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

DECEMBER 31, 1979

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Introduction and Summary

Section 1

Purpose

This report to the Nuclear Regulatory Commission provides the current status of implementing the commitments made by PG&E as a result of the lessons learned from the accident at Three Mile Island, Unit 2.

Background

PG&E's 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. As a part of that report, PG&E committed to keeping the NRC informed of the progress in implementing these commitments on January 1 and July 1 of each year until all commitments have been completed.

Following the issuance on July 19, 1979 of NUREG-0578, PG&E issued a "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'", dated August 27, 1979. The schedule for implementing the recommendations of NUREG-0578 includes numerous Category A items that are to be accomplished by January 1, 1980, or prior to OL, whichever is later.

During the subsequent reviews of NUREG-0578 by the NRC and the ACRS, the NRC issued letters on September 13 and 27, 1979, on the subject of Follow-up Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident. The additional recommendations of these letters were addressed in PG&E's Addendum dated October 26, 1979, to the August 27, 1979 report.

On November 9, 1979 the NRC issued clarifications to some of the recommendations of NUREG-0578 resulting in modifications to some of the implementation commitments.

Scope

This report sets forth the current status of PG&E's efforts in implementing PG&E commitments and recommendations as stated and clarified or modified in the above documents. Unless specifically mentioned in this report, all clarifications have been addressed in PG&E's previous responses. This report follows the sectional numbering of NUREG-0578 and provides the Task Force's position and NRC clarifications to facilitate review. Section 3 of the report follows the numbering scheme of the Summary of Actions and Plans in the Introduction and Summary Section of PG&E's July report.

Summary

This status report is a comprehensive compilation of PG&E's actions to implement the short-term recommendations of the Lessons Learned Task Force and the comments raised by the ACRS. This effort further demonstrates PG&E's commitment to comply with the recommendations.

Status of Implementation

of NUREG-0578

Section 2

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 (TFP) on Pressurizer Heater Power Supply

Task Force 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.)

Task Force 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 capability for the connection of the pressurizer heaters. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

Section 2.1.1. (Continued)

Task Force 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.)

Task Force 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.)

Clarification

1. In order not to compromise independence between the sources of emergency power and still provide redundant capability to provide emergency power to the pressurizer heaters, each redundant heater or group of heaters should have access to only one Class 1E division power supply.

Section 2.1.1 (Continued)

2. The number of heaters required to have access to each emergency power source is that number required to maintain natural circulation in the hot standby condition.

3. The power sources need not necessarily have the capacity to provide power to the heater concurrent with the loads required for LOCA.

4. Any change-over of the heaters from normal offsite power to emergency onsite power is to be accomplished manually in the control room.

5. In establishing procedures to manually reload the pressurizer heaters onto the emergency power sources, careful consideration must be given to:

   a. Which ESF loads may be appropriately shed for a given situation.

   b. Reset of the Safety Injection Actuation Signal to permit the operation of the heaters.

   c. Instrumentation and criteria for operator use to prevent overloading a diesel generator.

6. The Class IE interfaces for main power and control power are to be protected by safety-grade circuit breakers. (See also Regulatory Guide 1.75.)

Section 2.1.1 (Continued)

7. Being non-Class IE loads, the pressurizer heaters must be automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. (See Item 5.b. above.)

PG&E Status for TFP1

The Diablo Canyon design has been changed to provide emergency power to the pressurizer heaters. The design change will be implemented subsequent to completion of the seismic testing of the circuit breakers. It is anticipated that the testing will start January 7, 1980, and be completed by January 15, 1980.

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

PG&E Status for TFP2

PG&E will develop the required procedures and implement the proper training of the operators. The procedures will be written and approved, and the operators trained, by May 1, 1980.

Section 2.1.1 (Continued)

PG&E Status for TFP3

Since the design will be changed to allow transferring the power supplies from the Control Room, there is no problem in meeting the Westinghouse estimated time requirement of providing pressurizer heaters on emergency power within one hour after the accident. The operating and emergency procedures will be changed to reflect these requirements and changes by May 1, 1980.

PG&E Status for TFP4

The seismic testing mentioned in PG&E status for position 1 above will qualify to safety grade requirements the interface between the pressurizer heater motive and the emergency bus.

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

Task Force 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.)

Section 2.1.1 (Continued)

Task Force 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.)

Task Force 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 January 1, 1980, whichever is later.)

Task Force 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.)

Section 2.1.1 (Continued)

Clarification

1. While the prevalent consideration from TMI Lessons Learned is being able to close the PORV/block valves, the design should retain, to the extent practical, the capability to open these valves.

2. The motive and control power for the block valve should be supplied from an emergency power bus different from that which supplies the PORV.

3. Any changeover of the PORV and block valve motive and control power from the normal offsite power to the emergency onsite power is to be accomplished manually in the control room.

4. For those designs where instrument air is needed for operation, the electrical power supply requirement should be capable of being manually connected to the emergency power sources.

PG&E Status of TFP 1, 2, 3 and 4

A design is currently underway to incorporate clarification 2 into the present design. The design will be completed by February 1, 1980. The design conforms to all other clarifications. See PG&E response, August 27, 1979, to NUREG-0578. Construction will be completed by April 1, 1980.

Section 2.1.2 - <u>Performance Testing for BWR and PWR Relief and Safety Valves</u>

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).

Clarification

1. Expected operating conditions can be determined through the use of analysis of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70.

Section 2.1.2 (Continued)

2. This testing is intended to demonstrate valve operability under various flow conditions, that is, the ability of the valve to open and shut under the various flow conditions should be demonstrated.

3. Not all valves on all plants are required to be tested. The valve testing may be conducted on a prototypical basis.

4. The effect of piping on valve operability should be included in the test conditions. Not every piping configuration is required to be tested, but the configurations that are tested should produce the appropriate feedback effects as seen by the relief or safety valve.

5. Test data should include data that would permit an evaluation of discharge piping and supports if those components are not tested directly.

6. A description of the test program and the schedule for testing should be submitted by January 1, 1980.

7. Testing shall be complete by July 1, 1981.

Section 2.1.2 (Continued)

PG&E Status

PG&E is participating in an ongoing EPRI safety and relief valve test program. This program will assure that safety and relief valves can perform to prevent over-pressurization of the reactor coolant system pressure boundary. The hydraulic and structural performance of the valves and the associated discharge piping will be verified by testing. This testing will demonstrate: (1) that the valves open and close correctly; (2) that the flow capacity is sufficient; and (3) that piping integrity is maintained. The valves to be used at Diablo Canyon are included in this program. The desired capability to test valves at full pressure and full flow does not currently exist in the industry. Multiple test facilities are being modified or developed to allow for this testing capability. Initial testing with small safety valves and piping is scheduled to begin in April, 1980. This part of the test program (incorporating steam, water, and two phase flow) will be completed by July, 1981. The full capacity test facility is scheduled to be available by July, 1981, and will explore over-pressure protection system phenomena. It will be possible to test both a safety valve and a power-operated relief valve simultaneously, as well as flow phase transitions.

The entire EPRI safety and relief valve test program is anticipated to take four years and to be completed in 1983.

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.)

Clarification

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis and action.

Section 2.1.3.a (Continued)

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached. If the seismic qualification requirements cannot be met feasibly by January 1, 1980, a justification should be provided for less than seismic qualification and a schedule should be submitted for upgrade to the required seismic qualification.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift). If the environmental qualification program for this position indication will not be completed by January 1, 1980, a proposed schedule for completion of the environmental qualification program should be provided.

PG&E Status

The pressurizer PORV's have a position indicating system that complies with the Task Force Position and all clarifications except clarification 2. To comply with clarification 2, an alarm will be provided, and will be operative by May 1, 1980.

With regard to position indication for pressurizer safety valves, PG&E is currently evaluating proposals from several manufacturers for various means of providing valve position indication. PG&E plans to purchase the necessary equipment by January 15, 1980. Installation is expected to be completed by June 1, 1980.

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

A. Task Force Position on Subcooling Meter

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.)

Clarification

1. The analysis and procedures addressed in Paragraph 1 above should be submitted to the NRC for review.

Section 2.1.3.b (Continued)

2. The purpose of the subcooling meter is to provide a continuous indication of margin to saturated conditions. This is an important diagnostic tool for the reactor operators.

3. Redundant safety grade temperature input from each hot leg (or use of multiple core exit in T/C's) are required.

4. Redundant safety grade system pressure measures should be provided.

5. Continuous display of the primary coolant saturation conditions should be provided.

6. Each PWR should have: (A.) Safety grade calculational devices and display (minimum of two meters) or (B.) a highly reliable single channel environmentally qualified, and testable system plus a backup procedure for use of steam tables. If the plant computer is to be used, its availability must be documented.

7. In the long term, the instrumentation qualifications must be required to be upgraded to meet the requirements of Regulatory Guide 1.97 (Instrumentation for Light Water Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident) which is under development.

Section 2.1.3.b (Continued)

8. In all cases appropriate steps (electrical, isolation, etc.) must be taken to assure that the addition of the subcooling meter does not adversely impact the reactor protection or engineered safety features systems.

9. The attachment provides a definition of information required on the subcooling meter.

Section 2.1.3.b (Continued)

ATTACHMENT
INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	____________
Display Type (Analog, Digital, CRT)	____________
Continuous or on Demand	____________
Single or Redundant Display	____________
Location of Display	____________
Alarms (include setpoints)	____________
Overall uncertainty (°F, PSI)	____________
Range of Display	____________
Qualifications (seismic, environmental, IEEE323)	____________

Calculator

Type (process computer, dedicated digital or analog calc.)	____________
If process computer is used specify availability. (% of time)	____________
Single or redundant calculators	____________
Selection Logic (highest T., lowest press)	____________
Qualifications (seismic, environmental, IEEE323)	____________
Calculational Technique (Steam Tables, Functional Fit, ranges)	____________

Input

Temperature (RTD's or T/C's)	____________
Temperature (number of sensors and locations)	____________
Range of temperature sensors	____________

Section 2.1.3.b (Continued)

Uncertainty* of temperature sensors (°F at 1) ____________
Qualifications (seismic, environmental, IEEE323) ____________
Pressure (specify instrument used) ____________
Pressure (number of sensors and locations) ____________
Range of Pressure sensors ____________
Uncertainty* of pressure sensors (PSI at 1) ____________
Qualifications (seismic, environmental, IEEE323) ____________

Backup Capability

Availability of Temp & Press ____________
Availability of Steam Tables etc. ____________
Training of operators ____________
Procedures ____________

PG&E Status

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.

In addition, in order to fulfill the requirements of the clarifications another single safety grade device with single temperature inputs from each loop and multiple incore thermocouples will be added. It will have two pressure inputs and meet the requirements of Regulatory Guide 1.97, Revision 2. PG&E plans to purchase this additional instrumentation by January 15, 1980. Steam tables will be used for backup.

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

B. Task Force Position on Additional Instrumentation

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 in January 1, 1981.)

Section 2.1.3.b (Continued)

Clarification

1. Design of new instrumentation should provide an unambiguous indication of inadequate core cooling. This may require new measurements to or a synthesis of existing measurements which meet safety-grade criteria.

2. The evaluation is to include reactor water level indication.

3. A commitment to provide the necessary analysis and to study advantages of various instruments to monitor water level and core cooling is required in the response to the September 13, 1979 letter.

4. The indication of inadequate core cooling must be unambiguous, in that, it should have the following properties:

    a. It must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high void fraction pumped flow as well as stagnant boil off).

    b. It must not erroneously indicate inadequate core cooling because of the presence of an unrelated phenomenon.

5. The indication must give advanced warning of the approach of inadequate core cooling.

Section 2.1.3.b (Continued)

6. The indication must cover the full range from normal operation to complete core uncovering. For example, if water level is chosen as the unambiguous indication, then the range of the instrument (or instruments) must cover the full range from normal water level to the bottom of the core.

PG&E Status

PG&E is presently evaluating technical proposals from Westinghouse for a reactor vessel level indication system. PG&E plans to purchase equipment for such a system by January 15, 1980, to be installed by January 1, 1981. 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.

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.)

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

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 (Continued)

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.)

Clarification

1. Provide diverse containment isolation signals that satisfy safety-grade requirements.

2. Identify essential and non-essential systems and provide results to NRC.

3. Non-essential system should be automatically isolated by containment isolation signals.

4. Resetting of containment isolation signals shall not result in the automatic loss of containment isolation.

PG&E Status

PG&E considers the Diablo Canyon design to fully comply with the Task Force Positions and clarifications on containment isolation. Refer to PG&E's NUREG-0578 response August 27, 1979.

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.)

Clarification

1. This requirement is only applicable to those plants whose licensing basis includes requirements for external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere.

2. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

Section 2.1.5.a (Continued)

3. The dedicated penetration or the combined single-failure proof alternative should be sized such that the flow requirements for the use of the recombiner or purge system are satisfied.

4. Components necessitated by this requirement should be safety grade.

5. A description of required design changes and a schedule for accomplishing these changes should be provided by January 1, 1980. Design changes should be completed by January 1, 1981.

<u>PG&E Status</u>

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 external 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.

Section 2.1.5.a (Continued)

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.

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 Status

No action is required. Both Diablo Canyon Units 1 and 2 are Westinghouse PWR's.

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.)

Clarification

1. This requirement applies only to those plants that included Hydrogen Recombiners as a design basis for licensing.

2. The shielding and associated personnel exposure limitations associated with recombiner use should be evaluated as part of licensee response to requirement 2.1.6.b, "Design review for Plant Shielding".

Section 2.1.5.c (Continued)

3. Each licensee should review and upgrade, as necessary, those criteria and procedures dealing with recombiner use. Action taken on this requirement should be submitted by January 1, 1980.

PG&E 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.

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.)

Section 2.1.6.a (Continued)

Clarification

Licensees shall, by January 1, 1980, provide a summary description of their program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. Examples of such systems are given on page A-26 of NUREG-0578. Other examples include the Reactor Core Isolation Cooling and Reactor Water Cleanup (letdown function) Systems for BWRs. Include a list of systems which are excluded from this program. Testing of gaseous systems should include helium leak detection or equivalent testing methods. Consider in your program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to you regarding North Anna and Related Incidents dated October 17, 1979.

PG&E Status

A summary description of this program will be provided prior to the January 9, 1980 review meeting scheduled with the NRC.

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 (i.e., the equivalent of 50% of the core radioiodine and 100% of the core noble gas inventory are contained in the primary coolant), 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.)

Section 2.1.6.b (Continued)

Clarification

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. In order to assure that personnel can perform necessary post-accident operations in the vital areas, we are providing the following guidance to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. Source Term

The minimum radioactive source term should be equivalent to the source terms recommended, in Regulatory Guides 1.3, 1.4, 1.7 and Standard Review Plant 15.6.5. with appropriate decay times based on plant design.

a. Liquid Containing Systems: 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory and 1% of all others are assumed to be mixed in the reactor coolant and liquids injected by HPCI and LPCI.

b. Gas Containing Systems: 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For gas containing lines connected to the primary system (e.g., BWR steam lines) the concentration of radioactivity shall be determined assuming the activity is contained in the gas space in the primary coolant system.

2. <u>Dose Rate Criteria</u>

The dose rate for personnel in a vital area should be such that the guidelines of GDC 19 should not be exceeded during the course of the accident. GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. When determining the dose to an operator, care must be taken to determine the necessary occupancy time in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines provided occupancy is not required at the location of the hot spot. These doses are design objectives and are <u>not</u> to be used to limit access in the event of an accident.

a. <u>Areas Requiring Continuous Occupancy</u>: $\leq$15mr/hr. These areas will require full time occupancy during the course of the accident. The Control Room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in SRP 6.4.

Section 2.1.6.b (Continued)

b. Areas Requiring Infrequent Access: GDC 19. These areas may require access on a regular basis, but not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant Radiochemical/Chemical Analysis Laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples where occupancy may be needed often but not continuously.

PG&E 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 scenerios. 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.

Section 2.1.6.b (Continued)

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.

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

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:

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

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.

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

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

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.

Section 2.1.7.a (Continued)

6. The ac 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.

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

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

Clarification

Control Grade (Short-Term)

1. Provide automatic/manual initiation of AFWS.

2. Testability of the initiating signals and circuits is required.

3. Initiating signals and circuits shall be powered from the emergency buses.

Section 2.1.7.a (Continued)

4. Necessary pumps and valves shall be included in the automatic sequence of the loads to the emergency buses. Verify that the addition of these loads does not comprimise the emergency diesel generating capacity.

5. Failure in the automatic circuits shall not result in the loss of manual capability to initiate the AFWS from the control room.

6. Other Considerations

   a. For those designs where instrument air is needed for operation, the electric power supply requirement should be capable of being manually connected to emergency power sources.

PG&E Status

PG&E considers the Diablo Canyon design to fully comply with the Task Force Position and clarifications. Refer to PG&E's NUREG-0578 response August 27, 1979.

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:

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

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

Clarification

A. Control Grade (Short-Term)

1. Auxiliary feedwater flow indication to each steam generator shall satisfy the single failure criterion.

2. Testability of the auxiliary feedwater flow indication channels shall be a feature of the design.

Section 2.1.7.b (Continued)

3. Auxiliary feedwater flow instrument channels shall be powered from the vital instrument buses.

B. <u>Safety-Grade (Long-Term)</u>

1. Auxiliary feedwater flow indication to each steam generator shall satisfy safety-grade requirements.

C. <u>Other</u>

1. For the Short-Term the flow indication channels should by themselves satisfy the single failure criterion for each steam generator. As a fall-back position, one auxiliary feed water flow channel may be backed up by a steam generator level channel.

2. Each auxiliary feed water channel should provide an indication of feed flow with an accuracy on the order of ± 10%.

<u>PG&E Status</u>

Design changes have been made to upgrade existing indicators to safety grade requirements.

Section 2.1.7.b (Continued)

PG&E considers the Diablo Canyon design to fully comply with the Task Force Position and clarifications. Refer to PG&E's NUREG-0578 response August 27, 1979.

This action will be completed by May 1, 1980.

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.

Section 2.1.8.a (Continued)

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 and description of proposed modifications will be completed by January 1, 1980. Category B: Implementation of plant modifications and preparation of revised procedures will be completed by June 1, 1980.)

<u>Clarification</u>

The licensee shall have the capability to promptly obtain (in less than 1 hour) pressurized and unpressurized reactor coolant samples and a containment atmosphere (air) sample.

The licensee shall establish a plan for an onsite radiological and chemical analysis facility with the capability to provide, within 1 hour of obtaining the sample, quantification of the following:

1. Certain isotopes that are indicators of the degree of core damage (i.e., noble gases, iodines and cesiums and non-volatile isotopes),

2. Hydrogen levels in the containment atmosphere in the range 0 to 10 volume percent,

Section 2.1.8.a (Continued)

3. Dissolved gases, (i.e., H_2, O_2) and boron concentration of liquids,

or have in-line monitoring capabilities to perform the above analysis. Plant procedures for the handling and analysis of samples, minor plant modifications for taking samples and a design review and procedural modifications (if necessary) shall be completed by January 1, 1980. Major plant modifications shall be completed by January 1, 1981.

During the review of the post accident sampling capability consideration should be given to the following items:

1. Provisions shall be made to permit containment atmosphere sampling under both positive and negative containment pressure.

2. The licensee shall consider provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for passive flow restrictions to limit reactor coolant loss or containment air leak from a rupture of the sample line.

Section 2.1.8.a (Continued)

3. If changes or modifications to the existing sampling system are required, the seismic.design and quality group classification or sampling lines and components shall conform to the classification of the system to which each sampling line is connected. Components and piping downstream of the second isolation valve can be designed to quality Group D and nonseismic Category I requirements.

The licensee's radiological sample analysis capability should include provisions to:

a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Lessons Learned Item 2.1.6.b. Where necessary, ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/gm to the upper levels indicated here.

b. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

Section 2.1.8.a (Continued)

c. Maintain plant procedures which identify the analysis required, measurement techniques and provisions for reducing background levels.

The licensees chemical analysis capability shall consider the presence of the radiological source term indicated for the radiological analysis.

In performing the review of sampling and analysis capability, consideration shall be given to personnel occupational exposure. Procedural changes and/or plant modifications must assure that it shall be possible to obtain and analyze a sample while incurring a radiation dose to any individual that is as low as reasonably achievable and not in excess of GDC 19. In assuring that these limits are met, the following criteria will be used by the staff.

1. For shielding calculations, source terms shall be as given in Lessons Learned Item 2.1.6.b.

2. Access to the sample station and the radiological and chemical analysis facilities shall be through areas which are accessible in post accident situations and which are provided with sufficient shielding to assure that the radiation dose criteria are met.

3. Operations in the sample station, handling of highly radioactive samples from the sample station to the analysis facilities, and handling while working with the samples in the analysis facilities shall be such that the radiation dose criteria are met. This may involve sufficient shielding of personnel from the samples and/or the dilution of samples for analysis. If the existing facilities do not satisfy these criteria, then additional design features, e.g., additional shielding, remote handling, etc., shall be provided. The radioactive sample lines in the sample station, the samples themselves in the analysis facilities, and other radioactive lines of the vicinity of the sampling station and analysis facilities shall be included in the evaluation.

4. High range portable survey instruments and personnel dosimeters should be provided to permit rapid assessment of high exposure rates and accumulated personnel exposure.

The licensee shall demonstrate their capability to obtain and analyze a sample containing the isotopes discussed above according to the criteria given in this section.

Section 2.1.8.a (Continued)

PG&E Status

A shielding review of the Reactor Coolant System (RCS) and containment atmospheric sampling systems was performed to determine their shielding adequacy. The very high dose rates postulated by assuming TID 14188 source terms would have necessitated a major system redesign with extensive shielding additions to the existing sampling stations. The alternative chosen was to identify a remote sample location for emergency applications and upgrade the facilities of the chemistry laboratory to analyze high activity samples consistent with the requirements of NUREG-0578, Sections 2.1.6.b, 2.1.8.a, and subsequent clarifications.

In-line monitoring to remotely perform chemical and radiological spectrum analysis was considered. Problems with accurate dilution, low background requirements for equipment, measurement accuracy and reliability, and time constraints, necessitated the analysis to be performed in the chemistry laboratory by a method utilizing a retrievable sample.

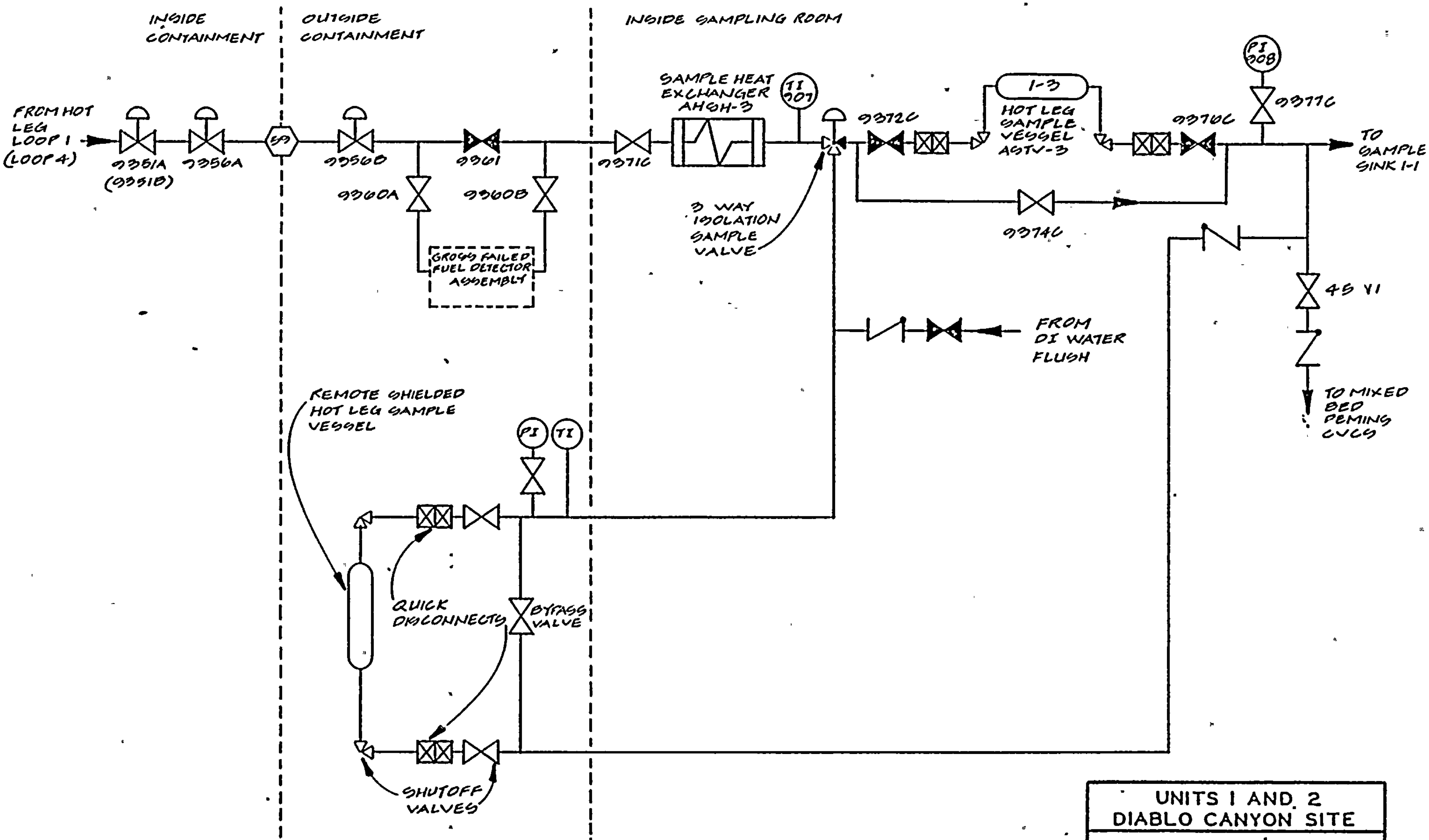
Three alternative remote sample station locations were identified for obtaining retrievable samples. A location was chosen on the 115 foot elevation because of low potential radiation considerations, spaciousness, and access to and from grade level. It is intended that both atmospheric and liquid sampling will be located here along with a semi-permanent hydrogen analyzer and remote sampling valve control panel. Both Units 1 and 2 will have separate, mirror image, remote sample stations.

Section 2.1.8.a (Continued)

Preliminary system flow diagrams for the remote RCS liquid and containment atmospheric sampling systems are shown in Figures 2.1.8.a-1, -2. The RCS liquid sample line will be connected downstream of the sample cooler heat exchanger. This line will be capable of being flushed with clean demineralized water. The containment atmospheric sample line will be connected in parallel with the containment particulate air monitor, and a semi-permanent hydrogen analyzer. Flushing was not provided for the atmospheric sample line to prevent interference with the use of the hydrogen analyzer.

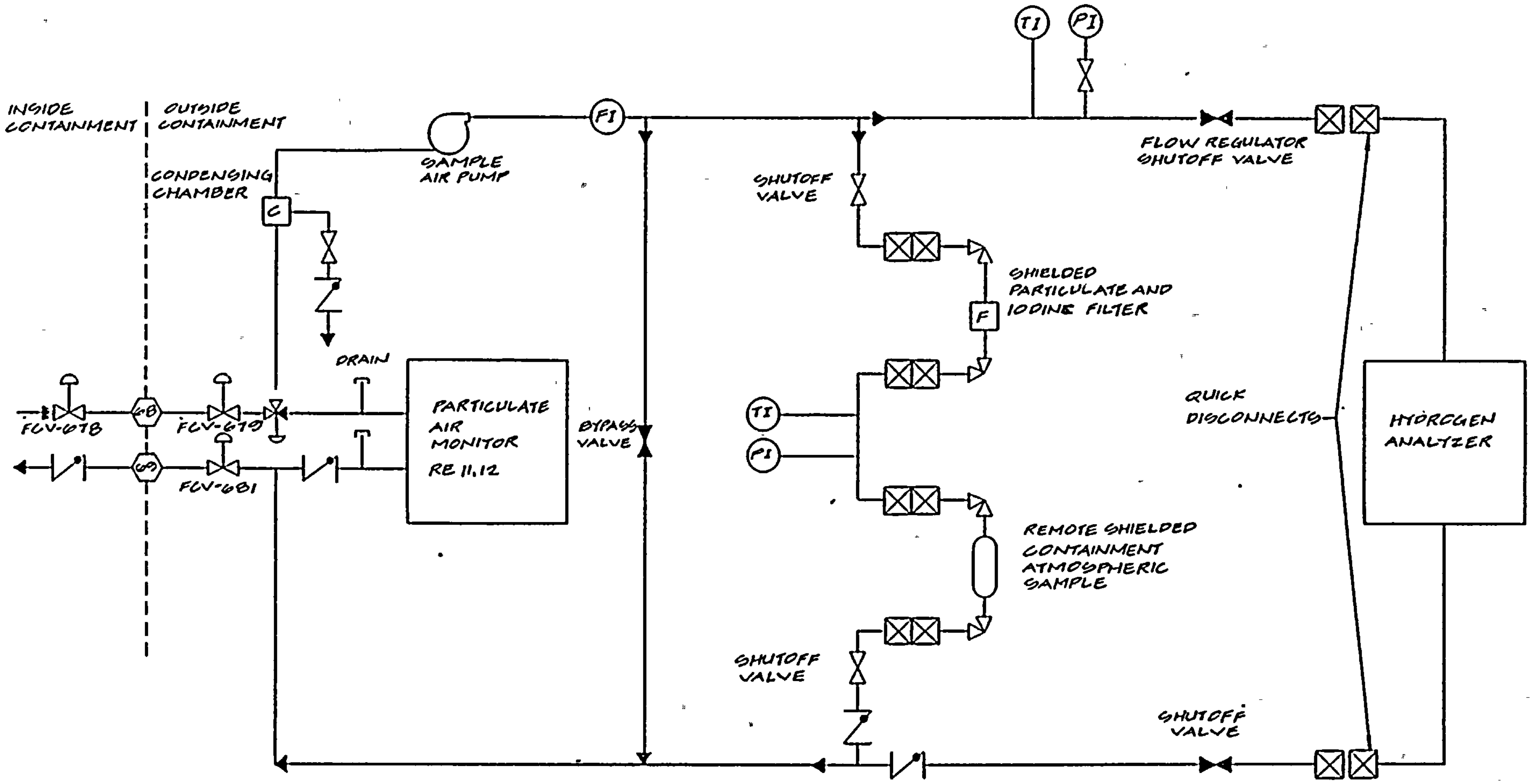
Liquid and atmospheric samples will be taken in shielded sample vessels to be transported to the chemistry laboratory for analysis. PG&E is obtaining an automatic titration system, additional lead shielding, and remote handling equipment to enhance the capability of high activity sample analysis.

Final designs for the remote sampling system will be completed by January 30, 1980. Installation of the remote sampling system, all analysis equipment and required procedural changes will be completed by May 1, 1980.



UNITS 1 AND 2
DIABLO CANYON SITE

PRELIMINARY FLOW DIAGRAM RCS
REMOTE LIQUID SAMPLING SYSTEM
FIGURE 2.1.8.a-1



UNITS 1 AND 2
DIABLO CANYON SITE

PRELIMINARY FLOW DIAGRAM REMOTE CONTAINMENT ATMOSPHERIC SAMPLING SYSTEM
FIGURE 2.1.8a-2

Section 2.1.8.b - Increased Range of Radiation Monitors

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.

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 concentrations extending from normal condition (ALARA) concentration 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 should overlap by a factor of ten.

Section 2.1.8.b (Continued)

2. Since iodine gaseous effluent monitors for the accident condition 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 adsorption on charcoal or other media, followed by onsite laboratory analysis.

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 complete by January 1, 1981.)

Clarification

1. Radiological Noble Gas Effluent Monitors

   a. January 1, 1980 Requirements

      Until final implementation in January 1, 1981, all operating reactors must provide, by January 1, 1980, an interim method for quantifying high level releases which meets the requirements of Table 2.1.8.b.1. This method is to serve only as a provisional fix with the more detailed, exact methods to follow. Methods are to be developed to quantify release rates of up to 10,000 Ci/sec for noble

gases from all potential release points, (e.g., auxiliary building, radwaste building, fuel handling building, reactor building, waste gas decay tank releases, main condenser air ejector, BWR main condenser vacuum pump exhaust, PWR steam safety valves and atmosphere steam dump valves and BWR turbine buildings) and any other areas that communicate directly with systems which may contain primary coolant or containment gases, (e.g., letdown and emergency core cooling systems and external recombiners). Measurements/ analysis capabilities of the effluents at the final release point (e.g., stack) should be such that measurements of individual sources which contribute to a common release point may not be necessary. For assessing radioiodine and particulate releases, special procedures must be developed for the removal and analysis of the radioiodine/particulate sampling media (i.e., charcoal canister/ filter paper). Existing, sampling locations are expected to be adequate; however, special procedures for retrieval and analysis of the sampling media under accident conditions (e.g., high air and surface contamination and direct radiation levels) are needed.

It is intended that the monitoring capabilities called for in the interim can be accomplished with existing instrumentation or readily available instrumentation. For noble gases, modifications to existing monitoring systems, such as the use of portable high range survey instruments, set in shielded collimators so that they "see" small sections of sampling lines is an acceptable method for meeting the intent of this requirement. Conversion of the measured dose rate (mR/hr) into concentration (μCi/cc) can be performed using standard volume source calculations. A method must be developed with sufficient accuracy to quantify the iodine releases in the presence of high background radiation from noble gases collected on charcoal filters. Seismically qualified equipment and equipment meeting IEEE-279 is not required.

The licensee shall provide the following information on his methods to quantify gaseous releases of radioactivity from the plant during an accident.

(1) Noble Gas Effluents

a) System/Method description including:

i) Instrumentation to be used including range or sensitivity, energy dependence, and calibration frequency and technique,

ii) Monitoring/sampling locations, including methods to assure representative measurements and background radiation correction,

iii) A description of method to be employed to facilitate access to radiation readings. For January 1, 1980, Control room read-out is perferred: however, if impractical, in-situ readings by an individual with verbal communication with the Control Room is acceptable based on (iv) below.

iv) Capability to obtain radiation readings at least every 15 minutes during an accident.

v) Source of power to be used. If normal AC power is used, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous readout for 7 consecutive days.

b) Procedures for conducting all aspects of the measurement/ analysis including:

i) Procedures for minimizing occupational exposures.

ii) Calculational methods for converting instrument readings to release rates based on exhaust air flow and taking into consideration radionuclide spectrum distribution as function of time after shutdown.

iii) Procedures for dissemination of information.

iv) Procedures for calibration.

b. January 1, 1981 Requirements

By January 1, 1981, the licensee shall provide high range noble gas effluent monitors for each release path. The noble gas effluent monitor should meet the requirements of Table 2.1.8.b.2. The licensee shall also provide the information given in Sections 1.A.1.a.i, 1.A.1.a.ii, 1.A.1.b.ii, 1.A.1.b.iii, and 1.A.1.b.iv above for the noble gas effluent monitors.

2. Radioiodine and Particulate Effluents

a. For January 1, 1980 the licensee should provide the following:

(1) System/Method description including:

a) Instrumentation to be used for analysis of the sampling media with discussion on methods used to correct for potentially interfering background levels of radioactivity.

b) Monitoring/sampling location.

c) Method to be used for retrieval and handling of sampling media to minimize occupational exposure.

d) Method to be used for data analysis of individual radionuclides in the presence of high levels of radioactive noble gases.

e) If normal AC power is used for sample collection and analysis equipment, an alternate back-up power supply should be provided. If DC power is used, the source should be capable of providing continuous read-out for 7 consecutive days.

(2) Procedures for conducting all aspects of the measurement analysis including:

a) Minimizing occupational exposure.

b) Calculational methods for determining release rates.

c) Procedures for dissemination of information.

d) Calibration frequency and technique.

b. For January 1, 1981, the licensee should have the capability to continuously sample and provide onsite analysis of the sampling media. The licensee should also provide the information required in 2.a above.

Section 2.1.8.b (Continued)

3. Containment Radiation Monitors

Provide by January 1, 1981, two radiation monitor systems in containment which are documented to meet the requirements of Table 2.1.8.b.2. It is possible that future regulatory requirements for emergency planning interfaces may necessitate identification of different types of radionuclides in the containment air, e.g., noble gases (indication of core damage) and non-volatiles (indication of core melt). Consequently, consideration should be given to the possible installation or future conversion of these monitors to perform this function.

TABLE 2.1.8.b.1

INTERIM PROCEDURES FOR QUANTIFYING HIGH LEVEL ACCIDENTAL RADIOACTIVITY RELEASES

- Licensees are to implement procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off scale.

- Examples of major elements of a highly radioactive effluent release special procedures (noble gas).

  - Preselected location to measure radiation from the exhaust air, e.g., exhaust duct or sample line.

  - Provide shielding to minimize background interference.

  - Use of an installed monitor (preferable) or dedicated portable monitor (acceptable) to measure the radiation.

  - Predetermined calculational method to convert the radiation level to radioactive effluent release rate.

TABLE 2.1.8.b.2

HIGH RANGE EFFLUENT MONITOR

- NOBLE GASES ONLY
- RANGE: (Overlap with Normal Effluent Instrument Range)

- UNDILUTED CONTAINMENT EXHAUST	10^{+5}	µCi/CC
- DILUTED (>10: 1) CONTAINMENT EXHAUST	10^{+4}	µCi/CC
- MARK I BWR REACTOR BUILDING EXHAUST	10^{+4}	µCi/CC
- PWR SECONDARY CONTAINMENT EXHAUST	10^{+4}	µCi/CC
- BUILDINGS WITH SYSTEMS CONTAINING PRIMARY COOLANT OR GASES	10^{+3}	µCi/CC
- OTHER BUILDINGS (E.G., RADWASTE)	10^{+2}	µCi/CC

- NOT REDUNDANT - 1 PER NORMAL RELEASE POINT
- SEISMIC - NO
- POWER - VITAL INSTRUMENT BUS
- SPECIFICATIONS - PER R.G. 1.97 AND ANSI N320-1979
- DISPLAY*: CONTINUOUS AND RECORDING WITH READOUTS IN THE TECHNICAL SUPPORT CENTER (TSC) AND EMERGENCY OPERATIONS CENTER (EOC)
- QUALIFICATIONS - NO

*Although not a present requirement, it is likely that this information may have to be transmitted to the NRC. Consequently, consideration should be given to this possible future requirement when designing the display interfaces.

TABLE 2.1.8.b.3

HIGH RANGE CONTAINMENT RADIATION MONITOR

- RADIATION: TOTAL RADIATION (ALTERNATE: PHOTON ONLY)
- RANGE:
  - UP TO 10^8 RAD/HR (TOTAL RADIATION)
  - ALTERNATE: 10^7 R/HR (PHOTON RADIATION ONLY)
  - SENSITIVE DOWN TO 60 KEV PHOTONS*
- REDUNDANT: TWO PHYSICALLY SEPARATED UNITS
- SEISMIC: PER R. G. 1.97
- POWER: VITAL INSTRUMENT BUS
- SPECIFICATIONS: PER R. G. 1.97 REV. 2 and ANSI N320-1978
- DISPLAY: CONTINUOUS AND RECORDING
- CALIBRATION: LABORATORY CALIBRATION ACCEPTABLE

*Monitors must not provide misleading information to the operators assuming delayed core damage when the 80 KEV photon Xe-133 is the major noble gas present.

Section 2.1.8.b (Continued)

PG&E Status

New instrumentation is being purchased to meet the Task Force Positions and clarifications. If the potential suppliers meet current delivery date commitments, installation is expected to be completed by October 1, 1980. For operation prior to October 1, 1980, PG&E will provide the necessary instrumentation and methods as required by the clarifications.

Section 2.1.8.c - Improved In-Plant Iodine Instrumentation Under Accident Conditions

Task Force Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident. (Category A: Implementation shall be completed prior to OL, or January 1, 1980, whichever is later.)

CLARIFICATION

1. Use of Portable versus Stationary Monitoring Equipment

   Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments for the following reasons:

   a. The physical size of the auxiliary/fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

   b. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

c. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

d. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high dose rate areas.

2. <u>Iodine Filters and Measurement Techniques</u>

a. The following are short-term recommendations and shall be implemented by the licensee by January 1, 1980. The licensee shall have the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single channel analyzer (SCA). The SCA window should be calibrated to the 365 keV of ^{131}I. A representative air sample shall be taken and then counted for ^{131}I using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

b. By January 1, 1981: The licensee shall have the capability to remove the sampling cartridge to a low background, low contamination area for further analysis. This area should be ventilated with clean air containing no airborne radionuclides which may contribute to inaccuracies in analyzing the sample. Here, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble bases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples and effluent charcoal samples under accident conditions.

PG&E Status

PG&E will meet the requirements of the Task Force Position and clarifications by locating an additional multichannel analyzer in a low background, low contamination area near the Technical Support Center. This addition to the system will give plant personnel the capability to accomplish the iodine analysis in the event that the chemistry lab becomes contaminated. This additional capability will be provided by May 1, 1980.

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 (schedules 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).

Section 2.1.9 (Continued)

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 operation 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).

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

Section 2.1.9 (Continued)

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.

Section 2.1.9 (Continued)

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 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 Status

All of the analyses have been completed or are underway and the status and schedule have been provided to the NRC by the Westinghouse Owners Group. Procedures identified by these analyses are presently being written. The procedures and operator training will be completed by May 1, 1980.

ACRS Comment No. 1

Containment Pressure Indication

Task Force Position

A continuous indication of containment pressure should be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

Clarification

1. The containment pressure indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment pressure monitor shall be installed by January 1, 1981.

PG&E Status

High level containment pressure transmitters are on order and will be delivered by February 1, 1980. The instrumentation will be installed by April 1, 1980, along with appropriate control board readouts.

ACRS Comment No. 2

Containment Water Level Indication

Task Force Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

Clarification

1. The narrow range sump level instrument shall monitor the normal containment sump level vice the containment emergency sump level.

2. The wide range containment water level instruments shall meet the requirements of the proposed revision to Regulatory Guide 1.97 (Instrumentation for Light-Water Cooled Nuclear Power Plant to Assess Plant Conditions During and Following a Accident).

ACRS Comment No. 2 (Continued)

3. The narrow range containment water level instruments shall meet the requirements of Regulatory Guide 1.89 (Qualification of Class 1E Equipment of Nuclear Power Plants).

4. The equivalent capacity of the wide range PWR level instrument has been changed from 500,000 gallons to 600,000 gallons to ensure consistency with the proposed revision to Regulatory Guide 1.97. It should be noted that this measurement capability is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

5. The containment water level indication shall be installed by January 1, 1981.

PG&E Status

The existing sump level instrumentation meets the requirements for narrow range, except that the level range is from the bottom of the recirculation sump to the highest level used for the recirculation mode. Since the operator is using this instrumentation for recirculation control, it is not advisable to reduce the range. A wide range sump level instrument will be purchased by February 15, 1980. The instrumentation will be installed by September 15, 1980.

ACRS Comment No. 3

Containment Hydrogen Indication

Task Force Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

Clarification

1. The containment hydrogen indication shall meet the design provisions of Regulatory Guide 1.97 including qualification, redundancy, and testability.

2. The containment hydrogen indication shall be installed by January 1, 1981.

PG&E Status

Containment hydrogen monitors will be ordered by March 15, 1980. The monitors will be installed by January 1, 1981, along with appropriate control board readouts.

ACRS Comment No. 4

Reactor Coolant System Venting

Task Force Position

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

1. A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

ACRS Comment No. 4 (Continued)

2. Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1), and Standard Review Plan Section 6.2.5.

3. Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

Clarification

A. General

1. The two important safety functions enhanced by this venting capability are core cooling and containment integrity. For events within the present design basis for nuclear power plants, the capability to vent non-condensible gases will provide additional assurance of meeting the requirements of 10 CFR 50.46 (LOCA criteria) and 10 CFR 50.44 (containment criteria for hydrogen generation). For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of non-condensible gas without the loss of core cooling or containment integrity.

ACRS Comment No. 4 (Continued)

2. Procedures addressing the use of the RCS vents are required by January 1, 1981. The procedures should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be based on the following criteria: (1) assurance that the plant can meet the requirements of 10 CFR 50.46 and 10 CFR 50.44 for Design Basis Accidents; and (2) a substantial increase in the plants ability to maintain core cooling and containment integrity for events beyond the Design Basis.

B. BWR Design Considerations

1. Since the BWR owners group has suggested that the present BWR designs inherent capability of venting, this question relates to the capability of existing systems. The ability of these systems to vent the RCS of non-condensible gas must be demonstrated. In addition the ability of these systems to meet the same requirements as the PWR vent systems must be documented. Since there are important differences among BWR's, each licensee should address the specific design features of his plant.

ACRS Comment No. 4 (Continued)

2. In addition to reactor coolant system venting, each BWR licensee should address the ability to vent other systems such as the isolation condenser, which may be required to maintain adequate core cooling. If the production of a large amount of non-condensible gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

C. PWR Vent Design Considerations

1. The location for PWR Vents are as follows:

   a. Each PWR licensee should provide the capability to vent the reactor vessel head.

   b. The reactor vessel head vent should be capable of venting non-condensible gas from the reactor vessel hot legs (to the elevation of the top of the outlet nozzle) and cold legs (through head jets and other leakage paths). Additional venting capability is required for those portions of each hot leg which can not be vented through the reactor vessel head vent. The NRC recognizes that it is impractical to vent each of the many thousands of tubes in a U-tube steam generator. However, we believe that a

procedure can be developed which assures that sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the reactor coolant system. Such a procedure is required by January 1981.

c. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations especially during natural circulation.

2. The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly large range of sizes. The criteria for sizing a vent can be developed in several ways. One approach, which we consider reasonable, is to specify a volume of non-condensible gas to be vented and a venting time i.e, a vent capable of venting a gas volume of 1/2 the RCS in one hour. Other criteria and engineering approaches should be considered if desired.

3. Where practical, the RCS vents should be kept smaller than the size corresponding to the definition of a LOCA (10 CFR 50 Appendix A). This will minimize the challenges to the ECCS since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation although it may result in leakage beyond Technical

ACRS Comment No. 4 (Continued)

Specification Limits. On PWRs the use of new or existing valves which are larger than the LOCA definition will require the addition of a block valve which can be closed remotely to terminate the LOCA resulting from the inadvertent opening of the vent.

4. An indication of valve position should be provided in the control room.

5. Each vent should be remotely operable from the control room.

6. Each vent should be seismically qualified.

7. The requirements for a safety grade system is the same as the safety grade requirement on other Short Term Lessons Learned items, that is, it should have the same qualifications as were accepted for the reactor protection system when the plant was licensed. The exception to this requirement is that we do not require redundant valves at each venting location. Each vent must have its power supplied from an emergency bus. A degree of redundancy should be provided by powering different vents from different emergency buses.

8. For systems where a block valve is required, the block valve should have the same qualifications as the vent.

ACRS Comment No. 4 (Continued)

9. Since the RCS vent system will be part of the reactor coolant systems boundary, efforts should be made to minimize the probability of an inadvertent actuation of the system. Removing power from the vents is one step in the direction. Other steps are also encouraged.

10. Since the generation of large quantities of non-condensible gas could be associated with substantial core damage, venting to atmosphere is unacceptable because of the associated released radioactivity. Venting into containment is the only presently available alternative. Within containment those areas which provide good mixing with containment air are preferred. In addition, areas which provide for maximum cooling of the vented gas are preferred. Therefore the selection of a location for venting should take advantage of existing ventilation and heat removal systems.

11. The inadvertent opening of an RCS vent must be addressed. For vents smaller than the LOCA definition, leakage detection must be sufficient to identify the leakage. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10 CFR 50.46.

ACRS Comment No. 4 (Continued)

PG&E Status

The long lead time equipment required for a head vent system has been ordered by Westinghouse for PG&E. Design details are presently under discussion between PG&E and Westinghouse. These details are expected to be finalized by February 1, 1980. Equipment and piping for Reactor Coolant System venting will be installed, procedures finalized and operator training completed by January 1, 1981.

Section 2.2.1.a - Shift Supervisor's Responsibilities

Task Force Position

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 cleary 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.

Section 2.2.1.a (Continued)

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.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and this 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.)

Section 2.2.1.a (Continued)

CLARIFICATION

This Table provides clarification to the above position.

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.a)

NUREG-0578 POSITION (POSITION NO.)	CLARIFICATION
Highest Level of Corporate Management (1.)	V.P. for Operations
Periodically Reissue (1.)	Annual Reinforcement of Company Policy
Management Direction (1.)	Formal Documentation of Shift Personnel, All Plant Management, Copy to IE Region
Properly Defined (2.0)	Defined in Writing in a Plant Procedure
Until Properly Relieved (2.B)	Formal Transfer of Authority, Valid SRO License, Recorded in Plant Log
Temporarily Absent (2.C)	Any Absence
Control Room Defined (2.C)	Includes Shift Supervisor Office Adjacent to the Control Room
Designated (2.C)	In Administrative Procedures
Clearly Specified	Defined in Administrative Procedures
SRO Training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative Duties (4.)	Not Affecting Plant Safety
Administrative Duties Reviewed (4.)	One Same Interval as Reinforcement: i.e., Annual by VP for Operations

Section 2.2.1.a (Continued)

PG&E Status

A review of the administrative procedures defining the shift supervisors' responsiblities is under way. The review includes the subject of chain of command in the event of temporary absence of the shift supervisor. These procedures will be completed and approved by March 1, 1980.

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.)

Section 2.2.1.b (Continued)

Clarification

1. Due to the similarity in the requirements for dedication to safety, training and onsite location and the desire that the accident assessment function be performed by someone whose normal duties involve review of operating experiences, our preferred position is that the same people perform the accident and operating experience assessment functions. The performance of these two functions may be split if it can be demonstrated the persons assigned the accident assessment role are aware, on a current basis, of the work being done by those reviewing operating experience.

2. To provide assurance that the STA will be dedicated to concern for the safety of the plant, our position has been that STA's must have a clear measure of independence from duties associated with the commercial operation of the plant. The would minimize possible distractions from safety judgments by the demands of commercial operations. We have determined that, while desirable, independence from the operations staff of the plant is not necessary to provide this assurance. It is necessary, however, to clearly emphasize the dedication to safety associated with the STA position both in the STA job description and in the personnel filling this position. It is not acceptable to assign a person, who is normally the immediate supervisor of the shift supervisor to STA duties as defined herein.

3. It is our position that the STA should be available within 10 minutes of being summoned and therefore should be onsite. The onsite STA may be in a duty status for periods of time longer than one shift, and therefore asleep at some times, if the ten minute availability is assured. It is preferable to locate those doing the operating experience assessment onsite. The desired exposure to the operating plant and contact with the STA (if these functions are to be split) may be able to be accomplished by a group, normally stationed offsite, with frequent onsite presence. We do not intend, at this time, to specify or advocate a minimum time onsite.

4. The implementation schedule for the STA requirements is to have the STA on duty by January 1, 1980, and to have STAs, who have all completed training requirements, on duty by January 1, 1981. While minimum training requirements have not been specified for January 1, 1980, the STAs on duty by that time should enhance the accident and operating experience assessment function at the plant.

PG&E 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.

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).

Section 2.2.1.c (Continued)

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 and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

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

PG&E Status

These procedures will be completed and approved by April 1, 1980.

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 Status

These procedures will be completed and approved by April 1, 1980.

Section 2.2.2.b - Onsite Technical Support Center

Task Force Position

Each operating nuclear power plant shall maintain an onsite 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.)

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.)

Section 2.2.2.b (Continued)

Clarification

1. By January 1, 1980, each licensee should meet items A-G that follow. Each licensee is encouraged to provide additional upgrading of the TSC (items 2-10) as soon as practical, but no later than January 1, 1981.

   a. Establish a TSC and provide a complete description,

   b. Provide plans and procedures for engineering/management support and staffing of the TSC,

   c. Install dedicated communications between the TSC and the control room, near site emergency operations center, and the NRC. Provide, between the TSC and the control room, a capability for the transmittal of some data. This requirement could be satisfied by closed circuit television or process computer printout,

   d. Provide monitoring (either portable or permanent) for both direct radiation and airborne radioactive contaminants. The monitors should provide warning if the radiation levels in the support center are reaching potentially dangerous levels. The licensee should designate action levels to define when protective measures should be taken (such as using breathing apparatus and potassium iodide tablets, or evacuation to the control room),

Section 2.2.2.b (Continued)

e. Assimilate or ensure access to Technical Data, including the licensee's best effort to have direct display of plant parameters, necessary for assessment in the TSC,

f. Develop procedures for performing this accident assessment function from the control room should the TSC become uninhabitable, and

g. Submit to the NRC a longer range plan for upgrading the TSC to meet all requirements.

2. Location

It is recommended that the TSC be located in close proximity to the control room to ease communications and access to technical information during an emergency. The center should be located onsite, i.e., within the plant security boundary. The greater the distance from the CR, the more sophisticated and complete should be the communications and availability of technical information. Consideration should be given to providing key TSC personnel with a means for gaining access to the control room.

3. <u>Physical Size and Staffing</u>

The TSC should be large enough to house 25 persons, necessary engineering data and information displays (TV monitors, recorders, etc.). Each licensee should specify staffing levels and disciplines reporting to the TSC for emergencies of varying severity.

4. <u>Activation</u>

The center should be activated in accordance with the "Alert" level as defined in the NRC document "Draft Emergency Action Level Guidelines, NUREG-0610" dated September, 1979, and currently out for public comment. Instrumentation in the TSC should be capable of providing displays of vital plant parameters from the time the accident began (t = 0 defined as either reactor or turbine trip). The Shift Technical Advisor should be consulted on the "Notification of Unusual Event" however, the activation of the TSC is discretionary for that class of event.

Section 2.2.2.b (Continued)

5. Instrumentation

The instrumentation to be located in the TSC need not meet safety-grade requirements but should be qualitatively comparable (as regards accuracy and reliability) to that in the control room. The TSC should have the capability to access and display plant parameters independent from actions in the control room. Careful consideration should be given to the design of the interface of the TSC instrumentation to assure that addition of the TSC will not result in any degradation of the control room or other plant functions.

6. Instrumentation Power Supply

The power supply to the TSC instrumentation need not meet safety-grade requirements, but should be reliable and of a quality compatible with the TSC instrumentation requirements. To insure continuity of information at the TSC, the power supply provided should be continuous once the TSC is activated. Consideration should be given to avoid loss of stored data (e.g., plant computer) due to momentary loss of power or switching transients. If the power supply is provided from a plant safety-related power source, careful attention should be given to assure that the capability and reliability of the safety-related power source is not degraded as a result of this modification.

Section 2.2.2.b (Continued)

7. Technical Data

Each licensee should establish the technical data requirements for the TSC, keeping in mind the accident assessment function that has been established for those persons reporting to the TSC during an emergency. As a minimum, data (historical in addition to current status) should be available to permit the assessment of:

Plant Safety Systems Parameters for:

- Reactor Coolant System
- Secondary System (PWRs)
- ECCS Systems
- Feedwater and Makeup Systems
- Containment

In-Plant Radiological Parameters for:

- Reactor Coolant System
- Containment
- Effluent Treatment
- Release Paths

Offsite Radiological

. Meteorology

. Offsite Radiation Levels

8. Data Transmission

In addition to providing a data transmission link between the TSC and the control room, each licensee should review current technology as regards transmission of those parameters identified for TSC display.

Although there is not a requirement at the present time, each licensee should investigate the capability to transmit plant data offsite to the Emergency Operations Center, the NRC, the reactor vendor, etc.

9. Structural Integrity

a. The TSC need not be designed to seismic Category I requirements. The center should be well built in accordance with sound engineering practice with due consideration to the effects of natural phenomena that may occur at the site.

Section 2.2.2.b (Continued)

b. Since the center need not be designed to the same stringent requirements as the Control Room, each licensee should prepare a backup plan for responding to an emergency from the control room.

10. Habitability

The licensee should provide protection for the technical support center personnel from radiological hazards including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4.

a. Licensee should assure that personnel inside the technical support center (TSC) will not receive doses in excess of those specified in GDC 19 and SRP 6.4 (i.e., 5 Rem whole body and 30 Rem to the thyroid for the duration of the accident). Major sources of radiation should be considered.

b. Permanent monitoring systems should be provided to continuously indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. The monitoring systems should include local alarms to warn personnel of adverse conditions. Procedures must be provided which will specify appropriate protective actions to be taken in the event that high dose rates or airborne radioactive concentrations exist.

c. Permanent ventilation systems which include particulate and charcoal filters should be provided. The ventilation systems need not be qualified as ESF systems. The design and testing guidance of Regulatory Guide 1.52 should be followed except that the systems do not have to be redundant, seismic, instrumented in the control room or automatically activated. In addition, the HEPA filters need not be tested as specified in Regulatory Guide 1.52 and the HEPA's do not have to meet the QA requirements of Appendix B to 10 CFR 50. However, spare parts should be readily available and procedures in place for replacing failed components during an accident. The systems should be designed to operate from the emergency power supply.

d. Dose reduction measures such as breathing apparatus and potassium iodide tablets cannot be used as a design basis for the TSC in lieu of ventilation systems with charcoal filters. However, potassium iodide and breathing apparatus should be available.

PG&E Status

The onsite technical support center will be located in the Unit 2 Turbine Building buttresses. The design criteria are now being formulated and will be completed by January 11, 1980. Detailed design will be issued by February 1 with construction and installation of equipment to be completed by August 1, 1980. A complete description of the onsite technical support center, and plans and procedures relating to its use, as described in clarifications 1.a through 1.g, will be included in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

Section 2.2.2.c - Onsite Operational Support Center

Task Force Position

An area to be designated as the onsite 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. 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 Status

The operational support center will be located in the Security Building. The design criteria are now being formulated. Information required by the Task Force Position will be provided in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

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.

Section 2.2.3 (Continued)

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.

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.

Section 2.2.3 (Continued)

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.)

PG&E Status

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

Status of Commitments

From PG&E's

July 1979

Response Programs Following the

Accident at Three Mile Island

Section 3

A.1 - Safety Injection System Initiation

Description

Remove coincident pressurizer low water level and low pressure initiation of safety injection by permanently modifying the initiation logic. The modification will provide automatic initiation of safety injection on any two of four low pressure signals.

PG&E Status

A design modification for Safety Injection Actuation has been developed so that it will be initiated on any two of four low pressurizer pressure signals. Installation of this design modification is in progress and will be completed by February 1, 1980.

A.2 - Pressurizer Heaters

Description

Provide operating instructions and training to assure that plant operators are knowledgeable of the procedure for loading the pressurizer heaters onto the onsite emergency power system.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.1 Task Force Position 2.

A.3 - Pressurizer Relief Valves

Description

Review and revise, as necessary, the testing and maintenance procedures to minimize power operated relief valve opening.

PG&E Status

This review is currently in progress and will be completed by May 1, 1980.

A.4 - Pressurizer Relief Valves

Description

Investigate the possibility of automatically closing the PORV isolation valves below some minimum pressure and establish a schedule for implementation if appropriate.

PG&E Status

It has been determined that any such automatic closure would invalidate the safety function of the PORV's because of the slow operating times of the block valves. Therefore no such change will be incorporated. This item is considered completed.

A.5 - Post-Accident Monitoring Instruments Review

Description

Review with Westinghouse the adequacy of the post accident monitoring instruments in light of the TMI accident. This review will make use of an evaluation of parameters essential for post accident monitoring which is in progress for the Westinghouse Operating Plant Owner's Group.

PG&E Status

Investigations into the instrumentation requirements for this activity are in progress, in conjunction with the review of Regulatory Guide 1.97, Revision 2. Implementation will be commensurate with the Regulatory Guide requirements. See also PG&E status for NUREG-0578, Section 2.1.3.b.

A.6 - Control of Hydrogen in Containment

Description

Conduct a review of previous analysis of post-accident hydrogen generation and control in light of the TMI accident. If it is determined that current methods of hydrogen control are no longer acceptable, interim measures will be taken to provide for the use of portable offsite recombiners and long-term measures will be taken which will include the installation of a permanent onsite recombiner system.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.5.c.

A.7 - Plant Ventilation Systems

Description

Conduct a complete review of the safety analyses and operating procedures for the auxiliary building and control room ventilation systems in light of the TMI accident. A schedule for implementation of any changes that are identified as necessary will be established.

PG&E Status

Control Room Ventilation Review Progress

Positive pressurization capability is currently being added to the Control Room ventilation system. As soon as this capability has been added this item will have been completed. This is expected to be completed by May 1, 1980.

Auxiliary Building Ventilation Review Progress

The NRC (D. B. Vassallo) letter of November 9, 1979, contained further definition and clarification of the requirements of NUREG-0578. This letter also established the requirement that consideration be given to leakage release paths such as those identified in NRC letter (D. G. Eisenhut) of October 17, 1979, regarding the incident and release at North Anna on September 25, 1979.

A.7 (Continued)

Feasibility of various modifications are ~~being~~ investigated. A course of action will be determined by February 1, ~~1980~~, and modifications deemed necessary will be completed by May 1, 1980.

A.8 - Emergency Power Systems

Description

Review the availability of alternate emergency power sources and establish and instigate plans for providing alternate offsite or additional onsite power within a short period of time following an accident.

PG&E Status

A revew is scheduled for completion by January 31, 1980.

A.9 - Site Emergency Planning - Communications

Description:

Augment existing communications by installing dedicated telephone communications from the plant control room and other selected onsite locations to the San Luis Obispo County Emergency Operations Center, the Headquarters of the California Office of Emergency Services, the NRC Headquarters, and the NRC Regional I&E Office.

PG&E Status

The above activities are being accomplished and specific information will be included in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

A.10 - Site Emergency Planning - Supplies

Description

Review emergency plan supplies and equipment.

PG&E Status

The above activities are being accomplished and specific information will be included in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

A.11 - Site Emergency Planning - Emergency Notification

Description

Develop and add to the Diablo Canyon Emergency Plan more definitive criteria for plant operators regarding notification of state and local officials for events that are of potential public interest although having minimal offsite consequences.

PG&E Status

The above activities are being accomplished and specific information will be included in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

A.12 - Operator Training - Staff

Description

Increase the full-time training staff from two to three by providing an additional Assistant Training Coordinator.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.2.1.b.

A.13 - Operator Training - Supervisor of Operations

Description

Increase technical expertise of the Operations staff by adding an engineer to assist the Supervisor of Operations.

PG&E Status

Refer to PG&E Status of NUREG-0578, Section 2.2.1.b.

A.14 - Normal and Emergency Operating Procedures

Description

Modify operating and emergency procedures based on current PG&E, Westinghouse, and Westinghouse Operating Plant Owner's Group Studies.

PG&E Status

Refer to PG&E Status NUREG-0578, Sections 2.1.3.b, and 2.1.9.

A.15 - Control of System Status - Indentifying Inoperative Equipment

Description

Improve identification of inoperable equipment through modification of existing and installation of new equipment status boards.

PG&E Status

Refer to PG&E Status of NUREG-0578, Section 2.2.1.c.

A.16 - Control of System Status - Clearance Procedures

Description

Review and revise test and clearance procedures to ensure that the redundant component or system is operable before a safety-related component or system is removed from service.

PG&E Status

PG&E will develop the required procedures and implement the proper training of the operators. The procedures will be written and approved, and the operators trained, by April 1, 1980.

A.17 - Control of System Status - Operational Checklist

Description

Develop procedures formalizing the current practices for using checklists to verify important process control settings and for marking piping diagrams to show complicated or unusual lineups.

PG&E Status

PG&E will develop the required procedures and implement the proper training of the operators. The procedures will be written and approved, and the operators trained, by April 1, 1980.

A.18 - Shift Turnover

Description

Review shift turnover procedures to formalize the requirement for walkdown of control panels and the use of checklists.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.2.1.c.

A.19 - Operating Experience Review

Description

Review and revise plant procedure for evaluation and dissemination of information to plant personnel to more formally include all areas of operating experience.

PG&E Status

PG&E will develop the required procedures and implement the proper training of the operators. The procedures will be written and approved, and the operators trained, by April 1, 1980.

A.20 - Control of Access to the Control Room

Description

Issue procedures to formalize existing practices for restricting access to the control room.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.2.2.a.

A.21 - Review, Approval and Use of Procedures

Description

Review and revise as necessary the administrative procedures for development and use of operating and test procedures.

PG&E Status

PG&E will review the required procedures and revise as necessary. The procedures will be written and approved by April 1, 1980.

B.

Prior to Initial Power Operation

B.1 - Containment Isolation

Description

Conduct a review to determine if containment isolation should be provided on high radiation level for potentially radioactive flow paths which do not currently have this provision and establish a schedule for any modification found to be necessary.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.4.

B.2 - Incore Thermocouple

Description

Provide the operator with the ability to read incore thermocouple temperatures directly over the full range (to 1,900°F) using the plant computer.

PG&E Status

This item has been completed. Due to input limitations, 1600°F is the maximum temperature readable using the plant computer.

B.3 - Residual Heat Removal

Description

Conduct a comprehensive evaluation of the part of the RHR system used for normal cooling and of the non-ECCS portion of the system to determine if changes are required or desired in light of the TMI accident. In addition, evaluate the possible use of the steam generators as water-to-water heat exchangers for long-term cooling.

PG&E Status

An industry-wide review of residual heat removal schemes, both high and low energy is currently in progress. This review, scheduled for completion by January 1, 1980, will be followed by the development of an action plan for Diablo Canyon by March 1, 1980.

B.4 - Reactor Coolant Saturation Indication

Description

Provide control room indication of the margin between reactor coolant system pressure and the saturation pressure for the reactor coolant system temperature.

PG&E Status

A computer program has been written to provide margin to saturation pressure on demand. The program is now fully operational on the Unit 2 computer and will soon be transferred to the Unit 1 computer. See also PG&E status for NUREG-0578, Section 2.1.3.b.

B.5 - Reactor Coolant Sampling System

Description

Complete a comprehensive review of the reactor coolant sampling system with respect to radiation exposure.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.6.b.

B.6 - Radwaste Systems Emergency Capacity

Description

Review radioactive waste handling and treatment systems and operating procedures to identify and implement necessary changes prior to power operation.

PG&E Status

Review is scheduled to begin February 15, 1980. The review and any resultant procedural changes will be completed by May 1, 1980.

B.7 - Steam Generator Level Instrumentation

Description

Complete the review of steam generator water level instrumentation performance with reference legs heated due to high energy line breaks inside containment. Adjust the steam generator low-low water level trip setpoint and revise emergency operating procedures as appropriate. This potential problem was reported to the NRC by Westinghouse on June 21, 1979 under 10 CFR 21.

PG&E Status

This analysis was a Westinghouse Part 21 review that has been completed. This analysis is currently being reviewed by PG&E and will be submitted to the NRC by January 18, 1980.

B.8 - Corporate Response Plan

Description

Finalize the Corporate Emergency Response Plan and implementing procedures.

PG&E Status

Comments on the draft issued October 16, 1979 have been received. Ongoing work on the Corporate Emergency Response Plan involves resolution of the aforementioned comments and coordination of corporate planning with site, state and local planning efforts, and writing the response plan implementing procedures. This work will be completed by April 1, 1980.

B.9 - Emergency Response Center - Offsite

Description

Establish an Offsite (local) Incident Response Center. Establish an Interim Onsite Incident Response Center and in Interim Plant Operations Support Center in the present temporary Administration Building. Make arrangements with the Company's General Construction Department for office and living trailers to be available on short notice for these sites.

PG&E Status

Refer to PG&E Status for NUREG-0578, Sections 2.2.2.b and 2.2.2.c. This information will be provided in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

B.10 - Emergency Response Center - Corporate

Description

Establish a Corporate Incident Response Center.

PG&E Status

Refer to PG&E Status, Section B.8 of this report.

B.11 - Operator Training

Description

Add simulator training to the licensed operator requalification program.

PG&E Status

Since this activity concerns operator requalification program, it will be implemented after plant operation. See also PG&E Status, Section C.4 of this report.

C.

As Soon
As Practicable

C.1 - Reactor Vessel Vent System

Description

Determine the necessity for a reactor vessel vent system and, if required, establish a schedule.

PG&E Response

Refer to PG&E Status for ACRS Comment No. 4.

C.2 - Plant Ventilation System

Description

Implement any changes to the control room and auxiliary building ventilation systems that are identified as necessary by the pre-fuel loading study.

PG&E Status

Refer to PG&E Status, Section A.7 of this report.

C.3 - Site Emergency Planning

Description

Stock thyroid-blocking pills at the site for plant personnel.

PG&E Status

This information will be provided in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

C.4 - Operator Training - Simulator

Description

Conduct simulator training for all operating personnel who are hot license candidates.

PG&E Status

Training will be conducted at the Westinghouse Zion training facility from January to April 1980.

C.5 - Operator Training - Fifth Shift

Description

Add a fifth shift of operators to accommodate the increased level of training.

PG&E Status

It is anticipated that the fifth shift of operators can be added by July 1, 1980.

D.

Complete During First Refueling Outage

D.1 - Post-Accident Monitoring Instruments

Description

If modifications to wide range or post-incident monitoring instrumentation are determined to be necessary by a pre-power operation evaluation, the modification will be completed.

PG&E Status

Refer to PG&E Status for Section A.5 of this report.

D.2 - Wide Range Monitoring Instruments

Descriptions

Complete installation of the Regulatory Guide 1.97 instrumentation. (After the NRC has provided guidance on positions C.1 and C.2 of the guide, changes will be implemented but not necessarily during the first refueling outage.)

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.3.b and Section A.5 of this report.

E.

Long Term

E.1 - Pressurizer Heaters

Description

Pressurizer heater requirements identified by a current Westinghouse study for the Westinghouse Operating Plant Owner's Group will be met by implementing appropriate modifications and procedural changes.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.1.

E.2 - Reactor Vessel Level Indication

Description

Continue reviewing the desirability and practicality of a reactor vessel fluid level indication system. This review is currently in progress and has indicated that power operation will not preclude the installation of a fluid level indication system.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.3.b.

E.3 - Residual Heat Removal

Description

Implement any changes to the RHR system or steam generator systems determined to be necessary following completion of the evaluation in B.3 above.

PG&E Status

Refer to PG&E Status, Section B.3 of this report.

E.4 - Reactor Coolant Sampling System

Description

Implement any changes to the coolant sampling system following completion of the evaluation in B.5 above.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.8.a.

E.5 - Control of Hydrogen in Containment

Description

Install a permanent onsite recombiner system if so indicated by the review of previous analysis of post-accident hydrogen generation which is to be completed prior to initial fuel loading. Interim measures, if required, would provide for use of portable offsite recombiners.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.1.5.c.

E.6 - Radwaste Systems Emergency Capability

Description

Implement any changes to the radioactive waste handling and treatment systems which were determined to be necessary following completion of the evaluation in B.6 above.

PG&E Status

Refer to PG&E Status for Section B.6 of this report.

E.7 - Steam Generator Level Instrumentation

Description

Implement any changes necessary to the steam generator water level indication system based on a current Westinghouse investigation of potential problems due to reference leg heating due to a high energy line break inside containment. Appropriate interim action will be completed prior to power operation.

PG&E Status

Refer to PG&E Status for Section B.7 of this report.

E.8 - Site Emergency Planning

Description

Resolve emergency planning issues raised by Governor Brown's Nuclear Power Plant Emergency Review Panel.

PG&E Status

The activity described above is being evaluated in the context of an update that is currently underway to the existing site emergency plan.

E.9 - Corporate Response Plan Charter

Description

Expand the charter of the President's Nuclear Advisory Committee to conduct annual reviews of the Corporate Response Plan.

PG&E Status

The President's Nuclear Advisory Committee will conduct annual reviews of the Corporate Response Plan.

E.10 - Corporate Response Plan - Training

Description

Establish training program to provide continued assurance that the Corporate Response Plan can be properly implemented.

PG&E Status

A training program will be established after the Company's Corporate Response Plan (Section B.8 of this report) is finalized, on or about April 1, 1980.

E.11 - Emergency Response Centers

Description

Install a permanent Onsite Incident Response Center and a permanent Plant Operations Support Center in the permanent Administration Building which is currently in the final stages of design. (These facilities are provided for in the present temporary Administration Building.)

PG&E Status

Refer to PG&E Status for NUREG-0578, Sections 2.2.2.b and 2.2.2.c. This information will be provided in the revised Diablo Canyon Power Plant Units 1 and 2 Emergency Plan to be submitted in January 1980.

E.12 - Operator Selection

Description

Update operator selection methods based on results of EEI study currently in progress.

PG&E Status

The EEI study is scheduled for completion by March 1, 1981.

E.13 - Operator Training

Description

Evaluate installation of a plant specific operator training simulator.

PG&E Status

The installation of a training simulator for Diablo Canyon is being evaluated.

E.14 - Normal and Emergency Operating Procedures

Description

Modify operating and emergency procedures based on continuing industry and government TMI studies.

PG&E Status

Procedure modifications are in progress. Procedures will be written and approved by May 1, 1980.

E.15 - System Status Monitoring

Description

Evaluate a more comprehensive system for status monitoring of equipment.

PG&E Status

Refer to PG&E Status NUREG-0578, Section 2.2.1.c.

E.16 - Control of Access to the Control Room

Description

Evaluate the feasibility and advisability of using the computer-controlled card key access control system to restrict the number of persons having access to the control room during emergencies.

PG&E Status

Refer to PG&E Status for NUREG-0578, Section 2.2.2.a.