NUREG-0675
Supplement No. 22

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 22 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate Diablo Canyon Nuclear Power Plants, Unit 1 and 2 (Docket Nos. 50-275 and 50-323), has been prepared jointly by the Office of Nuclear Reactor Regulation and the Region V Office of the U. S. Nuclear Regulatory Commission. This supplement provides the criteria that were used by the staff to determine which of the allegations that have been evaluated must be resolved prior to Unit 1 achieving criticality and operating at power level up to 5 percent of rated power (i.e. low power operation). The supplement also reports on the status of the staff's investigation, inspection and evaluation of 219 allegations or concerns that have been identified to the NRC as of March 9, 1984, excluding those recently received under 10 CFR 2.206 petitions.

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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 Plants, Units 1 and 2. The SER has since been supplemented by Supplements No. 1 through No. 21. SSER 18, 19 and 20 presented the staff's safety evaluation on matters related to the design verification efforts 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. SSER 21 presented the program and the status of the staff review and evaluation of allegations and concerns identified to the NRC as of December 19, 1983. This is SER Supplement No. 22 (SSER 22) and is based on allegations and concerns identified to the NRC as of March 9, 1984.

This supplement provides the criteria that were used by the staff to determine which of the allegations that have been evaluated so far must be resolved prior to Unit 1 achieving criticality and operating at power level up to 5 percent of rated power (i.e. low power operation).

SSER 22 also presents the staff's safety evaluation of these 219 allegations. The staff evaluation of allegations and concerns is presented as Appendix E to the Safety Evaluation Report, consistent with the format of SSER 21. As of March 9, 1984, 219 individual allegations or concerns have been addressed by the staff. In addition, submittals were received in the form of 2.206 petitions from the Government Accountability Project (GAP) on February 2, 1984 and on March 1, 1984 which contain additional allegations. The staff has not yet been able to evaluate or categorize these new submittals in depth.

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APPENDIX E

STATUS OF STAFF RESOLUTION

OF

ALLEGATIONS OR CONCERNS

ABOUT

THE CONSTRUCTION

AND

OPERATION OF DIABLO CANYON

UNIT 1 AND 2

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1.0 Introduction

In early 1982 during the course of the Diablo Canyon Unit 1 design verification program certain allegations were made to the staff regarding the design and operation of the Unit 1 component cooling water system and certain other design aspects. The staff reviewed and evaluated the allegations on the basis of discussions with the individual expressing the concerns and issued its safety evaluation in Supplement No. 16 to the Safety Evaluation Report (SSER 16). Since then numerous additional allegations have been made and concerns expressed regarding the design, construction and operation of the Diablo Canyon Nuclear Power Plant and the licensee's management of these activities. In many cases the allegations include some aspect of quality assurance or quality control. The allegations were received by the NRC staff in the Region V Offices and at Headquarters as well as by the Commission. They were made by a variety of sources, including private citizens, former and current workers at the plant and at the PG&E and Bechtel Offices, news media, intervenors, and Congressional Offices. In some cases the source has remained completely anonymous to the NRC, in some cases the source is known only to the NRC, however, in most cases the source has been publicly identified. In many cases one source identified many items in a single submittal. In some cases the same allegation or concern was raised by more than one source. However, such same allegations from different sources were not combined in order to maintain a record of each item separately.

As a result of the numerous allegations the Commission directed the staff on October 28, 1983 to pursue all allegations and concerns to resolution and requested a status report on the investigation, inspection and evaluation effort prior to its decision regarding authorization of criticality and low power testing. The staff subsequently developed the Diablo Canyon Allegation Management Program (DCAMP) which was provided to the Commission on November 29, 1983 in a memorandum from the Executive Director for Operations. A summary of the program and the methodology applied are presented in Section 2 of this report. The program was described in detail in SER Supplement 21.

The staff is performing its investigation, inspection and evaluation of the allegations in accordance with the DCAMP. In late December the staff provided a status of its efforts in SSER 21 on those allegations that had been received by the NRC as of December 19, 1983. The staff provided the Commission with written summaries of its ongoing efforts on January 4, 1984 (SECY 84-3) and February 6, 1984 (SECY 84-61) and verbally briefed the Commission on January 23 and February 10, 1984.

SSER 21 included, as an attachment, an Individual Assessment Summary for each of the allegations. In some cases the summary contained sensitive information or was predecisional in nature, in that the disclosure could impair the staff's ability to initiate and/or conduct appropriate investigations or inspections. These summaries were issued separately, with a limited distribution consistent with the Commission's August 5, 1983, Statement of Policy on Investigations and Adjudicatory Proceedings (48 Fed. Reg. 36358).

As of March 9, 1984, 219 individual allegations or concerns have been addressed by the staff. In addition, submittals were received in the form of 2.206 petitions from the Government Accountability Project (GAP) on February 2, 1984 and on March 1, 1984, which contain many additional allegations. The staff has not yet been able to evaluate or categorize these new submittals in depth. This supplement provides the criteria that were used by the staff to determine which of the allegations that have been evaluated must be resolved prior to Unit 1 achieving criticality and operating at power level up to 5 percent of rated power (i.e. low power operation). SSER 22 also presents the staff's safety evaluation of these 219 allegations.

2. Diablo Canyon Allegation Management Program

2.1 Scope

The Diablo Canyon Allegation Management Program (DCAMP) encompasses all allegations or expressions of concern which may be construed as allegations, which pertain to the design, construction, and operation of safety-related structures, systems and components at the Diablo Canyon Nuclear Power Plant, and which pertain to the PG&E management of the Diablo Canyon Nuclear Power Plant project. In this regard the DCAMP also includes concerns raised by the public and 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 desire of an alleger 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.

2.2 Approach

The fundamental approach in addressing the allegations to date has been to focus on two basic questions.

> Firstly, does the allegation present a technical problem which could affect safety of the plant?
>
> Secondly, does the allegation reveal any significant defects in the licensee's or his contractor's management or quality systems?

The general sequence of steps was as follows:

<u>Confirmation of Allegation</u>:

> As each allegation or concern was received an effort was normally 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. When 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.

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. In some cases additional, independent measurements and evaluations were performed where appropriate.

Technical Reviews

Consideration of allegations in technical areas previously reviewed by the staff included detailed evaluations using licensing documents, regulations, standards, additional information provided by the licensee, and independent analyses as necessary. In some cases additional audits were performed at the site or in the offices of the licensee and its contractors as necessary.

Interviews:

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

Public Meetings:

Where significant technical meetings were held, verbatim transcripts were generally taken to maintain an appropriate record.

Feedback from Allegers:

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

Allegation Management Instruction:

Region V's instruction on allegation management was used as guidance for this process. The draft instruction (entitled "Management of Allegations") was provided as Attachment 4 to SSER No. 21.

The staff examined in detail almost all of the first 180 allegations.[1/] The purpose in doing this was to gain an overall perspective of not only the technical aspects of the problems raised but also to use the specific allegation as a vehicle for assessing whether the licensee and its major contractors acted responsibly over the years. Considerable insight was developed on the licensee's and contractor's management control and quality control activities.

[1/] The allegations were not addressed in the same sequence as presented in Attachment 1.

As the picture began to develop, the staff started using more discretion on which individual allegations merited a detailed review. The staff elected not to review about 30 allegations in detail. These are issues which are either very similar to those already reviewed in detail or, based on an assessment review, do not relate to significant safety issues. The reasoning was that to do so would not add significantly to the management or quality performance issue. The staff either has or plans to request the licensee to address most of these from a technical standpoint with the staff auditing the licensee's response. Allegations in this category are identified on the individual sheets in Attachment 4. The staff continued to look into those allegations which appear to be unique, or which seem to present management control or quality issues not previously considered or those where alleger confidentiality was an issue. The staff plans to use this more discretionary approach in reviewing the unaddressed and future allegations.

3. Status Summary of Staff Effort

The staff review has to date involved more than 40 NRC technical staff (inspectors, engineers and investigators) from all NRC Regional Offices and Headquarters including contractor personnel. Collectively, these individuals have expended in excess of 18,000 manhours since early November 1983 examining and evaluating the allegations or concerns. During its inspection and evaluation of allegations the staff did not restrict itself to the allegation itself, but expanded its efforts beyond the original scope of the allegation whenever it considered this to be necessary. These efforts provide the staff with a substantial basis for understanding the technical concerns raised and also the perspective necessary for making conclusions regarding the effectiveness of the management and quality systems employed at the site.

In summary, of the 219 allegations addressed 146 items are considered resolved, 73 are unresolved. Of the 73 unresolved items the staff has determined that none require a resolution prior to criticality and operation up to 5 percent power (see also Section 5 of this report), 16 must be resolved prior to exceeding 5 percent power, the resolution of 57 items does not impact low or full power operation, and there are no items for which the resolution status has not been determined. Attachments 2 and 3 provide an overview of the status in a diagram and table, respectively.

The staff action for allegations or concerns is summarized in the Individual Assessment Summaries, Attachment 4. As discussed in Section 1 of this report, in some cases the Individual Assessment Summary contains sensitive information or is predecisional in nature. These summaries are not included in Attachment 4, but are 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).

hand calculations for small bore piping supports was acceptably low. In light of these findings the staff will require that PG&E establish a program to review all computer analyses for small bore piping supports.

In partial response to those staff findings the licensee has reported the results of a review of approximately 130 small bore piping support computer analyses including the analyses in which the staff had previously identified errors. The licensee reported that, with errors corrected where necessary, all completed calculations showed final acceptability of the supports. The staff conducted a special inspection to evaluate the process used to re-review the small bore piping calculation packages. We found with minor exception, that the review process was comprehensive, was being carried out by qualified individuals, and was conducted in a manner to assure that the results could be accepted with high confidence.

Analyses of the type and significance of the deficiencies seen to date has led the staff to conclude that, although the design QA program for the OPEG is not up to acceptable standards, the impact in terms of design adequacy, has not been significant.

Based on the results of the staff's review to date and the types of errors that have been identified it is very likely that modifications, if any, would be minor and only to fully meet seismic criteria with little or no impact on operability of systems under the full range of plant operations.

Since some piping support modifications are normally required as a result of initial plant operation, due to unexpected thermal motions or operating requirements of attached or supported equipment, there is sound logic in conducting the required calculation review during low power operation so that any resulting modifications could be included in an orderly and consolidated program prior to full power operation.

3. Prior to exceeding 5 percent power those allegations or concerns must be resolved which offer specific new information, not previously available to the staff, and which may reasonably be expected to involve sizeable failures of systems that contain radioactivity or of the ECCS systems. In addition, sufficient technical information regarding these allegations or concerns is not presently available to the staff, or programs have not been developed or implemented to assure that regulatory concerns related to reactor safety will be resolved prior to exceeding 5 percent power.

In formulating these criteria the staff emphasized that the new information must be definitive, specific and credible. As the staff has gained experience in evaluating the first 200 allegations addressed in this report it developed reasonable confidence to conclude that the licensee and its contractors have acted responsibly over the years. Although there have been some lapses the quality and management systems related to construction have worked reasonably well. As a result of this perspective gained the staff feels that the burden has shifted somewhat such that allegations of a general or circumstantial nature should not be "assumed true until proven otherwise".

5. Allegations Related to Reactor Criticality Considerations

In SSER 21 and SECY 84-61 the staff identified seven areas of concern (involving 21 allegations) which required resolution prior to reactor criticality and low power operation. Since early of this year the staff has pursued the resolution of these issues with the highest priority and has devoted extensive effort to the inspections and evaluation of these matters. As a result the staff reviews have progressed to the point that the issues are either completely resolved or resolved to the point where they no longer warrant full resolution prior to reactor criticality considerations. The status of each of these issues is provided below.

5.1 Small Bore Piping Design Adequacy (Allegation: 55, 79, 82, 86, 87, 88, 89, 89, 95, 97):

In the course of investigating the numerous allegations concerning the design of small bore piping supports the staff reviewed a large quantity of material concerning general design practices, implementation of design control measures and the conduct of specific analyses. These efforts included inspections at the On-Site Project Engineering Group (OPEG), the essentially self-contained engineering group responsible for small bore piping design and analyses at the Diablo Canyon Site, and inspections at the San Francisco offices of PG&E and the Bechtel Corporation.

As a result of these inspections a number of the allegations related to the administration of the OPEG were substantiated in whole or in part. Specifically, allegations related to deficiencies in document control at the site, site specific training and effective use of deficiency reports were substantiated.

The principal technical finding is that the analyses performed by computer for small bore piping supports have been determined to have an unexpectedly large error rate, on the order of twenty percent as compared to ten or less percent that experience has shown is likely. On the other hand the error rate in the

At this time the Diablo Canyon Unit 1 reactor is fueled completely with new, unirradiated fuel without any fission products. During low power operation the amounts of fission products in the reactor would be approximately proportional to the power level for short-lived radioisotopes and to the total energy produced for long lived radioisotopes. Even after several months of low power operation, the fission product inventory would still be one to two orders of magnitude less than the amount assumed in our safety evaluation. Possible accident consequences would be further reduced since the decay heat is also decreased, not only in the rate at which it is released but also in the total amount available. The energy required to damage the reactor in a postulated accident and the capacity of the plant heat removal systems and safety features are not reduced during low power operation. Therefore, postulated accidents involving a failure of these systems would require much longer times to evolve and could be contained by equipment operating at only a few percent of its design capacity. In summary, the possible consequences of a reactor accident during low power operation are limited to a very small fraction of those possible at full power.

Taking these factors into consideration the staff applied the following criteria for assessing which allegation and concern requires resolution prior to criticality:

1. Prior to criticality those allegations or concerns must be resolved which offer specific new information, not previously available to the staff, and which appear to involve a discrepancy between design criteria, design, construction or operation of a safety-related component, system, or structure of such magnitude so as to cause the operability to be drawn into question. In addition, sufficient technical information regarding these allegations or concerns is not presently available to the staff, or programs have not been developed or implemented to assure that regulatory concerns related to reactor safety will be resolved prior to criticality

2. Prior to criticality those allegations or concerns must be resolved which offer definitive new information, not previously available to the staff, and which indicate a potential, significant deficiency in the licensee's management or quality assurance of safety-related activities. In addition, sufficient technical information regarding these allegations or concerns is not presently available to the staff, or programs have not been developed or implemented to assure that regulatory concerns related to reactor safety will be resolved prior to criticality.

In addition, the staff applied a third criterion as follows to determine which allegations or concerns must be resolved prior to exceeding 5 percent power:

4. Criteria for Priority Resolution of Allegations

During the staff evaluation of the first 219 allegations criteria evolved to be applied to identify those allegations which need to be pursued and resolved with the highest priority due to their significance regarding criticality and low power operation. Particular consideration was given as to whether or not an issue caused operability to be drawn into question or whether a significant deficiency in management or quality was indicated. During the preliminary review the following considerations were applied:

- Is the allegation a specific safety or quality issue or a generalized concern?
- Has the staff previously addressed this issue?
- Has the issue been previously dealt with or is it now being dealt with by the licensee?
- Is the allegation reasonable and does it sound competent?
- Does the allegation represent a significant safety or management concern?

In addition to these considerations the staff considered two specific aspects in making its determination as to whether the allegation must be satisfactorily resolved or not resolved prior to criticality and low power operation. The two aspects are experience gained and fission product inventory resulting from low power operation. Both are addressed below.

The operation of Diablo Canyon Unit 1 at low power utilizes most of the same systems as at full power. Furthermore, systems and components will operate and be exposed to design pressure and temperature. Operation at low power would therefore provide a means to determine and evaluate the plant performance under more realistic conditions. In particular, such operation would expose the plant to actual thermal stresses and would result in and identify any interferences between pipes and supports and restraints under operating conditions. Therefore, a systematic low power operation program would identify deficiencies or confirm analytically determined deficiencies, if any, that subsequently could be corrected.

The staff concludes that in the overall quality control inspectors were properly qualified for the tasks they performed. Accordingly, the staff considers that this issue has been adequately addressed for the purpose of licensing decisions.

5.4 Design Change Notice and Drawing Control (Allegation 61 and 102):

The staff examined the licensees and contractors programs for the control and issuance of design change notices and related drawings. The staff determined that the controls applied to these activities were generally adequate. At the time of issuance of SSER 21 the staff had identified a particularly complex design change notice and its related drawings for further analysis. This change notice involved approximately 130 major and minor revisions. At the staff's request the responsible engineering personnel met with the staff and presented documentary evidence that each revision was either completed, superceded, or voided. The licensee also showed the staff the completed start-up test reports for this system which demonstrated that the system operated as intended. Based upon these results and additional programmatic and technical reviews the staff concluded that change notices and related drawings were adequately controlled and implemented. This issue is considered adequately resolved for purposes of licensing decisions.

5.5 Falsification of Vendor Records (Allegation 99):

This allegation came to the NRC staff attention through a local San Francisco television reporter. Staff action was initiated at that time. In addition, the licensee initiated its investigation of this subject after viewing the television report. Since the original allegations were received the staff and the licensee, through their investigations, have received two groups of additional allegations.

The NRC staff response to the allegations includes a combined effort by the Office of Investigations, the Licensee Contractor and Vendor Inspection Program Branch of the Office of Inspection and Enforcement, and Region V. The staff position has been both one of monitoring how the licensee is conducting its investigation for the Diablo Canyon Project and independently reviewing the issues for generic significance (the company has provided products to multiple nuclear reactor projects).

The staff has addressed and closed the original allegation. A review of pertinent records established that the former inspector (who claims to have documented inspections he did not perform) is credited with performing 650 inspections while he was employed at the vendor. Fifteen of the 650 inspections involve safety-related material. These fifteen items were found to be supplied to Diablo Canyon Unit 2 and involve "stock" material (i.e. raw material items which do not involve welding). As of this writing the staff has inspected 14 of the 15 items and found them to conform with requirements. The staff is following up on the last item (plate washers).

5.2 Anchor Bolt Design Margins and Installation (Allegations 25, 58, 96, 142, 154, 176):

The concerns raised by these allegations involve the installation and inspection of concrete expansion anchors by the H. P. Foley Company (primary electrical contractor and construction completion contractor). A general and non specific concern with anchor bolts was supplied initially to the staff from an anonymous alleger. Subsequent interviews of onsite contractor personnel resulted in additional concerns with added detail in some cases. The staff approach to resolution of these isues was to: (1) review installation procedures, audits, nonconformance reports, discrepancy reports, and licensee correspondence relating to concrete anchor bolts; (2) have an independent NRC contract team (Lawrence Livermore National Laboratory) inspect a sample of 124 electrical raceway supports modified in 1982 (involving hundreds of anchor bolts); and (3) request the licensee to perform torque tests and ultrasonic examination on a sample of 40 installed anchor bolts to verify the adequacy of installation. The staff found that none of the allegations involved a substantive quality or management control problem. During the course of this review, however, the staff identified a number of their own technical concerns related to anchor bolt adequacy. In response to a staff request the licensee undertook an extensive test and evaluation program. The results of this program were reported to the NRC, concluding that adequate margins of safety were provided in the installed anchor bolts.

Based on the results of the test program the staff concludes that there is reasonable assurance that installed anchor bolts are adequate. Accordingly, the staff considers this issue adequately resolved for the purpose of licensing decisions.

5.3 Inspector Certification (Allegations 57 and 68):

In response to the allegations concerning certification of quality control inspectors employed by both the H. P. Foley Company and by the Pullman Power Products Company (primary piping installation contractor) at the Diablo Canyon project, the staff examined the contractor's programs and their implementation in effect during the companies' activities to assess whether appropriately qualified persons performed quality control inspections of safety related items. The staff concluded from their examination that there is reasonable assurance that individuals performing quality control inspection were qualified to perform their assigned tasks with the exception of a case involving Pullman Power Product Company during the 1973-74 time frame. In this case certain QC inspectors were found to have been performing inspections prior to completely satisfying prescribed certification requirements. All but two of these individuals had adequate backgrounds and experience in the areas of welding and quality control inspection. It does not appear that this problem was chronic or widespread. The licensee has committed to complete a sample reinspection of the inspectors' work prior to the time that they were fully certified to perform the related visual inspections. This effort will be completed by March 30, 1984.

The licensee has selected a 10% sample of the other (non-safety related) inspections related to the inspector and performed a reinspection (involving 940 welds). Seven of the 940 reinspected welds were found to have deviations from requirements, these are being properly addressed. Based upon the low defect rate the licensee has concluded that the structures and components installed at Diablo Canyon have not been adversely impacted by the former inspector's alleged performance. The staff concurs with this conclusion based upon a review of licensee actions and independent inspection of the fifteen safety-related items.

Neither the licensee nor the staff can determine conclusively whether the former inspector neglected to do the inspections.

The staff has completed a substantial amount of review on the second and third groups of allegations, and to date has not identified problems of safety significance, the reviews, however, are continuing (e.g. the staff has not completed their review of the operations at the vendors subsidiary). These allegations are mainly general in nature, lacking in specific examples thus requiring extensive interviewing and document reviews.

In a parallel effort the licensee has initiated an inspection of installed hardware to allow a direct assessment of material adequacy, separate from the management and programmatic concerns related to the vendor. Items that are being reinspected were selected by reviewing all shop drawings and selected purchase orders involving the vendor's material shipped to the jobsite since 1969 and includes samples of each material type supplied to Diablo Canyon with particular attention to items which are difficult to fabricate or involve special materials.

90% of the sampling has been completed and the licensee reports that the following trends and results are apparent:

a) General inspections are finding that the existing geometries and dimensions are in conformance with the shop drawings.

b) Hardness tests are indicating that correct materials were provided.

c) Visual weld inspections are indicating that vendor welding meets design requirements.

d) Records from the NDE documentation research show that full penetration welds by the vendor are satisfactory.

In addition to the licensee's reinspection the staff has independently inspected a small sample (14 types of components) of installed safety related hardware to obtain first hand evidence of product quality. The components were visually inspected for material damage, weld location, length, size, shape, reinforcement, appearance and type. The staff did not identify any discrepant material. Records related to this material were reviewed and appeared to be in order.

Investigations and reviews have been completed on the initial and most alarming allegations. This item is resolved. The reviews are continuing on the other two sets, but, to date significant safety problems have not been identified. Based upon staff findings to date and the acceptable results of reinspection of installed hardware it is the staff's opinion that this issue no longer requires full resolution prior to licensing decisions.

5.6 Weld Symbol Implementation (Allegation No. 126)

The staff received an allegation on December 20, 1983, that alleged that a major problem existed with the licensee's home office and site engineering because no welding symbol standard (such as AWS A2.4) had been implemented at Diablo Canyon. The staff reviewed the alleger's concern and determined that his concerns had merit. The staff subsequently requested that the licensee address this by providing the following information:

- Assessment of the safety significance of the inconsistent weld symbol application.

- Assessment of the weld symbol interpretations used by organizations engaged in welding activities in the field.

- Performance of such field examinations as deemed necessary to establish whether any inconsistencies in interpretations caused a failure of the field welding activities to conform to the designers intent.

On February 2, 1984, the licensee provided their position on the acceptability of the Diablo Canyon weld design and installation program. The staff's review indicated that though the licensee was not required to comply with AWS A2.4, the licensee's program generally met the criteria of AWS A2.4 for welding symbology. Additionally, the licensee did have usable alternate programs for the clarification and interpretation of weld symbols. The staff notes that the NRC inspection and reviews have not identified any instance where the failure by the licensee to fully implement the AWS A2.4 welding symbology, resulted in weldments which would not meet the designer's intentions. This issue is considered resolved.

5.7 Cable Spreading Room Platform Adequacy (There is no specific allegation related to this topic. A staff concern was identified in this area while examining documentation related to anchor bolts).

During a walkdown of cable tray and conduit supports on January 14, 1984, the NRC inspector identified two Class I Electrical Raceway Supports attached to the Non-Class I steel supporting a platform in the cable spreading room. The inspector also noticed several deficiencies in the installation of the concrete anchor bolts securing the structural steel to the concrete.

A review of records disclosed that the deficiencies in the anchorage of the structural steel had been previously identified by a Foley inspector on October 7, 1983. The inspector observed from his review of the records that the platform steel was not designated Class I (safety-related) despite the fact that this structural steel was being used to support Class 1E electrical panels in the cable spreading room.

The condition identified by the NRC inspection was documented in a nonconformance report and provided to engineering for assessment of technical adequacy.

This issue was addressed in the licensee's letter to Region V (No. DCL-84-047), dated February 7, 1984. The licensee determined the as-built condition of the cable spreading room platform installation. The as-built condition was analyzed by the licensee's engineering verifying that the installed condition was acceptable and conformed with design requirements. In assessing the generic implications of this issue it was determined that the unique nature of the steel-frame raised-floor configuration led to the acceptance of the design and material without the detailed type of as-builting and analysis that was performed for the other structures. This type of configuration exists only in the cable spreading rooms. All other platforms which support Class I equipment have been analyzed. Therefore, this installation is not a generic issue.

The staff concludes that the licensee has adequately demonstrated the acceptability of the cable spreading room platform installation. The staff considers that this issue is resolved and does not require further action.

6. Concerns Relating to Employee Intimidation

A few of the allegations received by the staff related to possible intimidation of workers at the plant. The staff took specific action to assess whether this condition was a widespread problem or concern at the facility. The staff effort on Diablo Canyon allegations involved several thousand staff man-hours on-site, where staff members have interfaced with hundreds of licensee and contractor crafts, quality personnel, engineering personnel, supervisors, and managers. During the course of this effort the staff was instructed to be alert and look for evidence of "corner cutting" or pressure by management that would be counter to good quality practice. The staff interactions with site personnel included informal one-on-one discussions, group discussions, and formal meetings. The staff also observed groups and individuals interacting among themselves in very casual situations (such as during plant tours, and lunch room and work area discussions). These types of observations have been useful in gathering a subjective sense for the overall plant "atmosphere" regarding issues such as freedom to discuss concerns or intimidation. In addition, approximately 250 site personnel were specifically questioned regarding such items as pressures to "cut corners", intimidation, or freedom to bring forth quality and safety related concerns. These interviews were conducted, in part, to determine if there was a generalized atmosphere to repress problems or safety concerns.

Based on the staff work in this area it appears that a few individuals feel strongly that they have been directly intimidated. Some have offered specific and detailed reports in support of their allegation. These cases are complex. The staff could not readily tell whether the cases involve intimidation, proper exercise of management perogatives, or just poor communication. As appropriate, these few cases (eight total) are being addressed through the Department of Labor regulatory process, and/or review by the NRC Office of Investigations. A few additional individuals were concerned about intimidation but indicated their views stemmed from events not directly related to them, such as general perceptions that the pressure was on to get the job done, or from the layoff or firing of another employee, or media reports of intimidation. The staff does not detect any widespread company attitude (either deliberate or inadvertent) to suppress employee concerns or corrupt the overall effectiveness of the Quality Assurance Program. The staff also found that in the vast majority of interactions employees are not afraid to come forward with reports of, and deal with, quality problems in a responsible manner both with their own organizations and with the NRC.

While the staff concludes that a widespready suppression problem does not exist at Diablo Canyon the staff is concerned with employee perceptions in this area. Licensee management shares this concern. The staff has reviewed this subject with licensee management and notes that the licensee has undertaken steps to make improvements. This effort includes such actions as the development of video tape presentations for all existing and new employees regarding surfacing of quality concerns; an "800" telephone number for receiving quality concerns; and a system for receipt and control of concerns. The licensee's activities in this area will be monitored by the staff.

7. Summary and Conclusions

1. As of March 9, 1984 a total of 219 allegations or concerns have been addressed by the NRC.

2. The staff has developed criteria that have been used to determine which allegations or concerns must be resolved prior to (a) criticality and low power operation and (b) full power operation.

3. As of March 9, 1984 the staff has concluded that none of these allegations require resolution prior to a reactor criticality decision. The staff has concluded that the final resolution of 12 separate allegations relating to two subjects can be deferred from precriticality to pre-full power.

4. The staff has concluded on the basis of its investigation, inspection, and evaluation, that there have been some lapses in the quality and management systems related to construction, however the systems have worked reasonably well. The staff has reasonable confidence that the licensee and its contractors have acted responsibly over the years.

5. The staff is continuing its investigation, inspection and evaluation of all unresolved allegations and concerns.

6. The staff effort is sufficiently complete regarding the 219 allegations to conclude that none of the allegations indicate problems of such a magnitude, either individually or collectively, that should preclude authorization for criticality and low power operation.

ATTACHMENT 1

DIABLO CANYON

LIST OF ALLEGATIONS OR CONCERNS

AS OF MARCH 9, 1984

LIST OF ALLEGATIONS

Allegation

1. Passing of contraband
2. Anti-nuclear demonstration
3. Seismic qualification of CCW
4. Single failure capability of CCW
5. Heat removal capability of CCW
6. I&C design classification

6a. Feedwater isolation classification

7. Seismic Category I/Category II interface
8. Seismic design of diesel generator intake and exhaust
9. NRC staff concern regarding USI-17: Systems Interaction
10. Tilting of containment
11. Classification of platform
12. High energy line break analysis 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 inadequate SSE load
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. Foley NCR's rejected without good cause
25. Deficiency in use of "Red Head" anchors for raceway support
26. Foley did not document NCR's issued by field inspectors
27. Welding and QA deficiency in "Super Strut"
28. Annulus structure reverification
29. Pipe restraint design inadequate
30. Inadequate documentation of safety-related equipment
31. QA procedures for structural analysis
32. Seismic analysis of containment
33. Turbine Building (Class 2) contains Class 1 systems & components
34. Incomplete as-built drawings
35. Lack of support calculations for fluorescent light fixtures
36. Resolution of fluorescent light fixture interaction
37. Solid state protection system relays
38. PG&E ignoring spurious closure of MOV
39. No control room annunciation of closed RHR suction valve
40. RHR hot leg suction does not meet single failure
41. Drawings inadequate
42. Licensee management unresponsive to problems
43. Licensee reporting failure

44. Licensee improper assessment of DCN

45. Design inconsistency in FSAR for RHR valves

46. Foley QA procedures voiding of NCR's incorrect

47. Plant paging/announcing system

48. Systems interaction study and associated modifications

49. Emergency sirens not seismically qualified

50. Plant security should have been retained

51. Risk of job action against allegers

52. Construction and hearings 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" anchor studs reported as improperly installed

59. Foley lost cable traceability

60. Foley purchased material through unapproved vendors

61. Lack of document control

61a. Foley used unapproved drawing

62. Foley lacks adequate sampling of cable-pull activities

63. Foley lost material traceability through upgrade of non Class 1 to Class 1

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

65. Foley QA documents prior to 1980 in question

66. Defective weld reports rejected by Foley

67. Negligence by PG&E regarding flooding in auxiliary building
68. NSC audit of Pullman-Kellog
69. Revision of "Draft Case Study C"
70. Inadequate 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 accumulator
76. U-bolts have failed
77. Flange bent on I-beam
78. Bracket bolted to wall with only one 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
82. Minimal orientation for new engineers at site
83. NRC was not effective in identifying problems
84. Lack of responsiveness by management to identified design problems
85. U-bolt design
86. "Code break" design
87. Calculations related to "code break" design destroyed
88. Undocumented modifications made because of "code break" problems
89. Interference of pipe supports (attempted use of uni-strut)
90. Defective concrete in intake structure

91. Alleged cover-up of defective material

92. Flare bevel welds undersized and not complying with Code

93. Inaccurate depiction of welds on drawings

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 required to use uncontrolled documents

98. Possible non-adherence of pentration seal procedure

99. Falsification of welding quality control records

100. No quality control program for coatings

101. Qualification of welders and procedures

102. Improper references on DCN

103. Structural shapes not listed on WPS

104. Materials not listed in AWS code

105. Weld joint geometry not specified by the WPS

106. AWS 1-1 technique sheet not utilized

107. AWS 1-1 technique sheet improperly authorized

108. AWS 1-1 technique sheet listed non-ANS code steel

109. Contract specification for pipe support welding not followed

110. Pipe supports not welded in accordance with AWS 1-1

111. Welders qualified to ASME 1X (ESD 216)

112. Welders qualified to AWS D1.1 (ESD 243)

113. Contract specification not officially changed

114. Notch toughness requirement not followed

115. Unauthorized change to UT requirement in contract specifications

116. Code 88/89 used to weld plate instead of pipe

117. Code 88/89 not qualified per AWS D1.1

118. Technique sheet AWS 1-1 allows CTAW

119. Technique sheet allows materials not used in AWS code

120. Pullman-possible intimidation of personnel

121. Pullman inadequacies in valve wall thickness measurement activities

122. Pullman inadequacies in nondestructive testing activities and audits

123. Improper acceptance of welder qualification tests

124 Responses to audits were not timely

125. Pipe rupture restraint welds were not tested per specification

126. Inconsistent set of weld symbols for engineers and contractors

127. Preheat requirements not followed for certain welds

128. Pullman did not properly accept problem reports

129. Improper activities related to Pullman welding

130. Pullman-possible intimidation of personnel

131. Pullman welded bolts and studs to containment liner w/o qualified WPS

132. Pullman welded plate to CCW piping while piping contained water

133. Foley did not properly accept/document reports

134. Foley did not invoke Part 21 on vendor contracts

135. Foley audits were not performed for an extended period

136. Foley audit findings were not properly handled

137. Foley did not audit procedure adequacy

138. Foley lost wire traceability for incore thermocouple circuits

139. Foley improperly performed tubing fabrication (socket welding and bending)

140. Foley used material purchased for one contract on another

141. Foley performed transverse welding across beams (installation of unistrut)

142. Foley inadequately installed and checked anchor bolts

143. Foley did not torque beam clamps at installation

144. Foley installs P1100 conduit clamps too close to channel edges

145. Foley did not specify raceway materials in details

146. Foley does not keep raceways free of damaging debris

147. Foley installs different vital systems on single support

148. Foley QC identifying unsatisfactory work

149. Foley did not submit HVAC as-built information during 1981/82

150. Foley may have falsified structural steel and tubing heat records

151. Foley installs too many conduits or supports

152. Concerns with installation of P1331 conduit clamps

153. Foley specifies 1/8" welds or 3/32" clamp material

154. Foley does not specifcy adequate inspection criteria for anchor bolts

155. Welding on embedded plates causes distortion

156. Foley-possible intimidation of personnel

157. Pullman-possible intimidation of personnel

158. Unit 2 annulus design-inadequate seismic load combinations

159. Unit 2 annulus design-steel members may be over stressed

160. Unit 2 annulus design-bracings carry axial loads and supports

161. Unit 2 annulus design many assumptions of Class II and small bore loads

162. Unit 2 annulus design-calculations changed by reviewers

163. Unit 2 annulus design-assumptions related to thermal expansion

164. Unit 2 annulus design-beams not checked for tearing failure mode

165. Unit 2 annulus design-code check did not account for torsional stresses

166. Foley correction to QA documents by QC with inadequate guidelines

167. Foley not reviewing all records in preparation for turnover

168. Foley did not properly grout base plate anchor bolts

169. Pullman failed to conduct support welds as required by procedures

170. Pullman lost pipe traceability, inadequate training of fab shop inspectors

171. Inadequate planning and routing of cables

172. Transfer of cable to alternate reels

173. Improper clearing of cable ways before pulling cables

174. Inadequate control of tension levels when pulling cables

175. Changes from interim "as built" drawings to final drawing

176. Anchor bolts (torquing of "red-head" bolts)

177. Potential damage to RHR pumps due to suction line valve control

178. Boron worth vs. temperature written in 1976 to 1978 may be in error

179. Concern that auxiliary salt water pump flow does not meet FSAR flow rates

180. CCW heat exchanger inlet valves were broken due to water hammer

181. Diesel generator surveillance test records are inaccurate and incomplete

182. Bolts on CVCS, RHR, RCS, PORV's and safety valves do not meet ASME Specs

183. Alleged use of hard drugs

184. Unqualified fire stop designs being used

185. No QA being practiced during fire stop installation

186. Operators do not know how to operate two component foam equipment

187. Many foam seals are not good

188. QA breakdown at Pullman

189. Magnaflux weld verification program accepted bad welds

190. Pipe support base plate installation do not define bearing surface

191. PG&E has attitude that "QC finds too many problems"

192. Acceptance criteria changed to decrease weld failure rate

193. Poor QC inspector selection and training

194. Document control is informal (rules made up as they go along)

195. Document control stamps are not controlled

196. Intimidation by a Foley QC person against a supervisor

197. Intimidation by a Foley QC person on subordinates

198. Foley QC person handles work packages incorrectly

199. Foley QC rushing work to meet schedules

200. NDE Reports inconsistent with contractors inspection reports

201. NDE Reports changed w/o proper approvals

202. Falsification of weld x-rays

203. Square tubing for seismic supports is uncontrolled

204. Contractor engineering modified PG&E drawings

205. Unqualified electrical splices on solenoids

206. Electrical conduit may not be controlled

207. Inadequate training for Pullman work activity

208. Unacceptable management attitude for resolution of deficiency reports

209. Pullman supervision qualification inadequate

210. Qualifications of other plant workers is questionable

211. Welding not in accordance with ASME Section IX

212. Weld materials not properly qualified

213. Inadequate design of all raceway supports and other allegations

214. Code 7/8 and 92/93 not technically the same

215. Code 92/93 not qualified for unlimited thickness

216. Code 7/8 and Code 92/93 not interchangeable

217. Pullman performed a QA coverup through use of 1978 memo

Note: counting allegations 6a and 61a there are a total of 219 allegations

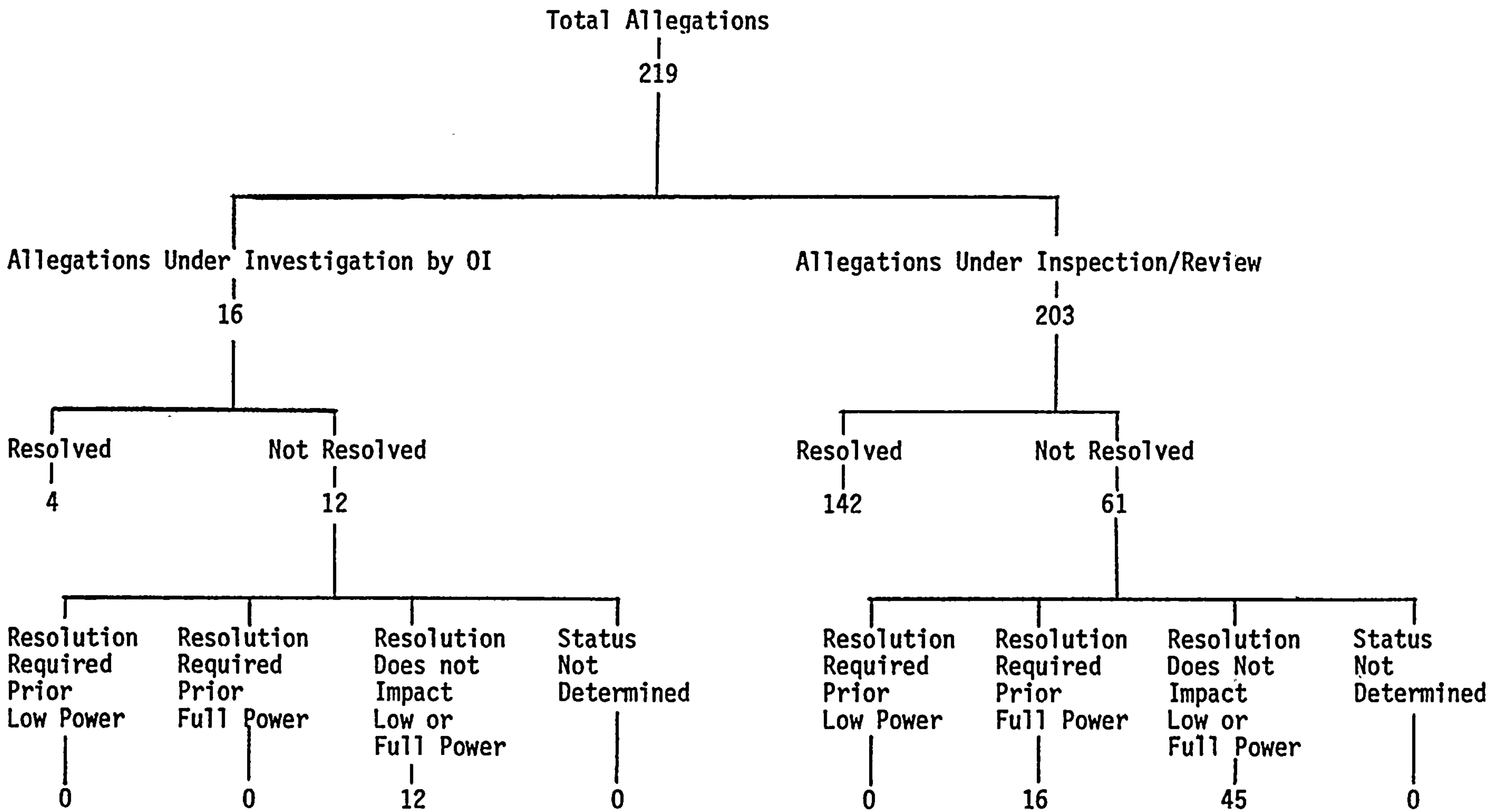
ATTACHMENT 2

DIAGRAM OF ALLEGATION STATUS

MARCH 9, 1984

Attachment 2

ALLEGATION STATUS AS OF MARCH 9, 1984



Low Power: Criticality and Operation Below 5% Power

Full Power: Operation above 5% power

ATTACHMENT 3

TABLE OF ALLEGATIONS STATUS

MARCH 9, 1984

Attachment 3

Table of Allegation Status

(March 9, 1984)

I. Total Allegations 219

A. Allegations under investigation by OI 16
B. Allegations under inspection/evaluation 203

II. Investigation Items 16

A. Resolved 4
(Allegation: 1, 2, 23, 53)

B. Not Resolved 12

1. Resolution prior to Low Power 0
2. Resolution prior to Full Power 0
3. Resolution w/o impact 12
4. Resolution not determined 0
(Allegation: 18, 19, 70, 81, 99, 120
130, 156, 157, 196, 197, 202)

III. Inspection/Evaluation Items 203

A. Resolved 142
(Allegation: 3, 4, 6, 6a, 7, 8,
9, 10, 11, 14, 15, 16,
17, 20, 21, 22, 24, 25,
26, 27, 28, 29, 30, 31,
32, 33, 35, 37, 38, 40,
41, 42, 43, 44, 46, 47,
49, 50, 51, 52, 54, 56,
57, 58, 59, 60, 61, 61a,
62, 63, 64, 65, 66, 67,
68, 69, 71, 72, 73, 74,
75, 76, 77, 78, 80, 84,
86, 90, 91, 92, 93, 94,
96, 98, 101, 102, 103, 104,
105, 106, 107, 108, 109, 110,
111, 112, 113, 114, 115, 116,
117, 118, 119, 121, 122, 124,
125, 126, 127, 128, 132, 133,
134, 135, 138, 142, 146, 154,
166, 167, 171, 172, 173, 174,
176, 178, 179, 180, 181, 182,
183, 184, 185, 186, 187, 190,
199, 203, 204, 205, 206, 207,
208, 209, 210, 211, 212, 213,
214, 215, 216, 217)

B. Not Resolved 61

1. Resolution prior to Low Power 0
2. Resolution prior to Full Power 16
   (Allegation: 5, 13, 34, 36, 48, 55, 79, 82, 85, 87, 88, 89, 95, 97, 100, 131)

3. Resolution w/o impact 45
   (Allegation: 12, 39, 45, 83, 123, 129, 136, 137, 139, 140, 141, 143, 144, 147, 148, 149, 150, 151, 152, 153, 155, 156, 158, 159, 160, 161, 162, 163, 164, 165, 168, 169, 170, 175, 177, 188, 189, 191, 192, 193, 194, 195, 198, 200, 201)

4. Resolution not determined 0

ATTACHMENT 4

INDIVIDUAL ASSESSMENT SUMMARIES

MARCH 9, 1984

Task: Allegation or Concern No. 1

ATS No: Q5-82-0004 BN No:

Characterization

Passing contraband

Implied Significance 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 Significance 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.

Action Required

No further action required on this allegation - refer to SSER 21

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.

Action Required

No further action required on this allegation - refer to SSER 21

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.

Action Required

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 will be incorporated in the Plant Technical Specifications prior to issuance of a full power license.

Task: Allegation No. 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).

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation No. 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).

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation 7

(previously addressed in SSER 21)

ATS No.: NRR-83-03 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.

Related Allegations: 9, 13, 36, 48

Implied Significance to Plant Design, Construction, or Operation

Previously provided in SSER 21

Assessment of Safety Significance

Previously provided in SSER 21

Staff Position

The staff position regarding PG&E's System Interaction Program is documented in Section 8.2 of SSER 11 as follows:

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

(b) The NRC will verify the completion of PG&E's program and the accetability of any plant modifications during the normal course of inspection activities.

(c) PG&E will provide, following the completion of the program, for NRC information copies of the final report of the program which will include an identification of all interactions postulated, all walkdown data, interaction resolutions and technical reports.

By letter dated October 13, 1983, PG&E submitted an information report on the status of their seismically induced systems interaction program (SISIP) within the containment of Unit 1. Included in the information report was the preliminary status of their study of Unit 2. The staff has discussed the progress of the program with PG&E since that submittal.

Based upon (a) the staff's understanding of the program which includes many details documented in SSER 11 and reinforced by extensive communication with PG&E, 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

The completion of the established systems interaction program and its review and evluation by the staff will be achieved in accordance with the position and schedule discribed above. The ongoing review will continue to take any necessary steps to assure that no misunderstandings occur which might be significant to the safe operation of Diablo Canyon.

Task: Allegation 8

(previously addressed in SSER-21)

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

The staff's assessment of the safety significance of this allegation was perviously provided in SSER 21. The staff concluded that a loss of efficiency in the operation of the diesel generators due to failures of intake and exhaust system piping resulting from a postulated Hosgri earthquake is not likely, provided that modifications to braces and piping supports are properly installed.

Staff Position

In SSER 21 the staff stated that the issue is satisfactorily resolved subject to completion of modifications. In a letter of February 2, 1984 PG&E informed the staff that the seismic modifications to the supports of diesel generator intake and exhaust system piping and to the exhaust silencer mounting brace have been completed. The FSAR will be updated to reflect that the system will perform its required safety function after the postulated Hosgri event. The staff concludes that this action resolves this allegation.

Action Required

In SSER 21 the staff stated that proposed modification to diesel generator silencer bracing and pipe supports should be completed prior to reactor power ascencion beyond 5 percent. PG&E has now completed the required modifications. No further action is required.

Task: Allegation 9

(previously addressed in SSER 21)

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

Characterization

This item is not an allegation but relates to Board Notification 83-17. The board notification involves the testimony of an NRC staff witness (J. Coran) in the Shoreham proceeding. In that testimony, Mr. Conran expresses his concerns in two areas, namely systems interaction and safety classification. The first concern has some potential generic implications due to the aspects which involve the resolution of Unresolved Safety Issues A-17. The second concern, safety classification is considered to be plant specific to Shoreham.

Related Allegations: 7, 13, 36, 48

Implied Significance to Plant Design, Construction and Operation

Previously addressed in SSER 21

Assessment of Safety Significance

As previously stated in SSER 21, 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 applicant's seismically induced systems interaction program. See also Allegation 48.

Staff Position

As previously stated in SSER 21 the staff position regarding Unresolved Safety Issue A-17 is reflected in the staff testimony in the Shoreham proceeding. This position is generic and applies to Diablo Canyon. In addition PG&E has completed over 90 percent of its seismically induced systems interaction program. The PG&E program goes beyond the requirements on Shoreham and will provide added assurance that Diablo Canyon can be operated safely. Modifications resulting from the program must be completed prior to full power operations as documented in SSER 11. For further detail on the program and its schedule see Allegation 7.

Action Required

The completion of the already existing systems interaction program and its review and evaluation by the staff will be accomplished in accordance with the staff position and schedule as described under Allegation 7. The final resolution of the generic Unresolved Safety Issue A-17 will be appropriately applied to Diablo Cnayon. No specific action is required for Diablo Canyon.

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.

Action Required

No further action required on this allegation - refer to SSER 21

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation 12

(Previously addressed in SSER 21)

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

Previously provided in SSER 21

Assessment of Safety Significance

Previously provided in SSER 21

Staff Position

Previously provided in SSER 21

Action Required

The required action, as previously stated in SSER 21, has not changed. If modifications to achieve substantial additional protection are found necessary, these modifications would be required before start-up after the first refueling.

Task: Allegation 13

(Previously addressed in SSER 21)

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

Characterization

Inadequate Seismic Systems

Related Allegations: 7, 9, 36, 48

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 safely shutdown the reactor and maintain it in a safe shutdown condition.

Assessment of Safety Significance

Previously addressed in SSER 21.

Staff Position

As described under Allegation 7, PG&E has established a seismically induced systems interaction program. The staff position and schedule for the completion of this program is documented in Section 8.2 of SSER 11 and summarized under Allegation 7. The completion of the program, including necessary modifications, will achieve the degree of safety addressed by the allegation. This will be accomplished through the use of various alternatives rather than upgrading Class II systems and components to Class I.

<u>Action Required</u>

The completion of the established systems interaction program and its review and evaluation by the staff will be achieved in accordance with the position and schedule as described under Allegation 7. No additional action regarding this allegation is required.

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.

Action Required

No further action required on this allegation - refer to SSER 21

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation No. 16

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

Characterization

Inadequate design of high energy rupture restraint crushable pads.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation No. 17

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

Characterization

Seismic criteria for Westinghouse items: NSSS SSE loads inadequate.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No.18

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

Characterization

QA/QC Allegations

Implied Significance 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 Significance 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

This concern does not have any implied significance to Design or Construction of the facility. It does have implied significance to Plant Operations. Failure to have adequately qualified Health Physics personnel could adversely affect the licensee's ability to implement a quality radiation protection program.

Assessment of Safety Significance

The NRC staff approach to resolving this issue was to examine the applicable Technical Specification and related standards; to review the licensee's implementation of these requirements; and to assess the licensee's compliance in assuring requisite qualifications of Health Physics personnel.

The licensee's Technical Specification 6.3.1 requires that each member of the Health Physics staff shall meet or exceed the minimum qualifications of ANSI standard 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.

The NRC staff has reviewed the qualifications of the Health Physics staff and found them to be adequate and in conformance with requirements. The qualifications of the Supervisor of Chemistry and Radiation Protection and those of his alternate were reviewed by the Office of Nuclear Reactor Regulation (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 Health Physics staff to insure that the ANSI N18.1-1971 requirements are met. Region V has reviewed this program and found it to be adequate. However an issue was identified 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 could be considered to be two separate specialties. The licensee, however, considers that a combined total of two years experience meets the intent of the ANSI standard. NRC has not specifically developed a position addressing whether 2 or 4 years of experience are appropriate for the disciplines of chemistry and radiation protection combined as a single specialty. There is precedent for both interpretations.

Staff Position

Region V concludes that the licensee's professional Health Physics staff meet the requirements of the Technical Specification. Notwithstanding the ANSI standard, the licensee intends to use only qualified technicians to fill responsible positions. The issue of the required number of years of experience for Chemistry and Radiation Protection technicians will be pursued on a generic basis by Region V.

Action Required

No further action is required relative to the specific allegation.

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

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 internal exposures to radioactive materials as low as reasonably achievable (ALARA). Specifically, (1) the air in 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 any time the workers want it.

Implied Significance to Design, Construction or Operation

These concerns do not have any implied significance to construction of the facility. The first concern implies that the proper air exchange was not considered when the chemistry laboratory was being designed. All three concerns have implications for proper operation of the facility. Poor practices in the respiratory protection program could lead to unneccessary internal exposure to radioactive materials.

Assessment of Safety Significance

The NRC staff's approach to resolving this issue was to review the licensee's procedures; to examine the chemistry laboratory; and to interview the cognizant licensee staff.

NRC's review of this matter found no basis to indicate the existence of unacceptable ALARA conditions or practices. (1) The fume hoods are not the only means of air exchange for the chemistry laboratory. Also, considering only the effect of the fume hoods, the number of air changes per hour exceed the OSHA requirements. (2) 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. (3) The licensee currently intends 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 on the specific equipment being used.

Staff Position

Region V concludes the specific concerns cited are not founded. In the inspector's opinion, the licensee is committed to a strong ALARA program. This commitment is reflected in statements in their procedures. The inspector note, however, that the ultimate performance can't be clearly demonstrated until the plant is operational.

Action Required

No further action is required relative to the concerns expressed. Region V will review the licensee's implementation of their operational ALARA program through the routine inspection program.

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

This concern implies that the stated monitors will be unable to accurately measure the radionuclides of Xenon-133 and Krypton-85 and consequently not all released activity will be accounted for.

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 NRC staff approach to resolving this issue was to examined the installed instrumentation; to review the applicable procedures; to review the correspondance between the licensee and the instrument vendor; and to interview the cognizant licensee staff.

Environmental shields are installed as alleged, however there are valid reasons for their presence. When the inspector examined the situation in detail the following was found:

The Gas Decay Tank Discharge Monitor monitors what may be relativly high concentrations of an undiluted stream. The Air Ejector Monitor is in a hostile environment: high humidity and temperature. The licensee procured environmental shields from the manufacturer of these monitors to protect them from the hostile environment, and to adjust the sensitivity to a proper operating range 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 are 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 intends to verify the vendor's response curves when the plant is operational.

Staff Position

Region V considers the licensee to have made an adequate review of the effects of the modifications to the monitors and the reduced sensitivity of these monitors does not adversely effect the ability of the instrumentation to perform its intended design function.

Action Required

No action is required.

Task: Allegation or Concern No.23

ATS No: Q5-83-017 BN No:

Characterization

QA Inspector concerns

Implied Significance 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 RV83A46

BN No: 83-164 (10/27/83)

Characterization

A site contractor (H.P. Foley (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 Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern Nos. 25, 58, 142, 154, 176

ATS No: RV-83-A-33, RV-83-A-57, RV-84-A-0015, RV-84 A-0017, RV-84-A-0007 BN No: N/A

Characterization

Alleged deficiencies in the installation of concrete expansion anchor bolts by site contractor H. P. Foley.

Implied Significance to Plant Design, Construction, or Operation

Improper installation of anchor bolts could result in reduced load capacity of the anchor bolts with attendant loss of design function during normal operation or design basis events, including seismic events.

Assessment of Safety Significance

This issue was discussed in SSER21 (NUREG 0675), pages 2-44 through 2-54 allegation No. 25. The staff's subsequent reviews of the issues in SSER-21 and other related allegations are documented below.

The concerns raised by these allegations involve the installation and inspection of concrete expansion anchors by the H. P. Foley Company. A general and non specific concern with anchor bolts was supplied initially to the staff from an anonymous alleger. Subsequent interviews of onsite contractor personnel resulted in additional concerns with added detail in some cases. The substance of the initial allegation and concerns obtained from interviews have been characterized by the staff as follows:

1. Phillips Red Head Stud Anchors have been forbidden for use in nuclear power plants.

2. Phillips Red Head Anchors are not good because at other nuclear power plants they have been removed or not used.

3. Many anchors have been installed improperly.

4. Anchors have not been torqued.

5. Phillips Red Head nuts are only tightened "finger tight".

6. Washers are not used on concrete anchors.

7. Quality Control does not inspect or inadequately inspects anchor bolt holes prior to installation.

8. Quality Control inspection of 10% of anchors on instrument supports is inadequate.

The staff approach to resolution of these issues was to: (1) review installation procedures, audits, nonconformance reports, discrepancy reports, and licensee correspondence relating to concrete anchor bolts; (2) have an independent NRC contract team (Lawrence Livermore National Laboratory) inspect a sample of 124 electrical raceway supports modified in 1982 (involving hundreds of anchors bolts); and (3) request the licensee to perform torque tests and ultrasonic examination on a sample of 40 installed anchor bolts to verify the adequacy of installation.

The results of the staff investigation, as applied to the general concerns listed above are:

(1)&(2) The concern that Phillips Red Head Stud Anchors are forbidden to use in nuclear power plants or are not good is not supported by any NRC criteria or industry standards. These anchor bolts are acceptable.

(3) The concern that many anchors have been installed improperly was not confirmed by field torque tests.

(4) The concern that anchors have not been torqued was confirmed, however, there is no NRC criteria which require anchor bolts installed in electrical, heating, ventilation and air conditioning, and instrument tubing applications to be torqued to any specified value. The staff also reviewed the licensee's electrical raceway qualification tests to determine if bolt torque was a significant parameter and concluded that anchor bolt torque would not impact the use of the qualification test results.

(5) The concern that Phillips Red Head nuts are only tightened "finger tight" was not confirmed. The "finger tight" check is a Quality Control inspection point. The anchor bolt installation procedures require nuts

to be tightened although a torque value is not specified. The staff field sample verified that all anchors tested were tight.

(6) No specific requirement exists for washers to be installed on concrete anchors. This is not considered a problem by the staff.

(7) The concern that Quality Control does not inspect or inadequately inspects anchor bolt holes prior to installation was not found to be a problem.

(8) The concern that Quality Control inspection of 10% of anchors is inadequate was not confirmed. The staff consider that a 10% sample size is technically adequate and that it is consistent with that used at other plants.

Although the staff investigation failed to confirm the alleged significant deficiencies in concrete anchor installations, the staff did come across some concerns as follows. (PG&E was aware of issues i, ii, and iii and had taken action prior to NRC involvement.)

i. Some Anchor bolts were too short to meet the licensee's minimum embedment criteria.

ii. Field surveys performed after 1974 identified numerous cases where the spacing between anchor bolts did not meet PG&E requirements.

iii. There existed a seemingly inappropriate disposition of several anchor bolt discrepancies involving angularity identified during PG&E's "Grid Program." (Note: The "Grid Program" was a 1978-1980 inspection of all Class IE raceway supports). The disposition in question was with the corrective actions taken by PG&E.

iv. Numerous installation deficiencies have been dispositioned over the years without incorporating these dispositions in the current engineering evaluations of raceway supports.

v. The appropriate factor of safety may not exist in the "as-built" condition on electrical raceway supports.

vi. H. P. Foley anchor installation procedures did not contain PG&Es criteria for slippage which can occur as one tightens the bolts.

In order to resolve the six issues listed above, the staff asked the licensee to perform various tests, evaluations and reviews. The staff then examined the majority of these results and concluded that the concerns probably do not involve violations of any NRC criteria. The inspector still has to complete his review of the PG&E information.

Staff Position

The concerns raised by the various allegers regarding H.P. Foley anchor bolt practices were not confirmed by the staff.

Action Required

The inspector still has to complete review of the PG&E material concerning the items questioned by the inspector. Further followup action by the staff will be performed as part of the routine inspection program.

Task: Allegation or Concern No. 26

ATS No: RV-83-A-0033

Characterization

Foley didn't document NCRs issued by Field Inspectors.

Implied Significance to Plant Design, Construction, or Operation

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. 27

ATS No: RV-83-A-33 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.

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation 28

(Previously addressed in SSER 21)

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

Charaterization

The annulus structure reverification is erroneous.

Implied Significance to Plant Design, Construction or Operation

Previously addressed in SSER 21

Assessment of Safety Significance

Previously addressed in SSER 21

Staff Position

The staff previously stated in SSER 21 that 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 of the supporting structural members if controversial in the opinion of some qualiried 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

The staff previously stated in SSER 21 that careful visual inspection of pipe supports and pipe support structures is necessary with the plant at operating thermal conditions, and that the licensee has such an inspection planned as part of the plant startup.

PG&E has advised the staff that system walk-downs with visual inspections have been performed during Mode 4 and Mode 3 at operating conditions. Some interferences due to thermal expansion were noted and necessary modifications have been completed. PG&E will continue this effort after criticality in accordance with normal surveillance requirements and will inform the staff of the results. The staff will review the walk-down and inspection results and may require additional surveillance in areas such as closly spaced supports, where clearances appear mininal or where significant torsional loads may occur. This effort must be completed prior to operation above 5 percent power.

Task: Allegation No. 29

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

Characterization

Pipe Restraint Design Inadequate.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No. 30

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

Characterization

Safety-related equipment has inadequate/untraceable documentation.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation 31

(previously addressed in SSER-21)

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

Characterization

(previously addressed in SSER-21)

Implied Significance to Plant Design, Construction or Operation

(previously addressed in SSER-21)

Assessment of Safety Significance

The staff's assessment of the safety significance of this allegation was previously provided in SSER 21. The staff concluded that although formal documentation of verification of some pre-and post-processor computer programs was not available no unusual results were found in the calculations.

Staff Position

In SSER 21 the staff stated that the issue is satisfactorily resolved and the licensee should document all pre-and post-processor computer programs. In a letter of February 17, 1984 PG&E informed the staff that the computer verifications are available for all technical options of the programs.

Action Required

The documentation of all pre-and post-processors computer programs by PG&E resolves this issue. No further action is required.

Task: Allegation 32

(previously addressed in SSER-21)

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

Characterization

(previously addressed in SSER-21)

Implied Significance to Plant Design, Construction or Operation

(previously addressed in SSER-21)

Assessment of Safety Significance

The staff's assessment of the safety significance of this allegation was previously provided in SSER-21. The staff concluded that there was no impact on low power testing or full power operation.

Staff Position

In SSER 21 the staff stated that the licensee should confirm that all penetrations have been or will be reviewed for structural adequacy. In a letter of February 17, 1984, PG&E informed the staff that calculations have been performed in accordance with the applicable licensing criteria to confirm that the all penetrations are structurally adequate. The calculations are documented in PG&E files.

Action Required

In SSER 21 the staff required the confirmation of all penetrations for structural adequacy. PG&E has completed the requirements. No further action is required.

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation for 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 Operation

Previously provided in SSER 21

Assessment of Safety Significance

Previously provided in SSER 21

Staff Position

Previously provided inSSER 21

Action Required

The required action is an accuracy review by Region V of "as-built" design drawings for operations personnel use prior to exceeding five percent power.

Task: Allegation No. 35

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

Characterization

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

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation 36

(Previously addressed in SSER 11)

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

Characterization

Analysis for resolution of fluorescent light fixture interaction assumed conduit connections to be hinged; however, inspection found connections to be fixed.

Related Allegations: 7, 9, 13, 45

Implied Significance to Plant Design, Construction or Operations

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 implications is that of adverse interaction between safety and non-safety equipment during and following a large earthquake.

Assessment of Safety Significance

The staff assessment of the safety significance of the specific issue was addressed in SSER 21.

Staff Position

This issue will be resolved pending satisfactory completion of the seismically induced systems interaction program. The staff position regarding this program including necessary actions and schedule is documented in Section 8.2 of SSER 11 A summary is provided under Allegation 7.

Action Required

No specific action is required regarding the allegation. The completion of the established seismically induced systems interaction program and its review and evaluation by the staff will be achieved in accordance with the position and schedule as described under Allegation 7.

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No. 38

ATS No: RV-83-A-47 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."

Action Required

No further action required on this allegation - refere to SSER 21.

Task: Allegation 39

(previously addressed in SSER 21)

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

Characterization

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

Related Allegations: 37, 40, 45, 177

Implied Significance to Plant Design, Construction, or Operation

Previously addressed in SSER 21.

Assessment of Safety Significance

In SSER 21 the staff stated that indication provided in the control room of RHR letdown line isolation includes position indication for two valves in series as well as RHR system flow, pressure, and pump status information. Although these features provide a capability to assess RHR status, the staff has recognized the need for installation of a RHR low flow alarm. Accordingly, the licensee was required to install a RHR low flow alarm during the first refueling.

Staff Position

In SSER 21 the staff stated that this allegation does not involve considerations that question plant readiness for power ascension testing or full power operation. In a letter of February 15, 1984 the licensee committed to install the RHR low flow alarm prior to entering Mode 1, i.e. operation above 5 percent power. The licensee also provided the administrative controls and procedures that are now in effect. Based on this committment, the staff finds these controls and procedures acceptable for the interim, i.e. until installation of the alarm. The staff concludes that the issue is resolved with regard to criticality and lower power operation.

Action Required

The staff requires that the low flow alarm be installed prior to entering Mode 1 and that the licensee advise the staff of the completion of the installation prior to Mode 1.

Task: Allegation No. 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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No. 41

ATS No: RV-83-A-47 BN No: 83-169 (20/29/83)

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).

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation or Concern No. 42

ATS No: R-83-A-47 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 removal (RHR) system. Closure of the valves disables operation of the RHR system for decay heat removal.

Action Required

No further action required on this allegation - refer to SSER 21.

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.

Action Required

No further action required on this allegation - refer to SSER 21

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.

Action Required

No further action required on this allegation - refer to SSER 21

TASK: Allegation 45

(Previously addressed in SSER 21)

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 requries power to be removed from these isolation valve operators during Mode 4 (hot shutdown, RCS cold leg temperature is less than 323°F), Mode 5 (cold shutdown) and Mode 6 (refueling). This requirement defeats the function of autoclosure interlock for the valves.

Related Allegations: 37, 39, 40, 177

Implied Significance to Plant Design, Construction or Operation

As stated in SSER 21, as the result of Technical Specification Section 3.4.9.3.a, the isolation valves 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 fuction. This will result in a situation in which insufficient isolation capability exists to prevent an intersystem LOCA between high pressure RCS and the low pressure RHR system.

Assessment of Safety Significance

As stated in SSER 21, the staff concluded in Diablo Canyon SSER 13 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.

Staff Position

As stated previously in SSER 21, to implement the staff position stated in SSER 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 anable to provide protection against intersystem LOCA.

In a letter dated February 15, 1984 the licensee committed to install the RHR low flow alarm prior to entering Mode 1, i.e. operation above 5 percent power. The licensee also provided the administrative controls and procedures that are now in effect. Based on the committment the staff finds these controls and procedures acceptable for the interim, i.e. until installation of the alarm. The staff concludes that this issue is resolved with regard to criticality and low power operation.

Action Required

The staff requries that the low flow alarm be installed prior to entering Mode 1 and that the licensee advise the staff of the completion of the installation prior to Mode 1.

Task: Allegation or Concern No. 46

ATS No: RV-83-A-0046 BN No.

Characterization

H. P. Foley QA Procedures Voiding NCRs Incorrect.

Implied Significance to Plant Design, Construction, or Operation

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 48

(Previously addressed in SSER 21)

ATS. No.: RV 83A34

Characterization

Status of Seismic Systems Interaction Study

Related Allegations: 7, 9, 13, 36

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 seismically induced 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 indications are that the modifications will be completed prior to operations.

Assessment of Safety Significance

As previously stated in SSER 21 the staff has re-examined both the status of the seismically induced systems interaction program and the activities related to the allegations.

Staff Position

The staff position regarding PG&E's program and pertinent to this allegation is documented in Section 8.2 of SSER 11 and follows:

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

(b) The NRC will verify the completion of PG&E's program and the acceptability of any plant modifications during the normal course of inspection activities.

(c) PG&E will provide, following the completion of the program, for NRC information copies of the final report of the program which will include an identification of all interactions postulated, all walkdown data, interaction resolutions, and technical reports.

Based on our review of the PG&E program, 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 a letter of September 10, 1983 from PG&E, and the commitment to complete these modifications prior to talking the reactor critical for the first time, the staff concluded that it is not necessary to complete all modifications prior to loading fuel.

Task: Allegation or Concern No. 47

ATS No: RV-83-A-34 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."

Action Required

No further action required on this allegation - refer to SSER 21.

Action Required

No additional action is required regarding this allegation. The completion of the established systems interaction program, audits, review and evaluation by the staff will be achieved in accordance with the position and schedule described above.

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."

Action Required

No further action required on this allegation - refer to SSER 21

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.

Action Required

No further action required on this allegation - refer to SSER 21

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 becuase their candor endangers their jobs and may subject them to public ridicule even if their allegations are true.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No. 52

ATS No.: RV 830034 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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No.53

ATS No: RV83A39 BN No:

Characterization

Welder qualification

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 54

ATS No: 83-A-38 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.

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation 55
(previously addressed in SSER 21)

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

Characterization

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

Related Allegations: 79, 82, 85, 87, 88, 89, 95, 97

Implied Significance to Plant Design, Construction or Operation

The safety and operation of small bore piping systems cannot be assured if analyses of the supports are unreliable and exceed required acceptance criteria.

Assessment of Safety Significance

1. Technical Approach to Resolution
   a. Review sample of small bore support design packages to verify allegation.
   b. Review PGE response to allegation
   c. Document findings

2. Work Performed and Findings Identified
   The staff reviewed a total of fifteen small bore support design packages. No direct evidence of directly altered documentation has

been found. Two instances were found in which a supervisor changed a fix proposed by the analyst, without supporting calculations. (Calculation MP-071, Hanger 2171-16 and Calculation MP-345, Hanger 2182-74). In both instances the supervisor signed the modifications, which appears to have been made based on judgement. These supervisor initiated changes appear to be reasonable. In addition, PG&E has also provided additional information in a letter dated February 7, 1984 regarding the circumstances under which these changes were made. The staff finds this information acceptable.

3. Additional Findings

In the course of verifying this allegation, and related Allegations No. 87 and No. 88, the staff identified, investigated and resolved certain technical issues. These issues and the corresponding findings are as follows:

a. Different penetration stiffnesses in static and dynamic analysis.

The documentation on which Allegation No. 55 is based indicated that in certain small bore piping stress calculations, rigid foam penetrations had been modeled with different stiffnesses under static and dynamic loading. PG&E stated in a submittal dated December 28, 1983, that this modeling assumption was applied at three wall penetrations involving seven small bore piping systems. They also stated that these piping systems were reanalyzed under the assumption that there is no thermal or seismic restraint at the penetration locations. The results show that the piping and pipe supports remain qualified for thermal and seismic loading under this assumption. The staff reviewed two piping stress calculations, 8-301 and 8-307, and found that the rigid foam penetrations had been modeled with different stifiness. However, in calculation 8-307 the staff verified the DCP assertion that the stresses increased but still met the required allowables under the assumption of no thermal or seismic restraint. The staff considers this issue resolved.

b. Different stifiness for the same rigid supports in static and dynamic piping analysis.

The staff reviwed piping stress calculation 8-304 and has verified that for two rigid supports a finite stiffness was used in the static (thermal) analysis while the stiffness of the same supports was taken as infinite in the dynamic (seismic) analysis. PG&E has stated in the letter of February 1, 1984, that this technique was used in four out of 129 computer based piping analyses to reduce the calculated thermal loads. To address this concern they stated stiffiness for both the static and dynamic analyses. The results of these analyses demonstrated that the stresses and supports meet the requisite licensing criteria. The staff has determined that current industry practice is to use the same support stifiness for both type of analyses. Therefore the use of different stiffnesses for the same support by the Diablo Canyon Project (DCP) is considered to be a modeling deficiency, and will require a commitment to modify the Design Criteria Documents (DCM) to preclude this practice in the future. The staff will consider this issue resolved on receipt of such a commitment by the DCP.

c. Calculational error and modeling deficiencies in support design packages.

Although no allegation was made regarding errors in support calculations, this issue resulted from a review of small bore pipe support design calculations to verify specific allegations. The staff reviewed a total of 12 small bore pipe support design packages, 9 of which showed either a design QA deficiency, design or modeling deficiency, or calculational errors. Three of the calculations indicated calculational errors (such as incorrect computer program input), two of which are known to be significant. The DCP has corrected these errors and redone these calculations, showing that the allowable stresses and loads are satisfied.

PG&E has stated in the letter of February 7, 1984 that the DCP has reviewed 110 support design calculations since the December 15, 1983 meeting with the staff in Bethesda, MD. They have determined that 22% of these calculations had significant discrepancies, and that these support calculations were acceptable on the basis of detailed calculations. They have also indicated that 74% of all discrepancies consisted of modeling, input or calculational errors. They have also stated that all revised calculations met the design requirements and that no modifications of any of these re-analyzed supports were required.

The number of deficiencies found in the sample of support calculations reviewed by the staff, and the sample re-reviewed by the DCP exceeds that which would have been expected at this stage of the effort. The staff therefore finds that this is a potentially significant safety issue.

Staff Position

The staff determined that the DCP did not purposely alter current documentation to approve pipe supports which had failed. Therefore the basic allegation is determined to be without basis.

The staff has found that there is a basis for the allegations that different penetration and support stiffnesses were used in static and dynamic piping analyses. The DCP has, however, demonstrated that no modifications were required, and that the pertinent design criteria were met when piping analyses were performed with same stiffnesses. The staff find this acceptable.

The staff has found that a significant number of support design packages contain modeling deficiencies or calculational errors. This finding has been substantiated by a review conducted by the DCP itself. The staff considers this to be a significant deficiency of the DCP reverification effort. However, since no restraints have been found which require modification, there appears to be no immediate impact on low power operation.

Action Required

1. The staff will require the DCP to modify the piping design procedures to insure that the same support stiffnesses are used in the static and dynamic piping analyses.
2. The staff will require the DCP to institute an in-house program to reverify in detail all small bore piping supports which were qualified by the DCP using computer analysis. This program should be completed before ascension to full power operation.
3. The NRC staff will audit the DCP reverification effort on a sample basis until this issue is satisfactorily resolved.

Task: Allegation or Concern No. 56

ATS No.: RV-83-A-0033 BN No.: 83-02/14

Characterization:

Pitting of Main Steam and Feedwater Piping

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation or Concern No. 57

ATS No: RV-83-A-0057 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 allegation was discussed in and partly resolved in SSER 21. At that time the staff concluded further examination of the facts concerning certification of H. P. Foley (HPF) inspector to ANSI N45.2.6 was needed. It appeared to the staff that Foley was required to meet ANSI N45.2.6 after late 1979; however, their state of compliance wasn't clear to us at the time SSER 21 was written.

Subsequent to publication of SSER 21, the staff examined further the licensee's and HPF's commitments and procedures, interviewed an additional ten

(10) personnel, examined an additional twenty (20) records of individual QC inspector qualifications, reviewed an additional eight (8) nonconformance and nine audit reports, and examined thirty (30) more work request packages.

On July 14, 1981, the licensee, in response to NRC generic letter 81-01, dated May 4, 1981, 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. Nevertherless, On December 7, 1982, in response to a licensee audit of August 1982, HPF established a new procedure providing for the qualification and certification of quality control inspectors and supervisors imposing ANSI N45.2.6 criteria. Based on the material, examined both prior and after the publication of SSER 21 the staff has found that prior to December 1982, HPF quality control inspectors were not required to be and were not certified in accordance with the classification system and recommendations outlined in ANSI 45.2.6.

Subsequently, in February 1983, a PG&E audit resulted in a reevaluation of the certification of 41 QC inspectors. Of these inspectors 22 were found to be lacking in qualifications for the assigned ANSI N45.2.6 classification, e.g., a welding inspector had been given full credit for job related experience as a welder, rather than only partial credit. These inspectors were then reclassified pursuant to the established procedure and assigned future inspection activities within their appropriate classification. To verify the acceptability of work examined by these 22 individual, 10 percent of their past work was reinspected. No significant deficiencies were found.

During the early years of nuclear construction, there were no recognized industry standards addressing QC inspector qualification (until the issuance of ANSI N45.2.6) and the readiness of an inspector to conduct inspection was based on the judgement of his supervisor. This was considered to be a uniformly accepted industry practice acceptable to the NRC. Quality Control inspection personnel were apparently hired by HPF based on consideration of the individual's education, experience documented in his resume' or application for employment, and the interviewer's impression of the individual's capability to learn and be trained. Training essentially consisted of reading pertinent procedures and on-the-job training which resembled apprenticeship, where a new inspector accompanied a more experienced inspector in the conduct of his inspections. Judgement of readiness to acceptably conduct the required inspections was, therefore, a supervisory perogative based on recommendations and observations of a more seasoned inspector. Reports of five (5) routine audits conducted by the licensee between 1975 and 1982, showed that HPF was generally complying with the requirements prescribed in their procedures governing qualification of QC inspectors at the time of the audits.

The licensee examination of this matter was summarized in a letter, dated February 17, 1984, to the Regional Administrator of Region V as follows:

> "From the beginning of construction until 1981, the number HPF inspectors on site varied from three to ten. Since the numbers of inspectors were small, the performance of inspectors was easily monitored by the HPF QA management. This overview provided assurance

that inspections were satisfactorily performed although the program did not meet all ANSI N45.6 requirements. HPF has always had approved procedures and training programs in place to assure appropriate inspections. Even though the earlier HPF program did not meet the requirements of ANSI N45.2.6, the level of training and documentation of inspector qualification met the licensing commitments of PG&E and was consistent with the intent of industry standards and requirements. Improvements have been made over time as has been done elsewhere in the industry, and today the program for Quality Control inspector qualification and certification is in complete compliance with ANSI N45.2.6."

Staff Position

The staff concludes that HPF was not committed to and did not use an ANSI N45.2.6 type qualification/certification program prior to December 1982. Since December 1982, the program has been audited by the licensee and deficiencies have been corrected. Even though, HPF QC inspector qualification program prior to December 1982, was not patterned after ANSI N45.2.6 recommendations, procedures were in effect to assure that inspectors were qualified to perform assigned inspections. These procedures and inspector qualifications were routinely audited by PG&E over the years and found acceptable.

Action Required

Based on NRC recent reinspection findings that provided the information documented above, no further examination of HPF inspection activities is warranted.

Task: Allegation or Concern No. 58

ATS No: RV-83-A-57 BN No: N/A

Characterization:

Foley allows "Red Head" Anchors Studs Reported Improperly Installed.

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 Significance 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 Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No.61

ATS No: RV83A57 BN No:

Characterization

Lack of Document Control

Implied Significance to Plant Design, Construction, 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 Significance 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 Significance 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-calss 1 material to class 1 material. (Specific examples were identified).

Implied Significance 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 Significance 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 Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 66

ATS No: RV-83-A-0052 BN No: N/A

Characterization:

Defective Weld Reports Rejected by Foley.

Implied Significance to Plant Design, Construction, or Operation

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. 67

ATS No. RV-83-A-0055

Characterization

Negligence by PG&E in response to flooding at 55 ft. level of Auxiliary Building pipe tunnel.

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation or Concern No. 68

ATS No. None BN No.

Characterization:

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

Implied Significance to Plant Design, Construction or Operation

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

Action Required

No further action required on this allegation - refer to SSER 21 and NRC inspection report 50-275/83-37 dated February 29, 1984.

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No.70

ATS No: Q5-83-019 BN No:

Characterization

Inadequate response to notice of violation

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No.71

ATS No: RV83A58 BN No:

Characterization

Use and sale of drugs

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 72

ATS No: N/A BN No: N/A

Characterization

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

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation or Concern No.73

ATS No: RV83A061 BN No:

Characterization

Selling of drugs

Implied Significance 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)

Action Required

No further action required on this allegation - refer to SSER 21

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No. 76

ATS No: RV-83-A-0063 BN No: N/A

Characterization:

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

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation or Concern No. 77

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

Characterization

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

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No. 78

ATS No: RV-83-A-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.

Action Required

No further action required on this allegation - refer to SSER 21.

Task: Allegation or Concern No. 79

(previously addressed in SSER 21)

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

Characterization

Site design engineers were not required to use controlled documents in the performance of their work. This resulted in different calculation bases, load ratings, and stress allowables being used in small bore (S/B) piping analyses.

Related Allegations: 55, 82, 85, 87, 88, 89, 95, 97

Implied Significance to Design, Construction, or Operation

Without uniform design bases, formulations, and acceptance criteria, the adequacy of plant system safety could not 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 standarization, currency and availability of design documents. Six design engineers performing on-site design activities were interviewed as part of the review. There were 30 hanger engineer and 25 piping stress engineerings working at the site. Three engineers from each group were selected for interview. Length of employment at the site was used as the criteria for selection. In each group the longest employed engineer, the newest employed engineer and an engineer with an intermediate length of employment was selected for interview. 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 did not represent the current procedure in effect.

There was additional 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 maintaince of procedures and drawings. The staff inspection also substaintiated the allegaion that site engineers were working without benefit of controlled design documents; in some instances for considerable periods of time.

Staff Position

The staff concluded that the administrative controls imposed on the site piping engineering activities have been inadequate and ineffective. The specific allegation items were substantiated.

Action Required

The licensee will be required to initiate a thorough review of the On-Site Project Engineering Group (OPEG) administrative procedures and correct all deficiencies noted. The impact of these administrative deficiencies on small bore piping support design is addressed by action required under allegation number 55.

Task: Allegation No. 80

ATS No: RV-83-A-64

Characterization

Letters dated 4 November 1983, 9 December 1983 and 9 January 1984 from Dr. Richard Kranzdorf, Spokesperson for Concerned Cal Poly Faculty and Staff, concluded that the licensing process for the Diablo Canyon Nuclear Power Plant (DCNPP) should cease until four primary issues regarding emergency planning by San Luis Obispo County/Cities are resolved:

1. The evacuation time calculations are not adequate because only 20% was added to the normal evacuation times to account for adverse weather conditions. Dr. Kranzdorf does not feel that the 20% factor represents the "worst case" possible which he considers may be dense fog.

2. The main evacuation transportation routes for the Baywood Park/Los Osos area are unacceptable because both are subject to flooding.

3. Sirens, as the primary means of notification, are not acceptable because they are powered by regular power lines and are, therefore, subject to periodic interruption. The back-up system (police cars with sirens) is not acceptable because it would not be as effective as a fully operational siren system.

4. The evacuation time estimates are inadequate because the effects of earthquakes (e.g., potentially greater evacuation times) have not been considered.

Implied Significance to Design, Construction or Operation

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

Assessment of Safety Significance

On 8 December 1983 a conference call involving Region IX of the Federal Emergency Management Agency (FEMA), the State of California Office of Emergency Services (COES), the San Luis Obispo County Office of Emergency Services (SOES), and NRC Region V was conducted to discuss and analyze the issues raised by Dr. Kranzdorf. Since FEMA has primary responsibility by Presidential Direction to take the lead in offsite planning for nuclear emergencies, FEMA Region IX agreed to coordinate the assessment of the allegations. Additionally, NRC Region V has performed an independent assessment of the allegations. The results of these assessments are as follows:

1. The evacuation time calculations are not adequate because only 20% was added to the normal evacuation times to account for adverse weather conditions. Dr. Kransdorf does not feel that the 20% factor represents the "worst case" possible which he considers may be dense fog.

Assessment

Several independent studies dealing with road capacities under adverse weather conditions concluded that a 20% reduction in speed and capacity is appropriate for a range of adverse weather conditions including heavy rain and fog. These studies were conducted in several different states including California (fog), New York (fog), Illinois (snow and rain) and Texas (rain). Since speeds during a fair weather evacuation are already reduced from maximum, an additional reduction of 20% appears to be reasonable. The 20% reduction factor is a widely accepted standard. Evacuation times during extremely adverse weather conditions (e.g., zero visibility fog) might be somewhat longer, however, the times noted in the San Luis Obispo County Emergency Plan for general adverse weather conditions are available to the decisionmakers so that during extreme conditions concurrent with a radiological emergency, appropriate protective measures could be taken based on these estimates. It should be noted that there is no requirement that evacuation time estimates be based on the worst possible weather conditions.

This issue was litigated in the licensing proceeding. In an initial decision regarding emergency planning for the DCNPP, dated August 31, 1982, the Atomic Safety Licensing Board (ASLB) in part stated:

- "The evacuation time estimate made by Applicant conforms with the requirements of Appendix 4 of NUREG-0654 and is therefore accepted for the purposes of this case. A second estimate of evacuation time, done independently by the TERA Corporation, leads to similar estimates as the above report.... The (Joint Intervenor) witnesses consistently urged the most conservative assumptions, however, which the Board concludes are not credible.... The time estimates by P.R.C. Voorhees were realistically made over a range of normal and adverse conditions.... We conclude that time estimates for emergency evacuation of the public within the plume exposure EPZ are valid and in conformance with Appendix 4 of NUREG-0654.... The board therefore finds that adequate protective actions can be taken both on site and off site in the event of an emergency and requirements of 10 CFR 50.47 and criteria of Part J of NUREG-0654."

2. The main evacuation transportation routes for the Baywood Park/Los Osos area are unacceptable because both are subject to flooding.

<u>Assessment</u>

These circumstances are addressed in the San Luis Obispo County Emergency Plan. The Plan acknowledges specific locations which have a tendency to flood and also notes duration of flood stage at those locations (normally 2 hours). County officials are prepared to consider temporary delays associated with these specific locations during flood conditions. Evacuation times would be extended in proportion to the lost capacity. In addition, the Plan has provided for a staged evacuation. This would help alleviate any added congestion due to the use of alternate evacuation routes. Evacuation time estimates for a staged evacuation are provided in the Plan and are, therefore, available to the decisionmakers. County Officials would use these data to take the most prudent protective measures when faced with the prospect of or actual flooding.

An important point to be considered is that under severe flooding conditions the most probable protective measure which would be employed in the Baywood Park/Los Osos area would be sheltering instead of evacuation since a) a radioactive plume from the plant would be diffused by the hills and distance between the plant and the Baywood Park/Los Osos area and b) the Baywood Park/Los Osos area is greater than five miles from the plant and c) a storm of this magnitude resulting in the flood conditions discussed above would in itself inhibit migration of the plume.

FEMA has evaluated this situation and found that the county plans are satisfactory.

3. Sirens, as the primary means of notification, are not acceptable because they are powered by regular power lines and are, therefore, subject to periodic interruption. The back-up system (police cars with sirens) is not acceptable because it would not be as effective as a fully operational siren system.

Assessment

The siren system for alerting residents within the offsite jurisdictions around the DCNPP is electrically powered by sources distributed through seven different electrical power substations. The potential for power failures has been considered and procedures exist to verify power availability. Should a substation outage be reported as a result of that verification procedure, those responsible would dispatch appropriate county staff to the affected area for personal notification to residents. This activity would be performed in accordance with the guidance provided in NUREG-0654/FEMA REP-1, Rev. 1 that specifies the county has 45 minutes to alert that portion of the public that did not receive the initial alert.

4. The evacuation time estimates are inadequate because the effects of earthquakes (e.g., potentially greater evacuation times) have not been considered.

Assessment

The effects of earthquakes, with respect to evacuation times, has been considered and data has been provided in the county plan. An estimate of the evacuation times has been provided for light, moderate and heavy damage levels. These data are available to the decisionmakers so that in the event of a radiological emergency, during and/or after an earthquake, appropriate protective measures could be taken based on these estimates.

Staff Position

Based on the results of the combined assessment efforts by FEMA, State, County and NRC personnel, the staff position is that all allegations have been responsibility evaluated and addressed by all of the appropriate authorities.

Action Required

Provide Dr. Kranzdorf with the results of the assessment of the allegations. This will be accomplished by letter, telephone or possibly a meeting with Dr. Kranzdorf.

Task: Allegation or Concern No.81

ATS No: RV83A063 **BN No:**

Characterization

Individual fired for whistle blowing

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation No. 82
(previously addressed in SSER 21)

ATS No. RV83A063 BN No. N/A

Characterization

There was minimal training for onsite pipe support engineers.

Related Allegations: 55, 79, 85, 87, 88, 89, 95, 97

Implied Significance To Design, Construction, or Operation

Without adequate indoctrination and training, piping stress and support design engineers may not effectively perform their assignments.

Assessment of Safety Significance

This issue was addressed through examination of training requirements, implementation records, interview with engineers, and a review of engineering calculations.

Personnel Authorities and Duties

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.

A followup inspection was conducted on 2/15-16/84 at the PG&E corporate office, San Francisco, CA.

Based on the results of the followup inspection the staff concluded that the design personnel authorities and duties were delineated in DCP Instruction No. 9; however, the site personnel were not familiar with the procedures, and as a result they were unable to address the inspector's questions during site interviews. Personnel qualifications and duties were prescribed in sufficient detail in Bechtel requisitions. PG&E requisitions require the contractor organization to submit the work force qualifications and capability for PG&E's review and acceptance. Both methods were considered to be acceptable by the staff.

Personnel Indoctrination and Training

The staff found that other than general site QA and technical training that were provided for the new employees, no project group specific program training was in place in either the piping stress or support design engineering groups. Further, the general QA and technical training received by the engineers was not timely and consistent. The bases for this determination are:

Work Group	Began Work Mo/Yr	Engineering Manual Survey Training Date	QA Indoctrination Date
A (Support)	10/82	02/18/83	05/05/83
B (Support)	04/83	07/15/83	05/04/83
C (Support)	09/83	12/16/82	10/23/81
D (Stress)	04/82	06/09/82	05/05/83
E (Stress)	02/83	04/19/83	05/04/83

The staff reviewed several design calculations which are identified in Allegation No. 79. Among the calculations reviewed, it was identified that design verification and checking were not adequate to catch calculational errors which is an indication of a lack of procedural knowledge and training in project controls. This allegation was substantiated.

Subsequent inspections were conducted at the PG&E corporate office, San Francisco, CA on February 15-16, 1984. The following training records were obtained for evaluation:

Staff Position

The lack of timely training, for the newly employed or assigned site, piping stress and support design engineers was primarily the result of site supervision not requesting training as required by EDP 2.1 and DCP PEI 15. There was an apparent lack of PG&E GA training for the S/B group employees. Comprehensive QA training was not conducted until May, 1983. There was no documented evidence that the project unique type training required by procedure PEI for site supervisory personnel had been provided to the support design and piping stress engineers.

Action Required

The licensee will be required to assure implementation of training procedures. This completion of action will be a specific subject of future inspections.

Task: Allegation or Concern No. 83

ATS No: RV83A063 BN No:

Characterization

The Nuclear Regulatory Commission was not effective in identifying problems with small bore pipe support design procedures.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation 85

(previously addressed in SSER 21)

ATS No.   BN No.:

Characterization

U-Bolt design inadequate.

Related Allegations: 55, 79, 82, 87, 88, 89, 95, 97

Implied Significance to Plant Design, Construction or Operation

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

Assessment of Safety Significance

1. Technical approach to resolution

   a. Review PGE response to allegation.

   b. Evaluate DCP technical bases for U-bolt allowables.

   c. Determine resolution of the allegation.

2. Work Performed and Findings Identified

   The alleger has stated that U-bolt allowables specified by the Diablo Canyon Project (DCP) for the qualification of small bore pipe supports are incorrect and exceed those specified by the manufacturer, and that an interaction equation relating axial and transverse loading is unconservative.

PGE has submitted background information on U-bolt allowables for small bore piping supports in letters of December 28, 1983 and February 7, 1984. A meeting with the DCP was held at the site on January 6, 1984, in which PG&E discussed and provided the technical basis for the U-bolt allowables specified in DCP design documents. This meeting was attended by personnel from PG&E, Bechtel-San Francisco and Bechtel-Gaithersburg.

The staff has reviewed the PG&E submittals and test data. The U-bolt allowables were determined in accordance with prescribed procedures specified in ASME Section III, Subsection NB-3260. A concern regarding the sample size used in the tests was satisfactorily resolved in that PG&E based the allowables on the lowest test loads and not on the average test loads. This is considered equivalent to the requirement in NF-3260 that test loads be derated by 10% if the test consists of a single specimen. PG&E also demonstrated satisfactorily that the interaction equation specified in the DCP design documents has a reasonably adequate technical basis. The DCP also provided at a meeting on January 9, 1984 the results of a study of 112 U-bolts which were randomly sampled from the small bore support design calculation packages on site. They indicated that roughly 75% of the sample U-bolts were loaded below the allowables specified by the manufacturer, and thus considerably lower than the DCP specified U-bolt allowables.

Staff Position

The information provided by PG&E regarding the adequacy of U-bolt allowables and the interaction equation specified for the design of small bore pipe supports is acceptable, and finds that there is no basis for the allegation.

Action Required

No further action required.

Task: Allegation No. 86

ATS. No.: BN No.:

Characterization

"Code break" design.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation 87

(previously addressed in SSER 21)

ATS No. 83A063 BN No.

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.

Related Allegations: 55, 79, 82, 85, 88, 89, 95, 97

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

1. Technical Approach to Resolution
   a. Review sample of small bore support design packages with alternate calculations to verify allegations.
   b. Review design logs.
   c. Review PG&E response to allegation.
   d. Document findings.

2. Work Performed and Findings Identified

   The staff has reviewed the design calculations provided by the alleger, the relevant Diablo Canyon Project (DCP) design calculation packages and the site design calculation logs, all of which were provided by the Region V staff.

The alleger has provided ten alternate calculations, which are not included in the design packages of records. Of these two pertain to supports which have been deleted (MP 416 and MP 285). One calculation (MP 345) pertains to the allegation on altered documentation (see Allegation 55). A review of the remaining calculations is summarized as follows:

	Calculation Package	Alleger Calculation	Calculation of Record
1	MP-988 Hgr 100-132	Rev. 1 shows base plate failure	Rev. 1 shows baseplate of and bolts acceptable; contains errors; different analyst
2	MP-301 Hgr 2182-93	Rev. 1 shows rigid frequency requirement not satisfied	Rev. 1 refers calculation to Hgr 169-12; different analyst
3	MP-302 Hgr 2182-94	Rev. 1 shows rigid frequency requirement not satisfied	Rev. 1 refers calculation to Hgr 169-12; different analyst
4	MP-268 Hgr 98-82	Rev. 1 shows bolt failure by hand calculation	Rev. 1 shows bolt acceptable based on computer calculation; different analyst
5	MP-357 Hgr 2182-91	Rev. 1 shows rigid frequency requirement not satisfied, based on hand calculation	Rev. 1 shows rigid frequency requirement satisfied, based on computer calculation; different analyst
6	MP-303 2182-64	Rev. 1 shows rigid frequency requirement not satisfied, based on hand calculation	Log indicates referral to calculation MP-997; different analyst
7	MP-277 2182-66	Rev. 1 shows failure in torsion	Log indicates referral to calculation MP-174; different analyst

The evidence provided by the alleger indicates that in all cases the initial calculation indicates that some design requirement was not satisfied, and which are not included in the design packages of record. However, the staff was not able to verify explicitly that on-site management has actually destroyed these calculations exclusively because failure was shown. The DCP has stated (letter of February 7, 1984) that the only calculations required to be retained are the final calculations which show the qualification of the design, in accordance with ANSI Standard N45.2.9 (1979). The same letter also provided information for the fact that certain calculations were performed by more than one analyst.

The calculation logs have also been reviewed to determine that names and dates match those of the calculation packages. There appear to be two logs, one of which is older and appears to be a subset of the current log. For design package MP-988 these logs show two different analysts for "Rev. 1", although both calculations are shown approved on the same date. A similar instance was found for design package MP-944. The DCP has stated that the older log was an informal log, kept as an aid by the Assistant Onsite Project Engineer, and was never updated. The current log, also termed the record calculation or master index log, is the only log which is required to be kept up to date.

Staff Position

The staff finds that the allegation that management has purposely destroyed documentation is not substantiated. The allegation that new staffers were assigned to reperform the calculations, and that the master log does not reflect the initial calculations, is verified. The circumstances which form the basis for the allegation need considerable clarification.

Action Required

The staff will conduct further investigations to clarify the conditions under which management is permitted to retain or dispose documentation, and to reassign design personnel.

Task: Allegation 88

(previously addressed in SSER 21)

ATS No. RV83A063 BN No.

Characterization

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

Related Allegations: 55, 79, 82, 85, 87, 89, 95, 97

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 original reviewers, could result in structures unable to perform their intended function.

Assessment of Safety Significance

1. Technical approach to resolution

   a. Review sample of small bore piping and pipe support design calculations.

   b. Review PG&E response to allegation.

   c. Determine resolution of issues.

2. Work Performed and Findings Identified.

   Based on an on-site meeting with the alleger on December 7, 1983 five issues were identified concerning this allegation. The staff has reviewed and evaluated various Diablo Canyon Project (DCP) documents pertaining to these issues. Following is a summary of the five issues and their evaluation.

1. Code-break locations have been revised in order to reduce the number of safety related supports, and many of those that failed were omitted in the review program.

   The issue of Code-break analysis has been addressed separately (see Allegation No. 86). However, to address the specific items described above, the staff reviewed piping design package 3-313 during a site audit on January 10, 1984, and performed an inspection of the code-break region. PG&E also submitted an extensive response to this issue in a letter dated February 7, 1984, which provided additional information and clarification on the DCP code-break analysis methodology. Based on this information the staff has determined that there is a basis for the allegation, but also that the final specification of code-break locations and the design of the related supports were reasonably determined based on proper engineering analysis.

2. Gaps have been assumed that did not exist and vice-versa.

   This issue pertains to the inclusion and modeling of as-built gaps in rigid restraints in piping stress analysis with the objective of reducing thermal loading in piping. The staff has reviewed information regarding this issue provided by PG&E in submittals of December 28, 1983 and February 7, 1984. PG&E has stated that the DCP reviewed all small bore piping analyses and determined that this modeling technique was used in 25 small bore piping analyses affecting a total of 64 supports. 16 of these analyses involved piping with service conditions below a temperature of 200°F, where thermal movement is of relatively minor concern. For the other 9 analyses the temperature exceeds 200°F, and these analyses include 16 affected supports. The DCP stated that in 15 of these supports gaps were specified and modeled to reduce the effects caused by thermal anchor movement of attached large bore piping. For the remaining supports the gap was modeled to relieve thermal loads induced by two opposing supports restraining the pipe in the same direction. The DCP also stated that the thermal anchor movements due to large bore piping expansion are repeatable throughout the life of the plant.

Based on this information the staff concluded the following:

a. Gaps were modeled in accordance with as-built conditions. There is no evidence that non-existing gaps have been assumed in thermal analyses, while ignoring existing gaps in thermal analyses represents a conservative approach. In addition, the alleger has not provided specific information where instances of non-existing gaps have been assumed. The staff therefore finds this allegation to be without basis.

b. The practice of modeling gaps in piping thermal analyses is acceptable only if these gap configurations can be shown to be present and repeatable throughout the life of the plant. Otherwise, a more conservative approach is to ignore these gaps in thermal analyses. Previous plant experience has shown that gaps in supports are not always repeatable.

3. Joint releases have been assumed for rigid connections, without removing the welds.

PG&E has provided information regarding this issue in its submittal of February 7, 1984, describing the engineering basis and the application of this technique in pipe support analyses. Based on this information the staff has determined that the allegation is substantiated. However, the staff also finds the engineering basis and approach as described by the DCP acceptable and in accordance with current engineering practice.

4. Calculations were performed 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.

PG&E has provided information regarding this practice in its submittal of February 7, 1984, in which they state that the technique of determining the maximum load carrying capacities of supports, which are then used iteratively in piping stress analyses, is analogous to calculating the load rating of standard supports. The staff has reviewed this submittal and finds that this procedure is in accordance with current engineering practice and is therefore acceptable.

5. New supports were added within six inches of unacceptable supports. The new supports contain inaccurate assumptions on restraining gaps, and did not have control or document numbers.

During an audit performed at the site on January 9, 1984 the staff selected a random sample of ten new piping supports from different piping analyses. The staff determined that three existing old supports were in possible close proximity. The calculations for these supports were reviewed to determine if these had not been qualified before the addition of the new supports. No deficiencies were noted. In all cases the new supports were added at the request of the piping stress group and were properly documented. PG&E also provided information in the submittal of February 7, 1984 which lists reasons for adding new restraints such as meeting Code-Break criteria and valve acceleration requirements. They also indicated that in some cases new supports were added near existing supports to reduce the loads on the existing supports. This is an acceptable procedure in accordance with current engineering practice. The staff has reviewed all this information and concluded that there is no basis for this allegation.

Staff Position

1. The issue of Code Break analysis is considered resolved.

2. PG&E will be required to verify and monitor the support gaps modeled in those thermal piping analyses where the operating temperature exceeds 200°F.

3. The issue of assumed joint releases for rigid connections is considered resolved.

4. The issue of the calculation of the load-carrying capacity of small bore pipe supports is considered resolved.

5. The issue of new supports added near old supports is considered resolved.

Action Required

PG&E will be required to develop and institute a program of inservice inspection to verify and monitor the support gaps modeled in those thermal piping analyses where the operating temperature exceeds 200°F.

Task: Allegation 89

(previously addressed in SSER 21)

ATS No. RV83A063 BN No. N/A

Characterization

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

Related Allegations: 55, 79, 82, 85, 87, 95, 97

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 for piping and supports being addressed under Allegations 79, 82, 84, 87, 88 and 95.

Staff Position

This concern is addressed by the resolution of Allegations 78, 82, 84, 87, 88 and 95.

Action Required

None

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.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation or Concern No. 91

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

Characterization

Alleged coverup of defective material use.

Action Required

No further action required on this allegation - refer to SSER 21

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 dihedral angle requirements.

Implied Significance to Design, Construction or Operations

Undersized welds on safety related structural components could impair the components ability to handle design loads.

Assessment of Safety Significance:

These issues were resolved in SSER No. 21, however new information was obtained as a result of discussions with the alleger on February 2, 1984 which identified the following concerns:

1. Pullman allegedly uses a 37.5° ASME weld preparation on all AWS steel and structures instead of the 45° preparation angle required by AWS D1.1.

2. It has been alleged that the Site and General Office design groups did not properly account for tube steel radii that actually exist in the field.

On February 29, 1984 the licensee addressed these concerns and provided the staff with their resolution of these issues. The staff's review of this response indicated that, (1) Pullman weld procedure no. 7/8 allows bevel angles of both 37.5° and 45° and (2) in each case cited by the alleger of improper tube steel radii, the licensee had introduced sufficient conservatism which met the designer's intent.

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.

Action Required

As stated in SSER No. 21, the staff will continue to monitor the results of Lawrence Livermore Laboratories inspections performed under contract to the NRC-Region V. These inspections consist of examinations of pipe support configurations to insure their "as-built" status.

Task: Allegation 93

ATS No: RV83A063 BN No:

Characterization

See Allegation 92

Implied Significance to Plant Design, Construction, or Operation

See Allegation 92

Assessment of Safety Significance

See Allegation 92

Staff Position

See Allegation 92

Action Required

See Allegation 92

Task: Allegation or Concern No. 94

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

Characterization

Pullman used pipe welding procedures to make structural steel welds.

Action Required

No further action required on this allegation - refer to SSER 21

Task: Allegation 95

(previously addressed in SSER 21)

ATS No. 83A063 BN No.

Characterization

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

Related Allegations: 55, 79, 82, 85, 87, 88, 89, 97

Implied Significance to Design, Construction, or Operating

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

Assessment of Safety Significance

1. Technical approach of resolution
   a. Review PG&E response to allegation
   b. Review technical basis for DCP unbraced angle steel length specification
   c. Document findings.

2. Work Performed and Findings Identified.

   PG&E has provided detailed background information on unbraced length specifications for angle beams in the submittal of February 7, 1983. PG&E also has provided two technical reports on investigations performed by Australian researchers on the structural analysis of angle beams. One report describes the theoretical investigation in the structural behavior of angle beams subjected to bending type loads, the other provides data of an experimental investigation in angle beam behavior subjected to the same type of loading. Based on these tests and the theoretical evaluation, criteria were developed for specifying safe, unbraced lengths of angle beams. These criteria were adopted by PG&E for the Diablo Canyon Project (DCP) evaluation of angle beams.

The staff is currently reviewing the PG&E submittal and the two technical reports which form the basis for the DCP angle beam unbraced length design criteria. The unbraced length criteria adopted by DCP from these reports exceed those specified by the American Institute of Steel Construction (AISC) Manual of Steel Construction. However, the AISC Manual does not provide guidance in the evaluation of angle beams greater than certain lengths and indicates that special investigations are necessary for laterally unsupported angle beams. In this sense the reports provided by PG&E satisfy this requirement. In addition, the general topic of structural analysis and the specification in the AISC Manual of Steel Construction for angle beams and columns subjected to general loading is an ongoing area of industry investigation.

Based on the review performed to date, the basis for the DCP criteria regarding unbraced lengths of angle beams appears to be technically sound. However, the acceptability of these criteria will be determined when the staff completes its in depth review. These criteria may therefore not be acceptable for all loading conditions and combinations and may have to be revised by the DCP to satisfy staff concerns regarding the safety of pipe supports containing angle members.

Staff Position

The basis provided by PG&E for the DCP criteria regarding unsupported lengths of angle beams appear to be technically sound. The staff has however not completed its full review of the subject, and therefore this issue is presently unresolved.

Action Required

The staff is to complete its review of this issue.

Task: Allegation No. 96

ATS No. RV-83-A-63

Characterization:

A discrepancy exists between manufacturer's recommendations and PG&E design criteria regarding the minimum spacing between concrete expansion anchors.

Implied Significance to Plant Design, Construction or Operation:

The spacing of concrete expansion anchor bolts too close together can reduce the pullout capacity of the anchor bolts due to interaction of the concrete shear cones.

Assessment of Safety Significance

The spacing concern raised in the Stokes' affidavit involves Phillips and Hilti shell-type concrete expansion anchor bolts used for piping supports. These anchor bolts consist of a threaded shell, set in the concrete, into which a bolt is then inserted and threaded to secure the attachment. The hole in the concrete which secures the anchor shell is larger in diameter than the nominal bolt size.

The alleger provided a copy of a discrepancy report initiated on October 5, 1983, which he had prepared to document the discrepancy between the manufacturer's recommendations and PG&E criteria. Manufacturers recommend basing minimum anchor spacing on the concrete hole size; PG&E criteria bases minimum spacing on the nominal bolt diameter.

The staff approach to resolution of this issue was to examine how PG&E dealt with the discrepancy report and independently assess the technical adequacy of the disposition. The staff investigation found that:

(1) PG&E design criteria for anchor bolt spacing is in fact based on nominal bolt diameter versus concrete hole size as stated in the alleger's discrepancy report.

(2) PG&E had initiated action to evaluate this concern based upon an advance copy of the alleger's discrepancy report (8/8/83). The discrepancy report disposition is provided in a Civil Engineering Department letter dated September 28, 1983. The disposition included as justification the current American Concrete Institute (ACI-349, Appendix B) criteria for anchor bolt capacity and the results of a test program to show that the interaction effects were not sufficient to reduce the capacity of supports installed in accordance with the PG&E criteria.

(3) The alleger's concern, as presented in the discrepancy report, was resolved in a technically satisfactory manner.

Staff Position

The PG&E criteria for anchor bolt spacing, although not consistent with manufacturer's recommendations, is equivalent to the criteria recommended by the American Concrete Institute and is further supported by test program results. The staff therefore finds the PG&E criteria acceptable. The staff also finds that the alleger's concern was addressed in a responsible manner.

Action Required

None.

Task: Allegation 97

(previously addressed in SSER 21)

ATS No. RV83A063 BN No.

Characterization

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

Related Allegations: 55, 79, 82, 85, 87, 88, 89, 95

Implied Significance to Plant Design, Construction, or Operation

See Allegation 79

Assessment of Safety Significance

See Allegation 79

Staff Position

See Allegation 79

Action Required

See Allegation 79

Task: Allegation or Concern No. 98

ATS No.: RV-83-A-0085 BN No.

Characterization

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

Implied Significance to Plant Design, Construction, or Operations

Improperly installed penetration seals may fail during accident conditions, resulting in a failure to contain or limit liquid or gaseous effluents to a room or section in the Auxiliary Building.

Assessment of Safety Significance

This allegation was received in Region V on December 2, 1983 and subsequently investigated and resolved in SSER No. 21 (dated December, 1983). The staff concluded in SSER No. 21 that BISCO had never performed any work either as a contractor or as a subcontractor at Diablo Canyon. Subsequent to the writing of SSER No. 21, BISCO contracted with PG&E to work on 179 safety-related pressure seals at the Diablo Canyon site. BISCO completed their contracted work on February 1, 1984 and is no longer onsite.

To resolve the allegation that BISCO improperly installed penetration seals, the staff examined the BISCO Quality Assurance Manual (BQAM), the BISCO quality control procedures, records of employee qualification and certification, records of completed work, and examined a number of the BISCO installed penetration seals.

The staff's examination of the BISCO Quality Assurance Program indicated that the program complied with the requirements of 10 CFR 50, Appendix B and ANSI N45.2, and that quality control personnel were certified and qualified in accordance with the program requirements. Examination of ten BISCO installed penetration seals in the Unit 1 Auxiliary Building did not reveal any discrepancy from the licensee and/or the BISCO requirements.

The staff's inquiry into the extent of the licensee's QA/QC surveillance of BISCO's in-process work revealed that no formal audit or surveillance had been performed. The licensee's explanation was that BISCO's work scope was so limited, i.e. for a short period of time, that a formal audit was not scheduled. However, the licensee is in the process of performing a quality control inspection of 5% of BISCO's work to verify proper installation of the pressure seals.

Staff Position

The staff concludes that this allegation does not represent a safety concern at Diablo Canyon. This conclusion is based on the staff's review of the BISCO's quality assurance program, quality control procedures implemented during BISCO's work at Diablo Canyon, and the field examination of the BISCO installed penetration seals.

Action Required

The licensee's audit results of BISCO's work will be reviewed during the regularly scheduled NRC Inspection program.

Task: Allegation or Concern No. 99

ATS No: Q5-83-024/RV83A0068 BN No.

Characterization:

1. Falsification of vendors records (a former inspector claims to have documented inspections he had not performed).

2. Additional allegations from other sources concerning welder qualification, training and QC deficiencies.

Implied Significance to Design, Construction or Operation

Allegations by a 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 may be of questionable quality.

Assessment of Safety Significance

This allegation came to the NRC staff attention through a local San Francisco television reporter. Staff action was initiated at that time. In addition, the licensee initiated their investigation of this subject after viewing the television report. Since the original allegations were received the staff and the licensee, through their investigations, have received additional allegations. These concerns can basically break down into three chronological groups:

1. 1981 to 1983 - The initial allegation involving a former inspector who claims to have documented inspections he did not perform.

2. 1979 to 1980 - Allegations regarding maintenance of welder qualifications, lack of QA training, shipment of discrepant material to nuclear projects, questionable subsuppliers, and working conditions.

3. 1977 to 1979 - Allegations regarding a Bostrom-Bergen subsidiary (Meddco Metals) relating to material traceability, removal of reject tags, and use of foreign produced steel.

The NRC staff response to the allegations includes a combined effort by the Office of Investigations, the Licensee Contractor and Vendor Inspection Program Branch of the Office of Inspection and Enforcement, and Region V. The staff position has been both one of monitoring how the licensee is conducting his investigation for the Diablo Canyon Project and independently reviewing the issues for generic significance (the company has provided products to multiple nuclear reactor projects). The staff's approach to these concerns includes:

1. monitoring licensee actions and independently conducting interviews with a number of current and former employees of the company and its subsidiaries;

2. examining of the quality program employed by the company currently, and in the past;

3. reviewing contractor and licensee records related to the work in question;

4. and physically inspecting material installed at Diablo Canyon.

The staff has addressed and closed the original allegation. A review of pertinent records established that the former inspector (who claims to have documented inspections he did not perform) is credited with performing 650 inspections while he was employed at Bostrom-Bergen. Fifteen of the 650 inspections involve safety-related material. These fifteen items were found to be supplied to Diablo Canyon Unit 2 and involve "stock" material (i.e. raw material items which do not involve welding). As of this writing the staff has inspected 14 of the 15 items and found them to conform with requirements. The staff is following up on the last item (plate washers). The licensee has selected a 10% sample of the other (non-safety related) inspections related to the inspector and performed a reinspection (involving 940 welds). Seven of the 940 reinspected welds were found to have deviations from requirements, these are being properly addressed. Based upon the low defect rate the licensee has concluded that the structures and components installed at Diablo Canyon have not been adversely impacted by the former inspector's alleged performance. The staff concurs with this conclusion based upon a review of licensee actions and independent inspection of the fifteen safety-related items.

Neither the licensee nor the staff can determine conclusively whether the former inspector neglected to do the inspections.

The staff has completed a substantial amount of review on the second and third groups of allegations, and to date has not identified problems of safety significance, the reviews, however, are continuing (e.g. the staff has not completed their review of the Meddco Metals operations). These allegations are mainly general in nature, lacking in specific examples thus requiring extensive interviewing and document reviews.

In a parallel effort the licensee has initiated an inspection of installed hardware to allow a direct assessment of material adequacy, separate from the management and programmatic concerns related to the vendor. Items that are being reinspected were selected by reviewing all shop drawings and selected purchase orders involving the vendor's material shipped to the jobsite since 1969 and includes samples of each material type supplied to Diablo Canyon with particular attention to items which are difficult to fabricate or involve special materials.

90% of the sampling has been completed and the licensee reports that the following trends and results are apparent:

a) General inspections are finding that the existing geometries and dimensions are in conformance with the shop drawings.

b) Hardness tests are indicating that correct materials were provided.

c) Visual weld inspections are indicating that vendor welding meets design requirements.

d) Records from the NDE documentation research show that full penetration welds by the vendor are satisfactory.

In addition to the licensee's reinspection the staff has independently inspected a small sample (14 types of components) of installed safety related hardware to obtain first hand evidence of product quality. The components were visually inspected for material damage, weld location, length, size, shape, reinforcement, appearance and type.

The staff did not identify any discrepant material. Records related to this material were reviewed and appeared to be in order.

Staff Position

Investigations and reviews have been completed on the initial and most alarming allegations. This item is resolved. The reviews are continuing on the other two sets, but, to date significant safety problems have not been identified. Based upon staff findings to date and the acceptable results of reinspection of installed hardware it is the staff's opinion that this issue no longer requires full resolution prior to a reactor criticality decision. The staff estimates that their investigations will be completed by mid May 1984.

Action Required

Complete investigations referenced in the staff position.

Task: Allegation 100

(previously addressed in SSER 21)

ATS No.: RV83A0069 BN No.: None

Characterization

No quality control program for painting inside containment.

Implied Significance to Design, Construction, or Operation

Unqualified coatings and qualified protective coatings which have been improperly applied, i.e. relative to applicable ANSI standards, without required quality assurance program can produce peeling, flaking, or chalking of protions of the coating under DBA conditions. The transport of the resultant degris to the containment sump may affect the performance of containment safety systems, e.g, containment spray or core cooling system.

Assessment of Safety Significant

Previously addressed in SSER 21

Staff Position

As previously discussed in SSER 21 the staff concluded that there was no formal quality control inspection program nor was there a quality assurance program applied to painting activities inside containment. Considering the importance of the containment coating, particularily with respect to the potential for flaking or release of sheets of coating, further staff evaluation is required.

The staff has estimated the gas and solid debris formation from zinc, aluminum, and organic coatings inside containment under DBA conditions. With respect to deberis formation, a conservative source term of about 34 cubic feet has been determined. The staff is reviewing the potential consequence with post-accident operations with these amounts of unqualified paints in containment.

Action Required

Based on the initial review the staff has concluded that this concern need not be resolved for operation below 5 percent power because any requirement for recirculation flow following onset of a LOCA would be minimal. Moreover, the challenges to the unqualified paints would likewise be minimal. A resolution is required before exceeding 5 percent power operation.

Task: Allegation or Concern No. 101

ATS No: RV-83A-0073 BN No:

Characterization:

Alleged deficiencies in H. P. Foley welding program including: (1) the use of unqualified welders; (2) the use of unqualified welding procedure specifications; (3) gross erros contained in welding procedures including nonconformance with industry codes; and (4) design errors. The nonconformance reports which document the substance of the allegations were prepared by the alleger while an employee of the H. P. Foley Company.

Implied Significance to Plant Design, Construction or Operation:

The H. P. Foley Company performed safety-related welding on: electrical components; heating, ventilation, and air conditioning systems; and instrumentation tubing. Improper welding could cause equipment failure and attendant loss of design functions during normal operation or during design basis events, including seismic events.

Assessment of Safety Significance:

The alleger provided the staff with a listing of specific weld procedure specifications and quality control procedures and copies of nonconformance reports to support his contentions. The staff approach to resolution of this issue was to examine how the H. P. Foley Company addressed the concerns and nonconformance reports issued by the alleger, and to independently assess the quality of the Foley welding program using the specific examples provided by the alleger, supplemented with additional investigation of related areas of the welding program.

The alleger's stated concern was that the adequacy of safety-related welding performed by the H. P. Foley Company was indeterminate in that:

(1) Unqualified welders are used.

Staff Note: The alleger's concern regarding welder qualifications was vague and unspecific but appeared, from nonconformance report 8802-924, item 2 as supplied by the alleger, to be directed towards lack of formal welder training rather than lack of strict compliance with the qualification requirements of the applicable codes.

(2) Welding procedures are not adequately qualified. See QCP-5A, welding procedure specifications (WPS) 31, 32, 35, 36, and 86 and see comment sheets for QCP-5A (Rev. 9).

(3) Welding procedures QCP-5A (Rev. 3, 4, 5 and 9), QCP-5B (Rev. 3 and 4), QCP-5C (Rev. 0), and QCP-5D (Rev. 0) contain gross errors and are not in accordance with the applicable codes. See NCR-8802-924, for example, relative to QCP-5A, Rev. 9.

(4) As indicated in NCR-8802-924, item 9, there are design errors.

The staff investigated the above stated concerns by: (1) conducting a thorough review of procedures specified by the alleger and associated procedures; (2) reviewing past NRC and NRC contractor (EG&G) inspection findings involving Foley weld quality; (3) interviewing, in private, over twenty responsible license and contractor personnel; (4) reviewing over 700 nonconformance reports issued by H. P. Foley; (5) reviewing qualification records for six welders; and (6) examining weld quality on over 100 components located in the Unit 1 containment, cable spreading room, turbine building, control room, and 480 volt switchgear rooms.

The staff investigtion of the alleger's concerns found that:

(1) The alleger's assertion that welders were unqualified was not confirmed. A staff review of qualifications for six welders found them in compliance with applicable codes with the exception of the minor deficiencies discussed below. A previous NRC investigation of Foley welder qualifications and procedure QCP-5 for AWS structural welding found that the procedure provides a system for qualifying and maintaining the qualifications of welders which was confirmed by a review of qualification records for seventeen welders (IE Report No. 50-275/83-13). Further, the staff concludes that the Foley resolution of the alleger's nonconformance report No. 8802-942, Rev. 2 properly dispositioned the apparent concern regarding inadequate welder training. Two minor deficiencies were identified by the staff during review of this issue:

(a) The welder qualification list indicated that welder "US" was qualified to limitation 1 whereas the welder qualification records indicated that he was qualified to the more restrictive limitation 2.

(b) The qualification record for welder "M27" for limitation 5 does not indicate completion of an acceptable fracture test as required by the ASME Code Section IX, Subsection QW-452.4.

(2) No significant deficiencies in the qualifications of QCP-5A procedure Nos. WPS-31, 32, 35, and 36 were identified by the staff. WPS-86 has been deleted from QCP-5A and the staff was informed that it was never used. The staff concluded that the alleger's concerns over procedure qualifications were properly addressed in nonconformance report No. 8802-924 and associated correspondence. The following minor deficiencies were identified by the staff in the reviews of welding procedure qualifications for procedures contained in other QCP's:

(a) The welding procedure qualification records for QCP-5C, WP-RS-4 do not indicate performance of the macroetch test required by the AWS D1.4-79 code.

(b) The welding procedure qualification record for QCP-5D, M05 does not list the actual preheat used as required by the ASME Code Section IX, Subsection QW-201.

(c) Procedure QCP-5C does not place any restriction on the carbon equivalent of reinforcing steel welded in accordance with a qualified welding procedure as required by the AWS D1.4-79 code.

(3) No gross errors were identified in the welding procedures. The procedural concerns described in NCR 8802-924 were dealt with in a letter from the Quality Director to the alleger dated September 29, 1983, in nonconformance report Nos. 8802-942R2, - 938R2, -951, -940R1, and -941, and in Engineering Disposition Request Nos. 1337 and 1432. The staff considers all of the concerns described in NCR 8802-924 to have been satisfactorily addressed by the licensee. A minor deficiency was identified by the staff during the procedure review regarding instructions from PG&E to the H. P. Foley Company to weld thin sheet metal to the requirements of the AWS D1.1-75 Code which is not intended for this application.

(4) The staff identified no design error associated with the alleger's specific concern.

Staff Position

The review of the alleger's concerns failed to identify significant welding program or hardware deficiencies or any instance in which the alleger's concerns were not addressed properly. The welds observed by the staff in installed components and structures, were judged adequate for their applications.

The evaluation of the licensee's compliance with regulatory requirements and applicable industry codes and standards conducted during the investigation identified some deficiencies. These deficiencies are not considered significant enough to suggest that the integrity of safety-related components is in question, however, they must be addressed and satisfactorily resolved by the licensee.

Action Required:

Evaluation of apparent deficiencies by the NRC staff for possible enforcement action and examination of licensee actions to resolve the identified deficiencies. This will be monitored through the routine inspection program.

Task: Allegation or Concern No. 102

ATS No: RV-83-A-0070 BN No:

Characterization

Improper References on DCN.

Implied Significance to Plant Design, Construction or Operation

See Task Allegation or Concern No. 61

Assessment of Safety Significance

See Task Allegation or Concern No. 61

Staff Position

See Task Allegation or Concern No. 61

Action Required

See Task Allegation or Concern No. 61

Task: Allegation or Concern Nos. 103, 104, 105, 106, 107, 108, 109, 110, 111, 112, 113, 114, 115, 116, 117, 118, 119, 214, 215, 216, and 217

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization

Multiple allegations associated with a failure of the licensee and Pullman Power Products to meet required codes and standards for welding pipe supports and pipe whip restraints.

Implied Significance to Design, Construction or Operation

The failure to meet stated codes and standards in the fabrication of pipe supports and pipe whip restraints may result in components which would not perform their intended safety function.

Assessment of Safety Significance

The allegations or concerns discussed in this section were received in the form of a 35 page letter from the alleger to a NRC Commissioner. Attached to the letter were numerous documents provided to support the allegers concerns. The staff's general approach to address these concerns was to interview the alleger, examine the contractors and licensee's written requirements, examine pertinent procedures, documentation, and to conduct interviews with personnel, as appropriate.

The alleger's written submittal and interview included multiple cross referencing of issues. The staff did not examine every example of each type of issue individually, but instead focused on the substantive technical and quality concerns by grouped topics. Many of the issues were topics which had been formally documented and addressed by the licensees and contractor's control programs. The staff directed special attention to where the licensee and contractors addressed these items in a responsible manner. The staff has placed the issues into 21 topics. These are discussed individually below.

1. Allegations 103, 104 and 105:

Pullman Welding Procedure Specification (WPS) 7/8 was inappropriately applied in that deviations from WPS 7/8 existed in the following areas:

(a) structural shapes,
(b) weld joint geometry,
(c) materials

Staff Position

(a) The alleger is correct that WPS 7/8 was used to weld structural shapes in addition to piping and plate as specified in the WPS. However, the structural shape of the member is not required to be included in the WPS. All structural shapes, such as W, H beams and angle iron, shall have the connecting sections prepared to conform to the weld joint configuration of the qualified WPS. The structural shapes are identified on the design drawings.

(b) The alleger is correct in stating that the WPS documents do not adequately illustrate all joint types which are welded. WPS 7/8 is qualified in accordance with ASME Section IX requirements which indicates in QW 402.1 that a change in joint type is a non-essential variable. Lack of description of all types of joints utilized is contrary to Section IX rules and requires a revision to the WPS. However, this is an administrative change only and does not require requalification of the WPS.

(c) In response to the allegation regarding unapproved welded materials, the staff reviewed each type of material identified by the alleger. Certain of these materials such as A500 and A307 were not listed in the published code but were approved for use by a separate code case. The staff is satisfied that all the materials of concern in this allegation were properly approved for ASME or AWS usage.

2. Allegations 106, 107 and 108:

The alleger stated that Welding Technique Specification No. AWS 1-1 was not applied to AWS welding in that, (a) AWS 1-1 was not referenced on every Pipe Rupture Restraint Welding Process Sheet, (b) AWS 1-1 was written and approved by an unqualified individual, and (c) AWS 1-1 specified an unlisted AWS code material.

Staff Position

(a) The alleger is correct that in some cases QC failed to clearly identify on the weld process sheets when welding was to be conducted to the WPS plus the Welding Technique Sheets. However, the use of Welding Technique Sheets to amplify and clarify WPS documents is an accepted standard industry practice. At Diablo Canyon the significant clarification made by the Welding Technique Sheet is the introduction of tighter controls on preheat. Whether this information was directly tied to the WPS through the technique sheet is of little consequence since the same information is clearly stated in other relevant documents (EDS 223 and EDS 243). As the preheat is covered in all cases, the inclusion of the exact document, whether it is the WPS or Welding Technique Sheet identification, is considered to have no engineering or quality related significance.

(b) The alleger expressed concern that a Welding Technique Sheet was prepared by an unqualified individual. In so doing Pullman utilized a QA/QC person to perform a function out of his area of expertise and permitted this individual to audit his own work. The staff found that there are no codes and standards requirements that state that a WPS or Welding Technique Sheet must be prepared by a specific individual. The only requirement is that the document adequately address the codes and standards variable rules i.e., essential and non-essential variables. The WPS documents and Welding Technique Sheets met the rules (with the exception of the QW 402.1 non-essential variable as previously discussed) and were properly approved by the licensee. QA/QC personnel normally monitor implementation of programs and procedures, the fact that they may have assisted in writing the implementing procedures does not support the conclusion that QA/QC is auditing its own work.

(c) The alleger is correct that ASTM A515 steel is not listed in AWS D1.1 as an approved welding material. The staff found that A515 is not listed in AWS D1.1 Structural Welding Code because A515 is normally considered as a pressure vessel material. However, A515 was properly qualified and is acceptable material for welding structures in compliance with AWS D1.1 rules.

3. <u>Allegations 109 and 110:</u>

The alleger states that structural steel pipe supports were not designed, fabricated and erected to the American Welding Society (AWS) code. He further states that the PG&E Contract Specification 8711 requires pipe supports to comply with the applicable standards of the ASTM, ANSI, ASME, MSS, AWS, and PFI. Additionally, he states there was no change to the PG&E contract specification to allow pipe support to be worked to a standard other the AWS.

<u>Staff Position</u>

The staff found that the pipe support work was properly done to the ASME code which is permitted by the AWS code. Supporting details of the staff's findings are as follows:

- The American Welding Society D1.1 permits the ENGINEER to "accept evidence of previous qualification." It is normal practice to interpret this as permitting ASME Section IX welding qualification in lieu of D1.1 qualification by testing. In addition, the 8711 Specification Section 3 (para 4.11 and 4.12) require performance and procedure qualification in accordance with Section IX. Based on staff reviews, the welding qualification methods utilized by Pullman meet ASME Section IX requirements.

- The materials for pipe support welding were: A36, A500, SA515, SA516, and bolting materials A307, and A108 (grades 1010-1020). The staff found that each of these materials is suitable and allowable for ASME pipe support welding.

The staff reviewed Pullman procedure qualification documentation for engineering justification for welding in accordance with current ASME Section IX and AWS D1.1 rules (through utilization of the ENGINEER'S prerogatives in paragraph 5.2). This review included the procedure qualifications for "as-welded" fabrications and the following types of welding: ASME P1 to P1 material using shielded metal arc welding (SMAW); AWS Group I to Group I, using SMAW; AWS Group II to Group I and II, using SMAW; Welding of SA500, A441, A588, using SMAW; welding ASME P1 to AWS Group I using gas tungsten are welding (GTAW), ASME P8 to P8 using SMAW; ASME P8 to P8 using GTAW; tack welding, using SMAW or GTAW. Various thickness ranges were included.

All WPS documents were properly qualified for AWS welding, all structural steel fabrication met AWS requirements. Therefore, no contract specification change was required or needed.

4. <u>Allegation 111 and 112:</u>

Contract Specification No. 8833XR was not officially changed/revised to reflect that procedure qualification in accordance with ASME Section IX may be used in lieu of AWS D1.0-1969.

<u>Staff Position:</u>

The staff found that no contract specification change was required because the AWS Code allows qualification of "other processes" and "evidence of previous qualification" of joint procedure qualification. In this case, Pullman Power Products provided evidence of qualification to ASME Section IX, which is allowed by the AWS Code. Therefore, no contract specification change or revision was needed since no deviation from the contract specification had taken place.

5. <u>Allegation 113:</u>

Contract Specification No. 8833XR requires welders to be qualified to the AWS Code, instead Pullman utilized welders qualified to ASME Section IX to perform the scope of work required by the contract.

Staff Position

The staff found that ASME Section IX qualified welders are qualified to AWS rules if the AWS thickness criteria is properly addressed. The staff found that the AWS thickness criteria was properly addressed and therefore, the Pullman welders were qualified in accordance with Contract Specification No. 8833XR requirements.

The licensee's and contractors practice of using ASME/AWS qualified welders is reasonable and acceptable in this case.

6. Allegation 114:

Pullman utilized welding procedures which have not been tested for notch toughness in the weld Heat Affected Zone (HAZ) for weldments made under Contract Specification 8833XR (pipe restraints). Contract Specification 8833XR requires in Section 3.6 such qualification. The Pullman practices in this area represent a deviation from the contract specification.

Staff Position

The alleger is correct in that Contract Specification 8833XR does require HAZ notch toughness verification. However, this requirement was clarified with a contract revision which indicated that notch toughness is required (only) if specified on the drawing.

Licensee correspondence and staff reviews indicate that HAZ notch toughness is not required, and therefore, the design of the rupture restraints does not require welding qualification documents demonstrating HAZ notch toughness. The licensee position that notch toughness verification is not required is documented in a licensee to NRC memo dated Janauary 18, 1984. Notch toughness in the weld HAZ is not a code or NRC requirement for rupture restraints.

Therefore, the alleger is correct that the Pullman practices in this area appear to represent a deviation from the contract specification, however, the staff found that because of the licensee correspondence referenced above no deviation from Contract Specification 8833X had occurred.

7. Allegation 115:

No Contract Specification Change Notice was issued authorizing the deletion of full penetration welds less than 9/16 inch effective throat from the ultrasonic examination program for the repair of pipe rupture restraints.

Staff Position

The staff's examination of licensee documents and discussions with engineering and quality assurance individuals revealed that the licensee's Engineering Department did not formally revise or process a design change allowing a deviation from Contract Specification 8833XR, paragraph 7.21. This item is not considered a safety problem because all the technical requirements and procedures for ultrasonic examination were reviewed and approved by the licensee. However, it does represent an unauthorized change which is not in strict compliance with Engineering Department Procedure No. 3.6 "Design Changes." This failure to formally change the contract specification appears to be an oversight on the part of the licensee, since all appropriate reviews were conducted, and approvals obtained.

Therefore, the alleger is correct that no contract specification change was initiated, however, based on the above no safety significance is attributed to this administrative oversight.

8. Allegations 116 and 117:

Pullman weld procedure code No. 88/89 was used to weld plate when the procedure was qualified for pipe welding under ASME Section IX. The Pullman weld procedure was never qualified in accordance with the AWS Code as required by Contract Specification No. 8833XR.

Staff Position

The staff found that no contract specification change was required because the AWS code allows qualification of "other processes" and "evidence of previous qualification" of joint procedure qualification. In this case, Pullman Power Products provided evidence of qualification of WPS 88/89 to ASME Section IX, which is permitted by the AWS Code. The AWS Code states that qualification on pipe shall also qualify for plate.

Therefore, no contract specification change or revision was needed because no deviation from contract specification had taken place.

9. <u>Allegations 118 and 119:</u>

Pullman Power Products uses a Welding Technique Sheet (AWS 3-1) to allow Gas Tungsten Arc Welding (GTAW) and a material (A515 steel). Neither of which are not allowed by the AWS Code.

<u>Staff Position</u>

This allegation is addressed as two parts as follows:

a. <u>Gas Tungsten Arc Welding is not allowed by the AWS Code (Allegation 118)</u>

The alleger is correct that the gas tungsten arc welding (GTAW) process is not specifically covered in the body of AWS D1.1. However, AWS D1.1 (paragraphs 1.3.4 and 5.2) permits qualification of "other processes" and "evidence of previous qualificiation" of joint procedure qualification. Pullman Power Products has demonstrated proper ASME qualification of this process and is, therefore, considered satisfactory for welding supports and restraints.

The GTAW welding process was qualified in accordance with AWS D1.1 provisions; therefore, there is no safety or quality management significance attributed to this allegation.

b. Grade A515 Steel is a Material not Listed as Approved in the AWS Code (Allegation 119)

The alleger is correct that ASTM A515 steel is not listed in AWS D1.1 as an approved welding material. A515 steel is not listed in AWS D1.1 because the steel is normally considered as a pressure vessel material. However, A515 was properly qualified and is acceptable for welding structures in compliance with AWS D1.1 rules.

10. Allegations 214, 215, 216, and 217:

The use of Code 92/93 to weld pipe rupture restraints when the process sheets specified Code 7/8 and Pullman's justification for this change is a major breach in the welding Quality Assurance Program.

Introduction

The alleger refers to a September 15, 1978 memorandum to file from the Assistant QA/QC Manager. This memorandum states, in part, "Both weld codes 7/8 and 92/93 are qualified to allow welding of unlimited thickness on structural members under AWS requirements. Technical aspects of both procedures are the same."

Assessment of Safety Significance

The staff examined the referenced memorandum and supporting documentation. Based on this review, it is clear that the alleger has four issues in question. The following is a characterization of these four issues along with the staff's conclusions:

a. Allegation 214:

The alleger's concern was that Welding Procedure Code 7/8 and 92/93 were not identical. He lists a number of welding parameters which are different between the two weld procedures. The staff found that the alleger is correct the procedures are not identical, though from a technical standpoint they are both acceptable for the work required (the rupture restraint work). This allegation appears to be an apparent misunderstanding on the alleger's part on the interchangeability of the welding procedures.

b. Allegation 215:

This concern is whether or not Code 92/93 is qualified to allow welding on unlimited thickness structural members under AWS requirements. Based on staff examination of AWS D.1-1 and Pullman's use of Code 92/93, the staff concludes that Code 92/93 has been properly qualified.

c. Allegation 216:

This issue is that Code 92/93 is not a suitable substitute for Code 7/8. As mentioned in item 1 above, even though, the two documents are not technically identical, they are both technically adequate for the work that was performed. Therefore, there is no safety significance associated with this issue.

d. Allegation 217:

Based on the alleger's concerns that the above three issues were safety significant, the alleger concluded that Pullman's QA/QC management attempted "to cover up a serious breach in the Quality Assurance program for welding Pipe Rupture Restraints...." However, because of the existence of the Assistant QA Manager's memorandum and the alleger's misinterpretations discussed above, the staff cannot see any objective basis for the conclusion that a "cover up" was attempted or existed. To the contrary the Pullman memorandum makes it a formal document available for all to see and review.

Staff Position

The allegation is not substantiated. It may have been generated, in part, because of a misinterpretation of the September 15, 1978, Interoffice Memorandum.

Action Required

None.

11. Further Allegations

A further staff examination of the alleger's submittal disclosed the following information:

This allegation relates to the installation of the Unit 1 containment spray ring piping in 1972. A review of the records associated with this activity resulted in the identification of discrepancies between the weld process sheets and weld rod requisition documents. These discrepancies were documented on Pullman Discrepancy Report (DR) No. 4713, dated April 14, 1983. The alleger contends that the Discrepancy Report misrepresents the discrepancies in order to cover up more significant Quality Assurance/Quality Control problems. More specifically the alleger states that:

a. DR No. 4713 did not identify the fact that the Production Department disregarded the process sheet and the specified weld procedure and substituted their own unauthorized and unapproved weld procedure (Code 15/16).

b. The DR does not address the failure to detect the discrepancies at the time they occurred.

c. The DR states that all welders were qualified, when, in the alleger's opinion one welder's (Welder "N") qualification status cannot be assured for the time period involved (since the Ninety Day Welder's Activity Log was not maintained from August 1972 to December 1972).

Staff Position

To address these issues the staff reviewed the DR, the contractor's response to the DR, examined evidence of weld procedure approval and interchangability, examination of welder activity logs, process sheets rod requisition documents, and other records. The allegers concerns are addressed below:

a. Use of unapproved and unauthorized weld procedure Code (Code 15/16).

A staff examination of the weld procedure in question (Code 15/16) disclosed that it had been properly qualified and approved. Therefore, the alleger's statement that unapproved procedures were used is incorrect. The statement is correct that the record discrepancies make it somewhat unclear as to which specific procedures were used. The staff, therefore, requested the licensee to perform a technical review of weld procedure interchangability. The conclusion of the review was that, for the weldments in question, any of the welding procedures listed could have been used to achieve acceptable welds. The staff concludes that there is no technical significance to the record discrepancies in this case. The general implications of record errors follows in item b.

b. Failure to detect the discrepancies at the time they occurred indicates a significant breakdown.

The alleger contends that since the personnel involved in the work at the time (crafts, QC/QA, supervisors) failed to detect the discrepancies and that this is indicative of a significant breakdown. The staff examined the situation to determine whether the record discrepancies were widespread (significant) or somewhat isolated. To assess the magnitude of the record discrepancy problem, 300 weld process sheets were reviewed. 100 for the Containment Spray System, 100 for Chemical and Volume Control System and 100 for Component Cooling Water system. These process sheets are for welds (piping to piping, attachments to piping, and pipe supports) completed between April 1972 and October 1975. There are 531 weld rod requisitions associated with these process sheets. The staff examined results of these reviews. The results showed that 20 weld rod requisitions records (15 Containment Spray, 5 Component Cooling Water) have a WPS listed on them that is not in agreement with the process sheet. This equates to 3.7 percent. Based on the results of the review it does not appear that record discrepancies were a widespread problem.

c. Welder "N" qualification status.

The alleger contends that the Ninety Day Welder's Log was not maintained current during August 1972 to December 1972 and therefore, Welder "N"s qualifications are in question. The staff notes that neither the AWS or ASME Codes require maintenance of a welder's activity log. The codes simply require that welder activity be maintained. The method of providing evidence of welding activity is not stated, and no explicit method is required to meet codes and standards rules. Based on NRC inspection there is evidence (e.g., filler metal issue slips, welding process sheets, etc) that welder N met the above codes and standards requirements.

In his written submittal the alleger further alleges that process sheet and rod requisition records cannot be relied upon to reconstruct evidence of welder activity. His basis is a discrepancy report identifying inconsistencies between process sheet and rod requisition records related to 1972 work involving the containment spray ring. An examination of the records, however, does not disclose widespread inaccuracies in the records (as discussed previously). Based on the above it appears that Pullman's and the licensee's approach to resolving the question of welder "N"'s activity is an acceptable approach.

Based on the results of the foregoing, the staff concludes that the individual discrepancies are not technically significant, that the discrepancies were not widespread, and that procedures and welders were appropriately qualified. It does not appear that Discrepancy Report No. 4713 misrepresents the scope of the problem. It appears that licensee and contractor management handled the problem in a acceptable manner.

Overall Staff Position

The staff's review of the above allegations disclosed that there were minor, isolated weaknesses in implementation of the contractor's and licensee's program. However, these discrepancies were not widespread and were primarily administrative in nature. The welding processes, welding procedures, welded materials, welders and nondestructive examinations were found to be in accordance with the required codes and standards.

In general, it appears that the licensee and his contractor managed their activities in a reasonable manner.

Action Required

None.

Task: Allegation 104

ATS No: RV-83-A-0074   BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 105

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 106

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 107

ATS No: RV-83-A-0074   BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 108

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 109

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 110

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 111

ATS No: RV-83-A-0074   BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 112

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 113

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 114

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 115

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 116

ATS No: RV-83-A-0074   BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 117

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 118

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation 119

ATS No: RV-83-A-0074 BN No: 84-009 (1/16/84)

Characterization:

See Allegation 103

Implied Safety Significance to Design, Construction or Organization:

See Allegation 103

Assessment of Safety Significance:

See Allegation 103

Staff Position:

See Allegation 103

Action Required:

See Allegation 103

Task: Allegation or Concern No.120

ATS No: RV83A074, Q5-84-012 BN No:

Characterization

Pullman - possible intimidation of personnel

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Allegation or Concern No. 121

ATS No. RV-83-A-0074 BN No. BN-84-014 1/24/84

Characterization

There are inadequacies in the implementation of the quality assurance program documenting the ultrasonic verification of valve wall thickness and the repair of valves found below minimum wall thickness. It was also alleged that pressurizer safety valves were designed to the incorrect code.

Implied Significance to Design, Construction or Operation

From a technical standpoint, the first concern implies that the wall thickness of the valves examined may not meet design wall thickness requirements and these valves may not be adequate for the intended service conditions; the second concern implies that the pressurizer safety valves installed in the plant may not be adequate for their intended service.

From a management standpoint, these concerns imply a failure of the Quality Assurance System at Diablo Canyon to track and control the construction program.

Assessment of Safety Significance

The alleger's concerns involve a 1972 AEC mandated inspection program which was executed in 1973. The alleger did an unscheduled audit of the records of the inspection in his capacity as Pullman internal auditor in 1982. The alleger is concerned that his audit findings were not properly considered in the Pullman response.

The staff's approach to resolving these issues was to examine the Pullman quality assurance audit report provided by the alleger, examine the Pullman response to each audit finding, assess whether the response was adequate, and assess the significance of each audit finding and response. The licensee (PG&E) subsequently responded to the entire allegation by letter dated February 29, 1984; those comments were also examined. On March 7, 1984, the staff discussed, by telephone, the measurement of valve body wall thickness with the individual who had performed about 50% of the wall thickness measurements. The results of those discussions are also presented, as appropriate, in the discussion of staff findings.

The allegations are addressed and assessed below, in the original order of presentation by the alleger.

A. There was no qualified procedure for ultrasonic thickness (UT) measurements of valve wall thicknesses.

   A1. There is no evidence of a procedure qualification record (PQR) documenting that the UT procedures "Engineering Specification Diablo" (ESD) 235 and ESD 244 were qualified by a proven demonstration, as required.

A2. A procedure qualification was not performed as required above, there was no demonstration of the maximum error in repeatability of the test.

A3. There was no procedure verification test documented that would have determined transducer requirements. Further, there is an Interoffice Correspondence (IOC) which specifically states that the required transducers and wedges were not available.

A4. Such UT valve wall thickness tests constitute special processes and as such must be controlled and accomplished using qualified procedures.

Staff Findings: Allegation A, while literally true, does not constitute a violation of the procedure qualification requirements. Wall thickness measurement is not a special process as the term 'special process' is intended by 10 CFR 50. Using a UT gauge for wall thickness measurement is akin to using a micrometer to measure a dimension, which similarly is not a special process. Moreover, for ultrasonic thickness measurement, the procedure is verified each time the preoperational calibration is made. Test blocks of correct material, grain refinement and known thickness were used to calibrate the test system before and after each series of measurements, this constitutes adequate procedure qualification and error verification.

The IOC identifying that proper equipment was not available on-site was dated 4/17/73; work was performed during June through August of 1973.

This IOC most probably initiated purchasing activity to acquire the correct equipment. If the correct equipment was not on hand, the results achieved would not have been possible. Transducers had sufficiently small diameters so that shoes and wedges were not required. The PG&E response dated February 29, 1984, PG&E Letter No. DCL-84-085, is adequate.

Discussions with the technician performing most of the thickness tests indicated that:

- specialized shoes and wedges were not necessary because acceptable results were obtained using stock items.

- specialized transducers were not necessary because acceptable results were obtained using 1/2" - 3/4" transducers.

- reference blocks were used of material as specified on the valve drawing. Each block had a discrete dimension recorded on the block.

- sound wave velocity compatibility was verified from the test block to the actual valve body by measurement of thickness with a micrometer and comparing this to the ultrasonically indicated thickness. This also constitutes a verification of repeatability.

B. An audit of 254 valve wall thickness data reports revealed excessive omissions and inaccuracies, implying reduced traceability.

B1. No data reports described transducers used for measurements.

B2. Most reports list transducer serial numbers, but the transducers are not traceable by serial number.

B3. Seven reports have no UT instrument serial number listed.

B4. Nineteen reports have transducer frequency missing.

B5. Nineteen reports list two types of instruments but have only one set of readings.

B6. Two hundred and seven reports listed UT instruments that could not be traced to calibration certifications.

Staff Findings: The staff did not confirm the alleger concerns of excessive omissions and inaccuracies in the data reports. With item B1, the allegers concern that information required to be recorded was not recorded is somewhat misleading. Neither the procedure nor the data sheet required the recording of such information. Furthermore, industry codes and standards do not require that such information be recorded for wall thickness verifications. It seems the alleger apparently has based his concern on his opinions of UT thickness measurement requirements.

The inspector did find some minor errors in the records as alleged in items B2-B5. A conversation with the technician who performed a major part of these thickness tests identified a possible cause of these errors as the union requirement for the use of laborers unfamiliar with UT as recorders during the testing. In any case, the absence of specific information on a transducer or instrument has no effect on the results of the measurement since the measuring system is completely verified before each measurement.

Traceability of instrument calibration certifications is not critical to test accuracy in UT thickness measurements. The only calibration aspect that might have an impact would be receiver and display horizontal linearity, but these are verified during calibration prior to each measurement. PG&E has however committed to locating the certifications of item B6.

The Pullman response in Audit Action Requests (AAR) numbers one and two to items B1-B6 are considered adequate.

B7. Fourteen reports are missing serial numbers for micrometers used in mechanically measuring wall thickness.

B8. Eighty-four reports reference serial numbers for micrometers that could not be traced to calibration certifications.

Staff Findings: The staff found the forms required the recording of micrometer serial numbers and further it is evident that the clerical requirements for recording the serial numbers in a clear, legible fashion was not well executed. However, these execution difficulties do not appear to make any technical difference.

According to Pullman and PG&E, all micrometers in use at DCNP were in a calibration program at the time these tests were performed.

Regarding item B8, there were seven micrometer identification numbers common to the eighty-four reports in question. However, review of these numbers shows omission of digits, recording a single digit in error, illegible printing, and transposition of digits most probably caused loss of traceability alleged by item B8. Item B7 is obviously due to complete omission by the recorder.

The inspector has determined that since these micrometers were used only for reference checks and not as final acceptance criteria, these omissions and recording errors do not represent technical problems. If there were inaccuracies in the micrometers being used, the error would have been discovered during the valve and stepwedge material verification steps. In letter DCL-84-085, PG&E stated that a complete review of the data reports showed no instance where UT equipment was readjusted as a result of the mechanical check performed by the micrometers.

Pullman responded adequately to these findings in AAR Number 3.

B9. Six reports do not list any information about the stepwedge calibration blocks.

Staff Findings: Data reports required recording of the calibration stepwedge material, since alloy and heat, as well as metal fabrication process, would effect sound velocity and, thus, measurement accuracy. In the six instances brought up by this allegation, the data was omitted from the record. Pullman responded to the audit finding in AAR number four by ensuring the stepwedge used did in fact match the material of the valve. This could easily be done because each stepwedge had unique actual step dimensions, and the data sheets had these dimensions recorded. Thus, it was a simple matter to verify that the proper stepwedge, of the proper material, had been used for calibration prior to measurement of the actual valve body.

B10. Eleven reports do not list pre or post operation calibration information.

Staff Findings: Preoperational and post operational calibrations were performed using calibration blocks and documented on the data reports in all cases. Item B10 is not true. Discussions with the individual who performed the thickness measurements further indicated that calibration verification checks were performed during the measurement process, in addition to those before and after measurements.

B11. Forty-two reports indicated valves below minimum wall thickness but sign offs occurred in spaces which indicated the valves were physically marked as acceptable.

Staff Findings: The pertinent line item on the data sheet was: "Valve identified per step 7.3.5 by _____". Step 7.3.5 of ESD 236 deals only with marking accepted valves, an initial in this space therefore indicates the valve was marked as acceptable. The inspector finds this item confusing - a technician could well have misunderstood the intent and signed off here even when marking a valve unacceptable.

The Pullman response via AAR number 6 was adequate. PG&E has recently reviewed all data sheets and confirmed that all valves identified as under minimum wall thickness were replaced, repaired, or accepted through engineering evaluation. (Refer to item C1 below). The inspector concludes these were recording inaccuracies and do not represent technical problems.

B12. Many reports had original information "whited out" and new information inserted with no explanation for changes.

Staff Findings: For a period of time this method of "white out" was used for making corrections. It was changed to single line-out subsequent to this testing. Discussions with the technician performing the majority of examinations indicated that during the examination care was taken to record the smallest thickness. Thus, movement of the transducer around on the valve body sometimes identified a smaller thickness than that previously recorded, and the previous entry was "whited out" in order to enter the smallest number observed.

The Pullman response by AAR number 6 was adequate.

B13. Eleven reports did not have a complete measurement inspection of all areas of the valve as required by the procedure. There is no documentation authorizing the incomplete measurements.

Staff Findings: PG&E Letter DCL-04-085 states that these eleven valves were new replacements for previously rejected valves and that they had been fully UT tested by Westinghouse prior to shipment. On receipt, although not required to do so, PG&E spot checked areas which were found questionable during a review of vendor documentation, which explains the incomplete ESD 236 documentation. These actions are documented in a licensee letter, dated December 5, 1973. The Pullman response by AAR number seven was adequate.

B14. Fourteen valves listed by Westinghouse letter #PG&E-2080 to be measured have no documented evidence of being examined.

Staff Findings: Pullman had direction to exclude these 14 valves from the measurement program. Eight valves were not primary pressure boundary valves and were deleted from the measurement program by Westinghouse. These were valves 1-8368 A, B, C, & D and 2-8368 A, B, C & D. The remaining six valves, 1-8010 A, B & C and 1-8010 A, B & C were not required to be examined in accordance with the exception in Westinghouse letter PG&E 2080. Pullman responded to this audit finding adequately in AAR number 7.

B15. Two of the twenty valves physically checked had serial numbers that did not match the data report serial numbers.

Staff Findings: B15 is true, but is a minor irregularity. The two valves were sight checked and serial numbers match the report serial numbers. The material heat number had been mistakenly recorded as serial number; traceability was not lost as a result of this error. Pullman's response in AAR number 8 was adequate.

C. The documentation on the disposition of valves found to be less than minimum wall thickness, including procedures used to repair any such valves, is inadequate.

C1. There are forty-seven reports that indicate valves were below minimum wall thickness. AAR#8 identified two valves that were weld repaired, but ESD 236 documentation packages do not specify which valves were weld repaired.

Staff Findings: PG&E Ltr DCL-84-085 claims 33 valves in both units were found to be under minimum wall thickness. The alleger's number of 47 is attributed to some valves receiving multiple tests and rejections. The licensee further points out that ESD 236 did not require indication of the final disposition of rejected valves, and that documentation of disposition of these valves is traceable through the deviation reporting system. Of the 33 valves; 15 were replaced, 9 were accepted "as is" through Westinghouse calculations, and 9 received weld build-up by the vendor. The staff considers the Pullman and PG&E responses to the alleger's concerns are adequate.

C2. There is no documented assurance of approved weld procedures or description of techniques used to verify acceptance of the repaired valves, as required by the AEC letter of 6-20-72.

Staff Findings: PG&E Ltr DCL-84-085 points out that ESD 236 documentation packages were not intended to document valve repair. All weld repairs were performed by Westinghouse, and the valves were retested on return to the site. These test results are in the ESD 236 documentation packages. Weld repair procedures were submitted to the AEC directly by Westinghouse, (Mr. Searls' letter of July 23, 1974). These approved procedures were retained by the vendor and were not available for the alleger's review. The Pullman and PG&E responses are adequate.

D. What is the relevant code or standard for minimum wall thickness to which reactor coolant pressure boundary valves should conform? The AEC letter of 6-20-72 defines wall thickness requirements as those specified by ASA B31.1 (1955), USAS B31.1.0 (1967), USAS B16.5, or MSS-SP-66 which were in effect on the date of the purchase order. The Westinghouse letter PG&E-2080 states that pressurizer safety valves were designed to ASME Boiler and Pressure Vessel Code Section III, Article 9 (1968) and were not designed to meet minimum wall thickness requirements of ANSI B16.5. The pressurizer safety valves do not comply with PG&E Code Specification 8711, Section 2.2.1 of that code calls out ANS B31.1 for design and fabrication.

The apparent nonconformance of the pressurizer safety valves to PG&E C.S. 8711 indicates noncompliance with 10 CFR 50 Appendix B Sections III, IV and VII.

Staff Findings: Reactor coolant pressure boundary valves were supplied as part of the Nuclear Steam Supply System in accordance with PG&E C.S.

8700, not 8711. There were no class 1 components supplied under C.S. 8711.

10 CFR 50.55 (a)(f)(i) calls out ASA B31.1 or USAS B31.1.0 for Class 1 valves. USAS B31.1 refers to USAS B16.5 for design and fabrication of valves, and the safety valves were designed and fabricated to this standard. The designer used ASME Boiler and Pressure Vessel Code Section III, Article 9 stress criteria only as a basis for establishing operational design stress levels. As defined by Westinghouse letter PG&E-2080, the safety valve bodies were not the pressure containing elements and were not designed to meet the minimum wall thickness requirements of the USAS B16.5 specification. The pressurizer safety valves were supplied under the correct code.

Staff Position

Pullman Power Products Unscheduled Internal Audit number 34 accomplished its intended purpose, turning up a number of inconsistencies and omissions in the documentation of ESD 236. The vast majority of these deficiencies received proper response via the Audit Action Request circuit as a result of the audit.

The staff draws the following conclusions:

1. There were no technical problems that resulted in improper valves being installed in the plant.

2. There was no significant irregularity in the control of work performed on these valve wall thickness measurements.

3. The items of concern generated by the 1982 internal audit were dealt with properly.

4. The allegations which were found to be valid were minor irregularities in paperwork and indicate less than desirable attention for proper documentation on the part of the NDE technicians involved.

The pressurizer safety valves were designed and supplied to the correct specifications and codes.

Action Required:

The licensee has committed to locating the UT instrument calibration certifications pointed out by allegation B6. This action will complete the issue of equipment traceability. The staff will examine this documentation in the conduct of the routine inspection program.

No further action is required.

Task: Allegation or Concern No. 122

ATS No: RV-83-074 BN No:

Characterization:

Some nondestructive evaluation (NDE) procedures used by Pullman were not properly qualified. One procedure was issued after inspections were performed. Pullman did not report difficult problems to PG&E, and the quality assurance (QA) system did not implement adequate or timely corrective actions for audit discrepancies.

Implied Significance to Plant Design, Construction or Operation

Lack of approved NDE procedures or use of nonqualified NDE procedures for inspecting pressure boundary welds, fasteners, or other retaining devices implies unreliable inspection and reduces confidence in the effectiveness of primary pressure boundaries. Inaccurate or untimely corrective actions to quality assurance audits may indicate a breakdown in the quality assurance program.

Assessment of Safety Significance

The alleger is an ex-employee of Pullman Power Products who, in his role as internal auditor in the QA group, performed a quality audit in January 1982, of various NDE work accomplished from 1972-1973. The alleger was concerned that the audit findings were not properly addressed by his employer, in their response in March 1983 that they were not addressed in a timely manner. The findings are itemized as issues in this allegation. The staff's approach to resolving these issues was to examine the Pullman quality assurance audit report provided by the alleger, examine the Pullman response to each audit finding, assess whether the response was adequate, and assess the significance of each audit finding and response. The licensee commented on the allegation by letter on February 29, 1984; and further reported by telcon on March 8 and 9, 1984. These comments were also examined:

A. A Pullman Internal Audit (No. 101) discovered five NDE procedures and two UT thickness measuring procedures that did not have Procedure Qualification Records (PQRs). The NDE procedures were Engineering Specification Diablo (ESD) 234, 241, 246, 247, and 270. The ultrasonic thickness (UT) measurement procedures were ESD 236 and 244. The alleger is concerned that these discrepancies were not adequately or promptly corrected.

<u>Staff findings:</u> The procedure titles are identified as:

- ESD 234 - Ultrasonic Inspection of Groove Welds
- ESD 241 - Ultrasonic Examination of Safety Yoke Rods on 3707RAXG-21 Safety Valves
- ESD 246 - Magnetic Particle Procedure/Dry/Continuous Coil B31.7
- ESD 247 - Magnetic Particle Procedure/Dry/Continuous Coil B31.1

- ESD 270 - Liquid Penetrant Examination Procedures, AWS for pipe rupture restraints

- ESD 236 and ESD 247 - Ultrasonic thickness measurement

The staff finds that there are no technical problems with the corrective action taken by Pullman. However, corrective action was not prompt, and this problem is addressed in item 10.

Procedure qualification tests and reports were required for each of the five NDE procedures, ESD 234. 241, 246, 247 and 270. The UT procedures, ESDs 236 and 244, did not require PQR's as they were not special processes and their procedures were actually qualified each time a test was made. This issue on UT procedures was completely addressed and resolved under item A of allegation number 121.

Pullman responded to the internal audit issues dealing with ESD 234, 246 and 247 by performing procedure qualification tests. ESD 234 PQR is dated October 1, 1982, ESD 246 and 247 PQR's are dated November 9, 1982. In each case procedure qualifications were performed satisfactorily, indicating the procedure was fully capable of detecting specified discrepancies.

Pullman responded to the lack of a PQR for ESD 270 by stating it was similar to ESD 210, which had a PQR, and that ESD 270 would be revised to add the PQR from ESD 210. It has been subsequently determined that ESD 270 was never used.

The details concerning ESD 241 are discussed in item B below.

B. There are a number of related allegations concerning ESD 241 (ultrasonic examination of safety yoke rods on 3707 RAX 6.21 safety valves):

B1. There was no PQR.

B2. The procedure was issued after inspections performed by the procedure were completed.

B3. The threaded portions of the rods were engaged in the mating hole during the test while a Dresser instruction specified that they not be.

B4. The rods were not dye penetrant (PT) testing or MT tested as recommended by Dresser and an internal Kellogg memorandum that was based on the Dresser recommendation.

B5. The Dresser and ESD 241 procedures differed in reference points for sensitivity determination.

B6. The test reports lacked the following information entry surface, calibration block description, calibration method, and calibration frequency.

B7. The test reports did not follow the format referenced in ESD 241.

B8. In addition to these technical issues, it was alleged that Pullman QA/QC did not properly responded to this audit finding. A memo attached to Audit Action Request (AAR) Number 1, dated January 13, 1983, states that ESD 241 is a special procedure to supplement PG&E and manufacturer's examinations. The memo further states that a nonconformance does not exist and a Deficiency Report (DR) is not necessary. "A written response indicating this will be considered basis for closing this portion of AAR." It was alleged that this action constituted a cover up of serious breach in the QA program.

Staff Findings:

This issue does not fall under the purview of the Pullman QA system. Dresser Industries, manufacturer of the steam generator safety valves, requested that PG&E perform UT inspections on yoke rods to assure certain discontinuities were not present. They requested that the tests be performed to Dresser procedure DC-66-3050-27-2. PG&E engineering complied and sent engineering instruction SP-52-166 to initiate the work under the Dresser procedure by Pullman. Pullman performed the work and supplied the data to Dresser.

This explains the lack of an internally generated procedure prior to tests being performed. ESD 241 was subsequently generated to narrow down the Dresser procedure, which was much broader in scope, in case it was ever needed again by Pullman. It was further determined as the procedure was being documented that a PQR was not necessary because the procedure was qualified each time the equipment was calibrated. The standard manufactured for the test contained a sample notch and flat bottomed hole that defined acceptance criteria, and that standard was used for setting system sensitivity and gain at each calibration.

Although the inspector has determined that allegation item B is not a safety concern parts B1 through B8 are addressed for the record:

B1. There was no PQR necessary, the test was qualified for ability to detect the unacceptable discontinuity each time the instument/transducer system was used to test a rod.

B2. ESD 241 was not called out for use in the test, it was merely formalized for future Pullman use after the Dresser initiated testing was completed.

B3. The Dresser instruction was a production test procedure. The valves were tested at DCNP after assembly. The Pullman Level III examiner responsible for the test determined that inspection with fasteners inserted would in no way compromise the accuracy of the inspection.

The staff agrees with this conclusion.

B4. The PT and MT requirements of the Dresser instruction and subsequent Kellogg memo were considered to be trivial compared to the sensitivity of the UT inspection, and therefore not done. This is not a violation of ESD 241.

B5. The allegation is true - ESD 241 required two reference sensitivity points while the Dresser instruction required only one. The difference was in a more conservative direction.

B6&B7 The documentation information and format did not comply with ESD 241 requirements because they were done to the Dresser procedure.

B8 In light of the information presented above, the memo attached to AAR number one can be seen to be valid - there was no nonconformance, no DR was necessary.

The staff concludes that there was no "cover up," nor was there "serious breach" in the QA program.

The staff further considers that the Pullman response for the allegees deficiency report was proper.

C. Only two of the seven ESD's identified as not having PQRs were reported to PG&E by DR for disposition. Pullman did not meet their specification requirements to report all the conditions adverse to quality to PG&E.

Staff Findings: Pullman assessed the findings of the audit and decided only two procedures out of the seven cited constituted possible problems to quality assurance. The inspector agrees that ESD 246 and 247 were the only procedures where a lack of PQR's would have any significant effect on safety or quality, and that Pullman's discussion is reasonable.

D. DR 4662 identified the lack of a PQR for ESD 246 and 247 to PG&E. PG&E accepted all work examined by ESD 247 "as is" but did not accept any work performed under ESD 246.

Staff Findings: There was no work performed under ESD 246. The Level III examiner that developed ESD 247 developed similar ESD 246 simultaneously, but ESD 246 was never used. Thus, there was no need to accept any work performed under this procedure.

E. NDE procedures that had PQRs performed after the inspections were completed did not have approval signatures by Pullman or PG&E management.

Staff Finding: The PQRs were signed-off by a Pullman employee certified to ASNI NDE Level III, but were not approved by Pullman and PG&E management. Subsequent to audit 101, these procedures were reviewed and signed off by required management. Additionally, all NDE records were reviewed for proper approval and signatures. Pullman's response here is adequate.

F. Concerning ESD 247, the MT procedure was written to ASME code B31.1 to inspect feedwater piping designed to ASME Section I.

Staff Finding: There is no discrepancy here. ASME Section I has no requirement for MT of pipe welds (PART PW SECT I, 1967-78 edition). The NDE requirements of ESD 247 thereby exceeded the requirements of the design code.

G. A magnetic particle inspector's certification records were incomplete - the composite MT examination grade was not supported by documentation of the three individual examinations.

Staff finding: Although the qualification certification sheet had spaces for general, specific, and practical exams, the revision of SNT-TC-1A in effect at the time the work was accomplished did not specifically require these individual exam grades. Moveover, an analysis by the inspector of various combinations of individual exam grades using both simple and weighted averages shows that a composite score of 98 percent could not have been achieved if any individual score was below the passing level of 70 percent.

PG&E searched the certification records and found the exams that this MT inspector had taken. His scores were: General-100 percent, Practical 93.3 percent, Specific 100 percent. A weighted average of these score is 98 percent.

H. The procedure qualification tests (PQT) for ESD 246 and 247 were performed using different equipment than was used during the inspection.

Staff Finding: The machine used for the actual test was a KH-15, with a 1000 amp capacity. The PQT was done with a P-90, with only a 700 amp capacity. The equipment used for the after-the-fact PQTs supplied less current than did the original equipment. Since direct current dry MT penetration and sensitivity is a function of amperage, the PQT could not have been more accurate or sensitive than the original test. The Pullman response by Discrepancy Report (DR) 4662 is adequate, the original tests were qualified by these PQTs.

I. The use of seven unqualified procedures puts the work examined into a questionable status. Can NDE procedures be qualified after they are used?

<u>Staff finding:</u> Obviously it would be more desirable to qualify a procedure prior to use. In cases where this was not accomplished, post inspection qualification does validate a procedure. If the post inspection qualification test were to indicate problems, a modified procedure would need to be written, and all applicable inspection would have to be repeated.

J. Internal audit (IA) 101 Audit Action Request (AAR) #1 findings were not corrected promptly. The QA/QC Manager is alleged to have <u>deliberately procrastinated on these findings to avoid identifying problems to management</u> for which he could not formulate proper corrective action. It is further alleged that the Pullman Corporate Director of QA failed to expedite corrective action after he became aware of the delay.

<u>Staff Findings:</u> The inspector agrees that the alleger's concern over the lack of a prompt response to AAR #1 is valid. The chronology of events presented in the allegation is accurate. The QA/QC Manager admits that this was not given appropriate priority; however, he attributes this to each of resources at the time.

The inspector notes that the issue were finally dealt with and closed out in March of 1983 and were handled responsibly.

PG&E has implemented the following corrective action: PG&E has reinstructed the Pullman QA/QC Manager regarding the desirability and necessity for prompt, effective and adequate response to potentially adverse findings. In addition, PG&E has taken action to increase their involvement in the process of supervising their contractor's QA/QC supervision.

The final claim of this statement, that the Pullman Corporate Director of QA failed to expedite corrective action, is not true. The alleger left one document (on which he was copied) out of the statement and exhibits of this allegation. That document was a memo from the Corporate Director of QA to the QA/QC Manager, dated June 24, 1982, reinstructing Pullman QA people to be more timely in response to audit findings.

Staff Position:

1. There are no technical problems which may have resulted in inadequate or inaccurate NDE testing.

2. With the exception of the prompt response to the AAR #1 issue in item J, none of these concerns (Items B through J) represent a significant irregularity in the control of work. The inspection could not confirm that the delay in resolving the audit was an effort to cover it up.

3. The Pullman response to the January 82 audit, completed in March 83, was responsive and appears technically adequate.

<u>Action Required:</u>

PG&E has initiated corrective action that should alleviate the recurrence of issues similar to item 10. The staff will examine this area in the conduct of the routine inspection program.

No further action is required.

Task: Allegation or Concern No. 123

ATS No.: RV-83-074 BN No.:

Characterization

Improper acceptance of welder qualification tests.

Staff Position

The staff considered this concern and observed that the alleger references a specific time period wherein a QC inspector was not present in the Pullman welder qualification area observing the conduct of welder qualification tests. The allegation is very narrow in scope and the staff considers that exhaustive staff examination would have a low potential for yield of any new management significant quality performance issue.

The staff had previously examined the general conduct of the welder qualification program (see NRC Report 50-275/83-37). These examinations likewise failed to yield any new management, quality performance or technical issues.

Action Required

This issue will be turned over to PG&E for resolution. The licensee will be required to provide written response of their findings and any necessary corrective actions.

Task: Allegation or Concern No. 124

ATS No: RV-83-A-0074 BN No: N/A

Characterization

Internal audit procedure, ESD 263, required a 10 day response time to audit findings. This requirement was not being met.

Implied Significance to Plant Design, Construction, or Operations

Represents an apparent lack of management control for the corrective action aspects of the Quality Assurance program.

Assessment of Safety Significance

The staff reviewed Pullman procedure ESD 263 entitled "Internal Auditing Procedure of Field QA Program by Field Staff." This procedure is used by the Pullman internal auditor for auditing of the quality assurance program implementation for erection of piping hanger and rupture restraints. Prior to September 1, 1983, ESD 263, para. 10.1 required that findings be responded to within ten (10) calendar days after receipt of report. On September 1, 1983, ESD 263 was revised to provide an extension of the 10 day requiement if justified and approved by the QA/QC manager or the internal auditor.

The staff also examined the two audits specifically mentioned by the alleger. Audits 32 and 35 violated the 10 day response time requirement, with no evidence of a request for extension in the documentation package. When questioned by the inspector, the QA/QC manager agreed that the 10 day requirement had not been met because the magnitude of the audit findings required a significant amount of research and verification and his current staff size did not allow resolution of this complex audit within 10 days. However, this condition did not appear to result in any safety concerns.

Staff Position

The allegation is true. Prior to September 1, 1983, the majority of audit findings were not responded to within the 10 day requirement.

It is the staffs position that the rigid 10 day requirement in procedure EDS-263 prior to September 1983 was unrealistic and that the contractors management should have recognized and dealt with this impractical requirement. earlier. It, however, does not appear that this condition resulted in any safety concerns or is indicative of a significant quality breakdown.

Action Required

None

Task: Allegation or Concern No. 125

ATS No.: RV-83A-074 BN No.:

Characterization

Pipe rupture restraint welds were not tested per specification.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Action Required

None - This is a restatement of Allegation No. 115

Task: Allegation or Concern No.126

ATS No: RV83A079 BN No:

Characterization

PG&E has not implemented a consistent set of weld symbols for engineers and contractors.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 127

ATS No.: RV-83-A0079 BN No.:

Characterization:

PG&E does not comply with AWS D1.1 regarding weld size and preheat on pipe supports.

Implied Significance to Design, Construction or Operations

Potentially significant to installed hardware.

Assessment of Safety Significance

The safety significance will be discussed in two parts; (a) weld size and (b) preheat requirements for pipe support welding.

a. Weld Size

The alleger expressed concern over the possibility of underbead cracking in the Heat Affect Zone (HAZ) due to rapid cooling of the base metal caused by insufficient weld metal. The weld size being smaller than the minimum fillet weld size specified in AWS D1.1.

On January 16, 1984, the inspector and a staff metallurgist met with the alleger to clarify this allegation. At that time, the alleger indicated that his concern arose from examining AWS D1.1 minimum fillet weld size requirements versus weld sizes called for in the PG&E specifications. Specifically, the alleger's concern resulted from his reading of the AWS Code and not from weld problems observed in the field.

The staff explained that the AWS Code was not required for pipe supports welding. Pullman Specification No. 8711, Section 3.0, paragraph 4.12, instead requires all welding to be performed in accordance with a procedure specification qualified in accordance with ASME Section IX. The inspectors found that the welds being questioned were qualified in accordance with the ASME Section IX and met all requirements. The inspector and alleger were satisfied that this was being done properly.

b. <u>Preheat For Pipe Support Welding</u>

Pullman Procedure ESD No. 223 requires preheat only on structural steel and does not require preheat on pipe support welding.

The allegation was found to be true, however, the staff found that the Procedure Qualification Records (PQR) for the pipe support welding were qualified without preheat, in accordance with ASME Section IX. Therefore, the licensee through the PQR's has demonstrated the weldability of the pipe support welding, without preheat, which is allowed and in accordance code requirements. This was reviewed with the alleger and he appeared satisfied.

Therefore, based on the foregoing items, "a" and "b" are considered to have been done properly and do not constitute violations of requirements or poor welding practices.

Action Required

None.

Task: Allegation or Concern No.128

ATS No: RV83A081 BN No:

Characterization

Pullman did not properly accept problem reports.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 129

ATS No.: RV-83A-081 BN No.:

Characterization

Improper activities related to Pullman Welding.

Staff Position

The staff made a face value assessment of the several specific concerns identified by the alleger and made the judgement that several of these had already been dealt with during the evaluation of other allegations. The staff considered that there was a low potential that these concerns would identify any new management or significant quality performance issues. (Refer to Allegations 103 to 119 and 214 to 217).

Action Required

These specific allegations will be turned over to the licensee for response. The licensee will be required to provide written response of their findings and necessary corrective actions.

Task: Allegation or Concern No.130

ATS No: RV84A009, Q5-84-011 BN No:

Characterization

Pullman - possible intimidation of personnel.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 131

ATS No: RV-84-A-006 BN No.

Characterization:

Pullman Power has welded A325 bolts to containment liner for use in pipe support structures without a qualified Welding Procedure Specification.

The alleger also expressed concerns regarding the welding of A307 bolts without a qualified welding procedure specifiction. This issue is discussed in Allegation 106.

Implied Significance to Design, Construction or Operations

The welding of material not qualified by an appropriate Welding Procedure Specification can result in the weldment not performing as intended. The failure of weldments in safety related areas could affect the reliability of safety related systems.

Assessment of Safety Significance:

To resolve this issue the staff examined the requirements of applicable codes and standards, the circumstances surrounding the welding of the A325 bolts, and had the licensee determine where the A325 bolts were welded during installation. In addition, the staff conducted several discussions with licensee personnel regarding this issue.

The staff determined that the allegation was true; however, the staff also determined that the A325 bolts had been welded to the Containment Fan Cooler support structure and not to the containment liner.

The staff determined the circumstances surrounding the welding of A325 bolts into pipe support structures and based upon discussion with licensee personnel determined the following. The licensee had assigned the responsibility for Component Cooling Water piping and support design, between the containment fan coolers and the first support located away from the fan cooler, to the fan cooler equipment supplier, Westinghouse. The Westinghouse design, provided to PG&E, called for welding A325 bolts to the containment fan cooler support structure for the installation of ten component cooling water piping supports. Westinghouse did not supply a qualified welding procedure with the design.

PG&E considered, in error, that the A325 bolts were compatible with an existing qualified welding procedure specification (no. 7/8) and, accordingly, welded A325 bolts to accomplish the Westinghouse design requirements on eight of the supports.

Thus, it appears that PG&E did not need the bolts using a qualified procedure. This is an apparent item of noncompliance.

The staff further determined that a discrepancy report had recently been written on this subject. In addressing the discrepancy report licensee conducted an extensive program, to establish whether the installed A325 welded bolts were capable of meeting design requirements. First three bolts were welded, using the procedure actually used in the field welding, to a test plate, of the same material as the containment fan cooler support structure. Because the concern with welding high carbon steel involves the potential for hardening and underbead cracking (underbead cracking, if it is going to occur, will usually occur within a few hours after welding) the licensee waited about 1 day prior to (1) loading the test welded bolts to design load conditions and (2) performing liquid penetrant examination on the welds. Both of these tests were completed with satisfactory results. The licensee then torqued all 80 installed A325 bolts to 47 ft. lbs., the design loading requirements. None of the as installed installations failed during this test. Thus the licensee concluded that the methods used for bolt welding resulted in an acceptable installation.

During research on this problem it was determined that a recent code case to the ASME Code (Code Case No. N71-9 (1644-9) of 7 January 1980) prohibits welding on materials such A325 bolts. Although the licensee is not committed to this code for this application (code applicable to PG&E would not have prohibited welding on the bolts), the licensee stated on March 10, 1984 that all 80 welded A325 bolts installed in the plant would be replaced, prior to power ascension even though the installed conditions are considered fully acceptable. The licensee stated that this additionally conservative action was being taken to provide additional assurance of installation conservatism throughout the life of the plant. The staff considers this to be a prudent and responsible decision.

<u>Staff Position</u>

The staff concludes that the licensee had welded A325 bolting material in a safety related installation without adequately using an appropriately qualified welding procedure specification. This is considered to be an item of noncompliance. The staff considers that the licensee has adequately demonstrated that (1) the welding procedure used to weld A325 bolts to the fan cooler supports, although not appropriately qualified a priori, was adequate to assure an acceptable installation, (2) that the installed condition of the welded A325 bolts were capable of meeting design load requirements, and (3) that the installed condition of the A325 bolts had not resulted in a potential construction deficiency. The staff considers that replacement of the A325 bolts, will provide additional assurance of design and installation conservatism throughout the life of the plant.

The staff further concludes that the licensee responded properly to this problem.

Action Required:

The staff will monitor the replacement of the welded A325 bolts, prior to power ascension, in the conduct of the routine inspection program. This issue will be referred to the NRC Vendor Inspection Branch for their use in determining whether Westinghouse has provided similarly discrepant designs to other facilities.

Task: Allegation or Concern No. 132

ATS No. RV-84-A-009 BN No.

Characterization:

Welding was performed on a portion of the Component Cooling Water System with the pipe full of water. Additionally, the alleger expressed concern that a Quality Control hold point had been by-passed during the subject welding.

Implied Significance to Design, Construction, or Operations

Water in the lines could provide an additional heat sink not accounted for by the weld process thereby resulting in weld discontinuities.

Assessment of Safety Significance:

The allegation is separated into two topics:

1. The welding of plates to piping which is full of water, could cause weld discontinuities due to the additional heat sink.

Assessment:

To determine the validity of this allegation, the inspector examined the documents provided by the alleger, the related licensee documents and the welding in question. The inspector also met with the licensee to confirm the inspector's results.

- On February 1, 1984, the inspector confirmed that attachment pads had been welded to the Component Cooling Water System when the pipe was full of water. On February 4, 1984, the licensee provided a basis for the acceptance of welds made on the Component Cooling Water System mentioned above. The licensee contends that because the required minimum preheat temperature was maintained, for the weldability of the materials, welding with the lines full of water does not cause or promote weld cracking, for the specific case described. The only precaution the licensee took was to insure that the welding occurred with the lines full of stagnant water, that is, assuring that the Component Cooling Water Pumps were off during the welding process.

- Examination of a document entitled a "Clearance Request and Job Assignment Sheet," for Unit 1 indicated that the Component Cooling Water Pump "B" was tagged off from July 28 through September 27, 1983. The welding in question was performed on August 13, 1983, on Train "B" of the Component Cooling Water System.

Staff Position

Based on the review of the licensee's position by the inspector and four staff metallurgists, no safety significance is attributed to this allegation. In addition the welding in question did not violate any codes or the welding procedure.

Actions Required

None

2. A Quality Control hold point had been by-passed during the subject welding.

Assessment

To determine the validity of this allegation, the inspector reviewed relevant documents provided by the alleger and the licensee. The inspector's findings are as follows.

Examination of the process sheet for the welding performed on line No. 1-K-104-20, Hanger No. 18-5R indicated that on May 5, 1983, the preparation for the welding of two steel plates was started on the referenced line, but apparently because of fit-up problems, the plates were subsequently removed. In the process of removing the plates a number of grinding gouges were created. On August 8, 1983, a Pullman Quality Control Inspector (the alleger) wrote a Deficient Condition Notice (DCN) DCN No. 1604-006 documenting that for Field Welds Nos. 1414 and 1377, grinding gouges, arc strikes, and a possible linear indication had been observed in the area (where the plates had been removed). The DCN was dispositioned on August 9, 1983, with the recommended disposition, to issue a process sheet, blend the gouges, remove the arc strikes, to perform Liquid Penetrant Testing (P.T.) and Ultrasonic Examination (U.T.) (to check if minimum wall was violated). On August 11, 1983 both the P.T. and the U.T. were performed and found that all areas examined were acceptable and that the minimum wall had not been violated. On August 12, 1983, the location of the plates were re-verified and the plate fit-up signed by the appropriate Pullman Quality Control Inspectors, including the fit-up for FW Nos. 1414 and 1377 which was verified by the alleger.

The issue of concern to the alleger is that the original plates (which were subsequently removed) had been fully welded, rather than just tack welded, prior to being checked by Quality Control for fit-up. The alleger bases this upon the size of the grinding marks left when the plates were removed. The inspector could not determine from such circumstantial evidence whether the grinding marks were of full welds (which would have been a missed Quality Control point) or of tack welds (which would have been proper). In summary, the inspector concludes that the work in question was redone and accomplished in proper order. The inspector could not find any conclusive evidence that any codes or procedures were violated.

Staff Position

In conclusion, no violation of NRC regulations occurred, since the alleged violation was found by Pullman's first line inspection force (the alleger), documented, and properly dispositioned in accordance with the approved quality program.

Actions Required:

None

Task: Allegation or Concern No. 133

ATS No.: RV-84A-0010 BN No.:

Characterization:

Foley did not properly accept/document reports (no specifics were provided).

Staff Position

This concern was received in a group conversation with certain plant workers. No specifics were provided related to the concern and it appears to be a restatement of concerns identified in Allegation 24. The information does not represent any new significant management or technical situation which has not been previously reviewed by the staff.

Action Required

None - Allegation is a restatement of concerns addressed in Allegations 24, 26, 46, 66.

Task: Allegation or Concern No. 134

ATS No.: RV-84A-0011 BN No.:

Characterization

Foley did not invoke Part 21 on vendor contracts.

Implied Significance to Plant Design, Construction, or Operation

The procurement of basic components subject to the regulations of 10 CFR 21 without referencing Part 21 on the purchase order may have resulted in products of inferior quality being supplied by a vendor.

Assessment of Safety Significance

This issue had been identified previously to upper level Foley management. The alleged discrepancy was based on PG&E direction provided to Foley.

The staff found that the allegation was true; however, the Foley procurements involved predominantly commercial grade, off-the-shelf items which are exempt from the requirements of 10 CFR 21. The staff examined about 50 Foley purchase orders, selected at random, and observed that these procured "parts" were exempt from the applicable 10 CFR 21 requirements.

Staff Position

The staff considers that the licensee and Foley had acted responsibly in this area of procurement and finds that PG&E and Foley appeared to comply with regulatory requirements.

Action Required

None

Task: Allegation or Concern No. 135

ATS No.: RV-84A-0011 BN No.:

Characterization

Foley audits were not performed for an extended period.

Assessment of Safety Significance

The staff found that the allegation was true; Foley QA audits had been suspended for a period of about 5 months in mid 1983. However, the staff found that this decision had been made by responsible Foley management, in order to restructure the audit program and minimize the number of restricted area access passes issued by security. PG&E was informed of this problem. The most compelling reason given for the restructure was that additional personnel were needed immediately in the first line quality control function to supplement the existing QC staff in response to the large increase in work activities. The licensee and Foley have since concluded that the decision was not fully appropriate. Based upon the staff's examination of several Foley related allegations and discussions with QC personnel, this audit suspension appears to have had minimal affect on the conduct of the overall quality program.

Staff Position

The staff considers that this action on the part of the licensee and Foley was not well thought out; however the staff could not find any evidence of quality program degradation which occurred as a result of this action.

Action Required

None

Test: Allegation or Concern No. 136

ATS No.: RV-844-40 BN No.:

Characterization

Foley audit findings were not properly handled. (Two examples were provided by the alleger.) The alleged problem seems to be that Foley response to the audit did not really address the finding.

Assessment of Safety Significance

The staff conducted a face value assessment of the two audit findings and concluded that part of the concern was the result of a misinterpretation of the resolutions provided by the QA Director and the allegers apparent distrust of the Foley QA Director's motivations. The inspector concluded that since the two resolutions were written in rather general terms although there was not sufficient evidence to conclude that these were improperly handled. The staff concludes that exhaustive examination of this issue is unlikely to result in any new management or quality performance issues and, thus, the additional expenditure of staff resources is not warranted.

Staff Position

The staff concludes that an audit of the resolutions provided by Foley in response to QA audit findings would be in order to more fully evaluate the acceptability of resolutions provided. This action would provide the assurance needed to resolve this issue.

Action Required

This item will be turned over to PG&E for accomplishment of the above action. The licensee will be required to provide written response of their findings and necessary corrective action.

Task: Allegation or Concern No. 137

ATS No.: RV-84A-40 BN No.:

Characterization

Foley did not audit procedure adequacy.

Staff Position

This allegation is general in nature, but appears to be a restatement of concerns identified and examined in allegation 68. Also, on page 4 of Report 50-275/83-37 dated February 29, 1984, this issue appears to have been addressed. The issue of concern here does not represent a new significiant management or quality performance issue which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 138

ATS No: RV84A0012 BN No.

Characterization

The incore thermocouple circuits were improperly upgraded to class 1 circuits in that wire traceability was lost. This item was discussed in SSER 21 under item 63. Since SSER 21 the staff obtained more information on this item. The conclusions remain the same.

Implied Significance to Plant Design, Construction or Operation

Inadequate cable material, and the improper upgrading of this material, may result in an inability to reliably monitor incore temperatures.

Assessment of Safety Significance

This item was examined through a review of engineering discrepancy report (EDR) No. 8938 (referred to by the alleger), material certificates, procurement records, cable schedules, interviews with personnel, and inspection of cable equipment installations.

The licensee stated that incore thermocouple circuits were upgraded to Class 1. To assure these circuits met Class 1 requirements all the circuits from the reactor head to the control room were removed, properly upgraded to Class 1 and reinstalled in new routings in accordance with quality controls.

The staff reviewed the circuit revisions and inspected the installations and records related to the upgrading of ten incore thermocouple circuits to Class 1 status, as accepted on EDR 8938. The staff concluded that traceability of the circuit pedigrees for the incore thermocouple was available, all records verified that the quality of materials was equivalent to Class 1, and the rerouting of the cables was performed to Class 1 criteria.

The staff noted that in providing acceptance for circuit materials, EDR 8938 failed to identify or refer to the basis for its determination of material acceptability, giving the appearance of the lack of such basis. This lack of information appears to be the source of the concern. The staff's examination of this issue finds that the EDR disposition had basis in documentation of acceptability.

Staff Position

The staff concluded that the thermocouple circuit upgrade is traceable and exhibits the features, design and installation required of Class 1 circuits.

Action Required

None.

Task: Allegation or Concern No. 139

ATS No.: RV-84-A-0013 BN No.:

Characterization

Foley Improperly Performed Tubing Fabrication (Socket Welding and Bending).

Assessment of Safety Significance

The allegers concern primarily involved Foley fabrication of the reactor vessel water level indication system tubing. The staff had previously examined the installation of this system and found nothing of particular significance or concern (See NRC Inspection Report Nos. 50-275/81-04 and 81-10). The staff requested that PG&E address this concern and evaluated PG&Es written response.
The staff's face value assessment indicates that this issue is of minimal safety significance.

Staff Position

The staff's evaluations indicate that this issue would not result in any new significant management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and resolution. The licensee will be required to provide the results of their evaluation, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 140

ATS No.: RV-84-A-0014 BN No.:

Characterization

Foley used material purchased for one contract on another. (No specific examples were provided)

Implied Significance to Plant Design, Construction, or Operation

The staff considers this issue to have mininal safety significance because Foley purchases were primarily restricted to commercial grade, off-the-shelf items.

Staff Position

The staff considers that exhaustive resolution of this issue will not result in any new significant management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation, and any necessary corrective actions, to the staff in writing.

Task: Allgeation or Concern No. 141

ATS No.: RV-84-A-0015 BN No.:

Characterization

Foley performed transverse welding across beams (Installation of Unistrut). (No specifics were provided)

Assessment of Safety Significance

This allegation is extremely vague. The alleger could provide no specific examples. Based upon the depth of examination and the associated findings of other welding related allegations, the staff's face value assessment is that exhaustive examination of this allegation would not result in any new management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 142

ATS No. RV-84-A-0015 BN No.

Characterization

Foley inadequately installed and checked anchor bolts.

Implied Significance to Plant Design, Construction or Operation

Improper installation of anchor bolts could result in reduced load capacity of the anchor bolts with attendant loss of design function during normal operation or design basis events, including seismic events.

Assessment of Safety Significance

See Task or Allegation No. 25.

Staff Position

See Task or Allegation No. 25.

Action Required

See Task or Allegation No. 25.

Task: Allegation or Concern No. 143

ATS No.: RV-84-A-0015 BN No.:

Characterization

Foley did not torque beam clips at installation.

Assessment of Safety Significance

Lawarence Livermore National Laboratory inspectors, under contract to the NRC, have examined the tightness of beam clips bolts and have found no evidence that these have not been torqued. The staff considers that exhaustive examination of this allegation would not result in any new significant management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluations, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 144

ATS No.: RV-84-A-0015 BN No.

Characterization

Foley installs P110 conduit clamps too close to channel edges and they may slip out.

Implied Significance to Plant Design, Construction, or Operation

Disengagement of the conduit clamp would result in a conduit being not supported as required by design criteria and may invalidate the assumptions of the seismic analysis.

Assessment of Safety Significance

The staff was aware of this concern. During plant tours conducted in the examination of other allegations, the "staff" examined the installed condition of over 100 P110 conduit clamps. This examination did not identify any instances of obvious concern for the clamp slipping out of the channel. Thus the staff's face value assessment does not indicate that this issue would result in any new significant management or quality performance issue.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 145

ATS No.: RV-84-A-0015 BN No.:

Characterization

Foley did not specify raceway materials in details - improper bolt heads may have been used. (No specifics were provided)

Assessment of Safety Significance

The alleger did not identify specific discrepancies, thus the allegation is vague. The staff had examined raceway installations in the past without identifying any significant potential for a material causing damage to a cable. Thus, the staff's face value assessment indicates exhaustive examination of this issue would not be likely to result in any new significant management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 146

ATS No.: RV-84-A-0015 BN No.:

Characterization

Foley does not keep raceways free of damaging debris. (No specifics were provided)

Assessment of Safety Significance

The staff was aware of this concern and during plant tours conducted to examine other allegations, observed about 50 cable tray installations containing safety related cables to ascertain whether these contained damaging debris. No such instances were identified; all cable trays observed with safety related cable appeared clean and free of damaging debris. The general nature of the expressed concern and the inspector's evaluations do not indicate that there is any substance or significance to this concern. The inspector concludes that the concern is not valid.

Action Required

None

Allegation or Concern Number: 147

ATS No. RV 84 A 0015

Characterization

Cable tray and conduits of independent and redundant trains were installed on common raceway supports. (No specific examples were given by the allegers).

Implied Significance to Plant Design, Construction, or Operation

The required independence of circuits that are essential to emergency reactor shutdown, containment and reactor heat removal, or otherwise essential in preventing significant release of radioactive materials to the environment is compromised by the possibility of common failure through the common support.

Assessment of Safety Significance

This issue was reviewed by; (1) by review of NRC regulatory requirements and industry standards within the topic area, (2) examination of approved licensee commitments as stated in the FSAR, and (3) extensive field inspections of Class I raceway supports to determine if common supports had in fact been utilized.

There is no generic requirement by the NRC to install redundant circuits on separate supports. Indeed, most facilities, even those most recently licensed, such as Washington Nuclear Project Number 2 (WNP 2), feature common supports of redundant safety related electrical divisions. The requirements are that the redundant safety related electrical circuits must be (1) electrically independent of each other and (2) physically separated from each other in order to preclude in the first case common electrical failures that would render both circuits inoperable or in the second case that common harms such as fire or missile hazards would affect both circuits. With respect to supports, if the support is seismically designed to withstand the design basis earthquake with its total load imposed, it is acceptable.

The adequacy of the tray supporting system is reviewed with respect to the ability to perform the intended safety function under the postulated seismic event. This review of safety related raceway supporting systems does not require inclusion of the independence criteria. The NRC position is expressed in REGULATORY GUIDE 1.29 which requires that safety related electrical systems have supports that are designed to withstand the effects of the safe shutdown earthquake and remain functional. There is no mention in this REGULATORY GUIDE of any requirement to provide independent supports.

The adequacy of safety related electrical systems with respect to electrical independence and physical separation is defined in IEEE 308 and IEEE 384 (REGULATORY GUIDE 1.75). These standards state the requirements for physical separation of redundant circuits in terms of distance or barriers, but remain silent as to any requirements of the raceway supporting system.

The specific separations of IEEE 384 (REGULATORY GUIDE 1.75) were not imposed upon the licensee because the licensee's proposed methods as stated in the FSAR Amendment 24 were found acceptable by the NRC Staff. (See Supplement No.

1 to the Safety Evaluation of the Diablo Canyon Nuclear Power Station Units 1 and 2) dated 31 January, 1975. The specific requirements are stated in FSAR Section 8.3.3 "Analysis of A-C Power Systems", "Separation Criteria for Class I Systems" on page 8.3-19.

On page 8.3-28 of the FSAR under the title "Supports" is a reference to section 3.10 for the seismic design and a statement that "Class I supports are not shared by mutually redundant Class I circuits".

Therefore, the inspector concludes that although there is no firm regulatory requirement to support different divisions on separate support systems, the licensee added this commitment to the FSAR to provide additional conservatism.

An NRC inspector conducted extensive examinations of Unit 1 areas containing large concentrations of safety related electrical cable raceway to determine whether the alleged condition existed. The inspector observed that several raceway supports in the cable spreading room supported conduit of redundant Class I divisions.

At the inspector's request the licensee evaluated this situation. The licensee stated by letter (DCL-84-064) dated February 17, 1984, that "supports in the cable spreading room under the control room and the K area, elevation 100'" were exceptions to the design approach of assuring that mutually redundant Class I conduits and trays were not supported by shared support systems. This response from the licensee also stated that Section 8.3.1 of the FSAR was in process of being updated to reflect this plant condition. The licensee's response however did not address the degree of compliance with the

FSAR commitment and the engineering justifictaion for failure to implement the FSAR Commitment. The licensee supplied additional information related to this issue by Letter No. DCL-84-092, dated March 7, 1984. The enclosure to this Letter states that "The FSAR statement that Class I supports are not shared by mutually redundant circuits was a design conservatism established by PG&E; however, deviation from this design standard was found to be required to show seismic qualification of raceways to the revised seismic spectra generated during the Diablo Canyon Phase 1 Verification Program. Prior to acceptance of this design standard change, reviews were performed which showed that no regulatory requirements, including those stated earlier, were impacted. The design of supports has sufficient margin to assure that loss of a single support will not cause loss of safety function. As stated in the previous submittal on this issue, an FSAR change will be submitted to clarify Page 8.3-28." Thus, it appears that the licensee had evaluated this change in design criteria, for compliance with regulatory requirements, with the result that the deviation from the additional conservatism, previously committed to in the FSAR, was justified based on analysis of regulatory requirements and industry standards. Furthermore, the licensee's engineering had brought this issue to the attention of the organization responsible for submitting requests for amendment of the FSAR. Although an amendment request had not yet been submitted this item was scheduled for inclusion in an amendment request. Therefore, the staff feels that the licensee acted in responsible manner as regards this situation; however, a more timely action to resolve the FSAR discrepancy would have been desirable. The staff feels that this situation does not represent a breakdown in the design process.

The failure to comply with the above referenced FSAR commitment is considered to be a Deviation.

Staff Position

Inspection of Unit 1 cable spreading room area indicated that the licensee did not comply with the provisions of the FSAR with respect to independence of supports for redundant safety related circuits. This represents a Deviation from an FSAR commitment.

Action Required

The matter of acceptability of the installed supports will be referred to the Office of Nuclear Reactor Regulation for use in their evaluation of the FSAR change, which PG&E will submit. No further regional action is anticipated.

Task: Allegation or Concern No. 148

ATS No.: RV 84A015 BN No.:

Characterization

Foley Q.C. identifying unsatisfactory work in progress were told to wait until completion, then reject. (No specific examples were provided by the alleger)

Assessment of Safety Significance

The staff considers that this concern, albeit excessively costly if implemented, would not necessarily result in an unacceptable final product. The PG&E and Foley philosophy regarding in process inspection has been that a hold point is assigned on a work process sheet if an inspection, critical to final quality and unobservable after work completion, is necessary and required by procedure. Based upon the staff's knowledge of past practices and philosophy in this area and the vague nature of the allegation, the staff considers that exhaustive evaluation of this issue would not likely result in any new management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluations, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 149

ATS No.: RV 84A016 BN No.:

Characterization

Foley did not submit HVAC as-built information during 1981/82; as-built may not be checked against design.

Implied Significance to Plant Design, Construction, or Operation

If true this concern may result in instances where the HVAC system or supports may not perform as intended by the designer.

Assessment of Safety Significance

The staff requested that the licensee conduct an evaluation of this concern. The licensee found that the installed condition of the duct work conformed to design. This was further reinforced based upon satisfactory completion of flow balance and pressure differential testing. The licensee stated that the as-built conditions of support structures was in the process of evaluation. Therefore, the staff feels that further evaluation of this concern would not likely result in any new management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation, and any necesssary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 150

ATS No.: RV 84A016 BN No.:

Characterization

Foley production may have falsified structural steel and tubing heat number records. (No specific examples were provided)

Implied Significance to Plant Design, Construction, or Operation

The staff's face value assessment is that this concern involves only minimal safety significance.

Assessment of Safety Significance

The alleger indicated that he knew of no specific examples of such falsification but stated that Foley production was in the process of assuring that quality documentation was in order. The alleger was told that one item being resolved by Foley production was in the area of steel and tubing traceability to material certifications and that in this process several instances required that a QC inspector inspect material in the field to verify that a material heat number was stenciled onto the installed piece. The alleger had heard that, if the material was not so stenciled, production would research the records and select a traceability number based upon material type, shape and time of issue. Thus, the alleger concluded that there was a possibility that traceability documentation of installed materials could be falsified.

The staff considers that, even if true, this concern involves only minimum safety significance because Foley structure steel was purchased as an off-the-shelf, commercial grade material which was supplied with, and receipt inspected for evidence of proper material physical and chemical properties. Stainless steel tubing is mainly 3/8 inch material which is similarly receipt inspected and supplied with evidence of conformance with specified chemical and physical properties and hydrostatically tested following installation.

Thus, the staff considers that exhaustive evaluation of this concern would not likely result in any new significant management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 151

ATS No.: RV 84A017 BN No.:

Characterization

(1) Foley installs too many conduits on supports; (2) inspection reject rate is too high for supports. (No specifics were provided)

Implied Significance to Plant Design, Construction, or Operation

The staff's face value assessment of this issue is that it constitutes minimal safety significance.

Assessment of Safety Significance

The staff's review determined: (1) The licensee has specified definitive design and installation criteria for the maximum number and size of conduits that may be installed on a particular support, and (2) this allegation is vague, with no specific examples provided. The alleger did not provide any documentation, conduit support locations, or other information to support this allegation. The staff and NRC consultants (Lawerence Livermore Laboratory) have examined several hundred conduit supports in the past without identifying any significant problems.

Staff Position

The staff's evaluations indicate that this issue would not result in any new significant management or quality performance issues.

Action Required

This item will be turned over to the licensee for evaluation and response. The licensee will be required to provide the results of their evaluation and any necessary corrective actions to the staff in writing.

Task: Allegation or Concern No. 152

ATS No.: RV 84A017 BN No.:

Characterization

Concerns with installation of P1331 conduit clamps (torque achievement, relocation, excess).

Staff Position

The alleger stated in a group meeting on January 17, 1984 that P1331, a 90° clamp for raceway supports is required to be torqued to 85 ft. lbs which: (1) cannot be achieved for the inner bolts, and (2) relaxes after several days, and (3) appears excessive.

The staff's face valve assessment of this issue indicates that there is not a major significant problem in terms of public health and safety or management breakdown. Also, clamp issues in general are known issues that have been responsibly handled.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 153

ATS No.: RV 84A017 BN No.:

Characterization

Foley specifies 1/8" welds on 3/32 clamp material.

Implied Significance to Plant Design, Construction, or Operation

The staff's initial assessment indicates that this issue is of minimal safety significance.

Assessment of Safety Significance

The alleger's concern is that an oversize weld is being specified (i.e. 4/32" (1/8) to 3/32 clamp material). The staff had previously examined welding in this area (uni-strut/superstrut) and found no significant problems.

The staff's evaluation indicate that this issue would not result in any new significant management or quality performance issue.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation and the necessary corrective action to the staff in writing.

Task: Allegation or Concern No. 154

ATS No: 'RV-84-A-0017 BN No:

Characterization

Foley does not specify adequate inspection criteria for anchor bolts.

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 Allegtion or Concern No. 25

Task: Allegation or Concern No. 155

ATS No.: RV 84A018 BN No.:

Characterization

Welding on embed plates causes distortion, may damage plate or anchors.

Implied Significance to Plant Design, Construction, or Operation

Embed can sustain sufficient damage so that the anchoring capacity of the studs will be less than designed.

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in the past. The issue of concern here does not appear to represent a new significant management or technical situation which has not been previously addressed. Similar issues were discussed in the time frame of March 1979 when IE Bulletin 79-02 was issued.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to thier findings and corrective actions.

Task: Allegation or Concern No.156

ATS No: RV84A019 BN No:

Characterization

Foley - possible intimidation of personnel.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No.157

ATS No: RV84A020, Q5-84-013 BN No:

Characterization

Pullman - possible intimidation of personnel.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation No. 158

ATS No.: RV-A-84-0001 BN No.

Characterization:

Unit 2 annulus design-inadequate seismic load combinations.

Implied Significance to Plant Design, Construction or Operation

Inadequate combination of orthogonal earthquake components, could underestimate seismic loads in the pipe supports and supporting annulus structural steel.

Assessment of Safety Significance

This allegation is currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 29, 1984. The staff is formalizing its position using the audit findings.

Action Required

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation No. 159

ATS No.: RV-A-84-0001 BNL No.:

Characterization:

Unit 2 annulus design-steel members may be over stressed due to additions.

Implied Significance to Plant Design, Construction or Operation

Members added to the original annulus frames for piping supports may be overstressed due to the members not being reviewed by the civil engineering group.

Assessment of Safety Significance

This allegation is currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 29, 1984. The staff is formalizing its position using the audit findings.

Action Required

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation No. 160

ATS No.: RV-A-84-0001 BN No.:

Characterization

Unit 2 annulus design-bracings carry axial loads and supports.

Implied Significance to Plant Design, Construction or Operation

Bracing added to the annulus steel frames carry piping loads as well as the framing bracing loads. The loads could overstress the bracing that is intended to carry only axial loads.

Assessment of Safety Significance

This allegation if currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 29, 1984. The staff is formalizing its position using the audit findings.

Action Required

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation No. 161

ATS No.: RV-A-84-0001 BN No.:

Characterization

Unit 2 annulus design - too many assumptions of Class II and small bore loads.

Implied Significance to Plant Design, Construction or Operation

The load used for the Class II and small bore pipe supports were assumed rather than obtained from the piping analyses. Additionally the support configurations were not adequately considered and thermal effects were neglected.

Assessment of Safety Significance

This allegation is currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 29, 1984. The staff is formalizing its position using the audit findings.

<u>Action Required</u>

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation No. 162

ATS No.: RV-A-84-0001 **BN No.:**

Characterization

Unit 2 annulus design-calculations changed by reviewer without consultation with originator/checker.

Implied Significance to Plant Design, Construction or Operation

Calculations were changed by reviewers without consultation with originator/checker. This practice could lead to unsafe structure depending on the nature of the modifications.

Assessment of Safety Significance

This allegation is currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 20, 1984. The staff is formalizing its position using the audit findings.

Action Required

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation No. 163

ATS No.: RV-A-84-0002 BN No.:

Characterization

Unit 2 annulus design-improper assumptions related to thermal expansion.

Implied Significance to Plant Design, Construction or Operation

The structural steel of the annulus frames is anchored on one end to the concrete crane wall. The stress in the annulus structural steel is effected by the thermal expansion of the steel as well as the concrete. If an improper evaluation is made of the thermal effects the stresses in the steel, it would not be properly predicted by the analysis. This may lead to overstressing the steel as well as the concrete members.

Assessment of Safety Significance

This allegation is currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 29, 1984. The staff is formalizing its position using the audit findings.

Action Required

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation No. 164

ATS No.: RV-A-84-0002 BN No.:

Characterization

Unit 2 annulus design-beams not checked for tearing failure mode.

Implied Significance to Plant Design, Construction or Operation

Beams that have the flanges open to facilitate the framing of one member to another can experience a failure of the web at the joint thru the bolt holes. The AISC limits this shear stress to such a value that this failure mode will not occur. If the stresses are in exceedance of the code allowables than a shear type failure could occur imposing additional loads on adjacent structures, systems and components.

Assessment of Safety Significance

This allegation is currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 29, 1984. The staff is formalizing its position using the audit findings.

Action Required

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation No. 165

ATS No.: RV-A-84-0002 BN No.:

Characterization

Unit 2 annulus design-computer code check did not account for torsional stresses.

Implied Significance to Plant Design, Construction or Operation

The annulus structural steel analysis did not directly account for torsional stresses in the members due to off center loads. Members were checked for torsional stresses if the non torsional stress level was 60% or greater of the code allowables. If the torsional stresses in the non checked members were large enough the member could be stressed beyond the code allowables.

Assessment of Safety Significance

This allegation is currently under review by the staff.

Staff Position

The staff audited the licensee's calculation books on February 28 and 29, 1984. The staff is formalizing its position using the audit findings.

Action Required

The staff is formalizing any required action necessary to resolve the allegation.

Task: Allegation or Concern No. 166

ATS No.: RV-84-A-0021 BN No.:

Characterization:

H. P. Foley Quality Control personnel are changing quality control documentation using general guidelines with no overall controls provided for problem documentation, review of changes by management, or management review of corrective actions.

Implied Significance to Plant Design, Construction or Operation:

Discrepant records may have been revised to reflect inspections and/or inspection criteria compliance, necessary to assure the verification of installation quality, which may not have been accomplished.

Assessment of Safety Significance:

This concern involves a program wherein H. P. Foley document analysts and QC personnel are performing a re-review of quality records generated since September 1981. The individual is apparently concerned that the reviewers can correct errors they find without subjecting these corrections to a management review. In addition, the individual is apparently concerned that the controls covering the error correction process do not provide sufficient specific guidance regarding the documentation of changes made.

The staff evaluation of this concern consisted of: (1) a review the procedures and criteria associated with the document review and revision process; (2) interviews with responsible licensee and contractor personnel; and (3) an examination of a sample of documentation packages to determine the types of record revisions which had been made and to evaluate if the revision made reflected the accomplishment of an activity which may not have been performed.

The staff examined the procedures and criteria associated with the document review and revision process. The records correction process in late July or early August 1983 and was performed qualified QC inspectors using sufficient basis for making the corrections. Changes were initialled and dated. Concurrently, H. P. Foley Quality Instruction No. 4 (titled "Records Correction") was developed and issued, dated August 18, 1983, establishing the methods and actions required to alter, change, correct or modify quality documentation. Additional guidance is provided by a Foley Inter-Office Memorandum, dated October 6, 1983. These documents provide the Foley Quality Control personnel, engaged in the review and revision of quality documents, with the guidelines and authority necessary to correct obvious discrepancies noted during the document review process. Both documents provide requirements to assure that: the changes made are justified and do not mask the accept/reject status of the item; and the reason for changes made, which if not obvious, shall be indicated on the document or a separate attachment. The above guidelines provided to H. P. Foley personnel performing the document turnover reviews require that items identified as discrepant be documented on inspection reports, nonconformance reports or document deficiency notices (DDN's).

The staff found that neither of the above documents specifically provide for management review of, and concurrence with, each document revision or change; however, recurring deficiencies in the quality documentation have been brought to the attention of management for resolution in the form of inspection reports, nonconformance reports and document deficiency notices (DDNs). The staff found that management had acted responsibly in evaluating and resolving these issues.

As an improvement in the review process, the licensee committed, in their letter No. DCL-84-080, dated February 29, 1984, to specifically provide for management reviews of document changes. The staff conducted discussions with responsible licensee and contractor personnel engaged in the document review and turnover process. These discussions indicated that, while senior level personnel were generally familiar with the total program for review and turnover of quality documents, the document review analysts did not have complete visibility of the total program and, thus, had reservations regarding the adequacy of resolutions provided in response to Document Deficiency Notices which they had written. These reservations seemed to be largely due to the lack of a complete explanation by the organization assigned to resolve the problem on certain document deficiency notices as to why the situations were resolved in the manner indicated on the forms. Therefore, when the completed package was returned to the document analysts they could not be sure the resolution was proper.

To assure that resolutions to identified problems are more adequately documented, the licensee committed, in their letter No. DCL-84-080, dated February 29, 1984, to revise Foley instructions to more adequately specify and provide for approval levels and documentation required for changes or corrections to quality records (this includes resolutions to deficiency notices). In the course of discussions with document turnover analysts, several examples of allegedly defective resolutions to document deficiency notices were provided to the staff. In order to resolve these concerns the staff examined the resolutions to 139 document deficiency notices (11 generic, 57 electrical, 40 mechanical and 31 civil) and examined 29 related purchase orders and 47 related file packages. In general, the staff was able to verify that the stated resolution was adequate. However, in 5 cases the resolutions provided did not appear to be justified. In two cases, Foley engineering had erred with the result that, in each case, the wire installed in a Class 1 circuit was not traceable to an accepted wire spool (Nonconformance reports were written documenting these discrepancies). One case involved a mere paperwork error with no effect on the installed circuits. In two cases, Foley engineering provided a response which could only be accepted if an actual field verification had been performed; however, there was no indication a field verification had been done. Subsequent reinspection by Foley verified that the two circuits were properly installed. Therefore, of the 139 deficiency notice resolutions reviewed only two DDNs were improperly dispositioned. PG&E elected to replace these cables rather than conduct a time consuming search of the record files to document the acceptability of these cables. In the inspector's opinion there is a high probability that these two instances merely represent a failure to record the proper wire spool number on the wire pull card.

The actual cables installed were a "color coded" cable. PG&E had originally purchased color coded cable to the requirements of applicable IEEE standards and did not accept this cable until documentation of conformance was supplied. Thus, cable reels released for installation in Class 1 circuits had assurance of conformance. In the process of circuit installation (cable pulling) the installing crew was required to log the cable reel number and traceability number on the pull card and have QC verification, in this manner providing traceability back to documentation of conformance supplied by the manufacturer. Therefore, because of the inspector's past knowledge of PG&E practice, and having examined the traceability of several circuits over the years, the staff has no real concern regarding whether or not the originally installed cable complied with quality class 1 material requirements.

The staff concludes that the overall controls, provided to quality control and document analysts, generally provided for adequate record discrepancy documentation. These controls could have been made more comprehensive and effective by specifically providing for management review of changes and management review of corrective actions. The staff found that, although not specifically required, management was involved in the document review and discrepancy correction process as evidenced by management's involvement in the review of recurring inspection reports, nonconformance reports and document deficiency notices.

The staff further concludes that the overall document review and discrepancy resolution program did result in an acceptable level of document review and discrepancy resolution, even though document analysts were apparently confused. This conclusion has basis in the results of the staff's examination of 139 of the more troublesome DDN resolutions, as detailed above. No real hardware problems were found.

Thus, the staff concludes that the identified concern, while true, is of only minimal importance and safety significance.

Staff Position

The staff considers that the general guidance provided personnel reviewing quality documentation, in preparation for turnover, to control the revision or changing of those quality documents was generally adequate. While the guidance did not specifically provide for management review and approval of each change, the staff finds that management was involved in the resolution of generic types of document changes and that management had generally provided controls over the types of changes which may be made and the documentation necessary to provide the justification for the change. With the further clarification of document change approval levels, the clarification of documentation required for quality record changes or corrections, and the increased training of quality control and document analysts, committed in the February 29, 1984 PG&E letter, the staff feels that the licensee's document review and turnover process will be further strengthened.

Action Required

None.

Task: Allegation or Concern No. 167

ATS No: RV-84-A-021 BN No:

Characterization

Foley is not reviewing all records in preparation for turnover; only post September 1981 records.

Implied Significance to Plant Design, Construction, or Operation

See Task Allegation or Concern No. 65

Assessment of Safety Significance

See Task Allegation or Concern No. 65

Staff Position

See Task Allegation or Concern No. 65

Action Required

See Task Allegation or Concern No. 65

Task: Allegation or Concern No. 168

ATS No.: RV-84-A-0022 BN No.:

Characterization

Foley did not properly grout base plate anchor bolts.

Implied Significance to Plant Design, Construction, or Operation

The staff's face value assessment is that this concern is of minimal safety significance and even if true would not seriously degrade the operability of the diesel fuel oil transfer system.

Assessment of Safety Significance

The alleger specifically referred to support No. 20/85R in the diesel generator fuel oil vault of Unit 1. Specifically, the alleger referred to an instance where a U-Bolt hole had been drilled through a weld attaching a shim plate to the support. Also, he stated that one of four anchor bolts in a baseplate had allegedly been improperly grouted, as evidenced by an excessively large amount of grout which had leaked out of the grout cap onto the surrounding floor area. Thus, the alleger concluded that the anchor bolt hole was not properly filled with grout. The alleger states that the first condition was wrongly accepted by field engineering and that Foley improperly accepted the anchor bolt grouting.

The staff considers that extensive evaluation of this concern is not likely to result in any significant new management or quality performance issues.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluations, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No. 169

ATS No.: RV-84-A-0022 BN No.:

Characterization

Pullman failed to conduct support welds as required by procedures.

Assessment of Safety Significance

The alleger stated that two W14X90 wide flange beams were welded together on support Nos. 2/45R and 2/49R (on the diesel generator exhaust system) by use of an unqualified welding technique. Specifically, the alleger stated that Pullman welding procedure specification (WPS) 7/8 was used to join the steel shapes without the use of the procedure required backing bar; in place of which a back-gouging was performed, contrary to the qualified technique. The alleger further stated that the Pullman QA/QC Manager wrongly approved the technique utilized.

Staff Position

Because welding related allegations had been extensively examined by the staff, an exhaustive examination of these two specifics would in the staffs opinion, add little to the management or quality performance issue.

Action Required

This item will be turned over to PG&E for evaluation and response. The licensee will be required to provide the results of their evaluation, and any necessary corrective actions, to the staff in writing.

Task: Allegation or Concern No.170

ATS No: RV84A022 BN No:

Characterization

Pullman lost pipe traceability due to inadequate training of fab shop inspectors.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 171

AITS No. RV-84-A0007 BN No. N/A

Characterization:

Inadequate planning and routing of cables within the plant giving rise to a potential for inadequate separation of redundant safety-related cables and loss of traceability.

Implied Significance to Plant Design, Construction or Operation

See statement below.

Assessment of Safety Significance

This concern is addressed in Allegation or Concern Nos. 54, 59 and 63 of SSER-21, including supplements thereto.

Staff Position

See Allegation or Concerns referenced above.

Task: Allegation or Concern No. 172

AITS No.: RV-84-A0007 BN No.: N/A

Characterization:

The transfer of cable to alternate reels - short sections of cable were frequently transferred from their original reel to other reels of cable as a convenience resulting in confusion regarding specific documentation of cable characteristics.

Implied Significance to Plant Design, Construction or Operation

See statement below.

Assessment of Safety Significance

This concern was addressed in Allegation or Concerns 54 and 59 of SSER-21.

Staff Position

See Allegation or Concerns referenced above.

Task: Allegation or Concern No. 173

ATS No: RV-84-A-0007 BN No: N/A

Characterization:

Improper clearing of cable ways before pulling of cables. Failure to adequately clear the cable ways have resulted in damage to cables when they were pulled through the cable ways.

Implied Significance to Plant Design, Construction or Operation

Pulling of cables into cable ways which have defective conditions, such as sharp edges on conduit or junction boxes, could result in unacceptable damage to the cable being pulled.

Assessment of Safety Significance

During an inspection at the site on January 31 through February 8, 1984, this concern was pursued by the staff. The staff's examination of QC records of the H. P. Foley Company, the electrical installation contractor, revealed that there were instances when cables were pulled into raceways prior to QC inspection and clearance of the raceways. These instances were documented in nonconformance reports (NCR's) by the contractor's QC department.

Dispositioning of the NCR's required thorough inspection of the raceways to determine their acceptability and that conditions did not exist which would have resulted in damage to the cables which were installed. An example was NCR No. 8802-975, dated December 12, 1983. In this instance cable had been repulled into a conduit following modifications to the conduit. The modified conduit was inspected on January 20, 1984 and found to be acceptable in terms of size, type, identification, support placement, installation detail and workmanship. Based upon the results of the raceway inspection the cable installation was accepted as installed. The quality records also include the results of satisfactory post-installation continuity and megger testing of the cables installed.

Staff Position

The licensee contractor disposition of the nonconforming conditions identified appears acceptable.

Action Required

None.

Task: Allegation or Concern No. 174

ATS No: RV-84-A-0007 BN No: N/A

Characterization:

Inadequate control of tension levels when pulling cables - inadequate control was exercised in pulling electrical cable through cable ways and could have resulted in damage to cables during installation.

Implied Significance to Plant Design, Construction or Operation

Inadequate control of pulling tension during the installation of electrical cables could result in unacceptable damage to the cables.

Assessment of Safety Significance

The allegation in this instance was not accompanied by detailed supporting information. However, during the staff's examination into other allegations at the Diablo Canyon site information of a related nature was obtained as follows.

The staff's review of H. P. Foley quality records revealed a condition identified by the QA department's review of quality records wherein it had not been documented that pulling tension had been measured directly by QC inspection as required. These instances, which involved the pulling of five circuits, were the subject of a nonconforming report (NCR No. 8802-1027) dated

January 19, 1984. All cable pulls in this case involved high temperature resistance (HTR) cables with "soft" jacket material - thus the requirement for direct measurement of pulling tension. The disposition of the NCR in this instance was "accept-as-is," based upon successful post installation electrical continuity and resistance tests and the fact that all pulls were made by hand.

Staff Position

The staff concludes that there were instances when QC inspections were not conducted in accordance with QA/QC program requirements regarding QC monitoring and witnessing of special cable pulls. These conditions were documented by the H. P. Foley QC department and acceptable dispositions were made regarding the nonconforming conditions identified.

Action Required

None.

Task: Allegation or Concern No. 175

ATS No. RV-84-A-007 BN No.

Characterization:

Changes from Interim "As Built" Drawings to Final Drawing - Inadequate Control has been exercised over the transition from Interim Drawings to Final Drawings of the station as actually constructed. No specifics were provided.

Assessment of Safety Significance

A face value assessment on the part of the staff indicates this issue is not of major significance in terms of public health and safety or management breakdown. Also, this issue appears to be a restatement of concerns identified and examined in allegation 61. The issue is a known issue and is being responsibly handled.

Staff Position

The issues of concern here do not appear to represent any new significant management or quality performance issues which have not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 176

ATS No: RV-84-A-007 BN No:

Characterization

Anchor Bolts (torquing of "Red-Head Bolts).

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 177

ATN No. None BN No.: None

Characterization

The allegation relates to the RHR pump common suction line valve control and a potential damage to RHR pumps due to loss of suction as a result of a single failure.

Related Allegations: 37, 39, 40, 45 (previously discussed in SSER 21)

Implied Significance to Plant Design, Construction or Operation

The RHR suction line from the RCS hot leg in the Diablo Canyon design contains two isolation valves (8701 and 8702) in series that are normally closed during power operation and hot standby condition (Modes 1, 2 and 3) The RHR suction line from the RCS hot leg is only used during Mode 4 (hot shut-down with RCS cold leg temperature less than 323 °F), Mode 5(cold shutdown) and Mode 6 (refueling). A postulated inadvertent closure of either isolation valve (8701 or 8702) in the RHR suction line during plant shutdown could cause potential damage to both RHR pumps.

Assessment of Safety Significance

This allegation overlaps concerns previously expressed in Allegations 40 and 45 which have been addressed by the staff in Diablo Canyon SSER No. 21. This concern also has been discussed by the staff at an ACRS meeting on February 10, 1984.

The potential damage of both RHR pumps due to loss of suction as a result of a single failure is prevented by the following provisions:

1. In response to the staff requirement in SSER 21 regarding Allegation 45, PG&E has committed, in a letter dated February 15, 1984, to install the RHR low flow alarm prior to entry into power operation (i.e. Mode 1 with associated decay heat generation). The low flow alarm will be set so that sufficient time would be available to alert the operators to trip the RHR pumps before pump damage occurs.

2. The current Technical Specifications and operating procedures for Diablo Canyon Unit 1 preclude the inadvertent closure of either of the two RHR pump suction line isolation valves (8701 and 8702) by maintaining the valves in an open position with power removed for the valve operators during Modes 4, 5 and 6.

The applicant stated at the ACRS meeting on February 10, 1984 that RHR pump damage could occur in 10 to 15 minutes following loss of suction flow. Operating experience from the Calvert Cliffs Nuclear Power Plant showed that the RHR pump seals were damaged approximately 15 minutes after loss of suction flow. The failure of both RHR pumps is an event beyond the design basis and its occurence is highly unlikely based on the plant specific design and administrative controls discussed above. However, if failure of both RHR pumps should occur during plant shutdown, the following steps could be taken to maintain a safe shutdown condition:

1. If both RHR pumps failed during the period when the decay heat level is still relatively high, then the plant conditions would permit decay heat to be removed by the steam generator(s). Condensate supplied from the condensate storage tank, raw water reservior, and the auxiliary salt water system (unlimited supply) via temporary connections could provide a long term source of auxiliary feedwater for decay heat removal.

2. If the steam generator(s) were not available, and the decay heat is relatively low, one RHR pump is generally used to remove decay heat with one pump in standby, in accordance with the requirements of Technical Specifications 3.9.8.2. In case the operating RHR pump is damaged due to closure of a suction valve, the standby RHR pump could be used to continue the decay heat removal function after the closed suction isolation valve(s) is manually opened by an operator. Analyses indicate that if all decay

heat removal capability were lost at the time of reactor trip, at least 2 hours would be available for the operators to restore decay heat removal capability before core uncovery. If decay heat removal capability were lost while on RHR cooling, considerably more time than 2 hours would be available for operator action to correct the situation.

3. If both RHR pumps were damaged while the steam generators were open for maintenance (or during any other period in which all steam genrators were unavailable), the charging pumps or safety injection pumps could be used to inject water into the RCS for core cooling. If the manways on the steam generator primary side were open for maintenance, water would flow out the manways and onto the floor of the containment. The containment spray system and the fan coolers, which are independent from the RHR system, could be used to remove decay heat inside containment to the ultimate heat sink via the component cooling water or the essential service water system.

4. Diablo Canyon Operating Procedure No. EOP-17 addresses the emergency procedure under the condition that both RHR pumps are damaged during plant shutdown.

In summary, the staff recognizes that closure of either of the two isolation valves in series in the RHR hot leg suction line would prevent the RHR system from performing its decay heat removal function and could result in damage to the RHR pumps if not corrected. Our evaluation has concluded that:

a. Although the staff did not specifically evaluate the Diablo Canyon RHR system against the criteria of BTP RSB 5-1 at the time the system was reviewed, the staff concludes that the system meets the intent of BTP RSB 5-1 for Class 2 plant implementation. The only deviation we have identified is the lack of a qualified auxiliary feedwater supply in excess of 8 hours. However, there are other diverse auxiliary feedwater sources available, which, while not designed to safety grade standards, nontheless provide a high degree of assurance that an ample auxliary feedwater supply will be available.

b) Technical Specifications and administrative procedures are in place at the plant to assure that the two series isolation valves in the RHR suction line are locked open with power sources removed from the valve operators. Moreover, a RHR low flow alarm will be installed and made operational prior to power operation to ensure that the operators will be alerted to any low flow condition that would occur in the RHR suction line, such as could occur from a closed isolation valve. Given spurious isolation valve closure as an initiating event, the failure of the operators to follow administrative procedures and technical specifications, combined with a failure of the low flow alarm or the operators to take corrective action in the presence of a low flow alarm must be postulated in order for RHR pump damage to result.

The staff considers that the need to postulate two independent failures to lose the RHR capability meets the intent of the single failure criteria.

The above capability combined with the additional capabilities to remove the decay heat even if the RHR system were lost, lead the staff to conclude that the RHR design of the Diablo Canyon Plant does not pose undue risk to the health and safety of the public.

The staff is currently conducting a generic re-evaluation of the requirements for shutdown decay heat removal systems. This work is being performed under Unresolved Safety Issue (TAP A-45). The effort includes a reassessment of the adequacy of the single RHR suction line from the hot leg and the interlocks on the suction line isolation valves.

Staff Position

Based on the staff evaluation and assessment of the safety significance as discussed above, the staff finds that this allegation does not involve considerations not previously considered for plant readiness for low power or full power operation.

Action Required

No specific action regarding Diablo Canyon is required. The staff is conducting a generic reevaluation as discussed above.

Task: Allegation or Concern No. 178

ATS No: RV-84-A-0006

Characterization:

The boron worth versus temperature curves written in 1976 to 1978 are incorrect.

Implied Significance to Plant Design, Construction or Operation:

Incorrect boron worth curves may affect calculation of the shutdown margin estimate of the reactor. Improper shutdown margin estimates could conceivably result in an inadvertent criticality.

Assessment of Safety Significance

The staff addressed this concern by interviewing the alleger, reviewing applicable procedures and records and interviewing plant engineers.

During discussions with the alleger, the alleger clarified his concern by stating that the boron worth versus temperature curves improperly assume a constant Doppler and Negative Temperature Coefficient. They do not take into account rod position or Effective Full Power Days. The alleger further stated that this problem may have been corrected after he left the site in July 1982.

Staff review of this issue disclosed that the licensee is handling this issue properly. It is a common practice to assume constant Doppler and Negative Temperature Coefficient for pre-operation estimates. This is a conservative estimate because these two parameters will actually decrease the reactivity as temperature rises. As actual reactor data becomes available from the preplanned reactor physics tests, these assumptions will be updated along with the effects of rod position and Effective Full Power Days.

Furthermore, the staff found that the licensee in fact initiated a new procedure in conjunction with Westinghouse in September 1983. This new procedure utilizes methodology acceptable to NRC for boron worth calculation and requires updating based on routine reactor physics test results.

Staff Position

The staff concluded that the licensee handled the procedure properly, i.e., the procedure development is consistent with plant operation status. Therefore, no significant safety or management problem is attributed to this allegation.

Action Required

None

Task: Allegation or Concern No. 179

ATS No: RV-84-A-0006

Characterization:

While attempting to measure the Auxiliary Salt Water (ASW) pump flow in order to meet the FSAR specified flow rates, PG&E was unable to prove the FSAR requirement was met after three years of effort.

Implied Significance to Plant Design, Construction or Operation

The flow rates specified in FSAR should be verified by actual measurement in order to confirm the performance of the ASW system.

Assessment of Safety Significance

The staff found that the flow rate measurement problem was corrected by 1) an improved flow measurement device and, 2) cleaning of the ASW pipe. Furthermore, the staff verified that flow rate measured during the surveillance of the ASW pump is consistent with the FSAR values. The staff found no significant quality or safety problem associated with this issue.

Staff Position

This allegation was substantiated. However, the problem was recognized and had been corrected by the licensee. Furthermore, no safety significance is attributed to this allegation. This topic was followed by the Resident Inspectors.

Action Required

None.

Task: Allegation or Concern No. 180

ATS No: RV-84-A-0006

Characterization:

During a pre-operational test, three Component Cooling Water (CCW) heat exchanger inlet valves of the Auxiliary Saltwater System (ASW) were broken due to water hammer. For some of these pre-operational tests, the extent of the pressure rise during testing was not accurately measured and documented.

Implied Significance to Plant Design, Construction or Operation

The operability of the ASW ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. Failure to measure the ASW pressure accurately could impair the operators ability to evaluate the test results for system operability and assess the structural damage due to pressure transients.

Assessment of Safety Significance

The staff contacted the alleger for clarification of the problems. The alleger indicated in this discussion that his main concern was with the measurement inaccuracy of the pressure instrument used in the ASW pre-operational tests. The alleger further stated that the occurrence of the pressure transients and the lack of proper documentation of these transients were less important than the measurement inaccurancy issue. To address this allegation, the staff reviewed applicable documents and discussed with the licensee the corrective actions taken.

With respect to the alleger's main concern of the pressure measurement accuracy, the licensee determined that the high pressure spikes generated during a preoperational test were in fact erroneously measured. Because the pressure measurement device was not vented, the instrument reading was higher than the actual pressure. The licensee verified this problem existed by performing a test. The licensee then took adequate corrective action.

Because of potential plant safety significance, the staff determined that in addition to addressing the alleger's main concern, it was necessary to evaluate the significance of the pressure transients and their proper documentation. The staff found the following:

a). Safety significance of the pressure transient that occurred during a preoperational test: The licensee determined that the damage to the valve discs of the CCW heat exchanger inlet valves was caused by water hammer. However, since the plant was not in operation, these damaged valves did not have any safety significance. To prevent the recurrence of water hammer in the ASW system, the licensee installed vacuum breakers and revised operating procedures. Furthermore, the Plant Safety Review Committee performed an engineering analysis to verify that there was no possibility of further structural damage to the ASW system due to the water hammer.

b). Proper documentation and followup: The licensee properly documented the damaged valve events. Furthermore, the licensee conducted tests to evaluate the ASW system's susceptablity to water hammer and reported the results to the NRC via LER 82-09. The licensee also documented the water hammer problem in nonconformance report DCO-82-MM-N059. Through routine inspection activities, the staff determined that all corrective actions have been satisfactorily completed.

Staff Position

The staff recognizes that the valve damage, the pressure transients and measurement errors indeed took place as identified by the alleger. However, the staff found that the licensee properly documented the deficiencies and corrected them. The licensee also took responsive actions to prevent recurrence of water hammer in the ASW system. Therefore, the staff concluded there is no safety significance attributed to this allegation.

Action Required

None.

Task: Allegation or Concern No. 181

ATS No.: RV-84-A-006

Characterization:

Poor, inaccurate and incomplete surveillance test records for the diesel generator system exist at Diablo Canyon.

Implied Significance to Plant Design, Construction or Operation:

Improper surveillance of diesel generators could result in existing defiencies remaining undetected and thus adversely affect a safety system required to achieve and maintain safe shutdown of the plant.

Assessment of Safety Significance:

The inspector discussed the concern with the alleger to obtain a firmer understanding of the allegation and reviewed the Regulatory Guide concerning periodic testing of diesel generators, which the licensee has committed to specific portions of in the Technical Specifications.

According to the alleger, the gist of this allegation is that failures of the diesel generators to perform during periodic tests are not being properly counted as failures. For example, a plastic cover placed on the diesel engine air intake was left in place during testing and caused a failure of the diesel to start. In another example, smoldering rags on the diesel exhaust caused the operators to terminate a test. These events were not considered valid failures of the diesel engines by the licensee and the alleger questions this decision.

Regulatory Position C.2.e. of Regulatory Guide 1.108 Revision 1, is referenced in table 4.8-1 of the Technical Specifications and establishes the requirements for determination of valid test failures and successes for the diesel generators. Section C.2.e(2) of the referenced Regulatory Guide states that, "Unsuccessful start and load attempts that can definitely be attributed to...malfunction of equipment that...is not part of the defined diesel generator unit design should not be considered valid tests or failures." Equipment not part of the unit design is interpreted to include such things as the plastic air intake cover and smoldering rags on the exhaust, and therefore, these things would not lead to valid test failures. Discussions with the alleger concluded that he agreed with the interpretation of the Regulatory Guide.

In conclusion, the staff recognizes that problems existed during surveillance testing that do not relate directly to the reliability of the diesel generators or reflect any irregularities in test records. The problems do indicate inadequate housekeeping and failure to maintain control of test conditions. The NRC resident reports that this is a situation that has received a lot of attention in the last few months and seems to be satisfactory at present.

Staff Position:

This allegation is not substantiated.

Action Required:

Monitor housekeeping practices and diesel generator surveillance testing as part of routine inspection activity.

Task: Allegation or Concern No. 182

ATS No: RV-84-A-006 BN No:

Characterization:

Several alleged improprieties occurred at Diablo Canyon that are included in this allegation. Specifically:

1). Standard design flanges were used on Class 2 piping instead of flanges designed in accordance with Subdivision 1-704.5 of USAS B31.7-1969.

2). Bolts on flanges in the CVCS, RHR, and RCS systems did not meet ASME code specifications in that they were overtorqued. The PORVs and safety valves were included in this problem.

3). An engineer by the name of Walt Scott was moved out of an engineering position into a warehousing position for identifying these problems.

The alleger indicated he had heard these items but had no direct knowledge.

Implied Significance to Plant Design, Construction or Operation

Overtorqued bolts and use of improper flanges can affect the integrity of systems required to achieve and maintain safe shutdown of the plant.

Assessment of Safety Significance

The staff addressed the allegation by interviewing Mr. Scott, reviewing the licensee's FSAR commitment to codes and standards, and reviewing the applicable code requirements. In addition, the Office of Nuclear Reactor Regulation (NRR) was consulted for their position concerning code requirements. The concerns will be addressed one at a time.

The interview with Mr. Walt Scott concluded that he had in fact identified these previously mentioned problems.

- One problem involves interpretation of USAS B31.7-1969, a standard that the licensee has commited to in their Final Safety Analysis Report. The above standard requires that flanges used in Class 2 piping be designed in accordance with Subdivision 1-704.5. The contention arises in that the licensee has used standard design flanges in accordance with Subdivision 1-704.7 of the above standard and USAS B16.5, vice Subdivision 1-704.5 as required. The licensee has verified this to be true, but considers that this is an alternate method deemed acceptable by USAS B31.7-1969.

  This topic was discussed with Bob Bosnak, Mark Hartzman and Frank Cherny of the Mechanical Engineering Branch in NRR and documented per telecon dated February 9, 1984. The issue was also raised with the same individuals per telecon on March 7, 1984. The staff concluded that the use of standard design flanges per Subdivision 1-704.7 is an acceptable alternative.

- Mr. Scott has pointed out that during assembly of flanges in the RHR system, bolts were torqued in excess of the code allowed yield strengths. This was done to assure firm seating of the flange gasket. However, this higher torque value did not exceed the actual yield strength of the bolts as determined from certified material tests that were corraborated by hardness tests.

  This issue was also discussed in a telecon with NRR on February 9, 1984, and again with the same individuals per telecon on March 7, 1984. The staff concluded, based on these discussions with the Mechanical Engineering Branch of NRR, that the torquing of bolts that exceeded code allowed yield strength, but not actual yield strength, is an acceptable approach allowed by USAS B31.7-1969.

  Mr. Scott indicated during the interview that he was independently pursuing this issue with the applicable ASME Code committee and will provide additional comments to NRC if he feels it necessary.

- In regard to the allegation that Mr. Scott was moved out of an engineering position for identifying these problems, an interview was conducted with Mr. Scott's former supervisor, as well as Mr. Scott, by the resident inspector, to determine its validity. Neither party confirmed the allegation.

Staff Position

1). Although the allegations concerning flange design and overtorquing of bolts are true, the staff found out these items had already been properly considered by the licensee. The staff considers this acceptably resolved.

2). The staff concludes that Mr. Walt Scott was not moved out of his engineering position into a scheduling position for identifying the above mentioned concerns, and therefore this allegation is unfounded.

Action Required

If the ASME code interpretation differs from NRR's evaluation, which is highly unlikely, the staff should reconsider the finding.

Task: Allegation or Concern No.183

ATS No: RV84A004 BN No:

Characterization

Alleger use of hard drugs in portable toilets on site.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 184

ATS No: RV-84-A-0023

Characterization:

Unqualified fire stop designs are being used.

Implied Significance to Plant Design, Construction or Operation:

Because fire may affect safe shutdown systems, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is necessary and must be assured through the use of adequately designed, manufactured and installed fire barriers.

Assessment of Safety Significance:

The alleger is concerned that the design of the fire stops being used is unqualified. Since this area is part of the licensing application previously evaluated by the Office of Nuclear Reactor Regulations (NRR), the staff reviewed related documents and discussed with the licensing reviewer at NRR to determine the validity of this allegation.

The staff found that the fire stop design was evaluated extensively by NRR. During that evaluation, PG&E submitted evidence that the fire stop has the required three hour rating. NRR determined that both the fire barriers cable penetration seals and Pycrocrete 102 fire barriers used were acceptable.

Along with the resolution of other fire protection related items, NRR concluded that all matters relating to the fire protection program had been resolved. This design approval is documented in NUREG-0675, "Safety Evaluation Report related to the operation of Diablo Canyon Nuclear Power Station Unit 1 and 2, Supplement No. 9", June 1980, page 9-2. In addition, the staff found no apparent practice by PG&E deviating from the approved design.

Staff Position

Based on the NRR's evaluation and the additional assessment, the staff concluded that this allegation is not subtantiated.

Action Required

None.

Task: Allegation or Concern Nos. 185, 186 and 187

ATS No.: RV-84-A-0023 BN No.:

Characterization:

These allegations concern improper foam fire stop installation techniques by the contractor, Plant-Thorpe, at Diablo Canyon. Specifically, 1) no QA is being practiced during installation of fire stops, 2) one installer does not know how to properly operate the equipment that actually formulates the two-component silicone foam and, 3) many of the foam seals are clearly no good.

Implied Significance to Plant Design, Construction or Operation:

Because fire may affect safe shutdown systems, the need to limit fire damage in systems required to achieve and maintain safe shutdown conditions is necessary and must be assured through the use of adequately designed, manufactured and installed fire stops.

Assessment of Safety Significance

To determine the validity of these allegations, the staff reviewed the foam seal specifications, installation procedure, and quality control and assurance records. The staff also interviewed one of the foam seal equipment operators to determine their training and experience. Furthermore, the staff independently inspected a small sample of foam seals in Unit 1. The allegations will be addressed one at a time.

- Concerning Quality Assurance, the staff discovered that the installing contractor does not have a quality assurance program of its own. The arrangement is for the licensee (PG&E) to provide the quality assurance program. The inspector verified they are performing quality control inspections of the contractor's installed foam seals.

  Quality Control Procedure #DCP-2 entitled "Silicone Foam Installation" is the applicable procedure used by the licensee's quality control inspector performing inspections of the foam fire stops. Foam quality is documented on a "Silicone Foam Test Report" and individual fire stop integrity is documented on a "Fire Stop Inspection Report". The staff reviewed 50 Fire Stop Inspection Reports and Silicone Foam Test Reports for compliance with quality control procedure DCP-2. No procedural violations were identified that were not corrected.

- Although the individual alleged to be incompetent was not available for interview at the time, one of the available operators was interviewed to determine the extent of their training, experience and knowledge. The individual interviewed indicated that there are currently only three people installing fire stop foam for the contractor at Diablo Canyon. He indicated that although they had no formal classroom training, all three had received on the job training and he believed that all were competent in installing fire stop foam.

Also, the resident inspector was able to observe the alleged individual operate the equipment and dispense some fire stop form. He concluded that the individual was familiar with the equipment and the procedures, and that his proficiency was satisfactory based on the quality of the foam dispensed.

- Concerning many of the foam seals being no good, the staff noted that the licensee has performed a 100% inspection of all fire stops which were used in nuclear safety related compartments. This inspection was conducted by nuclear plant operations (NPO) personnel who were trained specifically for this activity, and is required at least once per 18 months by the Technical Specifications. The staff reviewed inspection reports generated for three fire zones at random, and all reports reviewed revealed either adequate foam fire stops or, if deficiencies were identified, they were documented and corrected.

  In addition, the staff independently inspected 30 foam fire seals at random (approximately 1%) used for safety related cable penetrations in Unit 1. All of the seals inspected appeared acceptable.

Staff Position

1) Although the contractor does not have his own quality assurance program, the staff concludes that the licensee's quality assurance program appears adequate to ensure the quality of the foam fire stops being installed.

2) Based on the aforementioned interview and the resident inspector's observation of the alleged individual, the staff concludes that the operators have adequate knowledge to properly dispense the fire stop foam.

3) Based on the 100% inspection performed by the licensee and the independent inspection performed by the staff, there is a high degree of confidence in the quality of the existing foam fire stops.

In summary, the staff did not identify any obvious instances of wrongdoing and no significant breakdown of quality control procedures.

Action Required

None

Task: Allegation 186

ATS No.: RV-84-A-0023 BN No.:

Characterization

See Allegation 185

Implied Significance to Plant Design, Construction, or Operation

See Allegation 185

Assessment of Safety Significance

See Allegation 185

Staff Position

See Allegation 185

Action Required

See Allegation 185

Task: Allegation 187

ATS No.: RV-84-A-0023 BN No.:

Characterization

See Allegation 185

Implied Significance to Plant Design, Construction, or Operation

See Allegation 185

Assessment of Safety Significance

See Allegation 185

Staff Position

See Allegation 185

Action Required

See Allegation 185

Task: Allegation or Concern No. 188

ATS No.: RV-84-A-0024 BN No.:

Characterization

QA breakdown at Pullman.

Assessment of Safety Significance

The staff's assessment of this issue is that the alleger has identified issues which have been addressed and extensively examined in previous allegations 57, 68, and 103-119.

Staff Position

The issue of concern here does not appear to represent a new significant management or quality performance issue which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide written response to their findings and corrective actions.

Task: Allegation or Concern No. 189

ATS No.: RV-84-A-0025 BN No.:

Characterization

Magnaflux weld verification program accepted bad welds.

Assessment of Safety Significance

The staff's assessment of this issue is that the allegation is a known issue which has been responsibly handled in the past.

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in allegations 123 and 192. The issue of concern here does not appear to represent a new significant management or quality performance issue which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 190

ATS No.: RV-84-A-0025 BN No.:

Characterization

Pipe support base plate installation doesn't define bearing surfaces.

Assessment of Safety Significance

A face value assessment of this allegation indicates it is not of major significance in terms of public health and safety or management breakdown. The staff considers the placement of steel shims underneath base plates to be well within the purvue of the mechanics who install the base plates.

Staff Position

The issue of concern here does not appear to represent a new significant management or quality performance issue. Therefore, it will not be pursued further by the NRC staff.

Action Required

None

Task: Allegation or Concern No. 191

ATS No.: RV-84-A-0025 BN No.:

Characterization:

PG&E has the attitude that QC finds too many problems. PG&E has directed that shop welds are not to be inspected. No specifics were provided.

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in the past. The issue of concern here does not appear to represent a new significant management or technical situation which has not been previously addressed.

Action Required

This allegation will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 192

ATS No.: RV-84-A-0025 BN No.:

Characterization

Acceptance criteria changed to decrease weld failure rate.

Assessment of Safety Significance

The staff's assessment of this issue is that the allegation is a known issue which is being responsibly handled.

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in allegations 123 and 189. The issue of concern here does not appear to represent a new significant management or quality performance issue which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 193

ATS No.: RV-84-A-0025 BN No.:

Characterization

Poor QC inspector selection and training

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in the past and also identified and extensively reviewed in allegations 57 and/or 58. The issue of concern here does not appear to represent a new significant management or technical situation which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 194

ATS No.: RV-84-A-0025 BN No.:

Characterization

Document control is informal (rules made up as they go along).

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in the past and also identified and extensively reviewed in allegations 61 and 102. The issue of concern here does not appear to represent a new significant management or technical situation which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 195

ATS No.: RV-84-A-0025 BN No.:

Characterization

Document control stamps are not controlled.

Staff Position

This allegation is an issue that is vague and is one of minor significance in terms of the health and safety of the public. It does not represent a new significant management or technical situation which has not been previously addressed.

Action Required

This issue will be turned over the PG&E for response. The licensee will be required to provide written response to their findings and corrective actions.

Task: Allegation or Concern No.196

ATS No: Q5-84-010 BN No:

Characterization

Intimidation by a Foley QC person against a superior.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No.197

ATS No: Q5-84-010 **BN No:**

Characterization

Intimidation by a Foley QC person against subordinates.

Implied Significance to Plant Design, Construction, or Operation

Assessment of Safety Significance

Staff Position

Sensitive

Action Required

Task: Allegation or Concern No. 198

ATS No.: RV-84-A-0027 BN No.:

Characterization

Foley QC person incorrectly handles work packages.

Staff Position

This allegation is an issue which appears to be a one of minor significance in terms of the health and safety of the public. It does not represent a new significant management or technical situation which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 199

ATS No.: RV-84-A-0027 BN No.:

Characterization

Foley QC rushing work to meet schedules.

Implied Significance to Plant Design, Construction, or Operation

Rushing to meet schedules could reduce effectiveness of QC functions.

Assessment of Safety Significance

The allegation was in reference to the relatively large numbers of work items assigned to a QC group in a short period of time. The alleger provided examples of work performed. There was no indication, however, that the expediting of work in this case resulted in any violation of requirements. The implications of this concern were examined in part, through staff interviews of approximately 250 plant workers. Based on these interviews, the staff concluded that there was not a widespread or chronic problem of "corner-cutting" at Diablo Canyon.

The alleger did identify a related instance where there was direct perceived pressure to work contrary to the quality program. This is being separately addressed under allegation No. 197.

Staff Position

The staff review of the events in question did not disclose any improper actions being taken. Based on this, allegation is considered resolved.

Action Required

None

Task: Allegation or Concern No. 200

ATS No: RV-84-A-030 BN No:

Characterization:

NDE reports inconsistent with contractors inspection reports of welds.

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in the past. The issue of concern here does not appear to represent a new significant management or quality performance issue which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 201

ATS No: RV-84-A-030 BN No:

Characterization:

NDE reports improperly changed without proper approvals.

Staff Position

This allegation is an issue which appears to be a restatement of concerns identified in the past. The issue of concern here does not appear to represent a new significant management or quality performance issue which has not been previously addressed.

Action Required

This issue will be turned over to PG&E for response. The licensee will be required to provide a written response to their findings and corrective actions.

Task: Allegation or Concern No. 202

ATS No: Q5-84-003 **BN No:**

Characterization

Alleger states that five to seven years ago, he heard that a person he knows or a third party, when working at the Diablo Canyon Power Plant, falsified weld x-rays on piping. These welds reportedly were not "critical welds." The alleger has the impression this practice was condensed and "a lot" of people were involved in the fasification of these x-rays.

Implied Significance to Plant Design, Construction, or Operation

Piping has been installed in the Diablo Canyon Power Plant with rejectable welds.

Assessment of Safety Significance

An investigation must be conducted to assess the significance.

Statt Position

No assessment can be made at this time.

Action Required

Institute and complete the investigation.

Task: Allegation or Concern No. 203

ATS No: RV-84-A-0032

Characterization

An individual was concerned that square tubing used for a seismic support consisted of two pieces welded together and no record of the weld location retained.

Implied Significance to Design, Construction or Operation

The concern is that the weld may have to be reexamined in the future or that welding tube steel pieces together is not acceptable.

Assessment of Safety Significance

A concerned individual gave an inspector an H. P. Foley Engineering Disposition Request (EDR) No. 10070 on DCN DC1-EC-13762 dated October 6, 1983, which is an example of the expressed concern. The staff examined the EDR and determined that the extra weld was performed on a safety-related system. The work had been completed in accordance with approved HPF Weld Procedure Specification WPS-6 and that minor revision of DCN DC1-EC-13762 had been prepared to document this weld and to obtain PG&E engineering approval. When questioned regarding this issue licensee construction management personnel stated that this weld had been made as an aid to construction at the time. They also stated that the weld to a qualified procedure (WPS-6) should be as strong as the parent metal, that no further non-destructive examination was required or contemplated and that PG&E engineering had approved the minor revision which documented this extra weld.

Staff Position

The inspector determined that the alleger concern that the square support was made of two piece was confirmed. The inspector found; however, that this was done in a controlled and proper manner.

Action

None.

Task: Allegation or Concern No. 204

ATS No: RV-84-A-0032

Characterization

An individual expressed the concern that contractor engineering was modifying PG&E drawings by adding weld numbers.

Implied Significance to Design, Construction or Operation

The PG&E drawings are requirements based on engineering calculations and analyses which would be invalid if incorrectly installed.

Assessment of Safety Significance

The concerned individual gave the inspector an example in the form of Drawing SKC-HV2-320, Sheet 3 associated with DCN DC2-EC-14289, Rev. 0 and showed the inspector where weld numbers had been added by contractor personnel. The staff found that the practice of adding weld numbers was not described in licensee or contractor procedures and that the practice was also used by Pullman for rupture restraints and piping spools. When questioned regarding the issue, licensee personnel stated that having contractors weld numbers is a common industry practice and is necessary to track the work. The weld numbers (or weld map as it is sometimes called) may be used to assign specific Welding Procedures to specific welds and to provide a means to index weld inspection documentation when different welds on the same piece of work are completed to different weld procedures. This appears consistent with the inspection staff's prior inspection experience (i.e., that contractors make up weld maps) and with the contractor's weld inspection records, which are retained with the weld maps. Consequently, the staff was unable to identify any safety significance associated with the contractor assigning weld numbers to welds on PG&E furnished drawings.

Staff Position

The concern is true that contractor personnel were marking up PG&E drawings with weld numbers for their use. There appears to be no prohibition or safety significance associated with this practice in that it is a responsible orderly process that is used throughout the industry.

Action

None.

Task: Allegation or Concern No. 205

ATS No: RV-84-A-0032

Characterization

A concerned individual stated to an inspector that unqualified electrical splices had been observed on wires to solenoids and instruments in the containment. The alleger could not provide specific examples.

Implied Significance to Design, Construction or Operation

The concern is that environmentally unqualified splices may be used inside containment for devices required to be operable in a severe environment.

Assessment of Safety Significance

The staff examined panels and devices inside the containment listed below to determine the validity of the concern:

Panel/Elevation	Device	Environmentally Qualified Splices Used	Remarks
PM-46/115'	SV-66	Yes	Air to valve 8145, pressurizer auxiliary spray

	SV-79A	Yes	Air to valve 8149A, letdown orifice 1-1 outlet
	SV-245	No	Air to FCV-762-S/G 1-3 blowdown samp. valve (device fails safe on loss of power)
	SV-346	No	Air to valve 8875C - Acc.Tk. 3 vent isol. valve (device fails safe on loss of power)
	FT-532	Yes	S/G 1-3 flow transmitter
PM-45/115'	FT-522	Yes	S/G 1-2 flow transmitter
PM-83/85'	LT-501	Yes	S/G 1-1 level transmitter
PM-20/85'	PT-457	Yes	Pressurizer steam pressure transmitter

In all cases observed, the inspector found that the devices which required environmentally qualified splices were in fact proper. The staff also examined previous NRC inspections related to the expressed area of concern (Region V reports 50-275/78-03, paragraph 7, 50-275/79-06, paragraphs 2.n,o, and q, 50-275/79-07, paragraph 4.a, and 50-275/79-12, paragraph 5) and discussed these reports with the lead NRC inspector (Mr. D. Kirsch). Based on these examinations and discussions, previous NRC examinations in 1978 and 1979 have verified that properly environmentally qualified splices were used where required by the licensee. Because of the non specific nature of this allegation, the inspector considered the limited review sufficient.

Staff Position

The concerned individual's statement is true in that there are splices which are not environmentally qualified. However, based on the staff's examination of four panels in containment, examinations of previous NRC reports, and discussions with the lead inspector who inspected the environmentally qualified splices in the containment in the past, the staff concludes that environmentally qualified splices appear to have been used where required.

Action

None.

Task: Allegation or Concern No. 206

ATS No: RV-84-A-0032

Characterization:

An individual expressed the concern that electrical conduits may not be properly controlled.

Implied Significance to Design, Construction or Operation:

Loss of control of conduit identification and routing could result in violation of separation criteria within junction boxes and could result in invalid jet impingement and pipe whip analyses.

Assessment of Safety Significance:

When questioned as to what the concern related to the concerned individual showed the inspector H. P. Foley (HPF) conduit installation and inspection records described below:

HPF Installation Record of Conduit "KX467" dated April 21, 1980

HPF Inspection Record of Conduit "KX467" dated April 29, 1980

HPF Installation Record of Conduit "KX467" dated May 6, 1980

HPF Inspection Record of Conduit "KX467" dated May 29, 1980

The records dated in April 1980, indicate that the subject conduit is 3/4" in diameter and is routed between junction box PM-91 and containment penetration 14E. The records dated in May 1980 describe the conduit as 2" in diameter and routed between the containment hydrogen recombiner and junction box BJX200. The inspector visually confirmed that both conduits existed and were marked "KX467." The inspector examined the licensee's raceway schedules (PG&E drawings 103075 Rev. 44 page 125 and 103077 Rev. 44 page 60) and found that the 3/4" conduit was empty and had been abandoned. The inspector after identifying this item to the licensee asked licensee personnel how the error occurred and what prevented this type of error from occurring in other conduit.

Licensee personnel explained that the 3/4" conduit "KX467" had been properly installed in accordance with Design Change Notice (DCN) DCO-EE-512 which was issued on March 7, 1980 for conduit installation. Subsequently, DCN DCO-EE-512 was cancelled and the remaining work transfered to another DCN. Construction submitted as-built documentation to PG&E engineering showing that 3/4" "KX467" had been installed. However, engineering changed the PG&E raceway schedule listing this conduit as 3/4" KX467 to 3/4" KX469 and used the "KX467" designation for another DCN related to the installation of the Containment Hydrogen Recombiners. Since no wires were ever placed in the 3/4" conduit, engineering did not issue a new wire pull card to construction.

Licensee Construction Management personel explained that pull cards are issued whenever a circuit is installed (or changed) or when a conduit number with installed wires is changed. In addition, HPF procedures (QCP-E11) require a walkdown of a conduit and observation of correct conduit and terminations before any wire is installed. As a consequence any mislabeled conduit would be identified when wires were installed. Also, it would be obvious by visual observation if a mislabeled conduit bridged two divisions of safety-related electrical power since the conduits are coded with colored bands indicating electrical divisions.

Staff Position:

The concerned individual was found to be correct in that an example of a misidentified conduit was found, however, lack of control of conduits does not appear to be a problem.

Action Required:

None.

Task: Allegation or Concern No. 207

ATS No: RV-84-A-034 BN No:

Characterization

Inadequate training for all Pullman work activities.

Staff Position

This allegation is very general and appears to be a restatement of concerns identified in documentation for allegations 68 and 103-119. The issue of concern here does not represent a new significant management or quality performance issue which has not been previously addressed.

Action Required

None

Task: Allegation or Concern No. 208

ATS No: RV-84-A-034 BN No:

Characterization

Unacceptable management attitude in the administration of Discrepancy Reports.

Staff Position

This allegation is very general and appears to be a restatement of concerns identified in allegations 24, 26, 46, and 66. The issue of concern here does not represent a new significant management or quality performance situation which has not been previously addressed.

Action Required

None

Task: Allegation or Concern No. 209

ATS No: RV-84-A-034 BN No:

Characterization

Pullman Supervision (names given) qualifications inadequate.

Staff Position

This allegation came to the NRC third hand as a result of an individual, not involved with Diablo Canyon, overhearing conversations between individuals believed to be workers at Diablo Canyon. The allegation is vague, even though specific names are provided. In the past, the staff has reviewed the qualifications of various Pullman personnel. The issue of concern here does not appear to represent a new significant management or technical situation which has not been previously addressed.

Action Required

None

Task: Allegation or Concern No. 210

ATS No: RV-84-A-034 BN No:

Characterization

Qualifications of other plant workers is questionable.

Staff Position

This allegation came to the NRC third hand as a result of an individual, not involved with Diablo Canyon, overhearing conversations between individuals believed to be workers at Diablo Canyon. This allegation is very general and appears to be a repeat of concerns identified in allegation 68. In the past the staff has reviewed the qualifications of various Foley, Pullman and PG&E personnel. The issue of concern here does not represent a new significant management or technical situation which has not been previously addressed.

Action Required

None

Task: Allegation or Concern No. 211

ATS No: RV-84-A-034 BN No:

Characterization

Pullman welding not done in accordance with ASME IX.

Staff Position

This allegation came to the NRC third hand as a result of an individual, not involved with Diablo Canyon, overhearing conversations between individuals believed to be workers at Diablo Canyon. This allegation is very general, without specifics, and appears to be a repeat of concerns identified in allegations 101, 103-119 and 214-217. The issue of concern here does not represent a new significant management or technical situation which has not been previously addressed.

Action Required

None.

Task: Allegation or Concern No. 212

ATS No: RV-84-A-034 BN No:

Characterization

Pullman's welding of materials not properly qualified (no specifics provided).

Staff Position

This allegation came to the NRC third hand as a result of an individual, not involved with Diablo Canyon, overhearing conversations between individuals believed to be workers at Diablo Canyon. This allegation is very general, without specifics, and appears to be a repeat of concerns identified in allegations 103-119 and 214-217. The issue of concern here does not represent a new significant management or technical situation which has not been previously addressed.

Action Required

None

Task: Allegation or Concern No. 213

ATS No: BN No:

Characterization

Inadequate design of all raceway supports and others. (No specifics were provided.

Staff Position

This allegation was provided to the NRC during a telephone conversation with an anonymous alleger who indicated he was employed by Bechtel. The alleger would not provide specific information by telephone but requested to meet with the staff at 4:00 PM on Thursday, March 8, 1984. The alleger did not show up at the preestablished meeting place. Subsequent staff efforts failed to contact the alleger at the telephone number he provided. Because the staff, PG&E and the IDVP have previously examined raceway support design, the staff feels that additional effort to examine such a vague concern would not be prudent.

Action Required

None

Task: Allegation or Concern No. 214

ATS No: RV-83-A-0074

Characterization

Code 7/8 and 92/93 not technically the same.

Implied Significance to Plant Design, Construction, or Operation

See Task Allegation or Concern No. 103-119

Assessment of Safety Significance

See Task Allegation or Concern No. 103-119

Staff Position

See Task Allegation or Concern No. 103-119

Action Required

See Task Allegation or Concern No. 103-119

Task: Allegation or Concern No. 215

ATS No: RV-83-A-0074

Characterization

Code 92/93 not qualified for unlimited thickness.

Implied Significance to Plant Design, Construction, or Operation

See Task Allegation or Concern No. 103-119

Assessment of Safety Significance

See Task Allegation or Concern No. 103-119

Staff Position

See Task Allegation or Concern No. 103-119

Action Required

See Task Allegation or Concern No. 103-119

Task: Allegation or Concern No. 216

ATS No: RV-83-A-0074

Characterization

Code 7/8 and 92/93 not interchangeable.

Implied Significance to Plant Design, Construction, or Operation

See Task Allegation or Concern No. 103-119

Assessment of Safety Significance

See Task Allegation or Concern No. 103-119

Staff Position

See Task Allegation or Concern No. 103-119

Action Required

See Task Allegation or Concern No. 103-119

Task: Allegation or Concern No. 217

ATS No: RV-83-A-0074

Characterization

Pullman performed a QA coverup through use of 1978 memo.

Implied Significance to Plant Design, Construction, or Operation

See Task Allegation or Concern No. 103-119

Assessment of Safety Significance

See Task Allegation or Concern No. 103-119

Staff Position

See Task Allegation or Concern No. 103-119

Action Required

See Task Allegation or Concern No. 103-119

NRC FORM 335 (11-81) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0675 Supplement No. 22
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH March YEAR 1984
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Division of Licensing Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission Washington, D.C. 20555	DATE REPORT ISSUED MONTH March YEAR 1984
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Same as 9. above	10. PROJECT/TASK/WORK UNIT NO.
	11. FIN NO.
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15. SUPPLEMENTARY NOTES Docket Nos. 50-275 and 50-323	14. (Leave blank)

16. ABSTRACT (200 words or less)

Supplement No. 22 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 identified to the NRC as of March 9, 1984. It includes the criteria that were used by the NRC to determine which of the allegations that have been evaluated thus far must be resolved prior to Unit 1 achieving criticality and operating at power levelsup to 5 percent of rated power (i.e., low power operation).

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
17b. IDENTIFIERS/OPEN-ENDED TERMS	

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