REF 2 NUCLEAR BOILER SYSTEM
CR K648B REF 2 NUCLEAR BOILER SYSTEM
CR K648C REF 2 NUCLEAR BOILER SYSTEM
CR K612 NOTE 12
CR SELECTOR SWITCH NOTE 6
A1693 REACTOR PRESSURE
CR R606A SENSED LEVEL
CR K611 NOTE 12
CR K636 HIGH PRESS. ALARM UNIT
CR K633 HIGH/LOW WATER LEVEL ALARM UNIT
CR R606B SENSED LEVEL
CR K626A
CR K626B
FW PUMP TRIP INTER-LOCK/RECIRC LOOP "A" & "B" FLOW REDUCTION (NOTE 7)
CR R603A
CR K603
CR K608 REACTOR MANUAL CONTROL (REF 4)
CR R603B
A1727 REACTOR WATER LEVEL
LP N017
A1755 TOTAL STEAM FLOW
LP K650
CR R608 REACTOR LEVEL (0-180) WIDE RANGE
REACTOR LEVEL (0-60) NARROW RANGE
CR R607 STEAM FLOW FW FLOW
MODE SW (NOTE 6)
CR R600 LEVEL SET POINT STATION
CR K635 CR K634
CR R620 LEVEL SET POINT STATION
CR K645 CR K646
CR K602
CR K616 PROP AMP
CR K625A NOTE 8
CR R601A
BIAS MAN/AUTO TRANSFER STATION
CR K637A K638A
CR K625B NOTE 8
CR R601B MAN/AUTO TRANSFER STATION
CR K637B K638B
CR A CONTROL SIGNAL FAILURE
CR K607A
FEEDWATER CONTROL LOCKED IN POSITION
R
NOTE 13
120 VAC
MANUAL LOCK RESET SWITCH
CR
RFW "A" TURBINE CONTROL
CR K607B
FEEDWATER CONTROL LOCKED IN POSITION
120 VAC
NOTE 13
MANUAL LOCK RESET SWITCH
CR
RFW "B" TURBINE CONTROL
CR K653
FEEDWATER CONTROL LOCKED IN POSITION
120 VAC
MANUAL RESET S
CR
RFW VALVE CONTROL
A B C D E F H J K L M
1 2 3 4 5 6 7



AMENDMENT NO. 10
July 1980



FEEDWATER PUMP TRIP INTERLOCK/RECIRC LOOP A & B FLOW REDUCTION.

NOTES:

1. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY SYSTEM NO. C34 (TURBINE DRIVEN)UNLESS OTHERWISE NOTED.
2. FOR LOCATION, IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET (REF 10).
3. DEVICES K624A, B & C TRIP CONTACTS TO BE WIRED SO THAT 2 OUT OF 3 DEVICES MUST TRIP TO INITIATE MAIN AND AUXILIARY TURBINE STEAM STOP VALVE TRIP.
4. THE POWER SOURCE FOR THE FEEDWATER INSTRUMENTATION AND CONTROL SYSTEM SHALL HAVE AT LEAST THE SAME DEGREE OF RELIABILITY AS THE POWER SOURCE FOR THE REACTOR/FEED/BOOSTER/CONDENSATE PUMPS.
5. CUSTOMER TO PROVIDE A CIRCUIT BREAKER AUXILIARY CONTACT FROM EACH REACTOR FEEDPUMP CIRCUIT BREAKER. GE TO PROVIDE AUXILIARY RELAY LOGIC FROM CUSTOMER (AE C.B. AUXILIARY CONTACTS AND CONTACTS FROM CR/K626 (E4) TO INITIATE INDEPENDENT RUNBACK OF EACH REACTOR RECIRC PUMP, IN EVENT OF A 1 OUT OF 2 RF PUMP TRIP.
6. SWITCHES SHALL BE "SNAP ACTION" SWITCHES CONTACT OPERATION BEING INDEPENDENT OF SPEED OF CONTROL ROOM OPERATOR ACTION TO AVOID CONTROL SYSTEM TRANSIENTS DURING SWITCHING.
7. INDEPENDENT ALARM UNIT RELAYS OR INTERPOSING RELAYS TO BE PROVIDED TO ENSURE SEPARATION OF RECIRC LOOP A CONTROL CIRCUIT FROM LOOP B.
8. FUNCTION GENERATORS SPECIALLY CHARACTERIZED BASED ON TURBINE-RF PUMP UNIT MA SIGNAL VS FW FLOW LB/HR CHARACTERISTICS TO BE SUBMITTED TO GE (BWRSD) BY CUSTOMER/AE FOR BWRSD DESIGN COMPLETION.
9. FEEDWATER FLOW INTEGRATION IS AVAILABLE VIA NSSS COMPUTER.
10. FOR INDICATING INSTRUMENTS AND TRANSMITTER RANGES SEE INSTRUMENT DATA SHEET (NOTE 2).
11. REFERENCED DOCUMENTS 5 & 6 ARE TO BE SUBMITTED TO GE (BWRSD) BY THE CUSTOMER/AE FOR (BWRSD) DESIGN COMPLETION.
12. POWER SOURCES TO THE LEVEL TRIP CHANNELS MUST BE FROM INDEPENDENT SOURCES.
13. CONTROL SWITCH OPERATES AN INTEGRAL RELAY TO RESET FEEDWATER VALVE "LOCK-UP CIRCUIT" AND EXTINGUISH "LOCK-UP" PILOT LAMP.

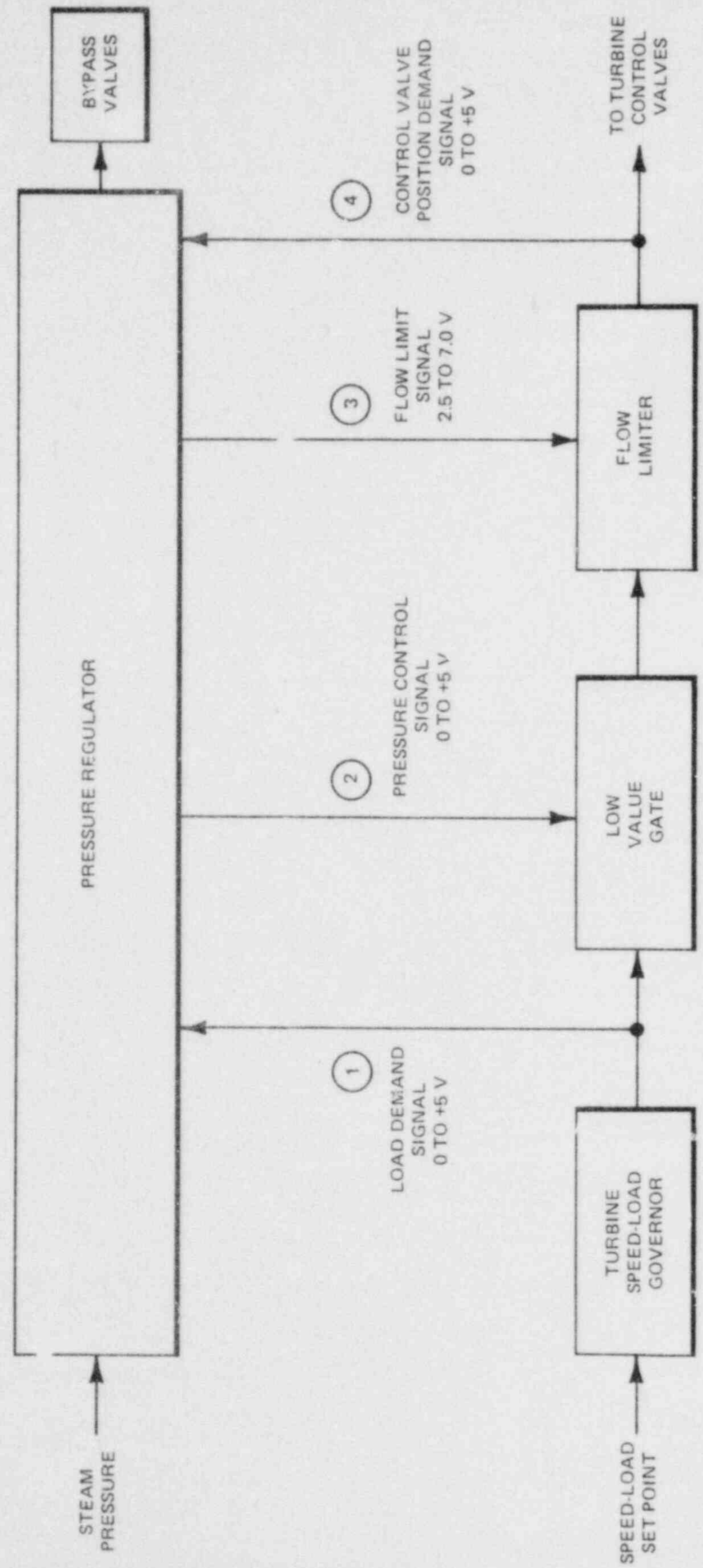
REFERENCE DOCUMENTS	MPL NO.
1. FEEDWATER CONTROL SYSTEM DESIGN SPEC	C34-4010
2. NUCLEAR BOILER SYSTEM P&ID	B22-1010
3. REACTOR RECIRCULATION SYSTEM FCD	B35-1020
4. REACTOR MANUAL CONTROL SYS ELEM DIAG (C12A)	C12-1030
5. FEEDWATER SYSTEM P&ID	BY OTHERS (NOTE 11)
6. MAIN TURBINE GENERATOR TRIP LOGIC ELEM	BY OTHERS (NOTE 11)
7. INSTRUMENT SYMBOLS	A42-1020
8. PIPING & INSTRUMENT SYMBOLS	A42-1010
9. PREOP. TEST SPEC	A15-3421
10. INSTRUMENT DATA SHEET	C34-3050

LEGEND:

- COMPUTER ANALOG INPUTS.

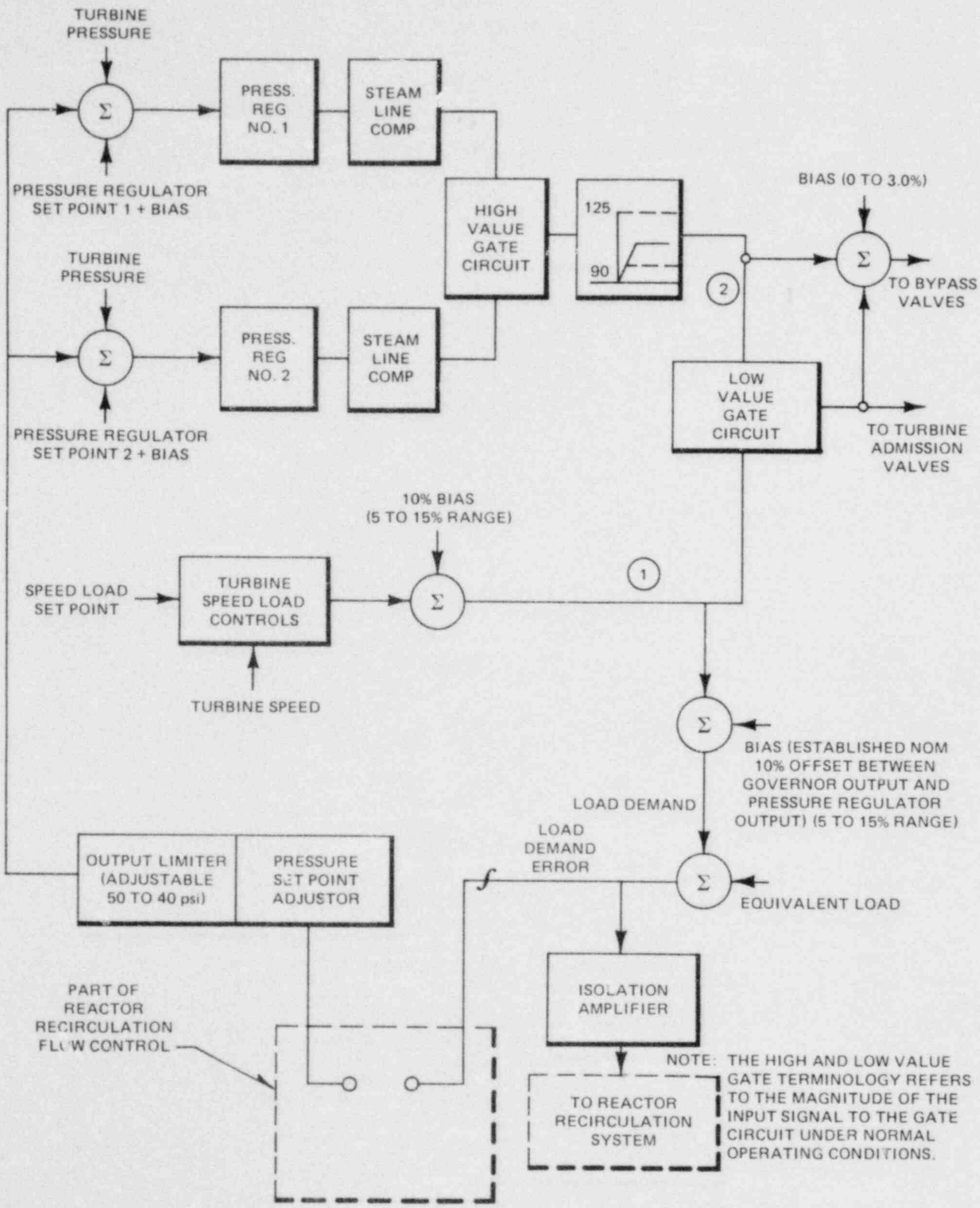
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FEEDWATER CONTROL SYSTEM IED SHEET 1	FIGURE 7.7-7

AMENDMENT NO. 10
July 1980



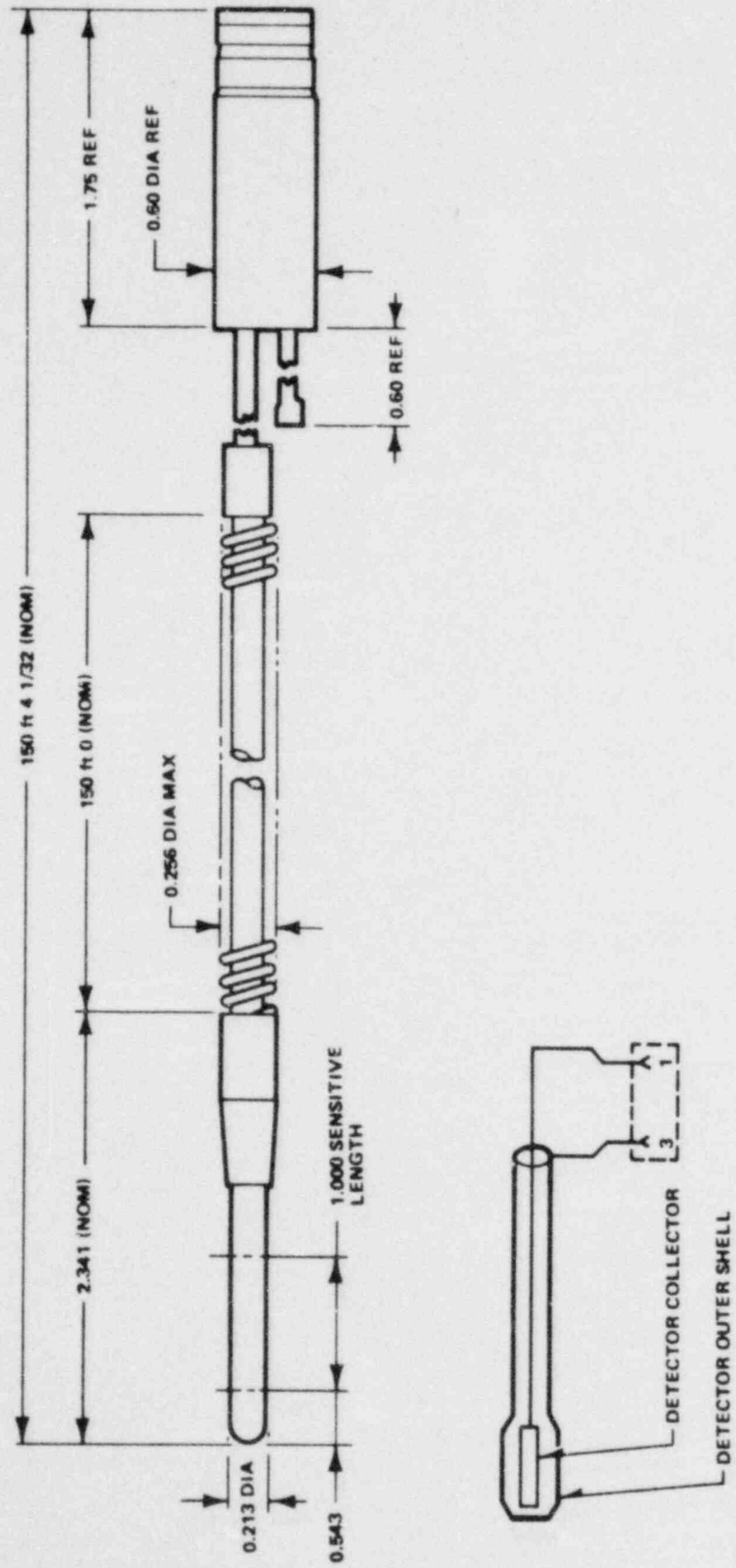
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SIMPLIFIED DIAGRAM PRESSURE CONTROL SYSTEM.	FIGURE 7.7-8

AMENDMENT NO. 10
July 1980



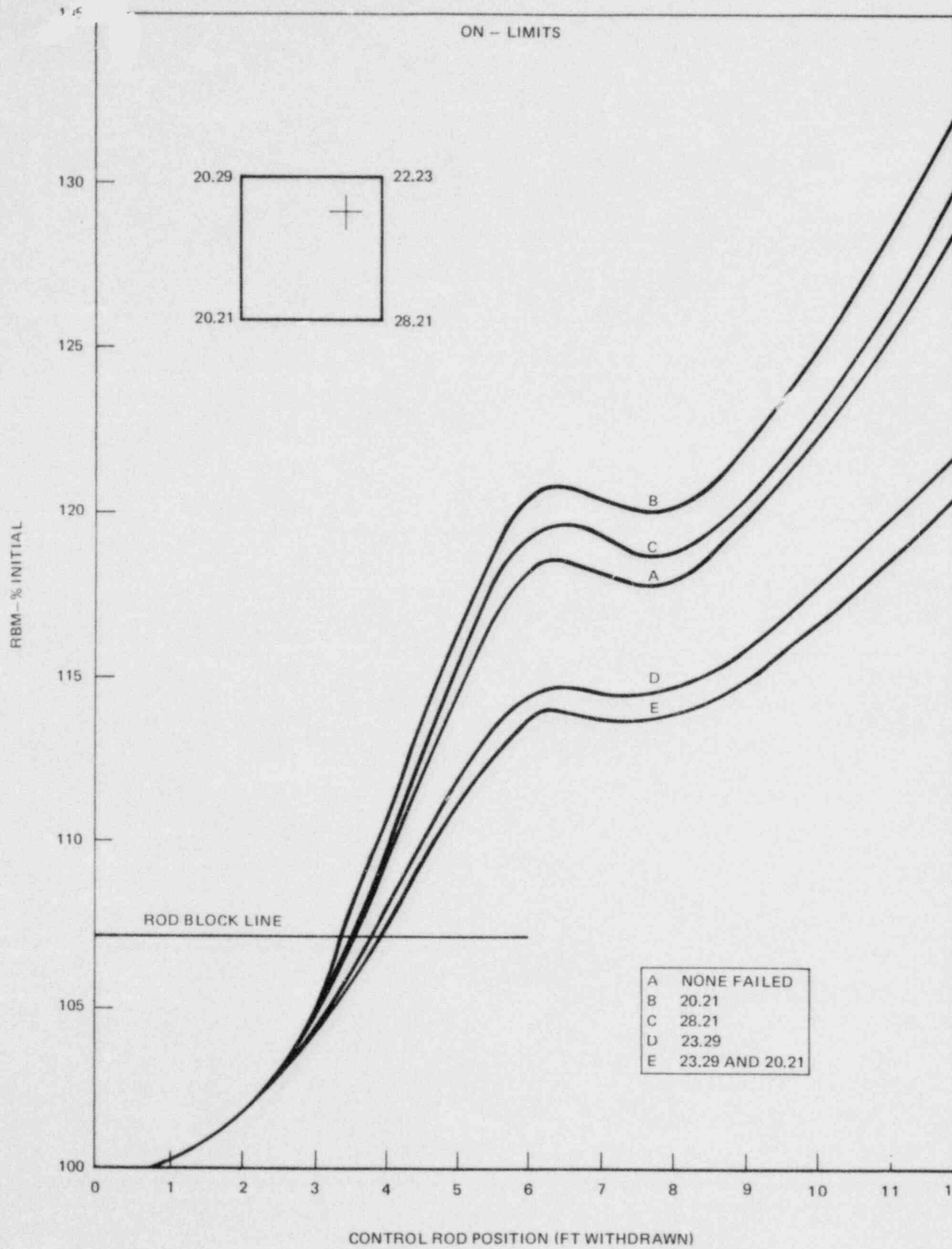
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SIMPLIFIED DIAGRAM, TURBINE PRESSURE AND SPEED LOAD CONTROL REQUIREMENTS.	FIGURE 7.7-9

AMENDMENT NO. 10
July 1980



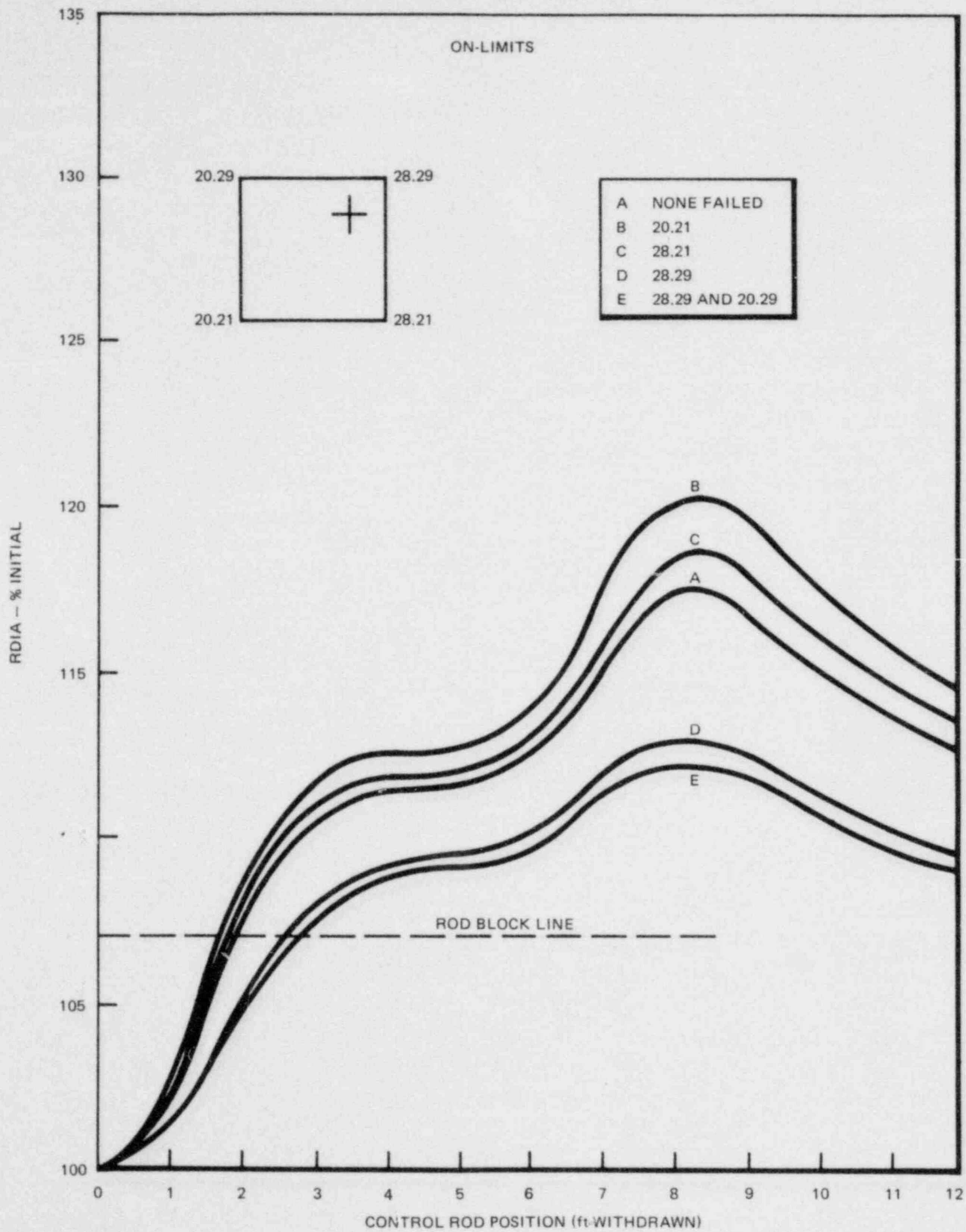
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	TRAVERSING IN-CORE PROBE ASSEMBLY.	FIGURE 7.7-10

AMENDMENT NO. 10
July 1980

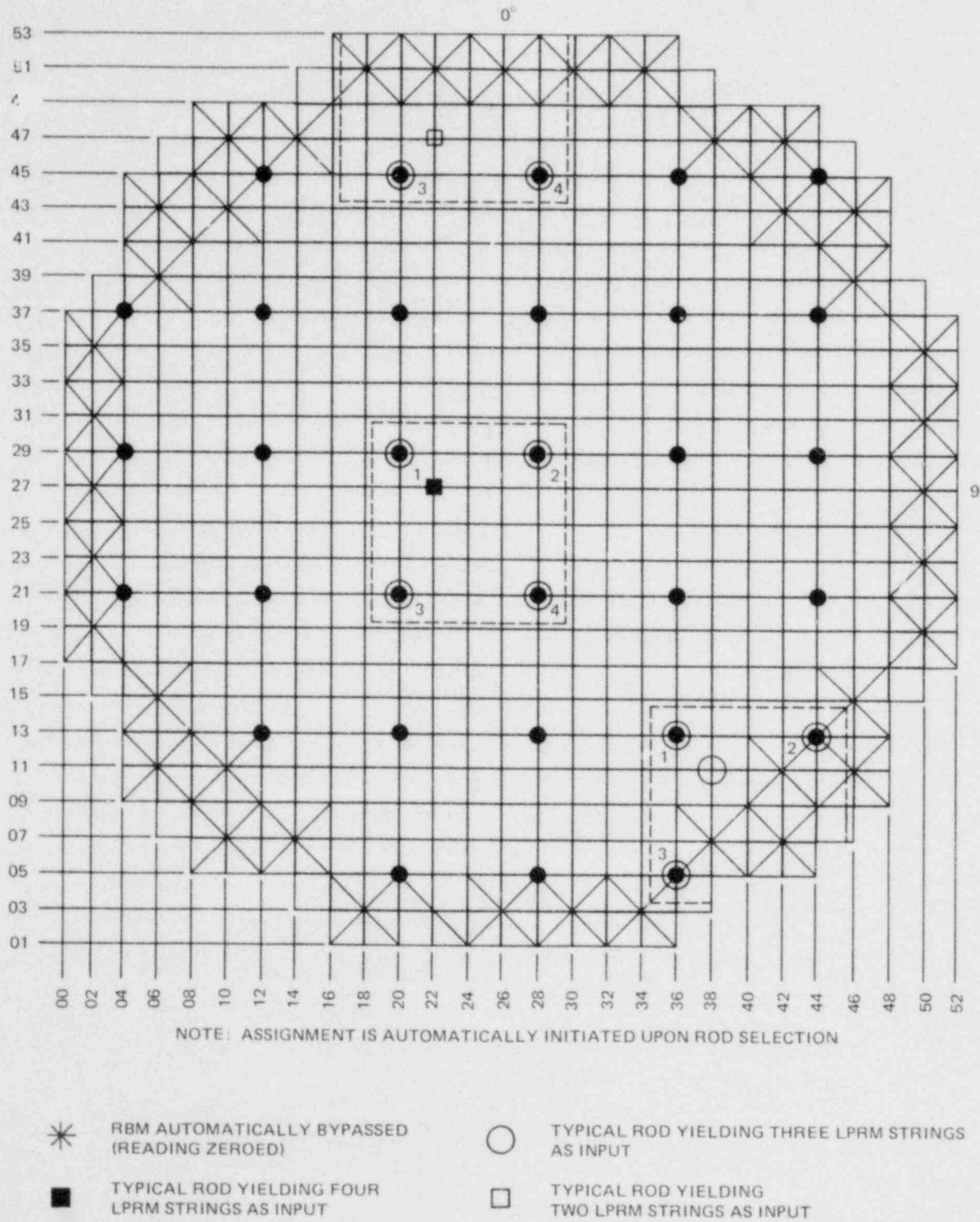


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RBM RESPONSE TO CONTROL ROD MOTION (CHANNEL A AND C).	FIGURE 7.7-11

AMENDMENT NO. 10
July 1980

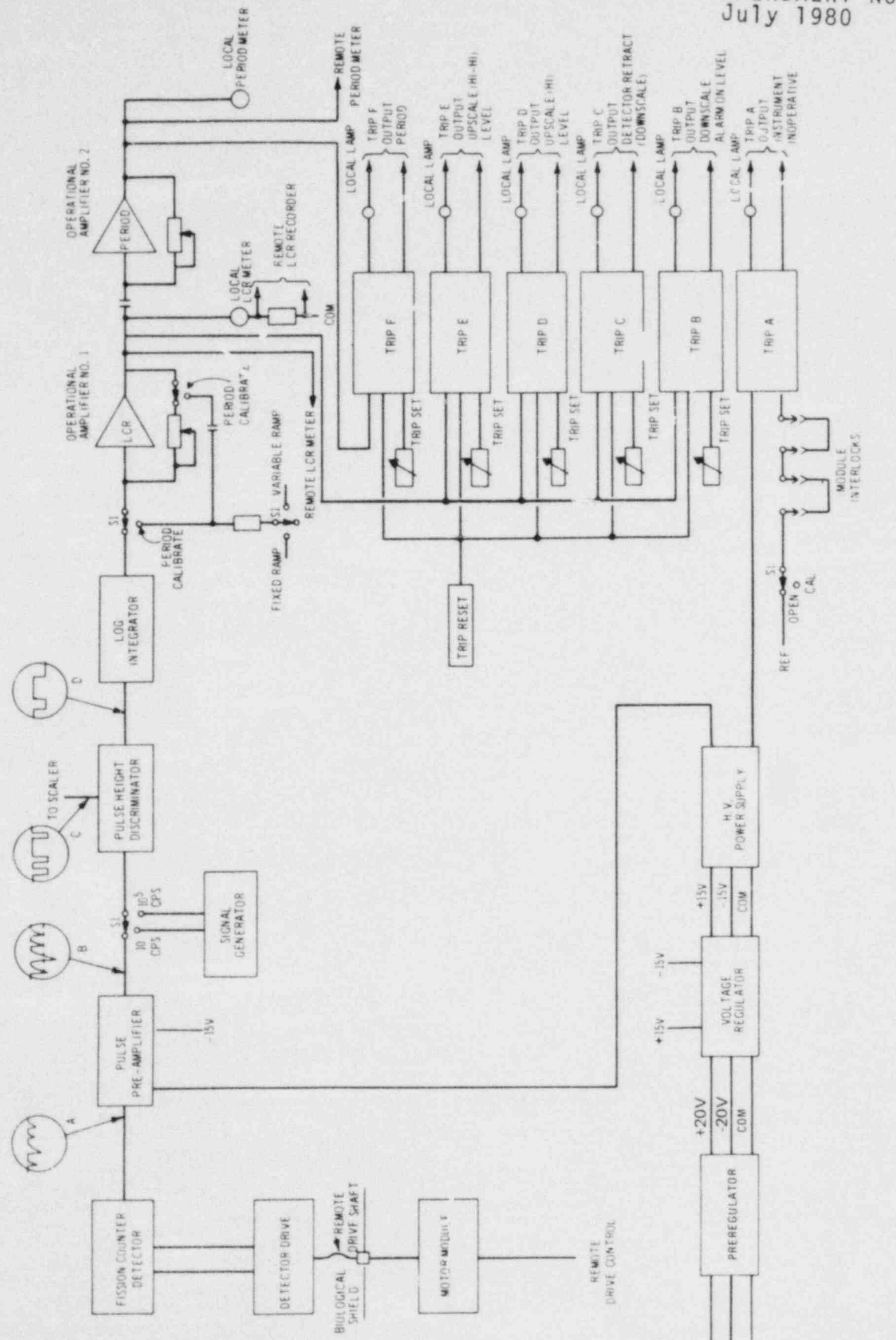


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RBM RESPONSE TO CONTROL ROD MOTION CHANNELS B AND D	FIGURE 7.7-12



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ASSIGNMENT OF LPRM INPUT TO RBM SYSTEM	FIGURE 7.7-13

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FUNCTIONAL BLOCK DIAGRAM OF SRM CHANNEL.	FIGURE 7.7-14

AMENDMENT NO. 10
July 1980

LATER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 7.7-15

TABLE 15.6-13

LOSS-OF-COOLANT ACCIDENT

(DESIGN BASIS ANALYSIS)

ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (CURIES)

ISOTOPE	1 MIN	30 MIN	1 HR	2 HR	4 HR	8 HR	12 HR	1 DAY	4 DAY	30 DAY
I-131	2.2E 07	2.2E 07	2.2E 07	2.2E 07	2.1E 07	2.1E 07	2.1E 07	2.0E 07	1.5E 07	1.6E 06
I-132	3.3E 07	2.8E 07	2.4E 07	1.8E 07	9.8E 06	2.9E 06	8.6E 05	2.3E 04	7.1E-06	0.
I-133	4.9E 07	4.8E 07	4.7E 07	4.5E 07	4.3E 07	3.7E 07	3.3E 07	2.2E 07	2.0E 06	1.9E-03
I-134	5.6E 07	3.8E 07	2.6E 07	1.2E 07	2.4E 06	1.0E 05	4.2E 03	3.2E-01	0.	0.
I-135	4.4E 07	4.2E 07	4.0E 07	3.6E 07	2.9E 07	1.9E 07	1.2E 07	3.5E 06	1.8E 03	0.
TOTAL	2.0E 08	1.8E 08	1.6E 08	1.3E 08	1.1E 08	8.0E 07	6.7E 07	4.5E 07	1.7E 07	1.6E 06
Kr-83m	1.4E 07	1.2E 07	9.9E 06	6.8E 06	3.2E 06	7.2E 05	1.6E 05	1.8E 03	3.4E-09	0.
Kr-85m	4.5E 07	4.2E 07	3.8E 07	3.3E 07	2.4E 07	1.3E 07	7.0E 06	1.1E 06	1.5E 01	0.
Kr-85	1.4E 06	1.4E 06	1.4E 06	1.4E 06	1.4E 06	1.4E 06	1.4E 06	1.4E 06	1.4E 06	1.4E 06
Kr-87	8.0E 07	6.1E 07	4.7E 07	2.7E 07	9.0E 06	1.0E 06	1.1E 05	1.6E 02	0.	0.
Kr-88	1.1E 08	9.8E 07	8.6E 07	6.7E 07	4.1E 07	1.5E 07	5.7E 06	2.9E 05	5.1E-03	0.
Kr-89	1.1E 08	1.9E 05	2.6E 02	4.9E-04	0.	0.	0.	0.	0.	0.
Xe-131m	9.0E 05	9.0E 05	9.0E 05	8.9E 05	8.9E 05	8.8E 05	8.7E 05	8.5E 05	7.1E 05	1.6E 05
Xe-133m	4.8E 06	4.8E 06	4.7E 06	4.7E 06	4.5E 06	4.3E 06	4.1E 06	3.5E 06	1.4E 06	4.3E 02
Xe-133	1.9E 08	1.9E 08	1.9E 08	1.9E 08	1.9E 08	1.9E 08	1.8E 08	1.7E 08	1.1E 08	3.8E 06
Xe-135m	5.1E 07	1.4E 07	3.6E 06	2.3E 05	1.0E 03	1.9E-02	3.7E-07	0.	0.	0.
Xe-135	1.9E 08	1.8E 08	1.7E 08	1.6E 08	1.4E 08	1.0E 08	7.5E 07	3.0E 07	1.3E 05	0.
Xe-137	1.5E 08	7.8E 05	3.5E 03	6.8E-02	2.6E-11	0.	0.	0.	0.	0.
Xe-138	1.6E 08	3.8E 07	8.8E 06	4.7E 05	1.3E 03	1.1E-02	8.8E-08	0.	0.	0.
TOTAL NG	1.1E 09	6.5E 08	5.7E 08	4.9E 08	4.1E 08	3.2E 08	2.8E 08	2.1E 08	1.2E 08	5.4E 06

AMENDMENT NO. 10
July 1980

TABLE 15.6-14

LOSS-OF-COOLANT ACCIDENT

(DESIGN BASIS ANALYSIS)

TOTAL ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

ISOTOPE	1 MIN	30 MIN	1 HR	2 HR	4 HR	8 HR	12 HR	1 DAY	4 DAY	30 DAY
I-131	1.8E 00	5.3E 01	1.1E 02	2.1E 02	4.6E 02	9.5E 02	1.4E 03	2.9E 03	1.0E 04	3.[illegible]E 04
I-132	2.7E 00	7.4E 01	1.4E 02	2.4E 02	4.0E 02	5.3E 02	5.7E 02	5.8E 02	5.8E 02	5.8E 02
I-133	3.9E 00	1.2E 02	2.3E 02	4.6E 02	9.7E 02	1.[illegible]E 03	2.7E 03	4.6E 03	8.0E 03	8.4E 03
I-134	4.6E 00	1.1E 02	1.9E 02	2.8E 02	3.4E 02	3.6E 02	3.6E 02	3.6E 02	3.6E 02	3.6E 02
I-135	3.6E 00	1.0E 02	2.0E 02	3.9E 02	7.6E 02	1.3E 03	1.7E 03	2.2E 03	2.4E 03	2.4E 03
TOTAL	1.6E 01	4.6E 02	8.7E 02	1.6E 03	2.9E 03	5.0E 03	6.7E 03	1.1E 04	2.2E 04	4.3E 04
Kr-83m	5.0E 01	1.4E 03	2.5E 03	4.3E 03	7.2E 03	9.3E 03	9.7E 03	9.8E 03	9.8E 03	9.8E 03
Kr-85m	1.6E 02	4.6E 03	8.8E 03	1.6E 04	3.4E 04	5.6E 04	6.8E 04	7.9E 04	8.2E 04	8.2E 04
Kr-85	5.0E 00	1.5E 02	3.0E 02	6.0E 02	1.5E 03	3.2E 03	5.0E 03	1.0E 04	4.1E 04	2.9E 05
Kr-87	2.8E 02	7.5E 03	1.3E 04	2.1E 04	3.1E 04	3.5E 04	3.6E 04	3.6E 04	3.6E 04	3.6E 04
Kr-88	3.9E 02	1.1E 04	2.1E 04	3.7E 04	7.0E 04	1.0E 05	1.1E 05	1.2E 05	1.2E 05	1.2E 05
Kr-89	4.3E 02	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03
Xe-131m	3.2E 00	9.5E 01	1.9E 02	3.8E 02	9.3E 02	2.0E 03	3.1E 03	6.3E 03	2.3E 04	9.0E 04
Xe-133m	1.7E 01	5.0E 02	1.0E 03	2.0E 03	4.8E 03	1.0E 04	1.5E 04	2.9E 04	8.0E 04	1.1E 05
Xe-133	6.8E 02	2.0E 04	4.1E 04	8.2E 04	2.0E 05	4.3E 05	6.6E 05	1.3E 06	4.4E 06	1.0E 07
Xe-135m	1.9E 02	3.1E 03	3.9E 03	4.2E 03	4.2E 03	4.2E 03	4.2E 03	4.2E 03	4.2E 03	4.2E 03
Xe-135	6.5E 02	1.9E 04	3.8E 04	7.[illegible]E 04	1.6E 05	3.1E 05	4.2E 05	6.0E 05	7.2E 05	7.2E 05
Xe-137	5.7E 02	3.4E 03	3.4E 03	3.4E 03	3.4E 03	3.4E 03	3.4E 03	3.4E 03	3.4E 03	3.4E 03
Xe-138	5.7E 02	9.1E 03	1.1E 04	1.2E 04	1.2E 04	1.2E 04	1.2E 04	1.2E 04	1.2E 04	1.2E 04
TOTAL NG	4.0E 03	8.3E 04	1.5E 05	2.6E 05	5.3E 05	9.8E 05	1.3E 06	2.2E 06	5.5E 06	1.2E 07

TABLE 15.6-15

LOSS-OF-COOLANT ACCIDENT

(DESIGN BASIS ANALYSIS)

ACTIVITY RELEASED IN ENVIRONMENT (CURIES) VIA MSIV-LCS ONLY

ISOTOPE	4 HR	8 HR	12 HR	1 DAY	4 DAY	30 DAY
I-131	4.1E 01	1.2E 02	2.0E 02	4.4E 02	1.7E 03	5.2E 03
I-132	2.6E 01	4.8E 01	5.4E 01	5.7E 01	5.7E 01	5.7E 01
I-133	8.5E 01	2.4E 02	3.7E 02	6.8E 02	1.3E 03	1.3E 03
I-134	1.1E 01	1.4E 01	1.4E 01	1.4E 01	1.4E 01	1.4E 01
I-135	6.2E 01	1.5E 02	2.1E 02	3.0E 02	3.3E 02	3.3E 02
TOTAL	2.3E 02	5.8E 02	8.6E 02	1.5E.03	3.3E 03	7.0E 03
Kr-83m	9.2E 02	1.6E 03	1.7E 03	1.8E 03	1.8E 03	1.8E 03
Kr-85m	5.5E 03	1.2E 04	1.6E 04	2.0E 04	2.1E 04	2.1E 04
Kr-85	2.7E 02	8.2E 02	1.4E 03	3.0E 03	1.3E 04	9.5E 04
Kr-87	3.2E 03	4.6E 03	4.8E 03	4.8E 03	4.8E 03	4.8E 03
Kr-88	1.0E 04	2.0E 04	2.4E 04	2.6E 04	2.6E 04	2.6E 04
Kr-89	3.6E-09	3.6E-09	3.6E-09	3.6E-09	3.6E-09	3.6E-09
Xe-131m	1.7E 02	5.1E 02	8.5E 02	1.8E 03	7.2E 03	2.9E 04
Xe-133m	8.9E 02	2.6E 03	4.2E 03	8.6E 03	2.4E 04	3.5E 04
Xe-133	3.7E 04	1.1E 05	1.8E 05	3.8E 05	1.4E 06	3.3E 06
Xe-135m	8.3E 00	8.3E 00	8.3E 00	8.3E 00	8.3E 00	8.3E 00
Xe-135	2.9E 04	7.4E 04	1.1E 05	1.7E 05	2.0E 05	2.0E 05
Xe-137	6.1E-07	6.1E-07	6.1E-07	6.1E-07	6.1E-07	6.1E-07
Xe-138	1.5E 01	1.6E 01	1.6E 01	1.6E 01	1.6E 01	1.6E 01
TOTAL NG	8.7E 04	2.3E 05	3.4E 05	6.2E 05	1.7E 06	3.7E 06

TABLE 15.6-16

LOSS-OF-COOLANT ACCIDENT

(DESIGN BASIS ANALYSIS)

RADIOLOGICAL EFFECTS

1. Total Effect (Including MSIV-LCS Leakage)

	WHOLE BODY DOSE (REM)	THYROID DOSE (REM)
Exclusion Area (1950M - 2 hr)	2.54	12.0
Low Population Zone (4827M - 30 days)	2.78	33.7

2. Effects from MSIV-LCS Only

	WHOLE BODY DOSE (REM)	THYROID DOSE (REM)
Low Population Zone (4827M - 30 days)	0.57	4.86

TABLE 15.6-17

ISOTOPIC SPIKING ACTIVITY

ISOTOPE NAME	THE 95th CUMULATIVE PROBABILITY SPIKING ACTIVITY (Ci/Bundle)
I-131	2.14
I-132	3.21
I-133	5.03
I-134	5.44
I-135	4.79
Kr-83m	9.04 - 1*
Kr-85m	2.23 + 0
Kr-85	4.70 - 1
Kr-87	4.33 + 0
Kr-88	6.12 + 0
Kr-89	7.96 + 0
Xe-131m	6.60 - 2
Xe-133m	3.26 - 1
Xe-133	1.16 + 1
Xe-135m	1.80 + 0
Xe-135	1.10 + 1
Xe-137	1.05 + 1
Xe-138	1.06 + 1

* 9.04 - 1 = 9.04×10^{-1}

TABLE 15.6-18

LOSS-OF-COOLANT ACCIDENT

(REALISTIC ANALYSIS)

TOTAL ACTIVITY AIRBORNE IN THE CONTAINMENT (CURIES)

ISOTOPE	1 MIN	1 HR	2 HRS	8 HRS	1 DAY	4 DAYS	30 DAYS
I-131	8.73E 01	8.70E 01	8.66E 01	8.48E 01	8.01E 01	6.18E 01	6.57E 00
I-132	1.36E 02	1.01E 02	7.45E 01	1.21E 01	9.38E-02	2.94E-11	0.
I-133	2.08E 02	2.02E 02	1.95E 02	1.60E 02	9.37E 01	8.46E 00	7.99E-09
I-134	2.31E 02	1.06E 02	4.79E 01	4.14E-01	1.30E-06	0.	0.
I-135	2.01E 02	1.81E 02	1.63E 02	8.67E 01	1.61E 01	8.23E-03	0.
TOTAL	8.63E 02	6.76E 02	5.67E 02	3.44E 02	1.90E 02	7.02E 01	6.57E 00
Kr-83m	6.86E 02	4.75E 02	3.27E 02	3.46E 01	8.65E-02	1.63E-13	0.
Kr-85m	1.70E 03	1.46E 03	1.25E 03	4.95E 02	4.16E 01	5.90E-04	0.
Kr-85	3.74E 02	3.74E 02	3.74E 02	3.74E 02	3.74E 02	3.74E 02	3.72E 02
Kr-87	3.28E 03	1.91E 03	1.11E 03	4.15E 01	6.55E-03	0.	0.
Kr-88	4.65E 03	3.65E 03	2.85E 03	6.44E 02	1.22E 01	2.14E-07	0.
Kr-89	4.88E 03	1.15E-02	2.19E-08	0.	0.	0.	0.
Xe-131m	5.04E 01	5.03E 01	5.02E 01	4.95E 01	4.76E 01	4.00E 01	8.90E 00
Xe-133m	2.49E 02	2.46E 02	2.43E 02	2.25E 02	1.82E 02	7.17E 01	2.22E-02
Xe-133	8.82E 03	8.78E 03	8.73E 03	8.45E 03	7.74E 03	5.21E 03	1.72E 02
Xe-135m	1.31E 03	9.06E 01	5.98E 00	4.95E-07	0.	0.	0.
Xe-135	8.36E 03	7.77E 03	7.20E 03	4.57E 03	1.36E 03	5.85E 00	0.
Xe-137	6.69E 03	1.58E-01	3.10E-06	0.	0.	0.	0.
Xe-138	7.74E 03	4.34E 02	2.32E 01	5.35E-07	0.	0.	0.
TOTAL	4.88E 04	2.52E 04	2.22E 04	1.49E 04	9.76E 03	5.71E 03	5.52E 02

TABLE 15.6-19

LOSS-OF-COOLANT ACCIDENT

(REALISTIC ANALYSIS)

TOTAL ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

ISOTOPE	1 MIN	1 HR	2 HRS	8 HRS	1 DAY	4 DAYS	30 DAYS
I-131	3.79E-06	2.59E-04	5.17E-04	2.12E-03	6.17E-03	2.18E-02	6.89E-02
I-132	5.80E-06	5.71E-04	6.10E-04	1.27E-03	1.39E-03	1.39E-03	1.39E-03
I-133	8.86E-06	6.10E-04	1.20E-03	4.47E-03	1.06E-02	1.84E-02	1.92E-02
I-134	9.86E-06	4.80E-04	7.33E-04	9.04E-04	9.06E-04	9.06E-04	9.06E-04
I-135	8.54E-06	5.68E-04	1.08E-03	3.34E-03	5.41E-03	5.88E-03	5.88E-03
TOTAL	3.68E-05	2.27E-03	4.11E-03	1.21E-02	2.44E-02	4.84E-02	9.63E-02
Kr-83m	2.08E-03	1.22E-01	2.05E-01	5.39E-01	5.67E-01	5.67E-01	5.67E-01
Kr-85m	5.14E-03	3.33E-01	6.19E-01	2.41E 00	3.31E 00	3.38E 00	3.38E 00
Kr-85	1.25E-03	7.88E-02	1.58E-01	9.20E-01	2.76E 00	1.09E 01	7.79E 01
Kr-87	9.91E-03	5.36E-01	8.47E-01	1.83E 00	1.86E 00	1.86E 00	1.86E 00
Kr-88	1.41E-02	8.71E-01	1.55E 00	5.00E 00	5.79E 00	5.80E 00	5.80E 00
Kr-89	1.62E-02	9.44E-02	9.44E-02	1.38E-01	1.38E-01	1.38E-01	1.38E-01
Xe-131m	1.52E-04	1.06E-02	2.12E-02	1.23E-01	3.61E-01	1.32E 00	5.09E 00
Xe-133m	7.50E-04	5.21E-02	1.03E-01	5.82E-01	1.58E 00	4.19E 00	5.83E 00
Xe-133	2.65E-02	1.85E 00	3.70E 00	2.12E 01	6.10E 01	2.01E 02	4.73E 02
Xe-135m	4.03E-03	9.89E-02	1.05E-01	1.55E-01	1.55E-01	1.55E-01	1.55E-01
Xe-135	2.52E-02	1.70E 00	3.27E 00	1.55E 01	2.85E 01	3.40E 01	3.40E 01
Xe-137	2.18E-02	1.52E-01	1.52E-01	2.22E-01	2.22E-01	2.22E-01	2.22E-01
Xe-138	2.38E-02	5.50E-01	5.80E-01	8.48E-01	8.48E-01	8.48E-01	8.48E-01
TOTAL	1.51E-01	6.45E 00	1.15E 01	4.94E 01	1.07E 02	2.65E 02	6.10E 02

TABLE 15.6-20

LOSS-OF-COOLANT ACCIDENT

(REALISTIC ANALYSIS)

ACTIVITY RELEASE TO ENVIRONMENT (CURIES)

VIA MSIV-LCS ONLY

ISOTOPE	8 HRS	1 DAY	4 DAYS	30 DAYS
I-131	6.65E-05	1.94E-04	6.84E-04	2.15E-03
I-132	3.96E-05	4.35E-05	4.35E-05	4.35E-05
I-133	1.41E-04	3.33E-04	5.79E-04	6.04E-04
I-134	2.84E-05	2.85E-05	2.85E-05	2.85E-05
I-135	1.05E-04	1.70E-04	1.85E-04	1.85E-04
TOTAL	3.81E-04	7.68E-04	1.52E-03	3.01E-03
Kr-83m	1.69E-01	1.78E-01	1.78E-01	1.78E-01
Kr-85m	7.56E-01	1.04E 00	1.06E 00	1.06E 00
Kr-85	2.89E-01	8.67E-01	3.46E 00	2.51E 01
Kr-87	5.76E-01	5.83E-01	5.83E-01	5.83E-01
Kr-88	1.57E 00	1.82E 00	1.82E 00	1.82E 00
Kr-89	4.32E-02	4.32E-02	4.32E-02	4.32E-02
Xe-131m	3.86E-02	1.14E-01	4.16E-01	1.63E 00
Xe-133m	1.83E-01	4.96E-01	1.32E 00	1.84E 00
Xe-133	6.67E 00	1.92E 01	6.35E 01	1.50E 02
Xe-135m	4.85E-02	4.85E-02	4.85E-02	4.85E-02
Xe-135	4.86E 00	8.95E 00	1.07E 01	1.07E 01
Xe-137	6.96E-02	6.96E-02	6.96E-02	6.96E-02
Xe-138	2.66E-01	2.66E-01	2.66E-01	2.66E-01
TOTAL	1.55E 01	3.36E 01	8.34E 01	1.94E 02

TABLE 15.6-21

LOSS-OF-COOLANT ACCIDENT

(REALISTIC ANALYSIS)

RADIOLOGICAL EFFECTS

	WHOLE BODY DOSE (REM)	THYROID DOSE (REM)
Exclusion Area (1950 Meters)	1.07E-04	3.03E-05
Low Population Zone (4827 Meters)	1.18E-04	7.49E-05

APPENDIX E

CROSS-INDEX OF FSAR FIGURES*

TO

ENGINEERING DRAWING NUMBERS

* Burns and Roe Figures only.
For GE Figures, see Table 1.8-2

When the pump is operating at 25% speed (15 Hz) the head provided by the elevation of the reactor water level above the recirculation pump and the jet pumps is sufficient to provide the required NPSH. When the pump is operating at 100% speed (60 Hz), most of the NPSH is supplied by the subcooling provided by feedwater flow. Accurate temperature detectors are provided in the recirculation lines and the steamlines (beyond the second isolation valves). The difference between these two readings is a direct measurement of the subcooling. If the subcooling falls below 10.1°F, the 60-Hz power supply is tripped to the 15-Hz supply to prevent cavitation of the recirculation pump and jet pump.

The transition to 100% speed is made after thermal power is sufficient to provide enough feedwater subcooling to prevent flow control valve cavitation. At this point the flow control valves are moved to the minimum positions and the pumps brought to 100% speed. The 25% to 100% speed transition is prevented by interlocks requiring sufficient thermal power and the control valve in minimum position.

During preparation for hydrostatic tests, the nuclear system temperature must be raised above the vessel nil ductility transition temperature limit. The vessel is heated by operating the residual heat removal pumps and by core decay heat. When NPSH is adequate, the recirculation pumps may be run at 100% speed to provide additional heat.

Each recirculation pump is equipped with mechanical shaft seal assemblies. Each of these assemblies consists of two seals built into a cartridge that can be readily replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for pump design pressures so that any one seal can adequately limit leakage in the event that the other seal should fail. The pump shaft passes through a breakdown bushing to reduce leakage in the event of a gross failure of both shaft seals. The cavity temperature and pressure drop across each individual seal is monitored.

Each recirculation pump motor is a vertical, solid-shaft, totally enclosed, air/water-cooled induction motor. The combined rotating inertias of the recirculation pump and motor determine the coastdown for both the loss-of-coolant accident and the turbine-generator trip transients. This inertia requirement is met without a flywheel.

The pump discharge flow control valve throttles the discharge flow of the pump. The recirculation loop flow rate can be changed rapidly within the expected flow range in response to rapid changes in system demand. The maximum required valve actuator stroking rate is 10% to 11% of full stroke per second.

A design objective for the recirculation system equipment is to provide units that will not require removal from the system for rework or overhaul. Pump casings and valve bodies are designed for a 40-year life and will be welded to the pipe. The pump drive motor, impeller, and wear rings and flow control valve internals are designed for as long a life as is practical. Pump mechanical seal parts and the valve packing are expected to have a life expectancy which affords convenient replacement during the refueling outages.

The recirculation system piping is of all-welded construction and is designed and constructed to meet the requirements of ASME Section III, Class I.

H.2.1.1.2 Piping and Instrumentation Diagram (P&ID)

See Figures 5.4-2a, b, and c.

H.2.1.1.3 Functional Control Diagrams

See Figure H.A-1a through H.A-1h

H.2.1.1.4 Elementary Diagrams

See Table 1.7-1 Drawing No. B35-1030.

H.2.1.2 Thermal, Hydraulic, and Physical Data

H.2.1.2.1 Recirculation Loop Piping

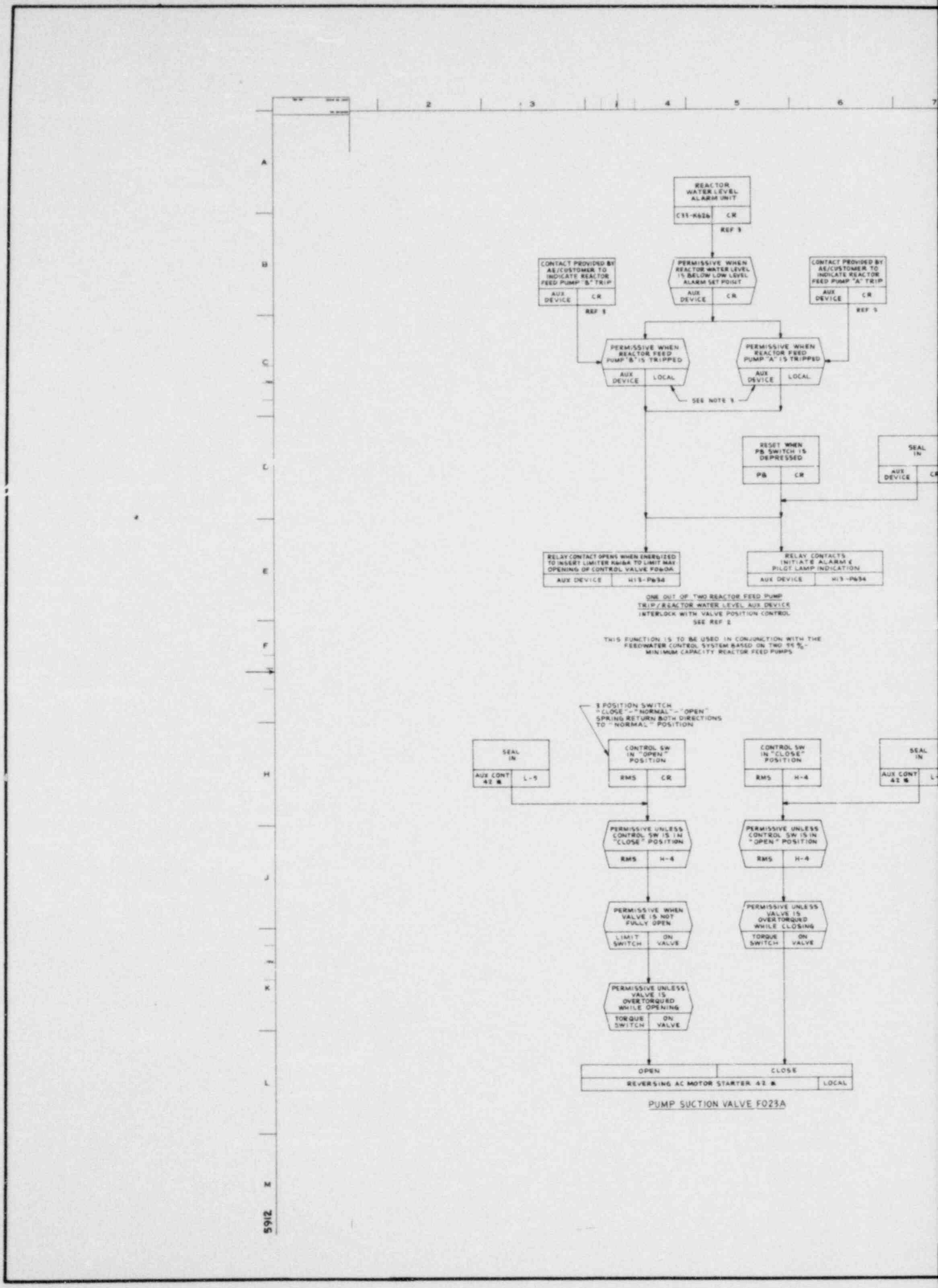
H.2.1.2.1.1 Piping Description

a.	Number of loops	2
b.	General arrangement	See Figures H.2.1-1 through H.2.1-5
c.	Pump suction line nominal size	24 in.
d.	Pump discharge line nominal size	24 in.
e.	Ring header nominal size	16 in.
f.	Number of risers	10
g.	External riser nominal size	12 in.
h.	Internal riser nominal size	10 in.

ATTACHMENT H.A. - REACTOR RECIRCULATION SYSTEM

DESIGN DRAWINGS

CONTENTS


REACTOR WATER LEVEL ALARM UNIT
C33-K626 CR
REF 3
CONTACT PROVIDED BY AE/CUSTOMER TO INDICATE REACTOR FEED PUMP "B" TRIP
AUX DEVICE CR
REF 3
PERMISSIVE WHEN REACTOR WATER LEVEL IS BELOW LOW LEVEL ALARM SET POINT
AUX DEVICE CR
CONTACT PROVIDED BY AE/CUSTOMER TO INDICATE REACTOR FEED PUMP "A" TRIP
AUX DEVICE CR
REF 3
PERMISSIVE WHEN REACTOR FEED PUMP "B" IS TRIPPED
AUX DEVICE LOCAL
PERMISSIVE WHEN REACTOR FEED PUMP "A" IS TRIPPED
AUX DEVICE LOCAL
SEE NOTE 3
RESET WHEN PB SWITCH IS DEPRESSED
PB CR
SEAL IN
AUX DEVICE
RELAY CONTACT OPENS WHEN ENERGIZED TO INSERT LIMITER K616A TO LIMIT MAX OPENING OF CONTROL VALVE F060A
AUX DEVICE H13-P634
RELAY CONTACTS INITIATE ALARM & PILOT LAMP INDICATION
AUX DEVICE H13-P634
ONE OUT OF TWO REACTOR FEED PUMP TRIP/REACTOR WATER LEVEL AUX DEVICE INTERLOCK WITH VALVE POSITION CONTROL
SEE REF 2
THIS FUNCTION IS TO BE USED IN CONJUNCTION WITH THE FEEDWATER CONTROL SYSTEM BASED ON TWO 55%-MINIMUM CAPACITY REACTOR FEED PUMPS
3 POSITION SWITCH "CLOSE"-"NORMAL"-"OPEN" SPRING RETURN BOTH DIRECTIONS TO "NORMAL" POSITION
SEAL IN
AUX CONT 42 L-5
CONTROL SW IN "OPEN" POSITION
RMS CR
CONTROL SW IN "CLOSE" POSITION
RMS H-4
SEAL IN
AUX CONT 42
PERMISSIVE UNLESS CONTROL SW IS IN "CLOSE" POSITION
RMS H-4
PERMISSIVE UNLESS CONTROL SW IS IN "OPEN" POSITION
RMS H-4
PERMISSIVE WHEN VALVE IS NOT FULLY OPEN
LIMIT SWITCH ON VALVE
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE CLOSING
TORQUE SWITCH ON VALVE
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE OPENING
TORQUE SWITCH ON VALVE
OPEN CLOSE
REVERSING AC MOTOR STARTER 42 LOCAL
PUMP SUCTION VALVE F023A
A B C D E F H J K L M
2 3 4 5 6 7
5912

8 9 10 11 12 13 CI

SEAL IN ONLY WHEN AUX DEVICE IS ENERGIZED

NOTES:

1. FUNCTION IS SHOWN FOR RECIRCULATION SYSTEM "A" & IS TYPICAL FOR RECIRCULATION SYSTEM "B" EXCEPT FOR LETTER SUFFIXES.
2. AUXILIARY DEVICES ARE SHOWN WHERE NECESSARY TO CLARIFY THE FUNCTION.
3. FOR DETECTION OF A ONE OUT OF TWO REACTOR FEED PUMP TRIP, AND COINCIDENT OR SUBSEQUENT LOW REACTOR WATER LEVEL, LOGIC IS PROVIDED TO INITIATE CLOSURE OF THE RECIRCULATION FLOW CONTROL VALVE TO REDUCE THE REACTOR POWER LEVEL TO WITHIN THE CAPACITY OF THE REMAINING FEED PUMP. THE MOTOR DRIVEN PUMP FAILURE CONTACTS SHALL BE PROVIDED BY THE AE/CUSTOMER.
4. NUMEROUS PILOT LAMP ALARM INDICATIONS ARE ALSO PROVIDED ON THE ANALOG CONTROL SYSTEM PANEL H13-P604 TO COMPLIMENT THESE REMOTE ALARMS. THE INDICATIONS ARE MORE EXTENSIVE AND INDICATE ALL FAULTS THAT GIVE RISE TO THESE FUNCTIONALLY GROUPED CONTROL ROOM ALARMS.
5. THE CUSTOMER/AE SHALL BE RESPONSIBLE FOR CONFIRMING DEVICES AS SHOWN ARE PROVIDED OR OTHERWISE CORRECTED ON THE FCD TO AGREE WITH CUSTOMER/AE INTERFACING SCOPE OF SUPPLY.

LEGEND:

SWGR = PUMP DRIVE MOTOR SWITCHGEAR

* = SWITCHGEAR DEVICE FUNCTION NO. ANSI SPEC. C37.2

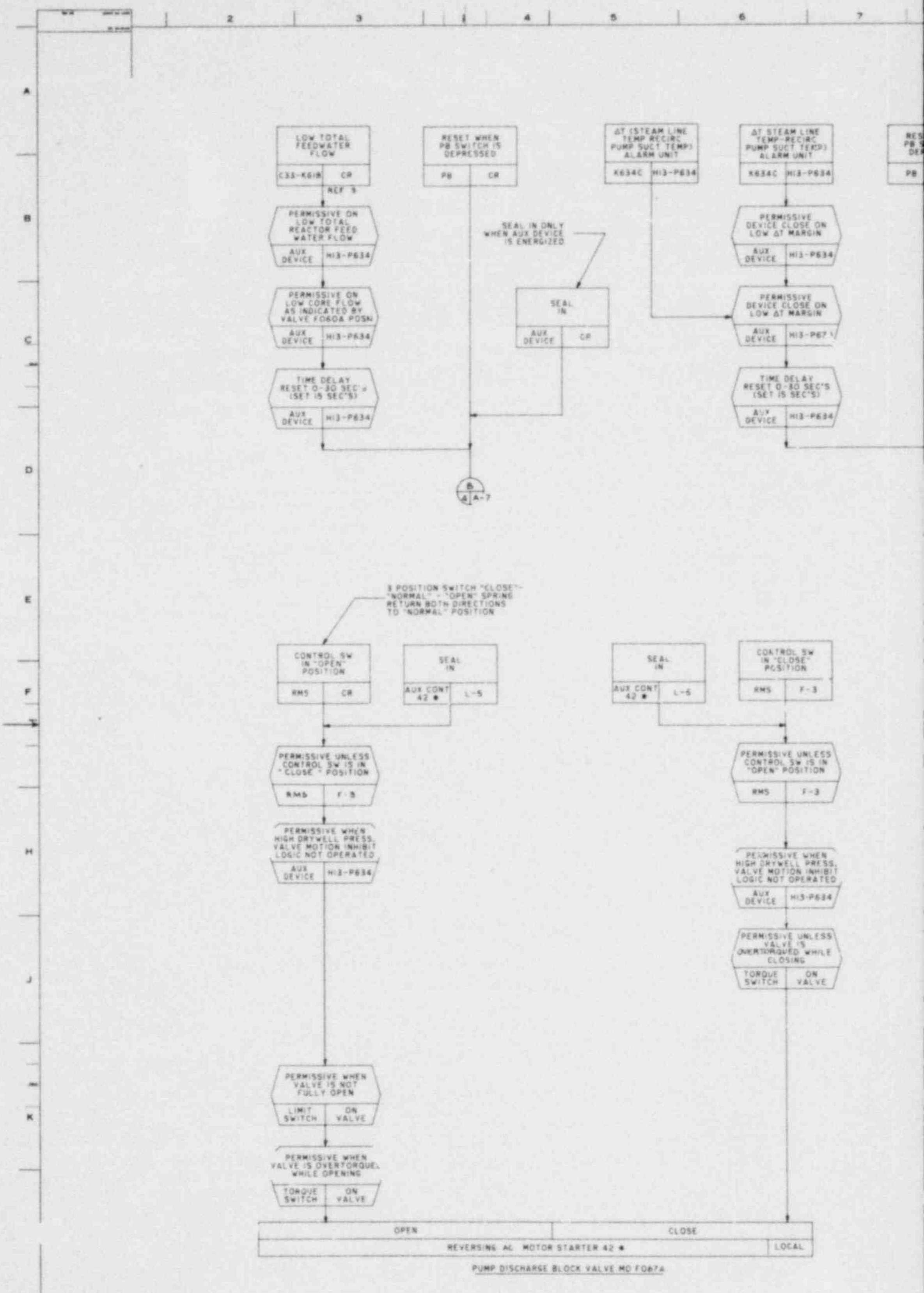
ATWS = AUTOMATIC TRIP WITHOUT SCRAM

REFERENCE DOCUMENTS:	MPL NO.
1. REACTOR RECIRC SYS VALVE FLOW CONT DES SPEC	B35-4010
2. REACTOR RECIRC SYS VALVE FLOW CONT P&ID	B35-1010
3. FEEDWATER CONTROL SYSTEM IED	C34-1010
4. NUCLEAR BOILER FCD	B22-1030
5. LOGIC SYMBOLS	A42-1030

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR RECIRCULATION SYSTEM FCD
SHEET 1

FIGURE H.A-1a



2
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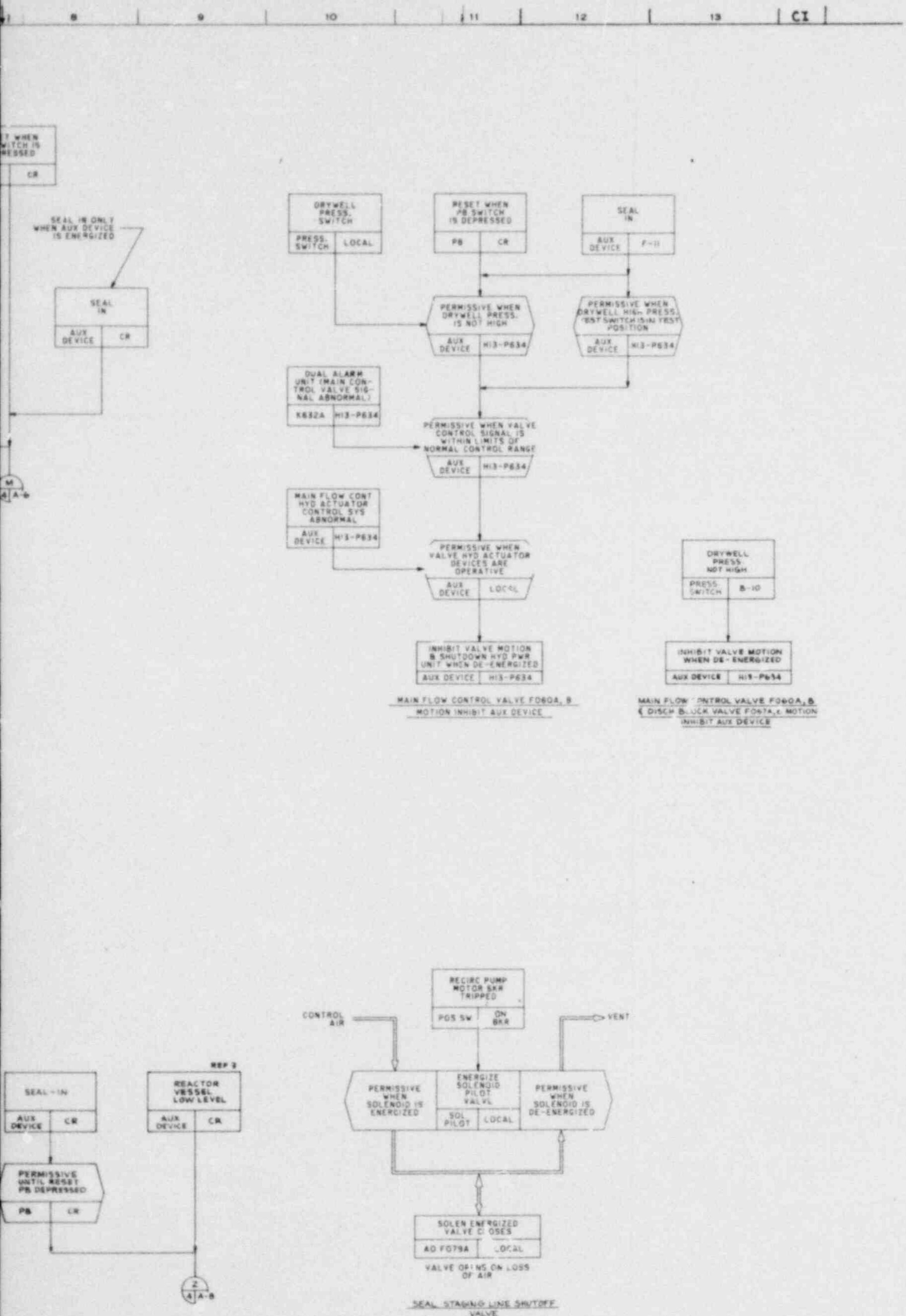
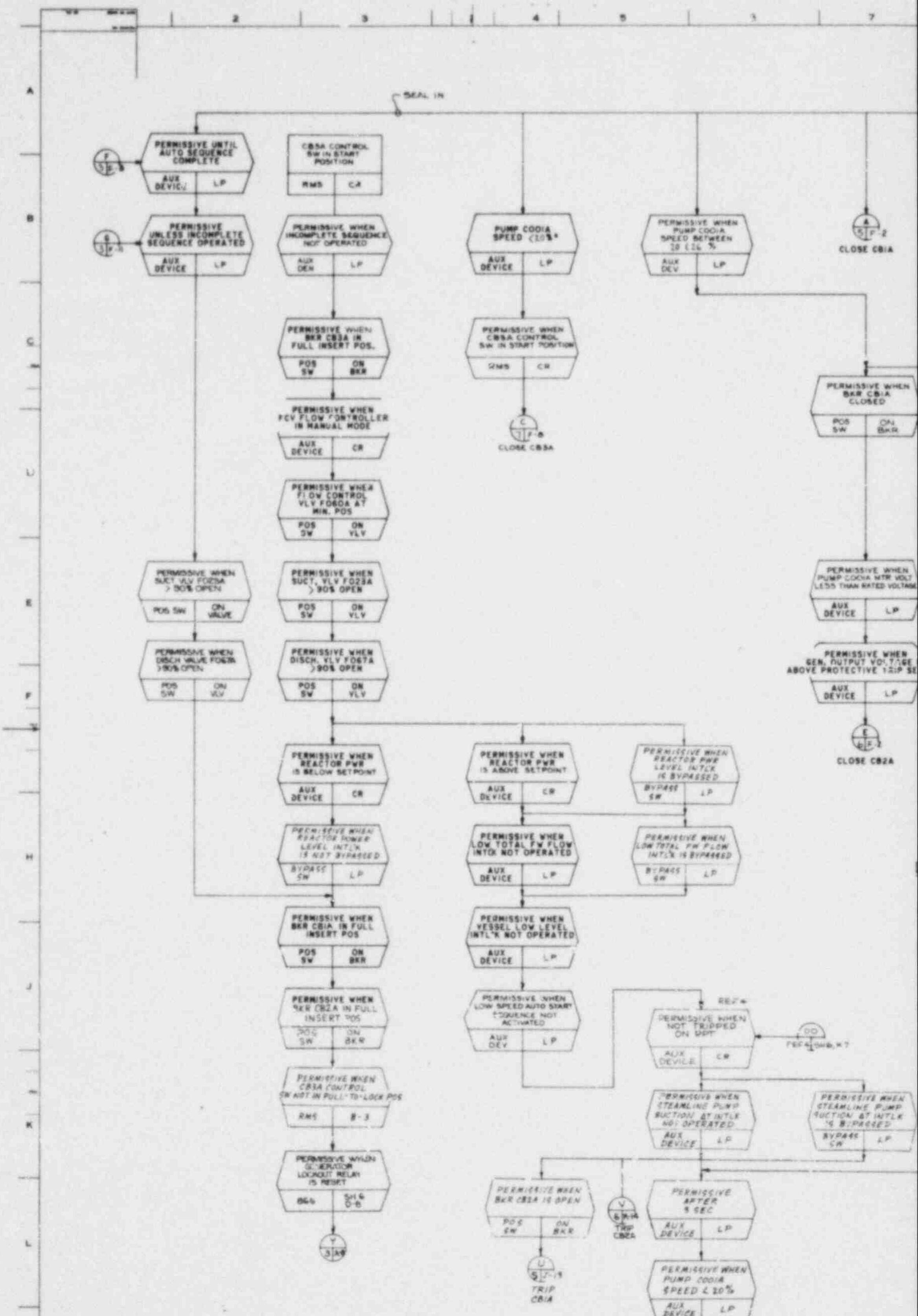
4
5
6
7
A
B
C
D
E
F
H
J
K
M
LOW TOTAL FEEDWATER FLOW
C33-K618
CR
REF 3
PERMISSIVE ON LOW TOTAL REACTOR FEED WATER FLOW
AUX DEVICE
H13-P634
PERMISSIVE ON LOW CORE FLOW AS INDICATED BY VALVE F060A POSN
AUX DEVICE
H13-P634
TIME DELAY RESET 0-30 SEC'S (SET 15 SEC'S)
AUX DEVICE
H13-P634
RESET WHEN PB SWITCH IS DEPRESSED
PB
CR
SEAL IN ONLY WHEN AUX DEVICE IS ENERGIZED
SEAL IN
AUX DEVICE
CR
A-7
ΔT (STEAM LINE TEMP RECIRC PUMP SUCT TEMP) ALARM UNIT
K634C
H13-P634
ΔT STEAM LINE TEMP-RECIRC PUMP SUCT TEMP) ALARM UNIT
K634C
H13-P634
PERMISSIVE DEVICE CLOSE ON LOW ΔT MARGIN
AUX DEVICE
H13-P634
PERMISSIVE DEVICE CLOSE ON LOW ΔT MARGIN
AUX DEVICE
TIME DELAY RESET 0-30 SEC'S (SET 15 SEC'S)
AUX DEVICE
H13-P634
3 POSITION SWITCH "CLOSE" - "NORMAL" - "OPEN" SPRING RETURN BOTH DIRECTIONS TO "NORMAL" POSITION
CONTROL SW IN "OPEN" POSITION
RMS
CR
SEAL IN
AUX CONT 42 *
L-5
SEAL IN
AUX CONT 42 *
L-5
CONTROL SW IN "CLOSE" POSITION
RMS
F-3
PERMISSIVE UNLESS CONTROL SW IS IN "CLOSE" POSITION
RMS
F-3
PERMISSIVE UNLESS CONTROL SW IS IN "OPEN" POSITION
RMS
F-3
PERMISSIVE WHEN HIGH DRYWELL PRESS. VALVE MOTION INHIBIT LOGIC NOT OPERATED
AUX DEVICE
H13-P634
PERMISSIVE WHEN HIGH DRYWELL PRESS. VALVE MOTION INHIBIT LOGIC NOT OPERATED
AUX DEVICE
H13-P634
PERMISSIVE UNLESS VALVE IS OVERTORQUED WHILE CLOSING
TORQUE SWITCH
ON VALVE
PERMISSIVE WHEN VALVE IS NOT FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE WHEN VALVE IS OVERTORQUEL WHILE OPENING
TORQUE SWITCH
ON VALVE
OPEN
CLOSE
REVERSING AC MOTOR STARTER 42 *
LOCAL
PUMP DISCHARGE BLOCK VALVE MO F067A

5911

AMENDMENT NO. 10
July 1980

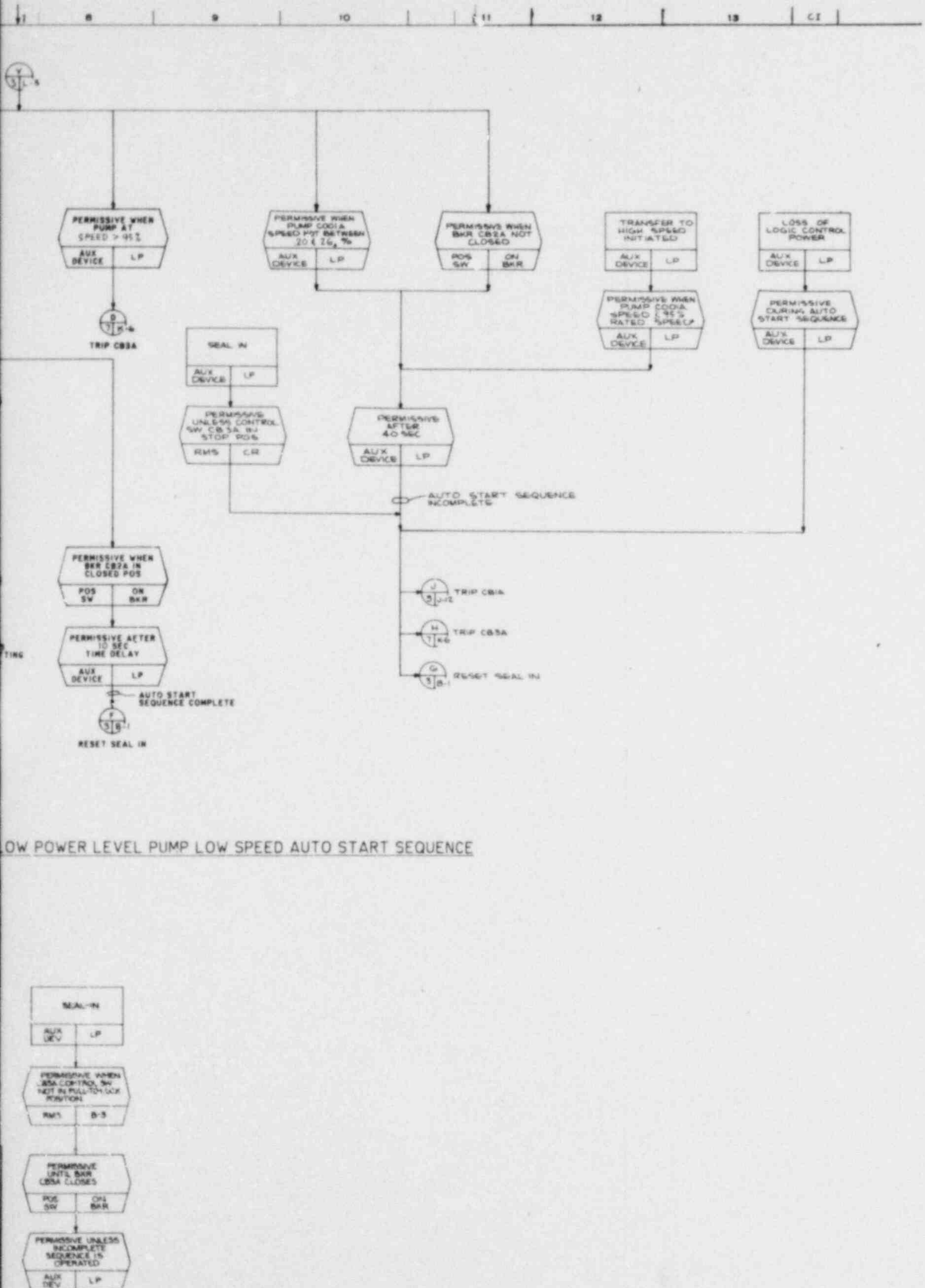


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 2	FIGURE H.A-1b



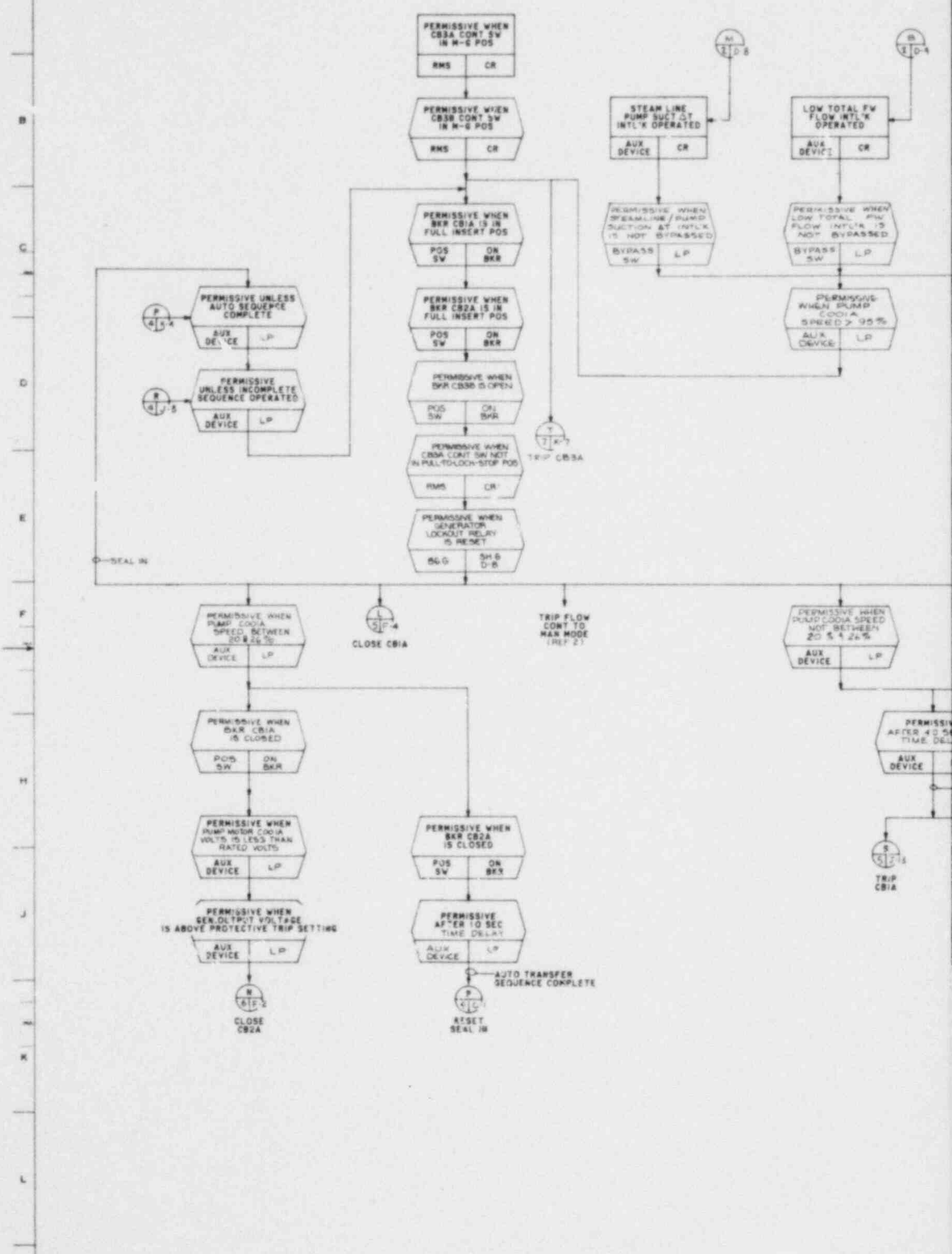
SEAL IN
PERMISSIVE UNTIL AUTO SEQUENCE COMPLETE
AUX DEVICE
LP
CB3A CONTROL SW IN START POSITION
RMS
CR
PERMISSIVE UNLESS INCOMPLETE SEQUENCE OPERATED
PERMISSIVE WHEN INCOMPLETE SEQUENCE NOT OPERATED
PUMP C001A SPEED
PERMISSIVE WHEN PUMP C001A SPEED BETWEEN
CLOSE CB1A
PERMISSIVE WHEN BKR CB3A IN FULL INSERT POS.
POS SW
ON BKR
PERMISSIVE WHEN CB3A CONTROL SW IN START POSITION
PERMISSIVE WHEN BKR CB1A CLOSED
PERMISSIVE WHEN FCV FLOW CONTROLLER IN MANUAL MODE
CLOSE CB3A
PERMISSIVE WHEN FLOW CONTROL VLV F060A AT MIN. POS
ON VLV
PERMISSIVE WHEN SUCT VLV F023A > 90% OPEN
ON VALVE
PERMISSIVE WHEN SUCT. VLV F023A > 90% OPEN
PERMISSIVE WHEN PUMP C001A MTR VOLT LESS THAN RATED VOLTAGE
PERMISSIVE WHEN DISCH VALVE F067A > 90% OPEN
PERMISSIVE WHEN DISCH. VLV F067A > 90% OPEN
PERMISSIVE WHEN GEN. OUTPUT VOLTAGE ABOVE PROTECTIVE TRIP SE
CLOSE CB2A
PERMISSIVE WHEN REACTOR PWR IS BELOW SETPOINT
PERMISSIVE WHEN REACTOR PWR IS ABOVE SETPOINT
PERMISSIVE WHEN REACTOR PWR LEVEL INTLK IS BYPASSED
BYPASS SW
PERMISSIVE WHEN REACTOR POWER LEVEL INTLK IS NOT BYPASSED
PERMISSIVE WHEN LOW TOTAL FW FLOW INTLK NOT OPERATED
PERMISSIVE WHEN LOW TOTAL FW FLOW INTLK IS BYPASSED
PERMISSIVE WHEN BKR CB1A IN FULL INSERT POS
PERMISSIVE WHEN VESSEL LOW LEVEL INTL'K NOT OPERATED
PERMISSIVE WHEN BKR CB2A IN FULL INSERT POS
PERMISSIVE WHEN LOW SPEED AUTO START SEQUENCE NOT ACTIVATED
AUX DEV
PERMISSIVE WHEN NOT TRIPPED ON RPT
PERMISSIVE WHEN CB3A CONTROL SW NOT IN PULL-TO-LOCK POS
B-3
PERMISSIVE WHEN STEAMLINE PUMP SUCTION AT INTLK NOT OPERATED
PERMISSIVE WHEN STEAMLINE PUMP SUCTION AT INTLK IS BYPASSED
PERMISSIVE WHEN GENERATOR LOCKOUT RELAY IS RESET
86G
PERMISSIVE WHEN BKR CB1A IS OPEN
TRIP CB2A
PERMISSIVE AFTER 5 SEC
TRIP CB1A
PERMISSIVE WHEN PUMP C001A SPEED < 20%
CLOSE CB3A
TRANSFER PUMP FROM LOW SPEED TO HIGH SPEED

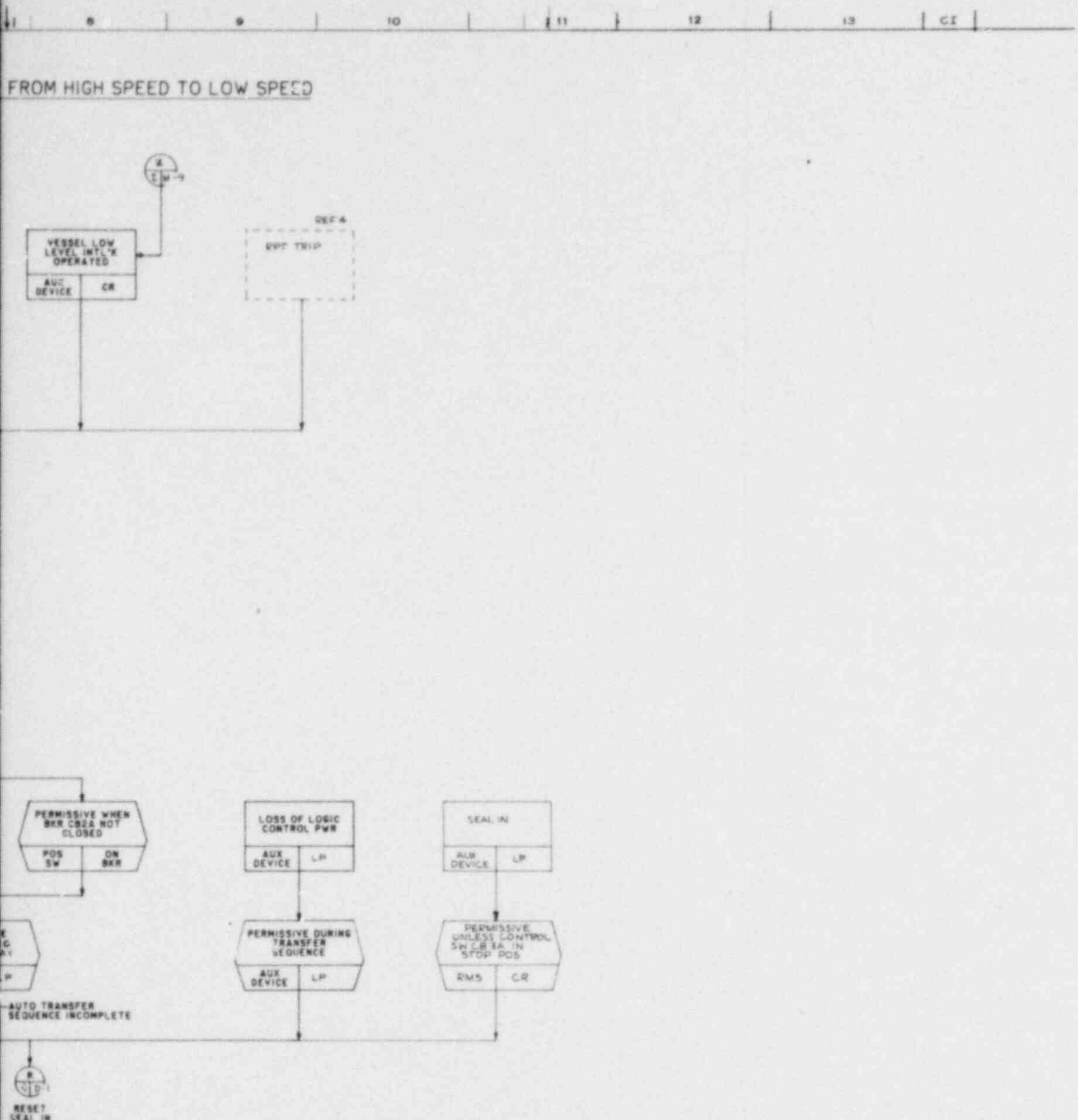
AMENDMENT NO. 10
July 1980



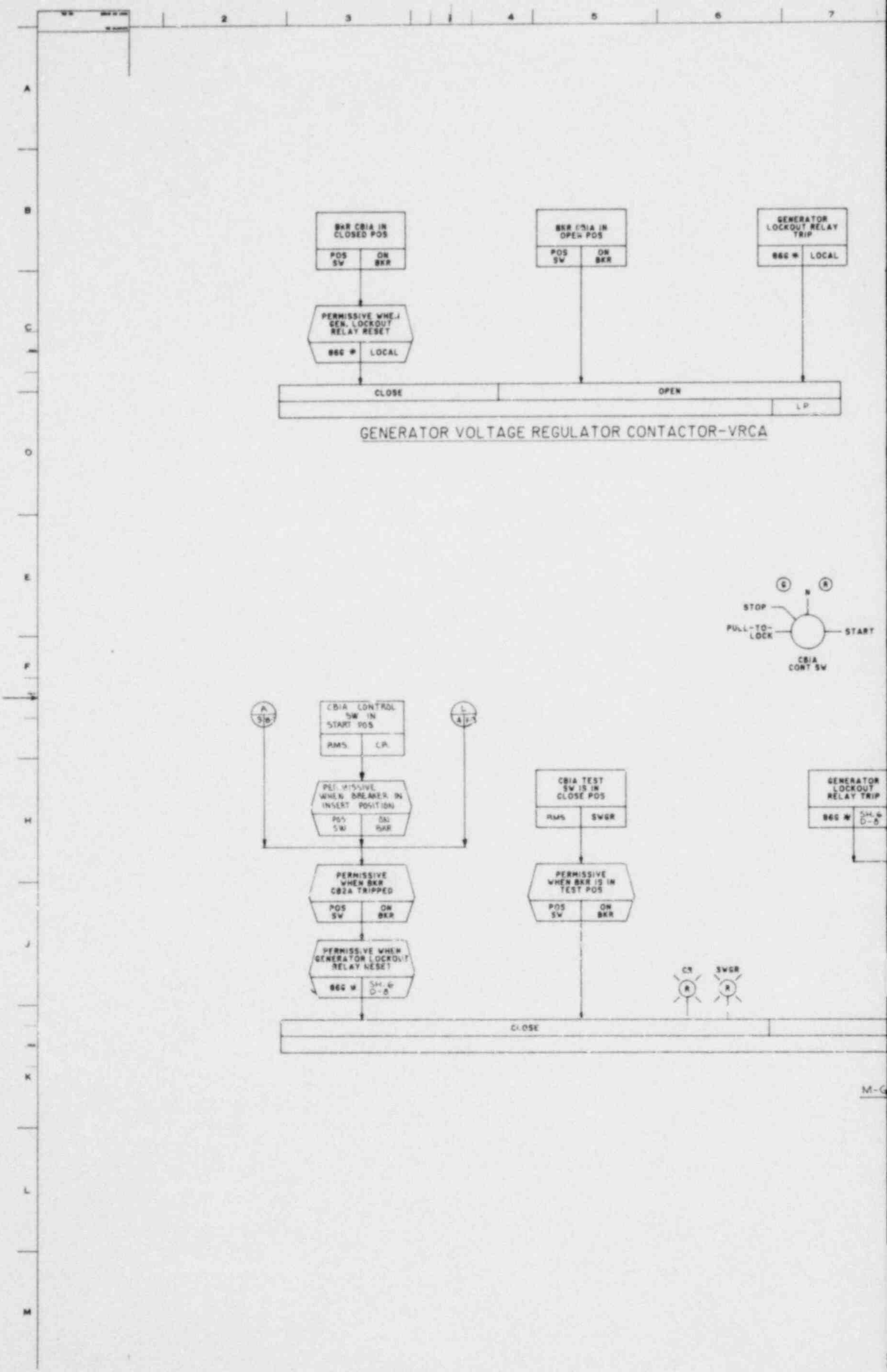
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHE... 3	FIGURE H.A-1c

TRIP OR TRANSFER PUMP





WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 4	FIGURE H.A- 1d



2
3
4
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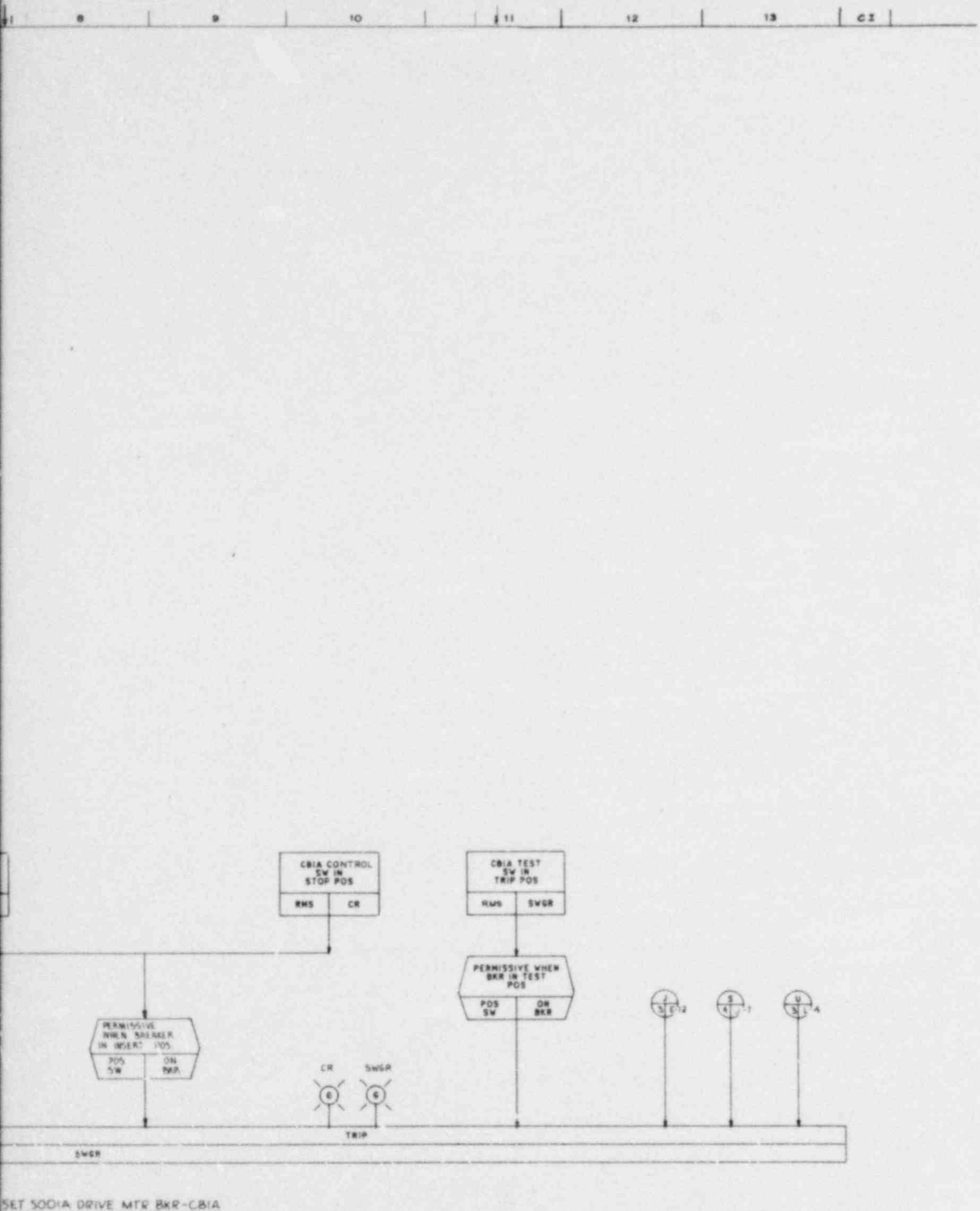
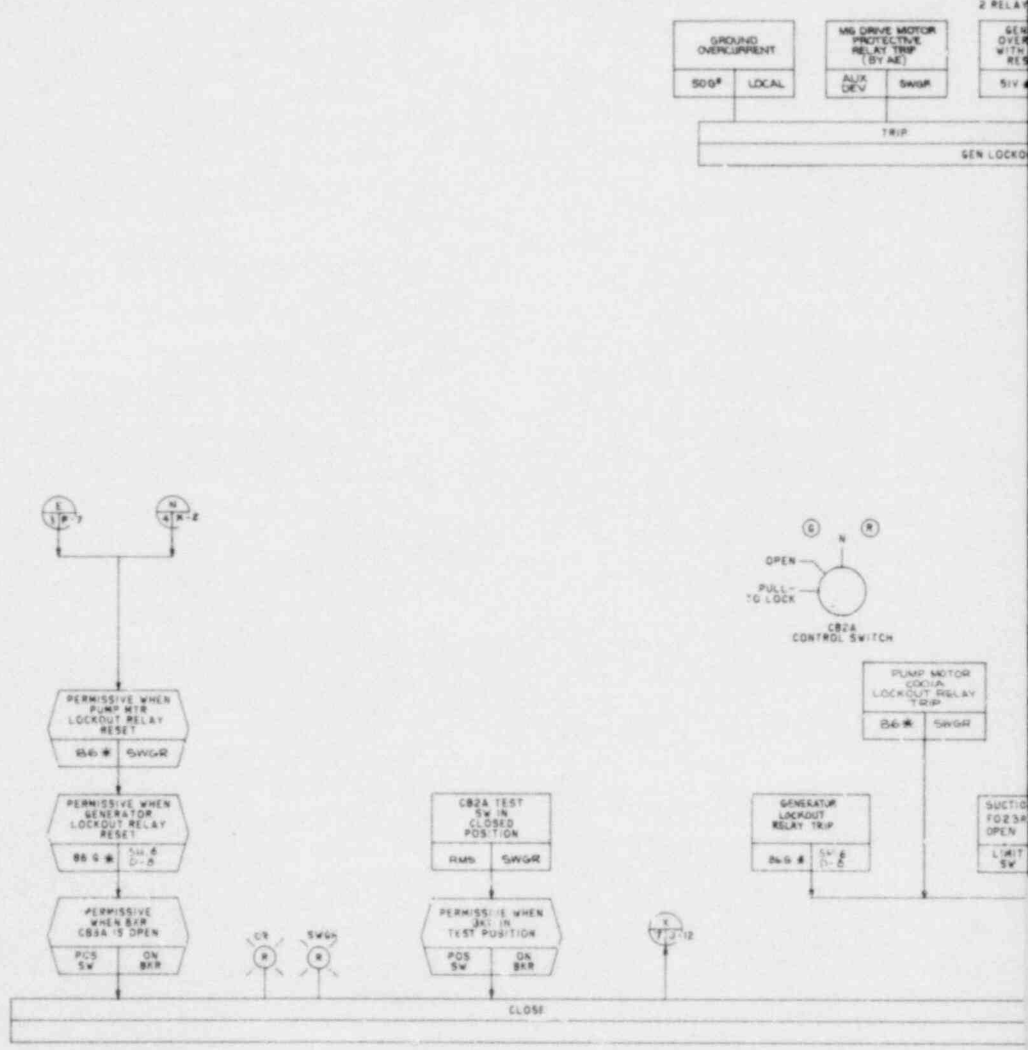
6
7
A
B
C
D
E
F
H
J
K
L
M
BKR CB1A IN CLOSED POS
POS SW
ON BKR
PERMISSIVE WHEN GEN. LOCKOUT RELAY RESET
86G #
LOCAL
BKR CB1A IN OPEN POS
POS SW
ON BKR
GENERATOR LOCKOUT RELAY TRIP
86G #
LOCAL
CLOSE
OPEN
LP
GENERATOR VOLTAGE REGULATOR CONTACTOR-VRCA
STOP
PULL-TO-LOCK
START
N
CB1A CONT SW
CB1A CONTROL SW IN START POS
RMS
CR
PERMISSIVE WHEN BREAKER IN INSERT POSITION
POS SW
ON BKR
PERMISSIVE WHEN BKR CB2A TRIPPED
POS SW
ON BKR
PERMISSIVE WHEN GENERATOR LOCKOUT RELAY RESET
86G #
SH. 6 D-8
CB1A TEST SW IS IN CLOSE POS
RMS
SWGR
PERMISSIVE WHEN BKR IS IN TEST POS
POS SW
ON BKR
GENERATOR LOCKOUT RELAY TRIP
86G #
SH. 6 D-8
CR
SWGR
R
CLOSE
M-G



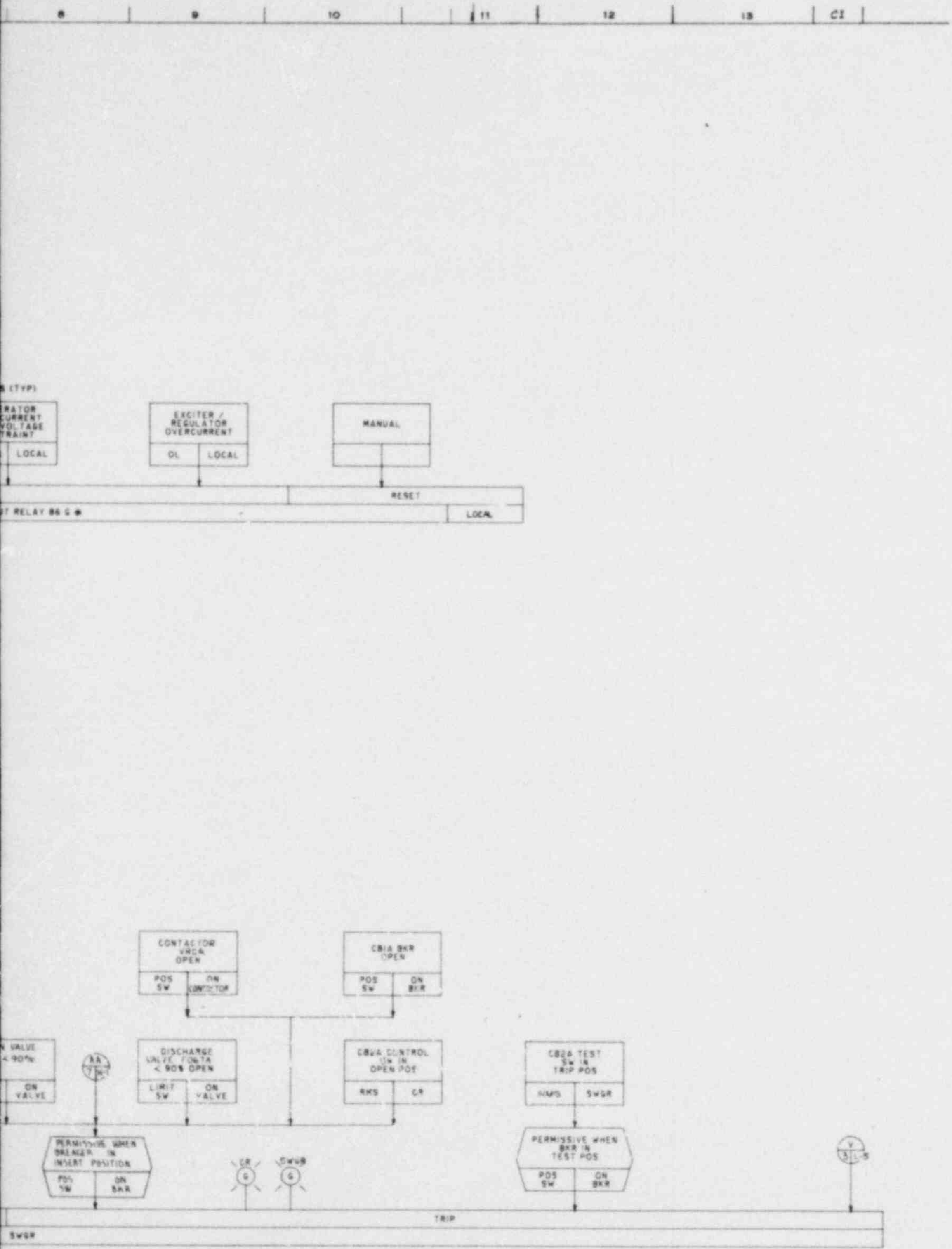
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 5	FIGURE H.A-1e



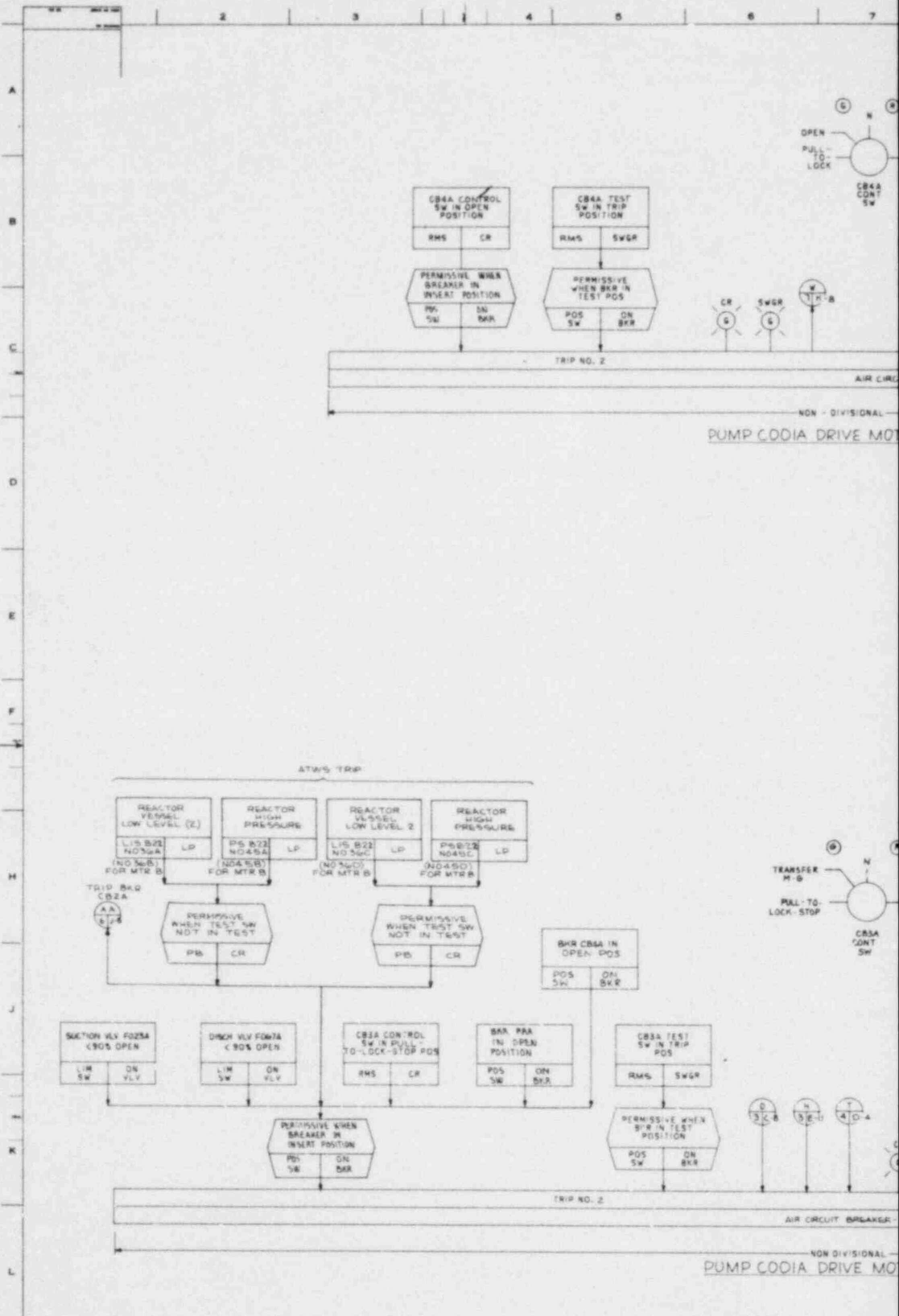
M-G SET 5001A GEN BKR-CB2A

NON-DIVISIONAL

AMENDMENT NO. 10
July 1980

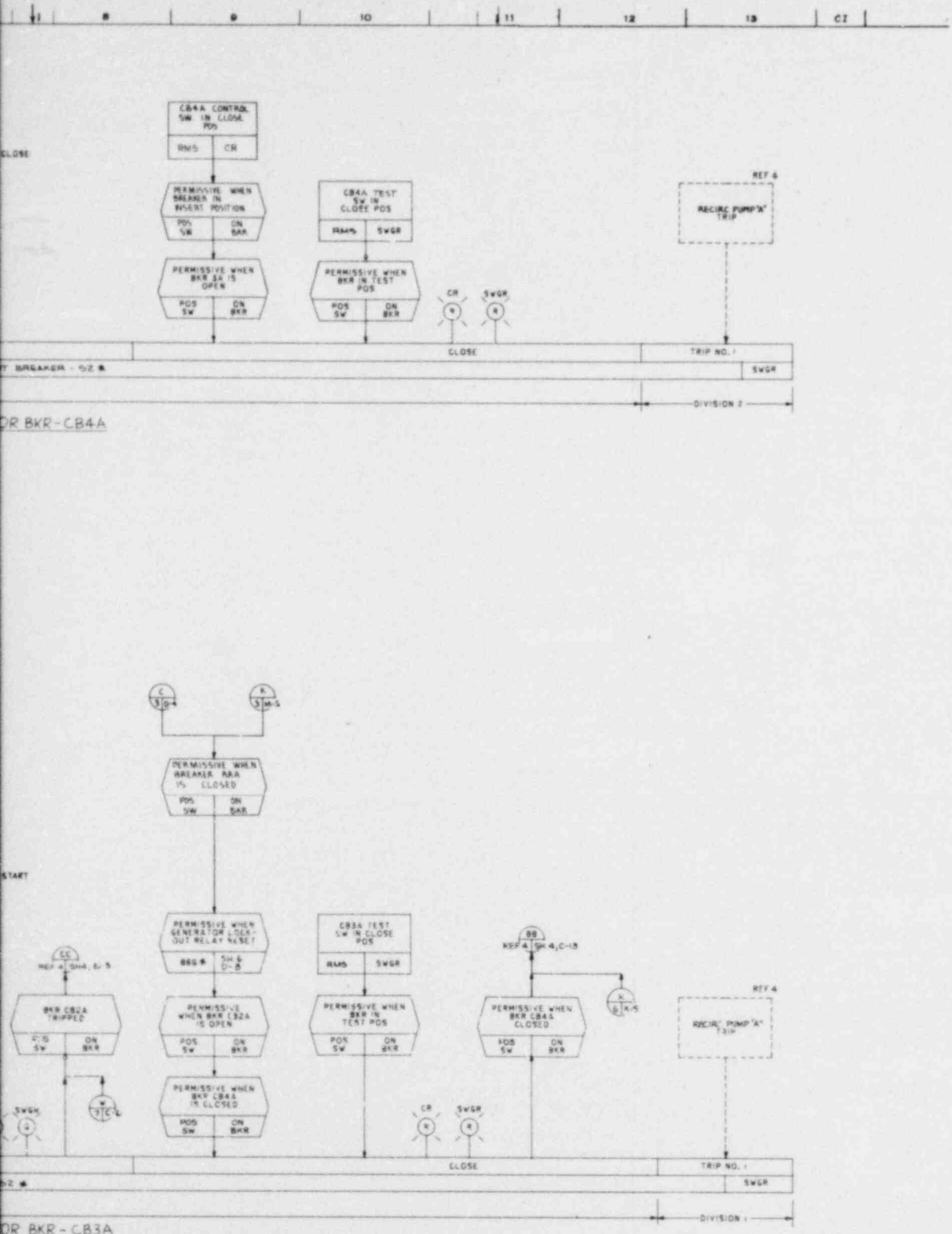


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 6	FIGURE H.A- 1f

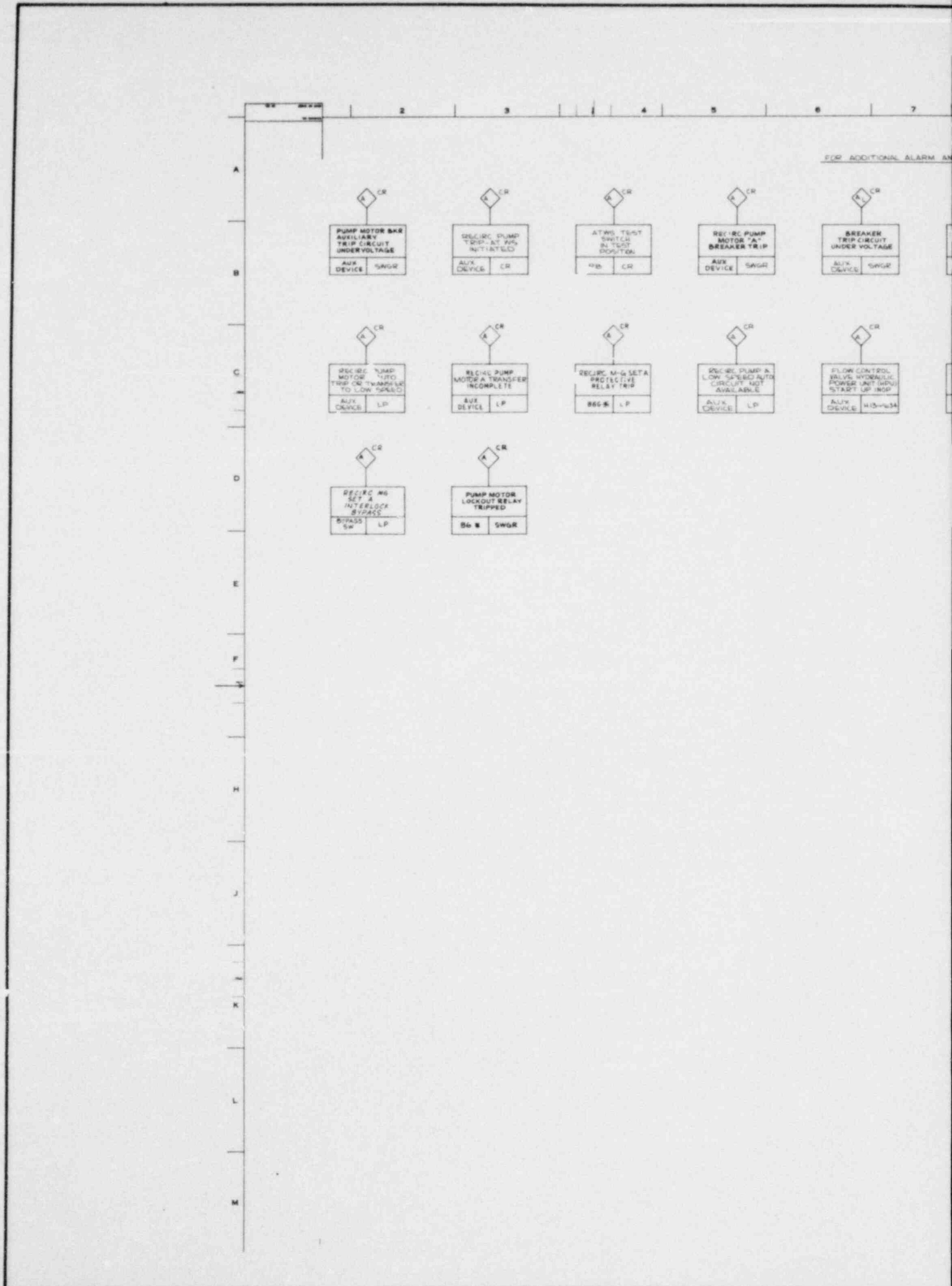


OPEN
PULL-TO-LOCK
CB4A CONT SW
CB4A CONTROL SW IN OPEN POSITION
RMS
CR
CB4A TEST SW IN TRIP POSITION
RMS
SWGR
PERMISSIVE WHEN BREAKER IN INSERT POSITION
POS SW
ON BKR
PERMISSIVE WHEN BKR IN TEST POS
POS SW
ON BKR
CR
SWGR
TRIP NO. 2
NON - DIVISIONAL
PUMP COOIA DRIVE MO
ATWS TRIP
REACTOR VESSEL LOW LEVEL (2)
LIS B22 NO36A
LP
(NO36B) FOR MTR B
REACTOR HIGH PRESSURE
PS B22 NO45A
LP
(NO45B) FOR MTR B
REACTOR VESSEL LOW LEVEL 2
LIS B22 NO36C
LP
(NO36D) FOR MTR B
REACTOR HIGH PRESSURE
PSB22 NO45C
LP
(NO45D) FOR MTR B
TRIP BKR CB2A
PERMISSIVE WHEN TEST SW NOT IN TEST
PB
CR
PERMISSIVE WHEN TEST SW NOT IN TEST
PB
CR
BKR CB4A IN OPEN POS
POS SW
ON BKR
TRANSFER M-6
PULL-TO-LOCK-STOP
CB3A CONT SW
SUCTION VLV F023A <90% OPEN
LIM SW
ON VLV
DISCH VLV F067A <90% OPEN
LIM SW
ON VLV
CB3A CONTROL SW IN PULL-TO-LOCK-STOP POS
RMS
CR
BKR PAA IN OPEN POSITION
POS SW
ON BKR
CB3A TEST SW IN TRIP POS
RMS
SWGR
PERMISSIVE WHEN BREAKER IN INSERT POSITION
POS SW
ON BKR
PERMISSIVE WHEN BKR IN TEST POSITION
POS SW
ON BKR
TRIP NO. 2
AIR CIRCUIT BREAKER
NON DIVISIONAL
PUMP COOIA DRIVE MO

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 7	FIGURE H.A-1g



2
3
4
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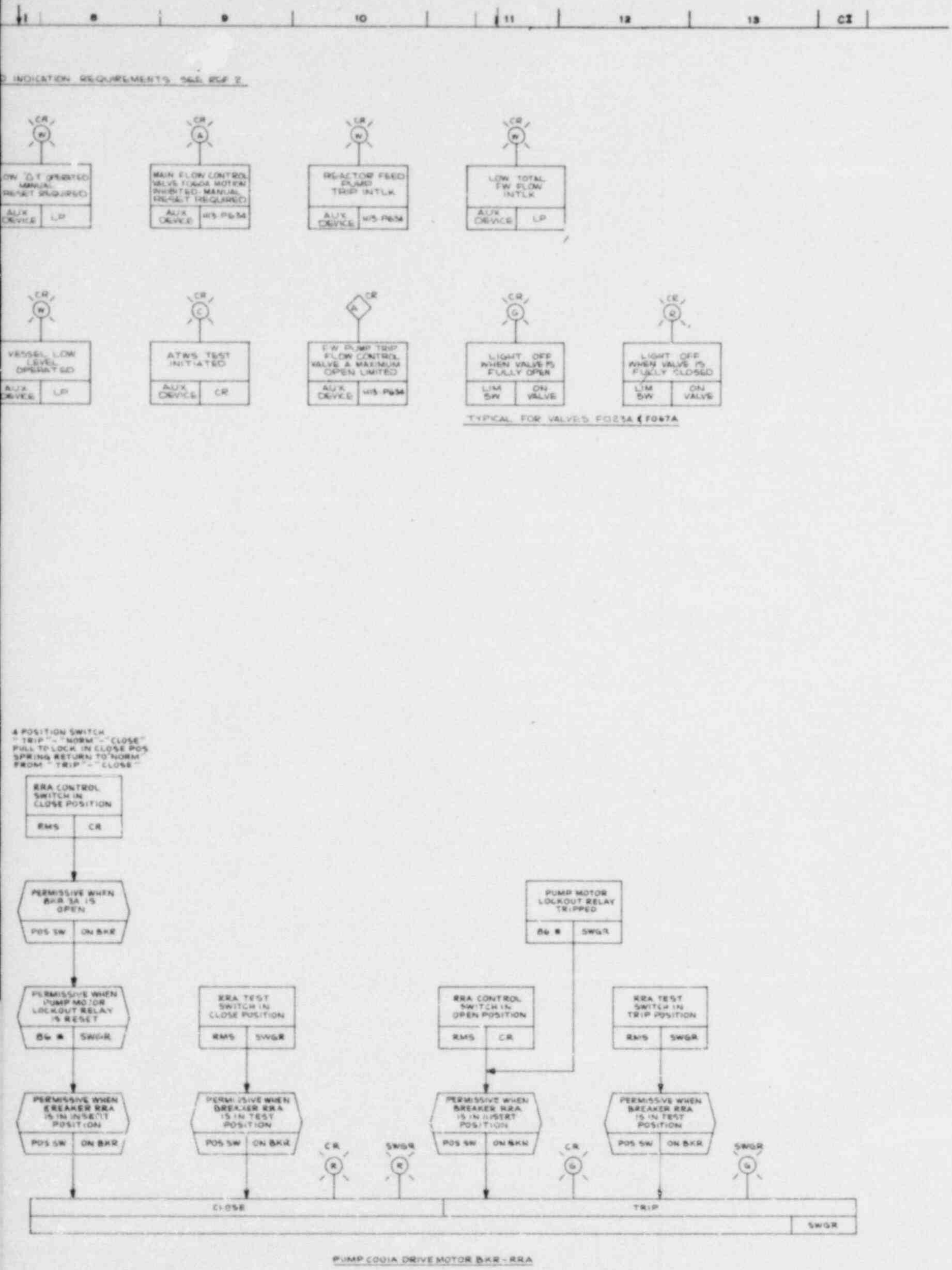
6
7
A
B
C
D
E
F
H
J
K
L
M
FOR ADDITIONAL ALARM AN
CR
A
PUMP MOTOR BKR AUXILIARY TRIP CIRCUIT UNDERVOLTAGE
AUX DEVICE
SWGR
RECIRC PUMP TRIP-AT WS INITIATED
AUX DEVICE
CR
ATWS TEST SWITCH IN TEST POSITION
PB
CR
RECIRC PUMP MOTOR "A" BREAKER TRIP
AUX DEVICE
SWGR
BREAKER TRIP CIRCUIT UNDER VOLTAGE
AUX DEVICE
SWGR
RECIRC PUMP MOTOR AUTO TRIP OR TRANSFER TO LOW SPEED
AUX DEVICE
LP
RECIRC PUMP MOTOR A TRANSFER INCOMPLETE
AUX DEVICE
LP
RECIRC M-G SET A PROTECTIVE RELAY TRIP
86G-B
LP
RECIRC PUMP A LOW SPEED AUTO CIRCUIT NOT AVAILABLE
AUX DEVICE
LP
FLOW CONTROL VALVE HYDRAULIC POWER UNIT (HPU) START UP INOP
AUX DEVICE
H13-P634
RECIRC MG SET A INTERLOCK BYPASS
BYPASS SW
LP
PUMP MOTOR LOCKOUT RELAY TRIPPED
86 B
SWGR

AMENDMENT NO. 10
July 1980



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	REACTOR RECIRCULATION SYSTEM FCD SHEET 8	FIGURE H.A-1h

QUESTIONS REFERENCING CHAPTER 1

Section 1.2

031.001(a)
031.080(a)

Section 1.7

031.002
031.089

QUESTIONS REFERENCING CHAPTER 2

Section 2.2

312.001

Section 2.3

372.001
372.003
372.004
372.005
372.006
372.007
372.008
372.009
372.010
372.011
372.012
372.013
372.014
372.015
372.017

Section 2.4

371.001
371.002
371.003
371.004
371.005
371.008
371.009
371.010
371.011
371.012
371.013
371.014

Section 2.5

360.001
360.002
360.003
360.004
360.005
362.001
362.002
362.003
362.004
362.005
362.006
362.007
362.008
362.009

QUESTIONS REFERENCING CHAPTER 3

Section 3.0

432.001

Section 3.1

031.001(b)
031.003
031.046

Section 3.2

010.003
022.040
432.002
432.003
432.004

Section 3.3

010.024
010.025
130.001
130.010
130.011

Section 3.4

010.010
031.099
130.006
371.005

Section 3.5

010.001
010.011
010.013
010.025
010.027
010.029
022.015
130.002
130.012
211.013
211.014
211.015

Section 3.5 (Continued)

211.015
211.023
312.001
312.002
312.003
312.004

Section 3.6

010.002
010.012
010.014
010.028
110.001
110.002
110.003
110.004
110.005
110.006
110.007
110.008
110.009
110.010
110.011
110.012
110.013
110.014
110.015
110.016
110.017
110.018
110.033

Section 3.7

110.035
130.003
130.004
130.005
130.006
130.008
130.013
130.014
130.015

QUESTIONS REFERENCING CHAPTER 3 (Continued)

Section 3.7 (Continued)

130.016
130.017
130.018
130.019
130.020
130.021
130.022
130.023
130.024
130.025
130.026
130.027
130.028
130.029
130.030
130.031
371.005

Section 3.8

022.009
022.020
031.001(p)
031.069
110.010
130.001
130.009
130.011
130.032
130.033
130.034
130.035
130.036
130.037
130.038
130.039
130.040
130.041
130.042
130.043
130.044

Section 3.9

110.019
110.020
110.021

Section 3.9 (Continued)

110.022
110.023
110.024
110.025
110.028
110.029
110.030
110.031
110.032
110.033
110.034
110.035
110.036
110.037

Section 3.10

031.004
031.005
031.006
031.055
031.081
110.035
110.036
110.037

Section 3.11

010.033
031.001(u)
031.005
031.006
031.028
031.055
031.056
031.057
031.080(a)
031.081
031.083
031.084
031.087
031.108

Section 3.12

130.007

QUESTIONS REFERENCING CHAPTER 4

Section 4.2

231.001
231.002
231.003

Section 4.3

232.002

Section 4.4

031.008
031.080(c,d)
031.108
221.001
221.002
221.003
221.004
221.005
221.006
221.007
221.008
221.009
221.010
221.011
221.012
221.013

Section 4.6

211.012
211.017
211.018
211.019
211.023

QUESTIONS REFERENCING CHAPTER 5

Section 5.2

031.009
031.024
031.046
121.001
121.002
211.002
211.003
211.004
211.005
211.006
211.008
211.009
211.010
211.048
423.025
423.026
432.005

Section 5.3

121.003
121.004

Section 5.4

031.001(f)
031.001(g)
031.008
031.010
031.108
211.012
211.020
211.021
211.022
211.024
211.026
211.027
211.028
211.029
211.030
211.031
211.032
211.033
211.034
211.035
211.036

Section 5.4 (Continued)

211.037
211.038
211.039
211.040
211.041
211.042
211.043
211.044
211.045
211.046
211.048

Section 5.5

211.011

QUESTIONS REFERENCING CHAPTER 6

Section 6.2

022.001
022.002
022.003
022.004
022.005
022.006
022.007
022.008
022.010
022.011
022.012
022.013
022.014
022.016
022.017
022.018
022.019
022.020
022.021
022.022
022.023
022.024
022.025
022.026
022.027
022.033
022.035
022.037
022.041
022.042
022.043
022.044
022.045
022.046
022.047
022.048
022.049
031.001(i)
031.001(j)
031.001(k)
031.001(l)
031.001(m)
031.001(p)
031.001(q)
031.011
031.012

Section 6.2 (Continued)

031.069
031.070
031.080(a,e,f,g)
031.088
222.001
222.002
222.003
222.004
312.017

Section 6.3

022.038
031.001(n)
031.001(o)
031.013
031.014
031.017
031.018
211.048
212.003
212.004
423.027

Section 6.4

312.006
312.007
312.010
312.011

Section 6.5

022.015
031.001(q)
031.011
031.070
321.003

Section 6.6

022.037

QUESTIONS REFERENCING CHAPTER 6 (Continued)

Section 6.7

010.015
031.017
031.076
031.080(a)
031.089

QUESTIONS REFERENCING CHAPTER 7

Section 7

031.001(s)
031.026

Section 7.1

022.022
031.001(t)
031.020
031.037
031.041
031.052
031.072

Section 7.2

031.001(d)
031.001(u)
031.005
031.016
031.018
031.021
031.022
031.023
031.024
031.025
031.032
031.033
031.072
031.080(b,i,j,o)
031.090
031.091
031.092
031.093
031.094
040.043

Section 7.3

031.001(c)
031.001(g)
031.001(h)
031.001(q)
031.001(r)
031.001(v)
031.001(w)

Section 7.3 (Continued)

031.011
031.013
031.016
031.018
031.019
031.021
031.027
031.028
031.029
031.030
031.031
031.033
031.034
031.035
031.036
031.037
031.038
031.039
031.040
031.041
031.043
031.044
031.069
031.070
031.071
031.076
031.078
031.079
031.080(m,o,p)
031.094
031.095
031.096
031.099
031.100
031.101
031.102
031.103
040.043
040.070

Section 7.4

031.001(y)
031.001(z)
031.001(aa)

QUESTIONS REFERENCING CHAPTER 7 (Continued)

Section 7.4 (Continued)

031.015
031.040
031.041
031.043
031.044
031.045
031.046
031.048
031.086
031.105
040.043

Section 7.5

031.025
031.047
040.009

Section 7.6

022.008
022.025
031.001(bb)
031.001(cc)
031.001(dd)
031.001(ff)
031.001(ii)
031.022
031.050
031.051
031.052
031.072
031.080(n,q,r,s,t,u)
031.082
031.085
031.088
031.098
031.103
031.106
031.107
031.112
040.043
211.002
211.004
211.006
211.007

Section 7.6 (Continued)

211.008
211.009

Section 7.7

031.001(dd)
031.001(ee)
031.001(ff)
031.001(hh)
031.010
031.053
031.054
031.074
031.075
031.085
031.088
031.109
031.110
321.001

QUESTIONS REFERENCING CHAPTER 8

Section 8.2

031.080(m)
040.003

Section 8.3

031.028
031.032
031.054
031.080(m)
040.001
040.002
040.004
040.005
040.006
040.007
040.008
040.010
040.021
040.023
040.036
040.037
040.038
040.055

QUESTIONS REFERENCING CHAPTER 9

Section 9.0

010.016

Section 9.1

010.004
010.017
010.018
010.019
010.020
010.021

Section 9.2

010.022
010.023
010.024
010.025
010.026
010.027
321.005
371.006

Section 9.3

010.028
031.007
031.073
031.088
211.001
321.005

Section 9.4

010.005
010.006
010.030
010.031
010.032
010.033
022.021
040.018
312.009

Section 9.5

010.007
031.111
040.011
040.012
040.013
040.014
040.015
040.016
040.017
040.018
040.019
040.020
040.022
040.023
040.024
040.025
040.045
040.046
040.047
040.048
040.049
040.050
040.051
040.052
040.053
040.054
040.056
040.057
040.058
040.059

QUESTIONS REFERENCING CHAPTER 10

Section 10.1

040.027

Section 10.2

040.028
040.029
040.030
040.031
040.032
040.060
040.061
040.062
040.063
312.002

Section 10.3

010.008
040.032

Section 10.4

010.009
010.034
040.030
040.033
040.064
040.065
040.066
040.067
040.068
040.069
040.070
040.071
040.072
040.073
040.074

QUESTIONS REFERENCING CHAPTER 11

Section 11.3

321.001
321.004

Section 11.6

031.059

QUESTIONS REFERENCING CHAPTER 12

Section 12.1

331.001
331.008
331.009
331.016

Section 12.2

331.003
331.017

Section 12.3

211.006
331.004
331.005
331.006
331.007
331.010
331.011
331.018
331.019

Section 12.4

331.012
331.020
331.021

Section 12.5

331.013
331.014
331.022
331.023
331.024

QUESTIONS REFERENCING CHAPTER 13

Section 13.1

422.003
422.004
422.005
422.006

Section 13.2

441.001
441.002
441.003
441.004
441.005

Section 13.4

422.007
422.008
422.009

QUESTIONS REFERENCING CHAPTER 14

Section 14.2

040.038
211.038
423.001
423.002
423.003
423.004
423.005
423.006
423.007
423.008
423.009
423.010
423.012
423.013
423.014
423.015
423.016
423.017
423.018
423.019
423.020
423.021
423.022
423.023
423.025
423.026
423.027
423.028

QUESTIONS REFERENCING CHAPTER 15

Section 15.0

211.020

Section 15.2

031.046
211.025
211.047

Section 15.4

031.098
031.113
232.001
232.003
232.004

Section 15.6

312.012
312.017
312.018
312.019

Section 15.7

031.049
312.013
312.014
321.002
321.005

QUESTIONS REFERENCING DAR

022.028	022.032
022.029	110.026
022.030	110.027

EMERGENCY PLAN

423.006
432.001 thru 432.015

SECURITY PLAN

022.034

FIRE PROTECTION

422.001
422.002

QUESTIONS REFERENCING QUESTIONS

Question	Question Referenced
031.058	031.001
031.059	031.001
031.060	031.001
031.061	031.039
031.062	031.001
	031.002
	031.023
031.063	031.006
	031.014
	031.016
	031.026
	031.047
031.064	031.009
031.065	031.010
031.066	031.009
	031.018
031.067	031.025
031.068	031.032
031.077	031.021
	031.037
031.078	031.050
031.083	031.006
	031.056
	031.059
031.090	031.032
031.094	031.033
031.097	031.026
031.099	031.030
312.016	022.048
423.011	423.002
	423.006
	423.007
423.024	423.023

QUESTIONS WITH NO REFERENCES

005.001
022.031
022.036
022.039
022.050
022.051
022.052
031.001(e)
031.001(x)
031.001(gg)
031.042
040.026
040.034
040.035
040.039
040.040
040.041
040.042
040.044
210.001
212.002
232.005
312.005
312.015
331.002
331.015
372.002
372.016
423.029
423.016

LIST OF NRC QUESTIONS

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
005.1	2	7
010.1	1	1
010.2	1	1
010.3	1	1
010.4	1	1
010.5	1	1
010.6	1	1
010.7	1	1
010.8	1	1
010.9	1	1
010.10	1	5
010.11	9	9
010.12	2	5
010.13	2	5
010.14	3	5
010.15	1	5
010.16	3	5
010.17	1	5
010.18	1	5
010.19	1	5
010.20	1	5
010.21	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
010.22	1	5
010.23	2	5
010.24	1	5
010.25	1	5
010.26	1	5
010.27	1	5
010.28	2	5
010.29	2	5
010.30	1	5
010.31	1	5
010.32	1	5
010.33	1	5
010.34	1	5
022.1	1	1
022.2	1	1
022.3	1	1
022.4	1	1
022.5	3	8
022.6	1	1
022.7	1	5
022.8	1	1
022.9	1	1
022.10	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.11	1	1
022.12	1	3
022.13	1	3
022.14	1	3
022.15	1	3
022.16	1	3
022.17	1	3
022.18	1	3
022.19	1	3
022.20	1	3
022.21	1	3
022.22	1	3
022.23	1	3
022.24	1	3
022.25	1	3
022.26	1	3
022.27	1	3
022.28	1	3
022.29	1	3
022.30	1	3
022.031	2	5
022.032	2	5
022.033	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.034	1	5
022.035	2	5
022.036	1	5
022.037	2	5
022.038	1	5
022.039	2	7
022.040	1	5
022.041	1	5
022.042	1	5
022.043	2	5
022.044	1	5
022.045	1	5
022.046	1	5
022.047	1	5
022.048	11	5
022.049	4	5
022.050	2	5
022.051	1	5
022.052	1	5
031.001(a)	1	0
031.001(b)	1	0
031.001(c)	1	0
031.001(d)	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.001(e)	1	0
031.001(f)	1	0
031.001(g)	1	0
031.001(h)	1	0
031.001(i)	2	0
031.001(j)	1	10
031.001(k)	1	10
031.001(l)	1	0
031.001(m)	1	0
031.001(n)	1	0
031.001(o)	1	0
031.001(p)	1	0
031.001(q)	1	0
031.001(r)	2	0
031.001(s)	1	0
031.001(t)	1	0
031.001(u)	1	0
031.001(v)	1	0
031.001(w)	1	0
031.001(x)	1	0
031.001(y)	1	0
031.001(z)	1	0
031.001(aa)	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.001(bb,cc)	1	0
031.001(dd)	1	0
031.001(ee)	1	0
031.001(ff)	1	0
031.001(gg)	1	0
031.001(hh)	1	0
031.001(ii)	1	0
031.002	1	0
031.003	1	0
031.004	1	0
031.005	1	0
031.006	2	10
031.007	1	0
031.008	1	0
031.009	4	0
031.010	4	0
031.011	1	0
031.012	1	0
031.013	1	0
031.014	1	3
031.015	3	0
031.016	2	0
031.017	2	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.018	2	0
031.019	1	0
031.020	1	0
031.021	2	0
031.022	1	0
031.023	1	0
031.024	1	0
031.025	2	0
031.026	1	10
031.027	1	0
031.028	1	0
031.029	1	0
031.030	2	0
031.031	1	0
031.032	1	0
031.033	1	0
031.034	2	0
031.035	1	0
031.036	1	0
031.037	1	0
031.038	1	0
031.039	2	0
031.040	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.041	1	0
031.042	1	0
031.043	1	0
031.044	2	0
031.045	1	0
031.046	1	0
031.047	1	3
031.048	3	0
031.049	1	0
031.050	3	0
031.051	1	0
031.052	1	0
031.053	2	0
031.054	1	0
031.055	1	10
031.056	1	10
031.057	1	10
031.058	3	3
031.059	2	10
031.060	1	3
031.061	1	3
031.062	1	3
031.063	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.064	1	3
031.065	1	3
031.066	2	3
031.067	1	3
031.068	1	3
031.069	1	3
031.070	5	3
031.071	1	3
031.072	1	3
031.073	1	3
031.074	1	3
031.075	1	3
031.076	3	5
031.077	1	3
031.078	2	3
031.079	2	3
031.080	7	10
031.081	2	10
031.082	1	10
031.083	1	10
031.084	1	10
031.085	1	10
031.086	1	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.087	5	10
031.088	1	10
031.089	1	10
031.090	1	10
031.091	1	10
031.092	2	10
031.093	1	10
031.094	1	10
031.095	1	10
031.096	1	10
031.097	1	10
031.098	1	10
031.099	1	10
031.100	25	10
031.101	1	10
031.102	1	10
031.103	3	10
031.104	2	10
031.105	2	10
031.106	1	10
031.107	1	10
031.108	2	10
031.109	1	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.110	1	10
031.111	1	10
031.112	1	10
031.113	1	10
040.1	2	0
040.2	1	0
040.3	1	0
040.4	1	1
040.5	1	0
040.6	1	0
040.7	1	0
040.8	1	0
040.9	1	0
040.10	2	0
040.11	1	0
040.12	1	0
040.13	1	0
040.14	1	0
040.15	1	5
040.16	1	0
040.17	1	0
040.18	1	0
040.19	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.20	7	0
040.21	1	0
040.22	1	0
040.23	1	0
040.24	2	0
040.25	1	0
040.26	2	5
040.27	1	0
040.28	1	0
040.29	1	0
040.30	1	0
040.31	1	0
040.32	1	0
040.33	1	0
040.34	2	7
040.35	1	7
040.36	2	7
040.37	1	7
040.38	1	7
040.39	4	7
040.40	1	7
040.41	1	7
040.42	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.43	1	7
040.44	2	7
040.45	5	7
040.46	1	7
040.47	1	7
040.48	1	7
040.49	1	7
040.50	1	7
040.51	1	7
040.52	1	7
040.53	2	7
040.54	1	7
040.55	1	7
040.56	1	7
040.57	1	7
040.58	1	7
040.59	1	7
040.60	2	7
040.61	1	7
040.62	1	7
040.63	1	7
040.64	1	7
040.65	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.66	1	7
040.67	1	7
040.68	1	7
040.69	1	7
040.70	1	7
040.71	1	7
040.72	1	7
040.73	1	7
040.74	1	7
110.001	1	9
110.002	1	9
110.003	1	9
110.004	1	9
110.005	1	9
110.006	1	9
110.007	1	9
110.008	1	9
110.009	1	9
110.010	1	9
110.011	1	9
110.012	1	9
110.013	1	9
110.014	1	9

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
110.015	1	9
110.016	1	9
110.017	1	9
110.018	1	9
110.019	4	9
110.020	1	9
110.021	1	9
110.022	1	9
110.023	1	9
110.024	1	9
110.025	1	9
110.026	1	9
110.027	1	9
110.028	2	9
110.029	1	9
110.030	2	9
110.031	1	9
110.032	1	9
110.033	1	9
110.034	1	9
110.035	1	9
110.036	2	9
110.037	2	9

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
121.1	3	5
121.2	8	5
121.3	1	5
121.4	1	5
121.5	1	5
121.6	1	5
121.7	1	5
121.8	10	10
121.9	1	5
121.10	2	7
130.1	1	1
130.2	1	1
130.3	1	1
130.4	1	1
130.5	1	1
130.6	1	1
130.7	1	1
130.8	1	1
130.9	1	1
130.010	1	8
130.011	1	8
130.012	1	8
130.013	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.014	1	8
130.015	1	8
130.016	1	8
130.017	1	8
130.018	1	8
130.019	1	8
130.020	1	8
130.021	1	8
130.022	1	8
130.023	1	8
130.024	1	8
130.025	1	8
130.026	1	8
130.027	1	8
130.028	1	8
130.029	1	8
130.030	1	8
130.031	1	8
130.032	1	8
130.033	1	8
130.034	1	8
130.035	4	8
130.036	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.037	1	8
130.038	2	8
130.039	7	8
130.040	1	8
130.041	1	8
130.042	8	8
130.043	1	8
130.044	1	8
210.1	1	0
211.002	1	8
211.003	1	8
211.004	1	8
211.005	1	8
211.006	1	8
211.007	1	8
211.008	2	8
211.009	1	8
211.010	2	8
211.011	1	8
211.012	2	8
211.013	1	8
211.014	2	8
211.015	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.016	3	8
211.017	2	8
211.018	1	8
211.019	3	8
211.020	1	8
211.021	2	8
211.022	1	8
211.023	1	8
211.024	1	8
211.025	1	8
211.026	1	8
211.027	4	8
211.028	2	8
211.029	1	8
211.030	1	8
211.031	4	10
211.032	1	8
211.033	4	8
211.034	1	8
211.035	1	8
211.036	1	8
211.037	1	8
211.038	3	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.039	2	8
211.040	2	8
211.041	1	8
211.042	1	8
211.043	1	8
211.044	1	8
211.045	1	8
211.046	1	8
211.047	1	8
211.048	2	8
212.1	1	3
212.2	1	3
212.3	2	5
212.4	1	3
221.01	2	7
221.02	1	7
221.03	1	7
221.04	1	7
221.05	1	7
221.06	1	7
221.07	1	7
221.08	1	7
221.09	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
221.10	2	7
221.11	1	7
221.12	3	7
221.13	1	7
222.001	1	8
222.002	3	8
222.003	1	8
222.004	1	8
231.001	1	3
231.002	3	3
231.003	1	5
232.001	1	3
232.002	1	5
232.003	1	5
232.004	1	5
232.005	1	5
312.1	1	1
312.2	1	1
312.3	1	1
312.4	1	1
312.5	1	1
312.6	1	1
312.7	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
312.09	1	8
312.10	1	1
312.11	1	1
312.12	2	1
312.13	1	1
312.14	1	1
312.15	1	8
312.16	1	5
312.17	1	5
312.18	1	5
312.19	1	5
321.1	1	1
321.2	1	1
321.3	1	5
321.4	1	5
321.5	4	5
331.1	1	1
331.2	1	1
331.3	2	1
331.4	1	1
331.5	1	1
331.6	1	1
331.7	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
331.8	1	1
331.9	1	1
331.10	1	1
331.11	1	1
331.12	1	1
331.13	1	1
331.14	1	1
331.15	2	5
331.16	1	5
331.17	1	5
331.18	[illegible]	5
331.19	1	5
331.20	1	5
331.21	1	5
331.22	1	5
331.23	1	5
331.24	1	5
360.1	1	1
360.2	1	1
360.3	1	1
360.4	3	10
360.5	6	10
362.1	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
362.2	2	3
362.3	1	3
362.4	2	3
362.5	1	5
362.6	1	5
362.7	1	5
362.8	1	5
362.9	2	5
371.1	1	1
371.2	1	1
371.3	1	1
371.4	1	1
371.5	1	1
371.6	1	5
371.8	1	5
371.9	1	5
371.10	1	5
371.11	1	5
371.12	1	5
371.13	1	5
371.14	1	5
372.1	1	1
372.2	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
372.3	1	1
372.4	1	1
372.5	1	1
372.6	1	1
372.7	1	3
372.8	14	5
372.9	1	5
372.10	1	5
372.11	1	5
372.12	1	5
372.13	1	5
372.14	7	5
372.15	4	5
372.16	5	5
372.17	1	5
422.01	2	7
422.02	1	7
422.03	1	7
422.04	4	7
422.05	1	7
422.06	3	7
422.07	2	7
422.08	2	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
422.09	1	7
423.1	1	1
423.2	1	1
423.3	1	1
423.4	1	0
423.5	1	1
423.6	2	1
423.7	1	1
423.8	1	1
423.9	1	1
423.10	1	1
423.11	4	7
423.12	1	7
423.13	1	7
423.14	1	7
423.15	1	7
423.16	2	7
423.17	1	7
423.18	1	7
423.19	7	7
423.20	4	7
423.21	3	7
423.22	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
423.23	11	7
423.24	1	7
423.25	1	7
423.26	1	7
423.27	1	7
423.28	1	7
423.29	1	7
432.01	7	5
432.02	2	5
432.03	2	5
432.04	3	5
432.05	1	5
432.06	1	5
432.07	1	5
432.08	1	5
432.09	1	5
432.10	1	5
432.11	2	5
432.12	1	5
432.13	2	5
432.14	1	5
432.15	1	5
432.16	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
441.01	3	7
441.02	1	7
441.03	1	7
441.04	1	7
441.05	1	7

Q. 031.001(j)

Provide justification for not seismically qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Seismic qualification documentation for the feedwater isolation valve actuators is presently under examination as part of an overall qualification review with results to be provided to your SQRT personnel. The control rod drive excess flow isolation valve has been deleted along with CRD Return Line.

Q. 031.001(k)

Provide justification for not environmentally qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Environmental qualification documentation for the feedwater isolation valve actuators is presently under examination as part of an overall qualification review to determine degree of compliance with NUREG-0588, the results of which will be provided to NRC personnel. The control rod drive excess flow isolation valve has been deleted along with the CRD Return Line.

Q. 031.005 (RSP)
(3.10)
(3.11.3)
(7.2.2.2)

A request for documentation of the seismic and environmental qualification of Class 1E equipment is contained in 3.10 and 3.11 of the Standard Format. This request is applicable to all engineered safety features, reactor protection systems and all supporting systems. It is not limited to those particular safety-related supporting systems supplied by General Electric. Accordingly, we require you to provide the information requested in 3.10 and 3.11 of the Standard Format for all Class 1E systems in accordance with: (a) the NRC Staff positions stated in Attachments 1 and 2; (b) IEEE Std. 323-1971; and (c) IEEE Std. 344-1971. Identify and justify any exceptions.

RESPONSE:

The requested seismic and environmental qualification data for Class IE equipment for all engineered safety features, reactor protection systems and all support systems is provided in the revised text and tables of 3.10 and 3.11.

For additional discussion refer to 7.2.1.2.7 (RPS), 7.3.1.1.1 (ECCS), 7.3.1.1.2.12 (PC and RVIS), and 7.3.1.1.4.11 (RHR/Containment Spray).

Additional discussion of conformance to IEEE 323-1971 is provided in 7.2.2.1.2.3.4 (RPS), 7.1.2.5.4 and 7.3.2.1.2.3.1.4 (ECCS), and 7.3.2.2.2.3.1.4 and 7.1.2.5 (PC and RVIS).

Additional discussion of conformance to IEEE 344-1971 is provided in 7.2.2.1.2.3.7 (RPS), 7.3.2.1.2.3.5 (ECCS), and 7.3.2.2.2.3.5 (PC and RVIS).

Q. 031.006

The staff requests that the following information regarding the qualification test program be provided for Class 1E equipment: (a) the equipment design specification requirements; (b) the test plan; (c) the test set up; (d) the test procedures; (e) the acceptability goals and requirements; and (f) the test results.

Provide this information for each of the following Class 1E components: (a) the 4.16 kV switchgear SM 7; (b) the damper operator for WMA-V-52C; (c) the fan WMA-FN-52B; (d) the logic equipment for the standby gas treatment system; (e) the diesel-generator control equipment; (f) the 480 V ESS switchgear MC-7A-A; and (g) the solenoid valve for the main steam line isolation valves.

Response:

An extensive seismic and environmental review program is presently underway encompassing BOP and NSSS scope, with a planned completion date in December 1980.

Within the BOP scope, the equipment documentation has been extracted from the contract files, copied and categorized for easy retrieval. Within the NSSS scope, contract negotiations are underway with GE to perform a similar function.

A list of all Class 1E equipment including splices, terminal blocks, termination cabinets and connectors is presently being compiled. This list will contain the following information:

1. Equipment location
2. Safety functional requirement
3. Manufacturer & Model No.
4. Qualification Method (test-analysis)
5. Environmental Extremes
6. Identification and location of qualification documents

The documentation will be reviewed to insure that the testing was adequate to meet the seismic and environmental extremes under which the equipment must either function or not fail.

A composite list will be included in the FSAR as equipment tables in 3.10 (seismic) and 3.11 (environmental).

The extensive review program underway will also satisfy the requirements of IE Circular 78-08, address the degree of compliance with NUREG-0588, and establish the conservativeness of seismic tests and analysis performed to IEEE-344, 1971.

The detailed results of this review will be made available to NRC SQRT and environmental review personnel during their site documentation reviews.

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Q. 031.026
(7)

Describe the installation, operation, and removal of the "Startrec" computer system which is used for startup testing of GE boiling water reactors, including the following topics: (a) specifications and qualification testing of electrical isolators; and (b) separation criteria for permanent and temporary wiring.

Response:

The Transient Data Acquisition System (TDAS) to support start-up transient testing will no longer be the GE STARTREC computer system. A re-evaluation of the WNP-2 data acquisition needs has led to the implementation of a permanent installation.

The WNP-2 TDAS control unit and analysis computer will be located in the main control room. Analog and digital signals will be isolated and conditioned in divisionalized remote units in the cabinet where they originate. Multiplexing and digitizing of all data will take place in these remote units before the data is transmitted over fiber-optic links to the control unit.

All signals originating from safety-related divisionalized equipment will be physically and electrically isolated such that faults occuring in the TDAS equipment cannot propagate back into safety-related circuits. The isolation devices will be qualified to the standards of Class 1E equipment and meet the intent of Regulatory Guide 1.75 concerning isolation devices.

Q. 031.055
(3.10.1.1)
(031.001)
(031.044)

Identify all Class 1E equipment which was not qualified by test. For each such item, provide the basis for assuming that it will not be spuriously operated, or fail to operate when required, during and after a seismic event.

Response:

A program is currently underway to re-evaluate the effects of vibratory loads on Class 1E equipment. This re-evaluation will consider the effects (including spurious operation) of seismic as well as hydrodynamic loads produced for Class 1E equipment.

We anticipate completing the re-evaluation phase of this program in calendar year 1980.

Q. 031.056
(3.11)

Describe the environmental qualification procedures and the environmental extemes of qualification for the following specific passive Class 1E components inside the drywell: (1) splices; (2) terminal blocks; (3) termination cabinets; and (4) connectors.

Response:

See response to Question 031.006.

Q. 031.057
(3.11)

Identify all Class 1E equipment inside the drywell, except the equipment cited in Item 031.056, and summarize the environmental qualifications for this equipment. The identification and summary for each item should include: (1) the safety funct'on and functional requirement; (2) the manufacturer, model number, type number, and any other identifying numbers; (3) the specific location of this equipment in the drywell; (4) the method of environmental qualification; (5) the environmental extremes, including the time period of testing, for which it is qualified; and (6) the identification and the location of the documents which are available so as to permit an independent evaluation of the adequacy of the environmental qualification.

Response:

See response to Question 031.006.

Q 31.058
(031.001)

Your response to Item 031.001(e) is incomplete. Describe the passive design features which prevent motion of the recirculation flow control valve after an accident. This motion could result from postulated damage to the hydraulic system during a loss-of-coolant accident.

Response:

As stated in the response to 031.001(e), the valve is considered inactive since, if it moves at all, it will move in the open direction in a LOCA.

The passive design features associated with the recirculation flow control valve to resist unaccountable motion in an accident are as follows:

1. Minimum exposure of components to the accident environment - The only components in the recirculation flow control valve system subject to the direct effects of a LOCA are the valve, valve actuator, and hydraulic lines. The balance of the system I and C is located outside primary containment.

2. Physical remoteness from potential pipe breaks -

   a) Referring to the response to 31.001(e), it is recalled the valve of concern is the one in the unbroken loop. Accordingly, for recirculation pipe breaks, there is virtually no potential for missile or jet impingement impact on the valve in the unbroken loop since it is on the opposite side of the reactor vessel within primary containment from the other loop.

   b) For other than recirculation pipe breaks, over 25 vertical feet of separation for the actuator and 15 vertical feet of separation for the hydraulic lines exist from the nearest high energy line.

Q 31.059
(031.001)
(11.6-1)

Your response to Item 031.001(h) presents a new design for the logic of the main steam line isolation valves which is different from that reviewed and accepted for licensing on similar boiling water reactors. Provide the manufacturing drawings for ASCO Valve No. 832320. Additionally, provide the results of the engineering analysis and the test results which demonstrate the ASCO Valve No. 832320: (1) is qualified for the environment in the drywell following a loss-of-coolant accident; (2) is seismically qualified; (3) meets the physical separation and the required electrical independence in accordance with the staff positions contained Regulatory Guide 1.75; (4) satisfies the single failure criterion (previous designs accepted for licensing have used two separate valves in a one-out-of-two logic for a reactor trip). Note that Table 1.6-1 of the FSAR states that the GE Topical Report, APED-5750, is applicable to the WNP-2 facility and that Table 7.1-1 indicates the main steam line isolation valves are designed and supplied by GE. Accordingly, provide justification for the change to the design which was previously reviewed and approved by the staff in our evaluation of the GE Topical Report, APED-5750.

Response:

The main steam line isolation valve logic is the same for WNP-2 as that supplied for previously reviewed and accepted for licensing BWR's.

Attachment - ASCO Dwg. #HVA-166-265

(1) ASCO Valve #832320 is a fail safe valve and, closes by current deenergizing. This valves application does not require it to function following a LOCA. The valve has been qualified for normal ambient conditions (VPF #3680-1) as follows:

Cycle tested at temperature (172-198°F)

100 cycles at 4 minute intervals
10 cycles at 12 hours intervals
5 cycles at 120 hours intervals

Total time at temperature 941 hours

(2) Qualification of valve for seismic. The solenoid valve was qualified for original seismic requirement when tested with complete valve (Wyle Laboratories -- Seismic Simulation Test Report #42610-1, dated 2/27/74).

The solenoid valve remained functional during all phases of the testing. Re-evaluation of the equipments original qualification will be performed as part of the overall seismic and environmental re-evaluation effort. See the response to Question 031.006.

(3) The protection system criteria of IEEE 279-1971 are met with this design; the requirements of Regulatory Guide 1.75 were not committed for this plant.

(4) The ASCO valves in question are not used in generating a reactor trip. The ASCO valves are used in a two-out-of-two logic for each MSIV. That is, in order for each MSIV to be isolated both ASCO solenoids must deenergize. The ASCO valves themselves are not single failure proof. Single failure criterion is preserved since each main steam line contains two valves in series. If a single failure occurs in one valve scheme the second will provide isolation.

There is no deviation from the commitments made in APED-5750.

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Q. 031.080
(1.2.1)
(3.10)
(3.11.2)
(4.4.3)
(6.2.4)
(6.7.1)
(7.2.1)
(7.3.1)
(7.6.1)
(7.6.2)
(8.2.1)
(8.3.1)
(031.001)

Your FSAR still contains many conflicting or confusing statements which must be resolved so that our review may proceed. Accordingly, provide the following information:

a. Clarify the discrepancy between the definition of passive failures in electrical, instrumentation, and control systems in Sections 1.2.1.1.1.1.2.L and 3.11.2.3, 6.2.4.1.1, and 6.7.1.2.c of the FSAR.

b. Clarify the discrepancy between your response to Item 031.001 (b) and Figures 7.2-1b and 7.2-1c of the FSAR.

c. Revise Section 4.4.3.3.3 of the FSAR to provide the actual values of the measured parameters which are to be used in the WNP-2 facility. This section indicates "typical" values; at the OL stage of review, we require actual values.

d. Clarify the discrepancy between the 25 percent pump speed interlock value described in Section 4.4.3.3.3.1 and the 20 percent value which is given in Figure 7.7-7c of the FSAR.

e. Describe the primary and secondary modes of operation of the isolation valves which are referenced in Section 6.2.4.2 of the FSAR.

f. Clarify the discrepancy between the description of the solenoid valves in Section 6.2.4.2 of the FSAR and the design which is presented in your response to Item 031.001(h).

g. Clarify the discrepancy between the isolation valve arrangement which is described in Section 6.2.4.3.2.1.2.1 and that which is shown in Figures 5.4-9a and 7.4-1a of the FSAR.

h. Provide a cross-reference between GE diagram numbers (EXX-XXXX) which are used in the FSAR diagrams and are included in the list of references on these diagrams, and the WNP-2 figures.

i. Clarify the reference to Figure 1.2-5 of the FSAR which is contained in Section 7.2.1.1.4.2.e by specifying the exact physical location and arrangement of the turbine generator oil line pressure switches and their sensing lines.

j. Indicate in Section 7.2.1.1.4.4.5 of the FSAR the delay time before the reactor mode switch scram is automatically bypassed.

k. Clarify the discrepancy in the instrumentation range between Note 4 of Table 7.1-2 and line 3 of Table 7.2-1 of the FSAR (i.e., reactor vessel lower water line).

l. Clarify the discrepancy between the response to Item 031.001 (s) and the content of Drawing 807E180TC, Sheets 1 through 9.

m. Clarify the reference in Section 7.3.1.2.7 of the FSAR to Sections 8.2.1 and 8.3.1. This clarification should clearly state the range of voltage and frequency for which all Class 1E instrumentation and control equipment is qualified and the range of voltage and frequency to which it will be exposed in the WNP-2 facility.

n. Clarify in Section 7.6.1.4.2 of the FSAR the divisional assignments which are made for the motor-generator sets of the reactor protection system. Specifically, justify the designation of these buses as "critical".

o. Clarify the discrepancy between Figure 5.2-6 and Tables 7.2-1, 7.3-2, 7.3-3, 7.3-4 and 7.3-5 of the FSAR with regard to the low level set point and range.

p. Clarify the discrepancies in the pressure trip setting between the Amendment 1 revision of Table 6.3-2 and other submittals of information in the FSAR such as Table 7.3-3 for the spray valve differential pressure.

q. Clarify the discrepancy between your statement in Section 7.6.1.8.1.2 of the FSAR regarding the uniqueness of the RPT and the statement in Item 37 of Table 7.1-2 which claims your RPT is identical to that of Zimmer.

r. Clarify the reference to four RPS divisions in Section 7.6.1.8.3.2 of the FSAR. It is our understanding that there are only two RPS divisions.

s. Clarify the discrepancy between the content of Sections 3.11 and 7.6.2.8.2.1.1.4 of the FSAR.

t. Clarify the discrepancy between Sections 3.10 and 7.6.2.8.2.1.5 of the FSAR. (Note that the RPT system is not listed in Table 3.10-1.)

u. Clarify the references in Section 7.6.1.7.8 of the FSAR to Table 3.11-4 for the reactor and control building environments.

Response:

a. There is no discrepancy intended. Systems designed in accordance with IEEE 279 include consideration of electrical passive failures as single active failures. Section 3.11.2.3 does not address single failure, only qualification of safety-related equipment.

b. There is no discrepancy. As stated in Question 031.001 (b) the reference to three trip logics has been revised. As you have noted by Figure 7.2-1 there are in fact two trip logics in each of the two RPS divisions, making a total of four logics.

c. Section 4.4.3.3.3 is revised as follows:

4.4.3.3.3.a - "...if the difference between steamline temperature and recirculation pump inlet temperature is less than a preset value (10.7°F)."

4.4.3.3.3.b - "...feedwater flow falls below a preset level (28 percent of rated) and the flow control valves are below a preset position (19 percent open)."

d. Due to a complete Chapter 7 rewrite in Amendment 10, Figures 7.7-7a through h have been moved to Appendix H.A.

There is no discrepancy; 4.4.3.3.3.1 describes the nominal speed at which the pump will be driven during steady-state conditions when powered from the LFMG set. Figure H.A-1c of the FSAR specifies speed ranges between which the functions described are permitted to occur during transient conditions existing when the pump is started, stopped, or transferred from one steady-state speed to another steady-state speed.

Typically, 4.4.3.3.3.1 describes the pump as being started on 100% power source, tripped when the pump approaches full speed, and allowed to coast down where the LFMG set then continues to drive the pump at 25% speed. Figure H.A-1c describes the same function but is more definitive in describing that the LFMG cannot start powering the pump until the pump speed is between 20 and 26% of rated. Since the LFMG output frequency is 25% of the pump motor rated frequency, the final steady-state speed will be 25% of rated.

e. The primary and secondary modes of operation for all containment isolation valves have been incorporated in Table 6.2-16 (see response to Question 022.044). The primary mode of operation is indicated under column headed "Isolation Signal" and the secondary mode under "Back Up".

f. See 7.3.1.1.2.4, 6.2.4.2, and response to Question 031.001(h). Due to a complete Chapter 7 rewrite in Amendment 10, 7.3.1.1.2.4 has been changed to 7.3.1.1.2.B.

g. The discrepancy is clarified in the revised text for 6.2.4.3.2.1.2.1.

h. See revised 1.8.

i. The revised FSAR Chapter 7 no longer makes reference to FSAR Figure 1.2-5. Section 7.2.1.1.4.2.e has been replaced by 7.2.1.1(2)e. However, the turbine governor valve hydraulic line pressure switches C72-N005 A, B and C72-N005 C, D are mounted on local instrument racks IR-10 and IR-11, respectively. The location of these instrument racks is shown on General Plant Arrangement Drawing, FSAR Figure 1.2-5.

A routing drawing of the pressure switch sensing lines is not available at this time as they are field routed and have yet to be installed.

j. Due to a complete Chapter 7 rewrite in Amendment 10, FSAR 7.2.1.1.4.4.5 has been changed to 7.2.1.1(b). This section has been revised to indicate a 10-second time delay.

k. There is no discrepancy. Note 4 of the FSAR Table 7.1-2 states that the active range for the reactor vessel water level is -150/0/+50 with zero at the top of the active fuel which includes wide and narrow range instruments. This is not defining any specific water level instrumentation, but rather the overall range of the water level instrumentation for the plant. Line 3 of FSAR Table 7.2-1 specifies a range of 0-60" for the low water level 3 instrument, a narrow range instrument, which is a portion of the reactor vessel water level range used for the RPS. Note 2 of the table states that "The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored." Therefore, the upper limit of 60" was selected as being above the maximum 50" active water level. Measurement below the top of the active fuel is of no interest or value for a RPS trip function.

Amendment 10, revising Chapter 7, moved the entry to which Note 4 applied to Table 7.7-2 which references Zimmer I as identical in design. Note 4 has since been deleted.

l. Drawing 807E180TC, Revision 12 (2/28/78), has deleted PRT and REVAB.

m. Section 7.3.1.2.7 has been replaced in the Chapter 7 rewrite by 7.3.1.2.F which now reads as follows:

"Refer to Tables 3.11-1 through 3.11-5 and paragraph 3.1.2.1.4.1 for environmental conditions. Refer to Sections 8.2.1 and 8.3.1 for the maximum and minimum range of energy supply to ESF instrumentation and controls; all ESF instrumentation and controls are specified and purchased to withstand the effects of energy supply extremes."

n. Due to a complete Chapter 7 rewrite in Amendment 10, FSAR 7.6.1.4.2 has been changed to 7.6.1. Reference to the RPS buses as "critical" has been eliminated.

o. The only actual discrepancy existed in Table 7.3-2. This table did not list the confirmatory reactor vessel low water level (Level 3) used in the ADS initiation logic. Amendment 10, revising Chapter 7, revised the tables to clarify water level trips and renumbered Tables 7.3-2, 7.3-3, 7.3-4 and 7.3-5 to 7.3-3, 7.3-5, 7.3-7 and 7.3-9 respectively.

p. Due to a complete Chapter 7 rewrite in Amendment 10, FSAR Table 7.7-3 has been changed to Table 7.7-5. As stated in Notes (3), (5), and (6) of Table 7.3-3, the instrument setpoints are subject to change to agree with Chapter 16, "Technical Specifications", which have not been submitted and are under development. The actual trip settings will be established when the Technical Specifications are submitted, after which Tables 7.3-1 (HPCS), 7.3-3 (ADS), 7.3-5 (LPCS), 7.3-7 (LPCI), 7.3-9 (PCRVCS), 7.2-1 (RPS) and 7.4-1 (RCIC) will be revised. The trip setpoints will take into account accuracy, calibration and drift allowances so that the required actuation will fall within the analytic or design basis limits.

q. The statement in 7.6.1.8.1 has been removed from the revision to Chapt. . Table 7.1-2 is correct as is.

r. The section discussing Recirculation Pump Trip (RPT) system instrumentation and controls has been rewritten to refer to Appendix H and 5.4. There are two RPS divisions.

s. Due to a complete rewrite of Chapter 7 in Amendment 10, FSAR 7.6.2.8.2.1.1.4 has been changed to 7.6.2.3.A.4 and reads as follows:

"Equipment Qualification (IEEE 279-1971, Paragraph 4.4)

Vendor certification requires that the sensor associated with each of the systems required for safety trip variable, manual switches, and trip logic components perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in conjunction with the existing field experience with these components in this application, will serve to qualify these components.

Qualification tests of the relay panels are conducted to confirm their adequacy for this service. In-situ operational testing of these sensors, channels, and the entire protection system will be performed at each project site during the preoperational test phase.

For a complete discussion of equipment qualification for the safety-related systems described in 7.6, refer to 3.5, 3.6, 3.10 and 3.11".

t. WNP-2 is presently reviewing qualification requirements of Class 1E equipment. A composite list of Class 1E equipment, including RPT initiation sensors, logic, breakers, etc., will be entered in Tables 3.10-1 and 3.10-4 or equivalent tables generated. The revision to Chapter 7 is removing the cross-reference between 7.6 and 3.10.

u. The Rod Sequence Control System referred to in 7.6.1.7.8 is a non-safety-related system. Environmentally-related statements have been removed from the text and the text has been moved to 7.7.

Q. 031.081
(3.10)
(3.11)

Identify each type of relay in the WNP-2 facility which must be energized or which must remain energized, during a seismic event. For each of these relay types, provide the following information: (1) the minimum voltage at which it must operate; (2) the voltage at which it was seismically qualified; (3) the normal operating voltage; and (4) the locations and functions of this type of relay. Where a particular relay was not qualified by test or was not tested in both the energized and de-energized state, justify the seismic qualification of the relay.

Response:

WPPSS has established a safety-related equipment re-evaluation program. This program will re-evaluate the equipment's original qualification. Its intent is to address the elements of qualification identified in new seismic and new environmental requirements. This program of re-evaluation is scheduled for completion during Decembr 1980, at which time the requested information will be provided to your SQRT personnel. Table 031.081-1 provides a list of the relays identified to date.

TABLE 031.081-1

MANUFACTURER	MODEL NO. OR TYPE	NOMINAL VOLTAGE RATING	OPERATING MINIMUM VOLTAGE	VOLTAGE AT WHICH SEISMIC QUALIFIED	PICK-UP VOLTAGE	DROP-OUT VOLTAGE	LOCATION	FUNCTION
Asea	RXMK1	120 VAC	Later	Later	96V	36V	Reactor Bldg.	Containment Isolation Valve
Struthers-Dunn	219 BBXP	120 VAC	Later	Later	80% DC 85% AC	None Specified	Control Room	Containment Atmosphere Control Isolation Valve Containment Supply Purge Indication
Agastat	GPI	120 VAC	Later	Later	92V @ 20°C	20-40% of Rated	Reactor Bldg.	Containment Atmosphere Isolation Interlock
	7012PH	125 VAC	Later	Later	100 VDC	10% of Rated	Control Room	RHR Heat Exch. Bypass Interlock
	7012AC	120 VAC	Later	Later	102 V	60V	Diesel Bldg.	DEA Vent after HPCS, Div. I & II DG Off
	7012AD	120 VAC	Later	Later	102 V	60V	Diesel Bldg.	Low Diff. Press. Trip DMA Fan for Cable Cooling System
	7022AI	120 VAC	Later	Later	102 V	60V	Diesel Bldg.	DMA Cooling after HPCS, Div. I & II DG Off
Denison	WE-74/EX-2	120 VAC	Later	Later	N/A	N/A	Reactor Bldg.	Containment Supply Purge Indication
Sylvania-Clark	7305-PM-5U4	120 V	Later	Later	85V	50V	Diesel Bdlg.	DMA & DEA Interlock with Div. I or Div. II Diesel Generator
GE	HEAG1	120 VDC	Later	Later	N/A	N/A	Diesel Bdlg.	HPCS Diesel Generator Lockout Relay
	HMA	120 VAC 125 VDC	Later Later	Later Later	90VAC 75VDC	6-12VAC 6-12VDC*	Control Room	Multi-System Logic Interlocks
	HFA	120 VAC 125 VDC	Later Later	Later Later	90VAC 75VDC	88VAC 3-13VDC*	Control Room	Multi-System Logic Interlocks
	12CFD22B2A	N/A	Later	Later	N/A	N/A	Diesel Bldg.	HPCS Diesel Generator Current Diff. Relay
	CR2820	120 VAC 125 VDC	Later Later	Later Later	98VAC 100VDC	75VAC 82VDC	Control Room	Minimum Flow Valve Time Delay
	CR105	115V 60 Hz	Later	Later	98V	75V	Control Room	Reactor Protection System Auto Scram

* After being continuously energized, pickup and dropout V increased by 10-20%.

AMENDMENT NO. 10
July 1980

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Q. 031.082
(7.6.2)

Demonstrate that the safety-related equipment discussed in Section 7.6.2.3.2.1 of the FSAR satisfies the requirements of General Design Criteria 1, 2, 3, 4, 13, 14, 16, 19, 23, and 55. Provide this demonstration of compliance with the requirements of Appendix A to 10CFR Part 50 in other sections of the FSAR where it is missing.

Response:

The applicable General Design Criteria that apply to the high pressure/low pressure interlocks are discussed in revised 7.6.2.2 and 7.1.2.2.

Q. 031.083
(3.11.3)
(3.11A)
(031.006)
(031.056)
(031.059)

Neither your response to Item 031.006 nor Appendix 3.11A of the FSAR satisfy our need for additional information on equipment qualification. In order to ensure that your environmental qualification programs conform with General Design Criteria 1, 2, 4, and 23 of Appendix A and Section III and XI of Appendix B to 10 CFR Part 50, and to the national standards (e.g., IEEE Standard 323-1971) mentioned in the Acceptance Criteria contained in Section 3.11 of the Standard Review Plan, NUREG-75/087, provided an amended response to Item 031.006 for:

a. The logic equipment for the standby gas treatment system (031.006, item (d) of the second paragraph).

b. The following sensors: (1) the rod block monitor flow transmitters; (2) the main steam line tunnel temperature thermocouple; and (3) B22-N024A.

c. All items listed in Questions 031.056 and 031.059.

Response:

a. The requested information for logic equipment for the standby gas treatment system will be provided as part of the overall re-evaluation program for seismic and environmental qualification. See response to Question 031.006.

b. The requested information will be provided as part of the overall re-evaluation program for seismic and environmental qualification. See response to Question 031.006.

c. 031.056 - See response to Question 031.006.

031.059 - See revised response to this Question.

Q. 031.084
(3.11A)

The specification requirements of Table 3.11A-1 of the FSAR are incomplete since they do not address the maximum and minimum values of all of the parameters which are cited in Section 3(7) of IEEE Standard 279-1971. Accordingly, provide the required data for all Class 1E components.

Response:

See response to Question 031.006.

Q. 031.085
(7.6.1)
(7.7.1)

Several systems (e.g., the safety/relief valve discharge line temperature monitoring system and the reactor vessel head leak detection system) are listed in both Sections 7.6 and 7.7 of the PSAR. However, Section 7.6 should describe only those systems required for safety while Section 7.7 should describe only those systems not required for safety. In conformance with the guidance contained in Section 7.7 of Regulatory Guide 1.70, Revision 2, safety-related systems should not be listed in Section 7.7 of the FSAR. Accordingly, revise your FSAR to eliminate such ambiguous design descriptions of safety-related systems and non-safety-related systems.

Response:

FSAR 7.6 and 7.7 have been revised as part of the Chapter 7 rewrite effort. Section 7.6 now describes only "all other systems required for safety" and 7.7 now describes only "control systems not required for safety" or non-safety-related instrumentation and controls of systems not described anywhere else in the FSAR.

Q. 031.086
(7.4.1)
(9.3.5)

The standby liquid control system (SLCS) is designated in Section 7.4.1.2.3.1 of the FSAR as a special plant capability event system in the WNP-2 facility. To assure the availability of the SLCS, you have provided in parallel, two sets of those components required to actuate this system. However, our review indicates that you have not provided redundant heating systems. Additionally, the heating equipment supply emergency bus is neither identified nor is it redundant. We have concluded, therefore, that your statement in Section 9.3.5.3 of the FSAR that "...a single failure will not prevent system operation..." is not true. (Note that this matter has been resolved in similar facilities by providing the redundant heating systems.) Accordingly, provide a modified design of the SLCS which satisfies the single failure criterion. Alternatively, justify your present design.

Response:

See also the response to Question 031.073.

The standby liquid control system is neither designed nor required to satisfy single failure criteria. The injection controls have redundant circuits, but only to enhance availability insofar as practical. The SLCS is a backup to CRD in the event that the rods cannot be inserted during normal plant operation by operator control of the Reactor Manual Control System. Hence, the SLCS by itself is not required to meet single failure criteria. Therefore, the system heaters are not required to be redundant nor does the power supply under normal conditions need to be powered from an emergency bus. Section 9.3.5.3 of the FSAR reads: "The SLC system is required to be operable in the event of a station power failure; therefore, the pumps, heaters, valves. and controls are powered from the standby A-C power supply. The pumps and valves are powered and controlled from separate buses and circuits so that a single failure will not prevent system operation."

Due to a complete Chapter 7 rewrite in Amendment 10, 7.4.1.2.3.1 has been changed to 7.4.1.2.

Q. 031.087
(3.11.2)

With regard to Section 3.11.2.3 of the FSAR, provide the following additional information and clarifications:

a. Provide a copy of the procedures for the following aging simulations: (1) thermal; (2) radiation; (3) operation; and (4) seismic.

b. Provide justification for the aging temperature which was used with respect to the maximum normal environmental conditions which are listed in Table 3.11-1 of the FSAR.

c. Indicate the thermal aging acceleration rate and provide the basis for this rate.

d. Indicate the thermal aging time used for each plant location listed in Table 3.11-1 of the FSAR which contains a valve that has been qualified in accordance with IEEE Standard 382-1972. Identify the valves which are so qualified.

e. Provide information similar to that requested in Items (b) through (d) above for radiation aging. In addition, describe how the effect of the neutron fluences was considered.

f. Provide your criteria for determining the limits of an actuator family including: (1) the definition of the limits of an actuator family; (2) the criteria which were used to assure that the sample valve operator is a valid representative of the family; and (3) a demonstration of how the criteria were applied.

g. Provide a table of the following information for all Class 1E valve actuators in the WNP-2 facility: (1) the equipment specifications in accordance with Section 3 of IEEE Standard 382-1972; (2) an identification of the family membership; and (3) an identification of the samples.

h. Indicate the number of operating cycles to which each test specimen was subjected.

i. Indicate the frequency range which was used in the seismic qualification and aging of the samples. (Note that the frequency range permitted by IEEE Standard 382-1972 does not agree with our acceptance criteria contained in Paragraph II.1.a of Section 3.10 of our Standard Review Plan, NUREG-75/087. We will require conformance with our positions in this latter document.)

j. Describe how you assure that equipment not qualified for all service conditions, will not spuriously operate during exposure to service conditions, including excessive exposure times during which this equipment is not required to function to mitigate the effects of accidents on other events.

Response:

a. The procedures for the following aging simulations are as specified in IEEE 382-1972:

(1)	Thermal:	Refer to IEEE 382-1972, Part II, Section 2, Page 10.
(2)	Radiation:	Refer to IEEE 382-1972, Part II, Section 1, Page 10.
(3)	Operation:	Refer to IEEE 382-1972, Part II, Section 3, Page 10.
(4)	Seismic:	Refer to IEEE 382-1972, Part I, Section 4, Paragraph 4.3, Page 8.

These IEEE-382 procedures provided the outline for valve actuator qualifications. Actual valve test parameters are discussed in the following sections.

b. The actual thermal aging qualification test parameters which were imposed for the NSSS safety-related actuator applications at WNP-2 were based on Part II, Section 2, Page 10 of IEEE 382-1972. This basis (i.e., 140°F for 40 years at 55% relative humidity) envelopes the normal average temperature of 3.11.1 for the worst case location of safety-related valve actuators on WNP-2. Note (6) of Table 3.11-1 states that the maximum (abnormal) temperature and humidity will occur less than 1% of the time and, therefore, this temperature is not used as the basis for the aging of valve actuators. During refueling and maintenance times (more than 1%) the temperature of the primary containment will be much less than the average temperature (135°F) which balances the maximum temperature conditions and durations. Therefore, the aging program basis of IEEE 382-1972, envelopes the WNP-2 requirements.

c. The NSSS actuators are qualified for thermal aging in accordance with IEEE 382-1972, with the aging acceleration rate justified by the application of the 10°C rule (standard industry practice).

d. All plant locations listed in FSAR Table 3.11-1 have the same aging time to qualify NSSS valves. The aging time is derived by using the 10°C rule and the accelerated aging temperature. This aging time/temperature is based on the Part II, Section 2 of IEEE 382-1972 which envelopes the WNP-2 requirements.

e. For radiation aging, air equivalence of neutron dose to gamma dose was determined so that the actual gamma dose used in aging is the summation of gamma dose and neutron/gamma dose equivalence.

f. As required by IEEE 382-1972, a type test demonstrated that the performance characteristics of the actuator adhered to the equipment specificatons and met all functional test requirements of IEEE 382-1972. The sample valve actuator was constructed using normal manufacturing processes and was then subjected to the test program. The test program for the sample valve actuator consisted of subjecting the actuator to the following sequence of conditions to simulate the design basis service conditions of the actuator: (a) aging, (b) seismic, and (c) accident. These test conditions are detailed in subparagraph 2 below. No maintenance was performed during this type test.

The manufacturer's equipment test specifications for the qualification sample are presented below and encompass the most severe conditions of equipment service.

(1) The valve operator was required to operate and remain operable during plant normal, test, design basis event, and post-design basis event conditions.

(2) The valve operator was required to provide rated mechanical force for the following conditions:

a. range of voltage - 230 to 460 volts;

b. range of frequency - 1 to 35 hertz @ 1.0g, (including seismic forces) 4 to 34 hertz @ 3.0g, 35 hertz @ 5.0g;

c. thermal conditions - see figure;

d. mechanical aging - 500 cycles, open and close; and

e. radiation exposure.

(3) The mounting configuration for the valve and operator was specified as mounted in a nominally horizontal run of pipe with the valve stem nominally vertical.

(4) Lubricants and seals have a minimum design life of 5 years.

(5) The design life of the valve operator is 40 years.

(6) Control and indicating devices contained on the valve operator include a torque switch and a limit switch.

All electrical valve actuators used in the NSSS design are in one family. The designation for this family has been established by the vendor as the "SMB" family. The sample valve actuator which was used to qualify this family was designated as follows:

Manufacturer:	Limitorque
Type:	SMB
Size:	0
Order No.:	360943A
Serial No.:	144068

g. Same as f above.

h. The test actuator was subjected to 500 cycles as indicated in IEEE 382-1972.

i. A search for resonance was performed by scanning the test specimen in the three major axes. Scanning was done in the range from 1 to 35 hertz at a maximum acceleration of 1g. This testing identified no resonance. Next, the test specimen was vibrated at even-integer frequencies from 4 to 34 hertz for a period of 10 seconds at an excitation of 3g in each of the three major axes. The test specimen was actuated at each dwell for one complete cycle (open and close). The test specimen was then vibrated at 35 hertz for 10 seconds in each of the three major axes at an excitation of 5g and was actuated for one complete cycle.

Specific quantification of actuator qualification is embodied in the qualification test reports which are available for review at GE-NED (San Jose).

j. Equipment qualification is conducted on the safety-related actuators to assure that equipment will not operate spuriously. Safety-related NSSS valve actuators are temperature qualified to IEEE 382-1972 by test for the equivalent active 40-year plant life plus LOCA conditions. Also, these valve actuators are qualified for radiation on the basis of integrated radiation doses from LOCA plus 40 years life conditions.

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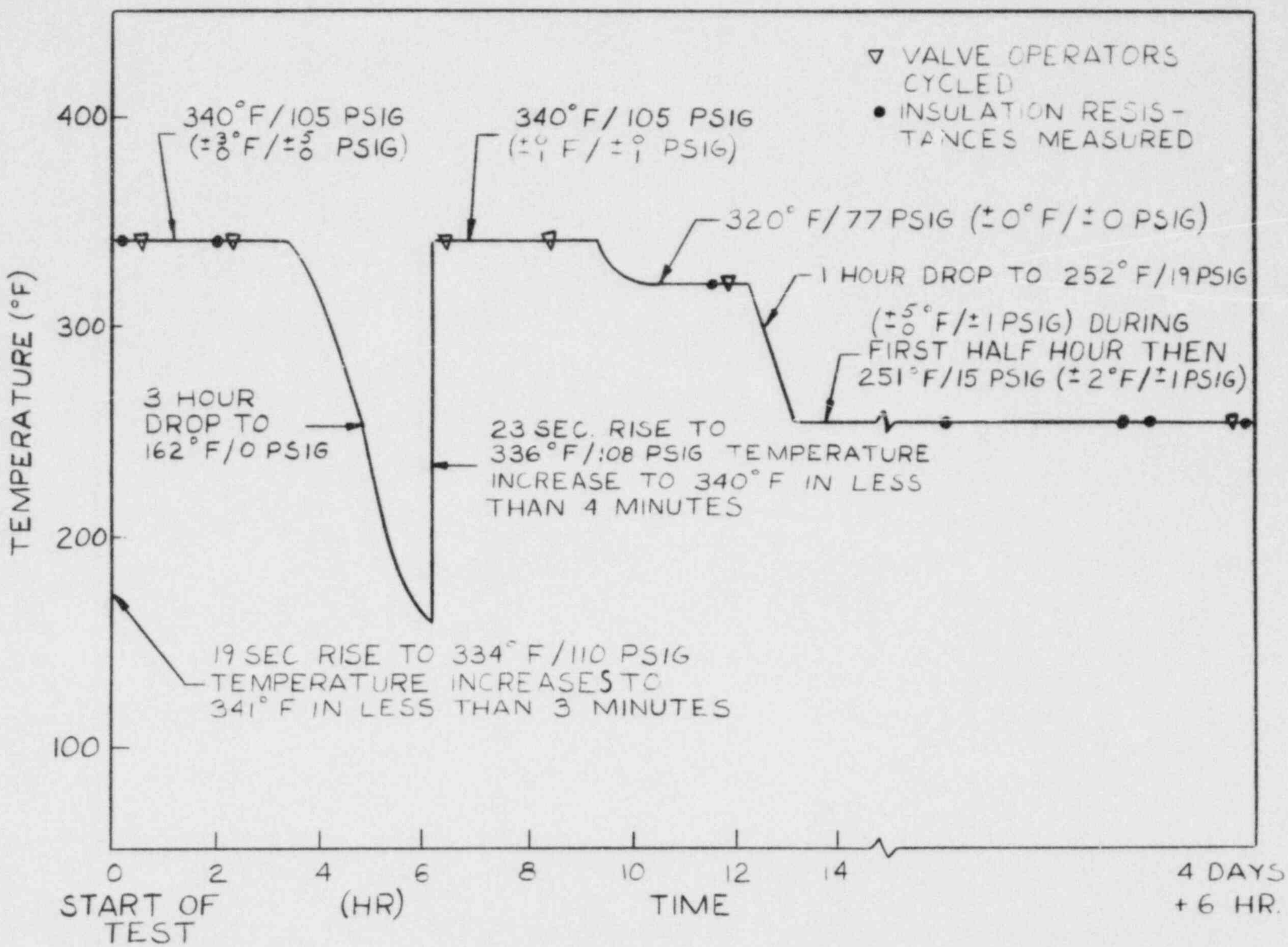
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2



ACTUAL STEAM EXPOSURE PROFILE

FIGURE
031.
087-1

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Q. 031.088
(6.2.4)
(7.6.1)
(7.7.1)
(9.3.5)

Clarify the discrepancies between the following sections and figures of the FSAR with regard to isolation of the reactor water cleanup system when the standby liquid control system is initiated: (1) Sections 6.2.4.3.2.1.1.7, 7.7.1.3, 7.6.1.4.3.6, and 9.3.5.2; (2) Figures 7.3-11a, 7.4-3, 7.7-14, and (3) Table 7.3-13.

Response:

Isolation of the RWCU system by SLCS initiation is not described in 6.2.4.3.2.1.1.7 and 7.6.1.4.3.6 or in Figure 7.3-11a because these sections and figure discuss provisions for assuring primary containment integrity following a RWCU system line break outside the containment. RWCU trip with SLCS initiation is provided, not to assure containment integrity, but to assure proper operation of the SLCS and is, therefore, only discussed in 9.3.5.2 and shown in Figure 7.4-3 as part of the SLCS system description.

The discussion of the RWCU system has been removed from 7.7 since RWCU is not a major plant control system.

Note that Amendment 10, revising Chapter 7, renumbered Figures 7.3-11a and 7.7-14 to 7.3-10a and 7.3-1, respectively. The discussion in 7.6.1.4.3.6 has been moved to 7.6.1.3.b and Table 7.3-13 has been removed from Chapter 7 and all information incorporated into Table 6.2-16.

Q. 031.089
(1.7)
(6.7.3)

The single failure analysis presented in Section 6.7.3.1 of the FSAR is inadequate. Accordingly, revise this section to include single failures of electrical components such as the spurious closing of relay contacts on K4. (Refer to GE Drawing 851E708TD.) Provide the electrical schematic and one-line drawings of this system for our review.

Response:

See response to Question 031.076.

Q. 031.090
(7.2.1)
(F7.2-1a)
(031.032)

In Section 7.2.1.1.2 of the FSAR, you state that the reactor protection system (RPS) is Class 1E. However, based on our review of similar facilities (e.g., Zimmer), we believe that this statement is incorrect since the WNP-2 motor-generator sets of the RPS are probably not Class 1E equipment. Accordingly, correct the discrepancy between Sections 7.2.1.1.1 and 7.2.1.1.2 of the FSAR regarding the qualification of the RPS motor-generator sets. Alternatively, demonstrate that all components of the RPS are Class 1E equipment.

Additionally, describe how the design and implementation of your RPS satisfies the requirements of Section 6.6 of IEEE Standard 379-1972, with special emphasis on the last paragraph of this section.

Response:

Section 7.2.1.1.2 has been deleted from the recently revised Chapter 7. The RPS motor-generator sets are now discussed in 7.2.1.1.

Class 1E protective devices (over/under voltage and frequency) have been included in the design to preclude adverse influence to the protection system from the motor-generator sets in the event a failure occurs.

Q. 031.091
(F7.2-9)

In facilities similar to WNP-2, the wiring from the RPS relay contacts 14A and 14C, via cabinet penetration Y, and 14E and 14G, via cabinet penetration Z, appears on terminal strip CC. (Refer to the GE drawing 807E-166TU.) This wiring is powered from two separate Class 1E d-c busses. Insufficient physical separation was provided between these busses on terminal strip CC, the associated cables, and in penetrations Y and Z, which also serve the plant process computer system. Our concern is that there may be insufficient physical separation in the RPS cabinets of the WNP-2 facility since it is our understanding that they are being manufactured by the same vendor. Accordingly, if this same problem exists in the RPS cabinets of the WNP-2 facility, we will require you to provide an acceptable design for the routing of Class 1E circuits inside the RPS cabinets. Alternatively, demonstrate that our concern on this matter is not applicable to the WNP-2 facility.

Response:

The separation review performed on WNP-2 designs resulted in the rework of the Reactor Protection System Cabinets. This rework included redesign of cabinet penetrations and contactor enclosures eliminating the problems described.

Q. 031.092
(F7.2-9)

In facilities similar to WNP-2, the cabinet lighting circuit which is not treated as an associated circuit, crosses cabinet penetration 187 in RPS cabinet A, and as a result, becomes associated with the containment isolation system wiring going to penetration 187. Our concern is that the physical separation provided in the RPS cabinets of the WNP-2 facility may not satisfy either the requirements of IEEE Standard 279-1971 or the WNP-2 separation criteria. Accordingly, if this problem of physical separation exists in the RPS cabinets of the WNP-2 facility, we will require you to take the following corrective actions:

a. Provide a modified design for the routing of non-Class 1E circuits in RPS cabinet A which satisfies the separation criteria.

b. Review the design of all other Class 1E cabinets for similar defects and indicate the cabinets which you reviewed.

c. Advise us of your findings and plans for the modifications necessary to satisfy the separation criteria.

d. Identify and justify all exceptions which you may take to items (a), (b) or (c) above.

e. Provide panel layout drawings and one line diagrams which show the routing and physical separation between the reactor trip sensors and: (1) the high level cutoffs for the HPCS and RCIC; and (2) the post-accident reactor vessel level indication system.

Response:

a. WNP-2 has recently completed a review of separation criteria versus actual installations. The review included the control room cabinets and panels as well as PGCC. See response to Question 031.100 for a description of the WNP-2 separation criteria. The RPS cabinets were reviewed and several modifications have been made. Cabinet lighting wiring has been separated from safety-related wiring. However, it should be noted that neither the WNP-2 separation criteria nor IEEE 279-1971 requires that non-Class 1E circuits be separated in any way from Class 1E circuits unless the non-Class 1E is supplied from a redundant division Class 1E power source.

b. Same as (a) above.

c. Same as (a) above.

d. Same as (a) above.

e. There is no criteria requiring physical separation between reactor trip sensors and HPCS/RCIC high level trips or reactor trip sensors and the reactor vessel level indication system, unless two or more noncompatible electrical divisions exist on a particular device. As part of the separation review, WNP-2 local instrument racks were evaluated and design changes incorporated to eliminate separation problems. However, two divisionally noncompatible systems (which are not electrically or functionally redundant), RPS and HPCS level trips, remain on a single instrument. Contact-to-contact separation has been employed with circuit wiring leaving the instrument housing via different feed-throughs and conduits to separated terminal boxes. The drawings showing these revisions are in the process of being updated.

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Q. 031.093
(7.2.1)
(7.2.2)

Provide in Sections 7.2.1.1 and 7.2.2.1 of the FSAR, the design criteria and a description of the scram discharge volume switches and their qualification testing, including the following information: (1) the manufacturer; (2) the type of float (i.e., whether it is self-equalizing or sealed); (3) the float material and the magnet material; and (4) the qualification test conditions including the water temperature; the pressure; the duration of the test conditions; the number of test cycles; the period between test cycles; the extremes of external temperature, pressure, and humidity; and the radiation source, strength, and dose.

Response:

The scram discharge volume switches are Quality Class 1E magnetic switches which use non-self-equalzing floats. They are manufactured by the Magnetrol Company and are identified as Model 5.0-751. The floats are made of 347 stainless steel, while the magnet material is ALNICO 5.

Test units were subjected to ten 64-hour exposure cycles, each consisting of 16 hours at 300°F dry heat, and 48 hours at 95 to 100% relative humidity between ambient and 100°F. Ten thousand cycles of on-off operation were performed. The units were exposed to 4.4 x 10^4 RAD integrated dosage to simulate 40 years of background radiation.

Any other information concerning qualification testing will be available to your SQRT or environmental review personnel when the qualification re-evaluation review has been completed. See responses to Questions 031.006 and 031.023.

Q. 031.094
(7.2.2)
(7.3.2)
(031.033)

Your response to Item 031.033 is incomplete since it does not indicate that the individual system level indicators can be actuated from the control room by the operator. Accordingly, revise Sections 7.2.2.1.2.1.5, 7.3.2.1.2.1.6.2, and 7.3.2.2.2.1.5.1.2 and all other similar sections of the FSAR, to describe the provisions you have incorporated into the design of the WNP-2 facility to satisfy Position C.4 of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems", May 1973. (Note that this position is not intended to address the testing of annunciators, but is intended to provide manual initiation of system level indication of inoperable and bypassed status.)

Response:

The WNP-2 FSAR Chapter 7 has been modified to include a generic discussion in 7.1 of the conformance to Regulatory Guide 1.47. All other discussions in 7.2, 7.3, 7.4, 7.5 and 7.6 have been deleted.

The generic discussion describes the capability of each system level bypass indication to be manually actuated. This provides system level indication for those bypass and inoperability conditions which are not automatically indicated.

<u>Q. 031.095</u>
<u>(7.3.2)</u>

Your discussion of how the instrumentation and controls satisfy the requirements of Section 4.1 of IEEE Standard 279-1971 is inadequate. Indicate the pick-up and dropout voltage values of the bus voltage relay.

<u>Response</u>:

The manufacturer, voltage characteristics, and seismic test criteria for WNP-2 relays is provided in the response to Question 031.081.

Q. 031.096
(7.3.2)

Provide justification in Section 7.3.2.1 of the FSAR for indicating a loss of power to the motor starters by deenergizing the indicating lamps. In your response, discuss how the reactor operator can distinguish between a failed lamp and a system bypass. Demonstrate how your design of these indicating lights satisfies the requirements of IEEE Standard 279-1971 with regard to providing the reactor operator with timely and unambiguous information.

Response:

The WNP-2 system level bypass indication system includes inputs from safety-related motor starter power monitoring relays. The de-energization of these relays results in annunciation of a safety system bypass condition.

The operator will then scan the main control board section involved looking for equipment with both indicating lamps extinguished. The use of these indicating lamps to identify valve power loss is only an aid to the operator in identifying the actual component causing the system bypass after receiving annunciation.

Q. 031.097
RSP
(031.026)

Your response to Question 031.026 is unacceptable. It is our position that isolation devices which are used to provide electrical independence between Class 1E and non-Class 1E equipment must: (1) be designed, qualified, and implemented in accordance with all of the requirements for Class 1E equipment; and (2) be an integral part of the system which they are intended to protect. Accordingly, we require you to revise your response to Question 031.026 and to provide all the information requested in this previous request. (Note that this matter has been resolved in similar facilities by modifying the design.)

Response:

Question 031.063 also requested an update of the response to 031.026. The updated response was submitted in Amendment 3 in March, 1979, and responds to the concerns of this question.

Q. 031.098
RSP
(7.6)
(15.4.1)

It is our position that the use of the rod worth minimizer (RWM) is unacceptable for safety-related functions since it does not satisfy the requirements of IEEE Standard 279-1971. Accordingly, we require you to delete this system from Section 7.6 of the FSAR. (Note that you claim credit for the RWM in Section 15.4.1.2.2.1 of the FSAR which implies that the RWM is a safety-related system.)

Response:

The Rod Worth Minimizer (RWM) does not provide a safety function, nor is it required for power generation. The FSAR has been modified to move the RWM description from 7.6 to 7.7.

Section 15.4.1.2.2.1 does describe the function of the RWM and the Rod Sequence Control System (RSCS) as the means of preventing an approach to criticality at low power levels by blocking the withdrawal of an out of sequence rod.

Credit is taken for these systems because the RWM and RSCS are redundant for this extremely unlikely event. However, the analysis provided in NEDO 23842 shows that the failure to block rod withdrawal by the RWM and the RSCS is backed-up by the scram which will be initiated by the Neutron Monitoring System (either IRM or APRM) inputs to the Reactor Protection System. Both these systems are safety-related. The analysis showed that the licensing basis criterion for fuel failure is still satisfied even when the RWM and RSCS fail to block rod withdrawal.

Q. 031.099
(3.4)
(7.3.1)
(031.030)

The response to Item 031.030(c) is incomplete since you do not discuss the consequences to electrical equipment in the event of internal flooding. Section 7.3.1.2.8.1 of the FSAR is similarly incomplete. Accordingly, provide a revised response to Item 031.030(c) which discusses the protection of Class 1E equipment from internal flooding (e.g., a failure of either the main condenser cooling line or of the fire protection system).

Response:

The protection of Class 1E equipment from internal flooding is discussed in FSAR 3.4.1.4.1.2 and 3.4.1.5.2, which were submitted in Amendment 5. The ECCS equipment in the reactor building basement, where the building sumps are located, are protected from internal flooding due to post-LOCA. ECCS passive failures by a Class 1E leak detection system are discussed in the response to FSAR Question 212.003. The passive failure is isolated before it has any additional effect on ECCS ope

The potential flooding and environmental effects from postulated through-wall leakage cracks in moderate energy fluid piping systems, and postulated rupture of high energy fluid pipings are currently being re-evaluated as stated in FSAR 3.6.

The effects of the internal flooding on electrical equipment are being taken into account in the re-evaluation. The results of this analysis will be furnished by amendment to FSAR 3.6. At that time a change to Question 031.030(c) and this question will be provided.

Q. 031.100

In Table 7.1-2 of the FSAR, you indicate that many of your instrumentation and control systems are identical to those of LaSalle and Zimmer. During the course of our review of these facilities, which are similar to the WNP-2 facility, we encountered a number of errors in the implementation of the basic GE design. Our concern is that these same errors, or similar errors, could occur in implementing the electrical design of the WNP-2 facility. In particular, we find that your analyses in 7.3.2.1.2.3.1 and 7.3.2.2.2.3.1.1 of the FSAR, to determine compliance with the requirements of IEEE Std. 279-1971, are too general in content. We provide guidance for the information we need in Section 7.2 of the Standard Review Plan, especially in Appendix 7.2.A. Specific examples of areas where we require additional information are presented in Items 031.081, 031.084, 031.091, and 031.092 of this enclosure. Accordingly, provide more specific analyses of how you have implemented, in detail, the basic GE electrical design in the WNP-2 facility. References to other sections of the FSAR are acceptable in lieu of repeating this information in 7.3.2.1.2.3.1.

Response:

The GE separation criteria has been integrated into the WNP-2 separation criteria. A copy of the criteria is attached.

Cable Separation Criteria

Objective

The installation of electrical cables shall be in accordance with the following design criteria. The purposes of these criteria are as follows:

a. To preserve the independence of redundant safety-related electrical systems.

b. To minimize the influence of a non-safety-related cable on safety-related cables.

c. To minimize the influence of various types of cables (instrumentation, power, etc.) on each other.

d. To give design and installation guidance to assure that separation and identification requirements are met.

Definitions and General Requirements - Balance of Plant and Nuclear Steam Supply Safety-Related Systems

Definitions

Power Cable

Power cables are defined as those cables that provide electrical energy for equipment motive power and heating requiring 14.4 kv, 6.9 kv, 4.16 kv, 480 volts, 240 volts, 120/208 volts, a-c, 250 and 125 volts d-c. (See Page 031.100-24 for further information.)

Power cables of different voltage ratings must be routed in different cable trays except as follows: (a) Common tray is permitted for 480 volt, 120/208 volt ac, 125 volt and 250 volt dc of compatible divisions; (b) Common tray is permitted for 4160 and 6900 volt power cables of compatible divisions. 480, 4160 and 6900 volt power cables are not to be installed in cable trays in the spreading area beneath the control room. If a run through this area is unavoidable, the power cable shall be installed in conduit.

Power cables shall be installed in raceways separate from control cables and low-level signal cables and where vertically stacked, the power cables shall be placed in the tray with the highest position in the tray tier. Stacking of multiple power trays shall be such that the voltage levels decrease sequentially from the top to the bottom tray in the stack.

Control Cable

Control cables are those cables using voltages 120 volts ac (or below) or 125 volts dc (or below), with normal current not in excess of 30 amperes, whose circuits are designed to supply control power for the plant systems. Included in the category of control cables are those cables used for intermittent operation to change the operating status of a utilization device of the plant system. Control cables include all cables which have any of the following functions: (See page 031.100-24 for further information.)

a. 125 volts dc or 120 volts ac feeds to switchgear, panel and local panel control buses. Wire types are to be power cable, type G2.

b. 125 volts dc or 120 volts ac feeds to solenoids.

c. 125 volts dc or 120 volts ac control and interlock circuits.

d. Annunciator circuits.

Instrument Cable (Low-level signals)

Instrumentation cables are those cables used to carry low-level analog or digital signals. Low-level signal cables require a specific degree of separation or segregation to perserve the accuracy of the transmitted signal. Low-level signal cables are run in raceways separate from all power and control cables, except within the Control Room Power Generation and Control Complex (PGCC) and as noted below. Instrument (signal) trays shall be of the enclosed (solid bottom and covers) type.

Analog and digital signal input cables shall be routed as follows:

Digital computer signals in the reactor building shall be run in Divisional control trays as applicable by the device being served. Non-Class 1E digital signals in other areas shall be run in instrumentation trays of Division B, unless they are routed through the reactor building.

Analog computer signals in the reactor building shall be run in Divisional instrumentation trays as applicable by the device being served. Non-Class 1E analog signals in other areas shall be run in instrumentation trays of Division A, unless they are routed through the reactor building.

Safety-Related Electrical and Instrumentation Systems and Equipment

Those electrical and instrumentation systems and equipment which are relied upon to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary. Safety-related systems and equipment are limited in this document to the Reactor Protection System and the Engineered Safeguards Systems.

Reactor Protection System (RPS)

The Reactor Protection System is the overall complex of instrument channels, trip system and trip actuators, and wiring which generates a reactor trip (scram) signal to initiate a reactor trip when a monitored parameter (or group of parameters) exceeds a setpoint value indicating the approach of an unsafe condition. The complete RPS is a Class 1E safety-related system.

The Reactor Protection System Power System, consisting of M-G sets, distribution panels, etc., is a separate, non-safety-related system which supplies power to the RPS itself.

Engineered Safeguards Systems (ESS)

This includes that combination of subsystems which take automatic action to provide the cooling necessary to limit or prevent the effects of fuel cladding melting, maintain the integrity of the containment, and insure that the exposure of the public to radiation will be below the limits of 10CFR100 in the event of a design basis reactor accident.

Nuclear Steam Supply Shutoff System (NSSS)

The instrument channels (except those common to RPS), power supplies, trip systems, manual controls, and interconnecting wiring involved in generating a NSSS system function. Instrument channels for the isolation functions which are shared with the Reactor Protection System are considered a part of the RPS as far as separation is concerned.

Instrument Channel

An arrangement of sensory and intermediate components as required to generate a single trip signal related to a particular plant parameter and introduce this trip signal into a trip system. The channel loses its identity upon combination of its trip signal with others.

Trip System

An interconnected arrangement of components making use of instrument channel outputs in the generation of a trip function when appropriate logic is satisfied.

Trip Actuator

The mechanism which carries out the final action of the protection system.

Redundant System

A system or sub-system whose function can be provided by another system or sub-system.

Standby Power Sources

Emergency "on-site" power sources designed for use when off-site power is not available. These include engine-driven generators and station batteries.

Single Failure

A single failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be part of the single failure. Systems providing safety functions are considered to be designed against an assumed single failure, if a single failure of any component does not result in a loss of capability of the systems to perform their safety function.

a. Active Failure: An active component failure is defined as the malfunction or loss of function of a component of electrical or fluid systems. The failure of an active component of a fluid system is considered to be a loss of component function as a result of mechanical, hydraulic, pneumatic, or electrical malfunction, but not the loss of component structural integrity.

b. Passive Failure: A passive component failure in the sense utilized in 3.6.1.21a refers to the failure of:

1) passive electrical equipment such as shorts in cables,

2) pump or valve seals for long term cooling requirements.

Isolation Device

A device in a circuit which limits the effects of events in one section of a circuit from causing unacceptable consequences in other sections of the circuits or other circuits. Some examples of isolation devices are relays, buffer amplifiers, isolation transformers, fuses, circuit breakers and fire stops.

Raceway

Any channel that is designed and used expressly for supporting wire, cables or bus bars. Raceways consist primarily of, but are not restricted to, cable trays, wireways and conduits.

Potential Hazardous Area

This is any area in the vicinity of potential missile and external fire risk, pipe whip, and jet impingement.

General Area

This is an area from which potential hazards of missiles, external fires and pipe whip are excluded.

General Requirements

Segregation of Cables

Outside of the Main Control Room separate cable trays shall be installed for the five types of cables, i.e., high voltage power, power, control, low-level signal and RPS, with not more than one of these types of cable permitted in any tray.

Separation Details for Raceways

The degree of isolation and/or separation between raceways varies with the potential hazards within a particular area of the station. These areas are classified as follows:

a. General Areas

b. Mechanical Damage (Missile) Area

c. Fire Hazard Area

d. Cable Spreading Room

e. Control Room

Minimum separation distances are for open ventilated trays providing the following is observed:

a. Cable splices in raceways are to be prohibited.

b. Cables and raceways are to be flame retardant.

c. Design basis is that cable trays will not be filled such that cables extend above tray side rails (this approximates a 50% tray fill on a random basis).

d. Hazards to be limited to failures or faults internal to electric cables.

General Areas

a. The minimum separation distance between open cable trays of redundant divisions or between an open tray of one division and a conduit of a redundant division routed above the tray shall be three feet free air space (horizontally) and five feet free air space (vertically). However, if no automatic area fire detection and extinguishing system exists, and the lower tray is the highest tray in a tier of more than three, the minimum vertical free air space for separation shall be eight feet. The minimum separation distance between an open cable tray of one division and a conduit of a redundant division where the conduit is routed below the open tray shall be one inch. Where equipment arrangement precludes maintaining the minimum separation distance, covers or barriers are to be provided between trays of redundant divisions, as shown on pages 031.100-15 thru 031.100-18. Circuits of redundant divisions can also be run in solid enclosed raceways, such as totally enclosed trays or rigid steel conduit, where the minimum established distance for open trays is not maintained.

b. In cases of crossover of one open tray over another of a redundant division where the minimum vertical separation criteria established in a. above is not maintained, barriers consisting of solid steel covers on bottom trays and solid bottom in top trays shall be provided. These covers shall extend to each side of both tray edges by a minimum distance equal to three times the width of the widest tray involved in either division. The length of the protective covers is taken along the tray centerline. At crossovers, a minimum vertical separation of one inch is to be provided between the top of the bottom tray and the bottom of the top tray.

c. In cases of crossovers of enclosed raceways and open trays of a redundant division, the minimum separation distance shall be one inch when the enclosed raceway is below the open tray. Otherwise, vertical separation established in a. above shall be maintained.

d. Fire stops shall be used where any raceway penetrates the slab into the control room, where any raceway penetrates designated fire areas, or where any raceway penetrates areas where an ambient pressure difference exists. In addition, fire stops shall be provided where any open vertical raceway penetrates floor or ceiling slabs. Both the penetration and the trays themselves shall be sealed with fire resistant material.

Mechanical Damage (Missile) Area

a. An analysis shall be performed to assure that routing, arrangement and/or protective barriers are such that no credible locally generated missile, pipe whip or jet impingement can damage a sufficient amount of safety-related cabling or equipment to cause loss of safe shutdown/accident mitigation capability when taken with a single active or passive failure.

b. Class I Electrical Systems Cables shall not be routed through other than Class I structures to protect against earthquake damage and exposure to tornado or flooding conditions unless analysis is performed to demonstrate that loss of such cables does not negate a safety function.

c. Installation of non-Seismic Category I equipment in areas containing Seismic Category I equipment should be avoided where practicable, or adequate barriers shall be provided to protect Category I equipment, or analysis shall be performed to demonstrate that failure of the non-Category I equipment will not lead to degradation of a plant safety function.

Fire Hazard Areas

a. Routing of cables and conduits for safety-related redundant systems through an area where there is potential for accumulation of large quantities of oil or other combustible material shall be prohibited.

b. Fire stops shall be provided for all tray and conduit (at the first available junction) penetrations passing through fire rated barriers at fire hazard area boundaries (both sides).

Cable Spreading Room

a. The cable spreading room is the area under the control room where cables leaving the panels are dispersed into their various raceways for routing to all parts of the plant.

b. The minimum separation distance between open trays of redundant divisions is to be one foot between trays separated horizontalaly and three feet between trays separated vertically assuming a fire detection and extinguishing system is present. If these distances cannot be maintained, fire barriers shall be installed.

c. The minimum separation clearance between conduits and open trays of redundant divisions is 6 inches free air space when the conduit is below or to the side of the open tray and 3 feet free air space when the conduit is located above the open trays.

Control Room

a. In general, no single control panel should include wiring essential to more than one safety related redundant function. If cabling of redundant functions must be terminated in the same panel or if cables of redundant divisions run through the same panel, a minimum separation of 6 inches shall be maintained between cables and components to prevent common damage, unless separated by a barrier or an isolation device. A sheet metal enclosure and/or conduit around the intruding division wiring or component is an adequate barrier. The enclosure(s) shall include the cables, terminal blocks and the actual device (e.g., switch, light) if required.

b. For PGCC see G.E. NEDO 10466.

c. In the area behind the PGCC termination cabinets and near the Control Room walls, cables will be routed in grounded flexible conduit and the area provided with a silicone foam fill or halon fire suppression system, or an alternate method of providing electrical separation/fire protection shall be furnished.

Identification of Panels, Racks, Junction and Pull Boxes, Cable, Cable Trays and Conduit

a. General

Equipment associated with the RPS, NSSS and ESS shall be identified so that two facts are physically apparent to the operating and maintenance personnel: first, that the equipment is part of Nuclear Safeguards System; and second, the grouping (or division) of enforced segregation with which the equipment is associated.

b. Panels and Racks

Panels and racks associated with the Nuclear Safeguards System shall be labelled with marker plates which are conspicuously different in color or color of engraving-fill from those for other similar panels. The marker plates shall include identification of the division of the equipment included.

c. Junction or Pull Boxes

Junction and/or pull boxes enclosing wiring for the Nuclear Safeguards System shall have identification similar to and compatible with the panels and racks considered in b. above.

d. Cable

Safety-related cables (Divisions 1 thru 7) shall be uniquely identified by number and color code. Each cable listed in the cable schedule shall be assigned a number for identification purposes. The number shall appear on the electrical installation drawing and on the wiring diagrams on which the terminations of the cable are shown.

Cable identification tags shall be made of a permanent material and permanently attached to all cables. Tags shall indicate the individual cable number, and the particular separation division to which the cable is assigned according to the marking characteristics shown in f. below.

Cables shall be tagged at fifteen foot intervals and at their terminations. This identification requirement does not apply to individual conductors or to cables which run in conduit only.

e. Identification of Cable Tray and Conduit

Each cable tray section shall be assigned an identification code number which is made of a plastic material and applied to the sides of the tray. Moreover, those sections that are assigned a separation code corresponding to the codes assigned to each safety system cable grouping shall have their respective code numbers marked on their sides in color.

Conduits shall be tagged in a manner similar to that used for cable identification.

All trays and conduits shall be identified at entrance and exit points of each room they pass through. Conduits shall be identified at the beginning and at the end, at all boxes, and at all discontinuities.

Tray/conduit marking characteristic code is shown in f. below.

f. Marking Characteristic Code

Division	Application	Tray/Conduit Characters	Inscription Characters	Background
1	P,C,I	Div. 1	Black	Yellow
2	P,C,I	Div. 2	Black	Orange
3	P,C,I	Div. 3	Black	Red
4	RPS-A1 NSSS-A1 NMS-A	R Ch. A1	Red	Lt. Blue
5	RPS-A2 NSSS-A2 NMS-C	R Ch. A2	Red	Green
6	RPS-B1 NSSS-B1 NMS-B	R Ch. B2	Red	Dk. Blue
7	RPS-B2 NSSS-B2 NMS-D	R Ch. B2	Red	Brown
A	P,C,I	Div. A	Black	Silver or Silver/ Yellow Stripe
B	P,C,I	Div. B	Black	Gold or Gold/ Orange Stripe

P - Power
C - Control
I - Instrumentation

Non-Class 1E circuits receiving power from Class 1E power sources which are not shed by an accident signal shall be identified by the addition of checkered black/silver or black/gold markers indicating the Class 1E division (Division 1 or 2 respectively) from which the circuit receives its power and identified as A'1 or B'2 (respectively) in the computerized cable schedule.

Specific Requirements for Separation of Cables for Nuclear Safeguards Systems

Reactor Protection System (RPS, NSSS and NMS)

Reactor Protection System (RPS, NSSS and NSSS, and NMS fail-safe wiring:

a. Fail-safe wiring outside of the main protection system cabinets shall be run in rigid or flexible conduits and/or totally enclosed trays used for no other wiring and shall be conspicuously identified at all junction or pull boxes. IRM, LPRM input, and RPS Scram Group output cables may be combined in the same wireway provided that the four divisional separation is maintained.

b. Wires from both RPS trip system trip actuators to a single group of scram solenoids may be run in a single conduit; however, a single conduit shall not contain wires to more than one group of scram solenoids. Wiring for two solenoids on the same control rod may be run in the same conduit.

c. Cables through the primary containment penetrations shall be so grouped that failure of all cabling in a single penetration cannot prevent a scram. (This applies specifically to the neutron monitoring cables and the main steam isolation valves position switches.)

d. Power supplies to systems which de-energize to operate (so called "fail-safe" power supplies) require only that separation which is deemed prudent to give reliability (continuity of operation). Therefore, the protection system flywheel motor generator (MG) sets and load circuit breakers are not required to comply with the separation requirements of this Specification for safety reasons even though the load circuits go to separate panels.

e. Wiring for the four RPS scram group outputs and the NSM LPRM inputs must be routed as four separate divisions.

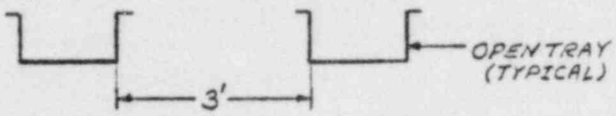
Non-Class 1E Circuits

Non-Class 1E circuits which receive power from Class 1E power sources shall be uniquely identified and comply with the same separation requirements placed on Class 1E circuits. For example, a Division A non-Class 1E circuit whose power source origin is a Division I critical bus must be separated from a Division B non-Class 1E circuit whose power source origin is a Division II critical bus.

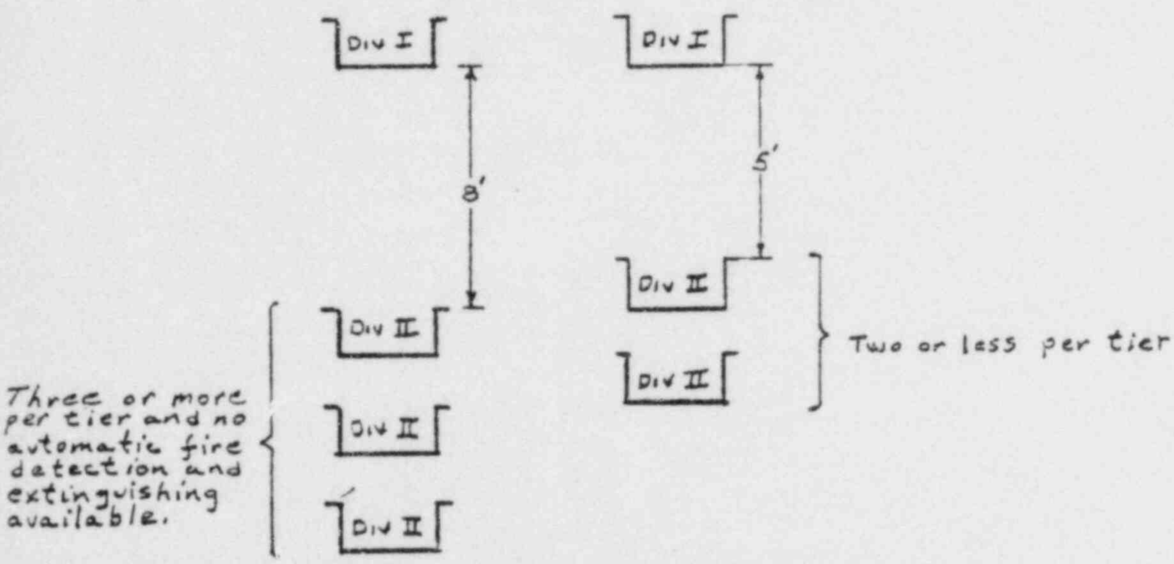
All other non-Class 1E circuits require no separation.

See Table IV for a description of acceptable non-Class 1E circuit routing.

A. Minimum horizontal separation requirements between any two redundant divisions.



B. Minimum vertical separation requirements between any two redundant divisions.



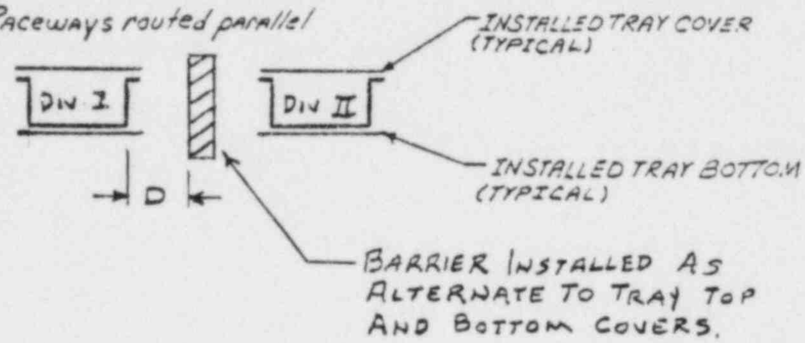
Note: Distances shown consider the ideal arrangement of two (2) raceways only. If more than two (2) trays exist in any particular arrangement, physical separation distances chosen must be based on the complete configuration.

GENERAL AREAS/OPEN TRAYS (see note, p. 031.100-15)

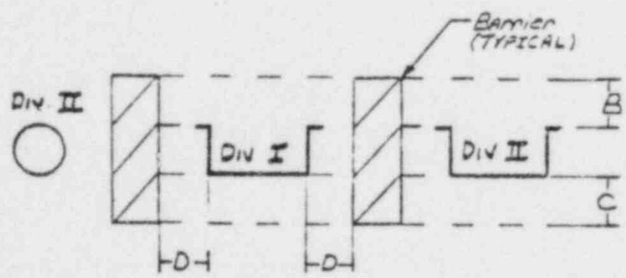
C. Where minimum separation requirements, between two raceways of redundant divisions are Not met, the appropriate solution depicted in the following illustrations shall be implemented.

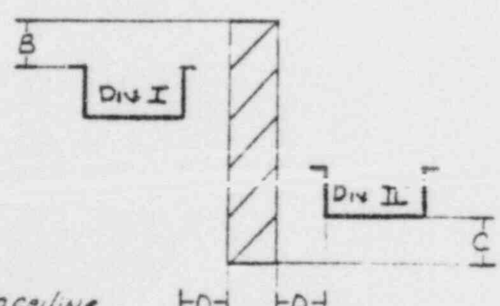
HORIZONTAL

1) Control & Instrumentation Raceways routed parallel



2) Power Raceways routed parallel





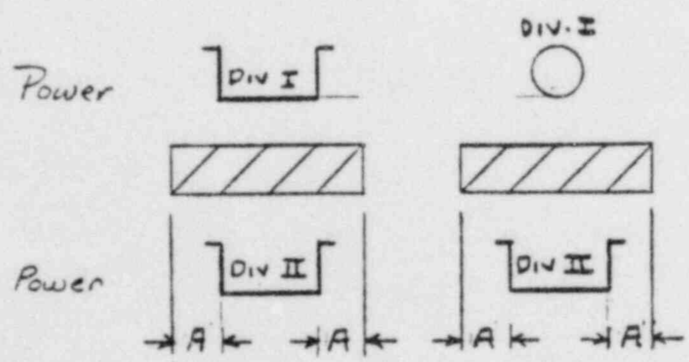
B = 12" Minimum or flush to ceiling
C = 12" Minimum or flush to floor
D = 1" Minimum

GENERAL AREAS/OPEN TRAYS

VERTICAL

See Note, P 031.100-15

Barriers & Tray Covers - Where 2 or more Power raceways, of redundant divisions are routed parallel above or below each other And where Control & Instrumentation raceways are routed Above or below Power raceways of redundant divisions.



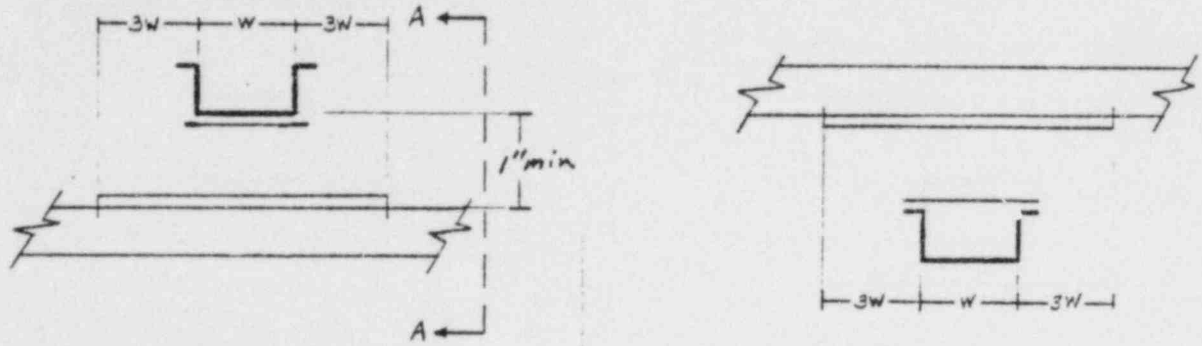
A = 12" minimum or flush to wall.

GENERAL AREAS/OPEN TRAYS

See Note, P.031.100-15

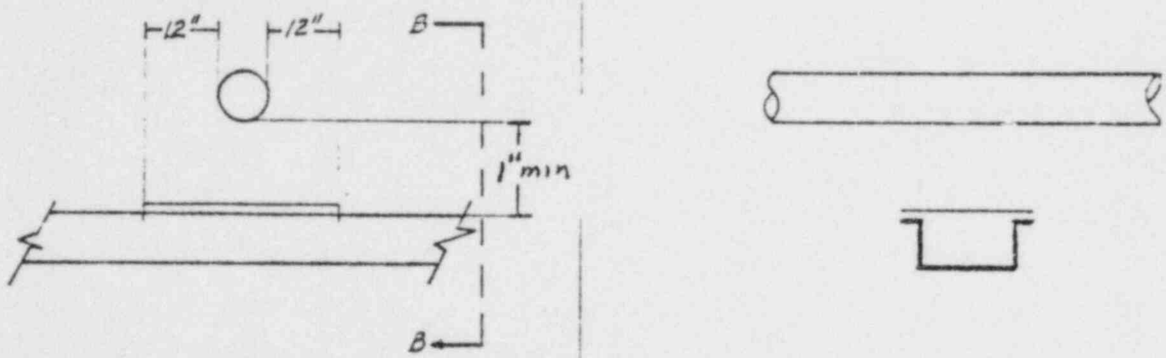
CROSSOVERS

Tray Covers shall be used for All Crossovers of **redundant** division raceway Systems. The schemes shown below shall be used regardless of the voltage level of the cables in a Crossover raceway system.

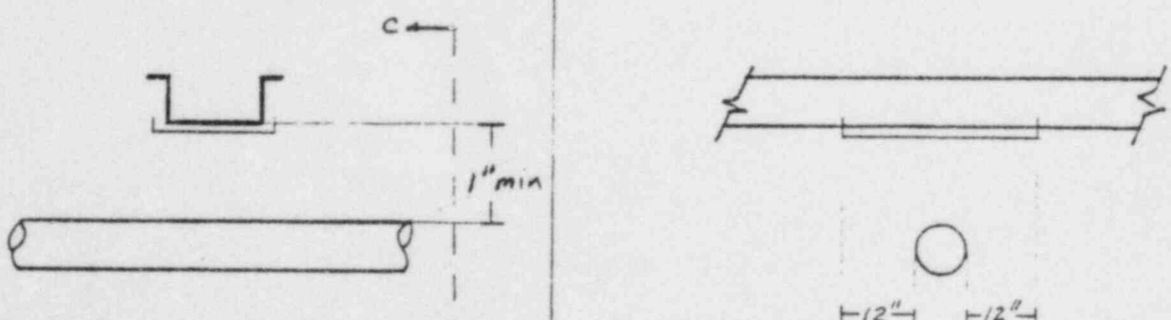


"W" is defined as the nominal tray width of the widest tray involved.
3W = 3 times the nominal tray width or flush to a wall

SECTION A-A



SECTION B-B



SECTION C-C

TABLE I

CABLE ROUTING CRITERIA
(Excluding of Redundant Channels)

GROUP (Note 1)	SYSTEM OR SERVICE	TYPE NON 1E	CLASS 1E	CONTROL	POWER	H.V. PWR.	SIGNAL	RACEWAY ENCLOSED	CONDUIT OR OPEN TRAY	REMARKS
1	RPS Controls		X	X				X		
2	RPS Scram Solenoids		X		X			X		
3	RPS Neutron Monitoring		X				X	X		
4	RTD's		X				X	X		
4	TC's		X				X	X		
4	Transducers		X				X	X		
4	Supervisory Inst.		X				X	X		
4	Radiation Monitoring		X				X	X		Note 2
4	Other Low Level Non-Dig.		X				X	X		
5	RTD's	X					X	X		
5	TC's	X					X	X		
5	Transducers	X					X	X		
5	Supervisory Inst.	X					X	X		
5	Radiation Monitoring	X					X	X		Note 2
5	Other Low Level Non-Dig.	X					X	X		
6	Digital		X				X	X		
6	Control Circuits		X	X					X	Note 3
7	Digital Ckts.	X					X		X	Note 4
8	Control Ckts.	X		X					X	Note 3
8	Communications Ckts.	X		X					X	
9	125-250 VDC, 120-480 VAC		X		X				X	
10	125-250 VDC, 120-480 VAC	X			X				X	
11	4.16-6.9 KV		X			X			X	
12	4.16-6.9 KV	X				X			X	

NOTES: 1. Same Group Number indicates a common raceway.
Different Group Number indicates separate raceway.

2. T.I.P. Cables may be combined with control cables.

3. See Par. 3.6.1.2 for control cable definition.

4. Non IE Digital Ckts. inside the reactor bldg. may be mixed with Control Ckts.

TABLE II

ASSIGNMENT OF SYSTEMS TO DIVISION OF SEPARATION

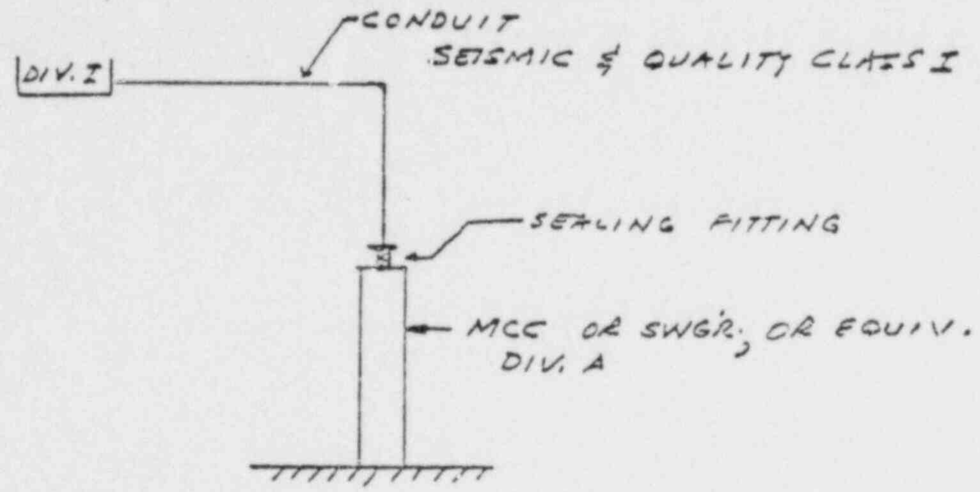
Division 1	Division 2	Division 3
RHR A	RHR B	HPCS
LPCS	RHR C	HPCS Diesel-Generator
Outboard Isolation Valves	Inboard Isolation Valves	125 VDC Battery 3
Standby Emergency Power 1	Standby Emergency Power 2	Standby Service Water C
RCIC		Safety-Related Display Instr. 3
Automatic Depressurization Div. 1 Controls	Automatic Depressurization Div. 2 Controls	
Standby Gas Treatment (Loop 1)	Standby Gas Treatment (Loop 2)	
250 volt dc Battery		
125 volt dc Battery 1	125 volt dc Battery 2	
24 volt dc Batery 1	24 volt dc Battery 2	
Standby Service Water Pump A	Standby Service Water Pump B	
MSIV-LCS (Inboard)	MSIV-LCS (Outboard)	
Leak Det. System 1	Leak Det. System 2	
CAC 1	CAC 2	
Cont. Inst. Air 1	Cont. Inst. Air 2	
SLCS 1	SLCS 2	
Mn. Cont. Rm. HVAC 1	Mn. Cont. Rm. HVAC 2	
Reactor Shutdown 1	Reactor Shutdown 2	
RPT 1 Output	RPT 2 Output	
Safety-Related Display Instr. 1	Safety-Related Display Instr. 2	
Suppression Pool Temp. Monit. 1	Suppression Pool Temp. Monit. 2	

TABLE III

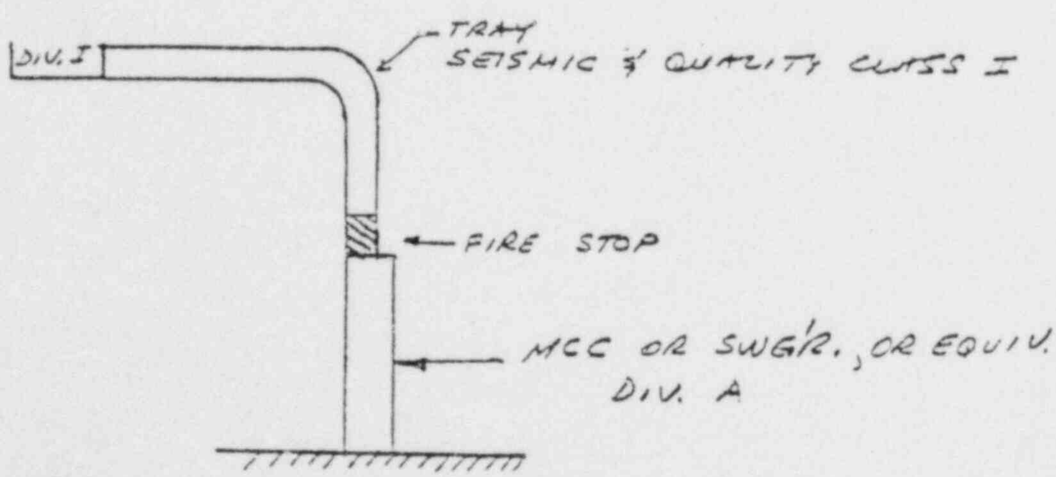
ASSIGNMENT OF RPS, NSSS AND NMS TO DIVISIONS OF SEPARATION
(FAIL-SAFE WIRING)

Division 4	Division 5	Division 6	Division 7
(PGCC Div. 1)	(PGCC Div. 2)	(PGCC Div. 1)	(PGCC Div. 2)
RPS A1	RPS A2	RPS B1	RPS B2
NSSS A1	NSSS A2	NSSS B1	NSSS B2
NMS A	NMS C	NMS B	NMS D

CASE NO. 3

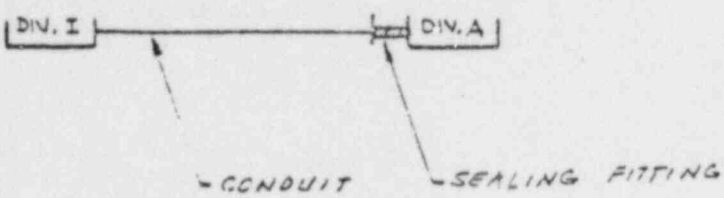


CASE NO. 4



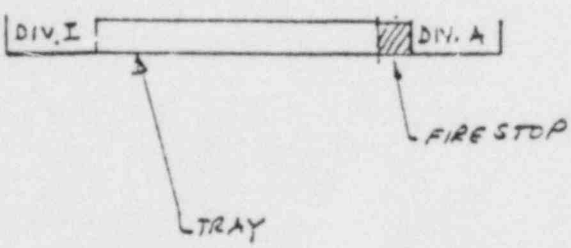
NON-CLASS 1E CIRCUITS

CASE NO. 1



SEISMIC
AND QUALITY
CLASS I.

CASE NO. 2



SEISMIC
AND QUALITY
CLASS I

	2	3	4	5	6	7	8	9	10	11	12
		LOAD TYPE									
CABLE SIZE (AWG)	SERVICE VOLTAGE (VOLTS)	MOTORS (ALL EXCEPT COL. 11	MOTOR OPERATED VALVES	SOLENOID VALVES	SPACE HEATER (INC. MOTOR HEATER)	PROCESS HEATER	TRANS (INC. PWR. AND LIGHT'G.)	FDR'S TO SWG'R & LOC. CONT. P.	METERING, PROTECTION & CONTROL CKTS.	SMALL MOTORS (* *)	
#10 AND SMALLER	120 VAC 125 VDC	P	P	C	see Note 2 C (up to 900W.)	P	P	(see Note 2) C (up to 30A. circuits)	C	C	
LARGER THAN #10	120 VAC 125 VDC	P	P	C	P	P	P	P	C	N.A.	
ANY	ABOVE 120 VAC, 125 VDC	P	P	N.A.	P	P	P	P	C	N.A.	

NOTES:

1. (* *) INCLUDED ARE: ELECTRO HYDRAULIC OPERATORS (EHO'S), HVAC DAMPERS, REACTOR START-UP RANGE DETECTOR DRIVE MOTOR, MOTORS UP TO 1/3 HP.

2. CONTROL DESIGNATION IS TO BE RETAINED FOR CABLES REQUIRING SIZES LARGER THAN #10 AWG FOR VOLTAGE DROP REDUCTION.

LEGEND

P - POWER
C - CONTROL
NA - NOT APPLICABLE

Prime designator (i.e., A', B' etc.) indicates a non-Class 1E circuit connected to a Class 1E power source.

TABLE IV

DIVISIONAL COMPATIBILITY

Divisions	1	2	3	4	5	6	7	A'	A' (9000)	B'	B' (9000)	XXX I'	XXX II'	XXX III'	XXX I	XXX II	XXX III	A	B
1	X			X		X		X	X			X			X	X	X	X	X
2		X			X		X			X	X		X		X	X	X	X	X
3			X											X	X	X	X	X	X
4	X			X		X		X	X			X			X	X	X	X	X
5		X			X		X			X	X		X		X	X	X	X	X
6	X			X		X		X	X			X			X	X	X	X	X
7		X			X		X			X	X		X		X	X	X	X	X
A'	X			X		X		X	X			X			X	X	X	X	X
A' (9000)	X			X		X		X	X			X			X	X	X	X	X
B'		X			X		X			X	X		X		X	X	X	X	X
B' (9000)		X			X		X			X	X		X		X	X	X	X	X
XXX I'	X			X		X		[illegible]	X			X			X	X	X	X	X
XXX II'		X			X		X			X	X		X		X	X	X	X	X
XXX III'			X											X	X	X	X	X	X
XXX I	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
XXX II	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
XXX III	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
A	X	X	X	X	X	X	X	X	X	X	X	[illegible]	X	X	X	X	X	X	X
B	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X

X - DENOTES NO SEPARATION REQUIRED (IE., COMPATIBLE DIVISIONS)

NOTE: CABLES SPECIFICALLY ASSIGNED TO NMS LPRM INPUTS, AND THE RPS SCRAM GROUP OUTPUTS, SHALL STRICTLY ADHERE TO FOUR DIVISIONAL SEPARATION

Q. 031.101
(7.3.2)

In Section 7.3.2.2.2.3.1.4 of the FSAR, you state that: "All components used in the isolation system have demonstrated reliable operation in ... industrial applications." Vague or general statements like this are unacceptable without the supporting basis. Accordingly, identify in Section 7.3.2 of the FSAR, the equipment which has been environmentally qualified by previous operating experience and, for each item, provide the basis for the extrapolation in accordance with the requirements of IEEE Standard 323-1971.

Response:

With the Chapter 7 rewrite, FSAR 7.3.2.2.2.3.1.4 has been removed and reference made to 3.10 and 3.11. However, in selecting such components, priority is given to selecting those which have in the past demonstrated reliable operation in nuclear applications. If such components do not exist, only those which have demonstrated reliable "industrial applications" are used. Furthermore, regardless of whether or not the component selected has a nuclear or industrial history, the above section states that "...qualification tests or analysis will be conducted (on all components) to qualify the items for this application. See 3.10 and 3.11."

Records of such qualification tests are available at GE, San Jose for NRC staff audit.

In addition, WPPSS has an environmental qualification review program underway. The results of that program will be factored into revised 3.10 and 3.11 which will respond to concerns such as those expressed in this question.

Q. 031.102
(7.3.2)

Describe in Section 7.3.2.1 of the FSAR, your proposed methods to provide for emergency operation of emergency switches and valves which are locked. (Refer to your discussion on locked safety equipment in Section 7.3.2.1.2.3.1.14(d) of the FSAR.)

Response:

Revised Chapter 7 discusses this subject in 7.3.2.1.2.14, which states:

> "Access to means of bypassing any safety action or function for the ESF systems is under the administrative control of the control room operator. The operator is alerted to bypasses as described in Section 7.1.2.4, Regulatory Guide 1.47.
>
> Control switches which allow safety system bypasses are keylocked. All keylock switches in the control room are designed such that their key can only be removed when the switch is in the "accident" or "safe" position. All keys will normally be removed from their respective switches during operation and maintained under the control of the shift supervisor. Further, the key locker will be audited once per day by the shift supervisor. Should a key be required to change a valve position, it will be obtained from the shift supervisor via approved key control procedures."

Q. 031.103
(7.3)
(7.6)
(F7.6-5)

Your description of the area temperature monitoring system in Sections 7.3 and 7.6 of the FSAR, is insufficient. Accordingly, provide the following additional information:

a. Identify the interfaces between the Class 1E and non-Class 1E parts of this system.

b. Describe how redundant components are electrically isolated and physically separated.

c. Describe how the electrical isolation devices were qualified and indicate the range of this qualification in terms of voltages and currents.

d. Provide the schematics for the Class 1E portions of this system, including the isolation devices.

e. Provide the bases and methods which were used to select the samples which were tested in accordance with the criteria identified in Item (c) above.

Response:

a. Interfaces of the area temperature monitoring system are shown on Elementary Diagram 807E154TC Revision 9, Leak Detection System, which were provided to you per Chapter 1.7 of the FSAR. The local temperature elements and control room temperature trip switches are Class 1E. The output of each temperature trip switch interfaces to a common non-1E meter module and meter and to the non-1E annunciator and computer circuitry. There is one set (meter module and meter) per control room panel, consequently one set per division of instrumentation. The separation interface with the annunciators and computer are protected by relays (contact-to-contact) or temperature switches (contact-to-contact).

b. Power sources, sensors, and wiring for essential circuits are physically separated and electrically isolated, as described in FSAR 8.3.1.4.

Even though separation of Class 1E and non-Class 1E circuits was not a design requirement for WNP-2, electrical isolation of power sources between essential and non-essential equipment is provided by relays (coil-to-contact). Refer to elementary diagram 807E154TC, Revision 9.

Redundant control room components are physically separated, where possible, by placing them in separate control room logic cabinets. Where it is not possible to place redundant control room components in separate cabinets, separation is achieved by surrounding redundant wiring and equipment in metal encasements or providing 6" physical separation as far as practicable.

b. Sensor devices are separated physically such that no single failure (open, closure, or short) can prevent the safety action. By the use of conduit and separated cable trays, the same criterion is met from the sensors to the logic cabinets in the control room. The logic cabinets are so arranged that redundant equipment and wiring are not present in the same bay of a cabinet (a bay is defined by adequate fire barriers). Redundant equipment and wiring may be present in control room bench boards, where separation is achieved by surrounding redun dant wire and equipment in metal encasements, or by 6" physical separation as far as practicable. From the logic cabinets to the isolation valves, separated cable trays or conduit are employed to complete adherence to the single-failure criterion.

c. No electrical isolation problem exists between the redundant components of the temperature monitoring system because of physical and electrical separation of the redundant channels.

The trip relays providing electrical isolation between essential and nonessential equipment were not originally required to be qualified as Class 1E isolation devices on the WNP-2 design.

However, the relays used in the WNP-2 design are qualified as Class 1E. The qualification was completed to IEEE 323-1971 and IEEE 344-1971 using GE Qualifications Specifications. The contact-to-contact rating of the Agastat GPI is as follows: 1000 MEGOHMS @ 500-Vdc insulation resistance and 1200 VRMS @ 60 hertz dielectric.

The rating for the GE type HMA relay was tested to ANSI 37.9 and has a contact-to-contact dielectric strength of 1500 VRMS @ 60 hertz.

The rating for the relay used in the temperature point modules is hermetically sealed with a dielectric strength of 500 VRMS between open contacts; 500 VRMS between coil and case; and 1000 VRMS between all other combinations.

The interface between the Class 1E temperature trip switch and the non-1E meter module takes place within the Class 1E temperature switch. An individual momentary toggle switch is contained in each temperature trip switch. Manual initiation of the toggle switch is required before any temperature trip switch electronics could interface with a signal going to the meter module. This manual toggle switch would be initiated during testing, calibration, or surveillance monitoring, and then only one switch at a time would be interfaced to the meter module. No specific qualification test has been conducted to classify this device as an isolation element, and it is not felt necessary due to the nature of the application. Switch vendor information identifies this switch as having a 1000-volt (RMS) dielectric strength, which is well above any credible power source available to this system.

d. Refer to the response to part "a" of this question and see attached Figures 1 and 2.

e. The bases and methods used to select the samples which were tested as noted in "c" above are specified in 225A6634, Qualification Specification for Essential Components, which can be made available if desired.

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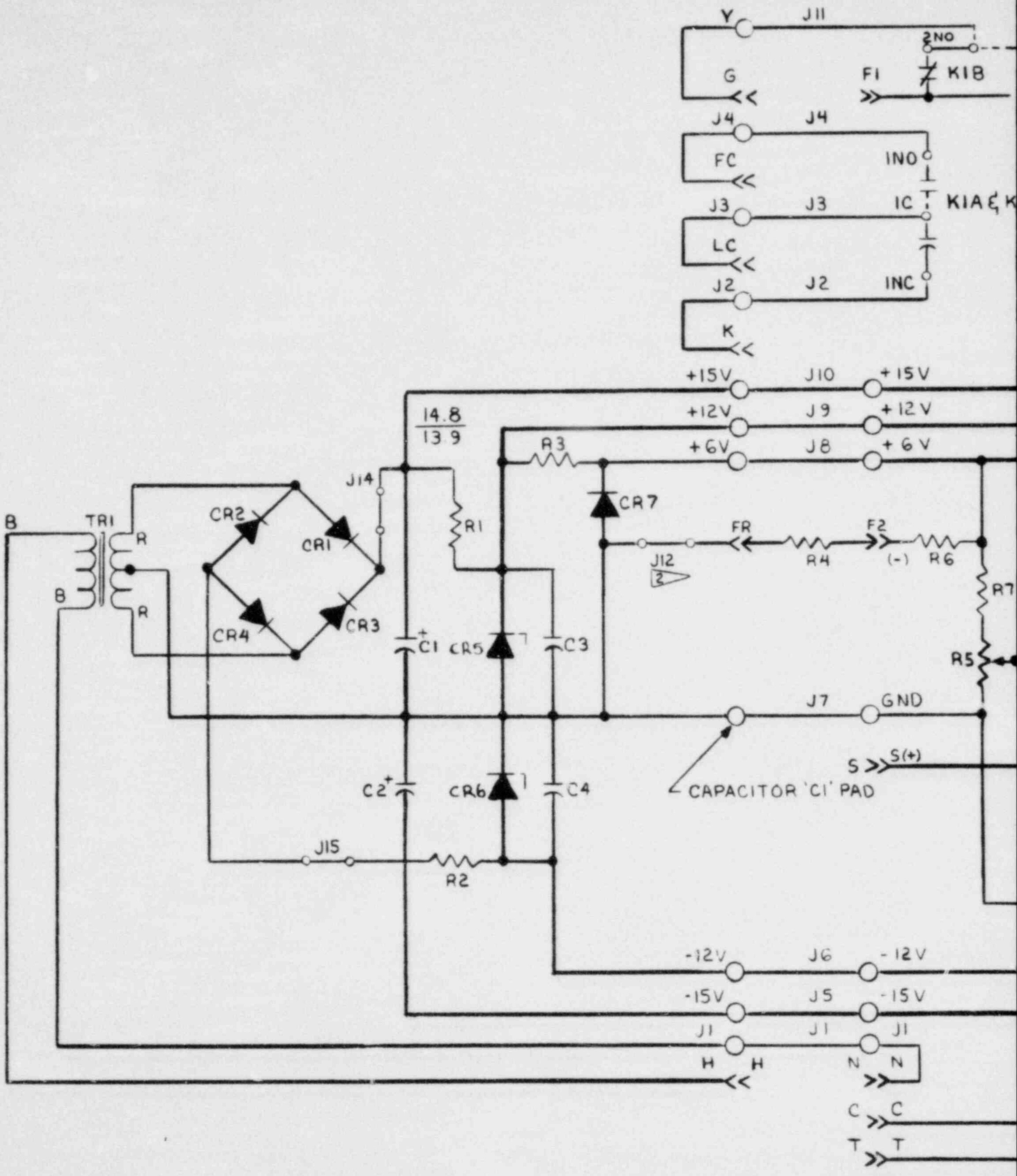
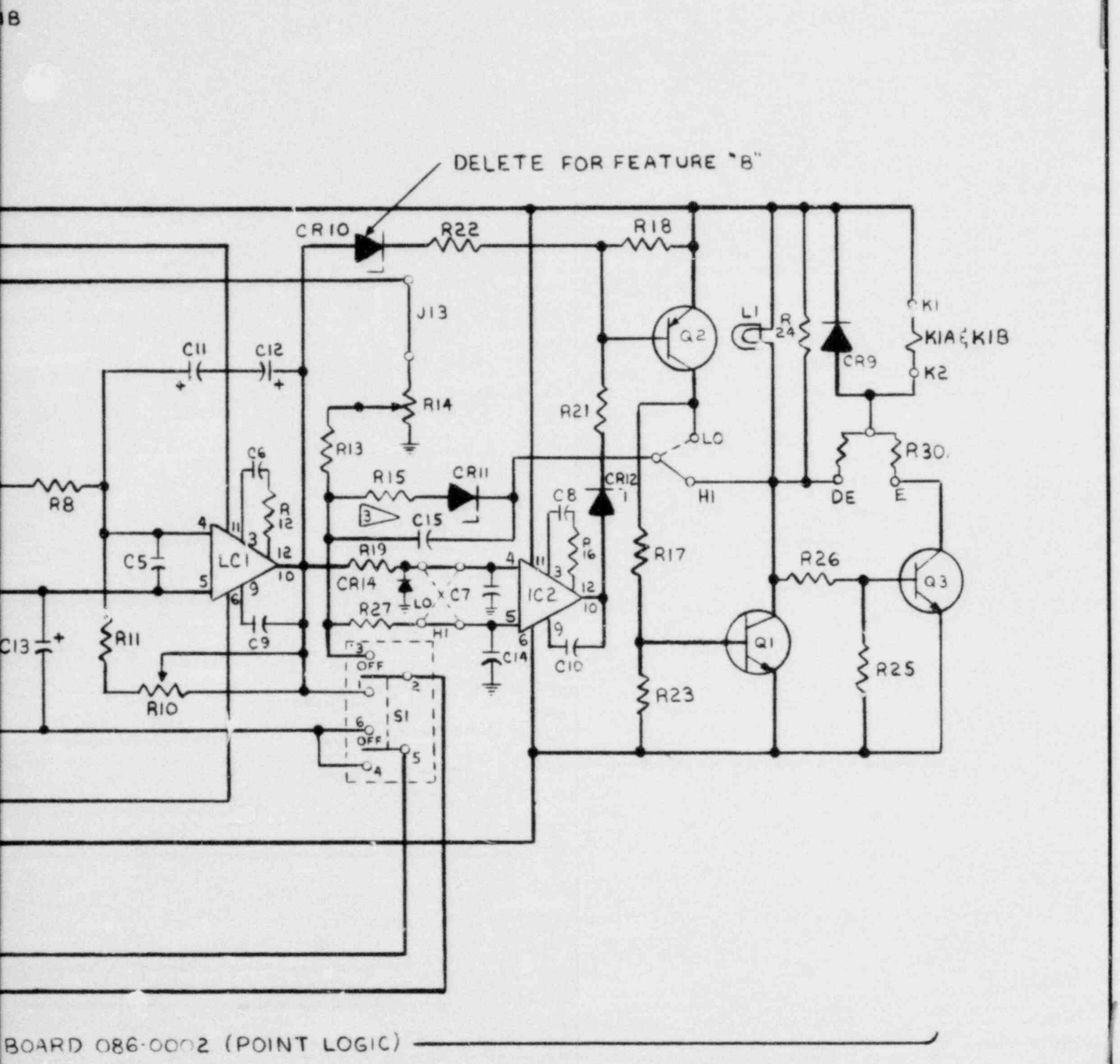


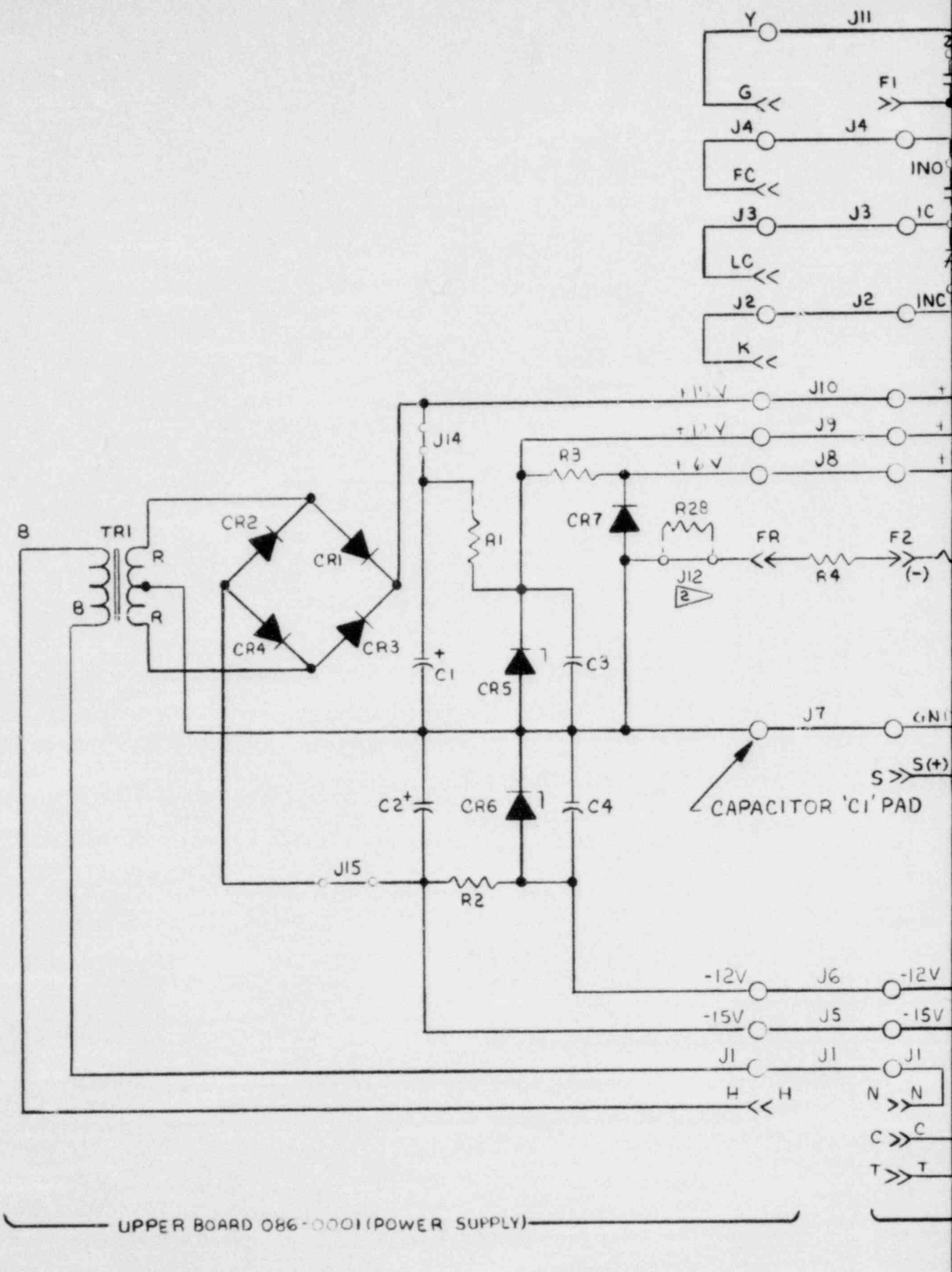
UPPER BOARD C86-0001 (POWER SUPPLY)

LOWER

NOTES
1. ADDITIONAL "FORM C" CONTACT FOUND ON OPTIONAL K1B RELAY
2. FOR "ISA DESIGNATIONS" R OR S, J12 BECOMES R29
3. ADD FOR FEATURE "G"

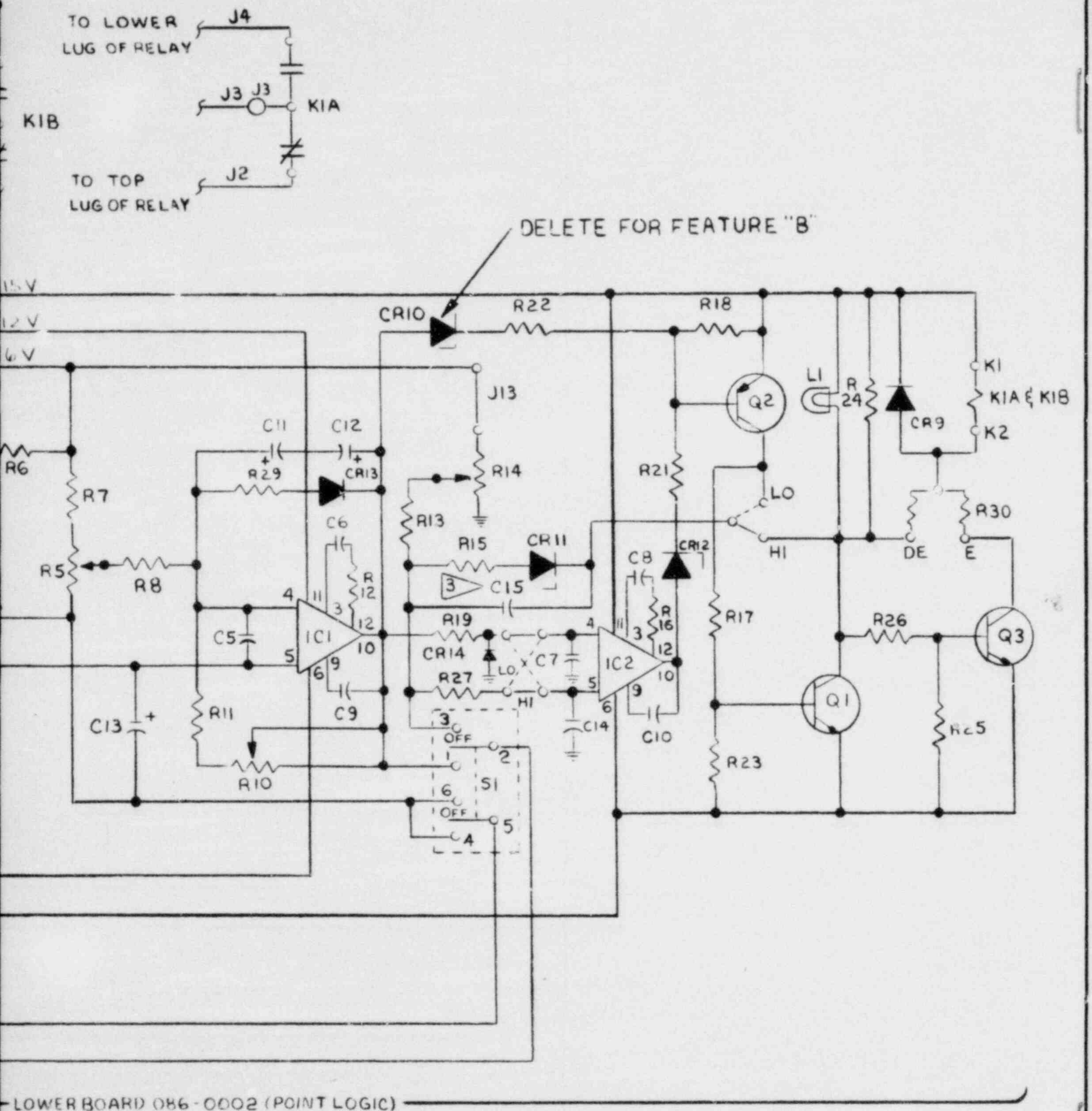


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	AREA TEMPERATURE MONITORING DEVICE SCHEMATIC	FIGURE 31. 103-1



UPPER BOARD 086-0001(POWER SUPPLY)

NOTES
1. ADDITIONAL "FORM C" CONTACT FOUND ON OPTIONAL KIB RELAY
2. FOR "ISA DESIGNATIONS" R OR S J12 BECOMES R29
3. ADD FOR FEATURE 'G'



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	AREA TEMPERATURE MONITORING DEVICE SCHEMATIC	FIGURE 31. 103-2

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Q. 031.104
(7.5)
(031.034)

The description of the control room in Section 7.5 of the FSAR is incomplete as are the figures in this section. (Refer to Item 031.034.) Accordingly, provide a layout drawing of the control room showing in sufficient detail, the following information:

a. The location and identification of each cabinet and panel.

b. The location and routing of each conduit and cable tray and pan.

c. The location and field of each emergency light.

d. The location and identification of each indicator and relay contact which satisfies the requirements of Sections 4.19 and 4.20 of IEEE Standard 279-1971.

Response:

a. The following WNP-2 drawings provide main control room panel and cabinet location and master parts list (MPL) identification numbers: E775 (Figure 6.4-1) and 127D1625TC.

   Panel MPL idents may be associated to a particular system identification through drawing 238XL93AD.

b. Location and routing of conduits within the main control room are shown on the following WNP-2 drawings: E751, E765, and E766.

   Location of PGCC floor ducts within the main control room is shown on WNP-2 drawing 127D1625TC (see Item a.).

c. The location and field of the WNP-2 main control room emergency lighting are shown on drawing E733.

d. All indications necessary for the plant operator to assess status of protective actions and status of protective functions are located in system level groupings, along with manual system controls, on front row main bench boards. These front row boards are identified on WNP-2 drawings E775 and P601, P602, P603, P800, P820, and P840.

Relays or analog outputs used to actuate indications of protective actions and status of protective functions are located within control room back row cabinets. These cabinets were identified by part a. above.

Copies of the above referenced drawings have been submitted to you under separate cover. The drawings are for information only as they are under continous review and update.

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Q. 031.105
(7.4.1.4)

Your description of the procedure for reactor shutdown from outside the control room is inadequate. Accordingly, provide the following additional information:

a. Provide plant layout sketches which show where the switches are located.

b. Describe the method which will be used to seal the transfer switches.

c. Describe the consequences of an inadvertent actuation of one or more of the switches.

d. Identify and justify each transfer switch which is not wired to the bypass and inoperable status indication system.

e. Describe the methods and indications available outside of the control room by which the operator can: (1) verify relief valve operation; (2) determine the reactor pressure, the coolant level and the coolant temperature; (3) determine the suppression pool level and the temperature of the water in the pool; (4) determine the containment pressure; and (5) determine the service water flow rate and the change in the coolant temperature through the RHR heat exchangers.

Response:

a. Please see revised 7.4.1.4.

   1. Radwaste and Control Building Control Room and Remote Shutdown Room Arrangement Sheet - 1 B&R Drawing No. E775, FSAR Figure Number 7.5-3

   2. BOP Remote Shutdown Board B&R Drawing No. M618 Sheets RS1 and RS2.

   3. Remote Shutdown Vertical Board and Legends GE Drawing No. 828E466TC and 163C1327TC Sheets 1 and 2.

b. The transfer switches will not be "sealed" as such. However, the Remote Shutdown Room will be locked and access administratively controlled.

In addition, movement of the transfer from the normal to the "transfer" position will result in an annunciator actuation in the Main Control Room as described in paragraph d below.

c. Inadvertent actuation of one or more transfer switches will result in loss of manual and automatic control, of associated equipment, from the Main Control Room. However, as described in paragraphs b above and d below, access to the transfer switches is administratively controlled and actuation of any transfer switch will result in annunciation in the Main Control Room.

d. Actuation of any Remote Shutdown Transfer Switch which controls safety-related equipment will result in actuation of the associated system level bypass and inoperability indication.

e. Methods and/or indications available to a remote shutdown operator are described in 7.4.1.4.

   1. Verification of relief valve operation by observation of the reactor pressure indicator;

   2. Determination of reactor pressure as noted in 1 above; determination of coolant level by the level indicators; and determination of coolant temperature by reference to saturated steam table curves;

   3. Determination of suppression pool level and water temperature by associated level and temperature indicators;

   4. Determination of containment pressure by associated pressure indicators;

   5. Determination of the RHR heat exchanger service water flow rate by associated flow indicators. Reactor pressure vessel parameters can be used to determine the operability of the RHR Heat Exchanger.

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Q. 031.106
(7.6.1)

Confirm in 7.6.1.13 of the FSAR that the primary containment atmosphere monitoring system, including its associated sensors, will be seismically and environmentally qualified. Identify and justify all exceptions. Describe your methods to seismically qualify the drywell hydrogen and oxygen monitoring system. Indicate the required response spectra for which the system is qualified and identify the limiting component.

Response:

The response to this question will be provided upon completion of the seismic and environmental re-evaluation. See response to Question 031.006.

Due to a complete rewrite of Chapter 7 in Amendment 10, the information in 7.6.1.13 has been moved to 7.5.1.5.

Q. 031.107
(7.6.2)

In 7.6.2.13.3.5.19 of the FSAR, you indicate that the post-LOCA containment monitors are in continuous operation. This appears to be a change from previous boiling water reactor (BWR) designs in which the hydrogen and oxygen subsystems were activated by an accident signal. Discuss this new feature of your design.

Response:

WPPSS recognizes that the hydrogen and oxygen analyzers are only required to perform their function after a LOCA. However, WPPSS has chosen to have this system on line during normal plant operation thereby assuring availability after a LOCA. The discussion of hydrogen and oxygen analyzers has been moved from 7.6 to 7.5.

Q. 031.108
(3.11)
(4.4.3)
(5.4.1)

Figures 5.4-2, 7.7-7, and 7.7-8 and Sections 3.11, 4.4.3.3, 5.4.1, 7.6.1.8, and 7.6.2.8 of the FSAR contain many discrepancies and are, therefore, unacceptable. Provide a consistent set of drawings and other information which represents the design of the reactor recirculation system for the WNP-2 facility. In your response to this item:

a. Provide setpoint information (i.e., the range of the instrument, its accuracy and its setpoint) for the following items: (1) the low total feedwater permissive (H13-P634); (2) the steam line recirculation pump differential temperature (K634); (3) C001A rated speed permissive for CB3A Trip 2; (4) a pump speed greater than 15 percent but less than 40 percent; (5) C001A less than rated voltage permissive for closing CB2A; (6) the generator protective trip voltage; and (7) the reactor power permissive for low speed start.

b. Identify which control option of note 7 in Figure 7.7-8 of the FSAR is applicable to the WNP-2 facility.

c. Identify the inputs which are received from Reference Document No. 4 of Figure 7.7-7a of the FSAR (i.e., C12-1050). Provide this reference and clarify the function of these trips and the ATWS trips shown on Sheet f of Figure 7.7-7.

d. Clarify the discrepancy between the ATWS trips shown in Figure 7.7-7 of the FSAR and the logic description given in Section 7.6.1.8.1.

e. Indicate the signal source for the "permissive when low speed auto start sequence is not activated". (Refer to Figure 7.7-7 of the FSAR.)

f. Indicate the signal source for the "transfer to high speed initiated" auxiliary device. (Refer to Figure 7.7-7 of the FSAR.)

g. Describe the initiating circuitry for, and the location of, the hydraulic line containment isolation valves.

h. Clarify the discrepancy between the setpoint stated in Sections 4.4.3.3.3.a and 5.4.1.3 of the FSAR.

i. Provide justification for not environmentally qualifying the 6.9 Kv switchgear.

Response:

A response to items a through f, and item h will be provided in a future amendment.

g. For the initiating circuitry and the location of the hydraulic line containment isolation valves please see Table 6.2-16.

i. The 6.9 Kv non-Class 1E switchgear supplies the non-Class 1E reactor recirculation pumps, cooling tower substations and auxiliary substations and, therefore, is not required to be environmentally qualified. (Reference FSAR 8.3.1.1.1.)

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Q. 031.109
(7.7.1)

Provide in Section 7.7.1.2 of the FSAR, the results of a failure mode and effects analysis for the reactor manual control system analyzer. Identify the design features which are provided to detect these failures. Describe the test procedures, including the test frequency, which will be used to detect these failures.

Response:

The limiting failure mode of the reactor manual control system (RMCS) leads to a continuous rod withdrawal. The rod worth minimizer (RWM) and rod sequence control system (RSCS) provide rod block inputs to the RMCS to mitigate this event in the start-up range (low power). However, as described in the response to 031.098, an analysis provided in NEDO 23842 shows that the neutron monitoring system scram (either IRM or APRM), which is independent of the RMCS, adequately terminates the event assuming RWM and RSCS failure to block rod withdrawal. In the power range, the rod block monitor (RBM) provides the block to the RMCS (see the response to 031.052).

The operation and design features of the RMCS are described in FSAR Chapter 7, 7.7.1.2.b.1. Briefly, the system is designed to continously monitor both the plant status for the presence of rod motion inhibits (rod blocks) and the operability of its various functions. In the event of any system failure, the default condition is a rod motion inhibit. The cause of any failure must be corrected before rod motion can proceed.

The ability of the control rod drive portion of the RMCS (Control Rod Drive Control System - CRDCS) to apply rod motion inhibits is demonstrated in conjunction with normal plant instrumentation surveillance on the neutron monitoring system (NMS). When the various NMS functions are tested they provide rod block inputs to the CRDCS. The generation of rod blocks indirectly confirms the operability of the CRDCS. The surveillance requirements for the NMS are described in Chapter 16, "Technical Specifications".

Q. 031.110
(7.7.1)
(15)

The information presented in Sections 7.7.1.5 and 15 of the FSAR is incomplete with regard to load following operations. Accordingly, describe the interfaces between the system dispatcher and the WNP-2 control systems (e.g., the turbine-generator and the recirculation flow control systems).

Response:

See revised 7.7.1.5.b.4.a).

Q. 031.111
(9.5.2)

The description of the intra-plant radio system inspection and testing in Section 9.5.2.2 of the FSAR is inadequate. Describe the preoperational and periodic testing which assures that radio transmissions will not cause spurious operation of relays and, as a result, negate the protective function of Class 1E equipment. (This question is similar to Item 031.123 on the LaSalle docket.)

Response:

During the preoperational test program any systems or components that are affected by radio transmissions during the normal course of testing and operations will be identified on a Startup Problem Report. The Test and Startup personnel and Operating staff will be briefed on the possible interaction between radio transmission and electronic gear, including solid state relays, and each preoperational test will include a precaution statement in 3.0 of the test procedure to alert personnel to possible interaction with protective relays. This will ensure that the appropriate action will be taken to assure that the protective function of Class 1E equipment is not negated or degraded during plant operation. The preoperational testing is described in FSAR 14.2.12.1. Security-related radio communications equipment will be surveillance tested periodically as required by Site Physical Security Plan.

Q. 031.112
(7.6.1)
(7.6.2)

Describe in Sections 7.6.1.1 and 7.6.2.1 of the FSAR, how the power cables and the refueling interlock circuits are separated on the refueling crane.

Response:

The refueling interlocks are not safety-related and no separation is required or provided between the power cables and the interlock circuits. See revised Chapter 7, 7.7.1.13.

Q. 031.113
RSP
(15.4.1.2)

It is our position that the rod sequence control system does not satisfy the requirements of IEEE Standard 279-1971 and, therefore, is unacceptable for the prevention of a control rod withdrawal accident. Accordingly, we require you to provide a modified design for the WNP-2 facility.

Response:

See response to Question 031.098.

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the nature of an indication discovered using the ultrasonic technique when the indication is suspected to result from service induced cracks on the nozzle inner surfaces. In the event an indication is discovered and found to result from service induced cracks propagating from the nozzle inner surfaces, the following actions will be taken:

a. All remaining feedwater nozzles will be examined using both ultrasonic (from the OD) and penetrant techiques during the refueling outage in which the cracking is verified.

b. All surface indications determined to be service induced cracks will be removed by local grinding.

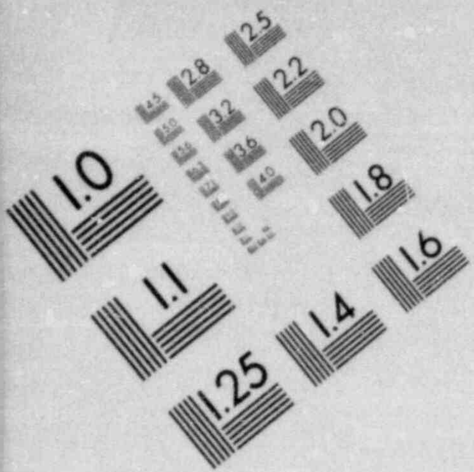
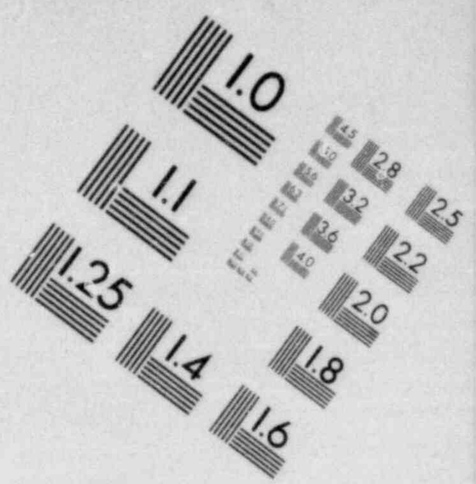
c. An inspection method, such as a leak test, will be used to determine the integrity of each of the RFW thermal sleeve to safe end joints.

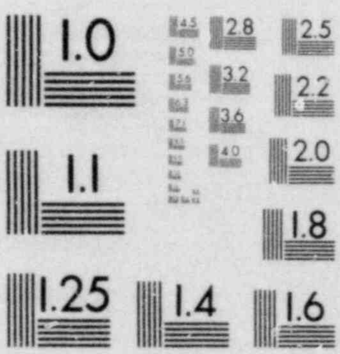
d. Appropriate corrective action will be taken as required and as practical to prevent recurrence of crack initiation. A program and schedule for implementing such corrective action will be prepared and submitted to the Commission prior to its implementation.

e. A RFW nozzle examination program for subsequent refueling outages will be modified to include an external ultrasonic examination of all feedwater nozzle inner radii, bore and safe end regions for each scheduled refueling outage for 3 consecutive outages. If no new indications are discovered, or if new indications are determined to not result from service induced cracks at the nozzle inner surfaces, the original Supply System program will be resumed. If after 3 additional outages no new indications resulting from surface induced cracks are detected, subsequent examinations will be performed in accordance with normal ASME Section XI requirements.

f. The conduct of surface examinations of accessible nozzle inner radius surfaces will continue to be used throughout plant life only to confirm or characterize new ultrasonic indications which are suspected to result from service induced cracks at the nozzle inner surfaces.

IMAGE EVALUATION TEST TARGET (MT-3)



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MICROCOPY RESOLUTION TEST CHART

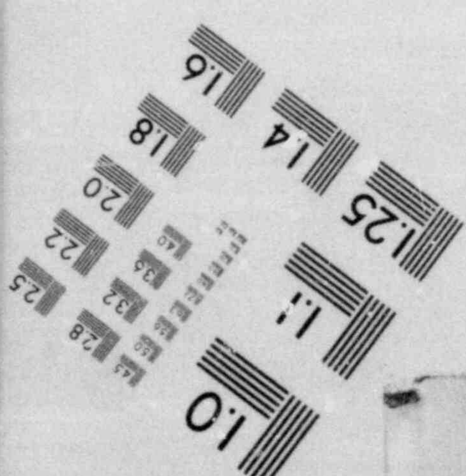
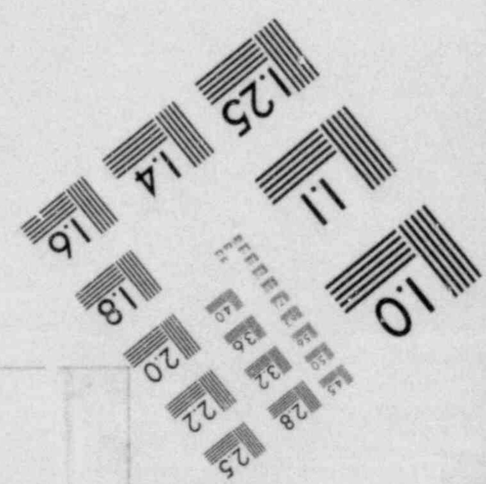
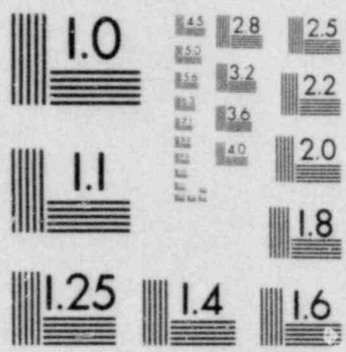



IMAGE EVALUATION
TEST TARGET (MT-3)



6″

MICROCOPY RESOLUTION TEST CHART

B.2 As stated in B.1 above, the Supply System will perform a surface (penetrant) examination of accessible inner surfaces on all RFW nozzles during the preservice examination program. Subsequent surface examinations of those surfaces will be performed only to verify the nature of an indication discovered using the ultrasonic technique when the ultrasonic indication provides evidence of previously unidentified service induced cracks.

B.3 See response to B.2 above.

If after the sixth planned refueling outage following commercial operation no indications resulting from service induced cracks are found, the subsequent inservice examinations will be performed in accordance with the normal ASME Section XI requirements. Any indications resulting from service induce cracks which are subsequently found will result in the corrective action described above.

C. Thermal Sleeve to Safe End Joint

As stated in B.1 above, the Supply System will perform an inspection of the thermal-sleeve-to-safe-end weld joint, such as a leak test, only if service induced cracks or some other anomoly is discovered which would bring the integrity of the joint into question. In that case, the feedwater piping will be filled with water and the area of the thermal-sleeve-to-safe-end joint will be inspected for indications of leakage.

II. ACCEPTANCE CRITERIA

A. The Supply System will comply with this criteria as stated in B.1 above.

B. The supply System will comply with this criteria as stated in B.1 above.

C. The Supply System will comply with option (b), in that both the examination personnel and the procedures to be used on the nozzles will be qualified on a full size nozzle mock-up. Supply System examiners will be trained by individual NDE specialists having previous experience with the General Electric Company procedures and their nozzle test program. These examiners will undergo further training, practice, and qualifications on a full size nozzle mock-up. The mock-up will be unclad if negotiations can be reached with a utility owning such a mock-up. As an alternative, the examiners will qualify on a clad mock-up owned by the General Electric Company. Following the qualification process, the examinations will be conducted under the direct supervision of the experienced NDE specialists responsible for ultrasonic technique and procedure development for the Supply System.

III. RECORDING AND REPORTING STANDARDS

The Supply System will record crack indications and report inspection results in compliance with the requirements stated in NUREG-0312.

121.8 JUSTIFICATION OF DEVIATION FROM APPENDIX A

I.B.1 Ultrasonic Examinations Frequency

The Supply System will examine only one RFW nozzle per refueling outage rather than all nozzles using an ultrasonic technique from the outside of the vessel. This is justified for the following reasons, which reflects a significant advance in the WNP-2 design and operating procedures towards the long term solution of the BWR nozzle cracking problems per NUREG-0312, Section 8.0, Part. 1.

a. Improved Design: The WNP-2 RFW welded thermal-sleeve-to-safe-end joint provides a "zero leakage" design. This design essentially eliminates the primary historical initiating source of nozzle cracking in BWRs.

b. No Nozzle Cladding: The WNP-2 RFW nozzle surfaces are not clad. The likelihood of crack initiation in unclad nozzles is more than a factor of a 5 less than for clad nozzles. All cracks in BWR feedwater nozzles have initiated in the clad metal.

c. Proven Examination Technique: The ultrasonic examination equipment and personnel to be used in performing both baseline and inservice ultrasonic examinations will be qualified on a full scale mock-up of the nozzle, simulating the nozzle geometry and anticipated fatigue crack defects. Since the WNP-2 reactor feedwater nozzles are unclad as stated in b) above, a more sensitive examination is possible due to lack of clad/basemetal interface.

d. Augmented Examination Frequency: The above stated program provides RFW nozzle examination coverage at nearly twice the frequency of the ASME Section XI requirements, i.e., all RFW nozzles will be examined within 6 years (approximately) rather than within 10 years.

e. Feedwater Temperature Controls: As previously stated, WNP-2 has incorporated a feedwater, low flow control valve. The advantages gained from low flow control are identified in Section 4.7 of NEDE-21821.

f. Projected Crack Growth Rates: As presented in Section 4.7 of NEDE-21821, the WNP-2 design should have greater than 35 years of operation, considering our low flow control, prior to an initiated crack reaching 1 inch in depth. This provides for a minimum of 4 examinations per nozzle before reaching a point requiring repair. Even if the extremely conservative (factor of 5) upper bound crack growth curve is applied, each RFW nozzle would be examined by the Supply System program prior to a crack becoming 1 inch in depth. It is clear that ample conservatism exists in the Supply System examination frequency of the RFW nozzles.

The above factors, when combined, provide a great deal of assurance that the factors which have led historically to BWR RFW nozzle cracking have been virtually eliminated. Furthermore, any cracking which might occur from unanticipated sources will be discovered before propogating to a significant depth due to low flow controls and an augmented examination schedule with state-of-the-art qualified ultrasonic examination techniques.

I.B.2&3 Surface Examinations

The Supply System will perform surface (penetrant) examinations of the accessible internal surfaces of RFW nozzles during the preservice inspection program. Inservice surface examinations will be performed only when indications of service induced cracking are found using the ultrasonic examination techniques. This is justified as follows:

a. Reduced probability of crack initiation and growth as stated in the justification under I.B.1a) through f) above.

b. Access: In order to obtain access to perform a penetrant surface examination of the RFW nozzle surfaces during a refueling outage, the vessel water level would have to be lowered below the level of the spargers and hydrolaser decontamination performed. A special shielded work platform would have to be devised to minimize radiation exposure. This technique was performed at Vermont Yankee resulting in about 15 man rem.

I.C Leak Test

Thermal-Sleeve-to-Safe-End Joint: The Supply System will perform inservice inspection to determine the integrity of the thermal-sleeve-to-safe-end joint only when indications of service induced cracking are detected using the ultrasonic examination technique. The justification for this exception is similar to the justification for not performing inservice surface examinations cited above, with the following additional justification:

a. Test Effectiveness: The maximum pressure which could practically be placed on the subject weld joint would be that available from the static head of a filled sparger, or approximately 6" of water. The effectiveness of this test to reveal throughwall cracks in the weld joint is questionable, since the weld experiences significantly higher differential pressure and temperature during operation. Furthermore, this test would not provide evidence of other than gross throughwall cracks which, if and when detected via such a test, will in all likelihood have been resulting in some degree of leaking for a significant period of time. As was previously demonstrated, any cracks developing as a result of such leakage will be detected prior to the crack propagating to a depth which would jeopardize the nozzle integrity. There is, therefore, no appreciable benefit from performing the leakage test of the spargers other than to determine the status of their integrity in the event service induced cracks are confirmed. Since there is no appreciable benefit, and the cost in dollars and man rem exposure for such a test is quite high, the performance of this test on a routine basis is not justified.

BLANK

5. Reactor building floor drain sump level for leakage rates greater than 50 gpm.

6. Reactor building floor drain leakage rate alarm (5 gpm).

7. ECCS pump room flood level instrumentation (Class 1E), installed to detect passive failures in the ECCS post-LOCA (Reference response to FSAR Question 212.003).

Notwithstanding these alarms, however, only about 13,000 gallons of water will spill out of the break before the reactor water level drops from the normal shutdown cooling level to Level 3, automatically closing MOF008 and MOF009 and isolating the break. No single active failure can prevent isolation of the break. During the time of the pipe break, HPCS and RCIC will not automatically initiate, because their initiation signal (Level 2) is about 50 inches below Level 3. Also, RCIC will not automatically initiate because the reactor pressure (135 psig maximum during shutdown cooling) is too low to operate the RCIC turbine.

For the largest pipe break (1000 gpm), which can only occur in the RHR A or B pump rooms, the flood level resulting from 13,000 gallons will not affect operation of either RHR pump A or B. In addition, the flooding would only affect the room in which the pipe break occurred because of the watertight integrity of the RHR A and B pump rooms. Therefore, no operator action is required to protect these pumps.

Pipe breaks in the RHR shutdown cooling mode which can affect other ECCS systems (LPCS, HPCS or RHRC) via flooding through the floor drain piping in the upper portions of the reactor building will have flooding rates less than 1000 gpm because they will be higher in the building (less static head), have less driving head due to friction losses, and smaller crack sizes (smaller diameter pipe). These pipe breaks in the upper portion of the reactor building will be immediately detected by the high flow (5 gpm) alarms in the floor drain downcomer piping. Regardless of what the pipe break discharge rate is, the flood level resulting from 13,000 gallons is not capable of affecting the operation of the LPCS, HPCS or RHRC pumps, assuming all of the water is spilled into each pump room. Again, no operator action is required to protect these pumps.

It should be noted that the environmental effects (pressure, temperature and humidity) of pipe breaks during shutdown cooling are being addressed by ongoing pipe break and missile study.

d. Core cooling is not a concern. Sufficient ECCS equipment remains functional to automatically keep the core covered at all times. The cold shutdown procedure, i.e., containment heat removal, needs to be resumed. If the pipe break disables the common shutdown cooldown suction line, cold shutdown can be assumed by the alternate shutdown cooldown path discussed in 15.2.9. If the pipe break in the shutdown cooldown line is downstream of the F006 valve, normal shutdown cooldown can be resumed using the redundant RHR shutdown loop.

e. For application of single failure criteria, see "a" above.

Q. 360.3

Expand the discussion in 2.5.2.2.2, including supporting data, to provide more complete tectonic bases for the proposed boundaries of the tectonic provinces identified in Figure 2.5-59 and in the referenced section of the FSAR. In particular, indicate why you believe there is sufficient justification to establish a tectonic boundary between the northern Cascades and the Columbia Basin province.

RESPONSE:

The text of 2.5.2.2.2 beginning on page 2.5-118, has been revised to incorporate the response to this question.

Q. 360.004

In the Weston Geophysical Research, Inc., report, "Qualitative Aeromagnetic Evaluation of Structures in the Columbia Plateau and Adjacent Cascade Mountain Area," March 28, 1978, Figure 13 shows several north to northwest trending aeromagnetic linears in the vicinity of Badger Mountain and Jump Off Joe Anticline. However, the Weston report does not discuss the origin or interpretations of these particular linears. The north trending linear crossing the Columbia River at the junction with the Snake River has an pparent offset of the magnetic low defining the Rattlesnake Hills anomaly. Since these aeromagnetic linears trend toward the WNP-2 site, provide: (1) an interpretation of these features, including but not limited to the potential for their continuation of the north to the near site are; and (2) a discussion of the fault parameters, if such an interpretation is proposed.

Response:

The north and northwest-trending aeromagnetic linears referred to in Question 360.004 are part of those shown on Figure 13 of the Weston Geophysical Research Report entitled "Qualitative Aeromagnetic Evaluation of Structures in the Columbia Plateau and Adjacent Cascade Mountain Area" and on Figure 2R I-14, Amendment 23, WNP-1/4 PSAR. These linears are interpreted by Weston as being primarily a manifestation of topography and are not the result of any single throughgoing fault-related structure.

Figure 1 (attached) shows each of the three linears referred to in Question 360.004 plotted on a 1:250,000 scale topographic map along with an estimate of the degree of correlation between the magnetic linear and topography as proposed by Weston Geophysical Reseach, Inc. It is Weston's opinion that the correlation of these linears with topography appears to be best illustrated along the middle portion (1/3 to 1/2) of each linear. Weston's definition of the linear as well as its correlatable topographic cause appears to decrease northward and southward. It is WPPSS' position that these linears as defined are spurious.

Figure 13 of the report entitled "Qualitative Aeromagnetic Evaluation of Structures in the Columbia Plateau and Adjacent Cascade Mountain Area" and Figure 2R I-14 Amendment 23, WNP-1/4 PSAR, show Weston's identification of some of the magnetic anomalies and less obvious magnetic linears. It is Weston's position that these magnetic anomalies and magnetic linears may be the manifestation of, or could be interpreted as resulting from one or a combination of hidden geologic features such as folds, faults, dikes, lithologic variations or buried topography.

The origin of all magnetic anomalies and magnetic linears is the spatial distribution of materials with variable magnetic properties, either remnant magnetization or susceptibility. On Figure 13 and Figure 2R I-14 dark and light tones are used to indicate relatively continuous zones of high and low magnetic values. Generally, these anomalies are in an east-west direction and are indicative of the more obvious geologic and topographic structures such as Saddle Mountains and Rattlesnake Hills.

The specific linears referred to in Question 360.004 are shown on Figure 13 and 2R I-14 as dark continuous lines. These particular linears are not due to continuous alignments of high or low magnetic anomalies but instead represent localized disruptions of the normally smooth contours. Because of the large contrast in susceptibility between rocks of the Columbia Plateau and air (approximately 5000 x 10^{-6}, cgs emu), topographic variations can and do produce significant magnetic anomalies. Close examination of the magnetic profile data in addition to the topographic contour maps for the area, which includes the magnetic linears referred to in Question 360.004, reveals that these linears have a variable magnetic signature along their length. An index of the relative degree of correlation with topography as determined by Weston is also shown on attached Figure 1. The aeromagnetic profiles crossing these linear features are shown on attached Figures 2 through 7. These data definitely indicate that the linears are not the result of any throughgoing structure. The linears in question are also shown superimposed on the aeromagnetic contour map (Figure 8 attached). The degree of certainty used by Weston in constructing each segment of the linears is also indicated.

The "apparent offset" of the magnetic low associated with the Rattlesnake Hills-Wallula structure by the northerly-trending linear crossing the Columbia River at the junction with the Snake River is apparent and does not in our opinion represent any physical offset of the structure. The change in size of the magnetic anomaly at the junction of the Rattlesnake Hills and The Butte is due to a coincidental change in both topography and the distribution of magnetostratigraphic units exposed at this local (see Figure 2R 4.5-1, Amendment 23, WNP-1/4 PSAR). Although a fault has been identified at The Butte, its strike is parallel to the Rattlesnake Wallula trend (see response to Question 360.005) and is therefore compatible with the hypothesized aeromagnetic linear.

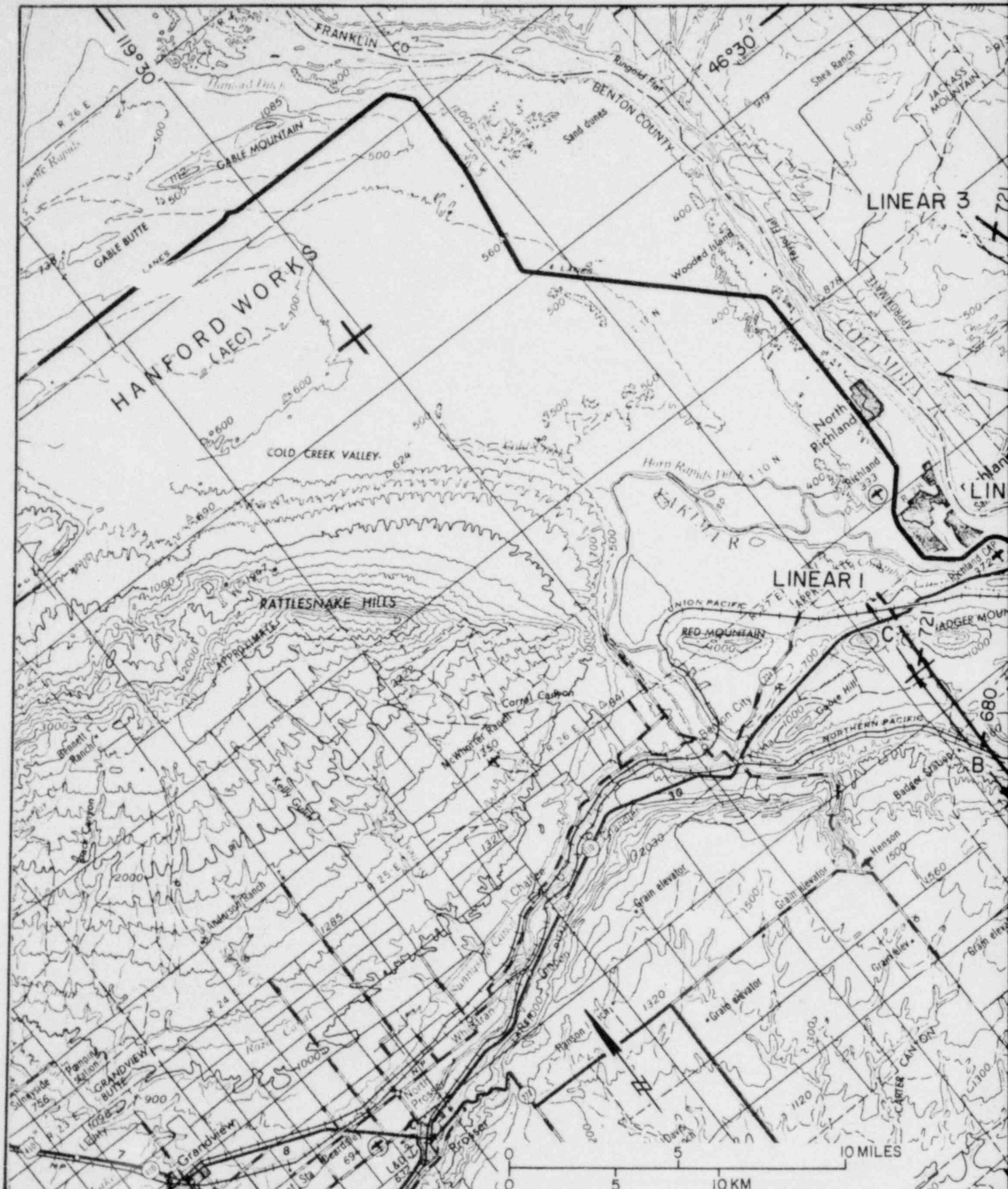
In addition to an examination of selected aeromagnetic profile segments crossing the three linears referred to in Question 360.004, four continuous aeromagnetic profiles crossing the linears were examined in more detail by two dimensional mathematical modeling utilizing specific magnetic susceptibility and natural remnant magnetism data, appropriate for the basalts of this area. Basics of the modeling technique can be found in M. Talwani and J. R. Heirtzler, "Computation of Magnetic Anomalies Caused by Two Dimensional Structures of Arbitrary Shape," Computers in the Mineral Industry, Part I, Vol IX, No. 1, Stanford University Publication, 1964.

The location of the four profiles relative to the three linears referred to in Question 360.004 is shown in attached Figure 8. The results of the modeling for profiles 360, 450, 510, and 570 are shown in attached Figures 9 through 12.

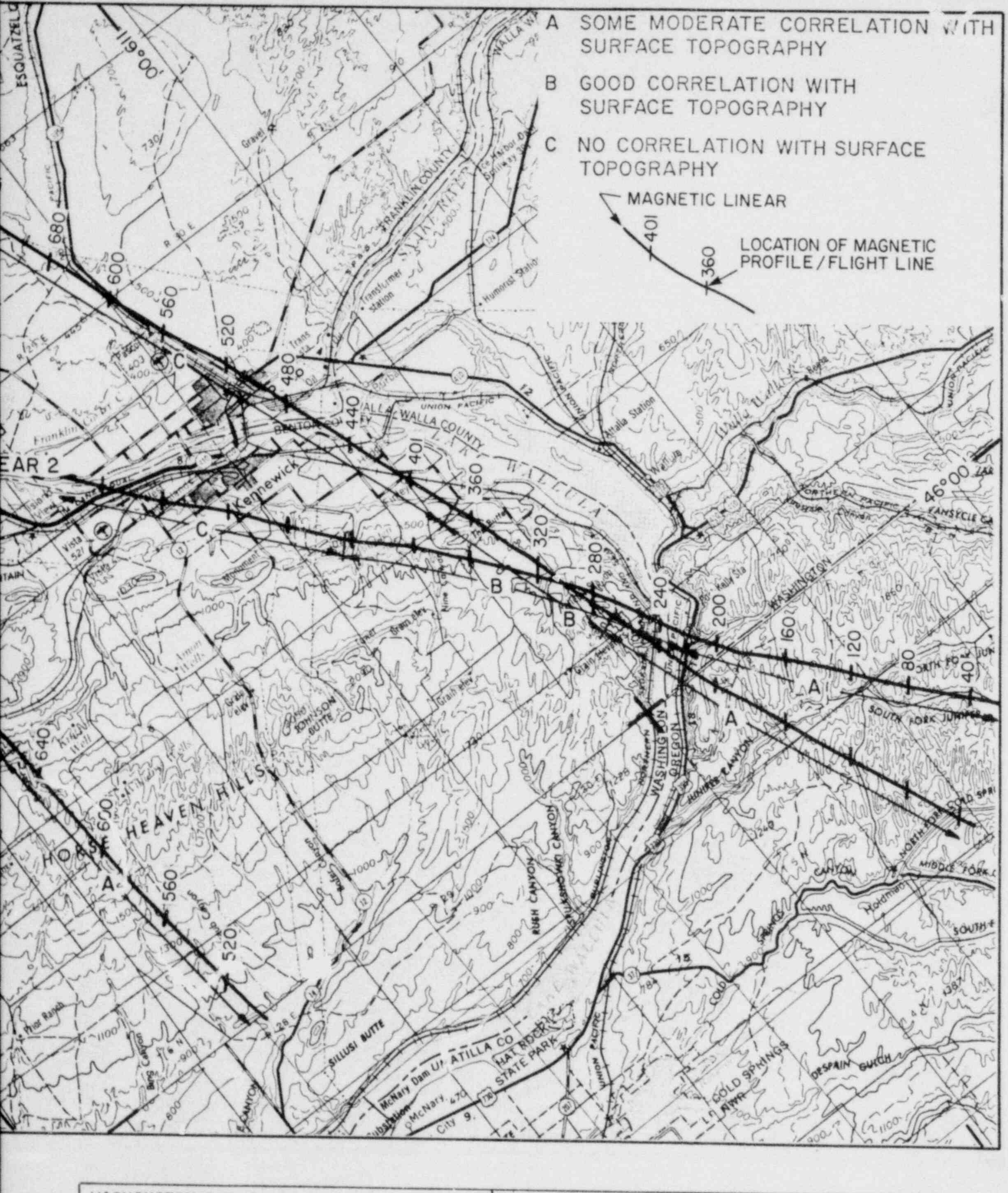
Profiles 370, 450 and 510 cross aeromagnetic linears 2 and 3, Question 360.004; the Kennewick-Cold Creek linear identified by C. E. Glass, page 2R K-10, Amendment 23, WNP-1/4 PSAR; the Rattlesnake-Wallula trend; and Zintel Canyon. No aeromagnetic anomaly can be observed on profile 370 or 450 in the vicinity of the intersect with the Kennewick-Cold Creek linear. On profile 510 an aeromagnetic anomaly does occur within approximately 5000' of the projected location of the Kennewick linear. Attempts to model this anomaly as a fault, assuming various offsets between 50' to 200', with a fault plane dipping between 45° NE and 70° SE, were unsuccessful. It is Weston's opinion that this anomaly probably represents an erosional feature in the basalt surface.

An aeromagnetic anomaly that is coincident with the Zintel Canyon fault, shown on Figure 2R H.5.1, Amendment 23, WNP-1/4 PSAR, can be observed on profiles 370, 450 and 510. This anomaly was successfully modeled as a fault, assuming 200' and 300' offsets and a vertical dipping fault plant.

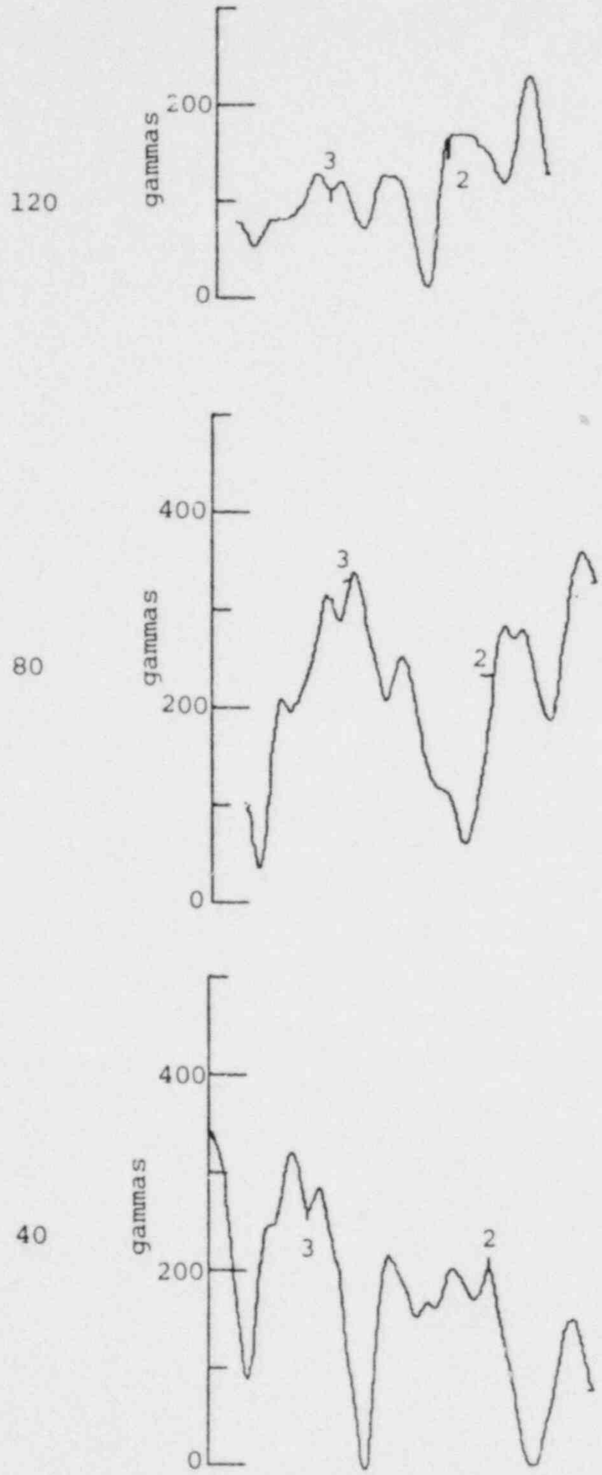
The Rattlesnake-Wallula trend of aeromagnetic anomalies was successfully modeled as a fault on all four profiles with 200' of offset assumed on profiles 370, 450 and 510, and 100' of offset assumed on profile 570. All profiles were modeled as a vertical fault with the northeast side down. Results of the modeling of profile 570, Figure 12 (attached) are incomplete at this time for the southwest portion where the profile intersects aeromagnetic linear 1, referred to in Question 360.004. The results of additional aeromagnetic modeling and a revised interpretation of that part of the aeromagnetic data for blocks A and D that includes the Hanford Reservation will be included in a revised version of Chapter 2.5 being prepared for submittal as part of the WNP-1/4 FSAR.



FRANKLIN CO
BENTON COUNTY
119°30'
46°30'
R 26 E
GABLE MOUNTAIN
GABLE BUTTE
Sand dunes
Shea Ranch
JACKASS MOUNTAIN
LINEAR 3
HANFORD WORKS
(AEC)
Wooded Island
COLUMBIA
North Richland
Richland
COLD CREEK VALLEY
RATTLESNAKE HILLS
APPROXIMATE
LINEAR I
UNION PACIFIC
RED MOUNTAIN
C
B
Corral Canyon
McWhorter Ranch
Benton City
NORTHERN PACIFIC
Badger Slough
Kelly Gulch
Anderson Ranch
Bennett Ranch
Rock Canyon
Grain elevator
Henson
Grain elev.
CARTER CANYON
GRANDVIEW BUTTE
Sunnyside
Grandview
North Prosser
Prosser
0
5
10 MILES
0
5
10 KM



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LOCATION MAP OF AEROMAGNETIC PROFILES ACROSS SELECTED AEROMAGNETIC LINEARS (WHICH ARE DUE TO TOPOGRAPHY)	FIGURE 1

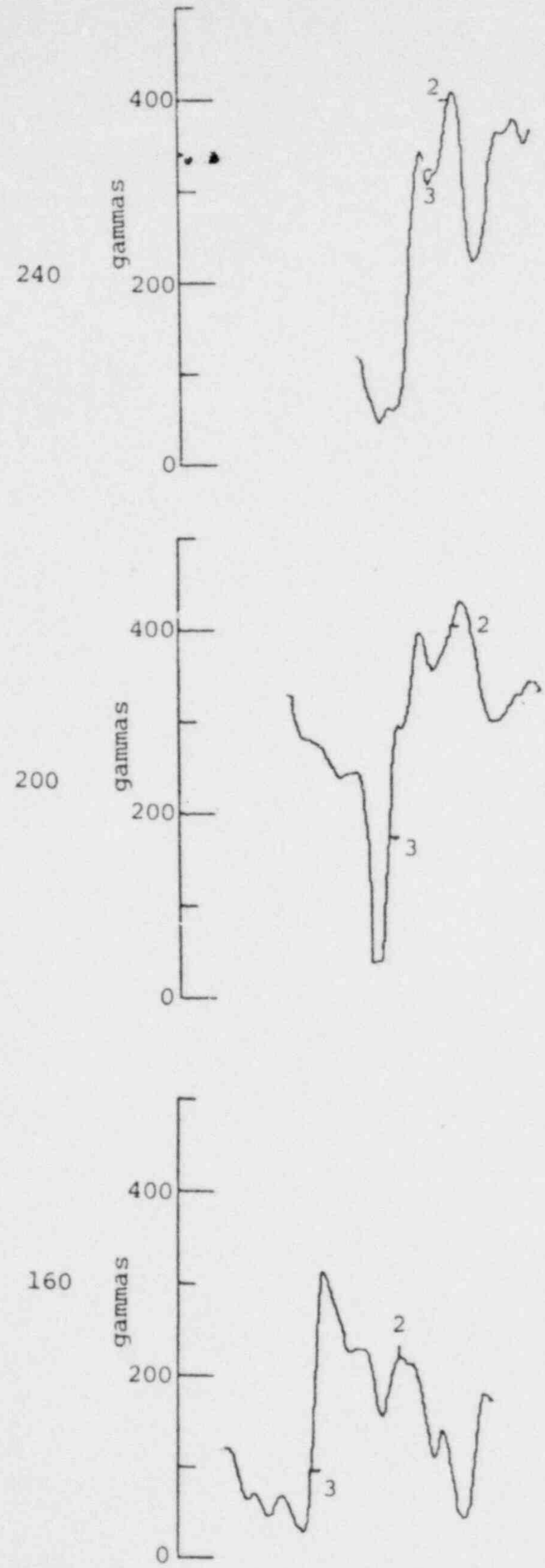


Each profile has
an arbitrary datum.

Horizontal Scale 1:250,000

Vertical Scale 1 inch = 200 gammas

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAGNETIC PROFILES 40, 80, 120	FIGURE 2

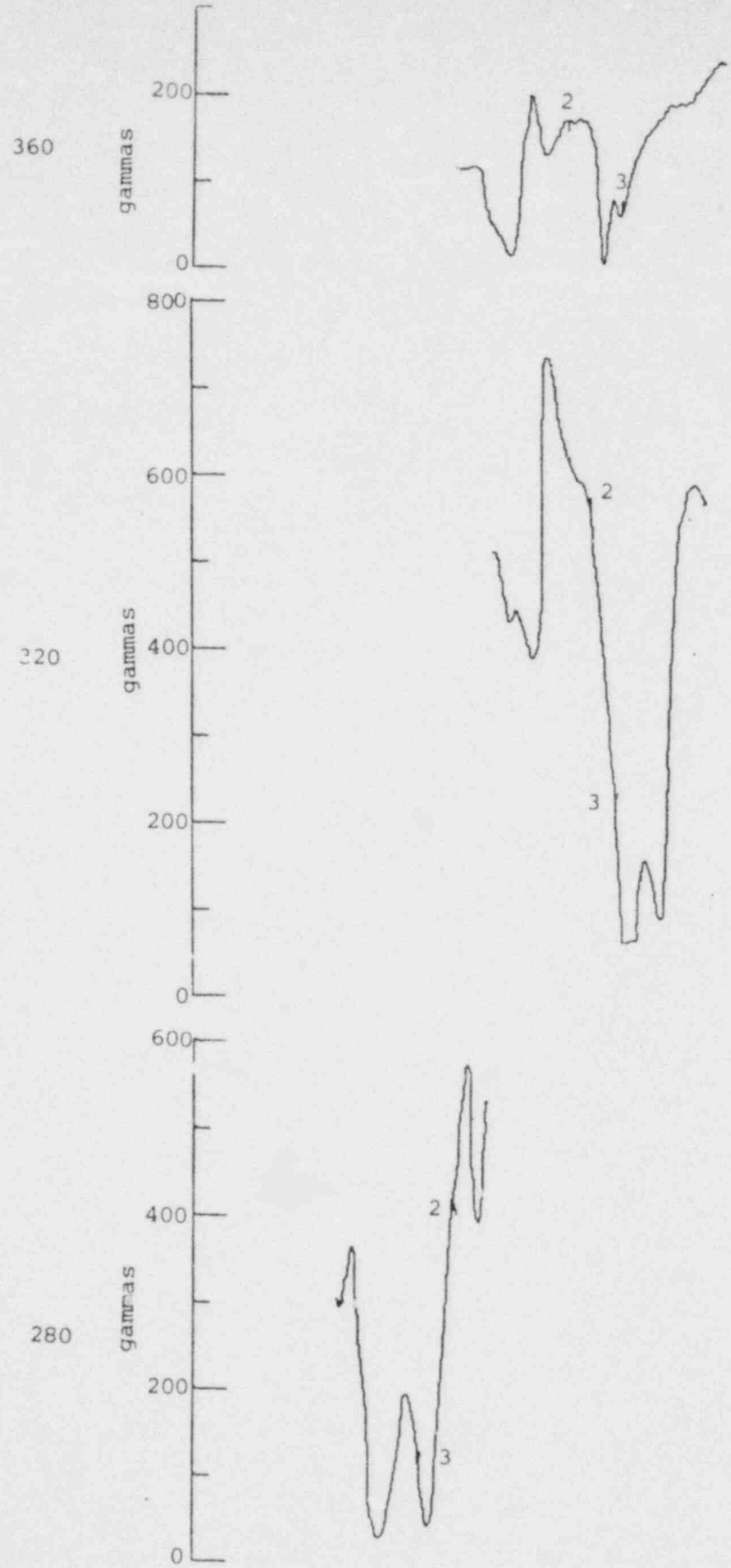


Each profile has
an arbitrary datum.

Horizontal Scale 1:250,000

Vertical Scale 1 inch = 200 gam

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAGNETIC PROFILES 160, 200, 240	FIGURE 3

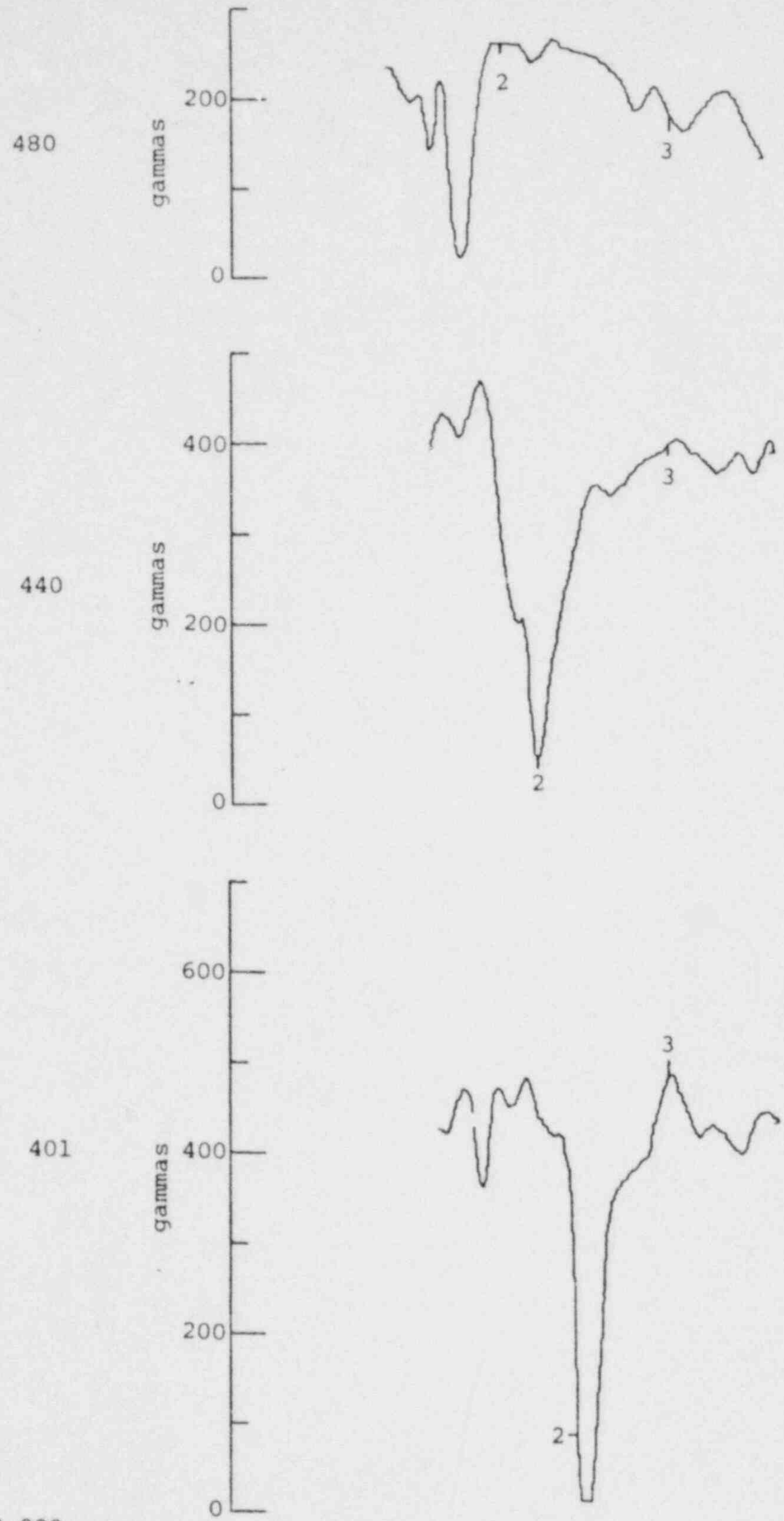


Each profile has
an arbitrary datum.

Horizontal Scale 1:250,000

Vertical Scale 1 inch = 200 gammas

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAGNETIC PROFILES 280, 320, 360	FIGURE 4



Each profile has
an arbitrary datum.

Horizontal Scale 1:250,000

Vertical Scale 1 inch = 200 gammas

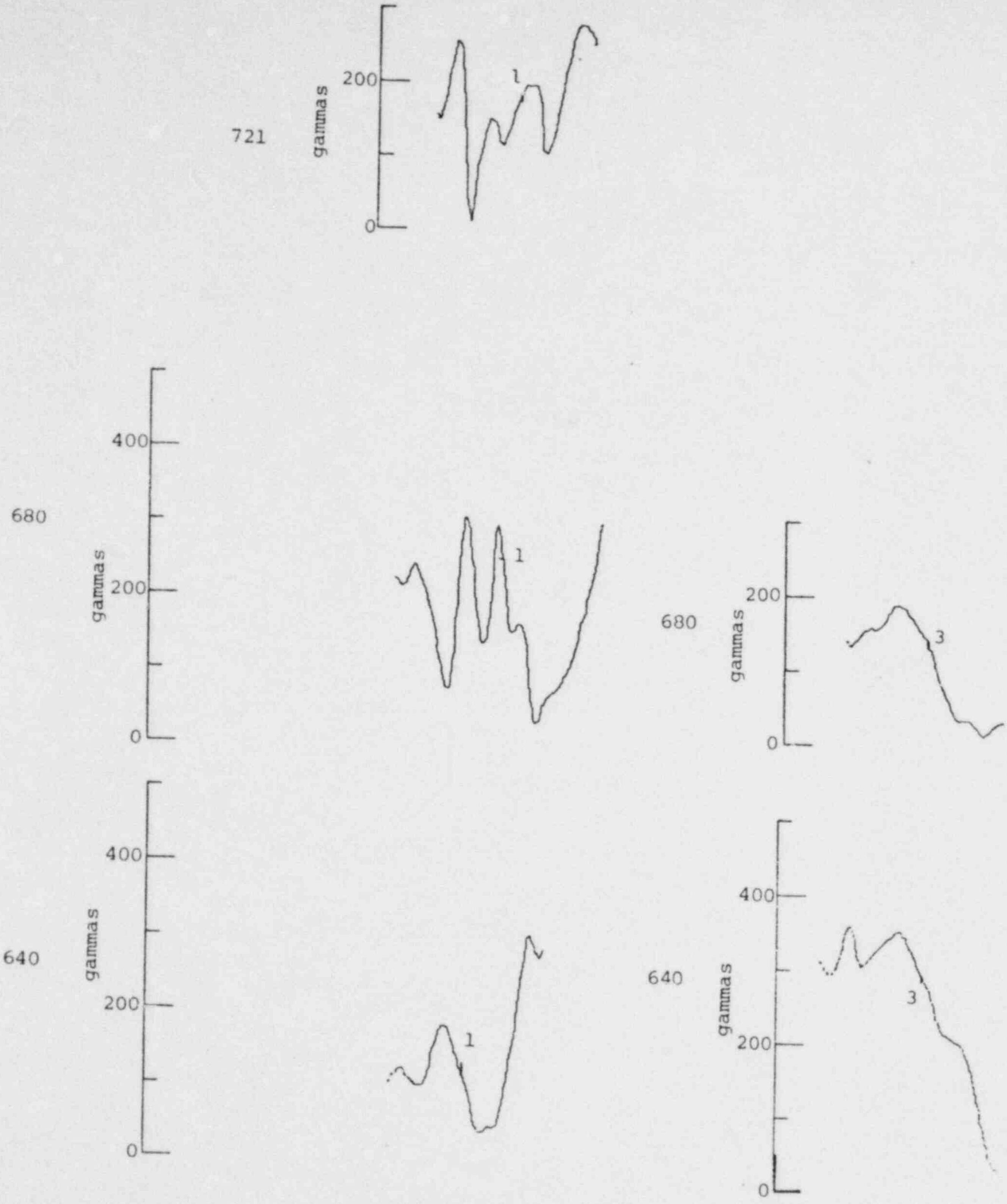
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAGNETIC PROFILES 401, 440, 480	FIGURE 5

Each profile has an arbitrary datum.

Horizontal Scale 1:250,000

Vertical Scale 1 inch = 200 gammas

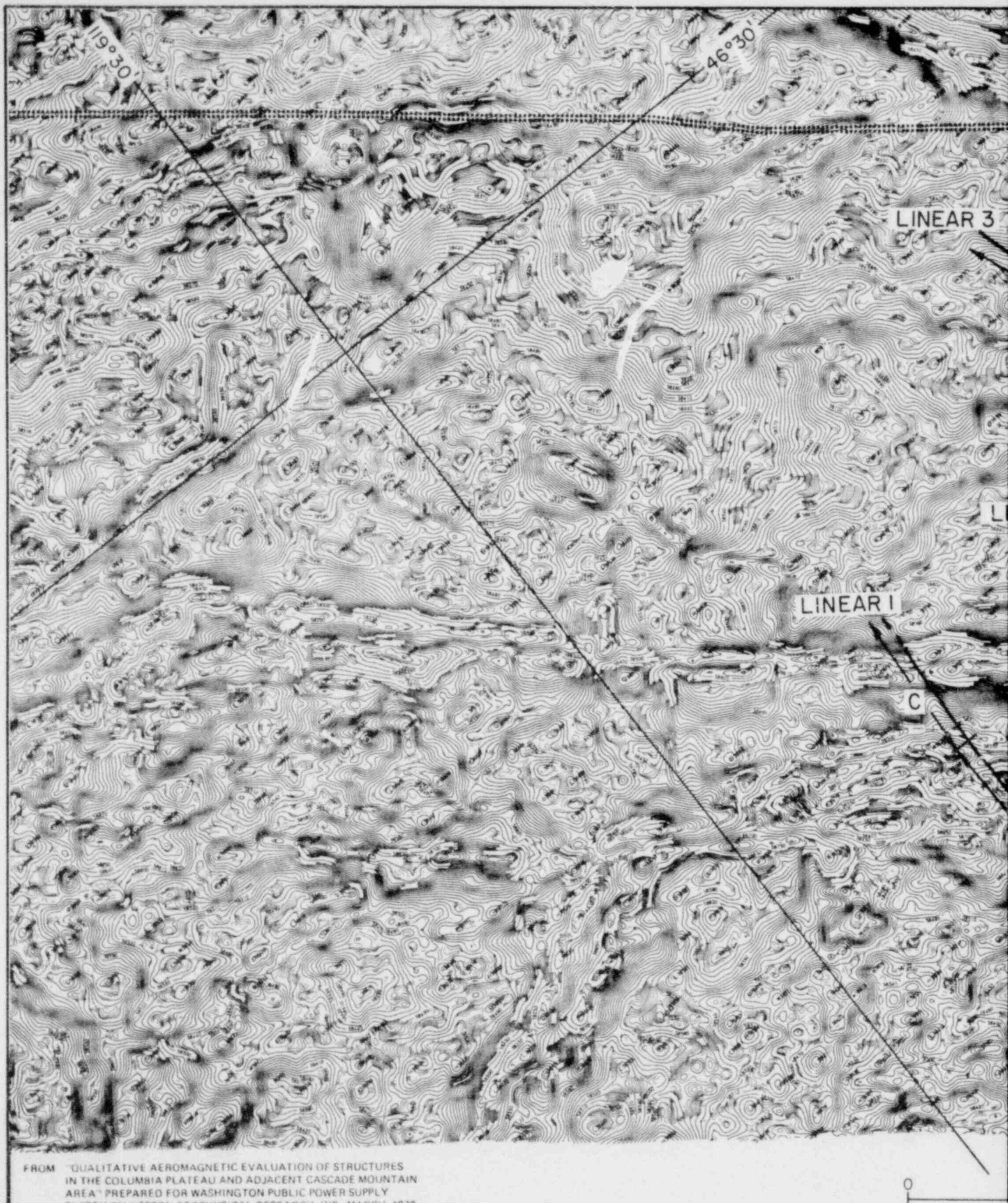
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAGNETIC PROFILES 520, 560, 600	FIGURE 6



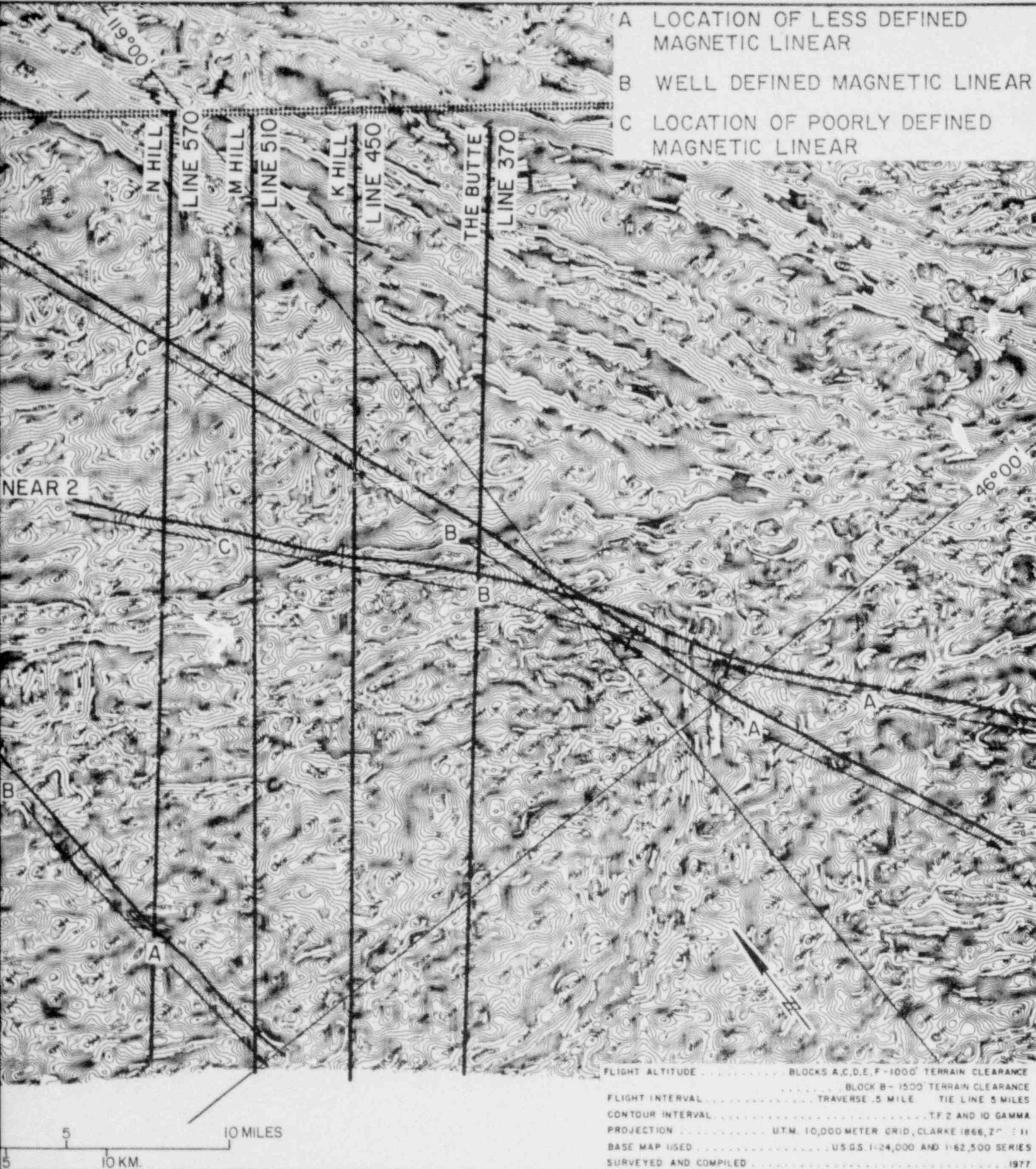
Each profile has an arbitrary datum.

Vertical Scale 1 inch = 200 gammas
Horizontal Scale 1:250,000

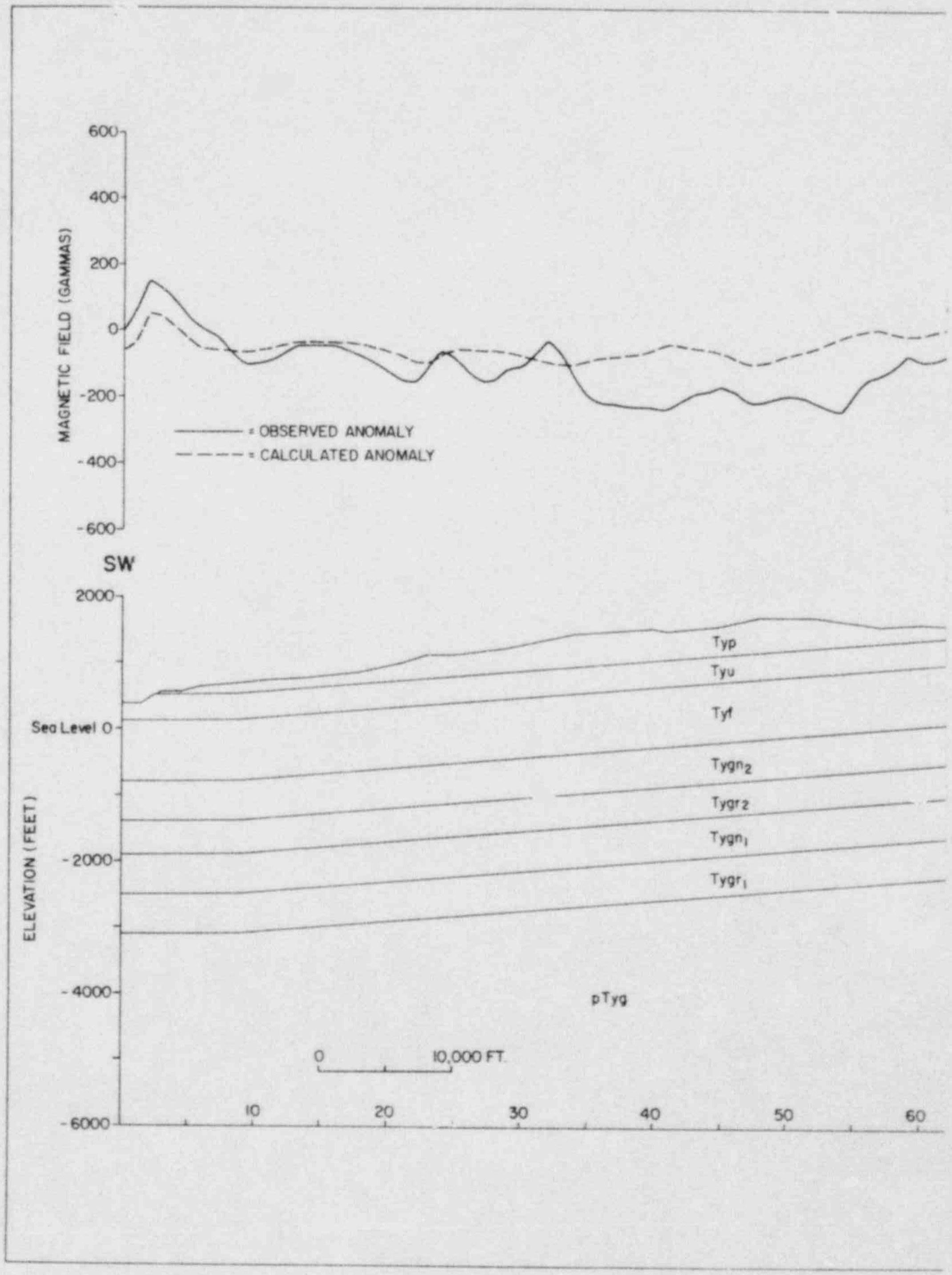
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAGNETIC PROFILES 640, 680, 721	FIGURE 7



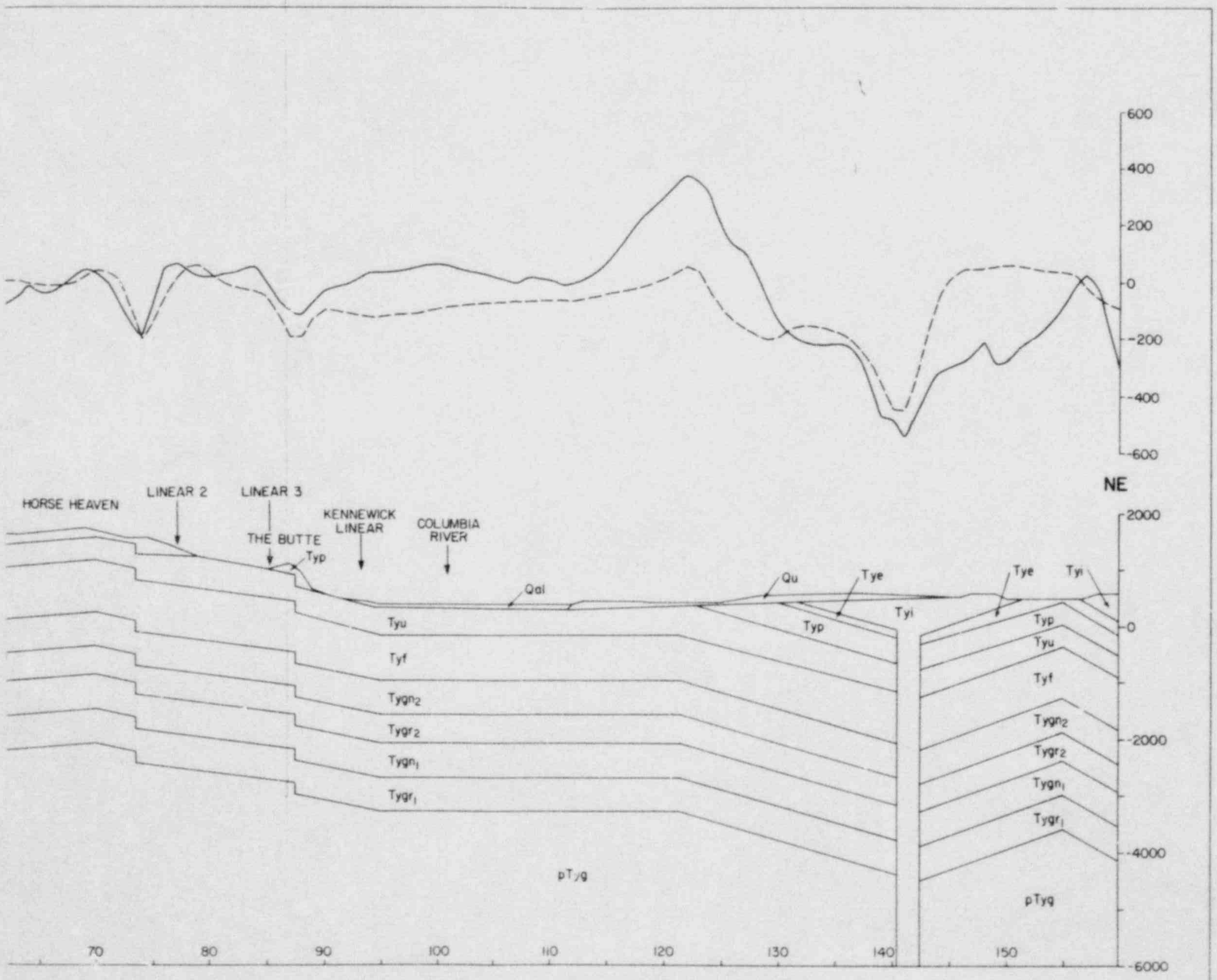
FROM "QUALITATIVE AEROMAGNETIC EVALUATION OF STRUCTURES IN THE COLUMBIA PLATEAU AND ADJACENT CASCADE MOUNTAIN AREA" PREPARED FOR WASHINGTON PUBLIC POWER SUPPLY SYSTEM BY WESTON GEOPHYSICAL RESEARCH, INC., MARCH, 1978.



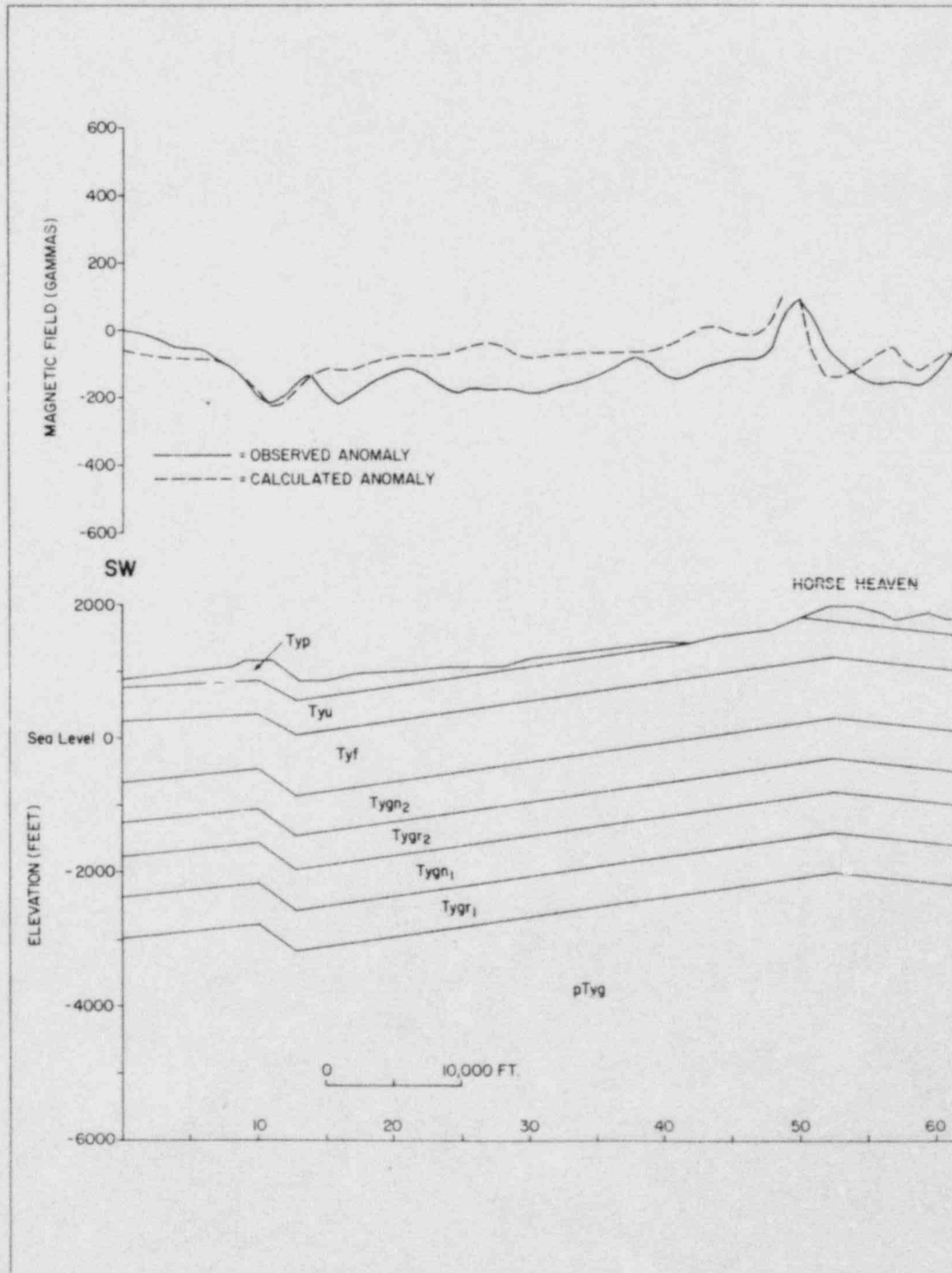
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	LOCATION MAP, MAGNETIC MODELS (WITH RESPECT TO AEROMAGNETIC LINEARS AND ANOMALIES) WALLULA GAP, WASHINGTON	FIGURE 8



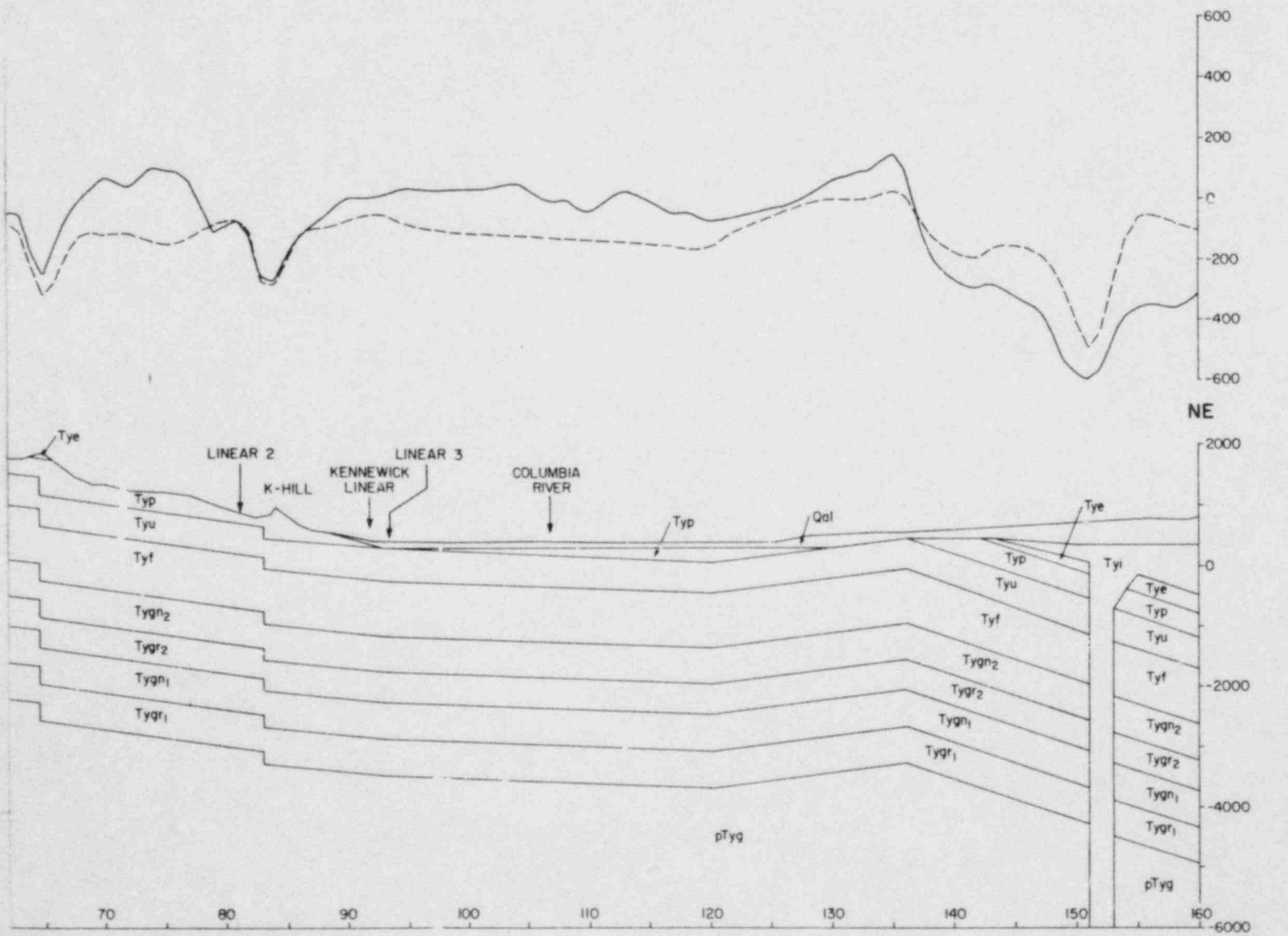
MAGNETIC FIELD (GAMMAS)
600
400
200
0
-200
-400
-600
= OBSERVED ANOMALY
= CALCULATED ANOMALY
SW
ELEVATION (FEET)
2000
Sea Level 0
-2000
-4000
-6000
Typ
Tyu
Tyf
Tygn2
Tygr2
Tygn1
Tygr1
pTyg
0
10,000 FT.
10
20
30
40
50
60



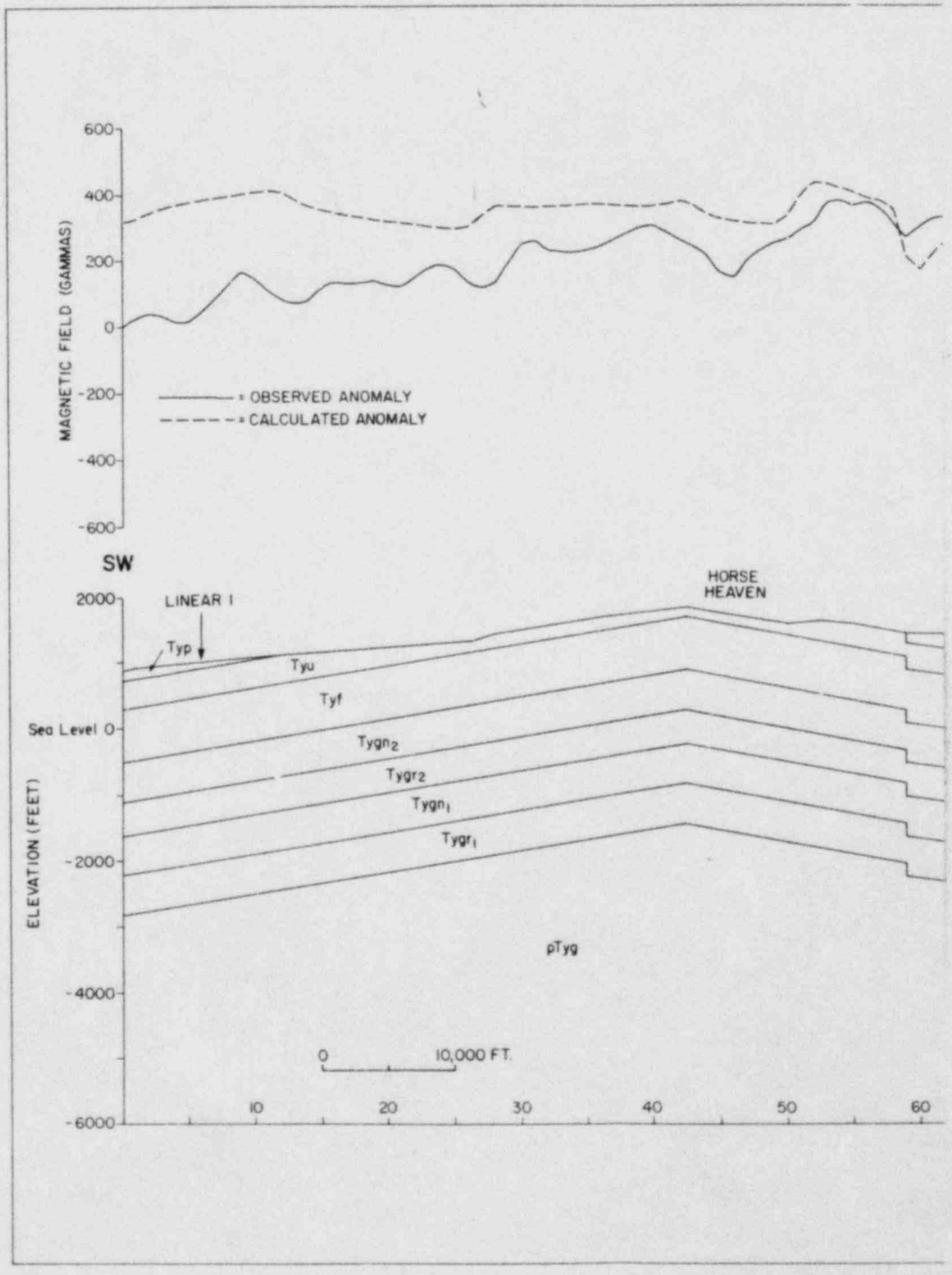
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GEOLOGIC PROFILE AND MAGNETIC PROFILES, LINE 370, HORSE HEAVEN HILLS, BUTTE, ICE HARBOR DAM, WASHINGTON	FIGURE 9



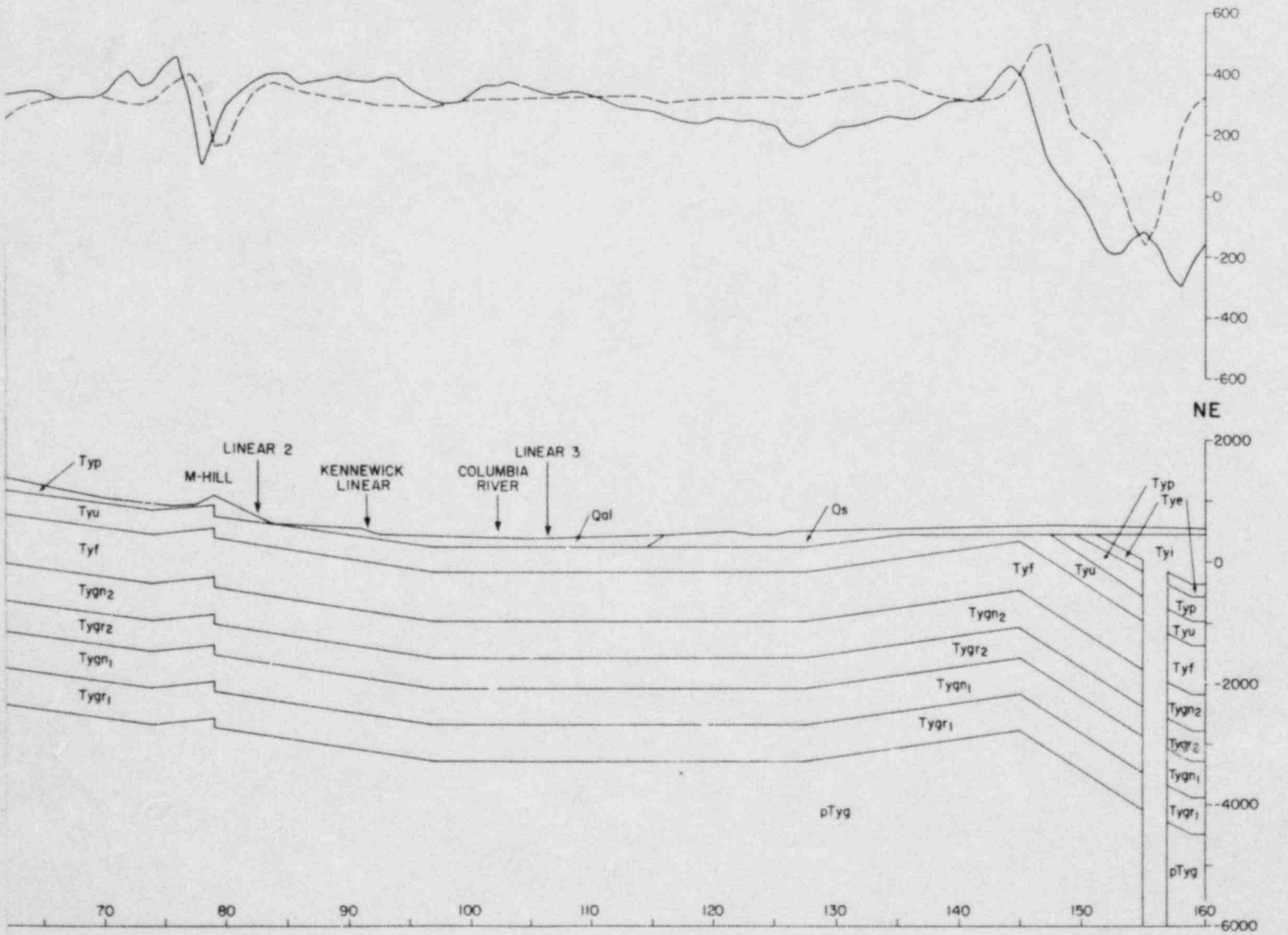
MAGNETIC FIELD (GAMMAS)
600
400
200
0
-200
-400
-600
= OBSERVED ANOMALY
= CALCULATED ANOMALY
SW
HORSE HEAVEN
ELEVATION (FEET)
2000
Sea Level 0
-2000
-4000
-6000
Typ
Tyu
Tyf
Tygn2
Tygr2
Tygn1
Tygr1
pTyg
0
10,000 FT.
10
20
30
40
50
60



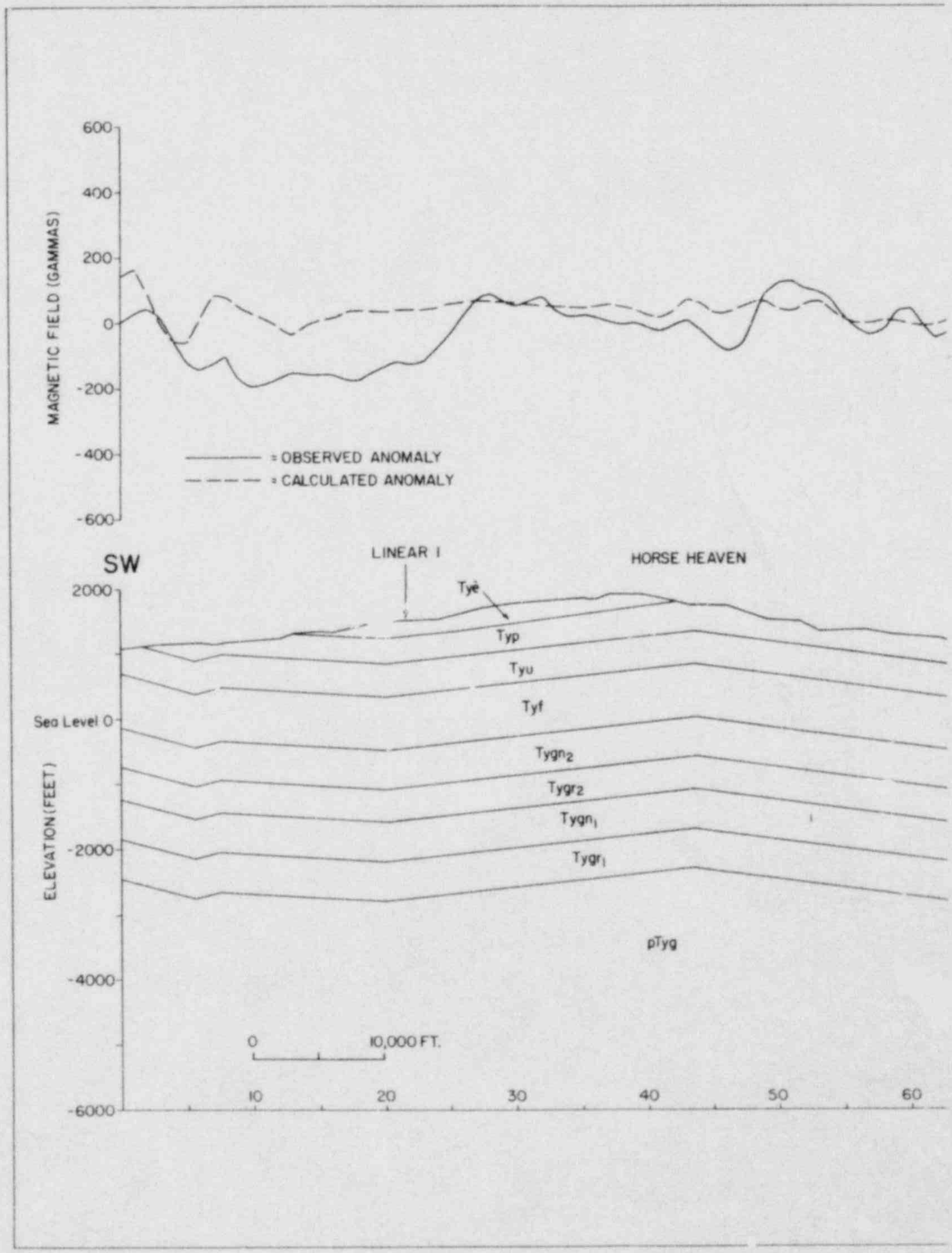
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GEOLOGIC PROFILE AND MAGNETIC PRO- FILES, LINE 450, HORSE HEAVEN HILLS, K. HILL, STRAWBERRY ISLAND, WASHINGTON	FIGURE 10



MAGNETIC FIELD (GAMMAS)
600
400
200
0
-200
-400
-600
= OBSERVED ANOMALY
= CALCULATED ANOMALY
SW
LINEAR I
HORSE HEAVEN
Typ
Tyu
Tyf
Tygn2
Tygr2
Tygn1
Tygr1
pTyg
ELEVATION (FEET)
2000
Sea Level 0
-2000
-4000
-6000
0
10,000 FT.
10
20
30
40
50
60



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GEOLOGIC PROFILE AND MAGNETIC PRO- FILES, LINE 510, HORSE HEAVEN HILLS, M. HILL, PASCO, WASHINGTON	FIGURE 11



MAGNETIC FIELD (GAMMAS)
600
400
200
0
-200
-400
-600
= OBSERVED ANOMALY
= CALCULATED ANOMALY
SW
LINEAR I
HORSE HEAVEN
ELEVATION (FEET)
2000
Sea Level 0
-2000
-4000
-6000
Tyė
Typ
Tyu
Tyf
Tygn2
Tygr2
Tygn1
Tygr1
pTyg
0
10,000 FT.
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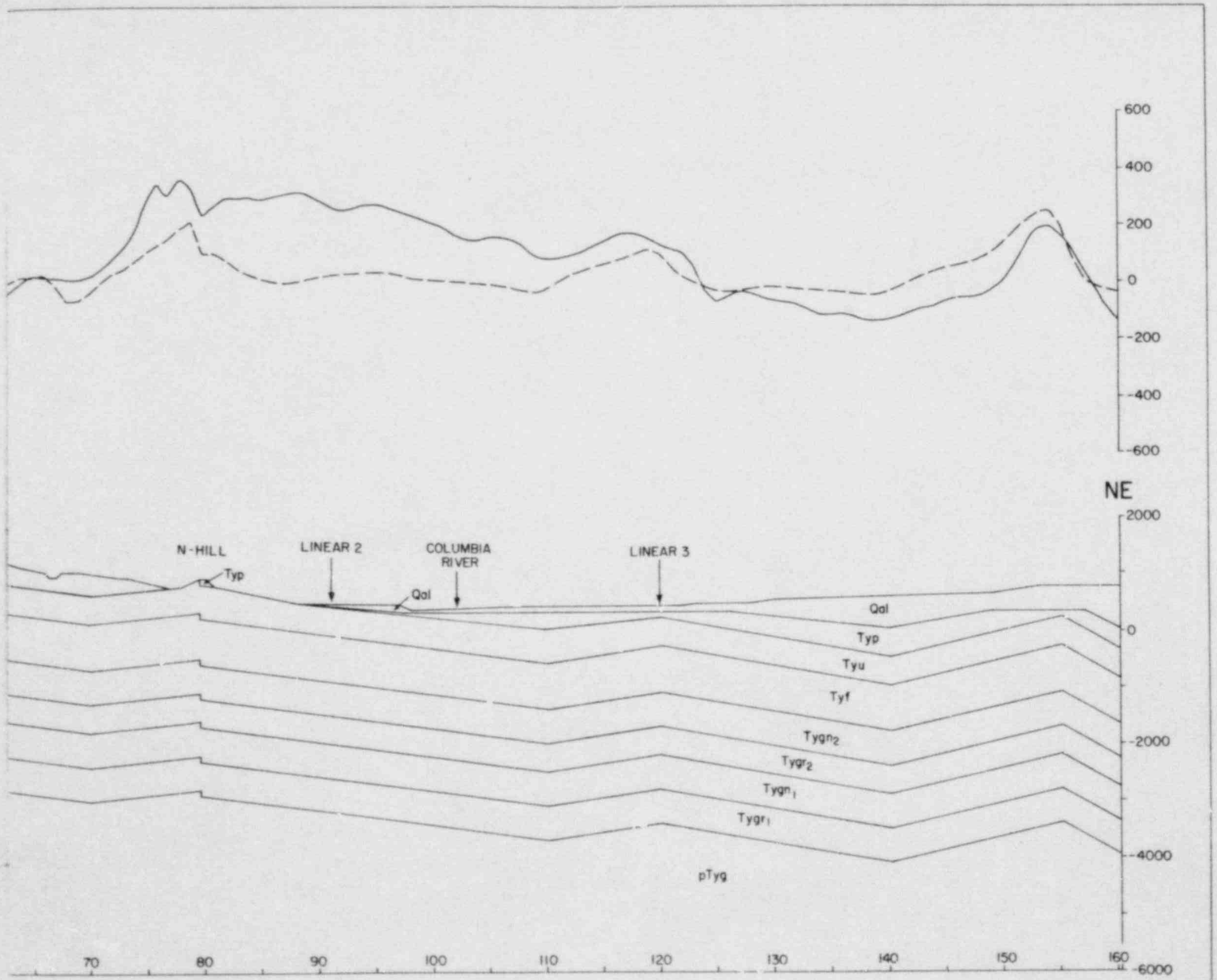

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GEOLOGIC PROFILE AND MAGNETIC PROFILES, LINE 570, HORSE HEAVEN HILLS, N. HILL, COLUMBIA RIVER, WASHINGTON	FIGURE 12

Q. 360.005

Some of the data and discussions in the FSAR of those Columbia Plateau structures relevant to the WNP-2 site are slightly different from the information provided in Amendment 23 to the WNP-1 and 4 PSAR (Docket Nos. 50-460 and 50-513). For example, with regard to the Wallula Gap Fault, your FSAR states that the "...probable fault movement occurred after the deposition of the Touchet beds, and thus less than 12,000 years ago." However, in Appendix 2R H.4 of the WNP-1 & 4 PSAR (Amendment 23), you indicate that the fault is older than the Quaternary Kennewick fanglomerate based on trenching. Additionally, in this same amendment to the WNP-1 & 4 PSAR, you indicate that the faulting along the Horse Heaven Hill Anticline ocurred about 3.5 million years before the present (mybp). The WNP-2 FSAR does not discuss this particular point but, rather, questions the existence of faulting along the Horse Heaven Hill Anticline and indicates that it could be the sole result of folding. Clarify these apparent discrepancies and provide cross-references in the WNP-2 FSAR to the appropriate sections of the WNP-1/4 PSAR.

Response:

The low-level seismicity of the Columbia Plateau attests to ongoing deformation of the region, although in most areas seismicity is not associated with surface manifestations of faulting (Woodward-Clyde Consultants, 1980a,b). Surface faulting of Quaternary age, essentially unknown in 1977 at the time Amendment 23, WNP-1/4 PSAR and the WNP-2 FSAR was submitted, has since been recognized in two general areas of the Plateau: (1) Toppenish Ridge, approximately 80 km west of the WNP-1,2 & 4 sites (Campbell and Bentley, 1980) and (2) about 25 km southeast of the WNP-1,2 & 4 sites, from the vicinity of Wallula Gap on the Columbia River southeastward to the Walla Walla/Milton-Freewater area (Shannon and Wilson, 1979a,b, 1980). The two areas of deformation can both be related to Plateau structural elements, but they appear to have little in common and are geographically distinct.

In the Columbia Plateau of Washington, it is believed that most plateau folding and thrust faulting occurred after Elephant Mountain time (10.5 mybp) and before about 4 to 5 million years ago. The younger limit of this deformational interval is not as well defined as the older. In the Yakima area, the entire Columbia River basalt sequence, including the Ice Harbor Member of the Saddle Mountain Basalt, is generally folded conformably according to Bentley (1977). The Pliocene Thorp Formation, from which one tephra unit has been dated as 3.7 mybp (Bentley, 1977) may be represented by conglomerates on Manastash Ridge that were folded with underlying units. The most important folding on the Yakima Ridges appears to have occurred after deposition of the Elephant Mountain member. Deformation in the western part of the ridge appears to have ceased prior to the emplacement of a one million year old andesite flow (Tieton Andesite) in an erosional embayment across the north flank of the Yakima anticline. The flow exhibits neither faulting nor tilting.

Plio-Pleistocent Simcoe Lavas of unknown age (formational range 4.5 to 0.9 my, Shannon and Wilson, 1973) lie unconformably across fold and thrust structures of the Simcoe-Horse Heaven anticline in south-central Washington. The lavas are cut by a northwest-striking fault with dextral slip. In the same general area of the plateau, a similar fault appears to cut a 4.5 my old Simcoe flow, but is overlapped by a flow dated at 3.5 my (Shannon and Wilson, 1973). Shannon and Wilson interpret the 4.5 to 3.5 my old fault as being synkinematic with formation of the Columbia Hills and Horse Heaven anticlines, but the most recent work by the U.S.G.S. suggests otherwise.

Campbell and Bentley (1980) report that the summit, north flank, and alluvial fans at the base of the north flank of Toppenish Ridge are broken by nearly 95 surface ruptures up to 9 km in length. Most of the ruptures are less than 1 km long. Six of the ruptures have lengths in excess of 3 km. The faulted zone varies in width from 0.5 to 2.2 km and has a length, more-or-less parallel to the ridge, of 32 km. Most flank and summit ruptures are sub-vertical faults for which no strike-slip displacement is evident. Faults at the base of the north-flank are lobate in plan-view and are interpreted as comprising a thrust zone coincident with the older south-dipping Toppenish fault. Some cut glaciofluvival slackwater deposits of Touchet-type and others displace post-"Touchet" alluvial fans. Mt. St. Helens ash (12,800 ybp) is present in some of the slackwater deposits, but has not yet been shown to be faulted. Campbell and Bentley (1980) attribute the Toppenish faults to north-south compression, with thrusting at the northern base of the anticlinal hinge zone. Bentley (personal communication, 1980) believes that the 30 km-long faulted segment of Toppenish Ridge may be terminated at its western and eastern ends by northwest-striking strike-slip faults. These bounding faults may be responsible in some way for localizing Quaternary rupture along the anticline.

To the east, faulting in the vicinities of Wallula Gap and the southern Walla Walla Valley near Milton-Freewater, is apparently related to a zone of dextral transcurrent faulting

(Shannon and Wilson, 1979 a,b), that includes the topographically prominent Wallula fault. Bingham and others (1970) first noted features within the zone that suggested to them the possibility of Quaternary displacement. Recent studies (Shannon and Wilson, 1979 a, b, 1980) indicate the involvement in faulting of late Pleistocene and/or Holocene units (including undated colluvium, Palouse Formation, Touchet Formation, and younger loess) at eight separate localities. The four westernmost localities lie within a 20 km-long segment of the Wallula fault zone. The most western of these localities is at Finley Quarry west of Wallula Gap on the northern end of The Butte. Two of the four eastern localities lie along faults in the Milton-Freewater area (Barrett and Milton-Freewater faults, Shannon and Wilson, 1979 b) that are in general alignment with the Wallula zone farther west. These six localities could be considered as defining a linear zone of 55 km in length within which youthful (late Quaternary), faulting has occurred. There is no evidence, however, that this hypothetical zone is characterized by a single through-going fault, and some [illegible] exists to the contrary. No physical continuity between any two of the six on-trend fault localities can be demonstrated by field relationships.

The eight fault localities need to be kept in perspective, particularly the easternmost localities which might be considered to extend the length of the Wallula fault zone of youthful faulting from 20 to 55 km. For example, at two of the eastern localities are small faults that dip northward at low angles (26° and 30°) and thus have geometric differences with faults in the Wallula zone. Apparent displacements on the two are both quite small. On one, the Buroker fault east of Walla Walla, the base of the late Pleistocene Palouse Formation is offset with a throw of approximately 0.5 meter (Shannon and Wilson, 1979b). South of Umapine near Milton-Freewater, small faults believed to relate to the larger, but inferred Barrett fault, cut Touchet beds and clastic dikes across them with a maximum offset of 0.5 meters (Shannon and Wilson, 1979 b). However, clastic dikes also appear to cut the fault at the same locality. Youthful faulting 10 km farther to the southeast of the Umapine locality is suspected, but not documented. Shannon and Wilson (1979 b) found angular basaltic debris in loess along the trace of an inferred bedrock fault. They suggest that the basalt fragments may have been derived from a fault scarp and were subsequently mixed with surficial loess deposits. At the fourth locality (Little Dry Creek fault south of Milton-Freewater), basalt and Palouse beds are downdropped along a steep (75°) northeast-dipping fault of about 0.5 meters displacement. This fault lies south of an east-projected trace of the Wallula fault zone, and is not in alignment with it.

Some fault displacements within the Wallula zone are clearly pre-Holocene. Along one southerly strand of the zone near Yellepit, west of the Columbia River and southeast of The Butte, late Pleistocene Kennewick gravels (age ca. 55,000 ybp Woodward-Clyde Consultants, 1978) overly sheared and displaced basalts but are not themselves deformed (WNP-1/4 PSAR Amendment 23, Appendix 2R H.4). Collectively, the eight documented or inferred cases of late Pleistocene or Holocene faulting at Finley Quarry and in areas to the east, support the contention that a zone of diffuse and discontinuous dextral strain extends southeastward across the Columbia Plateau into southeastern Washington and northeastern Oregon. The Wallula fault is a segment of that zone, but appears to lose surficial expression at both ends. The folded and faulted (The) Butte, west of Wallula Gap, is both the southeasternmost of the doubly plunging brachyanticlines associated with the Rattlesnake structure and, apparently, the northwesternmost expression of the surficially continuous Wallula fault. Some recent workers studying the area (e.g., Shannon and Wilson, 1979 a) have suggested that the isolated anticlines, which extend like beads in a string southeastward from Rattlesnake Mountain, lie above a deep-seated zone of limited displacement. No throughgoing surficial fault connects the anticlines and no Quaternary displacements have been reported northwest of Finley Quarry.

Although all the geologic and geophysical field work that was planned by WPPSS for 1979 has been completed, the process of snythesizing the data has not been completed. It is planned that all of the data collected since Amendment 23, WNP-1/4 and the WNP-2 FSAR were filed October 1977, will be included and discussed more completely in a revised FSAR for WNP-1, 2 and 4, scheduled for completion by late spring or early summer, 1980.

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