UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
COMMISSIONER

June 11, 1984

(2)

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Roberts
Commissioner Bernthal

FROM: James K. Asselstine

SUBJECT: DIABLO CANYON

At the time the Commission voted to reinstate the low power license for Diablo Canyon, we had the benefit of an ACRS report, dated April 9, 1984, on the design control measures at that site. Several of the members, in appended views, urged the staff to prepare, prior to authorizing power operation above five percent,"...a document discussing in considerable detail how the various relevant issues raised by its inspectors and others have been handled." Those members also urged"...a careful examination (by the NRC staff) of a selected sample of actual construction details to help assure that the appropriate quality has been accomplished." I supported those views at the time but an explicit Commission position was not developed. It is unclear how, if at all, the staff will address the above. I believe we should direct the staff to carry out those recommendations through the attached memorandum. I ask that SECY obtain Commissioner responses by C.O.B. Wednesday, June 13, 1984.

June 13

Jim:

I agree with the thrust of the ACRC members' request. However, I would expect that the SSER prepared for the full power authorization to contain a full discussion of the resolution of Mr. Yin's concerns and resolution of the allegations. I would direct the staff to ensure that the SSER contains this documentation. Regarding the need for staff verification of quality, I believe the inspection program has done this. I believe that the staff should verify that its previous activities have included confirmation of quality as well as quality assurance.

If a majority agrees, I would redraft the SRM accordingly.

NJP

cc: SECY
OGC
OPE

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

MARCH 1984

See pp E-5 to E-7 for allegation resolution criteria.

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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.

The NRC Project Manager for the Diablo Canyon Nuclear Power Plant is Mr. H. Schierling. Mr. Schierling may be contacted by calling (301-492-7100) or by writing to the following address:

Mr. H. Schierling
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Copies of this Supplement are available for public inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C., and at the California Polytechnic State University Library, Documents and Maps Department, San Luis Obispo, California 93407. Availability of all material cited is described on the inside front cover of this report.

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).

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.

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:

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

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.

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 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).

What is status

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. VI

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 pre-criticality 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.

EXCERPT FROM
SSER # 21

any corner cutting, intimidation or harassment, nor did they have any concerns related to safety items. In general, responses indicated that management was responsive to concerns, accessible, quality oriented, had an open door policy and supportive of employee's concerns. The information obtained in the interviews has been and is being used to follow up on the specific technical allegations. Additional interviews and inspections will be performed, as necessary, to assure adequate evaluation of comments received.

3.4 ALLEGATION STATUS

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

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

3.4.1 Actions Required prior to Criticality

3.4.1.1 Small bore piping design adequacy.

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

3.4.1.2 Anchor bolt design margins and installation

As discussed in paragraph 3.2.1 and 3.2.2 above there are three allegations or concerns which have lead the staff to seek more information about the adequacy of anchor bolt design margins and installation. Concern for design margins was not a specific allegation but

was encountered while reviewing concerns related to items Nos. 25, 58, and 96. A preliminary assessment of adequacy cannot be made at this time. The collection and evaluation of additional information is in progress.

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

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

3.4.1.3 Inspector Certifications

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

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

3.4.2 Actions Required Prior to Exceeding Five Percent Power

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

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

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

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

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

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

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

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

Later

6 Small bore piping

7. High strength bolts

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

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

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

3.5 CONCLUSION AND RECOMMENDATIONS

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

Approximately 75% of the allegations currently received have been examined to a point where it is the staff's opinion that there is no significant safety issue or substantial breakdown of management or quality systems. The remaining allegations have been assigned to various elements of the NRC staff for evaluation and most have been

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'84 APR -3 P4:44

COMMISSIONERS:

Nunzio J. Palladino, Chairman
Victor Gilinsky
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal

DOCKETING & SERVICE BRANCH

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SERVED APR 4 1984

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY (Diablo Canyon Nuclear Power Plant, Units 1 & 2)	Docket Nos. 50-275 50-323

OGC is reviewing + evaluating comments

ORDER
(CLI-84- 4)

This order concerns the issue of the consideration of complicating effects of earthquakes on emergency planning in the Diablo Canyon licensing proceedings.

In the San Onofre proceeding, the Commission declared that

> current regulations do not require consideration of the impacts on emergency planning of earthquakes which cause or occur during an accidental radiological release. Whether or not emergency planning requirements should be amended to include these considerations is a question to be addressed on a generic, as opposed to a case-by-case, basis.

Southern California Edison Co. (San Onofre Nuclear Generating Station, Units 2 and 3), CLI-81-33, 14 NRC 1091, 1091-1092 (1981). In the interim, the Commission precluded consideration of this issue in individual licensing adjudications. Thus, the boards have properly excluded this issue from this adjudication.

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In response to the Commission's San Onofre decision, the NRC staff reported its view that generic consideration was neither necessary nor appropriate, but appears to believe that some specific consideration of the effects of seismic events on emergency planning may be warranted for plants located in areas of relatively high seismicity. See NRC staff memoranda, dated June 22, 1982 and January 13, 1984, attached hereto.

In view of this development, the Commission has decided to address whether to allow such consideration under the circumstances in this case. With respect to low-power operation, however, the Commission is satisfied that, pursuant to 10 C.F.R. 50.47(d), this issue need not be reviewed further because it pertains primarily to offsite emergency planning requirements which are not essential to low-power license decisions.

To help the Commission with its consideration of this issue, the parties are requested to provide their views on the following issues no later than 30 days after the date of this order.

Issues:

1. whether NRC emergency planning regulations can and should be read to require some review of the complicating effects of earthquakes on emergency planning for Diablo Canyon;

2. if the answer to question (1) is no, should such a review be performed for Diablo Canyon on the ground that it presents special circumstances under 10 C.F.R. 2.758. If so, what are the special circumstances that would permit consideration of the effects of earthquakes on emergency planning for Diablo Canyon?

3. if the answer to (1) or (2) is yes, then the following information should be provided:

(a) The specific aspects of emergency planning at Diablo Canyon on which the impacts of earthquakes should be considered.

(b) The specific deficiencies in the consideration already given to the impacts of earthquakes on emergency plans for Diablo Canyon. In this regard the NRC staff is directed to serve on the parties to the proceeding a copy of the Licensee's submittal regarding effects of earthquake on emergency planning. However, the Commission is not requesting the filing of contentions in response to this order. The matter of contentions will be handled by a Licensing Board if a proceeding is to be held.

(c) The appropriateness of limiting to the Safe Shutdown Earthquake the magnitude of the largest earthquake to be considered.

(d) The substantive criteria for reviewing the effects of earthquakes on emergency planning.

(e) The necessity for litigation of this matter, including the general scope of (i) proceedings, if any, that should be held, and (ii) issues that should be litigated.

The Commission notes that it is not now deciding whether any requirement for further hearings would require that interim operation of the plant be stayed. The stay determination, if and when it is presented, will be a matter for the equitable discretion of the Commission or Appeal Board. See e.g., Public Service Company of New Hampshire

(Seabrook Station, Units 1 and 2), CLI-77-8, 5 NRC 503 (1977). Parties need not address the stay question at this time.

Commissioner Gilinsky abstained from this decision.

It is so ORDERED.

For the Commission[1]

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington, DC,
this 3d day of April, 1984.

[1]Commissioner Asselstine was not present when this Order was affirmed, but had previously indicated his approval.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUN 22 1982

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Ahearne
Commissioner Roberts
Commissioner Asselstine

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: EMERGENCY PLANNING AND NATURAL HAZARDS

By memorandum dated March 1, 1982, the Secretary of the Commission requested the staff to consider several questions with regard to emergency planning.

1. Should the emergency planning activities of NRC licensees include consideration of the possible effects on emergency plans of a very large earthquake?

It is the judgment of the staff that for most sites earthquakes need not be explicitly considered for emergency planning purposes because of the very low likelihood that an earthquake severe enough to disturb onsite or offsite planned responses will occur concurrently with or cause a reactor accident. Planning for earthquakes which might have implications for response actions or initiate occurrences of the "Unusual Event" or "Alert" classes in areas where the seismic risk of earthquakes to offsite structures is relatively high may be appropriate (e.g., for California sites and other areas of relatively high seismic hazard in the Western U. S.).

2. If NRC requirements are to include this consideration, then what criteria should be applied in evaluating the adequacy of such plans in this respect?

In view of the staff response to question 1, current review criteria are considered adequate. Also the staff does not believe that rulemaking is necessary with regard to this issue based on the analysis conducted. The Hearing Boards have read the Commission ruling in the San Onofre case (CLI-81-33) to eliminate consideration of all earthquakes at California sites.* The interaction of earthquakes less than the SSE with emergency preparedness was considered in the staff SER for San Onofre and ultimately was not a matter in contention in the San Onofre proceeding.

Commissioner Ahearne requested several actions be taken by the staff and these requests were also transmitted in the March 1, 1982, memorandum from the Secretary of the Commission. These are addressed below.

*For example, Pacific Gas & Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), Memorandum and Order, December 23, 1981 (unpublished), directed certification denied by Commission Order dated March 5, 1982.

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1. The staff should, in conjunction with FEMA, develop an approach for checking the ability of emergency plans to cope with natural phenomena which would be expected to occur during the life of the plant. Examples are: earthquakes, blizzards, tornadoes, hurricanes, tsunamis, and floods that might be expected once every 40 years. FEMA and the staff should develop guidelines for examining plans for flexibility and should identify measures which can be used to assure flexibility.

As stated in the enclosure, a site emergency plan is expected to address all the site characteristics which may require an emergency response. Adverse conditions, which generally correspond to once in 20 to 40 year events, are considered in the evacuation time estimates called for in staff guidance (Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654/FEMA-REP-1) which was developed jointly by the staff and FEMA. The evacuation time estimates are used in the optimization of evacuation and shelter plans as well as being available to decisionmakers in emergency conditions. Continuing review of plans to assure flexibility is already provided by 10 CFR Part 50, Appendix E and 10 CFR §50.54(t).

2. The staff should develop a list of the once in a lifetime natural disasters most likely for each plant either holding an operating license or in the OL process.

Because of the relatively high risk, current practice calls for California licensees and applicants to consider the effects of earthquakes in their emergency planning and for the Trojan plant to consider the consequences of a Mt. St. Helens eruption in its plan. Other plants do consider adverse conditions in developing evacuation time estimates as discussed above but a consolidated listing does not appear to warrant the effort.

3. Existing emergency plans should be examined to determine whether adequate flexibility is present.

The emergency plan reviews and the onsite implementation appraisals which the staff has been conducting include examinations of the overall flexibility of a licensee's emergency response capability and the adequacy of evacuation time estimates, which include the consideration of adverse conditions. Therefore, no further review is believed to be necessary by NRC.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosure: Staff Analysis

cc: OPE
OGC
SECY

BASIS FOR CONSIDERATION OF NATURAL HAZARDS IN EMERGENCY PLANNING

A fundamental premise in the approach to emergency planning utilized by the Federal Emergency Management Agency (FEMA) and the Commisison is that the emergency planning basis must be capable of responding to a wide spectrum of accidents. This was the conclusion reached by the Task Force which authored NUREG-0396 (Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants). That Task Force report was subsequently endorsed by the Commission in its Policy Statement with respect to the Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents (Policy Statement). 44 Fed. Reg. 61123 (October 23, 1979). The concept is reiterated in NUREG-0654 (Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants). Consequently, as a single specific accident sequence for a light water reactor nuclear power plant could not be identified as a planning basis, both NUREG-0396 and NUREG-0654 emphasized that the most important element of any planning basis is the distance from the nuclear facility which defines the area over which planning for predetermined action should be carried out. Not only is this area, termed the Emergency Planning Zone or EPZ, crucial but the <u>characteristics</u> of the EPZ are significant.

> The need for specification of areas for major exposure pathways is evident. The location of the population for whom protective measures may be needed, responsible authorities who would carry out protective actions and the means of communication to these authorities and to the population are all dependent on the <u>characteristics of the planning areas</u>. (Emphasis supplied). NUREG-0654, p. 8.

It is, therefore, inherent in the planning approach utilized by FEMA and the Commission, i.e., the Emergency Planning Zone concept, that the characteristics of the Emergency Planning Zones themselves must be factored into emergency planning considerations. For example, if an EPZ is an area with singular adverse weather attributes, those attributes must be considered in emergency planning. This reasoning would extend to all attributes that might adversely affect an Emergency Planning Zone. Although neither 10 CFR 50.47 nor Appendix E explicitly state that the emergency plans must account for adverse weather conditions or adverse site characteristics, such conditions are covered by NUREG-0654, which the Commission has adopted to provide guidance in developing plans for coping with emergencies. NUREG-0654 calls for required evacuation time estimates to consider adverse conditions which might reasonably be expected to occur during the plant lifetime at a particular site and be severe enough to affect the time estimates for a particular event.

> Two conditions--normal and adverse--are considered in the analyses. Adverse conditions would depend on the characteristics of a specific site and could include flooding, snow, ice, fog or rain. (Emphasis supplied). NUREG-0654, pp. 4-6.

Thus, adverse site characteristics of a particular Emergency Planning Zone must be taken into account to satisfactorily implement the Commission's emergency planning regulations.

Explicit planning for emergency preparedness provides a base capability which can be expanded or contracted to address an actual emergency. Backup communications and feedback of damage estimates regarding transportation routes to decisionmakers after an earthquake would be generally available with or without specific advance planning. The general planning base would allow decisionmakers to choose specific actions from among available alternatives for a spectrum of events.

There is no explicit guidance in 10 CFR 50.47 or in Appendix E to Part 50 nor in NUREG-0654 as to the extent to which adverse earthquake conditions are to be taken into account in emergency planning at particular sites. The staff, however, believes the answer to this question is dependent upon the nature of the risk and the nature of the remedy to deal with the risk. Except in California and other areas of relatively high seismic hazard in the Western U. S., the staff's judgment is that the nature of the seismic risk is such that no explicit consideration of earthquake effects is needed in emergency planning. (This judgment is not based on a quantitative analysis but rather on qualitative observations of the relatively lower seismic risk to roads, bridges and communications facilities in the east versus the west.) The occurrence of earthquakes of a nature that could have implications for onsite or offsite response actions or initiate occurrences of the "Unusual Event" or "Alert" class is an adverse characteristic of the type discussed above. The NRC staff made requests to California facilities to consider earthquake effects in their emergency planning, and the NRC staff also requested FEMA to consider earthquake effects in its evaluation of offsite plans. On the other hand, the staff concluded that additional requirements such

as the design of additional facilities, structures and systems to specifically withstand earthquakes was not necessary for the reasons discussed above. In particular, no special seismic design of public notification systems, environmental monitoring capability or communications equipment is contemplated. Also, explicit consideration need not be given to a seismic event coincident with a significant accident at the plant from another cause because of the very low likelihood of such a coincidence.

With respect to offsite effects at California sites, the FEMA Radiological Emergency Preparedness staff believes there should be assurance of continued communication between the plant and outside agencies. In addition, the Emergency Operations Centers (EOCs) of each of the jurisdictions involved in the emergency planning effort for a specific nuclear facility should have suitably distant backup facilities to permit continued functioning of a jurisdiction's emergency response given the possible failure of its primary EOC.

In addition, for California sites the capability should exist to obtain damage estimates both to the plant and to transportation and communication facilities offsite to provide a data base to factor into the decisionmaking process. Finally, California licensees should have available a range of recommendations to offsite authorities, taking into account the degree of damage to the plant caused by the earthquake and to transportation and communication facilities offsite.

Given an earthquake of magnitude less than or equal to the SSE, while the earthquake could have impacts upon communications and transportation as a consequence of the earthquake, the plant would likely not pose an immediate radiological hazard. If, however, an earthquake substantially in excess of the SSE were to occur, then the potential exists for a radiological hazard complicated by the nonradiological impacts posed by a major earthquake. In the view of the NRC staff, such a contingency does not warrant specific emergency planning efforts because of the general planning base capabilities discussed above. We conclude that this general planning base is adequate because of the remote likelihood of an earthquake substantially in excess of the SSE. In addition, the characteristics of an accident which could theoretically be created by an earthquake substantially larger than the SSE would not be outside the spectrum of accident consequences considered in NUREG-0396 upon which the judgment on planning zone sizes and other planning elements was based. This unlikely sequence would not be unlike the case of a severe accident (not generated by an earthquake) occurring after a winter storm at a site in the northern U. S. Evacuation may not be a feasible option in such a circumstance. It also should be noted that to provide for a preplanned emergency response in all remote circumstances could require a commitment of substantial societal resources, e.g., to assure that houses and bridges would withstand very large earthquakes.

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 13 1984

MEMORANDUM FOR: Chairman Palladino

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: EMERGENCY PLANNING AND SEISMIC HAZARDS

On September 9, 1983, a meeting was held with you to discuss the Staff's views on the need for and extent of consideration of the potentially complicating effects of earthquakes in the context of emergency preparedness. Please recall that this issue emanates from the Commission's Memorandum and Order in the San Onofre proceeding, CLI-81-33, issued in December 1981, in which the Commission determined that "its current regulations do not require consideration of the impacts on emergency planning of earthquakes which cause or occur during an accidental radiological release."[1] The Commission further noted that it "will consider on a generic basis whether regulations should be changed to address the potential impacts of a severe earthquake on emergency planning" and, a memorandum from the Secretary to the

[1] In the San Onofre proceeding, the Licensing Board sought to raise, sua sponte, the issue of the effects of an earthquake exceeding the Safe Shutdown Earthquake on the applicants' and responding jurisdictions' abilities to carry out an evacuation in a timely manner and/or protect those in the EPZ pending evacuation. It had been the Staff's and FEMA's positions before the Licensing Board that in that proceeding, while consideration of the complicating effects of earthquakes up to the SSE was appropriate, consideration of the potential of earthquakes exceeding the SSE was not warranted. The Licensing Board rejected this view and instead affirmed its prior position calling for consideration of the potential effects of an earthquake exceeding the SSE. Thereafter, the Commission, as indicated above, reversed the Licensing Board's decision. Parenthetically, based on the Commission's San Onofre decision, the Licensing Board, in the Diablo Canyon proceeding rejected a contention regarding consideration of the effects of earthquakes on emergency preparedness. In an unpublished order issued on March 5, 1982, the Commission denied the Governor's request for interlocutory review of the Licensing Board's action. The Licensing Board's ruling was affirmed by the Appeal Board in ALAB-728, slip op. at 20-21, (May 18, 1983) and review by the Commission was denied (CLI-83-32, December 9, 1983).

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Executive Director for Operations, by memorandum of March 1, 1982, directed the Staff to undertake such consideration. By memorandum to the Commissioners dated June 22, 1982 (copy attached), the Executive Director responded to the questions posed in the Secretary's March 1 memorandum.[2]

After our September 9, 1983 meeting with you on this subject, you requested further technical discussion to provide a rationale for either including or not including specific emergency planning requirements for seismic events. The following thoughts are presented to respond to your request:

1. Offsite Damage Associated With Extreme Seismic Events

Offsite damage generated by earthquakes can significantly affect nuclear emergency response. The earthquake hazard and potential for such damage varies across the United States. Severe damage, such as the failure of buildings, bridges, and other engineered structures can typically be associated with large damaging earthquakes and their related ground motion levels. For a large part of the U.S. east of the Rocky Mountains, where most nuclear power plants are located, such ground motion levels would be well beyond the Safe Shutdown Earthquake (SSE). For areas associated with higher earthquake hazard, such as the West Coast, these ground motion levels could be at or even less than the SSE. Such high hazard areas may also exist in the east (for example, the New Madrid, Missouri, area), however, no nuclear power plants are presently sited within these areas in the east.

2. The Potential Impact of Offsite Damage on Emergency Response

The impact on emergency response capability from earthquakes is clearly site region dependent and is generally proportional to the degree of offsite damage. That is, the higher the intensity of the earthquake, the more extensive and severe is the damage it causes. For seismic events that result in significant and widespread damage to surrounding areas, the response capability would be degraded through extensive disruption of transportation and communication

[2] To very briefly summarize the Staff's position as expressed in its June 22nd response, the Staff concluded that the Commission's regulations do not require amendment since (1) for most sites there is only a very low likelihood that an earthquake severe enough to disturb onsite or offsite planned responses will occur concurrently with or cause a reactor accident, and (2) while planning for earthquakes which might have emergency preparedness implications may be warranted in areas where the seismic risk to offsite structures is relatively high (e.g., California sites and other areas of the Western U.S.), current review criteria set forth in NUREG-0654 (which are derived from the Commission's regulations in 10 CFR §50.47) are considered adequate.

networks, and from the failure of major structures. In this instance the range of protective actions and the capability of the offsite jurisdictions to initiate and implement them could be drastically reduced. The degree of this reduction would vary based on conditions in the region around the site. For example, even with substantial damage to all bridges, a site might have so few bridges in its vicinity that blockage of roads would not be significant.

3. Plant Damage Associated With Seismic Events

When considering the possibilities of plant damage from seismic events, it is important to understand the severity of seismic events, their range of probabilities, and the potential for reactor accidents caused by seismic events. Three classes of seismic events are considered in this discussion. The first class includes earthquakes of relatively low ground motion, up to the Operating Basis Earthquake (OBE). The OBE ground motion depends on plant location. These accelerations vary in the range of about .05g to .10g (higher in areas of high seismicity). During an OBE all plant systems would be expected to remain operating.

The second class of events includes earthquakes with ground motion higher than the OBE but equal to or less than the Safe Shutdown Earthquakes (SSE); the ground motion of the SSE is typically about twice that of the OBE. Probabilities of occurrence for the SSE have typically been estimated to be on the order of one in a thousand or one in ten thousand per year. NRC regulations require that plants be designed to achieve a safe shutdown after an SSE. Given an SSE, all seismically qualified equipment would be expected to function to bring the plant to safe shutdown. An earthquake up to and including an SSE would be cause for an alert emergency action level classification. However, only in the event of a coincident failure of a safety function (safety systems are designed for the SSE) or some undiscovered common cause failure mechanism (such as a major design error) would there be a chance of an accident which would require offsite emergency response. The probability of these two events (SSE and safety function failure) occurring simultaneously is very much lower than the probability of either one, perhaps on the order of one in a million per reactor year or less.

The final class of events includes all earthquakes with ground motion levels above the SSE. Fragility analysis is used to estimate the probability of failure as a function of ground motion associated with these earthquakes. The Zion, Indian Point, and Limerick Probabilistic Risk Assessments estimated that, in general, ground motion on the order of 0.5g to 0.75g acceleration would be required to damage a nuclear power plant to the extent that significant release of radioactivity could occur. Of course, some plants, such as those in high seismic regions, are designed to withstand earthquakes with ground motion this high; they would resist damage to still higher levels of ground motion. The probability estimates for such ground accelerations are significantly less than the probability estimates for the SSE for these plants (the Zion, IP, and Limerick SSEs are .17g, .15g, and .15g respectively). The absolute probabilities for earthquakes at and beyond the SSE are extremely difficult to estimate and thus have large associated uncertainties.

4. Current Emergency Preparedness Considerations

Seismic events are considered and evaluated to a limited extent as part of our current emergency planning reviews. The following planning standards, some of which explicitly address seismic events, are addressed by the licensee, state and/or local emergency plans as explained in the following sections from NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."

II.D.4 Emergency Classification System

"Each State and local organization should have procedures in place that provide for emergency actions to be taken which are consistent with the emergency actions recommended by the nuclear facility licensee, taking into account local offsite conditions that exist at the time of the emergency." (Emphasis added)

II.H.5.a Emergency Facilities and Equipment

"Each licensee shall identify and establish onsite monitoring systems that are to be used to initiate emergency measures in accordance with Appendix 1, as well as those to be used for conducting assessment.

This equipment shall include:

a. geographical phenomena monitors, (e.g., meteorological, hydrologic, seismic);"

II.H.6.a Emergency Facilities and Equipment

"Each licensee shall make provisions to acquire data from or for emergency access to offsite monitoring and analysis equipment including: (Emphasis added)

a. geographical phenomena monitors, (e.g., meteorological, hydrologic, seismic);"

II.J.10.k Protective Response

"The organization's plans to implement protective measures for the plume exposure pathway shall include:

k. Identification of and means for dealing with potential impediments (e.g., seasonal impassibility of roads) to use of evacuation routes, and contingency measures;"

For each of the emergency response classes given in Appendix 1 of NUREG-0654, severe natural phenomena (including seismic events) are included as part of the

example initiating conditions. The seismic events specifically included in this appendix are the Operating Basis Earthquake, and the Safe Shutdown Earthquake as well as "any earthquake felt in-plant or detected on station seismic instrumentation."

The preceding show that seismic events are considered in emergency planning but, as is evident, these review criteria are not very clear and clarification of them could lead to some improvements in emergency preparedness, perhaps by leading to more refined analysis of potential road blockage, etc. However, it is not clear that such improvements would substantially reduce the impairment of emergency response caused by seismic damage offsite.

The Federal Emergency Management Agency (FEMA) reviews offsite radiological emergency planning and preparedness to insure the adequacy of Federal, State, and local capabilities in such areas as emergency organization, alert and notification, communications, measures to protect the public, accident assessment, public education and information, and medical support. Detailed, specific assessment of potential earthquake consequences and response are not part of this process related to radiological emergencies. FEMA does, however, have an active program of earthquake preparedness which includes estimates of damage and casualties, planning for Federal response to a major earthquake, and assistance to State and local governments in their earthquake planning and preparedness activities. FEMA believes that these separate activities would complement each other in the event that a concurrent response to a major earthquake and a serious accident at a nuclear power plant was required.

5. Risk Perspectives

Recent PRAs (e.g., Zion, Indian Point) have indicated that very large earthquakes (much greater than the SSE) can dominate the risk from a nuclear power plant. Such earthquakes can cause massive plant damage leading to immediate offsite radiological hazards. In addition, massive offsite damage was assumed in these analyses which substantially degraded the emergency response.

Based upon the PRA results, the staff finds that for most earthquakes (including some earthquakes more severe than the SSE) the power plant would not be expected to pose an immediate offsite radiological hazard. For earthquakes which would cause plant damage leading to immediate offsite radiological hazards but for which there would be relatively minor offsite damage, emergency response capabilities around nuclear power plants would not be seriously affected. For earthquakes which cause more severe offsite damage, such as, for example, disabling a siren alerting system, the earthquake itself acts as an alerting system. For those risk dominant earthquakes which cause very severe damage to both the plant and the offsite area, emergency response would have marginal benefit because of its impairment by offsite damage. The expenditure of additional resources to cope with seismically caused offsite damage is of doubtful value considering the modest benefit in overall risk reduction which could be obtained.

6. Summary

Based on the preceding discussion the following summary points can be made:

a. In general, earthquakes up to and including the SSE are not expected to pose an immediate offsite radiological hazard.

b. Earthquakes beyond the SSE may cause plant damage and radioactive release under conditions where offsite damage impairs emergency response.

c. Further clarification or refinement of current requirements and guidance might reduce the impairment of emergency response indicated in b. above, but the value of such reduction is uncertain.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Attachment: As stated

cc: Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal
OGC
OPE
OCA

7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
CHAIRMAN

The Honorable Morris K. Udall
Chairman
Subcommittee on Energy and the Environment
Committee on Interior and Insular Affairs
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

On March 22, 1984, Dr. Henry Myers of your staff submitted to the NRC staff a list of comments and questions, derived from his review of NRC Inspection Report No. 50-275/83-37 and its draft. This list of comments and questions was revised by Dr. Myers and submitted to the Commission on March 26, 1984.

The staff has reviewed the March 22 and March 26, 1984 lists and notes that all items contained in the March 22, 1984 list are included as a subset of the March 26, 1984 list.

The staff has developed written responses to all of the items contained in the revised list of comments and questions, dated March 26, 1984. These are attached for your information.

I trust that the staff has been responsive to these inquiries.

~~Sincerely,~~

~~Nunzio J. Palladino~~
~~Chairman~~

OCA will sign off on this.
6/11/84

1. Question

"GC-1. A statement to the following effect is made repeatedly with respect to the Region V method used to inquire into the NSC findings: "The inspector's approach to resolving this issue was to assess the validity of the NSC finding and Pullman response, and evaluate the NRC findings for conformance with the specified Pullman program." (E.g. 83-37, Item 24.) This implies that there is a documented Pullman response to the NSC finding. (E.g. "The licensee conducted an audit of Pullman, during the period of April 2 through June 1, 1978, in response to the NSC audit and the Pullman response." See Draft 83-37, p. 37. This statement does not appear in the Final 83-37.) Where is the Pullman response? What interviews were conducted with PG&E, Pullman, and NSC past and present personnel in the course of preparing 83-37? How were such interviews documented? Where is the documentation?"

Answer

The Pullman response to the NSC audit report was submitted to PG&E by Pullman on April 11, 1978. This response was formally submitted to the Atomic Safety and Licensing Appeals Board by PG&E as an attachment to the Affidavit of Russell P. Wischow, dated September 21, 1983.

To better understand the response to this question, and those which follow, some background may be helpful.

On September 9, 1983, a filing to the Diablo Canyon Atomic Safety and Licensing Appeals Board was made by "Joint Intervenors" which included an audit of the Pullman Power Products (PPP) quality assurance program performed by Nuclear Services Corporation (NSC) and reported on October 24, 1977. The Pullman and Pacific Gas and Electric responses to the NSC audit were dated April 11, 1978 and June 16, 1978, respectively.

One of the significant aspects of the NSC audit was that it was almost exclusively limited to a records and paperwork review as opposed to a hardware or personnel performance review.

The NSC report contained many critical findings and drew far reaching conclusions. Both the Pullman and the PG&E responses took issue with many of the findings and conclusions of the NSC report. Our review of our own inspection reports over the years did not seem to corroborate many of the NSC conclusions.

The Region V Administrator elected to approach this inconsistency by examining, in depth, a large sample of the most significant NSC findings and the associated Pullman and PG&E responses to determine whether the NSC conclusions and findings could reasonably be drawn from the QA records which NSC reviewed. The Region V examination was limited to this sample and did not constitute a comprehensive reconstruction of the entire Pullman activity at Diablo Canyon. Consistent with this logic, our inspection did not rely, in any appreciable way, on personnel interviews; consequently no transcripts, tapes, etc., were made. Summary sheets do exist in our inspection file for three discussions involving five individuals. These discussions did not contribute appreciably to

the NRC conclusions. The discussions are referred to in inspection report 83-37 in paragraphs 20 and 25.

In summary, as discussed in several Commission meetings and in Report 83-37, the inspection of the NSC report was limited to a sampling of the significant NSC findings to determine if they could be reasonably supported by QA records. The NSC audit was almost exclusively based on review of QA records. The NRC examination was similarly focused. Except as noted in Report 83-37, we found no reasonable basis to expand the limits of the review.

2. Question

"GC-2. Inspection Report 83-37 refers to corrective actions taken in response to the NSC audit. It is unclear in certain instances as to whether the corrective actions were taken with respect to QA deficiencies that existed prior to the audit; e.g. to what extent did the corrective actions involve activity to insure that inadequate workmanship did not escape detection as a consequence of the QA deficiencies that existed prior to the NSC audit."

Answer

It is difficult to address this comment in detail in the absence of specific examples of concern. However, a few general comments addressing this item may be made.

The vast majority of NSC findings involved some kind of paperwork; ie: program, procedures or instructions rather than workmanship issues. In writing the report, and during the conduct of the inspection, the staff made every attempt to address and assess not only the adequacy of prospective work but also the degree of retroactive back-fitting that was appropriate. Therefore, the staff did consider the applicability of the NSC findings and Pullman responses to previous work.

The staff attempted to make clear that the majority of corrective actions taken as a result of the NSC report were programmatic improvements, or amplification of existing program descriptions, and did not necessarily condemn work performed prior to the improvement. In each case in the inspection report, the staff feels that, whenever a programmatic improvement was made subsequent to the NSC audit, the NRC made the finding that the program prior to the improvement was adequate or that no evidence was found to indicate that the program prior to the improvement resulted, or would likely have resulted, in an inadequate implementation condition. For example:

a. In paragraph 8 of the report the following conclusion is stated "The inspector found the QA program elements describing hanger package review and weld preheat were adequate and met the applicable code requirements," even though programmatic improvements were effected subsequent to the NSC audit.

b. In paragraph 15 of the report the following conclusion is stated "Furthermore, there is no evidence in the NSC, PG&E, and Pullman

corporate audits to suspect that any field changes made to pre-1977 documents and records impacted adversely on the quality of field construction," even though programmatic improvements were effected subsequent to the NSC audit.

In summary, during the conduct of the inspection the inspectors considered the effect on previous work during their examination of each item and programmatic improvement.

3. Question

"GC-3. Inspection Report 83-37 contains several references to the 90 day welders' log. Does the NRC have the log in its possession? If not, is it readily accessible? Where is it? What deficiencies exist in this log vis-a-vis the ASME code?"

Answer

The NRC does not have the 90 day welders' log in its possession. The log is readily accessible and stored in the Pullman QA records vault at the Diablo Canyon site.

The ASME Code does not mandate the use of a 90 day welders' log. The ASME Code only requires some sensible method of keeping track of welder qualification and welder activity. The 90 day welders' log is the mechanism adopted by Pullman, at Diablo Canyon, to track the welders, employed by Pullman, which were qualified during a particular time period. It has as its basis the original welder qualification record and the use of a particular process during a predefined previous period, as derived from the weld filler metal withdrawal slips. As discussed in paragraphs 9, 18, 19, 20 and 22 of Inspection Report 83-37, the staff found the Pullman system adequate in fulfilling the requirement of the ASME Code.

4. Question

"GC-4. Inspection Report 83-37 states in several places that Pullman practices were "consistent" with the ASME code. Does "consistent" mean "in compliance with"? Is it the NRC position that wherever "consistent" is used that it may be replaced by "in compliance with"?"

Answer

Yes, as used in inspection report 83-37.

5. Question

"GC-5. There is no indication of Region V having sought the views of NSC either to elaborate on the 1977 findings or to comment on the findings and conclusions of the Region V inquiry."

Answer

Region V has actively sought the views of NSC (now named Quadrex).

On two occasions the staff sought the views of the team leader (who is no longer associated with Quadrex) and of the President of Quadrex by telephone. In both cases they could not recall details. The details of these calls are contained in the transcripts of the Commission meetings of March 26, 1984 and April 13, 1984.

Further, Region V formally requested on April 4, 1984 that Quadrex review the NRC Inspection Report (Nos. 50-275/83-37 and 50-323/83-25) and extended the NRC invitation to appear at the Commission meeting on April 13, 1984.

The President of Quadrex responded by letter dated April 9, 1984 and indicated that he didn't believe Quadrex could add substantive information regarding the differences in the audits at this time.

The Commission again requested Quadrex to meet with them in a letter dated May 18, 1984, and they have agreed to meet.

6. Question

"GC-6. Page 3 of the draft states a sample of 25 stainless steel welds were sampled for delta ferrite and that 100 radiographs were selected to verify field weld and inspection review adequacy. What is the basis for selecting these welds? On what dates were these welds produced? Did these welds represent an adequate statistical sample?"

Answer

The basis for the sampling done was as stated in the report (83-37) on page 3 ..."to provide an independent feel for the Pullman work rather than solely relying on information provided by licensee records." In the instances cited (dealt with in paragraphs 25 and 33 of report 83-37) the inspector's conclusions were not dependent on the independent sampling. The sampling was not meant to be, nor was it advertised to be, statistically rigorous but was as stated in paragraph 25 "an additional check...."

7. Question

"Criterion I, NSC Audit Finding 3. (Final p.3, Draft, p.2-5.):

Did the fact of QA personnel writing and approving Engineering Specifications, performing welding engineering functions; and approving welding engineering changes constitute a violation of Appendix B requirements?"

Answer

No.

8. Question

"Criterion II, NSC Audit Finding 4. (Final p.4-5, Draft, p.2-5.):

Is it the NRC conclusion that upper management performed scheduled reviews of nonconformance reports, personnel qualifications, and corrective actions as required by NRC regulations for the time periods addressed by the NSC audit? Note handwritten notation in draft report: "In conclusion, factual records do not support the NSC finding." The corresponding statement in the final report is: "The inspector concludes the historical records of corporate management audits do provide evidence that reviews of nonconformance reports, personnel qualifications and corrective actions were performed." Note comment in final report: "In addition, Pullman Power Products has since proved programmatic improvements..." etc. What was the program prior to the improvements? What was it after the improvements were instituted?"

Answer

Yes, it is the NRC staff's conclusion that upper management did perform periodic reviews of nonconformance reports, personnel qualifications, and corrective actions as required by NRC regulations for the time periods addressed by the NSC audit. This was stated in paragraph 6 of inspection report 83-37.

It is important to keep in focus the purpose of the NRC inspection. As stated previously in answer to GC-1, that purpose did not include a diagnostic evaluation of the entire Pullman QA history. Accordingly, we did not compile a description of the Pullman program for each point of time in its evolution. In addition, the NRC inspection did not identify anything which would indicate a need for such a total reconstruction.

9. Question

"Criterion V, NSC Audit Finding 1. (Final p. 5, Draft, p. 39-40.):

NSC stated: "There is no requirement that activities affecting quality shall be prescribed by documented instructions, procedures, and drawings." Region V states, apparently in reference to fabrication of piping assemblies and erection of pipe in the plant, that KFP-8 established appropriate instructions and procedures. Region V seems to imply that KFPS-7 established procedures for pipe supports. KFPS-7, however, was not issued until December 1973. Were the QA procedures for installation of pipe procedures prescribed by documented instruction, procedures etc. prior to December 1973? Moreover, the draft states that ESD-264, dated 9/15/78, provided a specific procedure "to implement precisely the QA program elements of KFP-8 and KFPS-7." The latter statement does not appear in the final report. In that the specific procedures for implementing "precisely the QA program elements of KFP-8 and KFPS-7" were apparently not promulgated until only September 15, 1978 what is the basis for assurance that KFP-8 and KFPS-7 were adequately implemented prior to September 1978."

Answer

A brief history of the Pullman QA program applicable to pipe supports would be helpful here. The first pipe support work began during August 1971 with work begun on the first non-safety related pipe support (August

5, 1971) and safety related pipe support (August 16, 1971). As a result of a PG&E QA audit, performed during late 1973, it was identified that Pullman did not have a QA program covering the installation of pipe supports and that the QA program for the installation of pressure boundary piping was not fully applicable to pipe support work. A stop work order was issued on pipe hanger/rupture restraint work until an approved QA program covering pipe support/rupture restraint work was implemented.

As corrective action Pullman procedure KFPS-7 was issued on December 3, 1973 establishing and implementing a pipe support QA program for process planning and control. In addition, a Pullman Discrepancy Report (Nonconformance Report) was issued on February 11, 1974. This Discrepancy Report recognized that pipe support work was performed prior to establishing process planning and control. As corrective action all Class 1 pipe supports installed without process control were identified, reinspected and inspection findings resolved.

In the inspector's judgement, procedures KFP-8 and KFPS-7 were entirely satisfactory and met NRC requirements. This is documented in NRC Inspection Report 50-275/83-37, paragraph 7 which states, in part, that "The inspector concludes the program elements of KFP-8 and KFPS-7 did establish that documented instructions and procedures were required to be prescribed for control of Pullman's quality related construction activities." The establishment of ESD-264 subsequent to the NSC audit provided a programmatic improvement to an already acceptable system in this area, in that the details of process sheet completion were more precisely defined and prescribed by the ESD-264.

In addition, the NRC has contracted with Lawrence Livermore National Laboratory (LLNL) to provide additional inspection services, in the area of pipe support inspection, to supplement the regional effort. The laboratory inspectors have already examined a sizeable sample of Unit 1 pipe supports and found a very low discrepancy rate on accepted pipe supports. For example, the NRC staff and LLNL inspected about 550 safety related pipe supports, out of a total population of about 4300 modified supports, and identified only 5 items of noncompliance. The results of the laboratory inspections provide additional assurance regarding pipe support acceptability. Therefore, the staff feels that the licensee and Pullman have effected a satisfactory pipe support installation program.

10. Question

"Criterion V, NSC Audit Finding 2. (Final p. 5-6, Draft, p. 40-41):

NSC states that hanger package review was not described in procedures. Region V states that hanger package review was described in KFPS-2 dated December 3, 1973 and that supplementary requirements were incorporated into ESD-254 dated December 30, 1977. What was the basis for reviews conducted prior to December 3, 1973? The draft, but not the final report, states that ESD-253 provided additional detailed information concerning hanger drawing controls. What is the date of ESD-253? Is it NRC's position that hanger package review was described in a manner that

complied with the Appendix B requirements for all periods covered by the NSC audit?

NSC states that other activities not described in procedures included preheating for welding, use of Note-O-Grams, use of Rejection Notices, and maintenance of Field Quality Inspector Daily Logs. Is it the NRC's position that all such activities were described in procedures in a manner that complied with the Appendix B requirements for all periods covered by the NSC audit?"

Answer

Refer to the answer, provided above, to Criterion V, NSC Audit Finding 1 regarding pipe support QA program history. The Pullman Discrepancy Report was resolved, as indicated previously in answer to the question regarding Criterion 5, NSC Audit Finding 1.

It is the NRC staff conclusion that hanger package review was described in a manner that complied with Appendix B requirements during the time periods when hanger package review activities were in progress.

The staff found that preheating was appropriately prescribed on the welding procedure specification (see paragraph 28, page 26, of inspection report 50-275/83-37). It is the NRC staff's conclusion that other activities (such as use of Note-O-Grams, Rejection Notices, and maintenance of Field Quality Inspector Daily Logs) were not required to be prescribed and controlled by written procedures, as indicated on page 6 of inspection report 50-275/83-37.

11. Question

"Criterion V, NSC Audit Finding 3. (Final p. 6-8, Draft, p. 41-42):

NSC found that isometric package review was not sufficiently described. The draft of 83-37 states that "Field procedure ESD-254 (issued 5/6/75) appears to provide an adequate outline guide for review of isometric drawing packages." The final report adds that May 6, 1975 was the earliest date that could be found for ESD-254 and that while most piping installations had been completed prior to May 1975, the inspector found that the final complete document review of isometric drawing packages were performed after ESD-254 was in effect.

Note that draft states that post heat treatment requirements are prescribed in ESD-218. The draft does not indicate that ESD-218 was issued in October 1977. The final report states that post weld heat treatment requirements "were always prescribed by weld procedure specifications." The final report does not refer to specific procedures in effect prior to ESD-218. What is the basis for the conclusion that post weld heat treatment requirements were in compliance with Appendix B prior to issuance of ESD-218 in October 1977?

Is it the NRC staff position that, in the time period encompassed by the NSC audit, non-conformance reporting requirements complied with the requirements of Appendix B?

83-37 states that the "internal audit program, implemented by on-site personnel, (prior to 1978) was determined to be of marginal quality, a redundant program of comprehensive quality was performed concurrently." The redundant program appears to have been one "directed and conducted by corporate management personnel." Did the redundant program find that the internal audit program, implemented by on-site personnel to be of "marginal quality?" Did the corporate audits encompass involve review of weld and welder quality, and Q.A. programs applied to weld and welder quality? On what dates were the corporate audits conducted and what were their findings?

Region V concluded that, notwithstanding the deficiencies in the internal audit program and the failure of the corporate audit program to discover these deficiencies in a timely manner (e.g. they appear not to have been corrected until 1978), 83-37 concludes that "no major breakdown of the Quality Assurance program had occurred, nor had any significant problems gone undetected, due to deficiencies identified with the internal auditing program." Is it the NRC position that the NSC findings do not indicate that "significant problems (had) gone undetected" until 1974 and, to a lesser extent, between 1974 and 1977?"

Answer

The basis for the NRC staff's conclusion that post weld heat treatment requirements were in compliance with Appendix B prior to issuance of ESD-218 in October 1977 is as stated in the NRC inspection report, "Appropriate post weld heat treatment requirements were always prescribed by welding procedure specifications."

It is the NRC staff's conclusion that nonconformance reporting requirements complied with the requirements of Appendix B. Pullman procedure KFP-10 (issued March 19, 1971) did provide adequate instructions to establish nonconformance reporting requirements in compliance with Appendix B.

The statement that the internal audit program (i.e. those conducted by on site people) was of marginal quality was the opinion of the inspector. His basis was that the breadth and frequency of the internal audits were not entirely consistent with today's guidance. The inspector went on to say that the audits conducted by the corporate people compensated for this and the program as a whole met Appendix B requirements. As far as the inspector can remember the frequency and breadth of the internal audits was not commented upon by the corporate audits. Since the inspector concluded that the Pullman audit program as a whole met all requirements, a detailed catalog of all findings, dates, and resolutions was not made.

As stated in the last paragraph of item 9 in Inspection Report 83-37, it is the staff's position that the Pullman audit program as a whole met the requirements of Appendix B.

12. Question

"Criterion VI, NSC Audit Finding 9a. (Final p. 8-9, Draft, p. 54-55):

NSC stated that a Process Sheet for Isometric 2-14-77 was changed approximately 19 months after the work was done. Region V states that the process sheet (which Region V states should have been in reference to 2-14-47) that "no evidence could be found to indicate that there had been an attempt to alter the dates or signatures on either or both these documents." Does NRC believe the NSC finding to have been in error? How does NSC explain the apparent discrepancy between its findings and those of the NRC? How does Region V know that the sheets it examined were in fact the same sheets examined by NSC?"

Answer

Yes, the NRC staff believes that the NSC finding was in error. In particular, isometric package 2-14-77 did not fit the time frame identified in the NSC finding.

The efforts of the NRC to solicit NSC review and comment on the NRC inspection report have been dealt with in answering question GC-5. These are precisely the types of details that neither the former auditor nor the President of Quadrex could recall.

The NRC staff examined the available records for the referenced isometric packages and found no basis to conclude that the records had been altered since the NSC audit.

13. Question

"Criterion VI, NSC Audit Finding 9b. (Final p. 9-11. Draft, p. 61-64):

NSC concluded that FW-1673 was performed without normal controls. Region V stated in the draft, that "although it was not the usual practice" the weld was carried out in accordance with the then existing design change control system. Is it the NRC position that this departure from the usual practice did not violate the NRC's QA requirements? What is NSC's response to the NRC finding?"

Answer

It is the NRC staff's conclusion that the referenced departure from usual practice was adequately controlled and did not violate the NRC QA requirements.

14. Question

"Criterion VI, NSC Audit Finding 10. (Final p. 13-15. Draft, p. 38-39):

NSC found that no procedure or requirement prohibits changing or alteration of records and documents necessary to track work. Region V stated that prior to 1977, insufficient requirements existed to control the changing or alteration of quality records and documents. Region V also concluded in the final report that neither NSC, nor NRC nor Pullman audits had "identified any unapproved technical changes or other substantive changes which would have adversely affected construction quality." What was done to reach this conclusion? Note that the

conclusion as stated in the draft was less firm; it said: "Pullman's corrective action is complete and appears to be effective. Previous inadequacy of management policy or written instructions in this area is not considered to have resulted in any adverse impact on quality related activities."

Answer

As stated in Inspection Report 83-37, the staff examined the results of Pullman audits, the NSC audit and PG&E audits and related QA records. In addition, the NRC inspection staff was aware of this NSC finding and the documents examined during the inspection were reviewed with particular attention given to unapproved or substantial technical changes.

15. Question

"Criterion VIII, NSC Audit Finding 12. (Final p. 14-15. Draft, p. 59)

NSC stated that ESD-223 did not give adequate instructions for the identification and control of Class I pipe supports. Region V reviewed ESD-223 and stated that specific revisions were dated November 11, 1975 and May 25, 1976. Region V stated that the procedure revisions contained adequate QA/QC instructions for the control and identification of Class I pipe supports. Is it the NRC position that the instructions were adequate prior to the 1975 and 1976 revisions? What is the basis for confidence in the adequacy of instructions prior to the 1975 and 1976 revisions in ESD-223?"

Answer

The Region considers that instructions regarding the control and identification of pipe supports were adequate prior to the referenced revisions of ESD-223. The basis for this conclusion was discussed in the answer to the question regarding Criterion V, NSC Audit Finding 1.

16. Question

"Criterion IX, NSC Audit Finding 10b. (Final p. 17-18, Draft p. 6-7.):

NSC found that from August 1972 through December 1972 a ninety day Welders' Log was not maintained nor did a Weekly Qualified Welders' List exist for that time. Region V agreed there was no weekly log but that a 90 day log did exist. Did Region V seek to determine the reason for the discrepant findings?"

Answer

Yes, the NRC did determine the source of the discrepancy. The void in the 90 day log had been reconstructed by Pullman subsequent to the NSC audit by using the weld rod withdrawal slips for the period in question. It should be noted that the 90 day log is normally made up using these weld rod withdrawal slips. This seeming discrepancy was dealt with in SSER-22 and was discussed in the March 26, 1984 Commission meeting.

17. Question

"Criterion IX, NSC Audit Finding 10c. (Final p. 18-19, Draft p. 7-8):

NSC found that the Ninety Day Welders' Log was not sufficiently detailed to determine if the welder was qualified to perform certain procedures. The draft stated that the "90 day qualified welders' log was sufficiently detailed to determine whether a welder was qualified to perform certain procedures." The draft did not state that the 90 day welders' log complied with applicable code requirements. The final report, states, depending upon the manner in which a sentence is interpreted, either that the log complied with the code or, alternatively, that the welder complied with the code requirements. Is it the NRC position that for the period covered by the NSC audit that the 90 day welders' log was in compliance with code requirements?

Note also that Region V based its conclusion in part upon discussions with the former Authorized Inspector. Does a record of these discussions exist? What was the substance of such discussions?"

Answer

The ASME Code does not require a 90 day log - only that some reliable method of determining welder activity be maintained. It is the staff's position that the 90 day log based on the weld rod withdrawal slips constitutes a reasonable method of complying with this requirement.

The discussion with the former Authorized Inspector was not relied upon in reaching the conclusions presented in the NRC inspection report because the former Authorized Inspector was, at the time of interview, an employee of Pullman. The discussion results were merely considered another data point recognizing that, while the information was given by an industry professional, the information may be of dubious value. The inspector relied instead on the results of his examination of the 90 day welders' logs. A record of that discussion exists in the Region V files as discussed in answer to GC-1.

The substance of the former Authorized Inspector's information was that, in his opinion, Pullman had adequately tracked and documented welder qualifications and had used weld rod withdrawal sheets to verify whether a welder had used a particular process as a supplement to the original welder qualification record. This is stated in report 83-37.

18. Question

"Criterion IX, NSC Audit Finding 10d. (Final p. 19-20, Draft p. 9-10):

NSC states that no procedure stated what the Field Quality Assurance Inspector was to use as the primary means to determine welder qualification. Region V appears to agree that a procedure did not exist but that weld filler metal withdrawal sheets and welder qualification records were used to determine welder qualification and that this method satisfied code requirements? Is it the NRC staff position that, the absence of a specific procedure notwithstanding, the method used by

inspectors to ascertain welder qualifications complied with code requirements?"

Answer

Yes, the staff considers that the method historically used by Pullman (i.e., weld filler metal withdrawal sheets and welder qualification records) was sufficient and adequate to document and verify welder qualification, as required by the ASME B&PV Code, Section IX (refer to paragraphs 9 and 21 of inspection report 50-275/83-37).

19. Question

"Criterion IX, NSC Audit Finding 10h, 10i. (Final p. 22-23, Draft p. 13-16):

The draft focuses on question as to whether auditors' observations need be recorded on the "process sheet or the inspectors' daily work sheet." The draft does not indicate that the inspector examined the welder audit sheets. The final does state that the inspector examined welder audit sheets but does not indicate the period covered by the examination. The final version of 83-37 states in 10h that the welder audits were "a Pullman program requirement in excess of the ASME code requirements" and twice in 10i that the program requirements "appeared" in excess of code requirements. The DRAFT did not mention that the code did not require a welder audit.

Draft 10i (p. 15) says "...records of the 9/73 revision and 11/73 implemented procedure are not available." Final drops this part stating (p. 23) "The November 1973 revision apparently was issued and implemented beginning in November 1973. ...welder audit sheets indicate that the required welder audits were performed beginning November 1, 1973." The following statement appears in the draft but not the final: "The welder audit sheets examined indicate the ferrite control measurements were performed on welds by the auditors." Why was this statement dropped? Is the statement accurate? Was there a requirement to make ferrite control measurements?

What is the significance of failing to adhere to ESD-219 if the ASME code does not require welder audits?

Note following statement in draft does not appear in final: "Since the record of the 9/73 revision is not available, the inspector could not determine when the procedure was approved for implementation and, thus, was not able to corroborate the Pullman statement that the September 1973 revision was made to initiate the auditing of welders." The draft and final state that "the inspector was not able to corroborate the NSC statement that Pullman was in noncompliance with the procedure for about 23 months."

Is the staff's conclusion that neither Item 10h nor Item 10i were identifiable items of noncompliance or deviation rest on the assumption that welders' audits were not required by the ASME code?"

Answer

The statement regarding ferrite control measurements was dropped from the final because it did not contribute in any meaningful way to addressing the finding identified by NSC. The draft version did not maintain a focus on the issue addressed by the NSC finding addressed in paragraph 24 of NRC Inspection Report 50-275/83-37. The statement is, however, accurate.

Ferrite control measurements were included as one of several suggested inspections on the welder audit sheet. The welder audit program was structured around a sampling approach and applied to in process welding activities. There was no requirement that each welder audit performed should address each and every suggested inspection attribute identified on the welder audit sheet. For example, even though the same welder audit sheet format would have been used for both carbon steel and stainless steel welding activities, a measurement of ferrite level on a carbon steel weldment would be quite meaningless. The intent was that a welder audit should sample the suggested attributes with emphasis on those suggested attributes which could be meaningfully examined at the time of the welder audit performance.

The NRC staff did not assess the significance of failing to adhere to the welder audit program of ESD-219 because the NRC found that Pullman did acceptably implement the ESD-219 specified welder audit program.

The staff's conclusion that Pullman had acceptably implemented the welder audit program was the basis for the determination, in items 10h and 10i, that no items of noncompliance or deviations were identified. The fact that the Code does not require such a welder audit program has no bearing on the finding of acceptable implementation, but the Code was mentioned merely to provide additional perspective.

20. Question

"Criterion IX, NSC Audit Finding 10j. (Final p. 23-24, Draft p. 16-18):

Note change from draft which relied on the examination of 25 welds to find that "...there is a high probability that other stainless steel welds installed in the plant comply with delta-ferrite acceptance criteria." The final report cites a "random sample of 25 stainless steel welds" as "an additional check". Primary reliance for the final report's conclusion that "the inspector was not able to corroborate that Pullman was in noncompliance with this procedure requirement for 12 months" was based on the assumption that stainless steel welding did not begin until early 1973. If it is true that on-site stainless steel welding did not begin until 1973, what is the relevance of the examination of the 25 welds since the NSC finding applied to the pre 1973 period?

Is there a documented basis for the statement "Based on discussions with PG&E personnel it appears that stainless steel welding on site began in early 1973?"

Answer

Contrary to the characterization above, the staff's conclusion was not based on the assumption that stainless steel welding did not begin until 1973. The staff's conclusion was based on the fact that ESD-219 became effective in November 1973 yet the severin gauges were on site as early as December 1972.

21. Question

"Criterion IX, NSC Audit Finding 10k. (Final p. 24-25, Draft p. 120-121.):

The NSC finding that "Hangers are not welded in accordance with Pacific Gas & Electric Company requirements" was not confirmed. Did NSC err in observing that hangers were welded to structural steel on the wrong side of the bracket? What was Pullman's response to the NSC finding? Would NSC agree that an error of this kind would be made in the audit? Was an effort made to determine whether the hangers might have been modified following the audit?"

Answer

Yes, the staff believes that the NSC finding was in error. This conclusion is also stated in the staff response to this NSC finding (Reference: NRC Inspection Report No. 50-275/83-37, page 24).

The Pullman response states, in part, that, "Pullman inspection personnel have reviewed Hangers No. 2023-IV and 2039-2V and found that they were welded in accordance with customer drawings."

The efforts of the staff to solicit NSC review and comment on the NRC inspection report have been dealt with in answering a previous question (GC-5).

Yes, the staff examined the available records for the referenced hangers and found no evidence to conclude that the hangers had been modified or reworked after the NSC audit.

22. Question

"Criterion IX, NSC Audit Finding 10n. (Final p. 26-27, Draft p. 20-21.)

NSC found that there was no procedure for preheating weld joints. The draft report (p. 21) states that a series of weld procedure specifications was examined and that each contained "an adequate definition of preheat, postweld heat treatment and interpass temperatures." The draft also states that "ESD-218 (Postweld Heat and Preheat Treatment Procedure) was revised 12/30/77 to prescribe preheat requirements and indicate preheat applicability." An adjacent handwritten comment (p. 21) asks "How about b/f 12/30/77?" Does this mean that the procedures were or were not adequate prior to 12/30/77?

The final report (p. 27) contains an additional statement to the effect that prior to early 1978, compliance with the preheat requirement was dependent upon the welder's knowledge etc. Did the procedure described

in the second paragraph on p. 27 comply with Appendix B? What was the basis for the added language? Was there discussion with Pullman or PG&E on this point beyond that which occurred during the inspection that ended on December 9, 1983?

The penultimate paragraph on this item states "while no separate and specific procedure for preheating of weld joints existed prior to December 30, 1977, preheating requirements were adequately prescribed by the welding procedure specifications and documented by signature on the welding block of the process sheet, which specified the applicable welding procedure." Was this in compliance with Appendix B?"

Answer

The handwritten comment does not mean that preheat procedures were inadequate prior to December 30, 1977. The handwritten comment was made by the Region V Administrator to instruct the inspector to make clear the situation that existed prior to December 30, 1977.

The finding that preheating was adequately prescribed is documented in paragraph 29 of the NRC inspection report which states "The inspector concludes that, while no separate and specific procedure for preheating of weld joints existed prior to December 30, 1977, preheating requirements were adequately prescribed by the welding procedure specifications and documented by signature on the welding block of the process sheet, which specified the applicable welding procedure."

This was in compliance with Appendix B, hence the finding in paragraph 29 of the NRC inspection report that "No items of noncompliance or deviations were identified."

The added language of the second paragraph on page 27 was to clarify the preheat prescription, implementation and documentation process. To the best of the inspector's recollection, there was no further discussion with Pullman or PG&E on this point beyond that which occurred during the inspection, which ended on December 9, 1983.

23. Question

"Criterion IX, NSC Audit Finding 10o. (Final p. 27-30. Draft p. 21-26.):

NSC stated that the initial results of welding auditing (from November 5, 1973 to February 1974) indicated the existence of 7 problems which, if they did exist, raised question about weld quality. NSC concluded on the basis of a review of these audits that "...there is no confidence that welding done prior to 1974 was performed in accordance with welding specification requirements."

The NRC inspector said he had "critically examined the records of welder audits performed between November 1, 1973 and April 1, 1974." On the basis of an examination of 183 audit records from this period, the NRC inspector concluded that the "aggregate of problem areas is not so pervasive such that support can be given to the NSC conclusion" that

there is no confidence that pre-1974 welding had been performed in accord with requirements.

83-37 states that "It is important to recognize that none of these were NSC findings, but were instead findings of the Pullman welder audit program, which was designed to detect program weaknesses and provide prompt corrective action during the early phases of site welding activity." The problem is that the welder audits referred to by the Region V inspector (which were found by Region V under 10h and 10i above to be beyond what was required by the code) were not initiated until November 1973. In addition, the NSC audit states that its findings were based on a review of Pullman's audits conducted in the period "from November 5, 1973 to February, 1974). Therefore, how could the audit program, upon which Region V relies "detect welding program weaknesses and provide prompt corrective action during early phases of site welding" if the audit program was not initiated until November 1974?

In sum, the NSC finding, based on findings obtained from a review of audits conducted after November 1, 1973, was that "...there is no confidence that welding done prior to early 1974 was done in accordance with welding specifications." Region V, on the other hand, based on a review of audit reports prepared during essentially the same period as the reports reviewed by NSC [and ignoring the above noted finding (final, p. 23) that "the required welder audits were performed beginning November 1, 1973"] concludes "no support can be given the [above quoted] NSC conclusion." Region V does not deal with neither (A) the fact of there having been no welder audits prior to November 1973 nor (B) the question of whether the types of deficiency discovered in the initial audits existed in prior years.

[At the March 19 Commission meeting, statements were apparently made to the effect that audits other than those that pursuant to the ESD-219 program were conducted prior to November 1973. If so, were the findings of such audits discussed in 83-37? Where? Why were these findings, rather than those in the post November 1973 period, used to refute the NSC findings?]"

Answer

Again, as discussed in our response to GC-1, the purpose of the inspection was to determine if the NSC conclusions could reasonably be drawn from the QA record they reveiwed. We did not undertake to reconstruct the entire quality history of the Pullman activity.

Dr. Myers correctly points out that both NSC and the staff looked at the record of welder audits from November 1973 through Spring of 1974. However, as stated in paragraph 30 of report 83-37, the NRC staff did not feel that many of the NSC conclusions could reasonably be drawn from the QA records they reviewed. Even though the welder audits did not start until November 1973, the Pullman internal audits and corporate audits (previously discussed) routinely examined in-process welding and were implemented from the beginning of work. As stated previously we found the basic audit program to be satisfactory and in compliance with Appendix B.

As a result of discussion at the March 26 Commission meeting the staff reviewed the Pullman audits and the PG&E audits done in the pre 1974 time period in more detail. The results are reported in Inspection Report 84-16 and confirm that the audit program met the requirements of Appendix B.

We have not seen this report

Many of the following questions deal with very specific hardware items during the pre November 1973 period. It is important to keep the inspection purpose in perspective. It was not the NRC staff's purpose to perform a detailed evaluation of hardware in the plant. Rather, the purpose was to assess whether the licensee and his contractors were doing a responsible job of controlling construction and assuring the adequacy of hardware evaluation. As stated previously, the basic system of audits applied to Pullman welding from the start of work was satisfactory and met Appendix B. The addition of the Pullman welder audit program in November 1973 was beneficial and improved the Pullman system of audits.

24. Question

"Criterion IX, NSC Audit Finding 10o, Item 1. (Final p. 28. Draft p. 23.):

The draft, without citing documents, appears to rely on the gas flow being "near the 20 cfm requirement" for its conclusion that defective welds might have resulted from inadequate shielding and purging. The draft states that excessively low flow rates would have been manifest in unacceptable porosity which would have been detected by NDE; the draft does not indicate the extent to which unacceptable porosity was found. The final does not state that the flow was near the 20 cfm requirement; it does state that "The vast majority of safety related stainless steel welds were radiographically examined and the film was reviewed and accepted by a qualified interpreter for code compliance." How many welds were not radiographically examined? How many were examined? Of those that were examined, what percentage exhibited excessive porosity? What was done to determine whether shielding and purging deficiencies that might have existed prior to the first welder audit? What was done to correct for such deficiencies?"

Answer

The NRC staff did not consider the specific deficiencies found in the purging and shielding area by the welder audits to be of much technical significance. Consequently, no effort was made to reconstruct the nondestructive examination history of the welds in question nor was it considered worthwhile to do so. The problem here was one normally encountered in a purge gas distribution system when welders hook into or drop off of it during the course of the work day. Pressure and flow variations are introduced in the various distribution outlets. Weld quality is not very sensitive to purge gas flow. As long as the purge gas flows are maintained at all, no ASME code violations are involved. The inspector found in his analysis of the situation that flows were maintained and in most all cases were reasonably near 20 cfm. As noted in our report, the critical welds in question would have been given

radiography and all pressure boundary welds are given a hydrostatic test which provides additional assurance of weld quality.

Prior to the initiation of welder audits in November 1973, in process welding was audited by the site and corporate audits as discussed in paragraph 23.

25. Question

"Criterion IX, NSC Audit Finding 10o, Item 2. (Final p. 28. Draft p. 23-24.)

What is the significance of 14 out of 183 audits identifying that welders did not have tempil sticks? Region V states that in each case that a welder was found not to have a tempil stick, one was provided. What was done to determine the extent to which welders did not have tempil sticks prior to November 1973? Does the code allow interpass temperature requirements to be met by the resumption of welding delayed until the welder "can touch the weld?" The draft, but not the final, states that "Tempil sticks were used by welders in the vast majority of cases." What constitutes a "vast majority?" What was done to determine whether there was a tempil stick problem prior to November 1973?"

Answer

The significance is that some fraction of the welders were not complying with their own internal procedures (i.e. having a Tempil stick in their possession). The Pullman audits were effective in identifying this. This had no real technical significance since no preheat or interpass temperature violations of the ASME Code were identified. The ASME code does not mandate the use of tempil sticks. Use of touch to ensure interpass limits were not exceeded would be allowed by Code. The inspector saw no reason to pursue the tempil stick issue back prior to November 1973.

26. Question

"Criterion IX, NSC Audit Finding 10o, Item 3. (Final p. 28. Draft p. 24.)

The draft states that in 4 out of 183 instances where amperages were not within the welding procedure specification limit, the welder corrected his amperage setting. The draft stopped there. The final adds statements to the effect that defects resulting from improper amperages would be found during inspections. The final also adds a statement that "...amperage is not an essential variable specified by the ASME code...." Does this mean that a welds produced with improper amperages could still be in compliance with the code? What about improper amperages that might have been used prior to November 1973?"

Answer

Yes, welds produced with improper amperages could still be in compliance with the code. Amperage is but one variable used by the welding engineers to obtain the proper welding heat input. Other variables are

voltage and travel speed. Each of these variables are normally specified in a welding procedure specification using fairly wide limits and a change in one variable is usually compensated for, by a journeyman welder, by a slight change in another.

In the judgement of the inspector, it did not appear to be a necessary or particularly fruitful exercise to attempt to assemble amperage data for the period prior to November, 1973.

27. Question

"Criterion IX, NSC Audit Finding 10o. Item 4. (Final p. 28-29. Draft p. 24.)

83-37 states concludes the "vast majority" of welders used welding procedures and knew where to obtain them. Those that did not have them were told to get them. Those that did not know where they could be found were given "an explanation of the location from where they could be obtained." This finding was based on welder audits conducted after November 1973. What is Region V's position with regard to those not members of the "great majority?" What is Region V's position with regard to the availability of procedures and welders' knowledge of where procedures could be obtained in the period prior to November 1973?"

Answer

The NRC staff position is that all welders should know where the procedures are and the Pullman audits properly identified and corrected the situation. The inspector noted that the welding auditors did not identify defective welding as a result of their original findings in this area.

Region V had no reason to believe that the situation was any worse prior to November 1973 and, thus, saw no reason to pursue this issue any further.

28. Question

"Criterion IX, NSC Audit Finding 10o, Item 5. (Final p. 29. Draft p. 24.)

NSC found that the oxygen analyzer was not available or not operative. Region V concludes that only one of the 183 audits reviewed "indicated a problem with the oxygen analyzer." What was done by Region V to determine the basis for the significant discrepancy between its finding and those of NSC? What documentation was examined?"

Answer

The staff's rationale for its conclusion is stated in NRC Inspection Report No. 83-37, paragraph 30, page 29. As stated previously in the answer to question GC-5 the staff has attempted to solicit review and comment from NSC on NRC Inspection Report No. 83-37.

29. Question

"Criterion IX, NSC Audit Finding 10o, Item 6. (Final p. 29. Draft p. 25.)

NSC concluded that "Oven rod temperature was not monitored by the welders." 83-37 states that 14 of 183 audits identified instances where rod oven temperatures were lower than those which were required. A note on the draft states: "With this many audit findings the rod oven temperature must have been too low much of the time." The NRC concludes that "The NSC finding that rod oven temperature was not monitored by the welders is not supported by the audits, although isolated instances of ovens being below temperature were identified by the audits." Is it correct that 14 out of 183 constitutes "isolated instances?" What is the NRC position with regard to temperature control during the period prior to the initial welders' audit?"

Answer

As stated in the report (page 29), the technical significance of this finding is minimal. Further, there was no code violation associated with the finding. The audit finding did point out that welders should have been more alert in monitoring their rod ovens. Pullman's audits recognized this condition and took corrective action.

Region V found no reason to believe the situation was any different prior to November 1973 and saw no reason to pursue this issue further.

30. Question

"Criterion IX, NSC Audit Finding 10o, Item 7. (Final p. 29. Draft p. 25.)

The NSC stated that "Many welders did not understand their duties and responsibilities." Region V states that "Of the 183 audits received, five welder audits indicated that the welder in question did not understand their (sic) duties and responsibilities." The final, but not the draft, contains a sentence: "The NRC considers that the reason these welder audits were done was to identify such instances and provide corrective action." The draft and final report state that "In each case the welder was reinstructed by the QA inspector auditing the welding...." 83-37 does not address the pre-November 1973 period during which audits were not conducted. What mechanism existed prior to November 1973 to identify situations where welders did not understand their duties and responsibilities? What is the basis for assurance that, prior to November 1973, welders understood their duties and responsibilities?"

Answer

As discussed in paragraph 23, there was an active audit program in existence prior to November 1973 which routinely examined in process welding.

31. Question

"Criterion X, NSC Audit Finding 5,6. (Final p. 30-31. Draft p. 26-28.)

NSC found that the inspection process is generally inauditable on the ground that there were acceptance signatures that did not permit a determination of whether the individual inspection requirements were fulfilled. Region V stated that acceptance process sheets identified the procedures necessary to perform a particular inspection and the acceptance signatures were sufficient documentation of these procedures having been followed. The final report, but not the draft, states that this practice was "in accordance with standard industry practice, and in compliance with ASME code requirements...." Was this practice employed at other plants under construction during this period? Did NSC consider this practice in compliance with the ASME code? What was Pullman's response?"

Answer

The staff has observed similar documentation practice at other nuclear plants under construction in that each inspection attribute is not contained as a separate line item on the work process traveler. The staff feels that to list each attribute in detail would unnecessarily complicate the process traveler system. The staff's conclusion that the Pullman practice was in compliance with ASME Code requirements is still valid.

The NSC audit did not address the issue of ASME Code compliance here.

Pullman stated that their program complied with the ASME Code and regulatory requirements and that their program was acceptable and auditable.

32. Question

"Criterion X, NSC Audit Finding 7. (Final p. 31. Draft p. 28-29.)

NSC found that a "large number of welds...were accepted for visual examination and thereafter accepted on surface NDE inspection.... Visual examination of those welds indicates that the surface is not acceptable for performance of surface NDE inspection." The final report, but not the draft, states "The inspector concludes that the NSC finding (that the surface of the welds was not acceptable for surface NDE inspection) was in error." What is the basis for these contradictory conclusions? Did NSC and NRC inspect the same surfaces? What evidence exists to demonstrate that remedial work was not carried out in the time between the NSC and NRC inspections?"

Answer

The staff's basis and rationale for its conclusion is stated in NRC Inspection Report No. 83-37, paragraph 32 - page 31. The staff cannot state what NSC's response to the NRC finding would be, though as stated previously the NRC has offered NSC the opportunity to review and comment on NRC Inspection Report No. 83-37.

The staff examined the weld surface of the welds contained in the referenced isometric package as indicated in the NSC Audit Report.

The staff examined the available records for the reference welds and found no document referencing rework or repair after or since the NSC audit.

33. Question

"Criterion X, NSC Audit Finding 9. (Final p. 31-32. Draft p. 28-29.)

The NRC disagreed with the NSC implied finding that inking "R1" onto a radiograph was not permitted by the code. NRC also disagreed with NSC that FW-83 contained a surface defect "that is questionable for acceptance under visual standards." Does NSC agree with NRC's findings?"

Answer

Refer to the answer to question GC-5 regarding NSCs views.

34. Question

"Criterion X, NSC Audit Finding 10a. (Final p. 32-33. Draft p. 28-29.)

NSC found that "Records of welder qualification prior to 1972 are not available." Thus, the inspector was not able to verify the validity of the Pullman response to the NSC audit finding." Region V found that 20 welders were qualified prior to 1972. Region V also found that the 90 day qualified welders log was started "at the beginning of 1972." The draft report, but not the final, states: "The inspector was not able to determine when the first production welding was performed or on what system the first weld was accomplished." The final report, but not the draft states: "The inspector concludes that records of welder qualification prior to 1972 were available and in acceptable order."

Does Region V now know when the first production welding was performed and on what system? In light of NRC having found records for 20 welders, has NSC been asked why they found that records were not available? Does Region V believe that the welder qualification records for this period are complete? How many active welders are shown on the initial 90 day qualified welders log? Is this log consistent with Region V's findings regarding the 20 welders?"

Answer

Note: This should have been titled Criterion IX, NSC Finding 10a, in the final report.

Yes, the first class 1 production pipe weld was performed on December 28, 1971, on the Component Cooling Water System.

The issue of NSCs views was addressed in the answer to question GC-5.

Yes, the staff believes that the records for the referenced period are complete. As stated in NRC Inspection Report No. 83-37, paragraph 34, page 33, "The inspector concludes that records of welder qualification prior to 1972 were available and were in acceptable order."

The NRC did not catalog and itemize the welders shown on the first 90 day log and saw no particular reason why this would have been necessary or desirable.

35. Question

"Criterion XIII, NSC Audit Finding 5. (Final p. 35-38. Draft p. 31-37.)

Note that last paragraph on Draft, p. 37 was dropped. The dropped paragraph mentions a PG&E audit of Pullman which identified programmatic and hardware discrepancies? What is the nature of these discrepancies? Was there a requirement that they be reported to the NRC? Were they reported to the NRC? Does Region V have a basis for concluding that appropriate corrective actions were taken. Note the reference to the inspector having discussed this matter with Pullman and PG&E personnel. What was the nature of these discussions? Do written summaries of these discussions exist? Is it Region V's position that for the entire period covered by the NSC audit, Pullman was in compliance with applicable NRC requirements pertaining to handling procedures?"

Answer

The nature of the discrepancies identified in the PG&E audit of Pullman are described in attachment 5 to the Affidavit of Russell P. Wischow, dated September 21, 1983, to the ASLAB.

These discrepancies were not reported to the NRC. The reporting requirements are described in 10 CFR 50.55(e). It is the NRC staff's conclusion that none of the identified discrepancies met the threshold defined in the regulation; thus, reporting these discrepancies to the NRC was not considered necessary.

As indicated on page 37 of the draft inspection report referenced by Dr. Myers, the inspector selectively examined the discrepancy resolutions and based upon those examinations obtained assurance that appropriate corrective actions were taken. This is documented on page 3 of the draft inspection report. The whole subject was dropped from the final report because it was irrelevant to the stated purpose of the inspection. During the course of this selective examination discussions were held with PG&E personnel regarding the location of the discrepancy reports and the corrective actions. These discussions were essentially an attempt to obtain the necessary documents for review. These discussions were not documented on written summaries because they contributed little to the overall NRC conclusion.

The NRC inspector did not attempt to reconstruct a history regarding compliance with handling procedures, nor apparently did NSC. The NSC statement was that "handling procedures do not exist" and the NRC's examination addressed whether or not such procedures did exist. The

inspector found that "appropriate and adequate handling requirements were in place."

36. Question

"Criterion XIV, NSC Audit Finding 1. (Final p. 38-39. Draft p. 59-60.):

NSC stated that the Field Process Sheet was inadequate. Region V reached a contrary conclusion. What is the basis for the discrepant findings? Is it the NRC staff position that the Field Process Sheet adequately controlled and specified required work activities?"

Answer

The staff's rationale for reaching the conclusion that the Field Process Sheet was adequate is contained in NRC Inspection Report No. 83-37, paragraph 40, pages 38-39.

Yes, the staff states in NRC Inspection Report No. 83-37 that, "...The inspector concludes that the use of the field process sheet adequately controlled and specified required work activities."

37. Question

"Criterion XV. NSC Audit Report, p. 36:

NSC found, among other things, that "Systems that circumvent the nonconformance system have been established." This finding was not addressed in 83-37. What is the NRC's response to this finding?"

Answer

During the inspection planning process the staff read the NSC finding, the Pullman response and the PG&E response. The staff determined that this item did not meet the selection criteria of paragraph 4 of report 83-37.

38. Question

"Criterion XVI, NSC Audit Finding 2. (Final p. 39. Draft 60-61.):

NSC stated that it appeared that a corrective action system had not been operative. Region V cited examples where corrective actions had been taken in response to audits. Was NSC's reference to a corrective actions system intended to encompass corrective actions in response to nonconformance reports? Are the samples cited by Region V sufficient to demonstrate that the Pullman did have an operative corrective action system?"

Answer

Refer to the answer to question GC-5.

Yes, the staff feels that the examples cited in the NRC inspection report were sufficient to show the breadth of the contractor's and licensee's corrective action system.

39. Question

"Criterion XVIII, NSC Audit Finding 3. (Final p. 39-40, Draft p. 45-46.):

NSC states that management audits were ineffectual. Region V stated that "there is no basis to suggest these audits were ineffectual." Why did NSC and Region V reach such disparate conclusions?"

Answer

The staff's rationale for reaching the conclusion that "There is no evidence to suggest these audits were ineffectual," is stated in NRC Inspection Report No. 83-37, paragraph 42, page 40. Refer to the answer to question GC-5 regarding NSC.