EDORF

DEC 23 1983

Docket Nos.: 50-275
50-323

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

FROM: William Dircks, Executive Director
for Operations

SUBJECT: DIABLO CANYON SSER 20 - CRITICALITY/LOW POWER AUTHORIZATION WITH RESPECT TO DESIGN VERIFICATION PROGRAM

Attached is SSER 20 for the Diablo Canyon Nuclear Power Plant, Unit 1. SSER 20 presents the staff safety evaluation of those issues that resulted from the design verification effort and which previously were discussed in SSER 18, SSER 19 and in my memorandum to you dated November 7, 1983.

We have concluded that those issues, flowing from the IDVP/ITP, which require a satisfactory resolution prior to authorizing criticality and low power testing (Step 2) have been satisfactorily resolved. Final resolution of the remaining open items will be achieved prior to full power authorization (Step 3).

We note, however, that the staff's status report related to our review of allegations is nearing completion and will be provided early next week. That transmittal will provide our position with respect to Step 2 as a result of our review of allegations.

(Signed) William J. Dircks

William Dircks, Executive Director
for Operations

Enclosure:
As Stated

XA Copy Has Been Sent to PDR

cc: SECY
OPE
OGC

8401030397

XA

NUREG-0675
Supplement No. 20

Safety Evaluation Report
related to the operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2

Docket Nos. 50-275 and 50-323

Pacific Gas and Electric Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

DECEMBER 1983

ABSTRACT

Supplement 20 to the Safety Evaluation Report for Pacific Gas and Electric Company's application for licenses to operate Diablo Canyon Nuclear Power Plant, Unit 1 and Unit 2 (Docket Nos. 50-275 and 50-323), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports on the verification effort for Diablo Canyon Unit 1 that was performed between November 1981 and the present in response to Commission Order CLI-81-30 and an NRC letter of November 19, 1981 to the licensee. Specifically, Supplement 20 addresses those issues and other matters identified in Supplements 18 and 19 that must be resolved prior to Unit 1 achieving criticality and operating at power levels up to 5 percent of rated full power. This SER Supplement applies only to Diablo Canyon Unit 1.

TABLE OF CONTENTS

	Page
ABSTRACT	iii
ABBREVIATIONS	vii
1 INTRODUCTION	1-1
APPENDIX C STAFF EVALUATION OF VERIFICATION EFFORT FOR DIABLO CANYON NUCLEAR POWER PLANT - UNIT 1	
APPENDIX D LIST OF CONTRIBUTORS	

ABBREVIATIONS

ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASLAB	Atomic Safety and Licensing Appeal Board
ASLB	Atomic Safety Licensing Board
ASME	American Society of Mechanical Engineers
ASW	auxiliary salt water
BIR	Blume Internal Review
BNL	Brookhaven National Laboratory
CAP	Corrective Action Program
CCW	component cooling water
CCWS	component cooling water system
CQA	construction quality assurance
CRVPS	control room ventilation and pressurization system
DCNPP	Diablo Canyon Nuclear Power Plant
DCP	Diablo Canyon Project
DDE	double design earthquake
DDOF	dynamic degree(s) of freedom
DE	design earthquake
DFOTS	diesel fuel oil transfer system
EDS	EDS Nuclear, Inc.
EOI	Error or Open Item
FOT	fuel oil transfer
FSAR	Final Safety Analysis Report
GDC	General Design Criteri(on) (a)
GFA	Guy F. Atkinson Co.
HLA	Harding Lawson Associates
HVAC	heating, ventilation, and air conditioning
IDVP	Independent Design Verification Program
IE	Office of Inspection and Enforcement (NRC)
IEEE	Institute of Electrical and Electronics Engineers
ITP	Internal Technical Program
ITR	Interim Technical Report

LCV	level control valve
LOCA	loss-of-coolant accident
MAFW	motor-driven auxiliary feedwater
MCB	main control board
MSS	main steam system
NRC	U.S. Nuclear Regulatory Commission
NSC	Nuclear Service Corporation
OBE	operating basis earthquake
OIR	Open Item Report
OWST	outdoor water storage tank
PG&E	Pacific Gas and Electric Company
QA	quality assurance
RFI	request(s) for information
RFR	R. F. Reedy, Inc.
RLCA	Robert L. Cloud and Associates
RRA	Radiation Research Associates
SEAOC	Structural Engineers Association of California
SER	Safety Evaluation Report
SIF	stress intensification factor
SIFPR	Supplementary Information for Fire Protection Review
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSE	safe shutdown earthquake
SSI	soil-structure interaction
SWEC	Stone & Webster Engineering Corporation
TAFW	turbine-driven auxiliary feedwater
TES	Teledyne Engineering Services
TMI	Three Mile Island
UL	Underwriters Laboratory
W	Westinghouse
W&B	Wismer & Becker
ZPA	zero period acceleration

1 INTRODUCTION

The staff of the U.S. Nuclear Regulatory Commission (NRC) issued on October 16, 1974, its Safety Evaluation Report (SER) in matters of the application of the Pacific Gas and Electric Company (PG&E) to operate the Diablo Canyon Nuclear Power Plant, Unit 1 and Unit 2. The SER has since been supplemented by Supplement Nos. 1 through 16 and Nos. 18 and 19. SER Supplement No. 18 (SSER 18) dated August 5, 1983, presented the staff's safety evaluation on matters related to the verification effort for Diablo Canyon Unit 1 that was the result of Commission Order CLI-81-30 and an NRC letter to PG&E of November 19, 1981. SSER 18 contained a number of open items that required resolution and a number of followup items that required some additional action by PG&E and the staff. SER Supplement 19, dated October 14, 1983, presented the staff's safety evaluation of those unresolved matters identified in SSER 18 which must be satisfactorily resolved prior to commencement of fuel loading operations at Diablo Canyon Unit 1 which is also known as Step 1 of the Diablo Canyon Unit 1 licensing process. (Supplement 17 has not yet been issued. It is not related to the design verification effort.)

This is SER Supplement No. 20 (SSER 20) and presents the staff's safety evaluation of those matters that must be satisfactorily resolved prior to Unit 1 achieving criticality and operating at power levels up to 5 percent of rated full power, i.e., full reinstatement of the suspended low power license. This is also known as Step 2 of the Diablo Canyon Unit 1 licensing process. The verification effort relates only to Unit 1 of the Diablo Canyon Nuclear Power Plant; therefore, this supplement applies only to Unit 1 unless otherwise stated. SER Supplement No. 20 is based on information available to the staff as of December 16, 1983.

The verification effort covers a wide range of subjects that cannot be presented effectively in the normal SER format. Therefore, the safety evaluation of the verification effort in SSERs 18 and 19 was reported in Appendix C to those supplements and the same format is used in SSER 20. (Appendix A was previously used for the chronology and Appendix B was used for the bibliography in the Diablo Canyon SER). Appendix D to this SER supplement includes the list of contributors to the report.

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

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

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

APPENDIX C

STAFF EVALUATION OF VERIFICATION EFFORT FOR
DIABLO CANYON NUCLEAR POWER PLANT - UNIT 1

TABLE OF CONTENTS

	Page
1. Background and Introduction	C.1-1
2. Quality Assurance	NA
3. Seismic Design Verification Effort	C.3-1
3.1 Introduction	C.3-1
3.2 Structures	C.3-1
3.2.1 Containment Annulus Structure	C.3-1
*OI 1: Free Hand Averaging of Spectra	C.3-1
OI 2: 20 Hertz Cutoff Frequency	C.3-1
3.2.2 Containment Interior Structure	C.3-2
3.2.3 Containment Exterior Shell	C.3-2
OI 3: AISC Code for Penetrations	C.3-2
OI 4: Stress Level at Openings	C.3-2
3.2.4 Auxiliary Building	C.3-2
OI 5: Floor Slab Model	C.3-2
OI 6: ACI Code Justification	C.3-4
OI 7: Soil Spring Influence	C.3-5
3.2.5 Fuel Handling Building	C.3-5
OI 8: Input from Auxiliary Building Response	C.3-5
OI 9: Procedure for Degree of Freedom Reduction	C.3-6
3.2.6 Intake Structure	C.3-7
3.2.7 Outdoor Water Storage Tanks	C.3-7
3.2.8 Turbine Building	C.3-7
OI 10: Load Combinations	C.3-7
OI 11: Roof Truss Modeling	C.3-7
OI 12: Continuous Exterior Wall	C.3-8
OI 13: Vertical Models	C.3-8
OI 14: Modal Combinations	C.3-9
OI 15: AISC Code Allowable Stresses	C.3-10
OI 31: Response Combinations	C.3-10
3.3 Piping and Piping Supports	C.3-11
3.3.1 Large Bore Piping and Supports	C.3-11

TABLE OF CONTENTS

Page

OI 16: Analysis of Supports. C.3-11
OI 17: Buckling Criteria C.3-11
OI 18: Additional Piping Analyses. C.3-12

3.3.2 Small Bore Piping and Supports. C.3-13

OI 19: Analysis of Supports. C.3-13

3.4 Equipment and Supports C.3-13

3.4.1 Mechanical Equipment and Supports C.3-13

OI 20: Equipment Qualification C.3-13
OI 21: Valve Nozzle Stresses C.3-14
OI 22: Pump Flange Stresses. C.3-15

3.4.2 Heating Ventilation and Air Conditioning Equipment. . . C.3-15
3.4.3 Electrical Equipment and Instrumentation and Supports. C.3-15

OI 23: Cable Tray Qualification. C.3-15
OI 24: Superstrut Welds. C.3-15

3.5 Other Seismic Design Verification Topics C.3-15

OI 25: Intake Structure Lateral Forces C.3-15

3.6 Brookhaven Analysis. C.3-17

OI 26: Buried Diesel Fuel Oil Tank C.3-17

4. Nonseismic Design Verification Effort C.4-1

4.1 Introduction C.4-1
4.2 Initial Sample C.4-1

4.2.1 Verification of Mechanical/Nuclear Design C.4-1

**FI 1: AFWS Runout Control System Test. C.4-1
FI 2: AFWS Drawing Update for Steam Trap C.4-2

4.2.2 Electrical Design C.4-2

FI 3: AFWS Electrical Circuit Coding C.4-2

4.2.3 Instrumentation and Control Design. C.4-3

OI 27: AFWS Isolation Valve Classification. C.4-3
OI 28: Steam Generator Blowdown Isolation Circuitry . . C.4-3

TABLE OF CONTENTS

	Page
FI 4: Equipment Qualification Report Update.	C.4-4
FI 5: Motor Capacitor Qualification Analysis	C.4-4
4.2.4 High Energy Line Break and Internally Generated Missiles.	C.4-5
FI 13: ANS 58.2 Jet Impingement Temperature Calculation Methodology	C.4-5
4.2.5 Effects of High Energy Line Cracks and Moderate Energy Line Breaks.	C.4-6
FI 6: Break in Steamline to Turbine Driven AFWS Pump.	C.4-6
FI 7: Jet Impingement Equipment Qualification Temperature.	C.4-6
FI 14: Environmental Qualification of Cables and Wires.	C.4-7
FI 15: Protection for CRVPS	C.4-8
FI 8: Protective Shields for CRVPS Valves.	C.4-9
4.2.6 Fire Protection	C.4-9
4.2.7 Radiation Environmental Qualification	C.4-9
4.2.8 Pressure and Temperature Environmental Analyses	C.4-9
4.3 Additional Verification.	C.4-10
4.3.1 Redundancy of Equipment and Power Supplies in Shared Safety-Related Systems	C.4-10
4.3.2 Selection of System Design Pressure and Temperature and Differential Pressure Across Power-Operated Valves	C.4-10
FI 9: Modifications to AFWS.	C.4-10
4.3.3 Environmental Consequences of Postulated Pipe Ruptures Outside Containment.	C.4-10
FI 10: Verification of Assumptions for Pressure/Temperature Calculations	C.4-10
FI 11: Pressure/Temperature Reanalyses Modifications and Documentation	C.4-11
FI 12: Confirmation of Environmental Qualification Documentation	C.4-12
4.3.4 Circuit Separation and Single Failure	C.4-13

TABLE OF CONTENTS

Page

4.3.5 Jet Impingement Effects of Postulated Pipe Breaks Inside Containment C.4-13

OI 29: Jet Impingement Loads on Piping Inside Containment. C.4-13

4.3.6 Rupture Restraints. C.4-14

OI 30: Rupture Restraints Inside and Outside Containment C.4-14

5. Summary and Conclusions C.5-1

6. References. . . . . NA

7. Chronology Pertaining to Diablo Canyon Unit 1 Verification Effort. C.7-1

8. Tables. C.8-1

* OI = Open Item
** FI = Followup Item

1 BACKGROUND AND INTRODUCTION

On August 5, 1983, the NRC staff issued SER Supplement No. 18 (SSER 18) which presented the staff evaluation of a design verification effort for Diablo Canyon Unit 1. The basis for this effort and a description of the process of this effort were presented in detail in SSER 18. In summary, the Commission Memorandum and Order CLI-81-30 (November 19, 1981) suspended the authorization to load fuel and perform low power testing granted by the Diablo Canyon Unit 1 Operating License No. DPR-76 because serious weaknesses had been identified in the implementation of the quality assurance programs of PG&E and its seismic, service related contractors. The Commission Order required that an independent design verification program (IDVP) of seismic, service related contract activities (pre-1978) be completed to the satisfaction of the NRC prior to lifting the suspension. In addition, the NRC staff issued a letter (November 19, 1981) which required an IDVP with respect to non-seismic, service related contract activities, PG&E internal design activities, and post-1978 seismic, service related contract activities, which must be satisfactorily completed prior to an NRC decision regarding a full power license. The activities associated with the Commission Order and the NRC letter have become known as Phase I and Phase II of the design verification, respectively.

The Diablo Canyon Unit 1 design verification effort consisted of two separate efforts. One was the IDVP as discussed above. It was conducted by organizations and individuals not associated with PG&E under the program management of Teledyne Engineering Services (TES). This effort was completed in October 1983. The other effort is the PG&E internal technical program (ITP) which is performed by PG&E's Diablo Canyon Project (DCP) which is a combined PG&E/Bechtel organization. This effort is still ongoing.

As stated in SSER 18, by the fall of 1982 it became evident that the earlier distinction between the pre-1978 and post-1978 effectiveness of design controls was no longer valid and thus the timing for completion of Phase I and Phase II activities was no longer necessary. PG&E proposed and the Commission approved a three-step process for reinstatement of the suspended low power license and issuance of the full power license as follows:

- Step 1: fuel load authorization (part of suspended low power license)
- Step 2: criticality and low power authorization (remainder of suspended low power license)
- Step 3: full power license

The activities that must be completed for each of the three steps were delineated in the PG&E submittal of December 3, 1982. In SSER 18 the staff presented its safety evaluation of the design verification effort, both IDVP and ITP, without specifically focusing on the requirements for the three-step concept.

The staff safety evaluation of the design verification effort in SSER 18 was based on information that had been submitted by the IDVP and PG&E as of June 30, 1983. At that time the effort had not been completed. Further analyses and verification effort by the IDVP and the DCP (including modifications by the DCP) were still in progress and the status was provided in SSER 18. Throughout SSER 18 the staff also had identified a number of open items which, based on the staff evaluation of the information provided, required further information, confirmation of data, additional justification or bases for an analysis or additional analyses or modifications. In addition, the staff also had identified in SSER 18 a number of followup items which required further documentation or verification based on commitments by the licensee to update the Final Safety Analysis Report (FSAR) or the need for staff verification of certain actions by the licensee.

On October 14, 1983 the staff issued SER Supplement 19 (SSER 19) which updated the staff evaluation of SSER 18 and, in particular, provided the staff evaluation of those matters that were identified as unresolved in SSER 18 and which required a satisfactory resolution prior to reinstatement of the licensee's authority to load fuel i.e., Step 1. SSER 19 was based on information submitted to the staff as of October 13, 1983. SSER 19 provided a listing of the 31 open items in SSER 18, including the licensing milestone for the resolution of each open item (i.e., Step 1, 2 or 3) and a listing of the 15 followup items in SSER 18.

On November 7, 1983 the staff provided the Commission with an evaluation of each of the 31 open items and the 15 followup items which concluded that all items designated for resolution prior to authorization to load fuel (i.e., Step 1) had been acceptably resolved for Step 1. The evaluation also gave the bases for the staff conclusion that resolution of the remaining items could be deferred to a later time. This evaluation was also issued as Board Notification BN-83-179, dated November 9, 1983. On November 8, 1983 the Commission issued Memorandum and Order CLI-83-27 which reinstated PG&E's license authorizing fuel loading and precriticality testing at Diablo Canyon Unit 1 in accordance with the conditions as prescribed in the Diablo Canyon Technical Specifications for Operational Mode 6 (Refueling) and Operational Mode 5 (Cold Shutdown).

This report is SER Supplement 20 (SSER 20) which is based on information provided to the staff as of December 15, 1983. This includes semimonthly reports from the IDVP and PG&E, the IDVP Final Report, the PG&E Final Reports for Phase I and Phase II, and Interim Technical Reports (ITRs) form the IDVP. A complete list of all ITRs issued is provided in Section 8. SSER 20 presents the staff safety evaluation of open items and followup items that must be satisfacorily resolved prior to Step 2 for the reinstatement of the licensee's authority for Diablo Canyon Unit 1 to achieve criticality and perform low power testing i.e. the reinstatement of the authority to perform all activities under license DPR-76.

Since the issuance of SSER 19 PG&E, the IDVP and the staff have continued to pursue the completion of the design verification effort and the resolution of issues, in particular those that require satisfactory resolution prior to Step 2. All open items and followup items are listed in Tables C.1-1 and C.1-2 respectively. During its review the staff met with the licensee and requested additional information on a number of items. The information requests and submittals are included in the chronology presented in Section 7. The status of each item and the basis and schedule for its resolution as presented in Board Notification 83-179 of November 9, 1983 have been included in this supplement.

SSER 20 is prepared in the same format as the SSER 18, the initial safety evaluation of the design verification effort. It presents the staff evaluation of additional information provided by addressing each Open Item and Followup Item of Tables C.1-1 and C.1-2 in the appropriate section where it originally had been identified in SSER 18. SSER 18, 19, and 20 address only those matters associated with the design verification effort and do not include other considerations pertinent to the reinstatement of the suspended low power license.

Table C.1.1

Diablo Canyon Design Verification Program
Open Items in SSER 18 (August 5, 1983)

1. Free-Hand Averaging of Spectra for Containment Annulus Structure
2. 20 Hertz Cutoff Frequency for Containment Annulus Structure
3. AISC Code for Penetrations in Containment Shell
4. Stress Level at Openings in Containment Shell
5. Floor Slab Model for Auxiliary Building
6. ACI Code Justification for Auxiliary Building
7. Soil Spring Influence for Auxiliary Building
8. Fuelhanding Building Input from Auxiliary Building Response
9. Reduction of Degrees of Freedom for Fuel Handling Building
10. Load Combinations in Turbine Building
11. Roof Truss Modeling of Turbine Building
12. Continuous Exterior Wall of Turbine Building
13. Vertical Models for Turbine Building
14. Modal Combination in Turbine Building
15. AISC Code Allowable Stresses in Turbine Building
16. Analysis of Large Bore Piping Support
17. Buckling Criteria for Large Bore Piping and Support
18. Additional Large Bore Piping Analyses
19. Analysis of Small Bore Piping Support
20. Equipment Qualification
21. Valve Nozzle Stresses
22. Pump Flange Stresses
23. Cable Tray Qualification
24. Superstrut Welds
25. Intake Structure Lateral Forces
26. Buried Diesel Fuel Oil Tank
27. AFWS Isolation Valve Classification
28. Steam Generator Blowdown Isolation Circuitry
29. Jet Impingement Loads Inside Containment
30. Rupture Restraints Inside and Outside Containment
31. Turbine Building Response Combinations

Table C.1.2

Diablo Canyon Design Verification Program
Follow-up Items in SSER 18 (August 5, 1983)

1. AFWS Runout Control System Test
2. AFWS Drawing Update for Steam Trap
3. AFWS Electrical Circuit Coding
4. Equipment Qualification Report Update
5. Qualification Analysis for Motor Capacitor
6. Break in Steamline to Turbine Driven AFWS Pump
7. Jet Impingement Temperature for Equipment Qualification
8. Protective Shields for CRVPS Valves
9. Modifications to AFWS
10. Verification of Assumptions for Pressure/Temperature Calculations
11. Modifications and Documentation from Pressure/Temperature Reanalyses
12. Confirmation of Environmental Qualification Documentation
13. ANS 58.2 Jet Impingement Temperature Calculation Methodology
14. Moderate Energy Line Break Environmental Qualification of Cables/Wires
15. Protection for CRVPS

3. SEISMIC DESIGN VERIFICATION EFFORT

3.1 Introduction

The approach to the seismic design verification effort was discussed in detail in this section of SSER 18. Initially the effort consisted of a design verification of a sample of seismic design activities by the IDVP. In early 1982 the DCP expanded its effort to a complete seismic design review of all safety-related structures, systems, and components. The IDVP then verified this effect on a sample bases. The staff efforts consisted of extensive review and analysis including the efforts by its consultant, Brookhaven National Laboratory, in particular with respect to the containment annulus structure.

3.2 Structures

3.2.1 Containment Annulus Structure

Open Item 1: Free-Hand Averaging of Spectra

In SSER 18 (page C.3-9) the staff stated that the free-hand averaging of spectra for the containment annulus structure should be performed in accordance with a staff-approved technique. The licensee provided additional information. In SSER 19 (page C.3-1) the staff stated that, based on its review and evaluation, this concern was resolved. PG&E had committed to provide additional spectra and appropriate information to confirm the spectra that had been provided at that time. The concern was further addressed in Board Notification 83-179. The information that had been submitted by the licensee in earlier submittals contained samples of spectra that had obviously been presented incorrectly although the text of the response and previous discussions with the engineering staff of the DCP had indicated that acceptable practices had been employed. Further spectra were submitted on October 14, 1983. Based on the review of the information the staff concludes that the spectra have been presented correctly. As stated in Board Notification 83-179 the staff considers this open item resolved.

Open Item 2: 20 Hertz Cutoff Frequency

In SSER 18 (page C.3-9) the staff had stated that the use of the 20 Hertz cutoff frequency for the generation of floor response spectra should be verified and/or justified. The licensee subsequently provided additional information. In SSER 19 (page C.3-2) the staff stated that based on its review and evaluation of the information the staff considered the concern resolved. PG&E had committed to provide additional analyses to confirm the results. In Board Notification 83-179 the staff provided the following further bases and a schedule for the complete resolution of this concern:

> "The annulus steel structure has been stiffened to assure that all structural members have a primary response frequency of 20 Hertz or above, referred to as a 20 Hertz cutoff frequency. The energy available in seismic ground motion drops off rapidly from approximately

> 20 Hertz to 33 Hertz. At 33 Hertz no amplification of the ground motion input occurs. In the 20 Hertz to 33 Hertz range some amplification can theoretically occur in the annulus structure with an attendant increase in the acceleration experienced by piping supported by the annulus members. The DCP has provided analyses of the combined structure-piping system that indicate very small effects due to amplification in the 20 Hertz to 33 Hertz range. Although some additional analyses may be requested by the staff to provide a well documented record for future reference, the staff review to date has progressed to the point where the likelihood of additional modifications is low. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore, it is acceptable to the staff to consider this matter resolved for Step 1."

The staff requested at a meeting with PG&E on December 6, 1983, that PG&E provide an additional analysis for a piping system in the high amplification area of the annulus structure, since the spectra for these locations show acceleration peaks in the 20-33 Hertz region. PG&E committed to provide to the staff for its evaluation a detailed outline for this additional piping analysis prior to initiating the analysis. The staff requires that this proposal be submitted prior to Mode 2 (criticality). The safety significance of this issue relates only to assuring that full design margins are achieved under loadings associated with the Hosgri event. In consideration of the low likelihood of occurrence of the Hosgri event during this time period and the small fission product inventory during low power operation (Mode 2), the staff has determined that the actual analysis and the comparison of the results of this analysis with similar results for the same piping system where the 20 Hertz cutoff criterion is applied can appropriately be deferred for completion before Step 3, i.e., issuance of the full power license.

3.2.2 Containment Interior Structure

No concern was identified in this section.

3.2.3 Containment Exterior Shell

<u>Open Item 3: AISC Code for Penetrations</u>

In SSER 18 (page C.3-17) the staff stated that the use of the AISC Code for containment penetration analysis should be justified. The resolution of this open item was presented in SSER 19 (page C.3-2).

<u>Open Item 4: Stress Level at Opening</u>

In SSER 18 (page C.3-17) the staff stated that the local yielding of steel plates around the opening in the containment exterior shell should be justified. The resolution of this open item was presented in SSER 19 (page C.3-2).

3.2.4 Auxiliary Building

<u>Open Item 5: Floor Slab Model</u>

The staff identified in SSER 18 (page C.3-22) the following concern regarding assumptions for the auxiliary building floor slab:

"The seismic model used by the DCP to predict the structural loads and produce the floor response spectra is of the generally accepted type for normal seismic analysis. However, the model has many simplifications and inherent assumptions. One assumption is that the floor slabs are rigid as compared to the walls. If floor slab flexibilities are to be used as justification for accepting an overstress condition, then these flexibilities should be incorporated into the dynamic model used to predict the structural loadings or show the flexibilities to be unimportant."

In Board Notification 83-179 the staff provided the following status, basis and schedule for the resolution of this concern:

"The seismic stick model used by the DCP to predict the structural loads and produce the floor response spectra is of the generally accepted type for normal seismic analysis and has many simplifications and inherent assumptions. One assumption is that the floor slabs are rigid as compared to the walls. The DCP use of a hand calculation method for distributing the stick model responses to the individual elements resisting the loads, resulted in higher stresses than allowable in the floor slabs. The DCP had used the concept of floor slab flexibility to redistribute loads as a basis for explaining this apparent overstress condition. This explanation was not acceptable to the staff. The DCP subsequently constructed a three-dimensional finite element model to more realistically distribute the stick model loads to the resisting elements. The results of this finite element analysis indicate the stresses in the floor slabs are within the Code allowables. The IDVP has verified and accepts the methodology used by PG&E. The staff is currently completing its review of this matter. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer the resolution of the auxiliary building floor slab apparent overstress to Step 2."

In response to the staff's concern, the DCP performed parametric studies by developing a three-stick model which allowed for the inclusion of inplane floor flexibility. Details of these studies are described in Section 4.2.15 of ITR-55 of the IDVP, and the results are summarized in the DCP submittal dated November 17, 1983.

The staff found these studies inadequate in that the results as reported in Tables 12 and 13 of ITR-55 showed considerable differences in frequencies and shear forces between the models with infinite beam stiffness and actual upper bound stiffness. The licensee has since performed additional studies in which the a 3-D finite element model of the auxiliary building was subjected to static loadings equal to the seismic-induced inertial loads. These inertial loads were obtained by multiplying the floor accelerations calculated from the dynamic response of the 2-D stick model of the building by the mass distribution in the 3-D finite element model.

The licensee provided additional information by letter of December 2, 1983. In addition the staff reviewed the licensee's "3-D Model Studies-Model C/C1" (Calculation No. 30.23.1.2.2) during a meeting with the licensee on December 6, 1983. Based on the above, the staff found that the resultant slab deformation from these studies departed from a rigid slab pattern by not more than 20 percent. Since a variation of this magnitude in the floor stiffness would generally not cause any significant difference in the overall response of the structure, the staff considers the 2-D stick model for the auxiliary building acceptable. This open issue is therefore resolved.

Open Item 6: Justification of ACI Code

The staff identified in SSER 18 (page C.3-22) the following concern regarding the use of different versions of the ACI Code:

> "The use of different versions of the ACI 318 Code for evaluation of the floor slabs and walls is not appropriate. The versions ACI 318-63 and ACI 318-77 are not the versions committed in the Hosgri evaluation criteria outlined in the FSAR. The use of the different versions of the Code and the modifications to the 1977 Code as described in Appendix 2a to the DCP Phase I Final Report should be justified by the DCP and evaluated by the IDVP."

In Board Notification 83-179, the staff provided the status, basis and schedule for the resolution of this concern as follows:

> "The ACI 318-63 Code is the basis for the acceptance criteria for design of the auxiliary building. This 1963 version of the Code does not explicitly provide guidance for evaluating in-plane forces in shear walls although Section 104 of ACI 318-63 allows criteria based on test data to be used for the design of structural components for loads not covered by its provisions. Initially, the ITP used the ACI 318-77 Code, which explicitly provides criteria for in-plane shear, until criteria could be developed consistent with the provisions of ACI 318-63. The ITP developed Appendix 2A to the Phase I Final Report to provide criteria for evaluating the in-plane loads. The provisions of this document are based on available test data for in-plane shear consistent with the ACI 318-63 Code originally accepted. The ACI 318-77 Code was used by the ITP for the final member evaluation of some members and is more conservative than Appendix 2A. By using ACI 318-77, the provisions of Appendix 2A (and, therefore, ACI 318-63) are also met. The IDVP has verified the methodology and accepted the use of Appendix 2A. The Appendix is currently under review by the staff. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems and components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer the resolution of the use of Appendix 2A to Step 2."

The staff has completed its review. During a meeting with the DCP on December 6, 1983, the staff also reviewed calculations contained in a PG&E calculation package (Calculation No. 3D.23.3.2.2) which illustrate the method used by the DCP in evaluating the strength of the slabs and walls in the auxiliary building. It was determined that the procedure as outlined in Appendix 2A of the Phase I Final Report was used for the qualification of all slabs and walls in the auxiliary building. The provisions of the ACI 318-77 Code were not applied. However, Appendix 2A does contain a shear stress restriction of $10\ f'_c$ where f'_c is the compressive stress of the concrete.

Futhermore, the use of Appendix 2A rather than the ACI 318-77 Code is considered justified since the approved licensing criterion in the ACI 318-63 Code which contains no provisions for shear walls and Article 104 of that Code permits the use of design practice justified by test or analysis for items not covered in the Code. While in some cases Appendix 2A is not always as conservative as the provisions in the ACI 318-77 Code, it is sufficiently conservative and in many cases it is more conservative. Based on its review and evaluation as discussed above the staff concludes that the use of Appendix 2A of the Phase I Final Report for qualifying the slabs and shear walls in the auxiliary building is acceptable. This open issue is therefore resolved.

Open Item 7: Soil Spring Influence on Seismic Response

In SSER 18 (page C.3-22) the staff stated that the discrepancy between the IDVP and DCP sensitivity of the soil spring influence on the seismic response of the auxiliary building should be reconciled, including the resolution of soil properties and documentation of parametric studies. This open item was resolved in SSER 19 (page C.3-3).

3.2.5 Fuel Handling Building

Open Item 8: Input from Auxiliary Building Response

In SSER 18 (page C.3-26) the staff identified the following concern regarding the translational and torsional response of the auxiliary building:

> "The use of the translational and torsional response of the auxiliary building as input to the base of the fuel handling building must be documented more completely in the Phase I report. Parametric studies to demonstrate the validity of the DCP approach should be included in the report."

In Board Notification 83-179, the staff provided the basis, status and schedule for the resolution of this concern as follows:

> "The dynamic analysis of the auxiliary building included a simple representation of the fuel handling building. The fuel handling building was decoupled from the auxiliary building and analyzed separately using a detailed three-dimensional finite element model and the coupled model motion at the fuel handling building base. This motion consists of two parts, the translation and the torsional motion to the finite element model. This method of decoupling structural systems is generally accepted by the profession and yields satisfactory results. The IDVP has verified and accepted the methodology used by PG&E. The results presented for the finite element model have been preliminarily reviewed by the staff and appear consistent with the coupled model. The staff is currently completing the review of this matter. The progress to date for the resolution of the staff concern indicates that the possibility of additional

significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore, it is acceptable to the staff to defer the resolution of the fuel handling building input motion to Step 2."

The staff has completed its review. The DCP used two detailed 3-D models for the seismic evaluation of the fuel handling building. The input to these models was obtained from the time history response of the auxiliary building at elevation 140 ft. This input consists of a translational and a rotational component. Since each column in the fuel handling building model is at a different distance from the center of rigidity of the auxiliary building, it will see a different input motion. In submittals dated November 17, and December 2, 1983, and during a meeting on December 6, 1983, the DCP further clarified the procedure used to develop column base inputs from the translational and rotational components. The evaluations were based on the STARDYNE computer code. The column bases of the fuel handling building were tied together to a rigid base. A single input was applied to this base at an offset location that correctly represented the translational and rotational component of the motion. The staff considers this procedure acceptable and this open issue is therefore resolved.

Open Item 9: Procedure for Reduction of Degrees of Freedom

The staff identified in SSER 18 (page C.3-26) the following concern:

"The total number of degrees of freedom contained in the dynamic models was reduced to 20-30 degrees of freedom before the dynamic analyses were performed. Some recent studies have indicated that this dynamic reduction often results in serious errors particularly with regard to member loads. The particular set of dynamic degrees of freedom selected for the models should be justified."

In Board Notification 83-179, the staff provided the basis, status, and schedule for the resolution of this concern as follows:

"A statement in the Phase I Final Report regarding the dynamic degrees of freedom in the fuel handling building finite element model was not clear. The staff interpretation of this statement was that the DCP had reduced the number of degrees of freedom from 156 to 20. The reason for the staff concern is that recent studies have shown that reductions of this type could result in errors in the individual member loads. The DCP used the public domain program STARDYNE to analyze the fuel handling building finite element model. Comparisons of results from the finite element model and the stick model appear to show the results to be consistent for base shears, roof accelerations and displacements. The staff is currently completing the review of this matter. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer the resolution of the degrees of freedom in the fuel handling building model to Step 2."

The staff has completed its review. It has been clarified that the two dynamic models which incorporate the final modifications to the fuel handling building as shown in Figures 2.1.3-19 and 2.1.3-20 of the DCP Phase I Final Report have a total of 156 and 162 dynamic degrees of freedom, respectively. The models which have 20 to 30 dynamic degrees of freedom are for the unmodified fuel handling building. For the models of the modified structure, the reduction of the dynamic degrees of freedom is accomplished by using the STARDYNE computer code which has the widely used Guyan reduction procedure. The staff considers the licensee's models acceptable and this open issue is therefore resolved.

3.2.6 Intake Structure

No concern was identified in this section of SSER 18.

3.2.7 Outdoor Water Storage Tanks

No concern was identified in this section of SSER 18.

3.2.8 Turbine Building

Open Item 10: Load Combinations

In SSER 18 (page C.3-36) the staff identified a concern regarding clarification of load combinations in the analysis of the turbine building. This open item was resolved in SSER 19 (page C.3-3).

Open Item 11: Roof Truss Modeling

In SSER 18 the staff raised the following concern regarding the turbine building roof truss model (SSER 18 page C.3-36):

> "The method of modeling the roof truss by two generalized uniaxial members and obtaining individual truss member responses from the uniaxial member model is questionable, since the action of the member is different from that of a truss and the maximum response of the model may not be the maximum response of each truss member."

In Board Notification 83-179 the staff presented the following status, basis and schedule for resolving this open item:

> "The staff was concerned that the method of modeling the roof trusses by using two generalized uniaxial members and obtaining responses from the members may not produce the maximum response in each individual truss member. PG&E stated that the generalized truss model was used only for calculating global responses and that individual member forces were obtained from a model that contained all truss members. The responses from the global model were applied to the individual member model as static loads. The IDVP has verified the calculations and concluded that the idealization of the roof trusses into two uniaxial members was done properly. The staff is currently completing the review of this matter. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any

modifications which may be necessary will not interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer modeling of the turbine building roof trusses to Step 3."

The staff has essentially completed this review. Upon receipt of acceptable confirmatory documentation this matter will be resolved. It is highly unlikely that any remaining efforts would interfere with activities associated with Modes 2, 3, 4, 5 or 6. Based on the above the staff concludes that final resolution must be achieved prior to Step 3, i.e. issuance of a full power license.

Open Item 12: Continuous Exterior Wall

The staff identified in SSER 18 (page C.3-37) a concern regarding the effect of a continuous exterior wall in the turbine building vertical models. In Board Notification 83-179, the staff provided the status, basis and schedule for the resolution of this concern as follows:

"The vertical seismic analysis of the turbine building utilized four different models. The basis for using the four different models is the fact that the large openings in the floors at the turbine pedestal divide the floors into separate areas. The staff concern was over the coupling of these models through the exterior walls and their effect on the final results. The vertical walls themselves are not continuous due to large openings in the walls. Where coupling could possibly occur, the walls are stiff in their own plane and do not significantly amplify the ground motion. Therefore, little or no coupling appears to occur between the separate models. The IDVP sample included only one of the four models but did verify that resistance of adjacent bays were properly accounted for. The staff is currently completing the review of this matter. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore, it is acceptable to the staff to defer the resolution of the effect of the exterior walls on the vertical response to Step 2."

The staff has completed its review. There are four vertical models used for the analysis of the turbine building. The exterior walls which extend from column lines 5.7 to 16 is the area represented by models 2 and 3. Hence there is no coupling between models 1 and 2. The coupling between models 2 and 3 is considered to be insignificant as demonstrated by the licensee's submittal dated August 30, 1983. In this submittal, two floor response spectra, generated at column line A (bent 9, elevation 140 ft) and at column line G, (bent 6.6, elevation 140 ft) respectively, were compared with the input ground response spectrum. The comparison showed that the amplification of ground motion through the walls was insignificant. The staff considers this justification for using four separate vertical models acceptable. This open issue is therefore resolved.

Open Item 13: Vertical Models

The staff stated in SSER 18 (page C.3-37) the following with respect to vertical models for the turbine building:

"The differences in modeling the steel frame and roof truss for vertical model 1 and vertical model 2 need clarification. Specifically, the reason for changing the roof truss, modeled as a truss in model 1, to uniaxial members in model 2. Furthermore, a basis should be provided why the nodes above 140 ft have 6 degrees of freedom for model 1, while they only have 3 degrees of freedom for model 2."

In Board Notification 83-179 the staff presented the basis, status and schedule for resolution as follows:

"The staff was concerned that the differences in the number of degrees of freedom for the nodes above elevation 140 used in the two vertical models of the roof trusses was not consistent with the response of the structure. PG&E stated that since the trusses near each end of the turbine building can produce horizontal motion from a vertical input while those nearer the center of the building could not, the models were appropriate. The IDVP sampled one of the models for their evaluation and found that model acceptable. The staff is currently reviewing this matter in further detail. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer the resolution of the degrees of freedom for the vertical models to Step 3."

The staff has completed its review and evaluation of this issue and concludes that this open item is resolved.

Open Item 14: Modal Combinations

The staff identified in SSER 18 (page C.3-37) a concern regarding the use of alternative procedures for modal combinations by the SRSS method. In Board Notification 83-179 the staff provided the status, basis and schedule for the resolution of this concern as follows:

"The Phase I Final Report contained the statement that alternative procedures are being reviewed to assure that turbine building modal combinations using the SRSS method are acceptable. The staff was concerned that some method other than the SRSS method was used to evaluate the structures. The DCP did in fact evaluate the dynamic response of the turbine building using the double algebraic sum method of combining modal responses in addition to using the SRSS method. The structure was shown capable of withstanding the loads calculated by either method and satisfies the FSAR requirements. The IDVP stated in their reports that the alternate method (double algebraic sum) was not used for final member evaluation in the IDVP sample. The staff is currently reviewing this matter in more detail. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6.

Therefore it is acceptable to the staff to defer the resolution of the concern regarding use of alternative procedures for modal combinations by the SRSS method to Step 2."

The staff has completed its review. In its submittal dated August 30, 1983, the DCP confirmed that the turbine building was evaluated for dynamic responses calculated on the basis of a modal superposition response spectrum analysis with modal responses combined on an SRSS basis. Additionally, the principal lateral and vertical force resisting elements of the turbine building were also evaluated for dynamic response calculated on the basis of a double algebraic sum combination of modal responses. The staff considers the approach acceptable. This open issue is therefore resolved.

Open Item 15: AISC Code Allowable Stresses

The staff identified in SSER 18 (page C.3-37) the concern that the use of the AISC Code 8th Edition is in violation of the acceptance criteria delineated in the FSAR. Therefore, the use of the increased allowable stresses should be justified. In Board Notification 83-179, the staff provided the status, basis and schedule for the resolution of the concern as follows:

> "The AISC Code, 7th Edition, shows certain values for the allowable bearing of bolts against the member material. For the Hosgri event the force resisting members are allowed inelastic deformation as indicated in the Hosgri Report. The provisions of AISC Code, 8th Edition, allow higher bearing values and could be acceptable criteria for meeting the conditions of the Hosgri Report commitments. The IDVP states that the lower of 1.7 times the AISC allowable stress or yield strength was used and the licensee criteria were met. The staff is currently reviewing this matter in more detail. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer the resolution of the use of AISC Code, 8th Edition, to Step 2."

The staff has completed its review. In submittals of November 17, 1983 and December 2, 1983, the DCP further demonstrated the applicability of the 8th Edition of the AISC Code by applying equations 1.16-1 and 1.16-2 of the Code to determine the ultimate capacity of the 3-bolt connection. During a meeting with the DCP on December 6, 1983, the staff reviewed calculations using the actual dimensions and material properties (calculation 65-T-004) and found the application of the 8th Edition of the AISC Code to be acceptable. This open issue is therefore resolved.

Open Item 31: Response Combinations

The staff identified in SSER 18 (page C.3-37) the concern that the statement in the PG&E Phase I Final Report, "Co-directional response due to the three orthogonal components of ground motion are combined on an SRSS basis, or equivalent," indicated some other material or component combination was used.

Therefore, the equivalent method needed further explanation. In Board Notification 83-179 the staff provided the status, basis and schedule for the resolution of the concern as follows:

> "The PG&E Phase I Final Report contained a statement that the codirectional response due to the three orthogonal components of ground motion are combined on an SRSS basis or equivalent. This statement indicated that the provisions of the FSAR may not have been followed and failed to specify what equivalent method was used. PG&E subsequently informed the staff that the equivalent method used was the full value of one component added to the sum of 40 percent of each of the other two components. This approach appears to lead to an acceptable resolution of this concern, and the staff is near completion of the review of this matter. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer this matter of codirectional responses to Step 2."

The staff has completed its review. In its submittal dated August 30, 1983, the DCP indicated that the equivalent method used in the turbine building evaluation was to add the full value of one earthquake component to the sum of 40 percent of each of the other two components. This method would in general result in more conservative results than those by the SRSS combination. In the case where two components are of the same magnitude and the third component is zero, this method will lead to slightly less conservative results but the difference is insignificant for all practical purposes. During a meeting with the DCP on December 6, 1983, the staff also reviewed PG&E calculations for "Beam 4-16" at elevation 119 ft and found the approach acceptable. Based on staff review of the additional information and the calculations, the staff concludes that this open issue is, therefore, resolved.

3.3 Piping and Piping Supports

3.3.1 Large Bore Piping and Supports

Open Item 16: Analysis of Supports

In SSER 18 (page C.3-48) the staff stated that the results of analyses of large bore piping supports should be verified. This open item was resolved in SSER 19 (page C.3-4).

Open Item 17: Buckling Criteria

The staff identified in SSER 18 (page C.3-48) the concern that buckling criteria for linear supports, specifically the Euler buckling equation for calculating critical buckling loads for all slenderness ratios, should be evaluated and justified. In SSER 19 (page C.3-5) the staff presented its evaluation of additional information that had been provided by the licensee and concluded that the issue was resolved for fuel loading. The licensee committed to provide additional analyses for confirmation. In Board Notification 83-179 the staff presented the status, basis and schedule for the resolution of this concern as follows:

> "This item relates to an interpretation of subsection NF of the ASME Boiler and Pressure Vessel Code. Specifically, the staff does not accept the upper range of the design curve employed by the DCP to implement the Code requirements for evaluation of allowable buckling loads (compressive stress) on some pipe support members. In a response provided during a transcribed meeting on September 28, 1983, the DCP stated that the matter was largely moot because the upper range of the design curve was rarely, if ever, used due to the nature of the supports at Diablo Canyon, i.e., short, stiff members with low compressive stresses. In a submittal dated October 6, 1983, the DCP provided a sample of calculated compressive stress for 24 supports demonstrating compressive stresses well within the range consistent with the staff's interpretation of the ASME Code intent. At a staff audit in San Franscisco on October 25-26, 1983, the DCP agreed to provide the compressive stress data for approximately 400 supports (on the order of 4000 individual supports are in Unit 1) throughout Unit 1 to provide final confirmation. The progress of the staff review to date indicates that the likelihood of additional modifications is low. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore, it is acceptable to the staff to consider this matter resolved for Step 1."

The DCP has provided the information requested during the audit on October 25-26, 1983. The DCP reviewed 459, or approximately 10 percent, of the large bore pipe supports with members within the B31.1 Code jurisdictional boundary. Sixty-seven supports were found with members (T-shoes or stanchions) which were subject to compressive loading. These members were checked for buckling according to the criterion proposed by the staff (two-thirds times the critical buckling stress, where the critical buckling stress, or load, is taken as the AISC compressive load equation without the factor of safety). In all cases these members were found to satisfy the staff criterion.

In the submittal of September 9, 1983, the DCP also provided the buckling criteria for the supplementary steel members which comprise the AISC portion of the pipe supports. These members are designed in accordance with the requirements of the AISC Manual for Steel Construction, 7th Edition, including the requirement for compressive loading. The staff reviewed this information and found it acceptable. The issue on buckling criteria is, therefore, considered resolved.

Open Item 18: Additional Piping Analysis

The staff stated in SSER 18 (page C.3-48) the request that calculations for selected piping systems analyzed previously in ITR-12 and ITR-17 should be repeated with revised support configurations and current loadings to verify that piping and supports satisfy corresponding design criteria. Results of piping system reevaluation with high thermal load should be verified. In SSER 19 the staff presented its evaluation of additional information and concluded that the issue was resolved for fuel loading. In Board Notification

83-179 the staff provided the status, basis, and schedule for the resolution of this concern as follows:

> "This matter concerns a commitment to perform confirmatory analyses of two piping systems. Because this is a long-lead item, the staff required a commitment to perform the required analyses prior to approval of Step 1 in order to ensure timely completion of the task. The procedures and criteria for all piping analyses performed by the DCP were reviewed and found acceptable by the IDVP. However, consistent with current practice, following the five-plant shutdown in 1979, the staff requires a final confirmatory analysis to be performed by an independent party, in this instance the IDVP. The IDVP has initiated confirmatory analyses on two piping systems selected by the staff. Based on the IDVP results, the staff concludes that no significant modifications are likely to be required. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. It is therefore acceptable to the staff to consider this matter resolved for Step 1."

The IDVP is presently conducting an independent reevaluation of two piping systems selected by the staff. Based on current information the staff concludes that no significant modifications are likely to be required, and any which may be necessary will not likely affect any safety related system or components. The staff therefore considers this concern resolved for Step 2. Final resolution is required prior to Step 3, i.e. issuance of a full power license.

3.3.2 Small Bore Piping and Supports

Open Item 19: Analyses of Supports

In SSER 18 (page C.3-57) the staff stated that the DCP Final Report for Phase I, is unclear as to the actual extent of the seismic review of the Class I small-bore piping. This concern was resolved in SSER 19 (page C.3-5).

3.4 Equipment and Supports

3.4.1 Mechanical Equipment and Supports

Open Item 20: Equipment Qualification

The staff identified in SSER 18 (pages C.3-59 and C.3-70) the concern that not all equipment listed in Table 2.3.1 of DCP Phase 1 Final Report was seismically qualified for nozzle loads and component configurations and should be verified. In Board Notification 83-179 the staff provided the status, schedule and resolution of this concern as follows:

> The results of the DCP mechanical equipment review are listed in Table 2.3.1-1 of the DCP Phase I Final Report. Each analysis is stated to have demonstrated that the equipment is qualified to perform its function without modification for the controlling spectra and load combination. However, this table also indicated that the following equipment had not yet been qualified for nozzle loads:

(1) boric acid tank
(2) CCW heat exchanger
(3) CCW pump lube oil cooler
(4) diesel generator
(5) diesel transfer filter
(6) waste gas compressor

The DCP anticipated that this equipment or connected piping supports may be modified or that the calculated loads could be reduced by further analysis. In addition, field verification of some component configurations had not been completed. Finally, because not all final spectra had been issued, some of the calculations might have to be revised to ensure that the affected equipment was qualified.

The IDVP has reviewed the DCP approach to resolution of nozzle loads and found this approach and a sample of results to be acceptable. In a response dated September 9, 1983, the DCP reported that all nozzle load allowables for items 2, 3 and 6 had been met and that further qualification for items 1, 4 and 5 was underway. This is a typical approach to resolution of nozzle loads and, based on the revision of October 11, 1983, to the DCP Phase I Final Report, corrective actions for all remaining items are well underway. Modifications beyond those currently underway are not anticipated. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer final resolution to Step 2."

The DCP has stated in Semi-Monthly Report 51 that the seismic design review for all Class I mechanical equipment within the PG&E design scope has been completed based on nozzle loads and spectra available as of November 30, 1983. The review continues to be updated and PG&E has stated that the effort will be completed prior to Step 2. PG&E will inform the staff at that time of the completion of the modification. The staff finds this acceptable and considers this open item resolved.

Open Item 21: Valve Nozzle Stresses

The staff identified in SSER 18 (page C.3-66) the concern that stresses in extreme fibers at the interface between the valve nozzle and the pipe should be evaluated and the results be documented. In Board Notification 83-179 the staff provided the status, basis and schedule for the resolution of this concern as follows:

"The initial IDVP review of valves sampled the PG&E approach for evaluation of the portions of valve structures that support the operators (commonly the most highly stressed portion of the valve body under seismic loading) and found this aspect of the PG&E design acceptable. An additional staff requirement for valves that must function during and after a seismic event (active valves) is that the maximum stress at the valve nozzle to pipe intersection remain below the yield stress to assure elastic action in the valve body. The IDVP has subsequently verified a sample of DCP valve analyses and found the stress levels acceptable. Subject to final review and evaluation, the staff considers this matter resolved."

The staff has reviewed the information in ITR-59, Rev. 1 where valve nozzle qualification is addressed, and has found the IDVP verification of the DCP valve analyses acceptable. The staff therefore considers this open item resolved.

Open Item 22: Pump Flange Stresses

The staff stated in SSER 18 (page C.3-69) that the IDVP review was not yet complete and that the stresses in pump flanges should be verified to be within allowable limits. In Board Notification 83-179, the staff provided the basis and schedule for the resolution of this concern as follows:

> "Early IDVP reviews identified some situations where the approach to evaluation of stresses in pump flanges was not fully acceptable and recommended reevaluation. In ITR-67, Rev. 1, dated September 9, 1983, the IDVP concluded that the seismic qualification of equipment, including pumps, was performed acceptably. In the IDVP Phase I Final Report the IDVP reported that the verification sample for stresses in pump flanges showed acceptable results. Subject to confirmation that the acceptable evaluation methods have been uniformly applied with regard to pumps requiring seismic qualification, the staff considers this matter resolved."

The staff has reviewed the ITR-67, Rev. 1 and the IDVP Phase I Final Report and finds these verification efforts acceptable. The staff considers this open item resolved.

3.4.2 Heating, Ventilation and Air Conditioning Equipment

No staff concern was identified in SSER 18 in this section.

3.4.3 Electrical Equipment and Instrumentation and Supports

Open Item 23: Cable Tray Qualification

In SSER 18 (page C.3-80) the staff stated that the qualification of cable trays and interaction of trays with supports should be addressed. The resolution of this open item was presented in SSER 19 (page C.3-6).

Open Item 24 Superstrut Weld Limits

In SSER 18 (page C.3-80) the staff stated that allowable limits for welds based on field samples should be used in qualification of trays supported by superstrut. The resolution of this open item was provided in SSER 19 (page C.3-6).

3.5 Other Seismic Design Verification Topics

Open Item 25: Intake Structure Lateral Forces

The staff stated in SSER 18 (page C.3-86) that the total lateral forces, the total resistance to sliding and the factor of safety against sliding of the intake structure should be fully evaluated. In Board Notification 83-179, the staff provided the status, basis and schedule for the resolution of the concern as follows:

> "The staff concern was that the total lateral forces on the intake structure were not completely evaluated. These forces consist of lateral static and dynamic earth pressure, hydrodynamic pressures and seismic forces. The structure is keyed into the underlying rock material and sliding at the structure rock interface is not of concern. The staff is concerned about sliding occurring in possible clay seams in the rock foundation. Some additional boring data have become available that can be used to address these concerns. The IDVP will use this data in the evaluation and review the findings and report them in a revision to ITR-40. It is expected that the results of this additional investigation will show the intake structure is stable against sliding. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to defer the resolution of the factor of safety against sliding to Step 2."

The staff concern was that the IDVP had not evaluated the sliding stability of the intake structure as reported in ITR-40. The structure is keyed into the underlying bedrock and the staff was concerned about the structure sliding along either the structure-rock interface or along a possible clay seam within the bedrock. In response to the staff comments, the IDVP issued ITR-68, Rev. 0 and Rev. 1. This report addresses the geotechnical concerns stated in Section 3.5.1 of SSER 18.

The staff has completed its review of ITR-68. During an audit on October 25-26, 1983 in San Francisco the staff also reviewed three field visit memos (1972) by the Harding and Lawson Associates (HLA) geologists which document the condition of the exposed bedrock. The staff performed an independent analysis of the sliding of the intake structure and concurs with the IDVP that the shear strength of rock required to result in a minimum factor of safety of 1.1 against sliding of the intake structure is 8.3 ksf. This is only 10 percent higher than the 7.5 ksf shear strength measured in the laboratory test on a sample of weathered tuff. The field visit memos document the results of four "Airtrac" holes drilled at the bottom of the foundation excavations. These did not reveal any soft zones or voids in the bedrock beneath the foundation. Geophysical tests in these holes resulted in an average compressional wave velocity of 7000 fps. The memos describe the rock as blocky to massive, moderately hard, moderately strong tuff/shale with minor weathering. The geologists noted that some minor blasting was necessary to excavate the bedrock and that the bedrock dipped steeply to the west. In the staff's judgment, based on the above and the considerably higher strength of tuff quoted in the literature, the in situ shear strength of the bedrock is probably higher than 8.3 ksf and will result in a minimum factor of safety of at least 1.1 against sliding. Since the bedrock dipped steeply to the west and the borings did not reveal any soft zones, sliding along a critically oriented clay seam within the bedrock is not a likely mode of failure. The staff therefore concurs with the IDVP that the intake structure is safe against sliding. The staff has requested RLCA to provide prior to Mode 2 (criticality) certain confirmatory information regarding ITR-68 Rev. 1 previously discussed with the staff.

The staff has reviewed the stability of the intake structure against bearing and overturning modes of failure and concurs with the IDVP that the intake structure foundation is stable against bearing and overturning. The IDVP has used the results of the structural analysis presented in ITR-58 in its sliding analysis of the structure. The staff concludes that the intake structure is stable under Hosgri loadings and satisfies the licensing criteria. The staff considers this open item resolved.

3.6 Brookhaven Analysis

Open Item 26: Buried Diesel Fuel Oil Tank

The staff stated in SSER 18 (page C.3-99) that additional analyses of buried diesel fuel oil tanks should be performed. In view of the BNL results, PG&E committed to perform the following further investigations:

(1) refined mesh computer runs will be made using YY section properties.
(2) runs with and without deconvolution will be made.
(3) a partially filled tank case will be examined.
(4) YY section properties in conjunction with the static analysis will be carefully examined.

In Board Notification 83-179 the staff provided the status, basis and schedule for the resolution of the concern as follows:

> "The staff consultant, Brookhaven National Laboratory (BNL), performed an independent analysis of the buried diesel fuel oil storage tank. The results of this analysis were compared to the PG&E analysis and showed that some deficiencies existed in the PG&E analysis. PG&E committed to reanalyzying the tanks using different models. The results of the later PG&E analysis showed the tanks can withstand the Hosgri event. The IDVP has verified that the PG&E reanalysis has addressed the deficiencies identified in the BNL analysis. The staff is currently reviewing this matter in further detail. The progress to date for the resolution of the staff concern indicates that the possibility of additional significant modifications to the structure is remote. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 or 6. Therefore it is acceptable to the staff to defer qualification of the buried diesel fuel oil tanks to Step 2."

The staff has completed its review. The additional analyses committed by the DCP were performed and documented in a HLA report dated August 19, 1983. Based on this report, refined models were used and several parameteric studies were carried out to assess the significance of various parameters on the safety of the tanks. Lumped mass representation of the diesel fuel oil was used. Results with and without deconvolution were obtained. The effect of the partially filled case was examined (i.e. tank was considered to be 50, 90 and 100 percent full). Several modes were used in this study to represent different sections along the tank axis with different section properties. The stresses for each of these cases were shown to be within the allowable limits. The staff concludes that the 1983 HLA analysis has corrected the deficiencies that existed in the previous analysis (1978 to 1982) and is acceptable. This open item is therefore resolved.

4 NONSEISMIC DESIGN VERIFICATION EFFORT

4.1 Introduction

In this section of SSER 18 the staff had identified 15 Followup Items which subsequently were listed in Table C.8.3 of SSER 19. In general, these items require some closeout action by the staff regarding actions taken by PG&E to resolve concerns originally identified by the IDVP. The items are of a confirmatory nature and include submittal of documentation, testing and as-built verification. Many items require an update (amendment) of the Final Safety Analysis Report (FSAR). While PG&E will update the entire FSAR during 1984, proposed revisions for specific items in this section were submitted by letter of December 6, 1983. These changes will be included in the FSAR update. This section also includes the resolution of 4 Open Items identified in SSER 18.

4.2 Initial Sample

4.2.1 Verification of Mechanical/Nuclear Design

Followup Item 1: AFWS Runout Control System Test

The staff stated in SSER 18 (page C.4-7) that PG&E had changed the pump discharge pressure setpoints and committed to perform a startup test of the runout control system to confirm dynamic stability. In Board Notification 83-179 the staff provided the basis and schedule for the test as follows:

> "An analysis performed by the IDVP indicated that the pressure control setpoints for the AFWS runout control system may not be low enough to permit minimum required flow to the steam generators when only one motor-driven AFW pump is operating. PG&E changed the low pump discharge pressure setpoints and committed to perform a startup test of the runout control system to confirm dynamic stability. The IDVP review of the new setpoints and startup test commitment indicated that the proposed resolution was acceptable. The staff concurred with this resolution in SSER 18. The test will be completed prior to entering Mode 3. The AFWS is not required to be operable by plant technical specifications prior to entering Mode 3. Therefore it is acceptable to the staff to defer the testing to Step 2."

By letter dated December 6, 1983 the licensee stated that a test of the AFWS runout control system and steam generator level control valves will be conducted during the startup testing of the AFWS at hot standby conditions (Mode 3) in order to verify proper component operability. The licensee will provide the results of this test prior to Mode 2 (criticality). The reactor will not be critical prior to test completion and therefore no safety concern exists during Modes 4 and 3. Based on staff review of the information, the staff finds the response acceptable and considers this item resolved.

Followup Item 2: AFWS Drawing Update for Steam Trap

The staff stated in SSER 18 (page C.4-5) that in response to a drawing discrepancy PG&E had indicated that the design drawings would be revised. In Board Notification 83-179 the staff provided the basis and schedule for closeout of this item as follows:

> "The IDVP performed a field walkdown of the AFWS to verify compliance of the as-built installation with the design documents. The as-built installation was confirmed to meet design drawings except that a steam trap on the turbine-driven AFW pump steam supply line was not provided. PG&E indicated that the design drawings would be revised to delete the steam trap on the steam supply line because satisfactory testing of the turbine-driven pump was completed without the need for the trap. The IDVP confirmed that the actual AFWS installation was acceptable and no technical concern existed. The staff concurred with the above resolution in SSER 18. The staff will verify incorporation of the drawing change and confirm as-built drawings prior to entering Mode 3. The AFWS is not required to be operable by plant technical specifications prior to entering Mode 3. Therefore it is acceptable to the staff to defer as-built confirmation to Step 2."

By letter dated December 6, 1983, the licensee submitted revised drawings indicating the as-built condition of the AFWS. Based on staff review of these drawings, this item is considered resolved.

4.2.2 Electrical Design

Followup Item 3: AFWS Electrical Circuit Coding

The staff identified in SSER 18 (page C.4-8) a concern regarding the AFWS electrical circuit color coding and separation. In Board Notification 83-179 the status, basis and schedule for closeout was summarized as follows:

> "Discrepancies regarding the as-built conditions for separation and color coding of AFWS electrical circuits was identified by the IDVP. PG&E committed to revise FSAR Section 8.3.3 to reflect acceptability of as-built conditions regarding separation and color coding. The staff concluded in SSER 18 that these concerns have been acceptably resolved and that plant modifications or additional verification is not required. FSAR revisions concerning AFWS electrical circuit separation and color coding will be submitted by PG&E prior to entering Mode 3. The AFWS is not required to be operable by plant technical specifications prior to entering Mode 3. Therefore it is acceptable to the staff to defer the FSAR revision to Step 2."

Regarding the concern of separation of electrical instrumentation circuits, the licensee provided, by letter dated December 6, 1983, revisions to Section 8.3.3 of Amendment 58 to the FSAR (Pages 8.3-19 through 8.3-20a). According to the revision, exposed wiring at end connections for instrument circuits may be separated by less than 5 inches of air space with no barrier separation. The staff interpreted the revisions to mean that bare, uninsulated instrumentation circuit wire can be touching mutually redundant bare, uninsulated circuit wire.

The licensee, by letter of December 12, 1983, provided a further revision to the FSAR which states that exposed terminals of low energy instrumentation devices are separated by at least one inch. Based on the low energy of the circuits and the rigid or fixed location of the exposed terminal connection, the staff concludes that one inch of separation provides sufficient independence and is, therefore, acceptable.

Section 8.3.3 of the FSAR has also been revised to indicate that unit cases are relied upon for adequate separation for low-energy devices. Based on this revision and further clarification provided by the licensee, the staff concludes that each of the mutually redundant instrumentation devices has its own unit case or surrounding enclosure and that separation between redundant devices is provided by a barrier of twice the thickness of the insulating material. This separation provides sufficient independence and is, therefore, acceptable.

In addition, Section 8.3.3 of the FSAR has been revised to indicate that any of nine methods defined in the FSAR for separation of mutually redundant circuits in boards and panels also applies to devices located on the boards and panels. The staff concludes that each of the nine methods provides sufficient independence and is, therefore, acceptable. This item is considered resolved.

With respect to color coding of electrical circuits, the licensee submitted, by letter of December 6, 1983 revisions to Section 8.3.3 of Amendment 36 to the FSAR (Pages 8.3-28b and 8.3-29) to indicate that non Class 1E control, indication, and annunciation circuits that are routed in Class 1E raceways are color coded and installed as Class 1E. The revision meets current review guidelines defined in the Standard Review Plan (SRP) for this aspect and is, therefore, acceptable.

The licensee also revised Section 8.3.3 of the FSAR to indicate that non Class 1E control, indication, and annunciation circuits routed in Class 1E raceways, (1) are designed with sufficient isolation to ensure that a single failure does not propagate to the mutually redundant device and, (2) are colored consistent with the safety-related device, train, or circuit so that the color may not reflect the color of their electric power source. Based on the staff review of this revision and clarification provided by the licensee, the staff concludes that these circuits have been routed with sufficient independence in accordance with single failure requirements and are, therefore, acceptable. This item is considered resolved.

4.2.3 Instrumentation and Control Design

<u>Open Item 27: AFWS Isolation Valve Classification</u>

In SSER 18 (page C.4-11) the staff stated that the control circuits for isolation valves FCV-37 and FCV-38 in the steam supply line for the turbine-driven AFWS pump should be classified as safety-related. The resolution of this concern was provided in SSER 19 (page C.4-1).

<u>Open Item 28: Steam Generator Blowdown Isolation Circuitry</u>

In SSER 18 (page C.4-12) the staff identified a concern that the auxiliary relay for automatic closure of redundant steam generator blowdown isolation valves should meet the applicable Westinghouse requirements. The resolution of this concern was provided in SSER 19 (page C.4-1).

Followup Item 4: Equipment Qualification Report Update

In SSER 18 (page C.4-12) the staff stated that PG&E will correct errors in the qualification report tables regarding flow transmitters and flow control valves in the AFWS. In Board Notification 83-179 the staff provided the basis and schedule for the resolution as follows:

> "The IDVP review of the environmental qualification of AFWS equipment indicated that a flow transmitter and flow control valve, which are exposed to a harsh environment resulting from a high energy line break, were not listed as located in harsh environments. PG&E responded by noting that the flow transmitter was identified under a different identification number and that the vendor provided justification for interim operation pending completion of the environmental qualification. The flow control valve was conditionally qualified, subject to an ongoing maintenance surveillance program, but was erroneously listed as a component not subject to a harsh environment. PG&E will correct errors in the qualification report tables. The IDVP withdrew its concern on this matter. The staff concurred with the IDVP resolution of this matter in SSER 18. Environmental qualification (EQ) documentation for the AFWS will be revised and submitted by PG&E prior to entering Mode 3. The AFWS is not required to be operable by plant technical specifications prior to entering Mode 3. Therefore, it is acceptable to the staff to defer EQ documentation update to Step 2."

By letter dated December 6, 1983, PG&E stated that the format of the equipment qualification tables has been completely revised so that there is one table which identifies the qualification documentation for all Class 1E equipment. This new table, identified as PG&E Drawing 050909, supersedes the previous table and correctly identifies the auxiliary feedwater flow transmitter and control valves and the qualification files that apply. The staff finds that the above information from PG&E adequately addresses this item. Therefore this followup item is satisfactorily resolved.

Followup Item 5: Motor Capacitor Qualification Analysis

In SSER 18 (page C.4-12) the staff stated that PG&E will conduct analyses to determine the qualified life of the motor capacitor for steam generator control valves. In Board Notification 83-179 the staff provided the basis and schedule for resolution as follows:

> "The IDVP review of the environmental qualification of AFWS equipment indicated that steam generator level control valves may not be qualified for harsh environments resulting from high energy line breaks as required because the motor capacitor qualification report was not yet complete. The qualification report did include justification for interim operation with replacement of this component following 20,000 hours of operation. PG&E indicated that an analysis to determine the qualified life of this components is being conducted. The

IDVP concluded that the PG&E response resolved this concern. The staff concurred with the conclusions of the IDVP on this matter in SSER 18. Submittal of the analysis regarding motor capacitor qualification life is required prior to entering Mode 3. The steam generator level control valves (on the AFW lines) are not required to be operable by plant technical specifications prior to entering Mode 3. Therefore it is acceptable to the staff to defer completion of the staff review of the analyses to Step 2. In any event, justification for interim operation has previously been submitted by PG&E."

During an earlier audit (August 21 to September 4, 1981) the staff had identified the aging qualification of the steam generator level control valves as a concern; however, an interim aging qualification for two years was found as adequately justified.

The licensee stated in a letter of November 28, 1983, that an analysis has been completed that demonstrates the qualified life. The analysis is contained in PG&E Qualification File IH-14. In a letter of December 6, 1983, the licensee identified specific references on which the qualification is based. NRC Region V staff reviewed those files during an audit on December 15 and 16, 1983. The issue was further discussed by the staff with the licensee on December 22, 1983. The licensee will provide the staff with a summary of the analysis performed regarding the qualified life of the motor capacitor and of any other component of the steam generator level control valves. This confirmatory information will be provided by PG&E prior to Mode 4 (hot shutdown). Based on the staff review and evaluation of the above information, including the earlier audit, the staff concludes that this followup item is resolved with respect to Step 2 (low power operation). Final resolution will be required prior to Step 3 i.e., full power operation.

4.2.4 High Energy Line Break and Internally Generated Missiles

Followup Item 13: ANS 58.2 Jet Impingement Temperature Calculation Methodology

In SSER 18 (page C.4-14 and page C.4-16) the staff stated that PG&E will revise the FSAR to incorporate the use of ANS 58.2 jet impingement temperature calculational method where applicable. In Board Notification 83-179 the staff provided the following basis and schedule for resolution of this issue:

"The IDVP review of high energy pipe crack concerns indicated that jet impingement may result in temperatures in excess of the qualification value for certain AFWS and CRVPS components. PG&E utilized the ANS 58.2 jet impingement temperature calculation method in lieu of that identified in the FSAR to verify that the qualification temperature was not exceeded. PG&E committed to revise the FSAR to incorporate use of ANS 58.2 jet impingement temperature calculational method. The IDVP reviewed this method and verified that it provides acceptable results. The staff concurred in this resolution in SSER 18. FSAR revisions confirming use of ANS 58.2 jet impingement temperature calculational method will be submitted by PG&E prior to exceeding 140°F. No environmental qualification concerns are present at low temperature. Therefore it is acceptable to the staff to defer FSAR revisions to Step 2."

By letter dated December 6, 1983 the licensee submitted changes to FSAR Section 3.6.4.3 (page 3.6-19) which states that ANS 58.2 methodology will be used in addition to the NSC method previously identified for determining jet impingement qualification temperatures for equipment. This issue is also related to Followup Item 7. Based on the above changes to the FSAR, the staff considers this issue resolved.

4.2.5 Effects of High Energy Line Cracks and Moderate Energy Line Breaks

<u>Followup Item 6: Break in Steam Line to Turbine-Driven AFWS Pump</u>

In SSER 18 (page C.4-16) the staff stated that PG&E will amend the FSAR to indicate that pipe breaks are not postulated in the steam supply line to the turbine-driven pump of the AFWS. In Board Notification 83-179 the staff provided the following basis and schedule for the resolution of this issue:

> "The IDVP review of high energy line cracks indicated that certain AFWS components were exposed to a postulated break in the steam supply line to the turbine-driven AFWS pump. PG&E reevaluated the high energy line crack analysis against the FSAR commitments (Giambusso letter dated December 18, 1972). It was determined that the line established in the IDVP analysis as a source affecting the motor-driven AFW pumps and pressure transmitters (located on the steam supply line to the turbine-driven AFW pump downstream of the flow control valve) was not subject to cracks because it is not pressurized during any normal plant operating conditions, including startup and shutdown. PG&E committed to revise the FSAR to indicate the above point. The IDVP agreed with the above resolution in SSER 18. FSAR revisions confirming AFWS turbine steam supply line pipe break resolution will be submitted by PG&E prior to entering Mode 3. The AFWS is not required to be operable prior to entering Mode 3. Therefore it is acceptable to the staff to defer the FSAR revision to Step 2."

By letter dated December 6, 1983, the licensee submitted changes to FSAR Section 3.6.1.2 (page 3.6-5) which state that pipe breaks are not postulated in the AFWS turbine steam supply line downstream of the steam supply valve because the line is not pressurized during normal plant operating conditions, including startup and shutdown. Based on the above change to the FSAR, this item is considered resolved.

<u>Followup Item 7: Jet Impingement Equipment Qualification Temperature</u>

In SSER 18 (page C.4-16) the staff stated that PG&E will amend the FSAR to include all changes for equipment qualification (CRVPS and AFWS) that resulted from the reanalysis of pipe break environments outside containment. In Board Notification 83-179 the Staff provided the following basis and schedule for closeout of this issue:

> "The IDVP review of high energy line cracks indicated that certain AFWS and CRVPS components may not have been qualified for the resulting environments. PG&E performed a reanalysis of the blowdown jet temperature from the postulated high energy line crack source

affecting the AFWS level valves using the ANS Standard 58.2 methodology in lieu of the NSC method documented in the FSAR. The results of this reanalysis showed a jet temperature below the qualification temperature for the valves. PG&E committed to revise the FSAR to incorporate this reanalysis. Additionally, for cables/wires and splices in the AFWS and CRVPS identified as targets by the IDVP, PG&E responded by providing documentation that indicated that the affected cables/wires and splices were environmentally qualified for the resulting high energy line crack blowdown jet environment and further committed to update environmental qualification documentation.

The IDVP concurred with the above resolutions. The staff concurred with the resolution in SSER 18. FSAR revisions confirming satisfactory resolution of jet impingement temperature methodology and cable/wire equipment qualification documentation will be submitted by PG&E prior to exceeding 140°F (Modes 5 and 6). No environmental qualification concerns are present at such a low temperature since a harsh environment cannot result. Therefore it is acceptable to the staff to defer documentation to Step 2. (Also see items 13 and 14 below.)"

By letter dated December 6, 1983, the licensee submitted changes to FSAR Section 3.6.4.3 (page 3.6-18) which state that ANS 58.2 methodology will be used in addition to the NSC methodology previously identified for determining qualification jet impingement temperatures for equipment. This issue is also related to Followup Item 14 regarding cable/wire jet impingement temperature qualification and to Followup Item 13. Based on the above changes to the FSAR, the staff considers this item resolved.

Followup Item 14: Environmental Qualification of Cables and Wires

In SSER 18 (page C.4-16) the staff stated that PG&E will revise equipment qualification documentation to include AFWS cable/wire other than that previously identified. In Board Notification 83-179 the staff provided the following basis and schedule for closeout of this item:

"The IDVP review of high energy pipe crack concerns indicated that cable/wire other than that previously identified as environmentally qualified for use in the AFWS was utilized and was subject to high temperature jet impingement. PG&E provided documentation which indicated that the cable/wire was qualified to the resulting jet impingement temperature. PG&E committed to revise the environmental qualification documentation. The IDVP reviewed the documentation and concurred with the resolution. The staff concurred with this resolution in SSER 18. Equipment environmental qualification documentation confirming satisfactory qualification of cables/wires will be submitted by PG&E prior to exceeding 140°F. No environmental qualification concerns are present at low temperature. Therefore it is acceptable to the staff to defer documentation to Step 2."

By letters of November 28, 1983, December 6, 1983, and December 12, 1983, PG&E provided additional information regarding the environmental qualification of cable/wires. PG&E stated that the following four Class 1E cables/wires are installed outside containment and have been environmentally qualified:

	Cable/Wire	Qualification Document
1.	Raychem Flametrol	Test Report EM-1030; September 24, 1974
2.	Okonite EPR/Hypalon	Okonite Letter Report; October 14, 1974
3.	Okonite XLPE	Engineering Report 367-A; January 7, 1983
4.	Rockbestos XLPE	Test Report S.O. 24408-5; March 3, 1983

No other types of Class 1E cables have been installed outside containment which potentially can be subjected to high energy line breaks. These four types of cables have been tested to 540°F with 480 Vac between lines for more than 48 hours. All four types passed the test. The staff reviewed the first two qualification reports and concluded that the Raychem Flametrol cable had been qualified as stated; however, the Okonite EPR/Hypalon cable had been demonstrated to be qualified for only 24 hours. Based on subsequent discussions with the licensee, including an audit of documentation by the staff at the PG&E offices in San Francisco on December 19 and 20, 1983 the staff determined:

1. The cables are enclosed in conduit and therefore, are not subject to direct jet impingement;

2. The consequences of jet impingement on those conduits that are essential targets are currently being reviewed by the staff under the same effort discussed under open item 29 in Section 4.3.5;

3 The qualification temperature of 540°F is based on the maximum temperature of the steam in the pipe prior to the postulated break; and

4. The cables are qualified for 24 hours at a temperature of 540°F. The operator will identify and isolate the break within less than 2 hours.

The licensee will submit the above information by letter prior to Mode 2 (criticality) Based on this commitment and based on the staff review and evaluation of the information during the audit, the staff concludes that this followup item is resolved.

Followup Item 15: Protection for CRVPS

The staff stated in SSER 18 (page C.4-17) that PG&E will revise the FSAR to incorporate results of moderate energy line break analyses on the CRVPS. In Board Notification 83-179 the staff provided the following basis and schedule for closeout of this item:

"The IDVP review of moderate energy line breaks indicated that PG&E had failed to meet its licensing commitment by not including the CRVPS in the original moderate energy line break analysis. PG&E provided a subsequent analysis indicating that only one CRVPS electrical train is affected by the postulated break identified by the IDVP. When combined with a single failure in the redundant electrical train, a loss of the CRVPS would occur, resulting in degradation of control room habitability. However, safe shutdown can be provided from the remote shutdown panel in the event the control room becomes uninhabitable. The IDVP concurred with this analysis. The staff also concurred with this resolution in SSER 18. FSAR revisions confirming satisfactory moderate energy line break protection for the CRVPS will be submitted by PG&E prior to initial criticality. Remote shutdown capability is provided in the event of loss of the CRVPS due to a moderate energy line break as indicated above. Moreover, because no fission product inventory is present, control room habitability is not of concern and offsite release consequences are not present. Therefore it is acceptable to the staff to defer the FSAR revision to Step 2."

By letters dated December 6 and 12, 1983, the licensee submitted revisions to the FSAR which state that moderate energy line break protection for the CRVPS is not required since safe shutdown can be achieved from the hot shutdown panel should the CRVPS be lost and the control room become uninhabitable. Based on the staff review and evaluation of the additional information, this item is considered resolved.

Followup Item 8: Protective Shields for CRVPS Valves

The staff stated in SSER 18 (page C.4-17) that PG&E will revise the FSAR licensing commitment regarding the need for protective shields for AFWS components (valves) against effects of moderate energy line breaks. In Board Notification 83-179 this item was closed out as follows:

"The IDVP review of moderate energy line breaks indicated that two AFWS valves were not provided with protective shields as documented in a licensing commitment. PG&E indicated that the flow control valves (suction supply valves from the alternate AFWS water source, the raw water storage reservoir) are not required to operate to ensure AFWS safety function following the postulated moderate energy line break; therefore, they are not required to be protected from the pipe break effects. PG&E committed to revise the licensing commitment to delete the need for protective shields for these valves. The IDVP agreed with this response. The staff concurred with the resolution in SSER 18. PG&E letter dated June 15, 1983, documents deletion of the protective shields for the long-term water supply valves for the AFWS. Therefore this concern has been closed out."

4.2.6 Fire Protection

No concern was identified in this section in SSER 18.

4.2.7 Radiation Environmental Qualification

No concern was identified in this section in SSER 18.

4.2.8 Pressure and Temperature Environmental Analyses

No concern was identified in this section in SSER 18.

4.3 Additional Verification

4.3.1 Redundancy of Equipment and Power Supplies in Shared Safety-Related Systems

No concern was identified in this section in SSER 18.

4.3.2 Selection of System Design Pressure and Temperature and Differential Pressure Across Power Operated Valves

Followup Item 9: Modification to AFWS

The staff stated in SSER 18 (page C.4-26) that it will confirm that any modifications required in safety-related systems with respect to pressure/temperature rating and power-operated valve operability are implemented. In Board Notification 83-179 the staff provided the following basis and schedule for closing out this item:

> "As a result of concerns identified by the IDVP regarding compliance with applicable design codes for the selection of the auxiliary feedwater system (AFWS) design pressure, isolation of low-pressure portions of the system from high-pressure portions, and the specification of low differential pressure for the motor-operated steam supply valves to the AFWS turbine-driven pump, the IDVP determined that additional sampling in these areas was required. PG&E undertook a review of the above concerns for all safety-related systems within their design scope. This generic review resulted in several modifications to safety-related systems as documented in PG&E letter dated October 7, 1983, which have been completed. The staff will verify that required modifications documented in PG&E letter dated October 7, 1983, are in place prior to Step 2. Prior to Step 2, the plant will not be in an operating condition which would result in pressure/temperature rating and power-operated valve operability considerations. Therefore it is acceptable to the staff to defer as-built verification to Step 2."

By letter dated December 6, 1983 the licensee confirmed that the modifications to the safety-related fluid systems as a result of the new determinations for the pressure/temperature rating, the differential pressure across power operated valves and the high/low pressure isolation provisions have been completed with the exception of the AFWS pump turbine overspeed setpoint change. The setpoint change will be accomplished when steam is available from reactor coolant pump heat during Mode 4 operations. The licensee will inform the staff of the change prior to Mode 2 (criticality). This is acceptable to the staff. This item is considered resolved.

4.3.3 Environmental Consequences of Postulated Pipe Ruptures Outside Containment

Followup Item 10: Verification of Assumptions for Pressure/Temperature Calculations

In SSER 18 (page C.4-27) the staff stated that PG&E will verify the assumptions regarding closing/opening of doors and the operation of ventilation systems in the continuing pressure/temperature environmental reanalysis. In Board Notification 83-179 the staff provided the following basis and schedule for closing out this item:

"As a result of the IDVP concerns regarding the method for establishing pressure/temperature environments following postulated high energy pipe breaks outside containment, PG&E undertook a reanalysis in this area. Specific concerns identified by the IDVP were with respect to assumptions regarding door positions and ventilation system operation. PG&E will provide verification of the assumptions regarding the above aspects of the reanalysis and will submit the reanalysis results including assumptions prior to exceeding 140°F. No environmental qualification concerns are present at low temperature. Therefore, it is acceptable to the staff to defer verification of the assumptions to Step 2."

By letter dated December 6, 1983, the licensee submitted information regarding the assumptions in the reanalysis of pressure/temperature environments following postulated high energy pipe breaks outside containment. The response verified that doors assumed to remain closed were designed for the resulting pressure. Replacement doors were required in order to assure the validity of the analysis for area GE/GW of the auxiliary building. The licensee also verified that failure pressures for doors in other plant areas were applied in the subcompartment pressurization analyses. In addition, the licensee determined that assumptions regarding ventilation system operation would not produce enhanced results as was originally felt but rather would reduce conservatism in the resulting environments. Therefore, in order to maintain the conservatism, the ventilation systems were not modeled in the reanalyses. The above satisfactorily confirms the pressure/temperature reanalyses assumptions. This item is considered resolved.

<u>Followup Item 11: Pressure/Temperature Reanalysis Modifications and Documentation</u>

In SSER 18 (page C.4-27) the staff stated that PG&E will make modifications and provide revised documentation as necessary based on results of pressure/temperature environmental reanalyses. In Board Notification 83-179 the staff provided the following basis and schedule for closing out this item:

"As a result of the IDVP concerns regarding the method for establishing pressure/temperature environments following postulated high energy pipe breaks outside containment, PG&E undertook a reanalysis in this area. The IDVP review of the resulting pressure and temperature transient conditions determined that the reanalysis methodology for the remaining auxiliary building areas was consistent with that used in areas GE and GW and in the turbine building. PG&E indiciated that results obtained are conservative for the break compartment. PG&E has committed to make any modifications necessary as a result of this reanalysis and provide revised documentation of this work. The IDVP concluded that the reanalyses satisfactorily resolved the IDVP concerns. Because of this conclusion, the IDVP determined that a further verification of the PG&E continuing effort in the selection of pressure and temperature conditions and associated environmental qualification of safety-related equipment was not necessary. PG&E will submit the results of the pressure/temperature environmental reanalysis and complete necessary modifications or provide justification for interim operation prior to exceeding 140°F. Any modifications required would be outside containment and would be expected to be of a minor nature. No environmental qualification concerns are present at low temperature. Therefore it is acceptable to the staff to defer completion of modifications to Step 2."

By letter dated December 6, 1983, the licensee submitted information regarding the results of the reanalyses of pressure/temperature transient environments resulting from postulated pipe breaks outside containment. The licensee has indicated that a number of modifications are required in order to assure that the resulting conditions remain within the equipment qualification envelope. By letter dated December 13, 1983 the licensee stated that these modifications will be completed prior to plant heat up (i.e. Mode 4 and Mode 3) with the exception of the installation of redundant Class 1E isolation controls and instrumentation for the CVCS letdown line and auxiliary steam line. The licensee further stated that prior to Mode 4 this modification will either be completed or justification for a later completion will be provided. The modification will be completed prior to Mode 2 (criticality). The reactor will not be critical prior to completion of this modification and therefore no safety concern exists during Modes 4 and 3. On the basis of the above commitment and staff requirement this item is considered resolved.

Followup Item 12: Confirmation of Environmental Qualification Documentation

In SSER 18 (page C.4-27) the staff stated that it will evaluate the results of the PG&E reanalysis with respect to assuring environmental qualification of equipment. In SSER 19 the staff provided the following basis and schedule for closing out this item:

— "As a result of the IDVP concerns regarding the method for establishing pressure/temperature environments following postulated high energy pipe breaks outside containment, PG&E undertook a reanalysis in this area. Included in the PG&E effort is a verification of environmental qualification of equipment to the environments resulting from the reanalysis. The staff will confirm satisfactory environmental qualification (EQ) has been provided to the reanalyzed environments (see item 11 above) or acceptable interim operation justification has been provided prior to exceeding 140°F. No environmental qualification concerns are present at low temperature. Therefore, it is acceptable to the staff to defer EQ confirmation to Step 2."

The staff has reviewed and found acceptable, the method of reanalysis used by PG&E to establish pressure/temperature environments following postulated high energy pipe breaks outside containment as previously indicated in SSER 18. The licensee provided additional information on this item by letters dated December 6 and 12, 1983. PG&E has stated that the reanalyzed environments are documented in Design Criteria Memorandum DCM M-73. In addition, PG&E has stated that the environmental qualification files were reviewed to ensure that the listed qualification temperatures for each device which would be subjected to the reanalyzed environments were greater than the temperatures listed in DCM M-73. For simplicity, the worst case temperature from DCM M-73 for any device was compared with the qualification temperature given in the file for that device. All devices subjected to the reanalyzed environments were found to be qualified to operate in the worst case environment in which their operation is required. Further, the devices are qualified to the revised environment within the margins required in NUREG-0588.

The staff conducted an audit of the PG&E environmental qualification files on December 19 and 20, 1983 in San Francisco to verify that the licensee had performed the necessary review and evaluation to ensure that the equipment is qualified to the reanalyzed environments. Although specific documentation was not available in the files the licensee stated that such an evaluation had been performed and the results will be included in the files prior to Mode 2 (criticality). The staff has determined, based on its audit of the licensee's equipment qualification files, that the equipment is qualified for the reanalyzed environments. Based on the above information, the results of the staff audit, and the PG&E commitment the staff concludes that this followup item has been resolved.

4.3.4 Circuit Separation and Single Failure

No concern was identified in this section in SSER 18.

4.3.5 Jet Impingement Effects of Postulated Pipe Breaks Inside Containment

Open Item 29: Jet Impingement Loads on Piping Inside Containment

The staff stated in SSER 18 (page C.4-29) that the review of jet impingement effects by the DCP and the IDVP had as yet not been completed and that consideration of jet impingement loads in design and qualification of all safety-related piping and equipment should be clearly demonstrated. In SSER 19 (page C.4-2) the staff provided an update of the ongoing effort and stated that the staff will complete the effort prior to full power authorization. In Board Notification 83-179 the staff provided the status, basis and schedule for the resolution of this concern as follows:

> "As noted in SER Supplement No. 19, the staff concluded that the licensee has met the FSAR commitment regarding the consideration of jet impingement loads inside containment, confirming the basis upon which the operating license was originally granted. Under the contemporary staff practice, aspects of jet impingement analyses that were judgmental for plants of the Diablo Canyon era are required to be demonstrated by deterministic analyses. To provide the basis for a jet impingement evaluation consistent with current practice, the DCP has completed a pipe break and jet target evaluation, and this effort has been reviewed and found acceptable to current standards by the IDVP. Based on this source and target evaluation, certain piping and structural members that could be subjected to jet loading, in the unlikely event that a large pipe rupture occurred inside containment, are currently being evaluated by analysis to determine what, if any, additional protection might be required to fully meet current requirements.
>
> In consideration of the possible impact on construction efforts, this item was upgraded from Step 2 as shown in Table C.8.1 of SER Supplement No. 18 to Step 1 as shown in SER Supplement No. 19. The process of the jet impingement evaluation discussed above is sufficient to demonstrate that the licensing basis for the Diablo Canyon Plant has been met and that significant modifications to fully meet current jet impingement protection requirements are unlikely. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to consider this matter resolved for Step 1."

The DCP provided additional information at a meeting on December 6, 1983 regarding the current status of the ongoing evaluation of essential safety-related targets subjected to jet impingement loads. Both the DCP and Westinghouse are conducting these evaluations, which are intended to supplement the information provided by the DCP in the submittal of October 12, 1983. This additional effort includes piping and supports, mechanical and electrical equipment, and conduits and is scheduled to be completed by January 1984. The licensee will inform the staff of the completion prior to Mode 2 (criticality). Any modifications which may be necessary will not likely affect system or components needed for criticality or low power testing. Therefore it is acceptable to the staff to consider this matter resolved for Step 2. This issue must be fully resolved prior to Step 3, i.e. prior to full power authorization (Mode 1).

4.3.6 Rupture Restraints

Open Item 30: Rupture Restraints Inside and Outside Containment

The staff stated in SSER 18 (page C.4-31) that it should be clearly indicated that rupture restraints inside and outside containment have been properly designed and installed. In SSER 19 (page C.4-3) the staff stated that the concern was resolved and that the design of the crushable bumpers would be audited prior to Step 2. In Board Notification 83-179, the staff provided the status, basis and schedule for the resolution of this concern as follows:

> "As noted in SER Supplement No. 19, the matters remaining for final closure of this item related to a staff audit of the test data and design criteria employed for those pipe whip restraints inside containment that contain crushable bumpers. Restraints of the type under consideration are required only in the unlikely event of a complete rupture of a pipe containing high energy fluid and are, therefore, not required during Modes 5 and 6, since all systems inside containment are essentially unpressurized. Further, the IDVP and staff review of this matter have progressed to the point where the likelihood of significant additional modifications is low. Any modifications which may be necessary will not likely affect systems or components needed for fuel load or otherwise interfere with activities associated with Modes 5 and 6. Therefore it is acceptable to the staff to consider this issue resolved for Step 1."

The design of the crushable bumpers was reviewed during an audit of the DCP files in San Francisco, California, on October 26, 1983. The staff examined the DCP Design Criteria Memorandum DCM-64, "Design of Rupture Restraints Inside Containment" and determined that the stated criteria for the design of the crushable bumpers is acceptable. In addition, the experimental data on which these criteria and their application are based was examined and found acceptable.

During the audit a· additional concern was raised regarding the acceptance criteria for compression members of the frame type rupture restraints which contain the crushable bumpers. The DCP provided during a meeting on December 6, 1983, extensive supplementary information which demonstrated that the criteria for compression members stated by the DCP conforms to the accepted industry-wide practices for the design of rupture restraints. The licensee will cleary identify or submit this information prior to Mode 2 (criticality). The staff has examined and accepted this information, and therefore, considers the matter of rupture restraint design resolved.

5 SUMMARY AND CONCLUSIONS

SSER 20 presents the staff evaluation of the Diablo Canyon Unit 1 design verification effort with respect to Step 2 of the three-step licensing process, which is the complete reinstatement of the authority granted by Operating License No. DPR-76 to conduct all activities up to and including operation of Diablo Canyon Unit 1 at a power level of 5 percent of rated power. In particular, this SSER 20 presents the staff safety evaluation of those concerns related to Step 2 which previously had been identified in SSER 18 and listed in SSER 19 as "open items" and "follow up items." Board Notification 83-179 of November 9, 1983 provided the status, basis and schedule for the resolution of these concerns.

The activities currently authorized by Step 1 include Mode 6 (refueling) and Mode 5 (cold shutdown) with a maximum 200°F for the average coolant temperature. The activities under Step 2 include Mode 4 (hot shutdown), Mode 3 (hot standby), and Mode 2 (startup). Modes 4 and 3 are with the reactor in a subcritical condition.

The staff has completed its review and evaluation of the concerns and concludes that 27 of the 31 open items have been completely resolved to the satisfaction of the staff. The remaining four open items have been satisfactorily resolved for Step 2 and the staff has concluded that final resolution can be deferred but must be completed prior to Step 3 (full power authorization). The remaining four open items are

- OI 2: 20 Hertz Cutoff Frequency for Containment Annulus Structure
- OI 11: Turbine Building Roof Truss Modeling
- OI 18: Additional Large Bore Piping Analyses
- OI 29: Jet Impingement Loads on Piping Inside Containment

The licensee has committed to perform additional analyses and evaluation for these items which will be completed prior to Step 3. The staff will evaluate the results and report its conclusions on these matters at the time of the full power licensing decision.

Regarding the 15 followup items, the staff concludes that 11 items have been completely resolved. In most cases the licensee has provided revisions to the FSAR that will be included in a forthcoming amendment, currently scheduled for September 1984. The remaining four followup items are:

- FI 1: AFWS Runout Control System Test
- FI 5: Motor Capacitor Qualification Analysis
- FI 9: Modification to AFWS - Overspeed Setpoint Change
- FI 11: Pressure/Temperature Reanalysis, Modifications and Documentation

The full resolution of items 1 and 9 cannot be achieved during Step 1 (i.e., Modes 6 and 5) because it requires the operation of the AFWS in Mode 3 and Mode 4, respectively. The resolution of item 5 is a matter of documentation. The modifications required by item 11 as a result of the pressure temperature reanalysis will be completed prior to entering Modes 4 and 3 with one potential exception, in which case the licensee will provide a justification for later completion. Because the reactor is not critical in Modes 4 and 3 and because no fission products have been generated the staff concludes that it is acceptable to defer the final resolution of the above four followup items to Mode 2 (reactor startup), when the reactor will achieve criticality. The staff will review the information to be provided by the licensee on the four items and will provide its evaluation of these matters at the time of the full power licensing decision.

7 CHRONOLOGY PERTAINING TO DIABLO CANYON UNIT 1 VERIFICATION EFFORTS

SSER 19 provided a chronology for the Diablo Canyon Unit 1 verification efforts from July 1, 1983 through October 10, 1983. The following is the continuation of the chronology.

October 11, 1983	Letter from R. L. Cloud transmitting report ITR-65, Rev. 1.
October 11, 1983	Letter from licensee regarding unresolved Item 30 in SER Supplement 18.
October 11, 1983	Letter from licensee transmitting update information on PG&E Phase I and Phase II Final Reports.
October 11, 1983	Letter from licensee regarding additional information on turbine building tornado loads.
October 11, 1983	Issuance of Amendment 6 to Facility Operating License No. DPR-76 regarding the security plan.
October 12, 1983	Letter from licensee regarding operational readiness with respect to containment integrity.
October 12, 1983	Letter from licensee regarding Item 29 in SER Supplement 18.
October 12, 1983	Letter from licensee regarding Item 2 in SER Supplement 18.
October 12, 1983	Letter to licensee advising of acceptability of August 3 letter concerning changes to security plan and safeguards contingency plan.
October 13, 1983	Letter from R. L. Cloud advising of no Open Item Reports.
October 13, 1983	Letter from licensee transmitting "Seismically Induced Systems Interaction Program (SISIP): Completion of Containment Activities."
October 14, 1983	Letter from licensee advising that responses to Open Items 1, 2, and 17 will be provided.
October 14, 1983	Letter from licensee advising of new Open Item regarding pipe restraint wedge nut test program.
October 14, 1983	Board Notification 83-135A - Information Item Regarding the Diablo Canyon Nuclear Power Plant (transmitting QA Case Study Working Paper - Class C).

October 14, 1983	Board Notification 83-156 - Information Items Regarding the Diablo Canyon Nuclear Power Plant (transmitting R. L. Cloud letters of October 1 (2 letters), October 2 (2 letters), October 4 (2 letters), October 5 and October 11 and Teledyne letters of October 10, 1983 (2 letters).
October 14, 1983	Letter from licensee transmitting 47th Semimonthly Status Report.
October 14, 1983	Board Notification 83-158 - Information Item Regarding the Diablo Canyon Nuclear Power Plant (transmitting SER Supplement No. 19).
October 14, 1983	Letter from Teledyne transmitting List of Effective Pages for IDVP Final Report.
October 14, 1983	Letter from Teledyne transmitting October 2nd Friday Semimonthly Report.
October 14, 1983	Letter from licensee providing additional information on fire protection.
October 14, 1983	Letter from licensee transmitting additional information on turbine building tornado loads.
October 14, 1983	Letter from licensee forwarding revision to figures concerning free hand averaging submitted as part of October 6, 1983 response to SSER 18.
October 14, 1983	Letter from licensee transmitting final update (Revision 14) to "Design Verification Program Phase I Final Report."
October 14, 1983	Letter from licensee in response to Phase I Technical Evaluation Report on control of heavy loads.
October 17, 1983	Letter to licensee regarding systems and technical specifications required for fuel load and low temperature testing.
October 18, 1983	Board Notification 83-161 - Allegations Concerning Errors in Design/Documentation of Safety Related Systems, Structures and Components.
October 19, 1983	Generic Letter 83-33, NRC Positions on Certain Requirements of Appendix R.
October 19, 1983	Letter from licensee transmitting replacement pages for October 14, 1983 revision to Phase I Final Report.
October 25, 1983	Audit (at Bechtel in California) of material related to allegations on design adequacy of certain structures, systems and components.

October 27, 1983	Board Notification 83-164 - Allegations Concerning Diablo Canyon (regarding QA/QC activities).
October 27, 1983	Board Notification 83-168 - Allegations Concerning Diablo Canyon.
October 27, 1983	Board Notification 83-169 - Allegations Concerning the Design of the Residual Heat Removal System.
October 27, 1983	Board Notification 83-170 - Allegations Concerning H. P. Foley Company.
October 27, 1983	Board Notification 83-171 - Allegations Concerning Small Bore Piping and Supports.
October 27, 1983	Letter to licensee transmitting Supplement 19 to SER.
October 27, 1983	Letter from licensee providing additional information regarding turbine building tornado loads.
October 28, 1983	Letter from licensee transmitting 48th Semimonthly Status Report.
October 31, 1983	Generic Letter 83-38, NUREG-0965, "NRC Inventory of Dams."
October 31, 1983	Board Notification 83-172 - Information Items Regarding the Diablo Canyon, Unit 1 Design Verification Program (transmitting R. L. Cloud letter dated October 13, 1983, Teledyne letters dated October 14, 1983 (2 letters).
November 1, 1983	Generic letter 83-37, NUREG-0737 Technical Specifications.
November 2, 1983	Generic Letter 83-35, Clarification of TMI Action Plan Item II.K.3.31.
November 4, 1983	Letter from licensee requesting technical specification changes on snubbers.
November 4, 1983	Letter from licensee forwarding revised information for Appendix R review.
November 7, 1983	Letter from licensee in response to Generic Letter 83-28.
November 8, 1983	Commission Memorandum and Order CLI 83-27 (reinstating license to load fuel and conduct precriticality testing).
November 8, 1983	Letter from licensee advising that technical support center is functional.
November 9, 1983	Board Notification 83-179 - Diablo Canyon Open Issues of SSER 19.

November 10, 1983	Board Notification 83-180 - Construction QA Concerns Reported to the NRC Staff by Henry Myers.
November 10, 1983	Issuance of Amendment No. 7 to DPR-76 (Technical Specification changes on inverters, containment isolation system, containment spray initiation, firewater pump and Halon system initiator replacement program).
November 10, 1983	Letter from licensee forwarding proposed technical specification changes on snubbers.
November 11, 1983	Letter from licensee providing status of compliance with certain license conditions.
November 11, 1983	Letter from licensee transmitting 49th Semimonthly Status Report.
November 14, 1983	Letter to licensee requesting information in BN 83-179 and information in connection with Contention 3.
November 15, 1983	Letter to licensee regarding NRC implementation dates for Amendment 1.
November 15, 1983	Letter to licensee requesting information related to test reports reviewed during site audit held August 31 - September 4, 1981 on environmental qualification program.
November 15, 1983	Letter to licensee transmitting request for additional information for NUREG-0737, Item II.D.1-Performance Testing of Relief and Safety Valves.
November 17, 1983	Letter to licensee forwarding Technical Evaluation Report on control of heavy loads (Phase II).
November 17, 1983	Letter from licensee providing response to November 14 letter regarding Contention 3.
November 23, 1983	Letter from licensee transmitting 50th Semimonthly Status Report.
November 28, 1983	Letter from licensee regarding status of items in Board Notification 83-179.
November 29, 1983	Letter from licensee responding to NRC letter of September 12, 1983 regarding deficiencies in quality assurance from fradulent actions by suppliers.
December 1, 1983	Letter to licensee on NRC review of emergency plan.
December 1, 1983	Letter from licensee advising that Design Verification Program is complete.
December 2, 1983	Generic Letter 83-32 - NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS.

December 2, 1983	Letter from licensee providing revised response to November 14 letter regarding Contention 3.
December 2, 1983	Form transmittal to licensee of monthly notice regarding applications and amendments to operating licenses.
December 6, 1983	Meeting with licensee to discuss open items in SSER 18.
December 6, 1983	Letter from licensee providing additional responses to NRC letter of November 14, 1983 regarding open items in SSER 18 and SSER 19.
December 8, 1983	Letter from licensee providing schedule for Unit 2.
December 9, 1983	Letter from licensee regarding public address system as discussed in meeting on September 7, 1983.
December 9, 1983	Letter from licensee regarding additional 6-month extension for FSAR update.
December 12, 1983	Letter from licensee providing information on items in SSER 19.
December 13, 1983	Board Notification 83-188 on inspection report on Diablo Canyon.
December 13, 1983	Letter from licensee providing information on follow up Item 11 in SSER 18.
December 13, 1983	Letter form licensee providing information on open Item 6 in SSER 18.
December 16, 1983	ACRS meeting with staff on status of Diablo Canyon Nuclear Power Plant.

Table C.8.1 (Continued)

Number	Title, IDVP organization, revision, and date
ITR-16:	Soils - Outdoor Water Storage Tanks (RLCA). Revision 0, December 8, 1982
ITR-17:	Piping - Additional Samples (RLCA). Revision 0, December 14, 1982
ITR-18:	Verification of the Fire Protection Provided for Auxiliary Feedwater System, Control Room Ventilation and Pressurization System Safety-Related Portion of the 4160 V Electric System (SWEC). Revision 0, December 13, 1982 Revision 1, May 24, 1983
ITR-19:	Verification of the Post-LOCA Portion of the Radiation Environments Used for Safety-Related Equipment Specification Outside Containment for Auxiliary Feedwater System and Control Room Ventilation and Pressurization System (SWEC). Revision 0, December 16, 1982
ITR-20:	Verification of the Mechanical/Nuclear Design of the Control Room Ventilation and Pressurization System (SWEC). Revision 0, December 16, 1982 Revision 1, April 26, 1983 Revision 2, July 25, 1983
ITR-21:	Verification of the Effects of High Energy Line Cracks and Moderate Energy Line Breaks for Auxiliary Feedwater System and Control Room Ventilation and Pressurization System (SWEC). Revision 0, December 15, 1982 Revision 1, May 3, 1983
ITR-22:	Verification of the Mechanical/Nuclear Portion of the Auxiliary Feedwater System (SWEC). Revision 0, December 17, 1982 Revision 1, April 26, 1983 Revision 2, July 25, 1983
ITR-23:	Verification of High Energy Line Break and Internally Generated Missile Review Outside Containment for Auxiliary Feedwater System and Control Room Ventilation and Pressurization System (SWEC). Revision 0, December 20, 1982 Revision 1, May 27, 1983
ITR-24:	Verification of the 4160 V Safety-Related Electrical Distribution System (SWEC). Revision 0, December 21, 1982 Revision 1, May 4, 1983
ITR-25:	Verification of the Auxiliary Feedwater System Electrical Design (SWEC). Revision 0, December 21, 1982 Revision 1, April 29, 1983
ITR-26:	Verification of the Control Room Ventilation and Pressurization System Electrical Design (SWEC). Revision 0, December 21, 1982 Revision 1, May 2, 1983

Table C.8.1 (Continued)

Number	Title, IDVP organization, revision, and date
ITR-27:	Verification of the Instrument and Control Design of the Auxiliary Feedwater System (SWEC). Revision 0, December 23, 1982 Revision 1, May 13, 1983 Revision 2, July 25, 1983
ITR-28:	Verification of the Instrument and Control Design of the Control Room Ventilation and Pressurization System (SWEC). Revision 0, December 23, 1982 Revision 1, May 13, 1983 Revision 2, July 25, 1983
ITR-29:	Design Chain - Initial Sample (SWEC). Revision 0, January 17, 1983
ITR-30:	Small Bore Piping Report (RLCA). Revision 0, January 12, 1983
ITR-31:	HVAC Components (RLCA). Revision 0, January 14, 1983 Revision 1, August 4, 1983
ITR-32:	Pumps (RLCA). Revision 0, February 17, 1983 Revision 1, April 1, 1983
ITR-33:	Electrical Equipment Analysis (RLCA). Revision 0, February 18, 1983 Revision 1, April 28, 1983
ITR-34:	Verification of DCP Effort by Stone & Webster Engineering Corporation (SWEC). Revision 0, February 4, 1983 Revision 1, March 24, 1983
ITR-35:	Independent Design Verification Program Verification Plan for Diablo Canyon Project Activities (RLCA). Revision 0, April 1, 1983
ITR-36:	Final Report on Construction Quality Assurance Evaluation of G. F. Atkinson (SWEC). Revision 0, February 25, 1983 Revision 1, June 20, 1983
ITR-37:	Valves (RLCA). Revision 0, February 23, 1983
ITR-38:	Final Report on Construction Quality Assurance Evaluation of Wismer & Becker (SWEC). Revision 0, March 1, 1983 Revision 1, March 16, 1983 Revision 2, June 20, 1983
ITR-39:	Soils - Intake Structure Bearing Capacity and Lateral Earth Pressure (RLCA). Revision 0, February 25, 1983

Table C.8.1 (Continued)

Number	Title, IDVP organization, revision, and date
ITR-40:	Soils Report - Intake Sliding Resistance (RLCA). Revision 0, March 9, 1983
ITR-41:	Corrective Action Program and Design Office Verification (RFR). Revision 0, April 19, 1983
ITR-42:	R. F. Reedy, Inc., Independent Design Verification Program Phase II Review and Audit of Pacific Gas and Electric Company and Design Consultants for Diablo Canyon Unit 1 (RFR). Revision 0, April 15, 1983
ITR-43:	Heat Exchangers (RLCA). Revision 0, April 14, 1983
ITR-44:	Shake Table Test Mounting Class 1E Electrical Equipment (RLCA). Revision 0, April 15, 1983
ITR-45:	Additional Verification of Redundancy of Equipment and Power Supplies in Shared Safety-Related Systems (SWEC). Revision 0, May 17, 1983
ITR-46:	Additional Verification of Selection of System Design Pressure and Temperature and Differential Pressure Across Power-Operated Valves (SWEC). Revision 0, June 27, 1983
ITR-47:	Additional Verification of Environmental Consequences of Postulated Pipe Ruptures Outside of Containment (SWEC). Revision 0, June 27, 1983
ITR-48:	Additional Verification of Jet Impingement Effects on Postulated Pipe Ruptures Inside Containment Revision 0, July 27, 1983
ITR-49:	Additional Verification of Circuit Separation and Single Failure Review of Safety-Related Electrical Equipment (SWEC). Revision 0, June 23, 1983
ITR-50:	Containment Annulus Structure Vertical Seismic Evaluation (TES). Revision 0, July 22, 1983
ITR-51:	Containment Annulus Structure Seismic Evaluation (TES). Revision 0, September 2, 1983 Revision 1, September 21, 1983
ITR-52:	Combined with ITR 68
ITR-53:	Combined with ITR 68
ITR-54:	Containment Building - Corrective Action (RLCA) Revision 0, September 11, 1983 Revision 1, October 3, 1983
ITR-55:	Auxiliary Building - Corrective Action (RLCA). Revision 0, Septembr 8, 1983 Revision 1, October 1, 1983

APPENDIX D

LIST OF CONTRIBUTORS

NRC Staff

M. Hartzman	Mechanical Engineering
B. Jagannath	Geotechnical Engineering
P. Kuo	Structural Engineering
J. Knox	Power Systems .
P. Morrill	Region V Reactor Inspector
H. Polk	Structural Engineering
H. Schierling	Licensing
H. Walker	Equipment Qualification
J. Wermiel	Auxiliary Systems

Table C.8.1 (Continued)

Number	Title, IDVP organization, revision, and date
ITR-56:	Turbine Building - Corrective Action (RLCA). Revision 0, September 9, 1983 Revision 1, September 24, 1983
ITR-57:	Fuel Handling Building - Review of DCP Activities (RLCA). Revision 0, August 1, 1983 Revision 1, September 8, 1983
ITR-58:	Intake Structure - Verification of DCP Activities (RLCA). Revision 0, August 8, 1983 Revision 1, October 1, 1983
ITR-59:	Large Bore Piping - IDVP Verification of Correction Action (RLCA). Revision 0, August 18, 1983 Revision 1, September 24, 1983
ITR-60:	Large and Small Bore Pipe Supports - IDVP Review of Corrective Action (RLCA). Revision 0, August 17, 1983 Revision 1, October 3, 1983
ITR-61:	Small Bore Piping - IDVP Review of Corrective Action (RLCA). Revision 0, September 10, 1983 Revision 1, October 2, 1983
ITR-62:	Combined with ITR-60
ITR-63:	HVAC Ducts, Electrical Raceways, Instrument Tubing and Associated Supports - IDVP Verification of Corrective Action (RLCA). Revision 0, August 22, 1983 Revision 1, October 2, 1983
ITR-64:	Combined with ITR-63
ITR-65:	Rupture Restraints - IDVP Verification of DCP Activities (RLCA). Revision 0, September 16, 1983 Revision 1, October 11, 1983
ITR-66:	Combined with ITR 63
ITR-67:	Equipment - IDVP Verification of Corrective Action (RLCA). Revision 0, August 12, 1982 Revision 1, September 9, 1983
ITR-68:	Verification of HLA Soils Work Revision 0, September 20, 1983 Revision 1, October 4, 1983

NOTE: The following reports were issued by RFR before the establishment of the ITR concept:

1: Review of ANCO Engineers, March 1, 1982.
2: Review of Cygna Energy Services, March 1, 1982.
3: Review of EDS Nuclear Inc., January 20, 1982.

Table C.8.1 (Continued)

Number	Title, IDVP organization, revision, and date
4:	Review of Harding Lawson Associates, January 26, 1982.
5:	Review of Pacific Gas and Electric Company, March 5, 1982.
6:	Review of URS/Blume and Associates. Engineers, March 5, 1982.
7:	Review of Wyle Laboratories, March 1, 1982.