NUREG-0675
Supplement No. 21

Safety Evaluation Report
related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2

Docket Nos. 50-275 and 50-323

Pacific Gas and Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

Supplement 21 to the Safety Evaluation Report for Pacific Gas and Electric Company's applicantion for licenses to operate Diablo Canyon Nuclear Power Plants, Unit 1 and 2 (Docket Nos. 50-275 and 50-323), has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission.

This supplement reports on the status of the staffs resolution of outstanding allegations or concerns pertaining to Diablo Canyon as directed by the Commission on October 28, 1983. The status of a number of the allegations or concerns are considered sensitive and not addressed here since disclosure would impede possible enforcement actions or identify allegers that have requested anonymity. Consistant with the procedures of the Commission Policy Statement of August 5, 1983 regarding Investigations and Adjudicatory (48 Fed. Reg. 3658, August 10, 1983) the staff has determined that their assessment will be provided only to the Commission and the Boards for their <u>in camera</u> consideration. The collective assessment provided does however consider the significance of the <u>in camera</u> evaluation as they impact the licensing of Diablo Canyon.

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1 INTRODUCTION

The staff of the U. S. Nuclear Regulatory Commission (NRC) issued on October 16, 1974, its Safety Evaluation Report (SER) in matters of the application of the Pacific Gas & Electric Company (PG&E) to operate Diablo Canyon Nuclear Power Plant, Units 1 and 2. The SER has since been supplemented by Supplement Nos. 1 through 16 and No. 18 through No. 20 (Supplement 17 has not been issued). SER supplement No. 18 (SSER 18) presented the staff's safety evaluation on matters related to a verification effort for Diablo Canyon Unit 1 that was the result of Commission Order CLI-81-30 and an NRC letter to PG&E of November 19, 1981. SER Supplement No. 19 (SSER 19) and presented the staff's safety evaluation of those unresolved matters identified in SSER 18 which has to be satisfactorily resolved prior to commencement of fuel loading operations at Diablo Canyon Unit 1. SER Supplement No. 20 presented the staff's safety evaluation of those unresolved matters identified in SSER 19 which had to be satisfactorily resolved prior to commencing low power testing of Diablo Canyon Unit 1.

This supplement is based on allegations and concerns available to the staff as December 16, 1983. The NRC Project Manager for the Diablo Canyon Nuclear Power Plant is Mr. H. Schierling. Mr. Schierling may be contacted by calling (301) 492-7100 or by writing to the following address:

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APPENDIX E

STATUS OF STAFF RESOLUTION

OF

ALLEGATIONS OR CONCERNS

ABOUT

THE CONSTRUCTION

AND

OPERATION OF DIABLO CANYON'S

UNIT 1 AND 2

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1. INTRODUCTION

On October 16, 1974, the staff of the U. S. Nuclear Regulatory Commission (NRC staff, the staff) issued its Safety Evaluation Report (SER) in the matter of the application of the Pacific Gas and Electric Company (applicant) to operate the Diablo Canyon Nuclear Power Plants, Units 1 and 2. The SER was supplemented by supplements (SSER's) 1 through 16, 18, 19 and 20. SSER 17 is in preparation. This is SSER 21.

1.1 Purpose

During a staff briefing of the Nuclear Regulatory Commission concerning the Diablo Canyon Unit 1 readiness for fuel loading on October 28, 1983, the Commission, recognizing a significant number of allegations or concerns have been received, directed the staff to pursue the outstanding issues to resolution. Further, the staff was requested to provide a status report on these matters to the Commission prior to a decision on authorization of criticality and low power testing. This SSER is prepared to serve as the report concerning the staff status in resolving the allegations and concerns.

1.2 Diablo Canyon Allegation Management Program (DCAMP)

In order to fulfill the Commission directive, the Executive Director for Operations instituted a Diablo Canyon Allegation Management Program (DCAMP). The program was specifically requested to recognize that resolution of these matters involves many of the Operations offices and to provide the quality of review consistent with the importance of these matters.

The DCAMP was given the following objectives:

(1) Conduct a systematic examination and analysis of allegations and expressions of concerns pertaining to design, construction, operation and management of safety-related structures, systems, and components at the Diablo Canyon Nuclear Power Plant.

(2) Provide for an assessment of safety significance of those allegations and concerns that question Diablo Canyon criticality readiness, prior to a Commission consideration of restoration of the license for reactor criticality and low power (less than 5% of rated power) testing; and

(3) Provide for an assessment of those allegations and concerns that question plant readiness for power ascension testing and full power operation, prior to a Commission consideration of this issue.

1.3 Scope

The Diablo Canyon Allegations Management Program (DCAMP) encompasses all allegations or expressions of concern which may be construed as allegations, which pertain to design, construction, operation, and management of safety-related structures, systems and components at Diablo Canyon. In this regard the DCAMP has also addressed certain concerns raised by the public, media, and provided by members of Congress. The program requires that all NRC Offices receiving new Diablo Canyon allegations forward them to the DCAMP staff in a timely manner.

The DCAMP maintains as one of its tenets that the allegers desire for confidentiality or anonymity will be protected by all means available. As a result of this requirement it is necessary for some allegations and concerns addressed to be provided in a separate, limited distribution document. The assessment in this report, however, does include consideration of such items.

This status report and the separate limited distribution document addresses approximately 100 items classified as allegations or concerns evaluated by the staff. They represent those received through December 19, 1983. Any new allegations received after this date will be reviewed and a status provided the Commission prior to further Commission consideration of plant licensing.

2. APPROACH

2.1 Diablo Canyon Allegation Management Staff

The responsibility for implementing the allegation and concern management plan was assigned to Mr. John B. Martin, Administrator for Region V. Mr. Thomas W. Bishop, Director, Division of Resident, Reactor Projects and Engineering Programs was assigned management responsibility with staff support from various Regional offices, OIE and ONRR. All NRC staff support necessary to resolve these allegations or concerns in a timely fashion have been made available.

2.2 Methodology

2.2.1 Confirmation of Allegation

As each allegation or concern was received every effort was made to contact the alleger to confirm our understanding of the matter. In many cases confirmation was through a sponsor due to the alleger's desire for anonymity. In some cases meetings were held with the alleger to confirm our understanding of the allegation. Where requested the alleger's identity has been withheld from public disclosure. In those cases where the alleger is unknown, the staff has made an effort to be reasonably broad in understanding the general deficiency or concern provided by the alleger.

2.2.2. Site Inspections

Many of the allegations required onsite inspections to verify construction practices, records, procedures and personnel qualification. These were handled by teams of staff personnel with appropriate consultants. Independent measurements and evaluations were performed where appropriate.

2.2.3. Technical Reviews

The technical reviews were accomplished by detailed evaluations using licensing documents, regulations, standards, additional information provided by the licensee, and independent analyses as necessary. In some cases audits were performed on site or in the offices of the licensee and his contractors as necessary.

2.2.4. Interviews

Interviews with site personnel (crafts, quality assurance personnel, engineers and management) were carried out as required to resolve the issues.

2.2.5. Public Meetings

Where significant technical meetings were held, verbatim transcripts were taken to maintain an appropriate record. These meetings were announced and open to the public.

2.2.6. Feedback to Allegers

Where practical, the staff attempted to discuss with the alleger the approach and findings of the staff's evaluation related to their allegation. The purpose here was to assure the staff thoroughly understood the concern and to demonstrate how the staff dealt with the concerns.

2.2.7 Allegation Management Instruction

Region V's draft instruction on allegation management was used as guidance for this process. The draft instruction (entitled "Management of Allegations") is provided as Attachment 4.

2.2.8. Status Report

The staff was to prepare the required status report for the Commission. The report was to consist of an SSER presenting the results of the allegation and concerns evaluation. The specific evaluation of those allegations where the requestor has asked for anonymity was provided the Commission through a separate limited distribution document, as necessary, to assure anonymity.

3. SUMMARY

3.1 Cataloging of Allegations or Concerns

The allegations and concerns addressed in this document were received by the staff through a variety of sources (including private citizens, former and current plant workers, media representatives, intervenors, and Congressional offices), and cover a broad spectrum of work activities and time periods. Attachment 1 provides a comprehensive listing of all allegations or concerns which were open during the period of November 18 through December 19, 1983. The allegations or concerns are addressed in four collective groupings below. These grouping are: Design, Construction, Project Management, and Other allegations. Individual assessment summaries are provided in Attachment 2. A table grouping the allegations is provided in Attachment 3. In some cases the Individual Assessment Summaries contain sensitive information or are predecisional in nature, in that their disclosure could impair the staff's ability to initiate and/or conduct appropriate inspections or investigations. These summaries have not been provided in Attachment 2, but have been provided to the Commission separately consistent with the Commission's August 5, 1983, Statement of Policy on Investigations and Adjudicatory Proceedings (48 Fed. Reg. 36358).

3.2 Summary of Individual Items in Grouped Format

3.2.1 Summary of Design Items

All allegations and concerns addressing design issues, or regulatory safety requirements normally reviewed in the licensing process were evaluated using licensing documents, NRC and industry standards, examination of licensee documentation, site audits, and meetings with allegers or their sponsors.

There were 48 allegations or concerns in this area. These issues were subdivided into two major categories of Design Control and Design Adequacy. These are summarized below together with the staff's position as to when the are to be resolved.

3.2.1.1 The Design Control area encompassed 15 allegations or concerns. The status of these items is as follows:

(a) Seven of the items are resolved with a conclusion that there is no safety concern associated with the issues, and therefore there is no impact on decisions regarding low power testing or full power operation. These seven are identified as Nos. 6, 6a, 30, 31, 44, 92, and 93 in Attachments 1 and 2.

(b) Five of the items relate to controls applied to the small bore piping and pipe support design process. This area is complex and the staff requires further information before an accurate assessment of the significance of these issues can be performed. These issues (Nos. 79, 82, 87, 88, and 97) are being examined in conjunction with related concerns of design adequacy, identified below, in paragraph 3.2.1.2.(d). It is the staff's position that this must be resolved prior to reactor criticality.

(c) Two of the items (Nos. 34 and 41) relate to plant drawings. While the staff has concluded that both the drawing quality and the general as-built drawing program were adequate the staff feels there should be additional verification of the accuracy and availability of as-built drawing to the plant operating. This action will be completed prior to exceeding 5% power, consistent with its safety significance.

(d) One item (No. 96), anchor bolt spacing, is related to two other issues discussed in paragraph 3.2.2.1, which also involves anchor bolts and will be considered in conjunction with the resolution of those two issues. It is the staff's position that this must be resolved prior to reactor criticality.

3.2.1.2 The Design Adequacy area encompassed 33 allegations or concerns. These were further subdivided as Seismic Adequacy (12), Systems Interaction (8), RHR Design Adequacy (6), Piping and Support Analysis (6) and Single Failure Criteria (1). Those items in Seismic Adequacy have been considered in light of the regulations appropriate to the licensing of Diablo Canyon, the Hosgri modifications and the Independent Verification required of PG&E.

(a) Seismic Adequacy Involved 12 Items.

(1) Under Seismic Adequacy (10) items (3, 10, 11, 13, 14, 17, 28, 29, 32, 33 and 35) were found not to involve significant safety or management problems and presently meet NRC safety criteria. In nearly all cases, original or new calculation data and description provided by the licensee to the staff or the Independent Design Verification Program was used to resolve the allegation or concern.

(2) Item 8 concerns seismic classification of the Diesel Generator intake and exhaust. The licensee demonstrated that these systems are qualified to the original Hosgri Spectra and current Hosgri Spectra where appropriate. However, modifications are necessary to braces and pipe supports. This work is underway and will be completed prior to exceeding 5% power.

(b) Seismic interaction involved eight items.

(1) Five of the items (7, 9, 15, 16 and 75) were found to offer no safety problems affecting licensing for low power test or full power operation.

(2) Item 36 considered the adequacy of control room fluorescent light fixtures under a seismic event. The item was reviewed and the staff concluded that it is satisfactorily resolved subject to completion of the safety and non-safety system interaction program which is required prior to full power operation.

(3) The staff review of allegation No.48, concerning completion of the Seismic Systems Interaction Study prior to fuel loading and operation resulted in a determination that the modifications required by that study would be required prior to full power operation but not low power testing.

(c) RHR design adequacy involved seven items;

(1) Five of the items (Nos. 37, 38, 39, 40, and 45) were found by the staff to have no safety significance and will not affect low power or full power operation.

(2) Item 5, concerning CCW (Component Cooling Water) heat removal capacity was found to require a full power technical specification as requested by the licensee requiring that the redundant CCW heat exchanger be aligned whenever the ocean water temperature exceeds 64°F. This will be included in the full power technical specifications to be issued with the full power licensing.

(3) One item (No. 42) concerns spurious closure of motor operated RHR pump suction valves. This involves a generic design issue. This will be resolved prior to exceeding 5% power.

(d) In the area of Piping and Support Analysis there are (6) items (55, 78, 85, 86, 89 and 95). All of these are associated with the small bore piping design and are being examined in conjunction with the related design control concerns in this area (items Nos. 79, 82, 87, 88, and 97). A collective assessment of adequacy cannot be made at this time and that more information is required. The collection and evaluation of additional information is in progress. At this time it is estimated that a staff assessment will be completed by January 18, 1984. This date is conditional upon subsequent review findings and responsiveness of the licensee. In any event this topic requires resolution prior to reactor criticality.

(e) The final Design Adequacy item is item (4) concerned with single failure capability of the CCW system. The staff review verified that the postulated event (Loss of Coolant Accident) with a concurrent single failure (not closing non-essential loop isolation valve) does not result in a heat load in excess of the design heat removal capability of the CCWs heat exchangers and therefore the concern is satisfactorily resolved.

3.2.2 SUMMARY OF CONSTRUCTION ITEMS

3.2.2.1 H. P. Foley Construction Activities

At the Diablo Canyon site, the H. P. Foley Company was primarily responsible for electrical system installation activities, including such actions as electrical cable tray and conduit installation, cable pulling, electrical cable terminations, and electrical equipment installations. This spanned the time period of 1971 to the present. Following the suspension of the Diablo Canyon license in November 1981, and the initiation of modifications, the H. P. Foley Company was tasked with responsibilities in other areas, such as implementing structural steel modifications.

Twenty of the allegations or concerns identified in Attachment 1 involve the H. P. Foley Company. Two of these, however, are only indirectly connected with Foley and were not considered in assessing the collective significance of concerns in the Foley area of responsibility. These two allegations are No. 27 (welding and QC concerns in vendor supplied Super-Strut cable tray support materials), and No. 18 (a sensitive issue which is currently the subject of review by the NRC Office of Investigation and is addressed separately).

Investigation or inspection has been initiated on all but one of the issues or concerns. In summary, the status is as follows:

(a) Ten items are resolved (Nos. 27, 54, 59, 60, 61a, 62, 63, 64, 65, and 66).

(b) Ten items require further actions as described below (Nos. 18, 24/26/46, 25, 57, 58, 61, 96, 101, and 102):

(1) Four of the items related to reporting of nonconformances and voiding of reports (nos. 24, 26, 46, and 66). The staff

concluded that, in general, nonconformance reporting and documentation is properly handled. However, in the course of this examination the staff identified three items which are candidates for enforcement action. These three items are not directly related to nonconformance reporting, but are included here because they were identified while reviewing this area. These items are:

(1) Three loose structural steel bolts
(2) Failure to post the required 10 CFR 21 form
(3) Use of an inappropriate weld procedure for welded studs.

These three items will be pursued through the NRCs routine inspection and enforcement program.

Item no. 66 was found not to involve a significant safety or management problem.

(2) Three of the items related to document control (nos. 61, 61a, and 102). It is the staff's opinion that two of these items (no. 61 and 102) require further examination to enable an accurate assessment. The licensee has been requested to provide additional information in this area. At this printing the staff estimates that an assessment can be completed by January 18, 1984. This date is conditioned upon subsequent review findings and responsiveness of the licensee. The staff recommends that an assessment be completed prior to reactor criticality. Item no. 61a was found not to involve a significant safety or management problem.

(3) Three of the items related to anchor bolting (nos. 25, 58, 96). It is the staffs opinion that this subject requires further examination to enable an accurate assessment. The licensee has been requested to provide additional information regarding installation practices and guidance. At this time the staff estimates that an assessment can be completed by January 18, 1984. This date is conditioned upon subsequent review findings and responsiveness of the licensee. The staff recommends that an assessment be completed prior to reactor criticality.

(4) Three items (nos. 18, 57, and 101) involved the certifications and qualifications of inspectors and crafts. One of these items (no. 57) identified several instances where inspections were performed by individuals prior to their certification. It is the staff's opinion that this area requires further examination to enable an accurate assessment. The licensee has been requested to take additional actions in this area. At this printing, the staff estimates that an assessment can be completed by January 18, 1984. This date is conditioned upon subsequent review findings and responsiveness of the licensee. The staff recommends that an assessment be completed prior to reactor criticality. Item 18 is the subject of an inquiry by the NRC Office of Investigation, however, it does not appear that this issue involves any significant safety issue or substantial breakdown of management or quality systems. Item no 101 was received late (December 8, 1983) in the evaluation period and has not been assessed by the staff. This concern is being evaluated in a timely manner.

In addition to the potential enforcement items, discussed above several other areas of Foley activity warrent followup action by the licensee. However, with the exception of the anchor bolting, drawing control issues, and certification of inspection personnel, none of these findings, either singularly or collectively, are of such a magnitude or predominance as to present any question regarding a significant safety issue or substantial breakdown of management or quality systems. This conclusion is based upon the significant sample of items inspected by the staff, the lack of significant equipment problems associated with the items, and the lack of substantial significance associated with the records deficiencies when considered collectively. This is not to say that these issues do not warrent thorough follow up by the licensee and appropriate monitoring by the staff. This will be accomplished through our routine program.

3.2.2.2 Pullman Construction Activities

Pullman is the primary mechanical equipment and piping installation contractor at Diablo Canyon, having responsibility for nearly all piping, pipe supports, and mechanical equipment exclusive of the Nuclear Steam Supply System.

Eight of the allegations or concerns identified in Attachment 1 involve the Pullman Company. Investigation or inspection has been initiated on all but one of the eight issues or concerns.

In most (6) of the cases the staff found the allegations or concerns to have some degree of substantiation. In evaluating the allegations or concerns in the Pullman area one item was identified which is a candidate for possible enforcement action (concern regarding certification of inspectors, identified during the examination of the NSC audit findings, item No. 68). It is the staff's opinion that additional information is required to assess the adequacy of this area, and that this should be done prior to reactor criticality. At this printing the staff estimates that an assessment can be completed by January 18, 1984. This date is conditioned upon subsequent review findings and responsiveness of the licensee. In addition to this item licensee follow-up action is required in one area (assessment of a nonsafety-related U-bolt installation, item No. 76).

With the exception of the inspector certification issue none of the specific allegations themselves or related areas inspected by the staff identified any question regarding a significant safety issue or substantial breakdown of management or quality systems.

The one Pullman concern (No. 103) which was not examined was received late in the evaluation period (December 14, 1983) and relates to welding activities. This concern is being evaluated in a timely manner.

3.2.3 SUMMARY OF PG&E PROGRAM MANAGEMENT ITEMS

Twelve of the allegations or concerns have been categorized as topics falling within the subject area of licensee program management. The 12 concerns have been evaluated in three groups: four items relating to management responsiveness to identified issues; three items relating to reporting of conditions to the NRC staff; and five items pertaining to quality assurance.

Investigation or inspection has been initiated on all but one of the issues. Inspection of those areas reviewed included an examination all pertinent documentation associated with the issues (nonconformance reports, design change documents, manuals, audit reports, contract specifications, letters, memoranda, and logs). In addition, where possible, personnel associated with the issues of concern were interviewed, and, as appropriate, physical inspection of components performed.

In many (6) of the cases the staff found the allegations or concerns have some degree of substantiation. However, with the exception of Item 100, discussed below, none of the specific allegations or concerns themselves, or related areas inspected by the staff, identified any question regarding a significant safety issue or substantial breakdown of management or quality systems.

In evaluating the allegations or concerns in this area one item was identified as a potential safety concern (concern with the coating/painting, item no. 100). It is the staff's opinion that further information is required to fully address the significance of this item. In addition to this item licensee followup action is in progress in two areas (responding to EDS/PAC audits, item No. 72; and, installation of a public address system, item No. 47). These items will be monitored by the staffs routine programs.

The one item (No. 99) which was not examined was received late (December 10, 1983) in the evaluation period and relates to the quality of vendor supplied structural steel (Bostrom-Bergen/Medco). This concern is being evaluated in a timely manner.

3.2.4 SUMMARY OF OTHER ALLEGATIONS OR CONCERNS

Seventeen of the allegations or concerns fell into areas not included in the topic areas discussed previously. These 17 can be subdivided into: health physics concerns (3); security concerns (6); emergency preparedness concerns (2); protection of allegers (2); and pipe pitting concern (1); concrete defect concern (1); a NRC effectiveness concern (1); and a concern about the authorization to load fuel while hearing action and construction activities are still in progress.

Investigation or inspection activity has been initiated on each of the concerns. Followup action is required and is in progress for some of the areas involved (health physics, security, and pipe pitting). However, none of these topics, either individually, or collectively are of such magnitude or significance as to indicate a substantial safety issue.

3.3 INTERVIEWS WITH SITE PERSONNEL

As an integral part of the team inspection conducted at the Diablo Canyon site from November 28, 1983 through December 9, 1983, the team members interviewed over 158 persons employed at the site. The interviews were informal, private and structured to determine if the individual:

- had experienced, or knew of any, improper management pressures to cut corners
- had been intimidated, or knew of any cases of intimidation
- had any concerns about the quality or safety of the plant

Members of the NRC inspection team chose the interviewees at random within the following guidelines. during the inspection. The interviewees represented all of the major organizations conducting work at the site and all the major disciplines/activities being conducted on site. Particular emphasis was placed on those disciplines/activities which were identified in allegations of problems at the site. This was done to attempt to identify specific examples of unacceptable conditions and to determine the perceptions of those persons closest to the actual activity.

The interviews did not identify any direct evidence of cutting corners or harassment/intimidation adverse to quality. Ten of the 158 individuals responded that they had heard rumors or sensed some management pressures to get the job done, however, none of the individuals indicated that these pressures had resulted in system deficiencies. Review of the ten individuals concerns indicated that:

(a) Most of the concerns (8) related to pressures to "get the job done," as opposed to circumventing quality programs.

(b) One concern related to "pressures" to void nonconformance reports. This topic was extensively examined (Item Nos. 24, 26, 46 and 66) by the staff. Inspection results indicated that the nonconformance reporting and voiding is in accordance with requirements.

(c) One of the items related to a concern that Pullman production has too much influence over the quality organization. Pullman activities were extensively examined during the evaluation of issue No. 68 (NSC audit of Pullman). This review did not provide any indication that production forces exercised control over the quality organization.

During one of the interviews an employee mentioned that quality class I procedures were not being required for the coating (painting) work. This item has been treated as a separate allegation or concern (No. 100). As indicated above, no direct evidence was offered by the interviewees concerning experiencing or knowing of any corner cutting, intimidation or harassment, nor did they have any concerns related to safety items. In general, responses indicated that management was responsive to concerns, accessible, quality oriented, had an open door policy and supportive of employee's concerns. The information obtained in the interviews has been and is being used to follow up on the specific technical allegations. Additional interviews and inspections will be performed, as necessary, to assure adequate evaluation of comments received.

3.4 ALLEGATION STATUS

In quantitative terms the majority of allegations or concerns have been fully addressed and require no further specific technical analysis, investigation, or inspection (although final staff report is required for a number of these items). Of the 103 allegations or concerns 58 fall into this category.

A number of the allegations or concerns which were examined require further action by the licensee and/or the staff; forty-five (45) of the allegations or concerns fall into this category. The majority of these items are being appropriately handled by the licensee's or staff's standard programs and are not of such significance that raise questions of the safety of reactor criticality or power operation. It is the staff's opinion, however, that certain actions should be performed prior to achieving reactor criticality or exceeding five percent power. These actions are of two types: first, areas where technical evaluations have been completed and specific actions are considered by staff to be required prior to these events; and second, areas where technical evaluations are incomplete and preliminary evaluations indicate there is a potential for a safety issue, necessitating action to provide a more comprehensive staff understanding of the issues involved before criticality or power operation. These actions are summarized below:

3.4.1 Actions Required prior to Criticality

3.4.1.1 Small bore piping design adequacy.

As discussed in paragraph 3.2.1 above, there are a number of allegations or concerns which have lead the staff to seek more information about the adequacy of small bore piping and pipe support design. A preliminary assessment of adequacy cannot be made at this time. The collection and evaluation of additional information is in progress.

3.4.1.2 Anchor bolt design margins and installation

As discussed in paragraph 3.2.1 and 3.2.2 above there are three allegations or concerns which have lead the staff to seek more information about the adequacy of anchor bolt design margins and installation. Concern for design margins was not a specific allegation but was encountered while reviewing concerns related to items Nos. 25, 58, and 96. A preliminary assessment of adequacy cannot be made at this time. The collection and evaluation of additional information is in progress.

3.4.1.3 Control and issuance of design change notices and related drawings.

As discussed in paragraphs 3.2.1 above there are two allegations or concerns (No. 61, and 102) which lead the staff to seek more information regarding this subject. It is the staff's opinion that a preliminary assessment of adequacy cannot be made at this time. The collection and evaluation of additional information is in progress. A preliminary assessment of adequacy cannot be made at this time. The collection and evaluation of additional information is in progress.

3.4.1.3 Inspector Certifications

As discussed in paragraph 3.2.2 above, inspection of allegations or concerns Nos. 57 and 68 identified several instances were inspections were performed by individuals not certified at the time of the inspection.

At this printing it is the staff's estimate that preliminary assessments regarding the above topics will be completed by January 18, 1984. This date is conditioned upon subsequent review findings and responsiveness of the licensee.

3.4.2 <u>Actions Required Prior to Exceeding Five Percent Power</u>

It is the staff's position that the following actions be completed prior to exceeding 5% power:

3.4.2.1 Implementation of a technical specification limit on the operation of the Component Cooling Water System whenever ocean water temperature exceeds 64° F.

This item was a result of staff examination into allegation or concern No. 5 and is addressed in detail in the Diablo Canyon Safety Evaluation Report, NUREG-0675, Supplement 16.

3.4.2.2 Completion of seismic modifications to the diesel generator silencer bracing and pipe supports.

This item was identified in conjection with the staff's examination into allegation or concern No. 8.

3.4.2.3 Completion of the inspection and verification of the as-built drawings located in the control room. This item was identified in conjunction with staff evaluation of allegation or concern No. 34

3.4.2.4 Complete modification resulting from the seismic systems interaction study, in progress, in accordance with commitments identified in SSER 11. This item was identified in conjunction with staff evaluation of allegation or concern No. 48.

3.4.2.5 Complete the analyses of significance of coating (painting) concerns discussed in concern item No. 100.

As indicated previously, there were a few allegations or concerns which were received late in the evaluation period and/or sufficient time was not available to effectively evaluated prior to the issuance of this SSER. Five allegations or concerns fall into this category, and are listed below:

Item No.	Subject
88	Undocumented modifications to small bore pipe supports
95	Angle members in small bore pipe supports
99	Falsification of Vendor Records (Bostrom Bergen/Medco).
101	Welding Qualifications (Foley Company)
103	Welding Qualifications (Pullman Company)

All of the above allegations or concerns have been entered into the established NRC tracking systems and are scheduled for investigation or inspection in a timely manner. The staff will provide the Commission an updated written status at six week intervals and will be prepared to provide an oral status report at any time.

3.5 CONCLUSION AND RECOMMENDATIONS

The allegation management program in place for current and future allegations related to Diablo Canyon has and should continue to provide a procedure for orderly and thorough yet timely examination of each concern raised.

Approximately 75% of the allegations currently received have been examined to a point where it is the staff's opinion that there is no significant safety issue or substantial breakdown of management or quality systems. The remaining allegations have been assigned to various elements of the NRC staff for evaluation and most have been partially examined. Examinations of these remaining allegations in sufficient detail to permit a staff conclusion relative to safety significance is expected to be completed during site inspections scheduled for January 4, 1984 through January 13, 1984. Approximately 15 professional staff will be active in these inspections.

The staff has not, at this time, identified any issue that would preclude authorization for operation up to and including testing at five percent power on the basis of public health and safety; particularly in light of the negligible fission product inventory that would be built up by such operation. There is good reason to extend the present limits of authorized activities at Diablo Canyon to allow sub critical operation at full system temperature and pressure. Evaluation of piping and pipe supports under full thermal loading, not allowed by present operating restrictions, will provide additional confidence in the evaluation of piping and pipe support design. There are, however, several areas where our examination of allegations has led us to require additional information. As a matter of prudence pending further evaluation of these matters, the staff has identified in Section 3.4 of SSER No. 21 four actions that we presently believe should be completed prior to authorizing criticality; five other actions have been identified for completion prior to authorizing operation above five percent power.

Recommendation

It is the staff's recommendation that the licensee be authorized to proceed to Modes 4 and 3 pending completion of staff assessments related to the areas of small bore piping design, anchor bolts, inspector certification, and design change notice/drawing control.

LIST OF ALLEGATIONS

Allegation

1. Passing of contraband
2. Anti-Nuclear Demonstration
3. Seismic qualification CCW
4. Single Failure Capability CCE
5. Heat removal capability CCW
6. I&C Design Classification

6a. Feedwater Isolation Classification

7. Seismic Category I/Category II Interface
8. Seismic Design of Diesel Gen. I & Exh.
9. USI-17 Systems Interaction Generic
10. Seismic Tilting of Containment
11. Classification of Platform (Category I/Category II)
12. HELBA did not meet FSAR, RG 1.46
13. Inadequate seismic systems
14. Loads on Annulus Structural Steel not calculated properly
15. Inadequate Tornado Load Analysis of Turbine Building
16. High energy pipe break restraint inadequate
17. NSSS SSE Load Inadequate
18. QA/QC Allegations
19. Guard Qualification
20. Health Physics personnel do not meet ANSI requirements
21. ALARA Program - Paper Tiger
22. Radiation Monitors lack sensitivity
23. QC Inspector Concerns
24. HPFoley NCR's rejected without good cause.
25. Deficiency in use of "Red Head" anchors for raceway support
26. Foley didn't document NCR's issued by field inspectors
27. Welding and QA deficiency in "Super Strut"
28. Annulus Structure Reverification
29. Pipe restraints design inadequate
30. Inadequate Documentation of Safety Related Equipment
31. QA Procedures for Struct. Analysis
32. Seismic analysis containment
33. Turbine Building (Class 2) Contains Class 1 systems & components
34. Incomplete as-built drwgs.
35. Lack of support calcs for fluorescent light fixtures

36. Resolution of fluorescent light fixture interaction
37. Solid state protection system relays
38. PG&E ignoring spurious closure of mot. valve
39. No control Room annunciation of closed RHR suction valve
40. RHR hot leg suction not single failure
41. Drwgs. inadequate
42. Licensee management unresponsive to problems
43. Licensee reporting failure
44. Licensee improp. assessment of Design Change Notice
45. Design inconsistency in FSAR RHR valves
46. HPFoley QA procedures voiding NCR's incorrect
47. Plant P.A. System
48. SI Study and associated Mods
49. Emergency Sirens not seismic qualified
50. Plant Security should have been retained
51. Risk of job action agaomst allegers
52. Construction & hrgs in progress after fuel load inappropriate
53. Welder Qualification
54. Wire traceability not evident for work by PG&E and Foley
55. Bechtel approved analysis of small bore pipe by altering failed analysis
56. Pitting of main steam and feedwater piping
57. Foley used uncertified and unqualified Q.C. inspectors prior to 1983
58. Foley allows "Red Head" anchors studs reported improprely installed
59. Foley lost cable traceability
60. Foley purchased material through unapproved vendors
61. Lack of document control
61a. H. P. Foley used unapproved drawing
62. Foley lacks adequate sampling of cable pull activities
63. Foley has lost material tracability through upgrade of non class 1 to class 1
64. Grout test sampling based on special tests rather than field tests
65. Foley documents prior to 1980 questioned No review required prior to 9/1981 license issuance date
66. Defective weld reports rejected by Foley
67. Negligence by PG&E flooding at 55 ft. elevation pipe tunnel
68. NSC Pullman-Kellog. audit
69. Revision of Draft Case Study "C"
70. Inadequate of response to NRC notice of Violation
71. Use and sale of drugs
72. Audits of PG&E (PAC/EDS)
73. Selling of drugs
74. Defective piping support
75. Discharge piping too close to accumulater
76. U-Bolts have failed
77. Flange bent on I-Beam
78. Bracket Bolted to wall with only bolt
79. Engineers are calculating stresses in piping in a variety of ways.
80. Concerns about the emergency response plan
81. Individual fired for whistle blowing

ATTACHMENT 1

DIABLO CANYON

LIST OF ALLEGATIONS OR CONCERNS

82. Minimal Orientation for New Engrs. at the site
83. NRC was not effective in identifying problems
84. Lack of responsiveness by management to identified problems relating to design
85. U-bolt design
86. "Code-break" design
87. Calc. related to "code break" design destroyed
88. Undocumented modifications were made because of code break problems
89. Interference of pipe supports (attempted use of uni-strut as a pipe support)
90. Defective concrete in intake structure
91. Alleged cover-up of defective material use
92. Flare bevel welds are undersized and do not comply with code dihedral angle
93. Inaccurate depiction of welds on drawings (symbolic)
94. Pullman used pipe welding procedures to make structural support welds
95. Angles of pipe support member are out of specification
96. Improper anchor bolt spacing ("Hilti" and "Red Head")
97. Site design engineers have been required to use uncontrolled documents resulting in different assumptions, etc.
98. Possible non-adherence of pentration seal procedure.
99. Falsfication of Welding Quality Control Records.
100. No quality control program for coatings
101. Qualification of welders and procedures
102. Improper references on DCN
103. Welding and welding program concerns

ATTACHMENT 2

INDIVIDUAL ALLEGATION

SUMMARY SHEETS

Task: Allegation or Concern No. 1

ATS No.: Q5-82-0004 BN No.:

Characterization

Passing contraband

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 2

ATS No.: Q5-82-006

BN No.:

Characterization

Anti-Nuclear demonstration

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 3

ATS No.: NRR-83-02 BN No.: 83-03(1/7/83)

Characterization

A concern was raised that the pressure boundary of the nonessential loop of the safety-related component cooling water system (CCWS) although not required to function following a safe shutdown earthquake (SSE) was not qualified for the SSE. This loop would therefore fail in an SSE resulting in loss of water and subsequent CCWS failure when a single active failure (to close) is assumed in the isolation valve to the nonessential loop.

Implied Significance to Plant Design, Construction or Operation

The potential loss of the CCWS as postulated in the concern would affect the ability to safely shutdown the plant following an SSE. Therefore, a reanalysis of the CCWS seismic qualification design and associated modifications to ensure its functional integrity following an SSE would be required.

Assessment of Safety Significance

The staff has evaluated the concern against the CCWS design information available in the FSAR and against information obtained in subsequent correspondence and meetings held with the licensee. The staff has verified that the CCWS including the pressure boundary of the nonessential loop is qualified to the SSE, and, therefore, no system failure of the type postulated in the concern should occur. That is, a postulated single active failure (to close) in the nonessential loop isolation valve will not result in an unacceptable condition in the CCWS because isolation of the nonessential loop following an SSE is not essential for ensuring the CCWS safety function. Refer to Diablo Canyon Safety Evaluation Report, NUREG-0675, Supplement No. 16 for further information.

Staff Position

The staff concludes that the CCWS design satisfies General Design Criteria 2, and 44 with respect to assuring its cooling water safety function following an SSE and concurrent single active failure. This concern has been satisfactorily resolved.

Action Required

None.

Task: Allegation or Concern No. 4

ATS No.: NRR 83-02 BN No.: 83-03(1/7/83)

Characterization

A concern was raised that a single failure (to close) in the isolation valve to the nonessential loop of the component cooling water system (CCWS) concurrent with a loss of coolant accident (LOCA) would result in an increase in the heat load on the CCW heat exchangers beyond their design heat removal capability because of failure to isolate nonessential heat loads.

Implied Significance to Plant Design, Construction or Operation

The potential inability of the CCWS to remove sufficient heat following a LOCA with a subsequently resulting higher than design allowable CCWS temperature could cause a failure of safety-related equipment to perform their function. Therefore, an evaluation of the consequences of this postulated occurrence with verification of satisfactory CCWS heat removal performance was required.

Assessment of Safety Significance

The staff has evaluated the concern against the CCWS design information available in the FSAR and against information obtained in subsequent correspondence and meetings held with the licensee. The staff has verified that the postulated event (LOCA with a concurrent single failure to close in the nonessential loop isolation valve) does not result in a heat load in excess of the design heat removal capability of the CCWS heat exchangers. Refer to Diablo Canyon Safety Evaluation Report, NUREG-0675, Supplement No. 16 for further information.

Staff Position

The staff concludes that the CCWS design satisfies General Design Criterion 44 with regard to assuring its cooling water safety function under the above assumed condition. This concern has been satisfactorily resolved.

Action Required

None

Task: Allegation or Concern No. 5

ATS No.: NRR 83-02 BN No.: 83-03(1/7/83)

Characterization

A concern was raised that with all redundant essential heat loads imposed on the component cooling water system (CCWS) following a loss of coolant accident (LOCA), the CCWS could not remove sufficient heat to maintain the design maximum CCWS temperature and assure a safe shutdown. This is because only one CCW heat exchanger is normally on line and operator action could not be taken soon enough to align the normally isolated redundant CCW heat exchanger prior to exceeding the allowable CCW temperature.

Implied Significance to Plant Design, Construction or Operation

The potential inability of the CCWS to remove sufficient heat following a LOCA with a subsequently resulting higher than design allowable CCWS temperature could cause a failure of safety-related equipment to perform their function. Therefore, an evaluation of the consequences of this postulated occurrence with verification of satisfactory CCWS heat removal performance was required.

Assessment of Safety Significance

The staff has evaluated the concern against the CCWS design information available in the FSAR and against information obtained in subsequent correspondence and meetings held with the licensee. The results of this review indicated that the originally assumed ultimate heat sink (Pacific Ocean) temperature of 70°F was too high for adequate heat removal following a LOCA with all essential equipment operable and one CCW heat exchanger on line assuming a concurrent single failure in an auxiliary salt water pump. Under this limiting condition from the standpoint of CCWS heat removal capability, a maximum ocean water temperature of 64°F must be assumed in order to assure that the design allowable CCWS temperature is not exceeded. The licensee has proposed a technical specification which requires that the redundant CCW heat exchanger be aligned whenever the ocean water temperature exceeds 64°F. Otherwise, the plant must be shutdown. The staff has accepted this technical specification and it has been incorporated in the Plant Technical Specifications. Refer to Diablo Canyon Safety Evaluation Report, NUREG-0675, Supplement No. 16 for further information.

Staff Position

The staff concludes that with incorporation of the above technical specification limit on CCWS operation that the CCWS design satisfies General Design Criterion 44 with regard to assuring its cooling water safety function under design basis accident conditions. This concern has been satisfactorily resolved.

Action Required

None

Task: Allegation #6

ATS No.: NRR 83-02 BN No.: 83-03 (1/7/83)

Characterization

Instrumentation and controls required to perform safety related functions do not conform to Seismic Category 1 requirements (e.g., component cooling water system surge tank level instrumentation).

Implied Significance to Plant Design, Construction, or Operation

In accordance with General Design Criterion 2 (Design Bases for Protection Against Natural Phenomena); the Diablo Canyon accident analyses assumes the proper functioning of instrumentation and controls used to mitigate the effects of accidents in conjunction with the effects of natural phenomena such as earthquakes. Instrumentation and controls relied upon to perform safety functions that are not seismically qualified cannot be assumed to function following a seismic event.

Assessment of Safety Significance

The component cooling water system (CCWS) at Diablo Canyon consists of three loops, A, B, and C. The CCWS is a Design Class 1 system except for non-vital components in loop C. An analysis has been performed by PG&E to demonstrate that the non-vital components will not fail as the result of a design basis seismic event (i.e., a safe shutdown earthquake; SSE). The CCWS surge tank is seismically qualified and is divided into two separate volumes to provide redundancy, thereby ensuring adequate cooling water to safety related loads following an accident. The surge tank level instrumentation is seismically qualified from a pressure boundary standpoint, however, it is not classified to function properly following a seismic event. The surge tank level instrumentation is used to automatically provide water from the Makeup Water System to the surge tank in the event of a low level. Since the CCWS is seismically qualified and therefore surge tank level is not expected to change during a seismic event, and since the surge tank level instrumentation is not used to perform a safety function, it need not be seismically qualified from an operational standpoint. Therefore, the staff concludes that this instrumentation is acceptable.

The licensee has stated and the Diablo Canyon SER notes that instrumentation and control components required to perform a safety function are designed to meet seismic Category 1 requirements. In accordance with the Standard Review Plan (SRP), the staff reviews the instrumentation and controls taken credit for by the accident analyses to assure they have been appropriately classified (i.e., as required to perform a safety function). Subsequent independent design reviews also verify the proper design classification of instrumentation and control components. Based on these reviews, the staff finds the instrumentation and controls at Diablo Canyon to be acceptable.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

Task: Allegation #6a

ATS No.: NRR-83-02 BN No.: 83-03 (1/7/83)

Characterization

Instrumentation and controls used to isolate main feedwater flow following a main steamline break are not safety related (i.e., do not conform to Class 1E and seismic requirements)..

Implied Significance to Plant Design, Construction, or Operation

The failure to isolate main feedwater flow following a main steamline break could result in an energy (steam) release to containment greater than assumed in the accident (FSAR Chapter 15); the concern here is overpressurization of the containment structure. Failure to isolate could also result in an additional (unwanted) cooldown of the reactor coolant system causing a reduction of core shutdown margin not considered in the accident analysis.

Assessment of Safety Significance

Isolation of main feedwater following a steamline break is mitigated by the Engineered Safety Features Actuation System (ESFAS). Isolation is accomplished by closing all main control valves, tripping the feedwater pumps, and closing the feedwater isolation valves. The feedwater isolation valves (also referred to as the backup feedwater isolation valves) are Category I containment isolation valves, i.e., they are designed to Class 1E requirements, including seismic qualification. The ESFAS is also designed to Class 1E requirements, including seismic qualification.

Physical separation is maintained between redundant ESFAS circuits, including field wiring. The tripping of the main feedwater pumps and closure of the feedwater control valves are redundant to closure of the safety Class 1 feedwater isolation valves and are not necessary for safety. The feedwater isolation valves, as well as all other containment isolation valves, were included in the PG&E Systems Interaction Program. The Diablo Canyon accident analysis for a main steamline break concludes that there is no consequential damage to the primary system or the core, and that there is no failure of the containment structure. The staff agrees with this assessment.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

Task: Allegation or Concern No. 7

ATS No: NRR-83-02 BN No: 83-03 (1/7/83)

Characterization:

PG&E appeared not to have a clear understanding of the scope of the targets and commitments to the NRC in the Systems Interaction Program.

Implied Significance to Plant Design, Construction, or Operation

If as alleged PG&E did not have a clear understanding of the scope of the targets and commitments to the NRC in the Seismically-Induced Systems Interaction Program (SISIP), then the misunderstanding might be significant to operation of equipment important to safety. At Diablo Canyon "Targets" refers to selected set of structures, systems and components that are important to safety and serve to either bring the plant to safe shutdown or maintain it in safe shutdown condition. A misunderstanding of the scope of the targets might affect the capability to safely shutdown the plant following the occurrence of a Hosgri event.

Assessment of Safety Significance

At the request of the Advisory Committee on Reactor Safeguards (ACRS) PG&E agreed to initiate a program to determine if seismically initiated failure of non-seismically qualified equipment and piping would cause interaction with safety-related systems which could prevent the plants from being safely shutdown following the occurrence of a Hosgri event.

PG&E, by letters dated May 7, July 1, July 15, August 19, and September 16, 1980, submitted drafts of their proposed program to the NRC staff for review and comment. The degree of PG&E's understanding including many details, e.g., target selection criteria, application of the target selection criteria, source identification criteria, application of source identification criteria, source-target interaction criteria, application of the source-target interaction criteria analysis for the resolution of postulated interactions, and the resolution of postulated interactions by plant modifications were contained in their draft program. These drafts were reviewed and comments submitted to PG&E as guidance for their use in improving their program. These reviews were described in Sections 2 through 5 of Supplement No. 11 to the Safety Evaluation Report (NUREG-0675, Supplement 11).

The staff performed an onsite audit of the program activities (reported in Sections 6 and 7 respectively of Supp. 11). Although the audit did not include a 100% review of PG&E's target list, it did include sufficient review to provide confidence that the list reflected the actual plant systems, components, structures and layout.

By letter dated October 13, 1983, PG&E submitted an information report on the status of their seismic systems interaction study within the containment of Unit 1. Included in the Information Report was the preliminary status of their study of Unit 2. PG&E has not yet completed its study of Unit 2 and the staff has not yet completed its review. However, the staff has not yet identified any misunderstanding of the original scope of the targets and commitments to the NRC in the PG&E program. In fact, there has been even more detailed understandings attained and more voluntary commitments made to the NRC. Therefore, the extent to which we have communicated with PG&E provides reasonable assurance that PG&E understands the scope of the targets and the commitments made by PG&E to the staff. The commitments are documented in Section 8.2, Supplement 11 to NUREG-0675 (SER):

(a) "PG&E will complete their program and any necessary plant modifications for each unit prior to the issuance of any license authorizing full-power operation of that unit."

(b) Region V, OIE, will verify "the completion of PG&E's program and the accetability of any plant modifications."

(c) "PG&E will ...provide for our information copies of their final report of their program which will include and identification of all interactions postulated, all walkdown data, interaction resolution, and technical reports."

Staff Position

Based upon (a) the degree of understanding between the staff and PG&E which includes many details documents in Supplement 11, NUREG-0675 and reinforced by extensive informal communication, and (b) the ongoing review of preliminary results, the staff has no basis to conclude that PG&E misunderstands the scope of the targets and their commitments to the NRC.

Action Required

No new action is required in response to this allegation. The ongoing review will continue to take steps to assure that no misunderstandings occur which might be significant to the safe operation of Diablo Canyon.

Task: Allegation #8

ATS No.: NRR 83-02 BN No.: BN 83-03 (1/7/83)

Characterization

Seismic Design of Diesel Generator Intake and Exhaust

Implied Significance to Plant Design, Construction or Operation

Availability of on-site power could be degraded and eventually interrupted and potentially hinder cold shutdown of reactor following a large earthquake event.

Assessment of Safety Significance

In response to a staff inquiry on an allegation concerning seismic design of emergency diesel generator intake and exhaust system, the licensee Pacific Gas and Electric Company (PG&E) provided additional information contained in a letter dated September 9, 1983 from J. O. Schulyer to D. G. Eisenhut. The staff has reviewed the additional information and in addition obtained further clarification through telephone conference on September 20, 1983. The approach in the staff review has been to determine the extent to which the diesel generator exhaust piping can maintain its integrity followinga large earthquake. Availability of on-site power following a large earthquake is important for maintaining reactor in a safe shutdown condition. The diesel generator intake air filter and air silencer are designed to withstand the safe shutdown earthquake. The concern with the integrity of the exhaust piping is that the operation and the efficiency of the diesel generator could be degraded, should the exhaust piping fail in an unusual way to block the pipe and build-up significant back pressure.

The licensee's commitment in the FSAR is that the diesel generator inlet and exhaust piping is classified as Design Class II, the intake air filter and air silencer are classified as Design Class I, and the engine exhaust silencer is classified as Design Class II. The criteria for Design Class I and II are defined in Section 3.2.1 of the FSAR. Design Class II components are considered important to reactor operation, but not essential for safe shutdown and isolation of the reactor. However, the diesel generator intake and exhaust system including filters and silencers have been qualified to the original Hosgri Spectra and current Hosgri Spectra where appropriate. Qualification models included explicit representation of exhaust silencer, piping and pipe supports. As a result of the Hosgri spectra qualification it has been determined that stresses in critical sections are within allowable values defined in ANSI B31.1-1967 standard. The Hosgri spectrum qualification has also identified the need for modification of piping as well as mounting braces of an exhaust silencer.

Based on the above discussion the staff concludes that any loss of efficiency in the operation of the diesel generators due to a large earthquake is not likely, provided that modifications to braces and piping supports are properly installed.

Staff Position

This issued is satisfactorily resolved subject to completion of modifications.

Action Required:

Proposed modification to diesel generator silencer bracing and pipe supports should be completed prior to reactor power ascencion beyond 5 percent.

Task: All egation or Concern No. 9

ATS No.: N/A   BN No.: 83-17

Characterization

This is not a allegation. It is a board notification.

Board Notification No. 83-17 involves the testimony of an NRC staff witness (J. Conran) in the Shoreham proceeding. In that testimony, Mr. Conran expresses his concerns in two areas, namely systems interactin and safety classification. The first concern, systems interaction does have some potential generic implications due to the aspects which involve the resolution of Unresolved Safety Issues-USI A-17. The second concern, safety classification is considered to be plant specific to Shoreham.

In response to the testimony of J. coran, the staff (F. Coffman, A. Thadani, R. Vollmer, C. Rossi, and R. Mattson) addressed both concerns. With respect to the systems interaction aspects, the staff stated: (a) the review of Shoreham against existing requirements provides reasonable assurance, pending the resolution of USI A-17, that the plant can be operated without undue risk to the health and safety of the public from potential adverse systems interactions: (b) the staff's program on A-17 is confirmatory in nature, and the staff continues to believe that reasonable progress toward a timely resolution of the USI is being made; (c) additional plant-specific systems interaction studies are not necessary as a predicate to licensing Shoreham.

Implied Significance to Plant Design, Construction and Operation

Based on the testimony of J. Conran, it is concluded that he had concerns on Shoreham in the two areas highlighted. There is the possibility that there would be similar concerns on Diablo Canyon with respect to USI A-17. The other concerns were plant-specific to Shoreham.

Assessment of Safety Significance

It is hard to assess the safety significance of Mr. Conran's concerns for Diablo Canyon because of some of the plant-specific aspects which are discussed in this testimony. Furthermore, in the case of Diablo Canyon the staff has placed additional requirements on the applicant based on the results of the appliant's seismic systems interaction program. See also Allegation 48.

Staff Position

The staff position (as summarized above under "Characterization") Unresolved Safety Issue (A-17) is reflected in the staff testimony in the Shoreham proceeding (Contention 7B). This position is generic and includes Diablo Canyon. In addition to the applicability of the staff's generic position, PG&E has completed over 90 percent of the seismic systems interaction program. The PG&E seismic systems interaction program goes beyond the requirements on Shoreham and will provide added assurance that Diablo Canyon can be operated safely. The modifications associated with the seismic systems interactions program must be completed prior to full power operations as documented in Supplement 11, NUREG-0675.

Action Required

None.

Task: Allegation No. 10

ATS No.: NRR 83-04 BN No.: 83-48 (4/4/83)

Characterization

Tilting of the containment structure under earthquake motions.

Implied Significance to Plant Design, Construction or Operation

Significant tilting of the containment structure could lead to an overstress situation at certain locations of the containment shell. It could also further amplify the floor response spectra in the vertical direction and thus cast doubt on the qualification of certain systems and equipment.

Assessment of Safety Significance

During a staff audit of the Diablo Canyon Project design documents on April 6, 1983, the calculations for the tilting of the containment structure, dated January, 1983 were examined. The calculations employed an approach consistent with that specified in Bechtel Topical Report BC-TOP-4A. This topical report has been reviewed and approved by the staff. The calculations indicated that there is an adequate factor of safety against tilting in the event of the Hosgri earthquake. Contrary to the allegation, cohesiveness between the foundation mat and the underlying rock and the low probability of occurrence with seismic excitation in the most critical direction were not considered in these calculations. The containment structure was shown to be stable against tilting.

Staff Position

The staff finds that the licensee's approach used for determining containment stability against tilting is acceptable and that the allegation presents no safety concern.

Action Required

None.

Task: Allegation No. 11

ATS No.: NRR 83-04 BN No.: 82-48 (4/4/83)

Characterization

Inadequate classification of the platform between the crane wall and the shield wall.

Implied Significance to Plant Design, Construction or Operation

If the platforms do not provide adequate support during a seismic event, the function of the Class I equipment and systems supported by the platform could be impaired.

Assessment of Safety Significance

A number of platforms supporting safety related equipment are located between the crane wall and the shield wall. Most of these platforms were not originally included on the "Q" list as seismic Class I items. Thus the need for seismic design, both original design and possible future modifications, may not have been recognized. However, the approximately 20 platforms in question were shown in the design drawings to be properly classified as seismic Class I platforms required to be designed and constructed to the proper seismic requirements. The omission of these platforms from the "Q" list was noted by the licensee in early April, 1983 and these platforms have since been added to the "Q" list. Based on a review of the design drawings for approximately 50 percent of the platforms the staff has concluded that this matter was properly resolved during the design review process employed by the Diablo Canyon Project.

Staff Position

The staff finds that this matter has been properly resolved and that there is no safety concern.

Action Required

None.

Task: Allegation #12

ATS No.: BN No.: 83-48 (4/4/83)

Characterization

The high energy line break (HELB) assessment did not meet the FSAR or R.G. 1.46 requirements.

Implied Significance to Plant Design, Construction or Operation

Safety related piping and requirement inside containment may not be properly designed to withstand jet impingement load from postulated pipe ruptures.

Assessment of Safety Significance

The staff has reviewed the results of the review and verification by the IDVP of the DCP effort on jet impingement effects inside containment. The IDVP reported the results of its verification in ITR-48, Rev. 0, "Additional Verification of Jet Impingement Effects of Postulated Pipe Rupture Inside Containment." The report provides a description of the work done, summary and evaluation of the results, and conclusions of the IDVP with respect to the concern of the jet impingement effects inside containment. The DCP responded to staff concerns by letters, including a letter of October 12, 1983, and in the meeting on September 28, 1983. Based on this information the staff has concluded that the licensee has met the FSAR commitment regarding the consideration of jet impingement loads inside containment, confirming the basis upon which the operating license was originally granted. However, under contemporary staff practice, aspects of jet impingement analyses that were judgemental for plants of the Diablo Canyon era are required to be demonstrated by deterministic analyses. To provide the basis for a jet impingement evaluation consistent with current practice, the DCP has completed a pipe break and jet target evaluation, and this effort has been reviewed and found acceptable to current standards by the IDVP. Based on this source and target evaluation, certain piping and structural members that could be subjected to jet loading, in the unlikely event that a large pipe rupture occurred inside containment, are currently being evaluated by analysis to determine what, if any, additional protection might be required to fully meet current requirements. The DCP has recently provided additional information and the current status of this effort. Both the DCP and W are conducting these evaluations which is scheduled to be completed in January 1984.

Staff Position

Upon completion of the ongoing jet impingement evaluation the licensee will submit a report to the staff identifying those targets for which additional protection would be required to meet current staff criteria. If modifications to achieve substantial additional protection are found necessary these modifications would be required before start-up after the first refueling.

Task: Allegation or Concern No. 13

ATS No.: NRR 83-04 BN No.: 83-48 (4/4/83)

Characterization: Inadequate Seismic Systems

Implied Significance to Plant Design, Construction or Operation

Failure to upgrade Class II equipment where its failure could damage Class I equipment might affect the capability to safety shut down the reactor and maintain it in a safe shutdown condition.

Assessment of Safety Significance

The concern indicates that there was a licensee commitment to upgrade any Class II system or components which if they failed during a seismic event would damage Class I systems or components such that they could not function properly. The commitment made by the licensee with respect to systems interaction was made as part of the seismic systems interactions study scheduled to be completed prior to exceeding 5% power operation. In this study the commitment made and accepted by the staff was quite flexible. It required alternatives including upgrading Class II items, relocating the Class item, protecting the Class I item, or relocating the Class I item. Fulfilling these alternatives will assure Class I systems and components will not present a safety concern by Class II systems and components failure during a seismic events.

Staff Position

The completion of the seismic systems interaction study and modifications identified will achieve the degree of safety desired by the allegation. This will be achieved through the licensee's commitment to use various alternatives rather than upgrading Class II systems and components to Class I. The completion of this study and modification is to be completed prior to exceeding 5% power operation.

Action Required

Staff review of study results and modifications. Completion of review is not an impediment of full power operation.

[illegible]

Task: Allegation No. 14

ATS No.: NRR 83-04 BN No. 83-48 (4/4/83)

Characterization

Analysis for the containment annulus structure did not include all potential loads.

Implied Significance to Plant Design, Construction or Operation

Failure to consider all potential loads could result in unconservative design of certain members in the annulus structure.

Assessment of Safety Significance

During staff audits of the Diablo Canyon Project design documents on April 6, and October 25/26, 1983, the staff examined several representative samples of design calculations selected at random from the entire set (several volumes) of design calculations for the containment annulus structure. The staff found that in each instance the licensee's final evaluation of the annulus structure was performed using the load combinations specified in the FSAR which included all potential loads expected for the annulus structure.

Staff Position

The staff finds that the concern expressed by this allegation is being addressed by the licensee in the normal design review and evaluation process.

Action Required

None.

Task: Allegation No. 15

ATS No.: NRR 83-04 BN No.: 83-48 (4/4/83)

Characterization

Inadequate tornado design criteria for the turbine building.

Implied Significance to Plant Design, Construction or Operation

Inadequate tornado design criteria could result in damage or failure of the masonry walls in the turbine building and therefore cause the loss of protection for class 1 equipment and systems against tornado wind.

Assessment of Safety Significance

All masonry walls in proximity to safety-related equipment are being re-evaluated for the loads resulting from a Hosgri earthquake using the appropriate response spectra. The design suction pressure of 0.86 psi is equivalent to 1.5g seismic load for an 8-inch thick masonry wall. The walls in question are being reviewed for a seismic acceleration of no less than 1.5g. Because the seismic loads are equal to or greater than the postulated tornado loads, the licensee concluded that the masonry walls located in the turbine building were adequately designed.

The switchgear and cable spreading rooms are located in the turbine building, and the separation between the individual rooms consists of 8-inch concrete block walls with all cells full of grout, number four reinforcing bars vertically, on 16-inch centers, and two number four reinforcing bars horizontally on 32-inch centers.

At the request of the staff, the licensee performed additional analysis to determine the capability of the walls to resist tornado loads. This analysis consisted of evaluation of the walls for a postulated 200-mph wind pressure, plus one-half of the associated atmospheric pressure drop. Because of the location of the equipment within the turbine building, the probability of a tornado-generated missile striking the vulnerable areas is small enough to be negligible and missiles were not included in the analysis. In the analysis, the licensee compared the required capacities of the walls with those which are available, using both criteria--those of IE Bulletin 80-11 and those of Standard Review Plan (SRP) Section 3.8.4, Appendix A (NUREG-0800). The analysis included comparison of shear stresses as well as those produced by the bending moment, which is governed by rebar tension stress. The allowable stresses of the material used in the evaluation were those obtained by the tests of the actual material installed instead of using code-specified minimum material properties.

In all cases examined, the available capacity of the walls exceed the required capacity.

The staff asked the licensee to provide additional information with regard to two items. The licensee must:

(1) demonstrate, by means of test records, or otherwise, that the material properties are consistant with those used in the analysis.

(2) demonstrate that the tornado loads that have been approved by the staff as appropriate for the site represent an upper bound of the loads that would be experienced by the subject masonry walls.

By a letter dated October 27, 1983, the licensee informed the staff that in January 1977, J. A. Blume and Associates provided a report to PG and E summarizing the actual strengths of various materials used in Diablo Canyon. The Blume report included investigation of the grade 40 rebar used in the turbine building masonry walls. Blume sampled 80% of the material test reports which were prepared by Pittsburgh Testing Lab. The data was analyzed and an average rebar yield stress of 51,390 psi was calculated. The Blume report contains a listing of the yield values from the lab test reports. Copies of these reports are available for review at PG and E.

With regard to the determination of the masonry strength, f_m, the licensee informed the staff that the PG and E specification for concrete block required tests to be performed as acceptance criteria for blocks to be used at Diablo Canyon. To satisfy this requirement, the block supplier, Air-Vol Block, Inc., provided a certificate stating that "all masonry units supplied...conform to all requirements of the plans and specifications." Block testing was performed by Central Coast Laboratories. Some representative Central Coast Lab test reports were sent to the NRC staff in a July 7, 1981 report addressing seismic qualification of masonry walls in compliance with IEB 80-11. These test reports indicate block compressive strength, based on the gross area of the block. Table 4.3 of the ACI 531 code identifies block compressive strength based on net area. Therefore, the applicant compiled the gross compressive strengths of the blocks from all the test data supplied by Air-Vol Block Inc. and then converted to a net area basis consistent with ACI 531. The average gross and net block compressive strengths are 1434 psi and 3830 psi, respectively.

Based on the above information the staff concludes that adequacy of the material properties used in the analysis have now been demonstrated and they are acceptable.

With regard to the second item, in a telephone conference call on November 3, 1983, PG&E confirmed that in their analysis of the internal walls the pressure drop (0.86 psi) is applied instantaneously. This is an upper bound on the pressure differential that any wall, internal or external, could experience. The rooms not being air-tight will communicate (vent) with the outside air. This venting will reduce the pressure differential on the internal walls to a value less than the 0.86 psi potential.

The staff reviewed the applicant's analysis of the pressure drop on the internal walls and conclude that the 0.86 psi upper bound is a conservative figure and is therefore acceptable (Ref. memorandum from O. D. Parr of NRC to G. Lear of NRC, dated November 9, 1983).

Staff Position

The staff finds that the licensee has provided reasonable assurance that the masonry walls will withstand the tornado loads. Therefore, the concern expressed by this allegation has been properly addressed.

Action Required

None.

Task: Allegation #16

ATS No.: NRR 83-04 BN No.: 83-48 (4/4/83)

Characterization

Inadequate design of high energy rupture restraint crushable pads.

Implied Significance to Plant Design, Construction or Operation

The crush pads provided as energy absorbers for some high energy pipe rupture restraints may have insufficient design margins thus increasing the calculated loads transmitted to structural steel members or concrete walls in the unlikely event of an instantaneous complete rupture of a high energy pipe.

Assessment of Safety Significance

The staff reviewed the design methodology for crushable pad restraints employed at Diablo Canyon during an NRC audit at PG&E/Bechtel offices on October 25, 1983. Based on the information gathered at this audit, the staff has found that the final crushable pad analysis and design were performed in a manner consistent with recognized engineering practice and staff requirments.

Staff Position

The staff finds that this matter was properly resolved in the design review process for the Diablo Canyon Project and that there is no safety concern.

Action Required

None.

Task: Allegation #17

ATS No.: NRR 83-04 SN No.: 83-48 (4/4/83)

Characterization

Seismic criteria for Westinghouse items: NSSS SSE loads inadequate.

Implied Significance to Plant Design, Construction or Operation

The Nuclear Steam Supply System (NSSS) equipment and piping is designed for a Safe Shutdown Earthquake (SSE) as originally defined by PG&E for the Double Design Earthquake (DDE). Thus, the Westinghouse SSE analyses were not systematically updated based on the new Hosgri SSE loads.

Assessment of Safety Significance

The NSSS and all safety related equipment and piping may not be qualified for Hosgri SSE loads.

Staff Position

The NSSS and all safety related equipment and piping within the scope of Westinghouse were qualified for the Hosgri event prior to the design verification effort. This information was documented in the Hosgri report which was reviewed and accepted by the staff. In addition, new spectra generated by DCP reevaluation effort were also transmitted to Westinghouse. The IDVP conducted a review of the PG&E/Westinghouse seismic interface and verified that appropriate controls for the transfer of information existed and that Westinghouse used the applicable information. Therefore, the staff finds the allegation does not present a safety concern in either low power test or full power licensing.

Action Required

None

Task: Allegation or Concern No. 18

ATS No.: Q5-83-001 BN No.: 83-55

Characterization

QA/QC Allegations

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 18

ATS No.: Q5-83-001 BN No.: 83-51, 83-55

Characterization

QA/QC Allegations

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 19

ATS No.: Q5-83-002 BN No.:

Characterization

Guard Qualification

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 20

ATS No. RV-83-A-018 BN No. N/A

Characterization

Licensee's Health Physics personnel are not qualified to American National Standard Institute (ANSI) requirements.

Implied Significance to Design, Construction or Operation

Technical Specification 6.3.1 requires that each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 except for the Supervisor of Chemistry and Radiation Protection who shall also meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. At other facilities failure to have properly qualified Health Physics personnel has resulted in poor implementation of the radiation protection program. A poor radiation protection program could result in personnel over exposures or release of materials to the environment above regulatory limits.

Assessment of Safety Significance

The qualifications of the Supervisor of Chemistry and Radiation Protection and those of his alternate were reviewed by NRR in February 1981 and found to meet both the ANSI standard and Regulatory Guide 1.8, September 1975. The individuals involved have had experience at another reactor facility and have been involved in the development of the radiation protection program at Diablo Canyon since its inception.

The licensee has a program for reviewing the qualifications of the unit staff to insure that the ANSI N18.1-1971 requirements are met. Our review of this program has found it to be adequate. Region V has raised a question regarding the experience requirements as it applies to Chemistry and Radiation Protection technicians. Section 4.5.2 of the ANSI standard states "technicians in responsible positions shall have a minimum of two years of working experience in their specialty." Chemistry and Radiation protection are usually considered to be be two separate specialties. The licensee considers that a combined total of two years meets the requirements of the ANSI standard. It is Region V understanding, however, that NRR has not established a firm position on this issue.

Staff Position

Region V staff believes that the licensee's professional staff meets the requirements of the ANSI standard and of Regulatory Guide 1.8, September 1975. The question of the required number of years of experience for Chemistry and Radiation Protection technicians in responsible position needs to be resolved.

Action Required

Region V submitted a request for guidance on the required experience for Chemistry and Radiation Protection technicians to IE on December 2, 1983. This issue has generic implications and needs to be reviewed in that light.

Implementation of this requirement at Diablo Canyon should be consistent with implementation at other reactor facilities.

Task: Allegation or Concern No. 21

ATS No. RV-83-A-018 BN No. N/A

Characterization

The licensee has poor practices as far as keeping exposures as low as reasonably achievable (ALARA). Specifically, (1) the air from the chemistry laboratory is only exhausted by means of the fume hoods and this is inadequate; (2) the licensee intends to permit all floors in the restricted area to become contaminated; (3) the licensee will not provide respiratory protection equipment to workers on demand.

Implied Significance to Design, Construction or Operation

Regarding the specific concerns there are no specific NRC requirements covering these subjects.

Assessment of Safety Significance

These concerns are not founded. The fume hoods are not the only means of removing air from the chemistry laboratory and the hoods alone exceed the OSHA required number of air changes per hour. Statements in the licensee's radiation control procedures indicate that corridors in the restricted area will not be permitted to remain contaminated, if they so become.

Finally the licensee currently plans to provide respiratory protection equipment to individuals who demand their use, even if the radiological conditions do not require respiratory protection. Individuals will have to have been tested and trained in the specific equipment being used.

Staff Position

A licensee's operational ALARA program cannot be clearly examined until the plant is operational. The licensee is committed to a strong ALARA program and this commitment is reflected in statements in their procedures.

Action Required

Region V will review the licensee's implementation of their operational ALARA program when the plant becomes operational.

Task: Allegation or Concern No. 22

ATS No. RV-83-A-018 BN No. N/A

Characterization

Modifications to the Air Ejector Discharge Radio-Gas Monitor (RE-15) and the Gas Decay Tank Discharge Radio-Gas Monitor (RE-22) have made these monitors insensitive to Xenon-133 and Krypton-85. An environmental shield has been placed over these monitors that prevents the detection of these nuclides.

Implied Significance to Design, Construction or Operation

The Air Ejector Discharge Monitor is used for indications of a primary to secondary system leak. If this monitor is not sensitive to Xenon-133 and/or Krypton-85 primary to secondary leaks would not be detected as promptly.

The Gas Decay Tank Discharge Monitor is used to monitor discharges from the gas decay tanks. This channel will alarm at the main control board and Auxiliary Building control board and close the gas decay tank vent valve on a high radiation level. Failure of this monitor to detect Xenon-133 or Krypton-85 could result in an unmonitored release or an unplanned release.

Assessment of Safety Significance

The Air Ejector Monitor is in a hostile environment, high humidity and temperature. The Gas Decay Tank Discharge Monitor monitors what may be relative high concentrations of an undiluted stream. The licensee procured environmental shields from the manufacturer of these monitors to protect them from the hostile environment, and to decrease the sentitivity, respectively. The manufacturer has provided the licensee with analysis of responses for Xe-133 and Kr-85 for these monitors. As expected the beta emissions from these radionuclides is completely shielded by the environmental shields. However, the gamma emissions (514 Kev for Kr-85 and 80 Kev for Xe-133) penetrate the shield and are detected by the monitor. The licensee stated that the vendor's response curves will be verified when the plant is operational.

Staff Position

Region V believes that the concerns expressed have been addressed and that the reduced sensitivity of these monitors does not constitute a significant safety issue.

Action Required

No action is required.

Task: Allegation or Concern No. 23

ATS No.: Q5-83-017 BN No.:

Characterization

QA Inspector concerns

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 24

ATS No.: RV83A28, RV83A33, & RV83A52 BN No.:

RV 83A46 83-164 (10/27/83)

Characterization

A site contractor (HPFoley (HPF)); (1) rejected nonconformance reports without justification, (2) was not documenting nonconformance reports issued by field inspectors, (3) has incorrect procedures for voiding nonconformance reports, and (4) incorrectly rejected defective weld reports. This characterization includes all of the above referenced allegations.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Predecisional

Action Required

Task: Allegation or Concern No. 25

ATS No. RV83A33 BN No. N/A

Characterization

Deficiencies in the use of Phillips Red Head anchors by a site contractor (H. P. Foley).

Implied Significance to Plant Design, Construction, or Operation

Improper use or installation of anchor bolts could result in reduced load carrying capacity of safety-related electrical cable tray supports and, consequently, the possibility of safety-related cable failures during a seismic event.

Assessment of Safety Significance

The staff examined project requirements and procedures related to anchor bolts, audits and descrepancy reports related to anchor bolts, conducted interviews with quality control inspectors, performed inspections of installed anchor bolts, and conducted independent verifications such as torque testing and ultrasonic length measurement of anchor bolts.

The staff reviewed the basic licensee's design and inspection criteria for electrical raceway supports. Acceptance criteria for electrical raceway supports is contained in Design Criteria Memorandum No. C-15, which refers to

PG&E Standard Drawing 054162, Rev. 3 for allowable loads on concrete expansion anchors. Standard Drawing 054162, Rev. 3 also contains PG&E criteria for the installation of concrete expansion anchors.

The staff found that the Foley concrete expansion anchor installation inspection criteria were not always consistent with PG&E specified installation requirements and, thus did not always provide assurance of an adequate installation. This is an apparent violation of 10 CFR 50, Appendix B, relating to the adequacy of contractor procedures. The staff performed a field sample (described below) and found the installations to be generally satisfactory.

The staff also reviewed related audits conducted by PG&E. PG&E Discrepancy Report DR-288 identified that HPF had installed anchor bolts that did not meet minimum embedment criteria. The disposition of this deficiency report was to accept-as-is, based on the results that less than 1% were improperly installed. The required embedment depths were reduced based on pull out tests conducted by PG&E. PG&E stated that the disposition of DR 288 was later evaluated by civil engineering and included a 100% review of HPF QC inspection sheets. This review resulted in several anchors being modified or dispositioned as acceptable; however, PG&E could not verify that 11 anchors, which did not meet minimum embedment criteria, were identified by this review and could not provide an estimate of the number of anchor bolts that potentially may not meet minimum embedment criteria.

PG&E Quality Assurance issued An Audit Finding Report, on 8/23/83, which stated "Requirements for the tensioning and associated testing of concrete anchor bolts used in Electrical, HVAC and Instrumentation support have not been specified by Engineering." PG&E's engineering response, dated 9/20/83, stated that a specification was not required. The staff has not completed its assessment of the safety implications of the apparent failures to: (1) provide adequate discrepancy report justification, and (2) provide adequate engineering resolution of identified deficiencies in the ends of anchor bolts.

With respect to interviews with QC inspectors, the staff found that many QC inspectors did not consider the Phillips Red Head Stud Anchors to be good anchor bolts, and however, the QC inspectors did not have specific examples pointing to deficient bolt installations. Their judgments were based on experiences at other jobsites and observations of some anchor bolts pulling out as they were tightened. Some QC inspectors erroneously thought the use of Red Head Anchors had been banned by NRC criteria.

An NRC independent inspection team from Lawrence Livermore National Laboratory (LLNL), under contract to Region V, inspected a sample of 124 electrical raceway supports modified in 1982. These supports contained approximately 1000 anchor bolts. The inspection was in accordance with the applicable Foley procedure in effect at the time of installation. No torque verification was performed by LLNL. The results of this inspection identified two loose anchor bolt nuts, one anchor bolt which did not meet alignment criteria, and one anchor bolt which did not meet minimum thread engagement criteria. The sample indicated a low failure rate.

To supplement the Livermore inspections, the staff requested that PG&E conduct torque tests and ultrasonic length measurement of forty one-half inch diameter anchor bolts in the Unit 1 cable spreading room to verify that the bolts could sustain design loadings. The actual test, witnessed by staff, were per formed by torquing and measuring anchor slip. No anchor bolt failures occurred and the maximum anchor slip was less than PG&E allowables. Ultrasonic testing, performed on each anchor bolt torqued (after torquing) to verify that minimum embedment criteria had been met, identified one anchor bolt that did not quite meet minimum embedment criteria. Overall, widespread failures were not found.

PG&E design engineers stated that they do not consider that the potential for undertorqued concrete anchor bolts would affect the seismic qualifications of the cable tray systems.

Staff Position

Installation Criteria

The lack of consistency between PG&E anchor bolt design and installation criteria and H. P. Foley's installation and acceptance criteria will be resolved in the followup of the licensee's action.

Embedment

The staff considers that all anchor bolts which do not meet minimum embedment criteria should be identified to verify that an appropriate factor of safety has been used in design. The design allowable loads in PG&E Standard Drawing 054162, Rev. 3 result in an approximate safety factor of of 3.

Installations

The Lawrence Livermore anchor bolts inspections and the NRC torque test results do not indicate widespread deficiencies in the installation of the Phillips Red Head Anchors.

The use of the Phillips Red Head Anchors does not violate any current NRC criteria.

Torquing

Since the Foley procedures and the PG&E criteria do not require tightening the Phillips Red Head Anchors to any specified torque value, the staff should require PG&E to formally verify that the potential lack of preload on the anchors does not affect the analysis or qualifications test results for the electrical raceway supports.

Action Required

The licensee will be required to formally address and justify their anchor bolt installation and acceptance criteria, and staff review and evaluate this issue prior will be performed prior to reactor criticality.

Task: Allegation or Concern No. 26

ATS No.: RV83A33 BN No.: N/A

Characterization

A site contractor (H. P. Foley) was not documenting nonconformance reports issued by field inspectors.

Implied Significance to Plant Design, Construction, or Operation

The site contractor in question (H. P. Foley) has been responsible for installation and modification of electrical, civil and mechanical design class 1 safety systems and/or structures which are necessary for the safe operation and shutdown of the plant.

Assessment of Safety Significance

See Task Allegation or Concern No. 24

Staff Position

See Task Allegation or Concern No. 24

Action Required

See Task Allegation or Concern No. 24

Task: Allegation No. 27

ATS No.: RV 83A33 BN No.: 83-02/14

Characterization

Inadequate welding procedure and quality of welders and materials used in Superstrut construction for cable trays, conduits and instrument supports.

Implied Significance to Plant Design, Construction or Operation

Inadequate welding procedure, substandard materials, and non-qualified welders could all result in load carrying capacity of cable trays, conduits and/or instrument supports less than what was designed for. Therefore, failure of these Superstrut structures could occur. As a consequence, the safety function of those systems and components supported by these structures could be severely impaired.

Assessment of Safety Significance

An inspector from the NRC Region IV Vendor Program Branch (VPB) inspected the "Superstrut" manufacturing facility during the period December 6-8, 1982. This facility manufactures mild steel fittings, brackets, and channels, some of which are used to construct cable tray, conduit, and instrument supports in nuclear power plants. The Region IV inspector informed the Region V staff of his findings at the manufacturing facility as follows:

(1) There was no formal quality assurance (QA) program before 1979.

(2) There were no records of the qualification of welding operators or welding procedures.

(3) Before 1980, spot welds were not sample tested and not controlled by procedures.

(4) There was no traceability of material.

(5) There were no quality records before 1980.

(6) Generally, the current QA program did not meet the intent of the criteria of 10 CFR 50, Appendix B.

The VPB inspector also informed the Region V staff that the Superstrut material manufactured at the manufacturing facility has been used at nuclear power plants in Region V, including Pacific Gas and Electric's Diablo Canyon nuclear power plant.

Region V dispatched an inspector on December 8, 1982, to conduct a special inspection of Diablo Canyon to determine the scope and potential impact of the VPB inspector's findings. The Region V inspector found (1) that the back-to-back double channels that were spot welded together, as well as the channels with welded end brackets, were widely used (up to 11,000 supports out of approximately 24,000 in the Diablo Canyon facility) and (2) that the Diablo Canyon engineering staff had treated the double channel Superstrut material as a composite member and not as two members acting independently.

The staff review has focused on the potential failure of the Superstrut spot welds. Failure of the spot welds would allow the strut material to act independently, reducing stiffness, changing the support frequency, and thus affecting the seismic qualification of the raceway system.

To establish the quantitative strength and quality of spot welds in the Diablo Canyuon cable tray supports, the licensee undertook a testing program on Superstrut spot welds. The test program consisted of sample selection, specimen preparation, and specimen testing. Three types of composite Superstrut are installed in the plant. They are all back-to-back and are identical in section width and material thickness, varying only in member depth: A-type (3-1/2 inches deep); E-type (4-7/8 inches deep); and H-type (6-1/2 inches deep). From a review of the support details, the applicant determined that 60% of all supports use A-type sections, 30% use E-type, and 10% use H-type.

Specimen preparation and testing was conducted and documented in the Bechtel Corporation Material and Quality Services Testing Laboratory. The final sample--consisting of 162 A-type members, 34 E-type members, and 9 H-type members--was tested to failure. This sample size and the method of selecting the samples have been confirmed by the staff to be statistically valid for assessing the acceptability of the as installed welds with a high level of confidence. For each specimen, shear load was applied to a single spot weld by applying a tensile load to the specimen at a slow uniform rate until the spot weld failed. All welds tested had strengths greater than 1600 pounds. During the shear tests, substantial elongation of the spot welds was observed, indicating the failure of the specimen to be ductile rather than brittle.

For each strut type, the minimum value of the various limits developed from the test program was chosen as the allowable shear per weld. The values for allowable shear per spot weld developed from this test program are based on the ultimate strength rather than the yield strength of the spot welds tested. THe allowable.values are less than 70% of mean ultimate. Reliance upon these allowables will preclude a collapse failure of the composite strut section.

The staff has completed its review of the licensee's test program and accepts the allowable limits for welds as established in the program.

Staff Position

As previously reported in Section 3.8.5.4.10 of SSER 17 and in Section 3.4.3 of SSER 19, the staff considers that the concern expressed in this allegation has adequately been resolved.

Action Required

Region IV has completed inspections at all known vendors of similar materials for nuclear plant use. Review of this matter is being pursued on a case by case basis as appropriate.

Task: Allegation No. 28

ATS No.: RV 83A41 BN No.: 83-161 (10/18/83)

Characterization

The annulus structure reverification is erroneous.

Implied Significance to Plant Design, Construction or Operation

There are a total of six specific concerns expressed by this allegation regarding the licensee's annulus structure verification effort. They are:

(i) The computer model used is incorrect. Members are missing, have wrong properties input and have boundary conditions different from existing conditions.

(ii) Loads from large bore piping are wrong and unrealistic in the way they are input in the model. Also, Fan Cooler loads are not considered.

(iii) Loading conditions used are incomplete without LOCA, rupture loads and proper thermal loads between polar crane wall and the annulus structure.

(iv) All members are not selected for design re-verification in Phase II design. Hand computations and the program CE 217 are used to identify "critical members". Both of these methods are unconservative in their treatment of torsion -- which governs in a few members. As a result of using this procedure, some overstressed members may not be checked for adequacy.

(v) Frequency modifications have been done on the structure. Additional bracings, stub columns and other members have been added without any backup calculations for the members or their connections.

(vi) Deficiencies in the CE 217 program preclude checking of composite members and built-up members. To date these have not been checked and may have some over-stressed members which have not been identified.

Concerns (i) and (ii) question the adequacy of the dynamic and static reanalyses for the annulus structure.

Concerns (iii) through (vi) question the design margin achieved for some structural members and connections in the annulus structure. In particular, the treatment of torsion loads in these members and connections are questioned.

<u>Assessment of Safety Significance</u>

During staff audits of the Diablo Canyon Project design documents on October 25 and 26, 1983 the staff with the aid of their consultant, Brookhaven National Laboratory, reviewed the seismic model used for the annulus structure and found that the model included the recent modifications to the annulus structure. Specifically, the global frequency analysis and Calculation Book 7140C, File No. 52.17 were reviewed. The staff verified that the model included all the new members. Calculation Book SK-112C-1 and SK-112C-2 were also reviewed to identify the type of connections used. Member properties (mass and stiffness) used for the model appear to be correct.

For the static analysis of the annulus structure, the DCP has divided the effort into two phases. Phase I included all major modifications, while Phase II was intended to perform final analysis including all the Phase I modifications. The Phase II work is still in progress. A 3-D BSAP model was developed and used for member stress evaluations for Phase II. It appears to be adequate. Some member connections were checked and found satisfactory. Several new members were verified to have been properly incorporated into the model.

Calculation Book 2024 C-1 for large bore piping loads was reviewed. Transfer of hanger/piping loads to structure members for Hangers 57N/60R and 57N/91R was specifically reviewed. For Hanger 57N/60R, loads resulting from Bechtel ME 101 program were applied to nodal points on the structural model. It appears to have been done properly. For Hanger 57N/91R, loads are supported by secondary members which are framed into main structural members. A STRUDL analysis was performed to obtain reaction forces for the secondary structural systems. These loads were then used to input into nodes on the main structural model. A general assumption was made that the major steel members of the annulus structure were rigid in torsion at the boundary between the major members and pipe supports. This assumption may not be fully consistent with the actual action of the major members particularly under large bore piping thermal loads and various qualified analysts will disagree on the need for explicit consideration of member torsional stiffness in this instance. Some additional analyses could be performed but the staff believes that a more meaningful assessment of the annulus steel member-pipe support systems can be made by careful visual inspection with the plant systems at full operating temperature.

Fan Cooler loads (File No. 5217) were reviewed. All loads appear to have properly been transferred to columns. The Fan Cooler loads were directly input to the columns in the 3-D BSAP model.

Calculation Book 2101C dated October 13, 1983 was reviewed. Jet impingement loads due to pipe breaks were identified for the two of those load cases. Stresses in the column resulting from jet impingement loading were reviewed. It appears that column stresses were small and within the allowable limits.

The staff reviewed hand calculations and CE 217 which were used for the stress evaluations of the members of the annulus structure. The results of the hand calculations was a list of stress ratios for all members of the annulus structure. Ratio above 0.5 were further evaluated with the 3-D BSAP model. The DCP identified 5 members overstressed and additional modifications will be made to reduce the stress level to within the allowable limits.

A typical beam calculation including the effect of torsion was reviewed. Specifically the calculation reviewed was for a tangential beam between columns 2 and 3 at elevation 117 ft. The beam was analyzed in two steps. In the first step, CE 217 was used to compute the bending and shear stresses without consideration of torsion. In the second step, the DCP performed a hand computation for torsion. The results for these two steps were then combined to evaluate the adequacy of the beam. The hand calculation for torsion was conservatively performed. The DCP is making 80 additional joint modifications to the annulus structure to account for torsion.

Calculation Book 2102C dated October 24, 1983 was reviewed. It contains stress evaluation for two of the strut columns and all of the bracings that were added to the annulus structure. The stresses in these members were shown to be within the allowables. Similar calculations are being extended to include all of the additional members.

An example of hand calculations of torsion (2092C-2) for a built-up member was reviewed. The standard AISC method was used to determine the torsional effect. Since this method is only applicable to symmetric I sections, the built-up member was converted to an equivalent I section having smaller sectional properties than the actual built-up section. This procedure will yield conservative results.

Staff Position

No safety concern was identified for most of the matters related in this allegation. However, the staff recognizes that the transfer of large bore piping loads to the main structure based on an assumption of a rigid boundary in torsion for the supporting structural members is controversial in the opinion of some qualified analysts. For this reason, as well as general prudence considering the numerous pipe support modifications made at Diablo Canyon, a careful inspection of the pipe support systems under operating thermal conditions is necessary.

Action Required

Careful visual inspection of pipe supports and pipe support structures by the licensee with the plant at operating thermal conditions. The licensee has such an inspection planned as part of the plant startup.

Task: Allegation #29

ATS No.: RV 83A41 BN No.: 83-161 (10/18/83)

Characterization

Pipe Restraint Design Inadequate

Implied Significance to Plant Design, Construction or Operation

Pipe rupture restraints may be inadequately designed to prevent pipe whip impact of safety-related equipment.

Assessment of Safety Significance

No safety significance until 5% power is exceeded due to negligible fission product inventory.

Staff Position

The staff has performed a review of PG&E/Bechtel design criteria documents relating to pipe rupture restraints design inside containment. These documents were reviewed as part of an NRC audit performed on designs at pipe rupture restraints inside containment for the purpose of addressing and resolving an open issue in SSER No. 18. The criteria in these documents are considered acceptable, and in accordance with current industry practice. Therefore, the staff finds the allegation does not present a safety concern that would effect licensing for either low power testing or full power operation.

Action Required

None

Task: Allegation or Concern No. 30

ATS No.: RV83A41 BN No.: 83-161 (10/18/83)

Characterization

Safety-related equipment has inadequate/untraceable documentation.

Implied Significance to Plant Design, Construction, or Operations

It is not possible to definitively assess the significance of this allegation due to its vagueness.

Assessment of Safety Significance

To assess the significance of the concern the staff evaluated: (1) project nonconformances identified in 1982 and 1983, (2) licensee procedures for assuring completion and documentation of equipment qualifications, and (3) engineering project files related to descriptive and qualification records for safety-related mechanical equipment. Records verified included, as specific examples, those associated with the qualification of the reactor vessel head vent solenoid, the reactor coolant subcooling meter and four auxiliary feedwater level control valves; and the seismic analysis of the diesel generators.*

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*This evaluation was performed by Region V at the PG&E offices in San Francisco on October 25, 1983 and November 3, 1983. In addition, discussions were held with the licensee management and engineering personnel. During the examination of documentation described above, the contacted individuals were asked if they were aware of any circumstances which might have caused the subject allegation to have been made.

Licensee personnel questioned in San Francisco stated that the Diablo Canyon Project was different from other Bechtel projects in that its large civil/structural scope necessitated that many equipment qualification analyses be assigned to support groups within Bechtel or PG&E rather than to the civil/structural group. They suggested that, since the civil discipline was not completing many of the qualifications that they normally would, this might be perceived by unknowing personnel as a failure to provide the qualifications.

Interviews with over 150 licensee and contractor site personnel (during 11/29-12/9) were also conducted by the NRC staff. This afforded a ready opportunity for individuals to express any specific concerns they had regarding adequacy and traceability of documentation for safety-related equipment. No concerns were raised other than those already being pursued by the staff.

On the basis of our reviews, the documentation of safety-related equipment appears satisfactory. In examination of nonconformance reports, one instance was noted in which engineering personnel had mistakenly disposed of some valve records (revisions of drawings). This resulted in inadequate documentation. However, the problem was identified and properly resolved by the licensee in that satisfactory replacement documentation was obtained.

Staff Position

On the basis of the data evaluated above, the allegation was not confirmed. The staff evaluation did not identify any unsatisfactorily addressed deficiencies related to this allegation. There is not any evidence to suggest that any further extensive investigation is warranted.

Action Required

None

Task: Allegation No. 31

ATS No.: RV 83A41 BN No.: 83-161 (10/18/83)

Characterization

Structural programs are being used without proper QA procedures.

Implied Significance to Plant Design, Construction or Operation

If computer programs had been used without or with poor QA procedures, the adequacy of the results obtained from these programs and thus the adequacy of the structural design would have been questionable. The design may or may not be adequately conservative.

Assessment of Safety Significance

During a staff audit on October 25 and 26, 1983, the staff selected four major, i.e., extensively used, programs from a list of 70 computer programs used for the analysis of Diablo Canyon Unit 1. The review was primarily concentrated on verification documentation of these computer programs. Specifically, programs CE 217, ANSENV, SECTSTR, and SPSS were reviewed. The staff concluded that these major programs have been reasonably documented and verified. The staff also determined that there were some pre- and post-processor programs used but were not formally documented. The staff reviewed a small sample of results obtained using the undocumented pre- and post-processor programs and found no unusual results.

Staff Position

Although some pre- and post-processors were not formally documented, no discrepancies were discovered. The staff finds that there is no safety concern.

Action Required

The licensee will be requested to document all pre- and post-processor computer programs employed in any and all analyses for Diablo Canyon Unit 1 to complete the record for future reference.

Task: Allegation No. 32

ATS No.: RV 83A41 BN No.: 83-161 (10/18/83)

Characterization

Dynamic analysis of the containment building, its roof and interaction with soil and adjoining structures is inadequate.

Inserts and attachments to containment walls are inadequate as well as connections to the liner plate.

Implied Significance to Plant Design, Construction or Operation

An inadequate dynamic analysis of the containment building could result in erroneous response in the shell and other structural elements. It could also result in unconservative response spectra which are used for the seismic qualification of systems and components.

An inadequate design of inserts and attachments as well as connections to the liner plate could lead to ultimate failure of these structural elements and/or excessive leakage of the containment.

Assessment of Safety Significance

During staff audits on October 25 and 26, the staff reviewed the Diablo Canyon Project design documents. A soil-structure interaction analysis (SSI) for the containment structure was performed by the licensee for the DDE earthquake. The approach used in general would not properly account for SSI effects for a soil site. Therefore, the concern expressed in the allegation is valid in that the model leads to essentially fixed-base response. However, at the DCP site, the foundation material is competent rock with shear wave velocity above 3500 ft/sec. Significant SSI effects are therefore not expected and a fixed-base analysis is appropriate.

With respect to the inserts and attachments to the containment as well as connections to the liner plate, Calculation Book 290 C-1 for connections between the liner and concrete containment was reviewed. Calculations appear to be reasonable. No anomalies were found for the anchor bolts to the concrete, steel plate to the hangers, and steel plate stress itself.

Penetration calculations (entitled "Other Major Penetrations" dated October 25, 1983) for the main steam line, MSL P-6 penetration was reviewed. The penetration is a 40 inch ID pipe with circumferential plate to resist pipe loads. The loads came from Westinghouse calculations dated June 22, 1983. The calculations considered bending and shear stresses in the sleeve and side plate, strength of the welds, and bearing stress in concrete. The calculations appear to be reasonable and stresses were within the allowable. It is noted that all work in the penetration area appears to have been performed very recently. It is not clear whether review for all penetrations was complete.

Calculation Book 280 C-1 for connections between piping supports and containment shell (two main steam lines and two feedwater lines between the containment and the turbine building) was reviewed. The stress analysis for the connections appear to be reasonable. Calculations for rebar stresses in the shell also appear to be reasonable.

Staff Position

The staff finds that the concern regarding the dynamic analysis of the containment structure for the DDE earthquake is valid with regard to general soil structure interaction analysis. However, because of the foundation material at the DCP site (rock) the analysis was acceptable for Diablo Canyon.

The staff finds no safety concern regarding the inserts and attachments to the containment shell and the connections to the liner plate. Therefore there is no impact on low power testing or full power licensing.

Action Required

The licensee should confirm that all penetrations have been or will be reviewed for structural adequacy.

Task: Allegation No. 33

ATS No.: RV 83A41 BN No.: 83-161 (10/18/83)

Characterization

The turbine building is designed as a Class 2 structure but contains Class 1 piping and equipment.

Implied Significance to Plant Design, Construction or Operation

The functions of the Class I piping and equipment could be impaired if the turbine building failed during a Hosgri event.

Assessment of Safety Significance

The turbine building was originally designed as a Class II structure. However, since it houses Class I equipment and piping, the turbine building was reanalyzed as if it were a Class I structure. Rigorous analysis were performed using the Hosgri seismic input and resulted in extensive modifications in the building and turbine pedestals.

Staff Position

Since the turbine building has been modified to meet Class 1 structural requirements, it does not present a safety concern,as alleged,that would effect licensing the plant for either low power testing or full power operation.

Action Required

None

Task: Allegation or Concern No. 34

ATS No.: RV83A41 BN No.: 83-161 (10/18/83)

Characterization

Incomplete and inaccurate as-built drawings.

Implied Significance to Plant, Design, Construction, or Operations

Accura~~ct~~te as-builts are required at Diablo Canyon 1) for design reconciliation of the as-constructed hardware and 2) to provide accurate, readily accessible information for operations personnel (including maintenance and engineering) on details of mechanical electrical and instrumentation systems.

Assessment of Safety Significance

The staff reviewed the status of as-built drawings and the effectiveness of the controls applied. This was done by review of procedures, through personnel interviews, reviews of as-built drawings and by a field check of the accuracy of modification as-builts. Additionally the results of a joint Region V/NRR as-built inspection conducted at the San Francisco offices on October 25, 1983 was utilized.

Note: Two "types" of as-built drawings are recognized and considered. The first "type" is the design drawing as-built which is generally an updating of the originally issued design drawing; e.g. piping isometrics, area structural steel drawings, and wiring schematics. The design drawing as-builts are of primary importance for operations personnel.

The second "type" of as-built drawing of importance at Diablo Canyon is the "modification as-built." The modification as-built is a contractor prepared drawing a of particular, usually small, modifcation such as a single beam added to the annulus steel structure, or a single pipe support. The modification as-builts are of primary importance for design reconciliation.

The staff examined both design drawing as-builts and modification as-builts.

The status of the backlog of incomplete as-builts appears to be reasonable. There is a current backlog in engineering of 90 design drawings as-builts to be updated and 336 modification as-builts to be reconciled to design.

The accuracy of modification as-builts was examined by detailed field inspections of about 500 modifications over the past several months by NRC contracted, Lawrence Livermore National Laboratory personnel. Although in general the modification as-builts were found to accurately represent the as-built condition 3 violations were found concerning such items as missing welds and bolting. Followup and resolution of these identified problems will be done through the routine program.

The accuracy of the design drawing as-builts (generated in the San Francisco offices) was not assessed at the site, but has been previously assessed by the staff and the IDVP.

The effectiveness of the use of the field generated modification as-builts by the design engineering group was assessed during a joint Region V/NRR inspection during the week of October 25, 1983 and was found satisfactory. In addition, this area was reviewed by the IDVP where in only minor problems were identifie d.

Staff Position

The field-generated modificjation as-built drawings for design reconciliation appear to have reasonable backlogs and engineering utilization controls.

The engineering generated design drawing as-builts, necessary for operations personnel use, have a backlog of incomplete drawings and a reasonable schedule for completion. The availability and accuracy of these design drawing as-builts should be verified prior to exceeding five percent power.

Action

Region V review the accuracy of design drawing as-builts, prior to exceeding five percent power.

Task: Allegation #35

ATS No.: ______________ BN No.: BN 83-168 (10/27/83)

Characterization

Lack of support calculations for support of fluorescent light fixtures (control room).

Implied Significance to Plant Design, Construction or Operation

Falling light fixtures as a result of a large earthquake could incapacitate operators.

Assessment of Safety Significance:

This issue was discussed with the licensee in a telephone conference call on December 6, 1983 in order to obtain pertinent background information. The light fixtures in the control room are not safety related. However, their gross failure in a manner that could incapacitate operators in the control room is not acceptable. The approach in the staff review has been to understand the general arrangement of the control room suspended ceiling and the fluorescent lighting fixtures, and to develop an engineering judgment as to the seismic capability of the control room ceiling and light fixtures.

The licensee described the general arrangement of the control room suspended ceiling and light fixtures during the conference call on December 6, 1983, and provided a sketch of the general arrangement which was received and reviewed on December 9, 1983. The licensee indicated that the suspended ceiling has been designed and constructed as a structural grid system to withstand earthquake loading from both vertical and horizontal components. The fluorescent light fixtures are attached to the structural grid system holding up the suspended ceiling and at an elevation several inches above the level of the ceiling tiles. Thus even if one of the fluorescent tubes comes off the fixture it should drop on the ceiling tile.

The staff did not review any calculations. However, based on the review of the structural details and the statement by the licensee that a proper evaluation of the seismic capability of the ceiling and fluroescent light fixtures for the control room had been conducted, the staff feels that the likelihood of a falling fluorescent light fixture and incapacitating an operator as a result of an earthquake is very low. Furthermore, there is a remote shutdown panel away from the control room providing alternate capability to bring the reactor to a hot shutdown condition.

Staff Position

This issue is satisfactorily resolved.

Action Required

None

Task: Allegation #36

ATS No.: ______________ BN No.: BN 83-16B (10/27/83)

Characterization

Resolution analysis of fluorescent light fixture interaction assumed conduit connection to be hinged-inspection found fixed connections.

Implied Significance to Plant Design, Construction or Operation:

Fluorescent light fixtures that are hung by their conduits may fail as a result of a large earthquake and fall on safety related equipment causing it to malfunction. The safety implication is that of adverse interaction between safety and non-safety equipment during and following a large earthquake.

Assessment of Safety Significance:

This issue was discussed with the licensee in a telephone conference call on December 8, 1983 in order to obtain pertinent background information. The basis for this concern is discussed in a letter from Steve Traisman of M. G. Jones Engineering Consultants, Inc. to L. W. Horn of Pacific Gas and Electric Company dated June 21, 1983. Since failure of non-safety lighting fixtures interfering with the function of safety equipment is clearly unacceptable, the approach in the staff review has been to understand broadly how safety and non-safety system interaction has been addressed by the licensee, to review typical details light fixtures involved, and to determine the adequacy of effort undertaken by the licensee.

The licensee indicated that a comprehensive program was conducted to review the potential for adverse interaction between safety and non-safety systems as a result of an earthquake and to eliminate those that were identified. The of falling lighting fixtures having an adverse consequence was identified and the licensee reviewed a large number of lighting fixture details throughout the plant in safety related areas. Resolution is very much dependent upon the details of the light fixture and what it orientation is with respect to fragile safety equipment. Licensee also indicated that the detailed process of checking is largely complete and in many cases chains have been provided to support the loads of light fixtures.

On December 8, 1983 staff also requested the resident NRC inspector to perform a plant walk-down of selected vital safety areas to determine the potential for falling light fixtures causing damage to the safety equipment during and following a large earthquake. Light fixtures were reviewed in 480KV Switchgear Room of Unit 1, 480V Vital Buses 1F, G, and H, Hot Shutdown Remote Control Panel, D.C. Switchgear Unit No. 1-1, Battery Room No. 1-1, D.C. Switchgear No. 1-2, Battery Room No. 1-2, D.C. Switchgear Units 1-2, 1-3, 2-3, Battery Rooms No. 2-1, 2-2, 3-1 and D.C. Switchgear No. 3-2. Also, the cable spreading rooms for both Unit 1 and Unit 2 were looked at. In various cases light fixtures are secured by chains attached at three points on the fixtures, in some cases chains are used to secure light fixtures at two attachment points. In some instances light fixtures are also supported by substantial conduits (3/4 to 1 inch in diameter) securely supported at regular intervals. In all cases reviewed, it was judged that no potential for any harmful interaction during and following an earthquake exists. The staff feels that adequate attention has been paid by the licensee to preclude adverse interaction between falling light fixtures and safety related equipment during and following a large earthquake.

Staff Position:

This issue is satisfactorily resolved pending completion of the safety and non-safety system interaction program.

Action Required:

Written Confirmation of a satisfactory completion of the safety and non-safety system interaction program, particularly with respect to the potential for light fixtures falling and causing malfunction of safety related equipment, is required prior to reactor power ascension beyond 5%.

Task: Allegation No. 37

ATS No.: RV 83A41 BN No.: 83-169 (10/20/83)

Characterization

The solid state protection system (SSPS) relays that initiate closure of RHR letdown isolation valves 8701 and 8702 perform no safety function, reduce the reliability of the RHR system, and cause a potential for RHR pump damage. Therefore, these relays should be removed.

Implied Significance to Plant Design, Construction, or Operation

The RHR letdown line contains two isolation valves (8701 and 8702) in series that are normally closed during power operation. These valves are opened when entering Mode 4 (hot shutdown) to allow the RHR pumps to take suction from the reactor coolant system (RCS) to the RHR heat exchangers for decay heat removal. Both valves 8701 and 8702 are interlocked so that they will automatically close to isolate the RHR system from the RCS if RCS pressure increases to a pre-determined setpoint. This automatic isolation function (performed by the Westinghouse designed SSPS) is provided to protect the low pressure RHR system piping from higher RCS pressures. Isolation is accomplished using a "fail safe" design (i.e., on a loss of SSPS power, valves 8701 and/or 8702 will automatically close). The concern here is that a loss of SSPS power will cause an unwanted (spurious) isolation of the RHR letdown line causing eventual RHR pump damage assuming no operator action.

Assessment of Safety Significance

Isolation of the low pressure RHR system from the high pressure RCS must be provided to prevent RHR system overpressurization that could potentially result in a loss of coolant accident (LOCA) outside containment. Therefore, RHR letdown line isolation is a safety function. The SSPS, including relays, which performs this function is safety related and designed to Class 1E requirements. Both valves 8701 and 8702 are provided with this automatic closure interlock on increasing RCS pressure so that a single failure will not prevent RHR letdown line isolation. Therefore, the relays used to initiate closure of these valves are essential and should not be removed.

Diverse indications and alarms are provided in the control room (including a RHR system low flow alarm to be installed during the first refueling outage) to allow the operator(s) to assess RHR system status and to alert them to potential system degradation. Technical Specification surveillance requirements at Diablo Canyon include periodic verification of RHR system flowrate when using the RHR letdown line. In addition, diverse means of decay heat removal (i.e., reactor coolant loops) can be readily made available should the RHR letdown line be inadvertently/spuriously isolated.

Based on the above, the staff concludes that the existing SSPS design regarding RHR letdown line isolation is acceptable.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

Task: Allegation or Concern No. 38

ATS No. RV83A47 BN No. 83-169 (10/20/83)

Characterization

PG&E is ignoring evidence that the spurious closure of a motor operated valve is not "impossible."

Implied Significance to Plant Design, Construction or Operation

The allegation suggests the licensee has not satisfactorily analyzed operational data.

Assessment of Safety Significance

The alleger has described operating events at the Diablo Canyon facility and other Westinghouse facilities during which motor operated valves in the residual heat removal (RHR) system have, upon spurious initiation of their automatic closure circuitry, moved from the normally open position (for RHR operation) to the closed position, these presenting the potential for damage to RHR pumps.

The staff has examined in depth the licensee's actions in response to an event involving the spurious initiation of RHR motor operated valve closure as well as the concerns expressed by the alleger regarding the potential for such event, and concluded that timely evaluation and corrective measures were taken to preclude repetition of such conditions. (See Allegation or Concern Nos.: 42 & 44).

Staff Position

The staff's position regarding the interlock cricurity which causes automatic closure of the RHR isolation valves is duscussed in Allegation or Concern No. 45. It does appear that the licensee is giving proper attention to the spurious closure of the valves in question.

Task: Allegation #39

ATS No.: RV 83A47 BN No: 83-169 (10/20/83)

Characterization

There is no control room annunciation provided to alert the operator(s) when the RHR letdown line has been isolated during Modes 4, 5, and 6 (hot shutdown, cold shutdown, and refueling respectively).

Implied Significance to Plant Design, Construction, or Operation

During modes 4, 5, and 6 the residual heat removal (RHR) system is aligned in the shutdown cooling mode by taking suction from reactor coolant system (RCS) loop 4 through the RHR letdown line to the RHR pumps. The RHR pumps direct flow through the RHR heat exchangers for decay heat removal via the component cooling water (CCW) system, and then back to the RCS cold legs. There are two isolation valves (8701 and 8702) in series located in the RHR letdown line. If one of these valves should inadvertently close, RHR pump suction would be lost. The concerns here are loss of decay heat removal capability and potential damage to the RHR pumps. It has been estimated that pump damage could occur as soon as 10 to 15 minutes following a spurious isolation of the RHR letdown line.

Assessment of Safety Significance

For those modes of operation where RHR shutdown cooling is used, only one RHR train or one filled reactor coolant loop is necessary to provide sufficient decay heat removal capability. The Diablo Canyon Technical Specifications require either two RHR trains be operable and/or two filled reactor coolant loops be available in order to allow for single failures. If both RHR trains are being used and the RHR letdown line becomes isolated, the operator(s) would have sufficient time to fill at least one coolant loop (assuming no loops are filled) for decay heat removal. Control room indications of loss of decay heat removal include RCS temperature, RHR system flow, and RHR pump discharge pressure. With less than the required number of reactor coolant loops and/or RHR trains operable, the Technical Specifications require immediate corrective actions to return the required loop/train to operable status as soon as possible.

Indication provided in the control room of RHR letdown line isolation includes position indication for valves 8701 and 8702 (red and green position status lights next to the valve control switches on the main control board) as well as RHR system flow, pressure, and pump status information. Although these features do provide a capability to assess RHR system status, the staff has recognized the need for installation of a RHR low flow alarm. Accordingly, the licensee is required to install a RHR low flow alarm during the first refueling. This requirement is documented in Supplement No. 13 of NUREG-0675, "Safety Evaluation Report related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2." The staff has concluded that the existing control room indications and procedures are sufficient to assure adequate decay heat removal in the interim.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None.

Task: Allegation #40

ATS No.: RV83A 47 BN No.: 83-169 (10/20/83)

Characterization

The question raised was with regard to whether or not the single RHR pump suction line from the RCS hot leg meets safety related standards. The newer PWRs are designed with redundant RHR pump suction lines from the RCS hot legs.

Implied Significance to Plant Design, Construction or Operation

The RHR suction line from the RCS hot leg in Diablo Canyon contains two isolation valves (8701 and 8702) in series that are normally closed during power operation. When the RHR system is operated as a part of the ECCS, the RHR pump suctions are aligned with either the RWST or the containment emergency sumps. The RHR suction line from the RCS hot leg is only used during modes 4 (hot shutdown while RCS temperature is less than 323°F), 5 (cold shutdown) and 6 (refueling). A postulated failure of either isolation valve (8701 or 8702) in the RHR suction line to open during plant shutdown could prevent the plant from reaching a cold shutdown condition.

Assessment of Safety Significance

In the Diablo Canyon SER Supplement No. 7, the staff states that the single RHR suction line from the RCS hot leg was acceptable. The staff conclusion was based on the following:

(1) The Diablo Canyon design has a safety related Auxiliary Feedwater System (AFWS). The condensate storage tank is the primary source of AFW with about an 8 hour water supply. In order to ensure the capability to remove heat via the steam generators for extended periods, provisions have been made to connect the raw water reservior to the suction line or the AFW pump. This will provide enough AFW to allow an additional 100 hours of steam generator operation for both units.

(2) The licensee has indicated that the combination of a mechanical failure of the RHR isolation valves and an earthquake results in a risk of about 10% of the core melt risk from all causes calculated in the Reactor Safety Study.

Branch Technical Position RSB 5-1 was not approved at time SSER No. 7 for Diablo Canyon was issued. In accordance with the implementation schedule of BTP RSB 5-1, the Diablo Canyon Units are considered class 2 plants which are not required to fully implement this BTP. Table 1 of BTP RSB 5-1 shows what is necessary to be implemented for class 2 plants. A single RHR suction line from the RCS hot leg is considered acceptable for a class 2 plant as long as a single failure could be corrected by manual actions inside or outside of containment, or the plant could be returned to hot standby until manual actions (or repairs) are accomplished. (page 5.4.7-16 of SRP 5.4.7). Also, BTP RSB 5-1 for class 2 plants requires that the RHR isolation valves have independent, diverse interlocks to protect against one or both valves being open during an RCS pressure increase above the design pressure of the RHR system. There was no assessment of the degree of compliance of the Diablo Canyon design against BTP RSB 5-1 documented in any staff SSER.

Based on the above facts, the staff evaluation of the subject allegation is as follows:
The RHR suction line from the RCS hot leg is not required for ECCS functionability. The RHR pumps take suction from RWST or containment emergency sumps, and serve the ECCS function during a LOCA. The suction line from RCS hot leg is used only for modes 4 (323°F), 5 and 6. GDC 34 of Appendix A to 10CFR 50 requires that the decay heat removal safety function should be accomplished assuming a single failure. THe Diablo Canyon design complies with this requirement by having a RHR system plus a safety related AFWs (with steam generators and atmospheric steam dump valves). Based on the above, we conclude that the Diablo Canyon design meets GDC 34 and the intent of BTP RSB 5-1. The current RHR design is adequate for safe operation at Diablo Canyon.

The staff is currently conducting a reevaluation of the adequacy of the decay heat removal system design of all LWRs. This work is being performed as an Unresolved Safety Issue (TAP-A-45), and the Task Action Plan is projected to be complete within one year. Diablo Canyon, will be subject to any new requirements that may result from the work of TAP A-45.

Staff Position

This allegation does not involve considerations that question plant readiness for power ascension testing or full power operation.

Action Required

None

Task: Allegation or Concern No. 41

ATS No.: RV83A47 BN No.: 83-169 (10/20/83)

Characterization

The power source of certain relays is not shown on certain drawings and this caused an operational problem, the failure (closure of RHR isolation valves).

Implied Significant to Plant Design, Construction or Operation

Sufficient information may not be readily available to plant operators or maintenance personnel regarding the effects of deenergizing certain portions of plant safety related systems causing unexpected plant behavior which, in turn, can be of safety concern.

Assessment of Safety Significance

Preliminary examination by the staff of the drawings and circuit schematics of concern to the alleger revealed that a detailed review of several drawings, circuit diagrams, and logic diagrams is necessary to fully comprehend the effect of the removal of power to the SSPS output relays. This removal of power can cause this RHR hot leg suction valves to close, resulting in potential damage to safety-related RHR pumps, and a condition which may not be detectable by operators in the control room.

The alleger's specific concern is that removal of power to a portion of the SSPS on September 29, 1981 did result in unexpected closure of the RHR isolation valves with an RHR pump running. (See Allegation or Concern No. 44).

Examination of facility records and discussions with licensee personnel knowlegable of the circumstances of the event of September 29, 1981 revealed the following. In preparation for "trouble-shouting" the cause of apparent power supply difficulties in a portion of the SSPS, a "...Clearance Request and Job Assignment Sheet" was prossed and approved, as required by plant administrative procedures, to authorize such activity. Subsequent disablement of the power supply (removal of a fuse) caused automatic closure of the RHR isolation valves thus interrupting RHR system flow. Initiation of the closure of the RHR valves had not been anticipated by either the operation supervisor or maintenance personnel involved in the activities Operations personnel did respond to the unpected closure of the RHR isolation valves in a resonably timely manner such that the RHR pump continued to operate without flow for approximately five minutes. The pump substained no detectable damage in this instance.

It was also revealed in discussions with licensee personnel that a simplified sketch of the RHR initiation circurtry has been constructed to clarify interactions between various components previously shown only on individual plant drawings and circuit diagrams. The construction of this simplified sketch has resulted in a much improved understanding of the cricuitry by the plant's maintenance as well as operations personnel.

Staff Position

Activities involving maintenance or texting of systems associated with the nuclear plant should be planned in advance sufficiently to anticipate the repsonse of such systems when these activities are undertaken. Adequate preplanning measures in this regard appear not to have been taken by the licensee in this instances. However, measures have been taken by the licensee to preclude a repitition of the specific occurrence in this instance.

No further specific action is required. The staff will focus attention in this inspection program to the preplanning and procedural precautions established by the licensee in carrying out maintenance and testing activities of a similar nature in the future.

Task: Allegation or Concern No. 42

ATS No.: RV83A47 BN No.: 83-169 (10/20/83)

Characterization

Licensee management was unresponsive to recommendations to prevent spurious closure of the isolation valves on the residual heat removal (RHR) system. Closure of the valves disables operation of the RHR system for decay heat removal.

Implied Significance to Design, Construction or Operation

A lack of appropriate response by the licensee, could indicate an undesirable level of management sensitivity toward employee concerns and recommendations aimed at improving operation of the reactor facility.

Assessment of Safety Significance

Facility records were examined, discussions were held with facility personnel, and observations were made by the staff. Periodic discussions were also held with the alleger. Since the alleger's concerns had been examined by Region V inspectors previously, reports of prior inspections were reviewed and discussions were held with Region V inspectors relating thereto. In addition to the specific concern (or allegation) characterized above, other concerns of the alleger, as discussed below, were also examined.

The alleger had documented concerns regarding spurious closure of the RHR isolation valves because of certain steps in an emergency operating procedure related to safe shutdown from outside the control room. The licensee's response consisted of the initiation of a nuclear plant problem report, and investigation of the alleger's concern. The licensee's resolution to the concern was to revise the emergency procedure.

A design change request (DCR) authored by the alleger addressed the alleger's more general concern of potential for inadvertant closure of the RHR isolation valves. A revision to the DCR was subsequently initiated by the alleger providing the Licensee Event Reports (LERs) of other facilities relating to instances of RHR system disablement due to spurious closure of the isolation valves similar to those which were the subject of the alleger's concern.

The alleger's preliminary evaluation of the DCR determined that the requested change involved an unreviewed safety question requiring prior NRC approval in accordance with 10 CFR 50.59. The DCR is still under consideration by the licensee's engineering department, the plant operating department and Westinghouse.

Preliminary discussions have been held between the licensee, Westinghouse and the NRC staff relating to an informal proposal by the licensee (supported by Westinghouse) to remove the RHR interlock circuitry from the Diablo Canyon facility. The proposals and actions required to resolve this DCR are still open.

The staff determined that other procedural changes have been made by the licensee in an effort to preclude closure of the RHR isolation valves from spurious actuation of the interlock circuitry.

The staff also reviewed a concern documented by the alleger in a memo in April 1981 to plant engineering regarding reactor coolant pump bearing oil level annunciators. In postulating a tube failure in the lube oil heat exchanger, the view was expressed that an incorrect alarm response procedure may lead the operator to take improper action. Written acknowledgement of the alleger's concern was provided by a plant engineer in June 1981, indicating that the procedures manual was being revised to resolve the concern. The alleger observed approximately eight months later, that no change to the Plant Procedures Manual had been made. The alleger documented this observation by an additional memo. The same plant engineer who had previously responded to the alleger responded to this memo. The engineer explained that the Plant Manual had been the subject of an extensive revision effort for the past year and all changes resulting from this effort were to be incorporated into the Manual "... in one major revision" which would be published "... definitely prior to low power physics testing." A major revision, which included the alleger's initial comment, was subsequently made to the Manual in September 1983.

During the intervening period between the time of the alleger's second memo and implementation of the major revision to the Manual NRC resident inspectors pursued the alleger's concern with licensee personnel. In response, the licensee implementing a temporary change to the specific procedure of concen. This temporary change was accomplished by the issuance of a Procedure ON-THE-SPOT Change in early 1983.

Staff Position

A period of approximately 2 years appears to be excessive in attempting to resolve the RHR concerns of the alleger. The issue is not yet fully resolved. However, unusual circumstances did exist in that resolution of the alleger's concerns regarding the RHR system and his specific recommendation to remove the interlock circuitry involve substantial safety analyses by the licensee, as well as NRC staff review and approval. In the interim, procedural changes had been implemented by the licensee which had substantially addressed the concern of the alleger. A similar period, approximately 2 1/2 years, to formally address the alleger's concern regarding the accuracy of an annunciator response procedure also, under normal circumstances, appears excessive. In this instance, however, the unusual circumstances of a major revision to the procedures manual was in progress.

It is the judgment of the staff that there is not a prevailing attitude by licensee management which in itself discourages employees from expressing concerns or making recommendations for improvement in facility operations.

Action Required

The Region V staff will give particular attention in its ongoing routine inspection program to evaluate the performance of licensee management in this area.

Task: Allegation or Concern No. 43

ATS No. RV83A47 . BN No. 83-169 (10/20/83)

Characterization

The loss of the residual heat removal (RHR) system on 9/29/81 due to unplanned closure of the RHR isolation valves was an event which should have been reported to the NRC in accordance with 10 CFR 50.72. The licensee's failure to make such a report was in violation of NRC regulations.

Implied Significance to Design, Construction, or Operation

The failure of the licensee to report this occurrence, would indicate a deficiency in the licensee's management control systems to provide adequate review and reporting of events to the NRC.

Assessment of Safety Significance

The circumstances associated with the event were examined by review of facility records and discussions with licensee personnel.

The loss of residual heat removal capacity during a time when significant fission product decay heat is present in the core would have safety significance. In this particular instance, fuel had not been loaded into the Diablo Canyon Unit 1. Therefore, no fission product decay heat was present and loss of RHR capability had no actual safety significance.

The intent of then applicable provision 10 CFR 50.72 of the NRC regulations was to insure that holders of operating licenses for power reactors report promptly by telephone to the NRC Operations Center significant events, such as those which involve intitiation of the licensee's emergency plan; the nuclear reactor not be in a controlled or expected condition; fatality or serious injury or radioactive contamination of personnel; or acts which seriously threaten the safety of the reactor or site personnel.

The event in question was reviewed by the staff and it was determined that this event is not required to be reported in accordance with 10 CFR 50.72. Licensee representatives did indicate that an informational report of the event was to be made in writing to the NRC.

Staff Position

The staff concludes that the event did not meet the reporting requirement of 10 CFR 50.72.

Action Required

None

Task: Allegation or Concern No. 44

ATS No. RV83A47 BN No. 83-169 (10/20/83)

Characterization

The licensee failed to properly process a Nuclear Plant Problem Report.

Implied Significance to Design, Construction, or Operation

The allegation, could indicate a weakness in the implementation of the licensee's Quality Assurance Program for Operations.

Assessment of Safety Significance

The Nuclear Plant Problem Report (NPPR) is the document used at Diablo Canyon to record events such as significant equipment failures and operational problems. The NPPR form becomes the record of the identification of a problem, its evaluation, and the action taken to correct and prevent recurrence.

On September 29, 1981, inadvertent closure of the residual heat removal (RHR) system isolation valves occurred while the RHR pump No. 1-1 was running. The alleger's concerns are that the NPPR which was initiated following this event was not processed properly in that it was, "signed off as complete without any plant management review...classified as 'non-reportable' and without any follow-up action such as an RHR pump inspection or investigation into the cause of the event."

The processing of the NPPR was assessed through an examination of facility records; discussions with facility personnel (including all those persons whose identity was provided by the alleger) and the alleger; and observations by the inspectors.

The NPPR record in question was examined. It was written on 9/21/81 and closed on 10/5/81.

The resolution of the three issues are as follows:

Signed-off without any plant management review

The inspector determined that licensee management, including the plant superintendent and operations supervisor, were involved in the review and evaluation of the NPPR.

The alleger's concern included the fact that when he examined the NPPR (after if had been completed) there was no signature to indicate the results of management's evaluation of cause and corrective action(s) taken. The alleger had called this discrepancy to the attention of a QC supervisor, who obtained the proper signature on the NPPR. When the NRC inspector examined the NPPR record (in December 1983) the Operation Supervisor's signature was found on the document. It was undated. In discussions with the NRC inspector, the Operations Supervisor stated he may have signed the NPPR after it had been closed, but he could not accurately recall the circumstances.

- NPPR classified as "non-reportable"

The inspector verified that the NPPR was in fact classified as "non-reportable" by licensee management. The classification is considered appropriate by the staff and is addressed in Allegation or Concern No. 43.

- <u>No followup action was taken into the cause of the event</u>

The NPPR indicated that a revision to operating procedures was necessary to prevent recurrence of the event, and that such a revision had been implemented. Facility records indicate that the NPPR relating to the event was the subject of review by the On-Site Safety Review Group (OSRG) on two occassions--October 19, 1981 and November 24, 1981. On October 29, 1981 the OSRG observed that the operating procedures had been changed, and that a proposed change to remove the RHR isolation valve initiating circuitry had been proposed. The latter, it was determined, was a Design Change Request (DCR) which had been initiated by the alleger (see Task Allegation or Concern No. 42). The OSRG determined that it would review the event further during a subsequent meeting. On November 24, 1981 the OSRC directed that an operational test of the RHR pump be conducted, and that the DCR not be approved since it would provide less protection for RHR over pressurization than presently existed.

<u>Staff Position</u>

The NPPR was properly processed and subsequently reviewed by the OSRG.

<u>Action Required</u>

None.

Reactor Systems Branch

TASK: Allegation #45

ATS NO.: RV 83A47 BN NO.: 83-169 (10/20/83)

Characterization:

Section 5.5. of the Diablo Canyon FSAR describes the autoclosure interlock for the RHR Suction line isolation valves (8701 and 8702). Section 3.4.9.3.a of the Diablo Canyon Technical Specifications requires power to be removed from these isolation valve operators during modes 4 (Hot shutdown when RCS cold leg temperature is less than 323°F), 5 (cold shutdown) and 6 (refueling). This requirement defeats the function of autoclosure interlock for the valves.

Implied Significance to Plant Design, Construction or Operation

As the result of Technical Specification Section 3.4.9.3.a, the isolation valves (8701 and 8702) will be left in an open position with power removed during low pressure/temperature operation of the plant. The automatic closure interlock to these isolation valves causes them to lose their design function. This will result in a situation in which there is in sufficient isolation capability feature to prevent an intersystem LOCA between the high pressure RCS and the low pressure RHR system.

Assessment of Safety Significance

Section 5.5 of the Diablo Canyon FSAR states that during low pressure/temperature operation, the isolation valves (8701 and 8702) between the RCS and the suction of the RHR pumps are interlocked with a pressure signal to automatically close the valves whenever the RCS pressure increase above approximately 600 psig. Section 3.4.9.3.a of the Diablo Canyon Technical Specification requires the RHR system isolation valves

(8701 and 8702) to be open with power removed from the valve operators while the positive displacement charging pump is in operation. The applicability of the T.S. is during mode 4 when the temperature of any RCS cold leg is less than or equal to 323°F, mode 5, or mode 6 with the reactor vessel head on this Technical Specification requirement defeats the automatic closure interlock function as designed.

Power removal from valves 8701 and 8702 while the RHR system is operating was required by the staff as the result of a meeting with the licensees on RCS low temperature overpressure protection (LTOP) and RHR pump protection concerns. Since the Diablo Canyon design has only one RHR suction line from the RCS, a spurious automatic closure of the isolation valve would result in loss of RHR pump suction flow and would result in a RCS pressurization as a result of the loss of letdown flow. However, there was no documentation (SSER, letter or meeting minutes) of the staff's basis for requiring power removal from those isolation valves during modes 4, 5 and 6.

In the Diablo Canyon SER Supplement No. 13, section 6.3. (ECCS), dated April 2, 1981, the staff concluded that the licensee should be required to provide an alarm to alert the operator to a degradation in ECCS (during long term recirculation). A low flow alarm was stated to be an acceptable method to satisfy this concern and the staff indicated that an alarm should be installed at the first refueling outage. Until then, procedures and dedicated operators were to be implemented during long term recirculation to manage and monitor ECCS performance. There was no documentation to indicate that the licensee committed to this staff position, nor was this staff position included in the Diablo Canyon low power license. SRP 5.4.7 (BTP RSB 5-1) requires an autoclosure interlock on the RHR suction line isolation valves. Without power to the valve operators, the autoclosure function is defeated.

Based on the above facts, the staff evaluation of the subject allegation is as follows:

Without power to the isolation valve operators, the plant design does not conform to BTP RSB 5-1, Position B.1.C, for the requirement of autoclosure interlock. By having power available to the isolation valves during shutdowns ensures an event V (intersystem LOCA) will not occur as a result of the operator failing to close both isolation valves during a return to power.

With power on the isolation valves, a spurious closure of the isolation valves would result in a loss of suction flow to the RHR pumps. However, the low flow alarm discussed in SSER No. 13 would enable rapid operator detection and mitigation. The licensee has informally indicated that a minimum of 10 minutes without adequate suction pressure would be available without pump damage. Also, there are numerous indications available to alert the operator to improper RHR valve alignment (A list is provided in staff evaluations to allegation No. 37 and 39).

<u>Staff Position</u>

To implement the staff position stated in SSER No. 13, the installation of a low flow alarm for RHR pump protection is being considered as a license condition in the Diablo Canyon full power license. Additionally, it is the staff position that power be available to the RHR MOVs when in a shutdown condition. However, there is a question as to when these requirements should be implemented. If the low flow alarm were not installed until the first refueling outage, reinstating power to the RHR MOVs in the meantime would result in the autoclosure interlock being enabled to provide protection against intersystem LOCA

However, the chances of spurious autoclosure and consequent loss of RHR suction pressure (without the low flow alarm) and of an overpressure event would be increased. If power restoration to the RHR MOVs were not implemented until the low flow alarm is installed at the first refueling outage, the chance of loss of RHR suction in the interim is reduced but there is a possibility of an intersystem LOCA. To determine which option results in the safest operation of the plant, the staff considered the following:

1. During the first cycle of operation, plants operate more frequently on the RHR system as a result of maintenance, testing and training requirements for a new plant. Thus, the period of vulnerability to a spurious RHR suction MOV closure may be greater than in subsequent cycles.
2. The RHR relief valve would open to relieve pressure if a plant startup were attempted with both RHR MOVs open. It is not, in the staff's judgment, credible to postulate plant startups with both MOVs left open. The operator would have to shut at least one MOV to continue the plant startup.
3. Failing to close the second RHR suction MOV would not, in itself, result in an intersystem LOCA. The first MOV must also fail. The

first MOV can fail in either of two ways by either the "open permissive" interlock failing along with the operator reinstating power to the valve, (it is required to be de-energized) then attempting to open the valve. The second mode of failure would be for the valve to rupture in such a way that flow between the two systems occurred. Both of these failure modes are judged to have an extremely low probability. However, the consequences of an intersystem LOCA could be severe.

4. There have been many occasions of spurious RHR suction valve closures on operating plants. This has resulted in not only a loss of decay heat removal, but also an overpressure event due to the loss of the letdown flowpath.

ACTION REQUIRED

Based on the above factors, the staff believes the best course of action is to continue the current technical specification for power to be removed from the RHR MOVs during Modes 4, 5 and 6 until the low flow alarm is installed. However, the staff position that would permit the licensee to wait until the first refueling outage before installing the low flow alarm was taken over two years ago. Staff will puruse with the licensee a commitment to a schedule for accomplishing this installation at the earliest possible time. In the interim, until the low flow alarm is installed, the staff believes that strict administrative controls should be developed and implemented to ensure that MOVS 8701 and 8702 are closed with power removed during plant startups when RCS pressure is above the RHR design pressure.

Task: Allegation or Concern No. 46

ATS No.: RV83A46 BN No.: 83-164 (10/27/83)

Characterization

A site contractor (H. P. Foley) has incorrect procedures for voiding nonconformance reports.

Implied Significance to Plant Design, Construction, or Operation

The site contractor in question (H. P. Foley) has been responsible for installation and modification of electrical, civil and mechanical design class 1 safety systems and/or structures which are necessary for the safe operation and shutdown of the plant. (See Task Allegation or Concern No: 24)

Assessment of Safety Significance

See Task Allegation or Concern No. 24

Staff Position

See Task Allegation or Concern No. 24

Action Required

See Task Allegation or Concern No. 24

Task: Allegation or Concern No 47

ATS No. RV83A34 BN No. N/A

Characterization

The licensee has not provided a plant voice paging/announcing system at the Diablo Canyon plant. Diablo Canyon is unique in this regard, since staff's experience is that other plants have such a system. The Joint Intervenor, in meeting with the staff and PG&E on September 6, 1983, expressed the view PG&E had placed this item among others " on the back burner."

Implied Significance to Design, Construction, or Operation

There is no specific NRC requirement for such a system. NRC experience, however, is that such a system provides/enhances communication, particularly when responding to unusual or emergency events.

Assessment of Safety Significance

A plant paging/announcing system could substantially improve communication in the plant and provide a more coordinated response of the operating crew during off-normal and/or emergency events in the operating plant.

The staff (both RV and NRR) has on several occasions discussed with the licensee NRC management's concern about the lack of plans for such a system at Diablo Canyon.

On December 6, 1983 the NRC was informed by senior licensee management that PG&E had decided to install a plant paging/announcing system at Diablo Canyon. The target date by the licensee is to have the system installed and operational by fuel loading on Unit 2. This commitment was confirmed by letter.

Staff Position

The staff has strongly urged the installation of this system by the licensee at Diablo Canyon. The licensee's commitment to install the system by fuel loading on Unit 2 appears reasonable and is satisfactory to the staff.

Action Required

Region V will follow the licensee's commitment and verify installation of the paging/announcing system.

Task: Allegation #48

ATS. No. RV 83A34

Characterization:

Status of Seismic Systems Interaction Study

Implied Significance to Plant Design Construction or Operation

The allegation that the safety of fuel loading and operations cannot be assured prior to completion of the modifications from the seismic systems interaction study is not significant to either fuel loading or operations, because: (a) the completion of the modifications prior to fuel loading is not required for safety, and (b) the completion of the modifications prior to operations is required, and all indications are that the modifications will be completed prior to operations.

Assessment of Safety Significance:

The staff has re-examined both the status of the seismic systems interaction study and the activities related to the allegation that: "The safety of operations is not assured if fuel load and operation of the plant occur before the seismic interaction study and associated modifications are complete."

During the re-examination the staff have assumed that "the study" is the P.G.&E. Seismically Initiated Systems Interaction Study with all its aspects including the criteria for postulating systems interactions. Also, it was assumed that "operation of the plant" means thermal power greater than five percent of design power.

Section 8.2, Supplement 11 to NUREG 0675 (SER) states the commitments pertinent to this allegation:

(a) "P.G.&E. will complete their program and any necessary plant modifications for each unit prior to the issuance of any license authorizing full-power operation of that unit."

(b) Region V, OIE, will verify "the completion of P.G.&E.'s program and the acceptability of any plant modifications."

(c) "P.G.&E. will ...provide for our information copies of their final report of their program which will include an identification of all interactions postulated, all walkdown data, interaction resolutions, and technical reports."

The important point to note is that no power operation of the plant will be authorized before the modifications are complete.

Although fuel loading is important to safety in other ways, it is not necessary to complete the modifications associated with the P.G.&E. study before loading fuel. Fuel loading, and its completion, means that only new fuel elements have been positioned in the reactor. Sustained fission has not occurred, therefore, fission products do not exist in the core in sufficient amount to require decay heat removal.

As additional safety precautions, P.G.&E. (in a letter dated September 10, 1983) states that (a) no modifications will be made inside containment during fuel loading, (b) during the period when the modifications are being made the plant will be in modes 5 and 6 (cold shutdown and refueling), (c) no modifications will be made to those systems or portions of systems required by Technical Specifications to be functional during these modes of operation, (d) post fuel-loading modifications will not be undertaken until the reactor vessel head and missile shield are in place to provide protection of fuel from any modification activity, and (e) the modifications will be completed prior to the first reactor criticality. Note that all such post fuel-loading work will be reviewed for the introduction of new interactions under P.G.&E.'s study.

The P.G.&E. study, as we accepted it, provides for follow-on activities during power operations to remain alert for adverse systems interactions. These follow-on activities should not be confused with the completion of the modifications identified during the pre-operating period. The follow-on activities provide for responsiveness to those adverse systems interactions that might be identified subsequently.

The staff review of the P.G.&E. reports provides an independent check of the P.G.&E. study. The staff review will provide assurance against adverse systems interactions from Hosgri events at Diablo Canyon and will consider the potential for generic implications from the findings of the P.G.&E. study.

By a letter dated October 13, 1983, P.G.&E. submitted an information report on the status of their seismic systems interaction study within the containment of Unit 1. Included in the Information Report was the preliminary status of their study of Unit 2. P.G.&E. has not yet completed its study of Unit 2 and the staff has not yet completed its review. However, neither P.G.&E. nor the staff has yet identified any seismically induced systems interaction that consists of a violation of the regulatory criteria within the applicable sections of the Standard Review Plan (NUREG-0800).

In summary, the staff concluded that power operations should be authorized only after all modifications are completed. It is not necessary to complete all modifications prior to fuel loading. Precautions are being taken to assure that the fuel loading is not vulnerable to modifications associated with the P.G.&E. systems interactions study. The safe operation of Diablo Canyon is not jeopardized by the seismic sytems interaction study and its associated activities.

Staff Position

Based on our review of the P.G.&E. seismic systems interaction study description, a site visit to observe the conduct of the system interaction walkdowns, the precautions being taken and the minor nature of the post fuel-loading modifications as described in the September 10, 1983, P.G.&E. letter, and the commitment to complete these modifications prior to taking the reactor critical for the first time, the staff concluded that it is not necessary to complete all modifications prior to loading fuel. We require that any necessary modifications for each unit be completed prior to issuing a license authorizing full-power operation of that unit.

Action Required

No new action is required in response to this allegation. The commitments identified in Supplement 11 to NUREG 0675 (SER), Section 8.2, continue to be required actions from our previous licensing review of the P.G.&E. seismic systems interaction study.

Task: Allegation or Concern No. 49

ATS No. RV83A34 BN No. N/A

Characterization

The Mothers for Peace representatives stated during an interview with NRC representatives that "Emergency Sirens are not seismic qualified."

Implied Significance to Plant Design, Construction, or Operations

The implied safety significance of this allegation is that the emergency sirens may not operate during a seismic event, which would have a detrimental effect on area evacuations during accident conditions.

Assessment of Significance

In a memorandum and order (CLI-81-33) regarding the San Onofre Nuclear Generating Station, dated December 8, 1981, the Commission decided that its regulations do not require consideration of the specific impacts on emergency planning of earthquakes which cause or occur during an accidental radiological release.

Staff Position

Emergency sirens are not required to be seismically qualified.

Action Required

None.

Task: Allegation or Concern No. 50

ATS No: RV83A34 .

Characterization

The Allegation states that the security plan should have been maintained and that imposing security just thirty days prior to fuel load is inadequate when one considers that there were several thousand workers onsite, one actual sabotage event, and many bomb threats.

Implied Significance to Plant Design, Construction or Operation

The Allegation implies that the staff's action in temporarily suspending the security plan at Diablo Canyon increased the opportunity for plant sabotage, thereby possibly affecting the plant's operability.

Assessment of Safety Significance

Upon receipt of a low power operating license in September 1981, Pacific Gas and Electric Company (PG&E) fully implemented the approved Diablo Canyon Physical Security, Safeguards Contingency, and Guard Qualification and Training Plans in accordance with the introductory paragraph of 10 CFR 73.55. On February 25, 1983, PG&E applied for an Amendment to their Facility Operating License No. DPR-76 that would authorize the temporary suspension of Section 2.E of the license relating to physical security. On March 11, 1983, the NRC issued Amendment No. 4 to the Facility Operating License that exempted the licensee from the requirement to maintain in effect the approved plans for a period ending thirty days prior to fuel loading, except for certain commitments regarding guard training, access to the plant protected areas, the intrusion alarm system, and protection of the fresh fuel on site.

The technical bases for issuing Amendment No. 4 are set forth in the notice of Exemption (48 FRN 12017, March 22, 1983). Among other things the Exemption statement notes that (1) there is no current potential for radiological sabotage at the facility since the reactor has never be operated and there is no irradiated fuel onsite, and (2) the licensee has committed to an extensive return - to - service alignment, test and inspection program of both vital plant equipment and the instrusion alarm system to insure that sabotage has not taken place and that sabotage materials have not been introduced into vital areas. The staff's action in this case was consistent with the intent of NRC physical security regulations in that there are no requirements for protecting the plant or equipment during the construction phase (from a practical standpoint Diablo Canyon was still in the construction stage in March of 1983). Nevertheless, the staff compiles data on security-related incidents at construction sites and reports the findings in NUREG-0525, Safeguards Summary Events List (SSEL). It is noted that vandalism, property damage and bomb threats are not uncommon at facilities being built, and the frequency of such events at Diablo Canyon after the exemption was issued was not unusual. However, one event, the apparently deliberate gouging of a reactor coolant pipe in April, was considered significant by both the utility and the NRC, and PG&E voluntarily increased surveillance and control in work areas. The staff reviewed the incident in connection with the advisability of continuing the

Exemption and concluded that there was no need to reinstate full scale security at the site. The conclusion was based on the following considerations:

- The damage was discovered shortly after it was inflicted by a routine inspection prior to the re-installation of the thermal insulation. This supported the view that the pre-start inspection program would be an effective technique in detecting acts of sabotage prior to start-up.
- The nature of the damage, which was described as hazard, did not indicate the presence of a sophisticated threat on site. It was more characteristic of vandalism.
- There were no other similar incidents reported, suggesting that the pipe damage was a random event and not part of a larger scenario.
- The incident had the positive effect of increasing security awareness at the site and highlighted the need for a thorough and extensive return-to-service effort regarding security matters.

Staff Position

The staff has concluded that temporarily exempting the licensee from maintaining the Physical Security and Safeguards Contingency Plans did not significantly increase the risk of radiological sabotage during the exemption period or over the life of the facility.

Action Required

None

Task: Allegation or Concern No. 51

ATS No: RV-83-A-0034 BN No:

Characterization

In a September 7, 1983 meeting among representatives of the staff and the joint intervenors, the representative of the joint intervenors expressed concern that plant personnel are reluctant to come forward with safety concerns because their candor endangers their jobs and may subject them to public ridicule even if their allegations are true.

Implied Significance to Plant Design, Construction, or Operation

Plant workers with knowledge of potential safety problems may be reluctant to speak of those problems, thereby resulting in a potential reduction in the flow of important safety information to the NRC.

Assessment of Safety Significance

During the staff team inspection of allegations conducted at the site from November 29, 1983 through December 8, 1983, NRC staff members interviewed over 158 site personnel representing the licensee and major contractors.

The interviewees were selected at random during the conduct of the inspection and represented most of the disciplines on site the majority of which were engineers (30 plus) quality assurance (50 plus) personnel, and crafts workers

(25 plus). Others included records clerks, purchasing agents and a variety of other personnel.

Our inspectors conducting the interviews stated it was their experience that the site personnel were not reluctant to talk to them candidly as evidenced by some new concerns which were expressed by the site personnel. In general those personnel who did have concerns did wish to have their identifies kept confidential.

Staff Position

Section 210 of the Energy Reorganization Act of 1974, as amended, prohibits discrimination against employees of a licensee or its contractors and agents for communicating safety information to the NRC. The statute affords employees a direct remedy against the employer for such discrimination through which the employee may be awarded, for example, reinstatement and back pay. NRC regulations (10 CFR 50.7) also prohibit licensees of production and utilization facilities and their agents from engaging in such discrimination. A violation of these regulations may result in imposition of civil penalties, denial, revocation or suspension of the license, or other enforcement action. The licensee is required to post NRC Form 3 on its premises, which provides notice to workers of these protections against discrimination.

If an employee is reluctant to speak to NRC representatives because of potential public ridicule, the NRC can offer confidentiality to a person who may have relevant information. Although confidentiality is not absolute, the

NRC is prepared to maintain the confidentiality of such communications to the extent permitted by law.

Action Required

No specific actions are planned regarding this specific statement of concern by the Joint Intervenors. The staff, however, is separately examining means to improve the flow of information from workers to the staff. (Within Region V this action includes such actions as: increasing NRC inspector visibility and availability to plant workers through site postings, expanded telephone book listings, and reevaluating the locations of NRC on site offices with respect to workers access).

Task: Allegation or Concern No. 52

ATS No.: RV 83A0034 BN No.:

Characterization

In a September 7, 1983 meeting among representatives of the NRC, Licensee, State of California and the Joint Intervenors, the representatives of the Joint Intervenors stated that she was concerned that loading of fuel might be permitted before construction is completed and that permitting fuel loading before holding hearings on the safety of the facility is inappropriate.

Implied Significance to Plant Design, Construction or Operation

Permitting fuel loading before the plant is completed or hearings are held might have an adverse effect on safety.

Assessment of Safety Significance

See "Staff Position".

Staff Position

The Commission has addressed the question of whether fuel loading should be permitted in its Memorandum and Order (CLI-83-27, at pages 6-7) dated November 8, 1983. The Commission decided to reinstate the Licensee's authority to load fuel and undertake pre-criticality testing. In reaching its determination, the Commission noted that the risk to public health and safety was extremely low because no self-sustaining nuclear chain reaction would take place which would create radioactive fission products. The Commission also found that there were no significant safety concerns material to fuel loading and pre-criticality testing that would warrant continuation of the suspension of these activities. The Commission also noted that its action would not prejudice future decisions on Diablo Canyon.

Action Required

None.

Task: Allegation or Concern No. 53

ATS No.: RV 83A39 BN No.:

Characterization

Welder qualification

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 54

ATS No. 83A38 BN No. 83-170 (10/27/83)

Characterization

Electrical cable traceability has been lost for work performed both by PG&E and H. P. Foley.

Implied Significance to Plant Design, Construction, or Operation

Treacability is required by IEEE Standards to insure that cable location can be identified in case of cable failure or potential cable failure.

Assessment of Safety Significance

This item was examined by reviewing licensee correspondence and procedural controls related to cable installation, inspecting cables and electrical equipment in the field, and reviewing of cable records.

Licensee documentation, a Foley interoffice memo, identified approximately 65 circuits that may have wire traceability problems. Many of these circuits on this memo dealt with circuits that initially had a defective cable installed which was subsequently replaced with new cable. The inspector examined 53-replaced cables and their records from these circuits. The field observations were compared to cable records.

The inspector reviewed the licensee's procedures QCPE-11, Cable and Wire Termination, and QCPE-10 Power, Control and Signal Cables.

The problem cable installation records were associated with a cable removal package identified by PG&E. PG&E stated that this work was done by different organizations with each organization maintaining their own quality assurance records. This multiplicity of work performing organizations, and the associated QC activities, made record traceability difficult but the staff determined that traceability was not lost.

The staff sampled 53 cable installations. No significant deficiencies were noted. Tasks 62, and 59 are related to traceability of class 1 circuits and cable and the findings of those tasks do not indicate any loss of traceability.

Staff Position

Results do not indicate any loss of traceability. The licensee has committed to consolidate the records to simplify traceability.

Action Required

Region V staff will perform additional verification inspections of cable traceability as a part of its routine inspection program.

Task: Allegation #55

ATS. No.: RV-83A50 BN No.: 83-171 (10/27/83)

Characterization

Bechtel has purposely approved analysis of small bore pipe supports that have failed by altering current documentation that shows failure of piping systems and pipe supports.

Implied Significance to Plant Design, Construction or Operation

Small bore piping is part of various systems required to monitor the status of the nuclear steam supply system and of systems required for component cooling and safe plant shutdown. Failure of such piping or its supports could preclude accurate monitoring of NSSS status.

Assessment of Safety Significance

The NRR has received from the anonymous alleger a set of sample problems where the alterations are alleged to have occured. The details are, however, unclear and are currently under review. A NRR representative reviewed a number of small bore design packages on site in conjunction with IE representatives. It was determined that in several cases incomplete documentation transfer existed for the source of loads for which the supports were designed. The significance of this deficiency is uncertain as insufficient detail is available as to its extent to reach a conclusion. However, if inadequate documentation exists, or is totally missing it indicates that a re-review of all small bore piping may again be needed before Step 2, criticality and low power testing.

Staff Position

The apparent or potential lack of proper documentation involving the small bore pipe supports is considered a deficiency supporting the allegation.

Action Required

NRR will review a representative sample of DCP small bore pipe support design packages to assess the current quality of these designs. This sample was provided by Region V. This item will be evaluated in conjunction with items 78, 79, 82, 85, 86, 87, 88, 89, 95 and 97).

Task: Allegation or Concern No. 56

ATS No. RV83A42 BN No. N/A

Characterization

Pitting in Main Steam and Feedwater Pipe.

Implied Significance to Design, Construction, or Operation

Severe pitting in plant systems could reduce pipe and component wall thickness and thereby may increase the probability for leakage and pipe breaks.

Assessment of Safety Significance

The licensee first identified the pitting on a Unit 2 main steamline elbow impingement sleeve, initiated internal problem reports to follow the resolution, and found that the cause of the pitting was the use of glue for insulation installation in an outdoor environment. The licensee performed qualitative chemical analyses of the glue, insulation and corrosion products. This analysis identified no corrosion inducing materials. The licensee's analysis concluded that the moisture at the impingement sleeve-glue-insulation interface for a prolonged period caused the pitting. (Insulation is not part of the quality program at Diablo Canyon). A repair program was initiated and completed.

The staff reviewed the above program and examined several pipe areas containing the surface-glue-insulation interface.

The licensee initiated effort to identify all areas where a potential for this pitting exists. The licensee found that under the impingement sleeves, glue was used to attach the insulation to the pipe. The pipe under the feedwater impingement sleeve was examined by the staff. The pitting depth was on the order of 1/32" and was not as extensive as observed on the impingement sleeve. The licensee has initiated a component identification and engineering analysis to assure that the pitting has not and will not violate minimum wall thickness requirements and committed to inspect locations where the potential for pitting exists during the first refueling outage.

Staff Position

The staff concurs with the current resolution given the satisfactory completion of the actions required.

Action Required

The Region V staff will followup and verify licensee commitments to: document the program to find all areas with the pitting potential, and complete the engineering analysis of all areas with pitting potential; to include minimum wall considerations for all pitted areas, an evaluation of the pitting potential and effect on areas inside buildings, and additional investigations of pitted areas to assure that worst case component is analyzed. This will be accomplished as a part of the staff's routine program.

Task: Allegation or Concern No. 57

ATS No. RV83A57 BN No. N/A

Characterization

Prior to 1983 a site contractor (H. P. Foley) used uncertified and unqualified quality control inspectors.

Implied Significance to Plant Design, Construction, or Operation

The site contractor in question (H. P. Foley) has been responsible for installation and modification of electrical, civil, and mechanical design class 1 safety systems and structures which are necessary for the safe operation and shutdown of the plant. The use of unqualified inspectors would raise questions as to the adequacy of installations.

Assessment of Safety Significance

This item was reviewed by examination of PG&E and HPF commitments and procedures, interviews of personnel, examination of training and qualification records, and review of nonconformance reports.

In the early 1970s Foley had no formal procedures regarding the determination or documentation of inspector certification or qualification. During this time the Foley Company was only required, consistent with 10 CFR 50, Appendix B, to perform an on-the-job training program to qualify QC inspectors. Staff inspections show that Foley complied with this requirement. A procedure was issued September 25, 1979 addressing, the qualification/certification of civil inspectors. This procedure was superceded by another procedure on April 25, 1980, which required indoctrination and training, but did not provide for formal documentation of qualification/certification. On July 14, 1981, the licensee, in response to NRC generic letter 81-01, committed to implement, with minor exceptions, ANSI N45.2.6, for quality control inspectors, and ANSI N45.2.23, for quality assurance auditors, prior to full power licensing of Unit 1. On December 7, 1982, in response to a licensee audit in August 1982, Foley generated a new procedure providing for the qualification and certification of quality control inspectors and supervisors imposing ANSI N45.2.6 criteria. Beginning in 1983, Foley QC inspectors have been qualified/certified to the standards of ANSI N45.2.6. The above conclusions are based upon the findings of the December 1983 NRC team inspection and the findings of an earlier inspection documented in Inspection Report 50-275/83-13.

The staff interviewed twelve HPF electrical inspectors and eight HPF civil/structural inspectors. Interviews with HPF management corroborated that there was no ANSI-type program until 1979 and that an effective program was only implemented in April of 1983. This lack of full effectiveness was previously detected by the staff and is addressed in NRC inspection reports (e.g., 50-275/83-13). The present program is administered by the Quality Analysis Section and requires a background search and on-the-job training for each potential inspector.

The staff further examined this area by reviewing a sample of 60 work packages which related to activities performed in late 1979 and later. This review identified eleven instances where inspections had been performed by individuals who did not have appropriate certification records on file during the 1979 to early 1982 timeframe (this is an apparent item of noncompliance). This finding warrants further examination to establish whether this condition is widespread and to determine the consequence of instances of this type.

It is noted that this topic has been the subject of previous reviews wherein the licensee performed 100% reinspections of structural work performed in late 1982 and 1983 and 10% reinspection of electrical work performed in late 1982 and 1983. The boundaries put on thes reinspection program were based upon rapid expansion of the work force, inspectro qualification errors, and material defect which were detected during this timeframe (late 1982 and 1983). Considering the current findings of untimely certification of certain inspectors, it appears that expanded reviews are warranted.

Staff Position

a. Lack of an ANSI N45.2.6 type qualification program:

The staff concludes that the H. P. Foley Company was not comitted to and did not have an ANSI N45.2.6 type qualification/certification program for inspectors up to late 1979. Although much of the construction was inspected by inspectors without benefit of a formal ANSI certification, there have been substantial reinspections, as-built reviews (external and independent), circuit continuity tests, and preoperational functional acceptance tests that provide additional assurance of the quality of the plant hardware.

b. Untimely certification of Inspectors:

Additional reviews are required to assess whether this condition is widespread and to determine the consequences of these conditions.

Action Required

The licensee has been requested to initiate an expanded examination of H. P. Foley inspection activities. The staff will monitor this work to assess the scope and consequences of untimely certification of inspection personnel.

Task: Allegation or Concern No. 58

ATS No.: RV83A57 BN No.: N/A

Characterization

A site contractor (H. P. Foley) allows the use of Phillips Red Head anchor studs, many of which are reported to be improperly installed and are subject to frequent dislodging.

Implied Significance to Plant Design, Construction, or Operation

See Task Allegation or Concern No. 25.

Assessment of Safety Significance

See Task Allegation or Concern No. 25.

Staff Position

See Task Allegation or Concern No. 25.

Action Required

See Task Allegation or Concern No. 25.

Task: Allegation or Concern No. 59

ATS No.: RV83A57 BN No.:

Characterization

The site electrical contractor (H. P. Foley) has lost the traceability of installed electrical cable in numerous cases. The production group has frequently used its own unauthorized stock of unmarked, nontraceable electric cable. Records are is not controlled.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 60

ATS No.: RV83A57 BN No.:

Characterization

A site contractor (H. P. Foley) has been purchasing material through unapproved vendors.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 61

ATS No.: RV 83A57 BN No.:

Characterization

Lack of Document Control

Implied Significance of Plant Design, Contruction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 61a

ATS No.: RV83A57 BN No.:

Characterization

H. P. Foley used unapproved drawing

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 62

ATS No.: RV83A57 BN No.:

Characterization

A site contractor (H. P. Foley) has not adequately performed sampling of cable pulling and termination program.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 63

ATS No.: RV83A57 BN No.:

Characterization

A site contractor (H. P. Foley) has lost material traceability through improper upgrading of non-class 1 material to class 1 material. (Specific examples were identified).

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 64

ATS No.: RV83A57 BN No.:

Characterization

Concrete grout test sampling by a site contractor (H. P. Foley) was based on a specially prepared test sample, as opposed to actual field samples.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 65

ATS No.: RV83A57 BN No.:

Characterization

A majority of H. P. Foley quality assurance (QA) records have not been reviewed by document analysts. QA record review checklists, which indicate problems, are to be destroyed. Records prior to the 1981 licensing of Unit 1 are not receiving any more attention regardless of probable inconsistencies.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 66

ATS No. RV83A52 BN No. N/A

Characterization

A site contractor (H. P. Foley) incorrectly rejected defective weld reports.

Implied Significance to Design, Construction, or Operation

The site contractor in question (H. P. Foley) has been responsible for installation and modification of electrical, civil and mechanical design class 1 safety systems and structures which are necessary for the safe operation and shutdown of the plant.

Assessment of Safety Significance

See Allegation or Concern No. 24

Staff Position

See Allegation or Concern No. 24

Action Required

See Allegation or Concern No. 24

Task: Allegation or Concern No. 67

ATS No. RV83A55 BN No. N/A

Characterization

A plant employee (security guard) telephoned the Region V office on November 8, 1982 and stated that the licensee was negligent in responding to a flooding occurrence in the plant's auxiliary building the preceeding day, in that the licensee did nothing to stop the flooding, cleanup the water or check for contamination.

Implied Signifiance to Design, Construction, or Operation

The allegation, if determined to be the case, could indicate a weakness in the licensee's response to abnormal conditions at the facility.

Assessment of Safety Significance

The NRC's Resident Inspector examined the circumstances of the occurrence by interviewing the personnel involved and review of available documentation.

Following flushing of a portion of the auxiliary feedwater piping, the system was prepared for draining when an unexpected quanity of water began to flow from the piping (due to a leakage past a valve which was later determined to be defective). The startup engineer who was in charge of the activity notified the control room operators.

During the course of the occurrence water from plant's condensate storage tank over-flowed the auxiliary building drain receiver tanks allowing water to accumulate on the floors of areas of the auxiliary building. Plant operators, observing this condition, determined that there was no threat of equipment damage of personnel hazard. Leakage was secured after approximately 3 hours. The water was not contaminated and cleanup consisted of pumping the water to the drain receiver tanks.

Discussions with plant operators revealed that they had been involved in what they considered at the time to be higher priority work during the initial approximate' 2½ hours of the occurrence.

Plant operators recalled the reports from security guards, but that they had at that time already verified that the leakage had been terminated.

The licensee's investigation into the event revealed that the day shift operators failed to turn over to the oncoming shift information relating to the ongoing flushing operations.

Staff Position

The Resident Inspector concluded from his examination of the occurrence that the licensee generally responded appropriately to the event considering the circumstances and other activities in progress at the time. However, the occurrence did reveal the need for improvements in the communication of information relating to activities in progress during shift changes, as well as overall communication among shift personnel.

Action Required

None

The Regional staff will give particular attention to the adequacy of overall communication among shift personnel. Followup will be performed to assure that the paging/announcing system which the licensee has committed to in Task Allegation or Concern No. 47.

Task: Allegation or Concern No. 68

ATS No. None BN No. None

Characterization

Nuclear Services Corporation (NSC) conducted an audit of Pullman Power Products, the prime piping contractor for Diablo Canyon in 1977. The audit findings implied a breakdown in the programmatic aspects of Pullman's QA program.

Implied Significance to Design, Construction, or Operation

The implication of the audit findings is that the Pullman QA program was not effectively implemented prior to 1977.

Assessment of Safety Significance

The staff reviewed all of the NSC audit findings and performed extensive reviews of those that could impact the quality of hardware. During this process approximately 70% of the findings were verified by the staff. The staff did not identify any significant breakdowns in the Pullman QA program although some records from the early 1970s could not be retrieved. To further assess the quality of work, an NRC independent contractor reviewed approximately 100 radiographs, performed independent measurements of weld attributes, and reviewed the records of Pullman work. No significant concern evolved.

The staff conducted indepth reviews of Pullman records and procedures on site, reviewed the QA program hierarchy of Pullman audits, and examined the licensee audits of Pullman activities: To supplement the records review and observations of hardware quality, the staff interviewed Pullman crafts, QA/QC and management personnel with particular experience at the site in the early 1970s. One apparent item of noncompliance, related to inspector qualification, was identified.

There were no significant findings which would be indicative of a programmatic breakdown.

The staff also assessed the response of the licensee and Pullman to the NSC audit and other QA/QC findings. The staff determined the corrective actions to be adequate.

The details of the staff review are documented in Inspection Report 50-275/83-37.

Staff Position

The staff found no evidence to conclude that there was a programmatic breakdown in Pullman Power Products QA program nor could the staff identify any safety concerns with the installed hardware. The staff is reviewing the reportability of this subject under 10 CFR 2.206.

Action Required.

Inspection Report 50-275/83-37 findings will be followed as part of the normal inspection program.

The staff response to the 10 CFR 2.206 will be completed in the near future.

Task: Allegation or Concern No. 69

ATS No. None

BN No. None

Characterization

Congressman Edward J. Markey raised questions related to the revision of Draft Case Study C based on the licensee's response to drafts provided to them by the NRC.

Implied Significance to Plant Design, Construction, or Operation

None

Assessment of Safety Significance

None

Staff Position

The staff position was explained in letters to Representative Markey dated October 7, 1982. Additional information is provided in a draft letter to Representative Markey sent to the Commission on December 1, 1983.

Action Required

None

Task: Allegation or Concern No. 70

ATS No.: Q5-83-019 BN No.:

Characterization

Inadequate respone to notice of violation.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 71

ATS No.: RV 83A58 BN No.:

Characterization

Use and sale of drugs

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegationor Concern No. 72

ATS No. N/A BN No. N/A

Characterization

Inadequate PG&E quality assurance (QA) program since license suspension.

Implied Significance to Plant Design, Construction or Operation

If PG&E's QA program has been inadequate since the license was suspended in November 1981, there is an implication that inadequate checks have been applied and verification done to assure the acceptability of the work performed since that time.

Assessment of Safety Significance

This allegation/concern is described in a letter to the Commissioners from Joel R. Reynolds, Counsel to the Joint Intervenors, Center for Law in the Public Interest, dated November 4, 1983. The letter is based on mid-1982 reviews of PG&E's QA program manuals by Project Assistance Corporation (PAC) and EDS Nuclear, Inc. (EDS). PAC was contracted by PG&E to review PG&E's Corporate Nuclear QA Manual vs. applicable NRC QA Regulatory Guides and the ANSI QA Standards referenced by these guides. EDS was contracted by PG&E to review PG&E's Departmental Manuals in a similar manner.

The staff reviewed and performed detailed examination of the Problem Statements (i.e., review findings) generated by PAC and EDS, as indicated below. The Problem Statements were classified by PAC and EDS reviewers in order of increasing significance.

Class	Problem Statements	Description
3	12	Efficiency item, i.e., one with no safety significance. (This class was not used by PAC in its review of the Corporate Nuclear QA Manual).
2	90	A potential problem or weakness in the manual being reviewed.
1B	85	A deficiency in the manual being reviewed determined by PAC or EDS to be either an open item (1B) or a nonconformance (1A).
1A	24	

The Class 3 Problem Statements were reviewed by the staff to verify that they have no safety significance. The Class 1A Problem Statements were reviewed 100% by the staff and 28 of the 175 Class 1B and Class 2 statements were examined. Each of the 28 statements included two or more examples of the indicated problem; a sampling of these examples were reviewed by the staff.

The staff review consisted of examining each Problem Statement and examining PG&E manuals to determine and assess the adequacy of procedures included therein as well as other licensee procedures which were not within the scope of the effort undertaken by PAC & EDS.

During its review of the Class 2 Problem Statements, the staff concluded that 19 have no safety significance and could be properly classified as Class 3.

The staff's review of the Class 1A, the Class 1B, and the other 71 Class 2 Problem Statements determined that findings of the PAC and EDS reviews did not reveal unacceptable deficiencies in the licensee's Quality Assurance Program. The staff will, however, continue to examine the licensee's response to findings by PAC and/or EDS.

The licensee has formed a QA task force to further evaluate and follow-up on the problem statements and the recommendations made by PAC and EDS.

Staff Position

The staff assessment, of findings in the PAC and EDS reviews, does not identify significant evidence that the licensee's QA program was inadequate or that the PAC or EDS reviews themselves disclosed any significant deficiencies in plant construction or operational capability.

Action Required

The staff will continue its review to determine the final adequacy of the licensee's procedures to meet the quality requirements and commitments.

Task: Allegation or Concern No. 73

ATS No.: RV 83A061 BN No.:

Characterization

Selling of drugs.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 74

ATS No. RV83A062 BN No.

Characterization

Defective pipe hangers. (See Task Allegation or Concern 91)

Implied Significance to Design, Construction, or Operation

Refer to Task Allegation or Concern 91

Staff Position

Refer to Task Allegation or Concern 91

Action Required

Refer to Task Allegation or Concern 91

Task: Allegation or Concern No. 75

ATS No.: RV83A063 BN No.: N/A

Characterization

The concern expressed was that the accumulator 1-2 discharge piping was routed too close to an adjacent operator valve support.

Implied Significance to Design, Construction or Operation

The significance appears to be that under certain undefined conditions contact may be made between the valve operator support and the accumulator 1-2 discharge piping and that this contact may provide adverse stresses to either the valve operators or the accumulator discharge line.

Addessment of Safety Significance

Examination of this issue involved field inspection of the condition and technical analysis of the failure mode. The staff inspected the installation and observed the valve operator support.

At the staff's request the licensee analyzed the predicted movements of the accumulator 2 discharge line under seismic and thermal conditions. This analysis showed that the predicted maximum line movement, in the direction of the support structure, is insufficient to cause contact between the components. Therefore, there is no safety significance to the apparent concern that the distance between the valve operator support structure and the accumulator 1-2 discharge line may not be sufficient to preclude contact between these components.

The valves supported by the proximate support are used in leak checking the Safety Injection check valve associated with the accumulator 2 discharge line. If the accumulator discharge line were to rupture, the leak check valves are no longer needed and would perform no further safety function. Therefore, there is no safety significance to the concern that a rupture of the accumulator 2 discharge line would render the leak check valves inoperable.

Staff Position

The staff concludes that the installation is adequate.

Action Required

None

Task: Allegation or Concern No. 76

ATS No.: RV83A0063 BN No.: N/A

Characterization

U-bolts have failed as evidenced by photographs of a deformed U-bolt supplied by the alleger.

Implied Significance to Design, Construction, or Operation

This concern is potentially significant in that the failure is implied to have been caused by overstressing the U-bolt as a result of excessive loading caused by thermal expansion of the pipe.

Assessment of Safety Significance

An interview and site tour with the alleger was performed on December 7, 1983. The alleger identified instances of deformed U-bolt installations including the deformed U-bolt in the photographs supplied by him, which was associated with a waste gas compressor.

The staff inspected class 1 areas in the plant with particular attention given to examining U-bolts attached to small bore class 1 lines for evidence of overstress caused by excessive thermally caused loading. The inspector did not observe any U-bolts supporting small bore class 1 lines which exhibited evidence of deformation caused by excessive pipe loading. Approximately 250

U-bolt installations on small bore class 1 lines were examined. The inspector noted that the waste gas system, particulary the waste gas compressors, are not nuclear safety-related or quality class 1 installations. As such quality control inspections were not performed on these U-bolt installations.

The inspector brought the above discrepant conditions to the licensee's attention. The licensee agreed to rework and/or replace the identified installations and bring these installations into conformance with specification requirements.

Staff Position

The waste gas system, particularly the waste gas compressors, are not nuclear safety-related or quality class 1 installations. The staff did not identify any deformed U-bolts on safety-related systems and concludes there is no safety concern.

Action Required

None by NRC staff. The above discrepant conditions were brought to the licensee's attention and the licensee agreed to rework and/or replace the identified U-bolt installations.

Task: Allegation or Concern No. 77

ATS No.: RV83A063 BN No.: N/A

Characterization

Steel plate valve support structure is bent, as evidenced by a photograph supplied by the alleger.

Implied Significance to Design, Construction, or Operation

If the installation is on class 1, safety-related system the discrepancy could have some significance.

Assessment of Safety Significance

The alleger was interviewed on December 7, 1983 at the site. On a site tour he identified the photographed location to be under the valve body of 1-PCV-75, the waste gas compressor number 01 suction valve. The support plate deformation, observed by the staff, appeared to be caused by some condition other than the valve weight. The waste gas compressor system is not nuclear safety-related or quality class 1, and therefore was not inspected by the licensee's quality control program for conformance.

Subsequent to the alleger interview, the staff examined about 50 dead load supports installed under valve bodies or operators in safety-related, quality class 1 systems. The staff did not identify any similarly deformed dead load supports.

Staff Position

The staff concludes that there is a low probability that deformed load supports exist in unit safety-related areas.

Action Required

None by NRC staff. The bent support plate installation was brought to the attention of the licensee who stated that the support structure would be evaluated for excessive loadings and reworked/replaced as necessary to prevent excessive forces from being imposed on the valve body.

Task: Allegation or Concern No: 78

ATS No.: RV-83A-063 BN No.: N/A

Characterization

Drain line support bracket bolted to the floor with only one anchor bolt in Unit 2 as evidenced by photograph supplied by the alleger.

Implied Safety Significance to Design, Construction, or Operation

The safety significance of this installation is minimal since drain lines feeding into the floor drain system, downstream of drain line isolation valves are not nuclear safety-related or quality class 1 systems.

Assessment of Safety Significance

The staff could not locate the subject installation in Unit 2. Drain lines downstream of isolation valves are not classifed as nuclear safety-related and are not quality class 1 installations. Therefore, they are not inspected by quality control for conformance to any specific requirements.

The licensee design provides a class 1 support (which is inspected by QC) immediately downstream of the class 1/class 2 code boundary transition. In the case of drain lines, this is the drain line isolation valve. All portions of the drain line downstream of the code boundary isolation are classified by the licensee as quality class 2 and are not within the ASME code jurisdication.

Discussions with the licensee's Engineering Supervisor on-site revealed that PG&E is conducting extensive reexaminations of the design and installation of small bore pipe supports installed in Unit 2. (This action was previously completed for Unit 1.) The licensee's new criteria requires an upgraded code boundary support evaluation. This reexamination is complete for class 1 supports and code boundary supports. The licensee estimates that 75% of the remaining support installation examinations are complete and have been reworked to more conservative criteria. Thus, it is highly probable that the installation photographed has been reworked. New supports installed to this criteria are full seismic class 1 supports and, as such, are inspected by QC for compliance with specification requirements.

Staff Position

The staff considers that the licensee's action completed to date, and which will be completed, provides assurance that the pictured installation, and similarly installed supports prior to this reexamination effort, will be upgraded to the new, more conservative criteria.

Action Required

Review licensee action as part of continuing inspection of Unit 2.

Task: Allegation or Concern No. 79

ATS No.: RV83A063 BN No.: N/A

Characterization

Site design engineers were not required to work using controlled documents resulting in different calculation bases, load rating, and allowables applied to their work.

Implied Significance to Design, Construction, or Operation

Engineers are calculating stresses in piping in a variety of ways. Without uniform design bases, formulations, and acceptance criteria, the adequacy of plant system safety cannot be verified and assured.

Assessment of Safety Significance

The staff reviewed engineering manuals, directives, and procedures located at onsite engineering offices to assess the degree of standardization, currency and availability of design documents. Six design engineers performing on-site design activities were interviewed as part of the review. The staff identified three instances of out-of-date engineering documents and several cases of the availability of technical articles and data not related to the design of Diablo Canyon.

It was determined that there was one set of controlled procedures maintained in the stress analysis group. Within this set, six procedures were selected for examination. Three of the six were not current, or should have been deleted and replaced.

There was evidence of inadequacies in document control such as inconsistencies in procedure lists maintained by different supervisors in design groups and confusion about who has responsibility for maintenance of procedures and drawings.

The staff reviewed several of the site design procedures for their calculated support structural frequency, the criteria contained in the widely used Bechtel procedures differ from the acceptance values included in the site design procedure.

The staff reviewed the calculations for severe different designs, most of which involved many revisions to the original calculations. The staff identified possible errors in design calculations. Further staff analysis is in progress to assess the significance and magnitude of these conditions.

Field inspection related to this subject revealed that large and small bore piping supports are somtimes installed very close to snubbers. Snubber operability may be affected by the installation of rigid restraints in close proximity to the snubbers. This subject is also the topic of further staff evaluation.

Certain of the above areas are candidates for possible enforcement action.

The staff orally presented the preliminary findings to the licensee in a management meeting on December 8, 1983. The licensee stated that some administrative controls may have been lacking, but that the final design is adequate and free of any significant design errors. Rationale for specific designs, such as the assumptions used, were discussed. There is some disagreement as to the acceptability of certain design assumptions, such as the affect of prequalified support members, between the staff and licensee engineers. The resoltuion of the acceptability of design assumptions will determine if the apparent errors detected by the staff are significant.

Staff Position

The staff concludes that the administrative controls imposed on the engineering activities require further examination. These items will be further examined in conjunction with the technical analyses related to this subject (nos. 55, 78, 82, 85, 86, 87, 88, 89, 95, and 97).

Action Required

The staff will perform an assessment of the licensee's small bore design practrus in conjunction with the technical issues identified above. It is the staff's recommendation that this assessment be completed prior to reactor criticality.

Task: Allegation No. 80

ATS No. RV-83-A-64 BN No. N/A

Characterization

A letter from Dr. Richard Kranzdorf as Spokesperson for Concerned Cal Poly Faculty and Staff concludes that the licensing process for the Diablo Canyon Nuclear Power Plant (DCPP) should cease until two primary issues regarding emergency planning by San Luis Obispo County/Cities are resolved:

- There is a perceived lack of public confidence in the feasibility of DCPP emergency response planning.
- Impediments to evacuation of the public exist which have either not been addressed in planning or have been inadequately or improperly addressed in planning.

Implied Significance to Design, Construction or Operation

Implied is that in event of a major nuclear emergency at the DCPP, planning is inadequate to insure the public health and safety through appropriate evacuation of some geographic areas within the emergency planning zone (EPZ) during cited inclement weather conditions or other natural physical phenomena (e.g., earthquakes, etc.).

Assessment of Safety Significance

A conference call involving Region IX of the Federal Emergency Management Agency (FEMA), the State of California Office of Emergency Services (OES), the San Luis Obispo County Emergency Coordinator's Office (SLO-EC), and NRC Region V analyzed and discussed the allegations and revelations in Dr. Kranzdorf's letter with the following interim conclusions.

- OES and SLO-EC after review and further investigation concluded that factors set forth by Dr. Kranzdorf were thoroughly studied and appropriately addressed in emergency planning for DCPP.

- Impediments to evacuation of the public are recognized and appropriate solutions have been developed and tested to insure feasibility.

- Dr. Kranzdorf postulates a "worst imaginable case" with no historical procedence as opposed to a "reasonable case" based on known capabilities and physical probabilities, recognizing and appropriately solving impediments to evacuation of the public under nuclear emergency conditions.

In that FEMA has primary responsibility by Presidential Executive Order to take the lead in offsite planning for nuclear emergencies, FEMA Region IX will prepare a letter of response to Dr. Kranzdorf's allegations for use by NRC in resolving the allegations and formulating an appropriate response to Dr. Kranzdorf. FEMA's initial evaluation is that the letter from Dr. Kranzdorf does not appear to disclose any reason to alter prior FEMA evaluations and conclusions regarding offsite planning for the public health and safety in event of a nuclear emergency at the DCPP.

Staff Position

Await the documented FEMA response and as appropriate, prepare a letter to Dr. Kranzdorf responding to his concerns.

Action Required

Upon receipt of FEMA input, prepare letter to Dr. Kransdorf setting forth a coordinated Federal, State, and local government response to his concerns.

Task: Allegation or Concern No. 81

ATS No.: RV83A063 BN No.:

Characterization

Individual fired for whistle blowing

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 82

ATS No. RV83A063 BN No. N/A

Characterization

There was minimal training for onsite pipe support engineers.

Implied Significance to Design, Construction, or Operation

Without adequate indoctrination and training, the engineers may not effectively perform their assignments.

Assessment of Safety Significance

This issue was addressed by examination of training requirements, implementation records, interview of engineers, and review of engineers work products.

The staff interviewed five onsite design engineers selected from the personnel roster. In addition, managers/supervisors of the various design groups were interviewed. There were no written job descriptions for any of the pipe stress and support group leaders, lead engineers, and engineers.

The staff found that, other than general site QA and technical training provided for the new employees, no project group specific program was in place in either the pipe stress, or the pipe support engineering group. The need for such training is being further evaluated by the staff. In addition, the general QA and technical trainings received by the staffers had not been timely and consistent in all cases. The bases for this determination are:

Work Group	Begin Work Mo/Yr	Engineering Manual Survey	Individual Indoctrination Date
A (Support)	10/82	02/18/83	5/5/83
B (Support)	04/83	07/15/83	5/4/83
C (Support)	09/83	record shown no longer with projects	record show no longer with projects
D (Stress)	05/81	06/9/83	none
E (Stress)	02/83	04/19/83	05/04/83

The staff reviewed several design calculations, which are identified in Task Allegation or Concern No. 55, 78, 84, 85, 86, 88 and 95. Among the calculations reviewed, possible errors were identified. The implication of any errors which are determined to exist will be considered in final evaluation of this area.

Staff Position

The staff concludes that training in project design standards and procedures has not been timely or consistent and may not have been adequate in some cases. This area will be the subject of further review.

Action Required

Further actions related to this issue will be handled in conjunction with issues 55, 78, 79, 84, 85, 86, 88 and 95.

Task: Allegation or Concern No. 83

ATS No.: RV83A063 BN No.:

Characterization

Site design engineer have not been required to work using controlled documents, resulting in the use of different design assumptions and other problems.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Predecisional

Action Required

Task: Allegation or Concern No. 84

ATS No. RV83A063 BN No. N/A

Characterization

Lack of management responsiveness to resolve the alleger's concern that he was not provided with controlled design procedures.

Implied Significance to Design, Construction, or Operation

The use of controlled design documents insures all works are performed to current relevant design, codes, and standard requirements.

Assessment of Safety Significance

The staff interviewed the alleger onsite on December 7, 1983 to clarify his concerns in this area. The alleger referred to a memorandum written by line management to upper management relative to his concern about a lack of controlled design procedures.

The staff interviewed project team general construction personnel in relation to the memo purportedly written by supervision. In discussion with the pipe support group leader, on 12/6/83, he denied that he had written a memorandum to Messrs. R. Oman and M. Leppke in December 1982, the On-site Project and Deputy Engineers, relative to the lack of controlled design procedures to be used in the pipe support group, in support of Mr. Stokes' concern.

Mr. Leo Mangoba agreed that he was aware of the subject concerns raised by a number of his staff, and had taken actions to obtain additional controlled design procedures.

In view of the task findings that were discussed in Task Allegation or Concern Nos. 79 and 82 that a large number of out-of-date procedures and drawings were found, the deficient document control system; the lack of training for the personnel relative to the use of up-to-date procedures, design revision, the management response to timely correct the problem and to prevent recurrence appears to be inadequate. The spirit of the allegation was substantiated.

Staff Position

The staff concludes that site management must improve its sensivity in addressing safety concerns and improve communication with the workers.

Action Required

None

Task: Allegation #85

ATS No.: BN No.:

Characterization

U-Bolt Design inadequate.

Implied Significance to Plant Design, Construction or Operation

U-Bolts act as load-carrying members of small bore pipe supports. As such they are used for supporting safety related piping which is required for plant safe shutdown.

Assessment of Safety Significance

An NRR representative has interviewed the alleger (C. Stokes) at the site on December 7, 1983. He stated that the installation of the U-Bolts was poor, the manufacturer's (ITT Grinell) load rating was exceeded by approximately a factor of four, and that the DCP interaction of tension and side loads was less conservative than the manufacturers. He also provided a DCP document which specifies the design load ratings and some DCP experimental data supporting these load ratings.

Staff Position

The staff has made a preliminary assessment and has concluded, based on actual observation of a sample of U-Bolt supports, that the U-Bolts appear to be installed in accordance with current industry practice. There may be some merit to the other parts of the allegation, but there is no safety significance until 5% power is exceeded due to negligible fission product inventory.

Action Required

The staff will assess the documents provided by the alleger as to technical adequacy. This issue will be further examine in conjunction with issues 55, 78, 79, 82, 86, 87, 88, 89, 95 and 97. A staff assessment on these items will be completed by January 18, 1984.

Task: Allegation #86

ATS. No.: BN No.:

Characterization

"Code break" design.

Implied Significance to Plant Design, Construction or Operation

A "code break" occurs within a piping system where Design Class I (seismic) and Design Class II (non-seismic) piping meet. The boundary is defined by a valve and certain support requirements, such as an anchor on the Class II side of the valve. These support requirements may induce stresses due to thermal constraint in safety related portions of piping.

Assessment of Safety Significance

An NRR representative interviewed the alleger at the site on December 7, 1983. The alleger stated that a "code break" deficiency had existed but that he was unaware if and when it had been resolved.

Staff Position

The "code break" deficiency was identified as a generic issue by the IDVP, and was addressed by the DCP during the reevaluation of the small bore piping. The IDVP verified, on a sample basis, that the DCP resolved this deficiency satisfactorily. The staff, therefore, finds no safety concern for this allegation. There is no impact on either low power testing or full power licensing.

Action

None.

Task: Allegation or Concern No. 87

ATS No. 83A063 BN No. N/A

Characterization

On site management destroyed those calculations showing certain supports will fail under design conditions, and assigned new staffers to reperform the calculations and show that these supports were adequate. The calculation logs did not refer to the original packages showing support failures.

Implied Significance to Design, Construction, or Operation

Management pressure to compromise system design safety margin. Falsification of records to cover up substandard design conditions.

Assessment of Safety Significance

The alleger was interviewed by the staff at the site on December 7, 1983 and additional information was obtained. The staff retrieved the original design calculation logs and design calculation packages from the licensee records vault. The records and logs are being reviewed and calculations evaluated by NRR and regional staff. An interface with the Office of Investigations has been established.

This task will be coordinated with the findings of Task Allegation or Concern Nos. 79, 82, 84, 88, and 95.

Staff Position

The staff evaluation has not progressed to the point where a position, either supporting or denying the allegation, can be taken.

Action Required

Complete staff technical review and OI examination.

Task: Allegation or Concern No. 88

ATS No. RV83A063 BN No. N/A

Characterization

There had been ways to accept supports designed on-site that were determined to be incapable of meeting the loading conditions.

Implied Significance to Design, Construction, or Operation

Management practice to compromise system design safety margin by juggling calculations and designs to accept supports, that had been rejected by calculations performed by the originial reviewers, could result in structures unable to perform their intended function.

Assessment of Safety Significance

The staff met with the alleger on site on December 7, 1983. Clarification and additional information concerning specific areas of his affidavit were obtained. The broad characterization of his concerns highlights the following detailed elements/ways the design group may compensate for unacceptable calculations:

a. Revising pipe code break locations in order to reduce the number of safety related supports, and omitting many of those that failed in the review program.

b. Assuming gaps that did not exist and vice versa.

c. Assuming joint release for rigid connections, but made no attempt to remove the welds.

d. Performing calculations to determine maximum support load carrying capacity. The results were then sent to the stress group for line model change to meet piping stress allowables.

e. Adding new supports within six inches of the unacceptable supports, the new supports consisted of inaccurate assumption of restraing gaps. The new supports did not have control or document numbers.

The staff has obtained the records, calculation logs and design calculations necessary to examine the above concerns. The issues will be reviewed jointly by NRR and regional staff and coordinated with Task Allegation or Concern Nos. 79, 82, 84, 87, and 95.

Staff Position

The staff evaluation has not progressed to the point where a position, either supporting or denying the allegation, can be taken.

Action Required

The staff will complete the technical review of design data and calculation packages to assess the significance of this concern.

Task: Allegation or Concern No. 89

ATS No. RV83A063 BN No. N/A

Characterization

The on-site design group has improperly resolved piping interferences.

Implied Significance to Plant Design, Construction, or Operation

Piping interferences or inadequate piping support could result in piping systems being overstressed during operational or design loading conditions.

Assessment of Safety Significance

The staff reviewed the disposition related to the allegation and determined that the matter was acceptably resolved. The staff inspected areas of the containment and auxiliary building looking for cases where pipes were resting on conduit supports. The staff did not observe any cases. Since no specific cases were cited in the affidavit, this concern relates to the more general concerns on design control on piping and supports being addressed by Tasks related to allegations 79, 82, 84, 87, 88 and 95.

Staff Position

This concern should be covered by the resolution of Tasks 78, 82, 84, 87, 88 and 95.

Action Required

The staff is to complete the technical reviews, as discussed above.

Task: Allegation or Concern No. 90

ATS No. RV83A063 BN No: N/A

Characterization

Embedded wood and defective concrete was discovered in a wall separating Unit 1 auxiliary saltwater system (ASW) pumps at the intake structure.

Implied Significance to Design, Construction, or Operation

Poetntial significant impact upon integrity of a structure important for protection and support of design class 1 components.

Assessment of Safety Significance

This concern was addressed by examining pertainent documentation, interviewing personnel, and inspecting and testing concrete.

On 7/22/83 while drilling holes for the installation of hanger anchor bolts, wood was discovered to be embedded within the north concrete wall of the ASW pump 1-2 room in the intake structure. Corrective action was taken to chip out the wood and grout the resultant enlarged hole and use thru-bolts instead of Hilth Kwik anchor bolts. Work was essentially completed on 8/9/83 for the hanger and the completed installation was accepted by quality control 9/22/83.

Individual interviews were conducted by the NRC inspector with those available personnel involved. Pertainent records were examined. The lead mechanical night shift engineer reported having observed two pieces of wood, characterized as approximately the size of a toothpick and a large pencil in magnitude, embedded within the concrete wall separating ASW pump rooms 1-1 and 1-2. Some concern was also expressed that the concrete consistency appeared soft and sandy. Foley cement masons chipped out the concrete (a hole approximately 4" in diamerer and 3/4" deep) removing all indications of wood. After inspecting the enlarged hole, the civil engineer reports concurring that all wood had indeed chipped away, and identified the quality of the exposed concrete as good.

A general visual inspection was performed by the staff, at the intake structure of the installed hanger, associated ASW pump room concrete walls, and ASW components. No discrepancies were observed.

Structural strength impact tests were performed on various locations in the intake structure. Ten Schmidt hammer impact tests at each of ten sites, located in the AWS 1-1/1-2 pump rooms and walls representative of the intake structure as a whole, were observed by the staff. The test data was evaluated with the determination that concrete strengths exceeded design requirements.

No evidence could be found to suggest that embedded wood in intake structure concrete is of a generic significance.

In conclusion, this allegation has no safety significance. Evidence indicates all the embedded wood was removed and the concrete wall was adequately repaired to support hanger installation. There is no justification to support the concern of soft or intrinsically defective concrete in the ASW 1-2 pump room.

Staff Position

The staff concludes that the intake structure concrete is of satisfactory quality and that the corrective actions taken to remove the wood were adequate.

Action Required

None

Task: Allegation or Concern No. 91

ATS No.: RV83A063 BN No. N/A

Characterization

Alleged coverup of defective material use.

Implied Significance to Design, Construction, or Operation

The effect of this allegation is to question the "workmanship" quality level of hardware brackets supplied for supports and the licensee's handling of a reported problem in this area.

Assessment of Safety Significance

The staff was provided by the alleger with a U-shaped support and supplemental metallurigical samples and a report.

The staff inspected the support, reviewed the metallurigical data, examined licensee procurement records, and analyzed corrective actions.

The laminations observed visually at the edge of the support were verified by metallurigical and nondestructive examinations (NDE). The working stresses applied to the support are parallel to the location of the lamination and therefore have minimum impact on the ability of the support to perform its design function.

The supplied support bracket and several others were procured from NPS Industries as nonsafety-related items, used in non-safety systems, and, therefore, there were no NDE requirements placed on them. The supports met nominal catalog dimensions and were fabricated from the specified material, SA36. The licensee performed NDE on several of the supports to determine the extent of lamination. These were subsequently reviewed by engineering and the supports were determined to be acceptable for service.

The staff inspected the support, reviewed NDE and metallurigical data, checked procurement records, and examined corrective action. The staff determined that the licensee action was acceptable and concurred that the supports were acceptable for service. The staff agreed that the workmanship exhibited by the supports was not of the quality that would be required if the support had been classified as quality class 1 and used in quality Class 1 systems.

Staff Positions

The staff concluded that the supports met the procurement specifications and were acceptable for the intended application.

Action Required

None

Task: Allegation or Concern Nos. 92 & 93

ATS No.: RV83A063 BN No.: N/A

Characterization

Flare Bevel Welds are undersized and do not comply with AWS Code requirements. Flare bevel welds were inadequately depicted on construction drawings.

Implied Significance to Design, Construction, or Operation

The implied significance is that actual weld sizes use below the sizes assumed in design and, therefore, unable to carry design loads.

Assessment of Safety Significance

The staff reviewed the design requirements and drawings for flare bevel welds and compared PG&E criteria with AWS D1.1. The alleger was interviewd on site on December 7, 1982 and identified specific welds of concern on the site tour.

The staff inspected over 100 flare bevel weld joints and determined that AWS requirements were satisfactorily met on all those inspected. The acceptance criteria of Attachment J to EDS 223 is only appropriate for welds that are not fully welded out to "flush" conditions. This was not the practice because all flare bevel welds observed were flush welded. The flare bevel joint requires no specific preparation to meet AWS D1.1 or ASME NF requirements. The weld quality for flare, skewed angle and fillet joints were of good quality, based on visual inspection.

Pullman Power Products provided the staff with results of an evaluation of bevel joints welded on tubular steel which showed that all welds made on various sizes of tubular steel members met, or exceeded, the AWS D1.1 effective throat requirements. The staff considers this agreed with visual inspection results.

The staff verified that the licensee's drawing did not specify any included angle for the angle bevel in partial penetration welds. However, interviews with ten (10) Pullman QC inspectors identified that their interpretation is that if the required bevel angle is not specified it would be 45-60°. Interviews with Pullman shop superintendent confirmed this same practice. Licensee design practice assumes an effective throat penalty for 45-60° partial penetrated welds which is in conformance with the AWS D1.1 Code. The staff concurs with this practice.

The Lawrence Livermore Laboratories is conducting an as-built review of pipe supports under contract with Region V. They have examined 280 pipe supports and identified four discrepant welds, none of which have been judged to have any safety significance.

Staff Position

The staff concludes that the flare bevel welds comply with AWS D1.1 requirements and that the quality of the welds is good.

The staff agrees that the licensee did not specify the included angle for partial penetration welds on the drawing. However, construction practices are such that the craft and their foremen are cognizant of the correct bevel angle to be used.

Action Required

The licensee will review and evaluate the discrepant welds identified during Lawrence Livermore Laboratory as-built inspection. The staff will monitor this review.

Task: Allegation or Concern No. 94

ATS No. RV83A063 BN No. N/A

Characterization

Pullman used pipe welding procedures to make structural steel welds.

Implied Significance to Design, Construction, or Operation

The implication is that welding procedures applicable to pipe welds may not be satisfactory for structural steel welding.

Assessment of Safety Significance

The staff reviewed a typical weld procedure specification (WPS) in question (WPS P1-BR-F4-SHAW-2G-5G also referred to as WPS Code Number 7/8) and the application of the procedure.

Pullman qualified the weld procedures to ASME Section IX requirements which is compatible with the AWS D1.1 requirement, (paragraph 5.2). The interpretation that the weldability and mechanical properties of the welded joint for AWS D1.1 welding can be so qualified is a standard industry practice. The WPS referenced above was qualified on the basis of two Procedure Qualification Records which qualified the process for 3/16" to 3/4" thick P1 carbon steel materials in the as welded condition. There is no requirement that the WPS specifically refer to welding of support structures. It is sufficient that the process be qualified to the proper base metal (P1), thicknesses and those filler metals specified in the AWS D1.1 Code.

Staff Position

The staff concludes that the practice of qualification of AWS and ASME WPSs to ASME Section IX is acceptable. The staff found no inconsistency in the WPS examined and that the use of the WPS for structural welding was acceptable.

Action Required

None

Task: Allegation or Concern 95

ATS No. 83A063 BN No. N/A

Characterization

Angles of pipe support member are out of specification. Unbraced angel steel members within a support framework exceeded AISC bending stress allowables, particulary those supports where a bundle of small bore pipes were attached.

Implied Significance to Design, Construction, or Operation

The angles could buckle under excessive loading, creating large system deformation and could result in piping overstress.

Assessment of Safety Significance

The staff interviewed the alleger on-site on December 7, 1983. During a site tour, the alleger identified one specific support installed at auxiliary feedwater pump (AFP) No. 11, steam supply trap drain line No. 443, located at elevation 100' in the auxiliary building.

The staff plans to evaluate the design criteria related to small bore pipe supports, and review calculations related to this type of support. This will be reviewed in conjunction with the licensee's small bore pipe support program.

Staff Position

The staff has not yet sufficiently examined the concern and cannot state a conclusion.

Action Required

NRR and the regional staff will review the above referenced pipe support, inspect additional similar supports, evaluate the design criteria, and review calculations related to this type of support..

Task: Allegation or Concern No. 96

ATS No.: RV 83A063 BN No.:

Characterization

Improper anchor bolt spacing for Phillips and Hilti shell anchors.

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Predecisional

Action Required

Under Review

Task: Allegation or Concern No. 97

ATS No. RV83A063 BN No. N/A

Characterization

Site design engineer have not been required to work using controlled documents, resulting in the use of different design assumptions among other problems.

Implied Significance to Plant Design, Construction, or Operation

See Task Allegation or Concern 79

Assessment of Safety Significance

See Task Allegation or Concern 79

Staff Position

See Task Allegation or Concern 79

Action Required

See Task Allegation or Concern 79

Task: Allegation or Concern No. 98

ATS No.: RIII83AXX BN No.: N/A

Characterization

A vendor in the nuclear industry (Brand Industrial Serices Company, BISCO) is improperly installing penetration seals. They may be involved with Diablo Canyon.

Implied Significance to Plant Design, Construction, or Operation

Improperly installed penetration seals may be installed at Diablo Canyon.

Assessment of Safety Significance

The staff determined that BISCO is not a contractor, or subcontractor at Diablo Canyon.

Staff Position

No safety concern exists at Diablo Canyon. The contractor in question has not worked at Diablo Canyon.

Action Required

None

Task: Allegation of Concern No. 99

ATS No. Q5-83-024 BN No:

Characterization

Falsification of vendors records

Implied Significance to Plant Design, Construction, or Operation

Allegations by former QC Inspector of Bostrom-Bergen Metal Products, Oakland, California, who has supplied safety-related hardware to Diablo Canyon, that he falsified nearly every QC inspection report between January 1981 and January 1983. Supplied material maybe of questionable quality.

Assessment of Safety Significance

OI has taken a signed sworn statement from the alleger. Additional interviews are scheduled. This statement will be given to the Regional Staff and OIE for followup to determine the significance and validity of the allegations.

Staff Position

Sufficient information is not available at this time to perform a safety assessment.

Action Required

Complete interviews of sources. Conduct coordinated technical review and OI investigation.

Task: Allegation or Concern 100

ATS No.: RV83A0069 BN No.: N/A

Characterization

Diablo Canyon painters have no Quality Control Program.

Implied Significance to Design, Construction, or Operation

Potentially significant to operations, specifically post-LOCA accident assumptions due to excessive zinc inside containment and potential clogging of drains.

Assessment of Safety Significance

The safety concern is that improperly applied coatings inside containment could flake or peel following an accident which could cause restrict core and/or containment spray flow paths.

The staff reviewed Specification No. 8848, "Final Painting at Diablo Canyon Units 1 and 2,: dated January 26, 1972.

The staff observed criteria in use in the field and interviewed PG&E staff and contractor personnel. It was determined that changes to the specification had been made but were never formally controlled. The painting specification was not classified as a quality class 1 activity and, therefore, there was no formal quality control inspection program nor was there a quality assurance program applied to painting activities in the auxiliary building or containment.

The licensee has prepared a draft FSAR revision which addresses changes to the previously calculated aluminum and zinc inventory in the containment. The zinc inventory is affected by the paint composition.

Staff Position

The staff concludes that the licensee has conformed with the original classification of painting as not being a quality class 1 activity. However, considering the importance of the containment coating, (particularily with respect to zinc inventory and potential for flaking) it is the staffs opinion that these aspects be further examined, as discussed below.

Action Required

NRR review the FSAR revision related to painting composition and inventory of zinc in the containment. NRR review the FSAR assumptions related to blockage of core flow paths and/or containment spray nozzles by flaked paint.

Task: Allegation or Concern No. 101

ATS No.: RV83A0073 BN No.:

Characterization

Qualification of welders and procedures

Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Predecisional

Action Required

Task: Allegation or Concern No. 102

Characterization:

PG&E references unissued drawings in Design Change Notices (DCN).

Implied Significance

The allegation implies a failure of the PG&E document control system to issue new drawings as controlled drawings.

Assessment of Safety Significance

The staff examined a Foley inter-office communication (IOC) and applicable DCNs, interviewed the IOC author, and discussed with PG&E document control personnel the apparent failure to issue the IOC referenced drawings to the H. P. Foley controlled files. PG&E records were reviewed by the staff to determine document status of the identified DCN's and drawings.

The author stated, in the IOC, that Foley document was not handling the new drawings as controlled documents. This practice by PG&E to list affected drawings on DCN;s, which are not in controlled distribution, is considered by the author to create problems of accessibility for field production personnel and makes it difficult for onsite engineering to validate drawings QS up-to-date.

The applicable DCNs were examined by the staff, for date of issuance, the description of change, and whether DCN is nuclear related and/or safety related. A complete review was made to cross-check the drawing revisions issued by different DCNs to the drawing revisions listed as the latest approved for construction, in the Corporate document control and the on-site drawing log.

(1) The staff concludes that drawings contained in a DCN are not distributed to the contractor's document control organizations as controlled documents. PG&E on-site document control does not maintain drawing revision status of previously issued DCNs. Corporate document control fails to issue (to the field) the latest revisions of drawings contained in these DCNs or to update DCN contained drawings to the latest revision.

(2) It would appear that work performed by the contractor to a DCN, may not be in accordance with the latest approved construction drawing revision.

Staff Position

The inadequate implementation of DCN drawing document controls could concervably affected safety-related work performed by contractors.

NRC Action

DCN/drawing controls shall be further evaluated by the staff in conjunction with Allegation or Concern No. 61.

Task: Allegation or Concern No. 103

ATS No.: BN No.: 83-48

Characterization

Welding and Welding Program Concerns
Implied Signifiance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Predecisional

Action Required

ATTACHMENT 3

DIABLO CANYON

ALLEGATIONS OR CONCERNS

BY

SUBJECT

Diablo Canyon Allegations by Subject

I. Design

A. Design Control

6. Instrumentation & Control Design Classification
6a. Feedwater Isolation Classification
30. Inadequate documentation
31. QA procedures for structural analysis
34. Incomplete as-built drawings
41. Drawings inadequate
44. Improper assessment of design change notice
79. Engineers calculating stresses in a variety of ways
82. Minimal orientation for new engineers at the site
87. Calculations related to "code break" design destroyed
88. Undocumented modifications were made because of code break problems
92. Flare bevel welds are undersized and do not comply with code dihedral angle
93. Inaccurate depiction of welds on drawings (symbolic)
96. Improper anchor bolt spacing ("Hilti" and "Red Head")
97. Site design engineers have been required to use uncontrolled documents resulting in different assumptions, etc. (Same as no. 79)

B. DESIGN ADEQUACY

Seismic Adequacy

3. Seismic qualification CCW
8. Seismic design of Diesel Generator intake & exhaust
10. Seismic tilting of containment
11. Classification of Platform (Category I/Category II)
13. Inadequate seismic systems
14. Loads on Annulus Structural Steel not calculated properly
17. NSSS SSE load inadequate
28. Annulus Structure Reverification Program inadequate
29. Pipe restraints design inadequate
32. Seismic analysis containment
33. Turbine Building (Class 2) Contains Class 1 systems & components
35. Lack of support calculations for fluorescent light fixtures
36. Resolution of fluorescent light fixture interaction

System Interaction

7. Seismic Category I/Category II Interface
9. USI-17 Systems Interaction (Generic)
12. HELBA did not meet FSAR, RG 1.46
15. Inadequate Tornado Load Analysis of Turbine Building
16. High energy pipe break restraint inadequate
48. System-interaction study and associated modifications
75. Discharge piping too close to accumulator

RHR Design Adequacy

5. Heat removal capability CCW
37. Solid state protection system relays
38. PG&E ignoring spurious closure of motor operated RHR suction valve
39. No control room annunciation of closed RHR suction valve
40. RHR hot leg suction not single failure design
45. Design inconsistency in FSAR RHR valves

Piping And Support Analysis

55. Bechtel approved analysis of small bore pipe by altering failed analysis
78. Bracket bolted to wall with only one bolt
85. U-bolt Design for small bore pipe supports
86. Small bore "Code-break" design practices
89. Improper support design (use of uni-strut in pipe support design)
95. Angles of pipe support members are out of specification

Single Failure Criteria

4. Single Failure Capability CCW

II. ELECTRICAL CONTRACTOR (FOLEY)

A. NON CONFORMANCE REPORTS

24. HPFoley NCR's rejected without good cause
26. Foley didn't document NCR's issued by field inspectors
46. HPFoley QA procedures voiding NCR's incorrect
66. Defective weld reports rejected by Foley

B. DCNS

61. Lack of document control
61a. HPFoley used unapproved drawing
101. Qualification of welders & procedures
102. Improper references on DCN

C. ANCHORS

25. Deficiency in use of "Red Head" anchors for raceways support
58. Foley allows "Red Head" anchor studs reported improperly installed

D. TRACEABILITY

54. Wire traceability not evident for work by PG&E and Foley
59. Foley lost cable traceability
63. Foley has lost material traceability throught upgrade of non-Class 1 to Class 1
18. QA/QC Allegations

E. QUALIFICATION OF QC

57. Foley used uncertified and unqualified QC inspectors prior to 1983

F. VENDORS

60. Foley purchased material through unapproved vendors

G. INSPECTION QUALITY

62. Foley lacks adequate sampling of cable pull activities

H. GROUT TESTS

64. Grout test sampling based on special tests rather than field tests

I. DESTROYING DOCUMENTS

65. Foley documents prior to 1980 questioned. No review required prior to 9/1981 license issuance date

J. SUPER STRUT

27. Welding and QA deficiency in "Super Strut"

III. PULLMAN

A. WELDING

53. Welder qualification
94. Pullman used pipe welding procedures to make structural support welds
103. Welding and welding program concerns

B. QUALITY ASSURANCE

23. QA Inspector concerns
68. NSC Pullman-Kellog. audit
74. Defective piping support
76. U-Bolts have failed
77. Flange bent on I-Beam

IV. PG&E MANAGEMENT

A. UNRESPONSIVE

42. Licensee management unresponsive to problems
47. Plant public address system
67. Negligence by PG&E flooding at 55 ft. elevation pipe tunnel
84. Lack of responsiveness by management to identified problems relating to design

B. REPORTING

43. Licensee reporting failure
70. Inadequate response to Notice of violation
91. Alleged coverings of defective material use

C. QUALITY ASSURANCE

69. Case Study "C"
72. Audits of PG&E (PAC/EDS)
98. Possible non-adherence to penetration seal procedure
99. Vendor inspection records
100. No QA for coatings

V. OTHER CONCERNS

A. HEALTH PHYSICS

20. Health Physics personnel do no meet ANSI requirements
21. ALARA Program - Paper Tiger
22. Radition Monitors lack sensitivity

B. SECURITY

1. Passing of contraband
2. Anti-Nuclear Demonstration
19. Guard Qualification
50. Plant Security should have been retained
71. Use and sale of drugs
73. Selling of drugs

C. EMERGENCY PREPAREDNESS

49. Emergency Sirens not seismic qualified
80. Concerns regarding the emergency response plan

D. PROTECTION OF ALLEGERS

51. Risk of job action against allegers
81. Individual fired for whistle-blowing

E. MISCELLANEOUS

52. Construction & hearings in progress after fuel load is inappropriate
56. Pitting of main steam and feedwater piping
83. NRC was not effective in identifying problems
90. Defective concrete in intake structure

ATTACHMENT 4

DIABLO CANYON

MANAGEMENT OF ALLEGATIONS

REGION V INSTRUCTION NO. 1303

Revision Responsibility - ORA

Region V Instruction No. 1303

MANAGEMENT OF ALLEGATIONS

A. Purpose

To ensure that allegations involving NRC licensed activities or activities within the jurisdiction of the NRC expressed to, received by, or reported to any Region V employee are properly and timely documented, evaluated, handled, controlled, and dispositioned.

B. Scope

This instruction provides for the actions to be taken by Region V employees whenever they may be the recipient or otherwise learn of an allegation that may adversely impact on the NRC or NRC licensed activities, or activities within the jurisdiction of the NRC including, but not limited to, reactor operation; reactor construction; radiography; control, use, and transportation of radioactive material; safeguards; environment; and employee discrimination complaints.

C. Definitions

1. Allegation is an assertion by an individual in the form of a statement, complaint, or concern that indicates a possible problem in connection with NRC licensed activities, or activities within the jurisdiction of the NRC.

2. Alleger is an individual who makes an allegation.

3. Allegation Panel is a group of Region V employees selected by a cognizant Division Director to evaluate and recommend actions to resolve an allegation. The Office Allegation Coordinator, and the Enforcement Officer shall serve on all allegation panels, when available.

4. Cognizant Division Director is the Division Director responsible for the inspection activities affected or otherwise involved in the allegation.

D. Responsibilities and Authorities

1. Regional Administrator is directly responsible to the Executive Director for Operations to ensure proper and timely execution of NRC policies and procedures related to receipt, action and disposition of allegations that fall within the jurisdiction of Region V.

2. Division Directors shall ensure that the instructions contained herein are properly and timely executed. In particular, the Cognizant Director shall upon receipt of an allegation by Region V personnel as appropriate:

a. Ensure that the Regional Administrator and the Office Allegation Coordinator are immediately informed of allegations
that he becomes aware of.

b. Serve as Chairman of the Allegation Panel.

c. Convene an Allegation Panel, evaluate available information, and within four (4) days of the receipt of the allegation formulate an action plan to appropriately dispose of the matter.

d. Notify the responsible Licensing Office (NRR-NMSS) within two (2) days of the receipt of the allegation.

e. Issue a PN and include in the Daily Report only with approval of the Regional Administrator.

f. Prepare, sign and/or concur in all written communications between Region V and the Alleger.

g. Establish a file for each allegation which provides current information that is readily retrievable throughout the course of an inquiry/inspection/investigation.

h. Determine need for confidentiality.

3. Office Allegation Coordinator serves as the focal point for the management of information received from allegers and assures that the Regional Administrator and all responsible parties are informed of and kept current on the status of allegations. In particular the Coordinator shall:

a. Serve as a member of all Allegation Panels.

b. Verify that allegations are entered into NRC Allegation Tracking System within two (2) working days of receipt.

c. Verify that written communications have been sent to alleger as prescribed in Section E.5 of this instruction. Sign and/or concur in the letters sent to an alleger at the discretion of the cognizant Division Director.

d. Upon disposition and/or closure of an allegation, verify that the file is complete and contains all necessary documentation pertinent to the allegation.

e. Provide the cognizant Division Director, Enforcement Officer, State Liaison Officer, and Public Information Officer with copies of the Allegation Data Form when the data is entered into the NRC Tracking System. Thereafter, inform the Director and Officers of all significant information subsequently obtained pertaining to the allegation.

f. Ensure proper implementation of Region V Instruction 1302, Allegation Tracking System.

g. Serve as the interface and principal contact person between the Region V staff and the staff of the Office of Investigations, Region V.

4. Enforcement Officer shall:

a. Serve as a member of all Allegation Panels.

b. Provide advise on potential violations and possible severity levels that may arise from the allegation.

c. Alert the Regional Administrator of alleged significant violations of regulatory requirements that potentially could result in escalated enforcement action.

5. State Liaison Officer shall inform state and local officials of information contained in allegations that fall within the jurisdiction of the state and local governments and provide notice to the appropriate officials of NRC findings and actions if and when appropriate as determined and directed by the cognizant Division Director, with due regard for the need to maintain the nature of the allegation and its source confidential.

6. Public Affairs Officer shall respond to requests from members of the media and other members of the public for information concerning allegations. Information pertaining to an allegation shall not be released without approval of the cognizant Division Director.

7. Region V Employees shall, upon receipt of an allegation, complete an Allegation Report and deliver it to the Office Allegation Coordinator.

E. ACTION:

1. Receipt of Allegation

a. Region V employees who receive an allegation in written form shall immediately deliver the document to the Office Allegation Coordinator or, in the Coordinator's absence, to the Regional Administrator.

b. Region V employees who receive allegations over the telephone or during discussions with individuals shall obtain, if possible, the following information.

(1) Full name of person.

(2) Telephone number where person can be reached (work - home).

(3) Mailing Address.

(4) Name of employer.

(5) Place of employment.

(6) Job or position title.

(7) Name of licensee.

(8) Name of facility.

(9) Nature of Allegation - obtain as many specific details as possible. In addition to who, what, when, where, why and how, attempt to expand and clarify all information so that issues are well defined and can be readily evaluated as to safety significance.

NOTE: Regardless of any personal opinions, employees shall communicate with allegers in a professional manner showing due respect and interest in any and all of the concerns expressed by the alleger. Even in areas where NRC clearly has no jurisdiction, NRC will assist the individual in reaching the appropriate authority.

c. A standard allegation report form is attached as Appendix A to this instruction and should be used to document all allegations. The Office Allegation Coordinator will assist, if needed, with the completion of the report and review all reports for completeness. At the same time, an Allegation Data Form should be completed per RV Instruction 1302. Both documents should then be immediately delivered to the Office Allegation Coordinator who shall then immediately notify the cognizant Division Director and provide the Director with copies of the Allegation Report.

d. If an employee receiving an allegation believes it would be better for the alleger to discuss the matter with another employee, and if the alleger consents, transfer the call to or refer the person to the Office Allegation Coordinator. If that individual is unavailable, then refer the alleger to another appropriate employee. However, before referring the alleger, be sure to obtain the alleger's name and phone number or how the individual can be reached in case of a disconnect.

e. Many persons reporting a particular matter to NRC wish to remain anonymous. If the alleger refuses to give a name, inform the person that:

(1) NRC will, if the alleger so requests, treat the individual's identify as confidential. (See paragraph 7 of this section for additional detail.)

(2) All matters involving public safety will be examined and evaluated. The individual's identity may, however, be revealed where required by law, when necessary to insure

public health and safety, pursuant to Congressional directives, where he himself makes the matter public or where the nature of the allegations or the limited number of people with access to the reported information may provide a basis for guessing their identity. This might be avoided if the NRC were aware of whose identity should be withheld.

(3) If the individual is alleging discrimination, refer the person to the Enforcement Officer or to an investigator in the Office of Investigations Region V. If this cannot be done, obtain as much information as possible about the problem and then inform the person that a complaint must be filed with the Department of Labor within 30 days of the acts complained of in order to obtain the Department's assistance. (See paragraph 6 of this section for additional detail.)

(4) NRC Region V policy is to send a letter to the alleger which documents the NRC's understanding of the allegation to assure that the NRC has correct information. The letter will be sent in a plain envelope with a return address shown as follows: OAC, 1450 Maria Lane, Suite 210, Walnut Creek, CA. 94596. (See paragraph 5 of this section for additional detail.)

(5) If the alleger insists on remaining anonymous, obtain as much information as possible and advise the individual to contact the Office Allegation Coordinator, collect, at (415) 943-3700 in about 30 days so that the matter may be further discussed and to ensure that the individual's concerns have been properly addressed.

2. <u>Evaluation of Allegation</u>

a. Except for those allegations that involve conditions that require immediate action such as theft of SNM, sabotage, and immediate threats to the health and safety of the public, governed by NRC Emergency procedures, within four (4) days of receipt of an allegation, the cognizant Division Director shall convene an Allegation Panel to evaluate the information and develop an action plan to resolve the matter. The Panel shall:

(1) Ensure that issues raised in the allegation are identified and understood.

(2) Evaluate the safety significance of all issues.

(3) Identify potential violations of regulatory requirements and potential enforcement action.

(4) Determine what additional information must be obtained.

(5) Consider time sensitivity.

(6) Determine how allegation should be handled, i.e., inquiry, routine inspection, special inspection or investigation. Allegations concerning technical matter, such as inadequacies in procedures, qualifications or training; inadequate implementation of procedures; inadequate corrective actions; radiation overexposures; etc. should be handled via the inspection program. Allegations involving wrongdoing such as record falsification; willful or deliberate violation of a regulatory requirement; material false statements, or improper conduct which affects licensed activities should be referred to OI for investigation.

(7) Recommend referrals to other NRC Offices, or to other Federal, State or Local governmental agencies.

(8) Identify the need for additional expert technical or investigative skills.

(9) Determine what information should be maintained confidential.

b. After evaluating the available information, the Allegation Panel shall provide the cognizant Division Director with recommendations in the form of an action plan as to what actions should be taken to appropriately resolve the matter.

c. If inquiry or inspection activities are conducted to verify or obtain additional information about an allegation, the activities should be clearly defined and the following shall be included in the action plan.

(1) If the inspection activities involve interviews of people, predetermine and include in action plan the minimum number of persons that will be interviewed; develop a series of questions to ask each individual, and record on a separate document at the time of an interview, the date, time, location, name of person, and the answers obtained plus any additional relevant information obtained during the interview. Offer the document to the interviewee to read upon completion of the interview and request the individual to sign and date the document. If the individual refuses, the inspector should so note on the document. All interview documents shall be signed and dated by the inspector. These original interview documents must be maintained as part of the official agency file.

(2) Information obtained from records to support or refute an allegation should specifically identify the source documents. If possible obtain a copy of the documents. When the records are numerous such as "weld rod issue data

forms" or "daily radiation survey forms," obtain only copies of selected samples.

For documents believed to contain vital information to support or refute an allegation, and if a copy cannot be obtained at the time, request permission to date and initial or otherwise mark the document for future identification and then hand copy or otherwise record all information contained on the document.

(3) If the inspection strategy involves sampling, make sure the technical basis of the sample size is clearly stated.

3. <u>Notifications</u>

As Chairman of an Allegation Panel, the cognizant Division Director shall assign individuals, as appropriate, to:

a. Notify Licensing Office(s) and transmit appropriate documents.

b. Notify State and Local authorities and refer issues to them that fall with their jurisdiction, e.g., OSHA violations.

c. Notify news media.

d. Issue PN or include in Daily Report (Must have Regional Administrator's approval).

e. Notify Department of Labor or other Federal Agencies.

f. Notify Director OI Region V of any potential wrongdoing by individuals that may require referral to the Department of Justice.

All notification decisions shall be made with due regard for the need to maintain the nature of the allegation and its confidential.

4. <u>Documentation</u>

a. Each allegation received shall be documented on an allegation report form prepared as called for by Appendix A to this instruction.

b. Results of evaluations of Allegation Panels shall be documented in a memorandum to files signed by the cognizant Division Director.

c. Action plans to resolve allegations shall be documented and approved by the cognizant Division Director.

d. Details and results of inquiries and inspections shall be documented in the standard IE formats except that all documents obtained during follow-up activities including original interview documents should be filed with the reports.

e. All documents including letters to and from an alleger relating to an allegation shall be filed in an appropriate facility docket file. Confidential and/or sensitive material should be marked as "official use only."

f. The purpose of all reports and other documents is to set forth sufficient facts and information in a manner such that a reasonable person will read and understand the allegation and the facts and circumstances that were found to exist or had existed concerning the matter. All reporting shall be factual and written in a style such that the NRC does not discourage persons from bringing matters to its attention. Under no circumstances is the report to be written such that it attacks or disparages the alleger. Pejorative language is to be avoided.

5. Letters to Allegers Acknowledging Receipt of Allegations

All allegations received from concerned citizens will be acknowledged by a letter to the individual who presented the allegation. This letter, in addition to stating an acknowledgement of the contact, will also contain a "Statement of Concerns" as an enclosure to the letter. The statement will detail the allegation as understood by the individual who received the allegation. The purpose of the letter is to assure the alleger that his concern will be examined as appropriate, and that the examination will address all of the specific concerns expressed by the alleger.

The Office Allegation Coordinator (OAC) is responsible for preparing acknowledgement letters to allegers. No members of the Region V staff will prepare and forward any correspondence to allegers without first coordinating such action with the OAC, to ensure that a single point of contact can be maintained for the alleger. Generally there are six types of letters which could be sent to allegers. These are as follows:

1. Normal first letter

2. Restatement of Concerns

3. Request for Additional Information

4. Close-out for Lack of Response

5. Close-out for Action Completed

Samples of the above letters are attached to this instruction as Appendix B.

6. Employee Discrimination Complaints

a. Background

A Memorandum of Understanding (MOU) signed by NRC and the Department of Labor (DOL) facilitates coordination and cooperation between the agencies in the processing of violations of the employee protection provisions of Section 210(a) of the Energy Reorganization Act. Subsequently, working arrangements were developed and points of contact established at regional and headquarters levels for each agency.

b. Working Arrangements

The working arrangements between NRC and DOL establish certain commitments that must be carried out by the regional contacts for the NRC. The working arrangements provide that NRC will refer complaints to DOL, advise DOL of complaints received concerning employee discrimination, inform DOL of investigations that NRC is conducting into these matters, and facilitate DOL investigations by assisting in gaining access to NRC-licensed facilities.

Section 210 of the Energy Reorganization Act prohibits any employer, including an NRC-licensee, applicant or a contractor or subcontractor from discriminating against any employee with respect to their compensation, terms, conditions or privileges of employment because the employee, assisted or participated, or is about to assist or participate in any manner in any action to carry out the purposes of either the Energy Reorganization Act of the Atomic Energy Act of 1954.

NRC and DOL agreed to cooperate with each other to the fullest extent possible in every case of alleged discrimination involving employees of NRC licenses, applicants, or contractors and sub-contractors. NRC will take all reasonable steps to assist DOL in obtaining access to licensed facilities and necessary security clearances. Each agency agreed to share and promote access to all information it obtains concerning a particular allegation and, to the extent permitted by law, will protect the confidentiality of information identified as sensitive that was supplied to it by the other agency.

c. Processing of Complaints

If a complaint is received concerning a possible violation of Section 210(a), the OAC will refer the complainant to the Enforcement Coordinator, the Region V point of contact responsible for the regional implementation of the NRC-DOL MOU. Region V will not normally initiate any action on such a complaint if DOL is conducting, or has completed, an investigation and found no violations; however, the matter will be documented on an Allegation Report and entered into the Region V Allegation Tracking System.

7. Confidentiality

a. Background

The ability of the NRC to obtain information, particularly adverse information from sources who wish to remain confidential, depends on the subsequent handling of such information by the NRC and its ability to protect the identity of individuals providing the information. While Public Law 95-601 makes it unlawful for employers to take retaliatory actions against employees reporting information to the NRC and provides the means for the employee to obtain legal remedies, the legal process can be lengthy, and burdensome so employees may still be reluctant to provide information for fear of being out of work for an extended period of time while going through the legal process.

Confidentiality is a means by which the NRC protects and withholds the identity of an individual who provides incriminating and/or adverse information to the NRC. It is NRC policy not to divulge to others the identity of individuals granted confidentiality, either during or subsequent to an inquiry based on the information provided to NRC.

b. Use of Confidentiality

Confidentiality should not be routinely offered to individuals making allegations or otherwise providing information during the course of an NRC inquiry, inspection or investigation. However, if a Region V staff member is of the opinion that he would not receive the information, or if the individual providing the information requests anonymity, then a grant of confidentiality will be proffered. Before confidentiality has been granted, the individual should be informed that, although the pledge is not absolute, it is NRC policy not to divulge the identity of people granted confidentiality. Also, the individual should be told that their name will not normally appear in the publicly released reports. The individual's identity may, however, be revealed where required by law, when necessary to insure public health and safety, pursuant to Congressional directives, where he himself makes the matter public or where the nature of the allegations or the limited number of people with access to the reported information may provide a basis for guessing their identity. In these cases, NRC will neither confirm or deny requests to verify the identity of a source of information. One point regarding promises of confidentiality should be clearly understood by all Region V staff members and explained to the individual providing information. A pledge of confidentiality shall not be made (or will not be honored if previously granted) if the individual provides information indicating that he intends to or has personally committed, or participated in criminal acts which may include a willful violation of NRC requirements. Should a Region V staff member grant confidentiality, all facts and circumstances surrounding the pledge must be documented in a memorandum to the OAC who will coordinate the information with the cognizant Division Director.

c. Restrictions

Within Region V, the identity of any individual making allegations, expressing concerns, or registering complaints shall be treated as "OFFICIAL USE ONLY" information. Their names shall not appear in any report (except as noted above regarding the preparation of Allegation Reports or related memorandum) or any internal memorandum or other document placed in normal mail distribution; nor will it be divulged to any NRC employee or outside individual who does not have a need for such information. If it is necessary to provide the name of an individual reporting information (alleger) to an inspector assigned to followup an allegation, or to other NRC offices, the OAC will coordinate the request for release with the cognizant Division Director. Every effort shall be made to preclude the inadvertent or premature disclosure of the identity of an individual providing information in connection with an allegation, complaint or concern.

In no case will the identity of such an individual be made known to a licensee employee without the specific approval of the cognizant Division Director. If the licensee correctly guesses the identity of the individual, the Region V staff members will respond that the NRC position is to neither confirm nor deny the validity of such guesses and refuse to discuss the matter further.

APPROVED: John B. Martin
Regional Administrator
Date:

Revision 0, October , 1983

APPENDIX A

PREPARATION OF ALLEGATION REPORT

A. Purpose

Allegation Reports (AR) serve as the basic document for initiating an allegation file within Region V. All allegations should be documented on the Allegation Report form in accordance with the following instructions:

B. Documentation

Region V personnel shall document information regarding an allegation, complaint or concern as follows:

a.	Name	-	Enter the full name of the individual providing the information.
b.	Address	-	Enter the mailing address of the individual providing the information.
c.	Phone	-	Enter the residential and/or business phone number of the individual providing the information.
d.	Allegation	-	Enter a concise statement describing the allegation, concern or complaint (e.g., improper welding procedures used in containment).
e.	Facility	-	Enter the name of the facility involved in the allegation, complaint or concern (e.g., Trojan).
f.	Docket No.	-	Enter the docket number of the facility if known.
g.	File No.	-	Leave blank. The Office Allegation Coordinator (AOC) will assign an NRC tracking number.
h.	Date and Time	-	Enter the date and time of initial contact with the individual who provided the information.
i.	Confidentiality Requested	-	If the individual who provides the information was granted confidentiality, so indicate and provide details.
j.	Summary of Information	-	Enter the details of the information provided by the individual.
k.	Prepared By	-	Enter your printed name and signature.
l.	Date	-	Enter date document was prepared.
m.	Action Required	-	Leave blank. OAC will use this space for internal administrative actions.

n. Reviewed By - Leave blank.

o. Date - Leave blank.

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16. ABSTRACT (200 words or less)

Supplement No. 21 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate the Diablo Canyon Nuclear Power Plant (Docket Nos. 50-275 and 50-323), located in San Luis Obispo County, California, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement provides information on the Commission's review of allegations and concerns about the design, construction and operation of Diablo Canyon.

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