EXHIBIT 1

PETITIONER'S REPLY TO RESPONDENTS' AND INTERVENOR'S RESPONSES TO PETITIONER'S MOTION TO SUPPLEMENT THE CERTIFIED INDEX OF THE RECORD

No. 14-1213 (D.C. Cir.)

Differing Professional Opinion – Appeal of Dr. Michael Peck
From Case File for DPO-2013-002 (Sep. 9, 2014) (ML14252A743)

NRC FORM 690 (11-2002) NRCMD 10.159

U.S. NUCLEAR REGULATORY COMMISSION

FOR PROCESSING USE ONLY

DIFFERING PROFESSIONAL OPINION -- APPEAL

1. DPO CASE NUMBER: DPO 2013-002

2. DATE APPEAL RECEIVED: 6/23/2014

INSTRUCTIONS: Prepare this form legibly and submit three copies to the address provided in Block 12 below.

3. NAME OF SUBMITTER	4. POSITION TITLE	5. GRADE
Michael Peck	Senior Reactor Technology Instructor	GG-14

6. OFFICE/DIVISION/BRANCH/SECTION	7. BUILDING	8. MAIL STOP	9. SUPERVISOR
OCHCO/ADHRTD/RTTBB	TTC		Steve Rutledge

10. DESCRIBE THE DIFFERING PROFESSIONAL OPINION. DESCRIBE THE PRESENT SITUATION, CONDITION, METHOD, ETC., WHICH YOU BELIEVE SHOULD BE CHANGED OR IMPROVED. (Continue on Page 2 or 3 as necessary.)

The agency failed to enforce statutory requirements following development of new seismic information by Pacific Gas and Electric (PG&E). This new information concluded that local earthquake faults could result in greater stress on plant equipment than considered in the Diablo Canyon Power Plant (DCPP) safe shutdown earthquake (SSE) safety analysis. These statutory requirements included:

1. Failure to promptly correct non-conforming safety analyses as required by 10 CFR 50, Appendix B.
2. Failure to obtain an amendment to the Operating License per 10 CFR 50.59 and 50.71(e).
3. Failure to enforce DCPP Technical Specifications requirements.

The DPO was written to draw attention to these issues leading to improved agency regulatory effectiveness and to ensure enforcement of the DCPP seismic design basis and technical specification requirements as defined by the facility Operating License.

11. DESCRIBE YOUR REASONS FOR SUBMITTING AN APPEAL (IN ACCORDANCE WITH THE GUIDANCE PRESENTED IN NRC MANAGEMENT DIRECTIVE 10.159). (Continue on Page 2 or 3 as necessary.)

Mr. Satorius, Executive Director for Operations:

Please take action to sustain the appeal of DPO 2013-002. The DPO Panel Report provided insufficient detail to support the conclusion that all statutory requirements were satisfied by PG&E.

Bases for Appeal

1. The conclusions presented in the Panel Report appeared to be based on a different facility design and licensing bases than described in the DCPP Final Safety Analysis Report Update (FSARU) and presented in the DPO. Resolution of the 10 CFR 50.71(e) and 10 CFR 50.59 DPO issues required a clear understanding *of the facility as described in the FSARU*. The DPO Panel Report did not include the bases for using current licensing bases (CLB) assumptions that deviated from those presented in the DPO and FSARU.

SIGNATURE OF SUBMITTER	DATE	SIGNATURE OF CO-SUBMITTER (if any)	DATE
[signature]	June 23, 2014		

12. SUBMIT THIS FORM TO:

Differing Professional Opinions Program Manager

Office of: Office of Enforcement/Concerns

Mail Stop: 4 A14A

13. ACKNOWLEDGMENT

13. SIGNATURE OF DIFFERING PROFESSIONAL OPINIONS PROGRAM MANAGER (DPOPM)	DATE OF ACKNOWLEDGMENT
Renée Pedersen	6/24/2014

14. DECISION

		DATE
☐ Appeal sustained ☐ Appeal denied (see attached)	Differing Professional Opinion Closed	

Continued Item 11

2. The Panel Report did not provide sufficient detail to support the conclusion that the licensee's actions were consistent with agency statutory requirements. The DPO detailed specific examples of the agency's failure to enforce certain regulatory and statutory requirements. The Panel Report responded to these detailed examples with general statements that regulatory requirements and safety objectives were satisfied.

Background

The DCPP seismic design and local geology is complex. However, the facility design control (10 CFR 50, Appendix B), License fidelity (10 CFR 50.59 and 10 CFR 50.71(e)), and operability (DCPP Technical Specification) issues raised in the DPO were not overly complex. These processes are well understood and routinely verified as part of the NRC Light Water Reactor Inspection Program and the Reactor Oversight Process.

In November 2008, PG&E reported discovery of a new line of epicenters located about a mile offshore from the DCPP.[1] The licensee stated that if this line of epicenters represented an earthquake fault, then the resulting ground motion would be bounded by the DCPP seismic design bases established by the Long Term Seismic Program (LTSP). The licensee committed to characterize the potential fault and evaluate the effect on plant structures, systems, and components (SSCs). This line of epicenters became known as the Shoreline fault.

In April, 2009, the NRC Office of Nuclear Reactor Regulation (NRR) released a preliminary review of the Shoreline fault.[2] This analysis concluded that ground motion that may be produced by the Shoreline fault would be within the plant seismic design bases (LTSP). NRC personnel, myself included, presented the results of this preliminary review at multiple public meetings held during the subsequent two years.

In September 2010, the NRC and PG&E held a public seismic workshop in San Luis Obispo, California. During the workshop, a PG&E seismologist presented the results of deterministic and seismic hazard characterization of the Shoreline fault. At the end of the presentation, I asked how the new ground motions compared to the facility SSE. The PG&E seismologist did not answer my question. The seismologist stated that LTSP established the facility seismic design basis. After the workshop, I reviewed the facility SSE as presented in the FSARU. I found that the seismic design basis documented in the FSARU was considerably different than both PG&E and the NRC personnel had described at the pervious public meetings. The FSARU stated that the LTSP was explicitly not part of the seismic design basis. I also found that the Shoreline fault deterministic ground motions, as presented at the workshop, were about 70 percent greater than those described in the facility SSE safety analysis.

Per Inspection Procedure IP 71111.15,[3] an operability evaluation was required because the new information called into question if the seismic design basis, as established by General Design

---

[1] NRC Event Number 44675, Offsite Notification and Media Briefing due to Potential Discovery of Off Shore Fault near Plant, November 21, 2008.
[2] Diablo Canyon Power Plant, Unit Nos. 1 and 2 – NRC Preliminary Review of Potential Shoreline Fault, April 8, 2009 (ML090930459).
[3] Inspection Procedure 71111.15, Operability Determinations and Functionality Assessments (ML112010663), "If operability is not justified then determine impact on any TS limiting condition for operation (LCO)."

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

Criteria (GDC) 2, was still satisfied.[4] To be considered operable, technical specification required SSCs must be capable of performing the required safety functions, as described in the safety analyses, at the higher seismic loadings. PG&E maintained that operability evaluation was not required because the new ground motions were within the bounds of the LTSP.

In November 2010, I presented my findings to Region IV management and the NRR project manager (PM). At this meeting the deputy director of Division of Reactor Projects (DRP) took an action to request PG&E to formally evaluate the operability of plant SSCs. PG&E again refused, stating that the LTSP established the seismic design basis for the facility.

I concluded that PG&E would likely not be successful demonstrating operability based on my previous experience with DCPP reactor head replacement inspections. These inspections identified that some reactor coolant pressure boundary and reactor head structural components failed to meet the American Society of Mechanical Engineers (ASME) Code[5] acceptance limits when evaluated against the existing double design earthquake (DDE) or SSE loads.[6] PG&E subsequently obtained an amendment to the Operation License allowing use of higher seismic damping values in the Code calculations.[7] This inspection revealed that insufficient Code margin was available to accommodate the higher loading represented by the Shoreline fault.

In December 2010, I reported back to the DRP deputy director that PG&E had not preformed the requested operability evaluation. The deputy director encouraged me to drop the issue. The deputy director suggested that, as an option, I could prepare a “white paper” detailing the concern.

In January 2011, PG&E submitted the completed reevaluation of the local geology on the DCPP Docket.[8] This report included deterministic evaluations concluding that three local faults, the Shoreline, Los Osos and San Luis Bay, were each capable of generating significantly greater ground motion than was used to establish the facility DDE/SSE.

In February 2011, I submitted a “white paper” to Region IV management.[9] The “white paper” described the facility seismic design bases and the extent the new ground motions exceeded the limiting values used the DDE/SSE safety analysis to seismically qualify plant SSCs. I included recommendations to initiate enforcement action against PG&E. These recommendations included

---

[4] NRC Inspection Manual, Part 9900: Technical Guidance, Operability Determinations & Functionality Assessments for Resolution of Degraded Or Nonconforming Conditions Adverse to Quality or Safety (ML 073531346), Section C.1, “Relationship Between the General Design Criteria and the Technical Specifications,” stated that the “failure to meet a General Design Criteria in the CLB should be treated as a degraded or nonconforming condition and, therefore, the technical guidance in this document is applicable. The Diablo Canyon CLB established the DDE as the GDC 2 SSE. The new ground motions exceeded the SSE ground motions described in the FSARU

[5] American Society of Mechanical Engineers Boiler and Pressure Vessel, Code, Section III, required per 10 CFR 50.55a. Meeting Code acceptance limits ensures the integrity of the reactor coolant pressure boundary following earthquakes and accidents

[6] Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2009005 And 05000323/2009005, February 3, 2010 (M100341199)

[7] Diablo Canyon Power Plant, Unit Nos. 1 And 2 -Issuance Of Amendments Re: Revision To Final Safety Analysis Report Update Section 3.7.1.3, "Critical Damping Values" (TAC NOS. ME4056 AND ME4057) (ML102530443)

[8] Report on the Analysis of the Shoreline Fault Zone, Central Coast California to the NRC, January 7, 2011 (ML110140400)

[9] White Paper, “Resident Inspectors Recommendation for Regulatory Disposition of the Failure of Pacific Gas & Electric to Perform an Operability Evaluation Following Discovery of the Shoreline Fault,” February 2, 2011, attached to e-mail to Geoff Miller, Subject: ACT: Diablo Canyon - Recommendation for Regulatory Disposition.

a potential greater than green finding associated with the failure of PG&E to evaluate and disposition SSC operability (10 CFR 50, Appendix B) and an escalated traditional enforcement violation (10 CFR 50.9) after PG&E provided incomplete and inaccurate information concerning the facility seismic design bases. This incomplete and inaccurate information was used by the NRR PM for the agency's conclusions presented in the April 2009 letter.

In March 2011, a meeting was held at Region IV to discuss the "white paper" recommendations. The branch chief from the NRR Division of Operating Reactor Licensing, the NRR PM and DRP management attended the meeting. A consensus was reached that PG&E had not evaluated the new seismic information against the facility design bases. The DRP division director expressed concern that enforcement action would conflict with the NRC position communicated in the April 2009 NRR letter.[10] To address this concern, I drafted a concurrence Task Interface Agreement (TIA) letter documenting agreement between NRR and Region IV that PG&E was required to evaluate the new seismic information against the facility design bases, including the DDE/SSE.[11] The failure of the licensee to perform an operability evaluation was documented as an unresolved item (URI) in the DCPP inspection report.[12]

Between December 2010 and June 2011, the NRC and PG&E held several public meetings to discuss how the new seismic information would be incorporated into the DCPP License. PG&E proposed using the Hosgri Evaluation (HE) methodology for the facility SSE. The HE described the plant response to a postulated 7.5 Magnitude earthquake on the Hosgri fault. The HE used different assumptions, methodology and acceptance limits than the existing DDE/SSE. The CLB described the HE as a response to a NRC question raised during original plant licensing. The HE bound the higher ground motions identified in the PG&E reevaluation of the local geology.

In October 2011, PG&E submitted License Amendment Request (LAR) 11-05 to designate the HE as the DCPP SSE.[13]

Also, in October 2011, PG&E concluded that all plant SSCs were operable in response to the URI and TIA.[14] However, the licensee only evaluated the new ground motions against the HE. The licensee stated that NRC operability policy provided for use of the HE as an "alternative method." Based on using the HE "alternative method," PG&E argued that the new ground motions did not have to be directly evaluated against the DDE/SSE safety analysis or acceptance limits. Based on

---

[10] Diablo Canyon Power Plant, Unit Nos. 1 and 2 – NRC Preliminary Review of Potential Shoreline Fault, April 8, 2009 (ML090930459). Letter stated that the LTSP established the seismic design bases

[11] Task Interface Agreement – Concurrence on Diablo Canyon Seismic Qualification Current Licensing and Design Basis," August 1, 2011 (ML112130665)

[12] Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011002 and 05000323/2011002, May 11, 2011, Unresolved Item: 05000275; 323/2011002-03, "Requirement to Perform an Operability Evaluation Following Receipt of New Seismic Information." URI updated in Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011003 And 05000323/2011003, August 10, 2011, Discussed URI 05000275; 05000323/2011002-08, Requirement To Perform An Operability Evaluation Following Receipt of New Seismic Information (Section 4OA2).

[13] License Amendment Request 11-05, "Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake" October 20, 2011 (ML11312A166).

[14] PG&E Notification: 50086062, Type: DA Work Type: EVAL AANS, Description: LTCA-Ident. of Seis. Lineament Offshore.

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

the PG&E operability evaluation, the NRC closed the URI and issued a violation associated with the failure to evaluate operability after initially developing the new seismic information.[15]

I disagreed that the HE satisfied NRC criteria for use as an "alternative method" for operability. I included a violation with DCPP Inspection Report 2011-05 to address PG&E's inadequate operability evaluation. Region IV management did not accept my recommended violation. The licensee stated that comparison of the new seismic information directly against the DDE/SSE safety analysis would result in "exceedances." In other words, operability could not be demonstrated by comparing the new seismic information with the GDC 2 design basis and safety analysis. This was a concern because the HE, while bounding for ground motion, was not bounding for the seismic qualification of technical specification required SSCs.[16]

I documented my concerns using the NRC non-concurrence process.[17] I included a detailed technical discussion addressing why the PG&E operability evaluation failed to meet the NRC standard. I expected Region IV to agree with the technical argument and issue the recommended violation. I also expected PG&E to follow up with a request for regulatory dispensation in the form of a waiver (10 CFR 50.12) and Code relief (10 CFR 50.55a) due to the lack of margin in the existing DDE/SSE safety analysis. The alterative required PG&E to perform a plant technical specification shut down pending disposition of the non-conforming safety analysis.

In response to the technical discussion in the non-concurrence, the agency stated:

> *" the seismic CLB did not provide a way to evaluate new information that becomes available. Therefore, the licensee has proposed a methodology to perform the full operability evaluation to the NRC as a license amendment request, and the staff is evaluating the best way to proceed."*
>
> *" the complete operability evaluation cannot be made by the licensee without the NRC agreeing on the correct way to perform the evaluation, what calculation method and design values are appropriate for the new data, and what plant capability must be demonstrated by this evaluation."*
>
> *"The NRC will not ask the licensee to use the new ground motion input data in the Design or the Double Design Earthquake (SSE) evaluations because the new ground motion data does not match the assumptions in those analyses. Attempting to do so would create a numerical result that is not technically justified."*
>
> *"The staff concluded the revised operability determination provided an initial basis for concluding a reasonable assurance that plant equipment would withstand the potential effect of the new vibratory ground motion."*

Rather than addressing the specific technical issues presented in the non-concurrence, Region IV presented an argument that PG&E did not have to meet technical specification operability requirements. Region IV's apparent argument was that operability cannot be demonstrated against the current safety analysis; therefore operability may be deferred until the NRC approves a method (LAR 11-05) that would have a successful result.

This was a concern because NRC policy did not provide for continued reactor operation outside of the bounds of limiting safety analysis unless the licensee clearly demonstrated SSC operability.

[15] NCV 05000275; 05000323/-2011005-02, Failure to Perform an Operability Determination for New Seismic Information (Section 1R15.2), Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011005 and 05000323/2011005 , February 14, 2012 (ML12040843).
[16] Detailed examples were provide in DPO 2013-002
[17] Non-Concurrence NCP-2012-001, DCPP IR 2011-05 (ML12045843)

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

NRC policy did not provide for an "initial basis" for operability or "deferment" until the License is amended. Continued reactor operation was only permitted after SSCs were demonstrated operable at that point in time. Plant SSC are considered inoperable, and the associated technical specification Limiting Condition for Operation not met when a nonconforming or unanalyzed condition results in an SSC unable to perform its specified safety function as described in the safety analysis.[18]

In February 2012, the NRC concluded that LAR 11-05 (requested to adopt the HE for the facility SSE) would not be accepted for review.[19] The staff rejected the LAR because:

1) The methodologies and acceptance limits for SSCs using HE differ from that specified in Standard Review Plan (NRC acceptance criteria for a facility SSE).

2) PG&E had not completed a reevaluation of the reactor coolant system for the seismic and LOCA loads (the HE didn't meet ASME Code requirements for the SSE).

3) PG&E did not provide a peer reviewed seismic probabilistic risk assessment.

4) Concerns about use of a seismic margins assessment for operability evaluations.

In October 2012, PG&E withdrew LAR 11-05 at the NRCs request.[20] Also, in October, the NRR PM provided PG&E written direction to update the FSARU to include the "Shoreline scenario as a lesser included case under the HE."[21] The PM's action essentially established the LTSP and HE as the de-facto SSE, circumventing the license amendment process per 10 CFR 50.90,[22] and bypassing the required public notice and hearing opportunities required for a change to the Operating License per 10 CFR 50.91.[23]

The PM justified this action by stating:

> *"As documented in RIL 12-01, the NRC staff's assessment is that deterministic seismic-loading levels predicted for all the Shoreline fault earthquake scenarios developed and analyzed by the NRC are at, or below, those levels for the Hosgri earthquake (HE) ground motion and the long term seismic program (LTSP) ground motion. The HE ground motion and the LTSP ground motion are those for which the plant was evaluated previously and demonstrated to have reasonable assurance of safety. Therefore, the staff has concluded that the Shoreline scenario should be considered as a lesser included case under the Hosgri evaluation and the licensee should update the final safety analysis report (FSAR), as necessary, to include the Shoreline scenario in accordance with the requirements of 10 CFR 50.71(e)."*

---

[18] NRC Inspection Manual, Part 9900: Technical Guidance, Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety (ML 073531346), Sections 3.8, 3.10 & 6.1
[19] FOIA/PA NO: 2014-0065 (Group B) (ML13354B992)
[20] Diablo Canyon Power Plant Units 1 and 2 – Withdrawal of an Amendment Request, October 31, 2012 (ML12289A076)
[21] Diablo Canyon Power Plant Units 1 and 2 – NRC Review of Shoreline Fault(ML120730106)
[22] NRR Office Instruction LIC-100, Revision 1, Control of Licensing Bases for Operating Reactors, Section 2.1.5.5 10 CFR 50.90, License Amendments (ML033530249)
[23] See the "Perry Decision," Commission Memorandum and Order CLI 96-12

Continued Item 11

As discussed in detail in the DPO, “demonstration to have reasonable assurance of safety” was not among the criteria used by NRC to determining if an amendment to the Operating License was required.[24]

In July 2013, I submitted DPO-2013-002, Differing Professional Opinion Involving Seismic Issues at DCPP. This DPO identified three concerns:

1) Incorporating the “Shoreline scenario” into the FSARU required prior NRC approval in the form of an amendment to the Operating License.

2) Region IV failed to enforce DCPP Technical Specification requirements for a plant shutdown after the licensee inadequately operability evaluation.

3) The Agency failed to adequately disposition the updated seismic information associated with San Luis Bay and Los Osos earthquake faults.

In May 2014, the DPO Panel Report was issued. I agreed with the Panel’s conclusion that issues raised in the DPO did not result in a significant or immediate safety concern. I also agreed that the potential ground motions from the nearby faults would not exceed the levels of ground motion considered during the licensing of the plant. However, I disagreed with the Panels other conclusion:

1) An amendment to the Operating License was not required for the new seismic information.

2) A lack of formal regulatory guidance exists for evaluating new information on natural hazards.

3) The licensee adequately demonstrated SSC technical specification operability.

Original Diablo Canyon Power Plant Seismic Design and Licensing Bases

An understanding of the facility licensing bases is needed before a effective review of the DPO Panel conclusion can be performed.

The FSAR (as amended) served as the principal reference document to support the PG&E Part 50 DCPP license application. The FSAR described the methods PG&E used to confirm that applicable NRC regulations were met and contained the technical information required by 10 CFR 50.34. This technical information included safety analyses that presented the design bases and the limits on operation for plant SSCs. 10 CFR 50.34(b) specifically required the FSAR to include safety analyses that demonstrated that the principal design criteria for the facility (GDCs) were met. This included the design basis and the relationship of the design bases to these principal design criteria (GDCs).

10 CFR 50.2 defined design bases as that information which identifies the specific functions to be performed by a facility SSC and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. 10 CFR 50.2 design bases included the bounding conditions under which SSCs must perform design bases functions, including protection against

[24] NRC criteria used to determining if an amendment to the Operating License is required is found in 10 CFR 50.59.

natural phenomena. For seismic, the design bases functional requirements were derived primarily from the principal design criteria contained in GDC 2 (the minimum standards set by Part 50, Appendix A) and NRC regulations that imposed functional requirements or limits on the plant design (10 CFR 100, Appendix A). These 10 CFR 50.2 design bases were a subset of the original licensing bases.

The original DCPP FSAR, including the 10 CFR 50.2 design bases, were presented in accordance with 10 CFR 50.34(b)[25] and were reviewed by the NRC in connection with granting the original license. These safety analyses (license application, FSAR Amendment 85) became the "original plant licensing bases" when the NRC approved the facility Operating License.

I've included exerts of the FSAR (license application, Amendment 85) in Appendix A. The original seismic licensing bases may be summarized as:

- The seismic design basis functional requirements were established by GDC 2[26] and 10 CFR 100, Appendix A. The DDE safety analysis (FSARU Sections 2.5, 3.7, 3.8, 3.9, 3.10, and 5.2) demonstrated that the GDC 2 and Part 100, Appendix A, SSE design bases functional requirements were satisfied.
- The earthquake design bases were defined as the DE and DDE (equivalent to the Part 100, Appendix A, operational basis earthquake and SSE).
- The GDC 2 safety analysis (FSAR 2.5.2.9) determined that the DDE was the maximum earthquake potential for the facility (considering all faults within 75 miles of the site). This safety analysis was consistent with the requirements 10 CFR 100, Appendix A. The Hosgri was not considered a "capable"[27] fault and excluded from the GDC 2 safety analysis.
- The HE was prepared to answer a NRC question. The HE was not included in the 10 CFR 50.34 safety analyses (FSAR Section 2.5) because the HE did not implement a regulatory requirement per 10 CFR 50.34. PG&E maintained the HE, a beyond design bases event, as a licensing bases commitment.[28]
- PG&E only committed to seismically qualify plant SSCs (needed to function for the SSE per Safety Guide 29, Seismic Design Classification) for the DDE.[29] Some plant SSCs were also qualified for the HE. In many cases the seismic qualification of plant SSCs were more limited

---

[25] Also consistent with PG&E's commitment to Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)

[26] FSAR stated that PG&E met GDC 2 (1997). However, Letter, from A. Giambusso, Director of Licensing, Atomic Energy Commission (AEC), to F.T. Searls, Pacific Gas and Electric, dated August 13, 1973, committed PG&E to address any deviations or exceptions taken to GDC 2 (Part 50, Appendix A, 1971). Letter: F. J. Miraglia, Division of Licensing, US NRC, from P. A. Crane, Pacific Gas and Electric, CHRON 131464, "Description of PG&E's compliance with the requirements 10 CFR 20, 50, and 100," dated September 10, 1981, included that DCPP seismic design bases did not include any exceptions to GDC 2 (Part 50, Appendix A, 1971).

[27] "Capable" defined per 10 CFR 100, Appendix A. At the time of OL, NRC and PG&E disagreed on the "capability" of the Hosgri fault (see DCPP SSER 7).

[28] Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," endorses use of NEI 97.04, Guidance and Examples for Identifying 10 CFR 50.2 Design Bases Appendix B, for" providing examples and guidance acceptable to the staff for providing a clearer understanding of what constitutes design bases information." NEI 97.04, Appendix B stated that design bases are explicitly tied to regulatory requirements, primarily the GDCs, and implemented by the 50.34 safety analyses. The HE does not implement a regulatory requirement or GDC and this not included within the GDC 2 design bases.

[29] Set of SSCs listed in Safety Guide 29 (Regulatory Guide 1.29, Seismic Design Classification), required to remain functional following a SSE.

Continued Item 11

for the SSE/DDE than the HE. As described in the DPO, this was based on differences in the assumptions, methods, and acceptance criteria used in the two analyses.

Diablo Canyon Power Plant Current Licensing Basis

FSARU, Revision 20, was the current FSARU when the DPO was written. The CLB seismic and design bases were very similar to the original licensing bases. In summary, the CLB:

- The DDE and supporting safety analysis satisfied the requirements of GDC 2 and were equivalent to the SSE described in 10 CFR 100, Appendix A.
- The licensee committed to ensure the plant SSCs listed in Regulatory Guide 1.29 (Seismic Design Classification) will remain functional following the DDE/SSE.
- The HE was an answer to an NRC question during original plant licensing. Regulatory Guide 1.29 does not apply to the HE.
- FSARU Section 3.7.6 established the HE shutdown path. Unlike the DDE/SSE (GDC 2), the HE did not assume a coincidental accident or fire. This section described the SSCs qualified for the Hosgri earthquake.
- As required by 10 CFR 50.55a, PG&E demonstrated that the combined accident and DDE/SSE loads did not exceed ASME Code acceptance limits for the reactor coolant pressure boundary.
- PG&E performed ASME Code calculations for the HE. However, PG&E did not include accident loads in these calculations. HE Code calculations were not required by NRC regulations. PG&E performed these calculations as part of a licensing bases commitment.
- The HE was not tied to meeting a regulatory requirement (GDC, Part 100, etc.). Because HE was not part of the design bases, the licensee was not required to include a 10 CFR 50.34 safety evaluation in the FSARU.[30]
- LTSP was explicitly excluded from the seismic design bases. PG&E maintained a licensing bases commitment to evaluate LTSP seismic margins during modifications of certain plant components.

I've included exerts of FSARU, Revision 20, in Appendix B.

PG&E implemented and maintained the CLB requirement for the SSE by the Plant Q-List. As shown in Appendix C, and required by 10 CFR 50, Appendix B; and the licensee's commitment to Regulatory Guide 1.26,[31] PG&E defined the facility SSE as the DDE in the facility design control management systems.[32]

---

[30] The HE is not defined as part of the design bases. Per NEI 97.04, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases, Appendix B, page B21, "Seismic Topical Design Bases" (ML003678532), design bases are explicitly established by regulatory requirements, primarily the GDCs. Since the HE is not tied to the GDCs or 10CFR50.55a, the HE is not part of the DCPP design bases. NEI-97.4 was endorsed by Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases." Maintaining selected plant SSCs qualified to the HE was a licensing bases commitment.

[31] Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, required establishing quality classifications for those plant SSCs credited for preventing or mitigating design bases events as defined in the safety analysis.

[32] Pacific Gas and Electric Company Nuclear Power Generation, Classification of Structures, Systems, and Components For Diablo Canyon Power Plant Units 1 And 2 (Q-LIST), Revision 27

Continued Item 11

In September 2013 (after the DPO was submitted), PG&E made extensive changes to FSARU Section 2.5, "Geology and Seismology." Many of these changes affected the description of the seismic design basis. These changes also included addition of the "Shoreline scenario as a lesser included case under the HE." PG&E did not screen these changes against the 10 CFR 50.59 criteria to determine if an amendment to the Operating License was required. PG&E justified omitting the required screen by stating these changed were derived from NRC correspondence:[33]

> "These enhancements are derived from correspondence with the NRC, NRC regulatory documentation, and specific USAFR text, therefore a 10 CFR 50.59 screen is not required."

Many of these changes indirectly addressed how SSC seismic safety functions were met. The 10 CFR 50.59 screening criteria required these changes to be evaluated:[34]

> *" methods of evaluation included in the UFSAR to demonstrate that intended SSC design functions will be accomplished are considered part of the "facility as described in the UFSAR." Thus use of new or revised methods of evaluation is considered to be a change that is controlled by 10 CFR 50.59 and needs to be considered as part of this screening step. Changing elements of a method of evaluation included in the UFSAR, or use of an alternative method, must be evaluated under 10 CFR 50.59(c)(2)(viii) to determine if prior NRC approval is required. Changes to methods of evaluation (only) do not require evaluation against the first seven criteria."*

These PG&E FSARU enhancements made to Section 2.5, "Geology and Seismology" may have contributed to the DPO Panels misunderstanding of the DCPP seismic design bases.

The Panel Assumed an Inappropriate Seismic Design Basis to Disposition the Issues Raised in the Differing Professional Opinion

The Panel depositions of the DPO issues were based on the underlying assumption that both HE and DDE ground motions established the GDC 2 SSE design basis for the facility. Using this assumption, the Panel concluded that the higher of the two ground motions, either the DDE or the HE, established the bounding condition for seismic design. The Panel used this logic to conclude that an amendment to the Operating License was not required because the new seismic information was already bound by the HE ground motion.

For the Panel's conclusions to be correct, then this underlying assumption must also be correct. Unfortunately, the Panel Report did not include sufficient detail to provide the reader an understanding of how the Panel formed this understanding of the facility design bases.

In June 2014, I met with the Panel members. At the meeting, I stated that the CLB presented in the Panel Report appeared to be conflict with the FSARU (see Appendix B) and the DPO. I requested that the Panel provide the bases for this underlying CLB assumptions used to disposition the DPO. The Panel Chairman stated that the FSARU clearly established the HE as part of the facility design bases and he referred me to FSARU (Revision 21) Section 2.5.5.9,

[33] DCPP Form 69-20108, UFSAR Change Request Section(s): 2. 5 (Seismology and Geology), June 2013 "These enhancements are derived from correspondence with the NRC, NRC Regulatory documentation and specific UFSAR test, therefore a 10 CFR 50.59 screen is not required."

[34] NEI 96-07, "Guidelines for10 CFR 50.59 Evaluations" (ML003636043), Section 4.2.1.3, "Screening Changes to UFSAR Methods of Evaluation," as endorsed by Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments, (ML003759710)

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

"Earthquake Design Basis." I've included this FSARU Section below with highlighted changes incorporated with Revision 21 and PG&E's annotations (September 2013).[35]

2.5.54.9 Earthquake Design Basis

The earthquake~~s postulated~~ design bases for the DCPP site are discussed in Section 2.5.32.9, ~~and~~ a discussion of the design response spectra is provided in Section 2.5.3.10, and the application of the earthquake ground motions to the seismic analysis of structures, systems, and components is provided in Section 3.7. Response acceleration curves for the site resulting from Earthquake B and Earthquake D-modified are shown in Figures 2.5-20 and 2.5-21, respectively. Response spectrum curves for the ~~7.5M~~ Hosgri earthquake are shown in Figures 2.5-29 through 2.5-32.

Edited for Clarity - Revised Section Number
Edited for Clarity
Added for Clarity
Edited for Clarity - Revised Section Number
Edited for Clarity
Added for Clarity – Section pointers revised to be more accurate.

A comparison of this FSARU Section with page A-6 (Appendix A), shows that PG&E added the HE as part of the seismic design bases description subsequent to plant licensing. This addition to the design basis description could be considered an acceptable change. However, the Panels use of this change to exclude the SSE/DDE requirements would be considered a change to the facility design bases and would require an amendment to the Operating License. 10 CFR 50.59 stated that an amendment to the Operating License was required before the licensee made a changed that "result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses."[36]

Consistent with the licensee's commitment to Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," FSARU Sections 3.1, Conformance with GDC, and 3.2.1, Seismic Classification, established the seismic design basis:

> *This section should identify those structures, systems, and components important to safety that are designed to withstand the effects of a Safe Shutdown Earthquake (see Section 2.5) and remain functional. These plant features are those necessary to ensure:*
>
> 1. *The integrity of the reactor coolant pressure boundary,*
> 2. *The capability to shut down the reactor and maintain it in a safe condition, or*
> 3. *The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.*

As shown in Appendixes A, B and C, the SSE for DCPP has always been the DDE, not the HE as described in the Panel Report..

The Panel's assumption that the HE was included in the SSE design basis provided insufficient justification to exclude comparison of the new information against the DDE/SSE safety analysis. If both analyses supported the facility SSE, as described in the Panel Report, then both analyses must be required for GDC 2 compliance. If both analyses are required for GDC 2, then the bounding condition for comparison would include the DDE <u>and</u> the HE, not the Panels position of the DDE <u>or</u> the HE.

---

[35] DCPP Form 69-20108, UFSAR Change Request Section(s): 2. 5 (Seismology and Geology), June 2013

[36] For additional detail see: Nuclear Energy Institute, Guidelines For 10 CFR 50.59 Evaluations, February 22, 2000, Section 4.3.8, "Does the Activity Result in a Departure from a Method of Evaluation Described in the UFSAR Used in Establishing the Design Bases or in the Safety Analyses?"

Continued Item 11

For the purposes the DPO disposition, it makes no difference whether or not the HE was or was not part of the GDC 2 design bases. The effect of the new information on the DDE/SSE licensing requirements and operability would still require disposition in terms of the license and operability. As discussed in the DPO, the DDE/SSE was more limiting for SSC seismic qualification than the HE. Given the 70-percent increase represented by the new ground motions, the limitations of the DDE/SSE safety analysis became even more pronounced.

The Panel Report Failed to Address the Specific Regulatory and Statutory Requirements Cited in the Differing Professional Opinion

The DPO identified the regulatory framework and specific statutory requirements that the agency failed to enforce at DCPP. Many of these requirements were related to *the facility as described in the Final Safety Analysis Report Update*. The Panel Report did not include adequate detail for the reader to conclude that these requirements were satisfied.

The DPO Panel Report stated that " *an FSARU change was likely not required at all, let alone, something that required a license amendment."*

However, Title 10 CFR 50.71(e) required the FSARU GDC 2 safety analysis to be updated:

> *" FASR originally submitted as part of the application for the operating license, to assure that the information included in the FSAR contains the latest material developed."*
>
> *"The updated dated FSAR shall be revised to include the effects of all changes made in the facility or procedures as described in the FSAR; all safety evaluations performed by the licensee.. and all analysis of new safety issues performed "*

Title 10 CFR 50.34(b) required the FSAR to include a safety analysis demonstrating that the GDC 2 design basis was satisfied:

> *"The FSAR shall include information that described the facility, presented the design bases and limits on its operation, and presents the safety analyses of the SSCs and of the facility as a whole."*

The Diablo Canyon license application (original FSAR, Amendment 85) included a safety analysis that demonstrated the GDC 2 and Part 100, Appendix A, SSE design basis was satisfied. This analysis included an evaluation of all earthquake faults within 75 miles of the site (with exception of the Hosgri fault). From this evaluation, this safety analysis developed a ground motion. The licensee used this ground motion as the *design bases controlling parameter* [37] to determine the amount of seismic stress plant SSCs would be exposed to following the DDE/SSE. The safety analysis, consistent with 10 CFR 50.34(b), included a description demonstrating that the functional design bases requirements of GDC 2 and Part 100, Appendix A, were meet for the SSCs listed in Regulatory Guide 1.29.[38]

[37] The DPO included a detained description of how this *design bases controlling parameter* was developed and used for SSC seismic qualification, consistent with NEI 97.04, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," Appendix B, for" providing examples and guidance acceptable to the staff for providing a clearer understanding of what constitutes design bases information."
[38] Per 10 CFR 100, App A, III(c) and 10 CFR 50.34(a)(3))

Continued Item 11

The licensee's new seismic information concluded that the existing *design bases controlling parameter* (ground motion) *as described in the FSARU safety analysis*, could be exceeded. PG&E was required to update the FSARU with this new information because the bounds of the safety analysis were challenged, calling into question the conclusion that the GDC 2 functional requirements were still satisfied. The new information raised the question if the plant SSCs, required by the design bases to remain functional for the DDE/SSE, would remain seismically qualified at the higher ground motions, within the context of the existing safety analysis.

The failure of PG&E to take prompt corrective action(s) to restore the bounds of safety analysis and plant SSCs to regulatory requirements and the design bases[39] was a violation of 10 CFR 50, Appendix B. Appendix B stated:

> *Criterion III, Design Control, required that "applicable regulatory requirements and the design basis (50.2) and as specified in the license application (FSAR), for those SSCs to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."*
>
> *Criterion XVI, Corrective Actions, required that conditions adverse to quality, such as failures, nonconformance's, are promptly identified and corrected."*

The new information resulted in the design basis (as specified in the license application for GDC 2) to be no longer correctly translated in the specifications, drawings, procedures, and instructions. The new seismic information rendered the FSARU SSE safety analysis non-conforming with GDC 2. 10 CFR 50.71(e) ensures that fidelity is maintained between new information, the FSARU safety analysis, and the GDC functional requirements establishing the design bases.[40]

The HE was unaffected by the new information for two independent reasons:

1) The CLB (FSARU) stated that the HE only applied to an earthquake on the Hosgri fault, and the new information was not related to the Hosgri fault, and

2) The HE was not used to establish the plant GDC 2 seismic design basis. The HE safety evaluation was not included in the FSARU. A 10 CFR 50.34 safety evaluation was not required to be included in the FSARU because the HE was not used to demonstrate that design bases or design basis functional requirements (GDC) were met.[41]

FSARU Change Required a License Amendment

The Panel Report did not address the specific issues identified in the DPO related to the failure of the licensee to obtain an amendment to the license supporting <u>the required</u> FSARU changes per 10 CFR 50.71(e). As an alternative, the Panel addressed the <u>actual changes</u> the licensee made to

[39] GDC 2 and Part 100, Appendix A, functional design based required: 1) integrity of the reactor coolant pressure boundary, 2) capability to shut down the reactor and maintain it in a safe condition, and 3) the SSCs needed to prevent or mitigate the consequences of accidents would remain functional given the maximum earthquake potential based on local geology.
[40] 10 CFR 50.71, "Maintenance of Records, Making of Reports," implemented by Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e), ML003740112, and Section 5 of NEI 98-03, Revision 1, Guidelines For Updating Final Safety Analysis. Changes to the FSAR may only be made after the licensee demonstrates that an amendment to the Operating Licensee is not required per 10 CFR 50.59.
[41] See footnote 30

Continued Item 11

the FSARU, Revision 21. The Report stated: "Consequently, there was insufficient basis to conclude that a license amendment was required to address the 2011 Shoreline report, and the NRC staff's recommendation for an FSAR updated was reasonable."

FSARU changes per 10 CFR 50.71(e), are subject to the previsions of 10 CFR 50.59.[42]
10 CFR 50.59 stated:

> *"A licensee shall obtain a license amendment pursuant to 50.90 prior to implementing a change, test or experiment if the change test or, experiment would:"*
>
> *" - Results in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety," or*
>
> *"- Results in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analysis"*

The new seismic information directly affected the information used in the FSARU safety analysis demonstrating that the GDC 2 design basis was satisfied. The licensee considered two cases.

For the first case, the licensee may update the existing FSARU safety analysis with the higher ground motions represented by the new seismic information. This update would result in the analyzed seismic stress to exceed ASME Code acceptance limits for reactor coolant system pressure boundary, major structures (reactor containment and auxiliary building), and the established qualification limits for important to safety SSCs (Regulatory Guide 1.29). NEI 96-07[43] (Section 4.3.2) stated that a change to *the facility as described in the FSARU* that results <u>in exceeding limits for seismic qualification</u> required prior NRC approval because of the increased likelihood of a malfunction of SSCs important to safety (during an earthquake).

For the second case, the licensee may use a different analytical method to demonstrate that the GDC 2 design basis was still satisfied given the increased ground motions. The licensee determined that HE methodology could be applied to the new ground motions without exceeding established plant SSC seismic qualification limits. This case also required prior NRC approval because the new or proposed method (the HE) <u>yielded results that were non-conservative when compared to the FSARU method</u> (NEI 96-07, Section 4.3.8).

As required by 10 CFR 50.59 and 10 CFR 50.90, the licensee requested NRC approval to use the HE method (LAR 2011-05) to demonstrate that the GDC 2 design basis was satisfied at the higher ground motions. The NRC subsequently concluded that the HE method was not appropriate for the SSE and requested that the licensee withdrawn the LAR.

Similarly, the licensee's action to revise the FSARU (Revision 21) to include the Shoreline (and presumably the San Luis Bay and Los Osos) fault(s) as lessor case(s) of the HE also required prior NRC approval. All of these faults are physically located within 75 miles of the site and are not

---

[42] Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e), ML003740112, and NEI 98-03, Revision 1,Guidelines For Updating Final Safety Analysis. Changes to the FSAR may only be made after the licensee demonstrates that an amendment to the Operating Licensee is not required per 10 CFR 50.59.
[43] Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments, (ML003759710) endorsed NEI 96-07, "Guidelines for10 CFR 50.59 Evaluations" ML003636043) as an acceptable method for implementation of 10 CFR 50.59.

Continued Item 11

associated with the Hosgri fault. As defined in the CLB (FSARU Section 2.5), deterministic ground motions that may be produced by these faults are within the scope of the GDC 2 SSE safety analysis. To limit the effect of these new faults on plant SSC to only the HE methodology was also a change to *the facility as described in the FSARU.* The end result was to exclude the Shoreline, San Luis Bay, and Los Osos faults from the GDC 2 design basis and safety analysis. This action also required prior NRC approval because the new or proposed method (the HE method) yielded results that were non-conservative when compared to the FSARU method (NEI 96-07, Section 4.3.8).

Technical Speciation Operability

The Panel Report stated:

> *"For situations without specific technical specification testing requirements, evaluations can be performed by the licensee to determine if the equipment can still perform its design function using appropriate evaluation methods. There is not a regulation that requires the methods used in the original design calculations must be used in these evaluations. Many times, engineering evaluation methods have changed since the original Construction Permit application was made. This is particularly true for seismic hazards. Modern methods are frequently used to show the equipment can still perform its function. Typical equipment installed at the facility had margin above the minimums that the design basis calculations required."*

The Panel concluded that NRC operability guidance (IMC 0326)[44] allowed the licensee to use an alternative method for demonstrating that SSC specified safety functions could still be met at the higher ground motions. The Panel Report stated that the use of the HE or LTSP "is attractive because the methods used in the LTSP are improved over those of initial licensing."

The Panel Report did not address the specific issues raised in the DPO related to the licensee's use of these "alternative methods." The DPO stated that the licensee's use of the HE (or the LTSP) was inappropriate for operability because these methods over-predict SSC performance when compared to the GDC 2 CLB analysis methods. The NRC provides use of "alternative methods"[45] to allow latitude for complex operability evaluations. The NRC restricts use of "alternative methods" that create additional margin when compared to the design basis method. For the new seismic information, the licensee had already established that SSC acceptance limits were exceeded using the GDC 2 design basis method. At this point, the licensee should have declared these SSCs inoperable and applied the required technical specification actions.

The DPO stated that the ASME Code acceptance limits are exceeded for reactor coolant pressure boundary components when the SSE seismic stresses are adjusted for the new higher ground motions. The Panel Report stated:

> *"The FSARU identifies both the DDE and the Hosgri as faulted conditions for use in the seismic stress levels for appropriate component and piping and demonstrates how it meets the appropriate ASME acceptance criteria. The use of both the DDE and the Hosgri in the evaluation is consistent with Panel's conclusion that both these limits are, at times, applicable as the limiting load."*

[44] Inspection Manual Chapter 0326, Operability Determinations and Functionality Assessments for Conditions Adverse to Quality or Safety (ML13274A578)
[45] (IMC 0326, Appendix C-04)

Continued Item 11

The Panel conclusion was based on the assumption that either the HE or SSE methodology could be used to satisfy Code requirements. Since the new ground motions were lower than those assumed for the HE, the HE method would result in meeting Code acceptance limits (assuming that the licensee included the required load combinations).

The Panel's conclusion did not consider the specific ASME Code and CLB requirements. The CLB, the Code, and 10 CFR 50.55a required the licensee to demonstrate that combined accident and SSE seismic loading be maintained below acceptance limits. Calculating the HE loading alone did not satisfy this requirement. The CLB clearly established the DDE as the SSE[46]. The HE was not the SSE. Neither the Code nor NRC Operability policy included provision to substitute the HE for the DDE/SSE to satisfy Code compliance. As a minimum, the DDE/SSE loads must meet acceptance limits. Also, as described in the DPO, for a given ground motion, the calculated stress will always be more limiting for the DDE/SSE method than for the HE. Because the Code specified that SSE loads be used, an amendment to the Operating License modifying the facility SSE design bases would be required before the HE could be used for Code compliance.

As described in the DPO, Code limits are exceeded when applying the new ground motions to the existing SSE Code calculations. Contrary to the Panel Report, IMC 0326, Appendix C.11, stated that a responsible expectation of operability cannot exist when Code requirements are not satisfied:

> *"ASME Class 1[47] components do not meet ASME Code or construction code acceptance standards, the requirements of an NRC endorsed ASME Code Case, or an NRC approved alternative, then an immediate operability determination cannot conclude a reasonable expectation of operability exists and the components are inoperable. Satisfaction of Code acceptance standards is the minimum necessary for operability of Class 1 pressure boundary components because of the importance of the safety function being performed."*

PG&E should have immediately declared ASME Class 1 components (reactor coolant pressure boundary) inoperable once they concluded "exceedances" existed with the higher ground motions.

The CLB stated that licensee demonstrated that Code limits were met for certain HE faulted cases. However, neither the ASME Code nor 10 CFR 50.55a required the licensee to perform these calculations. The license performed these calculations to meet a licensing bases commitment, not to satisfy design bases or a regulatory requirement.

Existing NRC Expectations Following Discovery of New Conditions Outside the Bounds of the Safety Analysis

The DPO Panel Report transmittal letter stated:

> "Finally, the Panel concluded that the lack of formal regulatory guidance for evaluating new information of natural hazards appears to be a contributing cause in creating many of the differing interpretations for potential significance of the information, along with confusion with regard to the regulatory process for evaluating the impact of new seismic information on system operability."

The agency has provided sufficient formal regulatory guidance for evaluating new information, including information affecting natural hazards. The DPO was written because the NRC staff failed

[46] See Appendix A and B of this report. DDE is the SSE for DCPP and HE did not include accident LOCA loads.
[47] Class 1 components make up the reactor coolant pressure boundary and pipe/component supports.

Continued Item 11

to follow this formal guidance during disposition of the Diablo Canyon seismic issues. This existing guidance included:

1) NRC Regulatory Issues Summary (RIS) 2013-05:[48] This RIS addressed questions raised about the relationship between licensing basis design requirements, the GDCs, and technical specification operability.

> "It is the staff's position that **failure to meet a GDC**, as described in the licensing basis (e.g., non-conforming with the CLB for protection against flooding, **seismic**, tornadoes) **should be treated as a nonconforming condition** and is an entry point for an operability determination if the non-conforming condition calls into question the ability of the SSCs to perform their specified safety functions(s) or necessary and related support functions(s)."

> "The safety analysis report describes the design capability of the facility to meet the GDC (or a plant-specific equivalent). The staff safety evaluation report documents the acceptability of safety analysis report analyses. The analyses and evaluation included in the safety analysis serve as the basis for TS issued with the operating license. The TS limiting conditions for operation, according to 10 CFR 50.36(c)(2)(i), "are the lowest functional capability or performance levels of equipment required for safe operation of the facility." Section 182 of the Atomic Energy Act of 1954, as amended and as implemented by 10 CFR 50.36, requires that those design features of the facility that, if altered or modified, would have a significant effect on safety, be included in the TS. Thus, TS are intended to ensure that the most safety significant design features of a plant, as determined by the safety analysis, maintain their capability to perform their safety functions, i.e., that SSCs are capable of performing their specified safety functions or necessary and related support functions."

> "Thus, an operability determination is appropriate upon identification of a degraded or nonconforming condition that calls into question the ability of SSCs to perform their specified safety function, including any nonconforming condition with a GDC included in either the CLB for an SSC described in TS or for a necessary and related support function required by the definition of operability. If the licensee determination concludes that the TS SSC is nonconforming but operable or the necessary and related support function is nonconforming but functional, it would be appropriate to address the nonconforming condition through the licensee's corrective action program."

2) Formal NRC regulatory guidance letter related to seismic hazard reevaluations:[49] This supplemental information reinforced agency regulations to address non-conforming conditions associated with the CLB:

> "During the course of stakeholder interactions regarding the hazard reevaluations, various questions were raised with respect to operability and reportability of systems, structures, and components (SSC) if the reevaluated seismic hazard is not bounded by the current seismic design basis."

> "However, as with any new information that may arise at a plant, licensees are responsible for evaluating and making determinations related to operability, and any associated reportability, on a case-by-case basis. Licensees should consider and disposition the information through their corrective action program or equivalent process. **If an error is identified in the current design or licensing basis during the performance of the requested seismic hazard evaluation, the staff expects that licensees would assess the operability of the affected SSC.** Additionally, licensees would need to determine if the situation is reportable pursuant to 10 CFR 50.72 and 50.73. Licensees would also be expected to determine whether aspects of 10 CFR 50.9, concerning the requirement to provide complete and accurate information to the NRC, would be applicable."

---

[48] RIS 2013-05, NRC Position on the Relationship between General Design Criteria and Technical Specification Operability (ML13056A077)

[49] Letter from E Leeds, Supplemental Information Related To Request For Information Pursuant To Title 10 of The *Code Of Federal Regulations* 50.54(f) Regarding Seismic Hazard Reevaluations For Recommendation 2.1 of the Near-Term Task Force Review of Insights From The Fukushima Dai-Ichi Accident, February 20, 2014 (ML14030A046)

Continued Item 11

At DCPP, PG&E developed new information that identified invalid inputs (errors) were used in the CLB safety analysis that demonstrated that the GDC 2 seismic design basis was met.

3) Inspection Manual Chapter 0326:[50] IMC provided formal regulatory guidance for evaluating new information of natural hazards. Section C.1 stated:

> "Failure to meet GDC, as described in the licensing basis (e.g., nonconformance with the CLB for protection against flooding, **seismic events**, tornadoes) should be treated as a nonconforming condition and is an entry point for an operability determination if the **nonconforming condition** calls into question the ability of SSCs to perform their specified safety function(s) or necessary and related support function(s). If the licensee determination concludes that the TS SSC is nonconforming but operable or the necessary and related support function is nonconforming but functional, it would be appropriate to address the nonconforming condition through the licensee's corrective action program. However, **if the licensee's evaluation concludes that the TS SSC is inoperable, then the licensee must enter its TS and follow the applicable required actions."**

4) The NRC enforced CLB GDC 2 flooding requirement's at Watts Bar.[51] Tennessee Valley Authority personnel identified that the spillway coefficient used to model flow from an upstream dam needed to be updated. Utility engineers found that the updated coefficient reduced the amount of spillway flow expected during periods of heavy rain. The reduction of spillway flow affected safety analysis inputs used to demonstrate that the facility met the GDC 2 design bases for maximum flood height. This case was very similar to the DCPP. At both facilities, new information affected the outcome of GDC 2 safety analyses and the capability of plant SSCs to perform the required safety functions. In the Watts Bar case, the new information resulted in a higher maximum flood height. In the DCPP case, the new information resulted in an increase in the amount of seismic stress affecting plant SSCs following an earthquake. In both cases, the licensees failed to take prompt corrective actions to correct the non-conforming safety analysis. However, for the Watts Bar case, the agency enforced statutory design control requirements. This enforcement action included:

   - A Severity Level III violation for failing to report an unanalyzed condition related to external flooding
   - A Yellow Finding following the failure to maintain an adequate abnormal condition procedure to implement the flood mitigation strategy
   - A White Finding following inadequate abnormal condition procedure for flood mitigation strategy.

5) The NRC also enforced GDC 2 CLB flooding requirements at several other facilities. For example, the NRC issued a Yellow Finding at the Monticello facility.[52] In the Monticello case, the licensee was unable to implement flood protection barriers consistent with the GDC 2 flooding safety analysis.

---

[50] IMC 0236, Operability Determinations and Functionality Assessments for Conditions Adverse to Quality or Safety (ML13274A578), Section 3.60 defined nonconforming condition and Section C-1 included the failure to meet a GDC as a non-conforming condition, Section C-11 defined the requirement to meet ASME

[51] Watts Bar Unit 1 Nuclear Plant - Final Significance Determination Of Yellow Finding, White Finding And Notices Of Violations; Assessment Follow-Up Letter; Inspection Report No. 05000390/2013009, EA-13-018, June 4, 2013.

[52] Final Significance Determination of A Yellow Finding With Assessment Follow up and Notice of Violation; NRC Inspection Report No. 5000263/2013009; Monticello Nuclear Generating Plant, EA-13-096, August 28, 2013.

Continued Item 11

Fukushima Term Task Force Recommendations 2.1 and 2.3

The Panel Report and Research Information Letter 12-01[53] both stated that the Fukushima Recommendation 2.1, Seismic Reevaluations,[54] will address the DCPP seismic issues. While the seismic reevaluations are designed to assess the seismic hazard for the facility, these ongoing activities do not address the concerns raised in the DPO. The DPO focused on the failure of agency personnel in enforce CLB requirements, not on how seismic hazards are evaluated. The requested seismic reevaluation will provide context for the agency to determine if the CLB should be modified.

In contrast, one purpose of Recommendation 2.3,[55] was to confirm that CLB seismic requirements were met while the seismic reevaluations are performed. Verification that the plant was operating within the bounds of the current design and licensing bases provided confidence that the plant was safe while the reevaluations are performed:

> "Structures, systems, and components (SSCs) important to safety in operating nuclear power plants are designed either in accordance with, or meet the intent of, Appendix A to 10 CFR Part 100 and Appendix A to 10 CFR Part 5O, General Design Criteria (GGC) 2. GDC 2 states that SSCs important to safety at nuclear power plants must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without loss of capability to perform their intended safety functions. The design bases for these SSCs are to reflect appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area. The design bases are also to reflect sufficient margin to account for the limited accuracy, quantity, and period of time in which the historical data have been accumulated."

> "In response to NTTF Recommendation 2.3, the Commission requests all licensees to perform seismic walkdowns in order to identify and address plant specific degraded, nonconforming, or unanalyzed conditions and verify the adequacy of strategies, monitoring, and maintenance programs such that the nuclear power plant can respond to external events. The walkdown will verify current plant configuration with the current licensing basis, verify the adequacy of current strategies, maintenance plans, and identify degraded, **nonconforming, or unanalyzed conditions."**

> "If any condition identified during the walkdown activities represents a degraded, nonconforming, or unanalyzed condition (i.e., noncompliance with the current licensing basis) for an SSC, describe actions that were taken or are planned to address the condition using the guidance in Regulatory Issues Summary 2005-20, Revision 1, Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Conditions Adverse to Quality or Safety," including entering the condition in the corrective action program. Reporting requirements pursuant to 10 CFR 50.72 should also be considered. Additionally, these findings should be considered in the Recommendation 2.1 hazard evaluations, as appropriate."

As detailed in the DPO, DCPP continues to operate in both unanalyzed and non-conforming conditions outside of the bounds of the CLB.

---

[53] Diablo Canyon Power Plant, Unit Nos. 1 And 2 -NRC Review of Shoreline Fault (TAC NOS. ME5306 AND ME5307), October 12, 2012 (ML120730106).
[54] Request For Information Pursuant To Title 10 Of The Code of Federal Regulations 50.54(F) Regarding Recommendations 2.1,2.3, And 9.3, of The Near-Term Task Force Review Of Insights From The Fukushima Dai-Ichi Accident (ML12053A340)
[55] See Footnote 51.

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

Summary

The existing regulatory framework for addressing the enforcement and operability issues raised in DPO 2013-002 are well established. NRC regulations[56] required PG&E to take prompt corrective action after developing new seismic information that concluded that the GDC 2 safety analysis was no longer bounding for the seismic qualification of plant SSCs. These actions also required the licensee to either incorporate the new seismic information into the existing safety analysis or establish a new methodology for demonstrating that the functional design bases requirements of GDC 2 remained satisfied.[57] Either approach required an amendment to the DCPP Operating License per 10 CFR 50.59[58] and 10 CFR 50.90.

PG&E requested that the NRC approve the HE, as a new method for the facility SSE. However, the NRC concluded that this new methodology was not appropriate for establishing the facility SSE and requested that the licensee withdraw the LAR. After the license amendment process was unsuccessful, the NRR PM provided the licensee direction to work around the amendment process by directly adding the new information to the FSARU. This action subverted the license amendment public notice requirements and hearing opportunities as prescribed by 10 CFR 50.91.

PG&E continued to operate the DCPP reactors following discovery of the unanalyzed condition and non-conforming safety analysis. The licensee was required to demonstrate that technical specifications SSCs would still be capable of performing the safety functions specified in the safety analysis at the higher seismic stress levels. The licensee's use of the HE "alternative method" for this demonstration was not consistent with NRC policy. The HE was inappropriate because for a given ground motion, the HE would always over-predict SSC seismic performance when compared to the SSE design basis method. Also, the licensee's use of the HE to demonstrate that reactor coolant pressure boundary integrity would be maintained during an earthquake was inconsistent with ASME Code requirements and 10 CFR 50.55a.

The DPO Panel concluded that an amendment to Operating License was not required to disposition the new seismic information. The Panel also concluded that the licensee satisfied all statutory requirements. The Panel's conclusions were based on the inappropriate assumption that GDC 2 SSE design basis was established by a combination of the DDE safety analysis and the HE. From this assumption, the Panel extrapolated that the new information was within the existing SSE GDC 2 design basis because the new ground motions were bound by <u>either</u> the DDE <u>or</u> the HE. The Panel Report did not include the bases for either of these assumptions.

This DPO Appeal demonstrates that the Panel's conclusions were incorrect because the underlying assumptions used to formulate those conclusions were inconsistent with the CLB. The CLB clearly described that the DDE was the facility SSE and the supporting DDE safety analysis demonstrated that the GDC 2 design basis was met. Even if the HE was considered part of the 10 CFR 50.2 design bases, then Panel Report provided inadequate justification to exclude the

[56] Appendix B to Part 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Criterion III. Design Control, and XVI. Corrective Action.
[57] 10 CFR50.71(e) required the FSARU to include all analyses of new safety issues affecting the originally license application to assure that the information included in the report contains the latest information developed
[58] 10 CFR 50.59 required an amendment to the Operating License for FSARU changes that "result in a departure from a method of evaluation described in the FSAR (as updated) used in establishing the design bases or in the safety analyses."

Continued Item 11

DDE/SSE safety analysis from the requirements of 10 CFR 50.59, 50.71(e), and Part 50, Appendix B. In either case, the new ground motions must be evaluated within the context of GDC 2 design bases and limiting SSC seismic qualification requirements.

Requested Action

Please take the following actions:

1. Disapprove the Panel Report depositing DPO 2013-002.

2. Initiate regulatory enforcement action to address the ongoing non-compliances with Part 50, Appendix B, 10 CFR 50.59, and plant technical specifications at DCPP.

3. Initiate a review to determine why the non-concurrence (NCP 2012-01) and the DPO process were not effective to address the outstanding DCPP seismic issues.

Thank you,
Michael Peck, Ph.D.

Attachments:
Appendix A, Original Diablo Canyon Seismic Licensing Bases
Appendix B, Current Diablo Canyon Seismic Licensing Bases
Appendix C, Pacific Gas and Electric Company Nuclear Power Generation, Classification of Structures, Systems, and Components for Diablo Canyon Power Plant Units 1 And 2 (Q-LIST), Revision 27

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

Appendix A

Original Diablo Canyon Seismic Licensing Bases

3.2.1 Seismic Classification

Criterion 2 of the July 1967 AEC General Design Criteria, and Appendix A to 10 CFR 100, Seismic and Geologic Siting Criteria for Nuclear Power Plants, require that nuclear power plant structures, components, and systems important to safety be designed to withstand the effects of earthquakes. Specifically, Appendix A to 10 CFR 100 requires that all nuclear power plants be designed so that, if the safe shutdown earthquake (SSE) occurs, all structures and components important to safety remain functional. Plant features important to safety are those necessary to ensure: (a) the integrity of the reactor coolant pressure boundary, (b) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (c) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

GDC 2 and Part 100, Appendix A, established the design basis requirements for the SSE.

These are the design basis functional criteria established by Part 100, Appendix A for the SSE for meeting GDC 2 seismic requirements.

The SSE of Appendix A to 10 CFR 100, is equivalent to the DCPP double design earthquake (DDE). Similarly, the operating basis earthquake (OBE) of Appendix A to 10 CFR 100, is equivalent to the DCPP design earthquake (DE).

The SSE is equivalent to the DDE

DCPP's capability to withstand a postulated Richter magnitude 7.5 earthquake centered along an offshore zone of geologic faulting known as the "Hosgri Fault" is reviewed.

In additional to GDC 2 and Part 100, Appendix A, PG&E review the effect of an 7.5 M earthquake on the Hosgri fault

DCPP UNITS 1 & 2 FSAR UPDATE

Guidance for determining the structures, components, and systems designed to remain functional in the event of an SSE is provided in AEC Safety Guide 29. These plant features, including their foundations and supports, are designated as Seismic Category I in Safety Guide 29. DCPP structures, components, and systems, and their seismic design classifications comply with the intent of Safety Guide 29. However, since DCPP design and construction had progressed substantially prior to the issuance of Safety Guide 29, different terminology is used. The terms Category I and SSE are not used.

PG&E committed to Safety Guide 29, "Seismic Design Classification" (Regulatory Guide 1.29) for those SSCs required to remain functional following the SSE/DDE (not the HE)

Plant features that correspond to seismic Category I, as identified in AEC Safety Guide 29, are designated as Design Class I, and these are designed to remain functional in the event of a DDE.

Structures, components, and systems not identified as Seismic Category I in AEC Safety Guide 29, are referred to by the guide as Nonseismic Category I features. The classification system for DCPP categorizes Nonseismic Category I as either Design Class II, IIA, or III.

Structures, components, and systems important to reactor operation but not essential to safe shutdown and isolation of the reactor, and failure of which would not result in the release of substantial amounts of radioactivity, are classified as Des'gn Class II.

Structures, components, and systems important to reactor operation but not safety-related may be designated Design Class IIA and will not necessarily have been designed or constructed under a quality program meeting all requirements of Chapter 17. However, activities such as repair, replacement, maintenance, or testing will be performed under the operational quality assurance program. Quality requirements administered shall be commensurate with the safety function of the structure, component, or system.

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

Structures, components, and systems not related to reactor operation or safety are classified as Design Class III.

Power and auxiliary service piping systems (as defined in ANSI standard B.31.1, Paragraph 100.1), which might otherwise be considered as Design Class III, are classified as Design Class II (i.e., Design Class III is not used for power and auxiliary service piping systems).

In addition, Appendix B to 10 CFR 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, requires that structures, components, and systems important to safety be designed and constructed in accordance with the quality assurance requirements described in Appendix B. Therefore, as described in Chapter 17, the requirements of the DCPP Quality Assurance Program apply to all structures, components, and systems classified as Design Class I. This ensures that plant features important to safety have met the requirements of Appendix B.

Part 50, Appendix B applies to Design Class 1 (tied back to the SSE and RG 1.29)

Piping Schematic Correlation

Piping Symbol	Design Class	Quality Code Class
A	I	I
B	I	II
[illegible]	I	II
C	I	III
D	I	III
E	II	None
F	IIA	None
G	I	None
G1	II	None
H	IIA	None
J	I	III

Those structures, components, and systems, including their foundations and supports, that have been classified as Design Class I and designed to remain functional in the event a DDE occurs, and to which the requirements of the Quality Assurance Program apply, are:

(1) The reactor coolant pressure boundary

(2) The reactor core and reactor vessel internals

(3) Systems [see note(1)][a], or portions of systems that are required for emergency core cooling, postaccident containment heat removal, or postaccident containment atmosphere cleanup

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

3.1.2.1 Criterion 1 - Quality Standards (Category A)

Those systems and components of reactor facilities that are essential to the prevention of accidents which could affect the public health and safety, or mitigation of their consequences, shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to ensure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Discussion

All systems and components of DCPP Units 1 and 2 are classified according to their importance in the prevention and mitigation of accidents. Those items vital to safe shutdown and isolation of the reactor, or whose failure might cause or increase the severity of a LOCA, or result in an uncontrolled release of excessive amounts of radioactivity, are designated Design Class I. Those items important to the reactor operation, but not essential to safe shudown and isolation of the reactor or control of the release of substantial amounts of radioactivity, are designated Design Class II or IIA. Those items not related to reactor operation or safety are designated Design Class III.

Part 50, Appendix B applies to Design Class 1 (tied back to the SSE and RG 1.29)

Design Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected, erected, and the materials selected to the applicable provisions of recognized codes, good nuclear practice, and to quality standards that reflect their importance. Discussions of applicable codes and standards as well as code classes are given in Section 3.2 for the major items and components. The quality assurance program conforms with the requirements of 10 CFR 50 Appendix B, Quality Assurance Criteria for Nuclear Power Plants. Details of the QA program are given in Chapter 17.

3.1.2.2 Criterion 2 - Performance Standards (Category A)

Those systems and components of reactor facilities that are essential to the prevention of accidents which could affect the public health and safety, or to mitigation of their consequences, shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect:

DCPP UNITS 1 & 2 FSAR UPDATE

(a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Discussion

All systems and components designated Design Class I are designed so that there is no loss of function for ground acceleration associated with two times the design earthquake (DE) acting in the horizontal and vertical directions simultaneously. The ESF are included in the above. The working stresses for Class I, Class II, and Class IIA items are kept within code allowable values for the DE. Similarly, measures are taken in the plant design to protect against possible effects of tsunamis, lightning storms, strong winds, and other natural phenomena.

GDC 2 was met by the SSE/DDE

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

3.2.2.1 Design Class I, Quality/Code Class I Fluid Systems and Fluid System Components
Section 50.55a of 10 CFR 50, Codes and Standards, requires that certain components of the reactor coolant pressure boundary be designed, fabricated, erected, and tested in accordance with the requirements for Class A[(a)] components of Section III of the ASME Boiler and Pressure Vessel Code, or the highest available industry codes and standards. Code Class I has been applied to those components of the reactor coolant pressure boundary and implements the quality standards that satisfy the requirements of Section 50.55a, 10 CFR 50. DCPP Code Class I components of the reactor coolant pressure boundary are listed in Table 3.2-3, along with the industry codes and standards used for their design, fabrication, erection, and test. The Code Class I classification includes the components of the reactor coolant pressure boundary identified as Safety Class 1 in ANSI N18.2, and Quality Group A in AEC Safety Guide 26.

RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants" ties 10 CFR 50.55a and ASME Section III Code requirements were tied back to the SSE

3.2.2.2 Design Class I, Quality/Code Class II Fluid Systems and Fluid System Components
Generally, Code Class II has been applied to include fluid systems and fluid system components that are either:

(1) Part of the reactor coolant boundary, but excluded from Code Class I requirements by Section 50.55a of 10 CFR 50

(2) Not part of the reactor coolant pressure boundary, but part of:

Code Class II fluid systems and fluid system components are listed in Table 3.2-3, along with the industry codes and standards used for their design, fabrication, erection, and testing. The Code Class II classification generally includes the fluid systems and components identified as Safety Class 2a in ANSI N18.2, and Quality Group B in AEC Safety Guide 26. However, the classification and quality standards for DCPP fluid systems and components were established prior to the existence of these documents and therefore do not always fall within their strict definitions. All Code Class II fluid systems and components are in accordance with the accepted industry codes and standards that were in effect during the design and construction of DCPP. If fluid systems and components were designed and constructed to codes and standards outside of the requirements of the above mentioned documents, additional quality standards have normally been applied so that their intent has been met.

3.2.2.3 Design Class I, Quality/Code Class III Fluid Systems and Fluid System Components
Generally, Code Class III has been applied to include fluid systems and fluid system components not part of the reactor coolant pressure boundary, nor included in Code Class II, but part of:

3.2.2.5 Summary of System Quality Group Classifications
Table 3.2-2 summarizes the design and quality group classifications applied to the DCPP structures systems and components, and their relationships to the other methods of classification.

Generally, codes and standards were applied prior to issuance of the latest codes and standards, such as the 1971 edition of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components. In some cases, fluid systems and components were designed and built to codes and standards outside the requirements of Safety Guide 26 or ANSI N18.2 definitions. The classification for those fluid systems and fluid system components that do not fall within the strict definition of AEC Safety Guide 26, and ANSI N18.2 were established prior to ANSI N18.2 and Safety Guide 26, and the issuance of revised industry codes and standards. For these fluid systems and fluid system components, the design specifications specified the accepted industry codes and standards in effect during the design and construction of DCPP.

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

DCPP UNITS 1 & 2 FSAR UPDATE

2.5 GEOLOGY AND SEISMOLOGY

This section presents the findings of the regional and site-specific geologic and seismologic investigations of the Diablo Canyon Power Plant (DCPP) site. Information presented is in compliance with the criteria in Appendix A of 10 CFR 100 and meets the format and content recommendations of Regulatory Guide 1.70, Revision 1[39].

Location of earthquake epicenters within 200 miles of the plant site, and faults and earthquake epicenters within 75 miles of the plant site for either magnitudes or intensities, respectively, are shown on Figures 2.5-2, 2.5-3, and 2.5-4. A geologic and tectonic map of the region surrounding the site is given on two sheets of Figure 2.5-5, and detailed information about site geology is presented on Figures 2.5-8 through 2.5-16. Geology and seismology are discussed in detail in Sections 2.5.1 through 2.5.4. Additional information on site geology is contained in References 1 and 2.

Detailed supporting data pertaining to this section are presented in References 3, 4, 8, and 9. Geologic and seismic information from investigations that responded to Nuclear Regulatory Commission (NRC) licensing review questions are presented in References 17 and 18 and in Chapter 3 of the Hosgri Report[34]. A brief synopsis of the information presented in the following sections is given below.

Seismic safety analysis demonstrating GDC 2 was in compliance with Part 100, Appendix A, and RG 1.70. RG 1.70 required any exception taking to GDC 2 (Part 50, Appendix A) to be identified. PG&E did not list any exceptions.

The GDC 2, SSE/DDE safety analysis included all faults within 75 miles of the plant site.

The Hosgri fault was an exception to GDC 2, "responded to NRC license review questions."

DCPP UNITS 1 & 2 FSAR UPDATE

3.7 SEISMIC DESIGN

3.7.1 Seismic Input

This chapter relates to the design earthquake (DE), the double design earthquake (DDE), and the postulated 7.5M Hosgri earthquake (HE).

3.7.1.1 Design Response Spectra

Section 2.5.2 provides a discussion of the earthquakes postulated for the Diablo Canyon Power Plant (DCPP) site, and the effects of these earthquakes in terms of maximum free-field ground motion accelerations and corresponding response spectra at the plant site. The maximum vibratory accelerations at the plant site would result from either Earthquake B or Earthquake D-modified, depending on the natural period of the vibrating body. Response acceleration spectra curves for horizontal free-field ground motion at the plant site from Earthquake B and Earthquake D-modified are presented on Figures 2.5-20 and 2.5-21, respectively.

For design purposes, the response spectra for each damping value from Earthquake B and Earthquake D-modified are combined to produce an envelope spectrum. The acceleration value for any period on the envelope spectrum is equal to the larger of the two values from the Earthquake B spectrum and the Earthquake D-modified spectrum. Vertical free field ground accelerations, and the vertical free-field ground motion response spectra are assumed to be two-thirds of the corresponding horizontal spectra.

The design earthquake (DE) is the hypothetical earthquake that would produce these horizontal and vertical vibratory accelerations. The DE corresponds to the operating basis earthquake (OBE), as described in Appendix A to 10 CFR 100[7].

In order to ensure adequate reserve energy capacity, Design Class I structures and equipment are reviewed for the DDE. The DDE is the hypothetical earthquake that would produce accelerations twice those of the DE. The DDE corresponds to the Safe Shutdown Earthquake (SSE), as described in Appendix A to 10 CFR 100[7].

The Pacific Gas and Electric Company (PGandE) requested by the Nuclear Regulatory Commission (NRC) to evaluate the plant's capability to withstand a postulated Richter magnitude 7.5 earthquake centered along an offshore zone of geologic faulting, generally referred to as the Hosgri fault. This evaluation is discussed in the various Chapters when it is specifically referred to as the "Hosgri Evaluation" or "Hosgri Event" (HE).

GDC 2 design bases is established by the DDE/SSE safety analysis

PG&E was requested to evaluate the HE, not part of the GDC 2 design basis

The SSE is equivalent to the DDE. Design Class 1 (from Section 3.1) was tied back to the SSE/DDE.

HE was provided in response to license questions, not the GDC 2 design basis.

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

DCPP UNITS 1 & 2 FSAR UPDATE

2.5.4.9 Earthquake Design Basis

The earthquakes postulated for DCPP site are discussed in Section 2.5.2.9, and a discussion of the design response spectra is in Section 3.7. Response acceleration curves for the site resulting from Earthquake B and Earthquake D-modified are shown on Figures 2.5-20 and 2.5-21, respectively.

DCPP UNITS 1 & 2 FSAR UPDATE

(4) Level IV - Potential for earthquakes and aftershocks resulting from crustal movements that cannot be associated with any near-surface fault structures: such earthquakes apparently can occur almost anywhere in the region.

2.5.2.10 Ground Accelerations and Response Spectra

The maximum ground acceleration that would occur at DCPP site has been estimated for each of the postulated earthquakes listed in Section 2.5.2.9, using the methods set forth in References 12 and 24. The plant site acceleration is primarily dependent on the following parameters: Gutenberg-Richter magnitude and released energy, distance from the earthquake focus to the plant site, shear and compressional velocities of the rock media, and density of the rock. Rock properties are discussed under Section 2.5.1.2.6, Site Engineering Properties.

Development of the maximum ground accelerations for GDC 2 (the Hosgri was excluded)

The maximum rock accelerations that would occur at the DCPP site are estimated as:

Earthquake A	0.10 g	Earthquake C . . .	0.05 g
Earthquake B	0.12 g	Earthquake D . . .	0.20 g

In addition to the maximum acceleration, the frequency distribution of earthquake motions is important for comparison of the effects on plant structures and equipment. In general, the parameters affecting the frequency distribution are distance, properties of the transmitting media, length of faulting, focus depth, and total energy release. Earthquakes that might reach the site after traveling over great distances would tend to have their high frequency waves filtered out. Earthquakes that might be centered close to the site would tend to produce wave forms at the site having minor low frequency characteristics.

In order to evaluate the frequency distribution of earthquakes, the concept of the response spectrum is used.

For nearby earthquakes, the resulting response spectra accelerations would peak sharply at short periods and would decay rapidly at longer periods. Earthquake D would produce such response spectra. The March 1957 San Francisco earthquake as recorded in Golden Gate Park (S80°E component) was the same type. It produced a maximum recorded ground acceleration of 0.13 g (on rock) at a distance of about 8 miles from the epicenter. Since Earthquake D has an assigned hypocentral distance of 12 miles, it would be expected to produce response spectra similar in shape to those of the 1957 event.

Large earthquakes centered at some distance from the plant site would tend to produce response spectra accelerations that peak at longer periods than those for nearby smaller shocks. Such spectra maintain a higher spectral acceleration throughout the period range beyond the peak period. Earthquakes A and C are events that would tend to produce this type of spectra. The intensity of shaking as indicated by the maximum predicted ground acceleration shows that Earthquake C would always have lower spectral accelerations than Earthquake A.

Continued Item 11

Since the two shocks would have approximately the same shape spectra, Earthquake C would always have lower spectral accelerations than Earthquake A, and it is therefore eliminated from further consideration. The north-south component of the 1940 El Centro earthquake produced response spectra that emphasized the long period characteristics described above. Earthquake A, because of its distance from the plant site, would be expected to produce response spectra similar in shape to those produced by the El Centro event. Smoothed response spectra for Earthquake A were constructed by normalizing the El Centro spectra to 0.10 g. These spectra, however, show smaller accelerations than the corresponding spectra for Earthquake B (discussed in the next paragraph) for all building periods, and thus Earthquake A is also eliminated from further consideration.

Earthquake B would tend to produce response spectra that emphasize the intermediate period range inasmuch as the epicenter is not close enough to the plant site to produce large high frequency (short-period) effects, and it is too close to the site and too small in magnitude to produce large low frequency (long-period) effects. The N69°W component to the 1952 Taft earthquake produced response spectra having such characteristics. That shock was therefore used as a guide in establishing the shape of the response spectra that would be expected for Earthquake B.

Following several meetings with the AEC staff and their consultants, the following two modifications were made in order to make the criteria more conservative:

(1) The Earthquake D time-history was modified in order to obtain better continuity of frequency distribution between Earthquakes D and B.

(2) The accelerations of Earthquake B were increased by 25% in order to provide the required margin of safety to compensate for possible uncertainties in the basic earthquake data.

Accordingly, Earthquake D-modified was derived by modifying the S80°E component of the 1957 Golden Gate Park, San Francisco earthquake, and then normalizing to a maximum ground acceleration of 0.20 g. Smoothed response spectra for this earthquake are shown on Figure 2.5-21. Likewise, Earthquake B was derived by normalizing the N69°W component of the 1952 Taft earthquake to a maximum ground acceleration of 0.15 g. Smoothed response spectra for Earthquake B are shown on Figure 2.5-20.

The maximum vibratory motion at the plant site would be produced by either Earthquake D-modified or Earthquake B, depending on the natural period of the vibrating body.

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

Appendix B

Current Diablo Canyon Seismic Licensing Bases

DCPP UNITS 1 & 2 FSAR UPDATE

2.5 GEOLOGY AND SEISMOLOGY

This section presents the findings of the regional and site-specific geologic and seismologic investigations of the Diablo Canyon Power Plant (DCPP) site. Information presented is in compliance with the criteria in Appendix A of 10 CFR 100 and meets the format and content recommendations of Regulatory Guide 1.70, Revision 1[79].

Description of the safety analysis used to determine the SSE/DDE ground motion.

Location of earthquake epicenters within 200 miles of the plant site, and faults and earthquake epicenters within 75 miles of the plant site for either magnitudes or intensities, respectively, are shown in Figures 2.5-2, 2.5-3, and 2.5-4. A geologic and tectonic map of the region surrounding the site is given in two sheets of Figure 2.5-5, and detailed information about site geology is presented in Figures 2.5-8 through 2.5-16. Geology and seismology are discussed in detail in Sections 2.5.1 through 2.5.4. Additional information on site geology is contained in References 1 and 2.

The DDE/SSE safety analysis was compliant with 10 CFR 100, Appendix A.

Included all epicenters within 200 miles and faults within 75 miles of the plant.

On November 2, 1984, the NRC issued the Diablo Canyon Unit 1 Facility Operating License DPR-80. In DPR-80, License Condition Item 2.C (7), the NRC stated, in part:

> "PG&E shall develop and implement a program to reevaluate the seismic design bases used for the Diablo Canyon Power Plant."

PG&E's reevaluation effort in response to the license condition was titled the "Long Term Seismic Program" (LTSP). PG&E prepared and submitted to the NRC the "Final Report of the Diablo Canyon Long Term Seismic Program" in July 1988[40]. Between 1988 and 1991, the NRC performed an extensive review of the Final Report, and PG&E prepared and submitted written responses to formal NRC questions. In February 1991, PG&E issued the "Addendum to the 1988 Final Report of the Diablo Canyon Long Term Seismic Program"[41]. In June 1991, the NRC issued Supplement Number 34 to the Diablo Canyon Safety Evaluation Report (SSER)[42], in which the NRC concluded that PG&E had satisfied License Condition 2.C (7) of Facility Operating License DPR-80. In the SSER the NRC requested certain confirmatory analyses from PG&E, and PG&E subsequently submitted the requested analyses. The NRC's final acceptance of the LTSP is documented in a letter to PG&E dated April 17, 1992[43].

The LTSP was completed in 1988.

The LTSP contains extensive data bases and analyses that update the basic geologic and seismic information in this section of the FSAR Update. However, the LTSP material does not address or alter the current design licensing basis for the plant, and thus is not included in the FSAR Update. A complete listing of bibliographic references to the LTSP reports and other documents may be found in References 40, 41 and 42.

The LTSP did not address or alter the current design licensing basis for the plant. The LTSP was not included in the FSARU because the information is not part of the seismic design basis or supporting safety analysis.

2.5.2.8 Description of Active Faults

Active faults that have any part passing within 200 miles of the site are described in Section 2.5.1.1.2.

2.5.2.9 Maximum Earthquake

Benioff and Smith, in reviewing the seismicity of the region around DCPP site, determined the maximum earthquakes that could reasonably be expected to affect the site. Their conclusions regarding the maximum size earthquakes that can be expected to occur during the life of the reactor are listed below.

(1) Earthquake A: A great earthquake may occur on the San Andreas fault at a distance from the site of more than 48 miles. It would be likely to produce surface rupture along the San Andreas fault over a distance of 200 miles with a horizontal slip of about 20 feet and a vertical slip of 3 feet. The duration of strong shaking from such an event would be about 40 seconds, and the equivalent magnitude would be 8.5.

(2) Earthquake B: A large earthquake on the Nacimiento (Rinconada) fault at a distance from the site of more than 20 miles would be likely to produce a 60 mile surface rupture along the Nacimiento fault, a slip of 6 feet in the horizontal direction, and have a duration of 10 seconds. The equivalent magnitude would be 7.5.

(3) Earthquake C: Possible large earthquakes occurring on offshore fault systems that may need to be considered for the generation of seismic sea waves are listed below:

The safety analysis considered all active faults passing within 200 miles from the plant when determining the "maximum Earthquake" (DD) for the facility.

Differding Professional Opinion--Appeal

(Continued)

Continued Item 11

DCPP UNITS 1 & 2 FSAR UPDATE

(4) Earthquake D: Should a great earthquake occur on the San Andreas fault, as described in "A" above, large aftershocks may occur out to distances of about 50 miles from the San Andreas fault, but those aftershocks which are not located on existing faults would not be expected to produce new surface faulting, and would be restricted to depths of about 6 miles or more and magnitudes of about 6.75 or less. The distance from the site to such aftershocks would thus be more than 6 miles.

A further assessment of the seismic potential of faults mapped in the region of DCPP site has been made following the extensive additional studies of on- and offshore geology of the last few years that are reported in Appendix 2.5D of Reference 27 of Section 2.3. This was done in terms of observed Holocene activity, to achieve assessment of what seismic activity is reasonably probable, in terms of observed late Pleistocene activity, fault dimensions, and style of deformation.

PG&E was requested by the NRC to evaluate the plant's capability to withstand a postulated Richter Magnitude 7.5 earthquake centered along an offshore zone of geologic faulting, generally referred to as the "Hosgri fault." The detailed methods, results, and plant modifications performed based on this evaluation are dealt with in Section 3.7.

2.5.2.10 Ground Accelerations and Response Spectra

The maximum ground acceleration that would occur at DCPP site has been estimated for each of the postulated earthquakes listed in Section 2.5.2.9, using the methods set forth in References 12 and 24. The plant site acceleration is primarily dependent on the following parameters: Gutenberg-Richter magnitude and released energy, distance from the earthquake focus to the plant site, shear and compressional velocities of the rock media, and density of the rock. Rock properties are discussed under Section 2.5.1.2.6, Site Engineering Properties.

The maximum rock accelerations that would occur at the DCPP site are estimated as:

Earthquake A	0.10 g	Earthquake C	0.05 g
Earthquake B	0.12 g	Earthquake D	0.20 g

In addition to the maximum acceleration, the frequency distribution of earthquake motions is important for comparison of the effects on plant structures and equipment. In general, the parameters affecting the frequency distribution are distance, properties of the transmitting media, length of faulting, focus depth, and total energy release. Earthquakes that might reach the site after traveling over great distances would tend to have their high frequency waves filtered out. Earthquakes that might be centered close to the site would tend to produce wave forms at the site having minor low frequency characteristics.

3.2.1 SEISMIC CLASSIFICATION

Criterion 2 of the July 1967 GDC, and Appendix A to 10 CFR 100, Seismic and Geologic Siting Criteria for Nuclear Power Plants, require that nuclear power plant SSCs important to safety be designed to withstand the effects of earthquakes. Specifically, Appendix A to 10 CFR 100 requires that all nuclear power plants be designed so that, if the safe shutdown earthquake (SSE) occurs, all structures and components important to safety remain functional. Plant features important to safety are those necessary to ensure (a) the integrity of the reactor coolant pressure boundary, (b) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (c) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

The SSE of Appendix A to 10 CFR 100 is equivalent to the DCPP double design earthquake (DDE) (see References 9 and 10 for final resolution of issues raised in Supplemental Safety Evaluation Reports 7, 8, and 31 relative to the SSE). Similarly, the operating basis earthquake (OBE) of Appendix A to 10 CFR 100 is equivalent to the DCPP DE.

DCPP's capability to withstand a postulated Richter magnitude 7.5 earthquake centered along an offshore zone of geologic faulting known as the "Hosgri Fault" has been reviewed. Guidance for determining the SSCs designed to remain functional in the

The Diablo Canyon seismic design bases was based on a magnitude 7.25 earthquake on the Nacimiento fault, 20 miles from the site (Earthquake B), and a magnitude 6.75 aftershock associated with a large earthquake on the San Andreas fault (Earthquake D).

The safety analysis did not include consideration of the Hosgri fault when determining the "maximum earthquake" for the facility. The Hosgri Evaluation (HE) is described as a response to an NRC question, not part of the SSE/DDE design basis.

The safety analysis concluded the maximum peak ground acceleration would be about 0.2 g (grounded at 100 Hz). PG&E designated the SSE/DDE at twice this value, or 0.4 g (grounding at 100 Hz). This approach was accepted by the NRC as "equivalent" to 10 CFR 100, Appendix A.

The Diablo Canyon FSARU establishes the CLB regulatory and design basis requirements for SSC seismic qualification.

Diablo Canyon complied with 1967 GDC 2 and 10 CFR 100, Appendix A. PG&E also stated that the facility conformed to Part 50, Appendix A, GDC 2 (see Footnote 24 and the Appendix to the DPO).

The DDE is equivalent to the 10 CFR 100, Appendix A, SSE.

PG&E committed to Safety Guide 29, "Seismic Design Classification," (Regulatory Guide (RG) 1.29), to determine the set of SSCs required seismically qualified for the SSE/DDE. RG 1.29 provided an NRC acceptable method for this determination.

Differing Professional Opinion--Appeal

(Continued)

Continued Item 11

event of an SSE is provided in SG 29. These plant features, including their foundations and supports, are designated as Seismic Category I in SG 29. DCPP SSCs, and their seismic design classifications comply with the intent of SG 29. However, since DCPP design and construction had progressed substantially prior to the issuance of SG 29, different terminology is often used.

Plant features that correspond to Seismic Category I, as identified in SG 29, are designed to remain functional during the design basis earthquakes that they are required to withstand: the DE (equivalent to the OBE of SG 29), the DDE (equivalent to the SSE of SG 29), and/or the postulated Hosgri earthquake (HE). Design Class I plant features are designed to maintain their structural integrity in the event of both the DE/DDE and HE. They may or may not be designed to remain operable for the DE/DDE or HE, the design basis function of the equipment determines whether it is qualified for active or passive function for a DE/DDE and/or an HE.

The DDE is equivalent to the SSE described Safety Guide 29, "Seismic Design Classification," and RG 1.29. Safety Guide 29 provided an NRC approved method for identifying plant SSCs required remaining functional following the GDC 2 SSE.

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 3.2-1

Sheet 2 of 2

Design Class I	Design Class II	Design Class III
Requirements		
1. Quality Standards - Plant features required to meet AEC GDC-1	1. Quality Standards - Plant features not required to meet AEC GDC-1.	1. Quality Standards - Plant features not required to meet AEC GDC-1
2. Quality Assurance - Plant features required to meet Appendix B to 10 CFR 50	2. Quality Assurance - Plant features not required to meet Appendix B to 10 CFR 50. Specific QA requirements may be applied to selected features.	2. Quality Assurance – Plant features not required to meet Appendix B to 10 CFR 50
3. Seismic Design – Plant features required to meet GDC-2 and Appendix A to 10 CFR 100. Plant features designed to withstand effects of double design earthquake (DDE). Features are also designed to maintain their structural integrity (and in some cases their operability) during a Hosgri earthquake.	3. Seismic Design - Plant features not required to meet GDC-2 and Appendix A to 10 CFR 100. Plant features not designed to withstand effects of design earthquakes except for items as required by RG 1.143, and for selected features where specifically designated.	3. Seismic Design – Plant features not required to meet GDC-2 and Appendix A to 10 CFR 100. Plant features not designed to withstand effects of design Earthquakes, except where specifically designated.

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

3.7 SEISMIC DESIGN

3.7.1 SEISMIC INPUT

This section describes the DE, the DDE, and the postulated 7.5M HE.

In addition to the above three earthquakes, PG&E conducted, as described below, a program to reevaluate the seismic design for DCPP. On November 2, 1984, the NRC issued the DCPP Unit 1 Facility Operating License DPR-80. In License Condition 2.C(7) of DPR-80, the NRC stated, in part: "PG&E shall develop and implement a program to reevaluate the seismic design bases used for the Diablo Canyon Power Plant."

PG&E's reevaluation effort in response to the license condition was titled the "Long Term Seismic Program" (LTSP). PG&E prepared and submitted to the NRC the "Final Report of the Diablo Canyon Long Term Seismic Program" in July 1988 (Reference 19). The NRC reviewed the Final Report between 1988 and 1991, and PG&E prepared and submitted written responses to NRC questions resulting from that review. In February 1991, PG&E issued the "Addendum to the 1988 Final Report of the Diablo Canyon Long Term Seismic Program." (Reference 20) In June 1991, the NRC issued Supplement 34 to the Diablo Canyon Safety Evaluation Report (SSER) (Reference 21), in which the NRC concluded that PG&E had satisfied License Condition 2.C(7) of DPR-80. In the SSER the NRC requested certain confirmatory analyses from PG&E, and PG&E subsequently submitted the requested analyses. The NRC's final acceptance of the LTSP is documented in a letter to PG&E dated April 17, 1992 (Reference 22).

The LTSP contains extensive databases and analyses that update the basic geologic and seismic information in this FSAR Update. However, the LTSP material does not alter the design bases for DCPP. In SSER 34 (Reference 21), the NRC states, "The Staff notes that the seismic qualification basis for Diablo Canyon will continue to be the original design basis plus the Hosgri evaluation basis, along with associated analytical methods, initial conditions, etc."

THE LTSP did not alter or change the DCPP design bases. Seismic qualification was based on the (DE/OBE & SSE/DDE) design basis and the HE. In addition to ground motion, the design basis includes the associated analytical methods, initial conditions, etc., applied to each analysis.

PG&E committed to the NRC in a letter dated July 16, 1991 (Reference 23), that certain future plant additions and modifications, as identified in that letter, would be checked against insights and knowledge gained from the LTSP to verify that the plant margins remain acceptable.

A completed listing of bibliographic references to the LTSP reports and other documents are provided in References 19, 20, and 21.

3.7.1.1 Design Response Spectra

Section 2.5.2 provides a discussion of the earthquakes postulated for the DCPP site and the effects of these earthquakes in terms of maximum free-field ground motion accelerations and corresponding response spectra at the plant site. The maximum vibratory accelerations at the plant site would result from either Earthquake B or Earthquake D-modified, depending on the natural period of the vibrating body. Response acceleration spectra curves for horizontal free-field ground motion at the plant site from Earthquake B, Earthquake D-modified, and HE are presented in Figures 2.5-20, 2.5-21, and 2.5-29 through 32, respectively.

Safety analysis results for ***maximum ground acceleration*** and response spectra – Earthquakes B or D-modified (DDE). This established the seismic design basis controlling parameter as defined in NEI 97-04.

For design purposes, the response spectra for each damping value from Earthquake B and Earthquake D-modified are combined to produce an envelope spectrum. The acceleration value for any period on the envelope spectrum is equal to the larger of the two values from the Earthquake B spectrum and the Earthquake D-modified spectrum. Vertical free field ground accelerations, and the vertical free-field ground motion response spectra are assumed to be two-thirds of the corresponding horizontal spectra.

The DE is the hypothetical earthquake that would produce these horizontal and vertical vibratory accelerations. The DE corresponds to the operating basis earthquake (OBE), as described in Appendix A to 10 CFR 100 (Reference 7).

The DE (design earthquake) is equivalent to the operational bases earthquake (OBE) defined in 10 CFR100, Appendix A. The OBE has about ½ the peak ground motion of the DDE/SSE.

To ensure adequate reserve energy capacity, Design Class I structures and equipment are reviewed for the DDE. The DDE is the hypothetical earthquake that would produce accelerations twice those of the DE. The DDE corresponds to the SSE, as described in Appendix A to 10 CFR 100 (Reference 7).

The safety analysis defined the SSE/DDE as meeting the 10 CFR 100, Appendix A, design basis (the HE was excluded from this analysis).

PG&E was requested by the NRC to evaluate the plant's capability to withstand a postulated Richter magnitude 7.5 earthquake centered along an offshore zone of geologic faulting, generally referred to as the Hosgri Fault. This evaluation is discussed in the various chapters when it is specifically referred to as the Hosgri evaluation or Hosgri event evaluation.

The FSARU refers to the HE as an answer to an NRC question during the original plant licensing process.

Differing Professional Opinion--Appeal

(Continued)

Continued Item 11

3.7.6 SEISMIC EVALUATION TO DEMONSTRATE COMPLIANCE WITH THE HOSGRI EARTHQUAKE REQUIREMENTS UTILIZING A DEDICATED SHUTDOWN FLOWPATH

3.7.6.1 Post-Hosgri Shutdown Requirements and Assumed Conditions

In response to a request from the NRC, PG&E evaluated the ability of DCPP to shut down following the occurrence of a 7.5M earthquake due to a seismic event on the Hosgri fault. This evaluation is presented in Reference 15, which was amended several times after it was first issued in order to respond to questions by the NRC and reflect agreements made at meetings with the NRC. The final document describes the method proposed by PG&E to shut down the plant after the earthquake, assuming a loss of all offsite power, but no concurrent accident, using only equipment qualified to remain operable following such an earthquake.

For this purpose, valves that are required to operate to achieve shutdown following the earthquake were qualified for active function to the Hosgri parameters, whereas other valves, which might have an active function for postaccident mitigation, but were not required to operate to achieve shutdown following the earthquake, were qualified for passive function (pressure boundary integrity) to the Hosgri parameters. This is consistent with the DCPP design basis stated in FSAR Section 3.7.1.1 that the DDE is the SSE for DCPP, and that the guidelines presented in RG 1.29 apply to the DDE.

In addition, pursuant to the NRC request, it was necessary to demonstrate that DCPP could be shut down following an HE in order to protect the health and safety of the public. The Hosgri evaluation presented in Reference 15 demonstrated this. To provide increased conservatism, PG&E has subsequently qualified all active valves for active function for an HE pursuant to a commitment made in Reference 17.

3.7.6.2 Post-Hosgri Safe Shutdown Flowpath

The flowpath qualified to enable shutdown of the plant following an HE is defined in Chapter 5 of Reference 15. For this purpose, safe shutdown was defined as cold shutdown. It assumes concurrent loss of offsite power, a single active failure, but no concurrent accident or fire. Local manual operation of equipment from outside the control room is acceptable for taking the plant from hot standby to cold shutdown.

3.7.6.2.4 Equipment Required for Post-Hosgri Shutdown

The equipment determined to be required to achieve post-Hosgri cold shutdown in the manner described above is presented in Sections 7.3 and 9.2 of Reference 15. Some minor revisions to the list of valves required have been made, and are reflected in the latest revision of the active valve list, FSAR Table 3.9-9. Instrument Class IA, Instrument Class IB, Category 1, and on a case-by-case basis, Instrument Class ID instrumentation are qualified to the Hosgri parameters, and assumed to be operable following an HE. Additional instrumentation determined to be required is presented in Section 7.3 of Reference 15. Some revisions have been made to that list; the revised list of required instrumentation is presented in Reference 16. The electrical Class 1E system is also qualified to the Hosgri parameters, and is assumed to be operable following an HE.

Discussion of the HE

The FSARU refers to the HE as an answer to an NRC question during the original plant licensing process.

The assumptions and methods used for the HE were based on agreements made at meetings with NRC (not regulatory requirements).

The HE demonstrated that the plant could safety shutdown following a 7.5 M earthquake on the Hosgri fault.

The FSARU again clarified that the DDE is the Diablo Canyon SSE and the list of SSCs to be seismically quailed to the SSE are compliant with Guide 1.29, "Seismic Design Classification."

In response to the NRC question, the HE established the scope of equipment needed be qualified for "safe shutdown" following an earthquake on the Hosgri fault. The HE safety functions are different than the specified by Part 100, Appendix A

!

Differing Professional Opinion--Appeal
(Continued)

Continued Item 11

5.2.1.3 Compliance with 10 CFR 50.55a

Codes and standards applicable to reactor coolant pressure boundary (RCPB) components are specified in 10 CFR 50.55a They depend on when the plant was designed and constructed. Construction permits for DCPP Units 1 and 2 were issued on April 23, 1968, and December 9, 1970, respectively. Therefore, codes and standards specified in 10 CFR 50.55a for construction permits issued before January 1, 1971, are applicable to the DCPP.

The FSARU stated that Diablo Canyon met code requirements (an earlier version of the Code is applicable in some cases)

The codes, standards, and component classifications used in the design and construction of the DCPP RCPB components are shown in Table 5.2-2 and are in accordance with the applicable provisions of 10 CFR 50.55a. These design codes specify applicable surveillance requirements including allowances for normal degradation.

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 5.2-6

LOAD COMBINATIONS AND STRESS CRITERIA FOR WESTINGHOUSE PRIMARY EQUIPMENT[a]

CONDITION	LOAD COMBINATION	STRESS CRITERIA[e]
Design	Deadweight + Pressure + DE	$P_m \le S_m$ $P_L + P_b \le 1.5\ S_m$
Normal	Deadweight + Pressure + Thermal	$P_L + P_b + P_e + Q \le 3\ S_m$[b]
Upset - 1	Deadweight + Pressure + Thermal + DE	$U \le 1.0$[b] $P_L + P_b + P_e + Q \le 3\ S_m$
	Deadweight + Pressure + Thermal	$U \le 1.0$[b] $P_L + P_b + P_e + Q \le 3\ S_m$
Faulted - 1	Deadweight + Pressure + DDE	Table 5.2-7
Faulted - 2	Deadweight + Pressure + DDE + LPR[c,d,g]	Table 5.2-7
Faulted - 3	Deadweight + Pressure + Hosgri	Table 5.2-7
Faulted - 4	Deadweight + Pressure + Other Pipe Rupture[f]	Table 5.2-7

The CLB requires the Code acceptance limits to be met for SSE/DDE loads combined with accident loads.

The HE did not include the accident loads (LPR) as required for the SSE.

HE load combinations and limits were negotiated.

(a) Steam generators, reactor coolant pumps, pressurizer
(b) Based on elastic analysis. For simplified elastic-plastic analysis, the stress limits of the 1971 ASME Code Section III, NB-3228.3 apply
(c) LPR = reactor coolant loop pipe rupture
(d) DDE and LPR combined by SRSS method
(e) For definition of stress criteria terms, see Additional Notes
(f) Pipe rupture other than LPR.
(g) While the original stress analysis considered this load combination, with the acceptance of the DCPP leak-before-break analysis by the NRC, loads resulting from ruptures in the main reactor coolant loop no longer have to be considered in the design basis structural analyses and included in the loading combinations, only the loads resulting from RCS branch line breaks have to be considered.

P_m = General membrane, average primary stress across solid section. Excludes discontinuities and concentrations. Produced only by mechanical loads.
P_L = Local membrane, average stress across any solid section. Considers discontinuities, but not concentrations. Produced only by mechanical loads.
P_b = Bending; component of primary stress proportional to distance from centroid of solid section. Excludes discontinuities and concentrations. Produced only by mechanical loads.
P_e = Expansions, stresses which result from the constraint of "free end displacement" and the effect of anchor point motions resulting from earthquakes. Considers effects of discontinuities, but not local stress concentration. (Not applicable to vessels)
Q = Membrane Plus Bending, self-equilibrating stress necessary to satisfy continuity of structure. Occurs at structural discontinuities. Can be caused by mechanical loads or by differential thermal expansion. Excludes local stress concentrations.
U = Cumulative usage factor

Continued Item 11

Appendix C,

Pacific Gas and Electric Company Nuclear Power Generation, Classification of Structures, Systems, and Components for Diablo Canyon Power Plant Units 1 And 2 (Q-LIST)

2. CLASSIFICATION SYSTEMS

2.1 GENERAL

PG&E established its own design criteria and classification requirements for structures, equipment, and systems used in the Diablo Canyon Power Plant because industry and regulatory standards were not developed. It is recognized that during the design and construction of Units 1 and 2, significant industry and regulatory progress was made in establishing common and agreed upon methods of classification. The newer methods of classification all differ slightly in detail from those for Diablo Canyon, but the form and intent of all are equivalent as shown in the FSAR Update [5]. Some of the major differences are summarized as follows:

(1) Use of the postulated double design earthquake (DDE) for seismic design criteria versus the safe shutdown earthquake (SSE) of Regulatory Guide 1.29 [13].

(2) Including all steam and feedwater piping from the secondary side of the steam generator up to, and including, the automatic containment isolation valves versus restricting pipe size to 2-1/2 inches or larger as in Regulatory Guide 1.29 [13].

2.2 PG&E CLASSIFICATION SYSTEM

2.2.1 Diablo Canyon Design Class

Appendix A to 10 CFR 100 requires that structures, systems, and components necessary to assure:

(1) the integrity of the reactor coolant pressure boundary

(2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to listed exposure guidelines.

be designed to withstand the safe shutdown earthquake (SSE) and remain functional.

Guidance for determining the structures, systems, and components required to remain functional in the event of an SSE was provided in AEC Safety Guide 29. For Diablo Canyon, plant features required to perform this function are designated Design Class I.

Plant features important to the operation of Diablo Canyon but not required to perform one of the three functions listed above are designated as Design Class II.

Other plant features not related to plant operation or plant safety are classified as Design Class III.