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ACCESSION NBR: 8001290255 DOC. DATE: 80/01/25 NOTARIZED: NO DOCKET #  
 FACIL: 50-275 Diablo Canyon Nuclear Power Plant, Unit 1, Pacific Gas 05000275  
 50-323 Diablo Canyon Nuclear Power Plant, Unit 2, Pacific Gas 05000323  
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SUBJECT: Forwards Addendum to 791231 rept, "Status Rept to NRC from Util, Responding to Nureg-0578: TMI-2 Lessons Learned Task Force Status Rept & Short-Term Recommendations & Clarifications." Addendum responds to re 800109 meeting

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1. The first part of the document discusses the importance of maintaining accurate records. It emphasizes that without proper documentation, it is difficult to track progress and identify areas for improvement.

2. The second part of the document focuses on the role of communication in a team. It states that clear and open communication is essential for ensuring that all team members are on the same page and working towards common goals.

3. The third part of the document addresses the challenges of time management. It suggests that creating a structured schedule and prioritizing tasks can help individuals make the most of their available time.

4. The fourth part of the document discusses the importance of flexibility. It notes that while having a plan is important, being able to adapt to changes and unexpected circumstances is equally crucial for success.

5. The fifth part of the document explores the concept of continuous learning. It encourages individuals to stay curious and open to new ideas, as this mindset is vital for staying relevant in a rapidly changing world.

6. The sixth part of the document discusses the value of networking. It highlights that building a strong professional network can provide valuable support, resources, and opportunities throughout one's career.

7. The seventh part of the document focuses on the importance of self-care. It reminds individuals that taking care of their physical and mental health is not just a luxury, but a necessity for maintaining high performance.

8. The eighth part of the document discusses the role of leadership. It suggests that effective leaders are those who inspire their teams, provide clear direction, and create a supportive environment for growth.

9. The ninth part of the document addresses the importance of accountability. It notes that holding oneself and others accountable for their actions is a key factor in achieving long-term success.

10. The final part of the document concludes by emphasizing that success is a journey, not a destination. It encourages individuals to stay motivated, persevere through challenges, and embrace the opportunities that come their way.

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Bethesda, Maryland 20014

Re: Docket No. 50-275  
Docket No. 50-323  
Diablo Canyon Units 1 and 2

Dear Mr. Denton:

Enclosed are 20 copies of an Addendum dated January 25, 1980, to the PG&E report entitled "Status Report to the Nuclear Regulatory Commission from Pacific Gas and Electric Company Responding to NUREG-0578: 'TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations' and Clarifications" which was submitted on December 31, 1979.

This Addendum responds to questions raised at the January 9, 1980 meeting with NRC representatives in Bethesda, Maryland.

Kindly acknowledge receipt of the above material on the enclosed copy of this letter and return it to me in the enclosed addressed envelope.

Very truly yours,

*Philip A. Crane*

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STATUS 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"  
AND CLARIFICATIONS

Docket # 50-275/323  
Control # 8001290255  
Date 1/05/80 of Document  
REGULATORY DOCKET FILE

December 31, 1979

ADDENDUM January 25, 1980



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A. Task Force Position 1

PG&E Response and Status

All of the four pressurizer heater groups can be supplied with power from the offsite power sources when they are available. In addition, provisions will be made to provide power to two out of four heater groups from the emergency power source through the Engineered Safety Features (ESF) buses when offsite power is not available. 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.

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

Circuit breakers 52-1G-72 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. The seismic qualification is based on PG&E's vast testing experience which has previously demonstrated that electromechanical equipment can withstand numerous seismic tests simulating high seismic events without damage and that the equipment will be available to perform its safety function after the seismic event.

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.

Design is currently underway to incorporate clarifications 4 and 7 into the present design. This modification will be completed by April 1, 1980.



NUREG-0578 ITEM 2.1.1

A. Task Force Position 2

PG&E Response and Status

PG&E will develop the required procedures and implement the proper training of the operators. 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. Loading of each ESF bus can be accomplished from the main control board (see TFP3 below). Procedures will identify under what conditions and which selected loads can be shed from the ESF bus to prevent overloading when the pressurizer heaters are connected. The procedures will include provisions to ensure that the heaters are transferred to the ESF power source as described in TFP3 below. The time required to transfer the power supplies will be less than 10 minutes and will expose the operator to no more than 10 mRem.

The procedures will be written and approved, and the operators trained, by May 1, 1980.



NUREG-0578 ITEM 2.1.1

A. Task Force Position 3

PG&E Response and Status

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 the 100 foot elevation 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, 52-1G-72) are open and manually throws the deenergized transfer switches. This action in itself will not connect the pressurizer heaters onto the ESF buses. Only when the Shift Foreman is informed that the transfers have been made, will the heaters again be controlled using the normal control devices provided on the main control console. Even with manual transfer, there is no problem in meeting the Westinghouse estimated time requirement of providing pressurizer heaters on emergency power within one hour after the accident. PG&E will provide control room indication of actual wattage being supplied to each heater group that has been transferred to the emergency power sources. The operating and emergency procedures will be changed to reflect these requirements and changes by May 1, 1980.



NUREG-0578 ITEM 2.1.3.b

Task Force Position A.

PG&E Response and Status

Procedures and Description of Existing Instrumentation

The Westinghouse Owners' Group, of which PG&E is a member, has performed analyses as required by Item 2.1.9 to study the effects of inadequate core cooling. These analyses were provided to the NRC "Bulletins and Orders Task Force" for review on October 31, 1979. As part of the submittal made by the Owners' Group, an "Instruction to Restore Core Cooling during a small LOCA" was included. This instruction provides the basis for procedure changes and operator training required to recognize the existence of inadequate core cooling and restore core cooling based on existing instrumentation. PG&E will incorporate the key considerations of this instruction into the LOCA procedures, and provide training to the operators in this area.

Subcooling Meter

PG&E is installing a Combustion Engineering PWR subcooled margin monitor. The equipment will be ordered by February 1, 1980. Equipment delivery is expected by May 1, 1980 and installation will be completed by June 1, 1980. Details of display, calculator and inputs are as follows:

Display

The display indicates either the temperature or pressure margin continuously on a digital type display and on an analog meter. The analog meter provides a redundant display. Both displays are located on the main control board in the control room. There is a low alarm at 30<sup>0</sup> F and a low-low alarm at 20<sup>0</sup> F. The uncertainty



### ITEM 2.1.3.b continued

in the monitor is 0.5 percent. The overall uncertainty is estimated to be  $\pm 10^{\circ}\text{F}$ . The range of the display is  $-40$  to  $+200^{\circ}\text{F}$ . The display is qualified to meet the seismic requirements of IEEE 344.

#### Calculator

The calculator module is a dedicated digital type. The plant process computer is not used. The calculator module is not redundant. The selection logic uses the highest temperature and the lowest pressure. The module is qualified to meet the seismic requirements of IEEE 344.

#### Inputs

##### 1. Temperature

The subcooled meter has 10 temperature inputs. Four temperature signals come from each of the four hot leg wide range RTD's and six temperatures are taken from core exit thermocouples. All temperature signals have a range of 0 to  $700^{\circ}\text{F}$ . The uncertainty is estimated to be  $\pm 5^{\circ}\text{F}$  with no data available to evaluate the difference between forced flow and natural circulation conditions. At the present time, none of the temperature inputs are safety grade, however, PG&E is upgrading these inputs to meet safety grade requirements.

##### 2. Pressure

Pressure is sensed by two safety grade reactor coolant loop Barton Model 763 pressure transmitters. Both pressure transmitters are located on the hot leg of loop 4. The pressure range is 0 to 3000 psi with an uncertainty of  $\pm 15$  psi. These



ITEM 2.1.3.b continued

transmitters are seismically qualified to the requirements of IEEE 344 and environmentally qualified to the requirements of IEEE 323.

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	<u>TSAT - T, P - PSAT</u>
Display Type (Analog, Digital, CRT)	<u>Digital and Analog</u>
Continuous or on Demand	<u>Continuous</u>
Single or Redundant Display	<u>Redundant</u>
Location of Display	<u>Control Board</u>
Alarms (include setpoints)	<u>30<sup>0</sup>F, 20<sup>0</sup>F</u>
Overall uncertainty ( <sup>0</sup> F, PSI)	<u>±10<sup>0</sup>F</u>
Range of Display	<u>-40, +200<sup>0</sup>F</u>
Qualifications (seismic, environmental, IEEE 323)	<u>Seismic</u>

Calculator

Type (process computer, dedicated digital or analog calc.)	<u>Dedicated Digital</u>
If process computer is used specify availability. (% of time)	<u>N/A</u>
Single or redundant calculators	<u>Single</u>
Selection Logic (highest T., lowest press)	<u>High T - low P</u>
Qualifications (seismic, environmental, IEEE 323)	<u>Seismic</u>
Calculational Technique (Steam Tables, Functional Fit, ranges)	<u>Steam Tables</u> <u>0-3200 psi</u> <u>0-999<sup>0</sup>F</u>

Input

Temperature (RTD's or T/C's)	<u>4 RTD's, 6 T/C's</u>
------------------------------	-------------------------



ITEM 2.1.3.b continued

Temperature (number of sensors and locations)	<u>Note 1</u>
Range of temperature sensors	<u>.0-999<sup>0</sup>F</u>
Uncertainty* of temperature sensors ( <sup>0</sup> F at 1)	<u>±5<sup>0</sup></u>
Qualifications (seismic, environmental, IEEE 323)	<u>Note 1</u>
Pressure (specify instrument used)	<u>Barton Model 763</u>
Pressure (number of sensors and locations)	<u>2 on Loop 4 Hot Leg</u>
Range of Pressure sensors	<u>0-3000 psi</u>
Uncertainty* of pressure sensors (PSI at 1)	<u>15 psi</u>
Qualifications (seismic, environmental, IEEE 323)	<u>Seismic, environmental</u>
<u>Backup Capability</u>	
Availability of Temp & Press	<u>Yes</u>
Availability of Steam Tables, etc.	<u>Yes</u>
Training of operators	<u>See below</u>
Procedures	<u>See below</u>

\*Uncertainties must address conditions of forced flow and natural circulation.

Note 1: See section on Inputs, above.

Procedures to recognize inadequate core cooling are presently being written. The procedures will be approved and the operators trained by May 1, 1980. As described in PG&E's NUREG-0578 response August 27, 1979, the process computer has been programmed to provide margin to saturation information.



Task Force Position B

PG&E Response and Status

Additional Instrumentation to Indicate Inadequate Core Cooling

The submittal referenced in A above described the capabilities of the core exit thermocouples in determining the existence of inadequate core cooling conditions and their superiority in some instances to the loop RTDs for measuring true core conditions. Other means of determining the approach to or existence of inadequate core cooling could be:

1. Reactor vessel water level
2. Incore detectors
3. Excore detectors
4. Reactor coolant pump motor currents
5. Steam generator pressure

A discussion of the possible use of these measurements is addressed below.

The use of incore movable detectors to determine the existence of inadequate core cooling conditions appears doubtful. The detectors could be driven in to the tops of the incore thimbles, which are located at the top of the core, following an accident in which concern for inadequate core cooling exists. The problem comes in the lack of sensitivity of the detectors to very low neutron levels and changes that would occur due to core uncovering. Gamma detectors could perhaps be employed, but they suffer from similar sensitivity problems, and the fact that gamma levels in the fuel region change insignificantly between the covered and uncovered condition. As a result, it does not appear worthwhile to pursue incore movable detectors as a means of determining inadequate core cooling conditions.



ITEM 2.1.3.b continued

The use of excore detectors has been mentioned as a possibility in responding to core uncover. The only detectors which would have the required sensitivity are the source range monitors. Since the intermediate and power range monitors are not sensitive enough to the low level changes resulting from vessel voiding. The use of the source range monitors will be investigated further as part of the more indepth study of inadequate core cooling being performed by the Westinghouse Owners' Group. However, their use is probably limited to those instances when significant voiding exists in the downcomer region, since normally water in the downcomer would effectively shield the detectors from the core region whether voids existed or not.

Reactor coolant pump motor current, which could be indicative of core voiding, is inappropriate for a reliable means of determining inadequate core cooling, since a loss of offsite power or pump trip due to a LOCA blowdown would shut down the pumps.

Steam generator pressure, which already exists, is useful in the case where heat transfer from primary to secondary is interrupted due to loss of natural circulation. This, however, does not satisfy requirements to indicate the approach to inadequate core cooling, nor does it indicate the true condition of the core.

Reactor vessel water level determination is the most promising of the items discussed to provide additional capability of determining the approach to and the existence of inadequate core cooling. Several systems for determining water level are under review by the Westinghouse Owners' Group. A conceptual design of one system is given below:



ITEM 2.1.3.b continued

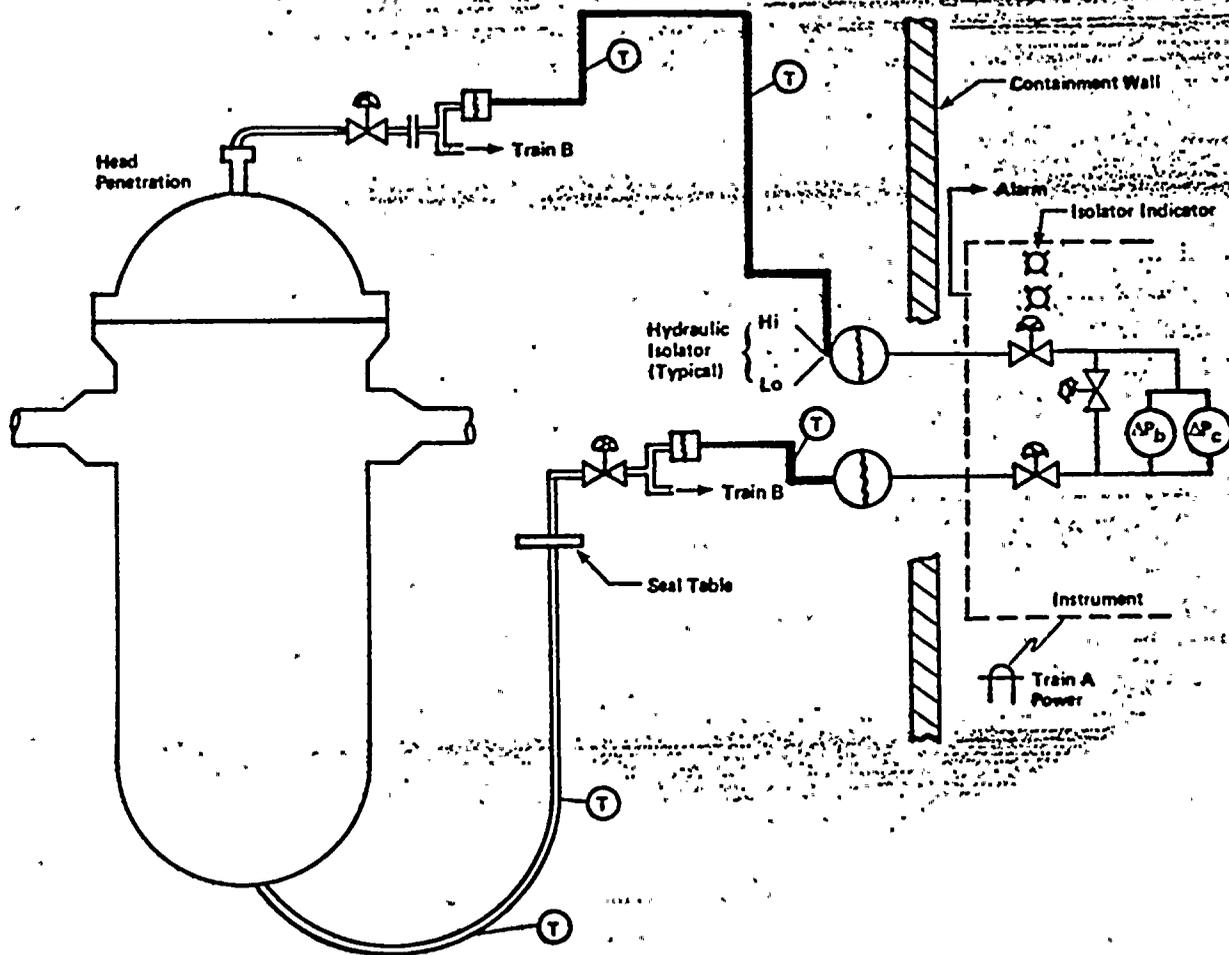
Vessel Level System Description.

After examining many different methods and principles for determining the water level in the reactor vessel, a basic delta pressure measurement from the bottom of the vessel to the top of the vessel appears to provide the most meaningful and reliable information to the operator. One of the reasons for choosing this system is that the sources of potential errors are better known for this system than for any other new or untested system.

The attached figure shows a simplified sketch of the proposed vessel level instrumentation system. The bottom tap of the instrument would use a thimble of the incore movable detector system either at the seal table or in the thimble below the vessel. Use of the thimble as part of the incore flux monitoring would not be lost. The flux thimble guide tube would be tapped below the vessel and an instrument line connection made. The instrument line would have an isolation valve and slope down to a hydraulic coupler connected to a sealed reference leg. The sealed reference leg would go to the differential pressure transmitter located at a higher elevation above the expected level of containment flooding. A similar sealed leg would go to the top of the vessel and penetrate the head using the vent line or a special connection on a spare RCC mechanism penetration. Two trains of vessel level instrumentation would be provided.

The behavior of the signal generated by this level instrument under normal and accident conditions is being evaluated. The usefulness of this instrument to provide an unambiguous indication of inadequate core cooling is being evaluated as part of Item 2.1.9. The potential errors and accuracy





PROCESS CONNECTION SCHEMATIC (TRAIN A)



ITEM 2.1.3.b continued

of a final system configuration are being evaluated to assess its usefulness to provide information to the operator for proper operation of a vessel venting system and for normal water level control during periods when the primary system is open and a water level may exist in the vessel. The connection of the level system to the vessel head should be designed to be compatible with the head vent system. Operation of the vent system should not upset all indications of vessel level. This can easily be avoided by using a separate instrument tap or by using more than one location.

PG&E believes that the proposed Westinghouse indication system to indicate adequate core cooling will provide unambiguous indication of adequate core cooling during all postulated reactor coolant system conditions. In addition the same indication system will provide indication of reactor water level during no flow conditions. Detailed analyses and system descriptions will be provided when available from Westinghouse. Delivery is not scheduled at this time.

PG&E is presently evaluating technical proposals from Westinghouse for a reactor vessel level indication system. PG&E and other near-term license applicants have met with the NRC to discuss the requirements given in the draft Regulatory Guide 1.97, Revision 2. These requirements in final form may well include additional instrumentation requirements. PG&E will comply with any new requirements.



NUREG-0578 ITEM 2.1.4

Task Force Position 4

PG&E Response and Status

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.

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

1. "Nonessential" process lines are defined as those which 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



ITEM 2.1.4 continued

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.

Table 2.1.4-1 of PG&E's August 27, 1979 response to NUREG-0578 provides the identification of nonessential, essential, and safety systems penetrating containment.

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.

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. 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



ITEM 2.1.4 continued

will remain closed even after the isolation signal is reset. FSAR  
Figure 7.3-43 shows a typical AOV circuit.

Control of Liquid Radwaste Isolation Valves is somewhat different than described above. Each valve does not receive an individual isolation signal. Instead four (4) valves inside the containment are closed by one signal from Train A and 5 valves outside of containment are closed by one signal from Train B.

As presently designed some of the valves could open automatically after manual reset of the isolation signal and the Radwaste Isolation Valves control circuit for each Train.

This scheme will be modified to require manual control of each valve after resetting the above signals. Implementation of these changes will be completed by May 1, 1980.



NUREG-0578 ITEM 2.1.5.a

PG&E Response and Status

Although recombiners located inside the containment are being provided for Diablo Canyon, PG&E has chosen to comply with the Task Force Position and clarifications concerning dedicated penetrations for external recombiner or post-accident purge system.

Design changes have been made to provide dedicated penetrations and isolation systems for the hydrogen recombiner and hydrogen purge systems. It is anticipated, depending upon equipment availability, these modifications will be completed by July 1, 1980.

The supply lines for the existing Containment Hydrogen Purge Systems use containment penetrations which are solely dedicated for that purpose. The exhaust lines previously were teed-off from the 48-inch containment purge lines. In response to NUREG-0578, PG&E has selected different containment penetrations (previously designated as spares) which will be solely dedicated to post-LOCA hydrogen control. New lines will be from those penetrations to the existing iodine filters and blowers of the Containment Hydrogen Purge exhaust lines.

The capability of adding external post-LOCA hydrogen recombiners will be facilitated by placing piping tees and blank flanges in the post-LOCA hydrogen purge lines at a place where it would be convenient to install external recombiners.

The areas selected for the control panels for the external recombiners and for the piping tee connections will be analyzed to evaluate the possible personnel exposures in connecting and operating the recombiners during the accident conditions. This analysis will be completed prior to March 15, 1980.



PG&E Response and Status

Hydrogen recombiners were not included as a design basis for licensing Diablo Canyon Units 1 and 2, however, PG&E has chosen to install hydrogen recombiners inside containment at Diablo Canyon. Westinghouse hydrogen recombiners and associated equipment have been purchased and are presently at the site. Installation of the recombiners will be completed by May 1, 1980.

The recombiners will be housed inside the containment and therefore will not present any shielding problems during operation of the recombiners. Procedures for operation of the installed hydrogen recombiners will be written and the operators trained prior to May 1, 1980.



PG&E Response and Status

At intervals not exceeding 18 months, operating pressure leak tests will be performed on appropriate portions of SIS, RHR, NSS Sampling, LRW and GRW Systems. Pressurized systems will be visually inspected for leakage into the building environment. Any observed leakage will be eliminated. An attempt will be made during the initial testing to measure system makeup during the period of pressurization, where the system configuration permits. System makeup rate provides some indication of system leakage, but may not be conclusive because of valve seat leakage. Initial testing will be conducted prior to fuel loading.

Where feasible, liquid containing systems will be pressurized with a hydro pump and makeup determined by measuring level changes in a graduated tank, or by normal operating pressure sources. During pressurization each system will be walked down and visually inspected for leaks. Visual inspection or other appropriate means will be used to differentiate system boundary leakage from test boundary leakage to determine actual leakage outside the system. There are portions of systems that will be pressurized with a system pump, and since makeup rate cannot be measured, any leaks discovered during the walkdown will be evaluated to determine the approximate leakage from that system. Baseline leakage rates will be recorded after appropriate measures to eliminate undesirable leakage paths have been taken.

For the Gaseous Radwaste System the immediate leak reduction plan will consist of a preoperational measurement, followed by an operational monitoring program.



ITEM 2.1.6.a continued

With the waste gas compressors running on recirc to maintain impeller seals the system will be maintained at maximum operating pressure using a regulated nitrogen source. The system makeup rate will be determined with a gas flowmeter at the nitrogen regulator. All non-welded connections in the system will be checked with a soap and water solution to locate leaks and appropriate means taken to eliminate them. This is the preoperational measurement. Thereafter (during operation), a routine radiation monitoring program will be implemented to detect minute changes in activity of air in the areas occupied by the system. If leakage is indicated by an increase in airborne activity, appropriate means, such as explosive gas detectors or soap, will be used to localize the leak.

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 by the reactor coolant system water inventory balance. Excessive leakage into controlled areas will be indicated by abnormally high airborne radioactivity levels.

Some portions of the CVCS are used for boron recycling during normal operation. The boron recycling system will not be used under post-accident circumstances, and is not included in this program.

Containment spray recirc piping and valves outside containment are leak rate tested with precision volumetrics equipment at a 24 month frequency as part of a containment isolation valve surveillance test.



ITEM 2.1.6.a continued

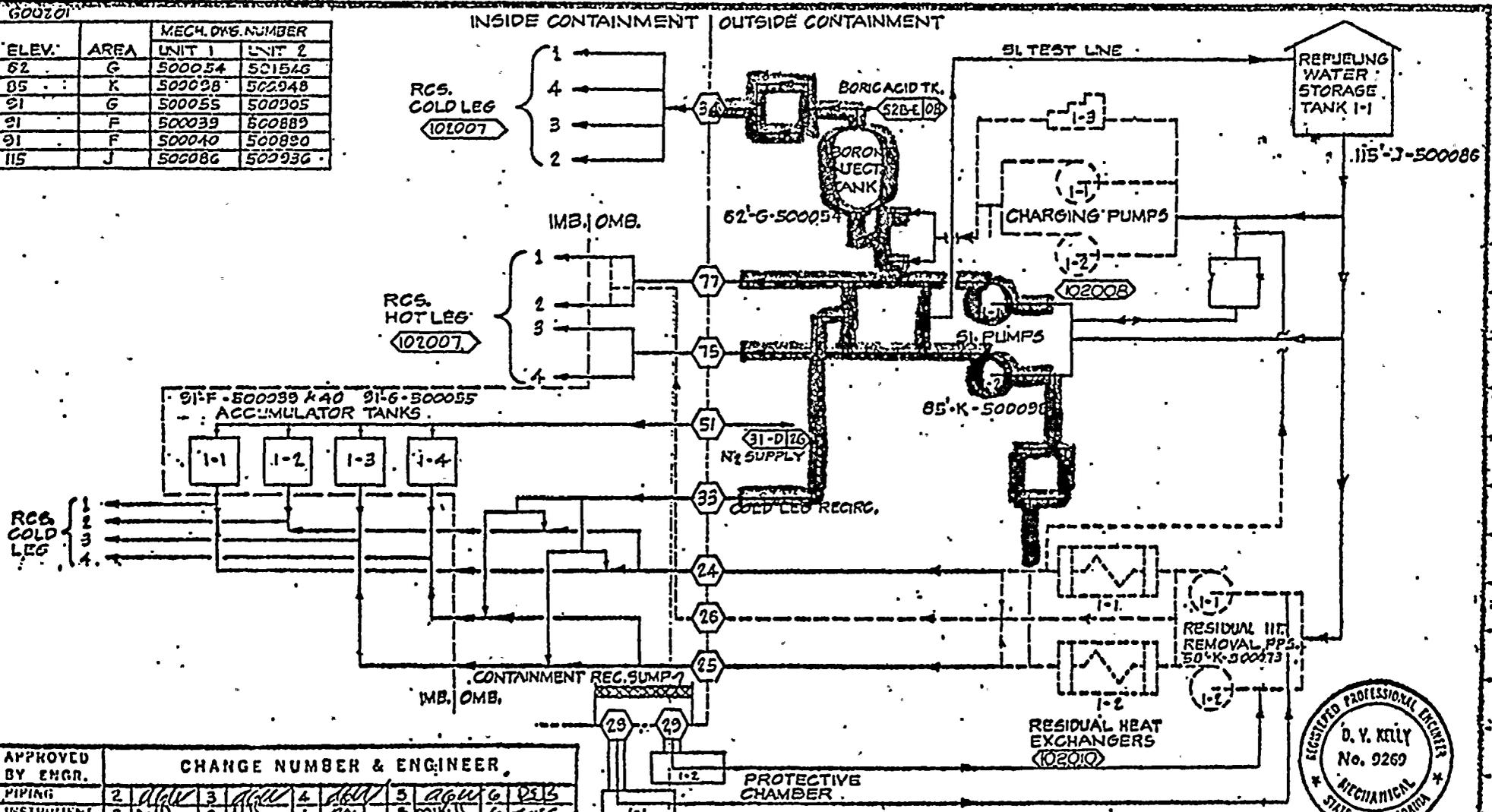
Individual Surveillance Test Procedures will be written for each system or each half system in the case of SIS and RHR. Each STP will list system boundary valves and specify test pressures. Each test will be performed prior to fuel loading, and at refueling outage intervals thereafter.

The portions of systems that will be monitored are indicated on the attached piping schematics.



600201

ELEV.	AREA	MECH. DWG. NUMBER	
		UNIT 1	UNIT 2
62	G	500054	501546
85	K	500088	500948
91	G	500055	500905
91	F	500039	500889
91	F	500040	500890
115	J	500086	500936



APPROVED BY ENGR.	CHANGE NUMBER & ENGINEER									
PIPING	2	AGW	3	AGW	4	AGW	5	AGW	6	DES
INSTRUMENT.	2	NR	3	WR	4	WR	5	WR	6	ZHC
SYSTEMS	2	WR	3	WR	4	WR	5	WR	6	ZHC
FRUIT/CHM 1	2	WR	3	WR	4	WR	5	WR	6	ZHC

NO.	DATE	DESCRIPTION	BY	CH.	APRD. ENGR.
6	5-3-78	SEE DESCRIPTION OF CHANGES SH 2B	AGW	LM	AGW
5	7-20-76	SEE DESCRIPTION OF CHANGES SH. 2B	AGW	LM	AGW
4	11-12-74	SEE DESCRIPTION OF CHANGES SH. 2A, 2B	J.M.	LM	J.M.
3	4-8-74	SEE DESCRIPTION OF CHANGES SH. 2A	J.M.	LM	J.M.
2	11-8-72	CHANGED SHEET 2 -> 5 PER DETAILS SHEET 2A	L.P.B.	PS	L.P.B.
1	5-8-71	APPROVED FOR CONSTRUCTION	L.P.B.	PS	L.P.B.

BASIC DIAGRAM UNITS 1 & 2 INDEXED REV 6



PIPING SCHEMATIC  
SAFETY INJECTION SYSTEM

DIABLO CANYON  
DEPARTMENT OF ENGINEERING  
PACIFIC GAS AND ELECTRIC COMPANY  
SAN FRANCISCO, CALIFORNIA

BILL OF MATERIAL	
DRAWING LIST	
SUPERSEDES	
SUPERSEDED BY	
SHEET NO. 107	9 1/2 SHEETS
DRAWING NUMBER	CHANGE
102009	6

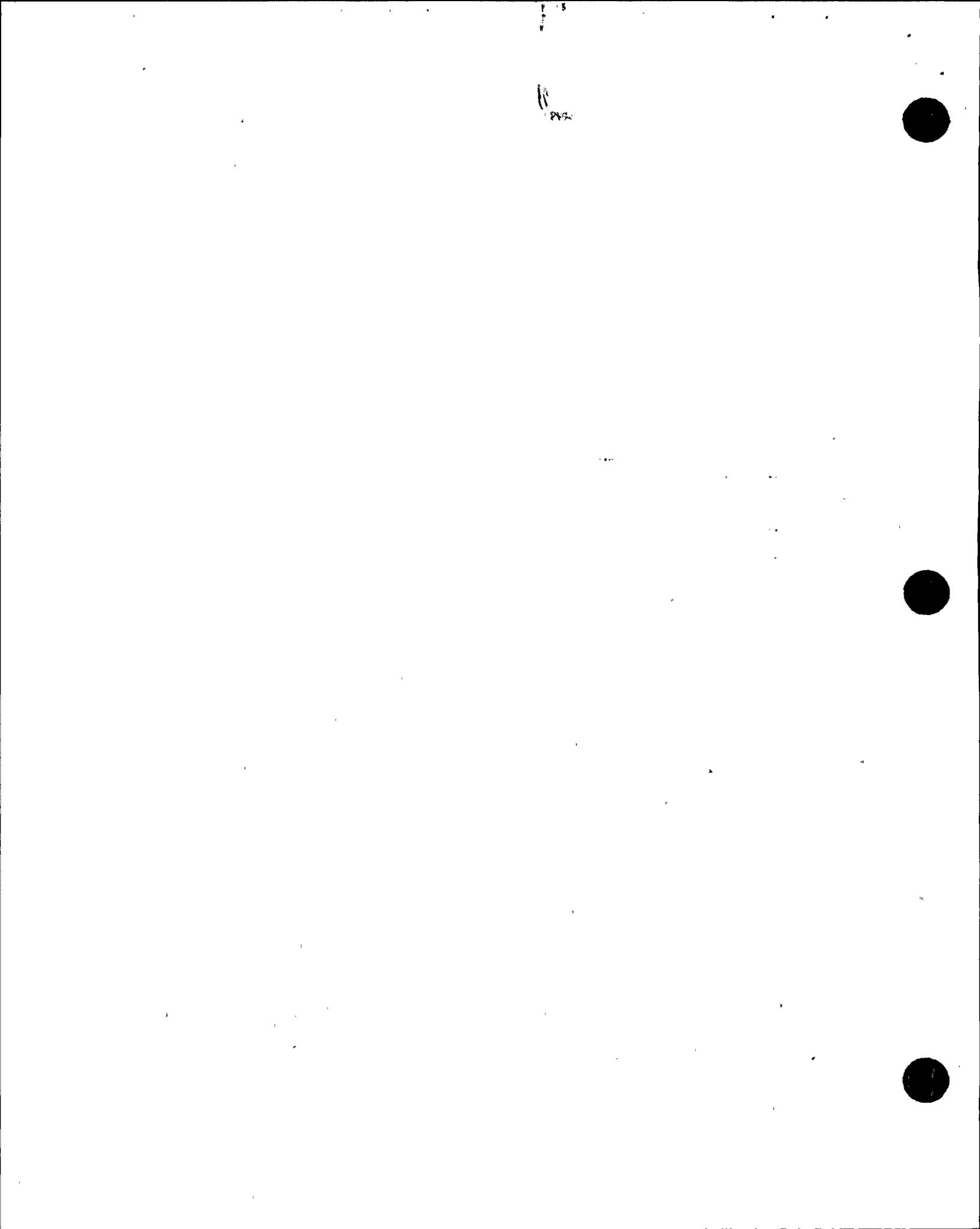
INTR. CHECK: CMV

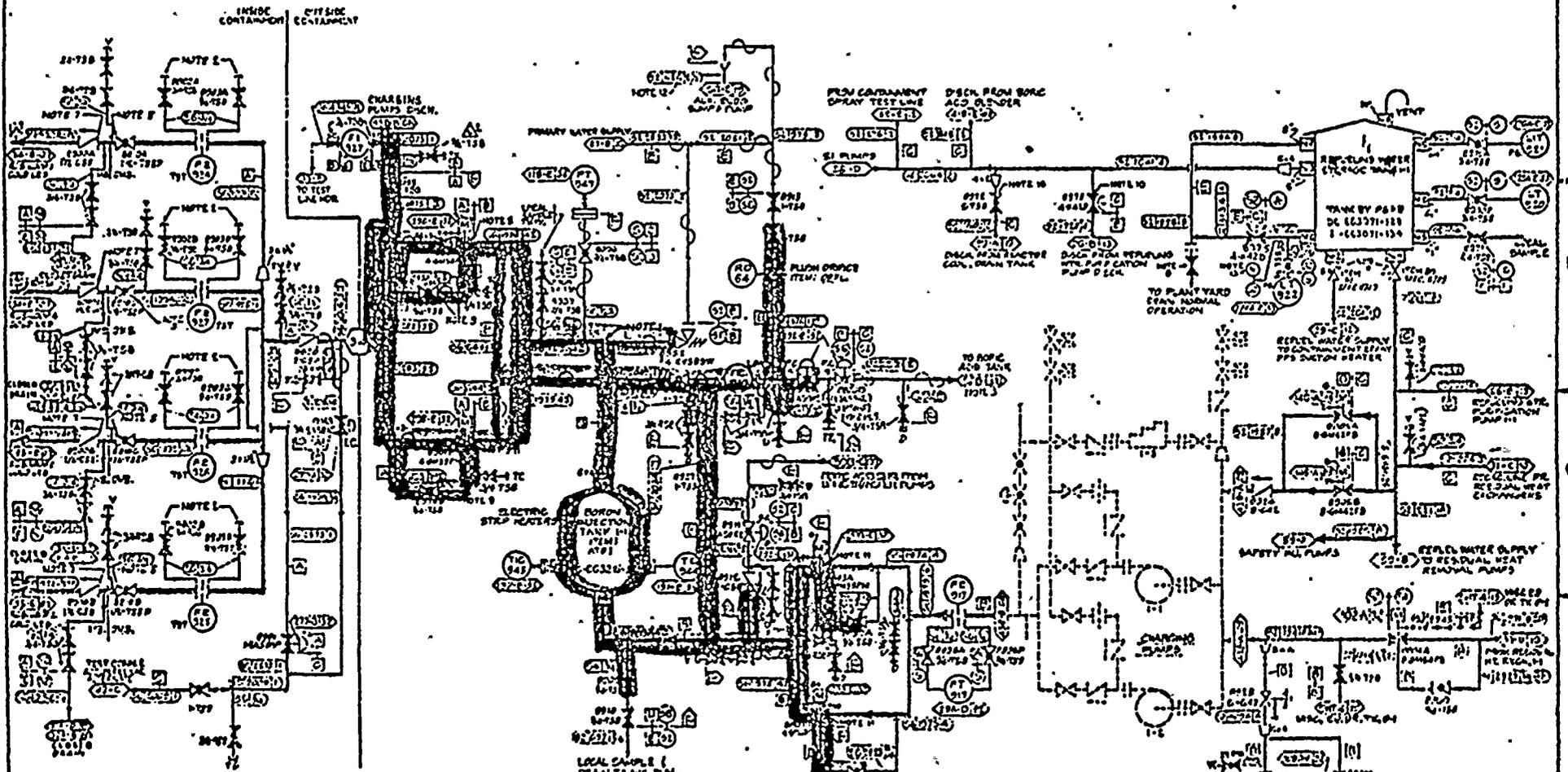
LINE & VALVE CHECK

SCALE CHECK

35 MAR 1971

466 27





**NOTES:**

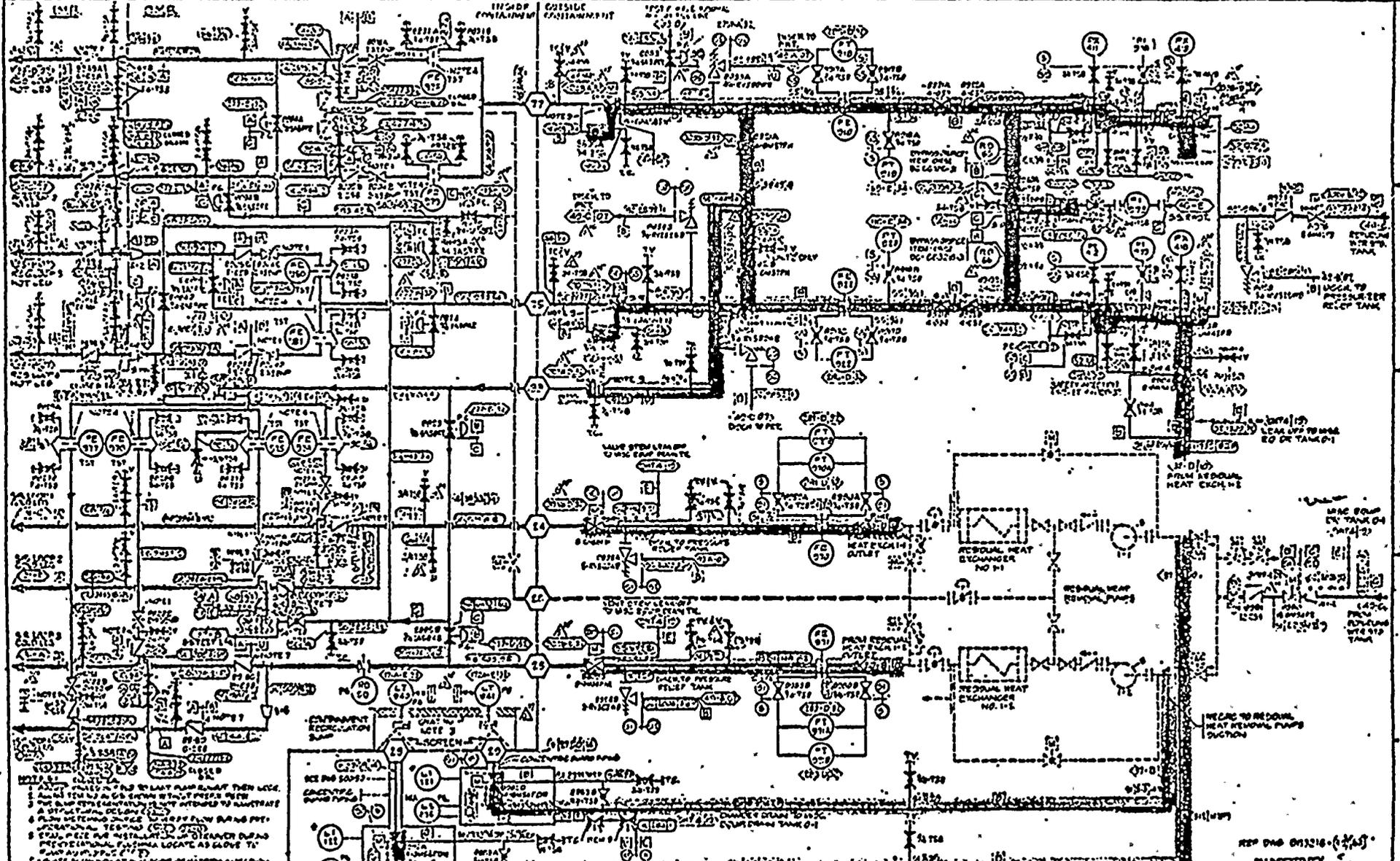
1. LOCATE CONNECTION CLOSE TO TANK (111)
2. FLOW METER RANG ORANGE TO VERIFY FLOW (111)
3. DURING PRE-OPERATIONAL TESTING
4. HEAT TRACE LINE ALL THE WAY TO TANK (111)
5. PS-1 INDICATES POSSE CO. SCOP
6. ADJUST VALVES IN FIELD TO OBTAIN PROPER FLOW (111)
7. RANGOUT FLOW FROM ROOMS FANS SHELL
8. ALL TESTS SHOULD BE MADE IN THIS AREA DURING WITHOUT PRESSURE (111)
9. SPECIAL LINE INDICATION INCLUDING OVER (111)
10. ANY WATER IN CONTAINMENT IS TO BE PLACED (111)
11. WITH THESE USES MUST BE INSTALLED DRAWING (111)
12. TO BE USED FOR DRAIN TANK ON (111)
13. FOR UNIT 1 - ONLY (111) (111) (111)
14. UNIT OPERATED (111) (111) (111)
15. USE DASHED LINE VALVES IN UNIT 1 AND SOLID LINE VALVES IN UNIT 2 (111) (111) (111)
16. LINE NO. 6 APPLICABLE TO UNIT 1 ONLY
17. UNIT 2 UNITS ENGINEERING LINE NO. 6 (111)
18. FIELD TO INSTALL BUSH PER 111-343 046-653033 (111)

REF. DWG. DC-65325 (1) (1)

P. G. & E. CO.	NO. CHANGE	
	102009	6
SHEET 4 OF 9 SHEETS		

-23-





-24-

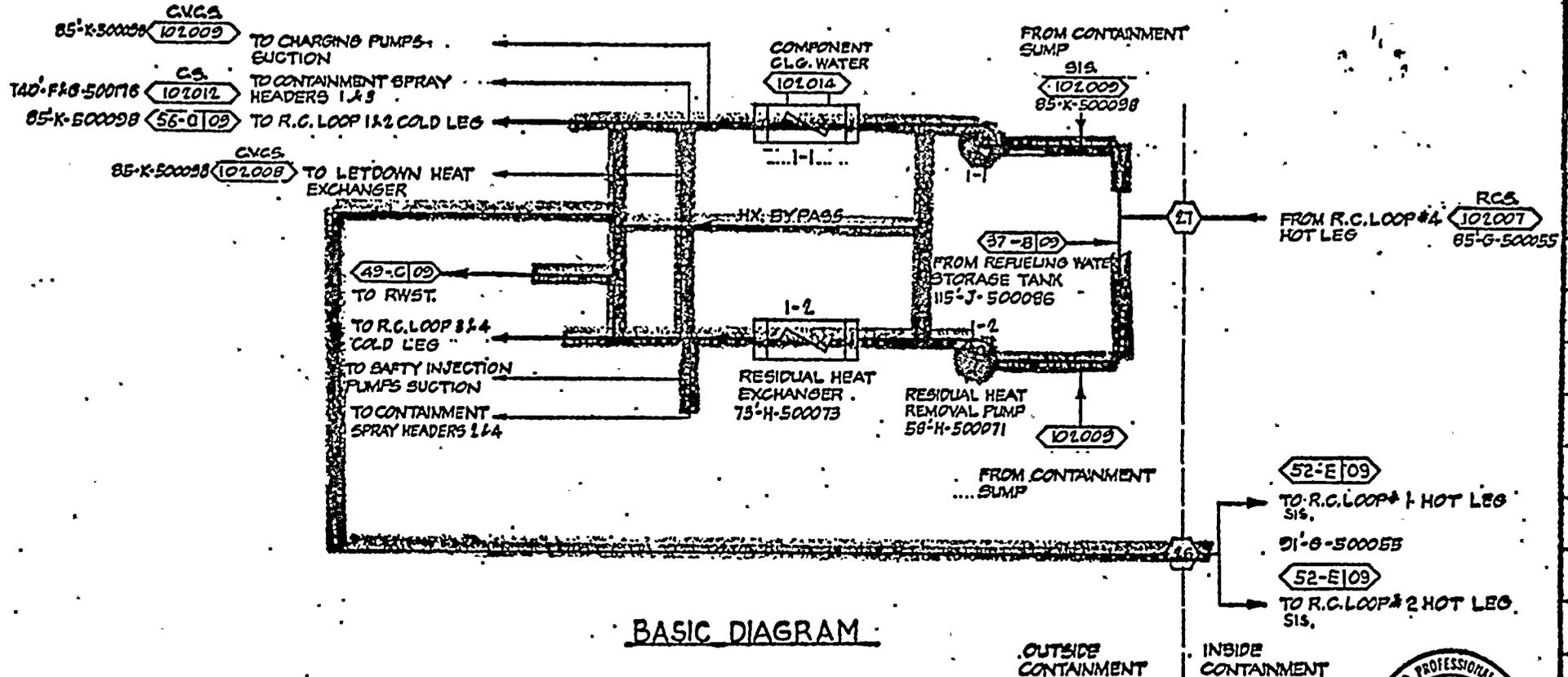
1. ALL PIPING TO BE INSTALLED AND TESTED WITH WATER.
2. ALL PIPING TO BE CLEANED AND DRYED OUT BEFORE USE.
3. ALL PIPING TO BE INSULATED TO PREVENT HEAT LOSS.
4. ALL PIPING TO BE SUPPORTED TO PREVENT SAGGING.
5. ALL PIPING TO BE PROTECTED FROM CORROSION.
6. ALL PIPING TO BE PROTECTED FROM VIBRATION.
7. ALL PIPING TO BE PROTECTED FROM OVERHEATING.
8. ALL PIPING TO BE PROTECTED FROM FREEZING.
9. ALL PIPING TO BE PROTECTED FROM MECHANICAL DAMAGE.
10. ALL PIPING TO BE PROTECTED FROM ELECTRICAL INTERFERENCE.

P. G. & E. CO. DRAWING NO. CHANGE  
 SHEET 5 OF 5 SHEETS 102009 5

157111



010201



BASIC DIAGRAM

OUTSIDE CONTAINMENT INSIDE CONTAINMENT



DCPP INFORMATION ONLY

DO NOT USE FOR OPERATIONS

COAST VALLEYS DIVISION

PIPING SCHEMATIC RESIDUAL HEAT REMOVAL SYSTEM

DIABLO CANYON

DEPARTMENT OF ENGINEERING PACIFIC GAS AND ELECTRIC COMPANY SAN FRANCISCO, CALIFORNIA

BILL OF MATERIAL
DRAWING LIST
SUPERSEDES
SUPERSEDED BY
SHEET NO. 1 OF 6 SHEETS
DRAWING NUMBER
102010
CHANGEL
4

ELEV.	AREA	MECH. DWG. NO.	
		UNIT 1	UNIT 2
58	H	500071	500921
85-91	G	500055	500905
79	H	500073	500923
85	K	500098	500948
115	J	500086	500936
140	F&G	500176	501510

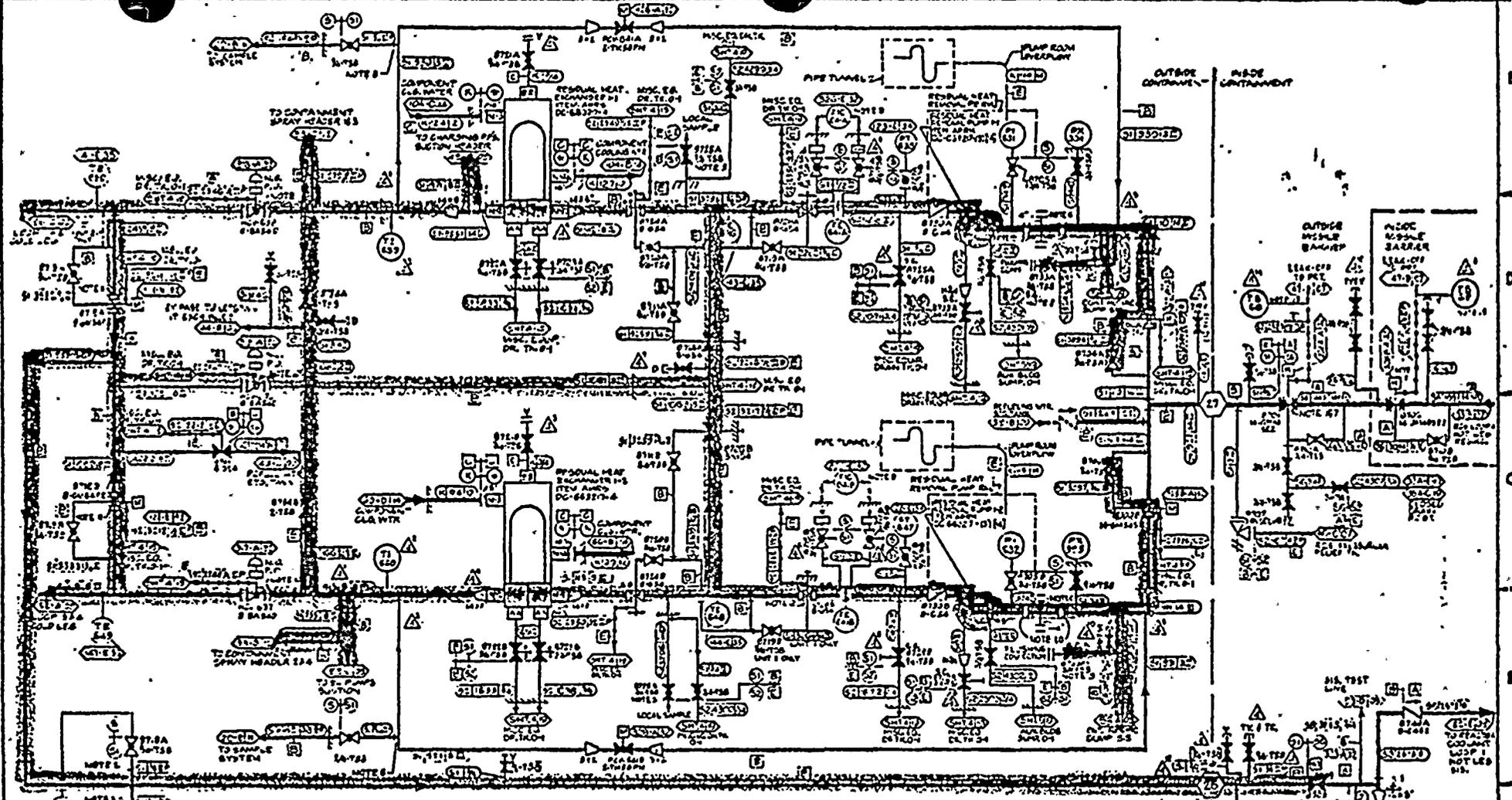
APPROVED BY ENGR.	CHANGE NUMBER & ENGINEER			
PIPING	2	3	4	
INSTRUMENT	2	3	4	
SYSTEMS	2	3	4	
PROJECT MECH	2	3	4	

NO.	DATE	DESCRIPTION	GM	BY	CH.	APPR. LEAD ENGR.
4	6-22-76	SEE DESCRIPTION OF CHANGES SH. 2	169972	J.M.	L.M.	RIL
3	2-6-76	SEE DESCRIPTION OF CHANGES SH. 2	169972	J.M.	L.M.	RIL
2	5-22-74	SEE DESCRIPTION OF CHANGES SH. 2	169972	J.M.	L.M.	RIL
1	5-17-72	APPROVED FOR CONSTRUCTION	169972	L.P.B.	M.G.G.	RIL

TABLE OF CHANGES

APPROVED BY	GM
DR. E. HORVATH	CH. L. MONALAC
O.K. A.H.E.	DATE 4-3-68
None	SCALE





**NOTES:**

- 1 VALVE INTERLOCKED WITH REACTOR COOLANT SYSTEM PRESSURE SIGNAL (12-0)
- 2 ELBOW TAPS FOR FLOW METERS (12-0)
- 3 LOCATE ABOVE RESIDUAL HEAT REMOVAL PUMP SHELDS (12-0)
- 4 ALL (12-0) ITEMS ARE SHOWN WITHOUT SPECIFIC PSE AC. REFER TO (12-0) STATUS REPORT.
- 5 LOCATE VALVE OUTSIDE SHIELD WALL SAMPLE LINE MUST BE AT A LOWER ELEVATION THAN THE 8 INCH PIPE BEING SAMPLED (12-0)
- 6 TEMPORARY STRAINER IS PLACED IN THE SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS. STRAINER MUST BE REMOVED BEFORE PLANT START-UP (12-0)
- 7 SPECIAL LINE ISOLATION INCLUDING VALVE (12-0)
- 8 LOCATE SAMPLE LINE CONNECTION AS CLOSE AS POSSIBLE TO THE SPEC. RUN IN DISCHARGE LINE. (12-0)
- 9 LOCATE ORIFICE FLOW METER LOCATION OUTSIDE OF SHIELD WALL. (12-0)
- 10 200" 20 50 P-N WELD (12-0) AND ATCS FLOW-ARREST SPOOLS PLACED ON LINE NO. 101-B
- 11 WELD 816 ALSO TO BE BARRING FOR UNIT 2 AND ALSO FOR UNIT 2 DC-64210-0211 (12-0)
- 12 USE CRANE VALVE WITH SOLID CSR FOR UNIT 2 DC-64210-440 (12-0)
- 13 T2-60.67 ARE TO BE STAPPED TO PIPE 16-24" FROM VALVE (12-0)

REF. DWG. DC-64210-8

P. G. & E. CO.	DRAWING NO.	CHANGE
	SHEET 3 OF 6 SHEETS	102010

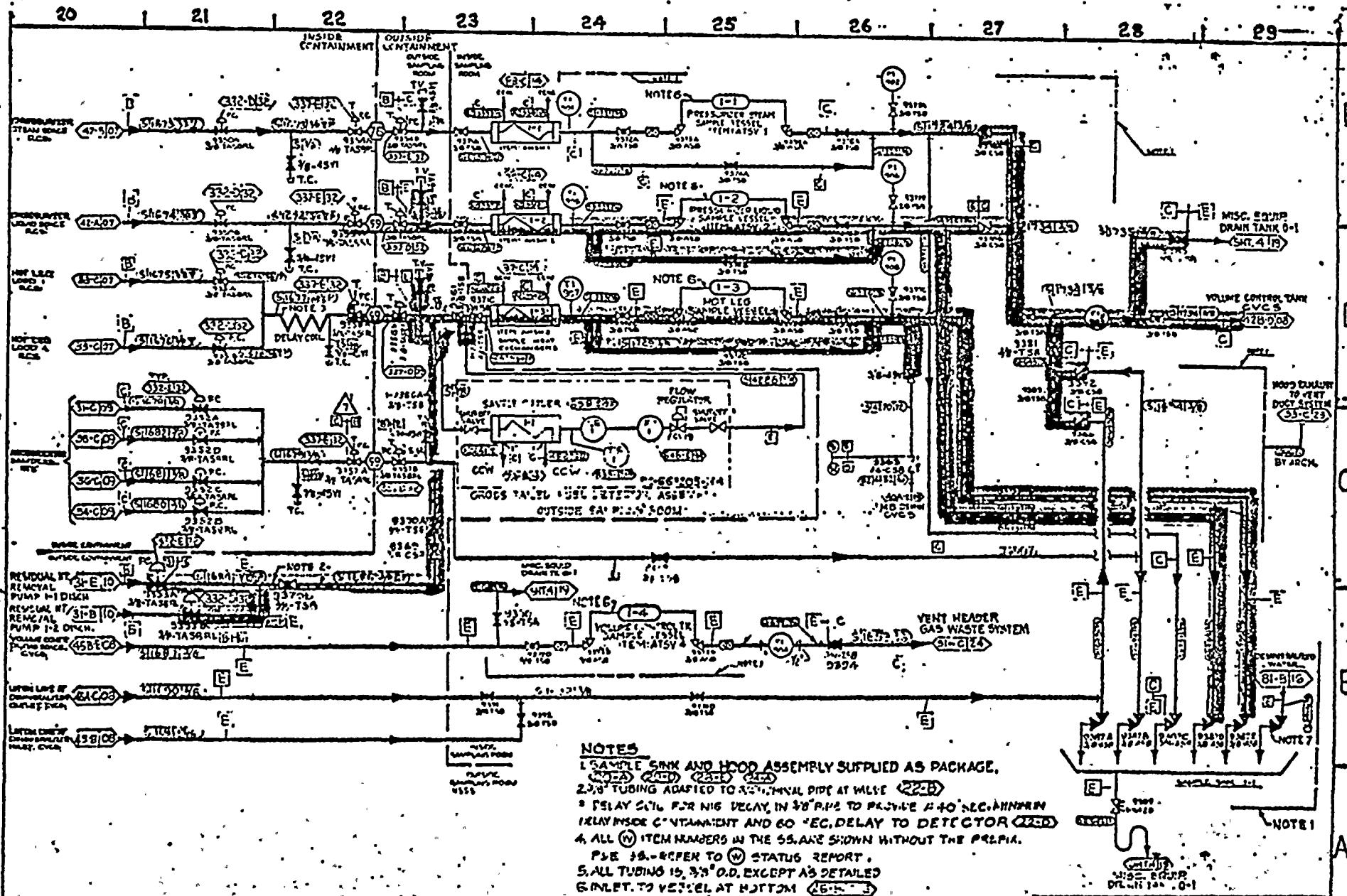
INLET





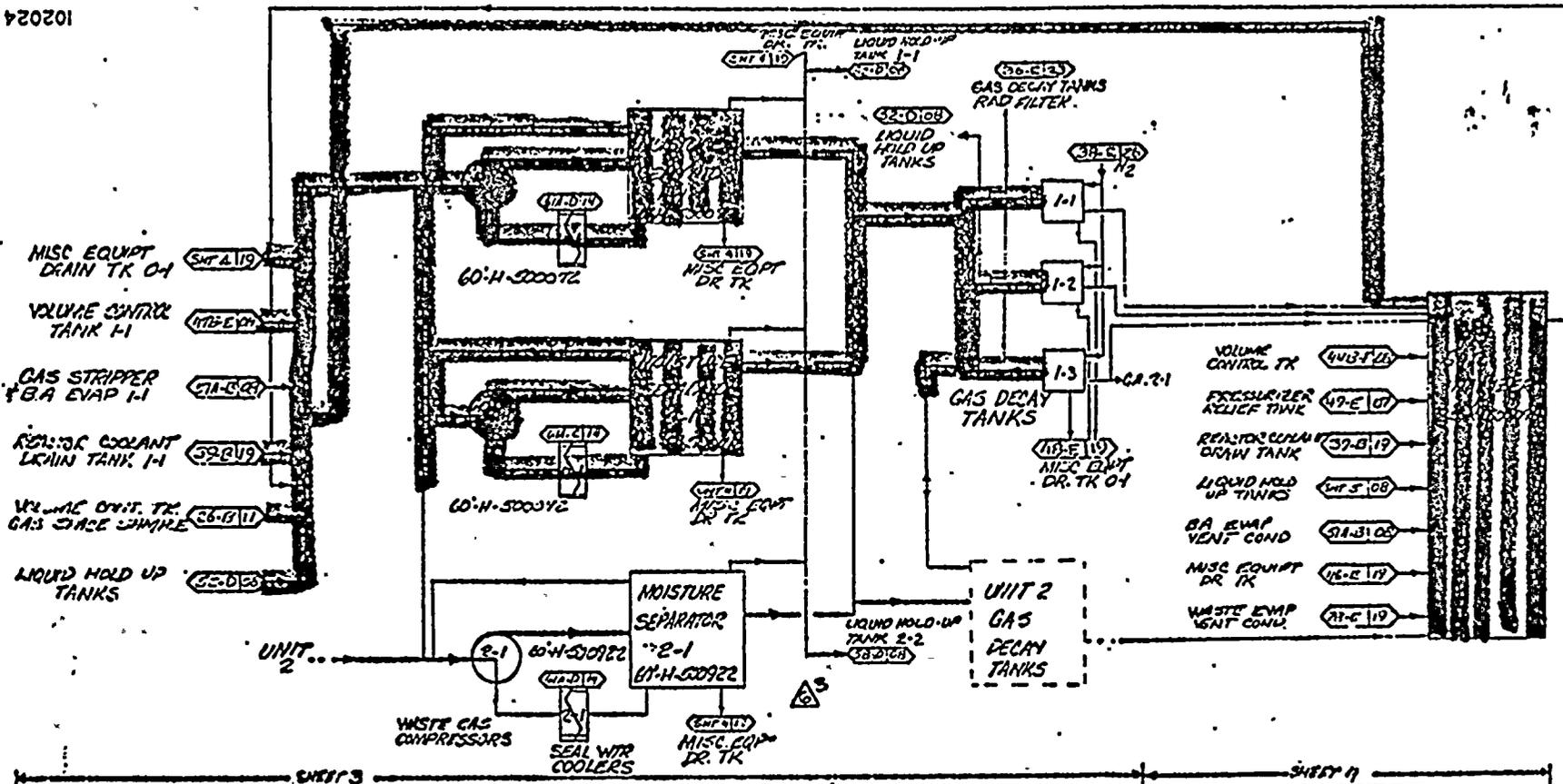


NOTE: Highlited paths represent portions of system used under post-accident circumstances.



- NOTES**
1. SAMPLE SINK AND HOOD ASSEMBLY SUPPLIED AS PACKAGE.
  2. 3/8" TUBING ADAPTED TO 1/2" MIN. PIPE AT VALVE (22-B)
  3. DELAY VALVE FOR NIS DELAY IN 3/8" PIPE TO POSITIVE #40 SEC. MINIMUM IN LAYERS OF CONTAINMENT AND 60 SEC. DELAY TO DETECTOR (22-B)
  4. ALL (W) ITEM NUMBERS IN THE SKETCH SHOWN WITHOUT THE PREFIX.
  5. P&E IS REFERRED TO (W) STATUS REPORT.
  6. ALL TUBING IS 3/8" O.D. EXCEPT AS DETAILED.
  7. LICE 'WHITEY' G-156-SPEC. 2602 VALVE ON LINE 194 (25-B)





BASIC DIAGRAM

ELEV	AREA	DNG NO (MECH)	
54	H	UNIT 1	UNIT 2
60	H	500071	500922
65	K	500072	500923
		500098	



APPROVED BY ENGR.	CHANGE NUMBER & ENGINEER					
PIPING	2	1661	3	1661	4	1661
INSTRUMENT	2	1661	3	1661	4	1661
SYSTEMS	2	1661	3	1661	4	1661
PROJECT CHIEF	2	1661	3	1661	4	1661

NO.	DATE	DESCRIPTION	GM	BY	CH.	APPD
6	5-12-73	SIZE DESCRIPTION OF CHANGES, SH. 2A	167027	COM	EMV	JSG
5	4-13-76	SEE DESCRIPTION OF CHANGES SH. 2A	167027	DOM	CV	TR. M.
4	10-3-74	CHANGED SHTS. 2A, 3, 4 PER SHEET 2A	167027	AS	LM	JSG
3	3-26-74	CHANGED SHTS. 2-4 PER DETAILS SHT 2	167027	RSP	LVN	JSG
2	1-22-73	REVISED SHTS 2-4 PER SHEET 2	167027	TAC	LPB	JSG
1	6-23-72	APPROVED FOR CONSTRUCTION	167027	PI	E.P.	J.S.

PER INDEXED REV. 6 UNITS 1 & 2

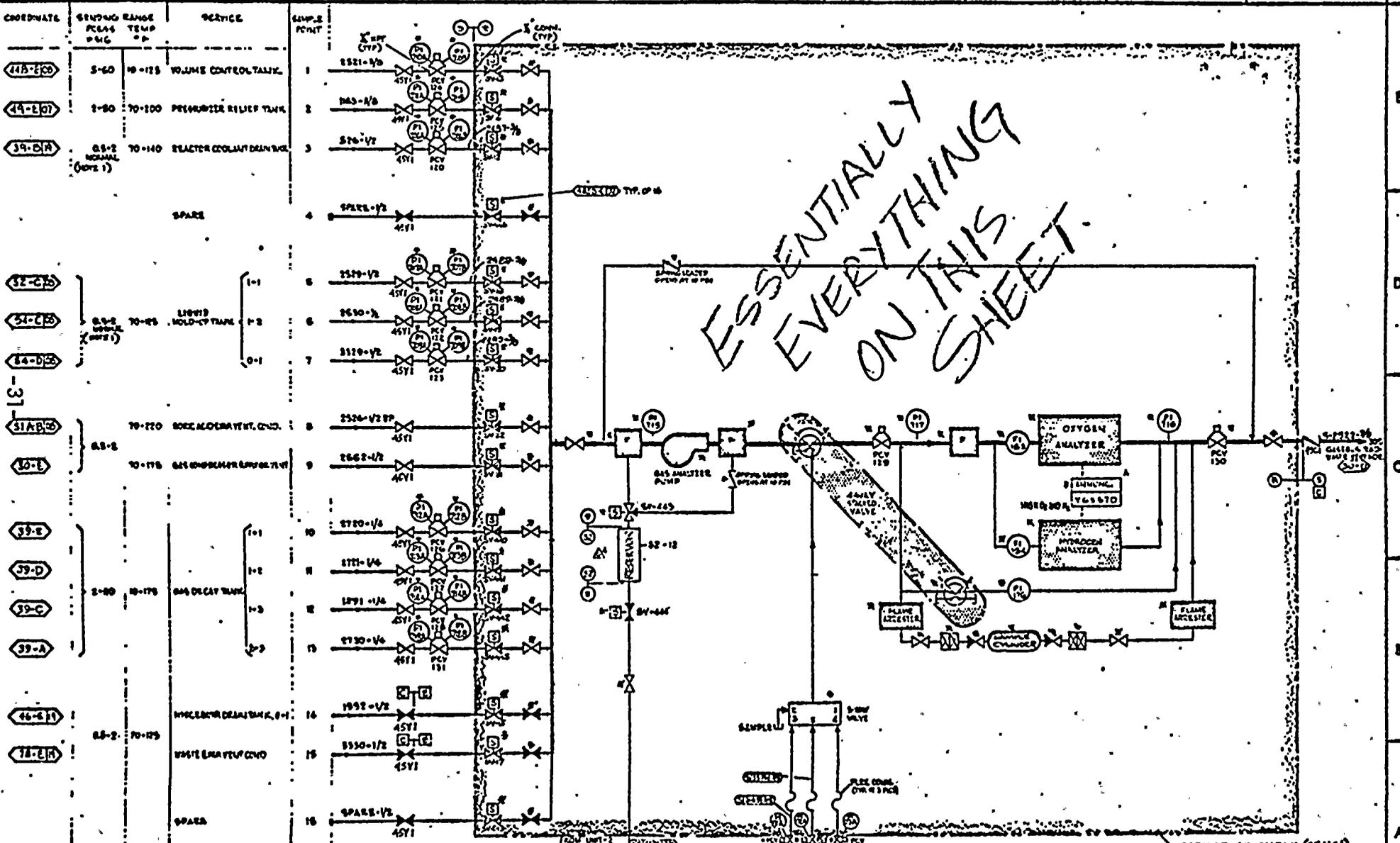
PIPING SCHEMATIC  
 GASEOUS RADWASTE SYSTEM  
 DIABLO CANYON  
 DEPARTMENT OF ENGINEERING  
 PACIFIC GAS AND ELECTRIC COMPANY  
 SAN FRANCISCO, CALIFORNIA

BILL OF MATERIAL	-
DRAWING LIST	-
SUPERSEDES	-
SUPERSEDED BY	-
SHEET NO. 1 OF 5 SHEETS	
DRAWING NUMBER	102024
CHANGE	6









ESSENTIALLY  
EVERYTHING  
ON THIS  
SHEET.

NOTES:  
1. PIPING TO BE ORDERED BY OTHERS IF  
SERVICED ON TANK DOES NOT REQUIRE.  
2. VALVES WITH P&E MARK NO. ARE SPECIFIED  
BY CPEC. 8102 - P&E DIA. NO. 053477  
3. ALL TUBING: P&E CLASS CODE IS UNLESS NOTED OTHERWISE.

4. ALL ROSELINES: P&E CLASS B,  
UNLESS NOTED.

**GAS ANALYZER 1-1**  
DC-663276-2

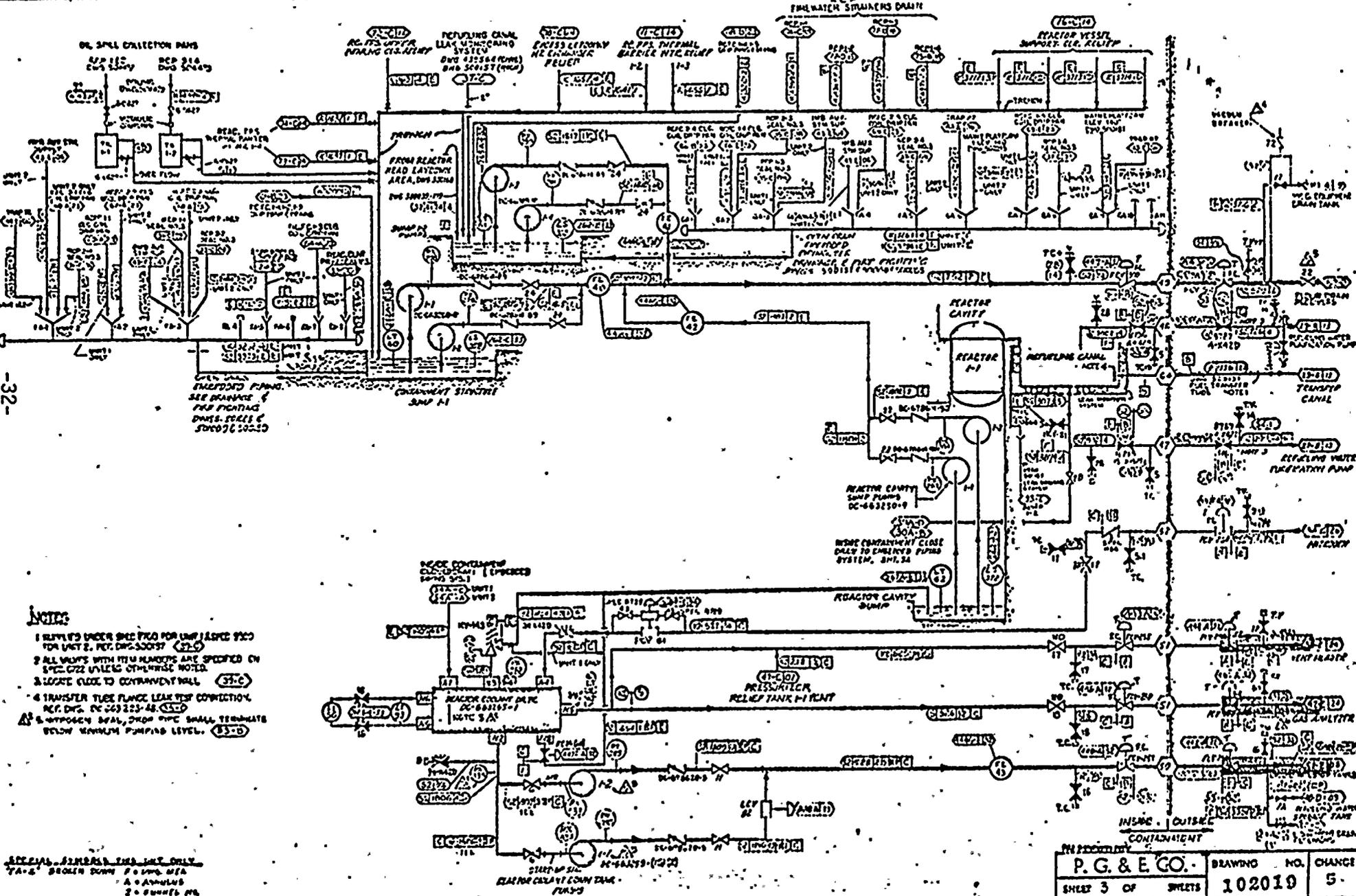
CABINET ASSEMBLY (BY MAA)  
PC-663276-1

<b>P. G. &amp; E. CO.</b> SHEET 4 OF 5 SHEETS	DRAWING NO. <b>102024</b> CHANGE <b>6</b>
--	--

CALIBRATION ONE VALVE (FIELD MARKED BY MAA)  
 LOCATED AT THE PLACE OF ANALYZER CONTROL PANELING SHALL BE 1119 00-00  
 LOCATED ABOVE THE NORTH WALL OF THE HYDROGEN TANK SHALL BE 1119 00-00

REVISED 6





-32-

SEE DRAWING & P&ID FOR PORTING DATA. SCALE & DATE 05/26/60

**NOTES**

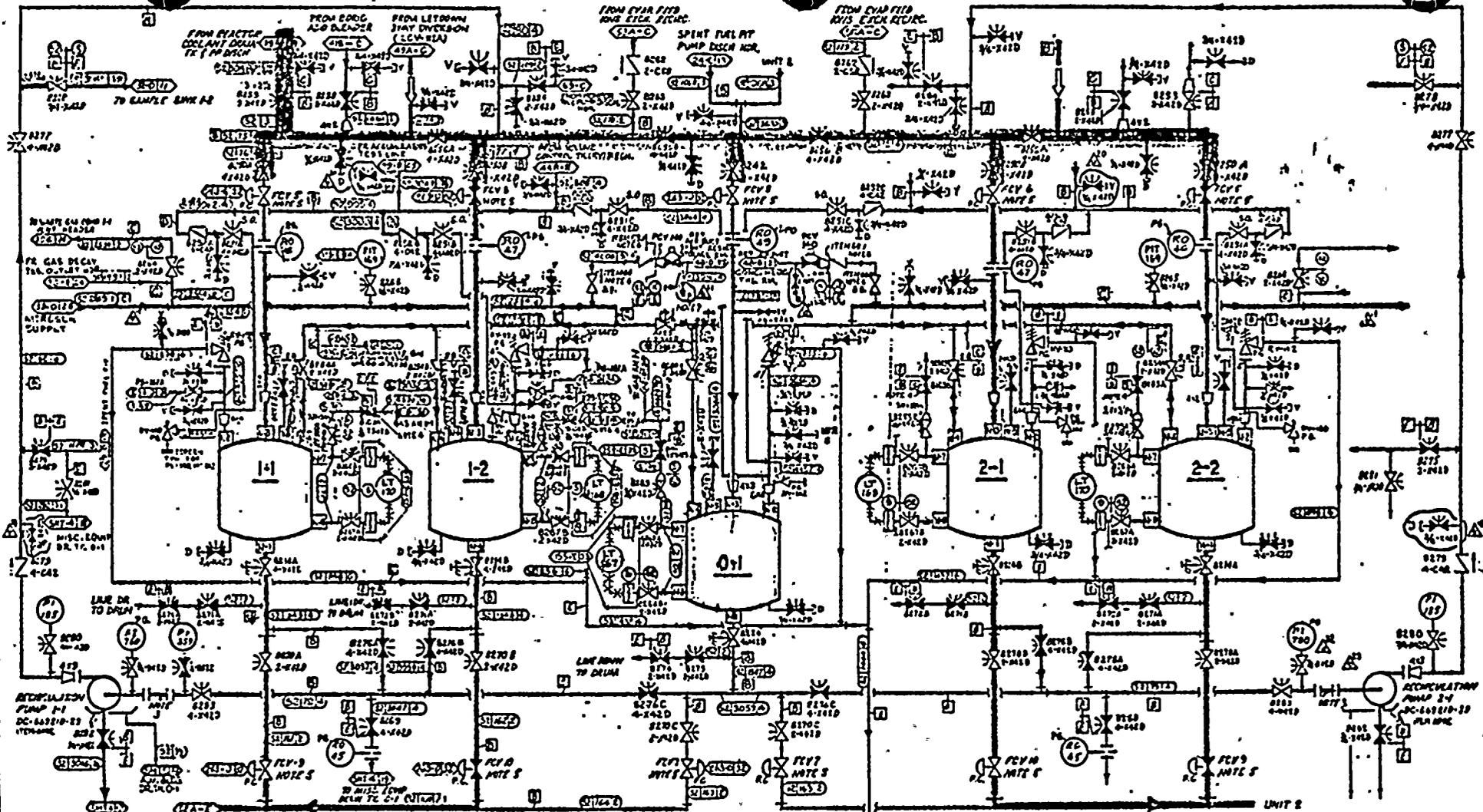
1. REFER TO UNDER SPEC P&ID FOR UNIT (SCALE 90%) FOR UNIT 2. REF. DWG. 30037 (31)
2. ALL VALVES WITH ITEM NUMBERS ARE SPECIFIED ON SPEC. Dwg. UNLESS OTHERWISE NOTED.
3. LOCKE CLOSE TO CONTAINMENT WALL (37C)
4. TRANSFER TUBE FLANGE LEAK TEST CONNECTION. REF. Dwg. DC 243 223-48 (32D)
5. APPROX. SEAL, DROP PIPE SHALL TERMINATE BELOW MAXIMUM PUMPING LEVEL. (33D)

SPECIAL SYMBOLS AND UNIT ONLY  
 TA-2 BROKEN DOWN PUMP MEA  
 A SYMBOL  
 2 SYMBOL ME

P. G. & E. CO.		DRAWING NO.	CHANGE
SHEET 3 OF 5 SHEETS		102019	5

T. R. BROWN





**NOTES:**

- 1. TANKS BY PAIR (1-1 & 1-2) & (2-1 & 2-2) TO BE USED FOR START-UP, THEN CAPPED.
- 2. STANDBY STRAINER IS PLACED IN THE SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS. STRAINER MUST BE REMOVED BEFORE PLANT START UP.
- 3. VALVE ACTUATED AUTOMATICALLY BY GAS ANALYZER, MC CHANNEL NO. AC-1067.
- 4. P&ID VALVES COVERED BY SPEC. 0732.
- 5. P&ID VALVES COVERED BY SPEC. 0729.
- 6. INTERLOCKED REACH DECS MAY BE USED TO INSURE ONE VALVE REMAINS OPEN AT ALL TIMES.

**LIQUID HOLD-UP TANKS**

NOTE 1 DC-663233-3

8. FUTURE CONNECTION TO BE MADE IN OPERATING CONDITION REQUIRED.

REF. DWG: DC-663210-16(169)

P. G. & E. CO.	DRAWING NO.	CHANGE
SHEET 5 OF 21 SHEETS	102008	6



PG&E Response and Status

A comprehensive site inspection and evaluation was conducted to identify those portions of the systems exposed to high activity reactor coolant both during the short-term post-accident cooldown mode and the long-term post-accident cleanup mode. Plant piping drawings and equipment arrangement drawings were reviewed and the systems and components were identified and labeled to locate all source terms during realistic accident scenarios. Personnel access pathways, equipment requiring access, and constantly or frequently manned areas of the plant during accident conditions were identified for dose rate calculations.

Initial area calculations have been made for the Control Room, Onsite Technical Support Center, Reactor Coolant Sampling Area and the penetration area between the Containment and the Auxiliary Building.

A computer model of the Auxiliary Building sources and calculation areas was developed. The initial calculations will be further refined using this computer model. Dose rates in those areas of the auxiliary and turbine buildings requiring personnel access will be calculated.

Calculations will be performed to evaluate exposure to equipment needed after the accident to insure adequate environmental equipment qualification. Evaluation of the new sampling equipment (see Section 2.1.8.a) for post-accident sampling will be conducted to insure personnel protection.

Shielding analysis and environmental qualification studies are currently in progress and will be completed by April 1, 1980. When the shielding studies have been completed (prior to July 1, 1980), a formal design review report of the results will be delivered to the NRC. If modifications or



ITEM 2.1.6.b continued

procedural changes are required as a result of the findings, details of the necessary changes will be provided to the NRC.

Redesign and addition of necessary shielding, if any, and changes in operating and administrative procedures will begin following completion of the evaluation effort.



NUREG-0578 ITEM 2.1.7.a

PG&E Response and Status

The auxiliary feedwater system is shown in 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 loss of offsite power. See 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 loss of offsite power and relay K634 provides starting on safety injection or low-low level in any steam generator. Loss of offsite power 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. 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).

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 Specifications.



ITEM 2.1.7.a continued

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, and are, 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 which is supplied by either on-site emergency generators or station emergency batteries. Each of the two redundant SSPS trains is supplied by separate power sources. Initiation and flow paths will be available upon loss of offsite power.

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.

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 on FSAR Table 8.3-2.

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.



ITEM 2.1.7.a continued

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



NUREG-0578 ITEM 2.1.7.b

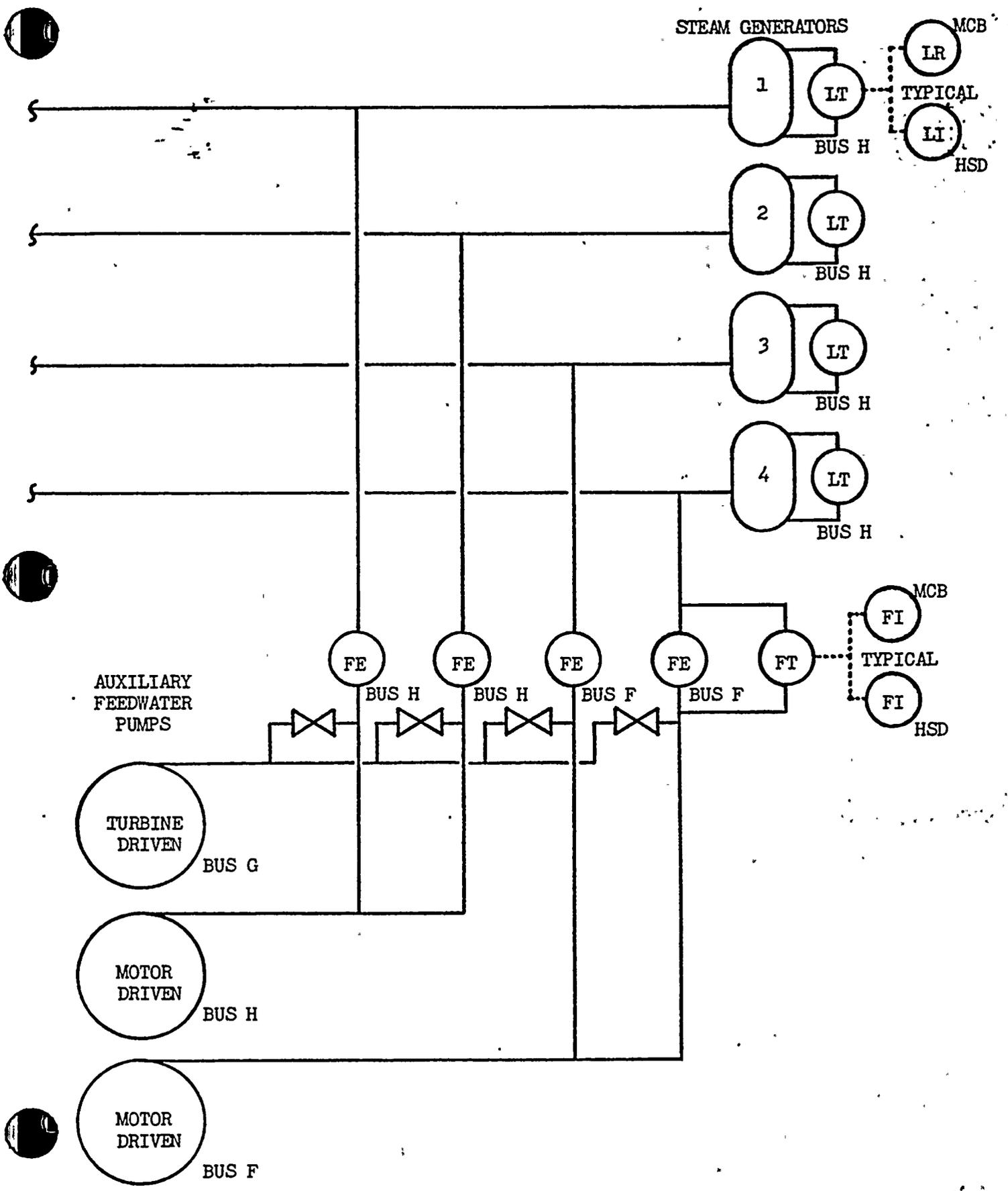
PG&E Response and Status

The flow from each primary motor-driven auxiliary feedwater pump is monitored by an indication system powered from the same primary source as the motor. This system provides indication, both at the main control board and the hot shutdown panel.

The flow from the turbine-driven auxiliary feedwater pump is monitored by the same indicators which monitor the motor-driven auxiliary feedwater pump flow.

An additional indication of auxiliary feedwater flow is provided by the steam generator wide range level indication. This provides recording on the main control board and indication on the hot shutdown panel. It is powered from the same bus as powers one of the motor-driven auxiliary feedwater pumps.







NUREG-0578 ITEM 2.1.8.a

PG&E Response and Status

SUMMARY

A review of existing reactor coolant and containment atmosphere sampling facilities has been conducted for Diablo Canyon Units 1 & 2. The purpose of this review was to evaluate the stations capability to obtain samples during accident and post-accident situations. NUREG-0578 Section 2.1.8.a, "Improved Post-Accident Sampling Station Capability" requirements were used as the basis for this evaluation.

This review and evaluation has resulted in a conceptual design providing a tentative Remote Sampling Facility which projected to be located at the 115 foot elevation on each unit. Shielded sample lines will be provided for both reactor coolant sampling and containment atmosphere sampling. A control panel will be provided where controls and indication may be located for the Remote Sampling System components. The system includes the capability for continuous monitoring of containment hydrogen concentration.

The described Remote Sample Station meets the requirements of NUREG-0578. Personnel radiation exposures have been minimized by shielding the sample lines and minimizing sample quantity to as low as practical limits. The final Sample Station design will allow Diablo Canyon Units 1 & 2 personnel to obtain RCS and Containment Atmosphere samples during post-accident conditions with a minimum of effort while also limiting personnel exposure.

Final designs for the remote sampling system will be completed by May 1, 1980. Installation of the remote sampling system, all analysis equipment and required procedural changes will be completed by January 1, 1981.



## ITEM 2.1.8.a continued

### INTRODUCTION

This report discusses the location of facilities to accommodate the sampling of reactor coolant and containment atmosphere during post-accident conditions. An alternate point, different from the existing sampling station location, is necessary due to the high radiation levels existing from consideration of source terms in NUREG-0578.

Conceptual designs are presented for the sampling of reactor coolant and containment atmosphere. These designs will be used as the basis for the final sampling systems to be installed at Diablo Canyon Units 1 & 2.

### REMOTE SAMPLE STATION LOCATION

Consideration was given to a number of locations where a remote sample station could be installed. A location was chosen on the 115' elevation because of low potential radiation considerations, spaciousness, and access to and from grade level.

The remote station will be installed at the 115' elevation adjacent to the steam generator blowdown demineralizers or the prefilter. Sample tubing for both containment air and reactor coolant sampling will be routed from the reactor coolant system (RCS) sample room (elevation 100'); vertically upward through an existing pipe chase to the steam generator demineralizer cubical. The tubing will extend over the top of the wall separating the demineralizers from the auxiliary building corridor. Shielding will be provided as required.

Radiation sources in the area result from the Chemical Volume Control System (CVCS) piping, the Volume Control Tank (VCT), and containment spray piping. These potential sources will be discussed in Section 3.0.



#### ITEM 2.1.8.a continued

A ventilation hood will be installed at the remote sample station above the sample points to remove any potential airborne activity and will be sized to provide a high flow rate. A blower will also be installed in the hood as opposed to simply branching into an auxiliary building return duct. This will insure quick removal and dilution of any airborne contamination.

A control panel containing switches for operating valves, status indication and direct telephone communications with the control room, will be provided at the remote sample station location. Section 6.0 discusses the design of the control panel further.

#### POTENTIAL BACKGROUND SOURCES

The remote sample station sampling tubing, CVCS piping, VCT, and containment spray piping contribute to personnel exposure during remote sampling.

Unshielded CVCS piping is visible from the corridor on the 115' elevation adjacent to the prefilter cubical. Because this pipe chase will also contain sample tubing, a labyrinth shield at the entrance to the cubical will be provided.

The volume control tank, located beneath the S.G. blowdown demineralizers, will become an important radiation source if a feed and bleed operation occurs using the assumed TID source terms.

The containment spray piping is located in the penetration area adjacent to the auxiliary building wall at the proposed sampling location. This piping will be a significant radiation source in post-accident operations.



### ITEM 2.1.8.a continued

A preliminary evaluation consisting of determining the radiation levels during the after the accident situation has been performed to determine shielding requirements for the area. A further, more specific evaluation will be conducted to determine the shielding designs.

### RCS LIQUID SAMPLING SYSTEM

Under normal operation, reactor coolant is continuously let down from either the hot leg of loop 1 or 4 to the Gross Failed Fuel Detector and to the hot leg sample vessel, ATSV3. Flow passing through the Gross Failed Fuel Detector returns to the sample heat exchangers. The flow continues through the sample line upstream of its connection to the CVCS Mixed Bed Demineralizers or the sample sink; depending on valve line-up. The remote sampling system utilizes this existing system and adds the capability for remote sampling.

#### 1. Remote RCS Liquid Sampling System

The Remote Sample System, shown in Figure 1, will utilize new isolation sample valves downstream of the sample heat exchanger to divert flow to a shielded, removable sample vessel for obtaining a reactor coolant sample. The shielded sample vessel may be used to obtain coolant samples from either the RHR pump discharge or hot leg 1 or 4 depending on valve line up. Capability to control valves 9370A and B must be provided on the remote sample system panel, due to the fact that the radiation level of the lines those lines are mounted on would preclude personnel access. The reactor coolant returns to the CVCS mixed bed demineralizers to allow the sample to flow to the liquid radwaste system. This minimizes potential gas releases from the sample sink to the auxiliary building atmosphere.



### ITEM 2.1.8.a continued

The remote sample system will also have the capability to be flushed with demineralized water using manual valves controlled at the sample panel. This is necessary in order to flush any high radiation level sample remaining in the lines prior to taking a new sample. FCV 19 is a local, manually operated flow control valve which maintains letdown flow at 1 GPM. This valve does not require control at the remote sample panel location assuming that it is operating to control letdown flow.

The remote, shielded sample vessel contains flexible lines on one end to allow the operator to easily connect the vessel to the quick disconnects by compensating from any minor mis-alignment. Pressure and temperature indicators for the sample are also provided for system operation and to allow calculations to be made relating to the sample mass.

### 2. RCS Liquid Sample Vessel

The RCS liquid sample vessel must have the capability to withstand hydrostatic testing to 3750 psig and operating temperatures and pressures of 130<sup>o</sup>F and 2250 psig, respectively. It will be constructed of 304 stainless steel or equivalent. The sample vessel must be capable of containing about 35 ml of liquid required for analysis purposes. This total results from requirements for 25 ml for a chloride analysis, 5 ml for boron and 1 ml for activity.

Two sample container attachments were considered. The 100 ml sample vessel supplied by the Kerotest Company and used for normal RCS sampling, could be adapted for use at the remote sampling station. It would require additional shielding but has the advantage of being an off-the-shelf item. Figure 3 shows a graph of dose rate vs. shield thickness for a cylindrical shield placed around the kerotest vessel.



#### ITEM 2.1.8.a continued

It should be noted that an 8" radius shielded vessel weighs about 1,000 lbm, and has an expected surface dose rate of about 62 mR/hr.

The other alternative allows collection of sample in a length of shielded tubing designed to obtain a 35 ml sample with shutoff valves and quick disconnects as shown in Figure 1. The smaller sample volume minimizes shielding and handling requirements, reduces the exposure potential, and would be simpler to fabricate. The connections at the sample station will be designed to accept both the 100 ml kerotest vessel and a 35 ml "sample tube" so that the capability exists to take either small or large RCS samples. The shielded sample tube will be the primary sample collection method used due to the advantages discussed previously.

#### CONTAINMENT ATMOSPHERE SAMPLING SYSTEM

Containment atmosphere sampling is normally accomplished by connecting a sample vessel with a flexible line upstream of the particulate air monitor (RE 11 and 12). The existing particulate monitor is normally used to determine radiation levels by passing the containment atmosphere through filter paper and reading levels of particulates which do not pass through. The existing monitor also measures radiation levels of any noble gases in the containment atmosphere. For the worst case, post-accident situation, noble gases, particulates, Halogens and Hydrogen may exist along with a sizable quantity of wet steam. Modifications must be made to the existing monitor as it cannot withstand this environment. In addition, the existing system isolates on high containment pressure since it is not capable of withstanding it.



ITEM 2.1.8.a continued

1. Remote Containment Atmosphere Sampling System

The design for the Containment Atmosphere Sampling System (CASS) which meets NUREG-0578 requirements and clarifications is shown in Figure 2.

The primary design features of the system are the redundant active components to meet single failure criteria which are required for hydrogen sampling, the capability to obtain samples for particulate, iodine analysis and a permanently connected hydrogen analyzer.

Containment atmosphere is drawn into the system through the existing FCV 678 and 679 valves by air sample pumps. Redundant valves are provided to meet single active failure criteria. Valves are also provided to isolate the existing particulate monitor in order to protect it from accident environmental conditions. The sample continues through to either the containment iodine, particulate and atmospheric sampling section or to the hydrogen analyzer, depending on valve lineup. It is not expected that both atmospheric sampling and hydrogen monitoring will be conducted concurrently; thus a common air pump and supply line is provided.

A shielded charcoal/paper filter is provided upstream of the sample vessel to remove particulates and iodine for separate analysis. Both the paper and charcoal filters require a moisture-free atmospheric sample to avoid deterioration. In addition iodine will plate out on the sample lines if the temperature of the sample drops below the saturation point. To minimize deterioration of the sample and plate out, heat tracing is used on all CASS lines prior to the iodine sampling point. Temperature and pressure indicators between the filter and the sample vessel may be used to calculate the air density and, when used in conjunction with the flow



### ITEM 2.1.8.a continued

indicator, the mass flow rate passing through the filter assembly (or hydrogen analyzer) may be determined.

Any required corrections for the volume fraction of steam (if present) can be determined by cooling the sample vessel and measuring the liquid content. Quick disconnects are provided for both the filter assembly and sample vessel to allow removal of that equipment for analysis.

Flexible lines are provided on the sample vessel and filter assembly as described in the RCS sampling section. The sample continues to the hydrogen analyzer where hydrogen content within the containment atmosphere is determined.

A connection to instrument air is provided near the remote sampling system locations. A flexible hose may be connected from the instrument air supply or the instrument air purge connection on the sample system to allow this air flow to remove any sample plate out or particulate matter. Bypass lines containing quick disconnects will be attached to the sample points to allow purge air to flow through the containment atmosphere/iodine-particulate sampling section.

### 2. Hydrogen Monitoring

An existing hydrogen analyzer detects between 0-100% of the concentrations of combustible gases or vapors required for an explosion to occur.

The detector, a Mine Safety Apparatus (MSA) model 501 system, utilizes a Wheatstone Bridge connected to heated filaments. Burning the combustible portion of an air sample produces heating of the filaments and causes a proportional change in electrical resistance. This device has two limitations: (1) it will not operate in the presence of steam



### ITEM 2.1.8.a continued

or inert atmospheres, and (2) it can only measure combustible mixtures up to the explosive limit (for hydrogen 5 - 10%). This device does not meet the intent of the clarifications of ACRS Comment #3, which requires a measurement capability of hydrogen concentrations over a 0-10% range.

The modified CASS is designed to include a new hydrogen analyzer with the capability for monitoring the 0-10% range. The hydrogen analyzer would also operate in the presence of high temperature steam-air and inert gas mixtures. Investigations are continuing with vendors to determine if hydrogen analyzers with this capability are available.

Redundant components are provided to meet single active failure criteria for hydrogen analysis. For example, Figure 2 shows dual FCV's on the inlet and outlet sample lines, air sample pumps and dual hydrogen analyzers.

### 3. CASS Sample Vessel

The shielded sample vessel for use in obtaining containment atmospheric samples will consist of a pipe with a septum closure nipple installed on the sidewall. After opening the shutoff valve and allowing sample to flow through the pipe, a shielded hypodermic will be used to withdraw 1 cc of atmospheric sample from the septum closure. This can be transported in a shielded carrier for analysis using existing laboratory methods and analytical techniques.

The shielded particulate and charcoal filter is located upstream of the atmospheric sample. Sample would flow through the filter upon opening of the shutoff valve. The filter would then be transported in a shielded container to the chemistry laboratory for analysis to determine iodine.



ITEM 2.1.8.a continued

and particulate levels using standard techniques.

REMOTE SAMPLE SYSTEM CONTROL PANEL

Due to the high radiation levels which could exist on the sample lines, remote control of certain valves and equipment is required at the remote sample station. In addition, temperature and pressure, valve status and other indications of system performance are required. A panel containing these components will be located at the point where remote sample station tubing is being routed as discussed in the location section.

1. Panel Mounted Instrumentation and Controls Required for Reactor Coolant Sampling System

The following control and indication will be added and/or mounted on the panel:

a. Valves

- (1) Manual valves, shown in Figure 2, will be mounted such that these handles are shielded from any sample line radiation level.
- (2) Control valves which close on receipt of a containment isolation signal.
- (3) Valves 9370B, 9370A will have solenoid operators added to allow remote operation.
- (4) Valves 9351A, 9356A and 9356B will have operators added to allow operation from the remote sample panel.
- (5) Sample isolation valves.



ITEM 2.1.8.a continued

b. Indication

- (1) All new pressure and temperature indication.
- (2) Status indication for all remotely operated valves.

2. Panel Mounted Instrumentation and Controls Required for Containment Atmospheric Sampling System

Indication and controls for the following devices will be mounted on the panel:

a. Valves

- (1) Valves receiving a safety isolation signal which require reopening to allow sampling during post-accident conditions: FCV-678, 679 and 681.
- (2) New control valves installed in parallel with FCV-678, 679 and 681.
- (3) New manual valves, shown in Figure 1 will be mounted such that the handles are shielded from any sample line radiation.

b. Indication

- (1) New pressure, temperature and flow indication.
- (2) Status lights for heat tracing.
- (3) Hydrogen analyzer status and monitoring indicators.
- (4) Valve status lights for all remote, operating valves.
- (5) Sample air pump's status.



ITEM 2.1.8.a continued

c. Controls

(1) Sample air pump controls.

(2) Fume hood ventilation fan.

REMOTE SAMPLE STATION SHIELDING

Two alternatives were considered for shielding sample tubing at the remote sample station. The first of these was to place the tubing coaxially inside larger diameter lead-filled pipes. The second was to place it behind sheet steel. Sheet steel will be used because of the ease of fabrication and it affords greater design flexibility. The shielding should be removable for maintenance and inspection purposes by placing all penetrations for valve stems, instrument lines, etc., along the seam between movable shield plates.

Preliminary calculations by Radiation Research Associates (RRA) indicate that in order to limit external contact dose rates to below 1 R/hr for an 8 ft. run of sample tubing, 6.7" and 2.8" of iron are required for reactor coolant system liquid samples and containment atmosphere air sample lines, respectively. The upper limit for the thickness of iron plate is then about 7". After a detailed design tubing layout is approved, shielding requirements will be finalized. The only unshielded portion of the sampling system will be the quick disconnects which must protrude from the steel plates. Movable shielding will be provided to cover the couplings after the sample vessels are removed. Personnel exposure will be minimized by using long-stemmed valves and instrumentation which does not create a radiation field at the gauge faces.

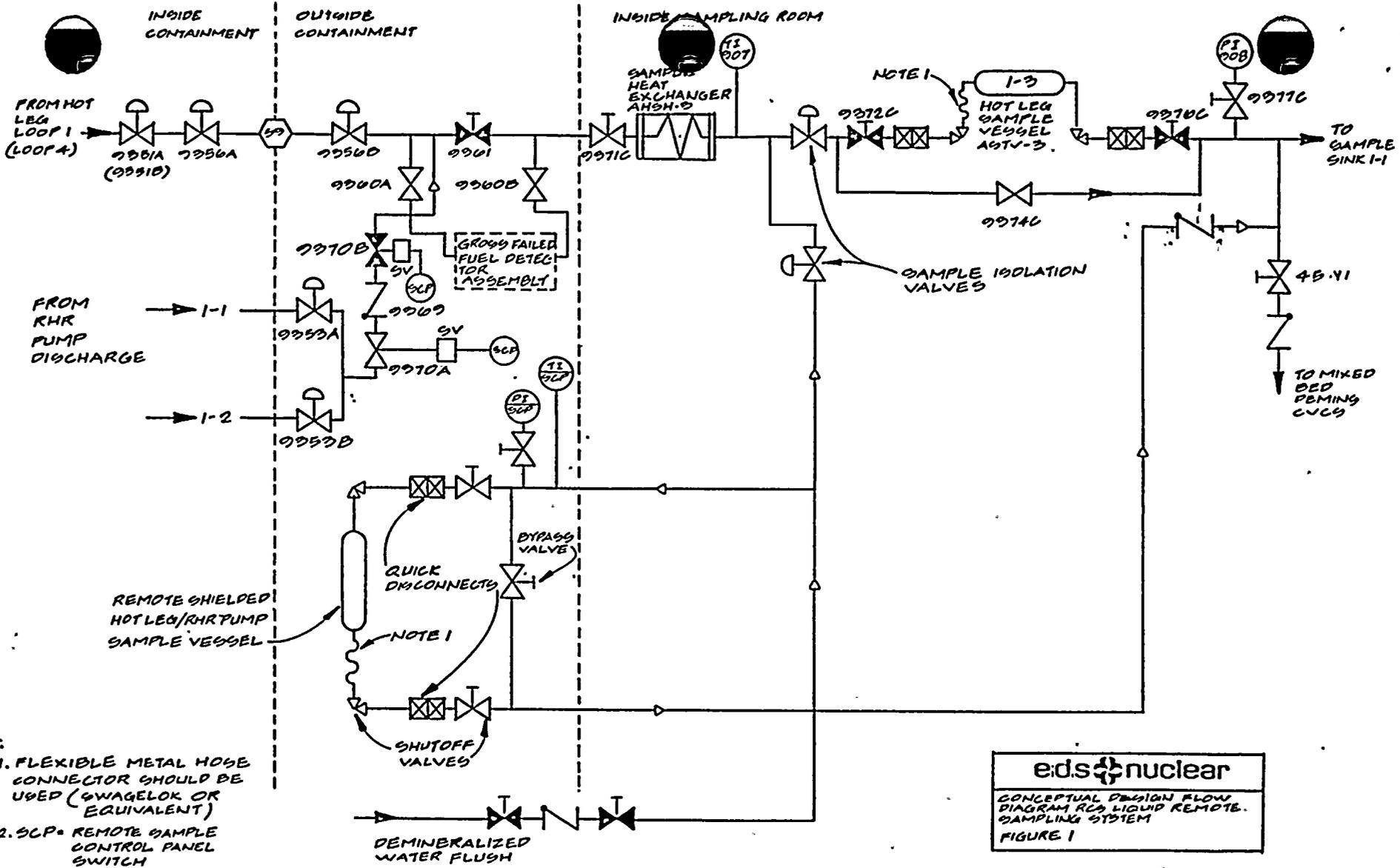


ITEM 2.1.8.a continued

REFERENCES

1. Letter 12/14/79, K. Warkentin (RRA) to S. Skidmore (PG&E)  
RRA Letter #RL4273-09
2. MSA Series 500 Combustible Gas Detection Systems;  
Instruction Manual
3. TMI-2 Lessons Learned Task Force Status Report and Short Term  
Recommendations, U. S. NRC, NUREG-0578, July 1979
4. Report to the Nuclear Regulatory Commission from PG&E Responding  
to NUREG-0578: TMI-2 Lessons Learned Task Force Status Report and  
Short-Term Recommendations, PG&E, August 27, 1979
5. Clarifications of NUREG-0578, Enclosure 1, USNRC, November 9, 1979.





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- NOTES:
1. FLEXIBLE METAL HOSE CONNECTOR SHOULD BE USED (WAGELOK OR EQUIVALENT)
  2. SCP = REMOTE SAMPLE CONTROL PANEL SWITCH
  3. VALVES NOT NUMBERED ARE NEW

eds+nuclear  
 CONCEPTUAL DESIGN FLOW  
 DIAGRAM FOR LIQUID REMOTE  
 SAMPLING SYSTEM  
 FIGURE 1

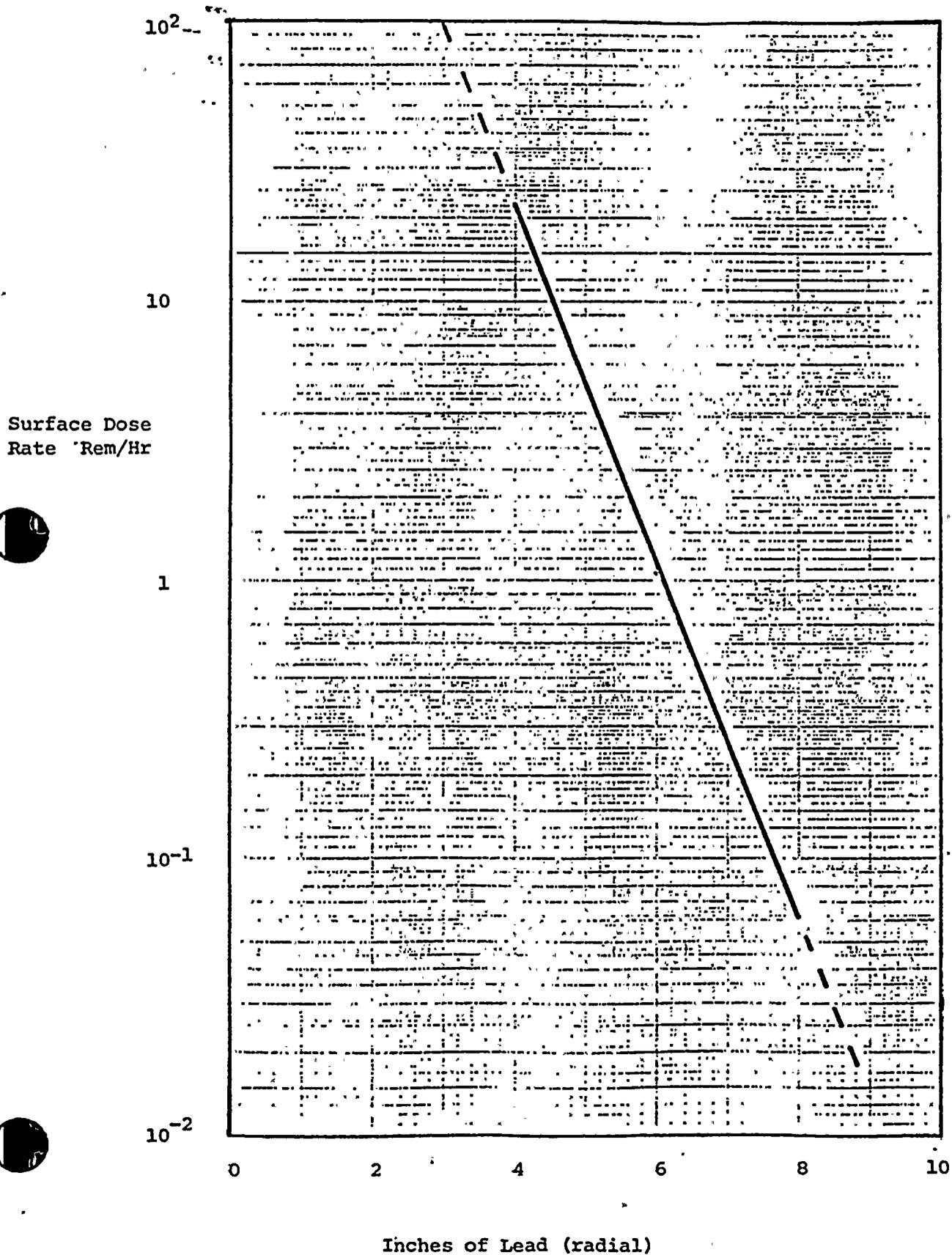


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Figure 3  
Reactor Coolant Liquid  
Surface Dose Rate vs Radial Shield Thickness





PG&E Response and Status

Extended Range Radiation Monitors

Each containment radiation monitor (there are two) is manufactured by Victoreen and consists of an ionization chamber inside containment and a readout in the control room. The unit can respond to photon energies as low as 60 KeV and has a range of  $10^0$  to  $10^7$  R/hr.

The plant vent monitor is manufactured by Victoreen, and consists of a particulate prefilter, a silver zeolite filter, a noble gas monitoring chamber, and appropriate air moving components. The iodine filter has a sodium iodide crystal monitor with a range of  $10^{-8}$  uCi/cc to  $10^{+2}$  uCi/cc, and the noble gas chamber has a sodium iodide crystal monitor with a range of  $10^{-5}$  uCi/cc to  $10^5$  uCi/cc based on Xe-133.

Readouts are in the control room.

All parameters are recorded in the control room.

If our supplier meets current delivery date commitments, installation is expected to be completed by August 1, 1980. For operation prior to August 1, 1980, PG&E will provide the necessary instrumentation and methods as required by the clarifications.



PG&E Response and Status

Analyses of small break loss-of-coolant accidents, symptoms of inadequate core cooling and required actions to restore core cooling, and analysis of transient and accident scenarios including operator actions not previously analyzed are being performed on a generic basis by the Westinghouse Owners' Group, of which PG&E is a member. The small break analyses have been completed and were reported in WCAP-9600, which was submitted to the Bulletins and Orders Task Force by the Owners' Group on June 29, 1979. Incorporated in that report were guidelines that were developed as a result of small break analyses. These guidelines have been reviewed and approved by the B&O Task Force and have been presented to the Owners' Group utility representatives in a seminar held on October 16-19, 1979. Following this seminar, each utility has developed plant specific procedures and trained their personnel on the new procedures.

The work required to address the other two areas--inadequate core cooling and other transient and accident scenarios--has been performed in conjunction with schedules and requirements established by the Bulletins and Orders Task Force. Analysis related to the definition of inadequate core cooling and guidelines for recognizing the symptoms of inadequate core cooling based on existing plant instrumentation and for restoring core cooling following a small break LOCA were submitted on October 31, 1979. This analysis is a less detailed analysis than was originally proposed, and will be followed up with a more extensive and detailed analysis which will be available during the first quarter of 1980.

With respect to other transient accidents contained in Chapter 15 of the Diablo Canyon FSAR, the Westinghouse Owners' Group has performed an evaluation of the actions which occur during an event by constructing sequence of event trees for each of the non-LOCA and LOCA transients. From these event trees a list of decision points for operator action has been prepared, along with a list of information available to the operator at each decision point. Following this, criteria have been set for credible misoperation, and time available for operator decisions have been qualitatively assessed. The information developed was then used to test Abnormal and Energy Operating Procedures against the event sequences and determine if inadequacies exist in the AOPs and EOPs. The results of this



ITEM 2.1.9 continued

study will be provided to the Bulletins and Orders Task Force on March 31, 1980, as required.

The Owners' Group has also provided test predictions analysis of the LOFT L3-1 nuclear small break experiment. This analysis was provided on December 15, 1979, in accordance with the schedule established mutually with the Bulletins and Orders Task Force.

Procedures identified by these analyses are presently being written. The procedures and operator training will be completed by May 1, 1980.



ACRS ITEM #4

PG&E Response and Status

A head vent system meeting the requirements of this comment will be installed at Diablo Canyon. The system design is being purchased from the nuclear steam supply system manufacturer, Westinghouse. The remotely operated valves, which are the item controlling the schedule, have been on order for several months and will be delivered in September, 1980.

The head vent system will use the existing manual vent connection to the reactor head, which is three-quarter inch diameter. Four one-inch electrically operated valves meeting the requirements of IEEE-323-74 and IEEE-344-75 will be installed in a series-parallel arrangement satisfying the single failure criterion. These valves will be operable from and position indicated in the control room. They will fail closed. A flow diagram showing the arrangement schematically is attached.

three-eighths inch diameter orifice will be installed in each of the parallel vent lines between the existing manual valve and the new electrically operated valves. The new valves, orifices, and connecting piping will be supported from the head lifting structure and will be qualified seismically for this location. Instrumentation and control will meet the requirements of IEEE-279.

The orifices in the vent lines would restrict the flow thru a postulated break in the new vent system to that which normal makeup can maintain. This would preclude the possibility of the occurrence of a small loss of coolant accident in the head vent system. (A rupture in the existing manual head vent line is covered by the small break LOCA discussion in Chapter 15 of the FSAR and by Westinghouse WCAP-9600 on the same subject which has been submitted to the NRC staff by Westinghouse.)

In a Westinghouse reactor coolant system, the head vent will also serve to remove non-condensable gases from the "hot leg" and "cold leg" reactor coolant system piping. Should the existence of non-condensable gases in the tubes of a steam generator be suspected, it is proposed to "bump" the related reactor coolant pump so as to move the gases to the reactor vessel, where they may be removed by the head vent system.



ACRS ITEM #4 continued

Equipment and piping for Reactor Coolant System venting will be installed, procedures finalized and operator training completed by January 1, 1981. Preliminary calculations confirm that a gas volume greater than one-half the reactor coolant system volume could be vented by either train in less than one hour. On the other hand, the procedure will call for venting to be terminated before the hydrogen concentration in the containment could exceed 4% under the most adverse assumptions. Venting would not be continued until the actual containment hydrogen concentration achieved had been determined.

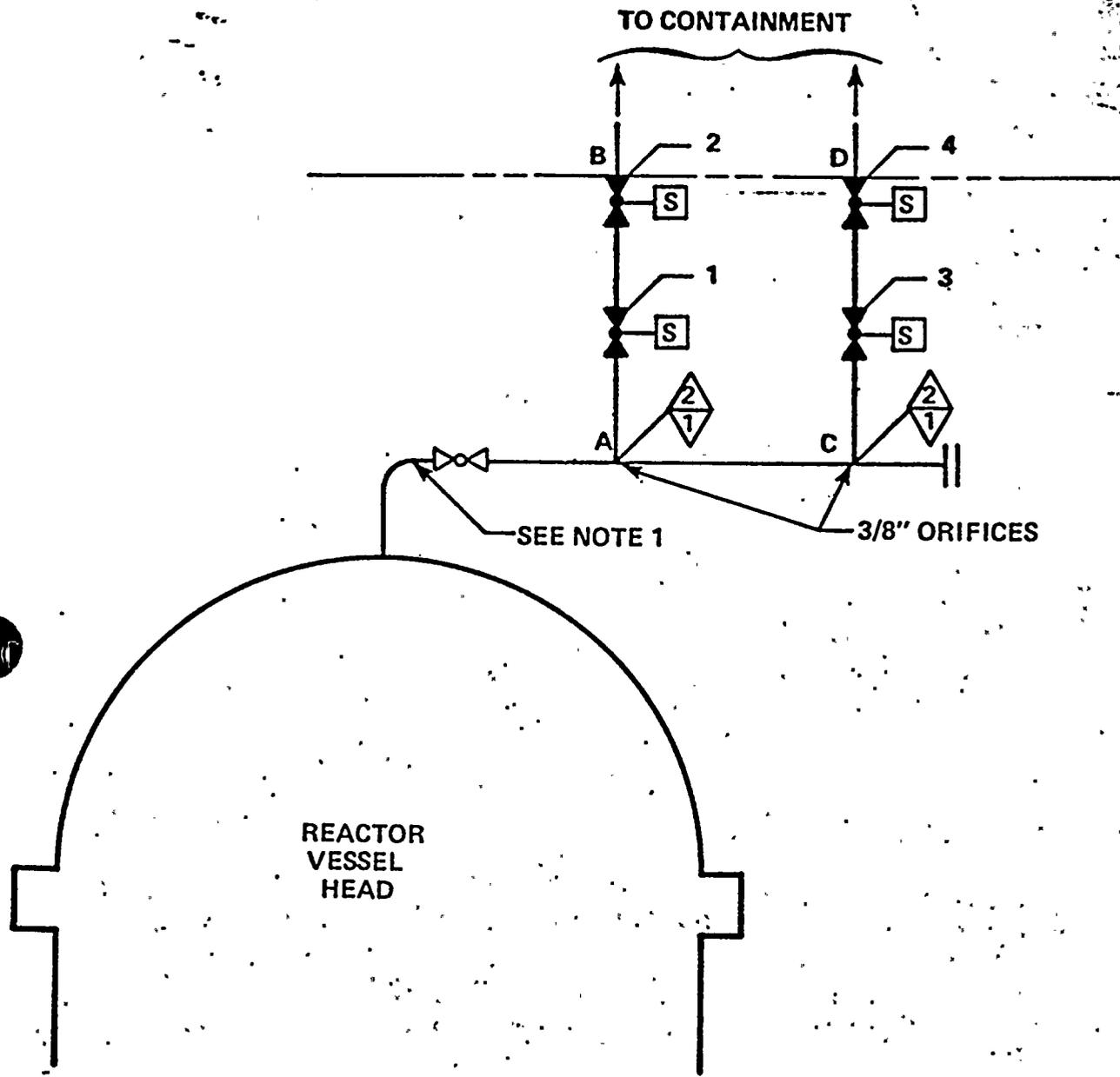
The head vent system described would vent into the area immediately surrounding the reactor vessel head. This area is swept by ventilating air flow and gases released would be mixed into the containment volume by this air flow. Consideration has been given to piping the vented gases to other locations, for example, the pressurizer relief tank. At this time, the identification or advantage of such an alternate vent release location is not known. However, should an alternate location be required in the future, an extension of the described system could be added.

Inadvertent opening of the head vent system would be detected by the valve position lights in the control room and by the leakage monitoring system which is described in Section 5.2.4 of the FSAR.

Venting of the pressurizer, should that be necessary, can be carried out by the use of the pressurizer power operated relief valves, which at Diablo Canyon are seismically qualified and have qualified control circuits.



SECRET



REACTOR  
VESSEL  
HEAD

NOTES:  
1. EXISTING VENT LINE

Flow Diagram of the Reactor Vessel Head Vent System

1  
2  
3  
4



PG&E Response and Status

The employment positions have been identified and authorized by PG&E. PG&E is attempting to locate six qualified individuals to hire for these positions. Every effort will be made to fill these positions as rapidly as possible. In any event, these positions will be filled and the personnel trained to allow operation by May 1, 1980. The STA job description is as follows:

Basic Responsibility

To ensure increased plant safety by providing an accident assessment function in the control room at all times and to provide on-shift operating experience assessment capability.

Specific Duties

1. Provides sound technical advice to the Shift Foreman during transient or accident conditions.
2. Evaluates plant equipment performance and makes recommendations for improvement.
3. Reviews operating experience and equipment performance at other nuclear plants.
4. Ensures that operating and emergency procedures are revised and operators trained based on his review of operating experience.
5. Provides assistance to the Shift Foreman in reviewing surveillance test results.
6. Assists the Shift Foreman in first aid and fire protection training.
7. Advises the Shift Foreman on the plant Technical Specifications.
8. Conducts operator training as specified by the Training Coordinator when his shift is on a training assignment.
9. Assist the Shift Foreman in the preparation of Special Work Permits.
10. May perform independent audits of operating practices and procedures.
11. Provides a critique of fire and other emergency drills.
12. Provides liaison between the operating and technical groups at the plant.

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ITEM 2.2.1.b continued

Relationships

1. Reports directly to the Shift Foreman and is responsible and accountable to him in his advisory capacity.
2. Reports indirectly to the Senior Power Production Engineer (Operations Engineer) and is responsible and accountable to him in administrative and technical matters.
3. Maintains close working relations with the Maintenance and Technical Departments.
4. At the direction of the Shift Foreman or Operations Engineer, confers with other departments and agencies both inside or outside the Company.

Authorities

1. To carry out his assigned duties.
2. To take action in an emergency to the full extent justified. Such action will be reported to the Shift Foreman and Operations Engineer as soon as practical.
3. To consult with and/or advise other plant supervisors on activities related to plant operations.

