ENCLOSURE

REGULATORY GUIDE 1.97 REVIEW PROJECT FINAL REPORT

DIABLO CANYON POWER PLANT UNITS 1 AND 2

DECEMBER 1993

PACIFIC GAS AND ELECTRIC COMPANY

EXECUTIVE SUMMARY

Regulatory Guide (RG) 1.97 is titled "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Pacific Gas and Electric Company (PG&E) has completed the review and evaluation activities for a RG 1.97 Review Project for Diablo Canyon Power Plant (DCPP). This is the final report for the project.

The project was initiated in response to issues identified through design basis reconstitution activities and Nonconformance Report NCR investigations. PG&E apprised the Nuclear Regulatory Commission of these issues, and of the plan to perform additional reviews, at a meeting with Office of Nuclear Reactor Regulation (NRR) personnel in Rockville, Maryland, on December 18, 1991. At a second meeting with NRR, on November 19, 1992, PG&E presented the status and preliminary results of the project. PG&E Letter DCL-93-122, dated May 18, 1993, provided an interim project status report.

The project systematically reviewed the design and qualification of post-accident monitoring instrumentation at DCPP with the various attributes for the instrumentation specified in the RG. The review comprised two components. First, because of their relative importance, PG&E reviewed the instrumentation for all of the 20 variables listed in Table 3.3-10, "Accident Monitoring Instrumentation," of the DCPP Technical Specifications (TS), which apply to both Units. Second, PG&E completed a sampling review of the total population of RG 1.97 instrument loops in both Units. The sampling review, by itself, provided a "95/5" confidence level (see Definitions, Section 6.1) of the adequacy of the entire RG 1.97 instrument population. Because the TS variable reviews were in addition to the sampling reviews, the actual overall confidence level provided by the project was greater than 95/5.

The project identified various findings. These generally fell into four areas, as described in this report: electrical isolation (power and signal), electrical separation, environmental qualification, and seismic qualification. The findings that were determined to be "potential problems" or "discrepancies" (as defined by the project instructions) were reviewed for safety significance, operability, and reportability. The overall number of findings was considered very low when compared to the total number of individual components and attributes reviewed. None of the findings were considered to be safety significant. The health and safety of the public were not affected.

Not all actions resulting from the project reviews have been completed. However, project finding evaluation activities have been completed such that a "resolution path" has been established for each project finding.

The project documented various DCPP-specific "exceptions" to the RG. These, along with their associated technical bases, were submitted to the NRC by letters DCL-93-040 and DCL-93-284, dated February 17, 1993, and December 17, 1993, respectively. (the later letter encompassed the exception information contained in the first letter).

The project also identified a potential issue with 120 VAC power isolation. That issue, described in PG&E Letter DCL-93-078, dated April 6, 1993, is being tracked separately from the RG 1.97 Review Project. PG&E will complete the breaker review activities described in DCL-93-078 and submit a final report for that issue by December 31, 1994.

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1.0 INTRODUCTION AND PURPOSE

Pacific Gas and Electric Company (PG&E) has completed a self-initiated Regulatory Guide (RG) 1.97[1] Review Project for Diablo Canyon Power Plant (DCPP) Units 1 and 2. The project systematically reviewed the design and qualification of post-accident monitoring instrumentation at DCPP with the various attributes for the instrumentation specified in the RG. This is the final report for the project.

This report is essentially an updated version of PG&E's interim report for the RG 1.97 Review Project, which was transmitted to the Nuclear Regulatory Commission (NRC) by Letter DCL-93-122[25], dated May 18, 1993. Some additional findings were identified; however, these generally fell into the same broad finding areas as identified in the interim report. No new significant findings were identified. For ease of review, we have chosen to make this final report a "stand-alone" document, rather than a supplement or addendum to the interim report. That is, this final report is in the same general format as the interim report, and significant portions of various sections and appendices are reproduced unchanged from the interim report. Thus, reference to the interim report is unnecessary.

Information pertaining to RG 1.97 "exceptions" identified by the project has been submitted separately, in PG&E Letters DCL-93-040[23], dated February 17, 1993 and DCL-93-284[26], dated December 17, 1993 and is not discussed in detail in this report. Also, the project identified a potential issue with 120 VAC power isolation. This issue was the subject of PG&E Letter DCL-93-078[24], dated April 6, 1993, and is being tracked separately from the RG 1.97 Review Project and is not discussed in detail in this report.

The RG 1.97 Review Project has been the subject of two meetings in Rockville, Maryland, between PG&E representatives and NRC Office of Nuclear Reactor Regulation (NRR) personnel. At the first meeting on December 18, 1991, PG&E described the issues that prompted project initiation and outlined the plan to perform additional reviews. PG&E presented the preliminary results of the project at the second meeting on November 19, 1992.

Detailed information on the background of RG 1.97 implementation at DCPP and the issues that prompted PG&E to initiate this review are contained in Appendix A.

Note that PG&E is committed to Revision 3 of RG 1.97, dated May 1983 (as indicated in PG&E letters to the NRC dated September 9, 1983[2], and January 25, 1985[3]). Unless otherwise stated, all references to RG 1.97 in this report pertain to the <u>Revision 3</u> version.

2.0 PROJECT METHODOLOGY

2.1 <u>TS Variable and Random Loop Review Process</u>

The RG 1.97 Review Project comprised two principal components: Technical Specification[15] (TS) variable reviews and Random Loop (RL) reviews. The TS variable reviews examined the 20 instrumentation variables listed in TS Table 3.3-10 ("Accident Monitoring Instrumentation"). These 20 variables are listed in Table 1 of this report. Because of their relative importance, the TS variable reviews received highest priority.

The purpose of the RL reviews was to provide a high degree of confidence in the adequacy of the design and qualification of the remainder of RG 1.97 instrumentation at DCPP.

Figure 1 provides an overview of the RG 1.97 review process. The review methodology provides for the verification, on a loop-by-loop basis for each variable, of the design and qualification attributes in Table 1 of RG 1.97. The reviews were documented in "loop packages." A project instruction controlled the assembly, review, approval, and revision of the loop packages. Appendix B describes the review methodology and the contents of each loop package in detail.

For the TS variable reviews, there is one package for each variable. The package comprises the RG 1.97 loops associated with the variable for both DCPP Units.

The RL reviews were based on conventional statistical sampling methodology. There are over 700 total RG 1.97 instrument loops for both DCPP Units. This total population was reduced to approximately 530 by eliminating loops that were known to contain instances of potential deviation from RG 1.97 criteria, and that had already been extensively evaluated for that reason. For example, the containment isolation valve (CIV) position indication loops were eliminated because of the known issues (e.g., redundancy, separation, environmental qualification (EQ), and seismic qualification (SQ), depending on the particular valve) and consequent evaluations associated with those loops. The 530 loops formed the base from which 60 RLs were randomly selected. The 60 selected loops are listed in Table 2 of this report.

As the detailed reviews of the RLs commenced, it was determined that three of the 60 selected RLs were, in fact, not RG 1.97 instrumentation loops. These loops were RL25, RL26 and RL55. A calculation was prepared to demonstrate that the desired "95/5" confidence level (see Section 2.3, "Acceptance Criteria") could still be accommodated using a random sample population of 57 instrument loops.

Note also that there was some overlap between this set of 57 loops and the set of variables from TS Table 3.3-10. Specifically, 15 of the 57 RLs were part of the TS variable review. Thus, the RL reviews resulted in the preparation of 42 loop packages. The RL review packages were prepared in the same manner as the TS variable packages.

The reviews have been completed for both the TS variables (both Units) and the RLs. The resulting review packages are in the process of final review and approval. The reviews resulted in "findings." Findings were categorized as either potential problems, discrepancies, exceptions, or clarifications as defined in Section 6.2. The findings (other than clarifications and exceptions) were reviewed for safety significance, operability, and reportability. As discussed in Section 5.0, the findings were determined not to be safety significant. Not all actions resulting from the project reviews have been completed. However, project finding evaluation activities have been completed such that a "resolution path" has been established for each project finding. (Approximately 82 percent of the total scope of identified non-hardware actions resulting from the project have been completed.)

The results of the TS variable and RL reviews are discussed in Section 3.0.

2.2 Areas Beyond RG 1.97 Project Scope

2.2.1 Areas Previously Reviewed

Certain aspects of conformance with RG 1.97 had received extensive review prior to the initiation of the RG 1.97 Review Project and/or had been initially implemented with significant design conservatism. For these reasons, these areas were excluded from the scope of the project and were not examined individually for each successive TS variable or RL review. These areas are as follows:

(a) Human Factors -- PG&E addressed human factors engineering issues, in response to NRC Generic Letter (GL) 82-33[5], by performing a comprehensive Detailed Control Room Design Review (DCRDR). This program consisted of identifying departures from good human factors practices through the use of task analyses and operability walk-throughs, operator interviews, and checklist surveys of the control boards to human factors guidelines. RG 1.97 indications were reviewed during the performance of the DCRDR. As a result of the DCRDR, identification labels were attached to post-accident monitoring (PAM) indications in accordance with the requirements of RG 1.97.

Subsequent to the DCRDR, PG&E established a procedural requirement to ensure human factors review of design changes for instrumentation within the control room, including the RG 1.97 indications. The RG 1.97 indications are reviewed for suitability of range and readability of scales.

As an example of how human factors principles were applied to indicators and recorders in the control room, the indicator scales were reviewed for readability to meet human factors requirements. The recorders have good chart visibility, removable chart magazines, and disposable fiber-tip pen cartridges. The recorder scales have been selected in accordance with human factors guidelines. The scales provide information that is immediately useable by the operators. Indicators and recorders are labeled. Size graduated labeling and demarcation lines have been provided that meet human factors requirements. Unique identification (a "PAMS" label) is provided in accordance with RG 1.97 requirements for Type A, B and C variables in Categories 1 and 2. Based on the above, it was determined that "Human Factors" would not be further examined as part of the individual RG 1.97 project reviews.

[Note: Human factors were considered in the case of CIV position indication (white and red/green light indication) as discussed in Operability Evaluation (OE) No. 91-13. However, this was a concern related to the circuit design

logic and function; it did not indicate a need for further human factors review.]

(b) Electrical Separation Within Panels -- For many RG 1.97 variables, indication signals are produced from the output of the plant protection system cabinets. These cabinets were furnished by the nuclear steam supply system (NSSS) vendor (Westinghouse). The cabinet internals are similar to other previously licensed Westinghouse plants and have received extensive review by the industry and by the NRC. Other PAM inputs are typically derived from various systems that require redundant channels or trains. For these inputs, separate panels are typically provided for each channel or train, and internal separation is not required. In a few other cases, separation has been provided by other means in accordance with our Final Safety Analysis Report (FSAR) Update[17] description. Reviews and walkdowns (discussed below) provide confidence that these measures are extensive and adequate.

The readout devices for RG 1.97 variables are located in the control room. The main control room panels were supplied by Westinghouse and, in a few cases, other vendors. The design of the main control room panels followed Westinghouse experience from previously licensed nuclear plants. The design of the other main control room panels was determined by PG&E specifications that implemented detailed separation criteria.

Subsequently, a number of modifications have been made, along with a large number of walkdowns associated both with these modifications and in response to various Nonconformance Reports (NCRs). Separation of wiring and equipment has consistently been found to meet the single failure criterion. (In a few cases where the requirements for panel wiring described in the DCPP FSAR Update[17] were not met, the panels were upgraded accordingly.)

The post-accident monitoring panels PAM1 and PAM2 were vendor-supplied under PG&E specifications. As stated previously, the specifications provided detailed separation requirements for these panels. The panels were furnished under 10 CFR 50, Appendix B Quality Assurance programs.

By having nuclear industry vendors supply the panels or by having separate panels for each channel or train, followed by extensive reviews and walkdowns, PG&E believes that the separation within panels fully meets PG&E's commitments as described in the FSAR Update. As a result, it was determined that internal separation within panels would not be further examined as part of the individual RG 1.97 project reviews.

(c) Servicing, Testing and Calibration -- PAM instrumentation is tested using procedures that provide guidance for checks,

verifications, and calibrations. During the preparation of Design Criteria Memorandum (DCM) T-34 ("Post-Accident Monitoring Instrumentation"), DCPP Surveillance test procedures, instrument loop tests and maintenance procedures were reviewed to confirm that procedures and/or tests are in place that assure, on an ongoing basis, that the RG 1.97 PAM variables are and will remain functional. The review also verified the calibrated ranges of RG 1.97 Category 1 PAM variables to assure that the range corresponds to that listed in DCM T-34. The range verification also looked at several Category 2 and Class II variables that are included in DCPP Technical Specification 3.3.3.6. Based on the above, it was determined that "Servicing, Testing and Calibration" would not be further examined as part of the individual RG 1.97 project reviews.

(d) Seismic Qualification of Class 1E Raceway and Conduit Supports -- The general design approach for Class 1E raceway and conduit supports at DCPP was very conservative, as evidenced by the following:

- Conservative standard values were generally used for the dead weight loading (pounds per foot) of the raceway or conduit, even though the actual loading is much less. (In some specific cases, more exact dead weight loading information was used.)

- A conservative standard maximum span length of 8.5 feet was generally used, even though the supports are actually spaced closer together.

  (Regarding the above two items, DCM T-8, "Structural Design of Electrical Raceways and Class 1E Supports," Revision 0, Section 1.2 states: "A great majority of supports are spaced much closer and carry significantly less weight than allowed by the generic support design.")

- The current design requirements assure the structural adequacy of supports when subjected to either the "Double Design Earthquake" or Hosgri events (DCM T-8, Revision 0, Section 4.3.1).

- DCPP-specific shake table testing of electrical raceway and conduit supports has demonstrated that typical DCPP supports can withstand seismic accelerations that are significantly higher than the design basis (DCM T-8, Revision 0, Section 4.3.2.2).

- Observations of historical earthquake damage around the world show that electrical raceway supports have excellent earthquake resistance performance. This is attributed to high ductility and damping for the cable

tray/conduit-support system (DCM T-8, Revision 0, Section 4.3.2.2).

Based on the above, it was determined that the seismic qualification of Class 1E raceway and conduit supports would not be further examined as part of the individual RG 1.97 project reviews.

(e) Quality Assurance -- PG&E evaluated the need to reverify the quality assurance attribute in RG 1.97 Table 1 for every individual component reviewed. The evaluation indicated that PG&E had procured equipment in accordance with applicable quality assurance requirements since the commencement of plant construction. Quality assurance processes have been maintained through commercial operation, and continue to the present day. While documentation of the quality assurance attributes for certain sub-components did not have the same level of detail as required today, the methods used were acceptable, as described in PG&E's letter DCL-93-040[23], dated February 17, 1993 (see Section 2.2.2 below).

Because PG&E has demonstrated compliance with quality assurance practices throughout DCPP's construction and operational history, PG&E determined that it was not necessary to reverify the RG 1.97 quality assurance attribute.

2.2.2 Other Areas

PG&E Letter DCL-93-040[23], dated February 17, 1993, provided two positions that were part of PG&E's presentation to the NRC on November 19, 1992. The first pertained to ANSI/ANS Standard 4.5-1980. The second pertained to historical sub-component quality assurance documentation. These positions provide the bases for limiting the scope of our review in these areas.

2.3 Acceptance Criteria

The satisfactory results from the review of the selected RLs have provided a "95/5" confidence level (see Definitions, Section 6.1) in the adequacy of the design and qualification of the entire RG 1.97 instrument loop population for both DCPP Units. This confidence level has been achieved because there were no identified "potential problems" in any of the selected RLs for which a hardware design change was subsequently determined to be required to resolve. A calculation and an engineering evaluation were prepared that confirm the statistical bases for concluding that the 95/5 criterion has been satisfied.

Although the project instructions did provide for expanding the size of the sample by randomly selecting additional loops for review if necessary to assure the 95/5 confidence level, the satisfactory results of the RL reviews meant that no such sample expansion was required.

[Note that where a common issue was identified across the sample population, the project instructions allowed for removing that particular issue from further consideration by the RL review effort, and initiating a separate review directed specifically at the identified common issue. For example, the issue with 120 VAC power isolation (see PG&E letter dated April 6, 1993[24]) was removed from random sample consideration and is being resolved by the performance of a separate, 100 percent review.]

The overall RG 1.97 Review Project results have, in fact, exceeded the 95/5 criterion. This is because the RL reviews by themselves achieved the 95/5 criterion; and the 100 percent review of the TS variables, considered the most important, was in addition to the RL reviews. Therefore, the combination of the TS variable reviews and the RL reviews have assured a confidence level that is better than 95/5. [Note: At the November 19, 1992, meeting between PG&E and NRC, PG&E indicated that additional reviews would be performed such that all RG 1.97 Type A and Category 1 variables would be reviewed. In view of the numbers and significance of the findings identified by the project, PG&E determined that these additional reviews were not warranted.]

Note also that the 5 percent in the 95/5 criterion represents uncertainty, not failed equipment. Further, the uncertainty is "compartmentalized"; i.e., if an issue did exist outside of the 95 percent confidence envelope, it would be confined to only one aspect of conformance with RG 1.97 (EQ, SQ, separation, etc.). Thus, any such problem would not mean that the loop would necessarily fail in an accident.

Finally, PG&E evaluated all potential problems identified by the project. Should any potential problem have indicated that a significant safety issue existed, additional steps beyond those required by the sampling program would have been taken. No such significant problems were identified.

2.4 Classification and Evaluation of Findings

Any findings identified during the reviews were documented as either potential problems, discrepancies, exceptions, or clarifications, as discussed in Appendix B, and were tracked in accordance with PG&E standard quality practices. For each finding:

(a) The particular aspect of deviation from RG 1.97 (e.g., power or signal isolation, separation, seismic qualification, etc.) was identified.

(b) The finding was reviewed for safety significance, operability and reportability. Only one finding was determined to be reportable. That finding involved the potential for failure of surge suppressor diodes in the auxiliary building ventilation system (ABVS) dampers and fan motor control circuits, and was determined to be reportable in accordance with 10 CFR 50.73. (See DCPP Licensee Event Report (LER) 2-92-005-01[18].) Although identified by the RG 1.97 project, that finding was beyond the scope of the RG 1.97 review since it affected control system functions other than indication. Hence, it was addressed separately from the

RG 1.97 Review Project. (PG&E also submitted a report[24], following discussions with the NRC, on the potential issue with 120 VAC power isolation.)

(c) If required, Prompt Operability Assessments (POAs) and OEs were prepared to document operability determinations.

(d) A closure path for each finding was established.

(e) The finding was closed once the required actions had either been completed, or transferred to Action Requests (ARs) and/or Action Evaluations (AEs) in PIMS for tracking purposes.

3.0 RESULTS

3.1 Overview of Project Findings

The project has generated 198 total findings. Of these, 60 were clarifications and 13 were exceptions. Figure 2 provides a breakdown of the remaining 125 findings (potential problems and discrepancies). The left-hand pie chart shows that almost two-thirds of the findings (66 percent) are minor drawing or document discrepancies. For these, either an electronic database did not contain some information, or inconsistencies were found to exist between drawings. The lower right-hand pie chart provides an additional perspective. This pie chart shows that these drawing/document findings represent much less than one percent of the total number of individual drawing and document details that were reviewed.

The upper right-hand pie chart pertains to the remaining findings, identified as "attribute findings." These were either problems or discrepancies where some attribute of a reviewed instrument loop was found to deviate from the requirements of RG 1.97. These attribute findings were much less than one percent of the total number of opportunities for such findings; i.e., the total number of components reviewed times the number of RG 1.97 attributes reviewed for each component.

This low rate of finding identification is generally to be expected for a review effort of this depth. Nevertheless, even a low rate would have been unacceptable if any of the individual findings had impacted the safe operation of the plant. Individual examination of the RG 1.97 Review Project findings concluded that none were safety significant.

The findings generally fell into four areas: electrical isolation (power and signal), electrical separation, environmental qualification, and seismic qualification. Each of these areas is addressed in the following section.

3.2 Description of Principal Finding Areas

3.2.1 Electrical Isolation

Two general finding areas were identified with respect to electrical isolation, as follows:

A. Lack of power isolation between instruments of different classification supplied from a common 1E power source.

PG&E identified the potential for inadequate electrical isolation in certain 120 VAC power circuits for PG&E Class IA and IB1 instrumentation. (See Appendix A for a brief description of PG&E's instrument classification system.)

This condition affects only the power circuits, and not the process or position signals or their associated wiring. Inadequate isolation could prevent the instrumentation from meeting the single failure criterion. PG&E is committed to the single failure criterion for Class IA and IB1 instrumentation in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standards 279-1971[21] and 308-1971[22].

Evaluation concluded that the electrical isolation for each of the reviewed affected circuits is acceptable. Therefore, there is no specific known deviation from the design basis single failure criterion. However, agreement was reached at a September 25, 1992, meeting between PG&E and NRC Region V representatives that PG&E would provide a report containing a detailed description of this issue, including PG&E's plan for performing additional reviews. That report was provided by PG&E Letter DCL-93-078[24].

As stated above, PG&E has found no specific instances where single failure tolerance could not be demonstrated for Class IA or IB1 instrumentation. Based on experience with the analyses performed to date, PG&E expects that the additional breaker reviews described in Letter DCL-93-078, which are in progress, will continue to demonstrate adequacy in this area. The additional breaker reviews are separate from the RG 1.97 Review Project. PG&E intends to submit a final report to close this 120 VAC power isolation issue by December 31, 1994.

B. Lack of documented analysis of ability of devices to function as signal isolation devices.

In some instances, documentation that establishes the capability of I&C signal isolation devices was not readily retrievable in one location. PG&E is completing (currently in process of review and comment) a calculation file that documents the test and/or analyses that have been performed to substantiate the capability of I&C isolators.

In addition, one switch was identified as being used as an isolator. The application was reviewed and found to be acceptable, as described in Enclosure 4 of PG&E Letter DCL-93-040[23].

Thus, the identified devices were determined to be acceptable isolation devices. Therefore, no operability issue existed and this issue was not safety significant.

Note also that switches and relays used as isolation devices were discussed during the NRC's 1988 inspection of RG 1.97 implementation at DCPP, and they found to be acceptable.

3.2.2 Electrical Separation

Separation requirements are prescribed only for Category 1 instruments in RG 1.97, Table 1 (under "Redundancy"). The RG 1.97 refers to RG 1.75[16], "Physical Independence of Electric Systems"; however, as previously mentioned, PG&E is committed to IEEE Standards 279-1971[21] and 308-1971[22]. (There are some specific instances documented in the DCPP FSAR Update where PG&E has committed to the RG 1.75 guidelines. In addition, PG&E has committed that new designs will conform with RG 1.75 guidelines where feasible.)

The overall design basis for electrical separation at DCPP formally resides in DCM T-19, "Electrical Separation and Isolation." Section 3.0 of that DCM provides a historical perspective of DCPP's design for separation and isolation. Fundamentally, DCPP was designed such that mutually redundant (see Definitions) Class 1E equipment, devices, and circuits are physically separated and electrically independent from each other to meet the single failure criterion.

Project reviews identified that two of four Category 1 reactor coolant temperature T_{COLD} wide range (WR) circuits share common Class II raceways between the Hagan racks and the plant computer (see Figure 3). An engineering evaluation, which considered credible failure modes and effects, was performed. No credible failure mechanisms were identified that will degrade both circuits based on accident scenarios for which RG 1.97 instrumentation is required. The issue of Class II cables used in these loops is enveloped by this evaluation. The evaluation concluded that the single failure criterion is satisfied. (The formal evaluation is in the process of final review and approval and will be retained as part of the project files.)

Based on the evaluation described above, it was concluded that T_{COLD} WR, which comprises recorders TR-413, 423, 433 and 443, satisfied RG 1.97 Category 1 requirements and presented no operability issue. Analysis showed that under the worst-case scenario, TS requirements were satisfied with at least one recorder operable, thus confirming that this configuration was not safety significant.

(Note: Eagle 21 implementation (on schedule for 1R6/2R6) will provide signal isolation before the circuits leave the Hagan racks en route to the plant computer. Thus, Eagle 21 will resolve this issue and obviate the engineering evaluation discussed above.)

3.2.3 Environmental Qualification

Table 1 in RG 1.97 prescribes EQ requirements (under the heading "Equipment Qualification") for Category 1 and Category 2 instruments. The requirement is the same for both categories:

> "The instrumentation should be qualified in accordance with Regulatory Guide 1.89, 'Qualification of Class 1E Equipment for Nuclear Power Plants,' and the methodology described in NUREG-0588, 'Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.' ... Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. If the instrumentation channel signal is to be used in a computer-based display, recording, or diagnostic program, qualification applies from the sensor up to and including the channel isolation device."

Two general finding areas were identified with respect to EQ, as follows:

A. Lack of documented analysis to support EQ (see Figure 4).

The majority of findings related to EQ involved lack of documented analysis to substantiate qualification. Affected equipment included terminal blocks and cable. Upon identification, the affected components were determined to be operable and qualifiable, and EQ was subsequently formally documented. For example, the terminal block finding involved qualification for a harsh radiation environment only; review confirmed that the terminal blocks would remain functional and the EQ files were updated to document qualification.

B. Equipment not qualified for accident environments (see Figure 5).

Two findings regarding EQ were identified in which qualification could not be documented (diodes installed as surge suppressors on CIV solenoid valve circuits, and solenoids installed below the post-accident flood level). Further reviews indicated that qualification for these devices could not be established. For both of these findings, an engineering evaluation, which considered credible failure modes and effects, was performed. The evaluation considered how the failures of these devices would affect the ability of the associated systems to perform their respective safety functions. The evaluation determined that a failure of the individual devices would not prevent the systems from being capable of accomplishing their required safety functions. The results of the evaluation was formally documented and will be retained as part of the project files.

[Note: As part of the review of diodes installed on CIV solenoid circuits, a broader review was performed of solenoid valve voltage surge suppressor diodes installed in harsh environmental locations. This review identified other such diodes installed on pilot solenoid valve circuits which serve heating, ventilating and air conditioning (HVAC) dampers, component cooling water system flow control valves, and the main feedwater bypass valves. (The HVAC diode application was determined to be reportable in accordance with 10 CFR 50.73; see LER 2-92-005-01[18].) However, because these circuit configurations were beyond the scope of the RG 1.97 Review Project, PG&E is tracking the additional reviews and corrective actions related to diodes installed in harsh environments by separate NCRs, Nos. DCO-92-EN-N011 and DCO-92-EN-N030. (NCR DCO-92-EN-N011 has been closed.)]

3.2.4 Seismic Qualification

Table 1 in RG 1.97 prescribes SQ requirements (also under the heading "Equipment Qualification") only for Category 1 instruments, as follows:

> "The seismic portion of the qualification should be in accordance with Regulatory Guide 1.100, 'Seismic Qualification of Electric Equipment for Nuclear Power Plants.' Instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake."

Findings related to SQ involved lack of documented analysis to substantiate the qualification (see Figure 6). The components involved were determined to be qualifiable and documentation was subsequently established to support qualification. The components involved included monitor light boxes, relays and relay cabinets, cable trays, junction boxes, and transmitters.

In the case of transmitters, component seismic reviews found that the Required Response Spectra for a number of these devices did not completely envelop the seismic response spectra for the panel in which they were located. Nevertheless, the transmitters were still seismically qualified because of excess margin available in the qualification testing of these devices. The appropriate SQ files were revised to clarify the response spectra for the transmitters located within the panels.

4.0 PROGRAM CLOSURE

RG 1.97 Review Project closure activities comprised the following:

4.1 Completion of the Project Scope

The TS variable and RL reviews have been completed. The resulting review packages are in the process of final review and approval. Not all actions resulting from the project reviews have been completed. However,

project finding evaluation activities have been completed such that a "resolution path" has been established for each project finding.

4.2 Closure of Other Identified Items

The following additional items were included as part of the RG 1.97 Review Project scope:

4.2.1 Closure of DCM T-34 Open Items

The project has expedited the closure of the open items identified by DCM T-34, "Post Accident Monitoring Instrumentation." Remaining actions are tracked via ARs in PIMS (there is a separate AR assigned to each remaining open item).

4.2.2 Close Outstanding Actions from Initial NCR Investigations

Appendix A provides historical background information on the RG 1.97 Review Project, including the initial NCR investigations. The only outstanding action from those investigations is the upgrading of the white light monitoring system for CIV position indication. Enclosure 8 of PG&E's letter DCL-93-284[26] provides an overview of the design modification that PG&E is preparing for this system. Implementation of the modification remains scheduled for the seventh refueling outages for both DCPP Units (1R7 and 2R7), as discussed at the December 18, 1991, meeting between PG&E and NRR personnel.

Note: Appendix C provides the status of the completed and planned RG 1.97 related modifications to which PG&E has committed. No changes have been made to the previously stated schedules for these modifications.

4.2.3 Resolve Independent Review Comments

As discussed in Appendix A, the RG 1.97 Review Project was subjected to an independent review. The report of that review contained favorable conclusions, and made various recommendations. The project resolved these recommendations. Outstanding actions are being tracked by ARs/AEs in PIMS.

[Note: PG&E's interim RG 1.97 project report[25] indicated that this final report would delineate the independent review recommendations and PG&E's associated responses. That information comprises nearly 20 pages, and, for the sake of brevity, is not reproduced herein. The information is available for review in PG&E's offices, along with the rest of the project documentation.]

The project was also subjected to a recent independent project closure review. The results of that recent review were also favorable.

4.3 Configuration Management Expansion

As discussed in PG&E Letter DCL-93-078[24] to the NRC, PG&E is making enhancements to the Configuration Management Program for DCPP. These enhancements will assure that components with certain attributes (e.g., seismic qualification) that would not normally be required of the device by virtue of its functional classification will be recognized and maintained throughout future procurement, maintenance, and modification activities.

As the above improvements were being implemented, PG&E identified another enhancement related to electrical device configuration management. The RG 1.97 Review Project identified certain Class 1E electrical devices in panels (e.g., relays, switches, terminal blocks) that were not specifically listed in the plant's computer databases. For each identified device, the required RG 1.97 attributes were confirmed to be adequate. The new enhancement involves the identification and documentation of these devices for purposes of maintaining required attributes and anticipating future procurement, maintenance, and modification.

5.0 CONCLUSIONS

The RG 1.97 Review Project has validated the design basis for RG 1.97 implementation at DCPP. The findings identified by the project were thoroughly reviewed for safety significance, operability, and reportability. The overall number of findings identified by the project is considered very low when compared to the total number of individual components and attributes that were reviewed, and none of the findings was considered to be safety significant. One issue was determined to be formally reportable (HVAC diodes; see Section 3.2.3.B), and PG&E submitted a report on one other issue (120 VAC power isolation; see Section 3.2.1.A).

Based on the above (i.e., the overall low number of findings that were identified compared to the total number of components and attributes reviewed, and the determination that no findings were safety significant), PG&E concludes that the health and safety of the public was not affected by these findings.

6.0 DEFINITIONS

6.1 95/5 Criterion -- The criterion satisfied by the RG 1.97 Review Project random loop reviews. The review methodology provided a 95 percent probability that no more than 5 percent of the entire RG 1.97 instrument loop population could have a "potential problem" (see definition in Section 6.2 below) for which a hardware design change would be required to resolve. Because the TS variable reviews were in addition to the random sampling program, the overall confidence level provided by the project was greater than 95/5.

6.2 Finding -- A finding was a discrete instance where some aspect of the design or qualification of existing plant hardware was found to deviate from, or not clearly conform with, the requirements of RG 1.97, Revision 3. Findings were also initiated for drawing/documentation inconsistencies identified during project reviews even though no aspect

of conformance with RG 1.97 was in question. Findings were documented in accordance with project instructions, and they were categorized as follows:

- Potential Problem: Relatively more significant findings that might potentially have affected plant operability and might have required a hardware design change to resolve. This was the most significant category of findings.

- Discrepancy: Findings in which incorrect, incomplete, or conflicting information was identified (but for which no hardware design change was expected to be required).

- Exception: Instances in which a variable did not meet the requirements of RG 1.97, and PG&E took exception to the requirement. Several DCPP-specific exceptions to the RG had already been accepted by the NRC. New exceptions, along with their associated technical bases, were submitted to the NRC by PG&E Letters DCL-93-040[23] and DCL-93-284[26].

- Clarification: Clarifications were used to document acceptable configurations that were not intuitively obvious; for example, when the results of a variable evaluation might not be readily apparent to an independent reviewer.

6.3 Loop/Variable -- A variable is a process function that is monitored by one or more instrument loops. For example, pressurizer level is a variable monitored by three redundant instrument loops. The instrument loops may contain instrumentation that provides protection functions, control functions, and/or indication and alarm functions.

6.4 Mutually Redundant -- A piece of equipment or a system that duplicates the identical function of another piece of equipment or a system to the extent that either may perform the required function regardless of the state of operation or failure of the other.

6.5 Operability Evaluation (OE) -- PG&E's follow-up process of confirming the conclusion of a POA (see Section 6.6 below), formally documenting the basis for the conclusion, and obtaining management review and approval.

6.6 Prompt Operability Assessment (POA) -- PG&E's process of expeditiously determining and documenting whether a safety-related system, structure, or component (SSC) is operable or inoperable. (POAs are also initiated for nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the required functions of safety-related SSCs, or that are relied on in the safety analyses that are part of DCPP's current licensing basis.) PG&E's procedures require that the POA process shall be initiated immediately upon identification of a degraded condition that may adversely impact the operability of a SSC.

7.0 REFERENCES

1. NRC Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, dated May 1983

2. PG&E (Mr. J.O. Schuyler) letter to the NRC (Mr. D.G. Eisenhut) dated September 9, 1983, in ACTS File 1119, re: "Regulatory Guide 1.97 (Revision 3) Report"

3. PG&E (Mr. D.A. Brand) Letter DCL-85-024 to the NRC (Mr. G.W. Knighton) dated January 25, 1985, in ACTS File 1119, re: "Regulatory Guide 1.97 Compliance"

4. American National Standards Institute (ANSI)/American Nuclear Society (ANS) Standard 4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors"

5. Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," issued by the NRC as Generic Letter 82-33, dated December 17, 1982

6. PG&E (Mr. J.O. Schuyler) letter to the NRC (Mr. D.G. Eisenhut) dated April 18, 1983, re: "Generic Letter No. 82-33"

7. EG&G Idaho Falls, Inc., report, "Conformance to Regulatory Guide 1.97, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2," dated May 1984, transmitted to PG&E (Mr. J.O. Schuyler) by NRC (Mr. G.W. Knighton) letter dated June 27, 1984, at RLOC 01596-2887

8. PG&E (Mr. D.A. Brand) Letter DCL-84-298 to the NRC (Mr. G.W. Knighton) dated September 4, 1984, re: "Compliance with Regulatory Guide 1.97, Revision 3"

9. Supplement No. 31 to NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2" (SSER 31), dated April 1985

10. NRC (Mr. R.J. Pate) letter to PG&E (Mr. J.D. Shiffer) dated April 27, 1988, re: "NRC Inspection of Diablo Canyon Units 1 and 2" (transmitted one Notice of Violation, one Notice of Deviation, and NRC Inspection Report Nos. 50-275/88-02 and 50-323/88-02)

11. PG&E (Mr. J.D. Shiffer) Letter DCL-88-141 to the NRC dated May 27, 1988, re: "Reply to Notice of Violation and Notice of Deviation, and Three Unresolved Items Regarding Regulatory Guide 1.97 Implementation, in NRC Inspection Report 50-275/88-02 and 50-323/88-02"

12. NUREG-0737, "Clarification of TMI Action Plan Requirements," dated October 31, 1980

13. PG&E (Mr. J.D. Shiffer) Letter DCL-91-258 to the NRC dated October 25, 1991, re: "Additional Information Regarding Unresolved Item 88-02-01 in NRC Inspection Report 50-275/88-02 and 50-323/88-02"

14. PG&E (Mr. G.M. Rueger) Letter DCL-92-031 to the NRC dated February 7, 1992, re: "Resolution of Inspection Items in NRC Inspection Report Nos. 50-275/88-02 and 50-323/88-02"

15. DCPP Technical Specifications (Appendix A of DCPP Operating Licenses DPR-80 and DPR-82), through Amendments 84 (Unit 1) and 83 (Unit 2).

16. NRC Regulatory Guide 1.75, "Physical Independence of Electric Systems," Revision 2, dated September 1978

17. DCPP Final Safety Analysis Report (FSAR) Update, Revision 8, dated September 1992

18. DCPP Licensee Event Report (LER) 2-92-005-01, "Use of Environmentally Unqualified Surge Suppression Diodes in the Auxiliary Building Ventilation System Due to Personnel Error," transmitted to the NRC (Document Control Desk) by PG&E (Mr. G.M. Rueger) Letter DCL-92-163 dated July 21, 1992

19. NRC (Mr. H. Rood) letter to PG&E (Mr. G.M. Rueger) dated April 17, 1992, re: "Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M80245 and M80246)"

20. PG&E (Mr. J.D. Shiffer) Letter DCL-91-053 to the NRC (Document Control Desk) dated March 18, 1991, re: "License Amendment Request 91-01, Revision of Technical Specification 4.6.2.3 -- Revise Requirement to Verify Containment Fan Cooler Unit Dampers Transfer from the Normal to the Accident Position"

21. Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"

22. IEEE Standard 308-1971, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"

23. PG&E (Mr. G.M. Rueger) Letter DCL-93-040 to the NRC dated February 17, 1993, re: "Bases for Exceptions Identified by PG&E's Regulatory Guide 1.97 Review Project"

24. PG&E (Mr. G.M. Rueger) Letter DCL-93-078 to the NRC dated April 6, 1993, re: "Report of Potential for Inadequate Electrical Isolation in Certain 120 VAC Power Circuits for PG&E Class IA and IB1 Instrumentation"

25. PG&E (Mr. G.M. Rueger) Letter DCL-93-122 to the NRC dated May 18, 1993, re: "Regulatory Guide 1.97 Review Project Interim Report"

26. PG&E (Mr. G.M. Rueger) Letter DCL-93-284 to the NRC dated December 17, 1993, re: "Final List of RG 1.97 Exceptions and Bases Identified by PG&E's Regulatory Guide 1.97 Review Project"

Table 1

Accident Monitoring Instrumentation Variables from TS Table 3.3-10

1. Containment Pressure (normal range)
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)
4. Reactor Coolant Pressure - Wide Range
5. Pressurizer Water Level
6. Steam Line Pressure
7. Steam Generator Water Level - Narrow Range
8. Refueling Water Storage Tank Water Level
9. Containment Reactor Cavity Sump Level - Wide Range
10. Containment Recirculation Sump Level - Narrow Range
11. Auxiliary Feedwater Flow Rate
12. Reactor Coolant System Subcooling Margin Monitor
13. PORV Position Indicator
14. PORV Block Valve Position Indicator
15. Safety Valve Position Indicator
16. In-Core Thermocouples
17. Main Steam Line Radiation Monitor
18. Containment Area Radiation Monitor - High Range
19. Plant Vent Radiation Monitor - High Range
20. Reactor Vessel Level Indication System

Table 2

RG 1.97 Review Project "Random Loops"

LOOP NO.	UNIT	SYSTEM	DESCRIPTION OF LOOP
RL01	1	RCS	IN CORE THERMOCOUPLES TRAIN A
RL02	1	RCS	IN CORE THERMOCOUPLES TRAIN B
RL03	1	MN STM	SG 1-1 NARROW RANGE LEVEL
RL04	1	CNT SP	CONTAINMENT PRESSURE WIDE RANGE
RL05	1	HVAC	PLANT VENT HIGH RADIATION GROSS GAMMA
RL06	1	SI	ACCUMULATOR 1-1 WIDE RANGE LEVEL
RL07	1	SI	ACCUMULATOR 1-1 PRESSURE
RL08	1	SI	SI PUMP 1-2 DISCHARGE FLOW
RL09	1	EE	RCP 1-1 MOTOR CURRENT
RL10	1	MN STM	STEAM GENERATOR 1-1 PRESSURE
RL11	1	MN STM	STEAM GENERATOR 1-2 PRESSURE
RL12	1	FW	STEAM GENERATOR 1-4 FEEDWATER FLOW
RL13	1	FW	STEAM GENERATOR 1-4 FEEDWATER FLOW
RL14	1	CNT SP	CONTAINMENT SPRAY PUMP 1-2 FLOW
RL15	1	GWS	GAS DECAY TANK 1-3 PRESSURE
RL16	1	HVAC	EL 100/115' CORRIDOR SUPPLY DAMPER POS
RL17	1	HVAC	SAFEGUARDS BYPASS DAMPER POSITION IND.
RL18	1	HVAC	EXH FAN SUCTION CROSS-TIE DAMPER POS IND
RL19	1	HVAC	PRESSURIZATION FAN DISCHARGE DAMPER POS
RL20	1	HVAC	SUPPLY FAN ISOLATION DAMPER POS
RL21	1	HVAC	SUPPLY FAN ISOLATION DAMPER POS
RL22	1	EE	BATTERY 1-1 VOLTAGE
RL23	1	EE	BATTERY 1-3 VOLTAGE
RL24	1	EE	DG BUS G CURRENT
RL25	1	HVAC	NSSS SAMPLING ROOM AREA RAD MONITOR
RL26	1	HVAC	AUX BLDG CONTROL BOARD AREA RAD MONITOR
RL27	1	MISC	WIND SPEED AT 10 METERS
RL28	2	MN STM	SG 2-1 WIDE RANGE LEVEL
RL29	2	MN STM	SG 2-2 WIDE RANGE LEVEL
RL30	2	RCS	RCS COLD LEG LOOP 2-2 TEMPERATURE

Table 2 (Cont.)

RG 1.97 Review Project "Random Loops"

LOOP NO.	UNIT	SYSTEM	DESCRIPTION OF LOOP
RL31	2	RCS	RCS COLD LEG LOOP 2-4 TEMPERATURE
RL32	2	CNT SP	CONTAINMENT PRESSURE NORMAL RANGE
RL33	2	MN STM	SG BLOWDOWN TANK VENT RAD MONITOR
RL34	2	RHR	RHR HX 2-2 OUTLET TEMPERATURE
RL35	2	SI	ACCUMULATOR 2-2 TANK PRESSURE
RL36	2	SI	ACCUMULATOR 2-2 TANK PRESSURE
RL37	2	RCS	PRESSURIZER LEVEL
RL38	2	EE	PRESSURIZER HEATER BACKUP GROUP 3*
RL39	2	RCS	PRESSURIZER RELIEF TANK TEMPERATURE
RL40	2	MN STM	STEAM GENERATOR 2-2 PRESSURE
RL41	2	MN STM	STEAM GENERATOR 2-2 FLOW
RL42	2	MN STM	STEAM GENERATOR 2-3 FLOW
RL43	2	CNT SP	CONTAINMENT SPRAY PUMP 2-1 FLOW
RL44	2	SI	CONTAINMENT RECIRC SUMP TEMPERATURE
RL45	2	HVAC	LHUT AREA SUPPLY DAMPER POS INDICATION
RL46	2	HVAC	LHUT AREA SUPPLY DAMPER POS INDICATION
RL47	2	HVAC	EXH FAN FILTER INLET DAMPER POS IND.
RL48	2	HVAC	CHG PUMP ROOM VENT EXH DAMPER POS IND.
RL49	2	HVAC	CCW PUMP ROOM VENT EXH DAMPER POS IND.
RL50	2	HVAC	EXH FAN FILTER INLET DAMPER POS IND.
RL51	2	HVAC	CRPS PRESSURIZATION FAN DISCH DAMPER POS
RL52	2	HVAC	BOOSTER FAN SUCTION DAMPER POS IND.
RL53	2	EE	BATTERY 2-3 VOLTAGE
RL54	2	EE	BATTERY 2-2 CURRENT
RL55	2	HVAC	NSSS SAMPLING ROOM AREA RAD MONITOR
RL56	2	MN STM	MAIN STEAM LINE 2-3 RAD MONITOR
RL57	2	HVAC	PLANT VENT PARTICULATE RAD MONITOR
RL58	2	MISC	PLANT/ENVIRONS RAD (PHOTON) - PORT INSTS
RL59	2	MISC	WIND DIRECTION AT 10 METERS
RL60	2	MISC	RCS & MISC SUMPS GRAB SAMPLE - GROSS ACT

* Changed from Backup Group 4 to Backup Group 3 because the latter was determined to be the equipment requiring RG 1.97 indication capability.

PG&E Reg. Guide 1.97 Review Process

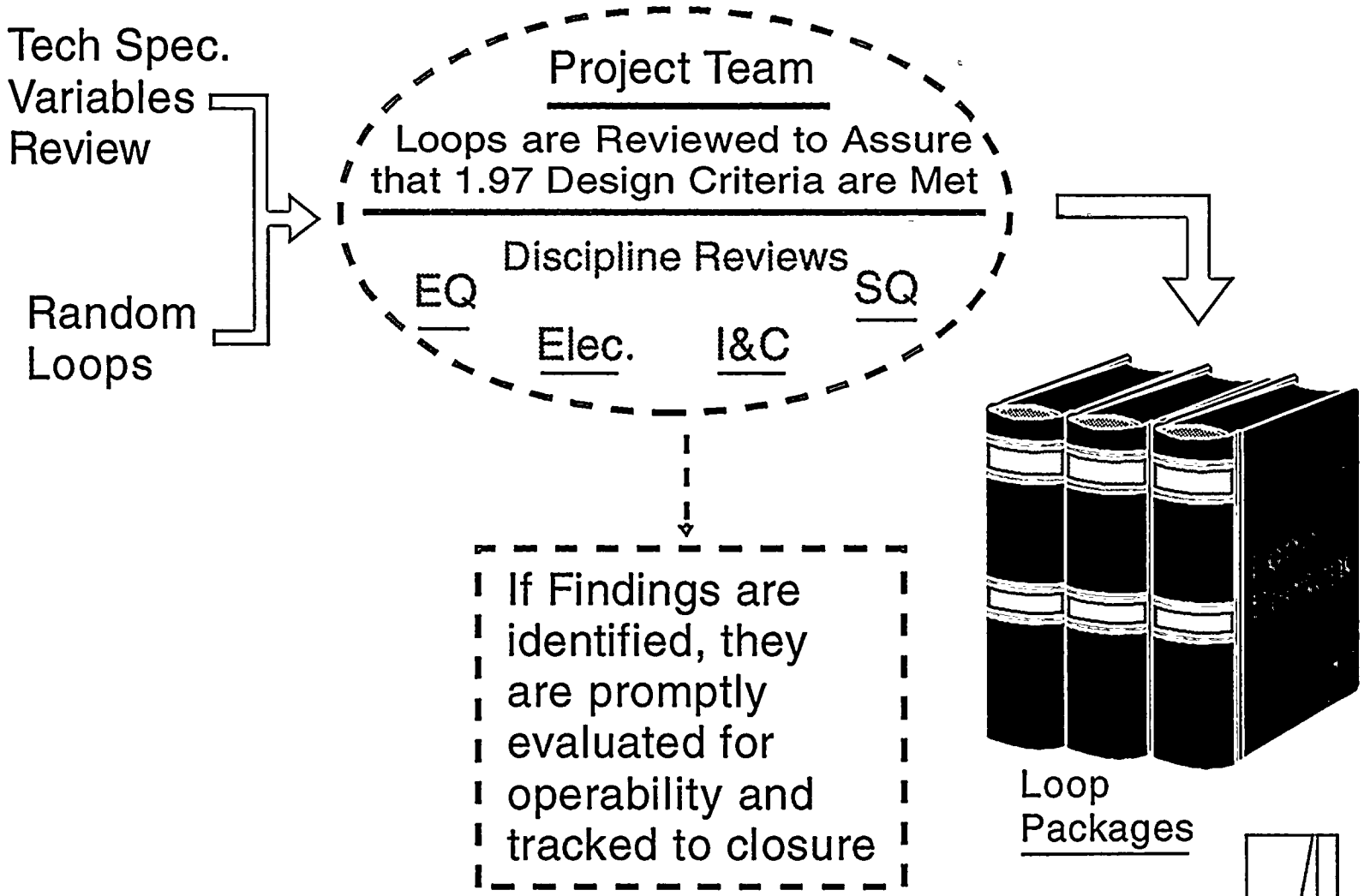

Figure 1

Summary of Findings from 1.97 Review

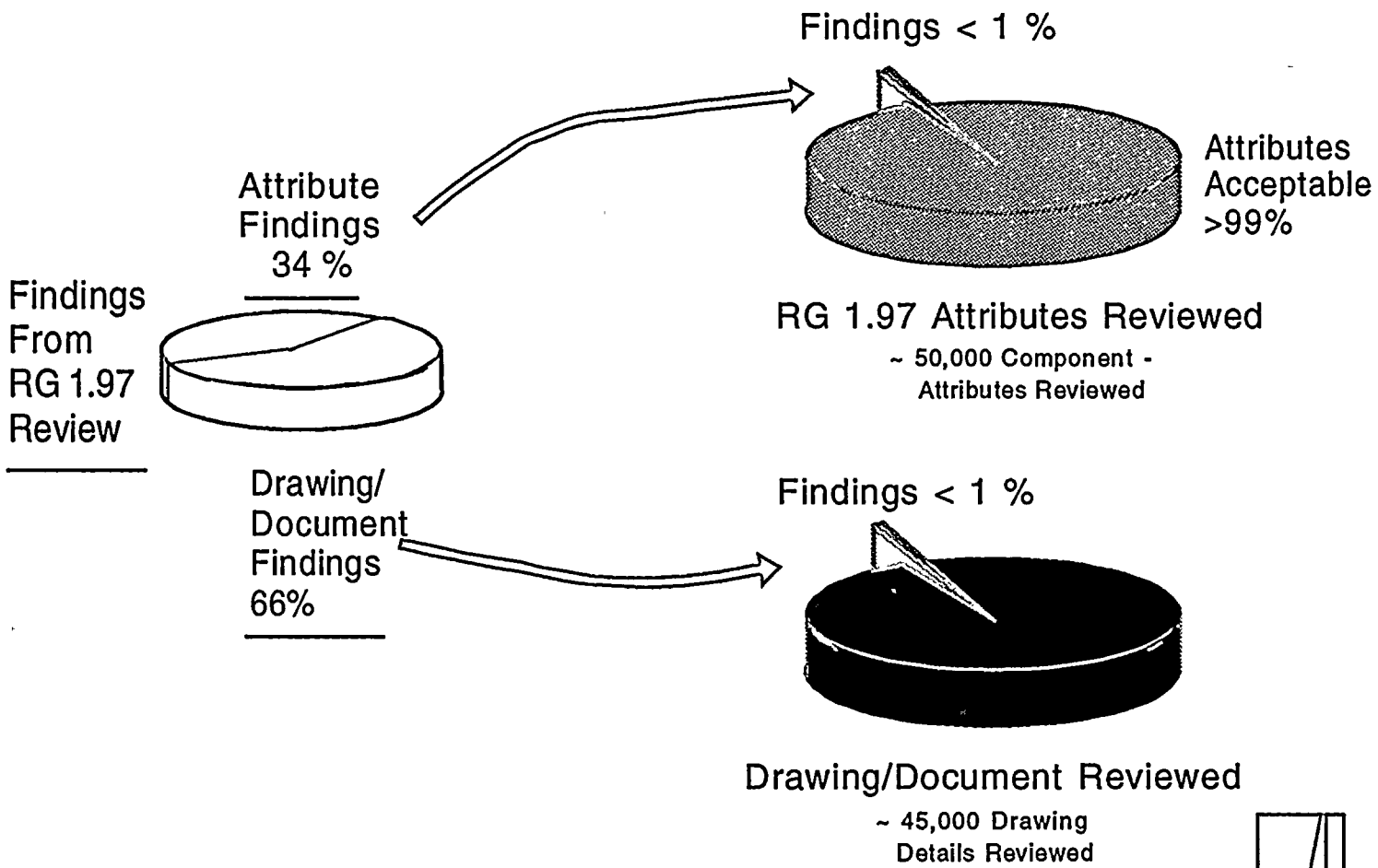

Figure 2

Separation Example Finding

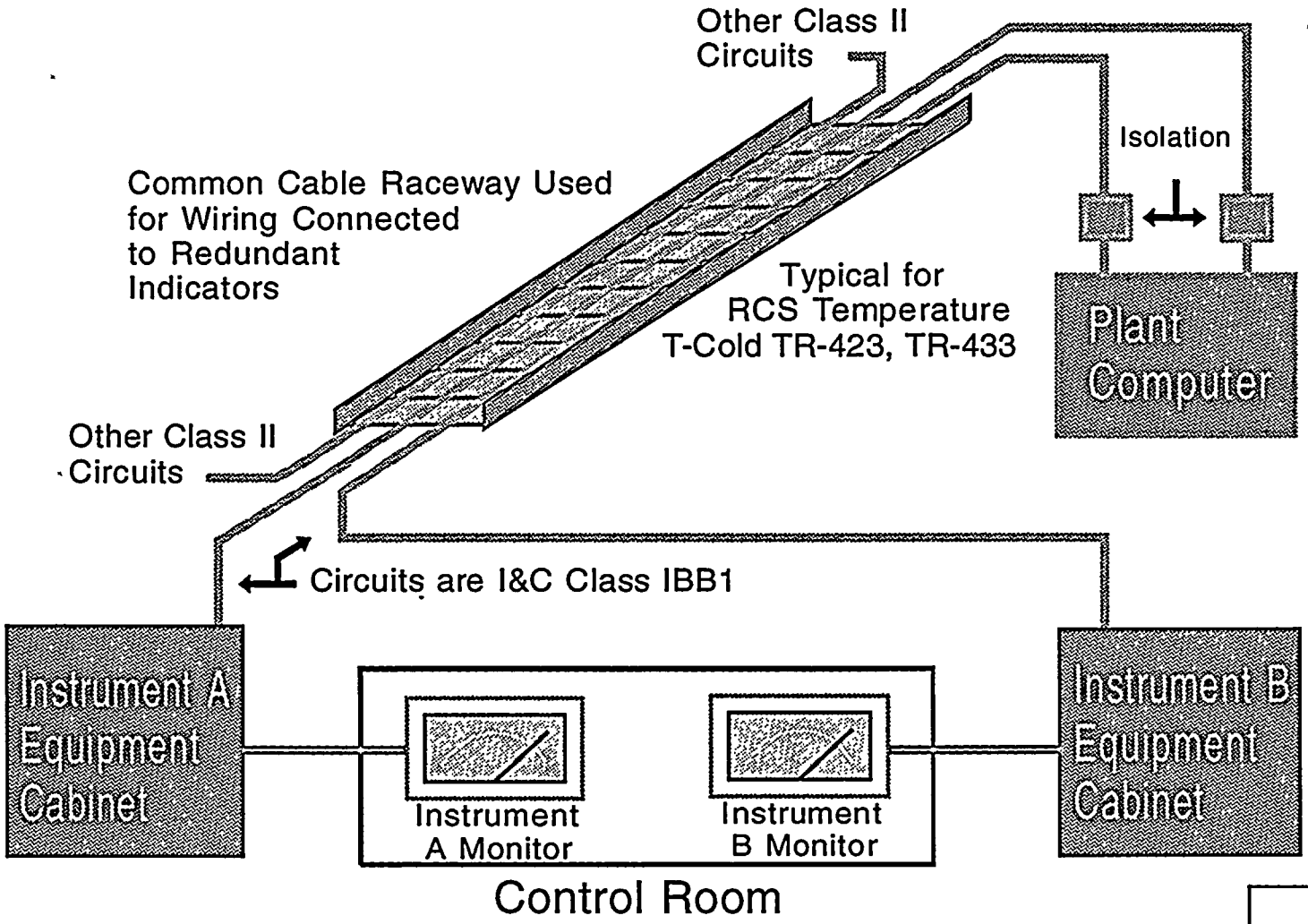

Figure 3

Environmental Qualification Findings

Finding

Documentation of EQ Not Established in Existing EQ Files, for the following types of equipment: (Typical Examples)

- o Cable
- o Terminal Blocks (Outside Containment)

Immediate Action:

Where required, prompt operability assessments were completed to document continued operability of installed equipment.....

Finding Resolution

Revise or Create EQ Files as required to document installed configuration.

Figure 4

Environmental Qualification Findings

Finding

Documentation of EQ Not Established in Existing EQ Files, for the following equipment:
(Typical Examples)

- o Diodes
- o Solenoid Valves

Immediate Action:

Prompt operability assessments were completed to document continued operability of installed equipment.....

Finding Resolution

Prepare a Failures Modes and Effects Analysis to justify system operability after the subject device becomes inoperable due to environmental DBE exposure.

FMEA

Figure 5

Seismic Qualification Issues

Finding

Documentation of SQ Not Established in Existing SQ Files, for the following types of equipment: (Typical Examples)

- o Relay Cabinet
- o Position Switches
- o Junction Box

Immediate Action:

Where required, prompt operability assessments were completed to document continued operability of installed equipment.....

Finding Resolution

Revise or Create SQ Files as required to document installed configuration.

Figure 6

Appendix A

Historical Perspective of RG 1.97 Implementation

Regulatory Guide 1.97

NRC Regulatory Guide (RG) 1.97[1] is titled "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Fundamentally, the RG endorses American National Standards Institute (ANSI)/American Nuclear Society (ANS) Standard 4.5-1980[4], "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors," subject to an extensive listing of clarifications. Most significantly, Table 3 of the RG provides a detailed listing of "variables" (e.g., steam generator level, containment pressure, wind direction) that must be monitorable during and after an accident at a pressurized-water reactor. The variables are segregated by "type" as follows:

Type A -- Those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events. Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures. A variable included as Type A does not preclude it from being included as Type B, C, D, or E, or vice versa.

Type B -- Those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are: (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control).

Type C -- Those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Type D -- Those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Type E -- Those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

For the variables in each type, the table in the RG gives the required range to be measured, the reason for inclusion of the variable, and the "category" of the variable.

The "category" (1, 2 or 3) assigned to each variable in Table 3 of the RG defines the extent of "design and qualification criteria" that apply to the variable. Table 1 in the RG delineates these criteria in detail for each category, in several areas (termed "attributes" in this report). These attributes include environmental and seismic qualification, redundancy, power source, instrument range, display and recording, and human factors. The three categories are as follows:

Category 1 - Imposes the most stringent requirements and applies to key variables.

Category 2 - Provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status.

Category 3 - Is intended to provide requirements that will ensure that high-quality off-the-shelf instrumentation is obtained, and applies to backup and diagnostic instrumentation.

PG&E's Instrument Classification System for DCPP

PG&E's instrument classification system is defined in Design Criteria Memorandum (DCM) T-24. Briefly, Instrument Class IA devices are those directly required to initiate safe shutdown of the reactor, mitigate the consequences of an accident, or prevent exceeding 10CFR100 offsite dose limits. In general, any instrument that automatically performs or allows operating personnel to manually perform a function that is necessary to complete a safeguards action is classified as Instrument Class IA.

Instrument Class IB devices are those that provide post-accident monitoring (PAM) functions in accordance with RG 1.97. This Class is then further segregated based on type and category as defined in the RG.

Instrument Class IC devices are those instruments that are attached to a PG&E Design Class I pressure boundary, are not classified as IA or IB, and have no safety function other than to maintain pressure boundary integrity.

Instrument Class ID is a new classification that is being applied to selected components having attributes (e.g., seismic qualification) that would not normally be required of the device by virtue of its functional classification. The Class ID assignment assures that these attributes will be recognized and maintained throughout future procurement, maintenance, and modification activities.

Instrument Class II is non-safety-related.

PG&E Implementation of RG 1.97

Compliance with RG 1.97, Revision 2, was prescribed by Supplement 1 of NUREG-0737 (Section 6), issued by the NRC in December 1982 as Generic Letter (GL) No. 82-33[5]. (NUREG-0737[12] was "Clarification of TMI Action Plan Requirements" -- the compilation of specific actions identified after the Three Mile Island accident that were approved by the NRC Commissioners for implementation.) The NRC sponsored a series of workshops to explain the GL and answer questions; PG&E representatives attended the workshop held in San Francisco on March 1, 1983. PG&E provided schedules for implementing the GL in a letter dated April 18, 1983[6], and provided the RG 1.97 "summary of compliance" table (required by the GL, Section 6.2) for DCPP Unit 1 in a letter dated September 9, 1983[2]. (This letter also contained the commitment to Revision 3 of the RG.)

Based on these letters, EG&G Idaho, Inc., under contract to the NRC, evaluated PG&E's conformance to the RG. Their report[7] found five deviations from the requirements of the RG to be acceptable and requested additional information. PG&E provided the requested information in a letter dated September 4, 1984[8]. That letter also identified the RG 1.97 Type A variables for DCPP and updated the Unit 1 "summary of compliance" table.

PG&E provided the NRC with a "summary of compliance" table for DCPP Unit 2 in a letter dated January 25, 1985[3].

In April 1985 the NRC published Supplement No. 31[9] to the Safety Evaluation Report (SSER 31) for DCPP. The SSER included as its Appendix B a Technical Evaluation Report (TER) prepared by EG&G Idaho, Inc., addressing conformance to RG 1.97. The SSER concludes (Section 4.5.4, p. 4-7) that: "Based on the staff's review of the PG&E submittals and the attached Technical Evaluation Report (Appendix B), the staff finds that the design of the Diablo Canyon Power Plant, Unit 1 and 2, is acceptable with respect to conformance with the guidelines of Regulatory Guide 1.97, Rev. 3."

The TER documented four additional acceptable deviations from the requirements of RG 1.97 (for a total of nine).

1988 NRC Inspection of RG 1.97 Implementation

During February-March 1988, the NRC conducted a detailed inspection of RG 1.97 implementation at DCPP. The Inspection Report[10] documented one Notice of Violation (NOV) (instruments past due for calibration), one Notice of Deviation (all wide-range [WR] steam generator [SG] level instruments powered from the same power source), and three unresolved items (use of resistor networks not proven by test for signal isolation, lack of a recorder for neutron flux, and inadequate tracking and documentation of calibration for the status of standby power instrumentation).

The Inspection Report also identified and documented the acceptance by NRC of four additional DCPP-specific deviations from RG 1.97 criteria (for a cumulative total of 13; these are all tabulated in Enclosure 1 of PG&E Letter DCL-93-040 to the NRC dated February 17, 1993[23]).

PG&E responded to the Inspection Report by letter dated May 27, 1988[11], which included the following information:

(1) The response acknowledged the instruments out-of-calibration NOV and the unresolved item regarding status of standby power, and committed to corrective actions to ensure that both the I&C and the Electrical Maintenance organizations will complete future calibrations within the required period.

(2) The response defended the lack of redundancy in the power source to the SG WR instruments by stating that narrow-range (NR) SG level was the key variable for secondary heat sink monitoring, that auxiliary feedwater (AFW) flow becomes the key variable if SG level drops below the range of the NR instruments, and that, to meet the redundancy requirement for secondary heat sink monitoring, the WR SG level was considered as a diverse variable to the AFW flow. The power supplies for these three variables, when considered in aggregate, were determined to satisfy the redundancy requirements prescribed by the RG.

To support and formalize this interpretation, PG&E committed to revise the applicable table in the DCPP FSAR Update[17] (Table 7.5-6) to include all three of these variables (as Category 1) with a note to explain the relationship between the three and how the redundancy requirements were satisfied. (Note: NR SG level is not listed in RG 1.97, Table 3, and had thus not previously been identified in any of PG&E's RG 1.97 submittals to the NRC.)

(3) The response regarding the use of resistor networks for signal isolation included an analysis to demonstrate "that failures within the Class II portion of any of the subject loops will not prevent the Class IB portion of the loop from performing its safety function." Further, because the networks comprise only simple linear components, "there are no complex failure mechanisms that require physical testing for evaluation."

(4) The response defended the acceptability of the lack of a recorder for neutron flux. This response was based on the observation that "appropriate safety features including rod insertion and boric acid addition would have taken place after an event and any subsequent changes in neutron flux level would be a slow process in which a panel meter would provide adequate indication."

In a subsequent letter to the NRC, dated October 25, 1991[13], PG&E committed (based on discussions with the NRC) to replacing the resistor isolators as part of the planned replacement of the existing Hagan Plant Protection System (PPS) with a Westinghouse "Eagle 21" PPS. Thus, "Following installation of the Eagle 21 PPS, isolation between the Class IB and Class II circuits will be accomplished by devices tested and qualified in accordance with the Class IB Category specified in RG 1.97 rather than by the resistor networks."

PG&E representatives met with the NRC on December 18, 1991 (see discussion below, under "Apprising the NRC"), to discuss RG 1.97 issues, which included three remaining outstanding issues from the 1988 NRC inspection. That meeting was followed by a letter, dated February 7, 1992[14], in which PG&E clarified and supplemented its responses for the three remaining inspection issues, as follows:

- PG&E clarified its letter dated October 25, 1991, concerning resistor networks and the replacement of the Hagan racks with the Westinghouse Eagle 21 PPS. The previous response was accurate for RG 1.97 instrument loops originating in the Hagan process protection racks. However, those Class IB Category 2 instrument loops that originate in the Hagan <u>control</u> racks are outside the scope of the Eagle 21 upgrade. PG&E committed to supplying qualified isolation devices for these loops. (Class IB Category 3 loops that originate in control racks do not require isolation and will not have isolation devices added.)

- PG&E committed to transferring the power source for two of the four WR SG level loops (SGs 3 and 4) to a different vital instrument AC power source (protection set III) to address the issue of the power source redundancy for this instrumentation. "While this approach is not in strict compliance with the specific RG 1.97 requirements for redundancy, isolation, and separation of redundant circuits, the intent of the regulatory guide is met."

- PG&E committed to install a seismically-qualified neutron flux recorder on panel PAM-1 in the control room.

Steam Generator Wide-Range Water Level Recorder Power NCR

Nonconformance Report (NCR) DCO-89-EN-N009 was initiated May 5, 1989, to document that the power source and cable for SG WR Level Recorders LR-517 and LR-537 were still Design Class II even though the signal side of this instrumentation loop had been upgraded to Category 1 requirements in conformance with RG 1.97. Justification for Continued Operation 89-21 was prepared to document the interim acceptability of continued operation. As a specific corrective action, Class 1E cables were added and the power source to the recorders was changed to a vital source fed from Channel IV. These design changes were implemented during 1R4 and 2R4.

Further, a broader review was initiated with respect to the use of black wiring (normally non-Class 1E) for RG 1.97 instrumentation. In a number of cases, black wiring was found, but was determined to be acceptable. However, as investigation for the NCR proceeded, additional inconsistencies with expected RG 1.97 design features were identified. This broader scope of RG 1.97 issues led PG&E management to initiate a second NCR (DCO-91-EN-N005) to track resolution of the initial wiring issues, and to complete the investigation and resolution of the other issues.

On December 11, 1991, the Technical Review Group for NCR DCO-89-EN-N009 documented that the action to correct the specific subject of the nonconformance (non-Class 1E cable and power supplies) was complete. (Generic considerations were transferred to the new NCR, DCO-91-EN-N005.)

Design Criteria Memorandum T-34, "Post Accident Monitoring Instrumentation"

DCM T-34 was under development concurrently with the investigations for both NCRs. [Note that a DCM is PG&E's equivalent to what many other nuclear plants call a "design basis document," or "DBD."] The development effort for the DCM identified some issues and helped clarify the various issues associated with PG&E's conformance with RG 1.97. The RG 1.97 Review Project helped expedite the closure of Open Items associated with DCM T-34 (see Section 4.2.1).

Initial Results of Investigations for NCR DCO-91-EN-N005

Preliminary investigation for the second NCR (DCO-91-EN-N005) revealed issues for four of the 72 RG 1.97 variables:

- Containment Isolation Valve Position -- Variable 15 (in DCM T-34)
- Accumulator Isolation Valve Position -- Variable 34 (DCM T-34)
- Containment Fan Cooler Performance -- Variable 53 (DCM T-34)
- Emergency Ventilation Damper Position Indication -- Variable 63 (DCM T-34)

The issues applied to a limited amount of RG 1.97 equipment, but did include lack of documented analysis and/or inadequate design in the areas of seismic qualification, environmental qualification, separation, isolation and human factors considerations. Safety analyses documented in the NCR and in associated Operability Evaluation (OE) 91-13 concluded that the monitoring systems would perform their specified function, or that alternate means existed to meet RG 1.97 requirements.

A substantial portion of the issues was related to lack of documented analysis showing seismic or environmental qualification, and/or clear identification and rationale for exceptions to RG 1.97 requirements. The root causes were identified as:

(1) Design basis rationale and interpretations not fully documented during implementation; and

(2) Lack of attention to detail in implementing design basis for RG 1.97.

Apprising the NRC

On December 18, 1991, PG&E representatives (Messrs. W. Fujimoto, R. Webb, and J. Tomkins) met with the NRC in Rockville, Maryland, to discuss the resolution of RG 1.97 implementation issues identified during the 1988 NRC audit, and identified by PG&E via nonconformance review and design basis reconstitution activities (DCM T-34). At this meeting, PG&E also outlined its approach for confirming the scope and resolving RG 1.97 implementation issues (i.e., performing a 100 percent review of all 20 variables in TS Table 3.3-10, and a sampling review of the total population of RG 1.97 loops).

The NRC generally accepted both PG&E's position that the issues were of low safety significance and PG&E's proposed resolution approaches. The NRC concurred that a subsequent report would be an appropriate vehicle to document the results of the TS and random loop reviews and any additional exceptions to the RG. They also requested that an independent review be performed of PG&E's RG 1.97 review methodology and implementation.

At the meeting, the NRC did not indicate any concerns with PG&E's schedule for complete resolution of RG 1.97 issues by the seventh refueling outage for both Units (1R7/2R7).

The RG 1.97 Review Project

Thus, to confirm the scope and define remaining issues associated with DCPP's implementation of RG 1.97, to ensure that a closure path is established and tracked for each identified issue, and to fulfill the commitments made to the NRC at the meeting on December 18, 1991, PG&E established the RG 1.97 Review Project. PG&E assigned a multi-discipline project team and provided a common area to facilitate interaction and assure a concentrated focus.

The overall review approach was to: (1) establish a methodology to verify, on a loop-by-loop basis for each variable, the design and qualification attributes delineated in Table 1 of RG 1.97; (2) implement the methodology for all 20 of the variables listed in TS Table 3.3-10 ("Accident Monitoring Instrumentation"); and (3) implement the methodology on a random sample of the total population (Units 1 and 2) of RG 1.97 instrumentation loops. Detailed project instructions were prepared to control the performance and documentation of the reviews.

Independent Review of the RG 1.97 Review Project

In June 1992, in its independent review of the RG 1.97 Review Project, EDAN Engineering Corporation concluded that: "... the DCPP Regulatory Guide 1.97 program is capable of validating the committed level of design and qualification for the

selected instrument loops. The program is well organized and adequate resources have been applied to the project to ensure that it can succeed in its charter."

The EDAN report also provided some recommendations. These recommendations were resolved by the project, as discussed in Section 4.2.3 in the body of this report.

Appendix B

Review Methodology and Loop Package Contents Description

A "loop package" was prepared for each instrumentation variable in Technical Specification (TS) Table 3.3-10 ("Accident Monitoring Instrumentation") for each DCPP Unit, and for each of a random selection of the total population (Unit 1 plus Unit 2) of Regulatory Guide (RG) 1.97 loops. The former are termed "TS variable loop packages"; the latter are called "random loop packages." Project instructions controlled the assembly, review, approval, and revision of the loop packages. Each package contains the following principal sections:

- Master Table -- The Master Table provides a composite summary of the review and evaluations performed for the variable. Each loop associated with the variable constitutes a line item in the table. The table documents "required" and "as-found" information for each RG 1.97 review attribute (e.g., redundancy, separation, isolation, environmental qualification, seismic qualification).

- Component and Loop Summary Tables -- The Master Table in the loop package is a summary of information contained in the Component and Loop Summary Tables (CLSTs) for the package. One CLST was prepared for each loop associated with the variable; i.e., if a variable had four loops, then the loop package has four CLSTs. Each line item on a CLST is an electrical or I&C component associated with the loop. As with the Master Table, "required" and "as-found" information was recorded for each RG 1.97 review attribute.

- Findings -- Project "findings" were identified, reviewed, and documented in accordance with project instructions. Each finding was classified as either a potential problem, discrepancy, exception or clarification, as defined in Section 6.2 of this report.

  Potential problems and discrepancies were reviewed for impact on plant operability and for reportability. One Operability Evaluation and several Prompt Operability Assessments were prepared in accordance with PG&E Nuclear Plant Administrative Procedure C-29. Only one item was determined to be reportable; that issue regarded the potential for failure of surge suppressor diodes in the ABVS dampers and fan motor control circuits, and was reported in accordance with 10 CFR 50.73. (See DCPP LER No. 92-005-01[18].) (PG&E also submitted a report[24], following discussions with the NRC, on a potential issue with 120 VAC power isolation.)

- Discipline-Specific Review Sheets -- The information contained on the CLSTs for each package was derived from the results of individual discipline reviews. These were documented on the following forms:

  - I&C Loop/Component Evaluation Sheets
  - Electrical Component Evaluation Sheets
  - Seismic Component Evaluation Sheets
  - EQ Component Evaluation Sheets

- Drawings Associated With the Loop -- The following drawings may be included in the loop packages to provide design references for the review and conclusions regarding RG 1.97 design adequacy:

  - Instrument Schematic
  - Electrical Schematic and/or Loop Diagram
  - Electrical Block Diagram (prepared per project instructions)
  - Diagram of Connections
  - Any other relevant drawings

Each loop package also contains an Open Item Log which lists the findings that were associated with the package. Closure of all of the findings for the package was not a prerequisite for package closure; however, in general, most of the findings associated with a package were closed prior to the final approval of the package. In addition, completion of all required action was not a prerequisite for finding closure; a finding could be closed so long as the required actions had been entered for tracking purposes into Action Requests (ARs) and/or Action Evaluations (AEs) in DCPP's Plant Information Management System (PIMS).

Loop packages will be retained as historical quality records in PG&E's Records Management System. RG 1.97 Review Project documentation is available for NRC review.

Appendix C

Status of Completed and Planned Modifications

In submittals to the NRC dated October 25, 1991[13], and February 7, 1992[14], PG&E committed to hardware changes to resolve outstanding issues from the NRC's 1988 inspection of Regulatory Guide 1.97 implementation. The status of these commitments is as follows:

- Resistor networks and the replacement of the Hagan protection racks with the Westinghouse Eagle 21: This work has been planned for both Units, development of design change packages is in progress, and the installation remains on schedule to be completed during the sixth refueling outages for both Units (1R6 and 2R6).

- Class IB Category 2 instrument loops that originate in the Hagan control racks and are outside the scope of the Eagle 21 upgrade: PG&E committed to supplying qualified isolation devices for these loops. This work was completed during outage 1R5 for DCPP Unit 1 and outage 2R5 for Unit 2. (Note: Class IB Category 3 loops that originate in control racks do not require isolation and will not have isolation devices added.)

- Power source redundancy of the four wide range steam generator (SG) level loops: PG&E committed to transferring the power source for two of the loops (SGs 3 and 4) to a different vital instrument AC power source (protection set III). The design change schedule has been finalized for implementation 1R6 and 2R6. The design change packages have been issued for both Units.

- Installation of a seismically-qualified neutron flux recorder on Panel PAM-1 in the control room: The design change schedule has been finalized for implementation during 1R6 and 2R6. The design change packages have been issued for both Units.

The status of other hardware changes discussed at the December 18, 1991, meeting between the NRC and PG&E is as follows:

- Upgrading of the white light monitoring system: Enclosure 8 of PG&E Letter DCL-93-284[26] provides an overview of the design modification that PG&E is preparing for this system. Implementation of the modification remains scheduled for the seventh refueling outages for both DCPP Units (1R7 and 2R7).

- Design change to secure the containment fan cooler unit normal and accident mode dampers in position: by letter dated April 17, 1992[19], the NRC approved PG&E's License Amendment Request[20] to change DCPP Technical Specification Section 4.6.2.3 to allow for the change. The work was completed during outage 1R5 for DCPP Unit 1 and outage 2R5 for Unit 2.