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 50-323 Diablo Canyon Nuclear Power Plant, Unit 2, Pacific Ga 05000323
 AUTH. NAME AUTHOR AFFILIATION
 CRANE, P.A., Pacific Gas & Electric Co.
 RECIP. NAME RECIPIENT AFFILIATION
 ENGELKEN, R.H. Region 5, San Francisco, Office of the Director

SUBJECT: Forwards revised respon to IE Bulletins 79-06A & 79-06A,
 Revision 1 furnished in response to verbal request from
 Ofc of nuclear Reactor Regualtion. Partially satifises
 Item II.K.C. of NUREG -0660.

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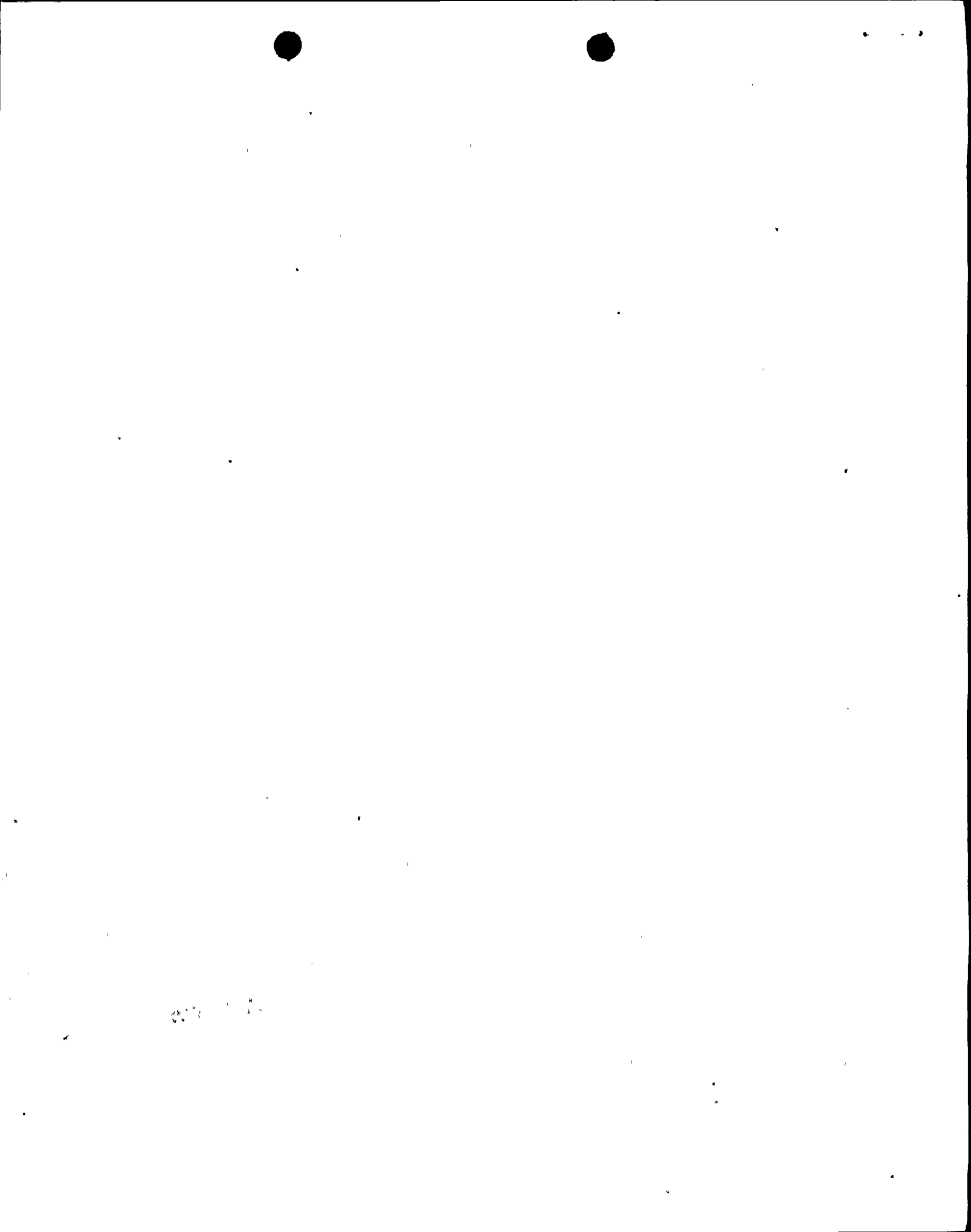
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PG&E + 77 BEALE STREET, 31ST FLOOR • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211

MALCOLM H. FURBUSH
VICE PRESIDENT AND GENERAL COUNSEL

ROBERT OHLBACH
ASSOCIATE GENERAL COUNSEL

CHARLES T. VAN DEUSEN

PHILIP A. CRANE, JR.

HENRY J. LAPLANTE

JOHN B. GIBSON

ARTHUR L. HILLMAN, JR.

CHARLES W. THISSELL

DANIEL E. GIBSON

ASSISTANT GENERAL COUNSEL

May 2, 1980

GILBERT L. MARRICK
GLENN WEST, JR.
JOSEPH I. KELLY
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ROBERT L. BORDON
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KENNETH YANG
ATTORNEYS

Mr. R. H. Engelken, Director
Office of Inspection and Enforcement
Region V
U. S. Nuclear Regulatory Commission
1990 North California Boulevard
Walnut Creek Plaza, Suite 202
Walnut Creek, California 94596

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

Dear Mr. Engelken:

Enclosed are copies of a revised response to IE Bulletin 79-06A and 79-06A, Revision 1. This revision is being furnished in response to a verbal request from the NRC Office of Nuclear Reactor Regulation, and in partial satisfaction of Item II.K.C.1 of NUREG-0660, "NRC Action Plans Developed as a Result of the TMI-2 Accident."

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

Very truly yours,

Philip A. Crane, Jr.

Enclosures

CC w/enc.: Mr. John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Service List

Boo!
S/H
ADD: C E
M WILLIAMS



IE BULLETIN 79-06A

UPDATED PGandE RESPONSE

ITEM 1

An extensive training presentation was developed by the Diablo Canyon operating staff covering the events that occurred at Three Mile Island Nuclear Plant. This presentation contained a detailed discussion and analysis of the events at Three Mile Island, emphasizing the serious effects of having both trains of auxiliary feedwater secured, the consequences of operator actions early in the event, the apparent personnel operating errors, and the failure to consult all information available from control room instrumentation during the transient. The presentation also included the consequences of overriding safety functions, the necessity for the use of all available instrumentation before making an operating decision, and the circumstances under which it is possible to have a low water level in the reactor without low pressurizer level.

Training for all licensed operators (or license candidates), plant management, and supervisors with operational responsibilities was held in June 1979. This training was also attended by the NRC Resident Inspector. Documentation of training session attendance was entered in the plant training records and is available for inspection in accordance with current requirements. This training was held on a one-time basis for the staff assigned at Diablo Canyon. In addition, PGandE videotaped the training sessions for use in educating additional sectors of the company.

Information resulting from the TMI-2 accident of lasting and permanent value to operators and other plant personnel was incorporated into the regular Diablo Canyon training program. In this manner, all new employees will continue to receive the benefit of the TMI-2 accident analysis.

ITEM 2

The Diablo Canyon operating procedures have been revised to incorporate the Westinghouse Owners' Group guidelines. These guidelines are contained in the recent revisions of the Westinghouse generic operating procedures, which were submitted to the NRC staff for review under cover of Westinghouse Owner's Group Letter OG-17 dated October 31, 1979. Included in the generic procedures reviewed were Procedure E-0 (Emergency Operating Procedure -- Immediate Actions/ Diagnostics) and E-1 (Emergency Operating Procedure -- Loss of Reactor Coolant). In addition, the staff intends to review generic Procedure A-6 (Abnormal Operating Procedure -- Natural Circulation at Hot Standby).

Revised PGandE emergency operating procedures (OP-0, OP-1, OP-2, and OP-3A) are appended to Section 2.1.9 of PGandE's "Response to NUREG-0578: Short Term Lessons Learned Requirements" ("NUREG-0578 Response"). Copies of these procedures have been sent to Westinghouse for a final review to assure conformance with the Westinghouse Owners' Group guidelines.



The procedures noted above include instructions for coping with transients and accidents as outlined in Bulletin Item 2. Reference should also be made to Section 2.1.3.b of "NUREG-0578 Response" which describes instrumentation available for detection of inadequate core cooling.

ITEM 3

The interim changes described in our initial response have been superceded by a permanent change in the initiating logic for safety injection. The modified logic incorporates a simple two-out-of-four logic to initiate safety injection on pressurizer low pressure.

ITEM 4

The subject of containment isolation, including the concerns of Bulletin Item 4, is discussed in "NUREG-0578 Response" Section 2.1.4.

ITEM 5

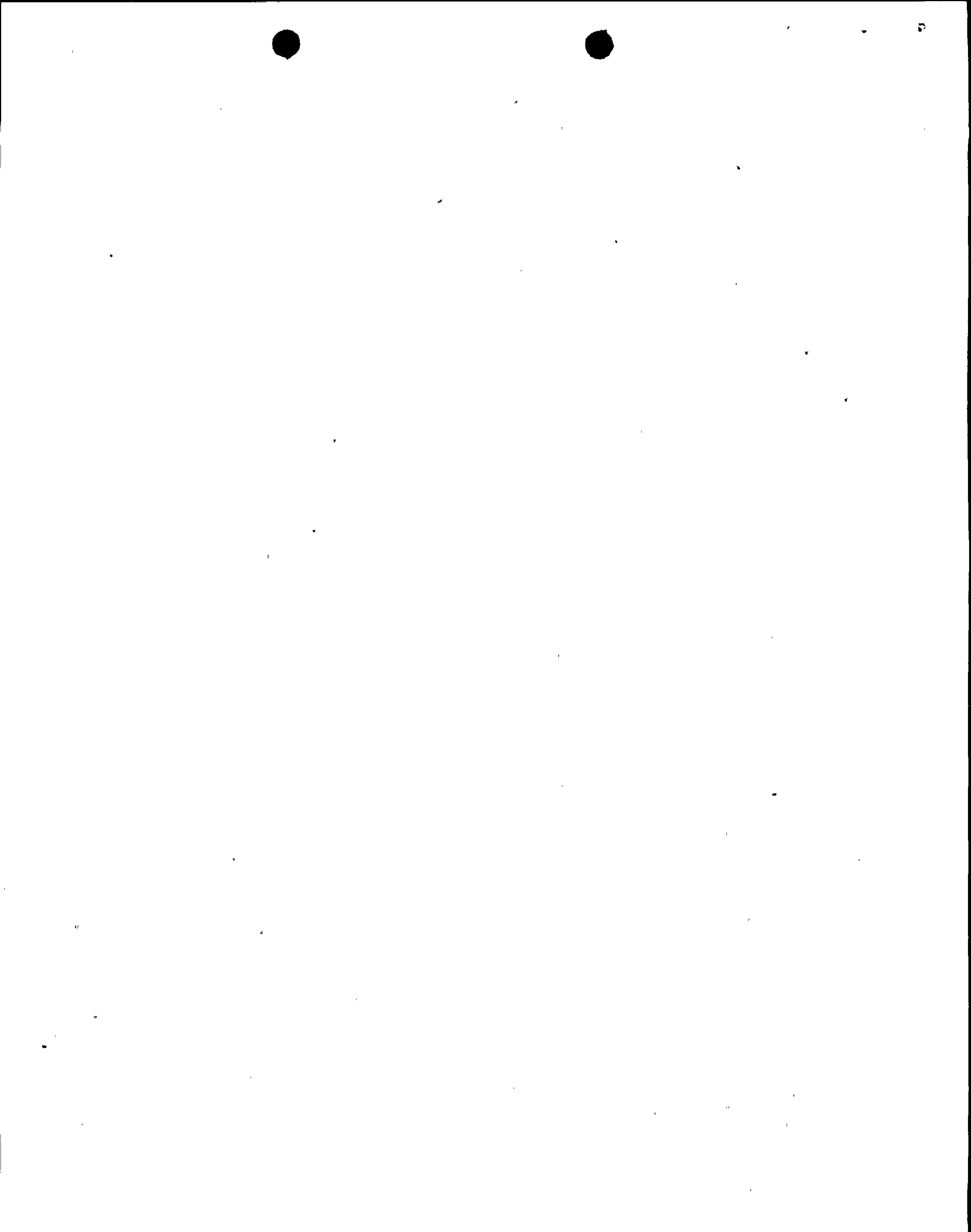
As indicated in our initial response, Diablo Canyon auxiliary feedwater system operation is initiated automatically; therefore, no action is required on this item.

ITEM 6

A discussion of direct position indication for PORVs and safety valves is contained in Section 2.1.3.a of "NUREG-0578 Response." As discussed there, in addition to direct position indication, valve position may also be inferred from temperature sensors in the pipe downstream from the valves and from temperature, pressure, and level sensors in the pressurizer relief tank. These indications are displayed on the main control board and are alarmed on the main annunciator.

ITEM 7

7a. Operators are made aware in their training that the automatic actions of engineering safety features are not to be overridden unless continued operation would result in hazardous plant conditions. Plant Emergency Operating Procedures OP-0 (Reactor trip with Safety Injection), OP-1 (Loss of Coolant Accident), OP-2 (Loss of Secondary Coolant) and OP-3A (Steam Generator Tube Rupture) contain specific criteria which must be met before automatic safety actions can be terminated. The criteria contained in these procedures conform to the criteria developed by the Westinghouse Owner's Group in its generic procedures.



- 7b. As indicated in our response to Sub-item 7a, specific criteria are identified for termination of automatic safety actions, including high pressure injection. The specific criteria which govern HPI termination differ from those listed in Sub-item 7b, but do conform to the criteria developed by the Westinghouse Owners' Group which have been approved by the NRC.
- 7c. It is our understanding that Bulletin Sub-item 7c is no longer effective. Conditions under which reactor coolant pumps should be tripped or allowed to remain in operation are identified in the procedure referenced in Sub-item 7a. These conditions are in agreement with the recommendations of the Westinghouse Owners' Group.
- 7d. Existing procedures and training emphasize that operators should not rely on a single parameter for evaluating plant conditions. The revised procedures referenced in Sub-item 7a require checking of several parameters during various types of emergency conditions.

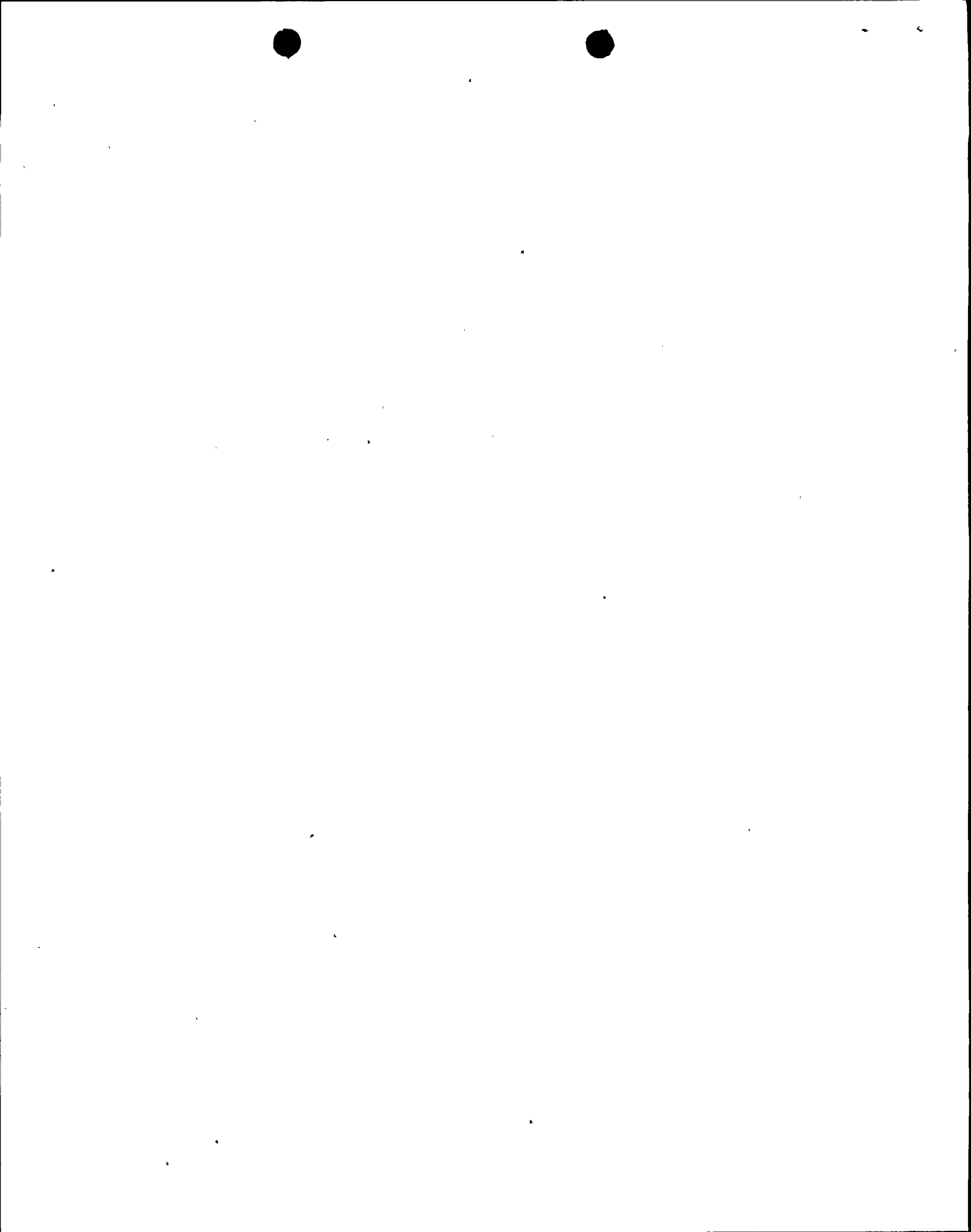
ITEM 8

We have reviewed the procedures covering the positioning of safety-related valves and have found the procedures adequate. They include the following features:

- a. Critical manual valves are sealed in position and a check list is maintained for inspection on a typical audit basis.
- b. When the Engineered Safety Features System operates, the misalignment of any remotely-operated critical valve in the system will be shown by a monitor light on the main control board.
- c. All safety-related valves which are operated remotely and whose purpose is to open or close (rather than throttle flow) have position indicating lights on the main control board. Valves with power removed from their motor operators during normal operation have continuously energized position indicating lights on the main control board which are redundant to those in (b.) above.
- d. All surveillance test procedures include checklists for returning the system to normal.
- e. A surveillance test is required after all maintenance to show that the valves work.
- f. Quality Control procedures require that any safety-related operations performed on one shift be verified on the following shift.

ITEM 9

A discussion of containment isolation appears in Section 2.1.4 of "NUREG-0578 Response." This discussion addresses task force positions and clarifications on containment isolation, and includes all Item 9 considerations.



ITEM 10

We have reviewed our maintenance and test procedures to assure that, before a safety-related component is removed from service, the redundant system is operable. This will be accomplished through checklists in the test procedures and in the clearance request procedures.

The procedures require the Shift Supervisor to give written authorization before equipment is removed from service and to acknowledge in writing its return to service.

Tests to show that equipment works properly after maintenance are also required (see answer to Item 8).

Shift and relief turnover procedures are discussed in Procedure NPAP A-101 ("Relieving the Watch"), which is appended to Section 2.2.1.c of "NUREG-0578 Response."

ITEM 11

We are currently in the process of revising our Administrative and Emergency procedures to provide for the notification requirements contained in Bulletin Item 11.

In the interim, it should be noted that there are currently two resident NRC inspectors at the Diablo Canyon plant, and that the inspectors have a direct communication link with NRC Region V headquarters.

ITEM 12

Hydrogen can be removed from the reactor coolant system by the following means:

- a. Hydrogen can be stripped from the reactor coolant to the pressurizer vapor space by pressurizer spray operation if the reactor coolant pump is operating.
- b. Hydrogen in the pressurizer vapor space can be vented by power-operated relief valves to the pressurizer relief tank.
- c. Hydrogen can be removed from the reactor coolant system by the letdown line and stripped in the volume control tank where it enters the waste gas system.
- d. In the event of a LOCA, some hydrogen would be vented with the steam to the containment.

The principal means of dealing with hydrogen in the primary system continues to be the prevention of hydrogen generation by the many design features and operating limits which limit the operating pressures and temperatures in the system. We have reviewed our operating procedures and training to be certain that hydrogen in the primary system is carefully considered.



In addition to the means enumerated above for removing hydrogen from the RCS, we are planning the installation of a reactor coolant system venting system, as discussed in detail in Section 2.1.9 of "NUREG-0578 Response."

We have reviewed our systems and procedures related to hydrogen in the containment building. These are described and evaluated in Chapters 6 and 15 of the Diablo Canyon Final Safety Analysis Report. Our conclusion is that the Three Mile Island accident has not shown that additional containment hydrogen control systems are required at Diablo Canyon. The information from Three Mile Island does not show a long-term rate of hydrogen production and accumulation in the containment in excess of the amounts for which our containment control system was designed. This is true even though the estimates of clad reaction at Three Mile Island significantly exceed those upon which the Diablo Canyon design criteria were based. This supports the expected case calculations of long-term hydrogen production in our Safety Analysis Report.

However, in order to provide additional control over the potential accumulation of hydrogen in containment, we have installed hydrogen recombiners inside the containment. In addition, although not required by NUREG-0578, dedicated penetrations are being provided for external recombiners. These items are discussed in Sections 2.1.5.a and 2.1.5.c of "NUREG-0578 Response."

ITEM 13

The technical Specifications for Diablo Canyon will be revised as necessary to reflect the information presented in the foregoing items.

