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

June 13, 1984

CHAIRMAN

MEMORANDUM FOR: Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal

FROM: Nunzio J. Palladino NJP

SUBJECT: REVISED RESPONSE TO CONGRESSMAN PANETTA (CR-84-43)

Attached is a revised response to Congressman Panetta regarding Diablo Canyon. This revision is based on a discussion that I had with OGC and OCA in recognition of their comments on the circulated draft. I propose that you consider this attached revision in lieu of the draft circulated on May 21.

cc: EDO
OGC
OPE
SECY

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

CHAIRMAN

The Honorable Leon Panetta
United States House of Representatives
Washington, DC 20515

Dear Congressman Panetta:

This responds to your letter of February 8, 1984 regarding the Diablo Canyon Nuclear Power Plant. We appreciate your interest in the licensing and safety of this plant. You have raised the following three issues in your letter: Commission consideration of the decision by the Atomic Safety and Licensing Appeal Board; NRC guidelines for resolving allegations on a priority basis; and NRC staff implementation of safety margins.

Regarding the first concern, on March 20, 1984 the Appeal Board issued its decision resolving the issues on design quality assurance regarding Diablo Canyon Unit 1 in favor of the Pacific Gas and Electric Company. The decision imposes a condition for the operation of the component cooling water system and also requires further analysis of the jet impingment effects inside containment. The Appeal Board decision is subject to review by the Commission, but the Commission has not yet decided whether or not to take review. The staff is continuing its evaluation of the jet impingment question and intends to resolve it prior to making a recommendation regarding operation above 5% power.

Your second concern regards the need for guidelines that will govern the evaluation of allegations. The staff provided these guidelines to the Commission in Supplement 22 to the Safety Evaluation Report (SSER 22, March 1984), a copy of which is enclosed. This report was used as part of the basis for reinstatement of the low-power license which the Commission made effective on April 19, 1984. The Commission understands that tne staff intends to use these same guidelines in the evaluation of allegations related to full power authorization.

Finally, you express a concern over an apparent tendency of our staff to assume that the margins of safety established by our criteria need not be adhered to for systems which are not pivotal to safety, and that less precise, <u>ad hoc</u> standards of safety can be applied. This concern appears to be related to a substantive issue involved in the reopened hearing before the Appeal Board on design quality assurance. As mentioned above, the Appeal Board decision is subject to review by the Commission. It is more appropriate, therefore, for the staff to respond directly to your concern. We have directed the staff to provide you with a separate response on this matter.

We trust that this letter and the separate staff letter are responsive to your concerns.

Sincerely,

Nunzio J. Palladino

Enclosure:
NUREG-0675: Supplement 22 to
Diablo Canyon Safety Evaluation
Report, March 1984

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

OFFICE OF THE
CHAIRMAN

First DRAFT
CIRCULATED
BY NJP

The Honorable Leon Panetta
United States House of Representatives
Washington, D. C. 20515

Dear Congressman Panetta:

The response to ~~Thank you for~~ your letter of February 8, 1984 regarding the Diablo Canyon Nuclear Power Plant. We appreciate your interest in the licensing and safety of this plant. You have raised the following three issues in your letter: Commission consideration of the decision by the Atomic Safety and Licensing Appeal Board, NRC guidelines for resolving allegations on a priority basis, and NRC staff implementation of safety margins.

We are ~~I am~~ certain you are aware that the Commission reinstated ~~on April 13, 1984~~ the low-power operating license for Diablo Canyon Unit 1 effective on April 19, 1984. In responding to your concerns we ~~I~~ would like to briefly discuss some of the events that preceded our decision. We ~~I~~ believe that the manner in which the NRC staff has resolved numerous issues during the past few months and the steps which the Commission has taken prior to its decision are indicative of our position and approach to your concerns as discussed below.

Regarding the first concern, on March 20, 1984 the Appeal Board issued its decision resolving the issues on design quality assurance regarding Diablo Canyon Unit 1 in favor of the Pacific Gas and Electric Company. The decision imposes a condition for the operation of the component cooling water system. We included this condition in our decision for reinstatement of the low-power license and the staff recently amended the Technical Specification accordingly. The Appeal Board decision also required further analysis of jet impingement effects inside containment. The staff is continuing its evaluation of this matter and it will be resolved prior to issuance of a full-power license.

Your second concern is the bases and guidelines the NRC staff applied to determine which allegations must be satisfactorily resolved prior to a Commission decision on low-power operation. The staff provided these guidelines in Supplement 22 to the Safety Evaluation Report (SSER 22, March 1984), a copy of which is enclosed. The underlying concept for authorizing any low-power operation is that fission product generation and build-up at these conditions are only a small fraction of the values assumed in our analysis of the design basis accident.

At this time we have received in excess of 500 allegations. Although many of these are identical or similar we treated them separately because they frequently were submitted by different sources. Our staff has evaluated in sufficient detail all of the allegations by considering the guidelines in SSER 22 and concluded that none of these allegations need a complete resolution prior to reinstatement of the low-power license. Some concerns were identified as requiring a resolution prior to issuance of a full-power license.

At a Commission meeting on March 26, 1984, Mr. Isa Yin, a member of the NRC staff that investigated and evaluated certain allegations regarding piping and piping supports, informed us that in his opinion the low-power license should not be reinstated because of deficiencies in design, document control and personnel training. An NRC staff peer review group further investigated and evaluated these concerns. The group determined that additional analyses should be performed by the licensee and additional inspections should be performed by the staff. The Advisory Committee on Reactor Safeguards (ACRS), at our request, also evaluated Mr. Yin's concerns and concurred with the staff determination. None of the required actions were found necessary to be completed prior to low-power operation but must be completed prior to exceeding five percent of rated power. Mr. Yin stated at the Commission meeting on April 13 that he agrees with the position.

Finally, you express a concern over an apparent tendency of our staff to assume that the margins of safety established by our criteria need not be adhered to for systems which are not pivotal to safety, and that less precise, ad hoc standards of safety can be applied. The testimony by the staff at the January 24, 1984 hearing relating to margins of safety should not be construed to mean that the staff accepts less than the margins of safety required by the Commission's regulations. The testimony was meant to relate that inspection and review efforts are more heavily focused on those aspects which have the greatest potential for affecting public health and safety.

~~I~~ We hope this letter is responsive to the concerns you raised. In reinstating the low-power license for Diablo Canyon Unit 1, we express our opinion that the health and safety of the public will not be jeopardized by the operation of the facility under these conditions.

Sincerely,

Nunzio J. Palladino
Chairman

Enclosure:
NUREG-0675: Supplement 22 to
Diablo Canyon Safety Evaluation
Report, March 1984

OELD
LChandler
4/ /84

*See previous concurrence.

*DL:LB#3	DL:LB#3	DL:LB#3	DL:AD/L	DL:DIR	NRR	NRR	EDO
HSchierling/yt	JLee	GWKnighton	TMNovak	DGEisenhut	ECase	HRDenton	WJDircks
4/ /84	4/ /84	4/ /84	4/ /84	4/ /84	4/23/84	4/ /84	4/24/84

SSER 22 19
(Abbreviated)

SEE COPY OF SSER 22

(CRITERIA FOR RESOLUTION
OF ALLEGATIONS)

IN BACKGROUND MATERIAL FOLDER

AIAB-763 20
(Design QA)

ALAB 763

DOCKETED USNRC

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'84 MAR 20 P4:16

ATOMIC SAFETY AND LICENSING APPEAL BOARD

OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

Administrative Judges:

Thomas S. Moore, Chairman
Dr. John H. Buck
Dr. W. Reed Johnson

March 20, 1984
(ALAB-763)

SERVED MAR 21 1984

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

Joel R. Reynolds, John R. Phillips and Eric R. Havian, Los Angeles, California, and David S. Fleischaker, Oklahoma City, Oklahoma, for the San Luis Obispo Mothers for Peace, et al., joint intervenors.

John K. Van De Kamp, Attorney General of the State of California, Andrea Sheridan Ordin, Michael J. Strumwasser, Susan L. Durbin and Peter H. Kaufman, Los Angeles, California, for George Deukmejian, Governor of the State of California.

Robert Ohlbach, Philip A. Crane, Jr., Richard F. Locke and Dan G. Lubbock, San Francisco, California, and Arthur C. Gehr, Bruce Norton and Thomas A. Scarduzio, Jr., Phoenix, Arizona, for Pacific Gas and Electric Company, applicant.

Lawrence J. Chandler and Henry J. McGurren, for the Nuclear Regulatory Commission staff.

DECISION

On April 21, 1983, we granted the motions of the joint intervenors and the Governor of California to reopen the record in this operating license proceeding. Instead of remanding to the Licensing Board for that purpose, we acquiesced in the request of the parties that we hear the

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ALAB 763

further evidence ourselves. This decision sets forth our findings of fact and conclusions of law based upon that evidence.

I. History of Proceeding

A. In July 1981, the Licensing Board issued a partial initial decision authorizing the Director of Nuclear Reactor Regulation to issue a license to the Pacific Gas and Electric Company (PG&E or applicant) to load fuel and to conduct low power tests up to five percent of rated power at its Diablo Canyon Nuclear Power Plant, Units 1 and 2.[1] After the Commission's favorable immediate effectiveness review for Unit 1 (conducted pursuant to 10 CFR 2.764(f)),[2] the Director issued a low power license for that unit on September 22, 1981.[3]

Shortly thereafter, while preparing a response to an agency request for information, the applicant discovered errors in the assignment of seismic design spectra for equipment and piping in portions of the containment for Unit 1. These errors, combined with the identification by the NRC staff of serious weaknesses in the implementation of

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[1] See LBP-81-21, 14 NRC 107 (1981). 7/17/81

[2] See CLI-81-22, 14 NRC 598 (1981).

[3] License No. DPR-76.

the applicant's quality assurance program, led the Commission to suspend conditionally the applicant's low power license. The license suspension was to remain in effect pending the applicant's satisfactory completion of an independent design verification program focusing upon the pre-1978 work of the service-related contractors utilized in the seismic design process of safety-related structures, systems and components for Unit 1.[4]

In addition to the Commission's enforcement action, the staff instructed the applicant to provide it with the results of a further independent verification program for Unit 1 to enable the staff to authorize operation above low power levels. This verification was to be aimed at the pre-June 1978 service-related contractors used by the applicant in the nonseismic design of safety-related structures, systems and components, the applicant's internal design activities, and the post-1977 service-related contractors utilized by the applicant for both seismic and nonseismic design of structures, systems and components.[5]

In order to secure reinstatement of its license and eventual authorization for full power operation, the

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[4]See CLI-81-30, 14 NRC 950 (1981).

[5]See Applicant Exhibit (App. Exh.) 87, letter from H. Denton, NRC, to M. Furbush, PG&E (Nov. 19, 1981).

applicant initiated a verification program to meet the Commission's order and the staff's directive. As subsequent events would reveal, the applicant's verification efforts expanded far beyond those originally envisioned and took more than two years to complete.

While the verification was ongoing, and while the joint intervenors' appeal from the Licensing Board's low power decision was pending before us, the joint intervenors, on June 8, 1982, filed a motion to reopen the record on the issue of the adequacy of the applicant's quality assurance program. That motion was based essentially upon the same information that prompted the Commission's enforcement action and the various deficiencies identified by the verification program up to that time.

Besides opposing the joint intervenors' motion on the merits, the applicant claimed that the Commission's enforcement order conditionally suspending its license had divested us of jurisdiction to reopen the record. Although unpersuaded by this argument, we certified the jurisdictional question, among others, to the Commission in order to avoid any unnecessary delay in the licensing process were it ultimately to be accepted.[6] In due course, the Commission

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[6] See ALAB-681, 16 NRC 146 (1982).

(Footnote Continued)

responded that it had not intended, and did not now intend, to divest the adjudicatory boards of jurisdiction to act on the motions and that they should be treated in accordance with applicable case law.[7]

We then directed the certification to us of a similar motion that had been filed by the Governor of California on August 2, 1982 with the Licensing Board.[8] After hearing argument on the motions, we concurred with the concessions of the applicant and the staff that, with respect to the issue of design quality assurance, the motions of the joint intervenors and the Governor met the standards for reopening

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(Footnote Continued)

In addition, we asked whether the Commission wished to relieve the adjudicatory boards of jurisdiction with regard to quality assurance issues at Diablo Canyon and whether the Commission had any other instructions with regard to the reopening motions.

[7]See CLI-82-39, 16 NRC 1712 (1982).

[8]See Order of January 5, 1983 (unpublished).

When the Governor filed his reopening motion with the Licensing Board, the Board had yet to issue its decision resolving all contested issues necessary for full power operation. Subsequently, on August 31, 1982, the Board issued its initial decision authorizing full power operation. See LBP-82-70, 16 NRC 756. There the Board noted that its action did not affect the applicant's license suspension and that it would hold the Governor's reopening motion in abeyance to await the answer to the jurisdictional questions previously certified to the Commission in ALAB-681. Id. at 760 and 763. All parties filed exceptions to the Licensing Board's initial decision and those appeals are currently pending before us. In addition, the Commission still must conduct its immediate effectiveness review of that Licensing Board decision.

the record.[9] Accordingly, we granted the motions on April 21, 1983.[10]

Although the motions to reopen were predicated on deficiencies in the applicant's design quality assurance program and the applicant's failure to comply with 10 CFR Part 50, Appendix B, the real issue in the proceeding quickly moved beyond that point.[11] As noted in our prehearing order of August 16, 1983,

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[9]The motions of the joint intervenors and the Governor also sought reopening on the issue of the adequacy of the applicant's construction -- as opposed to design -- quality assurance program. Because of the manner in which the issue was presented, we deferred ruling on it. See Memorandum and Order of April 21, 1983 (unpublished). Thereafter, the joint intervenors and the Governor filed new motions to reopen the record on the construction quality assurance issue. In ALAB-756, 18 NRC ___ (Dec. 19, 1983), we set out the reasons for denying these motions.

[10]See Memorandum and Order of April 21, 1983 (unpublished).

The granting of the motions to reopen the record had no effect on the Licensing Board's previously issued partial initial decision authorizing fuel loading and low power testing (LBP-81-21, 14 NRC 107 (1981)) or initial decision authorizing full power operation (LBP-82-70, 16 NRC 756 (1982)). Our action neither vacated nor stayed these decisions. We subsequently affirmed the Licensing Board's low power decision. See ALAB-728, 17 NRC 777 (1983). Similarly, the reopening of the record in the operating license proceeding had no effect on the Commission's enforcement action suspending the applicant's low power license.

[11]Indeed, as the applicant's counsel stated at the argument on the motions to reopen,

(Footnote Continued)

> the history and nature of the design quality assurance issue at Diablo Canyon make this reopened proceeding unusual. Normally, an effectively functioning design quality assurance program ensures that the design of a nuclear power plant is in conformance with the design criteria and commitments set forth in an applicant's PSAR [Preliminary Safety Analysis Report] and FSAR [Final Safety Analysis Report]. In the case of Diablo Canyon, however, this confidence has been seriously eroded by the existence of significant evidence that the design quality assurance program was faulty (i.e. it failed to comply with 10 CFR Part 50, Appendix B). Hence, there is now substantial uncertainty whether any particular structure, system or component was designed in accordance with stated criteria and commitments.[12]

The order then indicated we would take our lead from the Commission and permit the applicant's various verification efforts "to substitute for, and supplement, the applicant's design quality assurance program in order to demonstrate that the Diablo Canyon plant is correctly designed."[13] It concluded by stating that the "real issue . . . has, in

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(Footnote Continued)

> [w]e are willing to stipulate that there -- that there are, may have been, and have been deficiencies in design QA [Quality Assurance]. That is behind us. There is no sense in litigating design QA. Where does that get anybody? It doesn't accomplish anything.

Transcript (Tr.) of April 14, 1983 oral argument at 215. See Order of August 16, 1983 (unpublished).

[12]Order of August 16, 1983 (unpublished) at 4-5. The analysis of the issues involved in the reopened proceeding outlined in the August 16 order was subsequently incorporated into our August 26, 1983 prehearing conference order.

[13]Order of August 16, 1983 (unpublished) at 5.

effect, moved beyond the question of what deficiencies existed in the applicant's Diablo Canyon design quality assurance program to the question whether the applicant can demonstrate that [its verification efforts] verify the correctness of the Diablo Canyon design."[14]

Trial of the thirty-nine contested issues regarding the adequacy of the applicant's verification efforts commenced October 31, 1983 in Avila Beach, California near the reactor site and consumed fifteen hearing days.[15] The applicant presented twenty-five witnesses,[16] the staff fourteen, the

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[14] *Id*. at 6.

[15] We accepted fifty-six issues of those originally sought to be litigated in the reopened proceeding by the joint intervenors and the Governor. See Orders of August 26 and October 7, 1983 (unpublished). Prior to the hearing, the joint intervenors and the Governor withdrew seventeen, leaving thirty-nine contested issues. See Withdrawal of Certain Contentions By Governor Deukmejian and Joint Intervenors (Oct. 24, 1983); Withdrawal of Certain Additional Contentions By Governor Deukmejian And Joint Intervenors (Oct. 31, 1983). As numbered in our August 26, 1983, prehearing conference order, the following issues remained at the time of the hearing: 1(a), (b), (c), (d), (e), 2(a), (b), (c), (d), 3(f)(i), (ii), (iii), (iv), (v), (o), (p), (q), (r), (s), (t), 4(a), (b), (h), (i)(1), (2), (j)(1), (2), (k), (l), (q), (r), (s), (t), (u), 5, 6, 7, 8 and 9. These issues are set forth in Appendix A of this decision.

[16] Seven of the applicant's witnesses were members of the Independent Design Verification Program (IDVP), see pp. 12-13, *infra*.

joint intervenors one, and the Governor three.[17] The hearing produced some 3700 pages of transcript and better than 6000 pages of exhibits. At the conclusion of the hearing, the parties were ordered, pursuant to 10 CFR 2.754, to file proposed findings of fact and conclusions of law and were admonished that the failure to file proposed findings on any issue would be deemed a waiver of that issue.[18] The last of the parties' proposed findings was filed January 4, 1984. The joint intervenors and the Governor both failed to file proposed findings on sixteen issues.[19] In addition, the joint intervenors failed to file proposed findings on an issue that the Governor abandoned in his findings.[20] These

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[17]The applicant and the staff witnesses testified as panels. Because of the number of issues in the proceeding, the issues were treated discretely and the composition of the panels varied accordingly. A list of the witnesses, their education and their present position appears in Appendix B of this decision.

[18]Tr. D-3239. See Detroit Edison Co. (Enrico Fermi Atomic Power Plant, Unit 2), ALAB-709, 17 NRC 17 (1983).

[19]Those issues are as follows: 2(d), 3(f)(ii), (p), (s), (t), 4(a), (b), (h), (i)(2), (j)(1),(2), (k), (q), (r), (s), (u).

[20]The joint intervenors failed to file proposed findings on issue 3(f)(i) dealing with the boundary motion inputs for the applicant's soil structure interaction analysis of the containment building. See Joint Intervenors' Proposed Findings of Fact and Conclusions of Law (JI PF) (Dec. 23, 1983). The Governor's proposed findings now accept the applicant's results. See Proposed Findings of Fact And Conclusions of Law of Governor Deukmejian (Gov. PF) at 39-40 (Dec. 24, 1983).

issues are therefore waived, leaving twenty-two issues for resolution.[21]

In order to prevail on each of the remaining factual issues, the applicant's position must be supported by a preponderance of the evidence.[22] We do not decide, however, whether each element of the Commission's November 19, 1981 enforcement order (or other subsequent directives) has been met. That task is for the Commission itself.[23] Rather, we must independently determine whether the verification

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[21]The issues remaining for decision are as follows: 1(a), (b), (c), (d), (e), 2(a), (b), (c), 3(f)(iii), (iv), (v), (o), (q), (r), 4(i)(1), (l), (t), 5, 6, 7, 8 and 9.

[22]See Tennessee Valley Authority (Hartsville Nuclear Plant, Units 1A, 2A, 1B and 2B), ALAB-463, 7 NRC 341, 360 (1978), reconsideration denied, ALAB-467, 7 NRC 459 (1978); Duke Power Co. (Catawba Nuclear Station, Units 1 and 2), ALAB-355, 4 NRC 397, 405 n.19 (1976).

[23]The Commission's order suspending the applicant's low power license until the successful completion of a prescribed verification program was a Commission enforcement action. Because the applicant did not challenge that action, and the Commission did not otherwise direct, no enforcement proceeding was begun. Nor did the Commission, when responding to our certified questions, indicate that its enforcement action should become part of the operating license proceeding. See n.7, supra and accompanying text. Therefore, we believe it is clear that the Commission did not intend to leave enforcement of its order to the reopened licensing proceeding. Thus, the elements of the verification program contained in the Commission's enforcement order, like those contained in the November 19, 1981 staff letter to the applicant (see n.5, supra), may prove useful in assessing the overall adequacy of the applicant's verification program, but in these circumstances, they do not control our determination of the sufficiency of the applicant's verification efforts.

programs and their results placed before us in the reopened operating license proceeding are sufficient to verify the adequacy of the Diablo Canyon design. To do this, the applicant's efforts must be measured against the same standard as that set forth in the Commission's quality assurance criteria, 10 CFR Part 50, Appendix B: whether the verification program provides "adequate confidence that a [safety-related] structure, system or component will perform satisfactorily in service." If the applicant's verification efforts meet this standard, then there will be reasonable assurance with respect to the design of the Diablo Canyon facility that it can be operated without endangering the health and safety of the public.

B. A summary of the development and content of the Diablo Canyon verification efforts is helpful to an understanding of our resolution of the issues in Part II, *infra*.

Immediately after the discovery of the seismic design errors at Diablo Canyon, the applicant retained Robert L. Cloud and Associates, Inc. (Cloud Associates) to develop and implement an internal verification program to assess the adequacy of the plant's seismic design.[24] The initial Cloud

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[24] App. Exh. 90, Diablo Canyon Nuclear Power Plant-Unit 1, Final Report, Independent Design Verification Program, (Footnote Continued)

Associates' review indicated that the design problems were more pervasive than at first thought.

Subsequent to the issuance of the Commission order[25] calling for the establishment of an extensive and structured verification effort, the applicant, on December 4, 1981, proposed a program managed by Cloud Associates that would include the services of R.F. Reedy, Inc. (Reedy Inc.) for quality assurance verification and Teledyne Engineering Services (Teledyne) for overall review of the program and its implementation. This effort was to be directed at the seismic design work performed for the applicant under pre-June 1978 service-related contracts and was labeled the Phase I program.[26] Thereafter, in response to the broader matters raised in the staff letter, the applicant also submitted a Phase II program. This program included an examination of the nonseismic work performed for the applicant under pre-June 1978 service-related contracts, the applicant's own internal design activities, and all the nonseismic and seismic work performed for the applicant

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(Footnote Continued)
Vol. I (1983) (hereinafter IDVP Final Report), at 1.2-1 to -2.

[25] See CLI-81-30, supra, 14 NRC at 950.

[26] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-1. See also letter of December 4, 1981 from M. Furbush, PG&E, to H. Denton, NRC.

under post-1977 service-related contracts. The Phase II program also added the Stone and Webster Engineering Corporation (Stone and Webster) to the other organizations already proposed to conduct this review.[27]

The Commission's order required that the companies conducting the verification program possess the necessary technical competence and that they be independent of the applicant.[28] On March 4, 1982, the Commission approved the Phase I program but required that Teledyne be the program manager because Cloud Associates had previously done substantial work for the applicant.[29] In accordance with this Commission action, Teledyne prepared an Independent Design Verification Program (IDVP) Phase I Program Management Plan which integrated the earlier Cloud Associates' plan and included requirements for Teledyne's acceptance of work done prior to its takeover as program manager on March 25, 1982.[30] Under Teledyne's direction,

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[27] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-2. See also letter of January 13, 1982 from M. Furbush, PG&E, to H. Denton, NRC.

[28] CLI-81-30, supra, 14 NRC at 957.

[29] App. Exh. 156, SECY-82-89, and App. Exh. 158, Memorandum from W. Dircks to S. Chilk indicating Commission approval.

[30] App. Exh. 88, IDVP Program Management Plan, Phase I, Revision I (July 6, 1982) (hereinafter IDVP Phase I Management Plan); App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-5.

Cloud Associates would perform the review of seismic, structural and mechanical design and Reedy Inc. would review quality assurance.[31] The Phase I Plan included only the safety-related (Diablo Canyon Design Class I) buildings, equipment, piping and components that had been requalified in consideration of the Hosgri 7.5M earthquake.[32] The plan described the initial sampling and the requirements for any additional verification and sampling.[33] In a letter dated

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[31]App. Exh. 88, IDVP Phase I Management Plan, at 17; App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-5.

[32]App. Exh. 88, IDVP Phase I Management Plan, Appendix D at 2; App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-8.

[33]When a criterion used in the IDVP verification process was not met, the IDVP issued an Error or Open Item (EOI) File to track the resolution of the IDVP concern. Following further investigation, the IDVP would classify the item as either a deviation (i.e., a departure from standard procedure but not a mistake) or one of four categories of error (i.e., A, B, C, D). The safety significance, if any, of an error was not part of the classification scheme. Rather, an error was considered class A if design criteria or operating limits of safety-related equipment were not met and physical modifications or changes in operating procedures were required. An error was considered class B if it met the definition of class A but could be resolved by more realistic calculations or retesting, instead of physical modifications. A class C error was one in which incorrect engineering or incorrect installation of safety-related equipment was found, but no design criteria or operating limits were exceeded. An error was considered class D if safety-related equipment was not affected. An EOI file remained open until the IDVP determined that the item was in conformance with licensing criteria. App. Exh. 88, IDVP Phase I Management Plan, at 25, and Appendix E;

(Footnote Continued)

April 27, 1982, the NRC staff approved the IDVP Phase I Plan.[34]

Several months later, Teledyne developed an IDVP Phase II Management Plan and submitted it to the NRC.[35] This plan encompassed nonseismic, service-related contracts performed prior to June 1978, the applicant's internal design activities, and all service-related contracts after January 1978.[36] The participants and their general responsibilities were the same as those in the Phase I Plan but Stone and Webster was added to perform the review of nonseismic safety systems and analyses.[37] On December 9, 1982, the Commission approved the Phase II Plan.[38]

Shortly after the receiving approval of the Phase I program, the applicant retained Bechtel Power Corporation to work with it and act as Completion Manager for the Diablo Canyon facility. To align the verification activities with

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(Footnote Continued)
App. Exh. 89, IDVP Program Management Plan, Phase II (June 18, 1982) (hereinafter IDVP Phase II Management Plan), at 24; App. Exh. 90, IDVP Final Report, Vol. I, at 3.6-2 to -6.

[34] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-5.

[35] Id. at 1.3-6.

[36] App. Exh. 89, IDVP Phase II Management Plan, at 1.

[37] Id. at 8.

[38] App. Exh. 157, SECY-82-414; App. Exh. 159, Memorandum from W. Dircks to S. Chilk indicating Commission approval.

this development, the applicant developed an Overall Management Plan that, inter alia, adopted the IDVP Phase I Program Management Plan.[39] Under the Overall Management Plan, the joint Bechtel-PG&E team was referred to as the Diablo Canyon Project (DCP) and it was responsible for executing the Internal Technical Program (ITP). The purpose of the ITP was to (a) provide an additional design verification effort for the assurance of the overall adequacy of the design of the plant; (b) develop data and information in support of the IDVP; (c) respond to IDVP open items and findings; and (d) implement design modifications or other corrective actions arising from the verification program.[40]

Under the Phase I program, the seismic verification effort was initially based upon a sampling process.[41] The early findings of the sampling program led the applicant to review the entire scope of certain engineering activities. In order to save time and best assure final NRC approval of the verification effort, the applicant decided in the summer of 1982 to expand the seismic program to evaluate the total seismic design of safety-related structures, systems, and

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[39]App. Exh. 90, IDVP Final Report, Vol. I, at 1.4-1.

[40]Id. at 1.4-1 to -2.

[41]Id. at 1.4-2.

components.[42] This broad review enveloped the findings of the previous IDVP and ITP seismic reviews and made it unnecessary to review older analyses and calculations that were to be redone by the ITP. In view of the enlarged ITP seismic review,[43] the IDVP program was changed from one of sampling original designs to one of verifying the ITP's seismic work. The IDVP examined the scope, criteria and methodology of the ITP work for consistency with the license application and then verified samples of that work.[44] In

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[42] Id.; App. Exh. 91, ITP Design Verification Program Phase I Final Report (Oct. 19, 1983) (hereinafter ITP Phase I Final Report) at 1.5.2-1 to -2.

This phase of the work by the Bechtel-PG&E team is referred to as the Corrective Action Program (CAP). Thus, there are several labels which may be applied to work carried out by that group (i.e., DCP, ITP, CAP). Because our previous references to the work done by the Bechtel-PG&E team in the proceeding have been to the ITP, we shall continue to use ITP as a catchall phrase to denote work done both by the applicant subsequent to November 1981 as well as by the Bechtel-PG&E team.

[43] The complete ITP seismic review program is described in the ITP Phase I Final Report, App. Exh. 91.

[44] App. Exh. 90, IDVP Final Report, Vol. I, at 3.5-7.

The seismic design review resulted in thousands of minor modifications to steel frame structures and supports for piping, raceways, instrumentation, instrument tubing and equipment. App. Exh. 91, ITP Phase I Final Report, at 1.8.6-2 and Appendix 1E. A large number of modifications must be expected when seismic response spectra are changed, because many similar structural components are included in each individual seismic analysis and each component may be affected by a change in the seismic response spectra. For

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addition, the staff reviewed the seismic verification efforts of the ITP and the IDVP on a continuing basis.

The IDVP also selected samples of the original engineering design work for the Phase II nonseismic verification.[45] The samples were reviewed and analyzed by the IDVP against verification criteria from the program management plan. If the criteria were not satisfied, the initial samples were reanalyzed or additional samples were identified for verification. When the IDVP identified a potentially generic concern, the ITP was required to perform a review for that concern for all applicant-designed, safety-related systems.[46] The IDVP then evaluated these ITP

(Footnote Continued)
example, several pipe support modifications could result from a single change in one pipe analysis and that piping design may be repeated hundreds of times. See id. at 2.2.1-22 to -36 (Table 2.2.1-3), 2.2.1-37 to -51 (Table 2.2.1-4), 2.2.2-17 to -24 (Table 2.2.2-1). See also Moore Tr. D-412.

[45] App. Exh. 90, IDVP Final Report, Vol. I, at 1.3-8 to -9. The entire IDVP verification program (i.e., seismic and nonseismic) is documented in sixty-three interim technical reports (App. Exhs. 93 to 155) and a four-volume final reports that contains the IDVP's conclusions (IDVP Final Report) (App. Exh. 90).

[46] Only a few of the findings from the nonseismic design review resulted in modifications to plant systems and the alterations were minor. App. Exh. 92, ITP Phase II Final Report Design Verification Program (hereinafter ITP Phase II Final Report), at 3-2 to -3. For example, minor modifications were performed involving the following: (1) rerouting of certain electrical circuits to assure circuit independence; (2) electrical changes to the control room

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reviews and documented their findings in Interim Technical Reports (ITRs) for the staff to review. In addition to the nonseismic reviews performed by the ITP at the direction of the IDVP, the ITP independently conducted a functional design review that covered a portion of each of the applicant-designed, safety-related nonseismic systems.[47] Unlike the seismic review, the entire design of applicant-designed, safety-related systems was not reviewed.

II. Findings on Contested Issues

As previously noted, the real issue in this reopened proceeding is whether, in view of the conceded weakness of the Diablo Canyon design quality assurance program, the applicant's verification efforts demonstrate that the safety-related structures, systems and components of the plant are properly designed (i.e., conform to the various licensing criteria for the facility). Although the applicant presented evidence to establish that it verified the design of both Diablo Canyon Units 1 and 2, we make no

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ventilation and pressurization system to allow the single failure criterion to be met for Unit 1 without the availability of Unit 2 power supplies; (3) auxiliary feedwater system alterations to prevent inadvertent overpressurization of certain components; (4) strengthening of doors; and (5) installing flow limiters and dampers. Id. at 3-3 to -31.

[47]The modifications required by the ITP's functional design review are described in the ITP Phase II Final Report. App. Exh. 92 at 2-5 and Appendix B.

findings with respect to Unit 2. The two units are nearly identical from a design standpoint, but the applicant's verification efforts for Unit 2 differ from those for Unit 1. Significantly, the IDVP had no direct involvement in the Unit 2 verification program. Rather, the applicant has established an internal review organization for Unit 2 to evaluate deficiencies identified for Unit 1 and, if appropriate, to correct these deficiencies as they appear in Unit 2. The Unit 2 verification is still ongoing and has not been finally reviewed by the staff. Nor has the staff issued a safety evaluation report supplement on the Unit 2 verification. In the circumstances, we believe it is most appropriate to sever the question of the Unit 2 design verification from the proceeding and decide at this time only the issues related to Unit 1.

A. In issues 1 and 2, the Governor and joint intervenors challenge the scope of the applicant's verification program and, in effect, dispute the ability of the applicant's verification efforts to provide the same assurance of proper design as a satisfactory quality assurance program.

Specifically, in issues 1(a) and (b), the joint intervenors and the Governor assert that the scope of the IDVP review was too narrow, because it did not verify samples from each design activity and from each design group performing a particular design activity. Issues 2(a) and

(b) raise the same questions but with regard to the ITP verification efforts. The joint intervenors and the Governor also contend in issues 1(c) and 2(c) that the IDVP and ITP verification efforts were flawed because they did not have statistically valid samples from which to draw conclusions. Because there was a marked difference in the manner in which the seismic and nonseismic verifications were conducted, we first treat the seismic verification by the IDVP and the ITP, then in section 2, we deal with the nonseismic verification.[48]

1. The ITP essentially redid all of the seismic design for safety-related structures, systems and components, while the IDVP oversaw and verified selected samples of the work.[49] The ITP reanalyzed the design of portions of the containment, the auxiliary building, the fuel handling building, the turbine building and the intake structure. All large bore piping and pipe supports were reanalyzed, and small bore piping and pipe supports were reviewed either by sampling or on a generic basis. The ITP reviewed or reanalyzed the safety-related mechanical, electrical, and

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[48]See n.67 for discussion of issue 1(d). The remaining parts of 1 and 2, issues 1(e) and 2(d), pertain to Unit 2. See pp. 19-20, *supra*.

[49]Anderson *et al*. [This panel consisted of R. Anderson, G. Cranston, G. Moore, L. Shipley and W. White.] Tr. fol. D-224 at 5-6, 9-10; Cooper *et al*. [This panel consisted of W. Cooper, R. Cloud, J. Krechting and R. Reedy.] Tr. fol. D-1459 at 1/2-12 to -20; App. Exh. 100, ITR 8, at 1-2.

instrumentation and control equipment to assure that these components were seismically qualified. In addition, the ITP examined the design of all safety-related electrical raceways and heating, ventilation, and air conditioning (HVAC) ducts and supports. Finally, the ITP sampled the safety-related instrument tubing and supports to ensure their seismic qualification.[50] Thus, with respect to the seismic design, the work of the ITP became the design of record.[51]

The ITP's seismic design work was done under a quality assurance program that met the provisions of 10 CFR Part 50, Appendix B.[52] In addition, this work was independently verified by the IDVP. In each of the areas of seismic design addressed by the ITP, the IDVP verified the work by reviewing selected samples. The exact approach taken by the IDVP varied depending upon the nature of ITP work.[53] For

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[50]Anderson et al. Tr. fol. D-224 at 6; Seed et al. [This panel consisted of R. Anderson, H. Seed, L. Shipley and W. White.] Tr. fol. D-652 at 7-8; App. Exh. 91, ITP Phase I Final Report, at 1.5.1-3 to -4.

[51]Cooper et al. Tr. fol. D-1459 at 1/2-13.

[52]The adequacy of the quality assurance program covering the ITP's work is discussed subsequently. See pp. 89-98, infra.

[53]App. Exh. 100, ITR 8, at 1-2.

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all reviews, however, the IDVP first compared the scope of the ITP work with the applicable license criteria, and then ascertained that the analytical methods used by the ITP were valid, verifying such items as modeling techniques, model constraints, assumptions and the levels of model sophistication. In each seismic design area, the IDVP selected a sample of calculation packages for detailed review. The review was designed to investigate the specific concerns that the IDVP developed during earlier IDVP reviews, and to ensure the complete evaluation of the process utilized by the ITP. The calculation packages were verified by design review or by performing independent analyses, or a combination of these techniques. IDVP samples consisted of in-progress and completed ITP work. In certain instances, questions arose which caused additional samples to be evaluated by the IDVP. For each area of ITP work reviewed, the IDVP issued an ITR documenting the results of its review.[54] Thus, the final seismic design

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The ITP seismic verification work was divided into three categories according to the methods used: complete reanalysis (e.g., Fuel Handling Building); review of existing analyses followed by reanalysis of deficient items (e.g., large bore piping); and reviews of samples to demonstrate conservative design (e.g., small bore piping). Id.

[54] Cooper et al. Tr. fol. D-1459 at 1/2-13 to -20.

(Footnote Continued)

derived from the ITP's efforts and the IDVP review of those efforts subjected the design of Diablo Canyon to a measurably greater level of scrutiny than could have been provided by a quality assurance program complying with Appendix B.[55]

The Governor asserts, however, that the seismic verification was insufficient because the ITP's redesign efforts did not encompass all elements of the seismic design. Specifically, he claims that small bore (less than

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(Footnote Continued)
The following ITRs document the IDVP review of the ITP seismic verification work:

Applicant Exhibit #	ITR Number	Subject
142	50	Containment Annulus
143	51	Containment Annulus
144	54	Containment Building
145	55	Auxiliary Building
146	56	Turbine Building
147	57	Fuel Handling Building
148	58	Intake Structure
149	59	Large Bore Piping
150	60	Pipe Supports
151	61	Small Bore Piping
152	63	HVAC Ducts, Electrical Raceways, Instrument Tubing and Associated Supports
153	65	Rupture Restraints
154	67	Equipment
155	68	Soils

[55] Moreover, the nature and breadth of the seismic design review (i.e., essentially 100 percent) eliminates any reasonable argument that the review was flawed because statistically valid sampling techniques were not used.

2-inch diameter) piping was requalified not by 100 percent review, but through a program of generic reviews and sampling, and that instrument tubing supports were also requalified by sample calculations. He charges that the ITP reviewed equipment only if the response spectrum governing its seismic design had changed and, even then, the ITP only evaluated safety-related equipment designed by the applicant, not others.[56]

None of the Governor's challenges detracts from the adequacy of the applicant's seismic verification programs. Small bore piping at Diablo Canyon was designed by computer-based analysis or by the use of span criteria.[57] The ITP verification was carried out by "generic" reviews,[58]

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[56]Gov. PF at 29-31.

[57]Span criteria are analytically determined rules which govern the spacing between seismic supports in a run of piping (i.e., the length of the span of pipe between supports). App. Exh. 122, ITR 30, at 6 and A-6; "Seismic Evaluation for Postulated 7.5M Hosgri Earthquake" (Hosgri Report), Amendment No. 50 to operating license application (June 3, 1977), Vol. II, at 8-3 to -4 and Figure 8-1. The span rules were revised by the ITP to include the effect of insulation and spectra revisions, and to provide more user guidance. App. Exh. 91, ITP Phase I Final Report, at 2.2.2-6.

[58]The "generic" program encompassed small bore piping and piping analyses issues identified by the IDVP and ITP reviews as having a potential for causing modifications. Id. at 2.2.2-1. Specifically, the program included the following piping configurations: those previously analyzed by dynamic analysis; those in which safety-related valves

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and by sampling.[59] The ITP reported the results of some 80 piping analyses, involving approximately 1,550 piping spans,[60] carried out under the generic and sampling programs.[61] Noting the ITP's use of computer-based dynamic analysis and its limited use of the less conservative span rules, the IDVP concluded that the ITP methods and coverage were acceptable and the ITP analysis ensured that small bore piping was properly designed.[62] We agree. There is no need to test every repetitive pipe configuration. The ITP's

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are supported by pipes; those subject to thermal or seismic movement of anchors; those at boundaries between code requirements; and those pipes subject to thermal stresses previously qualified by span rules. Id. at 2.2.2-4 to -6. The generic review program was carried out primarily by dynamic analyses. Id. at 2.2.2-8 to -9.

[59] Id. at 2.2.2-1.

Under the sampling program a number of piping configurations designed using span criteria were selected to undergo dynamic analysis as well. The selection of samples was made to address a number of specific design configurations and issues not included in the generic review, and to demonstrate the qualification of piping that was designed using span criteria. Id. at 2.2.2-2 to -3, -6 to -8, -10 to -11.

[60] The computer analysis of a piping configuration generally includes many (typically ten to fifty) supports. Thus, a single piping analysis checks the design adequacy of many pipe spans. See id. at 2.2.2-17 to -24 (Table 2.2.2-1).

[61] Id. at 2.2.2-8 to -11, 2.2.2-17 to -24 (Table 2.2.2-1).

[62] App. Exh. 151, ITR 61, at 54, 60.

broad coverage in its generic and sampling reviews was sufficient to assure adequacy of the piping design.

The seismic design of instrument tubing supports, like that of small bore piping, need not be verified by 100 percent reanalysis. There are only a few basic seismic designs of instrument tubing supports, although there are many applications of each design. The ITP selected for review a sample of eighty-eight supports that represented worst case and enveloping situations. Of these supports, the analyses indicated that two were inadequate as a result of their specific cantilevered configuration. All tubing supports in the plant were then examined for this configuration and no other deficient cantilevered supports were found.[63] The IDVP review of the ITP effort confirmed that the analyzed tube support configurations included worst case situations, and concluded that the tube supports throughout the plant were adequate.[64] Once again, we agree with the IDVP's conclusion. Because of the repetitive nature of the instrument tubing support design, there is no need to test every support. The breadth of the ITP review,

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[63]App. Exh. 91, ITP Phase I Final Report, at 2.6-1 to -4.

[64]App. Exh. 152, ITR 63, at 47, 54; App. Exh. 90, IDVP Final Report, Vol. II, at 4.6.8.2-1 to -4.

which included worst case analysis, was sufficient to ensure proper instrument tube support design.

Finally, with respect to the Governor's last challenge to the sufficiency of the seismic review, we find that the seismic qualification of safety-related equipment was not deficient. The ITP determined new seismic response spectra for all structures except the containment shell. In reviewing the equipment for qualification to the new spectra, the ITP reviewed all safety-related equipment, even that in the containment, so that no equipment was overlooked.[65] Nor was the seismic review flawed because the IDVP did not review the qualification of safety-related equipment designed by Westinghouse. We deal with the question of the sufficiency of Westinghouse designed equipment subsequently.[66] Suffice it to say at this point

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[65]Seed et al. Tr. fol. D-652 at 7-8, 61-64; App. Exh. 91, ITP Phase I Final Report, at 2.3-1.

Mechanical equipment was checked by one of three methods: flexible items (having natural frequency less than 33 Hertz) were subjected to dynamic analysis; rigid items were qualified to equivalent static loads or by dynamic analysis; and some equipment was qualified by testing on a shake table. App. Exh. 91, ITP Phase I Final Report, at 2.3.1-5; App. Exh. 154, ITR 67, at 10, 17, 29. Similar methods were used to verify the electrical and HVAC equipment items. App. Exh. 91, ITP Phase I Final Report, at 2.3.2-2, 2.3.3-2 to -3.

[66]See pp. 77-82, infra.

that Westinghouse itself performed an adequate seismic design review of the equipment and systems it supplied.

We conclude, therefore, that the seismic redesign process carried out by the ITP and reviewed by the IDVP provides adequate confidence that the seismic design of the structures, systems and components at Diablo Canyon Unit 1 is proper and meets licensing criteria.[67]

2. Unlike the seismic verification under which essentially all of the Diablo Canyon seismic design was reviewed, the applicant's nonseismic design review efforts were less ambitious. Although both the IDVP and the ITP verified portions of the nonseismic design of the facility, their combined efforts did not encompass the entire nonseismic safety-related design. For example, neither the

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[67]In issue 1(d), the Governor also challenges the sufficiency of the IDVP seismic review program claiming that, instead of independently verifying analyses for Diablo Canyon, it merely checked data inputs to the applicant's design models. The record is replete with instances in which the IDVP carried out its own calculations, both in the seismic and nonseismic areas of the verification. The evidence also demonstrates that design reviews carried out in lieu of independent analyses were far more extensive than a mere checking of input data. Cooper et al. Tr. fol. D-1459 at 1/2-16, -19 to -20, -28 to -29, -34 to -35; Cloud Tr. D-1939-41, D-1944-45. Moreover, we find that the IDVP's approach of verifying samples by a combination of reanalysis and design review is sufficient to provide adequate verification of design. The value of independent recalculations is not disputed, but there is no indication that this approach is essential to provide assurance of design efficacy. Cloud Tr. D-1937-38; Roesset Tr. D-2247-48.

IDVP nor the ITP verified samples from each design activity and each design group performing that activity, as alleged to be necessary by the Governor and the joint intervenors in issues 1(a) and (b), and 2(a) and (b). Nor did the IDVP and ITP select the portions of the nonseismic design work they reviewed on a statistically valid basis (i.e., they did not randomly sample the universe of engineering design decisions), as urged by the Governor and the joint intervenors in issues 1(c) and 2(c). Because the nonseismic review was not all encompassing and not based on statistically valid sampling techniques, the Governor and the joint intervenors claim that the applicant's verification program is so seriously flawed that it cannot properly be used as a basis for reaching conclusions about the unreviewed portions of the nonseismic design. The applicant and the staff, on the other hand, assert that the scope and nature of the applicant's nonseismic design review are more than sufficient to support the conclusion that the Diablo Canyon design meets applicable licensing criteria.

Specifically, the Governor and the joint intervenors assert that because the design samples selected by the verification program were chosen deliberately on the basis of certain engineering judgments, and not randomly, the sample selection process was biased. Thus, the argument continues, no statistically valid conclusion regarding probabilities of errors or error rates can be drawn for the unreviewed portions of the nonseismic design; and, in order to verify satisfactorily the nonseismic design, the applicant must go back and either randomly sample the universe of nonseismic design decisions or review 100 percent of it.

This argument essentially overlooks the standard by which the applicant's program is to be judged. We must determine whether the nonseismic verification program provides "adequate confidence" that the nonseismic design of safety-related structures, systems and components is proper so that such structures, systems and components will perform satisfactorily in service.[68] This qualitative standard is

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[68]Pointing to the Commission's regulations, 10 CFR 50.57(a)(1), the Governor and the joint intervenors repeatedly assert in their proposed findings that the applicant's verification program, in order to be sufficient, must demonstrate that the design of Diablo Canyon meets its license application requirements or licensing criteria. The application requirements and licensing criteria for Diablo Canyon, like any nuclear power plant, are spelled out in the various documents comprising the operating license application including, most prominently, the applicant's Final Safety Analysis Report (FSAR). The FSAR is a multivolume description of the entire facility containing literally thousands of so-called "licensing criteria" ranging from safety significant ones to insignificant and extremely minor specifications or descriptions of details that have no safety implications. See 10 CFR 50.34(b). In their proposed findings of fact, the Governor and the joint intervenors do not distinguish between safety significant and nonsafety significant licensing criteria. For example, the Governor and joint intervenors argue, relying on the staff's and the applicant's witnesses, that the nonseismic design does not meet licensing criteria because it is a

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not numerically quantifiable into expressions of probability of errors or error rates, as the Governor and the joint intervenors would have it. Even if a statistically valid error rate were available to forecast the errors in the unreviewed portions of the nonseismic design,[69] in all but

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virtual certainty that there remain undetected design errors in the unreviewed portions of the design. JI PF at 14-16; Gov. PF at 5-8, 9-11. But the witnesses relied upon by the Governor and joint intervenors all testified that not only was it likely there remained some design errors, but that it was extremely unlikely any of the errors were safety significant. Cloud Tr. D-1543, D-1545; Schierling Tr. D-2662-63, D-2665; Knight Tr. D-2706. In effect, the Governor and joint intervenors champion form over substance. We reject their position. The central issue with respect to the proper design of Diablo Canyon, or any other facility, is the conformance of the design to the significant and substantive safety requirements and licensing criteria. To conclude otherwise would ignore reality and substitute "perfection" for the regulatory standards of "adequate confidence" and "reasonable assurance." See p. 11, <u>supra</u>.

[69] Dr. Stanley Kaplan, an engineer and applied mathematician, appeared as an expert witness for the applicant. Dr. Kaplan used the results of the nonseismic design verification work of the IDVP and applied Bayesian techniques to predict an error rate for the original design of the plant (<u>i.e.</u>, errors per design element). Also, using the judgment of the engineers associated with the verification effort that the errors identified by the verification were minor and of little safety significance, Dr. Kaplan applied his methodology to determine the likelihood of safety significant design errors remaining in the unsampled portions of the nonseismic design. Kaplan and Anderson Tr. fol. D-1161 at 56-63. Dr. Kaplan, however, cautioned (<u>id</u>. at 45) that his "numerical results are to be interpreted with a large grain of salt . . . ." See <u>id</u>. at 17-22.

The Governor's and the joint intervenors' expert witnesses, Drs. Apostolakis and Samaniego -- both
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certain obvious situations, such a rate would be of little utility in judging the adequacy of the verification of the nonseismic design of Diablo Canyon. In part, this is because no acceptable rate of design errors for nuclear power plants has ever been determined.[70] Thus, the ultimate determination regarding the adequacy of the plant's design remains a qualitative judgment and we must turn to the verification work that was performed to ascertain whether its scope and quality are sufficient to provide the requisite assurance of design adequacy.[71]

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statisticians -- reject out of hand Dr. Kaplan's projected error rate because it was calculated using nonrandomly selected samples. Samaniego Tr. D-2394-95; Apostolakis Tr. D-2343. Because we find little utility in the determination of error rates (or their accuracy) for the qualitative judgment we must make on the adequacy of the verification program for the nonseismic design, we need not decide the validity of Dr. Kaplan's calculations.

[70]Kaplan and Anderson Tr. fol. D-1161 at 67-70; Apostolakis Tr. D-2354, D-2369; Knight et al. [This panel consisted of J. Knight, H. Schierling and J. Wermiel.] Tr. fol. D-2649 (Contention 2) at 7-8.

According to the Governor and joint intervenors, the evidence indicates that, in spite of the verification program, there remain errors in the unreviewed portions of the nonseismic design that represent failures to meet licensing criteria. This fact, they claim, renders the verification program inadequate. Gov. PF at 9-11, 38-39; JI PF at 14. Thus, the Governor and the joint intervenors apparently would accept only a zero error rate. See n.68, supra.

[71]While it is unnecessary to consider the statistical question in more depth, we note our skepticism that a
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The IDVP and the ITP each took a different approach to verify the nonseismic design work. The IDVP chose for review three specific safety-related systems that included work from all the applicant's internal design groups and the service-related contractor who performed the most significant nonseismic design work.[72] It also selected two areas of safety-related analysis applicable to many other systems.[73] The majority of the IDVP's nonseismic

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statistically valid design verification program, as thorough as the applicant's verification efforts, could have been developed and implemented. No such program has ever been developed for a nuclear power plant. Apostolakis Tr. fol. D-2313 at 12; Samaniego Tr. D-2408-10, D-2451. Although theoretically possible, implementation presents formidable obstacles such as identifying and stratifying the many thousands of design decisions that went into the facility so they may be randomly sampled. Kaplan and Anderson Tr. fol. D-1161 at 5-6; Apostolakis Tr. D-2335-44. It must be borne in mind that the subject under investigation is the design adequacy of a complex facility consisting of a multitude of engineered systems, each with its own function and each with some potential for interacting in various ways with the other plant systems. Each "design element" or design decision for a particular system involves input from previous determinations for that system and for interacting systems. We are not persuaded that random sampling of such elements is necessarily the most effective means for addressing design adequacy. Rather, a coherent sampling scheme devised in view of a system's characteristics, its function, and its interaction with other systems appears to us to be a more acceptable method for ascertaining the adequacy of the design of a nuclear power plant. Cooper _et al_. Tr. fol. D-1459 at 1/2-14, -24 to -25; Anderson _et al_. Tr. fol. D-224 at 25-27.

[72] Cooper _et al_. Tr. fol. D-1459 at 1/2-24.

[73] Specifically, the IDVP selected the auxiliary
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verification involved the performance of independent

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feedwater (AFW) system, the control room ventilation and pressurization (CRVP) system, and the safety-related portion of the 4160 volt (V) electric distribution system for review. As stated by the IDVP:

> The AFW system was selected because its design represents an interrelationship of several design criteria and interfaces. Specifically, it involves interface with NSSS [Nuclear Steam Supply System] vendor criteria, with containment design criteria, interface of PGandE internal design organizations, and the methodology of determining a water system's mechanical, electrical, and control component design criteria. In addition, AFW systems often appear in the dominant accident sequences in various probabilistic risk assessment programs.
>
> The CRVP system was selected because it too represents an interrelationship of several design criteria and interfaces. Specifically, it involves interface with a service-related contractor, interface of PGandE internal design organizations, and interface with the control room habitability criteria. It also represents a contrast of design methods since it is an air system rather than a water system.
>
> The safety-related portion of the 4160 V electrical distribution system was selected because it is the basic power supply for safety-related electrical equipment. It also represents an interrelationship of several design criteria and involves the interfaces among several PGandE internal design organizations.
>
> The three sample systems were designed by different engineering groups within PGandE, thus providing for evaluation of a broad spectrum of the PGandE engineering organization.
>
> In addition, the IDVP selected two areas of safety-related analyses for review: the integrated dose analyses; and the temperature,

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calculations or analyses using models generally different than those employed in the original design.[74] When the IDVP identified a concern (*e.g.*, a design error) that it believed was generic (*i.e.*, having the potential for being repeated in other systems), this concern was then addressed by the

(Footnote Continued)
pressure and humidity analyses as they affect environmental qualification of equipment. These analyses were selected since this work was done almost exclusively by three service-related contractors and utilized by PGandE. The service-related contractors were different and their work involved a flow of design information through PGandE engineering groups.

For the three selected sample systems, a complete vertical verification of the system design was performed. The applicable licensing criteria were identified, and a system design chain was developed. The system's design was then reviewed to determine if the licensing criteria were satisfied. The review included the aspects of mechanical, electrical and instrumentation and control design.

In addition, the IDVP performed the following verifications of the sample systems. The IDVP verified the fire protection provided for the sample systems, including the separation, fire barriers, suppression and detection systems provided in areas containing sample system components. The IDVP verified that the AFW and CRVP systems were adequately protected from the effects of a high energy line break (HELB), high energy line crack (HELC), and moderate energy line break (MELB).

Cooper *et al*. Tr. fol. D-1459 at 1/2-21 to -23.

[74] *Id*. at 1/2-35.

ITP for <u>all</u> applicant designed systems.[75] In turn, the ITP's verification work was sampled by the IDVP and the results reported in an ITR.[76]

The IDVP verification samples for nonseismic design encompassed the work of the primary applicant engineering design groups (civil, mechanical, electrical, instrumentation and control, and heating and ventilation). It also covered the work of the three major service contractors in the nonseismic area: Quadrex (formerly Nuclear Services Corporation) - jet impingement and pipe whip analysis; EDS Nuclear Inc. - heating and ventilation system design and other activities; and Radiation Research Associates - radiation dose calculations.[77]

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[75]The ITP addressed the following concerns identified by the IDVP:

> all areas of analyses of pressure, temperature and humidity due to HELB; selection of system design pressure and temperature; selection of differential pressure across power operated valves; redundancy of power supplies for shared systems; separation and single failure criteria for mutually redundant circuits; and jet impingement effects of HELB inside containment.

<u>Id</u>. at 1/2-24.

[76]App. Exhs. 137 to 141.

[77]App. Exh. 90, IDVP Final Report, Vols. I and II, at 4.2.2-6 to -8, 4.7.1-1 to 4.7.7-5.

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In addition to reviews resulting from the identifica-

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The IDVP did not review the work of all service contractors. For example, it did not review the work of: Westinghouse, Western Canada Hydraulic Laboratories (Western Canada), Stafco Associates (Stafco), and the IDVP contractors, Cloud Associates and Teledyne. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.4-3; Reedy Tr. D-1486. We have reviewed each of these excluded contractors and conclude that because of the circumstances in each case, the exclusions were reasonable and do not render the verification efforts inadequate as claimed by the Governor and joint intervenors. As we discuss <u>infra</u>, pp. 77-82, Westinghouse had its own properly functioning quality assurance program that assured the adequacy of both the services it performed and the equipment it designed for the applicant. Although Western Canada did not have a proper quality assurance program (App. Exh. 157, SECY 82-414, Encl. 5, p. 5), Western Canada's work in vortex analysis -- the same work it performed for the applicant -- had been audited in a generic review and found sufficient by the NRC staff. App. Exh. 101, ITR 9, at A 52; Cooper Tr. D-1478-79, D-1481-82, D-1750-51. Stafco assisted in the preparation of a list of safety-related structures, systems and components and in updating the FSAR. Because Stafco did not perform design work, it was properly excluded from the design verification program. Reedy Tr. D-1486, D-1488. Finally, with respect to the IDVP participants, Teledyne had a satisfactory quality assurance program that attests to the sufficiency of its design work. App. Exh. 157, SECY 82-414, Encl. 5 at 4. In any event, the ITP reviewed the seismic work previously performed for the applicant by Teledyne. App. Exh. 91, ITP Phase I Final Report, at 2.2.3-5. Cloud Associates, on the other hand, did not have a quality assurance program. App. Exh. 157, SECY 82-414, Encl. 5 at 5. Of the three projects Cloud Associates performed for Diablo Canyon (a review of pipe whip restraints, a systems interaction program, and a research program on seismic capability of nonseismic design components), only pipe whip restraint comprised design work that would normally have been subject to review by the IDVP, but was excluded because Cloud Associates was a member of the IDVP. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.4-3; App. Exh. 156, SECY-82-89, Encl. 1 at II, p. 4. That Cloud Associates was not reviewed is not now important. As part of the complete seismic review, the ITP re-evaluated all rupture restraints
(Footnote Continued)

tion of concerns by the IDVP in the nonseismic design area, the ITP independently performed a functional design review of the applicant-designed, safety-related mechanical, electrical and ventilation systems. The instrumentation and controls for mechanical systems, and all of the safety-related mechanical, electrical and ventilation systems were reviewed to assure adequate protection against a series of postulated hazards.[78] This nonseismic evaluation was performed in accordance with an NRC-approved quality assurance program meeting the criteria of Appendix B.[79]

(Footnote Continued)
inside and outside containment to assure they were properly designed and installed. Staff Exh. 37, SSER 19, at C.4-2 to -3. Thus, the exclusion of these five service-related contractors does not render the applicant's verification efforts insufficient.

[78]Anderson et al. Tr. fol. D-224 at 17-19; Kaplan and Anderson Tr. fol. D-1161 at 64-66.

[79]Anderson et al. Tr. fol. D-224 at 7; Staff Exh. 36, SSER 18, at C.2-3 to -4; Dick et al. [This panel consisted of C. Dick, M. Jacobson, S. Skidmore and T. de Uriarte.] Tr. fol. D-847 at 9.

The Governor seeks to have us discount (as an applicant trial ploy not worthy of belief) that portion of the ITP review work that was not performed for, and reviewed by, the IDVP. Gov. PF at 31-34. The Governor argues that prior to the hearing none of this ITP work was represented by the applicant as an additional verification effort and, in any event, the ITP review was neither documented to the same extent as the IDVP reviews, nor done to the same depth as the IDVP work. But the existence of the separate ITP review is evident from the applicant's semi-monthly reports as early as February 1982, and contrary to the Governor's assertions, the applicant's June 1983 Phase II Final Report

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While the scope of the nonseismic review of the Diablo Canyon safety-related systems was not as complete as the seismic review, an appreciable portion of the nonseismic design was verified.[80] There were errors identified that

(Footnote Continued)
clearly identifies this ITP review effort. App. Exh. 92, ITP Phase II Final Report, at iv. Moreover, the fact that the verification work is not documented in the same fashion as the work carried out in conjunction with the IDVP is a reflection of the fact that the latter program had reporting requirements imposed upon it by the Commission and the NRC staff. The ITP review work is recorded in the applicant's files and open items (i.e., errors) found during the course of this review are discussed in the Phase II final report. Anderson Tr. D-1426. App. Exh. 92, ITP Phase II Final Report, at 3-22 to -31. Thus, absent a valid showing that the work is flawed, or an objection to its admission as evidence, neither of which was made, the ITP's functional design review stands as significant evidence of the adequacy of the Diablo Canyon nonseismic design.

[80]The applicant's witness Anderson estimated that the total nonseismic design review (IDVP plus ITP) encompassed about seventy-five percent of the engineering work at Diablo Canyon. Tr. D-1419-20, D-1425. He readily admitted, however, that his figure could be characterized as rather "soft." Tr. D-1441, D-1426-27, D-1429-33. The Governor and the joint intervenors take issue with Mr. Anderson's estimate. Gov. PF at 31-35; JI PF at 2-3. They object because the figure is an estimate, not a precise number, and because the ITP functional design review component of Mr. Anderson's estimate was neither mandated nor reviewed by the IDVP. They assert that if the latter component of the estimate is discarded, the IDVP only reviewed twenty-three percent of the design elements of the nonseismic work. This argument overlooks the review by the ITP performed at the direction of the IDVP (see pp. 36-37, supra), and as previously indicated, there is no reasoned basis for discarding the ITP functional design review. See n.79, supra, and accompanying text. Further, Mr. Anderson's seventy-five percent estimate dealt with total engineering work covered by both the IDVP and ITP reviews, not design elements. Anderson Tr. D-1419-20, D-1427, D-1436,
(Footnote Continued)

required reanalysis and, in some instances, physical modifications were necessary in order to comply strictly with licensing criteria,[81] but all the errors were judged to be of minor safety significance.[82]

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D-1438-39. The two are vastly different. There are numerous design elements of varying significance in the engineering work involved in a project of this magnitude.

Moreover, it is not the exact quantification of work reviewed that is critical. The important consideration is that the scope and implementation of the nonseismic verification program was sufficient to test thoroughly the design process in order to discover any defects in that process. Here, the applicant's verification program encompassed three systems in their entirety (covering the spectrum of applicant's in-house design groups and the interrelationships of all significant design criteria and interfaces) and parts of all the remaining nonseismic systems. This slice of the nonseismic design process was sufficient to uncover any significant inadequacies in the design process.

[81]See nn.46 and 47, supra.

[82]The Governor and joint intervenors object to this characterization of the nonseismic design errors that were discovered because no formal analysis was performed to assess their seriousness or their potential for reducing the plant's margin of safety. Gov. PF at 16-17; JI PF at 15. They contend that the latter determination requires the performance of a probabilistic risk assessment. See Apostolakis Tr. fol. D-2313 at 10-11. But neither the Governor nor the joint intervenors presented any direct evidence to dispute the expert opinions of the staff and applicant witnesses that none of the errors found by the verification program was safety significant. Anderson et al. Tr. fol. D-224 at 12-14; Anderson Tr. D-345-46, D-1420; Knight Tr. D-2696-97, D-2819; Cooper et al. Tr. fol. D-1459 at 1/2-32. We find that the expertise of the applicant's and the staff's witnesses in the design, construction and operation of nuclear power plants qualifies them to evaluate the safety significance of such nonseismic errors, at least
(Footnote Continued)

The IDVP's sampling method involved a complete (vertical) review of three dissimilar systems, followed by an analysis by the ITP (across all systems-horizontally) to search for generic problems suggested by errors found in the vertical review. By using this approach, the IDVP was able to obtain a broad and comprehensive understanding of the nonseismic design of Diablo Canyon Unit 1.[83] As a result of this review, the IDVP concluded that while there may exist errors in the unsampled portion of the design, the likelihood of the existence of a safety significant error was small.[84]

To a certain degree, this judgment was tested by the ITP's own review and found valid. The ITP review sampled an additional portion of the safety-related, nonseismic design of the plant and, once again, it found only a small number of errors, none of which was safety significant. This led

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to the point of determining whether the errors warrant a quantitative evaluation. While we agree that as a general proposition only a formal analysis can provide a quantitative assessment of an error's significance, our review of the nonseismic errors identified by the IDVP and ITP leads us to concur in the judgment of the applicant's and the staff's experts that the errors are of minor safety significance.

[83] Cooper _et al_. Tr. fol. D-1459 at 1/2-25.

[84] _Id_. at 1/2-32.

the ITP to conclude, with a high degree of assurance, that the nonseismic design of the plant was adequate.[85]

We are not without some puzzlement as to why, having reviewed so much of the plant, the applicant did not carry through to a total review of the nonseismic design. In light of the history of the Diablo Canyon facility and the considerable time and resources already expended by the applicant on the verification programs, such an additional undertaking might well have proven a provident step in order to dispell the inevitable speculation as to the adequacy of the unreviewed portions of the nonseismic design. Nevertheless, on the basis of all the evidence, we find that the verification efforts of the IDVP and ITP were sufficient to provide adequate confidence that the nonseismic design criteria have been met for Unit 1. Through sampling which covered all the engineering disciplines and types of analyses, and which encompassed a major portion of the plant's design, the IDVP and ITP concluded that the original design process was efficacious. The NRC staff concurred with this conclusion. The errors found were few, of minor significance, and did not indicate a pervasive weakness in any design area. We concur in the judgments of the ITP,

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[85] Anderson et al. Tr. fol. D-224 at 19-21; App. Exh. 92, ITP Phase II Final Report, at 5-1.

IDVP and staff that the level of assurance provided by the applicant's verification efforts is comparable to that which would be afforded by a properly functioning quality assurance program.[86]

B. Issue 3(f)(iii) concerns a possible phenomenon known as "uplifting." In theory, uplifting may occur when a seismic event produces a high horizontal acceleration. In some circumstances this acceleration tends to produce a shift in the center of mass of a rigid structure, perhaps to the extent of causing a building tilt. In a rigid building such as the Diablo Canyon containment, if uplifting occurs, one side of the base mat would lift away from the underlying rock or soil thereby causing increased vertical acceleration in the structure. No specific analysis was done by the ITP or IDVP concerning the effect on equipment of increased vertical acceleration caused by uplift. Only the effect of

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[86] We note that, in theory, a design quality assurance program will provide 100 percent review of the design work. The record is clear, however, that such a program can never assure that there will be no design errors. Anderson and Kaplan Tr. D-1176-81; Hubbard Tr. D-2130-31, D-2134-35; Apostolakis Tr. D-2376-77. Indeed, Appendix B only provides that the purpose of a quality assurance program is "to provide adequate confidence that a [safety-related] structure, system or component will perform satisfactorily in service." See 10 CFR Part 50, Appendix B, Introduction. Here, the applicant's verification program has provided that level of confidence.

uplift on the containment mat was studied by the ITP.[87] Joint intervenors and the Governor assert that the applicant should have analyzed the effects of uplift on equipment qualification in the Diablo Canyon containment.[88]

The uplift phenomenon is a relatively recent concept that evolved from discussions among seismologists, rather than from observation of seismic events.[89] It has never been identified as a source of actual damage to a structure and there is no NRC regulation or staff guidance requiring that the seismic analyses for a nuclear power plant include this phenomenon.[90]

The Governor's expert witness, Dr. Jose M. Roesset, opined that some uplifting of the Diablo Canyon containment would occur at the peak ground acceleration established for the site.[91] According to Dr. Roesset, such uplift would

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[87]White Tr. D-828.

The applicant's analysis of the possible uplift of the containment mat found that the maximum stress on the reinforcing steel was within allowable limits stated in the FSAR and Hosgri Report. Seed et al. Tr. fol. D-652 at 71-72.

[88]Gov. PF at 43-45; JI PF at 20-21.

[89]Seed Tr. D-687-88; Roesset Tr. fol. D-2206 at 6.

[90]White Tr. D-669-71, D-680-682; Seed Tr. D-684; Cloud Tr. D-1890; Polk Tr. D-2506.

[91]Roesset Tr. fol. D-2206 at 7-8.

(Footnote Continued)

amount to only a fraction of an inch and would cause only a small (approximately ten percent) increase in vertical acceleration. Such uplift also would result in a negligible increase in seismic displacement and velocity.[92] He noted further that most of the effects of uplift are beneficial.[93]

The applicant's expert witnesses were unwilling to concur in Dr. Roesset's opinion that uplifting of the Diablo Canyon containment would occur at design basis ground accelerations.[94] Rather, these experts were only willing to concede that uplift was possible at such accelerations.[95]

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In part, Dr. Roesset based his opinion on the work of R.P. Kennedy that indicated an increase in the response spectra of a high temperature gas-cooled reactor (HTGR) in the high frequency range, and on the work of J.P. Wolf that indicated the possibility of uplift of a "typical reactor" building at a peak ground acceleration of 0.167g if the containment were on a rock base. Roesset Tr. fol. D-2206 at 7; See Kennedy, Short, Wesley and Lee, Effect of Non-Linear Soil Structure Interaction due to Base Slab Uplift on the Seismic Response of a High-Temperature Gas-Cooled Reactor (HTGR), 38 Nuclear Engineering and Design, No. 3 (1976); Wolf and Skrikeru, Seismic Excitation With Large Overturning Moments: Projecting Base Mat or Lifting-off?, paper presented at the Conference on Structural Analysis, Design and Construction in Nuclear Power Plants, Porto Alegre, Brazil (Apr. 1978).

[92] Roesset Tr. D-2273-74, D-2276-77.

[93] Roesset Tr. fol. D-2206 at 8; Tr. D-2271.

[94] Seed Tr. D-687; White Tr. D-668-69; Holley Tr. D-1874-76; Biggs Tr. D-1881.

[95] See e.g., White Tr. D-671, D-675; Seed Tr. D-687;
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But, like Dr. Roesset, the applicant's experts (as well as those of the staff) agreed that should uplift occur at Diablo Canyon it would amount to only a fraction of an inch and would cause only very small increases in the vertical acceleration of the reactor building.[96] Thus, there is agreement among all the expert witnesses that the effects of uplift on the vertical accelerations of the Diablo Canyon reactor building would be extremely small.

The equipment inside the containment is seismically qualified for a total acceleration obtained by taking the square root of the sum of the squares of each of three

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Holley Tr. D-1874-75.

In general, the applicant's expert witnesses did not endorse Dr. Roesset's opinion that uplift would occur because, if the mathematical model was expanded to include all of the relevant factors, uplift would most likely not be shown to occur. In other words, the less detailed model relied upon by Dr. Roesset necessarily predicts the phenomenon. White Tr. D-668, D-671; Holley Tr. D-1874-76. See also Polk Tr. D-2503. In reality, uplift has never been found to occur in a structure. White Tr. D-669-70; Seed Tr. D-684. To model the uplift phenomenon properly would be an exceedingly complex and time consuming task. Seed Tr. D-687-89, D-694; White Tr. D-682, D-691. Moreover, the Diablo Canyon reactor building base mat is constructed with a deep concrete key poured into a rock foundation that would have to be torn before uplift could occur. White Tr. D-691; Seed and White Tr. D-695-96.

[96] White Tr. D-672, D-682; Holley Tr. D-1876; Biggs Tr. D-1882-83; Miller Tr. D-2507-508; Kuo Tr. D-2504-505.

accelerations (two horizontal and one vertical).[97] A small increase in the vertical acceleration on the order of that resulting from any uplift of the Diablo Canyon containment (i.e., ten to fifteen percent) would cause an insignificant increase in the total acceleration obtained from the combination of the horizontal and vertical accelerations.[98] This result is reduced even more because the uplift acceleration is not in phase with the seismic vertical acceleration and cannot be considered additive to the peak vertical acceleration.[99] Moreover, the equipment inside the containment is already seismically qualified within margins more than sufficient to accommodate any increase in vertical acceleration as a result of uplift.[100] We find, therefore, that, even if uplift should occur, its detrimental effects would be insignificant. The applicant need not include as part of its seismic verification any seismic modeling and

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[97] Biggs Tr. D-1882-83; Kuo Tr. D-2514; Cloud Tr. D-1886-87.

[98] Biggs Tr. D-1882-83; Kuo Tr. D-2514.

[99] Biggs Tr. D-1885; Roesset Tr. D-2282.

[100] Cloud Tr. D-1886-87; Knight Tr. D-2512-14.

Dr. Roesset called for an analysis of the effects of uplift on the equipment inside containment for the "sake of completeness." Tr. D-2273-74. But he conceded that he did not know what equipment was in containment and how it was qualified for vertical acceleration. Tr. D-2214-15.

analysis of the effects of uplift on equipment inside the reactor building.

C. Issues 3(f)(iv) and (v) concern the modeling of soil springs for the Diablo Canyon auxiliary building. The term "soil springs" is applied to the methodology used in seismic analysis to represent motion resistance and damping characteristics of the foundation media around a structure. In other words, the soil media is assumed to act like a spring in a seismic event. Here, the Governor and joint intervenors complain that the soil properties used to establish soil spring constants were not properly specified and that the use of soil springs did not adequately account for all soil structure interactions.

At Diablo Canyon, the auxiliary building has foundations at elevations 85 feet and 100 feet above sea level. In modeling the building for the seismic reanalysis, the ITP assumed the base to be rigid at the lower (85-foot) elevation and soil springs were used to represent the intervening rock between that level and the 100-foot level. The Hosgri Report and the FSAR allow a rigid or fixed-base analysis for stiff rock, indicated by a shear wave velocity at or above 3500 feet per second (fps).[101] Soil springs are used to model less stiff soil.

The ITP, relying on data supplied by Harding Lawson Associates,[102] determined the shear wave velocity at 100 feet to be about 2700 fps, supporting the use of soil springs at this elevation.[103] The ITP had examined the Harding Lawson data and found it to be reasonable and comparable to data obtained by other companies doing work involving the soil under the auxiliary building.[104] As part of its review, the ITP performed parametric studies in which the soil geometry and the soil springs under the 100-foot foundation were varied for a number of stiffness values. The shear wave velocities ranged from 6,000 fps (very rigid) to 2,000 fps (less rigid), with the value 2,775 fps serving as

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[101] See FSAR Section 3.7A at 6.

Soil stiffness may be determined from measured values of shear wave velocity of the soil.

[102] The engineering firm of Harding Lawson Associates did the major soils analyses (i.e., geotechnical studies) for the Diablo Canyon site. A review conducted as part of the seismic verification program found that Harding Lawson Associates had not implemented a quality assurance program for the soil work for Diablo Canyon. App. Exh. 155, ITR 68, at 2.

[103] Seed *et al*. Tr. fol. D-652 at 20-21; White Tr. D-700-01.

[104] White Tr. D-774; Seed and White Tr. D-811-13.

the base case.[105] The results of these analyses showed that there was little variation in the shear stresses for the auxiliary building walls and that generally the base case yielded the highest values.[106] In other words, the auxiliary building was found to be qualified for shear forces associated with all credible soil stiffness properties.

Because Harding Lawson Associates had not implemented a quality assurance program for their work at Diablo Canyon, the IDVP developed an extensive program to verify the soils work.[107] That program again verified the reasonableness and reliability of the soils data.[108] The IDVP also reviewed the ITP's auxiliary building analysis and found it acceptable.[109] We find that the ITP properly addressed the

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[105] App. Exh. 145, ITR 55, at 20, 21 (Table 3), 23 (Table 5); Seed and White Tr. D-700-706.

[106] White Tr. D-706; App. Exh. 145, ITR 55, at 23 (Table 5).

[107] App. Exh. 155, ITR 68, at 2, 4-5; Cloud Tr. D-1942-43, D-1997-99, D-2013-14.

[108] Cloud Tr. D-2002-2003, D-3124; Cloud et al. [This panel consisted of R. Cloud, J. Biggs, M. Holley and R. Wray.] Tr. fol. D-1843 at 3-8.

[109] App. Exh. 90, IDVP Final Report, Vol. I, at 4.4.2-9 to -10; Cloud Tr. D-1848-49.

The IDVP performed parametric calculations similar to the ITP's to determine the spring constants and their (Footnote Continued) soil properties in its modeling of the auxiliary building and that the model used was appropriate.

The Governor and the joint intervenors assert, however, that the ITP erred by using soil springs in modeling the auxiliary building at the 100-foot elevation. They claim a fixed-base analysis at that elevation should have been used and that such an analysis would show increased shear wall forces that have not been analyzed.[110] The Governor's and joint intervenors' position is not supported by the evidence.[111] The foundation material at elevation 100 feet has a shear wave velocity of 2500 to 2700 fps (not 3500 fps as the Governor and the joint intervenors assert); accordingly, the ITP's use of soil springs in its modeling

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results were in reasonable agreement with those of the ITP. App. Exh. 145, ITR 55, at 25; Cloud Tr. D-1905. The staff also concurred in the ITP's conclusions. Kuo et al. [This panel consisted of J. Knight, P. Kuo, H. Polk, C. Miller, A. Philippacopoulos, C. Costantino and P. Wang.] Tr. fol. D-2463 at 16-18.

[110]Gov. PF at 40-41; JI PF at 21-22.

[111]Although the Governor and the joint intervenors do not identify the source for their claim, they apparently reach their conclusion that the ITP should have used a fixed-base analysis for the 100-foot elevation by taking out of context from App. Exh. 145, ITR 55, at 24 (Table 7), "IDVP Soil Parameters," the IDVP's best estimate of 3500 feet per second shear wave velocity for the auxiliary building foundation material. They ignore, however, the IDVP's explanation of those soil parameters and the IDVP's conclusion that the values of the soil springs used by the ITP were acceptable. App. Exh. 145, ITR 55, at 25.

for that elevation is justifiable.[112] Moreover, the parametric studies performed as part of the seismic reanalysis demonstrate that there would be a negligible change in shear wall forces even if the foundation at the 100-foot elevation were to be considered fixed. All such forces are well within the margins for which the shear walls are qualified.[113]

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[112]See p. 50, supra.

The Governor's expert agreed that if the soil properties printed in App. Exh. 145, ITR 55, at 23 (Table 5) were correct, then the ITP's analysis was proper. Roesset Tr. D-2217-19, D-2249-50. Subsequently, one of the applicant's expert witnesses rechecked those numbers and found them correct. Cloud Tr. D-3111.

[113]App. Exh. 145, ITR 55, at 21 (Table 3), 23 (Table 5); Cloud et al. Tr. fol. D-1843 at 3-9; Biggs Tr. D-1907-08; White Tr. D-713-19.

Alternatively, the Governor and the joint intervenors argue that the ITP should have considered a softer soil spring in modeling the auxiliary building because the material underlying the foundation has a shear wave velocity of only 1500 fps. They claim that, in the event of an earthquake, the auxiliary building would be subject to rotational effects (i.e., a rocking motion) because the structure is embedded at different elevations in soils of widely varying stiffness. Gov. PF at 41-43; JI PF at 21-22. This argument, like the other, is unsupported by the evidence and footed on an inappropriate reference. The Governor and joint intervenors rely upon App. Exh. 155, ITR 68, Figure 15 at 81 to conclude that the material underlying the auxiliary building is soft. But this figure applies to the soil underneath the diesel fuel oil tanks, not the auxiliary building. App. Exh. 155, ITR 68, at 28-30; Cloud Tr. D-3110-11; White Tr. D-3136. Further, the seismic refraction tests used to produce the data in Figure 15 are not as reliable as cross-hole and up-hole testing techniques

(Footnote Continued)

D. Issue 3(o) questions the modeling of the fuel handling building. Specifically, it challenges the use of the translational and torsional response of the auxiliary building as input to the fuel handling building and the number of dynamic degrees of freedom used in the model.

The fuel handling building is, in essence, a small superstructure to the auxiliary building and will experience the motion of the auxiliary building during a seismic event.[114] The ITP therefore modeled the auxiliary building, including the fuel handling building, and then used the appropriate response of that building as input to a separate model of the fuel handling building in order to determine local responses.[115] The IDVP concluded that the modeling

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(Footnote Continued)
used to generate the data relied upon by the ITP. Cloud Tr. D-1996-2002, D-3112, D-3122, D-3125; White Tr. D-3136; Roesset Tr. D-2269. Accordingly, the ITP need not have considered such a soft soil spring in its modeling.

[114] Seed _et al_. Tr. fol. D-652 at 81-82; Cloud _et al_. Tr. fol. D-1843 at 3-18.

[115] Seed _et al_. Tr. fol. D-652 at 81-82; Cloud _et al_. Tr. fol. D-1843 at 3-18.

The seismic analysis of the auxiliary building, including the fuel handling building superstructure, was performed by using a lumped mass-spring model. Seed _et al_. Tr. fol. D-652 at 81. The model used a five percent eccentricity of mass to account for the effects of accidental torsion and the appropriate translational time-history was applied at its base. _Id_.

The fuel handling building was decoupled from the
(Footnote Continued)

was consistent with good engineering practice and acceptable.[116] The staff concurred in the IDVP's conclusion.[117]

In order to analyze the fuel handling building, the ITP first developed a large static model, which was then divided into two smaller dynamic models. The total number of degrees of freedom was reduced to make the analysis more manageable.[118] This reduction was accomplished by using standard industry procedures.[119] A sufficient number of

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(Footnote Continued)
auxiliary building and analyzed separately using three-dimensional finite element models. The seismic input motions at the base of these models consisted of acceleration time-histories (translational and torsional) from the auxiliary building dynamic analysis developed at the center of mass at elevation 140 feet. The geometric eccentricity of the fuel handling building relative to the 140-foot elevation center of mass was accounted for by applying the translational time-history together with the eccentric distance times the torsional time-history. Id.

[116] Cloud et al. Tr. fol. D-1843 at 3-18 to -19.

[117] Kuo et al. Tr. fol. D-2463 at 21.

[118] Staff Exh. 36, SSER 18, at C.3-26; Seed et al. Tr. fol. D-652 at 81-83; Cloud et al. Tr. fol. D-1843 at 3-18 to -19.

[119] Cloud et al. Tr. fol. D-1843 at 3-18 to -19. See also NUREG-0675, Supplement No. 20, Safety Evaluation Report related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2 (Dec. 1983) at C.3-6 to -7.

dynamic degrees of freedom were included to determine adequately peak accelerations.[120] We find, therefore, that the modeling of the fuel handling building was appropriate.[121]

E. Issue 3(q) concerns the soils analyses for the buried diesel fuel oil tanks at the facility site. The soils analyses for the diesel fuel tanks, like those for the auxiliary building dealt with in issue 3(f)(iv), were done by Harding Lawson Associates.[122] They performed the

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[120]Cloud _et al_. Tr. fol. D-1843 at 3-19.

[121]At the time of the hearing, no written confirmation of a certain aspect of the ITP's input into the model of the fuel handling building had yet been provided to the staff, as the staff had requested. Miller Tr. D-2528-29. On the basis of the applicant's oral representation, the staff concluded that the ITP procedures were acceptable. Kuo _et al_. Tr. fol. D-2463 at 20-21. Because no written confirmation had been received from the applicant, however, the Governor states that "no findings are yet possible." Gov. PF at 45. That is not the case. The applicant has fully met its burden of proof on this issue. The fact that the staff sought written confirmation from the applicant that separate time histories were applied at the base of each column in the fuel handling building model is irrelevant. Though it is without effect on our findings, we note that the written confirmation requested by the staff has since been provided by the applicant. See letter from J. Schuyler, PG&E, to G. Knighton, NRC, dated November 17, 1983, at 2-3. Moreover, by not filing adequate proposed findings on the issue, the Governor, in effect, has waived it. The joint intervenors filed no proposed findings on this issue. See n.18 and accompanying text, _supra_.

[122]Because Harding Lawson had not implemented a quality assurance program for their original work at Diablo Canyon, the IDVP performed an extensive review of the soils work for

(Footnote Continued)

original seismic qualification analyses for the diesel tanks. In 1983 as part of the ITP seismic verification, Harding Lawson reanalyzed them.[123] The IDVP then reviewed the Harding Lawson re-qualification analyses and conducted several alternative analyses including parametric studies covering a range of soil properties for the backfill around the diesel fuel tanks.[124] The IDVP found the Harding Lawson work acceptable and concluded that the diesel tanks meet licensing criteria.[125]

The Governor and the joint intervenors assert, however, that the variation in the Harding Lawson soils data for the backfill around the diesel tanks, and the variation between the data for the rock underlying the fuel tanks and that under the auxiliary building, demonstrate that the original data are unreliable and should not be used for qualifying

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(Footnote Continued)
the fuel tanks and found it reasonable and acceptable. Cloud _et al_. Tr. fol. D-1843 at 3-23 to -24; App. Exh. 90, IDVP _Final_ Report, Vol. II, at 4.9.2-1 to -2, -6 to -9; App. Exh. 155, ITR 68, at 28-30, 36-37, 41. That soils works was also reviewed by the ITP. White Tr. D-767, D-774; Seed Tr. D-770, D-772-73.

[123] App. Exh. 155, ITR 68, at 33-34.

[124] _Id_. at 34-40; Cloud _et al_. Tr. fol. D-1843 at 3-23.

[125] App. Exh. 155, ITR 68, at 41; App. Exh. 90, IDVP Final Report, Vol. II, at 4.9.2-6 to -9; Cloud _et al_. Tr. fol. D-1843 at 3-24.

the tanks.[126] We disagree. The Harding Lawson soils data were checked and rechecked, and the IDVP's parametric studies demonstrate that the qualification of the fuel tanks is not sensitive to the variation in the backfill soil properties about which the Governor and the joint intervenors complain.[127]

The properties of the rock under the diesel tanks and auxiliary building vary because the structures are in different locations.[128] The seismic analyses for the fuel tanks included properties for the underlying rock obtained using seismic refraction tests. These tests are considered to be relatively unreliable for the Diablo Canyon site, and would give results indicating the rock is less rigid (softer) than is actually the case.[129] The use of soft rock properties in the seismic analysis results in greater calculated strains in the tanks than would the use of more rigid rock. Thus, the analyses performed were conservative.[130] Additionally, the rock below the tanks

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[126] Gov. PF at 46-48; JI PF at 22-23.

[127] App. Exh. 155, ITR 68, at 29, 37, 41; Cloud _et al._ Tr. fol. D-1843 at 3-23 to -24; Cloud Tr. D-1982-3, D-1988.

[128] Cloud Tr. D-1998, D-3112, D-3122-3; White Tr. D-3136.

[129] Cloud Tr. D-1998-99; see n.113, _supra_.

[130] Cloud Tr. D-2001, D-3124-25.

has a small effect on the tanks' response.[131] Accordingly, we find that the data used for the diesel fuel tank analyses were adequate to demonstrate that the tanks are properly qualified.[132]

F. Like issue 3(q), issue 3(r) also questions the soils analysis of backfill material.[133] In particular, the Governor and the joint intervenors challenge the soils analysis of the backfill covering the circulating water intake (CWI) conduits and the auxiliary saltwater (ASW) piping used in the seismic qualification of these components.[134]

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[131] App. Exh. 155, ITR 68, at 41.

[132] The joint intervenors also question the soil properties underlying the diesel fuel oil tanks. The joint intervenors have misapprehended the data. They compare an equation for compressional wave velocity for the rock under the diesel fuel tanks (App. Exh. 155, ITR 68, at 38) with one that expresses shear wave velocity under the circulating water intake conduits and auxiliary saltwater piping (App. Exh. 155, ITR 68, Figure 25 at 91). These equations must be converted to the same component of wave velocity before any comparision is made.

[133] Gov. PF at 48-52; JI PF at 23-24.

[134] Each of the two circulating water intake conduits (one for each unit) is a 16 ft. by 30 ft. reinforced concrete structure containing two parallel, essentially square, tunnels (approximately 12 ft. by 12 ft.). The conduits parallel one another and extend approximately 1600 feet from the turbine building to the intake structure. The conduits are located in trenches excavated into rock and covered on top with some twenty feet of backfill. Two 24-inch diameter steel auxiliary saltwater pipes, placed one

(Footnote Continued)

The soils analysis for the ITP's seismic qualification of the CWI conduits and the ASW piping was performed by Harding Lawson Associates. As we have previously found, the Harding Lawson soils work was reviewed by the IDVP and found acceptable.[135] In this instance, Harding Lawson took test borings of the backfill and then performed laboratory tests on the specimens to arrive at soil property values.[136] The IDVP reviewed the seismic analyses and found that the Harding Lawson soil and rock properties were acceptable and that the CWI conduits and ASW piping meet licensing criteria.[137]

(Footnote Continued)
over the other, run parallel to and on one side of each of the CWI conduits in a narrow, shallow trench. One side of the trench is formed by the concrete sidewall of the CWI and the other by rock. The bottom of the trench consists of a concrete lip projecting from the CWI concrete sidewall. The ASW pipes are attached to this concrete lip at 40 foot intervals and are surrounded in the trench by compacted sand. The ASW pipes are then covered with the same backfill material as the CWI conduits. App. Exh. 155, ITR 68, at 42 and Figure 21, at 87; Seed Tr. D-837-40.

[135] See pp. 51-52, _supra_.

[136] App. Exh. 155, ITR 68, at 42-45.

As part of the seismic analyses, the data were used to plot soil shear modulus against strain and soil damping ratio as a function of strain. App. Exh. 155, ITR 68, at 89-90 (Figures 23 and 24). Those test data were then matched to a standard soils curve (_i.e._ Seed & Idriss 1970 sand curve) and that curve was employed in the qualification analyses. Seed _et al_. Tr. fol. D-652 at 85-86.

[137] App. Exh. 90, IDVP Final Report, Vol. II, at 4.9.2-10; App. Exh. 155, ITR 68, at 51.

The Governor and the joint intervenors charge that the soils data do not represent the actual backfill at the site over the conduits and piping, because no correction factor was applied to the data to adjust for the sample disturbance that occurs when laboratory values for soil properties are used. If a correction factor were applied, they claim, the soil properties would be different than the ones used in the qualification analysis.[138] The properties of the backfill over the CWI conduits and ASW piping about which the Governor and the joint intervenors complain are negligible factors, however, in the seismic qualification of these components.[139] The conduit and pipes are surrounded by rock on the sides and bottom (see n.134, supra) and the rock determines the seismic response of these components, not the backfill on top of them.[140] Because the effect of the backfill on the seismic response of these components is insignificant, it was acceptable for the ITP and IDVP to use the Harding Lawson data without correction factors in the

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[138]Gov. PF at 50-52; JI PF at 23-24.

[139]Seed et al. Tr. fol. D-652 at 86; White Tr. D-805; Seed Tr. D-836-40, D-3142-43.

[140]Seed et al. Tr. fol. D-652 at 86; Seed Tr. D-836-39, D-3142-43.

Indeed, the Governor's expert conceded that the effect on the seismic response of backfill over the conduits and piping was small. Roesset Tr. D-2254, D-2256.

seismic qualification of the CWI conduits and the ASW piping.

G. The joint intervenors allege in issue 4(i)(1) that the IDVP unjustifiably accepted a deviation from the FSAR fire protection licensing criteria for the room containing the motor driven auxiliary feedwater (AFW) pumps, because there is a large pipe chase covered with a grate in the ceiling of that room.[141] This purported deviation was discovered by the IDVP during its review of the auxiliary feedwater system and resulted in the issuance of EOI 8038. The EOI was issued because the description of the AFW pumproom in the FSAR, if read literally, was subject to misinterpretation (i.e., the ceiling was described only as a 2-foot thick concrete slab), and the existence of the ceiling pipe chase appeared to violate the FSAR fire zone separation licensing criteria.[142]

The ITP responded to the IDVP's concern by submitting a fire hazards analysis demonstrating that a fire is unlikely to propagate through the pipe chase because of the actual

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[141]The Governor failed to file proposed findings of fact on issue 4(i)(1). See n.18, supra and accompanying text.

[142]Krechting and Cooper Tr. fol. D-2040 at 4-16; App. Exh. 110, ITR 18, at 4-1, 5-3. See "Fire Protection Review, Units 1 and 2 Diablo Canyon Site" (Amendment No. 51 to operating license application) at 4-18 to -19.

combustible loading in the pumproom, the movement of air through the ceiling opening and the curbing surrounding the pipe chase in the area above. The ITP analysis confirmed that the plant was originally designed to meet the fire zone separation criteria with the open pipe chase in the AFW pumproom ceiling.[143] The IDVP concurred in the ITP's analysis. It agreed that the FSAR fire zone commitment had been met and that a fire in this area of the facility would not adversely affect the safe shutdown functions of the AFW system.[144] On the basis of the uncontroverted evidence of the origins of this purported deviation and the ITP's fire propagation analysis, we find -- like the IDVP -- that there is no deviation from licensing criteria because of the open pipe chase in the AFW pumproom.[145]

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[143]Connell et al. [This panel consisted of R. Anderson, E. Connell and G. Moore. W. Vahlstrom was subsequently added to the panel. Tr. D-531.] Tr. fol. D-487 at 22; App. Exh. 110, ITR 18, at 4-1, 5-3.

[144]App. Exh. 110, ITR 18, at 4-1, 5-3.

[145]The joint intervenors presented no witnesses or documentary evidence on this issue and did not cross-examine any of the applicant's witnesses on it. They claim, nevertheless, that the FSAR fire zone separation license criteria are not met based solely on a selective use of staff witness Kubicki's answers to their questions on cross-examination. See JI PF at 24-25. They assert that the existence of the open pipe chase precludes a complete fire barrier. Because the staff had no knowledge of the air flow patterns for the AFW pumproom, and it is possible for the products of combustion to travel through the pipe chase

(Footnote Continued)

H. Issue 4(l) addresses the adequacy of the applicant's analysis of possible jet impingement effects on the design and qualification of safety-related equipment and piping inside the containment. Because a break or crack in a line carrying high energy steam or water might result in damaging jets from the failed pipe, the NRC has long required that safety-related structures, systems and components in the vicinity of potential break locations be analyzed for (and, if necessary, protected from) the effects

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(Footnote Continued)
thereby propagating fire to another part of the plant, the joint intervenors contend there is no assurance the licensing criteria are met. Id. In addition to its other flaws, the joint intervenors' argument ignores the uncontradicted testimony of the applicant's witnesses that there are insufficient combustibles in the pumproom for a fire to propagate. Connell et al. Tr. fol. D-487 at 22. More important, however, is the fact that we do not rely upon the staff's testimony to reach our conclusion that there is no deviation from the FSAR fire zone separation criteria.

We note that the staff did not review the IDVP's analysis in reaching its conclusion that the open pipe chase in the ceiling of the AFW pumproom did not present a deviation from licensing criteria. Kubicki Tr. D-2873. Rather, the staff conducted an independent review of the Diablo Canyon fire protection program using 10 CFR Part 50, Appendix R (Fire Protection). As a result, the staff concluded that the pipe chase opening did not represent a significant degradation of the level of fire safety in the room and that any fire propagation would not represent a significant threat to adjoining areas. This conclusion was based on the defense in depth concept, administrative controls limiting combustibles, the existence of firewalls and the availability of automatic and manual fire protection systems. Kubicki Tr. D-2874-75.

of jet impingement.[146] In like manner, the Diablo Canyon FSAR requires that the applicant protect all safety-related structures, systems and components from the damaging effects of jet impingement.[147]

When the IDVP reviewed Diablo Canyon records, however, no documentary evidence of jet impingement analyses for safety-related systems, structures and components inside containment was found. Consequently, EOI 7002 was issued.[148] In response to this EOI, the ITP performed an extensive analysis of jet impingement effects of high energy line breaks.[149] That ITP analysis was verified by the IDVP on a sampling basis. In addition, the verification included a review of the ITP reanalysis procedure, a review of the ITP field review (including an independent walkdown to verify identification of safety-related targets) and a

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[146]See 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 4.

[147]FSAR Section 3.6.

[148]App. Exh. 140, ITR 48, at 3-1.

It should be noted that the degree of analysis and documentation of jet impingement required today is greater than for earlier-designed plants such as Diablo Canyon. Connell *et al*. Tr. fol. D-487 at 26.

[149]Connell *et al*. Tr. fol. D-487 at 25-26; Krechting and Cooper Tr. fol. D-2040 at 4-21 to -22.

review of the ITP safety evaluation of impinged targets.[150] On the basis of its verification (reported in ITR 48), the IDVP concluded that the ITP analysis met the FSAR licensing criteria.[151]

The Governor and the joint intervenors challenge the ITP analyses and the IDVP conclusion that the applicant has met the FSAR jet impingement licensing criteria, claiming that analyses were not performed for postulated breaks in each line inside the containment required by the FSAR.[152] In accordance with its interpretation of the FSAR, the ITP performed jet impingement analyses for high energy lines with a temperature above 200°Fahrenheit(F) <u>and</u> pressure above 275 pounds per square inch gage (psig).[153] The Governor and the joint intervenors assert that the FSAR

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[150] Krechting and Cooper Tr. fol. D-2040 at 4-21 to -22.

[151] App. Exh. 140, ITR 48, at 7-1.

After identifying several potential problems and seeking additional information from the applicant, the staff agreed that the applicant met the FSAR jet impingement licensing criteria. Staff Exh. 36, SSER 18, at C.4-29; Staff Exh. 37, SSER 19, at C.4-2. But the staff has yet to close its review of the applicant's jet impingement analysis. The matter still under investigation, however, is not pursued in issue 4(l). Staff Exh. 37, SSER 19, at C.4-2.

[152] Gov. PF at 54-56; JI PF at 25-27.

[153] Connell <u>et</u> <u>al</u>. Tr. fol. D-487 at 25-26; Connell Tr. D-584.

actually requires jet impingement analysis for postulated breaks in all lines exceeding 200°F <u>or</u> 275 psig. The difference between the two interpretations results in the exclusion of three lines inside containment from the ITP's analysis that the Governor and the joint intervenors would include.[154] Thus, those three lines were not analyzed as part of the ITP jet impingement analyses. These three lines were, however, "looked at" informally by the ITP.[155]

We believe the most prudent interpretation of the FSAR is that one which requires jet impingement analysis on lines having a temperature above 200°F <u>or</u> a pressure above 275 psig.[156] Therefore, in order to comply with the FSAR licensing criteria, the applicant must formally analyze (<u>i</u>.<u>e</u>., in the same fashion it analyzed the other lines inside containment) the three lines identified by its

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[154] Connell Tr. D-613-14.

[155] Connell Tr. D-616-17.

[156] The most rational, technically based criteria for postulating line breaks for the purpose of jet impingement analysis appear in the FSAR section that addresses lines outside of containment. Recognizing that jets may result from line cracks or breaks, this section requires such failures to be postulated for lines with service temperature above 200°F, or design pressure exceeding 275 psig. See FSAR at 3.6-16 to -17. This same rationale should control the analyses inside containment.

witness.[157] The applicant must report the results of its analyses to the staff and, if necessary, make any appropriate modifications prior to commercial operation.[158]

I. In issue 4(t), the joint intervenors assert that the IDVP accepted without proper justification a deviation from licensing criteria because the short circuit current for the circuit breakers on three 4160 volt (V) safety-related switchgear buses was calculated to be greater than the nameplate ratings on the breakers.[159] This situation was identified by the IDVP during the nonseismic design review of the 4160 V electrical distribution system and

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[157]Because the applicant's witness Connell did not elaborate upon his remark that these three lines "have been looked at, but they are not part of the formal jet program re-analysis . . . " (Tr. D-616-17) and there is no other evidence concerning these lines, we are unable to conclude on the basis of the record evidence that the applicant has complied with the FSAR licensing criteria.

[158]Although the applicant's witness did not identify the three lines inside the containment by name, it appears the lines were identified by the applicant in its response of October 12, 1983 to the NRC staff concerning the ITP jet impingement analyses. See letter from J. Schuyler, PG&E, to D. Eisenhut, NRC, dated Oct. 12, 1983 at 1-2; see also Staff Exh. 37, SSER 19, at C.4-2. If our assumption is correct that the lines are the excess letdown line, reactor coolant pump seal vent and leakoff lines, and reactor coolant pump seal water injection line, then it appears that the lines are small diameter ones with relatively low energy content that would not be expected to fail or to produce high energy jets.

[159]The Governor failed to file proposed findings of fact on issue 4(t). See n.18 and accompanying text, supra.

became the subject of EOI 8022.[160] The nameplate rating of the 4160 V circuit breakers is listed as 33 kiloamperes (kA), but the calculated short circuit current that the breakers might be required to interrupt is 39kA.[161] The IDVP declared the matter resolved when the ITP provided it with 1976 test data and a letter from the breaker manufacturer indicating the breakers can be relied upon to interrupt current up to 45kA.[162]

The joint intervenors object to the IDVP's resolution of this matter.[163] They assert that breakers are normally warranted only for the nameplate rating and, because it is not known whether the manufacturer is willing to warrant greater capacity for the breakers, we should find that the applicant's failure to install breakers with an adequate nameplate rating constitutes a violation of licensing criteria.

The FSAR, however, requires only that the applicant protect the emergency power supply with circuit breakers adequate to interrupt the calculated short circuit

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[160] App. Exh. 116, ITR 24, at 5-1.

[161] Moore Tr. D-524.

[162] Krechting and Cooper Tr. fol. D-2040 at 4-30; Krechting Tr. D-2052-55.

[163] JI PF at 27-28.

current.[164] Here, the manufacturer's 1976 test data demonstrate that the breakers in question will perform above the nameplate rating and interrupt the required short circuit current.[165] Moreover, in February 1983 the manufacturer explicitly responded to the applicant's inquiry concerning the interrupting capacity of the breakers stating that the breakers in question will interrupt 45kA.[166] It is clear, therefore, that the nameplate rating of the breakers in question is only a nominal rating and that the breakers will perform their intended function.[167] Accordingly, we find that the IDVP did not accept any deviation from licensing criteria.

J. In issue 5, the Governor and the joint intervenors charge that the applicant's verification program has failed to substantiate that the Diablo Canyon facility, as built,

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[164] See FSAR Section 3.1.

[165] Connell _et al_. Tr. fol. D-487 at 35; Moore Tr. D-524-25; Vahlstrom Tr. D-532.

[166] Krechting Tr. D-2054-55.

The question whether the manufacturer will warrant the breakers is, in fact, irrelevant, because nothing in the appropriate FSAR licensing criteria concerns manufacturer warranties. Hence, it is unnecessary for us to reach the question whether the manufacturer's 1983 written response to the applicant's inquiry is an express warranty.

[167] Moore Tr. D-524-26.

conforms to the actual design drawings and analyses.[168] In particular, they assert that a number of past deficiencies in the PG&E program, combined with more recent purported lapses in configuration control uncovered by the IDVP, establish the applicant's continuing failure to reconcile design documents with the plant as built. Although the evidence indicates a past weakness in this area, the applicant's revised configuration control procedures under which all modifications have been done, coupled with the extent of the verification efforts of the ITP and the IDVP, demonstrate that the applicant's reconciliation of design documents is in conformity with Appendix B.[169]

As part of its seismic design verification program, the ITP performed field walkdowns of the Diablo Canyon structures, equipment, piping, and electrical, pipe and HVAC supports to ensure that the design documents of record reflected the actual physical conditions of the plant. Any deviations identified by the ITP were incorporated into the design documents of record.[170] Similarly, as part of its

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[168] Gov. PF at 56-59; JI PF at 28-33.

[169] 10 CFR Part 50, Appendix B, III and VI.

[170] Anderson _et_ _al_. Tr. fol. D-224 at 31; Moore Tr. D-363-64.

nonseismic review of design pressures and temperatures, and redundant electrical circuits, the ITP conducted field verifications of the design documentation of PG&E designed safety-related systems.[171]

Further, the applicant modified its configuration control procedures in 1981 and again in 1983 to improve past weaknesses in reconciling design documents with the plant as built.[172] The present procedure (Engineering Department Procedure 3.6 ON) deals with the initiation, processing, approval and documentation of design changes during the operating life of the plant. Specifically, it requires Priority I drawings of design changes (i.e., documents pertaining to safety-related structures, systems and components that are required for the safe operation and maintenance of the plant) be revised to reflect as-built conditions within thirty days of the engineering department's acceptance of the design change completion package. All other drawings must be revised within ninety days.[173] The modification work performed at the site has

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[171] Anderson et al. Tr. fol. D-224 at 31; Moore Tr. D-363-64. See App. Exh. 90, IDVP Final Report, Vol. II, at 4.8.3-1, 4.8.6-1.

[172] Moore Tr. D-362; App. Exh. 161, Engineering Department 3.6 ON and 3.7.

[173] App. Exh. 161, Procedure 3.6 ON (May 14, 1983) at 1, 10 and Procedure 3.7 at Attachment A; Moore Tr. D-348.

conformed to this new procedure.[174] Experience under the procedure has shown that the construction department generally provides to the engineering department the as-built information within two weeks of the completion of the modification. The engineering department's acceptance requires one to three additional weeks depending on the nature of the modification.[175]

The IDVP also performed extensive field inspections to verify that the plant, as analyzed, is in conformity with the plant, as built, respecting both its seismic and nonseismic design. In its initial reviews of the seismic design of the facility, the IDVP conducted field verifications to validate the physical configurations. This verification was repeated on a sampling basis after the ITP's seismic reanalysis and the completion of necessary modifications.[176] The IDVP authenticated the as-built

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[174] Anderson et al. Tr. fol. D-224 at 32.

[175] Moore Tr. D-354-56, D-360-61.

[176] Cooper et al. Tr. fol. D-1459 at 5-2 to -4; Anderson et al. Tr. fol. D-224 at 31-32; App. Exh. 142, ITR 50, at 17-19, 24; App. Exh. 143, ITR 51, at 7, 19; App. Exh. 144, ITR 54, at 5; App. Exh. 145, ITR 55, at 46; App. Exh. 146, ITR 56, at 33; App. Exh. 147, ITR 57, at 22; App. Exh. 148, ITR 58, at 18; App. Exh. 149, ITR 59, at 3-3; App. Exh. 150, ITR 60, at 8; App. Exh. 151, ITR 61, at 8-9, 13-15; App. Exh. 152, ITR 63, at 14-19; App. Exh. 153, ITR 65, at 8. See also App. Exh. 128, ITR 36, at 4-9; App. Exh. 130, ITR 38, at 2-1, 3-2 to -6, 4-3, 6-1.

condition of the auxiliary feedwater system, control room ventilation and pressurization system, and the 4160 V electrical distribution system (i.e., the nonseismic systems it reviewed). It verified samples of the nonseismic review work performed by the ITP at the IDVP's direction and it substantiated the as-built condition of all modifications resulting from the IDVP's nonseismic verification program.[177] On the bases of its verification efforts, the IDVP concluded that, with respect to the portions of the facility it field verified, the as-built plant properly implemented the essential design elements.[178] Finally, the IDVP audited the applicant's process for controlling the update of engineering documents which included both the method for controlling design changes and the update of

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[177] Cooper et al. Tr. fol. D-1459 at 5-1 to -2; Anderson et al. Tr. fol. D-224 at 32; App. Exh. 106, ITR 14, at 3-9 to -11; App. Exh. 110, ITR 18, at 3-1 to -2; App. Exh. 111, ITR 19, at 7; App. Exh. 112, ITR 20, at 2-3, 6-3; App. Exh. 113, ITR 21, at 2-2, 3-1; App. Exh. 114, ITR 22, at 2-2, 6-2; App. Exh. 115, ITR 23, at 3-1 to -8; App. Exh. 116, ITR 24, at 1-2, 3-4 to -5; App. Exh. 117, ITR 25, at 3-1 to -4; App. Exh. 118, ITR 26, at 3-1 to -4; App. Exh. 119, ITR 27, at 3-1 to -3; App. Exh. 120, ITR 28, at 3-1 to -4; App. Exh. 140, ITR 48, at 6-9 to -14; App. Exh. 141, ITR 49, at 4-1, 5-1.

[178] Cooper et al. Tr. fol. D-1459 at 5-4.

documents to reflect the as-built condition. It concluded that the program was being effectively implemented.[179]

The Governor and the joint intervenors, however, point to a number of purported as-built discrepancies reported by the IDVP in various ITRs and assert that, as in the case of applicant's past weaknesses in this area,[180] these errors

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[179]Cooper et al. Tr. fol. D-1459 at 5-3 to -4; Reedy Tr. D-1640.

The Governor questions the IDVP's conclusion that the applicant's procedure is being effectively implemented. He asserts that the initial audit was unable to substantiate the implementation of the applicant's design control process because of a lack of information and that the follow-up audit did not attempt to verify the procedure because it was limited solely to verifying the correction of a number of other specific deficiencies found in the initial audit. Gov. PF at 58-59. The initial audit was unable to verify the process. Reedy Tr. D-1636; Gov. Exh. 48 at 33; Gov. Exh. 49 at 33. The follow-up audit was not limited to the matters claimed by the Governor. The applicant's quality assurance expert was emphatic that the follow-up audit specifically verified the efficacy of the applicant's process. Reedy Tr. D-1636-37. We credit that testimony. Further, the applicant's primary difficulty in the area of configuration control was its inability to revise affected documentation in a timely manner. Gov. Exh. 11 at 10. Thus, we do not view the applicant's May 1983 amendment of engineering procedure 3.6 ON to prescribe thirty and ninety-day limits on conforming documents (see pp. 72-73, supra) as inconsistent with the IDVP's March 1983 audit conclusion that the configuration control process was being effectively implemented.

[180]As we previously stated, the evidence indicates the applicant had difficulties with configuration control. Gov. Exh. 11, Institute of Nuclear Power Operations Startup Assistance Visit to Diablo Canyon Nuclear Power Plant (Feb. 12, 1982), at 10; Moore Tr. D-361-62; Morrill Tr. D-2948-49. We note, however, that a number of the documents relied on

(Footnote Continued)

demonstrate that the applicant's configuration control process is still inadequate.[181] None of these asserted deficiencies had any quality assurance implications or demonstrated a pattern of inadequate configuration control procedures.[182] Indeed, the Governor's expert upon whose claims the Governor's assertions are based, conceded that, in general, the applicant's as-built drawings reflect the actual physical configuration of the plant.[183]

The Governor and the joint intervenors also cite these as-built discrepancies as evidence that certain analyses did not conform to the as-built configuration of the plant.[184]

(Footnote Continued)
by the Governor and joint intervenors to establish these past deficiencies fail in that regard. Their reliance on a Brookhaven National Laboratory analysis of vertical response of the containment annulus structure (JI Exh. 130) is misplaced. Rather, that report uncovered a modeling discrepancy, not an as-built discrepancy. JI Exh. 130 at 11; Knight Tr. D-2948. Similarly, Gov. Exh. 36 (EDS Nuclear, Inc. Project Summary Report (June 7, 1982)) does little to enhance their position. That report describes an EDS review of the quality control manuals of each of the applicant's departments to determine the self-sufficiency of each manual. See pp. 94-97, <u>infra</u>. The EDS review was not an audit of the applicant's configuration control process. de Uriarte Tr. D-3148-49; Stokes Tr. D-3189.

[181] Gov. PF at 57; JI PF at 29-31. See Hubbard Tr. fol. D-2084 at Tables 5-1 and 8-1.

[182] Reedy Tr. D-1640-41; Morrill Tr. D-2948-49.

[183] Hubbard Tr. D-2157.

[184] Gov. PF at 57; JI PF at 29-31; See Hubbard Tr. D-2156.

In some instances, however, the discrepancy was the result of a worst-case assumption being used in the analysis which would not necessarily reflect as-built conditions.[185] In a large majority of the cases cited as examples of configuration control discrepancies, the IDVP determined that the plant's licensing criteria were met when the as-built condition was analyzed. In a few instances, modifications were required. Our review of these discrepancies reveals that many of them were attributable to modeling differences. Further, our review leads us to conclude, as did the IDVP, that this type and number of discrepancies are not unusual for the scope of activities undertaken.[186] Nor do we believe these discrepancies represent a pattern of inadequate configuration control. Accordingly, we are satisfied that applicant's reconciliation of design documents with the facility and with the design analyses is in compliance with the Commission's regulations.

K. In issue 6, the joint intervenors and the Governor charge that the applicant failed to verify that the design of Westinghouse-supplied, safety-related equipment met licensing criteria. Westinghouse was the vendor of the

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[185] Hubbard Tr. D-2157.

[186] App. Exh. 90, IDVP Final Report, Vol. III, at 5.6-4.

nuclear steam supply system (NSSS) at Diablo Canyon. As part of the verification program, the IDVP reviewed the Westinghouse-PG&E interface for the use of Hosgri spectra, but the applicant's verification program did not specifically validate the qualification of Westinghouse-supplied equipment. Accordingly, the joint intervenors and the Governor claim there is no meaningful assurance that the Westinghouse design of safety-related equipment at Diablo Canyon meets applicable licensing criteria.[187]

Contrary to this claim, however, the assurance that the Westinghouse-supplied equipment meets licensing criteria is provided by the Westinghouse quality assurance program. That program was sufficient during the time the NSSS was designed and at subsequent times when the Hosgri spectra reevaluations were performed. Consequently, the applicant's verification efforts were not deficient even though the scope of its program did not include review of Westinghouse-supplied equipment.

Inasmuch as the construction permit for the first Diablo Canyon unit was issued in 1968, much of the Westinghouse design work on the NSSS took place prior to the promulgation of the agency's quality assurance regulations,

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[187] JI PF at 33-36; Gov. PF at 59-62.

10 CFR Part 50, Appendix B.[188] During that period Westinghouse nuclear design work was carried out in compliance with the requirements of MIL Q 9858, which was the quality assurance specification used by the navy nuclear program and includes most of the requirements later incorporated into Appendix B.[189] Subsequent to the issuance of Appendix B, the Westinghouse program was conformed to the regulation but this modification did not change the basic characteristics of the Westinghouse program.[190]

The Westinghouse quality assurance program has been audited many times by utilities, architect-engineers and professional organizations, as well as the NRC.[191] Indeed, a number of NRC audits of the Westinghouse program occurred while the vendor was performing the reanalysis of the Diablo Canyon NSSS for the Hosgri spectra in the late 1970's and then again in the early 1980's. There is no record of unsatisfactory performance.[192] In addition, Westinghouse has designed the NSSS for some fifteen, four loop nuclear

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[188] Haass Tr. fol. D-2906 at 2.

[189] Kreh Tr. D-1151.

[190] Id. at D-1131.

[191] Id. at D-1129-31.

[192] Id. at D-1089, D-1114, D-1116, D-1129; Wiesemann Tr. D-1115.

power plants similar to Diablo Canyon which have been licensed by the NRC.[193]

The Governor and the joint intervenors point, however, to several asserted design errors at Diablo Canyon which they claim proves the inadequacy of Westinghouse's quality assurance program.[194] First, they point out that Westinghouse inappropriately used tau-filtered spectra[195] instead of unfiltered spectra in its design reanalysis for the Hosgri spectra. As a result of the IDVP interface review, two areas where Westinghouse inappropriately applied

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[193]Hoch et al. [This panel consisted of J. Hoch, R. Wiesemann and E. Kreh] Tr. fol. D-1088 at 3-4.

Additional confirmation of the adequacy of the Westinghouse quality assurance program is provided by the IDVP's recent verification of the PG&E-Westinghouse interface for the transmittal and use of the Hosgri spectra. The IDVP found that the appropriate information had been transmitted to Westinghouse and that, with one minor exception, the vendor correctly implemented the Hosgri data in their qualification and evaluation process. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.3-1; App. Exh. 103, ITR 11, at 18. Similarly, as part of its nonseismic review of the AFW system, the IDVP again examined this interface and concluded that Westinghouse correctly used the applicant-calculated design parameters provided to the vendor for accident analyses. App. Exh. 90, IDVP Final Report, Vol. I, at 4.1.3-1 to -2; App. Exh. 114, ITR 22, at 3-1, 3-4.

[194]Gov. PF at 60; JI PF at 33-35.

[195]For an explanation of the tau-effect, see Pacific Gas and Electric Co. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-644, 13 NRC 903, 962-65 (1981), petitions for review denied, CLI-82-12A, 16 NRC 7 (1982).

tau-filtered spectra in the Hosgri reanalysis were discovered. But this was a communication (i.e., interface) problem between the applicant and Westinghouse, not a problem with the vendor's quality assurance program. Once the information was interpreted by Westinghouse, it was applied in analyses in these two areas consistent with the vendor's properly functioning quality assurance program.[196]

Second, the Governor and the joint intervenors assert that a review by Brookhaven National Laboratory[197] of the IDVP interface verification found that thirty percent of the samples reviewed by the IDVP contained errors. This claim misconstrues the Brookhaven report. That report simply states that in thirty percent of the samples, the Westinghouse spectra did not completely envelop the Hosgri spectra.[198] Because Westinghouse has a generic seismic

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[196] App. Exh. 103, ITR 11, at 18-19; Cooper et al. Tr. fol. D-1459 at 6-1 to -2; Wiesemann and Kreh Tr. D-1136-41.

The Governor also charges that the Westinghouse quality assurance program was deficient because there was inadequate identification of the specific Diablo Canyon unit for design information transmitted between the applicant and Westinghouse. The NSSS vendor, however, had its own number and letter designation system for documents that distinguished between the two units. Reedy Tr. D-1650.

[197] Gov. Exh. 12, Summary and Evaluation Report, PG&E-Westinghouse Seismic Interface Review.

[198] Id. at 4-5.

design, site specific spectra may exceed the generic ones -- the situation noted in the Brookhaven review. Where that occurs, the affected equipment is specifically evaluated by Westinghouse to ensure conformance with the site specific spectra.[199]

Finally, the Governor and the joint intervenors charge that recent modifications to the main control board were necessary because of errors by Westinghouse in its original seismic qualification analyses. The modifications, however, were solely the result of changes in the seismic floor response spectra for the auxiliary building.[200]

Thus, none of the matters asserted by the Governor and the joint intervenors demonstrates that the Westinghouse quality assurance program was not adequate at the time of the original NSSS design or the subsequent reanalyses for the Hosgri spectra. The applicant's verification effort was not flawed by its exclusion of Westinghouse supplied equipment and the verification program could justifiably rely upon the existence of the Westinghouse quality assurance program to ensure the adequacy of the nuclear steam supply system.

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[199] Wiesemann Tr. D-1121, D-1135-36.

[200] Hoch Tr. D-1122-23.

L. In issue 7, the Governor and joint intervenors claim that the verification program did not identify the root causes of the failures of the applicant's design quality assurance program and did not determine if such failures raise generic concerns. They correctly assert that without the identification of the causes of the various design errors, and a determination whether the errors have generic implications, there can be no confidence that further design errors do not exist.[201]

The root causes of the failures in the Diablo Canyon's design quality assurance program have been, however, adequately identified and analyzed as part of the applicant's verification efforts. In November 1981, the applicant began "lookback" reviews of its own quality assurance program and that of its service-related contractors.[202] From these reviews, the applicant determined that the basic causes for its own quality assurance failures were its inadequate control of changes in FSAR descriptions, inadequate control of documents, and inadequate documentation of design inputs.[203] For service-related contractors, the applicant found the basic

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[201] Gov. PF at 62-68; JI PF at 36-37.

[202] Dick *et al*. Tr. fol. D-847 at 1-2.

[203] *Id*. at 3.

causes were PG&E's failure to require quality assurance controls prior to mid-1978, its failure to control transmitted information, its inadequate record disposition and its inadequate interface control.[204] In response to the basic causes identified by its review, the applicant then took appropriate corrective action.[205]

The IDVP and the ITP also evaluated the causes of the errors and deficiencies that were discovered in the design of Diablo Canyon. In addition to a group of isolated random causes, the IDVP identified two basic reasons for design errors: failure to control design interfaces and inadequate documentation of the original and revised design.[206] Further, the IDVP concluded that several factors related to the fact that the plant was designed over a fifteen year period during which requirements, criteria and engineering techniques were changing also contributed to design problems.[207] The IDVP then evaluated the cause of each EOI

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[204] Id.

[205] Id. at 3-5.

[206] Cooper et al. Tr. fol. D-1459 at 7-1 to -2; App. Exh. 90, IDVP Final Report, Vol. III, at 6.3-1, 6.3.2-1, 6.3.3-1, and 6.3.4-'.

[207] Specifically, the IDVP found that: (1) Safety-related systems for DCNPP-1 were seismically designed twice to meet two sets of design criteria, and with a substantial interval of time between the two designs. (2) In addition
(Footnote Continued)

and generally documented this evaluation in the EOI files.[208] Each EOI was also reviewed for quality assurance implications.[209] Because the IDVP assumed the applicant's quality assurance program had been deficient, whenever they opened an EOI file the IDVP looked for generic (plant-wide) implications, which necessarily included consideration of the cause of an error.[210] Similarly, the ITP identified, as causes for errors in its seismic design, the evolution of technology, criteria and requirements, difficulties with interfaces and communications, and several random

(Footnote Continued)
to two complete designs, the plant had substantial additional design work performed as a result of recent NRC IE [Inspection and Enforcement] bulletins and TMI [Three Mile Island] requirements. (3) This multiple design work has occupied 15 years of calendar time. (4) Seismic design technology had advanced from a rudimentary effort in 1967 to a reasonably mature, systematic and sophisticated process today. In the natural course of this evolution, methodology and criteria have changed significantly. (5) Nuclear plant design naturally requires the transfer of large amounts of design information from one design group to another. In the case of DCNPP-1, these design interfaces existed in especially large numbers both within PGandE and between PGandE and independent firms. (6) Design document control practices in use at the time of the original design were not consistent with the eventual duration and complexity of the design process.

App. Exh. 90, IDVP Final Report, Vol. III, at 6.3-2.

[208] Cooper Tr. D-1722.

[209] Reedy Tr. D-1642-43.

[210] Cooper et al. Tr. fol. D-1459 at 7-3 to -5; Hubbard Tr. D-2160; Jacobson Tr. D-987.

factors.[211] For seismic design errors, the ITP then correlated each Class A and B error identified by the IDVP and each Open Item identified by the ITP with its cause.[212] In the nonseismic design area, the ITP found only insignificant errors with random causes.[213]

The Governor also claims that, in addition to identifying the causes of the various design errors and their generic implications, the applicant's verification programs should have isolated the quality assurance failures that permitted each of the original design errors to occur. Additionally, he asserts that purported errors made by the ITP during its reviews that subsequently were found by the IDVP also were caused by quality assurance lapses that should have been specifically identified.[214] But the root

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[211] App. Exh. 91, ITP Phase I Final Report, at 1.8.2-2 to -3, 1.8.3-1 to 1.8.5-1; Dick _et al_. Tr. fol. D-847 at 5.

[212] App. Exh. 91, ITP Phase I Final Report, at Appendices 1C and 1D.

Based on the testimony of his expert witness, the Governor generally asserts that the verification program did not correlate basic causes to specific identified errors. Gov. PF at 63. Interestingly, the Governor's witness did not review the applicant's nonconformance reports (NCRs) even though each NCR contained a statement of the cause of each reported error. Hubbard Tr. D-2164; see, _e_._g_., Gov. Exhs. 43 and 44.

[213] App. Exh. 92, ITP Phase II Final Report, at 3-2 to -3.

[214] Gov. PF at 64-65.

causes of the original quality assurance program failures were identified in the "lookback" reviews and the applicant's quality assurance program was corrected to address these problems. Given this, and the search for generic causes carried out by the IDVP for each identified design error, no further specification of discrete quality assurance failures was necessary. Further, the asserted errors the Governor claims the ITP made as part of its verification effort were essentially differences between the ITP and the IDVP in modeling and they do not have quality assurance implications.[215]

We find, therefore, that the causes for the failures of the applicant's quality assurance program and the evaluation of those errors for generic concerns have been sufficiently addressed by the applicant's verification program to provide adequate confidence that no further significant design errors exist. We reach this result even though the verification efforts did not also identify as a root cause for the design deficiencies, PG&E management's lack of commitment to quality assurance as suggested by the Governor.[216] Similarly, the staff, while in general

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[215]Cooper et al. fol. Tr. D-1459 at 8.8; Reedy Tr. D-1640-41. See p. 77, supra.

[216]Gov. PF at 67-68.

agreement with the findings of the IDVP and ITP on the causes underlying the design errors, concluded that these causes should be more fundamentally attributed to the failure of PG&E management to recognize, at the time of the Hosgri reevaluation, the significance of the revised seismic design requirements and the attendant need to implement a well-controlled design effort.[217] Whether it was lack of commitment or lack of awareness, PG&E's management cannot escape responsibility for a quality assurance program that initially allowed for design errors of the type and number identified at Diablo Canyon by the verification program. The evidence indicates, however, that by the late 1970's significant improvements were being made in the applicant's quality assurance program.[218] Since that time, the applicant has instigated many more changes in its quality assurance program and carried out an extensive and unparalleled design verification program.[219] The painful lessons PG&E's management has learned from the huge

[217] Knight and Schierling Tr. fol. D-2906 at 4. See also Staff Exh. 54, Diablo Canyon QA Case Study (final) (Sept. 19, 1983) and JI Exh. 128, Diablo Canyon QA Case Study (draft)(July 1983).

[218] Dick *et al*. Tr. fol. D-847 at 3-5; de Uriarte Tr. 887-88; Staff Exh. 38, SECY 82-89, Encl. 1 at 1, n.1.

[219] Anderson *et al*. Tr. fol. D-224 at 16-21; Dick *et al*. Tr. fol. D-847 at 9-11.

expediture of resources required to verify the adequacy of the Diablo Canyon design have produced a present approach to quality assurance that is much improved and currently satisfactory.[220] As it must accept responsibility for past failings, PG&E management must also be credited for the significant improvements in its quality assurance program. For this reason, the failure of the applicant's verification program to include in its list of causative factors the past failings of PG&E management toward quality assurance is not fatal and does not alter our conclusion that the root causes have been sufficiently identified.

M. In Issue 8, the Governor and the joint intervenors maintain that the ITP did not timely develop and implement an adequate quality assurance program for performing its verification functions and the necessary physical modifications to the Diablo Canyon facility, and that the IDVP failed to oversee the ITP's program.[221]

From the time of its initiation in November 1981 until August 1982, the ITP carried out its design verification work under the PG&E quality assurance program. An NRC inspection in November 1981 found the structure of this program acceptable although some implementation deficiencies

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[220]See pp. 89-98, *infra*.

[221]Gov. PF at 68-73; JI PF at 37-45.

existed in the program.[222] When the Bechtel-PG&E team was formed in 1982, the managements of the two companies decided to use the Bechtel quality assurance program for the project. The applicant's program was not chosen because of the controversy surrounding PG&E quality assurance triggered by the suspension of its low power license. Rather, because Bechtel was the manager of the completion project and its quality assurance program had been accepted at other facilities by the NRC, the Bechtel program (i.e., the Bechtel Topical Report) was adopted and appropriately modified to reflect, inter alia, the organizational structure of the ITP.[223] Under the amended Bechtel program, however, the applicant's engineering procedures were used as implementing, or second tier, procedures.[224] The modified Bechtel quality assurance program was conditionally approved by the NRC on August 2, 1982, and placed in effect on August 20, 1982. Final NRC acceptance was granted on September 22, 1982. All design modifications performed by the ITP after

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[222] Dick et al. Tr. fol. D-847 at 9; de Uriarte Tr. D-895; Skidmore Tr. D-3170.

[223] Dick et al. Tr. fol. D-847 at 10, 15-16; Dick Tr. D-1016-18.

[224] Dick et al. Tr. fol. D-847 at 13-14; de Uriarte Tr. D-1015.

the August 20 date were done under the modified Bechtel program.[225]

After the ITP became a joint Bechtel-PG&E project, the relationship between the ITP and the applicant was essentially that of an architect-engineer and applicant. Before the Bechtel-based quality assurance program was put into effect, the applicant reviewed it for compliance with Appendix B and the applicant's licensing commitments. Once the program was adopted, the applicant's quality assurance department performed continuous audits of the ITP's activities.[226] The applicant also audited the IDVP contractors to ensure that each one had implemented an adequate quality assurance program to control the verification activities. Similarly, the applicant audited the IDVP-ITP interface to verify that it was adequately controlled.[227]

In order to ensure that the modified Bechtel quality assurance program was properly implemented, the ITP initially trained and indoctrinated all personnel performing quality-related activities in the requirements of the program and, while the verification was ongoing, performed

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[225] Dick _et_ _al_. Tr. fol. D-847 at 10-11.

[226] _Id_.; Skidmore Tr. D-851-52.

[227] Dick _et_ _al_. Tr. fol. D-847 at 11-12.

further training to remedy any program weaknesses identified by audits and other oversight activities. Throughout the program, numerous additional audits of the ITP verification program were performed by the applicant's quality assurance department, the ITP's own quality assurance personnel, Bechtel's San Francisco Power Division Quality Assurance, and the IDVP.[228] And, as a result of the various audit findings, the ITP took appropriate remedial and corrective actions.[229] Similarly, the staff reviewed the ITP's quality assurance program through a series of inspections while the verification activities were in progress.[230] The results of the audits and inspections demonstrate that the ITP's quality assurance program was effectively implemented.[231] No serious deficiencies were identified by the audits and the staff issued no notices of violation with respect to the ITP quality assurance program. In total, the various audits revealed less than 100 findings or conditions needing

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[228] Id. at 17-19; Cooper et al. Tr. fol. D-1459 at 8-1 to -5.

[229] Dick et al. Tr. fol. D-847 at 18.

[230] Morrill Tr. fol. D-2906 at 4-5.

[231] Id. at 5-6; Dick et al. Tr. fol. D-847 at 20; Cooper et al. Tr. fol. D-1459 at 8-3 to -5.

correction for a project involving 1200 technical people performing design work over an eighteen-month period.[232]

The Governor and the joint intervenors charge, however, that the design verification work performed under the PG&E quality assurance program (i.e., the program in effect from November 1981 until August 1982) is suspect because the PG&E program was inadequate. Additionally, they assert that deficiencies identified by audits of the ITP quality assurance program (i.e., the modified Bechtel program in effect after August 1982) show that that program also was insufficient.[233] A preponderance of the evidence does not support either position of the Governor and the joint intervenors.

The Governor and the joint intervenors rely upon two reports by PG&E consultants, Project Assistance Corporation (PAC)[234] and EDS Nuclear, Incorporated (EDS),[235] to support their claim that the applicant's quality assurance program in effect until August 1982 was inadequate. Both reports are generally critical of the relationship and coordination between the basic corporate quality assurance manual and the

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[232]Dick et al. Tr. fol. D-847 at 20-21.

[233]Gov. PF at 68-73; JI PF at 37-45.

[234]Gov. Exh. 35.

[235]Gov. Exh. 36.

various subordinate departmental manuals and other quality assurance documents. The two reports do not represent, however, the results of audits or evaluations of the pre-August 1982 PG&E quality assurance program.[236] Rather, each of the reports deals with a very limited review of the applicant's corporate quality assurance manual or the individual department quality control manuals.

The Commission's regulations do not require that all pertinent quality assurance or quality control documents be consolidated and integrated into a single manual or set of manuals. Under the applicant's quality assurance program none of the quality assurance and quality control manuals is self-sufficient (i.e., each must be read in conjunction with other documents). Because the PG&E quality assurance program is comprised of many documents and a large number of procedures, the applicant retained PAC to review the company quality assurance manual and outline the scope of the work necessary to make the manual self-sufficient.[237] PAC examined the corporate quality assurance manual, which consists of 40 procedures out of some 2400 procedures that

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[236] Stokes Tr. D-3147; Gouveia Tr. D-3149; de Uriarte Tr. D-3148-49, D-3173-74.

[237] de Uriarte Tr. D-3148-49.

make up the entire program,[238] and made a number of findings critical of the applicant's organization.[239] But the items identified by PAC as missing from the corporate manual can be found in other documents in the program.[240] Indeed, the PAC report indicated that within the company complete, yet separate, quality assurance programs were being implemented by various organizations.[241] Similarly, EDS was retained by the applicant to review the individual department quality control manuals to determine the work necessary to make each manual self-sufficient and properly coordinated with the other manuals.[242] Once again, EDS was critical of the applicant's organization,[243] but the applicant's review of the EDS findings identified no violations of 10 CFR Part 50, Appendix B.[244] Accordingly, these two limited reviews do not establish, as the Governor and the joint intervenors would have it, that the applicant's quality assurance program in effect from the beginning of the verification

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[238]Gouveia Tr. D-3149-51.

[239]Gov. Exh. 35 at 4-6.

[240]Gouveia Tr. D-3151-52; de Uriarte Tr. D-3152.

[241]Gov. Exh. 35 at 5.

[242]de Uriarte Tr. D-3149; Stokes Tr. D-3154.

[243]Gov. Exh. 36, Attachment at 1-2.

[244]de Uriarte Tr. D-3156.

program until August 1982 was inadequate. Hence, the verification design work performed under the PG&E program is not inferentially suspect.[245] Moreover, 95 to 100 percent of the design work that resulted in modifications to the Diablo Canyon facility was performed after August 1982.[246]

Nor do the conditions identified by the various audits of the ITP quality assurance program (i.e., the modified Bechtel program in effect after August 1982) demonstrate that the program was inadequate as charged by the Governor

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[245] The Governor and the joint intervenors assert, based on the testimony of a staff witness, that the PG&E quality assurance program in effect from November 1981 until August 1982 was also deficiently implemented. Gov. PF at 69; JI PF at 37. Although there were deficiencies in the implementation of the PG&E program, staff witness Morrill pointed out that as a result of staff inspections conducted prior to the verification program the deficiencies were known by the applicant and that corrective actions were taken and largely completed by mid-1982. Because of this, the staff witness did not consider the program implementation from November 1981 to August 1982 inadequate; rather he found it deficient only in certain particulars. Morrill Tr. D-3025-26.

[246] Moore Tr. D-3157-60.

The ITP's decisions to redo certain designs without reliance on any previous review work, were made over a period spanning the date when the modified Bechtel quality assurance program was adopted. The individual decisions were made at the following times: fuel handling building - May 1982; auxiliary building - June 1982; intake structure - June 1982; piping - July 1982; raceways and heating, ventilating and air conditioning (HVAC) supports - July 1982; containment annulus - January 1983. Additionally, the decisions on all Phase II reviews (including HVAC technical reviews, electrical reviews and mechanical reviews) were made in August 1982. Id.

and joint intervenors. Among others, the Governor and joint intervenors point to the twenty-four conditions identified by the IDVP in its initial audit of the ITP quality assurance program as establishing the inadequacy of the program. On the basis of that audit, and the subsequent follow-up audit of the previously identified conditions, the IDVP concluded that the ITP quality assurance program was being effectively implemented and none of the identified conditions would have an impact on the control of design for the ITP work.[247] Our review of the conditions noted by the IDVP, as well as the other audit findings relied upon by the Governor and the joint intervenors, convinces us that none of the conditions, singularly or in combination, shows that the ITP quality assurance program was inadequate. Typical of these conditions was ITP management's lack of action toward nine engineers who missed three scheduled training sessions. This condition was corrected after the initial audit and the IDVP's follow-up audit verified that the condition had been remedied. This minor deficiency and other similar ones simply do not demonstrate the program was unacceptable. Considering the extent of the ITP verification activities, such discrepancies are to be

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[247] Cooper _et al._ Tr. fol. D-1459 at 8-1 to -8; App. Exh. 90, IDVP Final Report, Vol. III, at 5.6.4; App. Exh. 133, ITR 41, at 1-2, 11.

expected and the very purpose of the auditing process is to ensure that they are caught and corrected. Thus, contrary to the charges of the Governor and the joint intervenors, the ITP quality assurance program, under which the vast majority of the design verification program was performed, was adequate.

N. In issue 9, the joint intervenors[248] maintain that the applicant has failed to provide assurance of component cooling water system (CCWS) heat removal capacity and that a technical specification limiting plant operation does not provide a level of safety equivalent to compliance with GDC 44.[249]

During the course of a review of the applicant's analysis of the CCWS, the NRC staff discovered that the most limiting single failure from the standpoint of CCWS performance, concurrent with the worst design basis accident heat load, had not been considered by the applicant as required by GDC-44.[250] Rather, the assumptions incorporated in the applicant's original analysis (including the use of a single heat exchanger) led the applicant to conclude that

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[248] JI PF at 45-46.

[249] 10 CFR 50, Appendix A, GDC-44.

[250] Wermiel Tr. fol. D-2864 at 1-2 (Contention 9); Staff Exh. 55, SSER 16, at 9-5 to -7.

adequate cooling for the CCWS would be available as long as the water temperature of the ocean, the ultimate heat sink for the Diablo Canyon reactors, did not go above 70°F. With the more stringent conditions assumed by the staff, however, the maximum temperature of the ocean under which the CCWS could meet the limiting conditions would be 64°F.[251]

To overcome this problem, the applicant proposed a technical specification requiring monitoring of the ocean water temperature. When the temperature approaches the maximum allowable limit of 64°F, the normally isolated second CCWS heat exchanger will be put on line to provide the additional heat removal capability needed to maintain an acceptable CCWS temperature in the event of the design basis loss of coolant accident.[252] In the event that the second heat exchanger, a passive unit with low failure probability, is unavailable or fails, the technical specification requires that the plant be shut down.[253]

We find the applicant's proposed technical specification is sufficient to meet the requirements of

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[251] Wermiel Tr. fol. D-2864 at 1-4 (Contention 9).

[252] Connell *et al*. Tr. fol. D-487 at 37.

[253] Connell Tr. D-546, D-551.

GDC-44.[254] Because the applicant's CCWS technical specification was presented to us as "proposed," the Director of Nuclear Reactor Regulation must ensure that the essential operating conditions set forth in the applicant's proposal are incorporated into the plant technical specifications before permitting operation.[255]

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[254]Connell *et al*. Tr. fol. D-487 at 37; Wermiel Tr. fol. D-2864 at 3-4 (Contention 9).

[255]The joint intervenors assert that the applicant's proposed technical specification is insufficient. They argue that if the recently experienced, above-normal ocean temperatures continue for long periods then, under the technical specification, the plant will have to shut down more frequently than originally contemplated. The joint intervenors then claim that each such unnecessary shutdown unacceptably challenges plant systems, thereby eroding the original safety margins of the facility. Thus, they argue the proposed limitation does not provide a level of safety equivalent to compliance with the requirements of GDC-44. See JI PF at 45-46. The sequence of events that must occur before shutdown is necessary is an unlikely one. The ocean temperatures must reach above-normal levels and the second heat exchanger (a passive component) must be unavailable for a period of at least six hours. Connell Tr. D-551. In these circumstances, the likelihood of any significant increase in the number of plant shutdowns because of ocean temperatures is exceedingly remote and the effect on the number of thermal cycles is inconsequential. Finally, we note that the applicant's technical specification could be amended and an additional heat exchanger added to the CCWS sometime in the future if the recent transient rise in ocean temperatures should become permanent or the facility should experience unexpected and repeated failures in the existing heat exchangers. Connell Tr. D-546.

III. Conclusion

For the reasons we have discussed in Parts I and II, we find that the scope and the execution of the applicant's verification programs have been sufficient to establish that Diablo Canyon Unit 1 design adequately meets its licensing criteria. The applicant's verification efforts provide adequate confidence that the Unit 1 safety-related structures, systems and components are designed to perform satisfactorily in service and that any significant design deficiencies in that facility resulting from defects in the applicant's design quality assurance program have been remedied. Accordingly, we conclude that there is reasonable assurance that the facility can be operated without endangering the health and safety of the public. As a result, the license authorization previously granted to the Director of Nuclear Reactor Regulation in the Licensing Board's August 31, 1982 initial decision, LBP-82-70, _supra_, 16 NRC at 854, remains in effect with respect to Unit 1. Before exercising that authority, the Director must ensure that the applicant has adopted an appropriate technical specification for the component cooling water system.[256] In addition, before allowing commercial operation, the Director must ensure that the applicant has performed the appropriate

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[256] See p. 100, _supra_.

jet impingement analyses for certain lines inside the containment.[257] Until we make our findings with respect to Unit 2, the license authorization previously granted for that unit is not effective.[258]

Our findings have been made on the basis of the record evidence in the reopened operating license proceeding. We note, however, that recent events may affect our findings. On February 14, 1984, the joint intervenors filed a second motion to reopen the record[259] citing, inter alia, a number of recently discovered, purported design deficiencies that they assert undermine the validity and integrity of the applicant's verification efforts and directly bear upon the issues in the proceeding. In support of their motion, the joint intervenors proffer the affidavits of several engineers who formerly worked at the Diablo Canyon site. The applicant and the staff oppose the joint intervenors' motion and have filed numerous affidavits of asserted experts rebutting joint intervenors' claims.[260] Although we

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[257]See pp. 67-68, supra.

[258]See pp. 19-20, supra.

[259]Joint Intervenors' Motion to Augment or, in the Alternative, to Reopen the Record.

[260]See Pacific Gas and Electric Company's Answer in Opposition to Joint Intervenors' Motion to Augment or, in the Alternative, to Reopen (Mar. 5, 1984); NRC Staff's
(Footnote Continued)

have initially reviewed the motions and the responses, our assessment of the parties' filings has not been completed. In addition, Supplement 21 of the staff's Safety Evaluation Report for Diablo Canyon indicates that the staff is currently investigating a large number of recent allegations concerning the Diablo Canyon facility including several that appear to relate to the adequacy of facility design.[261] In this regard, the staff informed us by a letter dated February 7, 1984, and again in its opposition to the joint intervenors' motion, that they are currently investigating matters relating to small bore piping at the facility that directly bear upon the issues in this proceeding. Therefore, some of these matters may require that we again reopen the record in the proceeding and hear further evidence.[262] Hence, it is possible that these findings may have to be amended or withdrawn in their entirety depending upon the nature of the new evidence.

(Footnote Continued)
Answer to Joint Intervenors' Motion to Augment or, in the Alternative, to Reopen the Record (March 15, 1984).

[261]NUREG-0675, Supplement No. 21, "Safety Evaluation Report related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2" (Dec. 1983).

[262]Because the joint intervenors' appeal from the Licensing Board's initial decision, LBP-82-70, supra, 16 NRC 756, is still pending before us and, in addition, the joint intervenors' latest motion was filed while the reopened phase of the proceeding was before us, we necessarily retain jurisdiction over the proceeding.

It is so ORDERED.

FOR THE APPEAL BOARD

C. Jean Shoemaker

C. Jean Shoemaker
Secretary to the
Appeal Board

Concurring opinion of Mr. Moore:

I write separately on an additional point in order to call it to the Commission's attention. In the reopened proceeding, the joint intervenors and the Governor sought to litigate several issues involving the adequacy of the applicant's verification efforts in light of the asserted failure of the applicant's quality assurance program to comply with 10 CFR, Part 50, Appendix A, GDC 1. Specifically, the joint intervenors and the Governor claimed, based on the applicant's FSAR, that the applicant had no quality assurance program to assure the design of structures, systems and components that were "important to safety" within the meaning of Appendix A. Rather, they asserted the applicant only had a quality assurance program to assure the design of structures, systems and components that were "safety-related" within the meaning of 10 CFR Part 50, Appendix B.

At the prehearing conference, we excluded these issues from the reopened proceeding. We ruled that the history of the Diablo Canyon operating license application showed that the two terms, "important to safety" and "safety-related," had been used synonomously by the applicant and the staff, and to the extent the quality assurance criteria are currently interpreted to distinguish between the terms, such distinction would not be retroactively applied to Diablo Canyon.*

I highlight this matter because on January 18, 1984 the staff issued Board notification 84-011 regarding the meaning of the terms "safety-related" and "important to safety." That notification contains a January 5, 1984 letter from the Director, Division of Licensing, to all operating licensees and applicants. The letter states that applicants are responsible for developing and implementing quality assurance programs that meet the requirement of Appendix A, GDC 1, for plant equipment "important to safety" as well as a program for "safety-related" equipment in accordance with

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*Transcript of August 23, 2983 prehearing conference at D-67-68.

Appendix B. The letter then suggests this interpretation of the regulations is not new but one that the staff has always followed. If the Director's position on this matter is now that of the Commission (including the asserted longstanding nature of the interpretation), then it would appear that the Governor and the joint intervenors must be given an opportunity to litigate the issues regarding the applicant's compliance with Appendix A.

APPENDIX A

Issues At Hearing in Accordance With Orders of August 26 and October 7, 1983 (unpublished)

1. The scope of the IDVP review of both the seismic and non-seismic aspects of the designs of safety-related systems, structures and components (SS&C's) was too narrow in the following respects:

(a) The IDVP did not verify samples from each design activity (seismic and non-seismic).

(b) In the design activities the IDVP did review, it did not verify samples from each of the design groups in the design chain performing the design activity.

(c) The IDVP did not have statistically valid samples from which to draw conclusions.

(d) The IDVP failed to verify independently the analyses but merely checked data of inputs to models used by PG&E.

(e) The IDVP failed to verify the design of Unit 2.

2. The scope of the ITP review of both the seismic and non-seismic aspects of the designs of the safety-related systems, structures and components (SS&C's) was too narrow in the following respects:

(a) The ITP did not verify samples from each design activity (seismic and non-seismic).

(b) In the design activities the ITP did review, it did not verify samples from each of the design groups in the design chain performing the design activity.

(c) The ITP did not have statistically valid samples from which to draw conclusions.

(d) The ITP has failed systematically to verify the adequacy of the design of Unit 2.

3. In various situations listed below the ITP used improper engineering standards to determine whether design activities met license criteria. In some of these situations, the IDVP either used or approved the use of such improper standards or did not verify them at all.

(f) The ITP's modeling of the soil properties for the containment and auxiliary buildings was improper in that:

(i) in the soil structure interaction analysis of containment for the DE [Design Earthquake] and the DDE [Double Design Earthquake], use of boundary motion inputs to the model were improperly used;

(ii) the soil structure interaction analysis for containment for the DE and the DDE uses a 7 percent damping value for rock, which is unconservative, especially for the DE;

(iii) the dynamic analyses of the containment for all earthquakes omit any analysis of uplifting of the foundation mat;

(iv) the modeling of the soil springs for the auxiliary building does not specify soil properties;

(v) in the modeling of the soil springs for the auxiliary building, the motion inputs to the lower ends of the springs does not account for all soil structure interaction phenomena that could be expected.

(o) The ITP has not demonstrated, and the IDVP has not verified, that the DCP modeling of the seismic response of the fuel handling building is proper, in that the DCP has not adequately justified the use of the translational and torsional response of the auxiliary building as input to the fuel handling building nor has it demonstrated the validity of the dynamic degrees of freedom selected.

(p) The ITP has not demonstrated, and the IDVP has not verified, that the DCP seismic model of the slabs in the auxiliary building is proper, in relation to the use of vertical and rotational springs to model the columns, and the motions used as input at the ends of the springs not connected to the slabs. In addition, in the study of the diaphragms, the ITP has not adequately accounted for the inplane flexibility of these slabs, and has not adequately demonstrated that stresses are within allowable limits at all elevations.

(q) The ITP has not demonstrated and the IDVP has not verified, that the soils analysis for the buried diesel fuel oil tanks is proper in that the values of the exponent shown in figure 14 of ITR 68 have not been demonstrated to be appropriate and the variation of shear velocity with depth is not properly justified.

(r) The ITP has not demonstrated and the IDVP has not verified that the soils analysis for the auxiliary saltwater piping and circulating water intake conduits is proper in that the selection of the modulus versus strain curve utilized is not justified.

(s) The ITP has not demonstrated and the IDVP has not verified that the seismic analysis of the turbine building is proper in that bolt bearing capacities were taken from an inappropriate source.

(t) The ITP has not demonstrated and the IDVP has not verified that the seismic analysis of the turbine building is proper in that the use of four different models for the vertical analysis has not been justified.

4. The IDVP accepted deviations from the licensing criteria without providing adequate engineering justification in the following respects:

a. Contrary to the requirements of FSAR Section 17.1 regarding compliance of the as-built installation with the design documents, the IDVP review of the AFWS disclosed that the as-built installation failed to meet the design drawings in that (i) a steam trap on the turbine-driven AFW pump steam supply line is not provided and (ii) there are discrepancies in the arrangement of the long-term cooling water supply line.

b. Contrary to FSAR Section 8.3.3, the electrical design does not fully comply with the commitments regarding separation and color coding.

h. Contrary to PG&E's September 14 and December 28, 1978 licensing commitments, CRVPS equipment identified in the FSAR as necessary to maintain control room habitability during safe shutdown has not been evaluated regarding the effects of a moderate energy pipe break.

i. The fire protection for the motor driven AFW pump room is not consistent with the PG&E licensing commitment for fire zone separation as stated in its November 13, 1978 Supplemental Information for Fire Protection Review ("SIFPR") in that:

   1) there is a large grated ventilation opening in the ceiling of the room;

   2) a fire damper has gaps when it is closed.

j. The fire protection for the AFW pump room is not consistent with the PG&E licensing commitment for cable separation as stated in its SIFPR of November 13, 1978 in that:

   1) the pumps for the motor driven AFW pumps and the control circuitry for a flow control valve necessary for operation of the turbine driven AFW pump are located in a single fire zone;

   2) cables for some AFW circuits are not routed in accord with descriptions in the SIFPR and four AFW circuits PG&E committed to identify and review in the SIFPR were not included in that document.

k. Contrary to the licensing commitment set forth in its SIFPR of November 13, 1978, each of the three 4160 volt cable spreading rooms has a ventilation opening leading up to the 4160 volt switchgear rooms.

l. Contrary to FSAR Section 3.6, possible jet impingement loads have not been considered in the design and qualification of safety-related piping and equipment inside containment.

q. Contrary to PG&E's December 28, 1979 licensing commitment letter to the NRC, modifications to protect two Auxiliary Feedwater valves from the effects of moderate energy line breaks were not implemented.

r. Contrary to the licensing commitment to maintain minimum system redundancy as stated in FSAR Section 3.6A (NSC evaluation of pipe break outside containment), four components were identified for which high energy line cracks could cause temperatures in excess of the specification temperatures of the components.

s. Contrary to the licensing commitment to maintain minimum system redundancy as stated in FSAR, Section 3.6A (NSC evaluation of pipe break outside containment), a conduit was identified whose failure due to a high energy line crack could eliminate redundant Auxiliary Feedwater system flow.

t. Contrary to the FSAR Section 8.3 commitment to provide switchgear buses with adequate short circuit interrupting capability, the calculated duties for circuit breakers on 4160 V buses F, G, and H were above the nameplate ratings for those buses.

u. Contrary to single failure criteria stated in FSAR Section 3.1.1, reviews of the Auxiliary Feedwater and Control Room Ventilation and Pressurization systems identified circuit separation and single failure deficiencies. Similar deficiencies were identified in additional verification reviews, which included other safety-related systems.

5. The verification program has not verified that Diablo Canyon Units 1 and 2 "as built" conform to the design drawings and analyses.

6. The verification program failed to verify that the design of safety-related equipment supplied to PG&E by Westinghouse met licensing criteria.

7. The verification program failed to identify the root causes for the failures in the PG&E design quality assurance program and failed to determine if such failures raise generic concerns.

8. The ITP failed to develop and implement in a timely manner a design quality assurance program in accordance with 10 CFR Part 50, Appendix B to assure the quality of the recent design modifications to the Diablo Canyon facility and the IDVP failed to ensure that the corrective and preventative action programs implemented by

the ITP are sufficient to assure that the Diablo Canyon facilities will meet licensing criteria.

9. Contrary to General Design Criterion 44 (GDC-44) of Appendix A to 10 CFR Part 50, PG&E has failed to provide adequate assurance of component cooling water system (CCWS) heat removal safety function capacity in that the maximum ocean water temperative of 64°F. is not conservative because it has already been exceeded in 1983. Furthermore a technical specification limitation which permits plant operation at reduced power levels in lieu of enlarging the capacity of the CCWS does not provide an equivalent level of safety as compliance with the requirements of GDC-44 (SSER, 16 (Aug. 1983) and September 1983 ocean water temperature readings).

APPENDIX B - LIST OF WITNESSES

APPLICANT'S WITNESSES

Anderson, Richard C.

Education:	B.S. Mechanical Engineering University of California at Berkeley
Present Occupation:	An Engineering Manager for Bechtel Power Corporation now assigned as the Engineering Manager for the Diablo Canyon Project

Connell, Edward C., III

Education:	M.S. Nuclear Engineering, 1974 Purdue University
Present Occupation:	Mechanical Group Supervisor (Bechtel) Diablo Canyon Project

Cranston, Gregory V.

Education:	B.S. Nuclear Science United States Naval Academy Annapolis, Maryland
Present Occupation:	Project Engineer (Bechtel) for Unit 2 of the Diablo Canyon Project

Dick, Charles W.

Education:	M.S. Electrical Engineering, 1948 Stanford University
Present Occupation:	Project Manager (Bechtel) and member of project management team of the Diablo Canyon Project

Gouveia, Leigh A.

Education:	B.S. Mechanical Engineering, 1968

	California State Polytechnic College San Luis Obispo
Present Occupation:	Project Engineer Project Assistance Corporation

Hoch, John B.

Education:	B.S. Mechanical Engineering, 1959 University of Idaho
Present Occupation:	PG&E Project Manager of Diablo Canyon Project

Jacobson, Michael J.

Education:	B.S. Civil Engineering, 1970 Sacramento State College
Present Occupation:	Project Quality Assurance Engineer (Bechtel) for Diablo Canyon Project

Kaplan, Stanley

Education:	Ph.D Mechanical Engineering and Applied Mathematics, 1960 University of Pittsburgh
Present Occupation:	President, Kaplan & Associates, Inc. - a consulting firm specializing in risk analysis and applied decision theory

Kreh, Edward J., Jr.

Education:	B.S. Mechanical Engineering Carnegie Institute of Technology (now Carnegie Mellon University) of Pittsburgh, PA
Present Occupation:	Senior Consulting Engineer with SMC O'Connell and Associates of Pittsburgh, PA

Moore, Gary H.

Education:	M.S. Mechanical Engineering, 1969 San Jose State University
Present Occupation:	PG&E Unit 1 Project Engineer of the Diablo Canyon Project

Seed, H. Bolton

Education:	Ph.D Civil Engineering, 1948 Kings College, London University
Present Occupation:	Professor, University of California at Berkeley

Shipley, Larry E.

Education:	B.S. Mechanical Engineering United States Merchant Marine Academy Kings Point, NY
Present Occupation:	Assistant Chief Engineer (Plant Design) in Bechtel's San Francisco Power Division and Technical Consultant to Diablo Canyon Project

Skidmore, Steven M.

Education:	M.S. Nuclear Engineering, 1969 Stanford University
Present Occupation:	PG&E Manager of Quality Assurance in the Nuclear Power Generation Department

Stokes, William J.

Education:	B.S. Mechanical Engineering, 1974 Drexel University
Present Occupation:	Partner, Shalako Energy Services (formerly with EDS Nuclear)

de Uriarte, Thomas G.

Education:	B.S. Civil Engineering, 1967 University of California, Berkeley
Present Occupation:	Senior Engineer, Quality Assurance Department, Pacific Gas and Electric Company

Vahlstrom, Wallace

Education:	Electrical Engineer (degree not specified)
Present Occupation:	Senior Electrical Engineer at Pacific Gas and Electric Company

White, William H.

Education:	Ph.D Civil Engineering University of Colorado
Present Occupation:	Engineering Specialist with Bechtel's San Francisco Power Division - Seismic Analysis and Assistant Project Engineer in the Diablo Canyon Project

Wiesemann, Robert A.

Education:	B.S. Mechanical Engineering, 1949 Case Institute of Technology
Present Occupation:	Manager of Regulatory and Legislative Affairs in the Nuclear Technology Division of the Westinghouse Electric Corporation

<u>IDVP WITNESSES</u>

Biggs, John M.

Education:	M.S. Civil Engineering, 1947 Massachusetts Institute of Technology

Present Occupation: Professor Emeritus of Civil Engineering, Massachusetts Institute of Technology and Partner in the Consulting Firm of Hansen, Holley and Biggs, Inc.

Cloud, Robert L.

Education: Ph.D Mechanical Engineering, 1964
University of Pittsburgh

Present Occupation: President, Robert L. Cloud Associates, Inc.
Berkeley, CA

Cooper, William E.

Education: Ph.D Engineering Mechanics, 1951
Purdue University

Present Occupation: Senior Vice President and Technical Director of Teledyne Engineering Services until 1976 now Consulting Engineer for Teledyne

Holley, Myle J., Jr.

Education: M.S. Civil Engineering, 1947
Massachusetts Institute of Technology

Present Occupation: Professor Emeritus, Massachusetts Institute of Technology and Partner in the Consulting Firm of Hansen, Holley and Biggs, Inc.

Krechting, John E.

Education: B.S. Naval Science, 1965
United States Naval Academy

Present Occupation: Employed by Stone & Webster Engineering Corporation - assigned as Project Engineer for the IDVP

Reedy, Roger F.

Education: B.S. Civil Engineering, 1956
Illinois Institute of Technology

Present Occupation: President, R.F. Reedy, Inc., Consulting Engineers
Los Gatos, CA

Wray, Ronald

Education: M.S. Engineering Science, 1962
Rensselaer Polytechnic Institute

Present Occupation: Theoretical Stress Analyst
Teledyne Engineering Services

GOVERNOR'S WITNESSES

Apostolakis, George

Education: Ph.D Engineering Science and Applied Mathematics
California Institute of Technology

Present Occupation: Professor, Engineering and Applied Science
University of California, Los Angeles

Hubbard, Richard B.

Education: B.S. Electrical Engineering, 1960
University of Arizona

Present Occupation: Vice President - MHB Technical Associates, San Jose, CA

Roesset, Jose M.

Education: D.S. Structures and Soil Mechanics, 1964
Massachusetts Institute of Technology

Present Occupation: Professor, University of Texas
Austin, Texas

JOINT INTERVENORS' WITNESS

Samaniego, Francisco J.

Education:	Ph.D Mathematics-Statistics, 1971 University of California at Los Angeles
Present Occupation:	Professor, Division of Statistics University of California at Davis

STAFF'S WITNESSES

Altman, Willard D.

Education:	Ph.D Mathematics, 1975 University of Virginia
Present Occupation:	Section Chief, Quality Assurance Branch, Division of Quality Assurance, Safeguards and Inspection Programs, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission

Costantino, Carl J.

Education:	Ph.D Civil Engineering, 1966 Illinois Institute of Technology
Present Occupation:	Professor, Civil Engineering City College of City University of New York

Haass, Walter P.

Education:	B.S. Mechanical Engineering, 1952 Stevens Institute of Technology
Present Occupation:	Assistant to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission

Knight, James P.

Education:	B.S. Mechanical Engineering, 1957 Northeastern University
Present Occupation:	Assistant Director for Components and Structures Engineering, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

Knox, John L.

Education:	B.S. Electronic Systems Engineering, 1971 University of Maryland
Present Occupation:	Senior Reactor Systems Engineer (Electrical), Power Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

Kubicki, Dennis J.

Education:	B.S. Fire Protection and Safety Engineering, 1974 Illinois Institute of Technology
Present Occupation:	Fire Protection Engineer, Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

Kuo, Pao-Tsin

Education:	Ph.D Civil Engineering, 1974 Rice University
Present Occupation:	Section Leader, Structural and Geotechnical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

Miller, Charles A.

Education:	Ph.D Civil Engineering, 1966 Illinois Institute of Technology
Present Occupation:	Professor, Department of Civil Engineering, City College of the City University of New York

Morrill, Philip J.

Education:	B.S. Nuclear Engineering, 1966 United States Naval Academy
Present Occupation:	Reactor Inspector, Division of Resident, Reactor Projects and Engineering Programs, Region V, U.S. Nuclear Regulatory Commission

Philippacopoulos, A.J.

Education:	Ph.D Civil Engineering, 1980 Polytechnic Institute of New York
Present Occupation:	Associate Scientist, Structural Analysis Division, Department of Nuclear Energy, Brookhaven National Laboratory

Polk, Harold E.

Education:	B.S. Civil Engineering, 1958 North Carolina State College
Present Occupation:	Senior Structural Engineer, Structural and Geotechnical Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

Schierling, Hartmut E.H.

Education:	M.S. Nuclear Engineering, 1963 Catholic University of America
Present Occupation:	Senior Project Manager, Division of Licensing, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

Wang, Ping-Chun

Education:	Ph.D Civil Engineering, 1951 University of Illinois
Present Occupation:	Professor, Civil Engineering Polytechnic Institute of New York

Wermiel, Jared S.

Education:	B.S. Chemical Engineering, 1972 Drexel University
Present Occupation:	Section Leader, Auxiliary Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION V
1450 MARIA LANE, SUITE 210
WALNUT CREEK, CALIFORNIA 94596

MAY 17 1984

Docket No. 50-275
License No. DPR-76
EA 84-42

Pacific Gas and Electric Company
77 Beale Street, Room 1435
San Francisco, California 94106

Attention: G. A. Maneatis, Executive Vice President

Gentlemen:

Subject: Notice of Violation and Proposed Imposition of Civil Penalties

This refers to the special inspection conducted by Messrs. M. M. Mendonca and M. L. Padovan of this office, during the period of April 7, through April 17, 1984, of activities authorized by NRC license No. DPR-76 at the Diablo Canyon Unit 1 facility. The report of the inspection was forwarded to you on April 27, 1984.

The inspection included an examination of the facts and circumstances associated with the unusual event on April 6, 1984, involving the disabling of the Boron Injection Tank (BIT) inlet and outlet valves which blocked the flow path of the emergency core cooling system between both charging pumps and the reactor primary cooling system. You provided prompt notification of this event to the Resident Inspector on April 7, 1984, and submitted the appropriate Licensee Event Report (LER) dated May 7, 1984.

The results of this inspection were discussed by Messrs. Mendonca and Padovan with Mr. R. C. Thornberry, and other members of your staff, on April 17, 1984. In addition, the circumstances associated with the event, and the apparent violation identified during this inspection, were discussed at an enforcement conference on May 1, 1984, by Mr. J. B. Martin and other members of the NRC staff with you and other members of your staff. As indicated in these discussions, the NRC is concerned that NRC licensed personnel, and other members of your staff, failed in this instance to exhibit an acceptable degree of awarene[illegible] your facility's technical specification requirements. For a per[illegible]xcess of fifteen hours, your staff did not recognize the inoperable s[illegible] of the Emergency Core Cooling System (ECCS) subsystems flow paths as a violation of your technical specifications. During that period the Unit 1 reactor was in Mode 3, without the capability of automatic high head ECCS injection capability.

The primary underlying cause of this event and the associated violation appears to involve the manner in which procedures are written, reviewed and approved. In this case, the controlling procedure had been revised in response to a problem report at another nuclear plant where during recharge of the BIT a

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RETURN RECEIPT REQUESTED

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safety injection occurred. Reportedly, the event resulted in loss of coolant out of the open vents and created a potential for "run out" of the charging pumps. We agree that all licensees can benefit from other plant experiences, however, whenever a change is made to a procedure, adequate technical analysis of the change must be completed to assure the change does not introduce adverse consequences as occurred in this case. The individual who prepared the revised procedure apparently did not do an adequate analysis of the consequences of isolating the BIT for recharging. Subsequent supervisory and plant safety committee reviews which are designed to detect errors also failed to adequately question the potential consequences of the revision. In addition, during implementation of the procedure, the licensed operators also missed the opportunity to challenge the appropriateness of the procedure at the time the BIT was isolated, which demonstrates the continued need for emphasis on maintaining and upgrading operator proficiency. During the enforcement conference several aspects of this review and approval process and opportunity for improvement were discussed. You should address this in detail in your response.

In order to emphasize the need to assure that procedures are adequate and consistent with regulatory requirements, and that operators are fully aware and cognizant of regulatory requirements, I have been authorized, after consultation with the Director of the Office of Inspection and Enforcement, to issue the enclosed Notice in the amount of Fifty Thousand Dollars ($50,000) for the violation set forth in the Notice. The violation has been categorized as Severity Level III in accordance with the NRC enforcement policy, 10 CFR Part 2, Appendix C. The base civil penalty for a Severity Level III violation is $50,000. While we recognize that the violation was self-identified and prompt corrective actions were taken, in view of the opportunities to prevent or identify the violation, no mitigation of the base civil penalty has been made.

You are required to respond in writing to the Notice. In preparing your response, you should follow the instructions specified in the Notice. Your written reply will be the basis for determining whether additional enforcement actions are warranted.

The responses directed by this letter and accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

Sincerely,

JBMartin

John B. Martin
Regional Administrator

Enclosure: Notice of Violation and
Proposed Imposition of Civil Penalty

cc w/enclosure:
F. Mielke, PG&E
J. Schuyler, PG&E
R. Thornberry, PG&E

MAY 17 1984

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Pacific Gas and Electric Company
Diablo Canyon Nuclear Facility

Docket No. 50-275
License No. DPR-76
EA 84-42

This Notice of Violation and Proposed Imposition of Civil Penalty involves an apparent violation at the Diablo Canyon Nuclear Power Plant related to the inoperability of portions of the Emergency Core Cooling System (ECCS).

On April 6, 1984, the Boron Injection Tank (BIT) was valved out of service and electrical power was removed from the valve operators to permit recharging of the tank to increase the boron concentration. The activity was performed in accordance with appfoved procedures. The action, however, violated the facility technical specification provisions that require that the charging pumps must be capable of injecting coolant through the BIT and into the reactor coolant system upon actuation of a safety injection signal whenever the reactor is being operated in Modes 1, 2 or 3.

To emphasize the need to assure that procedures are adequate and consistent with regulatory requirements, and that operators are fully cognizant and aware of regulatory requirements, the Nuclear Regulatory Commission proposes to impose a civil penalty in the amount of Fifty Thousand Dollars ($50,000) for the identified violation.

In accordance with the NRC Enforcement Policy, 10 CFR Part 2, Appendix C as revised, 48 FR 8563 (March 8, 1984) and pursuant to Section 234 of the Atomic Energy Act of 1954, as amended ("Act"), 42 U.S.C.2282, PL-96-295 and 10 CFR 2.205, the particular violation and the associated civil penalty is set forth below:

VIOLATION ASSESSED A CIVIL PENALTY

A. Technical Specification 3.5.2 reads, in part:

"Two Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:...

a. One OPERABLE centrifugal charging pump,...

e. An OPFRABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

8405290368

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in a least HOT SHUTDOWN within the following 6 hours."

Technical Specification 3.0.3 reads in part:

"When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour action shall be initiated to place the unit in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours."

Contrary to the above requirements on April 6, 1984 at about 7:10 P.M. the inlet and outlet valves to the Boron Injection Tank (BIT) were closed and disabled by securing the electrical power to the valve operators. This action blocked and rendered inoperable the flow path between the centrifugal charging pumps and the reactor coolant system for both ECCS subsystems. The valves were returned to service at about 10:10 A.M. on April 7, 1984. The reactor was in Mode 3 at all times during this period.

This is a Severity Level III violation (Supplement I) (Civil Penalty - $50,000).

Pursuant to the provisions of 10 CFR 2.201, Pacific Gas and Electric Company is hereby required to submit to the Director, Office of Inspection and Enforcement, USNRC, Washington, D.C. 20555, and a copy to the Regional Administrator, USNRC, Region V, within 30 days of the date of this Notice, a written statement or explanation, including: (1) admission or denial of the alleged violation; (2) the reasons for the violation if admitted; (3) the corrective steps which have been taken and the results achieved; (4) the corrective steps which will be taken to avoid further violations; and (5) the date when full compliance will be achieved. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, Pacific Gas and Electric Company may pay the civil penalty in the amount of $50,000 or may protest imposition of the civil penalty in whole or in part by a written answer. Should the Pacific Gas and Electric Company fail to answer within the time specified, the Director, Office of Inspection and Enforcement, will issue an order imposing the civil penalty in the amount proposed above. Should the Pacific Gas and Electric Company elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, such answer may: (1) deny the violation listed in this Notice in whole or in part; (2) demonstrate extenuating circumstances; (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty in whole or in part, such answer may request remission or mitigation of the penalty. In requesting mitigation of the proposed penalty, the five factors contained in Section IV (B) of 10 CFR Part 2, Appendix C should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate by specific reference (e.g., giving page and paragraph numbers) to avoid repetition. The Pacific Gas and Electric Company's attention is directed to the other provisions of 10 CFR 2.201, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due, which has been subsequently determined in accordance with the applicable provision of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Act, 42 U.S.C. 2282.

FOR THE NUCLEAR REGULATORY COMMISSION

JBMartin

J. B. Martin
Regional Administrator

Dated at Walnut Creek, California
this 17 day of May 1984

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

CHAIRMAN

June 1, 1984

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Nunzio J. Palladino NJP

SUBJECT: DIABLO CANYON HEARING

Congressman Udall, on May 23, 1984 sent us a letter describing his inquiry into the Diablo Canyon matter and listing items that NRC needs to address. I would like you to ensure that knowledgeable staff, e.g., those in IE headquarters who have QA expertise, focus carefully on the issues in the last paragraph in Congressman Udall's letter, especially those pertaining to the NSC audit and related questions. If there are any questions on the questions, staff should check with Congressman Udall's staff. Answers should be provided for the Commission to review well in advance of the hearing.

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

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