REPORT TO THE NUCLEAR REGULATORY COMMISSION

FROM

THE PACIFIC GAS AND ELECTRIC COMPANY

RESPONDING TO

NUREG-0578:
"TMI-2 LESSONS LEARNED TASK FORCE STATUS REPORT AND SHORT-TERM RECOMMENDATIONS"

50-275/323
Ltr 8-27-79
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AUGUST 27, 1979

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INTRODUCTION AND SUMMARY

Background

This report to the NRC presents the Company's formal response to the TMI-2 Lessons Learned Task Force's Short-Term Recommendations (NUREG-0578). Our first detailed assessment of the lessons learned from TMI was voluntarily submitted on July 5, 1979 in a Report to the NRC Describing Response Programs Following the Accident at Three Mile Island. That report documented PG&E's post-TMI review effort and listed the short-term and long-term commitments deemed necessary in the areas of plant design, emergency response, plant operating procedures, operator selection and training, and administrative controls. The implementation of these commitments has already begun, and semi-annual status reports will be provided.

Following the issuance on July 19, 1979 of NUREG-0578, PG&E met with the NRC Regulatory staff and agreed to assist the staff in expediting the Diablo Canyon review effort by redrafting those sections of our report corresponding to NUREG-0578 recommendations to conform to the NUREG-0578 format. Furthermore, PG&E has prepared this response in a format that minimizes references to other materials previously submitted on the Diablo Canyon docket.

Following the NRC Regulatory staff's presentation of the NUREG-0578 recommendations to the ACRS during its 232nd meeting on August 9-11, the ACRS sent to the NRC Chairman on August 13, 1979 a letter of qualified endorsement of the recommendations. In a memorandum to the NRC Executive Director for Operations, dated August 14, 1979, the ACRS discussed the relationship of the separate recommendations as they related to Diablo Canyon and other near-term operating licenses.

Purpose

The purpose of this report is to expedite the Diablo Canyon licensing process by demonstrating PG&E's commitment to comply with the NUREG-0578 recommendations and to address the ACRS comments.

Scope

The report is arranged by section numbers corresponding to those used in the NUREG-0578. In addition, Sections 2.1.5, 2.1.8, 2.2.1.b, 2.2.3 also address the ACRS comments on the issues of Post-Accident Hydrogen Control Systems, Instrumentation to Follow the Course of an Accident, Shift Technical Advisor, and Revised Limiting Conditions for Operation as they were addressed in the ACRS letter of August 3, 1979.

This report also address the ACRS comments about Diablo Canyon discussed during the Committee's 232nd meeting and alluded to in the August 14, 1979 memorandum. This concern deals with the seismic design implications of TMI (a subject which was not discussed in NUREG-0578). It is discussed generally in Section 3 of this report and in Section 2 discussions as it relates to specific NUREG-0578 recommendations.

On August 20, 1979, the Director of the Office of Nuclear Reactor Regulation sent to the Commission a memorandum with attachments addressing the NUREG-0578 recommendations, ACRS comments on NUREG-0578, and an additional requirement for short-term action. Also included was a set of errata and clarifying comments for NUREG-0578. We have been able to modify some sections of our report to address some of this additional material. However, it is not possible to address all of the material and still comply with our commitment to the Regulatory staff to submit this report by September 1, 1979. An addendum to this report addressing the remaining material will be submitted at the earliest possible time.

Summary

The following sections demonstrate PG&E's commitment to comply with the short-term recommendations of the Lessons Learned Task Force and with the comments raised by the ACRS. The concern for seismic design both at Diablo Canyon and throughout the PG&E Service Area is accommodated by higher seismic design requirements and more stringent building codes with the result that the accident risk at Diablo Canyon due to seismic events is not appreciably different than for other regions of the country.

RESPONSE TO NUREG-0578

SHORT-TERM RECOMMENDATIONS

Section 2.1.1 - Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWR's

A. Task Force Position on Pressurizer Heater Power Supply

Position 1.

The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response*

Power can be supplied to two of the four pressurizer heater groups from the offsite power source when available, or from the onsite emergency generators through the Engineered Safety Features (ESF) buses. Sufficient power is available from the ESF buses to energize enough heaters to establish and maintain natural circulation at hot standby conditions. Redundancy is provided by supplying each group of heaters from a different bus (see Figure 2.1.1-1).

Circuit breakers 52-1G-71 and 52-1H-74 will be added to 480 volt ESF buses 1G and 1H, respectively. These breakers will be seismically qualified and installed to meet safety-grade requirements.

Emergency power is generated by the onsite emergency diesel generators and supplied directly to the 4.16kV ESF buses. Power is then fed through a step-down transformer to the 480 volt ESF buses as shown on Figure 2.1.1-1.

*All of the equipment associated with pressurizer heater power supply described in this response is seismically qualified for the Hosgri event except for those devices specifically noted as non-safety grade.

Section 2.1.1 (Cont'd)

PG&E Response (Cont'd)

The normal power supply for the pressurizer heaters is through the unit auxiliary transformer No. 12 to the 4.16kV normal balance-of-plant (BOP) buses D and E and 480 volt BOP load centers 13D and 13E. If the unit is not generating, the power is supplied to the BOP buses from the 230kV transmission line via the standby startup transformer. The station electrical single line diagram is shown in Figure 2.1.1-2.

Control power for the new circuit breakers will be the same as for the existing breakers as shown in Figure 2.1.1-3. This power comes from the station batteries which supply both ESF and non-ESF loads (see Figure 2.1.1-4). These existing control circuits and switches are not designated Class IE (safety grade). However, they utilize components which are identical to safety-grade components used in other similar functions. The existing control circuits for the two back-up groups of heaters are redundant and the circuits for the new circuit breakers will be installed in accordance with safety-grade requirements. This will be completed prior to OL, or January 1, 1980, whichever is later.

Position 2.

Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capabity for the connection of the pressurizer heaters. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

Procedures and training will be completed to make the operator aware of when and how the required pressurizer heaters should be connected to the emergency buses.

Section 2.1.1 (Cont'd)

PG&E Response (Cont'd)

The normal power supply for the pressurizer heaters is through the unit auxiliary transformer No. 12 to the 4.16kV normal balance-of-plant (BOP) buses D and E and 480 volt BOP load centers 13D and 13E. If the unit is not generating, the power is supplied to the BOP buses from the 230kV transmission line via the standby startup transformer. The station electrical single line diagram is shown in Figure 2.1.1-2.

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Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capabity for the connection of the pressurizer heaters. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

Procedures and training will be completed to make the operator aware of when and how the required pressurizer heaters should be connected to the emergency buses.

Section 2.1.1 (Cont'd)

PG&E Response (Cont'd)

Loading of each ESF bus can be accomplished from the main control board. Procedures will be established to identify under what conditions and which selected loads can be shed from the ESF bus to prevent overloading when the pressurizer heaters are connected. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

Position 3.

The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

The proposed design modifications will provide for simple and rapid transfer of the heater groups to the ESF power source. Within 2 minutes after loss of offsite power, the onsite emergency diesel generators will have started and been connected to any required ESF loads.

When it is determined that the pressurizer heaters are required, the Shift Foreman need only to dispatch an operator to elevation 100' in the Auxiliary Building, which is just three floors directly below the main control room (two separate stairwells are available). Once in the area, the operator simply verifies that the source breakers (52-1H-74, 52-13D-6, 52-13E-2, and 52-1G-71) are open and manually throws a transfer switch. A simplified electrical diagram is given in Figure 2.1.1-1. As soon as the Shift Foreman is notified that the transfer has been made, the heaters can be controlled using the normal control devices provided on the main control console.

PG&E will provide control room indication of actual wattage being supplied to each heater group that has been transferred to the emergency power. This will be implemented prior to OL, or January 1, 1980, whichever is later.

Section 2.1.1 (Cont'd)

Position 4.

Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

Although power could be supplied to the pressurizer heaters from the emergency power sources without any plant modifications, the use of non-safety grade equipment would be required. To comply with this position, two new feeder circuit breakers will be added to the ESF 480 volt buses. Each breaker will feed one back-up group of heaters through a transfer switch and the existing distribution panels. The new circuit breakers and transfer switches will be qualified safety-grade equipment. Existing non-safety grade equipment utilized will be qualified or replaced with qualified equipment. This will be implemented prior to OL, or January 1, 1980, whichever is later.

B. Task Force Position on Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

Position 1.

Motive and control components of the power-operated relief valves (PORV's) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

FSAR Figure 3.2-07 shows the arrangement of the power-operated relief valves (PORV's). These valves are air-to-open, fail-closed valves. They are normally supplied by the plant air compressors with 80 psi air as shown in Figure 2.1.1-5. Two of the three valves have a back-up supply from the nitrogen system to function on loss of air and Class I high pressure accumulators which have sufficient capability to operate each valve 120 times after the loss of both air and nitrogen.

Section 2.1.1 (Cont'd)

PG&E Response (Cont'd)

The third PORV is not supplied with a back-up motive power supply. Its only function is to preclude the possibility of turbine-generator trip, ASME Code safety valve actuation, and reactor trip following the postulated loss of 100% of net plant electrical load.* Furthermore, this function is only required during those periods of early core life characterized by low inherent transient capability. Since the normal supply of valve motive power (i.e., a shared plant air system redundantly supplied by compressors powered from both units) would continue to be available unless both generators tripped, and since the combined probability of the initiating event occurring during the fraction of core life during which the valve is needed is low, the addition of a back-up air system for this valve will not significantly reduce the probability of ASME Code safety valve actuation and is, therefore, not considered necessary.

Each PORV is opened by a solenoid valve which is energized-to-open, spring-to-close. The circuits to the solenoid valves are supplied with redundant interlocks which prevent energization below normal operating pressures. These control circuits are powered from the emergency station batteries (see Figures 2.1.1-6 and 2.1.1-7.

Position 2.

Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

The PORV block valves are shown schematically in FSAR Figure 3.2-07. As shown in Figures 2.1.1-4 and 2.1.1-6, these valves are powered from ESF buses which are served by either offsite power or the emergency diesel generators. Each of the three valves is powered from a separate 480 volt ESF bus (Bus Sections 1F, 1G and 1H).

*Net plant electrical load is that portion of the generator ouput which does not power "house loads" such as the reactor coolant pumps, instrument air compressors, and other equipment necessary for continued plant operation.

Section 2.1.1 (Cont'd)

Position 3.

Motive and control power connections to the emergency buses for the PORV's and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements. (Category A: Implementation shall be completed prior to OL, or Janury 1, 1980, whichever is later.)

PG&E Response

The motive and control power connections for the PORV block valves are made with equipment qualified to safety-grade requirements. The motive power for the PORV's is air or nitrogen (see 1 above). The piping, accumulators, control power connections, and the solenoid valves are qualified in accordance with safety-grade requirements.

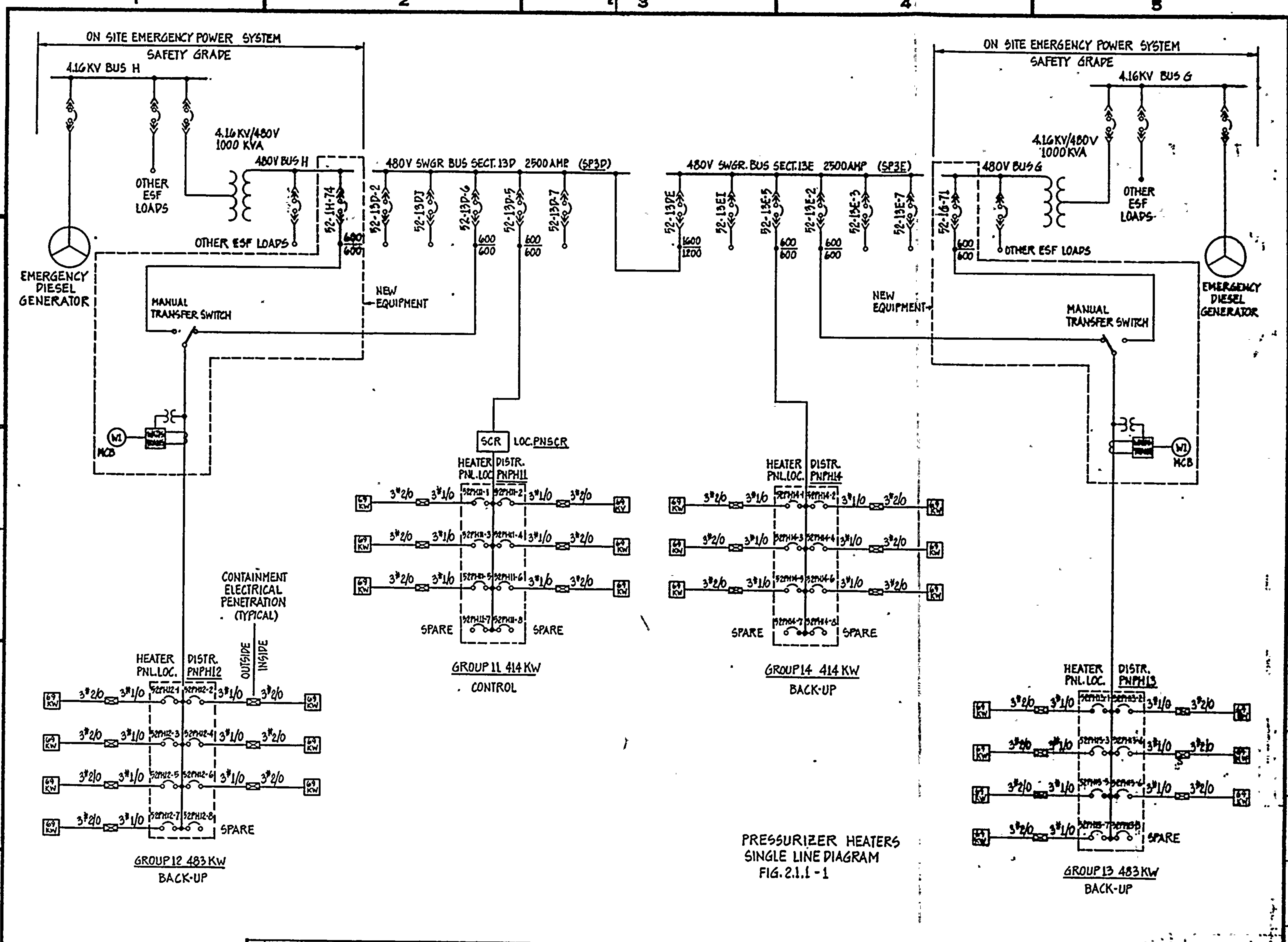
A description of the qualifications is given in the attached applicable pages of the Diablo Canyon FSAR Section 3.11

Position 4.

The pressurizer level indication instrument channels shall be powered from the vital instrument buses. These buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

The pressurizer level indication circuits are safety-grade and post-accident qualified. AC power for all Class IE instrument channels is supplied from inverters which are supplied from the ESF buses with automatic backup from the emergency batteries (see Figures 2.1.1-4, 2.1.1-8 to 2.1.1-10).



PRESSURIZER HEATERS
SINGLE LINE DIAGRAM
FIG. 2.1.1-1

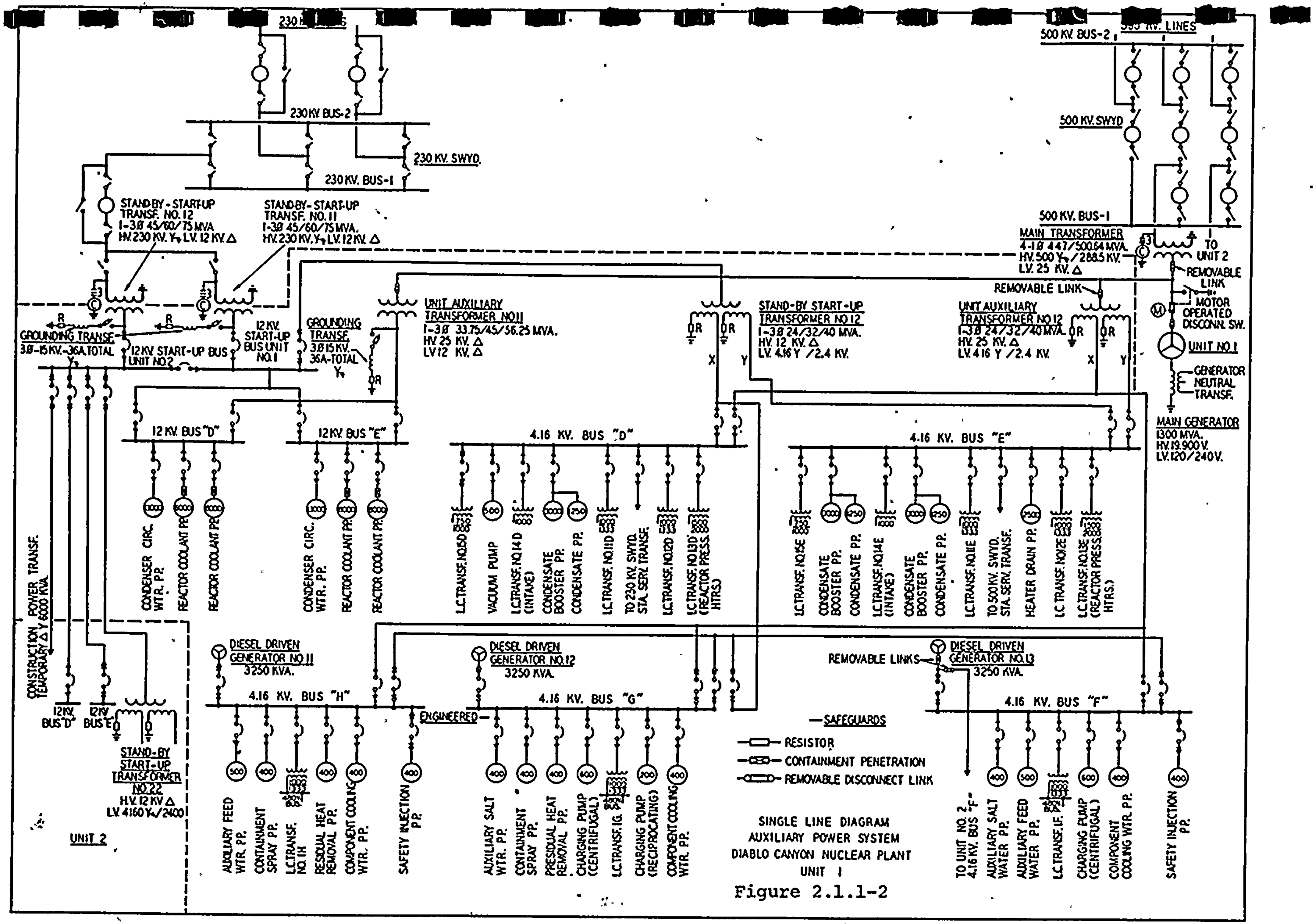
NO.	DATE	DESCRIPTION	GM/SPEC	DWN.	CHKD.	SUPV.	APVD. BY

REVISIONS

APPROVED BY: GM, SUPV., DSGN., DWN., CHKD., O.K., DATE, SCALES

PACIFIC GAS AND ELECTRIC COMPANY
SAN FRANCISCO, CALIFORNIA

BILL OF MATL
DWG LIST
SUPSDS
SUPSD BY
SHEET NO. SHEETS
REV.

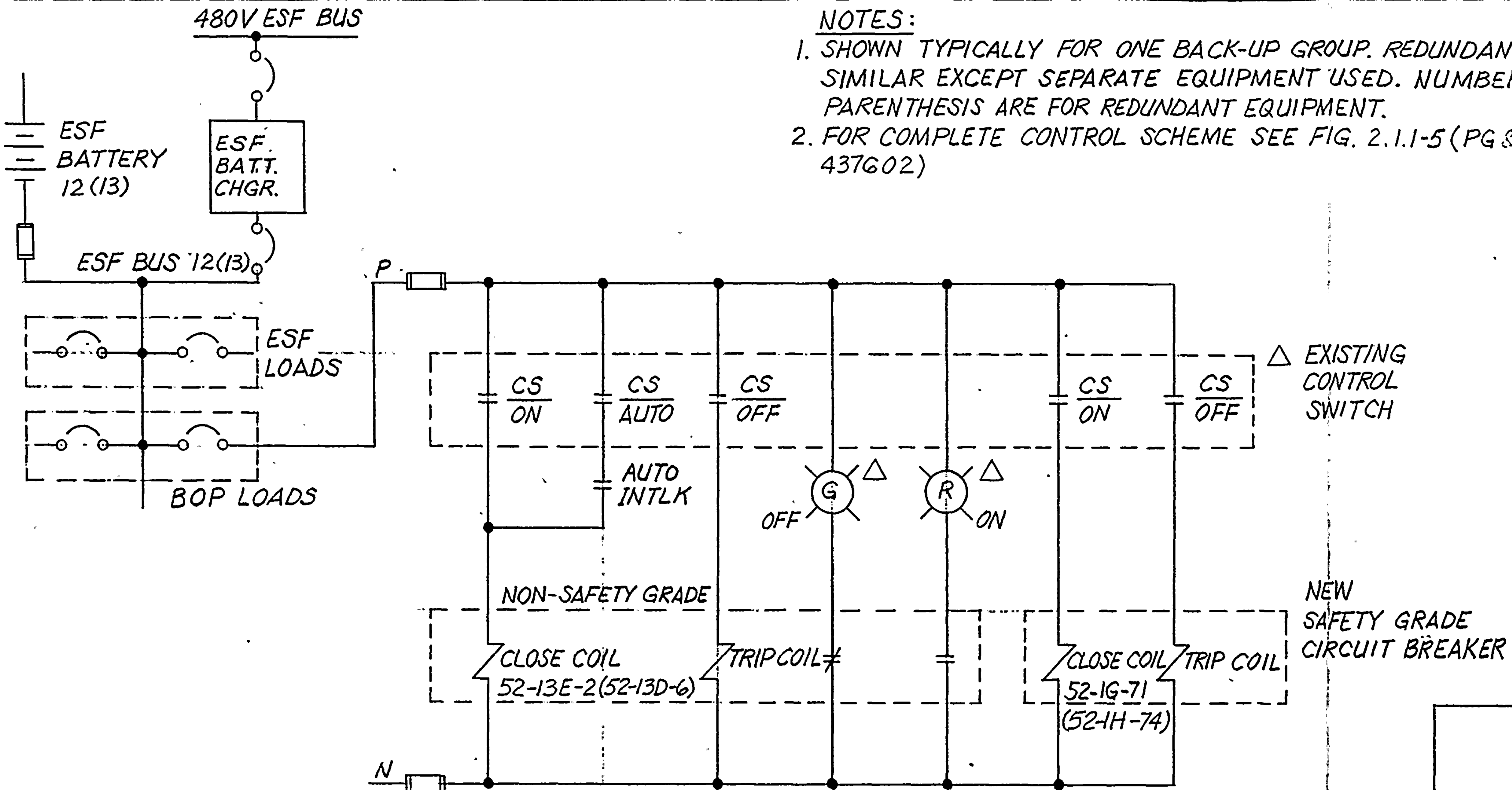


230 KV. BUS-2
230 KV. SWYD.
230 KV. BUS-1
500 KV. BUS-2
500 KV. SWYD.
500 KV. BUS-1
STAND-BY - START-UP TRANSF. NO. 12 1-3Ø 45/60/75 MVA HV. 230 KV. Y, LV. 12 KV. Δ
STAND-BY - START-UP TRANSF. NO. 11 1-3Ø 45/60/75 MVA. HV. 230 KV. Y, LV. 12 KV. Δ
MAIN TRANSFORMER 4-1Ø 447/500.64 MVA. HV. 500 Y/288.5 KV. LV. 25 KV. Δ
TO UNIT 2
REMOVABLE LINK
MOTOR OPERATED DISCONN. SW.
UNIT NO. 1
GENERATOR NEUTRAL TRANSF.
MAIN GENERATOR 1300 MVA. HV. 19,900 V. LV. 120/240 V.
GROUNDING TRANSF. 3Ø-15 KV.-36A TOTAL
12 KV START-UP BUS UNIT NO. 2
12 KV. START-UP BUS UNIT NO. 1
GROUNDING TRANSF. 3Ø 15 KV. 36A-TOTAL
UNIT AUXILIARY TRANSFORMER NO. 11 1-3Ø 33.75/45/56.25 MVA. HV. 25 KV. Δ LV. 12 KV. Δ
STAND-BY START-UP TRANSFORMER NO. 12 1-3Ø 24/32/40 MVA. HV. 12 KV. Δ LV. 4.16 Y/2.4 KV.
UNIT AUXILIARY TRANSFORMER NO. 12 1-3Ø 24/32/40 MVA. HV. 25 KV. Δ LV. 4.16 Y/2.4 KV.
12 KV. BUS "D"
12 KV. BUS "E"
4.16 KV. BUS "D"
4.16 KV. BUS "E"
CONDENSER CIRC. WTR. PP.
REACTOR COOLANT PP.
L.C. TRANSF. NO. 15D
VACUUM PUMP
L.C. TRANSF. NO. 14D (INTAKE)
CONDENSATE BOOSTER PP.
CONDENSATE PP.
L.C. TRANSF. NO. 11D
TO 230 KV. SWYD. STA. SERV. TRANSF.
L.C. TRANSF. NO. 12D
L.C. TRANSF. NO. 13D (REACTOR PRESS. HTRS.)
L.C. TRANSF. NO. 15E
L.C. TRANSF. NO. 14E (INTAKE)
L.C. TRANSF. NO. 11E
TO 500 KV. SWYD. STA. SERV. TRANSF.
HEATER DRAIN PP.
L.C. TRANSF. NO. 12E
L.C. TRANSF. NO. 13E (REACTOR PRESS. HTRS.)
CONSTRUCTION POWER TRANSF. TEMPORARY Δ Y 6000 KVA
12 KV. BUS "D"
12 KV. BUS "E"
STAND-BY START-UP TRANSFORMER NO. 22 H.V. 12 KV Δ L.V. 4160 Y/2400
UNIT 2
DIESEL DRIVEN GENERATOR NO. 11 3250 KVA.
DIESEL DRIVEN GENERATOR NO. 12 3250 KVA.
DIESEL DRIVEN GENERATOR NO. 13 3250 KVA.
REMOVABLE LINKS
4.16 KV. BUS "H"
4.16 KV. BUS "G"
4.16 KV. BUS "F"
ENGINEERED
SAFEGUARDS
RESISTOR
CONTAINMENT PENETRATION
REMOVABLE DISCONNECT LINK
AUXILIARY FEED WTR. PP.
CONTAINMENT SPRAY PP.
L.C. TRANSF. NO. 1H
RESIDUAL HEAT REMOVAL PP.
COMPONENT COOLING WTR. PP.
SAFETY INJECTION PP.
AUXILIARY SALT WTR. PP.
CHARGING PUMP (CENTRIFUGAL)
L.C. TRANSF. 1G
CHARGING PUMP (RECIPROCATING)
TO UNIT NO. 2 4.16 KV. BUS "F"
AUXILIARY SALT WATER PP.
L.C. TRANSF. 1F
SINGLE LINE DIAGRAM
AUXILIARY POWER SYSTEM
DIABLO CANYON NUCLEAR PLANT
UNIT 1

Figure 2.1.1-2

NOTES:

1. SHOWN TYPICALLY FOR ONE BACK-UP GROUP. REDUNDANT GROUP SIMILAR EXCEPT SEPARATE EQUIPMENT USED. NUMBERS IN PARENTHESIS ARE FOR REDUNDANT EQUIPMENT.
2. FOR COMPLETE CONTROL SCHEME SEE FIG. 2.1.1-5 (PG&E DRAWING 437602)



SIMPLIFIED CONTROL DIAGRAM
FIG. 2.1.1-3

△ ON OPERATOR CONTROL CONSOLE IN MAIN CONTROL ROOM.

NO.	DATE	DESCRIPTION	GM	DWN.	CHKD.	SUPV.	APVD.

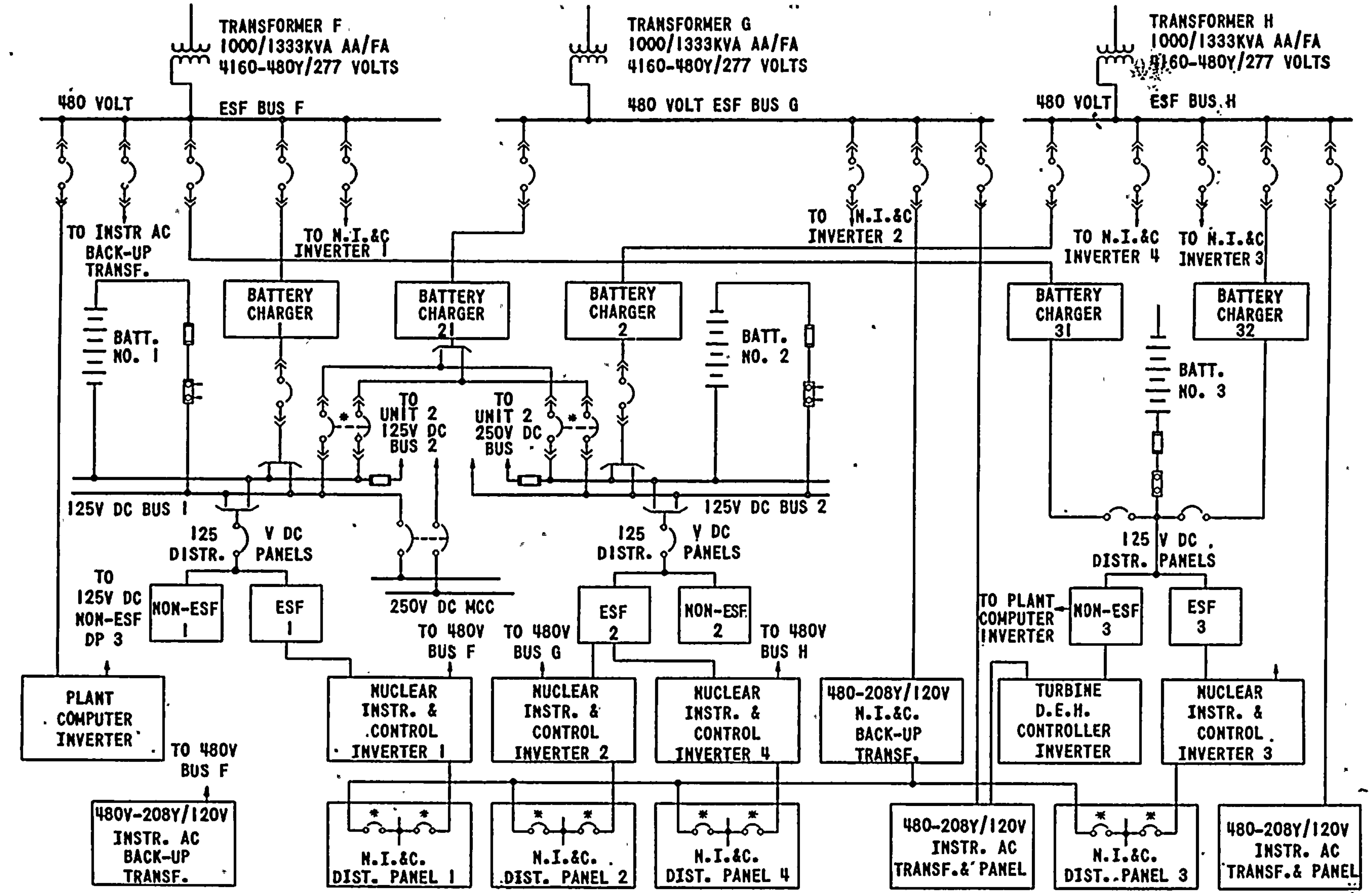
REVISIONS

APPROVED BY: GM, SUPV., DSGN., DWN., CHKD., O.K., DATE, SCALES

PACIFIC GAS AND ELECTRIC COMPANY
SAN FRANCISCO, CALIFORNIA

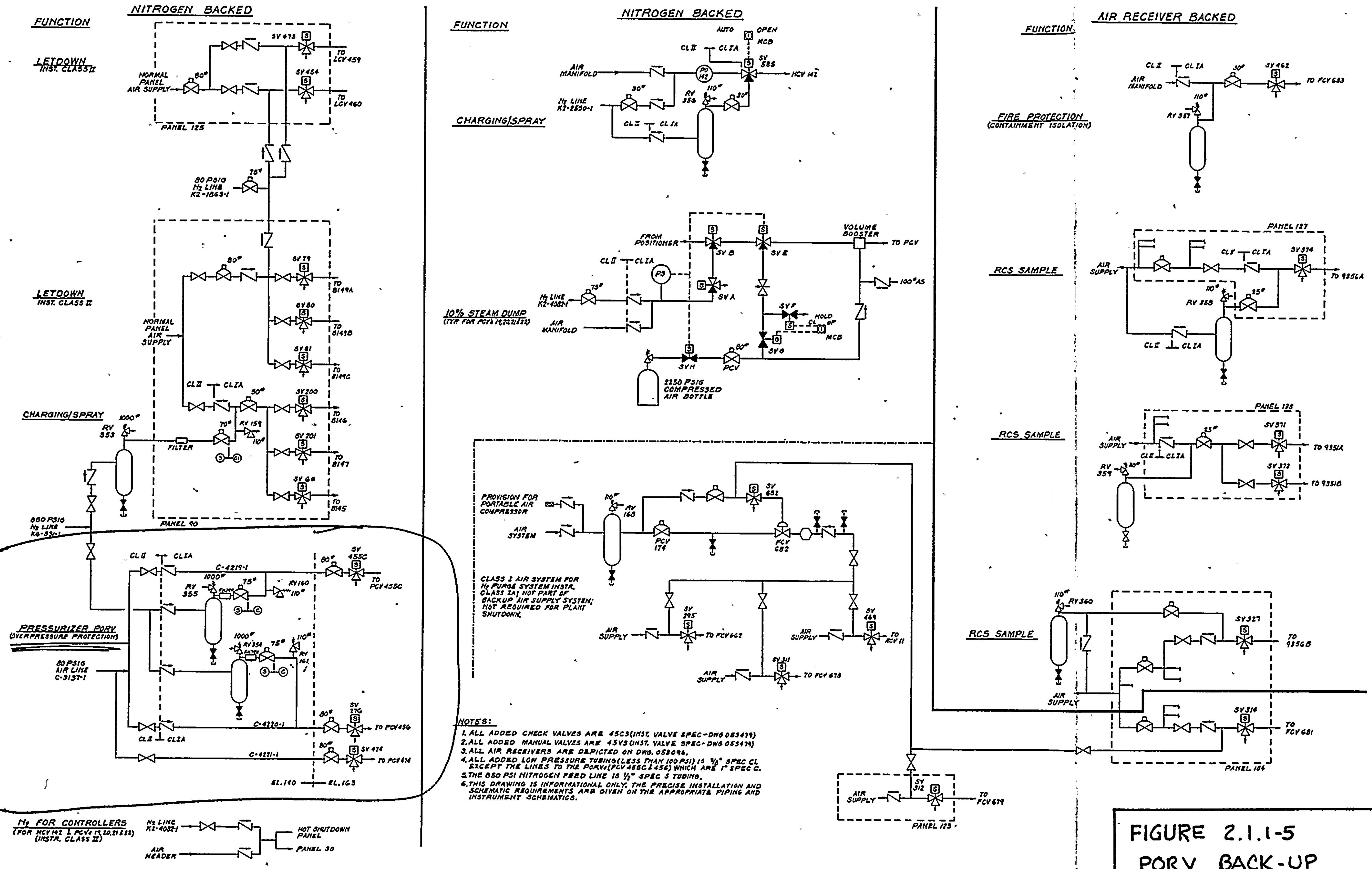
MICROFILM
BILL OF MATERIAL
DRAWING LIST
SUPERSEDES
SUPERSEDED BY
SHEET NO. SHEETS
REV.

62-1828 (REV. 12-74) PRINTED ON DIEPO NO. 1000H-10 CLEARPRINT FADE-OUT 4-75



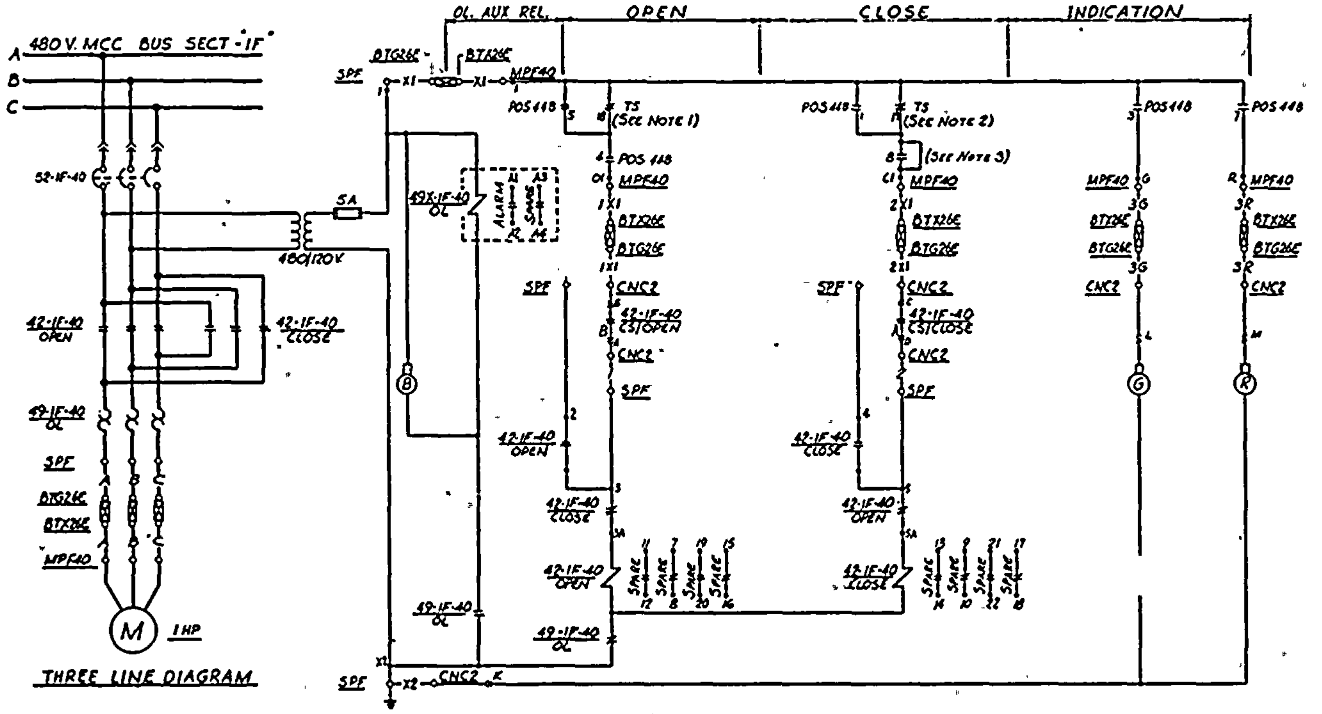
* BREAKERS MECHANICALLY INTERLOCKED

DIABLO CANYON SITE, UNIT 1
SINGLE LINE DIAGRAM
480V ESF & 250/125 DC
Figure 2.1.1-4



NOTES:
1. ALL ADDED CHECK VALVES ARE 45C3(INST. VALVE SPEC-DWG 053479)
2. ALL ADDED MANUAL VALVES ARE 45V3(INST. VALVE SPEC-DWG 053479)
3. ALL AIR RECEIVERS ARE DEPICTED ON DWG. 053096.
4. ALL ADDED LOW PRESSURE TUBING(LESS THAN 100 PSI) IS 3/8" SPEC CL EXCEPT THE LINES TO THE PORVs(PCV 455C & 456) WHICH ARE 1" SPEC C.
5. THE 850 PSI NITROGEN FEED LINE IS 1/2" SPEC S TUBING.
6. THIS DRAWING IS INFORMATIONAL ONLY. THE PRECISE INSTALLATION AND SCHEMATIC REQUIREMENTS ARE GIVEN ON THE APPROPRIATE PIPING AND INSTRUMENT SCHEMATICS.

FIGURE 2.1.1-5
PORV BACK-UP AIR SUPPLY



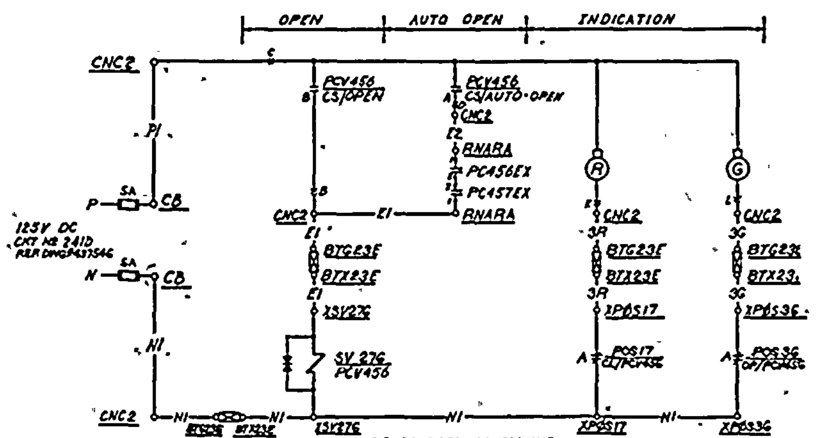
THREE LINE DIAGRAM

120V AC CONTROL SCHEMATIC

8000A PRESSURIZER POWER RELIEF STOP VALVE II - CKT Nº F40P00

PRESSURIZER RELIEF VALVES SIMILAR EXCEPT AS SHOWN IN TABLE.

VALVE No.	DESCRIPTION	CIRCUIT BKR No	CIRCUIT No.	BUS SECT	NORMAL POSITION	ENGINEER SAFETY VITAL FUNCTION	VALVE POSITION SWITCH-(POS) POS No.	ALARM CKT Nº OVER-CURRENT	PENETR. Nº	MOTOR Nº
8000A	Pressurizer Pwr Relief Stop V. II	52-1F-40	F40P00	1F	Open	———	448	A1440BDF	26E	MPF40
8000B	| | | | 12	52-1G-46	G46P00	1G	|	———	455	A1430BDG	19E	MPG46
8000C	| | | | 13	52-1H-33	H33P00	1H	|	———	462	A1420BDH	12E	MPH33

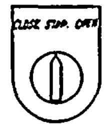


125V DC CONTROL SCHEMATIC

PCV456 PRESSURIZER POWER RELIEF VALVE - CKT. Nº RP053

PRESSURIZER POWER RELIEF VALVES SIMILAR EXCEPT AS SHOWN IN TABLE

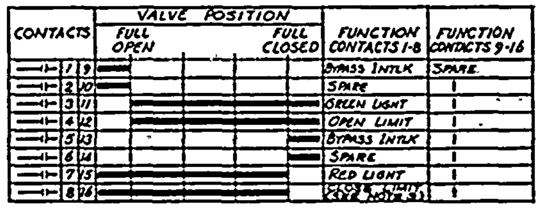
Valve No.	DESCRIPTION	D.C. Ckt. No.	Circuit No.	Valve Loc.	Sol. Vlv. No.	Sol. Vlv. Loc.	Normal Position	Engineer Safety Vital Function	Valve Position Switch (POS) Close	Open	Pressure Control Aux. Rel. Loc. of Relays RNARA Device	Contact	Device	Contact	Penn. No.
PCV456	Pressurizer Power Relief Valve	241D	RP053	1PCV456	276	XSV276	Closed	———	17	36	PC456EX	1—2	PC457EX	2—1	23E
PCV474	| | | |	241D	RP057	1PCV474	474	XSV474	|	———	18	37	PC457EX	3—4	PC456EX	4—3	|
PCV455C	| | | |	241D	RP049	1PCV455C	455C	XSV455C	|	———	16	35	PC455EX	1—2	PC474BX	2—1	|



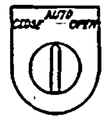
CONTACTS		POSITION CLOSE	STOP	OPEN	REMARKS
—||—	A	X			This Dwg.
—||—	B			X	This Dwg.

CONTROL SWITCH 42-1F-40 CS, 42-1G-46 CS, 42-1H-33 CS

USED AT LOCATION CNC2
NO SPRING RETURN



TYPICAL VALVE POSITION SWITCH CONTACT DEVELOPMENT



CONTACTS		POSITION CLOSE	AUTO	OPEN	REMARKS
—||—	A		X	X	This Dwg.
—||—	B			X	This Dwg.

CONTROL SWITCH PCV456 CS, PCV474 CS, PCV455C CS

USED AT LOCATION CNC2
NO SPRING RETURN

CONTACTS		VALVE POSITION CLOSE	OPEN & INTERM.	REMARKS
—||—	A		X	RED LT.
—||—	B	X		SPARE
—||—	C		X	SPARE
—||—	D	X		SPARE

VALVE CLOSE POSITION SWITCH
POS 16, 17, & 18

CONTACTS		VALVE POSITION CLOSE & INTERM.	OPEN	REMARKS
—||—	A	X		GREEN LT.
—||—	B		X	SPARE
—||—	C	X		SPARE
—||—	D		X	SPARE

VALVE OPEN POSITION SWITCH
POS 35, 36, & 37

TABLE OF DEVICES

DEVICE No	FUNCTION	RATING	MFR.	TYPE	CAT/DWG.#	REMARKS
42-1F-40, 1G-46, 1H-33	480V AC Motor Starter Reversible, Size 1	—	W.E.Corp	—	Ref. Dwg. #1,2	
49-1F-40, 1G-46, 1H-33	Motor Overload Relay	—	W.E.Corp	—	—	Part of Motor Starter
49X-1F-40, 1G-46, 1H-33	Motor Overload Auxiliary Relay	—	W.E.Corp	—	—	Part of Motor Starter
52-1F-40, 1G-46, 1H-33	480V AC Air Circuit Breaker	—	W.E.Corp	—	Ref. Dwg. #1,2	
FCV	Flow Control Valve	—	—	—	—	See Table for Valve Nos
PC456EX Etc.	Pressure Controller Aux. Rel.					
POS	FCV Position Switch	—	—	—	—	See Table for Sw. Nos
SV...	Solenoid Valve					
TS...	Torque Switch					

EQUIPMENT LOCATION NUMBERS

SPF - 480V Motor Control Center Bus Section 1F.
MPF40 - Motor Operated Valve 8000A
CNC2 - Control Board, Reactor Coolant.
BTG26E - Terminal Box, Area G, No 26E
RNARA - Rack Nuclear Aux. Relay Cubicle A

NOTES:

1. Switch Opens On Mechanical Torque During Opening C
2. Switch Opens On Mechanical Torque During Closing Cycle Or Fully Closed Valve (17)
3. Field To Install Jumper If Torque Seating Is Required

FIGURE 2.1.1-6
PRESSUZER PORVs & BLOCK VALVES

NOTES:

1- PRESSURIZER PC CONTACTS ARE IN SERIES THROUGH AUX. RELAY RACK.
2- SEE INSTRUMENT AIR PIPING REQUIREMENTS (SPECIAL AIR REQ'T.)

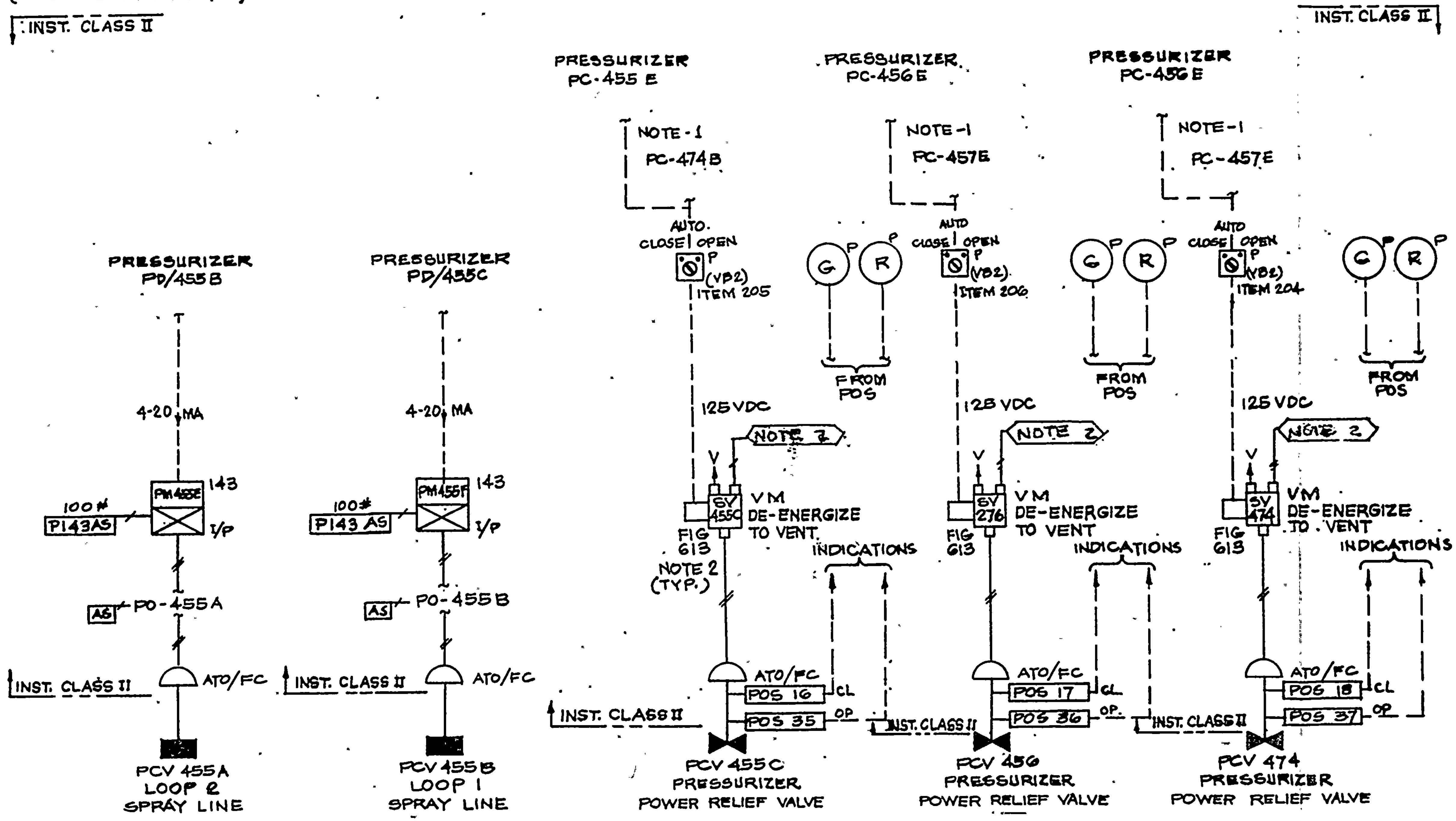


FIGURE 2.1.1-7
PRESSURIZER
PORV DIAGRAM

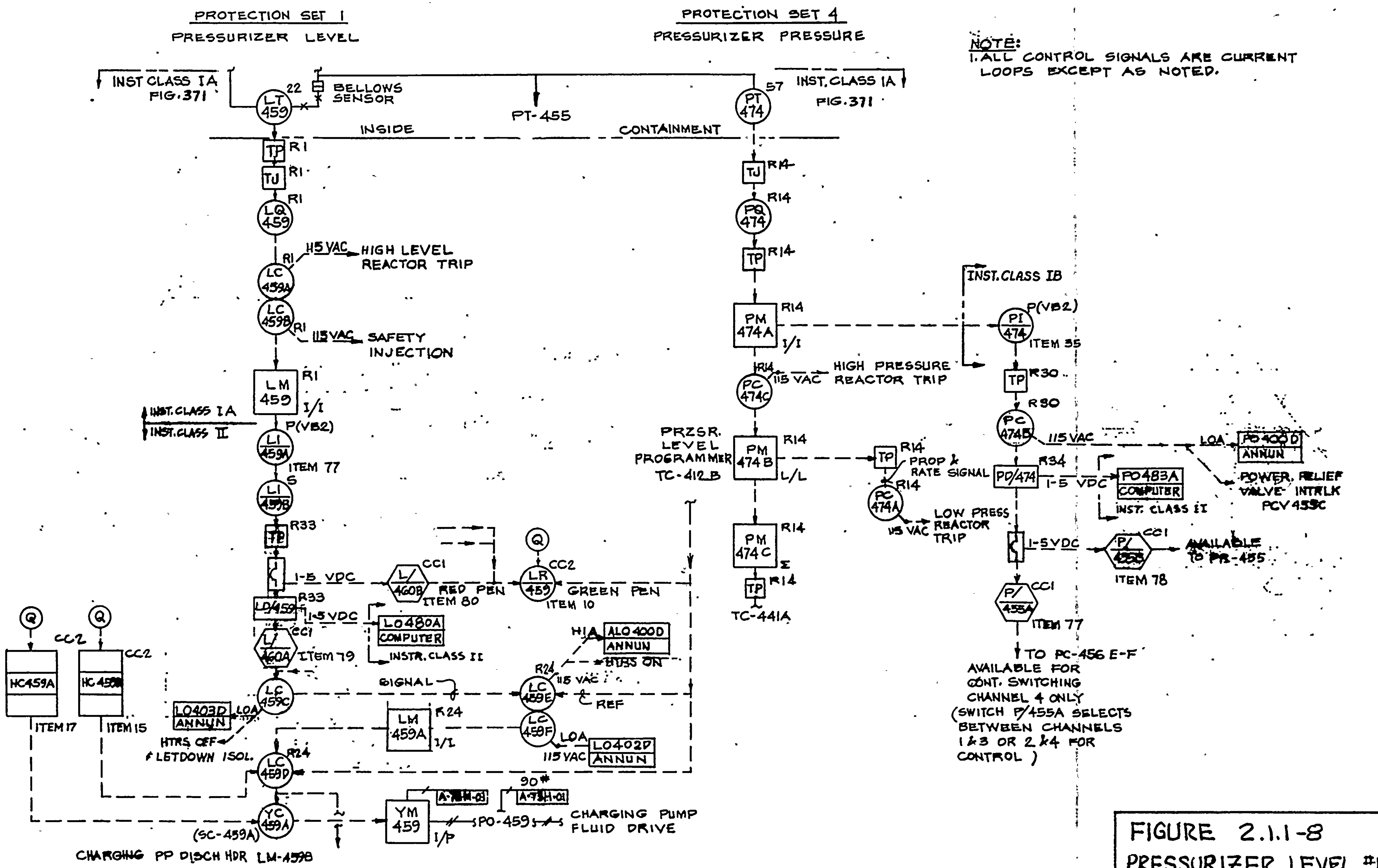


FIGURE 2.1.1-8
PRESSURIZER LEVEL, #1 & PRESSURE SET #4

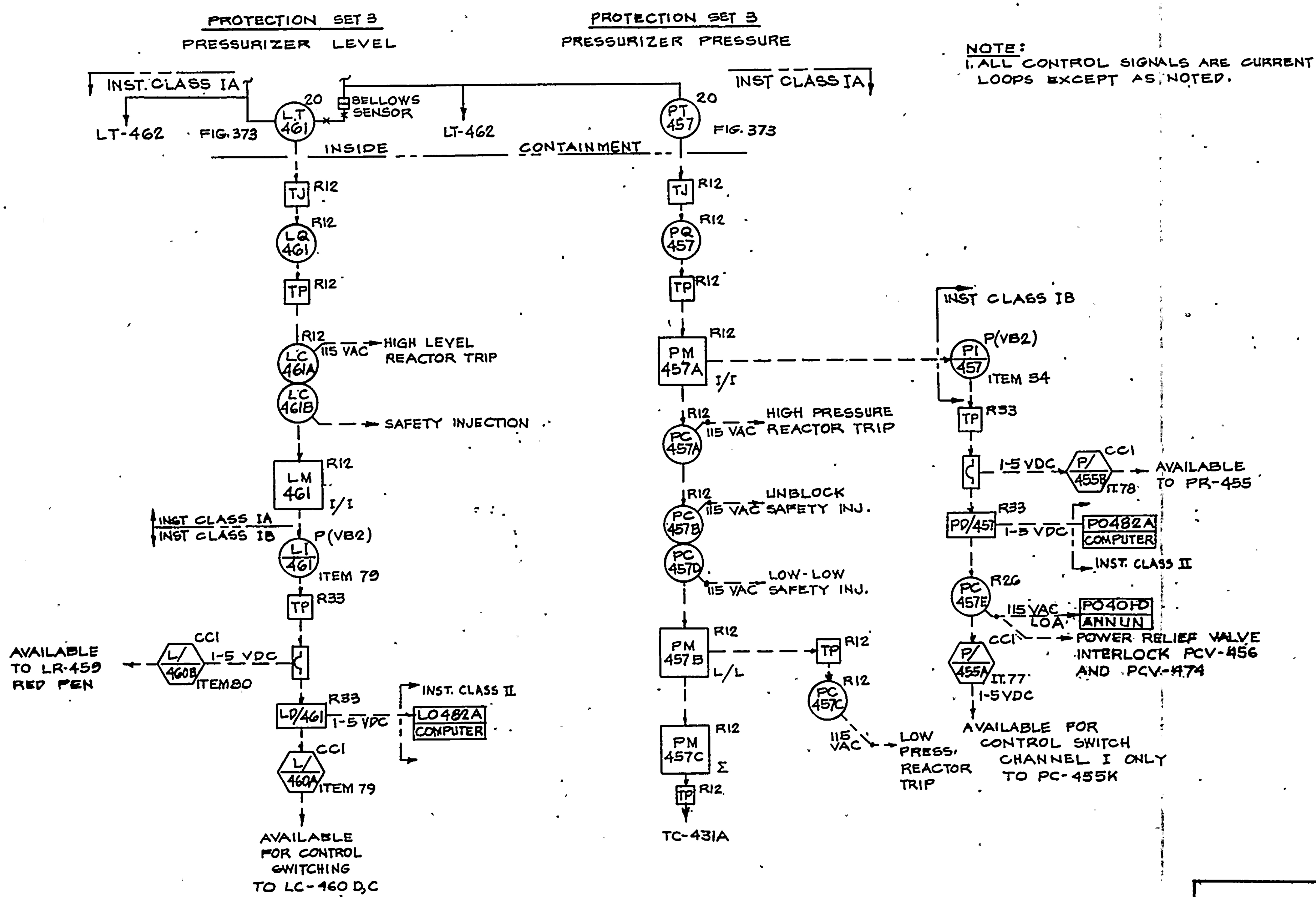
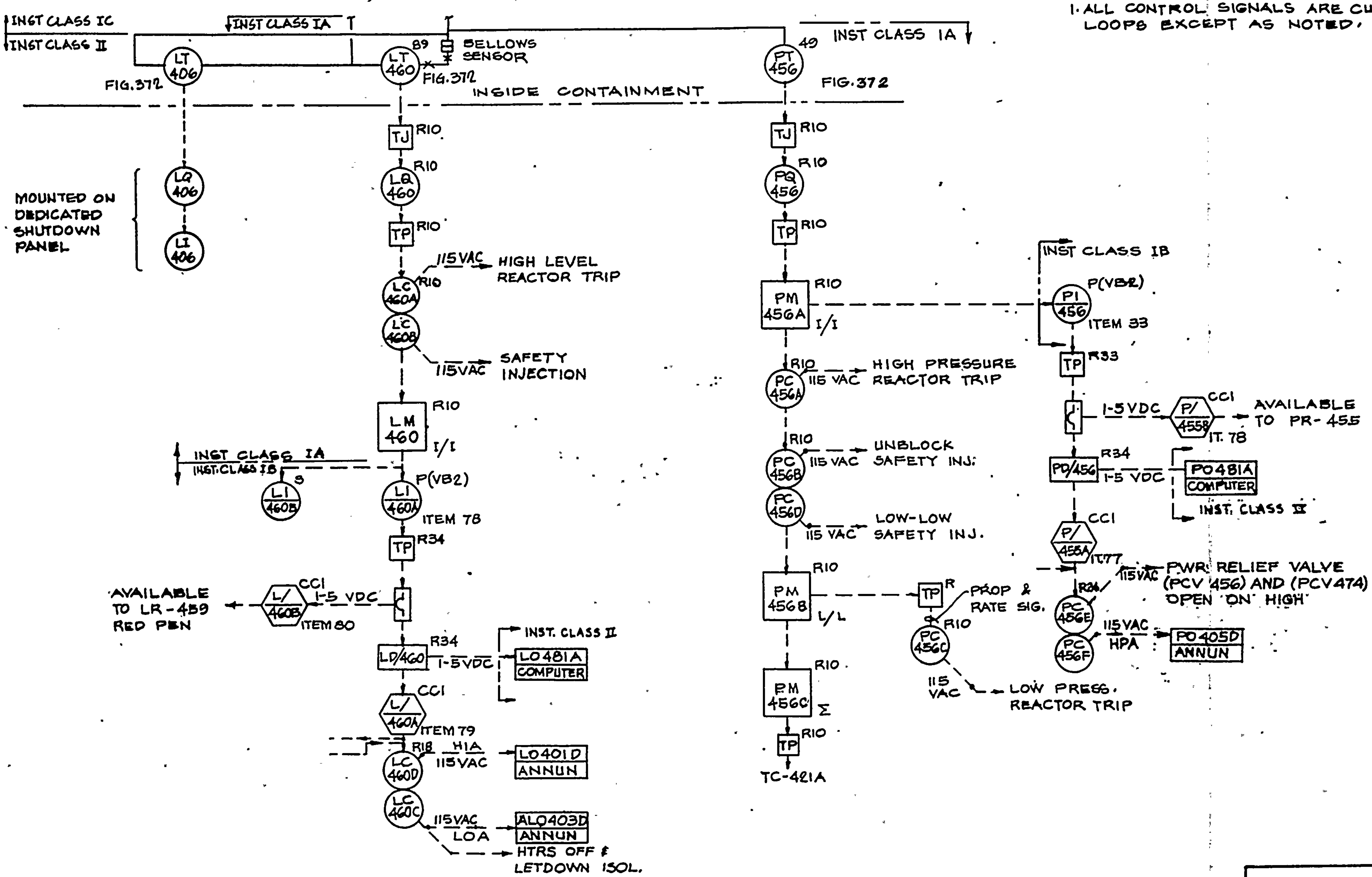
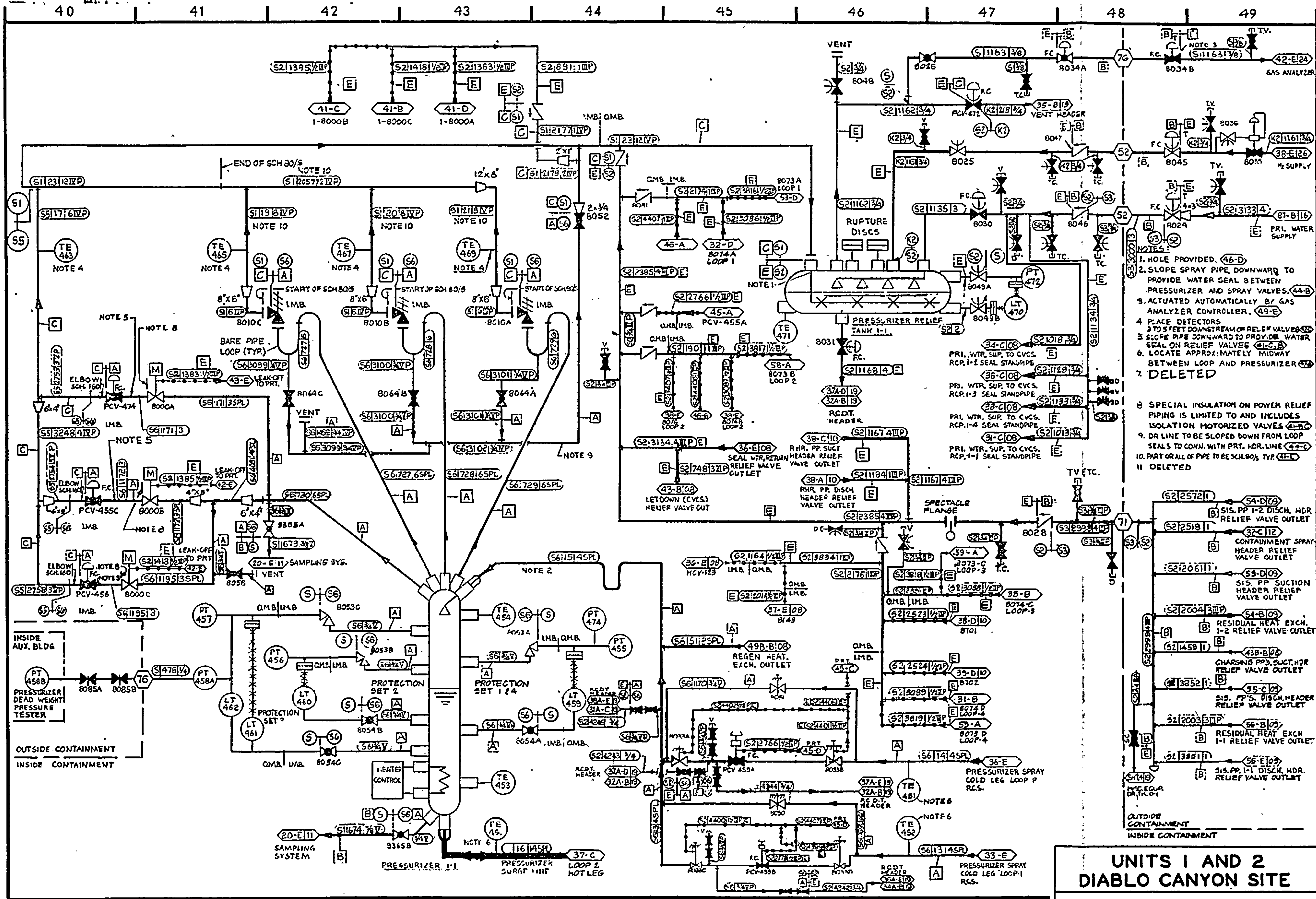


FIGURE 2.1.1-9
PRESSURIZER LEVEL AND PRESSURE, SET 3



NOTE:
1. ALL CONTROL SIGNALS ARE CURRENT LOOPS EXCEPT AS NOTED.

FIGURE 2.1.1-10
PRESSURIZR LEVEL AND PRESSURE SET 2



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.2-07 (Sheet 2 of 4)
PIPING SCHEMATIC
REACTOR COOLANT SYSTEM

3. Valve Operators - All Westinghouse-supplied safety-related electric motor valve operators which are required to operate in the post-accident containment environment (regardless of specified length of operation time) have class H insulation. These safety related valves are identified in Westinghouse Submittal NS-CE-847 dated November 24, 1975 (Eicheldinger to Vassallo)[6]. Westinghouse conducted environmental tests on a class H - insulated valve motor operator similar to those being used in the Diablo Canyon plant. The results of these tests demonstrate that the equipment will perform its required function in the post-LOCA environment[1]. As part of the Westinghouse Supplemental Program[4], Westinghouse provided additional information regarding the qualification of class H insulated valve motor operators in Reference 4. The operability, under severe accident conditions, of Westinghouse supplied solenoid valves is demonstrated by a Failure Modes and Effects Analysis and is documented in NS-CE-755[7]. 55

Electric valve operators for the balance of plant are qualified to withstand the same environment as that for which the Westinghouse-supplied operators are qualified.

4. Wire and Cable - Vital wire and cable of the type that is installed in the containment has undergone tests simulating conditions during a LOCA.

Low voltage power and control wire and cable, except for power cable to the containment fan cooler motors, have had the following tests made by the cable manufacturer:

Time of Cycle	Pressure, psig	Temp., Degrees F	Minimum Insulation Resistance, ohm-ft	Test Voltage
Initial	0	65	2.7×10^{11}	2200 v a-c
End of First Hour	90	333	5.4×10^{8}	"
End of Third Hour	60	290	2.04×10^{8}	"
End of 24 Hours	20	260	1.13×10^{8}	"
End of Fifth Day	7	230	1.26×10^{8}	"

The above test was performed after a total radiation dose of 5.5×10^7 R. No failures were encountered.

Cable for containment fan cooler motors was tested by Pacific Gas and Electric Company at its Emeryville Laboratory under conditions of high humidity, pressure and temperature. Temperatures up to 302°F at a steam pressure of 50 psig for a period of approximately 2 hours, followed by a temperature of 263°F at a pressure of 20 psig for 20 hours were applied. The insulation resistance was measured every half hour and was above 1×10^7 ohm-ft each time. This cable was not exposed to radiation before being subjected to the steam environmental test. Since this cable has the same Hypalon jacket as that used for the low voltage power and control cable, and since the insulation is Kapton with a gamma radiation resistance of 10^9 R, a special radiation test for this cable makeup is not needed. Vital instrumentation and thermocouple extension wire was supplied by three different manufacturers, utilizing three different types of insulation. All were type tested and passed simulated LOCA tests.

5. <u>Electrical Connections</u> - All splices and terminal connections made in the containment for safety related electrical circuits are low voltage and were made using a polyolefin heat shrinkable material, which has been type tested for loss of coolant accident conditions before and after exposure to nuclear radiation. The test sequence for assembled low voltage splices consisted of:

a. Heat aging at 121°C for 168 hours in a forced air oven.

b. Irradiation of splices with cobalt 60 gamma radiation at 0.27 Mrads per hour to total doses of 100 and 200 Mrads.

c. Subjecting irradiated assemblies maintained at maximum rated voltage to LOCA tests in a pressurized autoclave according to the following schedule:

(1) 5 hours at 360°F, 70 psig steam
(2) 6 hours at 320°F, 70 psig steam
(3) 24 hours at 250°F, 21 psig steam, 0.2% boric acid spray, buffered to pH of 10.
(4) 12 days at 221°F, 2.5 psig steam

Tests results show that when properly assembled, the splices have successfully withstood DBE and LOCA tests and will remain functional during a LOCA accident.

6. Electrical Penetrations - Low voltage power and control, medium voltage power, and shielded signal electrical penetrations were all successfully prototype tested by the manufacturer at a temperature of 281°F, 63 psig, and a relative humidity of 90 to 100 percent for a duration of 240 hours. Leakage rates were all less than 1×10^{-6} cc/sec.

7. Instrumentation - The instruments which are inside the containment and are required for action during and after the LOCA (pressurizer pressure, pressurizer level, and containment sump level) were environmentally qualified by test (testing described in Reference 1) to assure performance of their protective function. Supplemental testing, at conditions more severe than reported in Reference 1, did not confirm the long term survival capability of the instruments in a hostile environment. As a result, the company will replace those instruments required for long term survival in these severe environments (pressurizer level, wide range reactor coolant system pressure, narrow range steam generator level, and containment sump level) with instruments which have passed the supplemental testing (Ref. 8). This will occur during the first planned outage of one week or more after they become available. 55

Section 2.1.2 - Performance Testing for BWR and PWR Relief and Safety Valves

Task Force Position

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents. The licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports as well as the valves themselves. (Category A: Submit the program description and schedule prior to OL, or January 1, 1980, whichever is later, and complete the test program by July 1981).

PG&E Response

A full scale prototypical qualification testing program of relief and safety valves undergoing expected operating conditions for design basis transients and accidents involving two-phase slug flow and subcooled liquid flow will be undertaken on an industry-wide basis rather than by individual licensees. The Electric Power Research Institute (EPRI) has begun preliminary investigations into a relief and safety valve qualification testing program. PG&E will assist EPRI in their effort by providing financial support and technical assistance as requested. A program description and schedule for the qualification testing program for the Diablo Canyon safety and relief valves will be submitted prior to January 1, 1980, or OL, whichever is later. The testing program will be completed by July, 1981. Depending upon projected schedules, this effort will be done in conjunction with EPRI, the Westinghouse Plant Owners Group, or if necessary to meet the commitment date by PG&E alone.

Section 2.1.2 (Cont'd)

PG&E Response (Cont'd)

In addition to the valve qualification program, power operated relief valve control circuitry will be upgraded as specified in Section 2.1.1. Relief and safety valve downstream piping and supports will be reviewed and, if necessary, modified to accommodate the loadings expected for the design basis transients and accidents that result in two-phase slug flow and subcooled liquid flow. Completion of this work is necessarily dependent upon completion of the qualification program.

Section 2.1.3.a - Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWR's and BWR's

Task Force Position

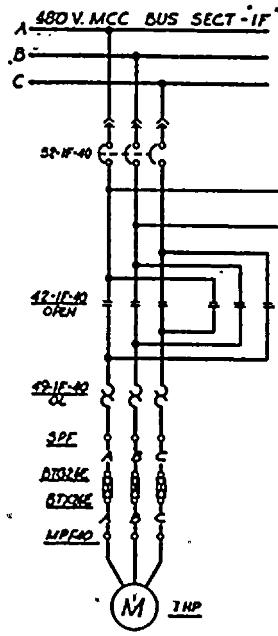
Reactor system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

(Category A: Implementation complete prior to OL or January 1, 1980, whichever is later.)

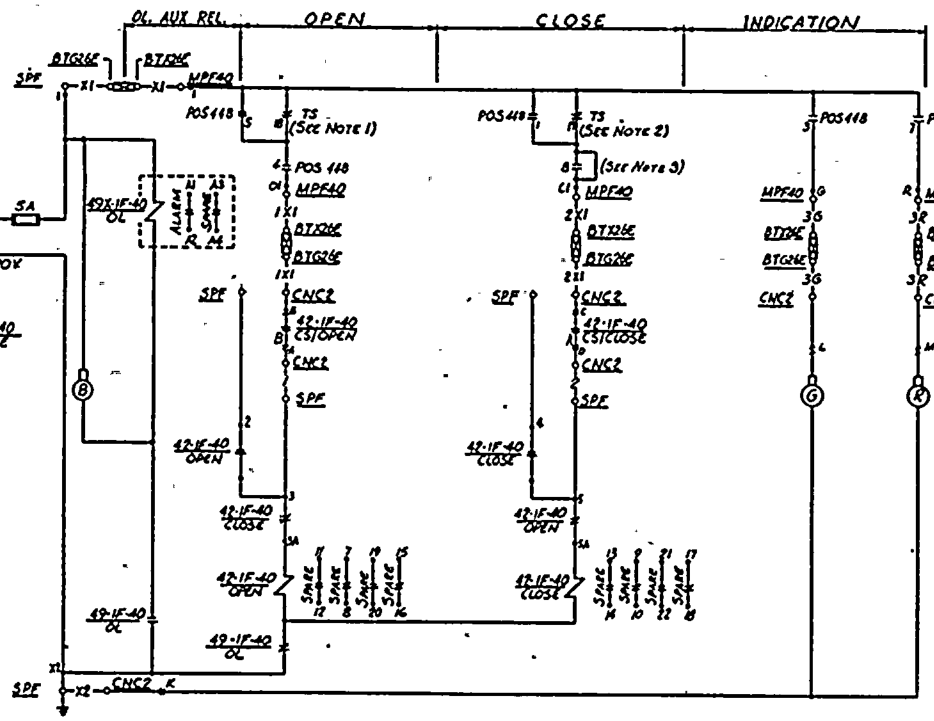
PG&E Response

The pressurizer PORV's presently have both open and close limit switches which control indicating lights mounted at their respective control switches on the main control board. The limit switches are snap acting, positive throw switches mounted on the valve yokes. They are operated by the actual valve stem motion. The indication circuits are powered from the station batteries. All devices have been seismically qualified to meet the postulated Hosgri Earthquake. (See attached FSAR Figure 7.3-21, Sheet 1.)

The pressurizer safety valves do not presently have position indication. Several means for detecting safety valve position are currently being considered. For example, an acoustic emission device has been tested by EPRI for this application. We will add an appropriate device as soon as is practical following its commercial availability.



THREE LINE DIAGRAM

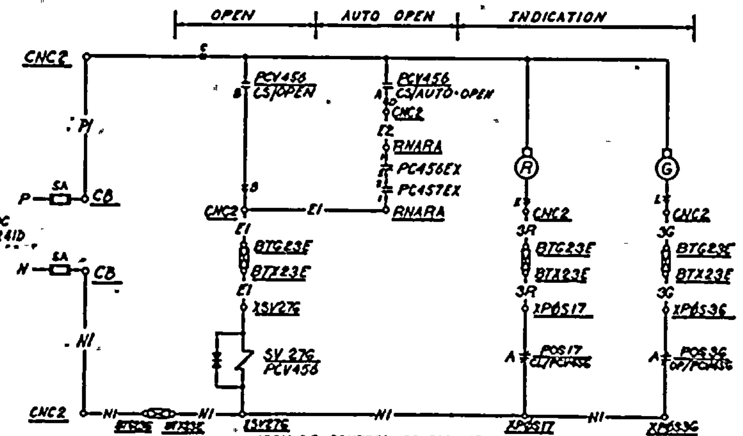


120V AC CONTROL SCHEMATIC

8000A PRESSURIZER POWER RELIEF STOP VALVE 11 - CKT No F40P00

PRESSURIZER RELIEF VALVES SIMILAR EXCEPT AS SHOWN IN TABLE

VALVE No.	DESCRIPTION	CIRCUIT BKR No	CIRCUIT No.	BUS SECT	NORMAL POSITION	ENGINEER SAFETY VITAL FUNCTION	VALVE POSITION SWITCH-(POS) POS No.	ALARM CKT No OVER-CURRENT	PENETR No	MOTOR No
8000A	Pressurizer Pwr Relief Stop Vlv 11	52-1F-40	F40P00	1F	Open	—	448	A14408DF	26E	MPF40
8000B	12	52-1G-46	G46P00	1G		—	455	A14308DG	19E	MPG46
8000C	13	52-1H-33	H33P00	1H		—	462	A14208DH	12E	MPH33

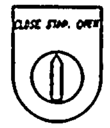


125V DC CONTROL SCHEMATIC

PCV456 PRESSURIZER POWER RELIEF VALVE - CKT. No RP053

PRESSURIZER POWER RELIEF VALVES SIMILAR EXCEPT AS SHOWN IN TABLE

Valve No.	DESCRIPTION	D.C. Ckt. No.	Circuit No.	Valve Loc.	Sol. Vlv. No.	Sol. Vlv Loc	Normal Position	Engineer Safety Vital Function	Valve Position Switch (POS) Close	Valve Position Switch (POS) Open	Pressure Control Aux Rel. Loc. of Relays RNARA Device	Contact	Device	Contact	Penn. No.
PCV456	Pressurizer Power Relief Valve	241D	RP053	1PCV456	276	XSV276	Closed	—	17	36	PC456EX	1—2	PC457EX	2—1	23E
PCV474		241D	RP057	1PCV474	474	XSV474		—	18	37	PC457EX	3—4	PC456EX	4—3	
PCV455C		241D	RP049	1PCV455C	455C	XSV455C		—	16	35	PC455EX	1—2	PC474BX	2—1	



CONTACTS		POSITION CLOSE	STOP	OPEN	REMARKS
—‖—	A	X			This Dwg
—‖—	B			X	This Dwg

CONTROL SWITCH $\frac{42\text{-}1F\text{-}40}{CS}$, $\frac{42\text{-}1G\text{-}46}{CS}$, $\frac{42\text{-}1H\text{-}33}{CS}$

USED AT LOCATION CNC2

NO SPRING RETURN

CONTACTS			FUNCTION CONTACTS 1-8	FUNCTION CONTACTS 9-16
—‖—	1	9	Bypass Intlk	Spare
—‖—	2	10	Spare	″
—‖—	3	11	Green Light	″
—‖—	4	12	Open Limit	″
—‖—	5	13	Bypass Intlk	″
—‖—	6	14	Spare	″
—‖—	7	15	Red Light	″
—‖—	8	16	Close Limit (See Note 3)	″

VALVE POSITION: FULL OPEN — FULL CLOSED

TYPICAL VALVE POSITION SWITCH CONTACT DEVELOPMENT

CONTACTS		POSITION CLOSE	AUTO	OPEN	REMARKS
—‖—	A		X	X	This Dwg.
—‖—	B			X	This Dwg.

CONTROL SWITCH $\frac{PCV456}{CS}$, $\frac{PCV474}{CS}$, $\frac{PCV455C}{CS}$

USED AT LOCATION CNC2

NO SPRING RETURN

CONTACTS		VALVE POSITION CLOSE	OPEN & INTERM.	REMARKS
—‖—	A		X	RED LT.
—‖—	B	X		SPARE
—‖—	C		X	SPARE
—‖—	D	X		SPARE

VALVE CLOSE POSITION SWITCH

POS 16, 17, & 18

CONTACTS		VALVE POSITION CLOSE & INTERM.	OPEN	REMARKS
—‖—	A	X		GREEN LT.
—‖—	B		X	SPARE
—‖—	C	X		SPARE
—‖—	D		X	SPARE

VALVE OPEN POSITION SWITCH

POS 35, 36, & 37

TABLE OF DEVICES

DEVICE No	FUNCTION	RATING	MFR.	TYPE	CAT/DWG.*	REMARKS
42-1F-40, 1G-46, 1H-33	480V AC Motor Starter Reversible, Size 1	—	W.E.Corp	—		
49-1F-40, 1G-46, 1H-33	Motor Overload Relay	—	W.E.Corp	—	—	Part Of Motor Starter
49X-1F-40, 1G-46, 1H-33	Motor Overload Auxiliary Relay	—	W.E.Corp	—	—	Part Of Motor Starter
52-1F-40, 1G-46, 1H-33	480V. AC Air Circuit Breaker	—	W.E.Corp	—		
FCV	Flow Control Valve	—	—	—	—	See Table For Valve No's
PC456EX Etc.	Pressure Controller Aux. Rel.					
POS	FCV Position Switch	—	—	—	—	See Table For Sw. No's
SV...	Solenoid Valve					
TS	Torque Switch					

EQUIPMENT LOCATION NUMBERS

SPF - 480V Motor Control Center Bus Section 1F
MPF40 - Motor Operated Valve 8000A
CNC2 - Control Board, Reactor Coolant
BTG26E - Terminal Box, Area G, No 26E
RNARA - Rack Nuclear Aux. Relay Cubicle A

NOTES:

1 - Switch Opens On Mechanical Torque During Opening Cycle (18)

2 - Switch Opens On Mechanical Torque During Closing Cycle Or Fully Closed Valve (17)

3 - Field To Install Jumper If Torque Seating Is Required

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 7.3-21 (Sheet 1 of 2)
SCHEMATIC DIAGRAM - REACTOR COOLANT MOTOR OPERATED VALVES

Amendment 78 March 1979

Section 2.1.3.b - <u>Instrumentation for Detection of Inadequate Core Cooling in PWR's</u>

Task Force Position 1.

Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of this appendix).

In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

(Category A - Implementation complete prior to OL or January 1, 1980, whichever is later.)

PG&E Response

The Working Group on Procedures of the Westinghouse Owners Group is currently engaged in the development of generic emergency and abnormal operating procedure guidelines for all Westinghouse plants. The first of these guidelines, involving accident diagnostics and LOCA response, were submitted to the NRC as part of WCAP-9600, "Small Break Analysis for Westinghouse NSSS Systems." As part of this effort, guidelines will be developed to enable the operator to recognize inadequate core cooling with currently available instrumentation. The guidelines so developed will be incorporated into the Diablo Canyon operating and emergency procedures. Because Diablo Canyon is basically a standard four loop design it is not anticipated that translation of the guidelines into plant specific procedures will entail any significant deviations from the guidelines.

Section 2.1.3.b (Cont'd)

PG&E Response (Cont'd)

The plant process computer is being programmed to provide an indication to the operator of margin to coolant saturation pressure. The computer will look at an operator selected incore thermocouple and will calculate saturation pressure for this temperature. The computer will then compare this calculated saturation pressure with actual RCS pressure to determine the margin to saturation. A low ΔP will alarm, and the operator has the option of displaying and trending either saturation pressure, ΔP, or rate of change of ΔP. Operating procedures will indicate that this information is to be used in conjunction with other monitored variables to assure proper subcooling. This will be completed prior to OL or January 1, 1980, whichever is later.

Task Force Position 2.

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

(Category B: Implementation will be completed by January 1, 1981.)

PG&E Response

Our recommendations for additional instrumentation or controls to indicate inadequate core cooling will be formulated after the analytical work in this area currently being performed by the Westinghouse Owners Group is complete. At the present time, Westinghouse has proposed a series of studies involving postulated circumstances which could result in inadequate core cooling, and will be meeting with NRC representatives in the near future to assure that the scope of this effort satisfies NRC concerns in this area.

Section 2.1.4 - Containment Isolation Provisions for PWR's and BWR's

Task Force Position 1.

All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

(Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

There are two phases of containment isolation at Diablo Canyon. Phase A isolates all nonessential process lines but does not affect safety injection, containment spray, component cooling water supplied to the reactor coolant pumps and containment fan coolers, and steam and auxiliary feedwater lines. Phase B isolates all process lines except safety injection, containment spray, auxiliary feedwater, and the containment fan coolers component cooling water system.

Phase A isolation is initiated by high containment pressure, high differential pressure between steam lines, low pressurizer pressure, or manual initiation. Phase B isolation is initiated by high-high containment pressure or manual initiation.

This system fully complies with Section II.6 of the SRP 6.2.4. Compliance with all remaining sections of SRP 6.2.4 is documented in Section 6.2.4 of the Diablo Canyon FSAR.

Task Force Position 2.

All plants shall give careful reconsideration to the definition of essential and nonessential systems, shall identify each system determined to be essential, shall identify each system determined to be nonessential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the reevaluation to the NRC.

Section 2.1.4 (Cont'd)

Task Force Position 2. (Cont'd)

(Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

PG&E has three levels of containment process penetrations. These are defined as:

1. "Nonessential" process lines are defined as those who do not increase the potential for damage for in-containment equipment when isolated. These are isolated on Phase A isolation.

2. "Essential" process lines are those providing cooling water and seal water flow through the reactor coolant pumps. These services should not be interrupted while the reactor coolant pumps are operating unless absolutely necessary. These are isolated on Phase B isolation.

3. Safety system process lines are those required to perform the function of the Engineered Safety Features System.

The attached Table 2.1.4-1 provides the identification of nonessential, essential, and safety systems penetrating containment.

Task Force Position 3.

All nonessential systems shall be automatically isolated by the containment isolation signal.

(Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

Section 2.1.4 (Cont'd)

PG&E Response

All nonessential systems use either manually sealed closed valves or else the valves are automatically isolated on a Phase A containment isolation signal. Additionally, all essential systems (defined in response 2 above) are automatically isolated on a Phase B containment isolation signal.

Task Force Position 4.

The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

The Diablo Canyon design complies with the position. There are two basic actuator methods, one for motor operated valves and one for air operated valves. For a motor operated valve (MOV), the trip signal operates the close coil until the valve closes. If the trip signal is removed, the valve will not operate unless the open coil is energized by an explicit operator action. The attached FSAR Figure 7.3-35 shows a typical MOV circuit.

For an air operated valve (AOV), the control switch must be held in the open position to open the valve. Once it is opened, a stem mounted position switch on the valve closes thus closing a latch-in circuit which holds the valve open when the control switch spring-returns to neutral. The isolation signal contacts are in this latch-in circuit. When an isolation signal is generated, the circuit opens and the valve is de-energized and closes. As soon as the valve begins to close, the position switch opens so that the circuit will remain open and the valve will remain closed even after the isolation signal is reset. The attached FSAR Figure 7.3-43 shows a typical AOV circuit.

TABLE 2.1.4-1
CONTAINMENT PENETRATIONS

Penetration Numbers	System	Priority	Safety Function (For Essential and Safety Systems)
1, 2, 3, 4	Feedwater	Nonessential/ Safety	Feedwater is nonessential. Auxiliary feedwater is necessary for safe shutdown.
5, 8	Main Steam	Nonessential/ Safety	Main Steam is nonessential.10% steam dump is required for safe shutdown.
6, 7	Main Steam	Nonessential/ Safety	Main Steam is nonessential.10% steam dump and auxiliary feedwater steam turbine are required for safe shutdown.
9-13	Component cooling water to Fan Coolers	Safety	Fan Coolers are required for post-accident containment cooling.
14-18	Component cooling water from fan coolers	Safety	Fan Coolers are required for post-accident containment cooling.
19	Component cooling water to Reactor Coolant Pumps	Essential	Reactor Coolant Pumps may be necessary for certain post-accident activities.
20	Component cooling water from Reactor Coolant Pumps	Essential	Reactor Coolant Pumps may be necessary for certain post-accident activities.
21	Component cooling water from Reactor Coolant Pumps	Essential	Reactor Coolant Pumps may be necessary for certain post-accident activities.
22	Component cooling water to excess letdown heat exchanger	Nonessential	

CONTAINMENT PENETRATIONS

Penetration Numbers	System	Priority	Safety Function (For Essential and Safety Systems)
23	Component Cooling Water to excess letdown heat exchanger	Nonessential	
24	Residual Heat Removal 1 Injection	Safety	Residual Heat Removal is required for cold shutdown.
25	Residual Heat Removal 2 Injection	Safety	Residual Heat Removal is required for cold shutdown.
26	Residual Heat Removal Hot Leg Injections	Safety	Residual Heat Removal is required for cold shutdown.
27	Reactor Coolant System Loop 4 Recirculation	Safety	Residual Heat Removal is required for cold shutdown.
28	Containment Sump Recirculation	Safety	Sump Recirculation is required for post-accident operation.
29	Containment Sump Recirculation	Safety	Sump Recirculation is required for post-accident operation.
30	Containment Spray System	Safety	Containment Spray is a safety function.
31	Containment Spray System	Safety	Containment Spray is a safety function.
32	Spare		
33	Safety Injection System	Safety	Safety Injection is a safety function.
34	Safety Injection System	Safety	Safety Injection is a safety function.

TABLE 2.1.4-1 (Cont'd)
CONTAINMENT PENETRATIONS

Penetration Numbers	System	Priority	Safety Function (For Essential and Safety Systems)
35	Regenerative Heat Exchanger to letdown Heat Exchanger	Nonessential	
36	Regenerative Heat Exchanger	Nonessential	
37, 38, 39, 40	Steam Generator Blowdown	Nonessential	
41, 42, 43, 44	Reactor Coolant Pump Seal Water Supply	Essential	The Reactor Coolant Pumps may be necessary in certain post-accident activities.
45	Reactor Coolant Pump Seal Water Return	Essential	The Reactor Coolant Pumps may be necessary in certain post-accident activities.
46	Refueling Canal Recirculation	Nonessential	
47	Refueling Canal Recirculation	Nonessential	
48	Spare		
49	Containment Sump	Nonessential	
50	Reactor Coolant Drain Tank Discharge	Nonessential	
51	Reactor Coolant Drain Tank Vent	Nonessential	
51	Reactor Coolant Drain Tank to Gas Analyzer	Nonessential	

TABLE 2.1.4-1 (Cont'd)
CONTAINMENT PENETRATIONS

Penetration Numbers	System	Priority	Safety Function (For Essential and Safety Systems)
51	Safety Injection System Test Line	Nonessential	
51	Nitrogen Supply Header to Accumulators	Nonessential	
52	Pressurizer Relief Tank Nitrogen Supply	Nonessential	
52	Pressurizer Relief Tank Makeup	Nonessential	
52	Reactor Coolant Drain Tank Nitrogen Supply	Nonessential	
52	Steam Generators Nitrogen Supply	Nonessential	
52, 53, 59, 76	Containment Pressures	Safety	Containment Pressure signal is required for certain safeguards actuation and post-accident monitoring.
53 (4 Lines)	Steam Generator Blowdown Sample	Nonessential	
54	Instrument Air Header	Nonessential	
55	Spare		
56	Service Air Header	Nonessential	
57	Spare		

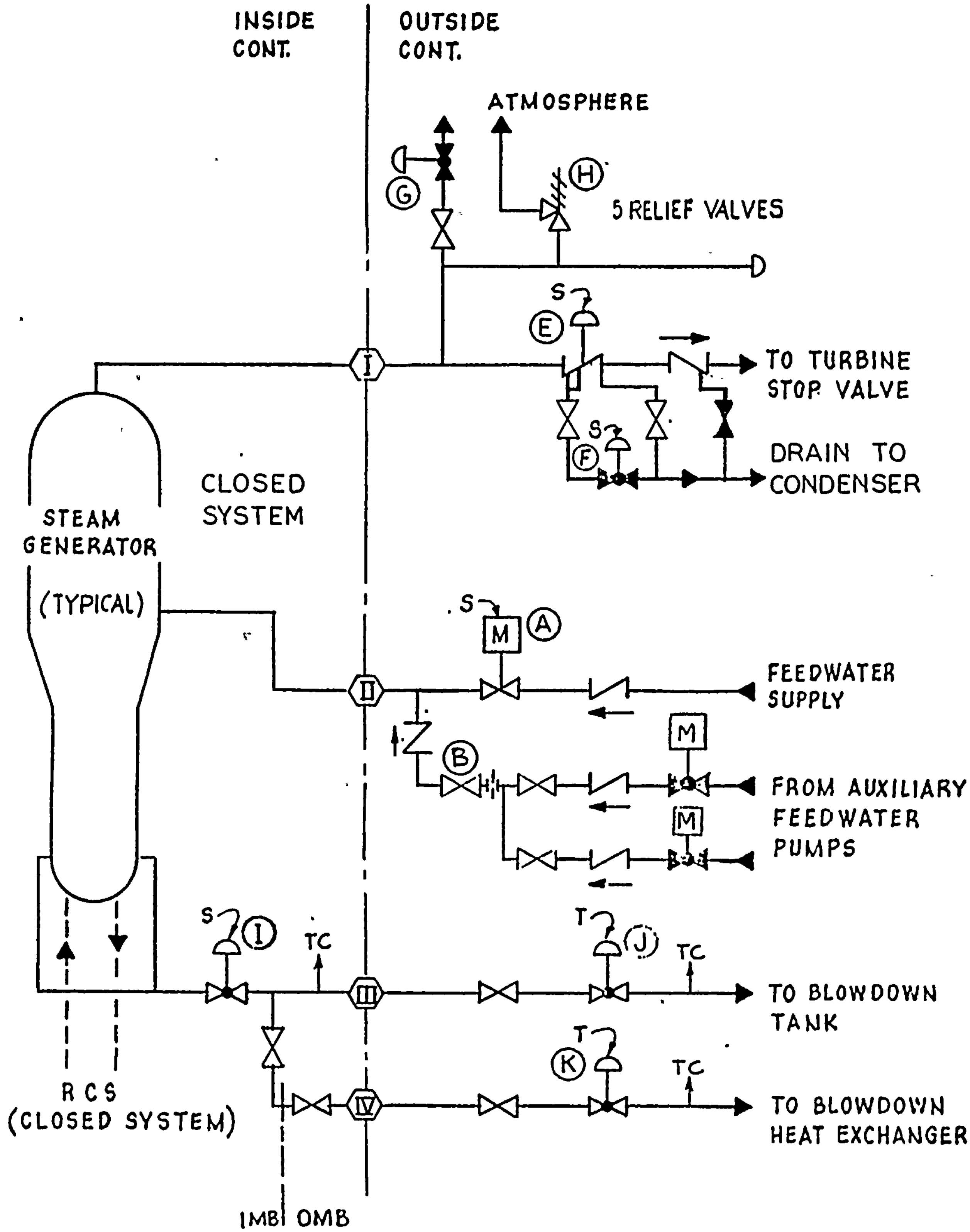
TABLE 2.1.4-1 (Cont'd)
CONTAINMENT PENETRATIONS

Penetration Numbers	System	Priority	Safety Function (For Essential and Safety Systems)
58	Not used		
59	Pressurizer Liquid Sample	Nonessential	
59	Hot Leg Sample	Nonessential	Although this is nonessential, motive power is available to operate the valves and samples can be taken by a continuous manual override until the sampling is complete.
59	Accumulator Sample	Nonessential	
59	Spare		
60	Not used		
61	Containment Purge Supply	Nonessential	
62	Containment Purge Exhaust	Nonessential	
63	Containment Pressure and Vacuum Relief	Nonessential	
64	Fuel Transfer	Nonessential	
65	Personnel Hatch	Nonessential	
66	Emergency Personnel Hatch	Nonessential	
67	Equipment Hatch	Nonessential	
68	Containment Air Sample	Nonessential	

TABLE 2.1.4-1 (Cont'd)
CONTAINMENT PENETRATIONS

Penetration Numbers	System	Priority	Safety Function (For Essential and Safety Systems)
69	Containment Air Sample	Nonessential	
70	Auxiliary Steam Supply	Nonessential	
71	Relief Valves	Nonessential	
72, 73, 74	Spare		
75	Safety Injection System Pump 2 Discharge	Safety	Safety Injection is a safety function.
76	Pressurizer Relief Tank Gas Analyzer	Nonessential	
76	Pressurizer Steam Sample	Nonessential	
76	Deadweight	Nonessential	
77	Safety Injection System Pump 1 Discharge	Safety	Safety Injection is a safety function.
78	Spare		
79	Firewater	Nonessential	
80	Class I Air System	Nonessential	This is required for long-term hydrogen control, but can be opened by the operator to effect post-accident hydrogen control.
81	Spare		

Penetration Numbers	System	Priority	Safety Function (For Essential and Safety Systems)
82	Chilled Water Supply	Nonessential	
83	Chilled Water Return	Nonessential	
83	Hydrogen Purge Supply	Nonessential	This is required for long-term hydrogen control, but can be opened by the operator to effect post-accident hydrogen control.
84	Spare		



ALL SYSTEMS ARE DESIGN CLASS I

(I) MAIN STEAM (5, 8)

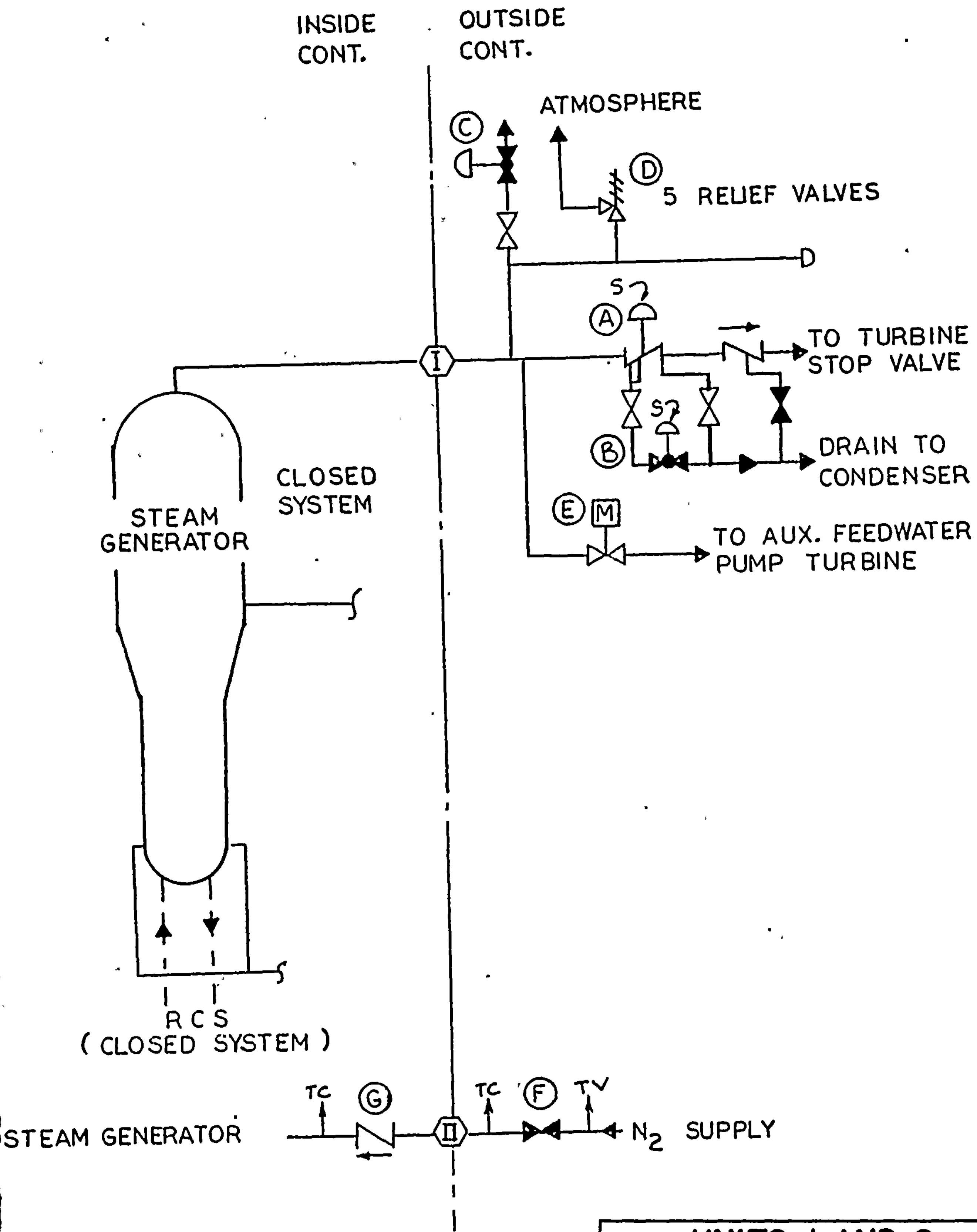
(II) FEEDWATER (1,2,3,4)

(III) STEAM GENERATOR BLOWDOWN (37, 38, 39, 40)

(IV) STEAM GENERATOR BLOWDOWN SAMPLE (53-3, 76)

UNITS 1 AND 2
DIABLO CANYON SITE

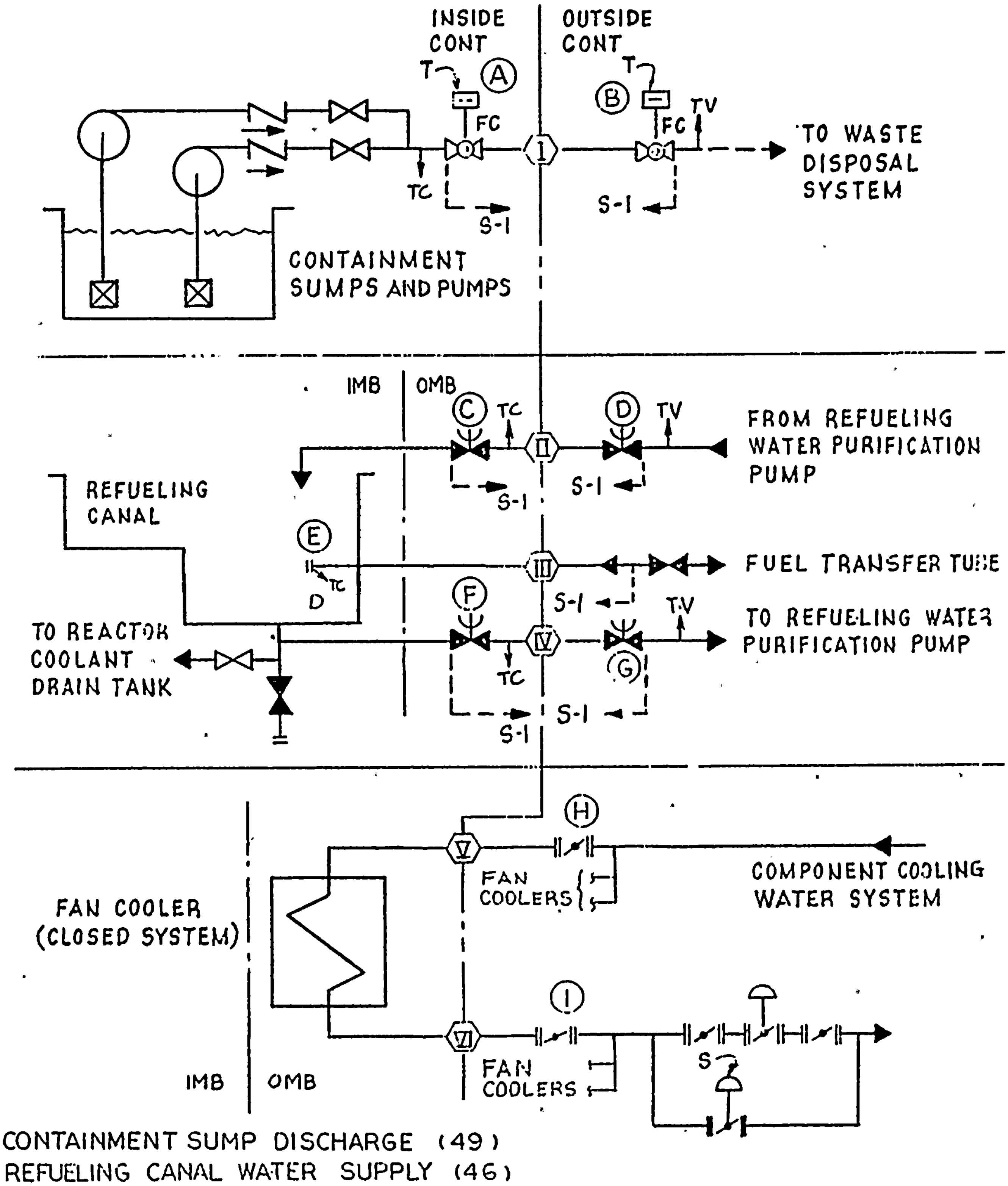
FIGURE 6.2-14
PENETRATION DIAGRAM
(SHEET 1 OF 23)



SYSTEMS ARE DESIGN CLASS 1
(I) MAIN STEAM (6,7)
(II) STEAM GENERATOR N_2 (52)

UNITS 1 AND 2 DIABLO CANYON SITE
FIGURE 6.2-14 PENETRATION DIAGRAM (SHEET 2 OF 23)

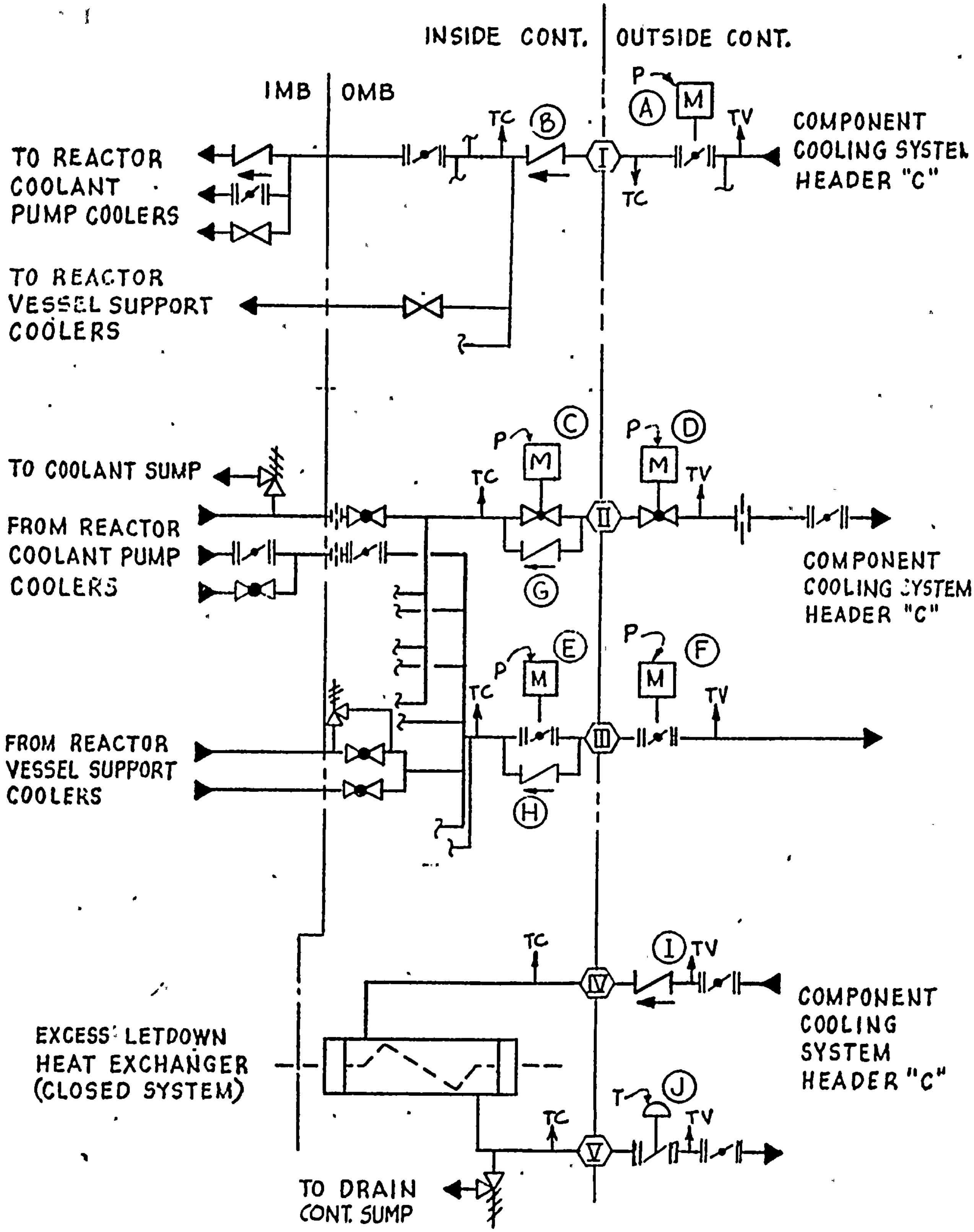
Amendment 36 October 1975



(I) CONTAINMENT SUMP DISCHARGE (49)
(II) REFUELING CANAL WATER SUPPLY (46)
(III) FUEL TRANSFER TUBE (64)
(IV) REFUELING CANAL DRAIN (47)
(V) FAN COOLER WATER SUPPLY (9-13)
(VI) FAN COOLER WATER RETURN (14-18)

UNITS 1 AND 2
DIABLO CANYON SITE

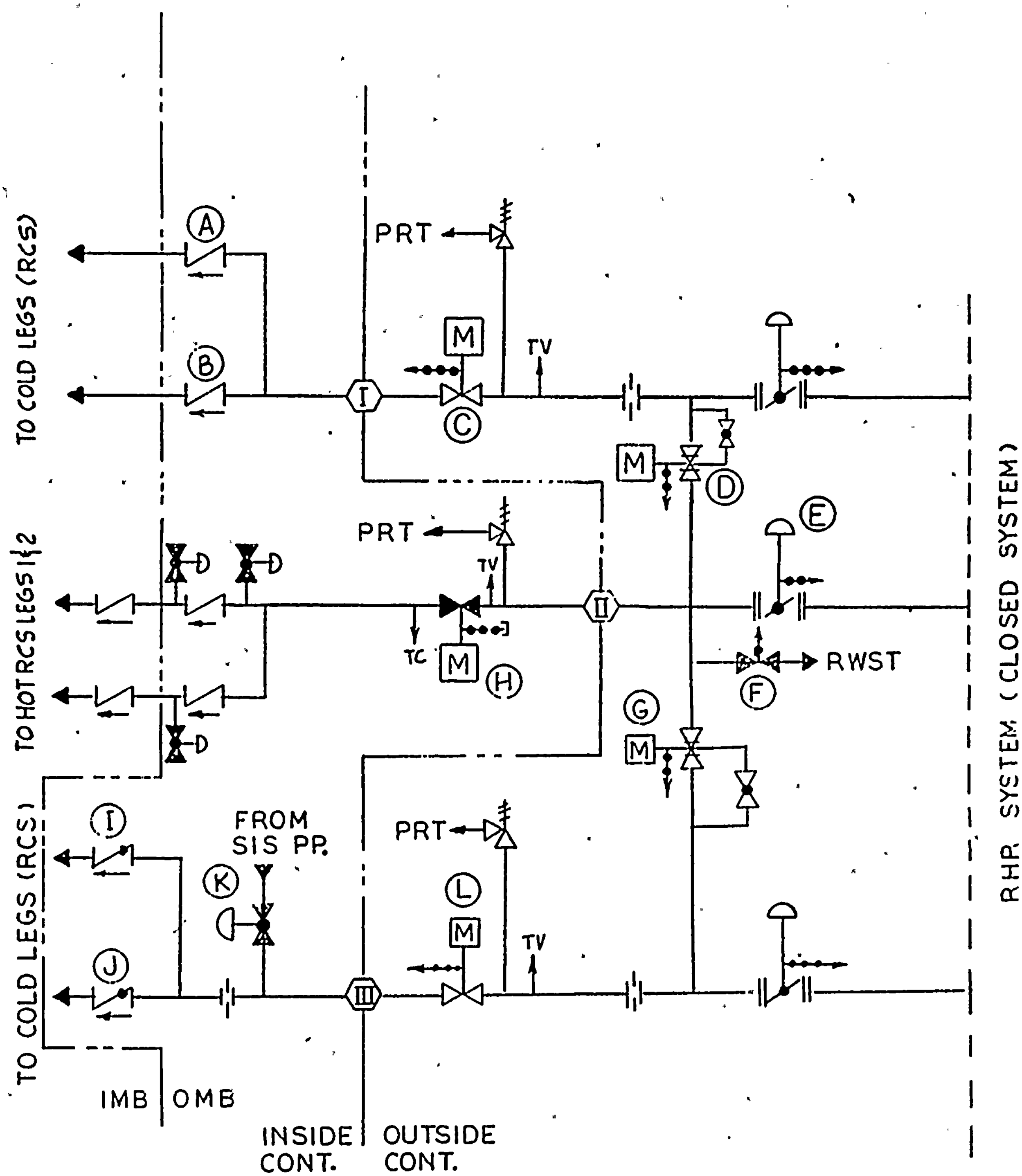
FIGURE 6.2-14
PENETRATION DIAGRAM
(SHEET 3 OF 23)



(I) CCW TO RCP (19)
(II) CCW FROM RCP (21)
(III) CCW FROM RCP (20)
(IV) CCW TO ELD HX (22)
(V) CCW FROM ELD HX (23)

UNITS 1 AND 2
DIABLO CANYON SITE

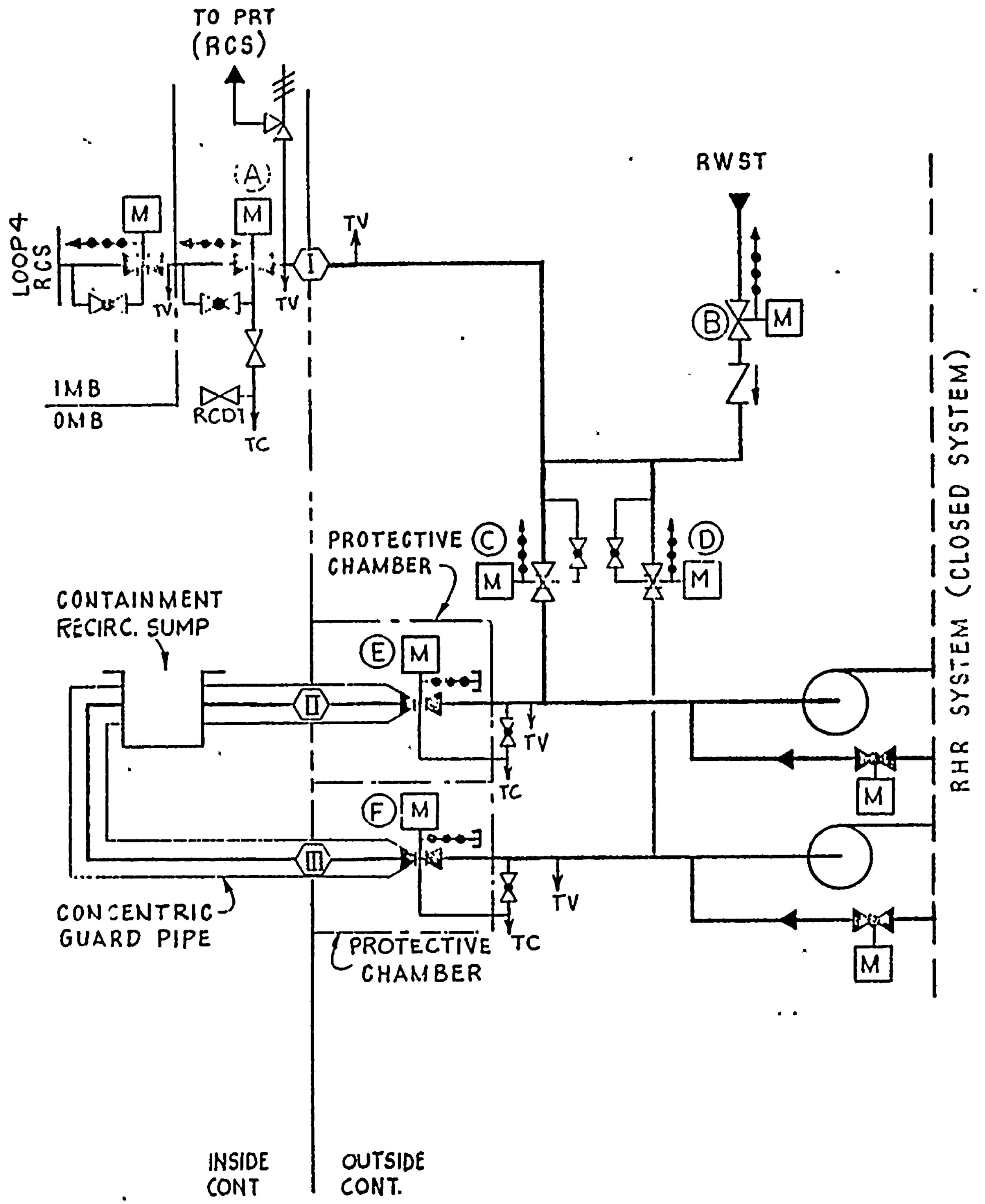
FIGURE 6.2-14
PENETRATION DIAGRAM
(SHEET 4 OF 23)



(I) RHR I INJECTION (24)
(II) RHR HOT LEG INJECTION (26)
(III) RHR 2 INJECTION (25)

UNITS I AND 2 **DIABLO CANYON SITE**
FIGURE 6.2-14 PENETRATION DIAGRAM (SHEET 5 OF 23)

Amendment 36 October 1975

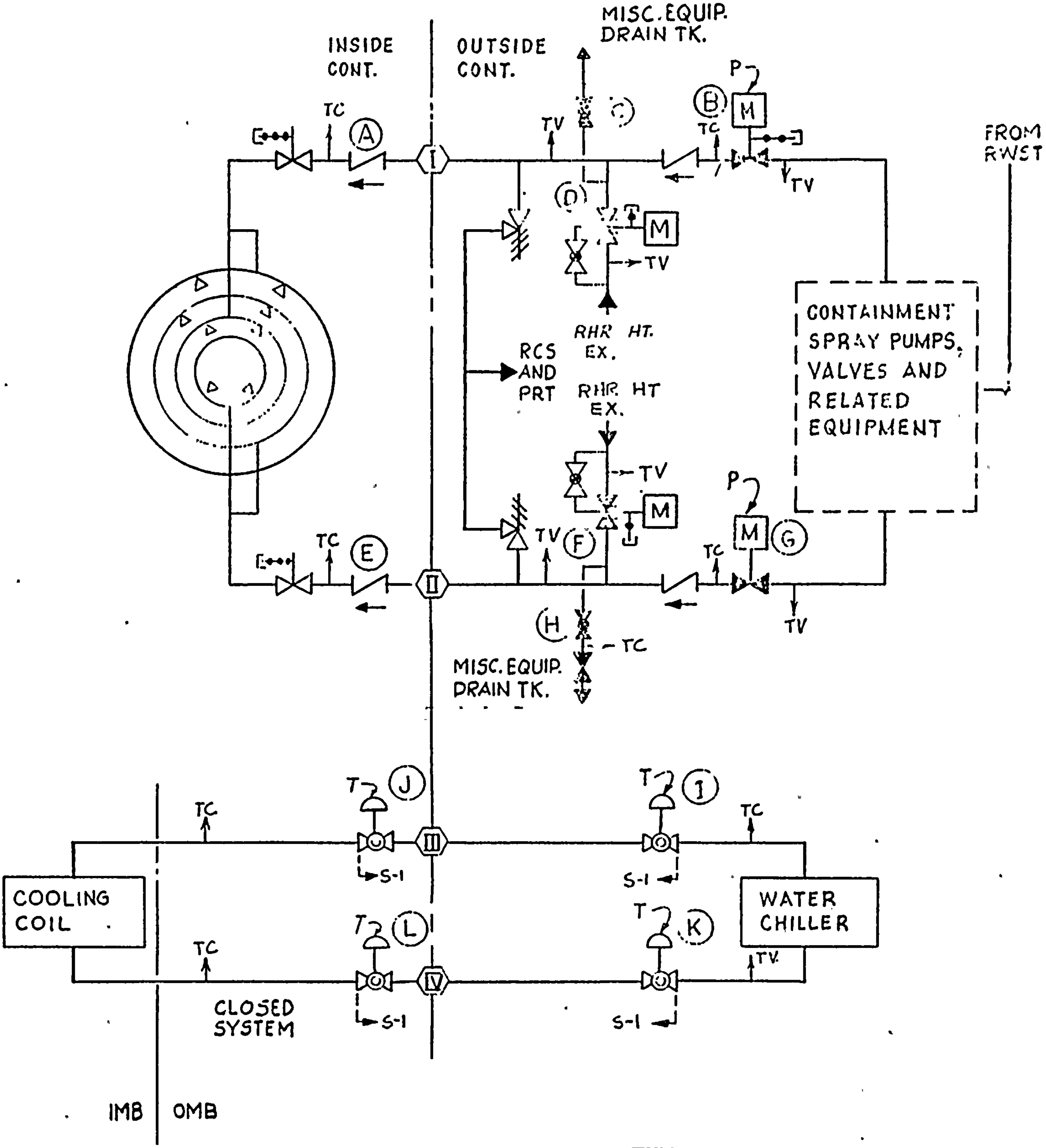


(I) RCS LOOP 4 RECIRC. (27)
(II) CONT. RECIRC. SUMP (28)
(III) CONT. RECIRC. SUMP (29)

UNITS I AND 2
DIABLO CANYON SITE

FIGURE 6.2-14
PENETRATION DIAGRAM
(SHEET 6 OF 23)

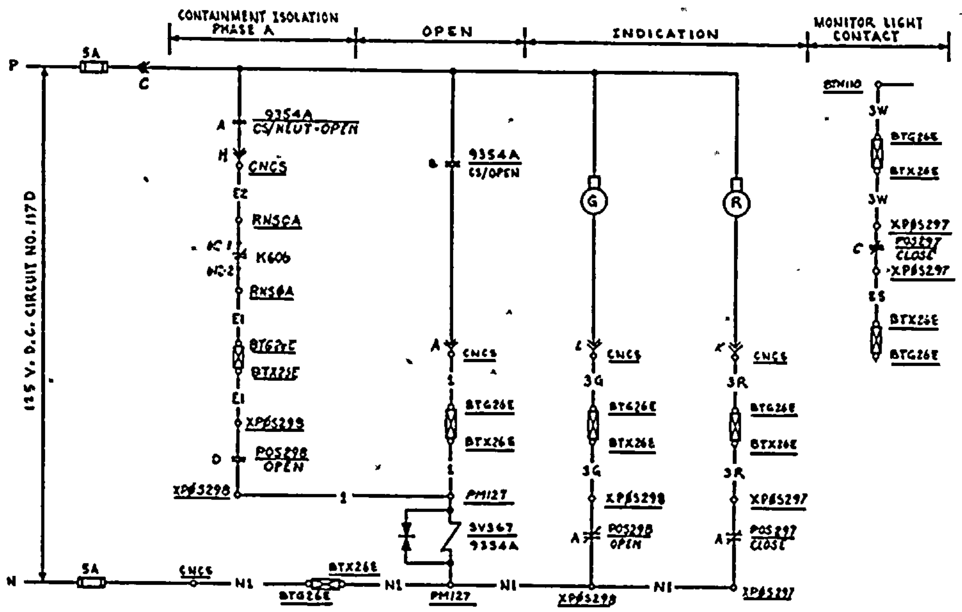
THIS SYSTEM IS DESIGN CLASS I



(I) CONT. SPRAY (30)
(II) CONT. SPRAY (31)
(III) CHILLED WATER SUPPLY (82)
(IV) CHILLED WATER RETURN (83)

UNITS 1 AND 2
DIABLO CANYON SITE

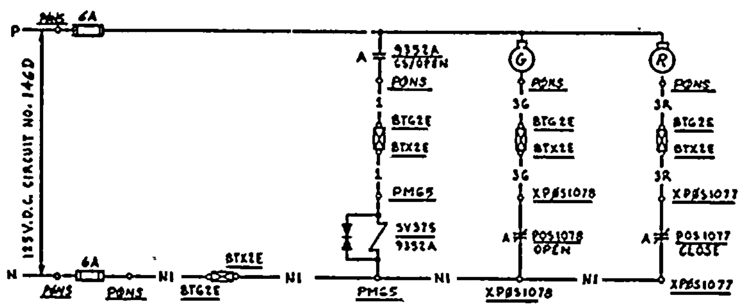
FIGURE 6.2-14
PENETRATION DIAGRAM
(SHEET 7 OF 23)



9354A PRESSURIZER STEAM SAMPLE ISOL. VALVE CIRCUIT NO. RSS043

OTHER SAMPLE SYSTEM VALVES SIMILAR EXCEPT AS SHOWN IN TABLE. CONTROL SWITCH FOR THESE VALVES — DETAIL B. CIRCUITS LISTED ON DRAWING 103072.

VALVE NO.	DESCRIPTION	D.C. CKT. NO.	VALVE LOC. NO.	SOLENOID VALVE NO.	SOL. VALVE LOC. NO.	ENG. SFGD FUNCTION	VALVE POSITION SW. (POS) CLOSED	OPEN			PENN NO.	CIRCUIT NO.	LOC. OF CONTROL	CONTAINMENT ISOL. PHASE A DEVICE	LOC.	CONTACT	NORMAL POSITION
9354A	PRZR. STM. SAMPLE ISOL. IN CNT.	117D	X9354A	367	PM127	T	297	298			26E	RSS043	CNCS	K606	RNS0A		OPEN
9355A	PRZR. LIQ SAMPLE ISOL. IN CNT	117D	X9355A	370	PM127		303	1044			26E	RSS049		K606	RNS0A		OPEN
9356A	RCS SAMPLE ISOL. IN CNT	117D	X9356A	374	PM127		1075	1046			26E	RSS055		K606	RNS0A		OPEN
9357A	ACCUM. SAMPLE ISOL. IN CNT	117D	X9357A	380	PM127		1087	1048			26E	RSS061		K606	RNS0A		OPEN
9354B	PRZR STM. SAMPLE ISOL. OUT CNT.	215D	X9354B	325	PM184		1312	1068			——	RSS008		K606	RNS0B		OPEN
9355B	PRZR. LIQ SAMPLE ISOL. OUT CNT.	215D	X9355B	326	PM184		1313	1070			——	RSS022		K606	RNS0B		OPEN
9356B	RCS SAMPLE ISOL. OUT CNT.	215D	X9356B	327	PM184		1314	1088			——	RSS029		K606	RNS0B		OPEN
9357B	ACCUM. SAMPLE ISOL. OUT CNT.	215D	X9357B	328	PM184		1319	1318			——	RSS036		K606	RNS0B		OPEN



9352A ACCUMULATOR 1 SAMPLE VALVE CIRCUIT NO. RSS079

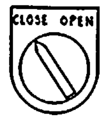
OTHER SAMPLE SYSTEM SOLENOID VALVES SIMILAR EXCEPT AS SHOWN IN TABLE BELOW.
CONTROL SWITCH FOR THESE VALVES — DETAIL A

VALVE NO.	DESCRIPTION	D.C. CKT. NO.	VALVE LOC. NO.	SOL. VALVE NO.	SOL. VALVE LOC. NO.	VLV. POS. SW. (POS) CLOSED	OPEN	PENN. NO.	CIRCUIT NO.	CONTROL LOC. NO.
9350A	PRZR. STM. SPACE IN CNT.	146D	X9350A	365	PM125	293	294	2E	RSS067	PØNS
9350B	PRZR. LIQ. SPACE IN CNT.	146D	X9350B	368	PM125	299	296	2E	RSS070	PØNS
9351A	RCS HOT LEG LOOP 1	146D	X9351A	371	PM138	1069	1041	2E	RSS073	PØNS
9351B	RCS HOT LEG LOOP 4	146D	X9351B	372	PM138	1071	1043	2E	RSS076	PØNS
9352A	ACCUM. 1 SAMPLE	146D	X9352A	375	PM65	1077	1078	2E	RSS079	PØNS
9352B	2	146D	X9352B	376	PM67	1079	1080	2E	RSS082	PØNS
9352C	3	146D	X9352C	377	PM68	1081	1082	2E	RSS085	PØNS
9352D	4	146D	X9352D	378	PM71	1083	1084	2E	RSS088	PØNS
9353A	RHR LOOPS 1 & 2	146D	X9353A	381	XSV381	1089	1090	——	RSS091	PØNS
9353B	RHR LOOPS 3 & 4	146D	X9353B	382	XSV382	1091	328	——	RSS094	PØNS

CONTACT		POSITIONS CLOSE	OPEN	REMARKS
	A		X	THIS DWG.
	B	X		SPARE

CONTROL SWITCH $\frac{9352A}{CS}$ --- DETAIL A

USED AT LOCATION PØNS
NO SPRING RETURN.



CONTACT		POSITIONS CLOSE	NEUT.	OPEN	REMARKS
	A		X	X	THIS DWG.
	B			X	THIS DWG.

CONTROL SWITCH $\frac{9354A}{CS}$ --- DETAIL B

USED AT LOCATION CNCS.
SPRING RETURN TO NEUTRAL FROM OPEN.

CONTACT		POSITION CLOSE	OPEN	REMARKS
	A		X	RED LIGHT
	B	X		SPARE
	C		X	MONITOR LT
	D	X		SPARE

VALVE CLOSE POSITION SWITCH

CONTACT		POSITION CLOSED	OPEN	REMARKS
	A	X		GREEN LT.
	B		X	SPARE
	C	X		SPARE
	D		X	SEAL IN

VALVE OPEN POSITION SWITCH

TABLE OF DEVICES

DEVICE NO.	FUNCTION	RATING	MFR.	TYPE	CAT/ DWG. NO.	REMARKS
SV367, 370, 374, 380, 365, 368, 371, 372, 325, 326, 327, 328, 375, 376, 377, 378, 381, 382	SOLENOID VALVE	——	——	——		
935__	FLOW CONTROL VALVE	——	——	——		SEE TABLE FOR VALVE NOS
POS___	VALVE POSITION SWITCHES	——	——	——		SEE TABLE FOR SWITCH NOS.
K606	CONTAINMENT ISOLATION PHASE A AUX. RELAY	——	——	——		

EQUIPMENT LOCATION NUMBERS

CNCS — CONTROL BOARD CONTAINMENT SPRAY
RNS0A — NUCLEAR SAFEGUARD OUTPUT RACK A
X9354A — AIR OPERATED VALVE 9354A
XSV381 — INSTRUMENT SV381 (LOCAL MOUNTED)
BTG26E — TERMINAL BOX, AREA G NO. 26E
PM127 — MECH. PANEL No 127, SAMPLE ISOL. VVS.
BTH110 — TERMINAL BOX AREA H No 110
PØNS — CONTROL PANEL NUCLEAR SAMPLE

VALVE NO'S.: 9350A, 9350B, 9351A, 9351B, 9352A, 9352B, 9352C, 9352D, 9353A, 9353B, 9354A, 9354B, 9355A, 9355B, 9356A, 9356B, 9357A, 9357B

UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 7.3-43
SCHEMATIC DIAGRAM
SAMPLING SYSTEM SOLENOID VALVES

Section 2.1.5.a - Dedicated Penetrations for External Recombiner or Post-Accident External Purge System

Task Force Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to service only, that meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that are sized to satisfy the flow requirements of the recombiner or purge system. (Description and implementation schedule is Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later. Category B: Implementation shall be completed by January 1, 1981.)

PG&E Response

PG&E will provide dedicated penetrations and isolation systems for the hydrogen recombiner and hydrogen purge systems at the Diablo Canyon Plant that meet the single failure redundancy and requirements of the NRC regulations and regulatory guides. This will require modification of existing systems such that they are not connected to, and are not branch lines of, the large containment purge penetrations. The description of the necessary modifications and implementation schedule for such modifications will be completed prior to OL. Modifications will be completed prior to January 1, 1981.

Section 2.1.5.b - Inerting BWR Containments

Task Force Position

It shall be required that the Vermont Yankee and Hatch 2 Mark I BWR containments be inerted in a manner similar to other operating BWR plants. Inerting shall also be required for near-term OL licensing of Mark I and Mark II BWR's.

PG&E Response

Both Diablo Canyon Units 1 and 2 are Westinghouse PWR's; therefore, the position does not apply.

Section 2.1.5.c - Capability to Install Hydrogen Recombiners at Each Light Water Nuclear Power Plant

Task Force Position (Minority View)

1. All licensees of light water reactor plants shall have the capability to obtain and install recombiners in their plants within a few days following an accident if containment access is impaired and if such a system is needed for long-term post-accident combustible gas control. Implementation schedules will be established by the Commission in the course of the immediately effective rulemaking. The Task Force recommends that the rulemaking process be initiated promptly.

2. The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2. (Category A - Implementation complete by January 1, 1980, or prior to OL, whichever is later).

PG&E Response

1. PG&E will install hydrogen recombiners. Completion of the modifications is dependent on availability of equipment. It is PG&E's intention to procure this equipment on an expedited basis. If equipment is available, the modifications will be completed prior to power operation.

2. The hydrogen recombiners will be located inside the containment structure. Consequently, personnel exposure is not a problem.

Section 2.1.6.a - Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems & Auxiliary Systems)

Task Force Position

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction

   Establish and implement a program of preventive maintenance to reduce leakage to as-low-as practical levels. This program shall include periodic integrated leak tests at a frequency not to exceed refueling cycle intervals.

   (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

The residual heat removal (RHR) system; portions of the containment spray (CS) system; portions of the safety injection (SI) system; portions of the chemical and volume control (CVCS) system including letdown, makeup and high pressure ECCS; primary sampling (SS) system; and gaseous radioactive waste (GRW) systems have been identified as systems which process primary coolant, and could contain high level radioactive materials. The following programs will be implemented to reduce and maintain leakage as low as practical.

Section 2.1.6.a (Cont'd)

PG&E Response (Cont'd)

1. The RHR, CS, SI, and CVCS systems are ASME Code Class 2 or 3, and are subject to the in-service inspection requirements of the ASME Boiler and Pressure Vessel Code, Section XI, including pressure tests.

2. At intervals not exceeding 18 months, operating pressure leakage tests with system makeup measured will be performed on appropriate portions of RHR, SI, CS, CVCS (ECCS) and SS systems. System leakage rates will be determined during the above tests and appropriate measures will be taken to reduce or eliminate undesirable leakage paths.

3. Portions of the charging system which are in service during normal operation (makeup and letdown) are monitored with the rest of the reactor coolant system for leakage during steady-state conditions by the reactor coolant system water inventory balance. Excessive leakage into controlled areas will also be indicated by abnormally high airborne radioactivity levels. Moreover, this system will be visually inspected on a routine basis.

   Some portions of the CVCS are used for boron recycling (feed and bleed to change RCS boron concentration) during normal operation. The boron recycling system will not be used under post-accident circumstances.

   Therefore, the boron recycle system, including liquid holdup tanks will not be subject to intensive leakage surveillance.

4. Excessive gaseous radwaste system leakage will be indicated by abnormal airborne radioactivity levels in the spaces occupied by the system. If excessive leakage is indicated, appropriate techniques will be used to localize the leakage so that it may be repaired. Because of the nature of the system, quantification of leakage rates is considered impractical and inconsistent with ALARA personnel radiation exposure considerations.

The necessary provisions to carry out these programs will be completed prior to power operation.

Section 2.1.6.b - Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations

Task Force Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4, each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility. (Category A: Complete the design review prior to OL, or January 1, 1980, whichever is later. Category B: Complete plant modifications by January 1, 1981.)

PG&E Response

The design basis accident considered in the present Diablo Canyon shielding design outside of the containment did not identify the potential for the spread of highly radioactive fluids and gases to systems not designed for such radioactivity. Therefore, these systems are being reviewed from the standpoint of this task force position, using a postulated high reactor coolant activity equivalent to the source definition in Regulatory Guide 1.4.

Specifically, a number of engineered safety feature systems and other systems located outside the containment may be required to function in a post-accident condition while containing highly radioactive coolant. These systems may become filled with highly radioactive coolant either prior to receipt of a containment isolation signal, or for example, as the result of deliberate actions taken to provide long term cooling, or to obtain coolant samples.

Section 2.1.6.b (cont'd)

Task Force Position (cont'd)

Based on the discussion provided in NUREG-0578, the systems to be reviewed include:

Reactor Coolant and Containment Atmospheric Sampling

Sampling Station and Radiochemistry Laboratory

Radwaste System

Chemical and Volume Control System

Residual Heat Removal System

Containment Spray System (Recirculation Mode Portions)

High Pressure Injection System (Recirculation Mode Portions)

Plant Ventilation System

Auxiliary Feedwater System

The design review specified for the shielding of spaces for post-accident operations will be completed prior to OL, or January 1, 1980, whichever is later. In addition, any changes in procedures for access control or systems operation determined to be required by the review will be implemented by the same time. All plant modifications will be completed, tested, and accepted, prior to January 1, 1981.

Section 2.1.7.a - Automatic Initiation of the Auxiliary Feedwater System

Task Force Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term: (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

Position 1.

The design shall provide for the automatic initiation of the auxiliary feedwater system.

PG&E Response

The auxiliary feedwater system is shown in the attached FSAR Figure 3.2-03, Sheet 2 of 4. The pumps are automatically started by low-low steam generator level, feedwater pump trip, safety injection, or station blackout. See attached FSAR Figures 7.3-8, 7.3-17, and 7.3-18. As shown on FSAR Figure 7.3-17, the motor-driven auxiliary feedwater pumps are started by closure of the Solid State Protection System (SSPS) output relay K633. Relay K633 is actuated by safety injection initiation or low-low level in any steam generator. Each pump is started by a separate relay from redundant SSPS trains A and B. The motor-driven pumps are also automatically started by trip of both main feedwater pumps.

The turbine driven auxiliary feedwater pump is started by opening steam supply valve FCV-95. As shown on FSAR Figure 7.3-18, this valve is opened by SSPS output relays K632 or K634. Relay K632 provides for starting on station blackout and relay K634 provides starting on safety injection or low-low level in any steam generator. Station blackout is determined by low voltage on the 12kV reactor coolant pump buses. An automatic starting signal is provided by redundant SSPS trains A and B.

The system valves are normally open and require no actions for system operation.

Section 2.1.7.a (Cont'd)

Position 2.

The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

PG&E Response

As shown in our response to Position 1 above, the auxiliary feedwater initiation circuitry is part of our Engineered Safety Features (ESF) system, and as such, is installed in accordance with IEEE Standard 279. This Standard is referenced in 10 CFR 50.55a(h).

Position 3.

Testability of the initiating signals and circuits shall be a feature of the design.

PG&E Response

The auxiliary feedwater initiation signals and circuitry are testable. Such testability is included in the surveillance test procedures for the plant as delineated in the Plant Technical Specification.

Position 4.

The initiating signals and circuits shall be powered from the emergency buses.

PG&E Response

As shown in our responses to Positions 1 and 2, the initiating signals and circuits are a part of the Plant Engineered Safety Features. The requirements for the ESF system dictate that the system shall meet the "single failure criteria". To accomplish this, the initiating signals and circuits must be powered from separate emergency buses.

The initiating sensors such as steam generator low-low level are powered from separate and redundant nuclear instrumentation and control panels, each of

Section 2.1.7.a (Cont'd)

PG&E Response (Cont'd)

which is supplied by either on-site emergency generators or station emergency batteries. See Figure 2.1.1-4. Each of the two redundant SSPS trains is supplied by separate power sources.

Position 5.

Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

PG&E Response

Manual initiation for each train exists in the control room. The manual initiation system is installed in the same manner as the automatic initiating system. No single failure in the manual initiation portion of the circuit can result in the loss of auxiliary feedwater system function. See FSAR Figures 7.3-17 and 7.3-18 for the circuitry.

Position 6.

The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

PG&E Response

As with all ESF equipment, the a-c motor-driven pumps and all valves in the system are automatically transferred to, and sequentially loaded on, the emergency buses on loss of offsite power. The sequence is shown in FSAR Table 8.3-2, attached.

Position 7.

The automatic initiating signals and circuits shall be designed so that their failures will not result in the loss of manual capability to initiate the AFWS from the control room.

Section 2.1.7.a (Cont'd)

PG&E Response (Cont'd)

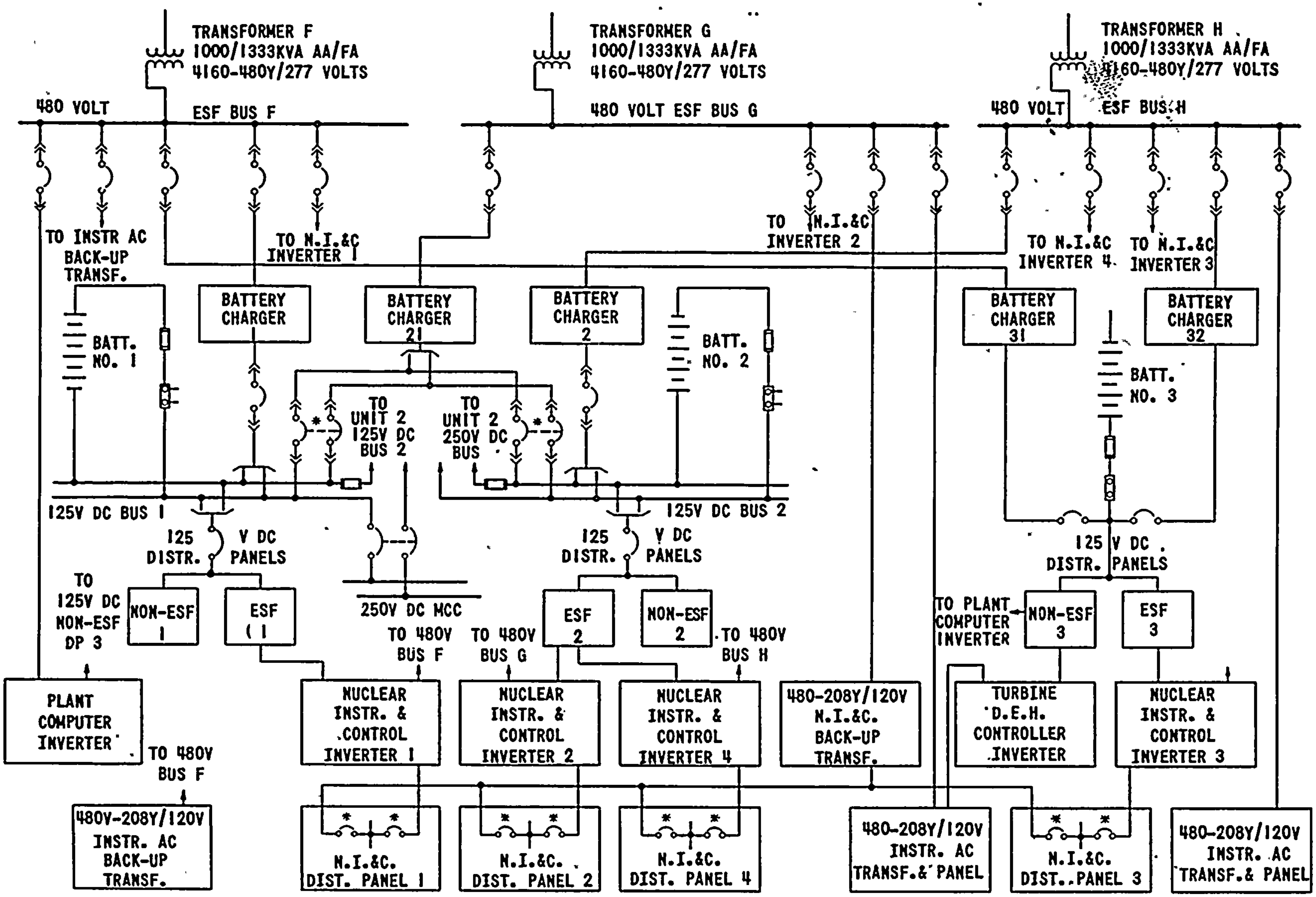
All automatic initiating signals and circuits are installed in accordance with regulatory requirements and are safety grade and redundant. No single failure in the automatic portion of the system will result in loss of the capability to manually initiate the AFWS from the control room.

Position 8.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety grade requirements.

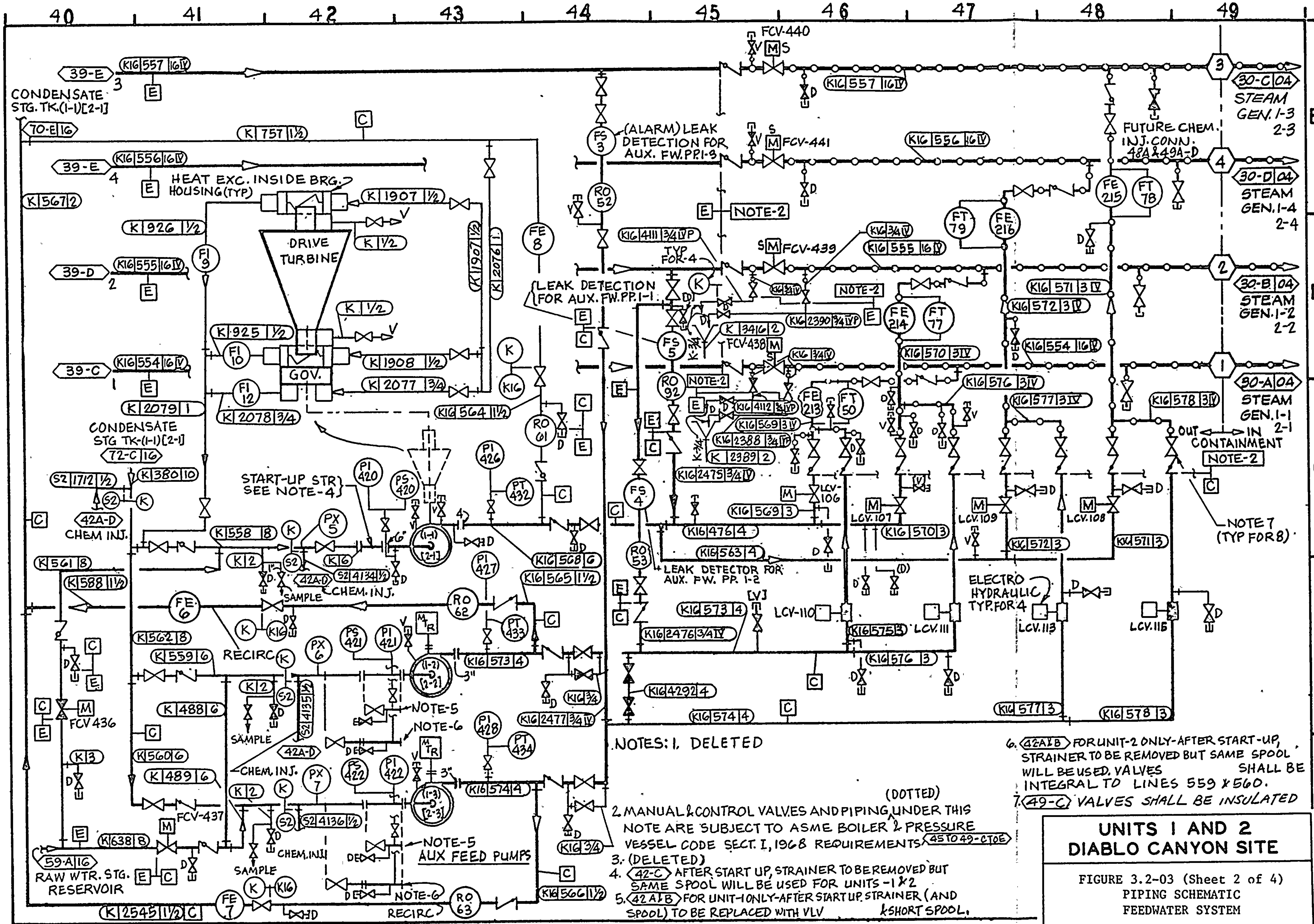
PG&E Response

As described in the preceding responses, the automatic initiating signals presently meet all safety grade requirements. No upgrading is required.



* BREAKERS MECHANICALLY INTERLOCKED

DIABLO CANYON SITE, UNIT 1
SINGLE LINE DIAGRAM
480V ESF & 250/125 DC
Figure 2.1.1-4



NOTES: 1. DELETED

2. MANUAL & CONTROL VALVES AND PIPING (DOTTED) UNDER THIS NOTE ARE SUBJECT TO ASME BOILER & PRESSURE VESSEL CODE SECT. I, 1968 REQUIREMENTS (45 TO 49-C TOE)

3. (DELETED)

4. (42-C) AFTER START UP, STRAINER TO BE REMOVED BUT SAME SPOOL WILL BE USED FOR UNITS -1 & 2

5. (42 A & B) FOR UNIT-1 ONLY-AFTER START UP, STRAINER (AND SPOOL) TO BE REPLACED WITH VLV & SHORT SPOOL.

6. (42A & B) FOR UNIT-2 ONLY-AFTER START-UP, STRAINER TO BE REMOVED BUT SAME SPOOL WILL BE USED. VALVES SHALL BE INTEGRAL TO LINES 559 & 560.

7. (49-C) VALVES SHALL BE INSULATED

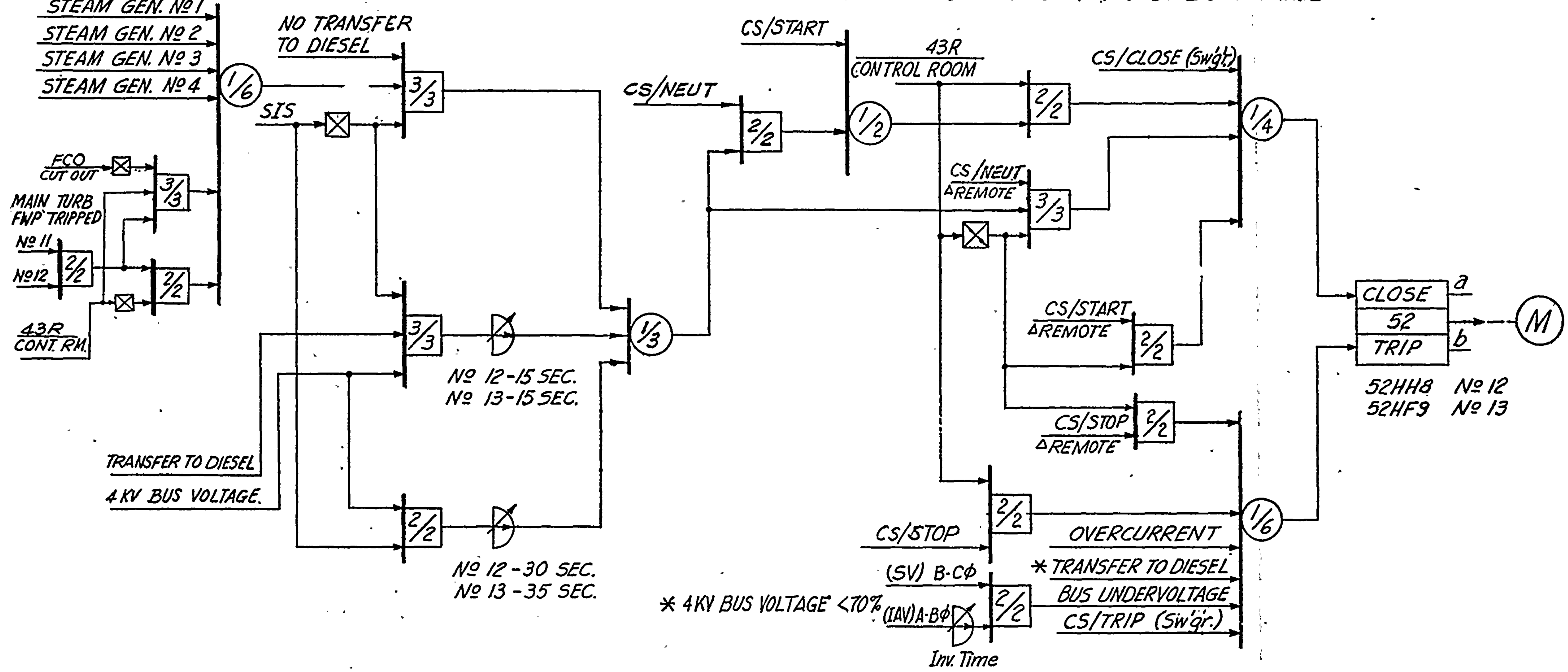
UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 3.2-03 (Sheet 2 of 4)
PIPING SCHEMATIC
FEEDWATER SYSTEM

Amendment 44 July 1976

NOTES

43R - TRANSFERS CONTROL FROM MAIN CONTROL BOARD TO HOT SHUT DOWN PANEL.
* - FOR COMPLETE TRANSFER TO DIESEL & 4KV BUS UNDERVOLTAGE LOGIC DIAGRAM - SEE 4160V BUS SECT. F, G, & H AUTOMATIC TRANSFER
Δ - REMOTE CONTROL SWITCH LOCATED ON HOT SHUT DOWN PANEL



UNITS 1 AND 2
DIABLO CANYON SITE

FIGURE 7.3-8
LOGIC DIAGRAM
AUXILIARY FEEDWATER PUMPS

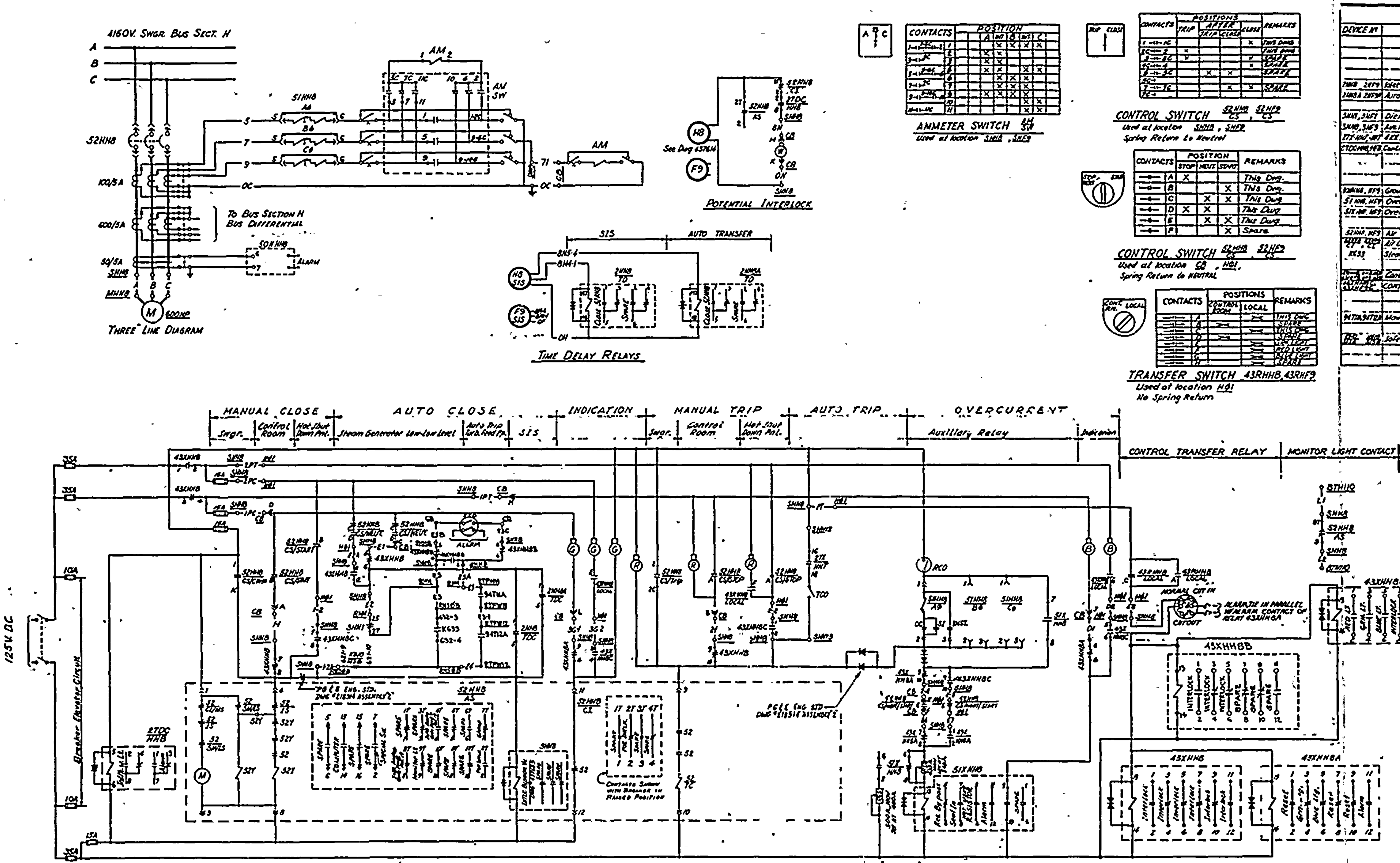


TABLE of DEVICES						
DEVICE Nº	FUNCTION	RATING	MFR	TYPE	CAT/DWG Nº	REMARKS
2HH8 2HF9	Safety Injection Signal Timing Rel.	120V AC	Agastat	2400	2412AD	Time Delay 2.5-50 Sec
2HH8A 2HF9A	Auto Transfer Timing Relay	120V AC	Agastat	2400	2412AD	Time Delay 2.5-50 Sec
3HH1, 3HF1	Diesel Auto Transfer Interlock Rel.	—	—	—		
3HH8, 3HF9	Aux Feedwater Pump Auto Start Rel.	125V DC	GE Co.	[illegible]	[illegible]	
27X-HHT, HFT	4KV Bus UV Trip Auxiliary Relay	—	—	—		[illegible]
2TDC HH8, HF9	Control Bus DC Undervoltage	125V DC	G.E. Co.	HGA		
50GHH8, HF9	Ground Overcurrent Sensor	—	—	—		Alarm Only
51 HH8, HF9	Overcurrent Relay	—	—	—		
51X HH8, HF9	Overcurrent Auxiliary Relay	[illegible]	G.E. Co.	HFA	[illegible]	
52HH8, HF9	Air Circuit Breaker	—	—	—		Rec. No. 663332
[illegible]	Air Ckt. Bkr. Cell Interlock	—	—	—		Part of Switchgear
K633	Steam Generator Lolo Level Aux Rel	—	—	—		
[illegible]	Control Transfer Relay	125V DC	G.E. Co.	HFA	[illegible]	
[illegible]	CONTROL TRANSFER RELAY	125V DC	G.E. Co.	HFA	[illegible]	
94T1A, 94T2A	Main Feedwater Pump Tripping Rel.	—	—	—		
[illegible]	Safety Injection Interlock Blk. Rel.	—	—	—		

EQUIPMENT LOCATION NUMBERS

- CB Control Board Condensate & Feedwater
- HS1 Hot Shut Down Remote Control Panel 1
- MHH8 Auxiliary Feedwater Pump Nº 12
- SHH8 Switchgear 4KV Bus H Cubicle 8
- RHH Engineer Safeguards Relay Board Bus H
- ETFWP Equipment Cabinet FWP Turbine Nº 8
- RNS8B Nuclear Safeguard Output Rack - Train B
- BTH110 Terminal Box Area H Nº 110

125 V DC CONTROL SCHEMATIC

AUXILIARY FEEDWATER PUMP Nº 12 CIRCUIT Nº H081100

Auxiliary Feedwater Pump Nº 13 Similar Except as Shown in Table

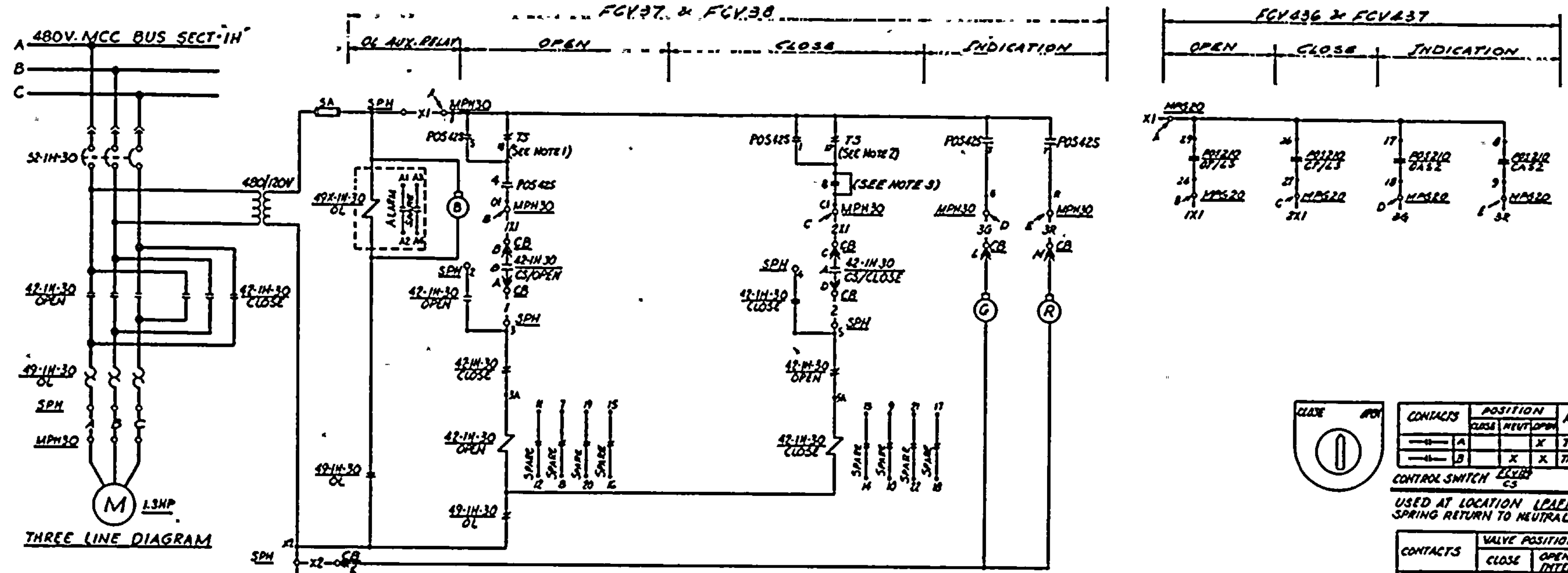
MOTOR	MOTOR LOC.	CKT BKR Nº	CIRCUIT Nº	BUS SECT	125 V DC CKT Nº	BUS UNDERVOLTAGE RELAY	BUS UNDERVOLTAGE LOC	BUS UNDERVOLTAGE CONTACT TRIP	AUTO TRANSFER BLOCKING RELAY	AUTO TRANSFER BLOCKING LOC	AUTO TRANSFER BLOCKING CONTACT INTERLOCK	FEEDWATER TURBINE Nº 11 TRIP RELAY	FEEDWATER TURBINE Nº 11 TRIP LOC	FEEDWATER TURBINE Nº 11 TRIP CONTACT INTLK	FEEDWATER TURBINE Nº 12 TRIP RELAY	FEEDWATER TURBINE Nº 12 TRIP LOC	FEEDWATER TURBINE Nº 12 TRIP CONTACT INTLK
Nº 12	MHH8	52HH8	H08H00	H	314D	27X HHT	[illegible]	[illegible]-16	3HH1	RHH	25-[illegible]-2T	94T1A	ETFW1	—‖—	94T2A	ETFW2	—‖—
Nº 13	MHF9	52HF9	F09H00	F	113D	27X HFT	SHF13	[illegible]-28	3HF1	RHF	35-[illegible]-3T	94T1A	ETFW1	—‖—	94T2A	ETFW2	—‖—

MOTOR	STM GEN LO LO LEVEL TRAIN A DEVICE	STM GEN LO LO LEVEL TRAIN A LOC	STM GEN LO LO LEVEL TRAIN A CONTACT	STM GEN LO LO LEVEL TRAIN B DEVICE	STM GEN LO LO LEVEL TRAIN B LOC	STM GEN LO LO LEVEL TRAIN B CONTACT	ALARM CIRCUIT Nº AUTO TRIP TURB FWP CUT OUT	ALARM CIRCUIT Nº OVERCURRENT	ALARM CIRCUIT Nº GROUND OVERCURRENT	ALARM CIRCUIT Nº TRANSFER OF CONTROL	ALARM CIRCUIT Nº CONT. BUS DC UNDERVOLT.	COMPUTER CKT. Nº RUNNING
Nº 12	—	—	—	K633	RNS8B	[illegible]	AY2442DH	AI 2415DH	AI 2417DH	AY5072 DH	AY4207DH	CY2845DH
Nº 13	K633	RNS8A	[illegible]	—	—	—	AY2441DF	AI 2416DF	AI 2418DF	AY5070 DF	AY4407DF	CY2846DF

MOTOR	SIS BLOCK DEVICE	SIS BLOCK LOC	SIS BLOCK CONTACT INTERLOCK	AUX. RELAY DEVICE Nº	AUX. RELAY Interlock Contact Ckt Nº	ANALOG GATE INTLKS CONTACTS 4 —‖— 4T	ANALOG GATE INTLKS CONTACTS 8 —‖— 8T
Nº 12	K610	RNS8B	[illegible]	3HH8	H08H14	PM87	PM86
Nº 13	K610	RNS8A	[illegible]	3HF8	F09H14	PM89	PM88

UNITS 1 AND 2 DIABLO CANYON SITE

FIGURE 7.3-17
SCHEMATIC DIAGRAM
AUXILIARY FEEDWATER PUMPS

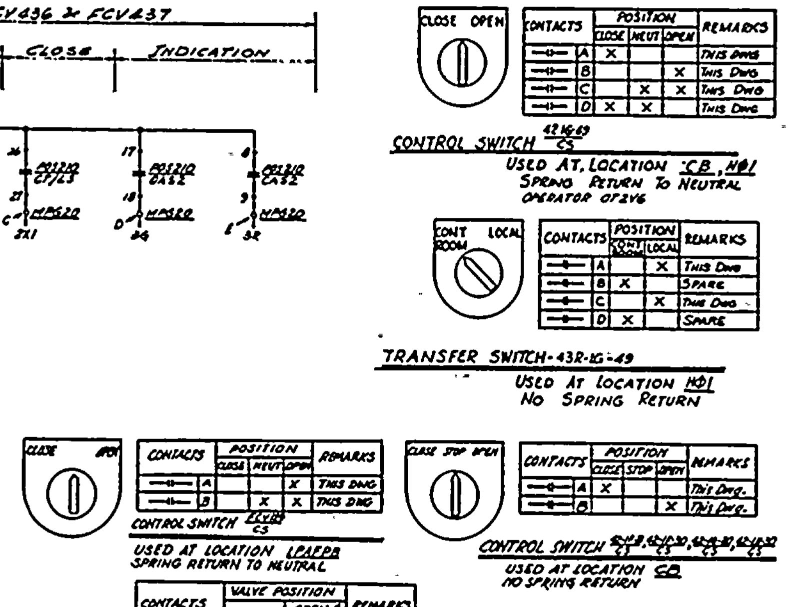


120V AC CONTROL SCHEMATIC

FCV 37 – AUX. FWP TURB II LEAD 2 STEAM SUPPLY M.O.V. CKT. No H30P00

AUX FWP MOV'S SIMILAR EXCEPT AS SHOWN IN TABLE

VALVE No	DESCRIPTION	CIRCUIT BKR. No	CIRCUIT No.	BUS SECT	NORMAL POSITION	ENGINEER SAFETY VITAL FUNCTION	VALVE POSITION SWITCH (POS) POS No.	ALARM CIRCUIT No. OVER CURRENT	VALVE CLOSED	VALVE OPEN	MOTOR LOC. No	MOTOR HP
FCV 37	AUX FWP TURB II LEAD 2 STM SUPPLY	52-1H-30	H30P00	1H	OPEN	—	425	[illegible]	[illegible]		MPH30	1.3
FCV 38	〃 〃 3 〃	52-1F-30	F30P00	1F	OPEN	—	433	[illegible]	[illegible]		MPF30	1.3
FCV436	PUMP II RAW WATER SUPPLY	52-1G-20	G20P00	1G	CLOSED	—	210	[illegible]	—	[illegible]	MPG20	0.25
FCV437	〃 〃	52-1F-21	F21P00	1F	CLOSED.	—	212	[illegible]	—	[illegible]	MPF21	0.25



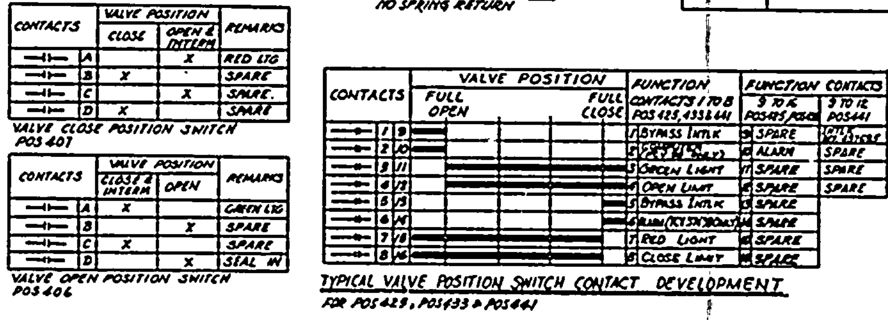
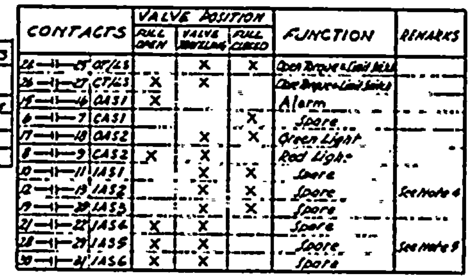
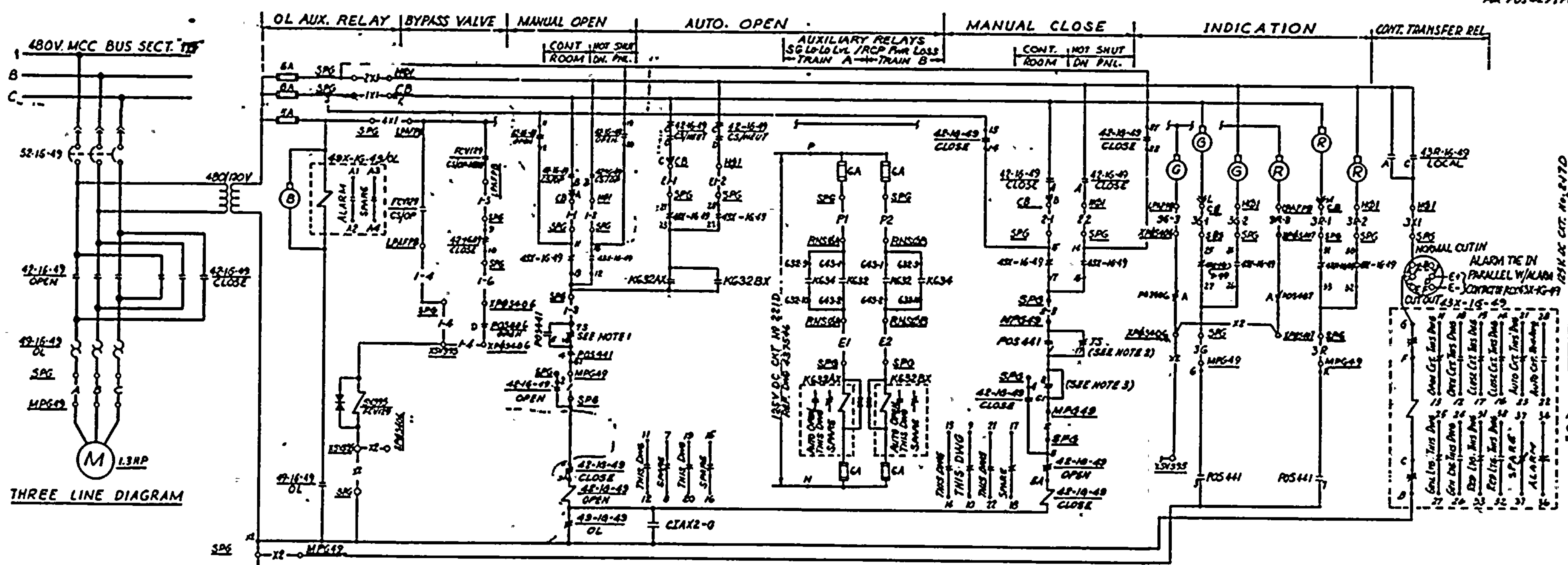


TABLE OF DEVICES						
DEVICE No	FUNCTION	RATING	MFR.	TYPE	CAT/DWG*	REMARKS
42-1F-21-30, 42-1G-20, 42-1G-49, 42-1H-30	480 V AC MOTOR STARTER REVERSIBLE. SIZE 1	—	—	—		
49-1F-21-30, 49-1G-20, 49-1G-49, 49-1H-30	MOTOR OVERLOAD RELAY	—	WE CORP	—	—	PART OF MOTOR STARTER
49X-1F-21-30, 49X-1G-20-49, 49X-1H-30	MOTOR OVERLOAD AUX. RELAY	—	WE CORP	—	—	PART OF MOTOR STARTER
52-1F-21-30, 52-1G-20-49, 52-1H-30	480V AC AIR CIRCUIT BKR	—	—	—		
K632	RCP BUS UNDERVOLTAGE AUX. REL.					
K634	STM GEN. LO-LO LEVEL AUX. REL.					
K632AX K632BX	SG LO-LO LVL /RCP PWR LOSS AUXILIARY RELAY	125VDC	CUTLER HAMMER	R. POWER	D40RRUAI	TO BE ADDED IN AUX RELAY PNL., MCC 1G.
43X-1G-49	CONTROL TRANSFER RELAY	120V AC	[illegible]	[illegible]	7803F	HAND RESET
FCV-	FLOW CONTROL VALVE	—	—	—	—	SEE TABLE FOR VALVE No's
POS-	FCV POSITION SWITCH	—	—	—	—	SEE TABLE FOR SW No's
C1AX2-0	[illegible]	—	—	—		[illegible]
SV	SOLENOID VALVE	—	—	—	—	
TS	TORQUE SWITCH	—	—	—	—	



TYPICAL VALVE POSITION SWITCH CONTACT DEVELOPMENT

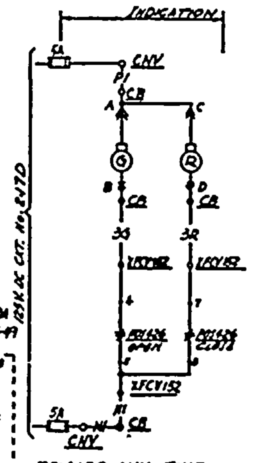
FOR POS210 & POS212



120 V AC CONTROL SCHEMATIC

FCV 95 – AUX. FWP TURB. II STEAM SUPPLY HEADER M.O.V. CKT. No. G49P00

ALARM CIRCUIT No's		COMPUTER CKT. No's	
OVER CURRENT	TRANS. [illegible] CONT	[illegible]	
[illegible]	[illegible]		C1284406



FCV152 AUX. FWP TURB. STEAM INLET VALVE CIRCUIT No. 1G152FCV

EQUIPMENT LOCATION NUMBERS:

SPH – 480V MOTOR CONTROL CENTER BUS SECTION "1H"
MPH30 – MOTOR OPERATED VALVE FCV 37
CB – CONTROL BOARD, CONDENSATE & FEEDWATER
HØ1 – HOT SHUTDOWN CONTROL PANEL. (FEEDWATER)
RNSØB – RACK NUCLEAR SAFEGUARD OUTPUT B
1FCV152 – INSTRUMENT No FCV 152 LOCAL MOUNTED

NOTES:

1.- SWITCH OPENS ON MECHANICAL TORQUE. DURING OPENING CYCLE (1S).
2.- SWITCH OPENS ON MECHANICAL TORQUE DURING CLOSING CYCLE OR FULLY CLOSED VALVE (1T).
3.- FIELD TO INSTALL JUMPER IF TORQUE SEATING IS REQUIRED.
4.- ADJUSTABLE TO OPERATE TOGETHER AT ANY POINT DURING OPENING TRAVEL.
5.- ADJUSTABLE TO OPERATE TOGETHER AT ANY POINT DURING CLOSING TRAVEL.

UNITS 1 AND 2 DIABLO CANYON SITE

FIGURE 7.3-18
SCHEMATIC DIAGRAM
AUXILIARY FEEDWATER PUMP
TURBINE CONTROL

Amendment 78 March 1979

TABLE 8.3-2

TIMING SEQUENCE AND INTERVALS - SAFETY INJECTION SIGNAL

Load	Starting Delay (Seconds) After Power is Restored to the Vital Buses (6)			Minimum Number Required
	Bus F	Bus C	Bus H	
Small Loads (480 and 120 Volt) on Vital 480 Volt Load Centers	0	0	0	2
Centrifugal Charging Pumps	5	5	-	1
Safety Injection Pumps	10	-	5	1
Containment Spray Pumps	-	10	10	1
Residual Heat Removal Pumps	-	15	15	1
Containment Fan Coolers	15, 20	20, 25	20	3
Component Cooling Water Pumps	25	30	25	2
Auxiliary Saltwater Pumps	30	35	-	1
Auxiliary Feedwater Pumps	35	-	30	1

Section 2.1.7.b Auxiliary Feedwater Flow Indication to Steam Generators for PWR's

Task Force Position

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented: (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

Position 1.

Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

PG&E Response

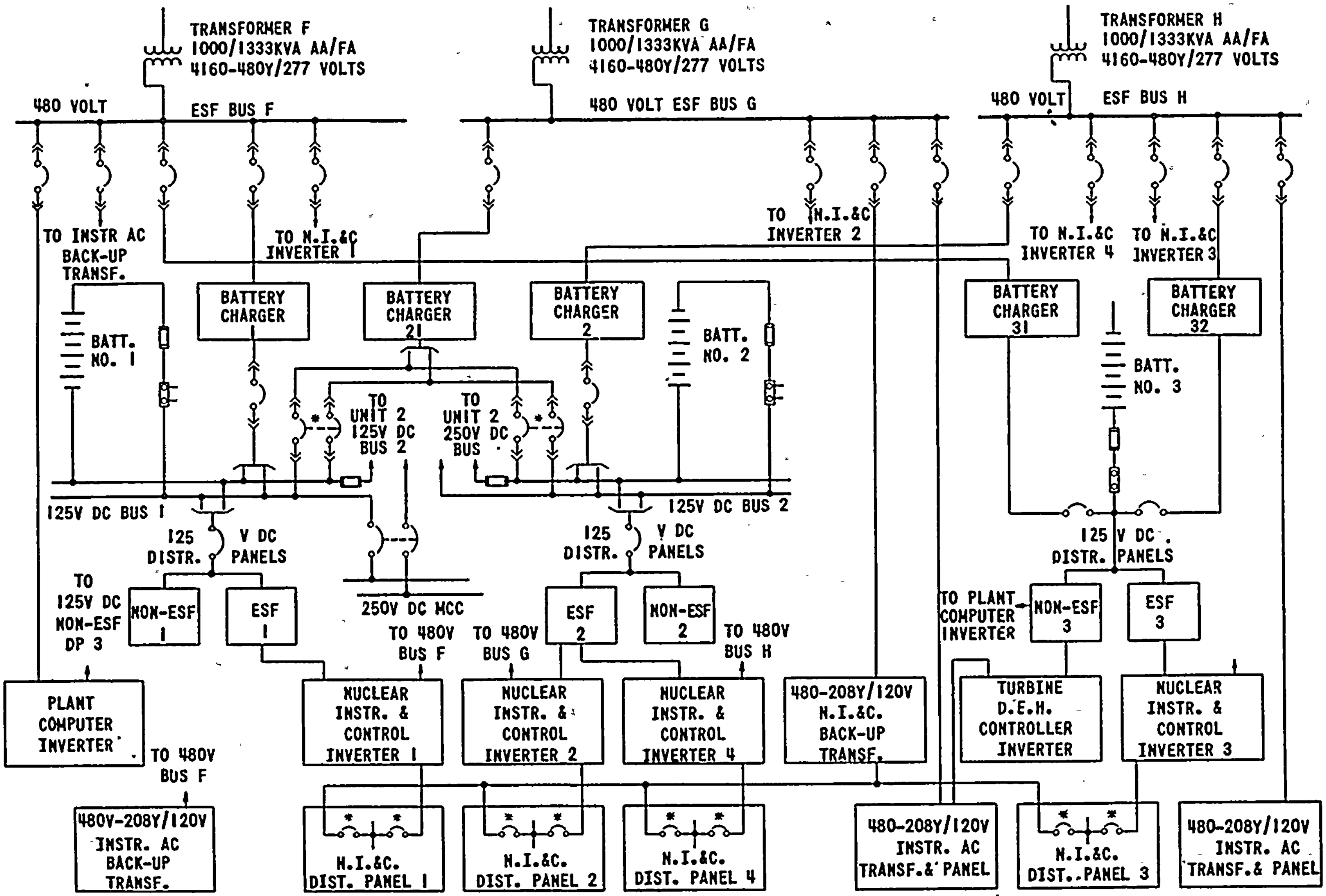
We have auxiliary feedwater flow indication for each of four auxiliary feedwater lines, one indicator per line. These presently are not safety grade and are all powered from the same nonvital bus. We will separate these so that each will be powered from the same vital bus as its associated motor driven auxiliary feedwater pump, and will upgrade the circuitry to safety grade requirements. This will be completed prior to OL, or January 1, 1980, whichever is later.

Position 2.

The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

PG&E Response

Auxiliary feedwater flow instrument channels will be powered from the same power bus that provides motive power for the auxiliary feedwater pumps. The instrument power is derived from separate inverters each of which is provided with 480 volts a-c from the on-site emergency power system and 125 volts d-c from the station emergency batteries (See Figure 2.1.1-4).



* BREAKERS MECHANICALLY INTERLOCKED

DIABLO CANYON SITE, UNIT 1
SINGLE LINE DIAGRAM
480V ESF & 250/125 DC
Figure 2.1.1-4

Section 2.1.8.a - Improved Post-Accident Sampling Capability

Task Force Position

A design and operational review of the reactor coolant containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

(Category A - Implementation of design reviews, description of proposed modifications, and preparation of revised procedures will be completed prior to OL, or January 1, 1980, whichever is later.)

Section 2.1.8.a (Cont'd)

Task Force Position (Cont'd)

(Category B: Implementation of plant modifications will be completed by January 1, 1981.)

PG&E Response

Review of Reactor Coolant and Containment Atmosphere Sampling Systems - A design review of post-accident sampling capability (obtaining and preparing samples in a form for transportation to the laboratory facilities) will be conducted. The review will assess the adequacy of the existing sampling capabilities for obtaining reactor coolant and containment atmospheric samples under post-accident conditions for the postulated accident involving significant core damage. The sampling system presently installed at Diablo Canyon was designed for a capability of obtaining samples during normal operating conditions based on one percent fuel defects. During postulated post-accident conditions, radiation levels will be significantly increased. Thus, the capability of obtaining and preparing the samples will be dependent on radiation doses to personnel. The review of sampling capabilities, shielding, and the potential for airborne contamination of the radiochemical laboratory and counting room will be conducted in parallel with the review of plant shielding (see Section 2.1.6.b).

As outlined in the Task Force Position, the review will consider the capability to:

1. Promptly obtain and prepare samples of reactor coolant and containment atmosphere for laboratory analysis (in less than one hour).

2. Maintain personnel exposures during the sampling collection and preparation to within the dose limits specified in 10 CFR 20 (see Section 2.1.6.b).

The following sequence will be followed in performing the design review:

Section 2.1.8.a (Cont'd)

PG&E Response (Cont'd)

1. An appropriate source term will be defined, as outlined in the shielding review section of this report.

2. Exposure to personnel obtaining and preparing samples will be determined based on the findings of the shielding review.

3. Sampling capabilities and techniques will be reviewed and dose rates estimated.

4. Using the estimated doses, any necessary modifications to the existing sampling systems will be developed and implemented.

The initial design review specified for improved post-accident sampling capability will be completed prior to power operation.

In addition, any changes in procedures for access control or systems operation determined to be required for obtaining and preparing reactor coolant and containment atmospheric samples will also be implemented prior to power operation.

All plant modifications to the containment atmosphere and reactor coolant sampling system shall be completed, tested, and accepted prior to January 1, 1981.

Review of Radiological Spectrum Analysis Capabilities -
Exisiting procedures provide for prompt (less than two hours) radiological spectrum analyses of noble gases, radioiodines, radiocesiums, and other non-volatile radionuclides. No difficulties are expected in performing these analyses provided samples are promptly prepared in the sampling area so as to avoid high background radiation or contamination levels in the counting room.

Review of Chemical Analysis Capabilities -
A review of existing boron and chloride analysis procedures indicates that they are adequate in their present form for highly radioactive samples and

Section 2.1.8.a (Cont'd)

<u>PG&E Position</u> (Cont'd)

are capable of being completed promptly (boron analysis within an hour and chloride analysis within eight hours). However, the dose received by a technician performing these analyses could approach the 3 Rem and 18 3/4 Rem values to the whole body and extremities, respectively. To improve this situation, an automatic burette system will be obtained for boron analysis, and a selective electrode will be obtained to allow chloride to be rapidly determined on a pH meter. Use of these techniques should significantly reduce the exposure received in these analyses. This equipment will be in place prior to power operation, which is the first time this capability will be needed.

Section 2.1.8.b - <u>Increased Range of Radiation Monitors</u>

Task Force Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

Position 1.

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

a. Noble gas effluent monitors with an upper range capacity of 10^5 µCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from a minimum of 10^{-7} µCi/cc (Xe-133) to a maximum of 10^5 µCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors shall overlap by a factor of ten.

(Category B: Implementation shall be completed by January 1, 1981.)

PG&E Response

We will add plant vent monitors with ranges of 10^{-5} µCi/cc to 10^5 µCi/cc to complement our existing monitors with ranges of 5 X 10^{-7} µCi/cc to 1 X 10^{-4} µCi/cc. This will provide a one-decade overlap. Multiple monitors will be required to cover the 10^{-5} to 10^5 µCi/cc range and will overlap each other by one decade. These will be installed prior to January 1981.

Section 2.1.8.b (Cont'd)

Position 2.

Since iodine gaseous effluent monitors for the accident conditions are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by on-site laboratory analysis. (Category B: Implementation shall be completed by January 1, 1981.)

PG&E Response

The monitors described in our response to Position 1 will have removable charcoal filters which can be analyzed in the counting room.

Position 3.

In containment radiation level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment. (Category B: Implementation shall be completed by January 1, 1981.)

PG&E Response

Mutually redundant, separated monitors will be provided to monitor to 10^7 R/hr. This value is equivalent to a factor of 8 over the maximum dose rate for a 100% core release to the containment environment. A 10^8 R/hr level would be two orders of magnitude above that dose rate. The ACRS has taken this into consideration. This matter was discussed with ACRS at its 220th general meeting on August 3, 1978.

These monitors will be mounted on steel hatches outside of the containment because of the great difficulties in qualifying them for a post-accident environment that might accompany such extreme levels of radioactivity. This equipment will be installed prior to January 1, 1981.

Section 2.1.8.c - Improved In-Plant Iodine Instrumentation

Task Force Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration throughout the plant under accident conditions.

(Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

Existing plant procedures provide for determination of airborne iodine concentration by counting the charcoal cartridges on the plant's multichannel analyzer, which employs a Ge(Li) detector. Chemistry and radiation protection technicians are trained in the use of this instrument and use it in their routine work.

Section 2.1.9 - Analysis of Design and Off-Normal Transients and Accidents

Task Force Position

Analyses, procedures, and training addressing the following are required:

1. Small break loss-of-coolant accidents;
2. Inadequate core cooling; and
3. Transients and accidents.

Some analysis requirements for small breaks have already been specified by the Bulletins and Orders Task Force. These should be completed. In addition, pretest calculations of some of the Loss of Fluid Test (LOFT) small break tests (scheduled to start in September 1979) shall be performed as means to verify the analyses performed in support of the small break emergency procedures and in support of an eventual long-term verification of compliance with Appendix K of 10 CFR Part 50.

In the analysis of inadequate core cooling, the following conditions shall be analyzed using realistic (best-estimate) methods:

1. Low reactor coolant system inventory (two examples will be required - LOCA with forced flow, LOCA without forced flow).
2. Loss of natural circulation (due to loss of heat sink).

These calculations shall include the period of time during which inadequate core cooling is approached as well as the period of time during which inadequate core cooling exists. The calculations shall be carried out in real time far enough that all important phenomena and instrument indications are included. Each case should then be repeated taking credit for correct operator action. These additional cases will provide the basis for developing appropriate emergency procedures. These calculations should also provide the analytical basis for the design of any additional instrumentation needed to provide operators with an unambiguous indication of vessel water level and core cooling adequacy (see Section 2.1.3.b in this appendix).

Task Force Position (Cont'd)

The analyses of transients and accidents shall include the design basis events specified in Section 15 of each FSAR. The analyses shall include a single active failure for each system called upon to function for a particular event. Consequential failures shall also be considered. Failures of the operators to perform required control manipulations shall be given consideration for permutations of the analyses. Operator actions that could cause the complete loss of function of a safety system shall also be considered. At present, these analyses need not address passive failures or multiple system failures in the short term. In the recent analysis of small break LOCAs, complete loss of auxiliary feedwater was considered. The complete loss of auxiliary feedwater may be added to the failures being considered in the analysis of transients and accidents if it is concluded that more is needed in operator training beyond the short-term actions to upgrade auxiliary feedwater system reliability. Similarly, the long-term, multiple failures and passive failures may be considered depending in part on staff review of the results of the short-term analyses.

The transient and accident analyses shall include event tree analyses, which are supplemented by computer calculations for those cases in which the system response to operator actions is unclear or these calculations could be used to provide important quantitative information not available from an event tree. For example, failure to initiate high-pressure injection could lead to core uncovery for some transients, and a computer calculation could provide information on the amount of time available for corrective action. Reactor simulators may provide some information in defining the event trees and would be useful in studying the information available to the operators. The transient and accident analyses are to be performed for the purpose of identifying appropriate and inappropriate operator actions relating to important safety considerations such as natural circulation, prevention of core uncovery, and prevention of more serious accidents.

The information derived from the preceding analyses shall be included in the plant emergency procedures and operator training. It is expected that analyses performed by the NSSS vendors will be put in the form of emergency procedure

Section 2.1.9 (Cont'd)

Task Force Position (Cont'd)

guidelines and that the changes in the procedures will be implemented by each licensee or applicant.

In addition to the analyses performed by the reactor vendors, analyses of selected transients should be performed by the NRC Office of Research, using the best available computer codes, to provide the basis for comparisons with the analytical methods being used by the reactor vendors. These comparisons together with comparisons to data, including LOFT small break test data, will constitute the short-term verification effort to assure the adequacy of the analytical methods being used to generate emergency procedures.

(Analyses, procedural changes, and operating training shall be provided by all operating plant licensees and applicants for operating licenses following the schedule in Table B-2 of NUREG-0578.)

PG&E Response

PG&E is participating as a member of the Westinghouse Owners Group in the review of the areas described in this Task Force Position. This review, coupled with further discussions with the NRC, will culminate in a plan and schedule to provide those additional actions relative to analyses, plant emergency operating procedures and operator training required by this position.

Section 2.2.1.a - <u>Shift Supervisor's Responsibilities</u>

<u>Task Force Position</u>

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the Shift Supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the Shift Supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the Shift Supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the Shift Supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

   b. The Shift Supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the Shift Supervisor shall be specified.

   c. If the Shift Supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

Section 2.2.1.a (Cont'd)

Task Force Position (Cont'd)

3. Training programs for Shift Supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

4. The administrative duties of the Shift Supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

(Category A: Implementation complete prior to OL or January 1, 1980, whichever is later).

PG&E Response

1. Prior to OL and at annual intervals thereafter, the Vice President - Electric Operations will issue a management directive that emphasizes the primary management responsibility of the Shift Foreman for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties as defined by the Administrative Procedures described below.

2, 3&4. Administrative Procedures are established which define the responsibilities and authorities of the Shift Foreman, and establish lines of succession. However, these procedures will be revised to more explicitly address the Task Force concerns, particularly those dealing with retaining breadth of perspective of operational conditions affecting safety and remaining in the control room under accident conditions to direct control room operational activities. These new provisions, as are all applicable administrative requirements, will be incorporated into the training program for the Shift Foreman.

Section 2.2.1.a (Cont'd)

PG&E Response (Cont'd)

A list of administrative duties of the Shift Foreman has been provided to the Vice President - Electric Operations, for his review and action. These activities will be completed prior to initial fuel loading which is the earliest time these provisions are needed.

Section 2.2.1.b - Shift Technical Advisor

Task Force Position

Each licensee shall provide an on-Shift Technical Advisor to the Shift Supervisor. The Shift Technical Advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the Shift Technical Advisors that pertain to the engineering aspects of assuring safe operations of the plant including the review and evaluation of operating experience. (Shift Technical Advisor on duty - Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later. Complete training for Shift Technical Advisor - Category B: Implementation shall be completed by January 1, 1981.)

PG&E Response

PG&E believes that its present shift operating organization, augmented by on-call nuclear engineers, represents a suitable alternative for accomplishing the objectives of the Task Force position with respect to the control room accident assessment function.

PG&E's Senior Reactor Operator Licensing Training Program provides the Shift Foremen with the same training as is received by our nuclear engineers. This training provides them with a broader knowledge of plant design, reactor core cooling and transient and accident analysis than is currently required by NRC operator licensing requirements. Continuing, additional training is being provided to our Shift Foremen and nuclear engineers based on evaluations of the lessons learned from the TMI accident being performed by the Company, industry and governmental groups. PG&E is actively supporting the development of the Institute of Nuclear Power Operation. We expect that the Institute will provide programs for enhanced technical training for on-shift supervisory personnel.

Section 2.2.1.b (Cont'd)

PG&E Response (Cont'd)

The formal on-call system for the Diablo Canyon Power Plant is used on weekends, holidays, or in any abnormal situation where the Plant Superintendent considers it necessary. In addition, trips away from the plant and vacations by key supervisory personnel are scheduled so that the necessary number of supervisory and technical personnel are available to respond to an emergency.

The individuals on call include a nuclear engineer on the plant staff or higher level plant staff person with a nuclear engineering background. There are ten positions on the plant staff normally filled by personnel who possess both a broad technical knowledge of the plant design and operation and possess a background in nuclear engineering. All of these individuals either possess or will obtain a NRC Senior Operator's License.

The on-call nuclear engineer is available to assist the Shift Foreman as requested. In addition, a nuclear engineer is in the control room for all approaches to critical and at all times during physics testing and refueling operations. It has been PG&E's policy to deploy nuclear engineers in this manner ever since initial startup of Humboldt Bay Unit No. 3 in the early 1960s. Recall of on-call personnel is effected through commercial telephone or through PG&E's private radio system using paging devices. The paging devices are actuated and are capable of receiving a voice message from the plant control room.

With respect to the Task Force position concerning the operating experience assessment function, PG&E has always considered this to be an extremely important function. Prior to the TMI accident, we have believed that our organization and procedures for accomplishing this function were adequate. As a result of our reassessment of requirements of this function, we are adding additional personnel to the staff of the Supervisor of Operations in the plant organization including a Senior Nuclear Engineer and an additional member for the training staff. In addition, a fifth shift of operators will

Section 2.2.1.b (Cont'd)

PG&E Response (Cont'd)

be added to provide required shift coverage for our expanded training program. One of the principal functions of the senior nuclear engineer will be that of assuring that plant and industry operating experience is imparted to shift operations supervision in a timely manner.

Section 2.2.1.c - Shift and Relief Turnover Procedures

Task Force Position

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptance status shall be included on the checklist).

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance of test that by themselves could degrade a system critical to the prevention and mitigation of operational transients (what to check and criteria for acceptable status shall be included on the checklist).

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

Section 2.2.1.c (Cont'd)

Task Force Position (Cont'd)

(Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

Existing shift and relief turnover procedures will be revised to incorporate the recommendations of the Task Force.

In addition, a system will be established to evaluate the effectiveness of the procedures which involves periodic independent verification of system alignments by the senior members of the Supervisor of Operation's staff (Relief Shift Supervisor, Senior Power Production Engineer (Operations)). This will be accomplished prior to initial fuel loading which is the earliest time these procedures are needed.

Section 2.2.2.a - Control Room Access

Task Force Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

   (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

An Administrative Procedure will be written to formalize existing policies which allow the Shift Foreman to restrict access to the control room during both normal operations and emergencies.

Existing Emergency Procedures establish authority and responsibilities in the control room in the event of an emergency, and provide for succession for the person in charge of the plant operations to possess a current NRC Senior Operator's License. Existing procedures also specify the locations to which all on-site personnel are to report in the event of an emergency. However,

Section 2.2.2.a (Cont'd)

PG&E Response (Cont'd)

these procedures will be revised to more explicitly address the Task Force recommendations, and to incorporate the changes necessitated by the establishment of the On-Site Technical and Operational Support Centers. These procedures will be developed and implemented prior to fuel loading which is the earliest time such procedures are needed.

Section 2.2.2.b - On-Site Technical Support Center

Task Force Position

Each operating nuclear power plant shall maintain an On-Site Technical Support Center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the Technical Support Center. (Category A: Establish center prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

PG&E will establish an On-Site Technical Support Center prior to initial power operation. The center will not be needed prior to that time. This center will be an area where Company and NRC supervisory and technical personnel not directly involved in the actions of the Emergency Control Center will work and will initially be located in the conference room of the Temporary Administration Building. It will have communication links with the control room and other response centers.

A permanent location for the On-Site Technical Support Center is now being evaluated. The permanent center will be habitable to the same degree as the control room for postulated accident conditions. In addition to the communication equipment that will be installed, monitoring equipment such as the process computer alarm typewriter, closed circuit television, and selected instrument repeaters will be incorporated in the design of the On-Site Technical Support Center. The status of the design of the permanent center will be provided to the NRC as part of the Diablo Canyon TMI Long-Term Studies Status Report. The first status report is scheduled for submittal on December 31, 1979.

Section 2.2.2.b (Cont'd)

PG&E Response (Cont'd)

The Diablo Canyon Emergency Plan will be revised prior to issuance of the OL to describe the existence and functioning of the On-Site Technical Support Center.

Task Force Position (Errata No. 9)

Records that pertain to the as-built conditions and layout of structures, systems and components shall be stored and filed at the site and accessible to the Technical Support Center under emergency conditions. Examples of such records include system descriptions, general arrangement drawings, piping and instrument diagrams, piping system isometrics, electrical schematics, wire and cable lists, and single line electrical diagrams. It is not the intent that all records described in ANSI N45.2.9-1974 be stored and filed at the site and accessible to the Technical Support Center under emergency conditions; however, as stated in that standard, storage systems shall provide for accurate retrieval of all pertinent information without undue delay. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

PG&E will have available as-built drawings and other appropriate records at the site which will be accessible to the On-Site Technical Support Center.

Section 2.2.2.c - On-Site Operational Support Center

Task Force Position

An area to be designated as the On-Site Operational Support Center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

PG&E will establish an On-Site Operational Support Center prior to initial power operation. The center will not be needed prior to that time. This will be an assembly area where plant operating and maintenance personnel will be gathered to be utilized as directed by the Emergency Control Center and will initially be located in the assembly room of the Temporary Administration Building. This center will have direct communication with the control room and the Emergency Control Center by intercom and phone. A permanent location for the On-Site Operational Support Center is now being evaluated and the status of the design of the permanent center will be provided to the NRC as part of the Diablo Canyon TMI Long-Term Studies Status Report. The first status report is scheduled for submittal on December 31, 1979.

Task Force Position

The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

PG&E Response

The Diablo Canyon Emergency Plan will be revised prior to issuance of the OL to describe the existence and functioning of the On-Site Operational Support Center.

Section 2.2.3 - Revised Limiting Conditions for Operation of Nuclear Power Plants Based Upon Safety System Availability

Task Force Position

All NRC nuclear power plant licensees shall provide information to define a limiting operational condition based on a threshold of complete loss of safety function. Identification of a human or operational error that prevents or could prevent the accomplishment of a safety function required by NRC regulations and analyzed in the license application shall require placement of the plant in a hot shutdown condition within 8 hours and in a cold shutdown condition within 24 hours.

The loss of operability of a safety function shall include consideration of the necessary instrumentation, controls, emergency electrical power sources, cooling or seal water, lubrication, operating procedures, maintenance procedures, test procedures and operator interface with the system, which must also be capable of performing their auxiliary or supporting functions. The limiting conditions for operation shall define the minimum safety functions for modes 1, 2, 3, 4, and 5 of operation.

The limiting conditions of operation shall require the following:

1. If the plant is critical, restore the safety function (if possible) and place the plant in a hot shutdown condition within 8 hours.
2. Within 24 hours, bring the plant to cold shutdown.
3. Determine the cause of the loss of operability of the safety function. Organizational accountability for the loss of operability of the safety system shall be established.
4. Determine corrective actions and measures to prevent recurrence of the specific loss of operability for the particular safety function and generally for any safety function.
5. Report the event within 24 hours by telephone and confirm by telegraph, mailgram or facsimile transmission to the Director of the Regional Office, or his designee.

Section 2.2.3 (Cont'd)

6. Prepare and deliver a Special Report to the NRC's Director of Nuclear Reactor Regulation and to the Director of the appropriate regional office of the Office of Inspection and Enforcement. The report shall contain the results of Steps 3 and 4, above, along with a basis for allowing the plant to return to power operation. The Senior Corporate Executive of the licensee responsible and accountable for safe plant operation shall deliver and discuss the contents of the report in a public meeting with the Office of Nuclear Reactor Regulation and the Office of Inspection and Enforcement at a location to be chosen by the Director of Nuclear Reactor Regulation.

7. A finding of adequacy of the licensee's Special Report by the Director of Nuclear Reactor Regulation will be required before the licensee returns the plant to power. (Implementation schedules will be established by the Commission in the course of the immediately effective rulemaking.) The Task Force recommends that the rulemaking process be initiated promptly.

<u>PG&E Response</u>

When this Task Force position or some alternate becomes a regulation, PG&E will comply.

SEISMIC DESIGN IMPLICATIONS OF THE TMI ACCIDENT TO DIABLO CANYON

In its report "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578), the NRC Regulatory staff does not explicitly address any implications of the TMI-2 accident with respect to seismic design. However, during its 232nd general meeting, the Advisory Committee on Reactor Safeguards (ACRS) indicated its interest in this subject in relation to the licensing of Diablo Canyon. Specific ACRS concerns have not yet been expressed and it is PG&E's understanding that the ACRS intends to examine this issue in future meetings. In this report, PG&E has attempted to anticipate some of the ACRS concerns and to address seismic design implications of the TMI-2 accident.

Individual sections of this report responding to the recommendations contained in NUREG-0578 also address seismic design considerations relating to those recommendations. In many cases, the Regulatory Staff has implicitly addressed seismic design considerations in the NUREG-0578 recommendations by including requirements for equipment and systems qualified in accordance with safety grade requirements.

This section of PG&E's report discusses the seismic design implications of the TMI-2 accident in a broader context, not limited to the recommendations of NUREG-0578. Those aspects of Diablo Canyon seismic design are summarized which are unique to Diablo Canyon or which differ significantly from operating plants and other near-term OL plants. The TMI-2 accident sequence, as given in "Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement" (NUREG-0600) is examined to identify equipment and systems whose failure or malfunction contributed to the accident or which were required to limit the accidents severity or mitigate its consequences. Finally, some conclusions are drawn concerning the seismic design implications of the TMI-2 accident in relation to Diablo Canyon.

The general conclusion of this section is that seismic considerations at Diablo Canyon do not result in any significant increase in risk when compared to other plants. Satisfaction of the short-term recommendations contained in NUREG-0578 should be a sufficient condition for licensing of Diablo Canyon as well as for other near-term OL plants.

The accident at TMI-2 was in no way related to any seismic event. Historically, to the best of our knowledge, no commercial power reactor in the United States has ever experienced an accident, incident, or even a transient which was caused or affected by a seismic event. This historical experience would certainly suggest that, for nuclear power plants in general, seismic risk is not a significant contributor to overall risk to public health and safety. Certainly, the probability of an accident such as that at TMI-2 being caused or affected by a seismic event is extremely low when compared to the probability of such an accident occurring from non-seismic causes.

The seismic design of the Diablo Canyon units is, indeed, significantly different from the seismic design for other operating and near-term OL plants. Diablo Canyon is located on the central California coastline, a region which has a history of seismic activity. As a consequence, a very large portion of the licensing process for Diablo Canyon has been concerned with the subjects of geology, seismology, and seismic design. The original seismic design basis for the plant, approved in connection with licensing proceedings leading to a construction permit required that structures, systems, and components important to safety accommodate high seismic inputs.

As a result of the identification of a geologic fault offshore from the plant site (the Hosgri fault), the operating licensing process for the plant was greatly expanded and extended. As a result of this process, the plant has been extensively modified to accommodate seismic inputs associated with a severe seismic event assumed to occur on the Hosgri fault. This effort was supported by a program of analysis, testing, and regulatory review unprecedented in the nuclear industry. The result of this program is a plant with a seismic design basis more severe than other operating and near-term OL plants but with greatly increased assurance of adequate capability to accommodate the seismic design basis. With respect to overall plant safety, the higher seismic risk at Diablo Canyon due to the seismic environment has been balanced by increased seismic capability and increased assurance that this seismic capability really exists. Consequently, the risk to the public health and safety at Diablo Canyon due to seismic considerations is comparable to the risk associated with other operating and near-term OL plants located in less seismically active areas of the United States.

The review of Diablo Canyon seismic design by the NRC Regulatory staff has been extremely thorough. In the process of its review, the Regulatory staff set forth a number of extraordinary requirements which have not been applied to other operating and near-term OL plants. One example of this is the staff's requirement that the Diablo Canyon units be capable of achieving cold shutdown, using only seismically qualified systems and components and assuming any single failure. This requirement has been met. These staff requirements, unique to Diablo Canyon, further reduce risk to public health and safety associated with the seismic environment of the plant.

PG&E has reviewed various descriptions of the TMI-2 accident, including the information contained in NUREG-0600, in an attempt to identify seismic design implications for Diablo Canyon.

The issue of primary concern in this respect is whether equipment and systems whose failure caused or contributed to the TMI-2 accident might also be subject to similar failures due to a seismic event. Of secondary concern is the question of whether equipment and systems required at TMI-2 for accident recovery or mitigation would be required for a similar accident at Diablo Canyon and, if required, whether they would be available following a seismic event.

Based on this review, it is PG&E's conclusion that the seismic design of Diablo Canyon, in combination with the recommendations contained in NUREG-0578, provides adequate assurance that a seismic event at Diablo Canyon would not cause an accident such as that at TMI-2. Additionally, if such an accident is presumed to occur at Diablo Canyon, equipment and systems necessary for accident recovery and mitigation would be available and functional following a seismic event.

PG&E is awaiting the ACRS expression of its concerns on this subject and stands ready to furnish any information needed by the Committee. We believe that, after consideration of the seismic design implications of the TMI-2 accident in relation to Diablo Canyon, the ACRS will conclude that licensing of Diablo Canyon can proceed, along with licensing of other near-term OL plants.