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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC CO. (Diablo Canyon Nuclear Power Plant, Unit 1)	)	Docket No. 50-275 OL

DOCKET NUMBER 50-275
PROD. & UTIL. FAC.
(2.206)

PETITION PURSUANT TO
10 CFR 2.206

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DATED: May 3, 1984

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BEFORE THE NUCLEAR REGULATORY COMMISSION

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In the Matter of	)	
PACIFIC GAS AND ELECTRIC CO.	)	Docket No. 50-275 OL
(Diablo Canyon Nuclear Power Plant, Unit 1)	)	

Pursuant to 10 CFR 2.206, on behalf of the San Luis Obispo, California, Mothers for Peace ("Mothers") the Government Accountability Project ("GAP") petitions the Nuclear Regulatory Commission to defer any decisions on whether to permit the Diablo Canyon Nuclear Power Plant ("Diablo Canyon" or "DCP") to begin ascension beyond 5% power until successful completion of the following relief, to be ordered by the Commission on or before May 11, 1984 --

(1) appointment and implementation by an independent third party of corrective action required in the Commission's Order to Modify Facility Operating License No. DPR-76 (Docket No. 50-275, April 18, 1984) ("April 18 Order"), with

(2) additional modification of item 6 in the April 18 Order to require comprehensive review of all "Pipe Support Design Tolerance Clarification" ("PSDTC") program and "Diablo Problem" ("DP") system activities; and to identify any deviations from federal regulations or licensing commitments, including but not limited to design, design control and either design or construction quality assurance ("QA") requirements;

(3) a full program of public participation for the selection and oversight of independent organizations described in #1-2 above, including NRC review and approval of independent organizations from nominations submitted by either PG&E or any interested member of the public, and creation of a public oversight committee consisting of equal representation by state and local representatives and the intervenors with the authority to obtain all requested information and to conduct legislative-style public oversight hearings;

(4) publication of a report from a Construction Assessment Team ("CAT"), whose members would not include any personnel from Region V or any other personnel previously assigned to Diablo Canyon --

(a) to determine through visual, destructive and non-destructive examination the scope and nature of deficiencies in the condition of Diablo Canyon resulting from alleged violations of regulatory or program quality assurance violations; and

(b) to determine the need for a comprehensive, third party reinspection program of all safety-related construction in Units 1 and 2, with full authority by the independent organization to identify and impose corrective action on any nonconforming condition that deviates from 10 CFR 50, Appendix B, the Final Safety Analysis Report ("FSAR") or plant specifications, through implementation of corrective action;

(5) development of a full factual record on Pacific Gas and Electric's ("PG&E") character and competence to operate the Diablo Canyon nuclear power plant, including

(a) a management audit by an independent organization, and

(b) publication of a report by the NRC Office of Investigations ("OI") after completion of its investigation to determine the causes of construction and design QA violations at Diablo Canyon, including issues such as harassment and retaliation, subordination of quality assurance to cost and scheduling concerns, destruction of records and false statements, and deliberate violations of the Atomic Energy Act, and

(c) review of the record compiled in any pending administrative hearings before the Department of Labor for alleged retaliatory personnel actions in violation of 42 USC 5851 at Diablo Canyon;

(6) Board Notification of transcripts of whistleblower interviews that began on April 3, 1984, until completion of all such interviews; 1/

(7) investigation by the Office of Inspector and Auditor ("OIA") to determine --

(a) whether there have been misleading or material false statements by the NRC staff to the Commission during the March 19, 26, 27 or April 13 briefings, or in Supplemental Safety Evaluation Reports SSER-21 (December 1983) or SSER-22 (March 1984), and

(b) the causes of the QA breakdown within the NRC staff responsible for Diablo Canyon

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1/ In order to facilitate manageable review of the voluminous transcripts, counsel will work with relevant witnesses to prepare summaries of issues raised and supported during the interviews.

The basis for this petition is evidence that the effects of a comprehensive design and construction quality assurance breakdown may not be corrected before Diablo Canyon begins ascension to full-power operations. Although Mr. Yin obtained a consensus for significant corrective action for design control and QA, there is no basis for confidence in the objectivity of the program. All corrective action is controlled by the licensee Pacific Gas and Electric, which has an inherent conflict-of-interest due to financial pressures for immediate operation. PG&E has exhibited its bias and lack of necessary character and competence to run the current reform program, through prejudging the results of prospective corrective action and through recent misleading or material false statements.

With respect to construction quality assurance violations, there has been no significant corrective action. Indeed, the staff has proposed to simply turn over hundreds of documented charges to PG&E for response, without first seriously reviewing the issues itself. (See, e.g., April 13, 1984 Commission transcript, p. 48).

During the limited efforts already completed, the current staff totally lost the confidence of nearly all participating witnesses. The credibility gap is so severe that retention of the current staff team threatens to dry up the flow of information to the NRC. 2/

2/ To preclude any confusion, this assessment does not apply to the NRC's Office of Investigations. OI personnel have represented the agency in a professional, responsible manner to date.

The Mothers adopt and incorporate by reference all 519 allegations and documentation in earlier February 2, March 1, March 23 and April 12, 1984 petitions to the Commission. Those disclosure are relevant bases for this petition and have not yet been seriously reviewed, let alone resolved. This petition is further supported by NRC inspector Isa Yin's draft reports on Diablo Canyon. See Diablo Canyon Board Notification No. 84-071 (April 3, 1984). Third, this petition is supported by the transcripts of witness interviews since April 3 which the Commission has not yet released. Those transcripts include hundreds of specific, new allegations.

Finally, this petition is supported by the evidence summarized below. Since the Commission's April 13 decision to permit low-power testing, counsel has received six additional affidavits , including two from GAP investigation Richard Parks and other affidavits from four current and former employees at the plant. [3/]

I. DESIGN QUALITY ASSURANCE BREAKDOWN

Although QA deficiencies for large- and small-bore piping are common in the nuclear industry, Diablo Canyon suffered a comprehensive QA breakdown in these areas. At the April 6, 1984, Advisory Committee on Reactor Safeguards ("ACRS") meeting, Inspector Yin put the violations in perspective. "What makes it uncommon with Diablo Canyon is that, first, all areas consist of deficiencies. . . " (Transcript of April 6, 1984 ACRS meeting, p.

3/ Two witnesses submitted affidavits on condition that that confidentiality would be protected through deletion of any identifying characteristics. In those cases uncensored versions of their statements will be provided to the NRC staff or the Office of Investigations.

338) ("ACRS transcript"). Even worse, these massive violations took place during the last major remedial design verification program, which was supposed to represent the final word on Diablo Canyon.

In his March 26, 1984, prepared statement to the Commissioners, Mr. Yin analyzed the cause of the QA breakdown: "At the time of the December 15, 1983 meeting, none of the issues was considered to be a problem by the D[iablo] C[anyon] P[roject]. However, during follow-up inspections, <u>all</u> of the above items had resulted in staff assessment of violation items. The event reflected DCP's lack of concern for establishment and implementation of a sound design control QA program." (Written Statement of NRC Inspector Isa Yin, presented at March 26, 1984, Commission Briefing on Diablo Canyon, p. 2) ("March 26 Yin statement"). (emphasis added).
The Mothers submit that under these circumstances, PG&E has forfeited any opportunity to serve as the judge and jury of its own work.

PG&E further disqualified itself through public statements immediately after the April 13 licensing vote, when chief executive officer Frederick W. Mielke Jr. predicted the plant would be ready for full power operation within four to six weeks. It is inherently uncredible to assume in advance that a design QA breakdown that affected "all areas" for large- and small-bore piping could be corrected in six weeks. It would take much longer merely to identify the deficiencies, let alone fix them.

To illustrate the scope of the problems, one of whistleblower Charles Stokes' allegations which Mr. Yin "fully substantiated" involved "thousands (more than thirty 2½" bindersfull)" of unreliable "Quick Fix" design changes. (Board Notification 84-0,. March 29, 1984 Yin draft, pp. 49,55). In light of the uncontrolled nature of the program, a sampling review by PG&E is unacceptable. Every "Quick Fix" must be examined by an objective reviewer. Every Quick Fix that is skipped or glossed over will represent a potentially dangerous question mark during commerical operations.

Finally, PG&E disqualified itself through misleading or material false statements to the Commission a week and a half prior to the April 13 licensing vote, on the same issues that were holding up the license. An April 30, 1984, affidavit from whistleblower Charles Stokes, enclosed as Exhibit 1, alleges 17 misleading or material false statements by licensee representatives at an April 2, 1984, public meeting on Mr. Yin's findings. Mr. Stokes' charges should be carefully considered due to his strong credibility to date on factual issues. As Mr. Yin pointed out on March 26, "Almost all of the Stokes allegations assigned to me for follow-up had been substantiated." (March 26 Yin statement, p. 1).

Clearly, an independent party free from conflicts-of-interest is a precondition for legitimate corrective action. The current program promises to turn into a public relations effort by the same organizations responsible for the plant's current quality indeterminate state. Unfortunately, a public relations effort will not solve the threats to public safety that remain dormant at Diablo Canyon.

II. CONSTRUCTION QUALITY ASSURANCE BREAKDOWN

A. Causes

The definition of the quality assurance program at Diablo Canyon reveals both the scope and causes of the quality assurance breakdown. An April 19, 1984, affidavit from a confidential witness, enclosed as Exhibit 2, reveals a decisive omission in the "PPP Employee Self-Study Book #2" issued by Pullman Power Products, one of the primary contractors at Diablo Canyon. In the Self-Study Book Pullman rewrote 10 CFR 50, Appendix B, Criterion I, the cornerstone of the NRC's quality assurance regulations. Pullman's version eliminated the portion of the law that guarantees necessary "authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations" for QA personnel. Pullman's version also deletes the requirement for QA access to project management, and dilutes the requirement to separate craft and QA requirements. (Exhibit 2, pp. 6-8).

The whistleblower, who still works at the plant, explained the significance: "Had Pullman complied with the legal version of 10 CFR 50, App. B, the proper respect for safety-related

work could have been maintained throughout the company. However, the Pullman version pervaded the attitude of the supervisors involved." (Id., p. 8).

B. Case studies

Individual case studies highlight the results of the QA breakdown -- ineffective corrective action and dormant hardware deficiencies that haven't been fixed.

1. Bolting

Three witnesses provided affidavits describing alleged QA violations for safety-related bolting throughout Diablo Canyon. An April 18, 1984, affidavit from a confidential witness, enclosed as Exhibit 3, charges the following violations of American Institute of Steel Construction ("AISC") requirements --

> (a) design drawings not specifying elongated holes;
>
> (b) hole sizes outside of Code specifications;
>
> (c) torquing method;
>
> (d) bolt reuse; [and]
>
> (e) examples of 'packing' violating foreign material specifications.

(Exhibit 3, p. 10).

To illustrate the effects of the QA violations, the former inspector reported his discovery of a Unit I pipe rack "where six of the eight mounting/bolt holes were elongated to the point where the washers could not cover the holes . . . In some instances I found the crafts had stuffed the holes with short sections of soft tie-wire to serve as packing." (Id., pp. 2-3).

The causes of the QA violations again were deficient procedures and management's attitude. The installation procedures, Engineering Specification - Diablo ("ESD"), that set the

standards for quality control ("QC") inspections "were supposed to conform to the AISC/ASTM [American Society of Testing Materials] codes, when in actuality they often conflicted with them." (Id., p. 2). When the whistleblower repeatedly identified the problem, he repeatedly received the same response: "[W]e had always done it this way, PG&E is aware of it and had accepted it as is." (Id., p. 3).

Under these conditions, corrective action was ineffective. In a November 19, 1980, Nonconformance Report ("NCR"), DC2-80-RM-002, PG&E identified and pledged to correct the bolting problem. Unfortunately, the NCR itself was deficient by failing to identify that procedures improperly deviated from the relevant contract commitment to honor the AISC Manual. Not surprisingly, "the resolution of the bolting problem was resolved by instructions to deviate from the requirements of the AISC Manual," according to an April 26, 1984, affidavit by former Pullman QC inspector Steven Lockert, enclosed as Exhibit 4. For A-325 and A-490 bolts on rupture restraints, the corrective action permitted the bolts to be tensioned to 55% and 25% of the minimum tensile strength, respectively, compared with the AISC requirement for at least 70% minimum tensile strength. (Exhibit 4, pp. 4,5,8).

In other instances, the corrective action commitments appeared to reflect, rather than neutralize, cost and scheduling pressures. For example, a current inspector explained that the same PG&E NCR "provided explicit instructions for the handling of accessable [sic] and fairly easily resolved problems and provided a built-in escape clause for problems that were inaccessable [sic] or required extensive rework." (Exhibit 2,

p. 6).

The result is that in late 1983, three years after NCR DC2-RM-002, significant procedure deficiencies remained. Mr. Lockert reports the following violations:

> [1.] The tables provided for the description of acceptable washers had not been updated per the requirements of AISC, Sec. 5, Page 191, Para. 2(a).
>
> [2.] Acceptance criteria for High Strength bolts was not defined in ESD 243. Field Inspectors did not know, nor were they legally able to reject bolts that were defective per ASTM A-490, ASTM-325, and ANSI B18.2 requirements.
>
> [3.] Bolt Torque Tables in ESD 243 were still out of compliance with AISC Manual requirements as late as December '83. Discussions with Pullman Field Engineers Dale Warren and Larry Werner indicated that although the tables had been recently updated, they still do not meet AISC Manual requirements.

(Exhibit 4, p. 6). Similarly, training remained deficient. (Id., p. 7).

The bottom line is that literally at the nuts and bolts level, hardware remains deficient. Between July and December 1983, when he has terminated, Mr. Lockert identified the following recurring violations:

> 1. Unauthorized modifications to fillet welds that encroached on bolt or washer land areas.
>
> 2. Oversize holes already QC accepted outside the tolerances of ESD 243 and AISC Manual.
>
> 3. Oversize holes in base plates packed with steel rods and wires without the benefit of an approved Pullman procedure. (This work was performed to a memo from Mr. Torstrom in violation of 10 CFR 50 App B, Criteria V and VI.)
>
> 4. Oversize welds beyond that allowed by AWS D1.1 and beyond that allowed by Pullman's ESD 243.
>
> 5. Defects in A-490 bolts had been found after the bolts had been "dedicated" by Pullman's QA Receiving Department and sent to the field for installation.

(Exhibit 4, p. 5).

2. <u>Reactor Coolant System welds and piping</u>

On an April 11, 1984, plant tour, a whistleblower confirmed problems with welding and minimum wall thickness on reactor coolant system piping. More specifically, the alleged deficiency exists adjacent to weld No. WIB-RC-2-16 on the safety injection accumulator (or core flood tank) line.

On the April 11, 1984, plant tour the witness confirmed to NRC inspector Kirsch and GAP investigator Richard Parks that the problems with weld RC-2-16 were still uncorrected. <u>See</u> Mr. Parks' April 17, 1984 affidavit, at pp. 1-3, enclosed as Exhibit 5, quoting inspector Kirsch's "Problem Desc:iption." The conditions confirmed by Mr. Kirsch's Problem Description obviously violate ASME Section III and are symptomatic of minimum wall thickness violations on the piping.

The significance of the issue is self-evident from a review of the public record. The licensee's five month delay from December 1982-May 1983 in reporting similar violations on a neighboring weld led to a citation, due to the necessity for "remedial action or corrective measures to prevent the existence or development of an unsafe condition." Indeed, the condition was so significant that it had to be reported in 24 hours. <u>See</u> Board Notification 50-275/83-83 (June 24, 1983). As pointed out in IE Report 83-20, at p. 5, the minimum wall thickness violations "if left uncorrected could have resulted in degradation or loss of integrity of the reactor coolant pressure boundary." Quite possibly, they still could.

Regardless of legalities, the issue is highly significant. As Mr. Parks, a startup engineer, explained in an April 30, 1984,

affidavit, enclosed as Exhibit 6,

> This weld connects the "Safety Injection Accumulators" (core flood tanks) to one of the Reactor Coolant Cold leg(s). This tank is required to be available during a "loss of coolant accident" to inject borated water into the core to ensure it is and remains in a safe, shutdown condition during an accident. This is because the borated water prevents the fission process by absorbing the neutrons required for fission. A failure of this line could "prevent an Engineered Safeguards Actuation System from performing its design function (maintaining the core shutdown)." A failure of this system would also violate the Accident Analysis of the Final Safety Analysis Review of the Plant, and every avenue should be pursued to assure this system boundary has not been violated.

(Exhibit 6, p. 8).

Unfortunately, PG&E didn't take that approach. In theory, this condition was resolved through a corrective action plan approved by NRC Region V. See IE Report 50=275/83-26 (August 5, 1983). In fact, the sampling program was not properly expanded (Exhibit 6, p. 5) and a relatively inaccessible weld such as RC-2-16 was missed.

Even more significant, PG&E knew better for this particular weld, despite the sampling deficiencies. In March or April of 1983, a whistleblower had identified the deficiency on-site. (March 23 supplement to Mothers for Peace 2.206 petition, Exhibit 12, p. 5).

These case studies raise the question of how many similar deficiencies remain dormant at Diablo Canyon. No one can answer with certainty. The breakdown in corrective action helps to explain the quality indeterminate state of the plant.

III. NRC QUALITY ASSURANCE BREAKDOWN

The failure of NRC-approved corrective action in the Reactor

Coolant System illustrates a basic lesson. The NRC staff responsible for Diablo Canyon is also responsible for the current condition of the plant. That may explain the perspective adopted by Region V in interviews with whistleblowers -- indifference to hostility. It also may explain the staff's perspective in Commission briefings -- advocate for PG&E.

Whatever the cause, Mr. Martin's April 13 briefing to the Commissioner eliminated the last shreds of Region V's credibility with whistleblowers. Mr. Martin did not have time to fully brief the Commission before the April 13 vote; he had a plane to catch "[a]s soon as I can get away. (Laughter.)" (April 13, 1984, Commission transcript, p. 47). But the Regional Administrator did find time to dismiss any significance from a plant tour with whistleblowers the night of April 11. He reported that after a second look the staff concluded, "at least preliminarily, that none of them violate any requirements." (Id., p. 46).

The second look must have been miraculous, in light of the five specific deficiencies recorded the night before. The items included the problems connected with weld RC-2-16. Mr. Parks' April 17 affidavit quotes NRC inspector Dennis Kirsch's "Problem Descriptions" of conditions that violated relevant code requirements for five items on the plant tower. (Exhibit 5). The problems -- underout, gouging, overwelding that led to pipe shrinkage and residual stress, ragged shopwelds, undersized welds, and fillet welds where full penetration welds are required -- were too serious to disappear legitimately overnight.

The staff's handling of the plant tour itself already had

eroded confidence. One whistleblower cancelled out of the tour due "to fears [of] my identity becoming known and my personal safety on-the-job jeopardized. I feel the potential existed for compromising my confidentiality because the NRC was callous and awkward in handling the details for the tour." (Exhibit 2, p. 1). By contrast, the staff refused to permit another whistleblower to attend the plant tour, despite prior agreement. As the witness explained,

> The NRC's position was that they had not spoken with me, and it had not been agreed to the night before so they couldn't let me go. Mr. Parks informed me that other witnesses had decided to back out because of fears that their identities would be compromised. I agreed to go in their place. I wasn't afraid, I would have been with NRC personnel and because I didn't work there anymore, I could not have been retaliated against later. The NRC declined again and made no effort to even speak with me about my concerns. I was astonished and angered.

(Exhibit 3, p. 5).

Region V has continued to exhibit bias since the April 13 decision on low-power testing. For example, on April 30 Region V official Thomas Bishop reported that "we did not see a widespread problem" with intimidation and unfair dismissal of whistleblowers (Thomas Hayes, "Diablo Canyon Reactor Starts Up Amid Protests and Industry Praise," The New York Times, April 30, 1984, p. B8). Mr. Bishop neglected to mention that those issues are under the jurisdiction of the Office of Investigations, which has just begun its probe of alleged retaliation. Mr. Bishop's exoneration was premature, to put it mildly.

Mr. Bishop also failed to identify "an undermining of the quality assurance program" from alleged intimidation and reprisals. In this instance, Mr. Bishop's reassurances have an

Alice-in-Wonderland flavor. For example, one of the reprisal victims, Mr. Charles Stokes, raised issues that led to Mr. Yin's findings of a design QA breakdown. The repression was so severe that Pullman's internal auditor Harold Hudson resigned and took a pipefitting job, rather than continue to endure the frustration and harassment. (February 2, 1984 Mothers for Peace 2.206 petition, Attachment 2, p. 13). As a new witness explained in his April 18 affidavit, "Finally, in . . . 1981 I had the opportunity for other work away from Diablo Canyon. I immediately took it even with a reduction in pay. I was relieved to be removed from the harassement [sic] and the batting of my head against a brick wall." (Exhibit 3, p. 4).

On balance, Mr. Hudson explained the impact of retaliation at Diablo Canyon: "[A] significant number of QA violations have gone unreported . . . Those who persist in reporting the violations are dismissed, or harassed relentlessly until they resign, or give up and stop trying." (February 2 petition, Attachment 2, p. 30). (See generally February 2 petition, pp. 32-38, and March 1 petition, pp. 15-17).

Region V's inability to "see" these problems raises concerns about its eyesight. The families living near Diablo Canyon sorely need help from an agency committed to defending public safety, rather than to defending the status quo. Unfortunately, the Region V management has discarded the former role and adopted the latter. It is imperative that a legitimate regulatory and corrective action program be instituted immediately before another fait accompli. Until now, the stonewall syndrome at Diablo Canyon has been a disappointment. After low-power testing,

it will become a threat to the public.

Respectfully submitted,

Thomas Devine

Thomas Devine
Counsel for
Mothers for Peace

EXHIBIT 1 CCS

AFFIDAVIT

My name is Charles C. Stokes. I am submitting this affidavit to the Nuclear Regulatory Commission (NRC) to inform them of material false and/or misleading statements made by PG&E/BECHTEL in the meeting on April 2, 1984 between PACIFIC GAS & ELECTRIC COMPANY and the NRC on DIABLO CANYON UNIT 1.

In the April 2 transcript on page 33, Mr. Shipley states "The supervisor trains the new employee, although new means new to Diablo and not new to the process. He trains that person on the job, carefully checking the first work that he does." During the time I spent under Mr. Mangoba, the Pipe Support Lead supervisor, I saw **new** people brought into the design group who were given other engineers' work to check before ever performing any design work of their own. This was a result of 1) pressure to get the work done and 2) the new people were slower as originators than the people who had been on the job longer 3) by giving the new personnel work to check instead of design, production was not effected. Employees still in Mr. Mangoba's trailer told CCS me that this practice Followed him in CCS the March 1983 move to the new unit 1 trailer. The trailer staff was comprised of a fifty-fifty split between new employees and old employees. As of that date none of the unit 1 calculations had been completed.

On page 35 Mr. Shipley continues by stating "I believe that Mr. Yin's approach to the problem would have been extremely conservative. I believe that the analyst's approach to the problem was a reasonable representation of the piping and support when taken together." I am aware of the problems which the NRC discovered in hanger 99-20 and I am sure that if the professors teaching in the engineering schools were polled on whether Mr. Yin or the PG&E personnel are taking the most reasonable approach, the results would show that Mr. Yin's would be considered the most reasonable, as I my self do.

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Mr. Kahler on page 65 testified that "In their investigations, they identified that in OPEG group, there were sixty three manuals containing one hundred and thirty three criteria documents, four hundred and twelve procedures, and fifty one instructions were review -- to give you an idea of the scope that was done for this particular issue. The results of that review showed that ninety percent of the documents were -- that were under control, were properly and correctly in place. In no cases, did they find any out of date criteria." Note the words used by Mr. Kahler following the second pause "that were under control". Was this an attempt to avoid making a false statement? Even though no statement was made as to whether any review was made of the documents of personnel who were not assigned control documents to see if they possessed old out-of-date control documents, evidently Mr. Kahler was aware that out-of-date documents did and probably still do exist in the employees' control and use.

On page 66 Mr. Kahler states that "engineers would receive a procedure, sign off that he had received it". This statement is either misleading or false depending on how Mr. Kahler used the word engineers. During my employment and as one of the few to have controlled documents I received many revisions and was asked to sign only once for receiving them. In using the term engineers was he indicating management and the clerks? I know it didn't apply to the casuals or job shoppers.

Mr. Oman continues with this ridiculous assertion on page 69 and I quote "and the control and distribution of those procedures was managed by the project administration group, using a system of signed returned receipts." The only way this statement can be true during the time I was in OPEG is that the project administration group signed the receipts themselves. I am assuming that the project administration group includes management and clerks.

On page 72 Mr. Oman states "there was always a return receipt system with

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distribution of instructions." The only return receipt I saw was when I received my first documents, never later.

Mr. Kahler again states on page 73 that "The requirement is that if an engineer wishes to keep an outdated procedure in his manual, he is required to mark it as a superseded procedure, clearly mark it as superseded." I was never instructed either orally or in writing that if I wished to keep the old procedures that I should write superseded on them.

On page 73 everyone attempts to get into the action when Mr. Vollmer asks "How often are the supervisors supposed to review their employees manuals for current status?" Mr. Oman answers "I believe the procedure either specifically states which I believe it does that it's a monthly requirement, that the supervisor review the manuals of the engineers under his supervision on a monthly basis." Then Mr. Tresler says that "I just spoke with Myron Leppke and he informed me that the procedure had been to perform this review on a monthly basis. Recently it was changed to a periodic basis,". (top of page 74) During my involvement with OPEG I never saw nor was otherwise made aware that my supervisors performed this inspection.

Mr. Tresler continues to be mistaken on page 74 about whether this review is documented. "I'm sorry, it is documented." "It's documented as a report by QA, those QA individuals assigned to monitor OPEG." Mr. Vollmer says "It's an audit function of theirs?" Tresler "Yes" "No, I say it is documented, it is documented in an audited report." Then following a pause "I'm sorry. As a clarification, this is Mike Tresler again. Apparently, the audits performed by the supervisors are not documented but there are audits performed by the QA organization within OPEG to verify that the audits being performed by the supervisors are effective." Mr. Tresler still doesn't give up. When asked by Mr. Vollmer "so, how do they audit an activity that's not documented?" Tresler

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says "they audit the manuals to verify that the supervisors' reviews are effective." To my knowledge this review was never documented nor conducted by my supervisors nor was any audit ever performed on my documents to see if they were up to date and even if they were in order that finding would not prove that the supervisor was performing this review.

I find the statements made on page 84 by Mr. Kahler that "In our reviews, we concluded that there was no effect on the design process." and was followed by Mr. Allison that "Not only on the product but on the process." to be ridiculous. This is in light of the following facts 1) that PG&E has admitted that they have found that approximately 74% of the small bore calculations have what they consider minor problems and an additional 22% which required completely redoing in order to be confident of the initial work, 2) that since I submitted my DR on generic welding problems on units 1 & 2 PG&E/BECHTEL have issued scores of memorandums and made procedure changes in an attempt to clear up many questions ranging from the design group to the field construction personnel, 3) PG&E has spent the last several months trying to explain away my allegations of QA problems, destruction of documents, technical deficiencies in the calculations (such as omission of eccentricies, secondary stresses from torsion, anchor bolt spacing requirements per the manufacturer and M-9 the Pipe Support Design Manual issued by PG&E, and the failure to limit structural angle members length per AISC Sect. 1.5.1.4.6b, the use of gaps to reduce thermal loads to supports, the placement of snubbers rigids and anchors close to other supports, and others) which were substantiated by Mr. Yin and many remain unresolved.

Mr. Manoli asks a pertinent question about the Diablo Problem (DP) program on page 93 "Did any of these DP's have dispositions on generic bases that effect other packages or more generic implications that you really need to document it so that you can handle it in all applicable cases, not just on a

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single case." Which is answered by Mr. Tresler "No." Each DP was specific to a discipline and was not a plant generic issue or concern". During the time I was employed at the site, I know management suppressed the use of Design Change Notices (DCN's), Discrepancy Reports (DR'S) and Non-Conformance Reports (NCR's). DP's were used to report problems on specific hangers, problems about a list of hangers, and frequently generic problems on both units 1 & 2.

On page 95 Mr. Shipley in explaining the lack of a procedure on the use of gaps, the lack of a procedure on "developing a 'KL over R' criteria, buckling, the engineer must determine what that end condition is and apply the appropriate factor in order to arrive at the proper result. It's a well-known engineering technique and it is not considered necessary to instruct the engineer precisely in each and every case which one he should use." He closes on page 95 with "we believe that a specific procedure is not required because it's common engineering practice." I have worked as a structural engineer for the past 9 years on many nuclear projects and even though these principles are taught in colleges, they are the most incorrectly used. They may be calculated close to correct on simple structures, but on complicated pipe supports when time is limited by the demand for quantity rather than quality almost no one performs these types of detailed analysis or get them right if they do attempt them. Procedures are needed to refresh memories, and provide consistency in application.

Mr. Soffell follows up on page 102 with "I'm wondering where cases of gaps and/or joint releases, that is the exceptions, are flagged so that the checker is kind of, so to speak, being asked, do you agree with what I've done here." This is responded to by Mr. Shipley "Okay. So there's a piece of paper that says, hey, I did this. In the computer model you would see a gap in the actual input to the analysis, in the output and so forth." The answer is **NO**, the exceptions are not flagged. The only way you would be able to find them is to

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know of their use (my method) or perform an in depth review of each analysis package.

Mr. Shipley again on page 112 misleads everyone with "It was a very well-controlled program". Careful review of specific information supplied by Mr. Tateosean demonstrates that Mr. Shipley's conclusion was false. It was not a "very well controlled program". Mr. Tateosean states on page 113 "On cited interferences, I've gone back and talked to the stress engineer who was on the walkdown". What criteria was established and followed to distinguish cited interferences from those which weren't cited as interferences? With only 10 people who performed the stress walkdown, why didn't Mr. Tateosean question them all, and not just the stress engineers? He also states "other interferences on these lines, but in his judgment, what he saw here was really interferences that weren't interferences because the -- it was such a slight interference." Was this program conducted on intuition as was the design calculations Mr. Shipley speaks of on page 147? What was the criteria which each member could apply to decide consistently what was an interference? Mr. Tateosean says on page 113 "Typically you had an inch and a half or so of insulation, and we're talking about calcium silicate insulation and it has the ability to crush that much or more." Had Mr. Tateosean's stress walkdown been provided criteria such as that provided in the FIELD ENGINEER POCKET HANGER REFERENCE which BECHTEL went to the trouble and expense to write and then changed their mind about issuing, even the crushing of calcium silicate insulation would have become important. I would like to quote from BECHTEL's proposed FIELD ENGINEER POCKET HANGER REFERENCE on pages 1-10 and 1-11 under "NOTES: PIPE INSULATION CHART".

Forth paragraph, "Most insulation failures are caused by water entering through breaks in the finish, such as expansion crack, or un-flashed openings, therefore, particular attention should be given to complete detailed speci-

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fications in regard to weatherproofing." CCS

From paragraph 5, "The usual insulating materials and jackets for heated piping and equipment allow the moisture to escape in the form of vapor. However in the medium temperature range, and where shut-downs are frequent, moisture in the insulation is not driven off and water damage is most likely to occur. For these conditions, the insulation should be thoroughly dry before applying the jacket, the surface of the pipe should be primed and painted, and corrosion-restraint wire or bands used for securing the insulation. If possible, insulation should be applied to high temperature piping while heated to insure the complete dryness of the completed installation."

From paragraph 6, "The layout of insulated piping and equipment should provide adequate clearances for proper application of the insulation and also safeguard against mechanical damage during normal operation and maintenance."

In some case's PG&E/BECHTEL gave up on making a false statement in the middle of the meeting. I question why the NRC didn't go back to consider the implication to the issues that were effected. For example on page 121, Mr. Tresler has the same problem as Mr. Shipley above in that he answers before the question is completed. When asked by Mr. Manoli "was the gap provided around the pipe all around--", Mr. Tresler states "Yes". Mr. Mangoli continues to elaborate on **all around** and on page 122 Mr. Tresler finally realizes the meaning of Mr. Mangoli's question. Quote of Tresler "360 degrees around the pipe." "I guess my answer to that is no." From the transcript it appears that this misleading narration was in innocence but could his replies be due to the pressure of having to answer questions by the NRC on issues which I raised or is this his standard level of competence? Is there more which should be looked into?

In his discussion of the Quick Fix program on page 128 Mr. Oman says, "they would, on a case by case basis, make a judgment based on their knowledge of

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M-9 which is the guide lines for design of Class 1 pipe supports and restraints for the project, the design criteria for pipe supports. They would make a judgment on a case by case basis whether an expanded tolerance, a deviation beyond that specifically allowed by ESD 223, could be made while still maintaining an acceptable support design." I personally know that some of the Quick Fix engineers were hired and placed in the group without ever performing any calculations or spending any time learning what was in M-9 or ESD 223 nor were they given a copy of Instruction 12 which supposedly defines the responsibilities and authorities of the Quick Fix group. Mr. Oman's statement is misleading in that he implies the engineers have knowledge of the documents mentioned above. Can we expect Mr. Oman to supply us with the negative elements as well as the positive without a specific question on point? Would the fact that the QF engineers were not trained in the performance of their assigned tasks bear on the quality of their work? Mr. Yin was not aware that some of the QF engineers had never worked in any aspect of the review program on Diablo Canyon before becoming Quick Fix engineers, until I pointed this out to him.

Mr. Oman states on page 127 "Also, those modifications which — or those hangers which a preexisting condition was determined to be unacceptable were not handled under this program. They were documented by discrepancy reports within Pullman Piping Contractor and General Construction." During the time I was in Quick Fix, almost none of the existing problems were written up on discrepancy reports. This was because I was the only QF engineer to have controlled documents for most of the program and I was the only QF engineer (to my knowledge) to have a copy of a memorandum which was written to clear up questions involving the operation of the program. This document stated that a DR had to be issued against existing supports before I could issue a Quick Fix (QF) resolving the problem. Often when I demanded a DR the field engineer for Pullman would walk

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away saying he had been instructed to get it resolved without having a DR issued. In discussions with the QF engineers on different shifts, I found that another Pullman engineer on their shift had gotten a QF from them without a DR being issued.

He continues on the bottom of 127 to state "Upon completion of construction of that support, the as-built package, the entire as-built package of that support, was included in the original design and any subsequent tolerance clarifications were all incorporated into one as-built package which was returned to engineering for acceptance of the final as-built condition in accordance with project procedures." In discussions with the unit 1 personnel, I was told that they never saw any QF's when approving an as-built, only the as-built drawing. I was told that hardly any one reviewed these in any detail; they just rubber stamped them OK.

On page 129, Mr. Oman states "the fact that every tolerance clarification is included in the as-built package and is reviewed as part of the final hanger acceptance, leads to the conclusion that particular finding would not affect the final qualification of the supports." See comments paragraph above.

Mr. Shipley states on page 145, "I'm actually reading from the February 7th submittal that acceptable with minor supplemental calculations or comments, is 78 percent. Acceptable with detailed calculations, which means that there was something found that the reviewer felt that without additional work, he was not able to justify it on the basis of the original calculation alone -- that was 17 %. And, unacceptable is zero.

That was at the time of this document. At that time there were six supports that had yet to be completed. They have since been completed and they are also acceptable. So, that would bring the 17 to 22 percent, today."

I would like to point that all through the April 2 transcript the 17

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percent figure has been used without any correction being proffered by PG&E/-BECHTEL. The first I believe is on page 42 when Mr. Yin and Mr. Shipley used it, the second was the quote above, the third is on page 156 when it was used by Mr. Faulkenburg and Mr. Shipley again, and the forth is on page 160 when it was used by Mr. Taylor and Mr. Shipley again. I am sure there are other locations where the 17 percent is used without a correction when the number should be 22 percent. Maybe I expect too much voluntary information but 22 seems more significant than 17.

Mr. Vollmer on page 147 asks "what sort of instructions are the checkers given, who perform that evaluation." Mr. Shipley replies on page 147 that "there is an intuitive ability of the designer, an experienced designer, to understand small bore piping." This point is followed up on by Mr. Manoli on page 154 with this comment: "So, it leaves, I think a hole here, where a person can just make judgments and thinks that the support is adequate." I would like to add that we were asked by group leaders to use our judgments on all most everything in the design. The worst use of this was when we all followed management's directive to take for granted that the supports as installed were installed under a valid Quality Assurance (QA) program. This I discovered was far from the truth. How much credibility can be given a reverification program which was based on intuition? There were so many assumptions which had no truth or basis which were never questioned in the review program that I can not see how anyone living in the vicinity of the plant can be safe with Diablo operating. The omission of information supplied by PG&E/BECHTEL similar to that supplied by me above, I feel is relevant for the companies' credibility.

On page 157 Mr. Tresler says "The judgments were used more in the small bore that it was in the large bore.

And I think that Larry is trying to point out also that this is industry

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practice. Is that correct?" Mr. Shipley replies "Yes". It is my experience that Diablo Canyon if it is industry practice to be at the lowest end of the scale and had I worked on any plant that I believed to be as unsafe as Diablo then I would never have gotten to work on Diablo for I would have become a WHISTLEBLOWER on that plant.

Mr. Tresler makes the statement on page 171 that "There was a very short period of time where the vehicle of phone calls were used in lieu of the normal process," and he continues on page 172 with "I don't know — a month or so, the work was expedited by use of the phone call, and the intent was that those calculations would not be finalized until the written information came through." I was on site from Nov. 8, 1982 until Oct. 14, 1983 and during this time the phone was consistently used to obtain necessary design information and almost none of the engineers documented these calls since no phone memorandum forms were available. Only a few of us indicated in the calculation that it was preliminary and that a written reply was necessary.

On page 175 Mr. Knight asks "Okay. So, for the record, .025 was the criterion?" and was answered by Mr. Shipley "Yes, sir." Mr. Knight asked again "And it was the only criterion that was employed?" and Mr. Shipley replied again "Yes". This is not true, we also used .009 inch. Both of these values were supplied to us in M-9. The .025 value was for 20 hertz and .009 was for 33 hertz.

Mr. Shipley on page 178 says "The 20 hertz is -- is — is only a criteria. It clearly doesn't set a pass/fail situation for the support -- ". As one of the criteria we were designing to, the support failed if it did not meet this requirement. I wonder now after considering Mr. Shipleys' statement if those supports which we failed due to insufficient stiffness were later changed to passing?

I have read the above 12-page statement and it is true and correct to the

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best of my knowledge and belief. I feel the disclosed false and/or misleading and/or misrepresented information is relevant to any future review work which is to be undertaken on Diablo Canyon by PG&E/BECHTEL.

Charles C. Stokes

Subscribed and sworn to before me this 30th day of April, 1984.

CARSEL
LIFORNIA
O COUNTY
My Comm... Expires Sept. 14, 1534

Notary Public in and for
the County of San Luis
Obispo, State of California

- EXHIBIT 2

A F F I D A V I T

My name is   I am providing this statement fully and voluntarily, without threats, inducements or coerceion to Richard Parks, who has identified himself to me as a volunteer investigator investigating alleged problems at the Diablo Canyon Power Plant.

I am submitting this statement to evidence the code and specification violations I could have identified to the NRC on the plant tour on 4-11-84. I did not accompany the tour as planned because I had reason to fear my identity becomming known and my personal safety on the job jeopardized. I feel the potential existed for compromising my confidentiality because the NRC was callous and awkward in handling the details for the tour.

I had intended to identify some examples of unacceptable workmanship with respect to the following three codes and specifications:

1. Vendor welds not complying with applicable AWS Code D1.1 Section 8.15 "Quality of Welds".

8.15.1 <u>Visual Inspection</u>. All welds shall be visually inspected. A weld shall be acceptable by visual inspection if it shows that

8.15.1.1 The weld has no cracks.
8.15.1.2 .....
8.15.1.3 All craters are filled to the full cross section of the weld.
8.15.1.4 Weld profiles are in accordance with 3.6.
8.15.1.5 Irrespective of length, undercut

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shall not exceed the value shown in Fig. 8.15.1.5 for the primary stress direction category applicable to the area containing the undercut. Further, the undercut may be twice the value permitted by Fig. 8.15.1.5 (for the applicable stress category) for an accumulated lengh of 2 in. in any 12 in. (51 mm in 305 mm) length of weld, but in no case may undercut on one side be greater than 1/16 in. (1.6 mm), the permitted length should be proportional to the actual length.

2. Violations of ASTM/AISC Codes governing bolting requirements on Rupture Restraints, and Class 1 structural steel installations. The Manual of Steel Construction (AISC), specification for "Structural Joints Using ASTM A325 or A490 Bolts", section 3, "BOLTED PARTS" states,

(a) The slope of surfaces of bolted parts in contact with the bolt head and nut shall not exceed 1:20 with respect to a plane normal to the bolt axis. Bolted steel parts shall not be separated by gaskets and shall fit solidly together after the bolts are tightened. Holes may be punched, subpunched and reamed, or drilled, as required by the applicable code or specification. Standard holes shall have a diameter nominally 1/16-in. in excess of the nominal bolt diameter.

Where shown in the design drawings and at other locations approved by the designer, oversize, short slotted, and long slotted holes(see Table 7 in Commentary) may be used with high-strength bolts 5/8-in. diameter and larger in connections assembled as follows:

1. Oversize holes may have nominal diameters up to: 3/16-in. larger than bolts 7/8-in. and less in diameter, 1/4-in. larger than bolts 1-in. in diameter, and 5/16-in. larger than bolts 1 1/8-in. and greater in diameter. They may be used in any or all plies of friction-type connections. Hardened washers shall be installed over oversize holes in an outer ply.

2. Short slotted holes are nominally 1/16-in. wider than the bolt diameter and have a length which does not exceed the oversize diameter provisions of subsection 3(a)1 by more than 1/16-in. They may be used in any or all plies of friction-type connections.

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or bearing-type connections. The slots may be used without regard to direction of loading in friction-type connections but shall be normal to the direction of the load in bearing-type connections. Hardened washers shall be installed over short slotted holes in an outer ply.

3. Long slotted holes are nominally 1/16-in. wider than the bolt diameter and have a length more than allowed in subsection 3(a)2 but not more than 2½ times the bolt diameter. The slots may be used without regard to direction of loading in friction-type connections but shall be normal to the direction of the load in bearing-type connections.

Long slotted holes may be used in only one of the connected parts of either a friction-type or bearing-type connection at an individual faying surface.

Where long slotted holes are used on an outer ply, a plate washer or continuous bar of at least 5/16-in. thickness with standard holes shall be provided. This washer or bar shall be of structural grade material, but need not be hardened. If hardened washers are required to satisfy Specification provisions, the hardened washers shall be placed over the outer surface of the plate washer or bar. These washers or bars shall have a size sufficient to completely cover the slot after installation.

(b) When assembled, all joint surfaces, including those adjacent to the bolt heads, nuts or washers, shall be free of burrs, dirt, and other foreign material that would prevent solid seating of the parts. Paint is permitted unconditionally in bearing-type connections.

5 INSTALLATION

(c) Turn-of-Nut Tightening

When the turn-of-nut method is used to provide the bolt tension specified in subsection 5(a), there shall first be enough bolts brought to a "snug tight" condition to insure that the parts of the joint are brought into good contact with each other. Snug tight is defined as the tightness attained by a few impacts of an impact wrench or the full effort of a man using an ordinary spud wrench. Following this initial operation, bolts shall be placed in any remining holes in the connection and brought to snug tightness. All bolts in

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the connection shall then be tightened additionally by the applicable amount of nut rotation specified in Table 4, with tightening progressing systematically from the most rigid part of the joint to its free edges. During this operation there shall be no rotation of the part not turned by the wrench.

(d) Calibrated Wrench Tightening

When calibrated wrenches are used, they should be set to provide a tension at least 5% in excess of the minimum bolt tension specified in subsection 5(a). The wrenches shall be calibrated at least once each working day for each bolt diameter being installed. Wrenches shall be recalibrated when significant changes are made in the equipment or when a significant difference is noted in the surface condition of the bolts, nuts, or washers. Calibration shall be accomplished by tightening, in a device capable of indicating actual bolt tension, three typical bolts of each diameter from the bolts being installed.

When adjusting the wrenches to provide the required tension, it shall be verified during actual installation in the assembled steelwork that the calibration selected does not produce a nut or bolt head rotation from snug tight greater than that permitted in Table 4. If manual torque wrenches are used, nuts shall be in tightening motion when torque is measured.

When using calibrated wrenches to install several bolts in a single connection, the wrench shall be returned to "touch up" bolts previously tightened, which may have been loosened by the tightening of subsequent bolts, until all are tightened to the prescribed amount.

(f) Reuse

A490 bolts and galvanized A325 bolts shall not be reused. Other A325 bolts may be reused if approved by the engineer responsible.

Retightening previously tightened bolts which may have been loosened by the tightening of adjacent bolts shall not be considered as a reuse.

6 INSPECTION

(a) The Inspector shall determine that the requirements of Sections 2, 3, and 5 of this Specifi-

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cation are met in the work. When the calibrated wrench method of tightening is used, the Inspector shall have full opportunity to witness the calibration test prescribed in subsection 5(d).

(b) The Inspector shall observe the installation of bolts to determine that the selected procedure is properly used and shall determine that all bolts are tightened. Bolts installed by the turn-of-nut method may reach tensions substantially above the value given in Table 3, but this shall not be cause for rejection.

COMMENTARY C5 INSTALLATION

Where long slotted holes are used, experimental evidence has shown that a plate washer or continuous bar of at least 5/16-in. thickness with standard holes is necessary to provide adequate bearing. This washer or bar shall be of structural grade material but need not be hardened. However, if hardened washers are required to satisfy Specification provisions, the hardened washer shall be placed over the outer surface of the plate washer or bar.

3. Examples of non-compliance with Pulman Power Products' own Engineering Specifications - Diablo (ESD's).

To refresh my memory and to help me identify some of the above mentioned problems and their locations in the plant, I had requested my DCN log, my file of DCN's (Deficient Condition Notices) and other readily available documents. I feel this could have contributed to the NRC's awkward approach in handling my anonymity.

I believe a thorough review and comparison of the actual hardware with the existing documentation would have revealed violations of Appendix B, 10CFR50, Quality Assurance Requirements for Nuclear Plants.

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My concerns relating to ASTM/AISC Bolting Requirements led to a review of an internal PPP document "Tensioning-ESD-243" Authored by R.L.Werner, which deals with the inadequacy of ESD 243 with respect to under tensioning and over tensioning of A325 and A490 bolts. This document also delt with the implementation of the disposition of NCR DC2-80-RM-002, dated 11-19-80. Page 3, paragraph 5 states:

> Bolts which have rejectable indications shall be discarded and replaced with new bolts and new nuts. If bolts are grouted in wall the connection shall be "As-Built" and the As-Built submitted to the assigned engineer for review and disposition.

This document leads me to believe that PG&E provided explicit instructions for the handling of accessable and fairly easily resolved problems and provided a built-in escape clause for problems that were inaccessable or required extensive rework.

Another document I reviewed was PPP EMPLOYEE SELF-STUDY BOOK #2, relating to Pullman's version of 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants." The Pullman version differs substantially from the legal version with respect to organizational structure for the QA program. The official version reads as follows:

> I. ORGANIZATION
>
> The applicant[1] shall be responsible for the establishment and execution of the quality assurance program. The applicant may delegate to <u>others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program,</u> or any part thereof, but shall retain responsiblitity therefor. The authority and duties of persons and organizations performing <u>activities affecting the safety-related functions of</u>

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structures, systems, and components shall be clearly established and delineated in writing. These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedon, including sufficient independence from cost and schedule when opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms provided that the persons and organizations assigned the quality assurance functions have this required authority and organizational freedom. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program at any location where activities subject to this appendix are being performed shall have direct access to such levels of management as may be necessary to perform this function.

(Footnote 1.) While the term "applicant" is used in these criteria, the requirements are, of course, applicable after such a person has received a license to construct and operate a nuclear powerplant or a fuel reprocessing plant. These criteria will also be used for guidance in evaluating the adequacy of quality assurance programs in use by holders of construction permits and operating licenses.

(NOTE: Those parts of 10CFR50, App.B, I. ORGANIZATION that are omitted or paraphrased in Pullman's version are underlined.)

The Pullman version is as follows:

The applicant shall be responsible for the establishment and execution of the quality assurance program. The ap-

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> plicant may delegate to <u>other organizations the work of establishing and executing the quality assurance program</u> or any part thereof, but shall retain responsibility therefore. The authority and the duties of persons and organizations performing <u>quality assurance functions</u> shall be clearly established and delineated in writting. <u>Such</u> persons and organizations shall have sufficient authority and organizational freedom to identify quality problems: to initiate, recommend, or provide solutions; and to verify implementation of solutions. <u>In general, assurance of quality requires management measures which provide that the individual or group assigned the responsibility for checking, auditing, inspecting, or otherwise verifying that an activity has been correctly performed is independent of the individual or group directly responsible for performing the specific activity</u>.
>
> (NOTE: Pullman's paraphrases are <u>underlined</u> in the above quote.)

The rest of appendix B is typed verbatim except for the omission of the words "fuel reprocessing plant" where they occur. My "official version"is ((35 FR 10499, June 27, 1970, as amended at 36 FR 18301, Sept 17, 1971; 40 FR 32100 Jan. 20, 1975.))

Had Pullman complied with the legal versi on of 10 CFR 50, App. B, the proper respect for safety related work could have been maintained throughout the company. However, the Pullman version pervaded the attitudes of the supervisors involved. Their attitudes served to restrict inspectors like myself from broadening our knowledge of the requirements and attempting to document and seek out resolution to safety-related problems. Pullman's arrogance in rewriting the law on Quality Assurance disturbs me. The lack of authority and independent freedom of the actual inspectors to cut through red tape and follow a problem to a conclusion can be traced back to the omissions and paraphrases of the legal Code.

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Pullman's omissions effectively placed the inspectors in a position of accepting only work shown to them rather than striving to prevent recurrence of problems in workmanship and design.

I was unaware of Pullman's omissions and thought they had given us a real copy of 10CFR50 App.B to study. In fact, in my first Affidavit I identified a requirement to maintain a separate QA/QC department as a requirement of 10CFR50 App.B even though this requirement is casually addresses in the Pullman relaxed version. It is clearely defined in the legal version. I am deeply concerned with Pullman's relaxed version because of the attitude of management to relax requirements even further in practice.

Based on my knowledge of what Pullman classifies as a QA program, I have serious doubts as to the ability of their version to "stand alone" under the real requirements of 10CFR50, App.B. This is not responsible behavior.

I have read the above 9-page statement, and it is true and complete to the best of my knowledge and belief.

Subscribed and sworn to before me this ____ day of April, 1984.

NOTARY PUBLIC-CALIFORNIA
PRINCIPAL OFFICE IN
SAN LUIS OBISPO COUNTY
My Commission Expires Jan. 25, 198[illegible]

Notary Public

My commission expires Jan 25 1985

EXHIBIT 3

A F F I D A V I T

My name is , I am providing this statement freely and voluntarily, without any threats, inducements or coercion to Richard Parks, who has identified himself to me as a volunteer investigator investigating alleged problems at the Diablo Canyon Power Plant.

I am providing this statement to document my concerns over the improper installation of rupture restraints, pipe supports and equipment foundations. I volunteered to personally identify these problems to the NRC representatives on the plant tour that took place on 4/11/84. My offer was declined. I felt it was necessary for me to accompany the tour to properly identify the locations of the problems that I knew existed before I terminated my employment with Pullman Power Products in 1981.

I was employed at the Diablo Canyon Plant from approximately 1978, until approximately 1979 with the G.F. Atkinson Company. My job status consisted of weld inspector. It was during that period that I functioned as an Inspector that I became intimately familiar with American Society for Testing and Materials (ASTM) and American Institute of Steel Construction (AISC) Codes relevant to bolting requirements on structural joints and surfaces

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and the documentation requirements delineated in the Quality Assurance Procedures.

In 1979, I went to work for Pullman Power Products as a Field Inspector. My primary responsibility consisted of visual inspection for welding and bolting requirements to assure they complied with the relevant codes. I soon discovered that PPP performed these inspections and installations to their own Engineering Specifications - Diablo (ESD's), rather than in strict compliance with the AISC/ASTM codes. The ESD's I was expected to perform my inspections to were supposed to conform to the AISC/ASTM codes, when in actuality they often conflicted with them. This is especially important because the ESD's did not reference any requirements pertaining to the shape or size of the hole the anchor bolts were mounted in.

I identified the deficiencies of the ESD to my supervisor, , on several occasions. In each instance I was instructed to inspect to the ESD's because Pullman worked to them and not to codes.

discovered a structural support on the Unit 1 pipe rack where six of the eight mounting/bolting holes were elongated to the point where the washers could not cover the holes. researched supervisors, fellow inspectors (old timers), engineers, and the design drawings. The

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design drawing showed no enlongated holes. In all cases the personnel advised that:

1. Work was performed by another contractor;
2. Not to worry;
3. PG&E knew about it, it was old work and was accepted as is.

had to accept these statements as being gospel, mainly because there was insufficient documentation in existence and available to dispute their claims.

This type of problem was widespread throughout the plant. I had discovered similar situations in Unit 1 Reactor Building and Unit 2 Reactor Building. In some instances I found the crafts had stuffed the holes with short sections of soft tie-wire to serve as packing. I could not understand this practice. When I questioned what document provided the instructions for this practice, none could be provided. I consulted the pipefitters involved, my supervisor, PG&E inspectors and the engineers. Their reply was that "we had always done it this way, PG&E is aware of it and had accepted it as is."

To me, this constituted covering up poor workmanship by virtue of oral procedure or at best by internal memo rather than by approved procedures or AISC/ASTM codes.

My persistance in persuing these examples of non-

compliance with the codes led to my being harassed in the performance of my job. Finally, in 1981, I had the opportunity for other employment away from Diablo Canyon. I immediately seized it even with a reduction in pay. I was relieved to be removed away from the harassement and the butting of my head against a brick wall. However the problems I had identified continued to bother me.

A colleague who was providing testimony to the NRC contacted me on 4/9/84 and informed me of a plant tour being conducted by the NRC to identify problems that he was discussing with them. He requested me to accompany him to help identify the exact locations of some of these problems. I was happy to finally have the opportunity to show the NRC what I considered violations of codes, poor workmanship, and possible problems. I had heard about the notices posted on site saying the NRC wanted to be notified about these things and I felt they would be happy to have my co-operation so they could decide if these problems were real or not.

My colleague informed me that the NRC was attempting to provide confidentiality. We agreed to request our DCN logs and DCN's (Deficient Condition Notices) to assist in the location/problem identification.

I was later informed by Mr. Parks, that upon his learning of my wish to accompany the tour, he contacted D. Kirsch of the NRC at approximately 8 A.M. the morning of the tour to make the arrangements. According to Mr. Parks, Mr. Kirsch did not

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state that there would be a problem with my going and had agreed the night before with Mr. Devine (GAP Legal Director), to allow another witness to go.

At approximately 7:30 P.M. on 4/11/84, I arrived at the meeting site. Shortly after my arrival, Mr. Parks and Mr. Schollenberger of the NRC were discussing my accompanying the tour. The NRC's position was that they had not spoken with me, and it had not been agreed to the night before so they couldn't let me go. Mr. Parks informed me that other witnesses had decided to back out because of fears that their identities would be compromised. I agreed to go in their place. I wasn't afraid, I would have been with NRC personnel and because I didn't work there anymore, I could not have been retaliated against later. The NRC declined again and made no effort to even speak with me about my concerns. I was astonished and angered.

Mr. Parks advised me to prepare a brief statement outlining my concerns and that he would ensure the NRC would receive it.

On 4/12/84, I presented a two page handwritten statement to Mr. Parks while he was meeting with the NRC. In this statement I stressed that I wished to speak with the NRC Office of Investigations. He asked me if I could wait around for a while and he would see if he could get the NRC to talk with me. Approximately 45 minutes later, Mr. Parks introduced me to three

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members of the NRC Investigation Team.

I discussed the bolting problem and a concern I had with "hilti-slip"; a condition where the anchor bolt doesn't anchor itself properly.

After my testimony was over I was called aside by Mr. Murphy of OI. He counseled me, in a subtle fashion, to not have anything to do with Mr. Parks, the Government Accountability Project or the Mothers for Peace if I wished to not be branded as an anti-nuke.

I could not understand this. To me, the NRC appeared to be more concerned with politics than with the technical, quality and safety problems I had raised. I wasn't talking anti-anything. I was talking pro quality and pro safety and I expected them to be in tune with that.

I feel that had I been allowed to accompany the tour I could have provided first-hand examples of workmanship that would have violated the following code requirements from the Manual of Steel Construction (AISC), Specification for "Structural Joints Using ASTM A325 or A490 Bolts", Section 3 BOLTED PARTS , Section 5, INSTALLATION; Section 6, INSPECTION ; and COMMENTARY, Section C5 :

3 BOLTED PARTS

(a) The slope of surfaces of bolted parts in contact with the bolt head and nut shall not exceed 1:20

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with respect to a plane normal to the bolt axis. Bolted steel parts shall not be separated by gaskets and shall fit solidly together after the bolts are tightened. Holes may be punched, subpunched and reamed, or drilled, as required by the applicable code or specification. Standard holes shall have a diameter nominally 1/16-in. in excess of the nominal bolt diameter.

Where shown in the design drawings and at other locations approved by the designer, oversize, short slotted, and long slotted holes (see Table 7 in Commentary) may be used with high-strength bolts 5/8-in. diameter and longer in connections assembled as follows:

1. Oversize holes may have nominal diameters up to: 3/16-in. larger than bolts 7/8-in. and less in diameter, 1/4-in. larger than bolts 1-in. in diameter, and 5/16-in. larger than bolts 1 1/8-in. greater in diameter. They may be used in any or all plies of friction-type connections. Hardened washers shall be installed over oversize holes in an outer ply.

2. Short slotted holes are nominally 1/16-in. wider than the bolt diameter and have a length which does not exceed the oversize diameter provisions of subsection 3(a)1 by more than 1/16-in. They may be used in any or all plies of friction-type or bearing-type connections. The slots may be used without regard to direction of loading in friction-type connections but shall be normal to the direction of the load in bearing-type connections. Hardened washers shall be installed over short slotted holes in an outer ply.

3. Long slotted holes are nominally 1/16-in. wider than the bolt diameter and have a length more than allowed in subsection 3(a)2 but not more than 2½ times the bolt diameter. The slots may be used without regard to direction of loading in friction-type connections but shall be normal to the direction of the load in bearing-type connections.

Long slotted holes may be used in only one of the connected parts of either a friction-type or bearing-type connection at an individual faying surface.

Where long slotted holes are used on an outer ply, a plate washer or continuous bar of at least 5/16-in. thickness with standard holes shall be provided. This washer or bar shall be of structural grade material, but need not be hardened. If hardened washers are re-

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quired to satisfy Specification provisions, the hardened washers shall be placed over the outer surface of the plate washer or bar. These washers or bars shall have a size sufficient to completely cover the slot after installation.

(b) When assembled, all joint surfaces, including those adjacent to the bolt heads, nuts or washers, shall be free of burrs, dirt, and other foreign material that would prevent solid seating of the parts. Paint is permitted unconditionally in bearing-type connections.

5 INSTALLATION

(c) Turn-of-Nut Tightening

When the turn-of-nut method is used to provide the bolt tension specified in subsection 5(a), there shall first be enough bolts brought to a "snug tight" condition to insure that the parts of the joint are brought into good contact with each other. Snug tight is defined as the tightness attained by a few impacts of an impact wrench or the full effort of a man using an ordinary spud wrench. Following this initial operation, bolts shall be placed in any remaining holes in the connection and brought to snug tightness. All bolts in the connection shall then be tightened additionally by the applicable amount of nut rotation specified in Table 4, with tightening progressing systematically from the most rigid part of the joint to its free edges. During this operation there shall be no rotation of the part not turned by the wrench.

(d) Calibrated Wrench Tightening

When calibrated wrenches are used, they should be set to provide a tension at least 5% in excess of the minimum bolt tension specified in subsection 5(a). The wrenches shall be calibrated at least once each working day for each bolt diameter being installed. Wrenches shall be recalibrated when significant changes are made in the equipment or when a significant difference is noted in the surface condition of the bolts, nuts, or washers. Calibration shall be accomplished by tightening, in a device capable of indicating actual bolt tension, three typical bolts of each diameter from the bolts being installed.

When adjusting the wrenches to provide the required tension, it shall be verified during actual installation in the assembled steelwork that the calibration selected

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does not produce a nut or bolt head rotation from snug tight greater than that permitted in Table 4. If manual torque wrenches are used, nuts shall be in tightening motion when torque is measured.

When using calibrated wrenches to install several bolts in a single connection, the wrench shall be returned to "touch up" bolts previously tightened, which may have been loosened by the tightening of subsequent bolts, until all are tightened to the prescribed amount.

(f) Reuse

A490 bolts and galvanized A325 bolts shall not be reused. Other A325 bolts may be reused if approved by the engineer responsible.

Retightening previously tightened bolts which may have been loosened by the tightening of adjacent bolts shall not be considered as a reuse.

6 INSPECTION

(a) The Inspector shall determine that the requirements of Sections 2,3, and 5 of this Specification are met in the work. When the calibrated wrench method of tightening is used, the Inspector shall have full opportunity to witness the calibration test prescribed in subsection 5(d).

(b) The Inspector shall observe the installation of bolts to determine that the selected procedure is properly used and shall determine that all bolts are tightened. Bolts installed by the turn-of-nut method may reach tensions substantially above the value given in Table 3, but this shall not be cause for rejection.

COMMENTARY C5 INSTALLATION

Where long slotted holes are used, experimental evidence has shown that a plate washer or continuous bar of at least 5/16-in. thickness with standard holes is necessary to provide adequate bearing. This washer or bar shall be of structural grade material but need not be hardened. However, if hardened washers are required to satisfy Specification provisions, the hardened washer shall be placed over the outer surface of the plate washer or bar.

The examples I could have identified to the NRC on the plant tour would have been Code violations with respect to:

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1. Design drawings not specifying elongated holes;
2. Hole sizes outside of Code specifications;
3. Torquing method;
4. Bolt reuse;
5. Examples of "packing" violating foreign material specifications.

I am extremely concerned with the NRC's performance and their apparent desire to be more concerned with formality and politics than to concern themselves with the problems I wished to discuss. Code violations existed in that plant up until I left. I believe they still exist. My utmost concern is having these problems addressed and resolved. I am convinced that I can identify these problems "in the field" easier than I can discuss them.

I am willing to co-operate with any good faith investigation of the Code violations at the Diablo Canyon Plant.

I have read the above 10-page document and certify that it is accurate and true to the best of my knowledge.

Subscribed and sworn to before me this 18th day of April, 1984.

Notary Public

My commission expires January 9, 1987

NOTARY PUBLIC
CALIFORNIA
Santa Barbara County
My Commission Expires Jan. 9, 1987

-EXHIBIT 4 SL

A F F I D A V I T

My name is Steven Lockert. I am making this statement, without threats or inducements, to Richard Parks who is a Volunteer investigating the Diablo Canyon Nuclear Power Plant.

I have reason to believe that the Bolting Program for Rupture Restraints in Units 1 and 2, conducted during late July to December of 1983, by the Pullman Power Product Corporation has failed to meet licensing requirements. I use the word "licensing" because the "Corrective Action" part of the Final Safety Analysis Report (FSAR) has not functioned as reported per 17.1.16 paragraph of the FSAR, "The Quality Assurance Program requires that conditions jeopardizing qualitybe promptly referred to responsible parties and that appropriate steps be taken to correct such situations."

A discussion of the Bolting Program for Rupture Restraints as practiced by Pullman is best discussed through Pullman D.R. 4342, PG&E Nonconformance Report DC2-80-RM-002, and my own inspection experience dated late-July to mid-December of 1983. PG&E required that Pullman adhere to Contract Specification 8833XR for structural steel erection (contract includes Pullman's Rupture Restraint Program). 8833XR specifically states that structural steel erection be conducted to the AISC Steel Construction Manual, Seventh Edition.

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AISC's specifications for structural joints using ASTM A-325 and A-490 High Strength Bolts has provided values for minimum fastener tension in Table 3, page 5-195. Basically, this Table requires that all A-325 and A-490 H.S. bolts be tightened to 70% of their tensile strength measured in tension. When turn-of-nut tightening is used the additional requirements of Table 4, page 5-196, are specified. Note that the turn-of-nut rotation is dependent on:

1) Disposition of outer faces of bolted parts.

2) Bolt length.

Additionally, thread pitch should be mentioned because it is a factor in the determination of the required turn-of-nut rotation to achieve the specified tensile bolt preload.

Pullman's ESD 243-1983 Torque Instructions per Charts A, A1, B and Field Process Sheets prepared by Pullman Field Engineers, simply, do not take into account the pre-requisites of the AISC Manual. Non Conformance Report DC2-80-RM-002 initiated by Robert Torstrom on 11/19/80 amd dated 12/12/80 for Corrective Action states:

SHEET 1: Cause of Non Conformance:

Pullman Power Products' Rupture Restraint Program has had inadequate design change control, inspection performance, and control.

SHEET 2: Description:

1) a. Out of tolerance gaps behind base plates... nuts not engaged per requirements.

b. ...There are cases of material and welds not conforming to the specification.

2) a. Welds exist which do not have documentation.

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b. Modifications have been performed... and have not been documented.

c. There are bolts that have 'Torque Seal' ...However, inspection records do not exist....

RESOLUTION:

Pullman Power Products shall perform a documented inspection of all bolted and welded connections and applicable documentation, required by the Specification, as set forth in approved contractor's ESD's in order to:

1) Identify connections which do not conform to specification requirements, and

2) Identify connections which do not have required documentation.

I would first like to point out that the cause of the NCR indicated a complete breakdown of Quality Assurance with respect to Pullman's Rupture Restraint Program meeting 8833XR Specification requirements. Of course, Mr. Torstrom did not use those exact words but one only has to look at the resolution of the NCR to see that PG&E required Pullman to do a 100% reinspection of "all bolted and welded connections and applicable documentation" required by Specification 8833XR.

Second, I would like to point out that Mr. Torstrom refers to the non-conforming conditions as Deficient Conditions; I do not feel deficient is the correct word. A departure from the requirements of 8833XR (a Procurement Document) is a "Deviation" defined by 10CFR21.3(e).

The deviations occured in work that had already been accepted by Pullman's Quality Assurance people as meeting the De-

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sign Drawings and 8833XR Specifications. Already being QA/QC accepted, the Rupture Restraints with deviations included were being offered to PG&E as an acceptable installation by Pullman. The deviations can now be spoken of as "Defects" per the 10CFR 21.3(d) definition. It should be pointed out that the defects were not reported per 10CFR21.21.

Now lets discuss the Resolution and Corrective Action in Torstrom's NCR of 12/12/80. Proper resolution required an identification of "all bolted and welded connections" which did not conform to 8833XR Specification requirements. Further, it was stated that:

> Pullman Power Products has developed and implemented a program which assures adequate control of design change. Training and indoctrination programs have been developed and implemented which assures adequate performance of inspection personnel.

Attachment 1 of NCR DC2-80-RM-002 correctly shows that the minimum tension for High Strength bolts (ASTM A-325 and A-490) is 70% of the minimum tensile strength. However, Anchor bolts used as "Through bolts" in concrete walls and floors and Anchor bolts cast in concrete are allowed to be tensioned to 55% and 25% of the minimum tensile strength, respectively. If the Anchor bolts happen to be A-235 or A-490 bolts, which I know for a fact that many of them are, then the instructions of the NCR are an apparent deviation from the requirements of the AISC Manual, paragraph 1.23.5, Table 1.23.5. In other words, the resolution of the bolting problem was resolved by instructions to deviate from the requirements of the AISC Manual.

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I do not know if NCR DC2-80-RM-002 had been closed by the time I was employed by Pullman (July of '83). I do know that I was not instructed in the resolution requirements of the NCR and that Pullman did not report defects that still existed in Rupture Restraints from July to December of 1983. Defects that I had noted that had not been previously reported were:

1. Unauthorized modifications to fillet welds that encroached on bolt or washer land areas.

2. Oversize holes already QC accepted outside the tolerances of ESD 243 and AISC Manual.

3. Oversize holes in base plates packed with steel rods and wires without the benefit of an approved Pullman procedure. (This work was performed to a memo from Mr. Torstrom in violation of 10CFR50 App B, Criteria V and VI.)

4. Oversize welds beyond that allowed by AWS D1.1 and beyond that allowed by Pullman's ESD 243.

5. Defects in A-490 bolts had been found after the bolts had been "dedicated" by Pullman's QA Receiving Department and sent to the field for installation.

(In addition to the above mentioned hardware problems, Pullman's ESD 243 of late 1983 had procedure problems written

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into the Rupture Restraint Program:)

6. The tables provided for the description of acceptable washers had not been updated per the requirements of AISC, Sec 5, Page 191, para. 2(a).

7. Acceptance criteria for High Strength bolts was not defined in ESD 243. Filed Inspectors did not know, nor were they legally able to reject bolts that were defective per ASTM A-490, ASTM A-325, and ANSI B18.2 requirements.

8. Bolt Torque Tables in ESD 243 were still out of compliance with AISC Manual requirements as late as December '83. Discussions with Pullman Feild Engineers Dale Warren and Larry Werner indicated that although the tables had been recently updated, they still do not meet AISC Manual requirements.

(Finally, the Quality Assurance Program did not function as reported in the FSAR and as defined in 10CFR50, App B. Examples provided are:)

9. Pullman Power Products did not develop nor implement a program to control design changes.

a) Design Drawings did not reflect unauthorized modifications to fillet welds because no As-Built Drawing was generated by Engineering when they were

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notified of such modifications.

b) Field Engineer Dale Warren issued the proper Washer Criteria for myself without notification or acceptance by Pullman or PG&E QA Departments. QA/QC Manager Harold Karner, when notified of out of date Washer Criteria in ESD 243, did not issue a Non Conformance Report nor update the present ESD 243.

c) Pullman did not have the proper Torque Tables in effect three years after the writing of NCR DC2-80-RM-002.

10. Pullman did not train nor indoctrinate inspectors to the requirements of the AISC Manual for Bolting. (Accidental reinspection of work accepted in late '82 or early '83 revealed hole sizes outside the tolerances of the AISC Manual.)

11. Defects in bolts were not reported per a NCR. I was unable to report the defects I had found in A-490 bolts because I was not allowed to consult the procurement documents needed to properly generate such a report. Pullman Supervisor, Russ Nolle specifically prevented me from referencing these documents by saying that I was out of my area. (See Oct. 17 indicent of Lockert Letter addressed to Mark Padovan, USNRC dated 1/2/84.)

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CONCLUSION:

** Diablo Canyon has had a long-standing bolting problem as evidenced by NCR DC2-80-RM-002.

** The Corrective Action that was stated to be taken as of 12/12/80 did not occur.

** If DC2-80-RM-002 was still open during my time as an Inspector, I was not informed of its requirements nor was I performing duties required for its resolution.

** Deviations to 8833XR requirements did not get reported in the Corrective Action.

** Rupture Restraint bolting is still defective.

I have read the above 8-page statement and it is true and correct to the best of my knowledge and belief.

Steven Lockert

Steven Lockert

Subscribed and sworn to before me

this 26th day of April 1984.

Lisa R. Wenter

Notary Public

My Commission expires 5/9/86

OFFICIAL SEAL
LISA R. WENTER
NOTARY PUBLIC - CALIFORNIA
SAN LUIS OBISPO COUNTY
My Comm. Expires May 9, 1986

[redacted]

- EXHIBIT 5

RP

My name is Richard D. Parks. I am submitting this affadavit to document the discrepant conditions identified, and corresponding violations of the applicable codes as a result of the plant tour conducted on April 11, 1984 with D. Kirsch and G. Hernandez of Region V, United States Nuclear Regulatory Commission (NRC) at the Diablo Canyon Nuclear Power Plant. I and three witnesses accompanied the NRC to provide "hands-on" examples of non-compliance with regulations, specifications and codes that form the basic cornerstone of a comprehensive Quality Assurance/Quality Control program.

Each example identified to the NRC was subsequently "tagged" for identification and a "report sheet" was filled out by the NRC. The "problem description" is a quote from the report sheet. The examples identified that violated applicable codes are discussed as follows:

<u>ITEM #1, Tag #2</u>: Elevation 116, Unit 1 Reactor Building. Line Designation NO.S2-254-10, in the area of Pressurizer and Reactor Coolant Pump 1-2.

<u>Problem Description:</u> Weld attaching Safety Injection Accumulator line to nozzle of the cold leg line (NO.S2-254-10). On the side facing Reactor Coolant Pump (RCP) is a grinding gouge in the pipe at the pipe-weld interface approximately 3/8 inches long, 1/8 inch at

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widest point and 1/16 inch deep (dimensions as visually determined by NRC Inspector - no measurements taken). Additionally, there appears to be a slight amount of undercut at two locations. The undercut is approximately 5/8 inches on the weld side facing the RCP and approximately 1 inch at 120° from the side away from the RCP.

Code Violation: American Society of Mechanical Engineers (ASME) Section III, "Rules for Construction of Nuclear Power Plant Components - 1977 edition, Division I General Requirements, Subsection NB, "Class 1 Components", para NB-4424 "Surfaces of Welds".

"As-welded surfaces are permitted, and for piping the appropriate stress indices given in Table NB-3683.2-1 shall be applied. However, the surface of welds shall be sufficiently free from coarse ripples, grooves, overlaps, and abrupt ridges and valleys to meet (a) through (f) below:

(a)...

(b)...

(c)Undercuts shall not exceed 1/32 inch (0.8mm) and shall not encroach on the required section thickness.

(d)...

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(e)...

(f)If the surface of the weld requires grinding to meet the above criteria, care shall be taken to avoid reducing the weld or base metal below the required thickness."

The discrepant condition identified by the witness violates the code requirements with respect to being "free from coarse ripples, grooves, overlaps, and abrupt ridges and valleys to meet (c) and (f)."

ITEM #2, Tag #4: Unit 2 Reactor Building, Elevation 115, Support 97-3R in vicinity of RCP 2-3.

Problem Description: "Excessive overweld has caused excessive shrinkage of SS line. This was supposed to be a full penetration weld with fillet cap and is as specified. The overwelding can damage the pipe because calculations don't account for residual stresses caused by such overwelding."

Code Violation: United States of America Standard (USAS) B31.7-1969 "Code for Pressure Piping - Nuclear Power Piping" (note: this standard now is known as ANSI-B31.7), foreword "FABRICATION REQUIREMENTS AND THEIR CORRELATION WITH DESIGN", page XVI paragraph 5. "Even hanger attachment details are covered. For Class 1 piping, complete penetration welds are required. The designer must consider all stresses in the attachment as well as their effect on the pressure

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retaining part."

The welds in question do not conform to the stated intent of the "Nuclear Power Piping" code with respect to the residual stresses induced by the overwelding. It is the concern of this particular anonymous witness that these residual stresses should have been but were not a factor in the design calculations.

ITEM #3, Tag #5: Unit 2 Reactor Building, large restraint wall attachment (around surge line), beneath Unit 2 Pressurizer.

Problem Description: "Shopwelding is supposed to conform to AWS D1.1 standards. The inner welds are excessively rough and of such a profile that they would not conform to AWS D1.1. The welds are ragged."

Code Violation: American Welding Society (AWS) Structural Welding Code - Steel, paragraph 8.15 "Quality of Welds", subparagraph 8.15.1 "Visual Inspection". "All welds shall be visually inspected. A weld shall be acceptable by visual inspection if it shows that

8.15.1.1 -The weld has no cracks

* 8.15.1.2 Thorough fusion exists between adjacent layers of weld metal and between weld metal and base metal

8.15.1.3 All craters are filled to the full cross section of the weld

* 8.15.1.4 Weld profiles are in accordance with (para.) 3.6 [weld profile]"

The weld in question does not conform to the requirements specified in paragraph 3.6 [weld profiles] or the evident thorough fusion requirements as stated in 8.15.1.2.

ITEM #4, Tag #6: Unit 2 Auxiliary Building, area GW, elevation 115, line No. 2-S2-265-8 (Containment Spray Discharge Pipe - 4 lug attachments between S and T line.)

Item Description: "Lug attachments are called out to be 1/2 inch fillet welds on three sides. Actual size is 7/16 inch fillet or less."

Problem Description: "Actual size is alleged to be less than or equal to 7/16 inch which is 1/16 inch less than required. The excessive welding used in the design of the lugs attachment welds, when welded to Schedule 10 stainless thin wall pipe, has caused excessive shrinkage. The excessive shrinkage causes residual stresses in the pipe which has not been accounted for in the design or stress analysis. The position of the clamp is such that there is a torsional force applied to

the lugs, because the clamp cannot contact the wall of the pipe due to the shrinkage. This torsional force is not accounted for in the design and compromises the pipe integrity."

Code Violation: Refer to "Code Violation" discussion in "ITEM #2, Tag #4".

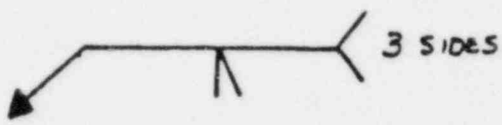
The welds in question do not conform to the stated intent of the "Nuclear Power Piping" code with respect to the residual stresses induced by the welding or the torsional force applied to the lugs due to excessive shrinkage. It is the concern of this particular anonymous witness that these stresses should have been but were not a factor in the design calculations.

ITEM #5, Tag #7: Unit 2, Auxiliary Building, Area 2H, support 413-131R around CCW line.

Problem Description: "Eight lug attachment welds are required to be full penetration welds on three sides. Actual weld is not a full penetration weld, but is, instead a fillet weld, contrary to the design."

Code Violation: American Welding Society (AWS) - A2.4 - 79 "Symbols for Welding and Non-Destructive Testing," paragraph 9.0 "Groove Welds," subparagraph 9.2.2 "Complete Joint Penetration Required." "When no depth of groove preparation or effective

throat is shown on the welding symbol for single-groove and symmetrical double-groove welds, complete joint penetration is required."



Symbol provided on "Detail" for weld(s) in question.

PG and E has stated in their letter, DCL-84-040, "The weld symbols used at Diablo Canyon are consistent with the standards specified in AWS..." and in an Interoffice Memorandum (file no. 930, 146.20, CA2) dated October 25, 1983 that "all pipe support as-builts issued by General Construction after October 15, 1983 should have all weld symbols in conformance with AWS A2.4."

The welds in question were incorrectly performed because of lack of proper interpretation of the weld symbol utilized on the design drawing. It is the concern of this particular anonymous witness that this discrepancy provided an example of code compliance violation due to a lack of intimate knowledge with AWS A2.4. These particular welds had been inspected and accepted by Pullman Quality Control and PG and E Quality Control prior to the discrepancy being identified by a Pre-Inspection Engineer.

I have read the above eight page statement. I have based the information contained therein either on personal knowledge or by reviewing the relevant information with the particular witness involved. This statement is true, correct and complete to the best of my knowledge and belief.

I declare under penalty of perjury that the foregoing is true and correct, and that the same was executed this 17th day of April, 1984 at San Luis Obispo, California.

Richard D. Parks

RICHARD D. PARKS, Declarant

STATE OF CALIFORNIA)
) ss.
COUNTY OF SAN LUIS OBISPO)

On April 17, 1984, before me, the undersigned, a Notary Public in for said State, personally appeared RICHARD D. PARKS, personally known to me and proved to me on the basis of satisfactory evidence to be the person whose name is subscribed to the within instrument, and that he executed the same.

WITNESS my hand and official seal.

OFFICIAL SEAL
LISA R. WENTER
NOTARY PUBLIC - CALIFORNIA
SAN LUIS OBISPO COUNTY
My Comm. Expires May 9, 1986

Lisa R. Wenter

Notary Public in and for
said County and State

EXHIBIT 6

RP

AFFIDAVIT

My name is Richard J. Parks. I am submitting this affidavit to expand the issue of an ASME III code violation identified in my previous affidavit (Dated 4/17/84). On page 1 of the 4/17/84 statement, under item 1, Tag #2, a weld (RC-2-16) was identified on the Safety Injection Accumulator line as having a grinding gouge and was undercut. This condition is in violation of ASME III para NB-4424 "Surfaces of Welds", with respect to the undercut and the "grinding gouge." The NRC Inspector, J. Kirsch, and I, along with the witness verified the visual deficiencies of the weld. Close inspection could not accurately determine how deep the gouge was. Thus from a visual inspection and limited capability to accurately measure the wall thickness remaining in the pipe, it was impossible to determine whether the minimum wall thickness requirements had been violated. This is significant for the following reasons:

1) The NRC knew the potential for this deficient condition existed as early as May 1983, and this potential problem was of major consequence. The following documents evidence the the NRC knowledge and concerns related to this type of discrepancy.

a. BOARD NOTIFICATION LETTER NO. 83-83: NOTICE OF VIOLATION CONCERNING REPORTING REQUIREMENTS FOR DIABLO CANYON (ATTACHMENT 1).

Appendix A (Notice of Violation) of Attachment 1 cites the licensee for failing to notify the NRC promptly when a problem of this nature was

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identified in the plant. The failure of the licensee to notify promptly was in violation of their Technical Specifications.

IE Report No. 83-20 (appended to Attachment 1) states on page 1, under "Licensee Action on Previous Inspection Findings", "On May 3, 1983 the licensee submitted LER No. 83-004 concerning the discovery of gouges and marks in the reactor coolant system piping. ...The licensee has completed an ASME Section XI repair plan which has been submitted to their engineering department for review and approval. The licensee has determined that the gouges can be removed by grinding and blending without violation of minimum pipe wall thickness criteria."

Page 3 of I&E Report 83-20 (Attachment 1) lists the chronology of events related to this type of weld deficiency under Section 5, "Violation of Minimum Wall Thickness on Reactor Coolant System Piping". The 2/16/83, 3/28/83, 4/1/83 and 5/9/83 entries pertaining to examinations of this problem confirm that the condition of weld RC-2-17 and piping adjacent to weld RC-2-17 violate minimum wall thickness requirements.

Page 5, first paragraph (Attachment 1) of the I&E Report summarizes the significance of the concern over these items of code violation the "Discrepancy ... which require resolution ... and ... if left uncorrected could have resulted in degradation or loss of integrity of the Reactor Coolant Pressure boundary".

b. Board Notification Letter No. 83-89: INFORMATION ITEM REGARDING ADDITIONAL POTENTIAL VIOLATION OF MINIMUM WALL THICKNESS OF REACTOR COOLANT SYSTEM PIPING AT DIABLO CANYON, UNIT 1. (Attach-

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ment 2)

This letter includes NRC Preliminary Notification-PNO-V-83-22A dated 6/23/83. The body of the PNO states that "On June 22, 1983, licensee personnel determined that the RCS piping in areas of welds other than reported on May 10, 1983 may be less than the minimum wall thickness specified by Code. A total of ten welds ... appear to be less than required minimum wall thickness.... All four RCS loops are involved."

The significance of this document is simply that the original problem thought to exist on one to three welds (per Attachment 1) has now expanded to include a possible total of 10 welds that are questionable.

c. <u>Board Notification Letter No. 83-96</u>: INFORMATION ITEM REGARDING ADDITIONAL POTENTIAL VIOLATION OF MINIMUM WALL THICKNESS OF REACTOR COOLANT SYSTEM PIPING AT DIABLO CANYON, UNIT 1 (Attachment 3).

Included in this Board notification letter is PG&E response "Report on Investigation of Reactor Coolant Pipe Weld Thickness at Diablo Canyon", dated 7/1/83. Page 1-1 of this report states in part, "performed ... a series of measurements across the weld and adjacent base metal for 31 of the 56 girth welds on the reactor coolant system piping of Diablo Canyon unit 1.... While the inside surface can be directly observed during PSI

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work, post operational access to these surfaces is not feasable due to high radiation levels. While it is possible to obtain this profile information post-operationally on an as-needed basis, this work would also involve additional personnel radiation exposure. These factors provided the incentive to identify geometric reflectors before the plant started up."

Page 1-2 states in part, "During the course of the ensuing UT investigation, it was discovered that there might be below minimum thickness areas on nine other welds in addition to 2-17."

Page 2-2 states in part that, "Analysis of Ultra-sonic (UT) thickness measurement capabilities and UT data and records show that UT measurements lack the precision required to verify the original micrometer wall thickness measurements on Reactor Coolant Piping."

Page 3-2 states in part that, "After design was complete and fabrication started, ASME Section III was expanded in scope to cover piping for Nuclear Power Plants. PG&E Specification 8752, issued for installation of the Nuclear Steam Supply Systems, was revised in March 1974 to incorporate requirements of ASME Section III, 1971 edition."

Page 5-1 states in part that, "After an assessment of alternative methods of measurement, it was concluded that the direct mechanical method using micrometers was the most accurate and valid method of measuring piping of this size and material..."

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The significance of this letter is:

(1) The licensee realizes that any deviations from the code (ASME III) would require corrective measures prior to criticality.

(2) ASME III was the code to inspect to and comply with. This is evident by the PG&E Specification (8752) revision requiring compliance with this code.

(3) UT determination is <u>not</u> an accurate method of detecting the deficiency, the more accurate method is "hands-on mechanical measurement."

(4) Of the 64 welds of this type in the Reactor Coolant System piping, the PNO figure left approximately 16% questionable. However only 56 of the 64 welds were considered, and only 31 were rechecked. Ten of the 31 welds were suspect (approximately 32%). With these initial results the reinspection program should have been expanded to include all welds on the Reactor Coolant System. It wasn't. In my opinion a 68% confidence level for the piping and welds in the Reactor Coolant System is <u>unacceptable</u>. Even if UT wasn't the most accurate technique, any similar indications should have been identified for mechanical measurement.

d. <u>Board Notification Letter No. 83-124</u>: NRC REGION V INSPECTION REPORT 50-275/83-26 RELATING TO APPARENT LESS THAN MINIMUM PIPING WALL THICKNESS. (Attachment 4)

(next)

RP

As stated in the Board Notification Letter - I&E Report #83-26 reviewed and investigated the licensee's corrective action program relating to the 10 identified weld deficiencies. Page 8 of the subject I&E report states in part, "For the above reasons the inspector considers that the licensee had inappropriately placed a high degree of reliance of the RCS thickness measurements obtained by the Ultrasonic nondestructive testing methods utilized in the identification and verification of the potential deviations from specified minimum wall thickness criteria."

The balance of the subject I&E report is devoted to the thoroughness of the I&E team in reviewing the documentation and testing of the 10 welds identified by the licensee. The report concluded that, "no items of noncompliance or deviations were identified."

The significance of this report is that:

(1) The NRC apparently concurs with the licensee that UT was an unreliable method for determining wall thickness minimums.

(2) The NRC could find "no problems" with the 10 welds identified and any related corrective activities.

2) <u>A thorough review of the documents available for public review does not identify, in any fashion, where weld RC-2-16 had been previously evaluated by the licensee or the NRC</u>. This weld (RC-2-16) was personally identified to the NRC on the plant walkdown on 4/11/84. This anonymous witness had submitted an affadavit ex-

(cont)

pressing his concern on this issue on 3/13/84. In consideration of the previous attention given to welds of this type (i.e.; possible wall thickness violations) by the NRC, it cannot be readily determined why the NRC Staff chose to recommend low power licensing on 4/12/84 without the same exhaustive requirements being implemented to determine if this gouge violated Reactor Coolant System Pressure Boundary Wall Thickness requirements. It was not possible to visually determine if a violation existed. The piping was inaccessible to take internal diameter (ID) measurements, the past "acceptable practice". In fact the piping would have been required to be cut open to perform this measurement. The previously accepted NRC position was that UT was unreliable and to be avoided. Yet approximately 36 hours after the plant tour was completed the NRC Staff testified to the Commission that the tour "identified no code violations", and the low power license was granted.

This pipe gouge should not be taken lightly for the following reasons:

1) .. On the April 11 plant tour, Inspector Kirsch informed me that he was not previously aware of the condition of weld RC-2-16 and adjacent piping. In other words, the combined licensee-NRC corrective action program had missed this problem which existed in the vicinity of the Reactor Vessel. How many similar items have been missed? We won't know without a comprehensive reinspection.

2) .. During the Spring of 1983 when the anonymous witness raised this issue on the job he was advised to "forget about it".

(next)

3) .. This weld connects one of the "Safety Injection Accumulators (core flood tanks) to one of the Reactor Coolant Cold leg(s). This tank is required to be available during a "loss of coolant accident" to inject borated water into the core to ensure it is and remains in a safe, shutdown condition during an accident. This is because the borated water prevents the fission process by absorbing the nuetrons required for fission. A failure of this line could "prevent an Engineered Safeguards Actuation System from performing its design function (maintaining the core shutdown)." A failure of this system would also violate the Accident Analysis of the Final Safety Analysis Review of the Plant, and every avenue should be pursued to assure this system boundary has not been violated.

I have read the above 8-page statement and swear the information contained therein is based on personal knowledge or review of the referenced documents. This statement is true, correct and complete to the best of my knowledge and belief.

Richard D. Parks

Richard D. Parks

Subscribed and sworn to before me this 30th day of April, 1984.

Sandy Carsel

Notary Public in and for
the County of San Luis Obispo,
State of California

SANDY CARSEL
NOTARY PUBLIC - CALIFORNIA
PRINCIPAL OFFICE IN
SAN LUIS OBISPO COUNTY
My Commission Expires Sept. 14, 1984

ATTACHMENT 1

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

JUN 24 1983

Docket No: 50-275

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

FROM: Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: NOTICE OF VIOLATION CONCERNING REPORTING REQUIREMENTS FOR DIABLO CANYON (Board Notification No. 83-83)

According to the procedures for Board Notification, the enclosed information is being transmitted directly to the Commission. The Boards and Parties are being notified by copy of this memorandum.

This information concerns a violation of NRC reporting requirements. The subject of the notice of violation was previously reported to you and was the subject of Board Notification 83-72.

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

cc w/encl:
See Next Page

8305110492

Contact:
B. Buckley, NRR
Ext. 28379

ATTACHMENT 1

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

JUN 17 1983

Docket No. 50-275

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

Attention: Mr. J. O. Schuyler, Vice President
Nuclear Power Generation

Gentlemen:

Subject: NRC Inspection of Diablo Canyon Unit No. 1

This refers to the routine inspection, conducted by Messrs. G. H. Hernandez and M. M. Mendonca of this office on May 23-June 6, 1983, of activities authorized by NRC License No. DPR-76 and to the discussion of our findings with Mr. R. C. Thornberry and other members of the Pacific Gas and Electric Company staff at the conclusion of the inspection.

Areas examined during this inspection are described in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspectors.

Based on the results of this inspection, it appears that one of your activities was not conducted in full compliance with NRC requirements, as set forth in the Notice of Violation, enclosed herewith as Appendix A.

Your response to this Notice is to be submitted in accordance with the provisions of 10 CFR 2.201 as stated in Appendix A, Notice of Violation.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosures will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

ATTACHMENT 1

Should you have any questions concerning this inspection, we will be glad to discuss them with you.

The responses directed by this letter and the 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,

D. F. Kirsch

for T. W. Bishop, Chief
Reactor Project Branch No. 2

Enclosures:
A. Notice of Violation
B. Inspection Report
Nos. 50-275/83-20

cc w/enclosure:
G. A. Maneatis, PG&E
P. A. Crane, PG&E
S. D. Skidmore, PG&E
R. C. Thornberry, PG&E (Diablo Canyon)
R. D. Etzler, PG&E (Diablo Canyon)

ATTACHMENT 1

APPENDIX A

NOTICE OF VIOLATION

Pacific Gas and Electric Company
P. O. Box 7442
San Francisco, California 94120

Docket No. 50-275
Licensee No. DPR-76

As a result of the inspection conducted on May 23-June 3, 1983, and in accordance with NRC Enforcement Policy, 10 CFR Part 2, Appendix C, the following violation was identified:

Technical Specification 6.9.1.11 states, in part, that "The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC..."

Technical Specification 6.9.1.12 lists the types of events which shall be reported by telephone within 24 hours to the Director of the Regional Office, or his designate, and confirmed by telegram, mailgram or facsimile no later than the first working day following the event, with a written followup report within 14 days.

Technical Specification 6.9.1.12.i describes a type of event which shall be reported pursuant to Technical Specification 6.9.1.12 and states as follows:

"Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition."

Contrary to the above requirements, on December 17, 1982 the licensee identified that certain areas of Weld No. WIB-RC-2-17 were less than the minimum wall thickness specified by design and the applicable codes. Weld No. WIB-RC-2-17 is in Loop No. 2 of the Reactor Coolant System. This condition was not reported to the NRC until May 10, 1983.

This is a Severity Level IV Violation (Supplement I).

ATTACHMENT 1

Purusant to the provisions of 10 CFR 2.201, Pacific Gas and Electric Company is hereby required to submit to this office within thirty days of the date of this notice, a written statement or explanation in reply, including: (1) the corrective steps which have been taken and the results achieved; (2) corrective steps which will be taken to avoid further items of noncompliance; and (3) the date when full compliance will be achieved. Consideration may be given to extending your response time for good cause shown.

JUN 17 1983
Date

D F Kirsch
D. F. Kirsch, Chief
Reactor Projects Section No. 3

ATTACHMENT 1

U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos. 50-275/83-20

Docket Nos. 50-275 License No. DPR-76

Licensee: Pacific Gas and Electric Company

77 Beale Street, Room 1435

San Francisco, California 94106

Facility Name: Diablo Canyon Unit No. 1

Inspection at: Diablo Canyon Site, San Luis Obispo County, California

Inspection conducted: May 23-June 6, 1983

Inspectors: [signature] 6/17/83
G. H. Hernandez, Reactor Inspector Dated

[signature] 6/17/83
M. M. Mendonca, Resident Inspector Dated

Approved by: [signature] 6/17/83
D. F. Kirsch, Chief, Reactor Projects Section No. 3 Dated

Summary:

Inspection during the period of May 23-June 6, 1983 (Report No. 50-275/83-20

Areas Inspected: Unannounced inspection by a regional and resident inspector of actions related to recent licensee identified items including indications of less than minimum wall thickness in certain areas of a Reactor Coolant Piping Loop weld; a leaking shop weld in the Component Cooling Water System; repair activities related to gouge and grinding marks in one area of Reactor Coolant Piping Loop No. 1-3; and resolution of Unit 1 Reactor Vessel preservice inspection indications.

The inspection involved 67 inspection-hours by two NRC inspectors.

Results: Of the areas examined, one item of noncompliance was identified in the failure of the licensee to promptly notify the NRC of a potentially reportable condition in accordance with the technical specifications (paragraph 5).

ATTACHMENT 1

DETAILS

1. Individuals Contacted

a. Pacific Gas and Electric Company (PG&E)

*W. A. Raymond, Assistant Manager for Nuclear Plant Operations
+*R. C. Thornberry, Plant Manager
+*R. Patterson, Plant Superintendent
*R. D. Etzler, Project Superintendent, General Construction
+*W. B. Kaefer, Technical Assistant to Plant Manager
~~*J. A. Sexton, Supervisor of Operations~~
*D. B. Miklush, Supervisor of Maintenance
*J. V. Boots, Supervisor of Chemistry and Radiation Protection
*R. G. Tadaro, Security Supervisor
+*R. T. Twiddy, Supervisor of Quality Assurance
*D. R. Bell, Quality Control Engineer, General Construction
*J. M. Gisclon, Power Plant Engineer
*E. M. Conway, Personnel and General Services Supervisor
*M. N. Norem, Lead Startup Engineer
*R. M. Luckett, Regulatory Compliance Engineer
T. D. Smith, Senior Quality Control Engineer
D. A. Gonzales, Quality Control Inspector

b. Bechtel Corporation (Bechtel)

+*J. W. Shryock, Site Completion Manager
D. O. Henery, NDE Level III Engineer

c. Pullman Power Products Corporation (Pullman)

H. W. Karner, Quality Assurance/Quality Control Manager

+Denotes personnel attending the exit management meeting of May 27, 1983.

*Denotes personnel attending the exit management meeting of June 3, 1983.

2. Licensee Action on Previous Inspection Findings

(Open) Followup Item (50-275/83-17/05): Degradation of the Reactor Coolant System Pressure Boundary

On May 3, 1983, the licensee submitted LER No. 83-004 concerning the discovery of gouges and marks in the reactor coolant system piping. The marks and gouge were discovered on the discharge side of the No. 3 Reactor Coolant Pump. Of the four blemishes discovered the most severe is a gouge approximately 0.150 inches deep by 2.0 inches long and 0.150 inches wide. The licensee has completed an ASME Section XI repair plan which has been submitted to their engineering department for review and

approval. The licensee has determined that the gouges can be removed by grinding and blending without violation of minimum pipe wall thickness criteria. The licensee's repair and inspection activities on this item will be examined during future inspections.

No items of noncompliance or deviations were identified.

3. Unit 1 Preservice Inspection (Baseline)

The Westinghouse Preservice Inspection summary for Unit 1 is currently under review by the licensee. Discussions with licensee personnel determined that the four indications requiring evaluation have been resolved. The four indications are located on the base material approximately four and one-half inches below the centerline of the flange-to-upper shell weld at vessel azimuths of approximately 25°, 115°, 205° and 295°. Evaluation of the indications suggest that they are the result of handling lugs which may have been attached during vessel manufacture.

No items of noncompliance or deviations were identified.

4. Component Cooling Water Weld Discrepancy

On May 23, 1983 the licensee submitted LER No. 83-007 concerning a discrepancy in a shop weld in the Component Cooling Water System. Licensee personnel observed that in the process of welding a reinforcement pad on a branch connection to the non-vital loop C of the Component Cooling Water System, water began leaking from the weld area. The discrepancy was determined to be a flaw in the root pass of the shop weld joining the branch connection to the header. The licensee indicated that repairs will be made using an approved ASME Section XI repair program. In the interim the branch connection has been cut out and sent to a materials laboratory to determine the failure mechanism.

As indicated in LER No. 83-007, the initial ultrasonic examination of the discrepant weld and ten other similar welds indicated that the code required full penetration welds may not have been made. However, when the branch connection with the discrepant weld was cut out, a full penetration weld was found. The presence of the full penetration weld raises questions regarding the applicability of this type of NDE method for this weld configuration. Examination of the weld detail for the discrepant weld determined that the weld detail did not clearly specify what type of weld was required. Based on these findings the inspector considers that a document review of other similiar type branch connection weld packages would assure that the welds complied with code requirements. In addition, the inconclusive results of the ultrasonic examination on the ten similiar type welds indicates that this type of nondestructive examination would not provide credible results.

The above expressed concerns and the licensee's repair program and inspection activities will be examined during a future inspection. (50-275/83-20/01)

ATTACHMENT 1

5. Violation of Minimum Wall Thickness on Reactor Coolant System Piping

On May 23, 1983, the licensee submitted LER No. 83-006 describing four areas of Weld No. WIB-RC-2-17 that were found to be less than the minimum wall thickness specified by design. Weld No. WIB-RC-2-17 is in Loop No. 2 of the cold leg between the reactor coolant pump and the reactor vessel. During this inspection the inspector was informed that subsequent examination by the licensee of twenty additional welds (located inside the biological shield) identified two more welds with potential minimum wall violations (Weld Nos. 2-19 and 4-17). The licensee is performing ~~additional confirmatory NDE examinations to verify~~ the validity of the initial findings. If a minimum wall problem is confirmed the licensee will document the discrepant welds on the same nonconformance report documenting Weld No. WIB-RC-2-17. This weld is documented on Nonconformance Report No. DCI-83-QC-N024.

During examination of licensee activities related to this item the inspectors became aware of an apparent failure by the licensee to follow their nonconforming report procedure for the identification of discrepant conditions. This failure resulted in a delay in notifying the NRC of a potentially reportable condition in accordance with the provisons of the Technical Specifications. The chronology of events, from the initial discovery of the discrepant condition to the May 10, 1983 NRC notification is described as follows:

- December 1982 - the licensee formulated plans to examine Reactor Coolant System Piping welds.

- December 7, 1982 - Weld No. WIB-RC-2-17 is measured with a NORTEC 131-D, ultrasonic tester.

- December 13, 1982 - a wall thickness measurement report is submitted. This report describes potential minimum wall problems with Weld Nos. WIB-RC-2-17 and WIB-RC-3-13. This report is distributed to appropriate levels of plant management including the plant manager, plant superintendent, and project superintendent.

- December 17, 1982 - a Nuclear Plant Problem Report (NPPR) is written on Welds Nos. WIB-RC-2-17 and WIB-RC-3-13.

- January 7, 1983 - Weld Nos. WIB-RC-2-17 and WIB-RC-3-13 are re-examined. Weld No. 2-17 is confirmed below minimum wall and Weld No. 3-13 is found to be within design limits.

- February 16, 1983 - Westinghouse attempts mechanical measurements from inside pipe, the results are not conclusive but some low areas are detected.

- March 28, 1983 - Weld No. WIB-RC-2-17 is examined again using a new ultrasonic tester (KB/USL-38). The weld is again confirmed below minimum wall.

ATTACHMENT 1

- April 1, 1983 - ISI/NDE weekly reports indicate that Weld No. WIB-RC-2-17 examined and confirmed below minimum wall.
- April 13, 1983 - Westinghouse Field Deficiency Report is issued with a proposed disposition.
- May 5, 1983 - a Nonconformance Report is written on Weld No. WIP-RC-2-17.
- May 6, 1983 - Licensee requests that examinations on Weld No. WIB-RC-2-17 be repeated to confirm and determine to what extent the condition exists.
- May 9, 1983 - Weld No. WIB-RC-2-17 is examined again using a KB/USL-38 ultrasonic tester and confirmed as being below minimum wall.
- May 10, 1983 - NRC informed of minimum wall conditions for Weld No. WIB-RC-2-17.

The inspectors noted that 10 CFR 50.36(c) specifies that Technical Specifications include items in the categories delinated therein. Category No. 4, which is entitled "Design Features" states that, "Design features to be included are those features of the facility such as material of construction and geometric arrangements which if altered or modified would have a significant effect on safety..." Section 5.4.1 of the Technical Specifications for the Diablo Canyon Nuclear Power Plant Unit No. 1 states that, "The reactor coolant system is designed and shall be maintained...in accordance with the code requirements specified in Section 5.2 of the FSAR..." Section 5.2.3 of the Diablo Canyon FSAR states in part that, "The minimum wall thickness of the pipe and fittings are not less than that calculated using ASA B31.1, Section 1 formula of paragraph 122...".

The inspectors reviewed Westinghouse Equipment Specification No. G-676341, dated April 4, 1967, which describes the design and construction of the reactor coolant system piping. Paragraph 3.2 of this specification requires that the minimum wall thickenss shall not be less than that calculated using ASA B31.1, Section I paragraph 122. Calculations by the inspectors verified the minimum wall design requirements for the cold leg of the reactor coolant piping is 2.22 inches. Southwest Fabricating Drawing No. 7524-F, Sheet 9 specifies the minimum wall for the RCS cold leg piping as 2.215 inches.

The inspectors further noted that Nuclear Plant Administrative Procedure No. C-12, "Indentification and Resolution of Problems and Nonconformances," states in part in paragraph C.1.a that, "The Plant Manager and the plant department heads have been delegated the responsibility for determining whether a problem identified in a Nuclear

Plant Problem Report is a nonconformance." A nonconformance is defined in paragraph C.1.b. of this procedure as a, "Discrepancy or departure from requirements in purchase specifications, drawings, approved practices, established Quality Assurance policies or procedures, or NRC requlations which require resolution...and...if left uncorrected could have resulted in degradation or loss of integrity of the reactor coolant pressure boundary." Appendix A of the procedure provides examples to aid in determining whether a problem or potential problem is a nonconformance. Paragraph E.1. of Appendix A to the procedure states that, "A discrepancy or departure from requirements in design documents or activities shall be identified and documented as a nonconformance if...it was identified after the design document had been approved and it was issued for use as a basis for further design." The departure of Weld No. WIB-RC-2-17 from specified design requirements appears to fall into the category of a nonconformance.

Therefore, as the previously presented chronology of licensee actions indicate that by December 13, 1983 appropriate levels of plant management were informed of the discrepancy. In addition, on December 17, 1982 a Nuclear Plant Problem Report No. DCI-82-QC-P0300 was written describing the discrepant condition, though it was not reported at this time as a condition requiring a nonconformance report. It appears that sufficient evidence was available at this time to warrent issuance of a noncoformance report. If a nonconformance had been written the Technical Review Group would have been required to review the discrepant condition for reportability under the Technical Specifications. Further, paragraph 6.9.1.12.i of the Technical Specifications requires that, "Discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition," be reported within 24 hours to the NRC. Discussions with the Office of Nuclear Reactor Regulation (NRR), Standard Technical Specification Group indicate that corrective measures includes engineering analysis. It appears that the late documentation of the condition in the nonconformance reporting system contributed to the untimely reporting of this problem to the NRC.

The failure of the licensee to comply with technical specifications and report the defective weld in a timely manner is considered an apparent item of noncompliance. (50-275/83-20/02)

6. Management Meeting

On May 27 and June 3, 1983, the inspectors met with licensee representatives denoted in paragraph 1. The scope of the inspection, the observations, and the findings of the inspectors were discussed. The licensee acknowledged the inspectors concerns and the apparent item of noncompliance as described in this report.

ATTACHMENT 2

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

JUN 2 9 1983

Docket No.: 50-275

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

FROM: Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: INFORMATION ITEM REGARDING ADDITIONAL POTENTIAL VIOLATION OF MINIMUM WALL THICKNESS OF REACTOR COOLANT SYSTEM PIPING AT DIABLO CANYON, UNIT 1 (Board Notification No. 83- 89)

In accordance with the present NRC procedures for Board Notifications, the following item is enclosed for information of the Commission:

NRC Preliminary Notification - PNO-V-83-22A, dated June 23, 1983. This PNO identifies possible additional deviations from Code requirements with respect to minimum specified wall thickness for the reactor coolant system piping. This issue was previously identified in Board Notification No. 83-72, dated June 20, 1983 and Board Notification No. 83-83, dated June 24, 1983. Region V is closely following the licensee's actions in this matter. We will promptly inform you of any further significant finding on this matter.

Darrell G. Eisenhut
Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: See next page

Contact: H. Schierling, ONRR
x27100

8305110500

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE--PNO-V-83-22A Date: 6/23/83

This preliminary notification constitutes EARLY notice of events of POSSIBLE safety or public interest significance. The information is as initially received without verification or evaluation and is basically all that is known by IE staff on this date.

ATTACHMENT 2

FACILITY: Pacific Gas & Electric Company
Diablo Canyon Unit No. 1
Docket No. 50-275
San Luis Obispo County, California

SUBJECT: POTENTIAL VIOLATION OF MINIMUM WALL THICKNESS ON REACTOR COOLANT SYSTEM PIPING

Licensee Emergency Classification:

____ Notification of Unusual Event
____ Alert
____ Site Area Emergency
____ General Emergency
X Not Applicable

On June 22, 1983, licensee personnel determined that the RCS piping in areas of welds other than reported on May 10, 1983 may be less than the minimum wall thickness specified by code. A total of ten welds in cold leg, hot leg, and cross over (between steam generator and reactor coolant pump) piping appear to be less than required minimum wall thickness. The condition was discovered while performing additional ultrasonic examinations at each 45 degree increments around the welds. The tests were to be used to improve the quality of the base-line preservice examination. The licensee identified the additional problems after determining that incorrect minimum wall thickness criteria had been applied to the hot legs and cross over piping. All four RCS loops are involved. The reduction in wall thickness appears to have occurred during the original preservice examination (1975-76) when welds were ground smooth to remove the potential for irrelevant indications when performing ultrasonic and/or radiographic examinations. The licensee plans to take additional thickness measurements at 3-inch circumferential steps around the welds to determine if any other minimum wall thickness conditions exist.

Region V will closely follow this situation.

Media interest is not expected. Neither the licensee nor the NRC plans to issue a news release at this time. Region V (San Francisco) received notification of this occurrence from the Resident Inspectors at about 11:00 a.m. on June 23, 1983. This information is current as of 2:00 p.m. on June 23, 1983.

CONTACT: D. F. Kirsch
463-3723

DISTRIBUTION:

H St. 4:24	MNBB ____	Phillips ____	E/W 4:20	Willste ____	Air Rights ____	Mail:
Chairman Palladino	EDO	NRR	IE	NMSS		ADM:DMB
Comm. Gilinsky	PA		OIA	RES		DOT: Trans Only
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Comm. Roberts	ELD					
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ACRS		Licensee: ____				
CA		(Reactor Licensees)				
PDR		Resident Inspector				

REGION V: FORM 21
(revised 3/14/83)

ATTCHMENT J

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

JUL 11 1983

#3

Docket No.: 50-275

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine

FROM: Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: INFORMATION ITEM REGARDING ADDITIONAL POTENTIAL VIOLATION OF MINIMUM WALL THICKNESS OF REACTOR COOLANT SYSTEM PIPING AT DIABLO CANYON, UNIT 1 (Board Notification No. 83-96)

In accordance with the present NRC procedures for Board Notifications, the following item is enclosed for information of the Commission:

Pacific Gas & Electric Company letter dated July 5, 1983 from Mr. Schuyler to Mr. John B. Martin transmitting its report on investigation of reactor coolant pipe weld thickness at Diablo Canyon. This issue was previously identified in Board Notification No. 83-72, dated June 20, 1983, Board Notification No. 83-83, dated June 24, 1983 and Board Notification 83-89 dated June 29, 1983. Region V is closely following the licensee's actions in this matter. We will promptly inform you of any further significant finding on this matter. As indicated in the above cited letter, copies of the enclosure have been provided to the service list.

for Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: See next page

Contact: B. Buckley, ONRR
x28379

8307110002

ATTACHMENT 3

PACIFIC GAS AND ELECTRIC COMPANY

PG&E 77 BEALE STREET • SAN FRANCISCO, CALIFORNIA 94106 • (415) 781-4211 • TWX 910-372-6587

J. O. SCHUYLER
VICE PRESIDENT
NUCLEAR POWER GENERATION

July 5, 1983

Mr. John B. Martin, Regional Administrator
U. S. Nuclear Regulatory Commission, Region V
1450 Maria Lane, Suite 210
Walnut Creek, CA 94596-5368

Re: Docket No. 50-275, OL-DPR-76
Diablo Canyon Unit 1

Dear Mr. Martin:

Enclosed is Pacific Gas and Electric Company's Report on Investigation of Reactor Coolant Pipe Weld Thickness at Diablo Canyon, dated July 1, 1983.

Sincerely,

J O Schuyler

Enclosure

cc w/enc: Document Control Desk (2)
U. S. Nuclear Regulatory Commission
Washington D. C. 20555

Mr. George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington D. C. 20555

Service List

ATTACHMENT 3

REPORT ON INVESTIGATION OF REACTOR COOLANT PIPE WELD THICKNESS AT DIABLO CANYON

July 1, 1983

ATTACHMENT 3

TABLE OF CONTENTS

ATTACHMENT 3

I. INTRODUCTON

In December 1982, members of the Inservice Inspection/Nondestructive Examination (ISI/NDE) section of the plant staff Quality Control group performed, using an ultrasonic technique, a series of measurements across the weld and adjacent base metal for 31 of the 56 girth welds on the reactor coolant system piping of Diablo Canyon Unit 1. The objective of this effort was to generate a plot of the contour of the weld root counterbore surface. These axial thickness "profiles" were made at several locations around each weld.

These measurements were taken when the ISI/NDE group was completing a general review of the plant's ASME B&PV Code, Section XI preservice inspection (PSI) prograr [illegible] ofile information is not required for PSI data, but does provide additi [illegible] formation useful in evaluating ultrasonic indications that might be obs [illegible] n future inservice inspections. The examiners were primarily intereste [illegible] locating irregularities on the inside surface of the pipe which could act as geometric reflectors and produce anomalous shear wave ultrasonic indications. While the inside surface can be directly observed during PSI work, post-operational access to these surfaces is not feasible due to high radiation levels. While it is possible to obtain this profile information post-operationally on an as-need basis, this work would also involve additional personnel radiation exposure. These factors provided the incentive to identify geometric reflectors before the plant started up.

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Review of the profile data revealed that four locations on weld WIB - RC-2-17 on the cold leg of Loop-2 might be below the specified minimum thickness. During the next several months various efforts were made to resolve this matter. On May 9, 1983, after these subsequent investigations failed to resolve the concern, a verbal notification was made to the NRC. A written Licensee Event Report (LER) was submitted on May 23, 1983. This LER was considered to be an interim report, with a final report to be submitted following completion of further investigation.

During the course of the ensuing UT investigation, it was discovered that there might be below minimum thickness areas on nine other welds in addition to 2-17. As a result of this development and its broader implications and in response to questions raised by the NRC, the ongoing investigation was intensified to resolve the problem. This investigation included a review of all quality records and controls related to these welds, mechanical measurements of pipe ID and OD using micrometers, visual inspection of a number of welds, a comprehensive UT measurement and evaluation program, and a review of the significance of apparent below minimum measurements on the adequacy of the welds.

This report summarizes the results and conclusions of the investigations which were performed. In addition, it includes a detailed chronology of the events.

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II. SUMMARY OF CONCLUSIONS

The principle conclusions of the investigation are summarized below.

- o Review of the construction quality records and controls showed:
  - A rigorous quality assurance program was followed in the manufacture and installation of the reactor coolant piping.
  - Inspections and tests verified that the pipe was made in compliance with the technical requirements.
  - Documentation of required inspections and tests are available in the quality records. The records show that the pipe and welds meets specified requirements including the minimum wall thickness requirements.
  - Records show that metal removal during grinding was controlled and met specification.
  - No evidence of failure to perform required checks or to complete required documents was found.
- o Reinspections conducted during this investigation have confirmed and coroborated the original records.
- o Micrometer measurements of the inside diameter of the weld corroborated records of original diameter measurements and control of inside surface grinding.
- o Micrometer measurements of the outside diameter, coupled with those made of the inside diameter, corroborate records of original pipe wall thickness and show that minimum wall thickness specifications are met in all cases.

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- Analysis of ultrasonic (UT) thickness measurement capabilities and UT data and records show that UT measurements lack the precision required to verify the original micrometer wall thickness measurements on reactor coolant piping.

The overall conclusions gained from this investigation are that (1) the reactor coolant loop piping and other design Class I piping in the plant meet minimum wall requirements, and (2) the associated concerns raised by the misleading UT readings have been completely resolved.

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III. SUMMARY OF CONSTRUCTION QUALITY RECORDS AND CONTROLS

When preliminary UT data indicated that as many as 10 reactor coolant loop piping girth welds might potentially have areas less than minimum specified thickness, an extensive investigation was made of the quality assurance records and controls for these welds. The purpose of this investigation was to:

1. Verify that a complete quality package was available,
2. Identify and review the adequacy of the quality controls on activities such as field grinding which may have caused the concern, and
3. Obtain any supplemental information which might substantiate, refute, or explain the preliminary ultrasonic results, which had been obtained.

In this section, the results of this quality investigation are described.

A. Background Information on Welds

1. Location and Type

The welds being examined are girth welds located in the reactor coolant system (RCS) loop piping which is comprised of 4 reactor coolant loops attached to the reactor vessel, as shown on Figures III-1, III-2, III-3 and III-4. Each loop is made up of a hot leg, a crossover leg, and a cold leg. The hot leg connects the reactor vessel to the steam generator, the crossover leg connects the steam generator to the reactor coolant pump, and the cold leg connects the reactor coolant pump to the reactor vessel.

As shown on the figures, each loop contains 14 girth welds, for a total of 56. Of the 14 girth welds on each loop, 4 are located on the hot leg (2.335 inch specified minimum wall thickness) of which 2 are shop and 2 are field welds; 6 are located on the crossover leg

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(2.495 inch specified minimum wall thickness) of which 2 are shop and 4 are field welds; and 4 are located on the cold leg (2.215 inch specified minimum wall thickness) of which 2 are shop and 2 are field welds.

2. Design, Fabrication and Installation

The RCS loop piping was designed to the USAS B31.1 Code, 1955 edition, with the addition of Code Cases N-7 and N-10 and Westinghouse (W) equipment specification G-676343. The piping was fabricated and examined by Southwest Fabricating and Welding Company (Southwest) in accordance with W specification G-676343 and ASME Section I, 1968 edition and was documented on a Form P-4A. The shop welds were made by Southwest in 1969 and 1970 for the hot and cold legs. The shop welds on the crossover legs were made by Southwest in 1973 and 1974. The hot and cold leg piping was received by PG&E in June of 1970. The crossover piping was received in the time period between September 1973 and February 1974.

After design was complete and fabrication started, ASME Section III was expanded in scope to cover piping for nuclear power plants. PG&E Specification 8752, issued for installation of the nuclear steam supply systems, was revised in March 1974 to incorporate requirements of ASME Section III, 1971 edition.

The RCS loop piping installation was performed by Wismer and Becker (W&B) and documented on a modified Form N-5. Although W&B was an ASME Section III "NA" stamp certified installer, the installation was not stamped because the design and fabrication had been done to different and earlier codes. The field welds were made by W&B during the period from May of 1973 to April of 1974. The W&B welds were performed to ASME Section I, 1968 edition and ASME Section III, 1971

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edition with Summer 73 addendum, W specification G-676496, PG&E Specification 8752, and under W&B's quality assurance program. RCS was released to the PG&E operations by PG&E General Construction on May 2, 1977.

B. Quality Records on RCS Loop Piping Welds

One of the main purposes of the investigation was to review the quality package on each weld for completeness. The reviewers found the records to be comprehensive and complete. The factual basis for much of the discussion which follows is contained in these records.

C. History and Controls on Grinding

If a less than minimum wall condition were proven to be true, it was p ated that the most likely cause was grinding on the welds to r surfaces for PSI and/or to remove indications found during PSI. A ult, a thorough review was made of the history and controls plac pon grinding operations on these welds. In this section, the results of this investigation are discussed.

1. Shop Fabrication

Shop welds were ground on the inside and outside by Southwest for surface preparation for liquid penetrant and radiographic examination and to remove penetrant indications. Southwest had procedures and inspection controls to measure pipe wall thickness at the weld bevel end prior to welding to insure minimum thickness specifications were met. This provided benchmark references for subsequent welding. Control of thickness during welding was by control of counterbore and outside diameter dimensions together with use of alignment jigs to maintain concentricity. Dimensional checks using micrometers were made prior to shipment.

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In 1970, PG&E General Construction Department inspected the as-received hot and cold legs and identified depressions in the pipe surface and "punch mark" identification markings. This was documented on PG&E Deviation Report Serial No. 39. This report addressed wall thickness and resulted in extensive optical, mechanical and ultrasonic thickness measurements being made. These measurements confirmed prior determinations that minimum wall thickness specifications were met. They also showed that ultrasonic thickness measurements were not always consistent on this pipe material, and were not of adequate precision for measuring to the specified tolerances.

2. <u>Field Welding</u>

The weld preparations and fit-up were checked prior to welding and the depth to the top of the root pass was measured as documented in the W&B Documentation Checklist-Traveler Packet. These measurements further verify that adequate wall thickness existed prior to field weld out.

PG&E Specification 8752 and W&B weld procedure specification 3500-1 required "The inside surface of the weld shall be clean, smooth, and free from the presence of sharp irregularities, lumps, and oxidation. Surface shall also be free of undercut." These criteria were included on the W&B visual weld examination checklist as item #18. The inside surfaces were polished clean and smooth. Metal

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removal was kept to a minimum under surveillance of W&B QC, PG&E inspectors and Westinghouse. Visual and liquid penetrant inspections of the inside surface are documented in the W&B Documentation Checklist - Traveler Packet. When grinding was done to remove penetrant indications, the depth of grinding was measured and weld metal added (if required) as documented in the Documentation Checklist - Traveler Packet to assure that minimum wall thickness was maintained. No direct wall thickness measurements were made after field weld completion.

3. Preparation for UT Inspection

Shop and field welds were ground on the weld crown in 1974 and 1975 by W&B at the direction of PG&E to facilitate ultrasonic inspection required by ASME Section XI for PSI/ISI. On several shop welds, indications required grinding and weld buildup. The weld buildup was performed by Southwest in the field in 1975.

Requirements for weld crown grinding to prepare for ultrasonic examination were specified by PG&E instructions. Confirmation was obtained from the pipe supplier and designer (Westinghouse) that weld reinforcement was not required to meet minimum wall thickness requirements. Welds were ground approximately flush (+1/16, -0) with immediately adjacent base material. A special grinding crew was selected and trained by W&B to do the work. PG&E General Construction inspectors provided continuous surveillance of the grinding to assure that grinding did not go below the immediately adjacent pipe or fitting surface, and that grinding was confined to

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the weld metal. This process assured that minimum wall thickness requirements were maintained. Special short straight edges were fabricated to gage flushness (+1/16,-0) and flatness across the welds. PG&E UT examiners inspected final surface finish for adequate smoothness for ultrasonic examinations. Liquid penetrant examination of the final weld outside surface was also performed by W&B.

4. Preservice Inspection

The PSI liquid penetrant examinations of the outside weld surfaces have not required further grinding. No inside weld surface liquid penetrant examinations were made for preservice inspection in 1975. Ultrasonic inspection conducted for preservice inspection in 1979 disclosed indications at the inside diameter of some of the reactor coolant loop girth welds. The indications were investigated by liquid penetrant inspection of the weld inside surfaces. These indications were removed by superficial spot grinding and by buffing and polishing under controlled conditions. This work was performed by Pullman Power Products (PPP) and PG&E in 1979.

The depth of grinding was measured and evaluated for impact on wall thickness by comparison with the depth of the weld counterbore. This check verifies that wall thickness requirements were met. In some cases ultrasonic thickness measurements were made of the adjacent pipe wall and the depth of grinding was measured with a machinist level and depth gage. This work and these inspections, measurements and evaluations are documented in the PSI data packages for the reactor coolant piping girth welds.

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D. Grinding on Girth Welds on Other Design Class I Piping Systems

An evaluation was also performed of grinding on girth welds in other design Class I piping systems to evaluate its impact, if any, on maintenance of minimum wall thickness.

Grinding on welds can be separated into inside and outside surface grinding. Reactor coolant pipe welds are accessible for inside grinding due to the pipe's large diameter and availability of access through the reactor vessel nozzles and steam generator manways. The inside surface of most other welds cannot be ground because there is no way to gain physical access. Special access provisions such as cutting the pipe to allow one to reach in would be required.

The inside surfaces of the four feedwater pipe to steam generator nozzle welds were inspected in 1977 and work performed as needed. Access for work on the inside surface of the welds was provided by cutting out a short section of pipe. The inside surfaces of the four main steam pipe to steam generator nozzle welds were inspected in 1977 and work was performed as needed. Access was provided by entering through the steam generator manway, removing internal parts and building scaffold on top of the moisture separator section. In addition, two branch line connections off the reactor coolant pipe had some minor inside surface grinding. These were the 14 inch surge line and the 14 inch residual heat removal branch connection welds.

In all cases, work on the inside surface of welds was closely monitored and inspected to assure compliance with Code and design requirements.

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The depth of any grinding was measured and, if necessary, a weld buildup added to assure maintenance of minimum wall thickness. Wall thickness verification was also made using ultrasonic methods. Work and inspections are documented in the PPP records. The outside crown of welds in other than the reactor coolant pipe were originally prepared for visual, surface and volumetric (liquid penetrant and radiography) examination and acceptance as required by the applicable fabrication and installation Codes. Some of these welds were selected for inservice inspection and had further outside surface grinding performed to facilitate PSI/ISI ultrasonic examinations. The outside surface preparation requirements for these welds did not require the weld to be ground flush with the pipe surface as the requirements for the reactor coolant pipe preparation did. These weld crowns could be "flat topped" and left higher than the adjacent pipe surface. It was not necessary to grind the weld crown to as smooth a blend or transition from the pipe surface to the weld surface. The weld crown preparation requirements for PSI/ISI were defined by PG&E and in PPP's procedures.

E. CONCLUSIONS

As the foregoing review demonstrates the shop fabrication, field installation and inservice inspection preparation work was all performed in accordance with appropriate quality procedures, was technically appropriate, and assured that minimum wall thickness requirements were met.

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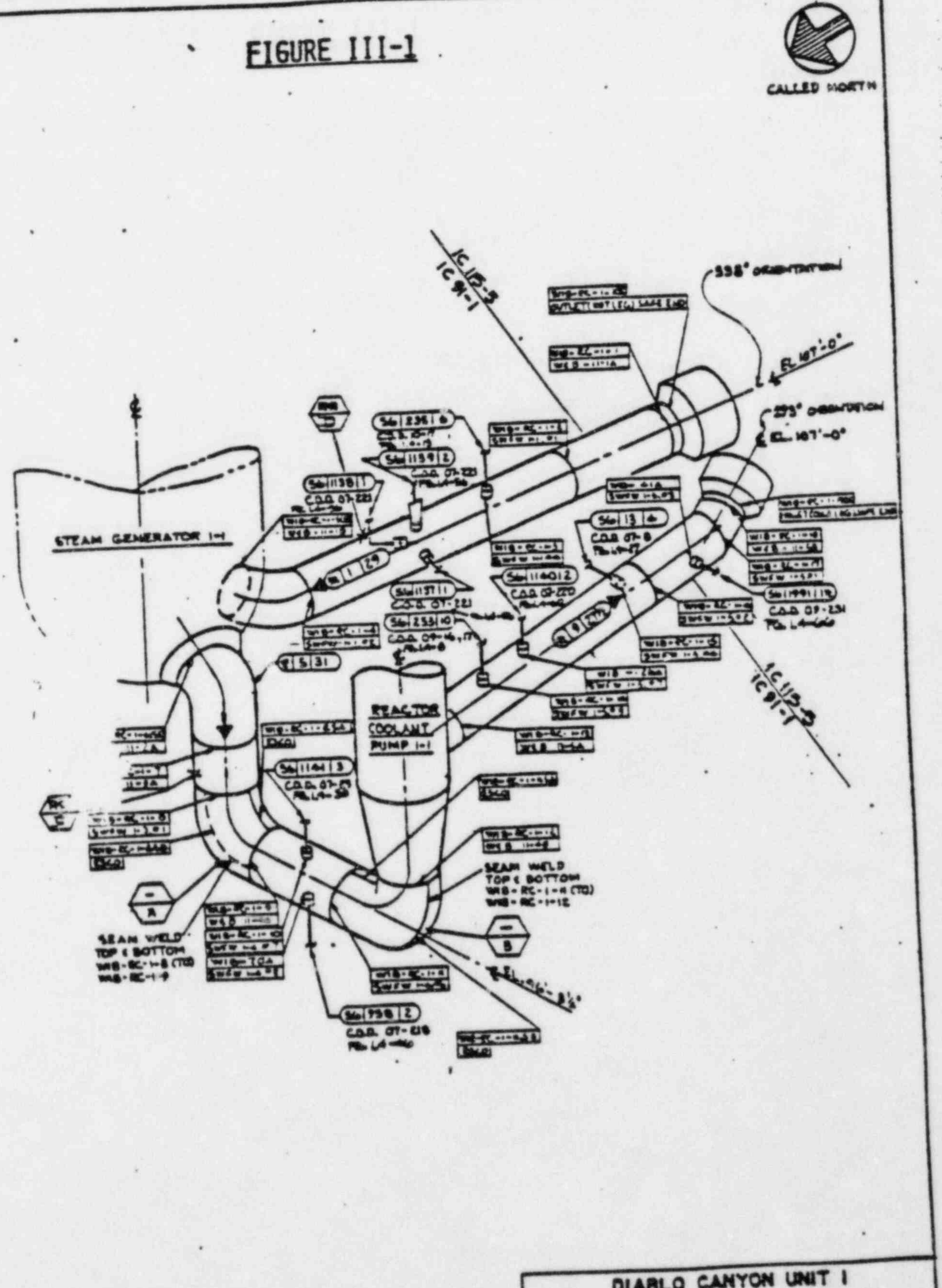
FIGURE III-1



DIABLO CANYON UNIT 1
INSERVICE EXAMINATION ISOMETRIC
PACIFIC GAS AND ELECTRIC COMPANY

	REACTOR COOLANT PP DISCH. LOOP 1
	REACTOR COOLANT PP SUCT LOOP 1
	REACTOR COOLANT OUT LOOP 1
	DESCRIPTION

SYSTEM: RCS
DRAWN: B. GATTISON
CHK'D:
APPR'D:
DATE:

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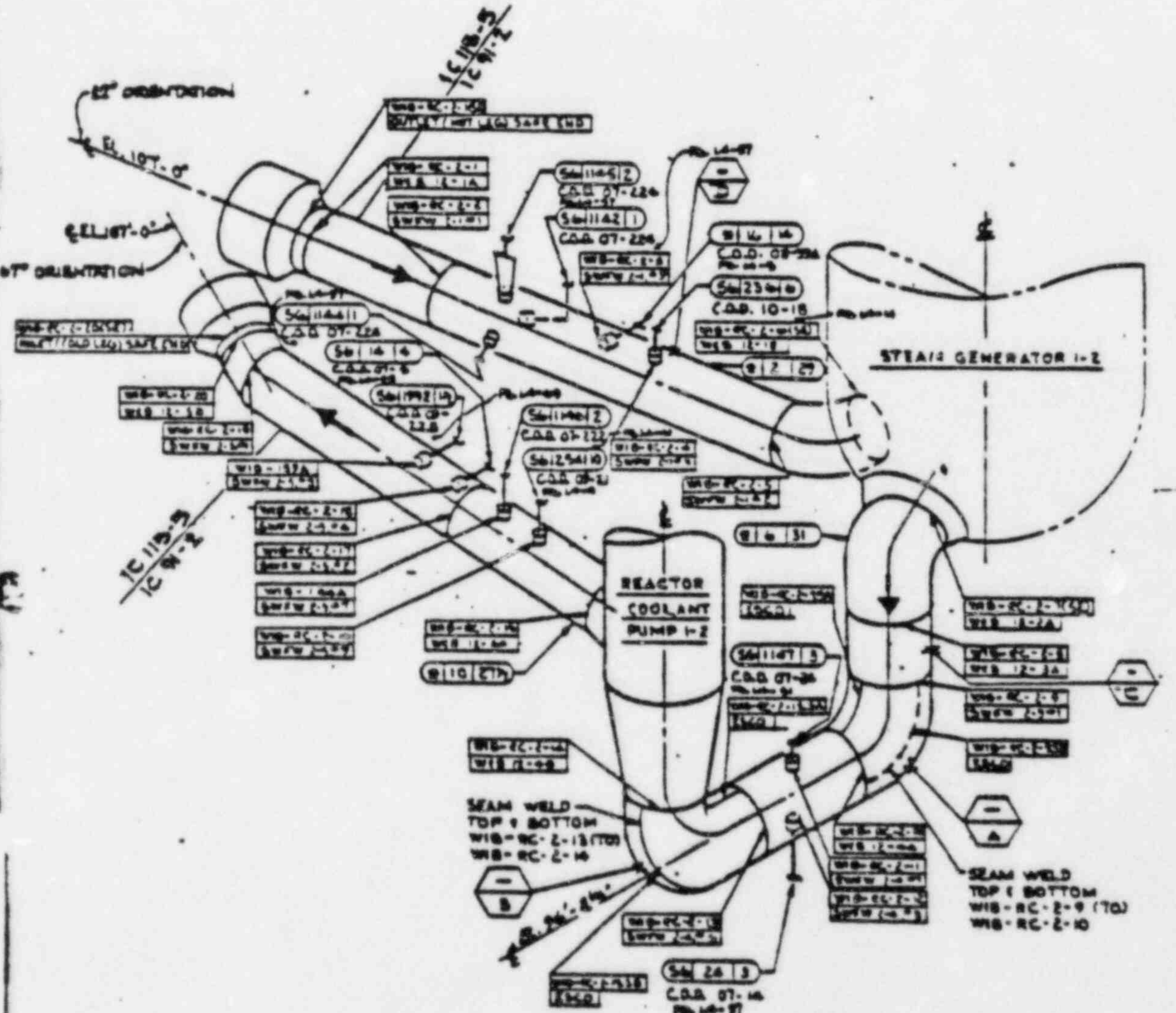
FIGURE III-2

CALLED NORTH



[illegible]	REACTOR COOLANT OUT LOOP 2
[illegible]	REACTOR COOLANT PP SUCT LOOP 2
[illegible]	REACTOR COOLANT PP DISCH LOOP 2
LINE NO.	DESCRIPTION

DIABLO CANYON UNIT I
INSERVICE EXAMINATION ISOMETRIC
PACIFIC GAS AND ELECTRIC COMPANY

AREA: [illegible]	[illegible]	[illegible]	SYSTEM: RCS
REF. DWG. RC 1-2	REPAIRS 010	LINE NO. [illegible]	DRAWN: [illegible] CHK'D: APPR'D: DATE:

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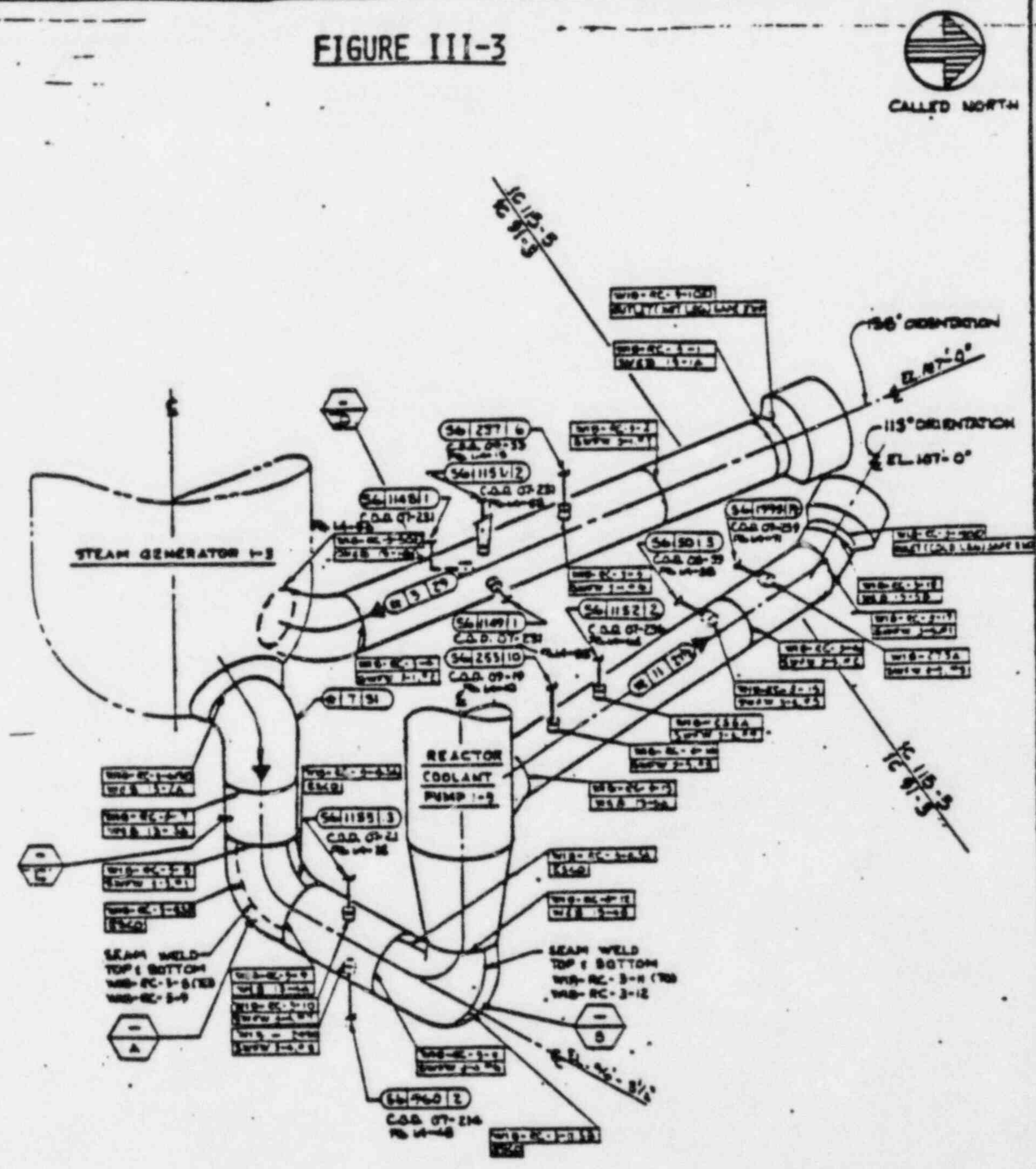
FIGURE III-3



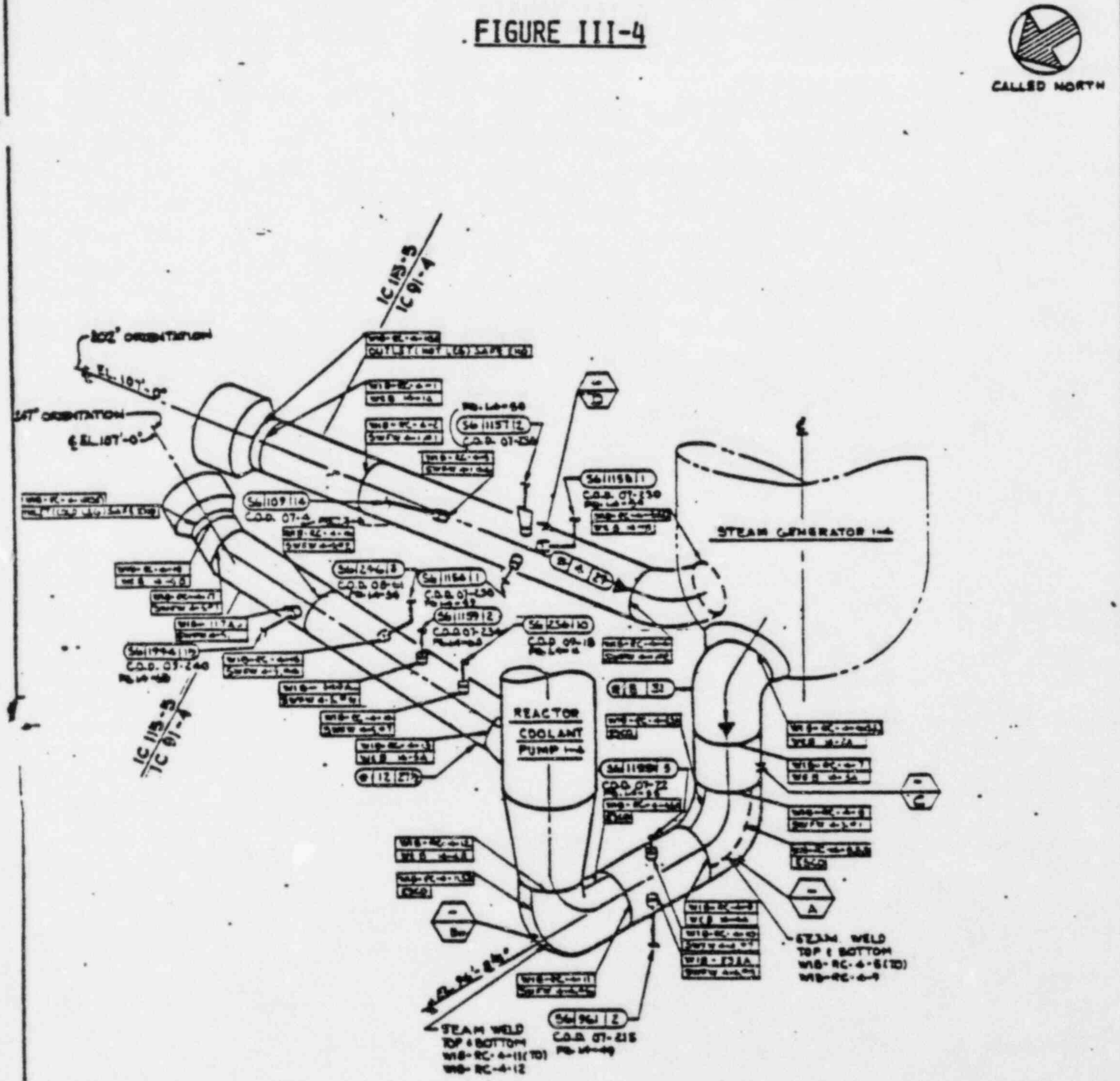
DIABLO CANYON UNIT I
INSERVICE EXAMINATION ISOMETRIC
PACIFIC GAS AND ELECTRIC COMPANY

	REACTOR COOLANT OUT. LOOP 3
	REACTOR COOLANT PP SUCT. LOOP 3
	REACTOR COOLANT PP DISCH. LOOP 3
LINE NO.	DESCRIPTION

SYSTEM: RCS
REF. DWG. RC 1-3
DRAWN: B. GATTISON
PAGE: 1.4-3

ATTACHMENT 3

FIGURE III-4



DIABLO CANYON UNIT I

INSERVICE EXAMINATION ISOMETRIC

PACIFIC GAS AND ELECTRIC COMPANY

SYSTEM: RCS

DRAWN: S.GATTISON

0-4-29 SP.	REACTOR COOLANT OUT LOOP 4
0-8-31 SP.	REACTOR COOLANT PP SUCT LOOP 4
0-12-27½ SP.	REACTOR COOLANT PP DISCH LOOP 4
	DESCRIPTION

REV. DWG.	REPORT	LINE NO.
RC 1-4	042	0-4-29 0-8-31 0-12-27½
		1.4-4

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IV. VISUAL INSPECTION AND RADIOGRAPH REVIEW

In order to further corroborate information contained in the quality records, a visual inspection was conducted on the suspect welds, and the acceptance radiographs for selected reactor coolant pipe welds were reviewed. The results of these investigations are discussed in this section.

A. Visual Inspection

On June 28 and 29, 1983, Mr. Jim Miller, Lead Welding Engineer for the Diablo Canyon Project Team, performed a visual inspection of nine (WIB-RC-1-1, 1-2, 1-8, 1-11, 1-16, 2-1, 2-17, 3-9 and 4-16) of the girth welds under investigation in the reactor coolant system on Diablo Canyon, Unit 1. Both the outside and inside surfaces of the welds were inspected. As a result of this inspection, Mr. Miller concluded that:

> "The inspected welds are visually acceptable to the specification and codes governing weld quality visual acceptance standards, workmanship is considered good and meets industry standards."

A copy of his report is included as Attachment IV-1. This report confirms the constructon visual inspection records for the welds.

B. Radiograph Review

The film quality of radiographs used for acceptance of the Unit-1 reactor coolant pipe welds was reviewed by D. R. Cady, Bechtel NDE Level III, and witnessed by PG&E. Radiograph film from four shop welds and five field welds, chosen at random, and weld WIB-RC-2-17 were reviewed. The completed checklist from the review is included as Attachment IV-2. All film reviewed was in compliance with the originally specified requirements.

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ATTACHMENT IV-I

INTEROFFICE MEMORANDUM

Diablo Canyon Project

PACIFIC GAS AND ELECTRIC COMPANY
BECHTEL POWER CORPORATION

To J. D. Shiffer - NPO

From J. A. Miller

Of General Construction

At Diablo Canyon Project Extension

Date June 30, 1983

File No.

Subject Visual Inspection Report Reactor Cooling Loop Piping Welds

On June 28 and 29, 1983, I performed the requested visual inspection of selected code Class I welds on the Reactor Coolant Piping in Unit #1 at Diablo Canyon. I am presently assigned as Lead Welding Engineer for construction at Diablo Canyon.

On June 28, 1983, I was accompanied by:

Jim Shiffer - Manager, Nuclear Plant Operations - PG&E
Frank Dodd - Senior Metallurgical Engineer - PG&E
Rick Cahoon - Site Mechanical Engineer - Westinghouse
Pete Broadnick - Nuclear Plant Operations Maintenance Foreman - PG&E
Bob Hindmarsh - Senior Construction Engineer - PG&E

On June 29, 1983, I was accompanied by Mr. Hindmarsh.

The attached is my report on the subject welds.

J. A. Miller
Lead Welding Engineer
Project Team
Diablo Canyon

JAMiller:fgm
cc: H. Friend w/o attachments w/report
J. Shryock " "
L. Rossetta " "
D. Rockwell " "
R. Etzler " "
R. Manley (M&QS) " "
J. Manning " "
R. Bain " "

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ATTACHMENT IV-2

FILM QUALITY REVIEW CHECKLIST

Check Point	Weld No. 1-1B	1-1A	4-1A	3-5A	7524F 9 W-1	7524F 6 W-1	7524F 13 W-2	7524F 9 W-2	7524F 4 W-1
Reader Sheet Complete	X	X	X	X	X	X	X	X	X
Penetrameter No.	X	X	X	X	X	X	X	X	X
Sensitivity	X	X	X	X	X	X	X	X	X
Pen. Location	Note 1	Note 3	X	X	X	X	X	X	X
Weld Indent.	X	X	X	X	X	X	X	X	X
Density*	X	X	X	X	X	X	X	X	X
Processing Quality	Note 2	X	X	X	X	X	X	X	X

NOTES:

1. 1-1B enetrameter is partly off edge of film but all T-holes are reada ere are 3 other penetrameters that are completely on the film. (Thre ameters required for panoramic exposures.)

2. Artifac tified on reader sheet and did not mask the interpretation for indicatio

3. 0-1-2 Pen. ok @ 0.
   2-3-4 Pen. in overlap @ 2 ok.
   4-5-6 Pen. in overlap @ 4 but 2 T hole at lap. Not ok.
   6-7-8 Pen. @ 8 partially off film but holes are readable.
   8-9-10 Pen. @ 10 partially off film but holes are readable.
   10-11-12 Pen. @ 10 in overlap area, same pen. as 8-9-10.
   12-13-14 Pen. @ 14 partly off film but holes are readable.
   14-15-0 Pen. @ 14 and 0 in overlap areas.

4. Yellowing from age is beginning to appear on some film.

5. 7524F, 9, W-2 is WIB-RC-2-17.

Reviewed by,

D.L. Cady

Witnessed by: F.J. Dodd

*Densitometer not available for density measurements.

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June 30, 1983

VISUAL INSPECTION REPORT

SUBJECT: Unit 1 Reactor Cooling Loop Piping Visual Inspection of Specific Welds.

SCOPE: Visual inspection both inside and outside of weld deposit and related exposed joint preparation condition. Weld joints to be inspected were numbers: 1-1, 1-2, 1-8, 1-11, 1-16, 2-1, 2-17, 3-9 and 4-16.

PROCEDURE: An overlay was prepared for both outside and inside circumferences. The inside overlay was adjusted to reflect the variation in the circumference and correlated to the master outside overlay.

The specific location of the welds to be inspected and identification was determined from general location drawings which physically located all welds on the Reactor Coolant Piping System by number. Free and open access was provided to all welds.

BASIC EQUIPMENT (TOOLS ETC.): Dial indicator, parallel bar, straight edge rule, cloth measure tape, adjustable profile, spirit level and supplementary hand flash lights.

GENERAL OBSERVATIONS (FOR ALL WELDS INSPECTED): All welds subject to visual inspection were identified by the associated weld number stamped into the base metal adjacent to the O.D. of the weld. Top dead center was indicated by a "V" stamp pointing in the direction the overlay was laid out.

The visual inspection of the internal weld joint, (i.e., the Weld Root Pass), revealed no areas of metal removal that visually impinged on existing wall thickness.

The back side of the root bead area was either flat or slightly convex in physical shape, with no visual evidence of base metal or weld metal removal other than that required to provide a smooth acceptable surface for non-destructive examination (NDE). The internal back bevel face of bevel is present with visual evidence of light grinding or buffing to blend in the parent base metal with the root bead reinforcement. This interface area is either flat or slightly contoured and uniform in shape throughout. At various locations the shop machining tool marks are present indicating the start and stop, width and plane of the internal bevel. Machining of the bevel planes are smooth and follow the general shape of the pipe. The internal shape of the weld joint is either totally flush with the base metal across the root weld area, flat between the beveled pipe ends or slightly contoured. All areas are acceptable to NDE preparation requirements.

External weld deposit area visual examination shows generally polished surface blended into the base metal at the toes of the weld. The deposited weld metal is flat or slightly convex above the base metal. Slight variances do exist in some areas at the toe of the weld, however, these variations are common and are basically the result of "flapper wheel" use to blend in the feather effect of the weld at the toe of the weld to the base metal. This blending is done to facilitate required NDE. Superficial areas of grinding were noted in the base metal area. These areas do not indicate a reduction in the basic pipe wall thickness.

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CONCLUSIONS: The inspected welds are visually acceptable to the specifications and codes governing weld quality visual acceptance standards, workmanship is considered good and meets industry standards.

ATTACHMENT #1 - Weld Locations

ATTACHMENT #2 - Resume - J. A. Miller

ATTACHMENT #3 - Field Inspection Notes

J. A. Miller
Lead Welding Engineer
Diablo Canyon Project Team

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V. SUMMARY AND ANALYSIS OF MEASUREMENTS AND INSPECTIONS

A. Reactor Coolant Loop Piping Weld Thickness Program

After an assessment of alternative methods of measurement it was concluded that the direct mechanical method using micrometers was the most accurate and valid method of measuring piping of this size and material. This conclusion was consistent with information developed in 1970, when measurements on the piping in question showed that the UT thickness measurements — when compared to mechanical measurements made at that time — were not consistent or of adaquate precision.

1. Mechanical Measurements

The outside (OD) and inside (ID) diameters of the pipe were measured. The two micrometers were aligned at the same circumferential locations on the weld centerline using an ultrasonic reference point. Because of the complexity and time consuming nature of these measurements, only welds with low ultrasonic readings were mechanically measured. The method consisted of measuring the OD and ID at the vertical and horizontal axis and one additional measurement at the thinnest point indicated by the previous ultrasonic examination.

Mechanical measurement results are listed in Table V-I for each weld identified by UT as possibly having wall thickness less than specified. In each case, the mechanical measurement indicates that the wall thickness is in fact above minimum requirements.

Table V-2 contains a comparison of measured diameters to specified values for "Max ID" and "Min OD" for unwelded pipe which was made to verify that pipe wall thickness was maintained in the weld area.

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The measured ID values confirm that inside grinding was well controlled and did not reduce the wall thickness below specified requirements. Comparing the measured OD and ID values at each location corroborates the records of original pipe wall thickness measurements and shows that minimum wall thickness specifications are met in all cases.

2. Ultrasonic Measurements:

All RCS ultrasonic data collected prior to the June 22, 1983 NRC notification was first reviewed and compared with the appropriate minimum wall thickness criteria. If the data indicated that a measurement was below the specified requirement, two teams of ultrasonic examiners independently performed a second and third set of independent thickness measurements.

Each weld's thickness was measured at points located on the weld centerline at 3 inch intervals around the entire circumference of the weld. The base metal thickness was also measured at both sides of the weld at these locations. The readings recorded were the lowest obtainable at each location. Any weld areas which appeared to be less than the minimum specified wall thickness were mapped. These areas were mapped along with those points immediately adjacent to them which met minimum wall thickness requirements. Also, similar thickness measurements were made for those welds that previously had only profile data measured.

Forty-four reactor coolant loop welds were ultrasonically measured. Seven had UT measurements below minimum requirements in the weld material. Three additional welds had low UT measurements in the base metal adjacent to the weld. Because weld number 1-8 had only one low reading which was not supported by additional measurements it was disregarded.

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For each weld where ultrasonic data indicated that a below minimum wall condition may have existed, extensive ultrasonic wall thickness measurements were obtained. Table V-1 also summarizes the pertinent data for each of these investigated welds.

B. Investigation of Other Design Class I Welds

The second part of the investigation consisted of a review of Design Class I piping welds not in the reactor coolant loop piping systems. In addition to the reactor coolant loop piping, other piping is required to have volumetric examinations of selective welds. Therefore only those selected welds would have had any exterior weld grinding for PSI/ISI preparation.

1. Visual Inspection

With the exception of the main steam and feedwater piping inside the containment isolation valves, the balance of the Code Class 2 piping is too small in diameter for internal grinding of welds. If external grinding complied with the applicable specifications and procedures, the welds would be flush with or higher than the adjacent pipe surface and adequate weld thickness would be maintained.

Sixteen Code welds were inspected by placing a straight-edge across the weld axially with the pipe to determine if the original grinding requirements were met.

Of the (16) welds inspected, 14 had exterior weld preparation (grinding). Four of the 14 welds were ground flush with the pipe base metal. There was no indication of weld grinding below the

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base metal. The other 10 had visible weld crowns (weld above base metal). The remaining two welds had not been ground for ISI. Visual examination results corroborate the construction quality records assuring that the wall thickness of other Design Class I welds meets thickness specifications.

Visual examination results corroborate the construction quality records assuring that the wall thickness of other Design Class I welds meets thickness specifications.

2. Ultrasonic Thickness Measurements

Because it was known that some interior examination and grinding was done on main steam pipe to steam generator nozzle welds and on main feedwater pipe to steam generator nozzle welds, three of ese welds were ultrasonically examined. These welds were: WICG 5-2 (F/W to SG 1-2), WICG 101-A1-4 (F/W to SG 1-4), and WICG -4 S/G to M/S 1-4). Two reactor coolant loop nozzle to branch pip welds were UT examined. These welds were Residual Heat Removal (RHR) pipe weld WIB - 227 and Pressurizer Surge Line Weld WIB - 65. All measurements were well above the minimum required thickness.

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TABLE V-1

SUMMARY OF MEASUREMENT RESULTS FOR INVESTIGATED WELDS

WELD NUMBER	(a) CODE ALLOWABLE MIN WALL	(b) UT MEAN WALL	(c) LOWEST UT MEAN MIN WALL	(d) MIN WALL BY MECH MEASURE
1-1	2.335	2.414	2.310	2.413
1-2	2.335	2.375	2.315	2.355
1-16	2.215	2.259	2.185	2.236
2-1	2.335	2.406	2.323	2.433
2-2	2.335	2.397	2.295	2.341
2-17	2.215	2.224	2.150	2.223
4-16	2.215	2.234	2.180	2.239
	Minimum Base Metal by UT			
1-11	2.495	2.564	2.470	2.660
3-9	2.495	2.638	2.465	2.560

Column Definitions:

(a) The minimum wall specified thickness.

(b) The mean wall thickness of the entire length of the weld, measured by ultrasonic techniques.

(c) The minimum average wall thickness of the three sets of measurments at the thinnest point. The readings recorded were the lowest recorded at each location.

(d) Wall thickness by mechanical measurement obtained at the minimum wall location.

Note: December 1982 and May 1983 UT data which showed low measurements on welds 2-9, 2-10, 2-13, 2-19, 4-1 and 4-17 could not be repeated in the most recent series of measurements.

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TABLE V-2
"MAX ID" - "MIN OD" COMPARISON REACTOR COOLANT PIPING

Reference: Westinghouse Spec. G676341
SW Fabrication Sheet Q89.7524

WELD	SPECIFICATION MAX** ID / MIN OD	MEASURED MAX. ID	MEASURED MIN. OD	MINIMUM WALL (2X) SPECIFICATION	MEASURED 2X MIN WALL
1-16	27.710/32.140	27.683	32.155	4.430	4.472
2-17	27.710/32.140	27.710	32.183*	4.430	4.473
		27.694*	32.140	4.430	4.446
4-16	27.710/32.140	27.691	32.168*	4.430	4.477
		27.660*	32.143	4.430	4.483
1-1	29.210/33.880	29.161	34.014*	4.670	4.853
		29.145*	33.970	4.670	4.825
1-2	29.210/33.880	29.175	33.885	4.670	4.710
2-1	29.210/33.880	29.173	34.030*	4.670	4.857
		29.147*	34.012	4.670	4.865
2-2	29.210/33.880	29.209	33.890	4.670	4.681
1-11	31.210/36.200	31.196	36.727*	4.990	5.531
		31.157*	36.477	4.990	5.320
3-9	31.210/36.200	31.098	36.533*	4.990	5.435
		31.030*	36.150	4.990	5.120

Notes

* - Value corresponding to Min. or Max. value.

** - Includes allowed 0.010 inches machining tolerance for ID.

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C. MICROMETER MEASUREMENT METHODOLOGY

1. Method

The method developed and used to determine pipe wall thickness using micrometer measurements supplemented by ultrasonic measurements is described below.

a. The outside and inside diameters (OD and ID) were measured with micrometers calibrated with standards traceable to NBS. To assure that the ID and OD measurements were obtained at the same location, plastic wedge UT damping location techniques were utilized.

b. Care was exercised by personnel making the measurements to assure that the maximum ID and minimum OD were obtained in the plane of measurement.

c. Measurements were made at the minimal wall location previously identified for each weld by ultrasonic means, in addition to a horizontal measurement and a vertical measurement at each weld.

d. The wall thickness was then determined as follows:

$$t = \frac{OD-ID}{2}$$

2. Justification

This method of measurement for piping welds of this size is justified as the most accurate and valid for the following reasons:

a. Care was exercised during pipe fabrication, fitup and assembly to assure that the pipe was concentric, not overbored, and that proper outside diameters were maintained. This was assured by the following:

- o The shop machining practices required the pipe to be counterbored within limits which assured minimum wall.
- o Strict tolerances were applied during both shop and field weld fit-up.

ATTACHMENT 3

b. All grinding performed on the exterior weld crowns (in preparation for inservice inspection) was controlled to assure that the crown was not removed to a depth below the adjacent piping OD.

c. Grinding on the weld ID was performed only to remove superficial blemishes. This is substantiated by the fact that the recent ID measurements show a reduction in ID from before the pipe was welded.

An alternative way to establish the wall thickness would be to utilize relative values of ultrasonically measured thickness. However, the accuracy of UT thickness measurement for piping welds of this type is considered less than the variation in wall thickness around the pipe measured at the spool ends prior to welding. Further, the scatter in the current UT measurements is larger than this variation. Reliable estimates of UT measurement accuracy are of the order of ± 5%, including the effects of both equipment accuracy and how personnel use it. Therefore, use of UT measurements to furthur adjust the micrometer measurement results cannot be supported.

D. CONCLUSIONS

Micrometer measurements are the most accurate and valid means to obtain minimum wall thickness for piping welds of this size and material. The micrometer measurements shown in Table V-1 and Table V-2 confirm previous data (prior to December 1982) that minimum wall thickness requirements were met.

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APPENDIX A-1

CHRONOLOGY OF MAJOR EVENTS RELATED TO DISCOVERY AND RESOLUTION OF MINIMUM WALL CONCERN

Date	Discussion
December 1982	NPO ISI/NDE group decided to make a series of thickness "profiles" axially across the reactor coolant loop piping girth welds to generate contour plots of the weld root counterbore surface. Thirty-one of the 56 RCS piping loop girth welds were profiled using an ultrasonic technique: WIB-RC-1-2, 1-4, 1-7, 1-8, 1-9, 1-11, 1-13, 1-16, 2-2, 2-5, 2-8, 2-9, 2-10, 2-13, 2-15, 2-17, 3-4, 3-7, 3-8, 3-9, 3-11, 3-13, 3-16, 4-2, 4-4, 4-7, 4-8, 4-9, 4-11, 4-13, and 4-16.
December 13, 1982	A review of the data revealed a point on the centerline of weld WIB-RC-2-17 and four points in the pipe downstream of weld WIB-RC-3-13 where the indicated thickness was suspiciously low. The recommendation of the ISI/NDE group was that the welds be reexamined after the RCS loops were drained[1] so that internal access could be provided.
December 17, 1982	Nuclear Plant Problem Reports were prepared for both welds to document the problem pending resolution.
January 7, 1983	An additional set of readings was made using both the original ultrasonic instrument and a newly acquired ultrasonic instrument which could be read to one more decimal place than the original instrument. The

(1) The reactor vessel was flooded from October 20, 1982 to January 15, 1983 for other work.

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Date	Discussion
	latter turned out to be effective on parent metal, but was unable to penetrate the weld material. On the retest, the suspect area adjacent to weld WIB-RC-3-13 measured thicker by both instruments. The retest of weld WIB-RC-2-17 continued to show a low reading at the weld centerline. The new instrument showed a previously unidentified area of concern 3" downstream.
February 16, 1983 (approximate)	An attempt was made to confirm the UT measurements on weld WIB-RC-2-17 by mechanical methods. The UT data was assumed accurate on the pipe metal, and a straight edge and depth gauge were used to measure the counterbore depth. The results were judged to be inconclusive.
March 28,	Using another new ultrasonic instrument and a 1" diameter, 2.25 MHz dual transducer, ISI/NDE personnel made another series of measurements on weld WIB-RC-2-17in the area of interest. A minimum thickness of 2.15" was indicated on the weld centerline.
May 5, 1983	NPO initiated a Nonconformance Report on weld WIB-RC-2-17.
May 6, 1983	The Plant Manager requested data on the entire circumference of weld WIB-RC-2-17, and to use a different team of examiners in order to obtain independent measurements.

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Date	Discussion
May 9, 1983	Data obtained on May 9 appeared to support previous data and a verbal notification was made to NRC Region V. A verbal commitment was made to examine the remaining welds to the extent feasible.
May 11, 1983	A Technical Review Group was convened to discuss the situation and lay out an investigation and corrective action program. This TRG included members from the plant staff, on-site QA, and on-site Westinghouse. This initial meeting was primarily to orient the TRG members to the issue.
May 13, 1983	A second meeting was convened including off-site personnel. At this meeting the TRG hypothesized that the cause was grinding preparation of the OD and ID weld surfaces to improve the finish for radiography.
May 20, 1983	Additional measurements were performed on UT thickness for weld WIB-RC-2-17 which supported earlier UT measurements.
May 23, 1983	LER 83-006 was submitted to the NRC discussing the findings on WIB-RC-2-17.
May 25, 1983	Beginning on May 25 and continuing on to June 2, plant staff ISI/NDE personnel took thickness measurements on welds located within the biological shield, which had not previously been examined.

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Date	Discussion
	Twelve additional welds were examined: WIB-RC-1-1, 1-17, 1-18, 2-1, 2-19, 2-20, 3-1, 3-17, 3-18, 4-1, 4-17, and 4-18.
June 3, 1983	Review of the previous data indicated that two cold leg welds, WIB-RC-2-19 and 4-17, each showed one small spot on the weld centerline which may have been below minimum wall, although the descrepancies were very small (_ 0.015"). The cause was assumed to be related to grinding.
June 22, 1983	It was determined that the crossover pipe and hot legs had higher minimum wall specifications than the cold leg (2.215" for the cold leg, 2.335" for the hot leg, and 2.495" for the crossover pipe). Prior to this time, the cold leg value had been used in identifying whether minimum wall specifications had been violated. This called into question seven additional welds: WIB-RC-RC-1-1, 2-1, 2-9, 2-10, 2-13, 3-9 and 4-1. An immediate telephone notification was made to the Manager, NPO. Although the results were recognized to be preliminary and unverified, the Manager, NPO instructed site personnel to inform the NRC site resident inspectors of this development to keep them appraised and to discuss reporting implications.

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Date	Discussion
June 23, 1983	Representatives of Region V arrive at Diablo Canyon to review the latest information regarding the reactor coolant piping welds.
June 24, 1983	At the routine weekly NRC exit interview the Plant Staff agreed to provide a written report regarding the reactor coolant piping welds to region V.
June 24, 1983	From the evening of June 24, 1983 thru July 1, 1983 an investigation team including project personnel and specialists from Bechtel and Westinghouse began a comprehensive investigation into the minimum wall concern. During this investigation the use of UT as a precise measurement technique was discarded due to its unreliability with regard to the piping in question. The investigation also confirmed that minimum wall thickness requirements had been met. The results of that investigation have been included in a written report to be submitted NRC Region V.

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APPENDIX - B

ENGINEERING EVALUATION

A. Scope of Evaluation

This appendix provides additional perspective on the significance of minor variations in piping wall thickness. This evaluation is not intended to depart from existing applicable code requirements, although certain later code concepts are used to describe conservative margins inherent in the existing design.

Minimum wall thickness is a code criterion used prior to fabrication to size pipe and pressure vessel wall thickness. The objective is to have a single number criterion which provides assurance that subsequent required calculations of operational stresses will show acceptable results. The concept includes inherent margins in material thickness to cover variations in manufacturing techniques, material properties, etc. The application of the concept for RCS loop pipe included additional margins.

An evaluation was done considering several aspects of design margin to assess the significance of potential minimum wall deviations of the size in question. The aspects considered included margin in the original piping stress analysis, and the margin inherent in the use of code, minimum allowable stress based on material properties rather than those measured for the actual pipe material. Also included was a parallel consideration of the recognized increases in recent years in published allowable stresses. Finally, consideration was given to the margin which could be shown by more explicit treatment of local geometry by current code techniques.

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B. Margin in Piping Stress Analysis

Along with the equations for minimum wall thickness, the USAS B31.1 code (B31.1) requires that additional evaluations be performed on a piping system to quantify the stress resulting from operating loads such as internal pressure, dead load, and seismic loads. The RCS loop piping was qualified for all of these loadings, and as summarized in Section 5 of the FSAR, the piping, met the applicable allowable limits in all cases. Meeting the stress allowables in the B31.1 code provides specific indication of the design adequacy of the reactor coolant piping; whereas the minimum wall thickness equations are used to size non-standard pipe and vessel products with some margin.

To demonstrate this design margin, an evaluation of the RCS loop piping was performed assuming a 10% reduction in the pipe wall thickness below the minimum defined by B31.1. For the evaluation, stresses of all of the highly loaded points in each of the RCS loop piping were re-calculated. The stresses resulting from the various conditions were combined in the same manner as the original evaluations and the results were compared with the allowable stresses.

The results of this evaluation are summarized in Table B-1. It demonstrates that for all loading conditions, the stresses calculated with an assumed reduction in wall thickness of 10% still meet the

ATTACHMENT 3

B 31.1 code allowables. In all cases there is sufficient additional margin to take a further reduction in the wall thickness without violating the code limits on calculated stress. This demonstrates the amount of extra material that the code minimum wall equation builds into the design. It also shows that the piping has sufficient wall thickness to operate safely with a significant reduction of material.

TABLE B-1
Summary of Piping Stress Analysis

Combined Stresses (FSAR)	FSAR Results (PSI)	Evaluation With 10% Wall Reduction (PSI)	Code Allowable Limit (PSI)
Normal Condit.	6,771	8,001	17,050
Upset Conditio	15,300	19,068	20,460
Faulted Condition	45,700	54,383	61,380

C. Margin in Allowable Stress Limits

1. Effect of Measured Material Properties

The reactor coolant loop piping is seamless extruded ASTM A376 Type 316 stainless steel. Transverse tensile tests were performed by Cameron Iron Works on both ends of each spool piece at ambient temperature and an additional tensile test was performed at 650° F on the pipe end exhibiting the lowest yield stress. Notarized test reports from Cameron Iron Works were examined and the results are reported in Table B - 2 below.

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TABLE B-2
Summary of Tensile Tests on Pipe Material

Test Temperature	Cameron Yield Test Results Average	Minimum
Ambient	40.9 ksi	32.5 ksi
650° F	23.5 ksi	20.5 ksi

The fittings were statically cast of ASTM-A351 CF8M by ESCO Inc. Tensile tests were performed at ambient temperature for each casting heat and the results are reported in Table B-3. There was also one certified test result at 650° F for one of the casting heats for a Diablo Canyon fitting which is reported below.

TABLE B-3
Summary of Tensile Tests on Fitting Material

Test Temperature	ESCO Yield Test Results Average	Minimum
Ambient	45.9 ksi	37.5 ksi
650° F	N/A	21.9 ksi

The pipe was designed to the requirements of USAS B31.1, 1955. The minimum wall requirements were set using the alternate formula from B31.1 which is based on the inside diameter of the pipe. The stress allowable used in the formula was obtained from Code Case N-7 using a design temperature of 650° F. For austenitic stainless steel such as that used in the reactor coolant loop piping, the stress allowable value for elevated temperature was set at 0.9 times the yield strength of the material at that temperature. Using the 0.9 factor and the actual yield

ATTACHMENT 3

data obtained for the A376 Type 316 pipe at 650° F, an "actual material" allowable stress of 18.45 ksi can be recalculated using the same B31.1 equation. If this "actual material" allowable is used in the alternate equation for minimum wall thickness, it results in approximately a 7% reduction in the Code minimum wall thickness. For the cold leg this would mean a reduction of the Code minimum wall thickness from 2.215 in. to 2.058 in. Similar reductions would be obtained for the other legs. Thus without any change in method, one can show a significant margin in the design of the RCS pipe . The reductions also show that if any areas of the pipe or weld were to be slightly smaller than the minimum original design they could be shown acceptable to the same Code requirements and they clearly would not have an adverse affect on the safe operation of the plant.

2. Evolution of Published Allowable Stress

In addition to the above information on actual material properties for the RCS loop piping, it should be noted that there has been a general increase in the published stress allowables for the stainless steels used in pipe and fittings over the last 20 years. When the A376 material was first included in B31.1, the allowable stress was set at 14.2 ksi. In later years it was increased to 15.9 ksi and by the time the piping was manufactured, a code case had raised it to 17.05 ksi, the value that was used to set the pipe thickness. The ASME code allowable for this material is currently 18.2 ksi. Use of this value requires

ATTACHMENT 3

that the material be nitrogen enriched. Discussion with W indicates that the pipe manufacturing included nitrogen enrichment. If the ASME Section III allowable stress value was used to calculate minimum wall requirements, it would result in close to a 7% reduction in requirements. This would be similar to the reduction obtained by using actual material properties. The forgoing indicates the conservatism in the B31.1 design basis for the RCS loop pipe.

C. Local Stress Effects

The ASME Boiler and Pressure Vessel Code Section III has long recognized that because of design considerations or manufacturing or fabrication processes, some localized areas of a pressure vessel may have thickness variations that produce stresses greater than the basic design limit on primary membrane stress. The ASME Section III addresses these areas with the local primary membrane stress limits defined in NB 1213.10. This paragraph allows for a 50% increase in the basic allowable stress limit of Sm. The increase is justified based on the fact that the stress is of a localized nature and does not effect the overall pressure boundary integrity. However, there are restrictions on the extent of the local area in the axial direction as well as restrictions on the axial proximity of adjacent areas of the same type. Applying the local membrane limits of ASME Section III NB 3213.10 to the reactor coolant loop piping and using the Sm allowable for this material given in ASME Section III, a reduction in the wall thickness of greater than 10% can be justified. The reduction would be limited to small areas but could extend completely around the inside surface or outside surface of the pipe. Areas such as this would not violate the intent of the code to limit the general primary membrane stress and protect the pressure boundary.

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The localized areas originally in question in this evaluation (based upon UT measurements) do not extend around the pipe and none of the suspect readings indicate anything close to a 10% reduction in thickness. All the areas identified meet the restrictions of ASME Section III for application of local limits. Thus, use of the latest availabe Code analysis would permit local "thin spots" both deeper and more extensive than any originally suspected with UT measurements.

ATTACHMENT 4

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

August 11, 1983

Thomas S. Moore, Esq., Chairman
Atomic Safety and Licensing
Appeal Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. W. Reed Johnson
Atomic Safety and Licensing
Appeal Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. John H. Buck
Atomic Safety and Licensing
Appeal Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Gentlemen:

The NRC Region V Staff has completed its inspection activities pertaining to Licensee Event Report 83-006 which identified a potential less than minimum wall condition in the reactor coolant system. This matter was the subject of PNO's V-83-22 and V-83-22A which were previously provided to the Appeal Board and parties. The Staff's inspection is documented in Report No. 50-275, transmitted under letter to the licensee dated August 5, 1983, copy attached. (The actual report number is 50-275/83-26, as indicated in the transmittal letter.)

Sincerely,

Lawrence J. Chandler

Lawrence J. Chandler
Deputy Assistant Chief Hearing Counsel

Enclosure: As stated

cc: See page 2

ATTACHMENT 4

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

AUG 23 1983

Docket No.: 50-275

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

FROM: Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: NRC REGION V INSPECTION REPORT 50-275/83-26 RELATING TO APPARENT LESS THAN MINIMUM PIPING WALL THICKNESS (Board Notification No. 83-124)

In accordance with NRC procedures for Board Notification, the following information is being transmitted directly to the Commission. By copy of this memorandum, we are also informing the Atomic Safety and Licensing Board and the Atomic Safety and Licensing Appeal Board.

The enclosed subject inspection report discusses NRC's review of concerns relating to the minimum wall thickness of reactor coolant system piping at Diablo Canyon. This has been the subject of previous Board Notification Nos. 83-72, dated June 20, 1983, Board Notification 83-83, dated June 24, 1983, Board Notification 83-89, dated June 29, 1983, Board Notification 83-96, dated July 11, 1983 and Board Notification 83-112, dated August 8, 1983.

for Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: See next page

8308040034

Contact: H. Schierling, ONRR
x27100

ATTCHMENT 4

AUG 05 1983

Docket No. 50-275

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

Attention: Mr. J. O. Schuyler, Jr.
Vice President, Nuclear Operations

Gentlemen:

Subject: NRC Inspection of Diablo Canyon Unit No. 1

This refers to the special announced inspection conducted by Messrs. J. Crews, A. Johnson, D. Kirsch, M. Mendonca, P. Morrill, J. Carlson, G. Hernandez and W. Wagner of this office during the period July 1-22, 1983, of activities authorized by NRC License No. DPR-76 and related to Licensee Event Report 83-006 which identified a potential less than minimum wall condition in the reactor coolant system.

Areas examined during this inspection are described in the enclosed inspection report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspectors.

No items of noncompliance with NRC requirements were identified within the scope of this inspection.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room unless you notify this office, by telephone, within ten days of the date of this letter and submit written application to withhold information contained therein within thirty days of the date of this letter. Such application must be consistent with the requirements of 2.790(b)(1).

ATTACHMENT 4

Should you have any questions concerning this inspection, we will be glad to discuss them with you.

Sincerely,

T. W. Bishop, Acting Director
Division of Resident, Reactor Projects
and Engineering Programs

Enclosure:
Inspection Report No. 50-275/83-26

cc w/enclosure:
W. A. Raymond, PG&E
R. C. Thornberry, PG&E
P. A. Crane, PG&E

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ATTCHMENT 4

U.S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No. 50-275/83-26 Docket No. 50-275

License No. DPR-76

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

Facility Name: Diablo Canyon Unit No. 1

Inspection at: Diablo Canyon Site, San Luis Obispo County, California

Inspection conducted: July 1-22, 1983

Inspectors:

G. Hernandez, Reactor Inspector	8/5/83 Date Signed
W. Wagner, Reactor Inspector	8/5/83 Date Signed
J. Crews, Technical Assistant to the Administrator	8/5/83 Date Signed
A. Johnson, Enforcement Officer	8/5/83 Date Signed
M. Mendonca, Resident Inspector	8/5/83 Date Signed
J. Carlson, Senior Resident Inspector	8/5/83 Date Signed
P. Morrill, Reactor Inspector	8/5/83 Date Signed

Approved by: D. F. Kirsch, Chief, Reactor Projects Sec. No. 3 — 8/5/83 Date Signed

Summary:

Inspection during the period of July 1-22, 1983 (Report No. 50-275/83-26)

Areas Inspected: Special announced inspection by regional and resident inspectors of the circumstances and facts relating to the licensee's discovery of apparent less than minimum code allowable wall thickness at or adjacent to welds in the reactor coolant system (RCS), which was initially identified in Licensee Event Report (LER) 83-006. The inspection involved 184 inspection-hours by seven NRC inspectors.

Results: No items of noncompliance or deviations were identfied.

ATTCHMENT 4

DETAILS

1. Individuals Contacted

   a. Pacific Gas and Electric Company (PG&E)

      G. Maneatis, Executive Vice President Facilities and Electric Resources Development
      J. Schuyler, Vice President Nuclear Power Generation
      R. Etzler, Field Construction Manager
      D. Rockwell, Assistant Project Superintendent
      R. Twiddy, Site Quality Assurance Manager
      J. Shiffer, Manager Nuclear Operations
      W. Raymond, Technical Assistant to the Vice President, Nuclear Power Operation
      F. Dodd, Senior Metallurgical Engineer
      S. Skidmore, Manager of Quality Assurance

   b. Bechtel Corporation (Bechtel)

      C. Dick, Project Management Team Member
      H. Friend, Project Completion Manager

2. Background

   On May 9, 1983 Pacific Gas and Electric Company (the licensee) representatives called the Region V staff to report that ultrasonic examination (UT) of RCS Weld Number WIB-RC-2-17 (in the Unit 1 RCS cold leg of loop No. 2) might be below specified minimum wall thickness. The licensee personnel committed to examine the remaining RCS girth welds in Unit 1 at that time. This telephone call was followed up with a LER (No. 83-006) dated May 23, 1983.

   On June 22, 1983 a member of the licensee's staff verbally informed the NRC that based upon additional ultrasonic measurements it appeared that minimum wall requirements might not be met in approximately nine additional weld areas. Members of the Region V inspection staff arrived at the Diablo Canyon site the following day and examined the latest information related to this issue. At the NRC exit meeting on June 23, 1983 the licensee committed to conduct a detailed investigation and to submit a report documenting these activities. This report (dated July 1, 1983) was submitted to the Region V office by PG&E letter "Schuyler to Martin" dated July 5, 1983.

   On June 29, 1983 the NRC contracted with Parameter, Inc. to conduct independent UT examinations of the subject RCS welds and to assess the adequacy of this technique for thickness measurements in this piping. During the week of July 5, 1983, three Parameter, Inc. personnel conducted these examinations which were documented in a report (dated July 14, 1983) and forwarded by a letter "Foley to Morrill" dated July 14, 1983.

Subsequently, Region V conducted a public meeting on July 14, 1983 in the Region V offices to discuss the licensee's July 1, 1983 report with members of the licensee's staff, members of the Independent Verification Program, representatives of the Governor of the State of California, and representatives of the joint intervenors. A transcript of that meeting was taken which was subsequently distributed to all parties to the Diablo Canyon licensing proceedings along with the Parameter Report, dated July 14, 1983.

Examinations of license records and measurements in progress had been examined on June 23-24, June 29 - July 1, July 7-8, July 12-13, and July 20-21, 1983 by the Region V staff. This report documents these inspection activities and the conclusions of the Region V staff.

3. Documents reviewed by the NRC included:

Westinghouse Specification No. G676341, Rev. 1, dated 4-11-67 "Reactor Coolant Seamless Pipe"

Westinghouse Specification No. G676342, Rev. 2, dated 4-6-67 "Reactor Coolant Cast Fittings"

Westinghouse Specification No. 676496, Rev. 0, dated 3-13-67 "Reactor Coolant Piping - Field Erection"

American Standard ASA B-31.1, 1955 Edition, Section 122 "Thickness of Pipe"

PG&E Deviation Report No. 39, written 10-7-70 and closed on May 5, 1971, to evaluate the effect of pipe spool marking depth on minimum wall thickness requirements

PG&E Procedure TG 83-01, Rev. 0, dated 6-29-83 "Temporary Procedure - RCS Piping Wall Thickness Measurements"

Mechanical measurement data for welds 1-1, 1-2, 1-4, 1-8, 1-11, 1-16, 2-1, 2-2, 2-17, 3-9, 3-13, 4-2, 4-16

PG&E Specification No. 8752 for Field Erection of RCS Piping (Wismer/Becker Specifiction)

PG&E Procedure N-UT-2, Rev. 0, dated 1-1-83, "UT Thickness Measurement Examination Procedure"

Southwest Fabricating & Weld Co. As-Built Drawings for pipe spools containing welds 1-1, 1-2, 1-11, 1-16, 2-1, 2-2, 2-17, 3-9, 4-16

Cameron Iron Works Data Sheets documenting minimum outside diameter, maximum inside diameter, maximum and minimum wall thickness measurements for pipe involved in RCS welds 1-1, 1-2, 1-11, 1-16, 2-1, 2-2, 2-17, 3-9, 4-16

Southwest Fabricating & Welding Company drawing no. SO.7524 Sheet Q giving details of shop and field weld tolerances for machining

ATTCHMENT 4

4. Evaluation of Reactor Coolant System (RCS) Piping Wall Thickness

(a) Examination of Shop Manufacturing and Fabrication Records

The inspector reviewed records generated during fabrication of the reactor coolant loop (RCL) piping. This was to determine the adequacy of the quality assurance program during fabrication, and to establish whether or not minimum wall was maintained prior to the pipe being received at the jobsite. Specific records reviewed and the general results are described below.

Westinghouse Equipment Specification G-676341, "Reactor Coolant Seamless Pipe" listed the requirements that the suppliers (vendors) were responsible to meet during fabrication of RCS piping; this included dimensional requirements for inside and outside diameters (I.D. and O.D. respectively), and minimum wall thicknesses.

Cameron Iron Work Material Certifications provide dimensional measurements of the O.D., I.D. and wall thickness. Based on this data the inspector verified that the dimensional requirements of Westinghouse Specification G-676341 were met. These verifications were made for the hot leg, crossover leg and cold leg piping.

Southwest Fabricating and Welding Company (Southwest) as-built drawing for fabricated spool piece number PGE DC-663219-167-3 was examined. The inspector verified that minimum wall met the drawing requirements and was correctly approved for construction.

The inspector also reviewed a Southwest document addressing final inspection, prior to shipment, of 8 pipe sections and 4 elbows. This document stated that "dimensions were checked throughout and were within allowable tolerances".

Westinghouse records show that numerous inspections were performed by Westinghouse of their reactor coolant piping vendors. One memo stated that mechanical readings at the shop and field are compatible.

PG&E weekly inspection reports were written by PG&E inspectors during fabrication at Cameron and Southwest. These reports indicate that RCS pipe dimensions were checked and found acceptable.

PG&E QA Audit of Southwest verified that as-built dimensions conform to appropriate specifications.

(1) Cameron Iron Works, Inc.

During the manufacturing process at Cameron Iron Works Inc., measurements were taken and documented on each pipe section and heat number manufactured. These measurements consisted of

outside diameter, minimum and maximum inside diameter, and minimum and maximum wall thickness. The measurements were taken at distances of one inch and two feet from each end of the pipe section.

Westinghouse E Specification No. G-676341 specified acceptance criteria for maximum and minimum inside diameter, minimum wall thickness and minimum outside diameter for each size of pipe manufactured (i.e., for nominal inside diameters of 27.5 inches, 29 inches and 31 inches).

The inspector examined the data documented by Cameron Iron Works for the pipe sections containing weld numbers: weld 1-1 (field weld), weld 1-2 (shop weld), weld 2-1 (field weld), weld 2-2 (shop weld), weld 1-11 (shop weld), weld 1-16 (shop weld), weld 2-17 (shop weld, weld 3-9 (field weld), and weld 4-16 (shop weld). The data recorded and documented by Cameron demonstrates compliance with dimensional acceptance criteria specified in Westinghouse E Specification No. G-676341.

The inspector also performed independent calculations of wall thickness remaining based on counterboring for the shop and field welds. The counterboring and shop welding was performed by Southwest Fabrication and Welding Company (see next subsection).

This calculation was performed using the following equation:

$$\text{Wall Thickness} = \frac{\text{(Minimum Outside Diameter)} - \text{(Maximum Specified Inside Diameter)}}{2}$$

Data for the minimum outside diameter was obtained from data recorded by Cameron Iron Works. The maximum inside diameter data was obtained from Southwest Drawing No. 80.7524, Sheet Q and Westinghouse E Specification No. G-676341. The Southwest drawing specifies weld preparation dimensions, counterbore dimensions, and tolerances.

The results of the inspector's calculations indicated that minimum wall thickness criteria were complied with in all cases. The results of these calculations were compared to the mechanically measured minimum wall thickness presented in Table V-1 of the PG&E Report on "Investigation of Reactor Coolant Pipe Weld Thickness at Diablo Canyon", transmitted to the NRC Region V on July 5, 1983. The results of the independent calculations, performed using worst case conditions, appeared consistent with the wall thickness obtained and documented by PG&E, and demonstrated compliance with the Westinghouse minimum wall thickness acceptance criteria.

(2) Southwest Fabricating and Welding Co.

This company machined the counterbore on the pipe sections manufactured by Cameron and completed the shop welds. The

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documentation indicates that the machining operations were performed as specified on Southwest detail sheet Q. Southwest has documented, by letter to Westinghouse, dated July 19, 1983, that wall thickness was checked with micrometers to verify that the minimum thickness specified on the detail sheet and sheet Q was satisfied and, further, that since this check was only to verify that thickness was adequate, actual thicknesses were not recorded. Southwest also states, in that letter, that in-service inspection preparation of welds was performed on the shop welds of the 31 inch inside diameter crossover legs while all other shop welds were furnished in the "as-welded" condition.

(3) Source Inspection Document Review

The inspector examined representative records of source inspections, performed by PG&E, of Southwest Fabricating and Cameron Iron Works. These records documented that PG&E inspectors made dimensional spot-checks and verified wall thicknesses of selected pipe spools.

The records documented that one pipe (4153 cold leg) was found to be less than minimum wall thickness in one location. It was subsequently repaired by welding and reinspected by Cameron.

.stinghouse Electric Company

Nuclear Steam Supply System supplier, Westinghouse furnished .e RCS piping including a quality control release form with ach piece. On these forms Westinghouse documented acceptance of dimensional records. However, the dimensional records were not included with the documentation package on shipment. PG&E, therefore based their acceptance on the documentation supplied by Westinghouse indicating that Westinghouse had accepted the dimensional records.

(b) Examination of Records of Field Erection and Welding of Reactor Coolant System Piping

Records of the erection and welding of the reactor coolant system (RCS) piping for Unit 1 were examined. Specific records which were examined included documentation for field weld numbers WIB-RC-1-1, 2-1 and 3-1.

The records indicated that weld fitup was examined and "signed-off" by three parties (Wismer & Becker, the California Code Inspector, and PG&E) for weld number 2-1. For Welds 1-1 and 3-9 the records indicated an additional sign-off of weld fitup by Westinghouse.

The records also indicated that measurements were recorded by Wismer & Becker inspectors of the pipe wall thickness after weld fitup. These measurements were recorded for each quadrant of the weld. According to PG&E General Construction Department personnel,

these measurements involved the placement of a mechanic's straight edge axially spanning the weld preparation area, with the depth of pipe wall determined by measurement from the straight edge to the top surface of the weld preparation land area at the root of the weld. The records indicated (with the exception of two quadrant measurements for weld 2-1, where the recorded value was not legible) wall thickness in each instance to be in excess of the minimum design wall thickness.

The inspector performed an independent calculation, using the data described above and the minimum allowed land thickness from drawing Sheet Q, to verify the wall thickness at the measured locations. The minimum allowed land thickness was 0.055 inches. Summing these dimensions indicates that the wall thickness remained above the specified minimum wall thickness in all locations measured by Wismer and Becker.

The records examined also included the logs of PG&E inspectors involved with inspection and surveillance of grinding of finished welds in the RCS during the period of early March 1975 through mid-May 1975. These records indicated essentially daily surveillance over this grinding activity. The records also contained acceptance criteria, established by PG&E's Engineering Department, for the grinding of the outside diameter of the welds. These criteria included the requirement that "...weld crowns should be ground smoothly down to the height of +1/16 inch max.,-0 inch min. from the adjacent pipe surface level...." The criteria also specified that grinding should be confined to the weld metal. The records indicated that this grinding was performed in preparation for ultrasonic inspection of the welds.

(c) Examination of Pacific Gas and Electric Company Deviation Report No. 39.

The inspector examined the subject deviation report. The report documents that, following receipt of the RCS piping spools at the warehousing area, PG&E became concerned that the observed depth of spool identification marking indentations may infringe on specified minimum wall thickness requirements.

Using ultrasonic wall thickness measurements PG&E rejected spool 1-1. The Westinghouse site manager made arrangements to measure wall thickness using state of the art optical and ultrasonic equipment. Optical measurements verified that wall thickness exceeded the specified minimum.

During these measurements a conflict developed between the data obtained ultrasonically and optically. The theory was advanced that the Type 316 SST material, used for the RCS pipe, was not homogenous in all heats thus causing the ultrasonic wave velocity to vary between heats.

When the UT instrument was calibrated to a known thickness of a specific heat number the material thicknesses (measured

ultrasonically) exceeded minimum wall specifications. However, data taken indicate that, even by calibrating the instrument to a specific heat, a difference of 2.0% to 4.5% existed between micrometer (mechanical measurement) data and UT data.

Westinghouse conducted an evaluation of the UT technique applied to extruded stainless steel material. The conclusions were: (a) the UT equipment used initially by PG&E was not accurate in the 2.5 inch range; and (b) the UT equipment <u>must</u> be calibrated on the same heat number (material) as the piece to be tested. The findings of this evaluation indicate that a sonic velocity difference of almost 4% existed from one heat number to the other. Furthermore, discussions with an industry expert indicated that sonic velocity variances of up to 10% had been observed, mainly due to the differences experienced by material in the heat treatment and stress level.

Examination of this Deviation Report indicates that Ultrasonic examination techniques were not a sufficiently reliable means for measuring wall thickness in this type of material.

(d) <u>Examination of Ultrasonic Test Procedure</u>

The inspector examined PG&E procedure no. N-UT-2, Rev. 0, dated January 1, 1983, titled "UT Thickness Measurement Examination Procedure." This procedure was utilized in the calibration of instruments and examination of the RCS piping.

The calibration section requires that an appropriate calibration block be used of the same material (material having similar chemical analysis, mechanical properties and microstructure) and product form (material manufactured by casting, rolling or forging for plate, etc.) as the material to be measured.

Furthermore the calibration section requires (following calibration to a step wedge), that the response of an intermediate thickness should not deviate by more than 1% of the range under test.

Discussions with licensee representatives involved in the UT process indicated that compliance with the above 1% criteria could not be consistently obtained.

The inspector questioned the validity of the ultrasonic measurement technique as applied to the RCS piping for the following reasons:

- The response of the UT instrument to an intermediate thickness could not be consistently maintained within 1% of the range under test.
- Data obtained in the resolution Deviation Report No. 39, in 1971, indicated that the ultrasonic method of wall thickness measurement was not reliable when applied to RCS piping.

- The material used in the calibration of the instrument potentially had a far different microstructure than the material under test. The sensitivity of the UT technique to different material heat numbers was amply demonstrated in the resolution of Deviation Report No. 39 in 1971.
- Use of a step wedge for calibration doesn't adequately provide a product form calibration standard since the material under test had a curved surface.

For the above reasons the inspector considers that the licensee had inappropriately placed a high degree of reliance on the RCS thickness measurements obtained by the ultrasonic nondestructive testing methods utilized in the identification and verification of the potential deviations from specified minimum wall thickness criteria.

(e) Verification of Mechanical and Ultrasonic Measurements

On July 1, 12 and 13, 1983, mechanical and ultrasonic measurements were observed and verified by an NRC inspector on five Reactor Coolant System girth welds. The licensee had previously identified nine Reactor Coolant System girth welds as being potentially below minimum wall in certain areas.

Mechanical measurements were performed on the inside and outside diameters of each weld area. The measurements were made at the horizontal and vertical axis of the pipe weld area, at the licensee identified minimum wall area, (as determined by ultrasonic examination) and at points selected by the NRC inspector. Ultrasonic thickness measurements were then performed for comparison with the mechanical measurements. The welds examined were welds Nos. 1-1, 2-1, 2-2, 2-17 and 3-9. For weld no. 3-9 the minimum wall point was determined to be in the heat affected zone of the weld.

The inspector observed that while the ultrasonic thickness measurements of the vertical and horizontal axis of each weld were consistent with previous licensee ultrasonic data, in most cases the previously identified licensee minimum wall point could not be relocated. In almost all cases a new minimum wall point was recorded.

The following tabulation is a comparison of minimum wall mechanical measurements obtained during the NRC inspection, with the data reported by the licensee in their report entitled, "Investigation of Reactor Coolant Pipe Weld Thickness at Diablo Canyon", dated July 1, 1983.

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Weld No.	Required Minimum Wall Thickness	NRC Observed minimum wall data	PG&E Reported minimum wall data
1-1	2.335	2.382	2.413
2-1	2.335	2.405	2.433
2-2	2.335	2.342	2.341
2-17	2.215	2.222	2.223
3-9	2.495	2.503	2.560

The mechanical measurements observed and verified by the NRC inspector indicated that the wall thickness was above minimum wall requirements for the five welds measured. The variations in the minimum wall data between the NRC and the licensee obtained data is attributed to the different persons taking the data, the cramped quarters involved in obtaining the data, and the difficulty of relocating the same spot on the RCS piping.

(f) Analysis of Mechanical Wall Thickness Measurements

The inspector performed an independent conservative verification of wall thickness by using the PG&E measurements of minimum outside diameter and the maximum allowed inside diameter (Drawing Sheet Q) to verify adequate wall thicknesses, in accordance with the following equation.

$$\text{Wall Thickness} = \frac{OD - ID}{2}$$

Where

OD = minimum recorded outside diameter
ID = maximum allowed inside diameter at bottom of weld land on counterbore (reference Drawing Sheet Q)

At one location at weld no. 3-9, the minimum measured outside diameter (at location 30°) was reported to 36.138 inches which was less than the 36.20 inches as specified in Westinghouse Specification No. G-676341. However, the mechanical measurements taken by the licensee and the NRC inspector (at this location) indicated that the minimum wall was 2.503 inches, which is greater than the required wall thickness of 2.495 inches. At another location on this pipe the licensee's data identified another point on the outside diameter which appears to be less than the required outside diameter. This point was reported as 36.167 inches, however the mechanical measurements at that location indicated a minimum wall thickness of 2.561 inches.

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5. Open Item

As a separate issue, the licensee has been requested to provide additional information regarding any instances where ultrasonic wall thickness measurements were used for quality acceptance in stainless steel piping systems. This area will be further examined in a subsequent inspection (50-275/83-26-01).

6. Conclusion

Based on the foregoing information the inspectors concluded that there is reasonable assurance that RCS piping wall thickness meets or exceeds design requirements.