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DENTON,H.R. Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards responses to allegations contained in affidavits attached to Joint Intervenor 840611 reply to answers to Joint Intervenor motions to reopen design & const QA issues.

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J. O. SCHUYLER
VICE PRESIDENT
NUCLEAR POWER GENERATION

June 26, 1984

PGandE Letter No.: DCL-84-239

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Dear Mr. Denton:

Enclosed are Pacific Gas and Electric Company's responses to the allegations contained in affidavits which were attached to the Joint Intervenors' June 11, 1984 Reply to PGandE and NRC Staff Answers to Joint Intervenors' Motions to Reopen Design and Construction Quality Assurance.

For each allegation, a "JIR" (Joint Intervenors' Reply) number has been assigned and the responses are numbered from JIR-1 to JIR-81, as shown in the enclosed index. All of the responses have been verified and the professional resumes of the verifiers have either been previously submitted or are attached to the enclosure. Additional responses to recent allegations from the Government Accountability Project which have not been specifically addressed previously will be submitted within the next week.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

B. Koch
for J. O. Schuyler

Enclosure

cc: D. G. Eisenhut
J. B. Martin
H. E. Schierling
Service List

A001
1/40

PGandE Letter No.: DCL-84-239

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It is alleged that:

1. The second paragraph on page four (4) states, "Supervisors may have relied on the administrative system too much, to assure training, and did not, in some cases, track the satisfactory completion of training. Supervisors also were not required to document how employees were informed of procedure revisions. This situation is being corrected."

In my opinion, to support that conclusion several questions must be addressed: 1) How are the employees to be instructed, verbal transmission by group leaders (which can vary), verbal transmission by section lead, or by individual distributed written memorandum? 2) Will there be a signature slip requirement for verbal and written instruction? 3) How will people who are absent from oral session be followed up on to insure [sic] that they are informed upon return? 4) Concerning the signing of signature slips, who will be required to sign indicating that an employee has been trained. Will it be the person trained or his supervisor? 5) If questions are raised during training which can not be answered during the session, how will these be tracked and responded to so that all trained employees are aware of the question and answers? 6) How will questions be addressed which concern the validity of the practices being taught? 7) Will employees who raise serious questions be part of team reviewing the procedure or technical issue to which he [sic] addressed?

These types of questions should be addressed for the following reasons: 1) They are necessary to ensure that all members of work force receive the required information. 2) That isn't the way it was done in the past, which helps to explain the breakdown in design control I exposed. The NRC should evaluate PG&E's implementation of corrective action in light of these criteria, to assure that similar errors do not recur. (6/1/84 Stokes Aff. at 1-2.)

In this allegation, Mr. Stokes merely suggests teaching methods and procedures which could be employed in applicant's revised training program at the On-site Project Engineering Group (OPEG). While these

suggestions may have merit, they in no way constitute requirements, and they are not the only way a training program may be carried out.

In viewing any training program, its purpose and the prior competence of those being trained must be kept in mind. At OPEG, the program is an orientation for experienced professional engineers. It is designed to introduce them to the engineering process at Diablo and to inform them how to work within that process. It is not, as Mr. Stokes implies, basic engineering training.

There was nothing false or misleading about applicant's statements concerning its training program.

JIR-2

It is alleged that:

2. The bottom of page five (5) includes the following claim. "Since the sampling of small bore analyses performed by OPEG engineers did not indicate any correlation between training and frequency of these minor errors there is no basis to expect a correlation in any other engineering area, especially large bore where the frequency of detected errors has been very low (2-4%)."

This statement is in my opinion a prime example of continuing to ignore the cause of a problem. My reasons for this statement are. 1) I have alleged in several past statements that an additional possible reason for many of the so-called errors was that they were the results of group leaders [sic] instructions. Group leaders are part of the same overall management leadership, which forms a link between responsibility for both small bore and large bore problems. 2) I have also alleged that training was non-existent or one on one (group leader to engineer). The lack of group instruction was the missing factor which allowed the group leaders to get away with instructing individuals differently. At present, these faulty instructions are included in the categories of technical errors or engineering judgment, rather than credited as the training problems they represent. 3) The breakdown in training was generic. Everyone was on their own, and in all areas of the job. As a result, there was no control group (well-trained employees) which could be used to assess the performance of poorly trained employees. In my opinion, PG&E's self-exoneration is impossible. (6/1/84 Stokes Aff. at 2-3.)

As discussed in the response to allegation JIR-1, a training program was in place and functioning at OPEG. The procedural orientation provided by this training, coupled with the pre-existing technical

qualifications of engineers which were verified prior to their employment, and augmented by on-the-job training given to familiarize new arrivals with specific project methods and criteria, resulted in pipe support designers suitably trained for their assigned work. Mr. Stokes' alternative claim that group leaders instructed engineers under their direction to make minor errors is ludicrous. The fact that certain assumptions, judgments, and analytical techniques were used which were unfamiliar to Mr. Stokes does not detract from the validity of these methods and certainly does not constitute an error. Such practices as modeling of thermal gaps, use of joint release in analysis of pipe support structures, and others which have been widely discussed elsewhere herein and in previous responses, have each been demonstrated to be clearly appropriate techniques for pipe support analysis despite Mr. Stokes' obvious unwillingness to accept them.

Additionally, this position has been further strengthened by reviews conducted in conjunction with NRC licensing conditions. Hundreds of supports have been subjected to the most intense detailed reviews. In no case has a modification been required to meet licensing conditions. This effort has been monitored and closely reviewed by the NRC staff.

JIR-3

It is alleged that:

3. On the bottom of page six (6) and the top of page seven (7) PG&E states, "We share the inspector's opinion that audit findings should receive proper evaluation and be resolved in a timely manner. These observations have no significance for piping design activities or for low power testing or commerical operations."

The lack of timely solution of many problems, some for years, permitted similar problems which developed to go undetected and uncorrected. These have never been documented. They still are not fixed. They have been institutionalized on-site as "old work" which has been defined out of the quality assurance system on a generic basis. They are significant in that they bear on the plant's ability to operate safely. The last line quoted above is a false statement before the plant goes commercial, there should be a full review of all work missed by the Audit program. Until such corrective action occurs, the quality of what we're left with in relevant safety systems will be indeterminate. (6/1/84 Stokes Aff. at 3-4.)

Mr. Stokes' allegation stems from his continuing habit of taking sentences out of context from the PGandE letter. The two sentences pointed to by Mr. Stokes are from the Summary Response Section of the letter, whereas the information in support of the conclusion claimed to be false, is found in the Detailed Responses on pages 7-11.

As noted in the quoted response, the reason PGandE has concluded that the timing of the resolution audit findings is not significant is because of the implementation of a system that reviews the findings

for significance, prioritizes the findings for resolution, and resolves the findings according to that priority.

Although it is impossible to determine what matters Mr. Stokes is complaining about, at the time he was employed at Diablo Canyon there were some audit findings that had not been closed for a long period of time. However, these findings had been prioritized for completion corresponding to specific Unit 1 milestones such as "prior to fuel loading" or "prior to power ascension" or "prior to commercial operation." Each and every finding has been timely resolved as each milestone has been reached. All quality problems that have been identified must be closed prior to operation or sooner if priority evaluation indicates.

Finally, Mr. Stokes proposes a full review of all work missed by the Audit Program, based upon vague, general references to the lack of timely resolutions of QA audit findings resulting in undocumented and uncorrected problems of a "significant," but unspecified, nature that bear on the plant's ability to operate safely. In effect Mr. Stokes is demanding a 100% QA Review of the plant (presumably both design and construction) without providing any basis for doing so.

JIR-4

It is alleged that:

On page thirteen (13) paragraph no.1 PG&E states, "ANSI N45.2.12-1977 (Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants) states that the audited organization shall respond as requested by the audit report but does not require 'documented justification' for delays."

In response to the above statement, I would like to quote from 10 CFR 50 Appendix B, XVI. "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." (Emphasis added) In my opinion the word promptly requires that if a delay occurs in a scheduled audit there should be documented justification. (6/1/84 Stokes Aff. at 4.)

Not only does Mr. Stokes add new meaning to the word promptly (i.e., "promptly" means "documented justification for delay"), he confuses the distinction between the timing of an audit and the timing of responses to the audit and confuses corrective action in Criteria XVI with audits in Criteria XVIII.

Mr. Stokes asserts, in the concluding sentence, that there should be documented justification when "a delay occurs in a scheduled audit. . . ." However, the DCP response Mr. Stokes takes issue with concerns in delays in responses to audit findings and not delays in the conduct of the audit itself. As previously pointed out in the DCP response, documented justification for delays in response to audit findings is not required by regulation or the DCP QA Program. The follow-up measures taken by the auditing organization (DCP Quality

Assurance) to achieve resolution of audit findings conform with the NRC-endorsed ANSI Standard N45.2.12-1977, and satisfy in part the "measures . . . established to assure that conditions adverse to quality . . . are promptly identified and corrected" as required by 10 CFR 50, Appendix B. To further provide for "prompt correction," the required action to correct any audit findings is established and agreed to by the audited organization even before the audit is formally concluded. This agreement allows corrective action to proceed immediately.

Setting aside Mr. Stokes' apparent confusion between "audits" and "audit findings," the allegation, as written, is an opinion that is not supported by regulatory requirements. ANSI N45.2.12-1977 does not require documented justification for delays in a scheduled audit. In the case of DCP Project Audits (the subject of the DCP response), the Bechtel Quality Assurance Department Manual provides as follows:

> The audit schedule may be adjusted during the quarter, if required by unanticipated conditions. Audits may be postponed if there is insufficient activity in the area. There is no requirement for documented justification of such schedule adjustments. Further, such adjustments do not affect the pre-established Master Audit Plan which provides for audits early in the life of the audited activity and for auditing of applicable elements of the QA Program at least annually.

Under standard and accepted QA interpretation, this allegation has no merit whatsoever.

It is alleged that:

5. On page nineteen (19) PG&E states, "The Project acknowledges that out-of-date procedures were in some controlled manuals at OPEG."

This statement is misleading, because it is incomplete. What about uncontrolled copies of the manuals? While I was in OPEG, the group leaders at times copied various sections of controlled manuals and gave them to us to use. At times, we discovered that a document existed which was relevant to the work which we were performing and we made our own copy. Have the uncontrolled documents have been [sic] removed from use? I know that in light of the heavy production drive which we were required to live up to, that we needed our own controlled documents in order to avoid spending unnecessary time searching out a document in someone elses [sic] possession. I would like to quote a line from 10 CFR 50 Appendix B, II. QUALITY ASSURANCE PROGRAM. "Activities affecting quality shall be accomplished under a [sic] suitably controlled conditions." I believe that the practices above are not in compliance with 10 CFR 50 Appendix B, II. (6/1/84 Stokes Aff. at 4-5.)

Mr. Stokes contends that the response to Mr. Yin's observation is misleading because it does not address uncontrolled copies of manuals. However, Mr. Yin's observation was that "Engineers were using out-of-date procedures for performing their work." (Draft Report pp. 10-12 S/B.) At the time of his inspection, Mr. Yin did not have any observations with respect to uncontrolled copies of manuals. Yet, Mr. Stokes claims PGandE's response is misleading because PGandE responded fully to the observations of Mr. Yin and did not address Mr. Stokes' allegations in that response.

More importantly, although Mr. Stokes poses the question as to the use of uncontrolled manuals, his examples do not involve problems

with manuals. While he may be technically correct in his assessment that a copy of a section of a controlled manual may create an uncontrolled document, it does not follow that there was anything improper about the practice. The difficulty is that Mr. Stokes misleads the reader by implying that procedures changed so often that making a copy of a procedure would inevitably lead the user to rely upon an out-of-date procedure in the event he used the copy a second time. This observation is misleading and totally without factual basis.

It has been acknowledged that out-of-date procedures were in the possession of some OPEG pipe support engineers. In some cases this included uncontrolled copies of controlled procedures. However, pipe support engineers were specifically directed to rely only on current, controlled copies of procedures in completing their work. Sufficient copies of controlled procedures were available for reference use in the OPEG work areas. Mr. Stokes' intimation that each engineer must continually refer to a procedure manual in the course of his work ignores the fact that, because of the basically repetitive nature of this work, engineers soon gain a working knowledge of procedures affecting their work and consequently need not refer to the procedures frequently. The number of controlled procedures made available to OPEG engineers afforded convenient reference access, while at the same time avoiding unnecessary complications in document control due to distribution of more copies than necessary to

accomplish the work. Contrary to Mr. Stokes' allegation, this practice does not violate any requirements of 10 CFR 50, Appendix B, and is not inconsistent with industry practice.

JIR-6

It is alleged that:

6. On page twenty (20) PG&E's statements raise serious questions. "Apparently the inspector believes that each OPEG engineer required his own controlled documents. Project documentation shows that, depending on the size of the group, there were never less than three and as many as eleven sets of controlled piping procedures assigned to the OPEG Stress Group. This constituted a sufficient number of controlled procedures for use by OPEG engineers."

From personal experience and discussion with OPEG engineers, I can testify that PG&E's conclusion is absurd. Under the conditions on-site--20 to 30 engineers trying to use three sets under a very demanding schedule. As a result, they made do without control documents. They used uncontrolled documents instead.

PG&E's assertion violates industry accepted engineering procedures. It is industry practice to assign each engineer his own controlled documents. At Bechtel's Gaithersburg office, I was given controlled documents within two weeks of my beginning work on the DAVIS-BESSEE [sic] project. When I went to work for Nuclear Services Corporation on the Zimmer project, we all received sets of control documents within two weeks of starting. I believe this is done for two reasons--1) to be in compliance with 10 CFR 50 Appendix B Criterion VI. DOCUMENT CONTROL; and 2) because production is very important to meet scheduled completion dates and time wasted looking for information should be avoided by providing each engineer with his own project documents. (6/1/84 Stokes Aff. at 5-6.)

Mr. Stokes asserts that from "personal experience" the number of controlled documents assigned to the OPEG stress group was insufficient. Mr. Stokes was never assigned to the OPEG stress group and his specific "personal experience" in the matter is therefore

vicarious, at best. In any case, this allegation is based on Mr. Stokes' basic misconception that each engineer must continuously refer to the procedure documents during the course of development of each calculation he originates or checks in order to remember how to correctly do the work. In fact, the calculational process involved in pipe stress and pipe support design is largely repetitive from one calculation to the next. It frequently involves iterative steps which are essentially identical. As a practical matter, an engineer gains a working familiarity with applicable procedures and criteria within a short period of time and will subsequently not need to refer to them on a frequent basis. The perception that all stress engineers must spend most of their time huddled around a few procedure manuals is, in Mr. Stokes' vernacular, "absurd."

The number of controlled procedures made available to OPEG engineers afforded convenient reference access while at the same time avoiding unnecessary complications in document control due to distribution of more copies than necessary to accomplish the work. Contrary to Mr. Stokes' allegation, this practice does not violate any requirements of 10 CFR 50, Appendix B, and is consistent with industry practice.

It is alleged that:

7. PG&E contends on page twenty-nine (29), " The need to specifically address thermal gap-limiting conditions in design procedures is obviated if an acceptable alternate method is used." They also state that "it is neither necessary nor consistent with normal engineering practice to specify in the design procedures the limiting conditions for the use of thermal gaps."

In response to these statements, I would like to quote from 10 CFR 50 Appendix B, III. DESIGN CONTROL. "Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in Section 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled." (Emphasis added in original)

Further, the contention that it is not normal engineering practice to specify in the design procedures the limiting conditions for the use of thermal gaps empirically is false. It is my experience that the engineering firms in which I have worked have had procedures directing the use of thermal gaps.

Unlike Diablo Canyon, industry practice is that a seismic restraint is not modeled with a thermal gap. Nor are the construction and radial gaps which provided for the pipe diameter growth used as thermal gaps as such, as they are at Diablo Canyon. The use of these small gaps, 1/16 to 3/16 of an inch, is not a normal practice and does require procedures to limit their use to ensure that deviations from such standards are controlled. (Emphasis added in original.) (6/1/84 Stokes Aff. at 6-7.)

Contrary to Mr. Stokes' allegation, the express language of 10 CFR 50, Appendix B, cited by him does not require procedures to limit the use of items such as thermal gaps to ensure that deviations are controlled. Instead, the language clearly permits their use so long as a controlled procedure exists for their use.

Mr. Stokes' confusion about the proceduralization of support gap modeling stems partly from the fact that he is selectively quoting from a previous response on this subject. Citing the entire response adds considerable clarity:

> The need to specifically address thermal gap-limiting conditions in design procedures is obviated if an acceptable alternate method is used. Such is the case on the Diablo Canyon Project. DCP Procedure P-11 requires that an as-built verification be performed whenever gaps are modeled in the piping thermal analysis. In accordance with this requirement, an experienced piping stress analyst reviews the installation of each unique configuration. This ensures that modeling of the thermal gaps is consistent with other design assumptions used in the calculation. Under these circumstances, it is neither necessary nor consistent with normal engineering practice to specify in the design the limiting conditions for the use of thermal gaps. (Underlined portion quoted by Mr. Stokes.)

Existing procedures governing the use of support gap modeling are adequate to ensure proper design control and are not in violation of 10 CFR 50, Appendix B, requirements.

It is alleged that:

8. On the bottom of page twenty-nine (29), PG&E states "The affected analyses have been rerun without gaps in these locations. The results indicate that all piping, supports and equipment remain qualified in accordance with licensing criteria."

I have a serious reservation about this statement. Before I state my reservation, I would like to state the standard industry practice in the use and application of thermal gaps based on my knowledge as a pipe stress engineer. First, it is standard practice to design a pipe system without thermal gaps if possible through good support location. Second when a thermal gap is used, it is because system (pipe, supports and connections to equipment) is over-stressed at some point, and without the use of the gap, design criteria (stresses) can not be met. I therefore believe the statement made by PG&E to be false. (6/1/84 Stokes Aff. at 7-8.)

Contrary to the assertions of Mr. Stokes, and consistent with good engineering practice, several analytical methods and refinements may be employed to successfully demonstrate piping and support qualification without consideration of support gaps. These include modeling of the actual stiffness of all supports in the analysis to reduce support loads, reducing the temperatures considered in the piping analysis through more detailed evaluation of the actual piping system operating conditions, or use of more detailed but time-consuming analyses of the pipe supports to show qualification to the higher previously calculated thermal loads. Contrary to

Mr. Stokes' opinion, support gaps were considered in the previous piping analysis, not as a "last resort," but simply because that method was and is considered to be a valid and legitimate analytical technique.

JIR-9

It is alleged that:

9. On page thirty (30) PG&E asserts, " A specific design procedure to describe the methods or limitations for modeling pin joints (joint releases) is not required. The use of joint releases is a well-accepted engineering practice that is standard in engineering evaluations of frame structures." This is followed in the next paragraph by "It can be pointed out that this well-accepted engineering technique was used very infrequently in the course of the small bore analysis."

It is false to imply that joint releases have been commonly applied in the industry in the context defended by PG&E at Diablo Canyon. Joint releases are used in structural steel design to represent specific bolted connections as pin-joints. They are almost never used in pipe support design, because typically all joints are welded all-around or on all accessible sides. Most pipe support engineers have never worked as structural steel designers and are not familiar with the use of joint releases. The use of joint releases should be spelled out in a procedure or instruction to ensure "that deviations from such standards are controlled." 10 CFR 50 Appendix B, III. It was not at Diablo Canyon. (6/1/84 Stokes Aff. at 8.)

Mr. Stokes states that joint releases are an acceptable practice in structural analysis. However, Mr. Stokes maintains that because most pipe support designers have never been structural steel designers, joint releases can't be employed on pipe support structures. This opinion is simply untrue. Precisely the same computer codes are used by structural engineers in each case.

The NRC Staff has reviewed this issue and also does not share Mr. Stokes' view. They have concluded, in SSER 22, "... the staff also finds the engineering basis and approach as described by the DCP acceptable and in accordance with current engineering practice," and later, "...the issue of assumed joint releases for rigid connections is considered resolved."

The use of joint releases is a well-accepted engineering practice that is standard in engineering evaluations of frame structures and a specific design procedure to describe the methods or limitations for modeling joint releases is not required.

It is alleged that:

10. On page thirty-two (32), PG&E states "The walkdowns for the 79-14 requirement were performed in 1980 and 1981. The NRC Staff signed off on this effort in 1981."

I question the validity of these sign-offs based on what has been learned since 1981. During that period, there have been many changes to the systems due to the reverification program. During this work, the as-builts did not reflect the installed hardware. These drawings in many cases had been as-built up to five (5) times and they still were not accurate. The Quick Fix Program exacerbated further, how much of the actual design is correctly as-built. No one can truthfully say that the drawings reflect what is installed at Diablo Canyon, even Mr. Yin found discrepancies in the as-built drawings.

Several lines from 79-14 demonstrate the significance of this issue. "The staff has determined, where design specifications and drawings are used to obtain input information for seismic analysis of safety-related piping systems, that is essential for these documents to reflect as-built configurations. Where subsequent use, damage or modifications affect the condition or configuration of safety-related piping systems as described in documents from which seismic analysis input information was obtained, the licensee must consider the need to re-evaluate the seismic analyses to consider the as-built configuration." (third para.) PG&E has conceded that 63% of the plant has been modified since the earlier walkdowns. 79-14 requirements should be reperformed. (6/1/84 Stokes Aff. at 8-9.)

USNRC IE Bulletin 79-14 required verification of the as-built condition of existing safety-related piping systems. This was accomplished at Diablo Canyon by comprehensive, proceduralized walkdowns performed in 1980 and 1981, long before Mr. Stokes was employed on the job. The NRC Staff reviewed the submittal

and documented the results of this program in 1981. At that time, Diablo Canyon procedures were revised to require Engineering review and acceptance of revised as-built conditions resulting from any changes to the previously evaluated configuration. The effect of these procedural requirements is to continually update the 79-14 walkdown as-built documentation. These procedures have been utilized during the recently completed reverification program and consequently all modifications made during that program have received appropriate Engineering review and acceptance as mandated by Bulletin 79-14. Contrary to Mr. Stokes' unfounded opinion, it is therefore not necessary to reperform the 79-14 walkdown and review.

JIR-11

It is alleged that:

In a discussion of the Quick Fix Program in the second paragraph on page thirty-eight (38), PG&E states "Field construction problems were referred to PSTDC [sic] team engineers who, based on their engineering judgement and knowledge of DCM M-9, would, on a case-by-case basis, determine whether use of expanded tolerance limits could be authorized to resolve the construction problem while maintaining an acceptable support design."

This is a false and misleading statement. I know one engineer was hired and assigned to the PSTDC [sic] group who had no previous knowledge of M9, nor did he have a copy of M-9 to consult in the field. I also know other engineers who had only worked in the San Francisco design office for several weeks prior to their assignment to the PSTDC [sic] group. They had very little knowledge of M-9. They were not assigned copies of M-9 to use in the field. Without knowledge of the design requirements contained in M-9, there was no way for these engineers to determine whether an expanded tolerance could be authorized to resolve a construction problem while maintaining an acceptable support design, other than through their engineering judgement alone. This is not enough. In this instance, the breakdown in training and the continued lack of necessary design information in the field means that, in an unknown number of cases, the Quick Fixes were just hunches. The review of all Quick Fixes is required to ensure that important design information shown on them is checked in the as-built review program. (6/1/84 Stokes Aff. at 9-10.)

Most nuclear power plant construction projects use some form of program that is similar in many ways to the PSDTC program, which relies on engineering judgment to develop field modifications of pipe support designs to facilitate their installation. Such programs utilize various forms of official, documented approval at a later

date, as does the Diablo Canyon Project through its "as-built" acceptance program. Contrary to the implication of the allegation, this type of program is certainly not unique to Diablo Canyon and many PSDTC engineers were familiar with similar programs.

The as-built drawing generated at the completion of the PSDTC process reflects the actual configuration of the support as it finally exists in the plant. The drawing is reviewed by Engineering to ensure that the actual installation meets design and licensing requirements. Any exceedence of allowable limits is subject to rejection by Engineering. For such an engineering review, each and every PSDTC document is not necessary as the information on each PSDTC has already been incorporated into the as-built drawing being reviewed.

The statement that PSDTC engineers resolved construction problems "based on their engineering judgment and knowledge of DCM M-9" is neither false nor misleading. Pipe support engineers, working out of OPEG in the PSDTC group, did have access to DCM M-9. Notwithstanding, the type of information contained in the document is not unique and is not foreign to experienced pipe support engineers but rather is common to the industry.

PSDTC Engineers, given their experience and knowledge of the principles in DCM M-9, had the ability to make the kind of design decisions required. All such design decisions were subsequently reviewed as part of the as-built process. The implication that the OPEG engineer made decisions on matters about which he was totally unfamiliar without the availability of information to make his decision is totally false.

Contrary to the allegation, the vast majority of the engineers who were assigned to the PSDTC group were already familiar with DCM M-9 at the time of their assignment by virtue of their prior experience in pipe support work involving its use. The few engineers who were exceptions gained familiarity with M-9 while working with other members of the PSDTC group. These few exceptions do not invalidate the program as Mr Stokes would suggest.

It is alleged that:

12. Still under the PSTDC [sic] discussion on page thirty-eight (38) but in paragraph no. four, PG&E would like all parties concerned in the safety of Diablo to believe that I don't know what I am talking about. "Notwithstanding Mr. Stokes' apparent lack of knowledge all the PSTDC [sic] group's modifications received final engineering review and approval as part of the as-built acceptance."

In reply to this, I would like to inform all interested parties that PG&E has never asked me to explain or clarify any statement I have made. Had they, I would have informed them that I was in touch with fellow Unit 1 engineers while I was in the PSTDC [sic] group. They told me that in performing the as-built reviews; PSTDC's [sic] were not part of the as-built review package. They were only given the as-built drawing to review and accept.

This is the same practice which was followd prior to March 1983 when I was part of the Unit 1 group. During this time, we were asked to approve as-builts which were not like the design drawings which we had issued to the field for construction. I was not aware that the PSTDC [sic] group was formed in January 1983.

I should also say that before March 1983 the engineer who designed a support also did the as-built acceptance of that support. This was changed in March because the Unit 1 group was split into two (2) groups. The engineers who were questioning management about the changes were assigned to Unit 2. This included myself. (6/1/84 Stokes Aff. at 10-11.)

Mr. Stokes continues to contend that PSDTCs are not incorporated in the pipe support as-built review package. His contention is incorrect. It is true that not every PSDTC sheet is individually reviewed and accepted by Engineering in the as-built acceptance process. However, PSDTCs are incorporated in the final hanger as-built drawing which does receive Engineering approval as part of the as-built acceptance process.

When construction of a pipe support is complete, Pullman Engineering prepares a final as-built drawing of the support to reflect the actual, final, as-installed configuration of the support. This drawing incorporates any PSDTCs which have altered the support design during the course of its construction and which account for its final configuration as it exists in the final installed condition. It is this as-built drawing which is then sent to Engineering for final acceptance. This comprehensive process of review and acceptance of changes made by PSDTCs as a part of the as-built acceptance program has been corroborated by recent NRC Staff audits.

Mr. Stokes' statements to the effect that an engineer did the as-built acceptance for supports he designed may have been true in some isolated cases but was not even remotely the normal case. As engineers might be transferred from one group to another such a thing could occur. The applicable procedures required an engineering group other than the PSDTC engineers to do the as-built acceptances.

Mr. Stokes' statement that engineers who questioned management were assigned to Unit 2 work when separate Unit 1 and 2 pipe support groups were formed in OPEG is incorrect and is not supported by the facts. (See PGandE response to JI's Motion to Reopen DQA). Engineering personnel to staff the two separate pipe support groups came from both newly hired individuals and from the existing OPEG personnel. The basic consideration in establishing the makeup of the

two separate teams was to provide each with an essentially equivalent mix of new assignees, engineers with more project experience, and supervisory personnel such that each project would be supported equally. At the time of the division, the four supervisors who were to be the principal leadership in the new Unit 1 and Unit 2 organizational structure held discussions to establish which of the more experienced engineers were to be assigned with the newer engineers in each of the squads of the new organization. Contrary to Mr. Stokes' claim, there was no discussion or consideration of any factor other than a balancing of experience-ability in the assignment of personnel to the two teams.

It is alleged that:

13. On page thirty-nine (39) in the third paragraph, discussing the PSTDC [sic] program PG&E states "In August 1983 an audit was conducted by the PG&E QA Department which resulted in the overall conclusion that the control of design changes by OPEG appeared to be effectively implemented. One finding was identified with respect to use of the PSTDC [sic] forms. In response to this finding, special training sessions were held in October 1983 for all PSTDC [sic] engineers to emphasize the limitations on the use of PSTDC [sic] forms and to assure that Design Change Notices would be initiated when required by the procedures." (Emphasis added.) Then in the fourth paragraph in explaining "the June 16, 1983 memo, referred to by Mr. Stokes," PG&E says, "Summarized, the memo states that the PSTDC [sic] program is not a corrective action program and may not be used in lieu of construction discrepancy reports (DRs and DCNs). This memo was not applicable to the PSTDC [sic] engineers and as such did not receive distribution to them." (Emphasis added.)

It appears to me based only on the above two PG&E quotations that the last line of the second quotation is a false statement. This is supported by PG&E's QA Audit finding. (See emphasized comments) I would like to quote item 1 from the June 16, 1983 memorandum. "If an item on the process sheet has been accepted by Quality Control and then it is determined that a PSTDC [sic] is necessary to authorize this change, Pullman will generate a DCN prior to requesting the PSTDC. PSTDC [sic] will monitor to assure that this is complied with." (Emphasis added.) Clearly, the document should have been given to the PSTDC [sic] engineers; if PSTDC [sic] was to monitor the generation of DCN's as stated in this memo.

To my knowledge, I was the only PSTDC [sic] engineer who had a copy of this memo other than my group leader Jeff Von Klompenberg. As a result, I was the only PSTDC [sic] engineer who consistently saw that the applicable DCN or DR was generated by Pullman. Many times when I asked for the DCN or DR to be generated before I wrote a PSTDC [sic], the Pullman Field Engineer, Pre-Inspect Engineer or Redliner would walk away not to return during my shift. I checked with the day shift PSTDC [sic] engineers the following day and found that they had issued the PSTDC [sic] without requiring the necessary DCN or DR. Had this memo been distributed to the PSTDC [sic] engineers in June 16, 1983,

many DCN's and DR's would have been generated, forming a record of existing defects discovered at Diablo Canyon during the reverification program. This information could have been used to correct other generic safety related defects. This knowledge has been lost. (6/1/84 Stokes Aff. at 11-13.)

This allegation implies that as a result of the implementation of the PSDTC program, there are undetected generic defects in the piping supports at Diablo Canyon. This is not so.

The procedural limitations against using a PSDTC in lieu of a Pullman Discrepant Condition Notice or Pullman Discrepancy Report are discussed in the reponse to Allegation JIR-14. While there may have been occasional exceptions to this procedural requirement, they were very limited in number.

However, regardless of whether DCNs or DRs were written, all installations as modified under the PSDTC program were verified to ensure that design and licensing criteria were satisfied, thus ensuring no defect. The PSDTC was never the final design qualification for a pipe support modification. All modifications authorized by the PSDTC were formally reviewed and approved by Engineering, using controlled procedures as part of the as-built acceptance program. Consequently, there are no undocumented and unreviewed conditions involving pipe supports which were created as a result of the PSDTC program.

Finally, as a basis for this allegation, Mr. Stokes cites a June 16, 1983, memorandum. The copy he cites, however, was a carbon copy and was not the corrected document which was actually issued to the contractor. A copy of each memorandum is attached as Exhibits 1 and 2, respectively. Overstrikes are apparent when they are compared. In the memorandum as released, Project General Construction, not the PSDTC Engineer, was directed to monitor the issuance of quality deficiency reports. Consequently, there was no reason for the PSDTC Engineer to receive a copy of the memorandum. The June 16, 1983, memorandum discussed by Mr. Stokes is a wonderful example of how rummaging around in other people's desks after hours results in obtaining the wrong information. Contrary to the allegation, the statement that the memorandum was not applicable to PSDTC was not false.

The June 16, 1983, memorandum quoted by Mr. Stokes refers to Discrepancy Reports (DRs) and Deficient Condition Notices (DCNs) which are quality documents that are initiated by the Contractor. The August, 1983, Audit quoted by Mr. Stokes refers to Design Change Notices which are initiated by Project Engineering. In this allegation, Mr. Stokes mixes stolen apples with oranges and hopelessly misses the mark.

PG&E

MEMORANDUM

EXHIBIT 1 OF RESPONSE TO JIR 13

PCD # 003559

Date June 16, 1983

TO P. Stieger/R. King/H. Karner Location DCPP Ext.

FROM R. L. Meredith/J. Arnold Location DCPP Ext. #3108

SUBJECT Pipe Support Design Tolerance Clarifications File No.

This memo is to clear up confusion which currently exists regarding the use of Pipe Suppo Design Tolerance Clarifications (PSDTC) when a hanger is in-process.

1. If an item on the process sheet has been accepted by quality control and then it i determined that a PSDTC is necessary to authorize this change, Pullman will genera a DCN prior to requesting the PSDTC. PIGC will monitor to assure that this is complied with.
2. If a hanger has been final accepted (green tagged) and it is determined that a PSDTC is necessary a DR must be generated by Pullman.

The above is PG&E's interpretation of ESD 268. If Pullman feels any ESD's require modification to comply with the above they should be revised and submitted prior to July 1, 1983. Attached find a flow chart illustrating the above.

In addition the following guidelines shall be adhered to by everyone involved with PSDTC' PSDTC's are to be treated in the same manner as the original "approved for construction" drawing. No changes outside of ESD 223 are permitted after the PSDTC area engineer signs it. All PSDTC's must be signed by a PSDTC area engineer on either the "constructio may proceed" or the "construction DR required" line of the PSDTC. The "contractor receip signature is only required on PSDTC's that are signed "construction may proceed". If a PSDTC is no longer required it must be voided. Only a PSDTC area engineer can void a PSDT The voided PSDTC remains with the package. In no case shall white-out or liquid paper be used.

cc: J. Bratton
R. Oman
G. Thomas
R. Tinkle

FILE
KARNER
SPEIR
KAPSALIS
KING
MOREY
HAMILTON
POLCHOK
DAVIS
MADRON

OLDENKAMP
RECTIFICO
FLEDERMAN
C. ROBERTS
LANKES
GUDLOY
MCGROW
SCHULTZ
HENDERSON
JACKSON

Unit #1 R. L. Meredith

Unit #2 J. Arnold

Rec'd by Lisa Brock 6-17-83

REPLY BY:

P G and E EXT. NO.

INSTRUCTIONS – REMOVE YELLOW COPY AND BOND WHITE AND PINK COPIES WITH CARBON INTACT. WHITE COPY WILL BE RETURNED WI REPLY. REMOVE SET FROM PAD BEFORE USING.

EXHIBIT 2 OF RESPONSE TO JIR 13

P G &E
MEMORANDUM

Date June 16, 1983 6

TO: Stieger/P. King/H. Karner Location DCPP Ext.

FROM: R. L. Meredith/J. Arnold Location DCPP Ext. 3108

SUBJECT: Pipe Support Design Tolerance Clarifications File No.

This memo is to clear up confusion which currently exists regarding the use of Pipe Support Design Tolerance Clarifications (PSDTC) when a hanger is in-process.

1. If an item on the process sheet has been accepted by quality control and then it is determined that a PSDTC is necessary to authorize this change, Pullman will generate a DCN prior to requesting the PSDTC. PSDTC will monitor to assure that this is complied with.
2. If a hanger has been final accepted (green tagged) and it is determined that a PSDTC is necessary a DR must be generated by Pullman.

The above is PG&E's interpretation of ESD 268. If Pullman feels any ESD's require modification to comply with the above they should be revised and submitted prior to July 1, 1983. Attached find a flow chart illustrating the above.

In addition the following guidelines shall be adhered to by everyone involved with PSDTC's. PSDTC's are to be treated in the same manner as the original "approved for construction" drawing. No changes outside of ESD 223 are permitted after the PSDTC area Engineer signs it. All PSDTC's must be signed by a PSDTC area Engineer on either the "construction may proceed" or the "construction DP required" line of the PSDTC. The "contractor receipt" signature is only required on PSDTC's that are signed "construction may proceed". If a PSDTC is no longer required it must be voided. Only a PSDTC area engineer can void a PSDTC. The worded PSDTC remains with the package, in no case shall white-out or liquid paper be used.

REPLY

cc: J. Bratton
R. Oman
~~G. Thomas~~
R. Tinkle

RECEIVED
JUN 19 1983
DIABLO CANYON

Unit #1 R. L. Meredith
Unit #2 J. Arnold

OMS-1334

REPLY BY: 6-17-83 P G and E EXT. NO.

INSTRUCTIONS – REMOVE YELLOW COPY AND SEND WHITE AND PINK COPIES WITH CARBON INTACT. WHITE COPY WILL BE RETURNED WITH REPLY. REMOVE SET FROM PAD BEFORE USING

JIR-14

It is alleged that:

14. Witnesses still on-site have described how Quick Fixes are still being used to disposition DCN's and DR's. This means that DCN's and DR's have been and are being closed out on the promise of future engineering review. As seen above, often the engineering reviews never took place. All DCN's and DR's dispositioned with Quick Fixes should be reopened to receive correct engineering resolution prior to their being closed out. (6/1/84 Stokes Aff. at 13.)

This allegation concerns proper dispositioning of two types of internal Pullman quality documents known as Pullman Discrepant Condition Notices (DCN) and Pullman Discrepancy Reports (DR). These were used to document pre-existing pipe support configurations found to be in noncompliance with appropriate design and construction documents, and were controlled in accordance with Pullman procedures. A PSDTC was not intended to be used to document such pre-existing discrepant conditions. However, once the appropriate Pullman DCN or Pullman DR was originated, a PSDTC could be used to document the engineering disposition. In short, PSDTCs were not to be used in lieu of quality documents (DCNs or DRs), but they could be used to disposition such documents. Mr. Stokes is categorically incorrect in stating that Engineering reviews of PSDTCs never took place. As discussed in the response to Allegation JIR-12, all PSDTCs were reviewed by Engineering as part of the pipe support as-built acceptance program.

JIR-15

It is alleged that:

15. On the top of page forty (40), PG&E goes further "The PSTDC [sic] engineer is not, however, required to monitor writing of discrepancy reports by construction."

The point is that the PSTDC [sic] engineers should have been, as seen by the discussion of No. 13 above and PG&E's comment on page forty-two (42) last paragraph. "The PSTDC [sic] program may have used used in situations where a Discrepancy Report (DR) or Design Change Notice (DCN) was necessary."

The last statement should say was used, not "may have." Based on my discussions with other witnesses, I believe that Quick Fixes have continued to serve as substitutes for quality reports well into 1984. (6/1/84 Stokes Aff. at 13-14.)

This allegation restates the point made by PGandE on page 42 of letter DCL-84-131 dated April 4, 1984; namely, that it is recognized that there were tolerance clarifications that were issued which exceeded the intended scope of the PSDTC program. However, these cases were the exception.

The quotation from DCL-84-131 was based upon an August, 1983, audit of the PSDTC program. In such audit there were no findings that related to use of the PSDTC program as a substitute for quality reports. The findings related only to use of the PSDTC program in place of a Design Change Notice (DCN), and <u>not</u> as alleged by Mr. Stokes "Discrepancy Report(s) (DR)."

Even in the instances where the PSDTC program was used in lieu of a Design Change Notice, all modifications authorized by PSDTC were formally reviewed and approved by Engineering, using controlled procedures as part of the as-built process.

There are no audit findings of the use of the PSDTC program as a substitute for quality reports. However, even if a PSDTC may have been used in lieu of a Discrepant Condition Notice or Discrepancy Report, the pre-inspect, red-line, as-built process would ensure that any discrepant condition would be documented on an as-built drawing. That as-built drawing would then receive full Engineering review which would catch any discrepant condition.

It is alleged that:

On the top of page seventy-seven (77) in PG&E's response on audits, PG&E states, "Program audits provide coverage of all QA Program elements as required by Regulatory Guide 1.33 and 1.144."

In my opinion, this is a false statement. This is based on reading Regulatory guides 1.33 and 1.144 for myself. The first point which is relevant is that Guide 1.33 is titled "QUALITY ASSURANCE PROGRAM REQUIREMENTS (OPERATIONAL)." The audits discussed on page seventy-seven (77) are during the construction phase not operation. The second point which is important is taken from the the third paragraph under B. DISCUSSION. " A separate regulatory guide addressing the quality assurance program for the preoperational phase will be issued." After reading Reg. Guide 1.33, I don't believe it should have been included in PG&E's comments.

Regulatory Guide 1.144 in Section B. DISCUSSION in the second paragraph, discusses two documents "WASH-1283, Revision 1, 'Guidance on Quality Assurance During Design and Procurement Phase of Nuclear Power Plants,' dated May 24, 1974 (Gray Book, Revision 1), and WASH-1309, 'Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants,' dated May 10, 1974 (Green Book, Revision 0)." In the third paragraph, it is stated that "ANSI/ASME N45.2.12-1977, subject to the regulatory exceptions in Section C of this guide, will replace N45.2.12 (Draft 3, Revision 4) contained in WASH-1283 and WASH-1309 in the evaluation of proposed auditing procedures for the applicant's quality assurance program included in construction permit and operating license applications." (Emphasis added.) From the sections quoted above, it appears Reg. Guide 1.144 does apply to the discussion of PG&E's audit procedures. Looking to Section C, first paragraph, "The requirements that are included in ANSI/ASME N45.2.12-1977 for auditing quality assurance programs for nuclear power plants are acceptable to the NRC staff and provide an adequate basis for complying with the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50, subject to the following." (Emphasis added)

Continuing in Reg. Guide 1.144 Section C.3.

> 3. Section 3.5.2 of ANSI/ASME N45.2.12-1977 requires that audits of quality assurance activities be regularly scheduled to ensure that the quality assurance program is adequate and that activities are being performed in accordance with the quality assurance program. The frequency of the scheduling of audits is dependent on the status and importance of the activities, and the following is considered acceptable scheduling:
>
> a. Internal Audits
>
> (2) Design and Construction Phase Activities --Applicable elements of an organization's quality assurance program should be audited at least annually or at least once within the life of the activity, whichever is shorter. In determining the scope of the audit, an evaluation of the area being audited may be useful. The evaluation may include some or all of the following: prior quality assurance program audits; results of audits from other sources; nature and frequency of identified deficiencies; and significant changes in personnel, organization, or quality assurance program. (Emphasis added.)

I don't think PG&E's audits fully met the intent of the above Section, especially the emphasized portions. They didn't comply with the frequency requirements per C.3.a.(2) "at least once within the life of the activity." Neither did they consider the emphasized parts of C.3.a.(2) "prior quality asurance program audits" (examples: deficient SITE INVESTIGATION--HOSGRI FAULT ANALYSIS pre-1981-82--MIRROR IMAGE PROBLEM--REVERIFICATION PROGRAM); "results of audits from other sources" (example: NSC AUDIT 1977); "nature and frequency of identified deficiencies" (examples: ALL USED SO FAR); "and significant changes in personnel, organization, or quality assurance program." "Significant changes in personnel, organization, or quality assurance program" accured [sic] almost weekly during the reverification program.

Continuing in Reg. Guide 1.144 Section C.3.b. External Audits

> (2) For other procurement actions not listed in Item C.3.b.(1), audits should be conducted as follows: Elements of a supplier's quality assurance program

> should (shall per C.4) be audited by the purchaser on a triennial basis with the audit implemented in accordance with Section 4, "Audit Implementation," of ANSI/ASME N45.2.12-1977. (Skip several lines) When a subsequent contract or a contract modification that significantly enlarges the scope of activities performed by the same supplier is executed, an audit should (shall per C.4) be conducted of the increased requirements, thus starting a new triennial period.

PG&E would have had to perform an audit almost weekly to have complied with the requirements of Reg. Guide 1.144 for Internal and External Audits. (6/1/84 Stokes Aff. at 14-16.)

Again, Mr. Stokes is rendering opinions in an area in which he is totally unqualified to do so. This lack of expertise is hopefully what leads him astray. Mr. Stokes is correct when he concludes that the audits discussed on page 77 were conducted during the construction phase. DCPP has not yet begun commercial operation. However, requirements issued by NRC and ANSI/ASME which are specifically for operating activities have been included in PGandE's program in preparation for operating DCPP. The PGandE quality assurance manual, issued in February 1981, is titled "Quality Assurance Manual for Operating Nuclear Power Plants." It is obvious that the requirements of Regulatory Guide 1.33 are just as applicable as the requirements of Regulatory Guide 1.144. Actually, Regulatory Guide 1.33 is <u>more</u> applicable for the QA program <u>for operations</u>, which is what PGandE is now implementing.

PGandE's audit program meets all requirements of both Regulatory Guide 1.33 and Regulatory Guide 1.144. The NRC, Region V, has

conducted detailed inspections of PGandE's audit program. These inspections have specifically inspected to the details of ANSI/ASME N45.2.12-1977 and have not raised any questions about audit frequency or any of Mr. Stokes' other concerns.

Mr. Stokes states, "I don't think PG&E's audits fully met the intent of the above Section...," but he doesn't give any details upon which to base his conclusion. The NRC, Region V, which has inspected the audit program in great detail, has however, found the program acceptable.

Mr. Stokes does give some "examples" but his examples only serve to amplify his lack of knowledge and expertise in the area of quality assurance. For example, on Stokes' page 16 of 18, he refers to:

- "prior quality assurance program audits": Mr. Stokes' "examples" are <u>not</u> audits.

- "NSC AUDIT": PGandE <u>did</u> conduct an audit after receiving a copy of the NSC audit and so did PGandE's contractor <u>at PGandE's direction</u>. (Refer to the enormous amount of correspondence on NSC.)

- "nature and frequency of identified deficiencies: "ALL USED SO FAR." Response: Everything Mr. Stokes has "used" to this point

is not pertinent information, so this point of his contention really does not require response.

- "significant changes in personnel...occurred almost weekly": Mr. Stokes seems to think that any change in personnel such as whenever a clerk quits or changes sections is significant! That patently is not true. Changes in the management organization described in the Nuclear Quality Assurance Manual or in the Quality Assurance Program are required to be approved by PGandE QA Department prior to being implemented. This requirement has been followed to date on the Diablo Canyon Project.

- "PGandE would have had to...audit almost weekly": Mr. Stokes simply does not understand what he is talking about. PGandE's procedures specifically cover frequency for both audits of suppliers and audits of internal activities and Region V has checked the implementation of that program and found it to be satisfactory.

JIR-17

It is alleged that:

17. On page eighty-one (81) and other locations in many responses PG&E states "technical QA audits are not a requirement of 10 CFR 50 Appendix B."

In response to PG&E's persistent statement, I would like to quote 10 CFR 50 Appendix B, Section XVIII.AUDITS dated June 17, 1970.

> A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits shall be performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audit results shall be documented and reviewed by management having responsibility in the area audited. Follow up action, including reaudit of deficient areas, shall be taken where indicated. (Emphasis added.)

I would also like to quote the last line of ANSI/ASME N45.2.12-1977 Section 4.2.2 Team Selection.

> 4.2.2 Team Selection. (Several lines omitted) In selecting personnel for auditing assignments, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personnal [sic] characteristics, and education. (Emphasis added.)

It is true that 10 CFR 50 Appendix B, Section XVIII does not say "technical audits are required" but I believe the reason for including "by appropriately trained personnel not having direct responsibilities in the areas being audited" was because it was and is intended that all seventeen Sections listed before Section XVIII. AUDITS were to be audited. (Emphasis added.) No one section was to be self-monitoring.

This is implied by the emphasized section of Section 4.2.2 of ANSI/ASME N45.2.12-1977 also. Why would the authors require personnel to have special abilities, specialized technical training, and prior pertinent experience if technical audits were not required?

> In practice, this commitment is honored outside of Diablo Canyon within the industry. Many design firms for which I have worked have included technical QA audits of engineering to ensure their compliance with the required CODES, STANDARDS AND the PSAR requirements. (6/1/84 Stokes Aff. at 16-18.)

Mr. Stokes has absolutely no professional experience or training to qualify him to render opinions in the area of quality assurance. Nonetheless, while apparently agreeing with DCP that 10 CFR 50, Appendix B, does not require "technical QA audits," Mr. Stokes recasts the issue by concluding that this approach results in one of the 10 CFR 50, Appendix B, criteria (presumably Criterion III) not being audited. This conclusion is wrong and unsupported. The DCP Audit Program clearly includes audits of design control activities in addition to the separate activity of verification of technical requirements in design output documents. As stated in the following letter DCL-84-046 dated February 7, 1984, from PGandE to the NRC:

> "In implementing Criterion XVIII of 10 CFR Part 50, Appendix B, the NRC has endorsed, with certain exceptions, ANSI N45.2 and ANSI N45.2.12. The latter document provides requirements and guidance for establishing a system of audits of quality assurance programs, and provides definition of various types of audits. Criterion XVIII mandates audits to verify compliance with the QA program and to determine its effectiveness. None of the above-cited references establish requirements for the performance of <u>technical QA</u> audits.
>
> On the Diablo Canyon Project, QA audits are conducted (in fulfillment of licensing commitments) to verify compliance with the project quality assurance program requirements.

> The Project audit program has been developed and implemented to comply with requirements of the Project Nuclear Quality Assurance Manual. This program, in turn, has been approved as being in compliance with Project requirements and Criterion XVIII of Appendix B. It calls for a system of audits, the scope of which has been widely accepted in the nuclear industry, to assure that the QA program is properly functioning. Relative to the OPEG group, this audit scope has included all the major areas of design activity such as control of calculations, control of design drawings, indoctrination and training, and design change control. In addition, PGandE, as the licensee, has conducted a series of Activity Audits covering OPEG activities.
>
> Since 1982 there have been some nineteen (19) audits of OPEG to verify compliance with Project QA requirements. Closeout and corrective actions related to audits is documented in the Project audit files.
>
> The verification of technical requirements in design output documents is performed by Engineering as part of the design control process. The type of verification can vary from checking to independent review by the Chief Engineer or an outside agency, depending on the significance of the document."

In the last complete paragraph on page 17 of Exhibit 1 to Joint Intervenors Reply dated June 11, 1984, Mr. Stokes attempts to connect the need for auditors to have "special abilities, specialized technical training, and prior pertinent experience" with a requirement for technical audits. However, the "special abilities, training and prior experience" referred to are, by accepted industry definition, those required to properly perform a QA audit, not those required to check and verify any design work performed on the Project. An entire standard, ANSI/ASME N45.2.23-1978, was written to define the requirements for training, prior experience, and abilities needed for individuals to be qualified as auditors or lead auditors.

The DCP QA Program complies with this standard, and with Stokes' quotation from ANSI/ASME N45.2.12-1977. The allegation has no merit.

At the top of page 18 of Exhibit 1 to Joint Intervenors Reply dated June 11, 1984, Mr. Stokes complains that Diablo Canyon does not do technical QA audits of engineering even though "many design firms for which I (Stokes) have worked have included technical QA audits...." This statement conveniently ignores the other significant technical reviews which take place because they are not labeled "technical QA audit" and are not performed by the QA organization. PGandE letter DCL-84-131 dated April 4, 1984, to the NRC summarized as follows:

> "The second observation concerns requirements for technical QA audits of OPEG work. While technical QA audits are not a requirement of 10 CFR 50, Appendix B, the value of technical reviews or audits is clearly recognized. Such reviews have been performed on various aspects of the design as part of, or in addition to, design verification measures even though they are not a part of the formalized QA audit procedures. In addition to normal checking and approval, small bore pipe support design was subjected to an independent design review by the IDVP, on a sample basis, and an IDVP audit with emphasis on technical interface control."

Finally, the NRC Peer Review Group that reviewed Mr. Isa Yin's draft findings apparently agrees with PGandE's position. At page 365 of the transcript of the ACRS Meeting of April 6, 1984, Mr. Jim Taylor acknowledged that QA standards "leave most of the auditing and verification to be done within the design organization performing the design work -- I refer you to the various ANSI N45.2.11 for that."

The allegation that PGandE's audit practices do not meet industry standards is false and has no merit.

It is alleged that:

In response to PG&E letters on the subject of Tube Steel Radius and Flare-Bevel Welds as addressed in the April 7, 1984 and April 11, 1984 to the members of the Appeal Board, and PG&E letter No.: DCL-84-083, DCL-84-164, DCL-84-166, and DCL-84-190.

In PG&E letter: DCL-84-164 Enclosure 7 on page 10, PG&E states "The adequacy of the DCP criterion was addressed in PG&E letters DCL-84-083, DCL-84-141, and DCL-84-153. As described in these letters, site inspections confirmed that the tube steel corner radii are, in fact, 2.0 t (or slightly larger)."

In PG&E letter: DCL-84-166 on page 47 under the subject "WELDING OVERVIEW WELD SYMBOLS," PG&E states "The allegations all fail for either a lack of substance, lack of context, technical errors, false or misleading statements, or a combination of these reasons." (Emphasis added in original.)

I hope to put an end to the verbal debate about the existence and safety significance of out-of-specification tube steel which is and has been used in constructing faulty safety-related pipe supports and other safety structures at Diablo Canyon. This problem is not limited to isolated occurrences, as claimed by PG&E. Until recently, I had only seen tube steel with radius less than 2 t at Diablo Canyon site before I was terminated last October 1983. However, that has changed in the last few days. A currently-employed Diablo worker who was aware of my ongoing debate with PG&E, provided my counsel Thomas Devine a piece of 3 x 3 x 1/4 tube steel which was cut from a piece of tubing being used to construct a Unit 2 class 1 pipe support. The worker informed me that tubing similar to this example was used during 1983 to construct Unit 1 class 1 pipe supports. The tube steel was given to Mr. Devine for my analysis to forward to the appropriate government authorities. I know that it was intended for use on safety-related work, because I examined the relevant support drawing. It indicated that the support was code class 1 and design class 1.

In PG&E letter no. DCL-84-166 on page fifty under topic "WELDING OVERVIEW FLARE BEVEL WELDS TUBE STEEL RADII" PG&E states that "different radii can be inferred for the tube corner dependent upon the measurement method used."

On page 51, PG&E states, "We have termed this the flat side intersection method (D dimension). Based on this D dimension, an apparent corner radius of RD is implied. (...) For the 3 x 3 x 1/4 tubes, the apparent radius, RD, is approximately 1.25 t to 1.5 t and the RA is always 2.0 t or slightly larger. For the 3 x 3 x 1/2 tubes, RD is approximately 1.0 t and RA is again 2.0 t."

This tubing (Attachment 1 sketches) not only has a corner radius less than 2t but has only one (1) corner which is 2t. Three corners are less than 2t, of which two (2) are only 1t as I and others have confirmed. It measures 1t using the Straight Ruler Method (per PG&E) and using the Concave Radius Gage Method. An additional flaw exists, which before seeing this piece, not even I was aware of. The tube wall has what appears to me to be a lamination crack running all the way around its' [sic] perimeter. This lamination crack is almost exactly at the center line of the tube wall.

On the 24th of May, I was in a meeting with the NRC staff in which I was asked how I felt the radius of tube steel should be verified? Should the Straight Ruler Method, the Concave Radius Gage Method or some other method be used? I am not going to propose a novel method. What is and has been my only concern is that the calculations which PG&E/BECHTEL perform are done with good engineering judgement based on obtaining conservative results. After measuring the sample tube steel referenced above, I recognized that no simple accurate method exists for measuring the radii, because this steel is so badly made that the radii changes noticeably along its' [sic] length. Depending on show you decide to measure the radii you can obtain three (3) possible results at each corner.

As an engineer and in accordance with the established guidelines of relevant professional CODES, I feel the above data should be factored into the methodology. In the case of pipe support steel design, I will and would like to see applied at Diablo Canyon the definition of tube steel radius as defined by either the (AISC) or the (AWS) CODES. Based on my knowledge, and after review of pages 4-131 to 4-149 and Section 1.17 "WELDS" on page 5-44 of the AISC manual, I decided that the answer must be within the AWS document. Both documents govern the construction of pipe support steel, but on page 4-149 and 5-44 the AISC references the reader to the AWS code for this issue.

Looking at AWS A2.4-79, the AWS document on SYMBOLS FOR WELDING AND NONDESTRUCTIVE TESTING Section 9.2.9.

"Flare-groove welds. The dimension S of flare-groove welds is considered as extending only to the tangent points indicated below by dimension lines. (See Fig. 32)" (See Attachment 2, 2 pages)

Based on my own measurements of the sample tube radii and the section 9.2.9, the only conservative method of measurement is the Straight Line Method per PG&E. By using a combination square with a Bevel Protractor Head with a blade graduated in 64ths in conjunction with a flashlight, a QC inspector or engineer can measure the radius to within 1/64 inch. I feel that it is important to state that two (2) values per corner typically can be measured, and it is good engineering practice to use the minimum value in the effective throat calculations to ensure a conservative design. The reason for the Bevel Protractor Head is that the tube steel walls are not necessarily 90° to each other, and the protractor will allow flush contact with the wall of the tube.

I can't provide PG&E or the NRC with an easy way to resolve the lamination problem. There is one question that should be addressed before any other. Should this defective steel be allowed to remain in the plant, or should a through [sic] search be made and all defective materials be removed from the plant? To accept this condition "as is" without compromising public safety, several questions must be satisfactorily addressed. First, Welding - What joint configuration, effective throat, or wall thickness will govern the failure mode which is the most critical? Second, member stresses - What section properly will be used in the analysis of member stresses? Third, how will confidence be established that this is the only defective material in the plant?

There is increasing empirical evidence that the answers will confirm a problem. Another Diablo employee told me that some NPS Struts and Snubbers which were for class 1 systems have spherical bearings which are cracked from what appears to be the swageing [sic] process when they were installed. Swageing [sic] is a process by which material is welded by pressure or hammering. My friend recalled that the spherical bearings are marked "made in Japan." I gave the NRC in December 1983 a large spring-can structural steel attachment which also had lamination cracks from cold rolling. I told them that I was aware that three (3) were defective of a random sample of four (4). Since December 1983, I have heard from workers at the plant that 25 defective spring-can attachments had been found in the plant. I should note that the NRC was not concerned with the defective spring-can attachment - it was not installed on a class 1 pipe system (just Main Steam!).

> The problem has been confirmed on a sufficient scale that to adequately protect the public safety we still must obtain answers before Unit 1 begins power ascension. Results from PG&E and the NRC's "samples" are sharply in conflict with hardware reports from workers in the field. Only a comprehensive review can resolve this dispute of fact. (6/1/84 Stokes Aff. at 1-5.)

The portions of these two allegations related to a single length of tube steel with defective corners and laminations are addressed herein in responses to JIR 70, 73 and 74.

The issues regarding tube corner radii and weld effective throats were previously addressed and should be moot as it has been shown that there is no real concern for tube corner dimensions because the weld effective throat used in the design calculations is always achieved and with significant margin. This issue was summarized in letter DCL-84-166 to the NRC, as follows:

> The curved portion of the tube corner is not fabricated to be exactly one quarter of a circle, as indicated by the dotted line in Figure A. In fact, it is actually less than a quarter of a circle as indicated by the solid line in the same figure. Therefore, different radii can be inferred for the tube corner dependent upon the measurement method used.
>
> If a straight ruler or mechanic's square is used, as shown in Figure B .... a corner dimension, D, is measured. We have termed this the flat side intersection method (D dimension). Based on this D dimension, an apparent corner radius of R_D is implied. As shown in Figure B, this may not be the actual radius of curvature. Alternatively, if a concave radius gauge is used, the measured radius is R_A, as shown in Figure C. R_A is the appropriate measurement of the actual radius of curvature. Only when the tube corner is an exact quarter circle will the two measurement methods yield the same radius. Examples of tubes with the smallest corner dimension, D, were selected by the Pullman QC for further measurement. The resulting measurements are

> summarized in Tables 1A and 1B .... The difference between R_D and R_A is clearly shown in these tables. For the 3 x 3 x 1/4 tubes, the apparent radius, R_D, is approximately 1.25 t to 1.5 t and the R_A is always 2.0 t or slightly larger. For the 3 x 3 x 1/2 tubes, R_D is approximately 1.0 t and R_A is again 2.0 t.
>
> The tubes selected by Pullman for detailed measurement (the "D" dimension was 1.0t to 1.5t) were also used for weld tests. The weld tests were performed using both 3/32" and 1/8" diameter electrodes in four welding positions. The test coupons were cross sectioned and the flare bevel weld effective throats were measured. The effective throat was measured as flush with the side of the tube. No credit was taken for weld reinforcement. The actual weld effective throat, as shown in Tables 2A and 2B of ..., is, in all cases, at least equal to or greater than the dimension required by the design calculations.
>
> The use of tube steel with corner dimensions D of 1.0t, 1.25t or 1.5t, measured by the flat side intersection method, has no effect on the adequacy of pipe supports or their flare bevel welds. As demonstrated by weld tests, corner dimensions with R_D less than 2.0t have no adverse effect on the weld effective throat. The effective throat used in the design calculation is always equalled or exceeded. Regardless of how the tube corners are measured, the flare bevel welds are more than adequate. (DCL-84-166, pp. 50-52)

Figures A, B, and C, and Tables 1A, 1B, 2A and 2B referred to above, are shown on Exhibit 1.

Mr. Stokes persists in deceptive semantic exercises and presents obviously absurd ideas. In pages 2 and 3 of his affidavit, he indicates he is not going to propose a novel method for measuring tube corner radii. Yet, one method he used to evaluate the <u>outside</u> corner radius condition was to use a weld fillet gauge on the <u>inside</u> corner (see Stokes Attachment 1, middle of page, left side) and not a

concave radius of curvature gauge as he implies. The welds are made on the outside corner radius, and this is the dimension which must be used in calculating the effective throat.

Mr. Stokes' assertion that the only conservative radius measurement method to use is the straight line method is to quibble over trivialities. The differences in corner measurements methods were studied and reported in letters DCL-84-141 and DCL-84-166 to the NRC, cited above. Test welds in all qualification positions were done using the same welding electrode sizes as those on the project. The tubes used in those tests were the worst case examples of tube corners that Pullman QC could find. The weld effective throats were measured and found to exceed the values used in the design calculation by 20 percent or more. This demonstrated that there was no basis for concern regarding tube corner measurement and weld effective throat.

Mr. Stokes has also alleged that there are cracks in spherical bearings. This allegation is not a new issue; it was identified and documented by Pullman's quality control program. It is currently under review by the Project on Pullman Discrepancy Report No. 8175 (initiated March 23, 1984) and PGandE Nonconformance No. DC2-84-RM-N007. The discrepancy resulted from the discovery of two sway strut paddles with cracked spherical bearings in Unit 2. The investigation addresses the cause, safety implications, reportability, steps to prevent recurrence, and corrective actions.

We do not understand Mr. Stokes's reference to swaging. Contrary to what Stokes says, swaging is not a pressure or hammer welding process; it is a forming process related to forging.

The discovery of two cracked spherical bearings in Unit 2 is simply another example of a properly functioning quality control program and not "increasing empirical evidence" confirming a problem as alleged by Mr. Stokes.

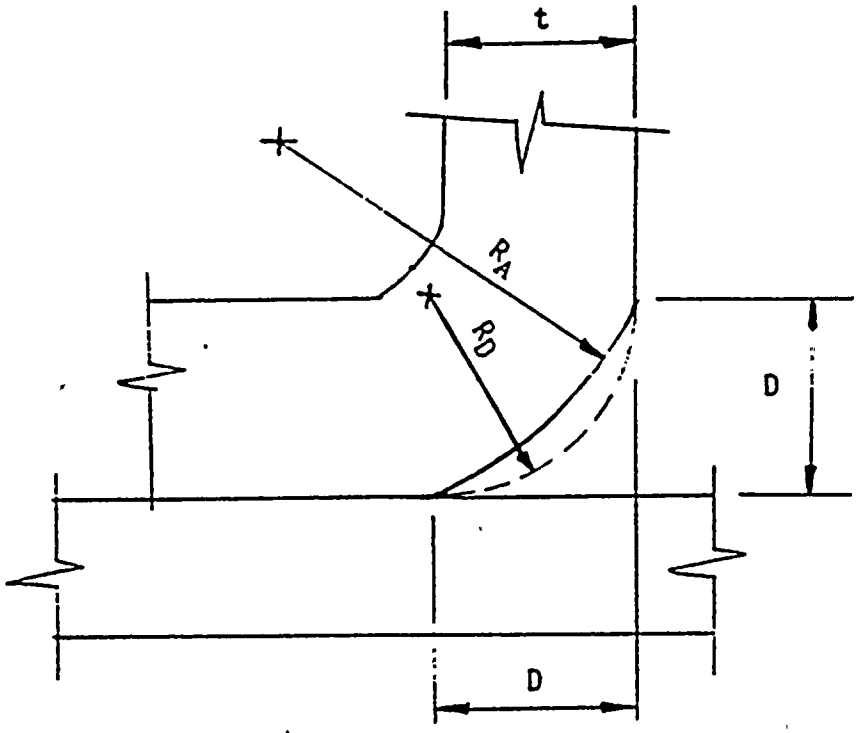
Mr. Stokes has alleged that spring-can attachments are defective. It must be noted that this is not a new issue; it was addressed in PGandE's Response to Joint Intervenors' Motion to Reopen DQA. (DQA Breismeister, et al., p. 25.)

> It involved support 1029-5CS, which is a constant rate spring support used as a dead load support on a 28-inch steam line (not 20-inch as alleged in the affidavit). The "cracks" mentioned in the affidavit were in fact laminations as determined by ultrasonic testing which commonly occur in this type of SA-36 plate, and it is not surprising that ultrasonic or magnetic particle testing would indicate this condition existed. These laminations do not detract from the component's load capacity. In addition, the pieces exhibited punching marks which the component manufacturer has certified do not affect the component's capacity or function.
>
> To be conservative, the contractor returned four of the ten brackets included in the order while the two on 1029-5CS were scrapped." (DQA, Breismeister et al., p. 25)

Additionally, the 25 "defective spring-can attachments" referred to in this latest variation of the previous allegation are undoubtedly

the six returned or scrapped brackets listed above and 19 brackets of other sizes exhibiting punching marks which were also returned and not, as implied by this allegation, spring-cans in use at the site.

Exhibit 1 of Response to JIR 18 and 19



t = wall thickness of the tube

R_D = implied radius

R_A = actual radius of the corner curvature

D = corner dimension

Fig. A

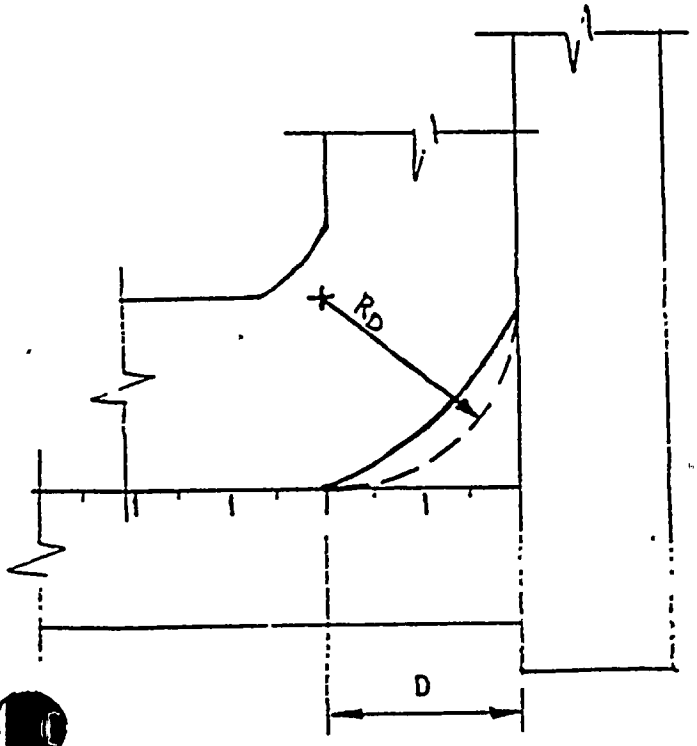


Fig. B

Straight Ruler Method

The implied radius R_D is taken as the measured D dimension

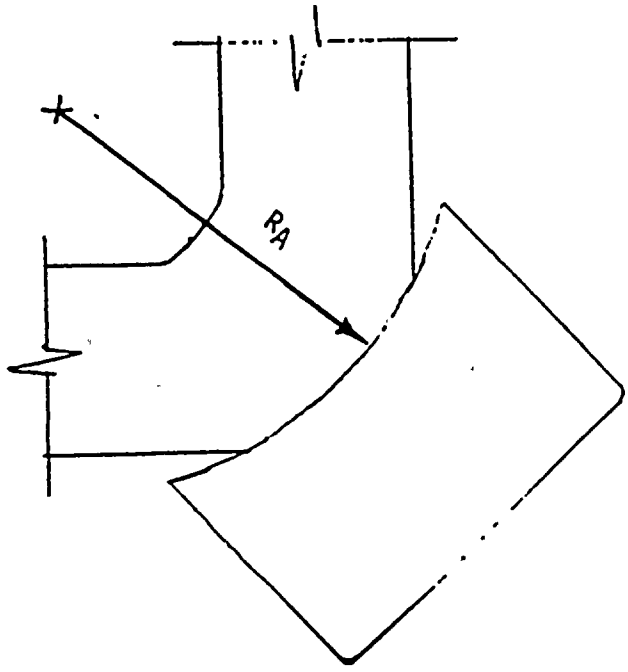


Fig. C

Concave Radius Gage Method

The actual radius of the curvature is measured

TABLE 1A

3 x 3 x 1/4 SQUARE TUBING

Number	Sample	Corner Dimension (D)*		Radius of Curvature (R_A)**
#1	A	11/32	13/32	1/2
	B	7/16	3/8	1/2
	C	3/8	7/16	1/2
	D	13/32	11/32	1/2
#2	A	11/32	13/32	1/2
	B	13/32	3/8	17/32
	C	3/8	13/32	17/32
	D	3/8	11/32	17/32
#3	A	3/8	13/32	17/32
	B	13/32	11/32	1/2
	C	15/32	13/32	1/2
	D	13/32	3/8	17/32
#4	A	3/8	7/16	17/32
	B	7/16	11/32	1/2
	C	3/8	3/8	1/2
	D	7/16	3/8	17/32

*Corner dimension, D, equals implied radius, R_D. (Tangent Intersection Method)

**Radius of curvature, R_A, always equals or exceeds 2.0 t.

Exhibit 1 of Response to JIR 18 and 19

TABLE 1B

3 x 3 x 1/2 SQUARE TUBING

Number	Sample	Corner Dimension (D)*		Radius of Curvature (R_A)**
#1	A	17/32	15/32	1
	B	1/2	17/32	1
	C	9/16	17/32	1
	D	17/32	9/16	1
#2	A	1/2	1/2	1
	B	17/32	9/16	1
	C	15/32	7/16	1
	D	7/16	15/32	1
#3	A	1/2	15/32	1
	B	9/16	1/2	1
	C	9/16	17/32	1
	D	15/32	9/16	1

*Corner dimension, D, equals implied radius, R_D. (Tangent Intersection Method)

**Radius of curvature, R_A, always equals or exceeds 2.0 t.

TABLE 2A

TEST RESULTS OF EFFECTIVE THROAT OF FLARE BEVEL WELDS

3 x 3 x 1/2 TUBE, CORNER DIMENSION = 1/2" (NOMINAL)

Position	1/8 Dia. Electrode	3/32 Dia. Electrode
1G	31/64, 30/64. 32/64	27/64, 26/64, 26/64
2G	28/64, 28/64, 29/64	26/64, 26/64, 27/64
3G	32/64, 31/64, 34/64	26/64, 24/64, 26/64
4G	31/64, 28/64, 30/54	26/64, 24/64, 25/64

NOTES:

1. Effective throat used in design calculations = 20/64".
2. Above test results show that the actual weld effective throats exceed the value (20/64") used in the design calculations.

TABLE 2B

TEST RESULTS OF EFFECTIVE THROAT OF FLARE BEVEL WELDS

3 x 3 x 1/4 TUBE, CORNER DIMENSION = 5/16" (NOMINAL)

	1/8 Dia. Electrode	3/32 Dia. Electrode
1G	14/64, 14/64, 16/64	17/64, 18/64, 17/64
2G	16/64, 16/64, 17/64	19/64, 18/64, 19/64
3G	18/64, 18/64, 17/64	19/64, 19/64, 14/64
4G	18/64, 19/64, 19/64	21/64, 20/64, 19/64

NOTES:

1. Effective throat used in design calculations = 10/64".
2. Above test results show that the actual weld effective throats exceed the value (10/64") used in the design calculations.

JIR-20

It is alleged that:

In PG&E letter No. DCL-83-166 on page 48 in the last paragraph, PG&E states "Due to the rapid expansion of the Diablo Canyon plant staff, specific training programs were conducted regarding AWS A2.4 weld symbols. Three hundred and fifty (350) engineers and QC inspectors were trained during May, June, and July of 1983. Additional pre-certification training was conducted for the AWS Certified Welding Inspectors Program in June-July and November-December, 1983." (Emphasis added.). [sic]

When you read PG&E's responses to valid problems such as this, their answers are humorous.

Maybe 350 people sounds like a lot. Let me put the number in perspective. How does 350 compare to 7000 relevant employees at the site? Mathematically the ratio is 1 in 20. When you consider that 350 were typically in management, the number left in the field to spread the knowledge around brings the ratio to more like 1 in 50.

[illegible or deleted] As of March 1983, according to PG&E Unit 1 modifications were complete. What good does it do to train the help after the job is finished? (6/1/84 Stokes Aff. at 5-6.)

In response to Stokes' rebuttal, as has been previously stated in response to previous allegations, there was weld symbol training:

> The allegation is ... false regarding lack of training. Weld symbol training was conducted for 350 personnel during May, June and July, 1983. This program was implemented to ensure that all personnel were using the same system and were aware of the symbols in use. In addition, pre-certification Certified Welding Inspector Training Courses were given to engineers and inspectors at the Diablo Canyon site in June-July, 1983, and November-December, 1983.
>
> Those individuals involved with welding at Diablo Canyon are assumed to have an adequate working knowledge of weld symbols due to previous education,

> experience and/or training and weld symbols are commonly available in reference documents. Because of the previous knowledge of all personnel involved in welding and related engineering and QA/QC activities, and the ready availability of applicable information, there is adequate confidence that the as-built drawings transmitted to Project Engineering both prior to and after February 15, 1984 are accurate." (DCL-83-166, pp. 79-80)

Mr. Stokes again brings up this point by once again distorting the issues with invalid comparisons. There were not 7,000 "relevant" employees on site as alleged. Stokes is absurdly counting sweepers, typists, maintenance personnel, guards, carpenters, pile drivers, etc., as "affected" by weld symbols knowledge. The 350 personnel trained represent engineers and inspectors involved in activities directly related to welding symbols, and are a large, significant percentage of "relevant" personnel. For example, at the time that there were 7,000 employees, there were a total of 818 engineers and 461 QA/QC inspectors. Obviously, not all engineers and inspectors work with welding. It must be recalled that knowledge of weld symbols has also been developed by previous education, experience, and/or training, and that the weld symbols are a commonly available reference.

This training was conducted because use of a common language for communication was deemed to be of some importance. The benefits, if any, will be realized in Unit 2 construction and Unit 1 maintenance. Another potential benefit of this training was to assist in looking for significant errors in Unit 1, if such errors existed, and no such errors were found.

It is alleged that:

1. AS-BUILD HANGER DRAWINGS - In summary there are many aspects of the as-builts which do not supply adequate information, it is PG&E's intention to get any information which is incomplete and required for the review. This appears to be acceptable to the staff. It [is] impossible for the engineer in an office to know if the drawing with which he has been provided is an accurate representation of what exists in the field. There are details which can tip him off to problems. The first place to look is at the concrete pour details showing embeds, conduit, ground cable, and drain lines.. These items could be the reason the bolt pattern is as shown on the drawing and in the case of drain lines may lower allowable bolt loads. The drawing usually shows the type and size of anchor bolt. When large bolts (1" and 1-1/4" or larger) are shown or the bolt embedment length is shown and the embedment requested to be installed is 6" or more. It is important to check the slab thickness because as a guide it should be twice the bolt embedment depth. The concrete drawings are necessary to determine the thickness of the slab or wall. Thin slabs and walls (those not twice the maximum bolt embedment used) may require calculations under the pipe support loads to determine if they are safe. Three bolt problems exist which have not been addressed in the calculations. 1) No review has been made checking the embedment length to slab or wall thickness for effects on bolt interaction on the opposite face of the slab or wall. 2) No review has been made of the effects to the concrete member subjected to a large load which is indicated by the use of the larger bolts and thru-bolts from a single support. 3) Similarly, no review has been made of the effects to the concrete member from multiple support loads.

Members shown intersecting other members at angles other than what appear on the drawing to be square should be a clue that fillet weld may not have full effective throat. This depends on whether the member intersects another at an angle less that [sic] 60°. Dimensions or and [sic] included angle should be requested from the field in any case where the drawing appears to show members intersecting at other than 90°.

Another item which should be included, was admitted by Mr. Tressler [sic] on page 215 of the May 9 transcript. "I guess another case is I remember seeing a DP where the field asked the engineer of [sic] three threads of nut

engagement was [sic] acceptable on this support. I don't know for sure what the answer was, but if the answer was yes, we don't as-built thread engagement." This problem requires that each support be checked for a full nut thread engagement. The engineers performing the review can not verify the adequacy of the bolted conncection with out [sic] knowing the thread engagement. Will the staff require that thread engagement be as-built knowing that some bolts exist with only three (3) thread engagement?

Other things to watch for are in [sic] accurate or insufficient detail. At times items will be shown but not in enough detail for an engineer to perform an analysis or evaluate the effects through the use of good engineering judgement. I have seen supports which looked to be quite strong on the drawing but upon a closer examination in the field were found to be completely free to slide in the direction of pipe thermal movement. Another example that comes to mind was a plate attachment near the edge of a wall. The original drawing indicated that the plate was supported under its' [sic] full length by concrete but after a field trip, it was found to be cantilevered out over the edge so that a large bending stress was placed in the plate. I should note that the anchor bolts were over stressed [sic] when considering the cantilever.

One form of information, which is available and could be used to provide the missing kinds of information on problems which must be reviewed can be provided by including the PSTDC [sic] forms or Quick Fix sheets with the As-Built Drawings. Hard to find information relevant to the design review process is frequently shown on the PSTDC [sic] form as the reason for a modification. It is true that the information was obtained thru "trial and error," but that process of finding necessary information for review might prove more efficient than trying to locate the applicable concrete pour drawing which would only provide approximate information. Construction drawings are not very accurate in many ways. For example, the slab dimensions are usually to a quarter inch, but the location of reinforcing steel at best is only good for a rough spacing requirement. Accuracy for the location of items like drain lines and ground wire is none [sic] existent. Typically drains [sic] lines are supposed to be place near the center of a slab but while in the PSTDC [sic] group, I had to write a Quick Fix when a construction crew drilled through a drain line when installing anchor bolts. The location and depth can be determined with more acuracy when checking the anchor bolts for reduced allowable load

reductions from the Quick Fix form than from any concrete drawing. Other important information might be that the slab or wall is only 10" when it was supposed to be 12".

Beyond the fact that the Quick Fix sheets show critical information which should be reviewed, there is the necessity to have engineering review these documents and give their approval of the installed change as is the standard procedure when a design change request is made. The only difference here is the issuer didn't perform any calculations. Thus requiring both the origination of some calculation or other basis for acceptance from the applicable engineering department [sic] . Without this act, we will uproot the foundation which has been laid for a good QA program.

The review of the PSTDC's [sic] and the DP's has no formal procedure by which to work. PG&E doesn't appear to be committed to reviewing either of these two critical programs. In the May transcript on page 204 Tressler states that 2,120 DP's were reviewed, we found 429 related to piping or pipe supports and out of these 55 were found to contain design information which applied to pipe or pipe supports. What were all the rest written against structural steel, concrete or what? If there was were [sic] 55 DP's which contained design information, why are't [sic] the others being reviewed by the appropriate design departments? In the case of PSTDC's, [sic] out of approximately 15,000 only 1100 have been reviewed of which only 20 small and 20 large bore were selected as having extensive modifications made by the PSTDC [sic] group. Here the staff was insistent that all be reviewed. The detail of this review is not clear nor is there a clear purpose behind it.

All of the examples used are hardware problems that have been confirmed in the field. Our engineering calculations did not take them into account because they weren't on the drawing. The effect on public safety is unknown, but could be devastating. Problems also loom over the NEW staff corrective action programs to verify the calculations but not the hardware. The drawings are still incomplete, so significant problems won't be considered. The conditions of the hardware must first be confirmed, before we can produce reliable engineering conclusions. (6/1/84 Stokes Aff. at 6-13.)

In this allegation, Mr. Stokes maintains that the accuracy and completeness of a pipe support as-built drawing is an important element in the computer analyzed small bore pipe support review process. DCP agrees with this point. The Project procedure developed to govern this program provided a detailed check list which included requirements to review the as-built drawing to confirm that all the information required to perform the support analysis was provided. Any necessary information that was missing was provided by onsite inspection of the installed support.

Mr. Stokes includes several specific concerns in this allegation which are addressed as follows:

A. The various concerns relating to bolt embedment, slab thickness, effects of bolting from opposite sides of concrete elements, and civil verification of support loadings previously were addressed in detail in Section 6 of letter DCL-84-203 dated June 1, 1984, to the NRC as follows:

> 6.3.2.1. DESIGN CONTROL PROCEDURE
>
> A question was raised regarding design control procedures and criteria used for relating pipe support designs and installations to civil engineering to achieve civil verification.
>
> When a pipe support design has been completed by the piping group, coordination with the civil group is effected in accordance with Procedure No. P-22, "Procedure for Coordination of Pipe Support Loads with Civil Engineering." The P-22 procedure provides

criteria for determining which pipe supports require approval by civil engineering. The criteria include consideration of (a) the magnitude of the pipe support loads imposed on the structure, (b) the location where the support is attached to the structure, and (c) the intricacies of the support configuration. These criteria were developed jointly by the piping and civil engineering groups.

For supports requiring civil approval, a review package, including a sketch of the support and a summary of the loads imposed on the structure, is sent to the civil group. Large bore pipe supports are transmitted via a CV (civil verification) form. For small bore pipe supports, which are designed by the Onsite Project Engineering Group, civil approval is requested via a field transmittal form. In accordance with piping Procedure P-6, "Procedure for Assembling Pipe Support Calculation Packages, Units 1 and 2," a calculation checklist is included in each pipe support calculation package. This checklist contains a block for indication of whether civil approval is required and provides a double-check for ensuring that all required pipe supports receive civil coordination.

Upon receipt of CVs or field transmittals, the pipe support loads imposed on the structure are reviewed in accordance with the applicable civil DCMs. The pipe supports with significant loads are plotted on structural drawings so that cumulative effects of numerous pipe supports in the same area can be readily accounted for in the support review. This plotting also aids in the evaluation of global structural effects as well as the local effects. When the piping loads are significant, civil calculations are prepared to ensure the adequacy of the structures. Civil calculations conform to the requirements for preparing design calculations contained in the PGandE Engineering Manual Procedures. Based on this evaluation, approval or rejection of the pipe support is noted on the CV or field transmittal and is signed by the cognizant civil engineer. Field transmittals are returned under the signature of the civil EGS. In cases where support designs are rejected, the reasons for rejection are identified and frequently design alterations are suggested. Revisions to support designs, whether due to (a) civil group rejection, (b) field construction problems, or (c) changes in the piping analysis, are again coordinated with the civil group in the same manner described above.

6.3.2.2. HILTI KWIK-BOLTS

A question was raised regarding the effect on the adequacy of pipe supports when (a) Hilti Kwik-Bolt expansion anchors are embedded deeply into thin concrete elements and (b) Hilti Kwik-Bolts are installed in close proximity to each other, but from opposite sides of concrete elements. In order to respond to these questions, PGandE consulted with Hilti Fastening Systems, Inc. engineers.

Regarding question (a), Diablo Canyon expansion anchor installation criteria do not limit the depth to which Hilti Kwik-Bolts may be installed. Hilti engineers have recommended to some of their clients that the embedment depth be limited to 80% of the concrete element thickness. This recommendation is based on Hilti's judgment that deeper embedments may result in spalling concrete on the back side of the concrete element when hammering the anchor into the hole. Neither Hilti nor PGandE is aware of any analytical or test data that validate this recommendation. Further, Hilti has not published this recommendation as it is not considered to be an installation requirement but rather is an optional precautionary measure.

At Diablo Canyon, QC inspections noted a few cases in which concrete was spalled during installation of Hilti Kwik-Bolts in a thin (12-inch) slab. The spalling occurred either while hammering the bolt into the hole or during the torque-setting operation. Subsequent inspection found that the spalled concrete did not extend into the concrete surrounding the anchor wedge. The anchors were set in accordance with normal installation procedures and held the final torque, 360 ft-lb. The spalled areas were then repaired by drypacking.

Anchoring of Hilti Kwik-Bolts is achieved by forcing spring steel clips at the wedge shaped base of the anchor into the surrounding concrete. Expansion anchors are not loaded in compression so the concrete below the clips carries no load. Further, spalling of the concrete below the clips does not affect the strength of the concrete shear cone that anchors the bolt when it is subjected to tensile loads. While spalling of the concrete below a Hilti Kwik-Bolt is not desirable, it does not affect the adequacy of the anchor. Thus, there is no technical reason to limit the depth of embedment.

Regarding question (b), Diablo Canyon expansion anchor installation procedures do not require mapping of anchor locations on opposing sides of concrete elements. In practice, it would be difficult to accomplish this mapping within the accuracy necessary to identify close spacing of anchors on opposite sides of walls and slabs. Since most of the concrete in the safety-related structures at Diablo Canyon is thick (greater than 12 inches), and most of the expansion anchors require installation at relatively shallow embedments (less than 6 inches), the potential for this type of overlapping to occur is extremely low.

Hilti is currently performing tests to quantify the effects of closely spaced anchors installed in opposite sides of concrete elements. These tests are being performed at the request of another utility and are being monitored by cognizant NRC personnel. A final report is not expected to be completed until August 1984, but preliminary indications are that there is no significant reduction in anchor strength, even when the anchors are installed as close as 1-1/2 bolt diameters (center-to-center). These preliminary results support the judgment of the Hilti engineers (and that of PGandE Engineering) that an overlapping condition has a negligible effect on the adequacy of the anchors.

B. As discussed previously, the support as-built was scrutinized as a part of this review program and any insufficient information or questionable details needing clarification to allow support analysis were confirmed by onsite inspection of the installed support. The concern raised about a single documented case of questionable bolt thread engagement is clearly an isolated occurence which was documented by a DP and does not warrant further generic review.

C. Mr. Stokes continues to contend that PSDTCs are not incorporated in the pipe support as-built review package. His understanding is incorrect. It is true that not every PSDTC sheet is individually

reviewed and accepted by Engineering in the as-built acceptance process. However, PSDTCs are incorporated in the final hanger as-built drawing which does receive Engineering approval as part of the as-built acceptance process.

When construction of a pipe support is complete, Pullman engineering prepares a final as-built drawing of the support to reflect the actual final, as-installed configuration of the support. This process double checks that the actual, final physical condition of the support is accurately reflected on the as-built drawing. This drawing incorporates any PSDTCs which have altered the support design during the course of its construction and which account for its final configuration as it exists in the final installed condition. It is this as-built drawing which is then sent to Engineering for final acceptance.

The as-built acceptance process involved review of the the revised support design and performance of necessary calculations for qualification of the design. Where qualification could not be shown, a new design was prepared and issued for Construction.

This comprehensive process of review and acceptance of changes made by PSDTC as a part of the as-built acceptance program has been corroborated by recent NRC staff audits.

D. The Diablo Canyon Project program to address and comply with Low Power License Condition 2.C.(11), Item 6, involving PSDTC and DPs is fully defined and the results described in the final report on this issue were submitted to the NRC by PGandE letter DCL-84-203 dated June 1, 1984. This program concluded that adequate Project and Contractor procedures and controls were in place to assure that all as-built designs are shown on drawings and are qualified by calculations as appropriate. The conduct and results of this program were reviewed by recent NRC Staff technical audits.

It is alleged that:

2. LOAD DETERMINATION - PG&E's response to the staff has been incomplete.

To adequately protect public safety through conservative analysis, the following factors must be considered. In order to begin any calculation or review process the first step should be to decide how we will determine the loads to be used. This is important to provide uniformity in load determination and consistency of application to the structure which is relevant for both small and large bore. Due to the fact that less than all possible combinations of loads were actually run in the analysis, I feel that some procedure is required to be followed to ensure that the maximum possible load combination has been analyzed. This problem was one which existed during our review work, which I felt was far too vague and thus allowed failing structures to pass. (See Attachment 3a,b,c)

Attachment 3a is the form supplied by management for the pipe support engineers to compute the loads for their calculations. I have added what I believe to be clarification letters to this form which were not part of the original, or were used by only a few other engineers. I added the (A) or (B) letters to the load cases on form 3a. Though it may not be obvious the difference between 1A and 1B, 2A and 2B, etc. is quite simple. All cases with an (A) are the most positive value that can be determined using the formulas and notes on the bottom of the form and all cases with a (B) are the most negative values, also based on the formulas and notes on the bottom of the form. Most of the time I only used this form for the most simple support, ie. [sic] one (1) pipe, 2 directional restraint. For this type support and using this form, four (4) combinations are possible for Load case 1, not just the two (2) which are on the sheet by the 1A or 1B, but also the diagonal cases.

Not all OPEG engineers were aware that pipe loads are dynamic and each direction should be assumed to be independent of the others unless a very detailed analysis has been performed demonstrating otherwise. At Diablo the practice was to also allow each engineer to perform his

analysis with whatever he was able to get a checker to also accept. In sum, the new review must have procedures directing analysis from all relevant directions. The current methodology does not guarantee this necessary analysis.

Forms 3b and 3c have been included in Attachment 3 along with a discussion of their development. They are intended to be used as an example of the kinds of detail which should be included in any procedures on load determination, as well as examples of changes to the forms which should be made so that all engineers are consistent in the load calculations. The types of supports addressed in this discussion are the most difficult and are the ones which are performed using STRUDL. These are being reviewed because of errors which were found by the staff.

When performing an analysis requiring the use of STRUDL on a simple support as described above, I usually ran four load cases to guarantee that I obtained the maximum stresses. For a gang (a number of individually numbered supports welded together at one or more points) support, it was not possible ro run all possible combinations. Gang supports are the most difficult supports to analyze. They require the best analysis and the most experienced engineers are assigned to them.

For these supports, I developed two (2) forms which caused me to spend a little more time in determining the load case I would analyze but reduced the number of combinations from each pipe to only one (1). Actually if anchor bolts were involved, there would be two (2). However two individual runs could be made, one to check stresses for frame members, welds, component hardware etc., and one to perform the base plate and anchor bolt calculations. I also wrote (on my own time) and had a co-engineer check a program for an HP41C to perform the combinations required in the flow chart on the top of Attachment 3b. This program provided the same results for load cases (A) or (B) as explained earlier. The reason for my doing all this was to keep up with the production requirements since I was usually given gang supports and multi-pipe supports (support restraining multiple lines but only assigned one support number) to analyze. Production credit was based on support numbers not number of pipe restrained. When I was laid off, I was and had been working on extremely large gang and

multi-support structures since approximately the 1st of March 1983 except while in the PSTDC [sic] group. Two which I had worked on just before going to the PSTDC [sic] group in June were reassigned to me. The structural steel loads had been sent to the city for approval and had come back rejected. The pipe support loads were causing too much torsional load on some fairly large W14's [sic] (structural steel) which were designed as pined end beams attached with angle clips.

I am not proposing that my forms be used, but I hope the NRC will see the necessity for procedures to ensure that all engineers can concistently obtain the same load values to be used in the calculations. That control still does not exist in the PG&E/STAFF program. The loophole means that the same deficiencies could repeat themselves, which would disqualify the results of the current corrective action. (6/1/84 Stokes Aff. at 10-13.)

In this allegation, Mr. Stokes presents his personal opinions on how he would document load cases used in the computer analyzed small bore pipe supports. The review program recently completed in response to Low Power License Condition 2.C.(11), Item 1, fully evaluated the load combination used in the calculations even though not specificaly required by the License Condition. The method of load combinations actually used by DCP during the review to determine the worst case loading condition on a pipe support meets licensing requirements and is consistent with industry practice. Also, it is basically consistent with the methods Mr. Stokes advocates in the allegation. The forms Mr. Stokes would use to document the results are different, but the methods and the results are the same. One can always suggest a different method of documenting the results of a calculation as

Mr. Stokes has done here, but that doesn't mean the method used was in any way improper as insinuated by Mr. Stokes.

Contrary to the allegation, the Diablo Canyon load combination methods provide conservative results that satisfy all licensing criteria. For example, the total combined load from the pipe is calculated by combining the dead load, thermal load, and dynamic loads. In the enveloping process for each pipe restraining direction, the thermal expansion load may not be used to decrease the combined loads (DCM-M9). This assures that the maximum load from the pipe will be calculated by one of the load combinations of Attachment 3A to Mr. Stokes' affidavit. Attachment 3B to the affidavit is merely another way to document a very simple arithmetic thought process. If Mr. Stokes feels he needs this additional documentation for such a rudimentary engineering computation, he is entitled to his opinion. However, such documentation is not required as the results obtained by using either method are the same.

Contrary to the allegation, the dynamic nature of the seismic loads was considered in the load combination. The DCP form that is Attachment 3A to the allegation requires the load combination to consider the seismic loads as + or - events. Even if the engineer

was unaware that the seismic loads are dynamic, the form forces the engineer to consider the dynamic nature. This, in combination with the worst case static loads, assures that the most conservative load combination will result.

It is alleged that:

3. STRUDL MODEL - Several details should be addressed which are critical to obtaining consistent results by all engineers performing the review. To date, they have not been covered by the PG&E/STAFF responses.

Guidelines should be defined for accuracy of dimensions, both of members and eccentric connections. The use of the word significant as a standard by which to build a nuclear plant has no reasonable basis, nor can a definiton be given to the word such that all the engineers could apply it consistently - one to another. A standard value of (+ or -) 1/2 inch would ensure that a consistent level of accuracy is maintained. On other projects which I worked, the standard was (+ or -) 1/4 inch. Procedures should be more specific and beyond misinterpretation.

Second, project specific instruction should be supplied on the determination of Beta angles. There were many non-controlled documents on how to determine the Beta angles from all over the country within OPEG. This was not any different from the San Francisco office, as employees there confirmed to me. Due to the number of angle members in the small bore supports, I feel it imperative that tables be given to the engineers which have been approved by management indicating the correct Beta angle depending on position of the member.

Third, instructions should be given the engineers on when to use joint releases. This should include sketches of the detail and how it should be modeled. There should be a free atmosphere within which questions may be asked both for clarification and to question the validity of the proposed instruction or any technical aspect of the review program.

Finally, guidance on modeling members which have had a part of the cross section cutout for any number of reasons should be provided as examples of good modeling practice. These could range from a hole through the web of a wide flange (WF) to the hole member having been cutout and

displaced in one direction or another. This problem with notches and cutouts are stress related. Typically, details are not paid very much attention. When a member has had its cross section reduced, stresses are intensified in the reduced zone and unless an engineer considers the transferral of stresses from one side to the other - an overstress can exist.

On balance the failure to provide methodological controls such as described above, means that there is no guarantee that the PG&E/staff corrective action program will catch all significant problems. These factors must be considered. (6/1/84 Stokes Aff. at 13-15.)

Mr. Stokes details four specific issues. He then speculates that inadequate attention will be given to them during the small bore pipe support review program for computer analyzed supports undertaken in response to License Condition 2.C.(11), Item 1. In fact, the review of these pipe supports has been done using a comprehensive procedural check sheet which includes review of the items identified by Mr. Stokes in addition to a wide number of other technical points. The review is essentially completed and in each and every case, even after any discrepancies were resolved, the evaluation showed that all supports fully met the licensing criteria and commitments. Furthermore, the adequacy of the support design review has been corroborated by extensive NRC Staff technical audits of this work.

Contrary to the allegation, it was appropriate to use a general guideline, such as significant, for the determination of dimensional accuracy. The reason being that no "across-the-board" standard or criteria will fit all cases. It is an engineer's responsibility to define the necessary accuracy as it applies to the specific case being reviewed.

In regard to the specification of proper Beta angles, the Project has done a study of equal leg angles to establish that in most cases Beta angles have an inconsequential impact on the qualification of the support. However, in order to assure consistency of approach among engineers in the application of Beta angles, the Project has developed Procedure I-52 to provide guidance to all engineers. The Project has reviewed all STRUDL analysis to assure the proper application of Beta angles.

The joint release issue has been treated at length in previous responses to Mr. Stokes' allegations. To reiterate:

1. "Joint Release" is a recognized structural analytical technique to consider rotational flexibility at member connections.

2. The use on Diablo Canyon is limited to unique cases that can be fully justified.
3. The re-review of all STRUDL analyses has been completed and the use of joint releases fully justified.

Contrary to the allegation, the Project has contacted other major A-E firms and consultants and have been told that joint releases are frequently used in the design of pipe supports and, with one exception, all of the firms stated they did not feel procedures were necessary to control its use as maintained by Mr. Stokes.

The Project recognizes that reductions in cross-sections due to as-built conditions should be modeled when necessary and clearly does so when necessary. In fact, reduced section members have been considered in conjuction with the STRUDL re-review and the stresses at the reduced cross-section satisfy Project criteria.

It is alleged that:

4. STRUDL STRESS SUMMARIES - These represent another problem not covered in the current corrective action plan. At Diablo, each engineer was allowed to check as little as his checker would agree to accept. Even the work at Zimmmer was better in this respect. Enclosed is a copy of the CHECKLIST FOR STRUDL FRAME ANALYSIS (Attachment 4a) which was used at Diablo by OPEG. Also attached is a copy of a form which was used at Quadrex Corporation for checking the stress combinations of members which were analyzed by Strudl. (See Attachment 4b) I adapted the basic form to the Diablo Title for my use at Diablo, but was not allowed to use it. This form was included as an example of how to properly combine the torsional shear stresses (see line 18) and also how to combine the warping normal bending stresses with the axial and bending stresses (see line 21). For the record, I would like to state that this form was part of the calculation package for all small bore supports which were performed by Quadrex on Zimmer Nuclear Plant. I would like to say that a fortran program based on the same technical literature as the Bethlehem Steel design guide on TORSION ANALYSIS of Rolled Steel Sections was used to compute the warping shear and warping normal stresses (lines 7 and 8 of attachment 4b) If there is any remaining question whether my challenge to the loophole is reasonable, in 1980 at the Gaithersburg, MD office of Bechtel, we used this same program. It was on Bechtel's own system. The subject of warping stresses is not new either to the engineers at Bechtel or PG&E.

Another point should be raised in discussing Strudl stress summaries, the use of strudl input forms at OPEG. These forms had all load combinations already written out. When I reviewed the combinations in detail to make sure all load cases were being combined which effected [sic] stresses in the Load Combination statements, I found that the [sic] of the structure was not being included in the stress results. "Self-weight" is the load of the structure itself. In some cases, the stresses on members from "self-weight" load alone will be sufficient to fail the support, and in all cases this component of load is important in evaluating the acceptance or failure of that support.

This was not obvious since in load case five (5) or so there were three (3) load cases written to provide the self weight in the X, Y, and Z directions. These were being run

and combined with the tributary load (also effective weight of pipe system in each of the respective directions) for output as the displacement by which we were to check the stiffness of the structure. Apparently, the writer of this form failed to add the necessary load cases in the load combination statements which are combined for stress output. After this discovery, I wrote my own input forms.

I would suggest that the NRC have an in [sic] depth look at this form if it is being used in the review work. The public record does not demonstrate that this problem has been corrected. If there is any question about the significance of this error, some supports will fail under their self-weight without any pipe load being applied.

Another point on this subject must be addressed, but is not clear from the public record. Why wasn't the deflection required to be checked? On all other plants with which I am familiar, a displacement limit was imposed at the point of the pipe attachment.

This was a secondary check beyond the stiffness requirement. It is important to verify the displacement under load for if the support moves in some cases too far the pipe may be overstressed. It is possible to have a support which meets the stiffness requirements and still will have large amounts of displacement. (6/1/84 Stokes Aff. at 15-17.)

In this allegation Mr. Stokes attempts to compare a form used by the Diablo Canyon Project for one purpose with a form, apparently from another project in his experience, obviously used for a different purpose. The two forms that have been included in this allegation serve totally different purposes and it is therefore difficult to draw a meaningful comparison. The first form, Attachment 4A to the Stokes affidavit, is to guide a checker in the performance of a review. The second form, Attachment 4B, is a form proposed by Mr. Stokes for use in documenting the stress combinations necessary to qualify a pipe support. It is <u>not</u> an official Project form. Most of

the calculations required by the form were performed by computer programs rendering Mr. Stokes' proposed form unnecessary.

The warping and torsional stress checks that are described in the allegation have been fully documented as required by Project Instruction I-9. It is interesting to note that none of the supports reviewed as part of the program to respond to License Condition 2.C.(11) item 7, exceeded allowable stress when warping was included. In 90 percent of the supports reviewed, the calculated stress was less than 60 percent of allowable stress when warping was included. This confirms that warping has an insignificant role in the structural capacity of pipe supports. There is no question that the phenomenon of warping exists and that there are computer programs that handle its effects. But there is no Bechtel standard computer program that evaluates warping. It is difficult to understand how it makes any difference considering: (1) the insignificance of the effect, and (2) the results of the recently completed additional review required by the NRC staff to investigate this issue.

The "self-weight" effects of a pipe support structure are very poorly explained in this allegation. It is believed the concern appears to be the same as that previously raised by the NRC staff. The concern centers around the ability of a pipe support to withstand the accelerations induced by an earthquake independent of any piping loads. As with warping, the "self-weight" effects are believed to be

of an insignificant magnitude, which is why they were not initially considered. However, the Diablo Canyon Project has committed to the NRC staff to evaluate these pipe support "self-weight" effects.

Mr. Stoke's proposal of a secondary check of support deflection, in addition to support stiffness requirements, is unnecessary. Compliance with the stiffness requirement obviates the need to check deflection with respect to pipe stress since such deflection would be limited by the high stiffness required to meet the 20 hertz natural frequency licensing criteria.

It is alleged that:

5. TORSIONAL WARPING NORMAL AND WARPING SHEAR STRESS - In PG&E's letter: DCL-84-164 Enclosure 7 on page 2 PG&E states "There are three considerations in the pipe support design at Diablo Canyon that tend to minimize the significance of the warping phenomena:

- The predominant use of wide flange sections rather than "I" sections or other sectional shapes having a lesser capacity to restrain torsional loads.

- The pipe supports are designed to use standard size members and a stiffness criteria that, in most cases, assure that the member stresses will not be the critical factor in the strength of the support.

- Small bore supports typically use angle or square tube section material that are not [sic] subject to warping." (Emphasis added.)

The statements by PG&E are ridiculous. Wide flange sections may be used predominantly in large bore, but even if they are that does not preclude many from being overstressed when torsional warping normal and warping shear stresses are added to the existing bending and shear stresses. This is especially true in large bore supports where very large loads are possible. The end connections are very important in how much torsion a member can safely carry. It is true that wide flanges have more torsional stiffness than an "I" beam, but it is equally true that a square, rectangular or ciruclar tube has far more torsional stiffness than a wide flange.

The second PG&E statement concerning the stiffness criteria may be somewhat relevant for the most simple support, i.e., a simple cantilever, but even here a counteracting force is at work. As mentioned in the paragraph above, the end condition has a lot to do with how much torsional loading a member can take. A beam which is Fixed-Fixed will carry more torsional loading than one which is Fixed-Pined, Pine-Pined, or Fixed-Free, providing only end conditions vary. Again the failure to consider all possible forces could be the difference between passing and failing the support.

The last statement has to be a misprint. I can't believe that PG&E woud allow such an obvious falsehood. I believe

almost any competent engineer can tell you that any material or shape will warp if loaded. In reply to the angle and tube shape comments of PG&E, I would like to quote several lines from a book published by the Lincoln Arc Welding Foundation "Design of Welded Structures." I selected this document because many engineers have one and use it regularly.

From Section 2.10-3 "Designing for Torsional Loading"

"The solid or tubular round closed section is best for torsional loading since the shear stresses are uniform around the circumference of the member.

Next to a tubular section, the best section for resisting torsion is a closed square or rectangular tubular section.

(Skip a paragraph)

The poorest sections for torsional loading are open sections, flat plates, angle sections, channel sections, z-bar sections, T-bar sections, I-beam sections, and tubular sections which have a slot." (Emphasis added.)

The truth of PG&E's amazing assertion should be tested fully, both for the validity of this theory and its effect on the design. I feel that when the evidence is laid out on the table that PG&E will have to admit their incompetence. (6/1/84 Stokes Aff. at 17-18.)

In this allegation Mr. Stokes displays difficulty in separating academic minutiae from practical engineering applications. The considerations identified by PGandE which tend to minimize the significance of warping in pipe support design at Diablo Canyon are not only valid, but have been verified in large measure by the recent review of large and small bore supports completed in response to Low Power License Condition 2.c.(11) items 1 and 7. The results of these reviews were reported in PGandE letters DCL-84-219 dated June 8, 1984, and DCL-84-223 dated June 11, 1984, to the NRC. The program included a thorough review of warping and concluded that

project criteria was satisfied for every case reviewed. The results of these reviews have been corroborated by extensive technical audits of the work by the NRC Staff.

Mr. Stokes' observation that "any material or shape will warp if loaded" is noted. In response the following are quotes from various sources:

1. From the appendix of Torsional Analysis by Bethelem Steel,

> For non-circular cross sections this rotation is accompanied by warping; that is, transverse sections do not remain plane. (An exception to this occurs in sections composed of plate elements having centerlines which intersect at a common point such as a structural tee).

Obviously, a structural angle would be an exception since it is also made up of plate elements, just as the structural tee.

2. From British Steel Design Manual, 4th Edition by Granada Publishing Limited; In Section 9, Beams in Torsion, the following is stated,

> The sections which remain plane after twisting may be defined as follows:
> (1) Round bars or cylindrical tubes
> (2) Open sections comprising two thin rectangles, the center lines of which intersect at a point, e.g., angle or tee sections.

As indicated in quote (1), sections remaining plane after twisting is the same as exhibiting no warping.

In view of this information it is obvious that Mr. Stokes has, at best, an incomplete understanding of warping stress and the types of cross sections that will exhibit this phenomenon.

It is alleged that:

6. ANGLE MEMBERS - In PG&E letter No: DCL-84-164 Enclosure 7 on page 3 papargraph (b), Differences Between AISC Code and Project Criteria, PG&E state "The so called 'differences' between AISC and the Project criteria using the Australian data, references 1,2, and 3, with respect to allowable stresses of angle sections in bending do not really exist."

While I have not had very much time to fully analysis [sic] the Australian papers, I have studied them sufficiently to conclude that they are not applicable for type [sic] of loading to which they are being applied by PG&E. I would like to quote several lines from the Australian paper "Laterally Unsupported Angles With Equal and Unequal Legs" Section 4. LOADING CASES. "Note that the location of the moment on the cross-section is not critical. However point loadings which are not applied through the shear centre will cause additional twisting (Section 12)." In Section 12. LOADS NOT THROUGH THE SHEAR CENTRE, additional formulas are given for the additional twist and the additional stresses due to twisting. On the bottom of Section 12, "A more exact and comprehensive solution to this problem can be found in (21)." Reference 21 is to an article entitled "Deformations of Geometrically Imperfect Beams" written by N. S. Trahair which was printed in Proc. ASCE, 95 (ST7) JULY 1969, PP.1475-1496.

I am sure that the Australian study was not intended to be used by the author in application to the types of end connections which we find in nuclear power plants. In Section 1. INTRODUCTION second paragraph, a discussion is provided as to the use of angles in structures such as transmission towers. The last line of this paragraph says "Even here, however, the underlying research has frequently been highly empirical with strut load capacities given for each member size under practical field conditions." From the qualifiers in the test, these studies were made srictly for expansion of knowledge in the area of transmission tower design.

I don't believe the Australian papers should be used in the design of nuclear power plants or the construction of any structure which would place the lives of people in danger for the following reasons: 1) The design of Class 1 nuclear pipe supports requires a much more conservative analysis than do transmission towers. 2) The joint details which

exist in constructing a transmission tower are easier to control than they are in pipe supports. Thus through a good joint design it is possible to apply loads to the shear centre avoiding the additonal twisting stresses inherent to a nuclear plant, discussed in the first paragraph. 3) The Bending moments to which the angles in the Australian papers are subjected are limited to those caused by wind combined with a small amount of axial load, neither of which require the same degree of evaluation as do the safety-related systems in a nuclear power plant.

On page 33 of PG&E letter: DCL-84-164 Enclosure 7 PG&E states "It should also be pointed out that the 18 pipe supports identified in the DR 83-042-S as discrepant have been reviewed. All of the angle beam spans are found within the Project Design Criteria."

This statement is false and misleading. This is not the DR which I signed and submitted to Leo Mangoba on 10/5/83. That document did not contain any pipe support numbers which I felt were discrepant. It did contain seven (7) pages which I had copied out of the ASIC. The pages which I included were marked up to show the sections which detailed the problem. I did give a list of supports to Jeff Van Klompenburg several days after I first submitted the DR in preliminary form back in 8/83. This was after I submitted the DR and it was returned with a note that I should provide a support number which was not within criteria as an example. In stead [sic] of one support, I submitted a list in rough form which contained approximately 100 Unit 1 supports which were not in compliance the [sic] AISC code section 1.5.1.4.6b or more specifically 76.0 b_f/ /F_y which for 2" angles = 25.3", and for 3" angles = 38", and for 4" angles = 50.68". I have no knowledge as to who added the 18 supports to the DR which I submitted to management on 10/5/83. (6/1/84 Stokes Aff. at 18-21.)

The first two paragraphs are Mr. Stokes' opinion that the use of Australian papers is not applicable to the type of loading used at Diablo Canyon and the studies were made to expand knowledge for transmission tower design.

The first point is covered quite well in Section 4 of "Laterally Unsupported Angles with Equal and Unequal Legs" by J. M. Leigh and M. G. Lay. Since this section is short, it is reproduced here in its entirety:

LOADING CASES

The behaviour of the beam is dependent on the axis about which the moment is applied (Fig. 4). Six loading conditions can be used vectorially to represent all possible cross-section loadings (Fig. 5). Taken individually they are:

Case I: (Fig. 6) Moment applied parallel to the long leg.

Case II: Moment applied parallel to the short leg.

Case III: Moment applied parallel to the U-U axis.

Case IV: Moment applied parallel to the V-V axis.

Case V: Moment applied parallel to an axis beween the U and Y axes.

Case VI: Moment applied parallel to an axis between the X and Y axes.

Note that the location of the moment on the cross-section is not critical. However point loadings which are not applied through the shear centre will cause additional twisting (Section 12).

In the following Sections each of these Cases will be examined individually.

As indicated by the second line in the first paragraph, the six loading conditions considered represent all possible cross-section loadings.

This statement, along with Figures 4 and 5, implies the loads are being applied at the ends of the members and are causing bending. For this method of loading the angle members, all possible loading is considered and no restrictions are necessary as to the type of structure of which the angle member is part.

The paragraph directly below Case VI states that if the point load (a load applied between ends of the member or support points) is not applied through the shear center, additonal twisting will occur. This additonal twisting has been included in the design of the angles.

Mr. Stokes further states the Australian studies were made strictly for expansion of knowledge in the area of transmission tower design. He does accurately quote from Section 1, second paragraph, but his conclusion is in error.

The second paragraph states:

> "The behaviour of angles as compression members has not been studied relatively extensively (e.g. 234) as a result of their widespread use in such structures as transmission towers. These towers are usually precisely analysed[3] for actual failure under well defined load factors and an accurate knowledge of member load capacity has been essential. Even here, however, the underlying research has frequently been highly empirical with load capacities given for each member size under practical field conditions."[6]

The third paragraph continues to give you the actual reason for the studies, an information gap existed for a case where angles are used

as laterally unsupported beam. This is stated quite clearly in the third paragraph:

> The case which presents the designer with his current major information gap occurs when the angle is used as a laterally unsupported beam. For example, the S.A.A. Steel Structures Code AS CA1[7] states in Rule 5.4.3:
>
> > The Standards Association of Australia is not prepared at this stage to make recommendations for angles which are not supported laterally".

The article continues with the subject of why the work was done in the first paragraph of Section 2. This paragraph states:

> 2. Current Investigations
>
> The purpose of the current Australian investigation was to develop rational but simple formulas for the design of laterally unsupported angles in bending. This should help fill the present, previously quoted, void in the S.A.A. Steel Structures Code, CA1 and thus permit the more widespread use of angles in building construction.

Note also the last sentence in the paragraph refers to information from these studies permitting wider use "of angles in building construction". This quote is a general statement, certainly not limiting the use of this information to design of transmission towers as surmised by Mr. Stokes.

Mr. Stokes' recollection of the history of DR 83-042-S is at variance with the Project record. Mr. Stokes originally submitted a draft copy of this DR in August 1983. In its draft form the DR did not identify any specific pipe supports exhibiting the alleged

discrepancy and consequently Mr. Stokes was asked to provide a listing of such supports. The project is unaware of any list of "approximately 100" supports prepared by Mr. Stokes. Mr. Stokes did provide a list of 18 such supports which were evaluated to establish whether or not they were in fact discrepant. This evaluation was completed and documented in a Project memo dated September 27, 1983, which summarizes the results and documents the acceptability of the supports reviewed. This conclusion confirmed that no discrepancy existed and therefore no Discrepancy Report was required. Mr. Stokes was not satisfied with the original version of the DR and submitted a revised version of the DR on October 5, 1983. The DR was dispositioned on the basis of the previous work described above.

It is alleged that:

7. BASE PLATE II - There are several points which should not be over looked [sic] in the Base plate calculations beyond those discussed by Mr. Hartzman and Mr. Manoly. Base Plate II must as a finite element program, in order to give accurate results have certain criteria met [sic]. These are: 1) The elements must be within a certain aspect ratio, i.e., height, length, width. Most finite element programs specify that the aspect ratio be not more than 1 to 4. This requirement if applied to a 1/2 inch thick plate, would require that the length and width dimensions not exceed 2 inches. I have been informed, that the analyses which were performed were not within the aspect ratio which is required for accurate results. 2) Some plates are not one piece of steel and have been welded together with partial penetration welds. The model should account for the reduced thickness where welded, so that the plate stress calculations will be accurate. Also, in the case of shell type anchor bolts the hole diameter should be used per the manufacturers requirements in determining the 10D distance. In the past, this was done incorrectly as the hole diameter was not supplied by management. This information is now supplied to the engineers in the ESD 223 and the calculation should be corrected during the present review. Failure to correct this could result in a safety-related support failing. (6/1/84 Stokes Aff. at 21.)

If Mr. Stokes had done his homework, he would know that Base Plate II is <u>not</u> a finite element program. It is quite simply a pre and post processor for Stardyne which is a finite element program. While this may appear to be a minor point, it is characteristic of the lack of knowledge that typifies these allegations.

Mr. Stokes' statements concerning aspect ratios are wrong. This program uses 2-dimensional elements -- (length and width) and not 3-dimensional (height, length, width) as Mr. Stokes seems to believe. The 4:1 ratio is applied as a guideline for determining the

length-to-width grid and has nothing to do with the thickness or height. Further, the program author, Control Data Corporation (CDC), has been consulted and they have stated that this aspect ratio is a guideline and is not an absolute measure of solution accuracy. Nonetheless, the DCP has followed these guidelines.

It is recognized that some baseplates are of composite construction and that discontinuities may result from welds (less than full penetration) and different material thicknesses. These variations are factored into the computer model as appropriate.

The subject of anchor bolts is fully covered in the DQA response (Breismeister, et al. Aff. at 34) on this issue as follows:

> 79. The capacity of a concrete anchor bolt is a function of the bolt length (embedment), bolt material, and concrete strength. Anchor bolt capacity relates to a shear cone of concrete originating at the end of the anchor bolt embedment. This cone projects at a 45^{0} angle to the surface. If two anchor bolts are placed close enough together that their shear cones overlap, some of the strength of the anchor bolts may be lost. The 10d (bolt diameter) criterion between anchor bolts was established to assure this would not occur.
>
> 80. All shell type anchor bolts on Diablo Canyon have an embedment of less than five bolt diameters. Since the anchor bolt center lines are ten bolt diameters apart, the shear cones can never overlap. Hence the anchor bolts retain their full capacity. The capacity of an anchor bolt is determined by test. The test for a shell anchor is normally performed on one anchor at a time. The anchor bolt will develop that full capacity so long as no adjacent anchor bolt is less than 10 bolt diameters away. In other words, the criteria that determines the required spacing is solely a function of concrete failure theory and test results which are categorized by bolt diameter.

81. Tests to validate this premise were conducted in 1962 on a Phillips shell type anchor. The results reported no reduction in capacity for ten bolt diameter spacing. It is true that the recommendation in the Hilti catalogue is to space the bolts 10 hole diameters apart. However, when the actual shear cone is developed, the results are bounded by the 10d bolt criterion.

It is alleged that:

1. WELDED STUDS

1.1 The subject of welded studs and welded "standard fasteners" that appear to be studs has been discussed extensively since January. Pullman, with PG&E's knowledge and approval, has welded:

ASTM A-307 grade B bolts
ASTM A-325 high strength bolts (type 1)
ASTM A-490 alloy steel, high strength bolts

(6/7/84 Lockert Aff. at 2-3.)

Mr. Lockert continues to resurrect the same issues which he has previously pressed and which have been responded to on numerous occasions. There is nothing here which is new, nor is it significant. In fact, many of the issues raised are items which have been thoroughly and accurately addressed more than once. Mr. Lockert has difficulty accepting the truth, no matter how incontrovertible the facts or the opinions of others with far more experience, when it runs counter to his espoused ideas. Nor is he willing to accept the fact that many issues have been reported to the NRC and dispositioned acceptably. He also continues to selectively compare apples and oranges and make a great deal out of the differences or to quote statements totally out of context.

This allegation is an old issue previously addressed in PGandE's response to JI's CQA Motion to Reopen and PGandE letter DCL-84-166, dated May 17, 1984, to the NRC Region V (pp. 7-9).

This allegation is false because the studs and bolting materials that have been welded as a normal practice are weldable and appropriate Welding Procedures Specifications (WPS) for the application were used. Previously, PGandE identified where approximately 80 A325 5/8-inch diameter bolts and 17 A490 (or equivalent) bolts were welded. These discrepancies have been identified, and Mr. Lockert's concerns have been evaluated. Contrary to his allegation, of the thousands of acceptable welded studs that have been installed approximately 100 discrepant weld cases have been identified by the quality assurance program and have been evaluated and corrected as necessary. In particular, the welding of A307 Grade B was correct and proper as explained in CQA response to JI Items 14 and 15 (Breismeister, et al. Aff.). In brief, A307 Grade B is listed as an PI material in ASME Section III and is weldable. Even though Mr. Lockert does not accept this uncontrovertable fact, the welding of A307 Grade B studs meets the specification and code requirements for supports and restraints at Diablo Canyon.

The welding procedure specification for A325 did not receive prior approval from PGandE. This discrepancy was documented. The 80 A325 welds in question were analyzed using extensive in-plant and laboratory tests. This analysis showed that these welded A325 bolts were technically acceptable. The Project decided however, to abandon the use of the A325 welded studs for design purposes to bring prompt resolution to the issue. This has been fully explained in PGandE

letters DCLs 84-067, -078 and -113 dated February 17, 29 and March 23, 1984, respectively, to the NRC, Region V. Mr. Lockert's misplaced technical concern for welding the 5/8-inch diameter, 2-inch long A325 studs was addressed in the response to Items 450 and 460 (as identified in NRC Region V letter of April 27, 1984, and contained in the GAP March 23, 1984 submittal), DCL 84-195, pages 108-115.

The other discrepant welds involving A490 (or equivalent) were four bolts and 13 pins. The bolts were identified on DR 4447 several years ago. The pins were addressed on a PGandE Nonconformance Report. All unacceptable welds were removed. The engineering response to the Nonconformance addressed the total significance of that welding. This has been reported to the NRC in DCL 84-166, pages 7-8.

As regards the toughness of P1 material such as A307 Grade B heat affected zones (HAZ), there is no technical concern because the material was controlled by type, and installed with correct WPSs. The codes used for Diablo Canyon construction did not, and ASME Section III today does not, require toughness testing for small diameter bolts. This is reasonable for P1 bolting because of the composition, size effects on toughness response to loading, and because the design loads are relatively low. The mild to warm

service temperature conditions at Diablo Canyon are another reason there is no technical concern for A307 Grade B weld heat affected zone toughness.

The HAZ toughness of the A325 bolts is a moot point because these are no longer used in the plant's design. However, the laboratory data on hardness, and the metallurgical micro structure (non hardened) indicate that those HAZs would not have been a toughness concern.

The A490 HAZs are not a concern because these were either removed or are in locations where there are no design loads. Thus there is no concern for the HAZ toughness of A490 material.

It is alleged that:

1.2 All the installations were improper, performed without the benefit of an approved welding procedure, and constitute code violations to both ASME and AWS code specifications. Yet, each installation (there are thousands) was signed off by both contractor and utility QA groups as meeting all contract and code specifications. (6/7/84 Lockert Aff. at 3.)

As set forth above, this is an old subject which Mr. Lockert continues to repeat despite incontrovertible facts to the contrary. In each of the limited cases where a problem existed, the quality assurance program identified the problem and corrective action was taken.

The welding of A307 Grade B was proper, as explained in PGandE's Response to JI's CQA Motion, Attachment C, Items 11, 12, 13, 14, and 15. A307 Grade B is listed as a weldable P1 material in ASME Section III and has been listed as a weldable material for which P1 WPS could be used in Code Cases.

The lack of approved welding procedures for A325 and A490 material was recognized, documented, and properly addressed. The WPSs which were used were subsequently accepted for the applications.

It is alleged that:

> 1.3 More distressing, after the facts have been shown the utility still insists that the installations are proper and acceptable. Pullman has researched the A-307 grade B bolts and correctly found that this is an unacceptable practice, Pullman's QA/QC Manager, Harold Karner, had deleted reference to A-307 Gr. B bolts in a Discrepancy Report sent to PG&E. This act can only be interpreted as some kind of cover-up of defective installations. (6/7/84 Lockert Aff. at 3.)

The allegation is totally incorrect in inferring that Mr. Karner was covering up defective installations. In fact, Mr. Karner correctly determined that A 307 Grade B bolting material was a P1 material and therefore the Pullman welding procedures being used to weld these bolts were correctly being applied. He, therefore, deleted the incorrect reference in the discrepancy report. The A307 Grade B material is listed in ASME Section III as a P1, weldable, material. This again is an old issue which was addressed in DCL 84-166, page 12.

It is alleged that:

1.4 The use of A-307 Gr. B "Mechanical Fasteners" is improper because the bolts were not ordered by Pullman to specifically [sic] made from A-36 steel. Heats of steel were not traced to each product batch because the distributer was not responsible for segr gation [sic] of different heats when selling bolts. A-307 non-headed anchor bolts can be ordered to A-36 steel specifications as stated in scope part (D) of ASTM-A-307-74 but, unfortunately, this was not done. Bolts purchased as mechanical fasteners without the proper metallurgical identity were made to appear as non-headed anchors when Pullman cut the head off the bolt and then presented the item in the field as a weldable stud. Contrary to Breismeister at 33, 34, and 35 such "Diablo made" welded studs are not P1 materials and cannot be welded with Pullman's present welding procedures. Contrary to PG&E's response to GAP's March 1, 2.206 Petition (Schuyler at 9, 30); A-307 Gr. B bol [sic] ordered "Mechanical Fasteners" with no supplementary requirements for checking limitations of carbon and manganese contents are not P1 materials. (6/7/84 Lockert Aff. at 3-4.)

The traceability issue was addressed in PGandE's response (DCL 84-195) to Items 476 and 477 (as identified in NRC Region V letter of April 27, 1984, and contained in the GAP March 23, 1984, submittal), and the weldability issue in PGandE's Response to JI CQA Motion, Items 14 and 15. Mr. Lockert simply refuses to recognize, accept, or fails to understand that heat traceability is not required. All of these materials were marked as to material type and were traceable by type/grade up to the time of installation. Thus all materials were controlled as required by the contract and specifications. Mr. Lockert obviously refuses to accept the fact that A307 Grade B is listed in Section III as a P1 material.

It is alleged that:

2.1 During my inspection activities, I came across many situations where welding procedures were being used outside the scope of their original limits. The most blatant violation of weld procedure limits was the welding of Rupture Restraints with WPS 7/8. As discussed by Breismeister at 9, changes in "essential variables" within the procedure require additional requalification measures. Also discussed by Breismeister at 12 iii) is the fact that joint design is an essential variable for the AWS D1.1 code. Breismeister states "both codes require that the allowed joint configurations be described in the WPS or in documents which are used with the WPS."

2.2 This is a false statement in that joint configurations (plural) requires WPSs (plural). Not WPS (singular) as indicated by Breismeister. D1.1-83 paragraph 5.1.2 states "All prequalified joint welding procedures to be used shall be prepared by the manufacturer, fabricator, or contractor as written procedure specifications...." Note the use of plural "specifications" rather than the singular "specification." This is because all changes in essential variables requires additional WPS's to be written.

2.3 Using AWS D1.1 Procedure Qualification Criteria the only joint design qualified by the 7/8 procedure is the joint design used in the Procedure Qualification Record (PQR). Apparently, no one has ever tried to track the essential variable, joint design, through the PQR because the traceability gets lost. Using the four PQRs used by WPS 7/8 all refer to the original joint dimension by pointing to sheet 2 of 10. Seeing how WPS 7/8 now has only 6 pages and page 2 shows eight different joint designs, the original qualified joint design per AWS requirements has been lost in the shuffle.

2.4 This means that the procedure 7/8 cannot be considered qualified within the AWS D1.1 code because the essential variable, joint design, has been lost. Further, Pullman and PG& have attached additional joint designs on to the 7/8 procedure through the Welding Technique Specification

(WTS) AWS 1.1. Breismeister himself states that changes to "essential variables" require requalification. Yet WPS 7/8 shows eight different joint designs and AWS 1.1 shows an additional nine joint designs all grouped under one procedure.

2.5 Breismeister states at 10 "If the WPS is written in accordance with restrictions within AWS D1.1, the WPS is termed "prequalified" and no PQR is needed." This statement is correct but Pullman's and PG&E's application of the above statement as a technique specification is wrong. Neither WPS 7/8 or AWS 1.1, separate or together, incorporate all aspects of AWS D1.1. One quick look at page three of WTS 1.1 will show that the AWS pre-qualified joint dimensions and tolerances have been subverted by PG&E's detail sketch. The point being--PG&E has taken authority reserved to AWS by AWS for determining prequalified status and given it to themselves. PG&E does not hold the authority to determine prequalified status over and above the code body.

2.6 Breismeister has made false statements at 42, 43, 45, and 51 all relating to the assumption that WPS 7/8 + AWS 1.1. exceeds AWS code requirements. Karner, similarly, makes false statements at 40 stating "There is no evidence that the use of WPS 7/8 on Rupture Restraints is inadequate." As pointed out to Russ Nolle, a QC supervisor, on Oct. 24 and Nov. 16 and Harold Karner, the QA/QC Manager, on Nov. 2 and Nov. 16 the use of 7/8 with or without WTS AWS 1.1 is a deficient procedure because changes in joint design are not adequately accounted for. (6/7/84 Lockert Aff. at 4-6.)

All of these allegations are repeated from earlier affidavits. These allegations were responded to in the Response to JI's CQA Motion Breismeister et al. answer to JI #10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 21, 23, 25, 48 and 49. Those responses, plus the introduction of the Breismeister et al. affidavit attached to PGandE's Response to JI's CQA Motion, clearly answer these allegations and show that, in every case, the WPS's and referenced documents fully meet the requirements for welding rupture restraints. AWS D1.1 paragraph

5.1.3, specifically allows the combination of qualified and prequalified joint welding procedures without qualification.

Mr. Lockert apparently has a differing personal opinion regarding the format for a WPS, and either does not understand or is unwilling to accept the latitude provided by the ASME and AWS Codes regarding WPS format.

It is alleged that:

3.1 The Diablo Canyon Rupture Restraint program is probably the most illustrative element of not just one failing in quality assurance but repeated, many, failings in construction quality assurance. The first failings started with the vendors who supplied the Rupture Restraints with nonconforming welds and oversize bolt holes. The repeated, numerous, failings that follow have been PG&E's incomplete resolutions to those same problems.

3.2 Breismeister at 55 makes a false statement when he states "All older heavy section welds in Rupture Restraints were re-inspected beginning in 1979 and repaired, as necessary, to requirements in AWS D1.1". Breismeister at 121 provides the proof to dispute what he stated as fact at 55 "The program was based on magnetic particle and ultrasonic examinations of enough welds on installed Rupture Restraints to obtain an adequate sampling for thorough engineering analysis, evaluation, and corrective action." An adequate sampling does not constitute a 100% inspection as indicated by Breismeister at 55. (6/7/84 Lockert Aff. at 6-7.)

The inspection and repair of rupture restraints is an issue which was fully documented and reported to the NRC several years ago. The issue was again fully addressed in PGandE's Response to JI's CQA Motion. (Breismeister et al. Aff. at 19 and 40-43) That response was necessarily fragmented because of the fragmented and multiple fashion in which the allegations were raised. What Mr. Lockert has chosen to do here is to grasp upon a portion of a response to one allegation issue and compare it to a portion of a response to another allegation issue without bothering to place the entire concern in proper perspective.

The referenced statement in paragraph 55 of the Breismeister affidavit related to the proper use of field construction WPS 7/8 and its supplement AWS 1-1. Mr. Lockert has misquoted paragraph 55, which in fact reads, "All older heavy section welds in Rupture Restraints were reinspected beginning in 1979 and repaired, as necessary, to requirements in AWS 1-1 (here he cites AWS D1.1)."

First, Mr. Lockert is factually incorrect in indicating AWS D1.1, which is the Structural Welding Code. The reference was to the AWS 1-1, a Pullman document with which Mr. Lockert is familiar.

The statement in paragraph 55 was that all heavy section field welds were reinspected and repaired as necessary, not all welds. This is completely consistent with the statement in paragraph 121 that a sampling plan was used for shop weldments. The sampling program concerned another set of welds with different problems, not those welds referenced in paragraph 55. In this case, Mr. Lockert is using answers to allegations about different subjects to "prove" that false statements were made. No such false statements were made. It would appear Mr. Lockert is intentionally engaging in pure sophistry in an attempt to support otherwise unsupportable allegations.

It is alleged that:

3.3 Contrary to the resolution of Nonconformance report DC2-80-RM-002 Pullman did not do a documented inspection of all bolted and welded connections and applicable documentation. I personally observed oversize bolt holes, nonconforming welds, and out of tolerance washers on Rupture Restraints as late as December of 1983. (6/7/84 Lockert Aff. at 7.)

The subject nonconformance report (NCR) applies to Unit 2 only. The corresponding NCR for Unit 1 was properly dispositioned and closed. The resolution set forth for NCR #DC2-80-RM-002 has been and is presently being implemented. Pullman has been and is presently performing documented inspections of bolted and welded connections for engineering evaluation of nonconforming connections. NCR #DC2-80-RM-002 applies to Unit 2 and has not been closed and will remain open until completion of all required activities set forth in the resolution.

The corresponding Unit 1 nonconformance report, NCR #DC1-79-RM-003, was resolved and closed per the stated resolution. As-builts, generated by Pullman, were reviewed by Engineering and modifications implemented as necessary. The steps taken for NCR closure are fully documented.

It is alleged that:

3.4 It's interesting to note that the 100% re-inspection of "all bolted and welded connections" was cut short as early as April 3, 1980 as evidenced by a M. E. Leppke memo presented as exhibit 7 to the Breismeister affidavit dated March 19, 1984. It should be pointed out here that the Leppke memo also instructs Pullman to deviate from AWS D1.1. Apparently Leppke holds Diablo Canyon above AWS requirements by stating "Pullman should use judgement when removing construction induced defects on shop welds. For example, it is not necessary to 'chase' defects uncovered by grinding arc strikes." Compare this statement to AWS D1.1 paragraph 3.10 requirements "Cracks or blemishes caused by arc strikes shall be ground to a smooth contour and checked to ensure soundness." (6/7/84 Lockert Aff. at 7-8.)

This allegation is once again an old issue which was addressed in PGandE's Response to JI's CQA Motion, #102, 103, and 104. The allegation neglects to place the circumstances in perspective and in fact does not accurately reflect instructions given by PGandE to the contractor. The identification of shop weld rupture restraint indications occurred during an extensive program developed and implemented to investigate, evaluate, and repair rupture restraint field weldments. This program was started because cracked field welds were found by Pullman and PGandE personnel in Unit 1 rupture restraints during late 1978. The shop program was based upon magnetic particle (MT) and ultrasonic examinations (UT) of enough welds on installed restraints to obtain an adequate sampling for thorough engineering analysis, evaluation, and corrective action. As a result

of these studies, Engineering identified all shop welds which required examination. All of these welds were magnetic particle or ultrasonically examined and any defective weldments were repaired. This included both Units 1 and 2. This program is documented in PGandE NCR DCL-79-RM-010. PGandE reported the deficiency to the NRC per 10 CFR 50.55(e) verbally on April 4, 1979, and by letter dated May 3, 1979, and reported the resolution in a final report for Unit 1 dated December 9, 1980.

This allegation neglects to point out that this major repair program was underway and the identified shop welds were part of this program. Thus, the allegation and affidavit is out of context. The PGandE memorandum states that sufficient data on shop welds had been received and that the Engineering Department would review the data and include its conclusions in the final rupture restraint report. It also states that Pullman need not report further test results on shop welds. The original Joint Intervenors' CQA allegation and affidavit falsely indicated that Mr. Leppke directed Pullman to stop issuing Discrepancy Reports on shop welds.

Mr. Leppke correctly instructed Pullman to use judgement and stop chasing the indications since they were to be part of the Engineering

evaluation program. Thus, the memo did not cut short the re-inspection program which was being implemented.

Mr. Lockert has again presented statements out of context. As may be seen from the full text, Mr. Leppke was indicating to Pullman that it was not important for Pullman to search out and develop extensive data on shop welded rupture restraints because those welds were already under separate evaluation and there was sufficient data for the evaluation. And, when Pullman's construction work induced defects or discontinuities in shop fabricated weldments under evaluation, it would be sufficient to note the defect. Pullman should not put forth unreasonable effort grinding on the material to develop more data, because the material was already in question and under evaluation. It would be sufficient to document the problems, but it was not necessary for Pullman to develop further extensive data. PGandE Engineering had sufficient data to evaluate the condition. This does not constitute an instruction to deviate from AWS D1.1.

It is alleged that:

3.5 Pullman QC Supervisor, Russ Nolle, also showed no interest in observing welds that did not conform to good workmanship standards. I informed Russ Nolle about welds under the unit two pressurizer that would be "absolutely

unacceptable under any code." Russ told me that it wasn't Pullman's job to inspect another contractor's work. I agreed that this was a reasonable stance for the company but in the interest of making the plant right someone should fix it. Russ told me that it was another contractor's work, already accepted, and to stay within Pullman's scope of responsibility. As I remember it, Russ Nolle did not suggest to me to write a DR as noted at 124 of Breismeister affidavit. (6/7/84 Lockert Aff. at 8.)

This is another old issue which was fully addressed in the CQA Motion Response. Mr. Lockert's reference to paragraph 124 should be to paragraph 123. The statement regarding Mr. Nolle telling Mr. Lockert he could write a discrepancy report is supported by Mr. Nolle's sworn affidavit which was referenced in paragraph 123 and was included as Exhibit 8. Mr. Lockert's convenient memory lapses are overriden by the Nolle affidavit.

It is alleged that:

3.6 Finally the statement by Schuyler in response to GAP, March 1, at 56 states "To the best of our knowledge, this established process of resolving each instance of an oversized bolt hole on a case-by-case basis has resulted in the resolution of all concerns involving oversized bolt holes, either by repair or by member replacement." This statement is totally false in light of my own inspection experience on Rupture Restraints in late 1983. Oversized bolt holes in rupture restraints were not being reported nor was corrective action being taken to ad dress [sic] the problem. I was severly [sic] reprimanded for an accidental reinspection and rejection of some other inspector's work where unreported oversized holes existed in 1983. (6/7/84 Lockert Aff. at 8.)

In accordance with the resolution set forth in NCR #DC2-80-RM-002, oversized bolt holes are to be identified and addressed. Pullman is presently reviewing Unit 2 bolts and bolt holes. This review has been and will be a continuing process until completion of the required documented review. Upon completion of the requirements set forth in the NCR resolution, the NCR will be closed.

The corresponding Unit 1 NCR was dispositioned and closed as set forth in response to JIR 30, 3.3 above.

It is alleged that:

4.1 As a welding inspector at Diablo Canyon I was required to check the welders' adherence to the procedures. One such procedure using the GTAW process requires an argon shielding gas flow over the weld of 20 cubic feet per hour (CFH). A welder was getting ready to start the process when I noticed the flow meter that he had with him was not attached at or near his torch. I requested that the flow meter be attached so that the inspector could verify compliance to the procedure. The welder refused the inspector's request, and when a QC supervisor was requested to help the inspector, the QC supervisor failed to support the inspector's request.

4.2 Breismeistr at 98 and 99 makes two false statements in response to the incident. At 98 "At no time did Pullman QC management establish a policy of deferring to construction (production personnel) when they [sic] voiced an objection." At 99"...the contract does not require checks of argon gas flow. The contract requires regulators, but not flow meters."

4.3 This incident is enlightening in that it shows the ignorance of not only the QC supervisor but the QC manager, welding engineers of both contractor and licensee, and the top Bechtel manager of Research and Engineering. I find this distressing because all the above mentioned individuals have heavy responsibilities toward insuring the plant is constructed to the commitmentss of code, contract, FSAR, and Federal Regulations.

4.4 The Welding Procedure Qualification [sic] (WPS) is the document that field engineers, welders and QC inspectors usually have to give them direction in the field. The WPS for the weld in question required a 20 CFH minimum flow rate of argon gas. My job as an inspector should have been to verify compliance to the WPS.

4.5 The welder refused to mount the flow meter at or near the torch. The welder wanted the inspector to run up two elevations to the 140' deck and check an unspecified gas bottle in an unspecified location. The gas bottle could have been one of many bottles on the 140 deck because there were many welders in containment.

4.6 The flow rate at the bottle did not necessarily mean the flow at the torch would be the same due to possible

junctions, cracks, crimps, and holes in the long hose. The flow rate of the gas at the bottle is meaningless. The flow rate should properly be measured at or near the torch. This was explained to the welder and the QC Supervisor at the time of confrontation. The QC Supervisor denied my request to mount a second, readily available, flow meter at the torch and in doing so prevented me from implementing the QA program.

4.7 Mr. Breismeister states the contract does not require checks of argon gas flow. Mr. Breismeister is wrong. Contract 8833XR requires that all Rupture Restraint welding be performed to the AWS code. The AWS code requires that all welding be performed within the stated parameters of the WPS. The AWS code further requires the inspector to verify that welding does conform to the WPS. The contract requires the inspector to check not only the shielding gas flow but the voltage, current, travel speed and any other process parameters that effect the quality of the deposited weld metal. (6/7/84 Lockert Aff. at 9-10.)

This allegation can be summarized as follows:

1. An inspection was required which Mr. Lockert would not complete.

2. The inspection was required by code and the contract.

3. Pullman QC management would not back inspectors to make sure their requirements were implemented when construction was inconvenienced.

4. Bechtel and Project Welding Engineering do not know contract and code requirements.

All of these allegations are false:

1. The inspection discussed by Mr. Lockert was a weld in process inspection required by Pullman internal specifications ESD 215 and 219. There are no requirements in PGandE Specification 8833XR or AWS D1.1 for checking gas flow during gas tungsten arc welding (GTAW).

2. The required inspection could have been easily completed by Mr. Lockert had he chosen to do so. At most, he would only have had to follow the gas line back to the bottle and then verify the gas flow at the flow meter. Mr. Lockert evidently felt he was above making this effort and requested that the welding equipment set up be changed by the welder for Mr. Lockert's convenience. When the welder refused, Mr. Lockert's supervisor was called in to determine the requirements and resolve the issue. As stated previously in PGandE's response to the JI CQA Motion (Breismeister et al. Aff. at 33 and Karner et al. Aff. at 18), the supervisor correctly determined that the ESD requirements could be met by inspection of the gas flow at the gas bottle. Mr. Lockert refused to accept this at the time and another inspector was brought in. This new inspector completed the required inspection. The weld was accepted. Mr. Lockert's statement that hose connections, kinks, or leaks in the long hose make it impossible to measure flow at the gas bottle is also incorrect. His tracing the gas line back to the gas bottle would have found any such gross problems. Further, inspection of the gas line would only have

been required if problems had been noted at the welding torch. Typical first line inspection of GTAW includes noticing the sound of gas emanating from the gas nozzle and feeling the gas velocity impinging on the hand or cheek. Additionally, the adequacy of the gas flow at the torch can be evaluated by the appearance of the tungsten electrode and the weld surface during welding. All of these normal and routine inspection activities, in addition to the check of the flow meter, would ensure the proper gas flow at the weld, and would ensure that requirements were met.

The above explanation clearly shows that Breismeister et al. did, in fact, know the requirements of the contract, the applicable codes, and the Pullman specifications that Mr. Lockert was supposed to be implementing. It is also clear that Pullman management was not deferring to construction but was attempting to have its employee (Mr. Lockert) complete his inspection in accordance with standard inspection procedure requirements. While Mr. Lockert did not complete this work, other inspectors were capable of following the applicable instructions in completing the inspection. Thus, no code or contract requirements were violated and the Pullman specification requirements were met when Mr. Lockert's replacement preformed the inspections.

Mr. Lockert is entitled to his opinion regarding the "ignorance" of the Licensee's affiants. He is incorrect, however, regarding the "top Bechtel manager of Research and Engineering" being involved in

the affidavits. The personnel involved in the affidavits were identified along with their titles and resumes. The "top Bechtel manager . . ." was not one of them. This is yet another example of Mr. Lockert's misconstruction of the facts leading to unfounded accusations.

It is alleged that:

5. WELDING EQUIPMENT STANDARDS

5.1 In January of 1984, I reported to the USNRC the use of deficient GTAW welding machines by Pullman Power Products Corporation. [sic] at Diablo Canyon Nuclear Power Plant. The licensee has responded to the charge by stating (Breismeister at 82) "... Pullman has never violated the contract, since PG&E supplies the welding equipment as allowed by contract 8711....."

5.2 Breismeister failed to supply applicable documentation of the stated paragraphs of contract specification 8711 showing how the contractor was relieved of its responsibility to use adequate, dated, welding equipment as described in paragraph 7.10.1. Since the Breismeister affidavit choose not to present the facts, one can only speculate why the utility choose [sic] to withhold the information.

5.3 The Breismeister affidavit references paragraph 3.21 and I related statements found in paragraph 7.10.1, both from section 1 of contract specification 8711. Paragraph 7.10.1 stated all GTAW welding shall be performed with a power supply equipped with:

1.) High frequency for all initiation.
2.) Rheostat for stepless control of current.

(6/7/84 Lockert Aff. at 11.)

The original PGandE response (see Breismeister et al. Aff. at 26-30) to the original JI CQA Motion was correct. The 8711 specification permits PGandE to provide the welding equipment, which PGandE does provide. This relieves the contractor of the responsibility to provide welding equipment. When it is beneficial to have high frequency arc initiation capability, or stepless current control, this equipment is provided as was stated in the original CQA

response. The original allegation and this followup allegation are more examples of Mr. Lockert omitting significant factual details, and making representations out of context.

There was no perceived need to present specification 8711, paragraph 3.21, which permits PGandE to provide the welding equipment. However, that paragraph is as follows:

> Company reserves the right to award separately any item in the Proposal Form and the right to award any furnishing, fabricating, and delivery quotation items hereunder separately without the erection. Company also reserves the right to furnish valves, pipe hangers, weld inspection service, welding gas, and arc welding equipment at its option.

It is alleged that:

> 5.4 I was forced to bring the matter to the NRC's attention because both Pullman and PG&E could not resolve the issue within their respective QA organizations. After my discovery of the contract requirements in September of 1983, I wrote a memo to Pullman's QA/QC Manager, Harold Karner. Mr. Karner was unwilling to put his response on paper but did tell me verbally "If PG&E doesn't enforce the contract Pullman doesn't intend to." (6/7/84 Lockert Aff.at 11.)

This was addressed extensively in the original PGandE response (Breismeister et al. Aff. at 28-29) to the JI CQA Motion and in the response to SSER 22. It should be clear to everyone that this was not and is not a quality issue. The contract terms and conditions permitted the equipment as used. Mr. Lockert is complaining about commercial issues which do not affect QA. The equipment which Mr. Lockert thinks is necessary, and about which he has wildly speculated

as affecting quality, is in fact unnecessary. When considered to be beneficial the subject equipment is selected for use. The welders were qualified on the less sophisticated equipment which was generally used for production. The production welds made with the less sophisticated welding equipment were subject to a variety of examinations and tests and have been accepted. The examinations have included radiography of thousands of welds. The pipe welds are also pressure tested at multiples of the design pressure. This simply is not a quality issue, it has no merit.

It is alleged that:

5.5 Mr. Karner, not surprisingly, has apparently had a lapse of memory regarding our conversation and now denies ever making such a statement (Brmr at 86.) Further, Mr. Karner now states that he contacted PG&E's QA Engineer Russ Taylor soon after I contacted Mr. Karner. This appears to be a false statement because I contacted Mr. Taylor with my concerns after being referred to him by another PG&E employee, Dave Stupi. From Mr. Taylor's reaction, it was apparent that as late as Oct. 25, 1983, he had not been contacted by Karner about the subject because he requested several days to become familiar with the subject.

5.6 Subsequently, Mr. Russ Taylor agreed that the contract stated all GTAW welding machines required the additional controls already mentioned. I asked what did PG&E intend to do about it since Pullman didn't seem to care. Russ Taylor stated that the contract would be changed.

5.7 This series of discussions does not show proper handling of the matter as alleged by Breismeister (at 87), but rather a pass the buck attitude resulting in a change of the contract instead of an upgrade of the welding equipment to the specification requirements. Breismeister stated at 87 "It is also clear that Mr. Lockert had gone far from normal channels in pursuing this baseless issue...." The actions I took on the matter dealt with people that went up the ladder of responsibility. The

contractor QA Manager had aptly expressed himself and it was clear to me that he had no corrective action in mind. Next, I went to the licensee's welding engineering representatives who were unwilling or unable to correct a contract violation. Final resolution of the matter resulted in a verbal promise to change the contract six years too late. This hardly shows a functioning QA system. Breismeister's answer that paragraph 3.21 relieves PG&E of its contractual commitments was now shown to be true. (6/7/84 Lockert Aff. at 12-13.)

The original PGandE responses (Breismeister et al. Aff. at 26-30) to the original JI's CQA Motion and the SSER 22 answers on these points were correct.

Contrary to this allegation, the 8711 contract was not changed regarding welding equipment in reponse to Mr. Lockert's concerns. There was no buck passing and there was no need to upgrade the welding equipment.

In the allegation Mr. Lockert states he went up the ladder of responsibility. His own statements do not support this. As was originally pointed out in the CQA response, after talking with the Pullman onsite QA/QC manager, Mr. Lockert jumped over to the PGandE operations group which had no construction or QA responsibility. Then he jumped over to PGandE QE. Mr. Lockert never pursued the issue upward, he simply jumped around sideways, as in "lateral arabesque."

Mr. Lockert's penultimate sentence contains one last unsupported misleading allegation, "Breismeister's answer that paragraph 3.21 relieves PGandE of its contractual commitments was not shown to be true." This implies that the answer was false. The original CQA answer was and is true. PGandE met its contractual commitment regarding the supply of welding equipment. Mr. Lockert has presented no contrary data. The applicable text of the specification is quoted at p. 2 of this response.

JIR-33 to 69 (Hudson Affidavit)

In his affidavit of June 5, 1984, Mr. Hudson has set forth a selectively detailed chronology of the Kellogg/Pullman pipe support and rupture restraint effort at Diablo Canyon. This historical document recounts the deficiencies uncovered in the course of the program and the numerous steps taken by PGandE and its contractors to revise and upgrade both the technical and quality control/quality assurance procedures and requirements associated with that effort.

Unfortunately, Mr. Hudson then appears to purposely ignore the December 19, 1983, decision of the Appeal Board in this case which addressed many of the same issues which he raises and found them "without merit" and to not "raise legitimate doubt as to the plant's capability of being operated safely."

Mr. Hudson continues to resurrect issues which have been raised over and over again. His dissertation contains numerous statements of opinion disguised and presented as fact and some statements which are simply incorrect. He appears unwilling to accept the fact that deficiencies, which were identified in the course of the construction process and which then resulted in appropriate corrective action being taken, do not indicate that the end product is of suspect quality or that the contractors were incompetent or less than honest. The fact that repeated actions had to be taken to address

certain problems is not indicative of deliberate malfeasance but of PGandE and contractors taking the necessary actions to ensure that the issues are addressed and resolved appropriately. The vast majority of his points have been previously addressed. There is nothing new or significant in his latest affidavit. He has conveniently failed to mention the various reinspections and reanalyses which, had they been set forth, would put his "allegations" in the proper perspective.

Many of Mr. Hudson's concerns center about the fact that certain procedures and programs at Diablo Canyon were not continuously updated to reflect the very latest criteria promulgated by the NRC and/or national standards groups. This concern is raised repeatedly in regards to a variety of plant documents not addressing 10 CFR 50, Appendix B, even though the Appeal Board in this case has already ruled:

> Although not expressly stated, seemingly implicit in movants' argument is the notion that the regulations required immediate compliance upon the effective date of Appendix B and that applicant's commitment was insufficient to ensure a properly constructed facility. We disagree.
>
> The Commission's predecessor, the Atomic Energy Commission, recognized in promulgating Appendix B in 1970 that the nature of the construction process for a plant already being built, such as Diablo Canyon, Unit 1, precluded the complete and immediate application of the quality assurance criteria. In the Statement of Considerations accompanying the final version of Appendix B, it stated that the criteria would be 'used for <u>guidance</u> in evaluating the adequacy

> of the quality assurance programs in use by holders of construction permits and operating licenses.'[28] Therefore, contrary to the movants' suggestion, the applicant was not required to conform the construction quality assurance program for Unit 1 to Appendix B upon the provision's effective date. Moreover, the applicant's commitment in the Final Safety Analysis Report (FSAR) to apply the Appendix B criteria to the extent possible for the construction of Unit 1 was completely reasonable.

Mr. Hudson points to the "large number" and "repetitiveness" of the errors discussed. However, when one recognizes that the Construction Permit for Unit 1 was issued in 1968, and thus the plant has been under construction for about 16 years, the number of deficiencies identified is not unusual. As the Appeal Board recounted the testimony of Mr. Bobby Faulkenberry, Deputy Regional Administrator of the Commission's Region V office:

> "...the inspection history of Diablo Canyon from 1969 to the present time -- a program amounting to some 20 to 25 man-years of effort and covering the activities of all contractors on the site -- did not find the applicant's noncompliance record out of the ordinary. Indeed, he found the noncompliance rate 'about average, or possibly even on the low side.'"

What Mr. Hudson has really done is to prepare a chronicle depicting a utility and its contractors doing all that was necessary to ensure that the project was constructed in the proper manner. When deficiencies were identified (or reidentified), their past, present, and future implications were evaluated and appropriate corrective actions were taken. In many cases, this resulted in the rewriting of procedures or the issuance of new instructions to inspectors and

craft personnel. In some cases, the corrective action required major rework efforts (Mr. Hudson at p. 30 recognizes the 80,000+ manhours involved in one of these efforts) to ensure that the plant was correctly constructed. But the necessary efforts were undertaken and the end product inspected and reinspected to ensure that it met all of the regulatory requirements and commitments applicable to Diablo Canyon.

It is in this perspective that Mr. Hudson's affidavit with its allegations must be viewed. The issues raised were addressed and corrected when they were first identified which, in some cases, was over 10 years ago. The fact that these previously identified deficiencies have been regurgitated once again in the guise of "new allegations" is misleading at best. In addition, many of the issues posed are well outside the realm of Mr. Hudson's demonstrated experience and expertise, which may well explain why he has such difficulty understanding some of the issues.

Response

A brief factual overview of the rupture restraint program and the installation, inspection, analyses, and reanalyses is presented here. We then proceed to address the specifics of Mr. Hudson's rambling discourse.

Although the design and construction of Diablo Canyon began in the late 1960s, it wasn't until a few years later that the rupture restraint concept was introduced and retrofitted to the plant. Rupture restraints are similar to, but different from, pipe supports and building frames. Restraints are neither simple nor easy to fabricate. They are very often extremely difficult to install. Diablo Canyon was on the frontier, so to speak, of an advancing nuclear plant technology. It was known that there would be problems with rupture restraints, but all of the specific problems and specific countermeasures could not have been predicted in advance.

Thus, the procedures for the welding QA/QC program for rupture restraints initially used the pipe support activity as a role model. The rupture restraint activities evolved in an appropriate and responsible manner as problems became apparent. Mr. Hudson has chronicled several of these problems and their resolution through the QA/QC programs of Pullman and PGandE. However, he has intentionally left issues hanging, as if they were and are unresolved. This is not the case.

As recounted by Mr. Hudson, the PGandE specifications for pipe supports (#8711) and pipe rupture restraints (#8833XR) were prepared and issued in 1971 and work in these areas began in late 1971. Neither specification made any reference to or made any commitment to comply with 10 CFR 50, Appendix B, as Appendix B was not a licensing

requirement for Diablo Canyon Unit 1 and thus, there was no need to specifically address it. However, the quality assurance requirements of both specifications, which were basically the same, meet the intent of Appendix B.

To address the QA requirements associated with the installation of pipe supports, Kellogg (the corporate predecessor of Pullman) prepared ESD 223 to augment their QA Manual in this area and to consolidate all of the related requirements in one document. Subsequently, a separate QA Manual/Program was established for this area. Pullman revised its QA program to specifically address rupture restraints in 1973. The program was approved on December 11, 1973. All pipe supports and rupture restraints that had been installed prior to that time were reinspected and approved, repaired, or replaced, as necessary.

The program did not address the newly issued ANSI N45.2, which the NRC approved in Regulatory Guide 1.28 as a guide to and enhancement of 10 CFR 50, Appendix B, as the new standard was also not a licensing requirement for Diablo Canyon Unit 1. Although good faith efforts were made to bring the program into conformance with the standard, it was unnecessary and impossible to totally do so.

The issuance of Regulatory Guide 1.29 (which previously had been available as Safety Guide 29) had no authoritative impact on Diablo

Canyon as: a) Regulatory Guides were promulgated by the NRC to provide guidance and were not binding on each licensee; and b) it was not the licensing basis for Diablo Canyon. As is stated in the footnote to the Regulatory Guide:

> Regulatory Guides are issued to describe and make available to the public methods acceptable to the AEC Regulatory staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations and compliance with them is not required. (Emphasis added). Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Having confused the regulatory requirements associated with the program, Mr. Hudson now goes on to confuse the actions taken by others as part of the program, many of which took place years before his arrival on site. Mr. Hudson alleges that in the 1974-75 time frame, rupture restraint erection continued with only cursory QA/QC participation. The fact is that the welding effort was, in general, documented as required. In his own dissertation, Mr. Hudson points out several Pullman and PGandE audits and follow-up actions during this time period. While there were errors made, the QA/QC participation was significantly more than "cursory."

Mr. Hudson later complains about an active, aggressive QA/QC program that was attempting to build quality into the job while work was in

progress rather than inspecting quality after the fact. He complains that "QA/QC (was) directing production work." As on any job, there is a need for communication between the quality groups and the production groups. Such communication, which may include detailed discussions of what is acceptable and unacceptable and the reasons therefor, does not imply that QA/QC was providing engineering services or directing the production efforts. Having QC inspectors suggest and recommend acceptable practices involving such subjects as preheating and weld sequences certainly does not interject QA/QC into the engineering function. These are valid standard practices. The memos from the QA/QC supervisor to the inspectors detailing their responsibilities ensured that any misconception about actual responsibilities on the part of either party would be cleared up.

Mr. Hudson first complains about cursory QA/QC participation, then he complains about active, positive QA/QC participation. Mr. Hudson appears to be a confused, constant complainer, unable to recognize significant effective activity.

During the same time period, inspectors were given permission to mark process sheets "N/A" (Not Applicable) when specific, in-process inspection points did not apply. This solution, which did not extend to final inspections, ensured that contrary to the allegation, all check points were addressed and that no holes existed in the documentation.

Mr. Hudson only casually mentions the extensive and intensive reinspection and re-examination program, which occurred in 1978-79, for all heavy section field rupture restraint welds. He has focused on the cracks documented on various specific DRs and NCRs but doesn't appear to understand the significance of the overall activity.

The heavy section field welds in which cracks had been found were all reinspected, reexamined, and repaired, as required. An engineering evaluation was performed to determine the significance of defects which had been found and to help understand their impact on the functioning of the restraints, if these were ever needed in the event of a pipe rupture. These analyses also triggered a re-inspection and evaluation program for the shop-welded portions of rupture restraints.

The evaluation of the field welds caused the inspection program to be modified so as to better focus on the defects which had been found. Magnetic particle examination (MT) was used in conjunction with ultrasonic examination (UT) to examine subsequent work.

Mr. Hudson has also carefully failed to mention that many rupture restraints have been redesigned and/or rebuilt over the years in response to installation problems and/or revised loading criteria. Many of the original carbon steel restraints which experienced the cracking problem have been replaced with an austenitic stainless steel bar system. He also neglects to mention a reverification and

reanalysis of rupture restraints in 1982-83 and another extensive ultrasonic reinspection program, which revealed only one problem weld configuration in approximately 300 welds that were checked. That configuration had an unusual design and was redesigned and rewelded as were all similar configurations. The results of the reverification program were further reviewed by the IDVP and found acceptable.

Mr. Hudson would have us believe that QA/QC and preheat failures were the cause of rupture restraint cracks and lamellar tears. This is a gross over-simplification. Failure to preheat may have contributed to some defects and others were caused by basic component design and detail problems. Redesign and new details were required for resolution of some of the problems. The QA/QC documentation errors did not cause any of the hardware problems, all of which have since been corrected.

Mr. Hudson has difficulty separating trivialities from substance. He mistakes the decision to institute a more detailed documentation system for complex weldments, which required an expanded identification weld numbering system, as having some real practical significance in the safety of the plant. He also mistakenly attributes great significance to welders stamping their identification on hardware and having unique multidigital stamps. This issue was addressed previously in response to a GAP allegation.

The AWS Code does not require welds to be stamped. The stamping errors constitute a program deviation, nothing more.

Mr. Hudson's complaints regarding Welding Procedure Specifications were also fully addressed previously in response to the JI's CQA Motion, specifically the affidavit of Breismeister et al., pgs. 1-6 and 9-26. These complaints have no significance.

Again, in complaining about lack of a wall thermometer to determine "powerhouse" environment temperature, Mr. Hudson demonstrates an inability to separate trivialities from substance. The 50°F environmental temperature has little technical significance for the materials, thicknesses, and filler metals to which it applies at Diablo Canyon. Whether the environmental preheat is 50°F or 40°F makes essentially no difference. When it is remembered that Diablo Canyon is on the central California coast, the concern for documenting a 50°F (versus 40°F or 60°F) preheat is ridiculous. This allegation about missing thermometers, when viewed in light of rupture restraint welding and especially, the heavy sections which experienced cracking, is absurd because those thicknesses require 150°F preheat or more. Wall thermometers are irrelevant to such issues. And so, while the managements of Pullman and PGandE were addressing real problems, Mr. Hudson is complaining about the lack of wall thermometers, which had no significance.

Mr. Hudson continues to reiterate inconsequential issues in regards to the acceptance of minor and temporary attachments welded to rupture restraints. This issue was addressed in PGandE's response to NRC item #464 (DCL-84-195). Mr. Hudson fails to recognize these were not the kinds of welds which were cracking, that engineering analyses indicate that these welds were not a concern in and of themselves or in relation to the rupture restraints, and he fails to accept the significant fact that a sample of more than 25% of the 1,200 welds was examined and accepted. The acceptance of the remaining minor and temporary welds without NDE was appropriate.

Mr. Hudson appears to be totally confused about the acceptability of indicating a specific in-process inspection point as not being applicable (N/A) to a specific fabrication, whereas it is not acceptable to "N/A" a final inspection for permanent plant work. The distinction between non-applicable in-process points and a final inspection should be obvious to everyone.

Mr. Hudson appears to overstate the facts as regards some of the causes for failures "as determined by PGandE and Pullman." None of the cognizant PGandE and Pullman personnel recall identifying the causes stated by Mr. Hudson:

In Contrast to the "conclusions" put forward by Mr. Hudson, the actual conclusions as contained in PGandE Report No. 411-80.93 are as follows:

Defects in excess of those allowed by the fabrication code (AWS D1.0) were found in Diablo Canyon Rupture Restraint Welds completed by Pullman Power Products. No single element, process or error can be indicated as the primary cause for these defective welds. The defective welds exist because of inadequate or improper control of a combination of the following:

1. Weld sequence control.
2. Welding parameter control including preheat and interpass temperature.
3. Specification of nondestructive examination techniques.
4. Control of nondestructive examination techniques.
5. Control of thermal cutting processes.
6. Control of welding repair operations.
7. Base material specification and acceptance.
8. Weld joint design.

The defective Pullman Power Products weldments have been identified, evaluated and repaired where required. All repair operations were carried out using materials, designs, and procedures which precluded the problems encountered during initial fabrication.

Mr. Hudson has taken a few elements of each and expanded upon them to suit his own purposes.

Mr. Hudson, a semantic wizard but technically uninformed, correctly identifies that PGandE <u>preferred</u> the magnetic particle (MT) yoke testing method and then attempts to make an issue out of the fact that the PGandE procedure reflected a second method. A preference is just that, a preference. It is not a requirement, as Mr. Hudson would have one believe. In actuality this is another triviality which Mr. Hudson apparently does not understand. Either the yoke or prod methods are more than adequate.

Mr. Hudson apparently does not understand that PGandE specification

8833XR is applicable to contractors other than Pullman and to work other than rupture restraints. Thus, it was appropriate for PGandE to provide specific direction to Pullman for rupture restraints without changing the specification applicable to other work. As the rupture restraint problems were for thicker weldments, it was appropriate to exempt welds less than 9/16-inch thick from UT examination. In any case, welds which were visually examined but not nondestructively examined as they should have been, were, by Mr. Hudson's own admission, appropriately examined when the procedure was revised.

Furthermore, Mr. Hudson is completely, and apparently hopelessly, confused regarding the need to qualify NDE procedures. The need for NDE procedure qualification was thoroughly addressed in PGandE's response to the JI CQA Motion, see Arnold et al. Aff. at 7-10. It is absurd to suggest that the lack of procedural qualification for ultrasonic examination (UT) contributed to the weld cracking problem.

In the areas of walkdown inspection and calibration of torque wrenches, Mr. Hudson takes exception to the manner in which things were accomplished but he fails to point out that appropriate audits were undertaken, in some cases by Mr. Hudson himself, and that, where needed, appropriate corrective action was taken. Specifically, Mr. Hudson fails to note that the "QC/Engineering Walkdown Sheet" has a specific place to note exceptions and that this space was used to

explain deviations from established procedures. A recent review of some of the Unit 1 packages indicates that many DCNs were written on conditions found during the walkdown but not every condition required a DCN.

To require a separate DCN for every item identified during a walkdown is taking the requirements of the procedure totally out of context. This is especially true when many of these items were arc strikes, minor gouges, etc. The methods used to identify and document items discovered during walkdown were developed to efficiently perform the walkdown function. DCNs were then generated as required. All items were identified and appropriately dispositioned.

As to torque wrenches, Mr. Hudson infers that their major use was for rupture restraints when, in fact, they were and are predominantly used on hangers. Mr. Hudson is also incorrect when stating that no quality and technical requirements existed for this calibration service well beyond 1974. The appropriate procedure, ESD 213, was in place in 1971.

PGandE did write a Minor Variation Report (MVR) on the calibration situation as stated, but Mr. Hudson neglects to indicate that the tolerances required by the Pullman procedures had been reviewed against the PGandE procedures and were found to be in agreement. The PGandE calibration reports also reflected all the requirements of the

Pullman procedures and the reports were required to be reviewed by Pullman QA to verify compliance to Pullman procedures and identify any out-of-tolerance conditions so corrective action could be implemented.

The MVR and corrective action ensured that, should the Pullman requirements change, PGandE would know immediately since the requirements would be attached to each request for a tool calibration.

Conclusion

In his closing statement, Mr. Hudson asks the somewhat sarcastic question "[P]erhaps quality assurance is all irrelevant?" The emphatic answer is that "Quality Assurance is not only relevant, it is necessary and it has been appropriately implemented." What Mr. Hudson has failed to recognize throughout is that the purpose of a strong QA/QC program is to catch errors and to improve the confidence level in the end product. He goes to great ends to point out problems which have been corrected and focuses on the documentation process rather than the actual work completed. His incredible attention to minutiae without an appreciation and understanding of the big picture renders his alleged findings of shortcomings moot. There can be no realistic expectation that a project as complex as any power plant will be perfect or that all corrective actions will be immediately and completely effective.

This is especially true when a project has a 16-year history, involves a state-of-the-art technology and a constantly changing regulatory atmosphere. In this environment, an effective QA/QC program is most certainly relevant. But the process adopted and followed must be a reasoned and practical one which contributes to improvement of the product rather than merely producing a paper morass which becomes fertile ground for nitpickers such as Mr. Hudson and adds not one iota to the overall quality. It is unfortunate that individuals such as Mr. Hudson, with his limited depth and breadth of expertise, can repeatedly pose his allegations and then demonstrate a lack of the technical skills necessary to understand the responses to those allegations.

The record at Diablo Canyon clearly reflects an on-going program which sought to and did address the real issues, not manufactured ones. The Diablo Canyon project reflects the necessary confidence, based on sound engineering practices and long experience, that the plant will function in a manner which, under any conditions, will not be inimical to the health and safety of the public.

It is alleged that:

SUBJECT:

Diablo Canyon 3" x 3" x 1/4" steel tubing analysis.

DESCRIPTION OF FACTS:

On May 29, 1984 I was given a 1-3/4" piece of 3" x 3" x 1/4" square steel tubing from Diablo Canyon that was reported from a lot of a material used to fabricate pipe hangers. It had an identification marking, P. O. 14817.

I was asked to assess the overall quality of the material.

The sample was visually examined first, followed by grinding and etching for macro examination. Finally the sample was dye penetrant inspected.

FINDINGS:

Visual:

1. Extremely wide variation in corner radii, external and internal. (Fig. 1 & II)

2. A seam visible midway between the internal and external surfaces, extending most of the way around the piece. (Fig. I & II)

3. The surface is somewhat rough, lightly rusted and coated with what appears to be gray primer.

Macro Examination:

1. A seam was shown to extend around the entire piece for its full length. The seam was very close to midway between the inside and outside surfaces. (Fig. 1 & II)

2. The seam included a considerable quantity of inclusions for approximately 75% of it's [sic] length. (Fig. III)

Dye Penetrant Inspection:

1. Dye penetrant inspection revealed a massive discontinuity that extended approximately 75% of the

distance around the piece, and that there was little if any continuity in this region. (Fig. IV)

Dimensional:

1. Thickness varied from .231 to .242 in. Nominal thickness is .250. (Fig. V)

2. Radius variation was very large, from approximately 1/4 R. to 4/8 R. with none of the radii being uniform either internally or externally. (Fig. V) (5/4/84 Clark Aff. at 1-2.)

Dimensional variations undoubtedly exist throughout the entire lot of material from which this sample was obtained. Dimensions are inherently related to the rolling process. If the rolling process is adjusted to yield unacceptable dimensions, then the unacceptable dimensions will continue to be produced.

Steel quality within a heat of material will vary considerably. This particular piece of material is of very poor quality. It could be expected that some of this heat would be of better quality but also that some could be worse. (5/4/84 Clark Aff. at 3.)

The issues raised in these two allegations had been addressed by the Project quality program prior to receipt of the allegation. At the end of May, 1984, Project/Pullman found a single nonconforming 20 foot length of 3x3x1/4 inch structural steel tube in the normal course of work prior to installation and this was documented on a discrepancy report (DR 8488) dated May 30, 1984. The DR notes both the deviant corner dimension and the mid thickness seam or lamination. This allegation and those portions of Mr. Stokes allegations which relate to laminations contribute nothing further.

The defective condition is limited to a single 20 foot length of tube. The jobsite personnel immediately initiated an inspection

program to confirm the extent of the problem. All 1000 feet of 3x3x1/4 inch tube steel from this order are being inspected, including the tube in storage and the tube which has been installed. As of this submittal, no additional defective tube has been found. Specifically 200 feet, not yet installed at the time the condition was revealed has been found to have acceptable corners and no visible laminations. All the installed tubes have acceptable corners. The 200 feet which had not been installed has been cut up for installation and no laminations were revealed on the cut ends. This is significant because the cutting activity revealed the original length of defective tube. Thus a random 20% sample has been examined and found to be free of laminations.

This material was ordered in 1984 for use in Unit 2. This material has not been used in Unit 1.

Laminations in tubular products rarely occur and are the result of a transitory condition at the start of the roll of skelp material from which the tube is made. Center line laminations occur when the raw steel ingots and blooms are insufficienctly cropped and trimmed to remove the top center shrinkage which occurs during the inital steel solidification.

Thus this is a rather limited condition within an ingot. As the thick steel gets rolled to thinner thicknesses the length of the

center line defect increases within limits. Such a defect would not, because of its inherent nature, run the full length of the skelp (plate/strip) which is made into the round pipe.

The round pipe is then run through a forming operation to make the square or rectangular shape. At the start of a production run, while the forming rolls are being adjusted to provide the correct size and squareness, the tubes may not develop the required shape. This is a transitory condition while adjustments are made.

The nature of both conditions, laminations and corners, indicate that the length of tube in which these were found should not have been shipped. Because of the transitory nature of the defects it is not likely that additonal defective material will be found on site, but an intensive investigation is under way, and being documented under DR 8488.

It is important to note that this defective material was revealed in the normal course of events and reported properly. The quality program continues to work effectively. The wall thickness variation 0.231 to 0.242 inches for 1/4 inch nominal thickness is permissable as the ASTM A500 specification has a 10% thickness tolerance.

JIR-71 and JIR-72

It is alleged that:

ASSESSMENT OF QUALITY:

Material quality was extremely poor as indicated by the gross seam with massive inclusions.

Overall fabrication technique was very poor considering the gross dimensional variations and material discontinuities. (5/4/84 Clark Aff. at 2.)

SOURCE OF MATERIAL AND COMPARISON WITH U.S. PRODUCED MATERIAL:

It is my opinion that the types and severity of defects in this material are consistent with those found in various foreign produced materials. The dimensional and material related items noted above are often encountered in foreigm [sic] materials that are available at bargain prices.

The gross nature and combination of defects are consistent with foreign manufacture or the lowest quality domestic product, sold as 'seconds'.

Domestic mills generally are noted for producing high quality material, and would not normally produce material with such a combination of significant defects. This sample is not representative of even average quality domestic steel. (5/4/84 Clark Aff. at 2.)

We agree that the sample is of poor, unacceptable quality and it had previously been rejected on or before May 30, 1984. We also agree that this is not representative of domestic materials as used at Diablo Canyon. The fact is that this was produced in a domestic Chicago area mill. The materials test reports are on file

at the site. The manufacturer has been contacted to determine how the defective length of tube was shipped and to prevent reoccurence. The manufacturer has confirmed that these defect types are the result of transitory conditions at start up of the manufacturing process.

It is alleged that:

WELD STRENGTH:

It is my understanding that this type material is typically welded to other structural members using simple fillet welds (not full penetration).

If this piece of material were placed in tension with simple fillet welds the load bearing capacity would be seriously reduced because of the effective reduction of cross sectional area due to the seam in the material.

If the seam were located closer to the surface, which it very possibly could be, in other pieces of material, the load bearing capacity would be even further reduced. (5/4/84 Clark Aff. at 4.)

As stated above, this condition had been identified on DR 8488. It was identified prior to installation. It is confined to a single length of tube steel which was not installed. While it is not anticipated that any more defective material exists, there is an on-going investigation to account for this material. Therefore, effects of laminations on structural performance need not be addressed.

It is alleged that:

This statement evidences my concern for inadequate controls on early concrete pours at the Diablo Canyon Nuclear Power Plant and any lingering effects on the condition of the Plant, particularly if the plant should be subjected to the stresses from an earthquake. I am also concerned about the effects on consumers who have to pay for Diablo Canyon, if the waste that I observed continued during the ten years after I left.

I worked for Pacific Gas and Electric Company (PG&E) at Diablo Canyon from 1968-January 1973, initially as a surveyor but primarily as an inspector for early concrete pours and installation of rebar at Unit I and Unit II. I also served as night shift inspector for the breakwater, and as an inspector for cadwelds. I resigned in 1973. (6/1/84 Kinney Aff. at 1.)

Contrary to this allegation, Class I concrete pours at the Diablo Canyon Nuclear Power Plant were placed in accordance with the design drawings, specifications, and quality assurance programs and all records have been maintained. The concrete was tested on a routine basis during placement and in all cases, the strength met design, licensing, and regulatory requirements.

As stated in his affidavit, Mr. Kinney was an employee of PGandE who resigned on January 31, 1973. Several PGandE employees who worked at Diablo Canyon at the same time as Mr. Kinney indicate that Mr. Kinney told them that he resigned to continue and/or expand his outside activities as a construction/renovation contractor. It is curious indeed that he has waited more than 11 years before raising his voice.

It is alleged that:

Failure to review as-built drawings. Supposedly as-built drawings were checked after being signed off. I suspected that was not the case, however and decided to test my suspicions. In mid-late 1972 I signed drawings as "Roy Rogers," "Gene Autry," "Donald Duck" and "John Wayne." I then waited to be challenged. No one noticed. (6/1/84 Kinney Aff. at 1.)

Mr. Kinney had authorization to sign "as-built" reinforcing steel payment sepias. These sepia drawings were to document for pay purposes the work performed by Pacific States Steel Corp., the reinforcing steel subcontractor. They were not quality documents at all.

Subsequent to Mr. Kinney's departure from PGandE, these payment sepias were reviewed and compared to the final "as-built" (original tracings) drawings. No fictitious signatures were found during this review.

JIR-77

It is alleged that:

Field inspectors from Pittsburg Testing were hired to check reinforcement bar steel. They regularly complained to me of threats from PG&E's steel contractor, Pacific States Steel, if they wrote up problem reports for conditions such as excessive numbers of voids in cadwelds. I recall the reports of harassment as follows: "We pay your check, so keep your mouth shut or you'll be fired." (6/1/84 Kinney Aff. at 2.)

In response to the allegation, PGandE inquired of the supervisors in charge of Mr. Kinney at the time he was employed and no one has any knowledge of any reports of harassment of field inspectors by Pacific States Steel.

Former Pacific Testing Laboratory (PTL) employee, Dave Aitken, who worked with Mr. Kinney, stated that he had not experienced or even heard of any instances of harassment of PTL employees by Pacific States Steel employees.

It is alleged that:

> One of the main reasons I resigned was that there were too many things that I couldn't sign off, but management wouldn't let me reject. Management consistently overrode my attempts to stop work. One example involved reinforcement bar steel that had been bent back and forth. I wrote up the practice, because it establishes break points and compromises the strength of the steel. This is a basic rule of concrete work. Even on this issue however, management overrode me. (6/1/84 Kinney Aff. at 2.)

Contrary to the allegation, individuals who knew Mr. Kinney onsite recall that he stated the main reason he left Diablo Canyon was to pursue his own business interests.

Civil Instruction No. C-1A states "Inspector will stop or reject work which, in his opinion, is below specified quality standards only after noting that contractor's inspector is not taking proper action to document such deviations as well as initiating corrective measures. Should work have to be stopped or rejected, inspector shall immediately inform his supervisor (field engineer). The above remarks shall apply should inspectors find reinforcing steel with significant specification discrepancies on non-Quality Assurance systems. In this case, inspectors can stop or reject the work." PGandE inspectors were so instructed by the Resident Civil Engineer.

A complete review of Mr. Kinney's logbooks shows that there were instances where he stopped or rejected work and was supported in his stop-work order to contractor by PGandE management. All problems noted in Mr. Kinney's logbook were reviewed by supervisory personnel to assure that the problem resolution was in accordance with the Quality Program. No instance of noncompliance was observed.

It is alleged that:

Construction supervisors offered me $100 per night to keep my eyes shut during the breakwater work. I declined but was disillusioned both with the offer and myself for briefly considering it. (6/1/84 Kinney Aff. at 2.)

Supervisors in charge of DCP when Mr. Kinney was employed have no knowledge of bribes being offered. A complete review of Mr. Kinney's daily logbooks has revealed no notations of bribes being offered.

There is no evidence available to verify whether the attempted bribe occurred or not. If this incident occurred as Mr. Kinney indicated in his affidavit, he did the proper thing in declining the offer, but he acted improperly in not immediately reporting it to his supervisor.

In any event, the breakwater is not a Class I safety-related structure, has been redesigned, and substantially reinforced in conjunction with the repair of storm damage during the past several years.

It is alleged that:

5) Advance warning and gag orders for Atomic Energy Commission inspections. In my four years on-site, management always gave us at least one day advance warning before all Atomic Energy Commission (AEC) inspections. Management told the inspectors not to ask questions or volunteer any information beyond what was specifically requested by the AEC. (6/1/84 Kinney Aff. at 2.)

Mr. Kinney is correct as to advance notice for announced-only visits of the AEC/NRC. Such visits are a common part of the AEC/NRC inspection process. This notice was for the purpose of having personnel available who are familiar with areas that the AEC had scheduled to inspect during their visit.

PGandE inspectors were instructed to provide any information that the AEC requested about which they had knowledge. If they did not know answers to any AEC question, they were instructed to tell the AEC that they did not know and refer the AEC to their supervisors, who would get answers as required.

At no time were PGandE employees ever instructed not to ask or answer questions of the AEC.

It is alleged that:

> I feel strongly that a significant amount of the construction costs were indefensible. One example involved $30,000 that was spent to build the guards shack and fence. I told my boss that it was unacceptable to spend so much for so little. to the best of my recollection, the message that he communicated to me could be summarized as follows: "You might as well accept it, because we can just go to the Public Utilities Commission and get it all back. The plant will be paid off six months after it begins operating." (6/1/84 Kinney Aff. at 3.)

This item does not present a quality concern and does not present a safety problem. However, contrary to the implication of the allegation, security fencing, in the form of chain-link fences and guardposts, is cost-effective as a means of protecting millions of dollars worth of records, materials, and equipment.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

AFFIDAVIT OF F. C. BREISMEISTER, M. E. LEPPKE, R. G. OMAN AND L. E. SHIPLEY

STATE OF CALIFORNIA)
) ss.
COUNTY OF SAN FRANCISCO)

The above, being duly sworn, depose and say:

I, Fred C. Breismeister, am Manager of the Research and Engineering/Materials and Quality Services Department, San Francisco Area Office, for the Bechtel Group.

I, Myron E. Leppke, am Onsite Project Engineer for the Diablo Canyon Project.

I, Robert G. Oman, am an Assistant Project Engineer for the Diablo Canyon Project, and from August 1982 to October 1983, acted as Onsite Project Engineer at the Diablo Canyon Nuclear Power Plant.

I, Larry E. Shipley, am Technical Consultant for Piping for the Diablo Canyon Project.

Our responses to the following allegation numbers in PGandE Letter No. DCL-84-239, dated June 26, 1984 are true and correct to the best of our knowledge, information, and belief.

JIR-1, 2, 3, 5, 6, 7, 8, 9, 10, 11, 12, 14, 15, 18, 19, 20, 21, 22, 23, 24, 25, 26, and 27.

Dated: June 26, 1984

F. C. Breismeister

M. E. Leppke

R. G. Oman

L. E. Shipley

Subscribed and sworn to
before me this 26th day
of June, 1984

C. T. Neal-Madison

Cynthia Neal-Madison
Notary Public in and for the
City and County of San Francsico
State of California
My commission expires
December 27, 1985

C. T. NEAL MADISON
NOTARY PUBLIC — CALIFORNIA
CITY AND COUNTY OF
SAN FRANCISCO
My Commission Expires Dec. 27, 1985

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

AFFIDAVIT OF T. G. De URIARTE AND M. J. JACOBSON

STATE OF CALIFORNIA)
) ss.
COUNTY OF SAN FRANCISCO)

The above, being duly sworn, depose and say:

I, Thomas G. De Uriarte, am Director of Program Management, Quality Assurance Department, Pacific Gas and Electric Company.

I, Michael J. Jacobson, am Project Quality Assurance Engineer for the Diablo Canyon Project.

Our responses to the following allegation numbers in PGandE Letter No. DCL-84-239, dated June 26, 1984 are true and correct to the best of our knowledge, information, and belief.

JIR-4, 16, and 17.

Dated: June 26, 1984

T. G. De Uriarte

M. J. Jacobson

Subscribed and sworn to
before me this 26th day
of June, 1984

C. T. Neal-Madison

Cynthia Neal-Madison
Notary Public in and for the
City and County of San Francsico
State of California
My commission expires
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

AFFIDAVIT OF D. A. ROCKWELL AND L. G. RASMUSSEN

STATE OF CALIFORNIA)
) ss.
COUNTY OF SAN FRANCISCO)

The above, being duly sworn, depose and say:

I, Donald A. Rockwell, am Special Projects Engineer for the Pacific Gas and Electric Company at the Diablo Canyon Nuclear Power Plant.

I, Laurence G. Rasmussen, am Manager, Civil-Hydro Construction Department, Pacific Gas and Electric Company.

Our responses to the following allegation numbers in PGandE Letter No. DCL-84-239, dated June 26, 1984 are true and correct to the best of our knowledge, information, and belief.

JIR-13, 75, 76, 77, 78, 79, 80, and 81.

Dated: June 26, 1984

D. A. Rockwell

L. G. Rasmussen

Subscribed and sworn to
before me this 26th day
of June, 1984

C. T. Neal Madison

Cynthia Neal-Madison
Notary Public in and for the
City and County of San Francsico
State of California
My commission expires
December 27, 1985

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

AFFIDAVIT OF F. C. BREISMEISTER, H. W. KARNER, AND D. A. ROCKWELL

STATE OF CALIFORNIA)
) ss.
COUNTY OF SAN FRANCISCO)

The above, being duly sworn, depose and say:

I, Fred C. Breismeister, am Manager of the Research and Engineering/Materials and Quality Services Department, San Francisco Area Office, for the Bechtel Group.

I, Harold W. Karner, am Quality Assurance/Quality Control Manager for the Pullman Power Products Corporation.

I, Donald A. Rockwell, am Special Projects Engineer for the Pacific Gas and Electric Company at the Diablo Canyon Nuclear Power Plant.

Our responses to the following allegation numbers in PGandE Letter No. DCL-84-239, dated June 26, 1984 are true and correct to the best of our knowledge, information, and belief.

JIR-28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, 71, 72, 73, and 74.

Dated: June 26, 1984

F. C. Breismeister

H. W. Karner

D. A. Rockwell

Subscribed and sworn to
before me this 26th day
of June, 1984

C. T. Neal-Madison

Cynthia Neal-Madison
Notary Public in and for the
City and County of San Francsico
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My commission expires
December 27, 1985

C. T. NEAL MADISON
NOTARY PUBLIC — CALIFORNIA
CITY AND COUNTY OF
SAN FRANCISCO
My Commission Expires Dec. 27, 1985

PROFESSIONAL QUALIFICATIONS

Professional qualifications for the following affiants have been previously submitted in the PGandE Response to Joint Intervenors' Motion to Reopen the Record on Design Quality Assurance. (March 6, 1984)

Fred C. Breismeister
Myron E. Leppke
Michael J. Jacobson
Robert G. Oman
Larry E. Shipley

Professional qualifications for the following affiants have been previously submitted in the PGandE Response to Joint Intervenors' Motion to Reopen the Record on Construction Quality Assurance. (March 19, 1984)

Harold W. Karner
Donald A. Rockwell

In addition, statements of the Professional Qualifications of the remaining affiants are attached.

Thomas G. DeUriarte
Laurence G. Rasmussen

PROFESSIONAL QUALIFICATIONS OF

THOMAS G. DEURIARTE

My name is Thomas G. DeUriarte. I am a senior engineer in the Quality Assurance Department with Pacific Gas and Electric Company in San Franciso.

I have been with PGandE since 1967, when I graduated with a B.S. degree in Civil Engineering from the University of California in Berkeley. I am a registered civil engineer and quality engineer in California and a member of the American Society for Quality Control, the American Society for Nondestructive Testing, and the California Association of Professional Engineers.

My current assignment as senior engineer with the Quality Assurance Department began in July 1977. I have been responsible for the administration of the audit program, scheduling audits, and assigning personnel to perform audits of Company and supplier activities. I am responsible for managing these audits, which verify compliance with applicable regulations of the NRC and other requirements, such as nuclear industry guides and standards, and corporate procedures and specifications. I am also responsible for the supervision of department personnel for the completion of special projects, departmental participation in reviewing discrepancies, and development of departmental training and participation in licensing hearings as a representative of the QA Department.

From 1976 to 1977, I was an engineer in the Quality Assurance Department with responsibility for performing audits of Company and supplier activities, developing procedures for the corporate quality program, reviewing suppliers' quality assurance/control programs, developing corporate specifications for suppliers' quality assurance programs, and coordination with regional NRC inspectors.

For the previous two years, I was a scheduling engineer in PGandE's Station Construction Department for the power plant at Pittsburg, California. From 1967 to 1970, I was a field engineer in PGandE.

PROFESSIONAL QUALIFICATIONS OF

LAURENCE G. RASMUSSEN

My name is Laurence G. Rasmussen. I am employed by Pacific Gas and Electric Company (PGandE) as Manager, Civil-Hydro Construction Department. I am a graduate of the University of Arizona and a Registered Engineer in California. Prior to joining Pacific Gas and Electric Company - General Construction, in 1968, I worked five years for the County of Napa and two years for private engineering firms.

I spent seven years at PGandE's Diablo Canyon Nuclear Power Plant. During this period, my responsibilities included Field Engineering for the construction of two containment structures and the supervision of construction inspection and contract administration for civil contracts.

I also worked at PGandE's Humbolt Bay Nuclear Power Plant and Pittsburg Power Plant. I was Resident Civil Engineer on PGandE's Helms Pumped Storage Project. I supply Construction Department input for power plant licensing processes.

My current responsibilities include managing work at PGandE's Diablo Canyon Power Plant, the Geysers Power Plants, and Kerckhoff Underground Power Plant.

I am a member of the American Society of Civil Engineers and serve on the National Committee for Social and Environmental Concern in Construction. I also serve on the Environmental Committee for the California Associated General Contractors.