DPO Case File for DPO-2013-002

The following pdf represents a collection of documents associated with the submittal and disposition of a differing professional opinion (DPO) from an NRC employee involving seismic issues at Diablo Canyon.

Management Directive (MD) 10.159, “The NRC Differing Professional Opinions Program,” dated May 16, 2004, describes the DPO Program.
http://pbadupws.nrc.gov/docs/ML0417/ML041770431.pdf

The DPO Program is a formal process that allows employees and NRC contractors to have their differing views on established, mission-related issues considered by the highest level managers in their organizations, i.e., Office Directors and Regional Administrators. The process also provides managers with an independent, three-person review of the issue (one person chosen by the employee). After a decision is issued to an employee, he or she may appeal the decision to the Executive Director for Operations (EDO).

Because the disposition of a DPO represents a multi-step process, readers should view the records as a collection. In other words, reading a document in isolation will not provide the correct context for how this issue was considered by the NRC.

The records in this collection have been reviewed and approved for public dissemination.

Document 1 – DPO Submittal

NRC FORM 680 (11-2002) NRCMD 10.159

U.S. NUCLEAR REGULATORY COMMISSION

DIFFERING PROFESSIONAL OPINION

FOR PROCESSING USE ONLY

1. DPO CASE NUMBER: DPO-2013-002

2. DATE RECEIVED: 7/19/2013

INSTRUCTIONS: *Prepare this form legibly and submit three copies to the address provided in Block 14 below.*

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

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

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

Please see attachment for the DPO.

Please note: This DPO involves Region IV, Reactor Projects, and NRR, DORL. These issues were developed while I was the senior resident inspector at Diablo Canyon. My supervisor was Neil O'keefe.
I was subsequently reassigned to the TTC.

11. DESCRIBE YOUR DIFFERING OPINION IN ACCORDANCE WITH THE GUIDANCE PRESENTED IN NRC MANAGEMENT DIRECTIVE 10.159. *(Continue on Page 2 or 3 as necessary.)*

Please see attachment.

As discussed in MD 10.159, please make this DPO available to the public.

Thank you,

12. Check (a) or (b) as appropriate:

- [✓] a. Thorough discussions of the issue(s) raised in item 11 have taken place within my management chain; or
- [ ] b. The reasons why I cannot approach my immediate chain of command are:

SIGNATURE OF SUBMITTER	DATE	SIGNATURE OF CO-SUBMITTER *(if any)*	DATE
[signature]	July 18, 2013		

13. PROPOSED PANEL MEMBERS ARE *(in priority order)*:
1. Gerond George
2. Larry Criscione
3. Rudy Bernhard

14. Submit this form to:
Differing Professional Opinions Program Manager
Office of: OE/CRB
Mail Stop: 4 A15A

15. ACKNOWLEDGMENT

THANK YOU FOR YOUR DIFFERING PROFESSIONAL OPINION. It will be carefully considered by a panel of experts in accordance with the provisions of NRCMD 10.159, and you will be advised of any action taken. Your interest in improving NRC operations is appreciated.

SIGNATURE OF DIFFERING PROFESSIONAL OPINIONS PROGRAM MANAGER (DPOPM): Renée Pedersen

PRE-CONDITIONS MET: [X] YES [ ] NO

DATE OF ACKNOWLEDGMENT: 7/31/2013

NRC FORM 680 (11-2002)

Differing Professional Opinion – Diablo Canyon Seismic Issues

1.0 Summary

In 2011, Pacific Gas and Electric (PG&E) submitted a report to the NRC that included a reevaluation of the local geology surrounding the Diablo Canyon Power Plant.[1] This report included deterministic evaluations concluding that three local earthquake faults are capable of generating significantly greater vibratory ground motion than was used to establish the facility safe shutdown earthquake (SSE) design basis. In response to this issue, NRC staff actions have been inconsistent with existing regulatory requirements and the facility design bases and Operating License.

a. Less than Adequate Corrective Actions to Incorporation the New Seismic Information Into the Current Licensing Basis (CLB)

Prevailing Staff View: The NRC concluded that potential earthquake ground motions from the Shoreline fault are at or below those levels for which the plant was previously evaluated and demonstrated to have a "reasonable assurance of safety."[2] The staff stated that PG&E should incorporate Shoreline scenario into the Final Safety Analysis Report Update (FSARU) as an included case under the Hosgri evaluation (HE).

Alternate View: Incorporating the Shoreline scenario into the FSARU will require an amendment to the Diablo Canyon Operating License. A license amendment is required because the change results in more than a minimal increase in the likelihood of a malfunction of a structure, system, or component (SSC) important to safety than previously evaluated in the FSARU. A license amendment is also required because this change represents a departure from the FSARU method of evaluation used to establish the seismic SSE design basis. PG&E previously submitted a license amendment request to modify the plant design bases and safety analysis to accommodate the new seismic information. However, this request was not accepted by the NRC for review. The staff's conclusion of a "reasonable assurance of safety" does not provide an acceptable basis for not enforcing existing NRC quality assurance, safety analysis, and license requirements. The staff's corrective action also failed to address the Los Osos and San Luis Bay faults. The new seismic information concluded that these faults were also capable of producing ground motions in excess of the current plant SSE design basis.

Recommended Action: The NRC to initiate enforcement action to ensure PG&E complies with NRC quality assurance requirements to take prompt corrective action to correct the nonconforming FSARU safety analysis.

b. Failure to Demonstrate Plant Technical Specification Required Structures, Systems, and Components (SSCs) are "Operable"

Prevailing Staff View: The NRC concluded that all Diablo Canyon technical specification required plant SSCs were "operable" at the higher ground motions.[3,4] The staff based this conclusion on a comparison of the new seismic information with the ground motion spectrums used in the HE and the Long Term Seismic Program (LTSP).[5] While the new ground motions exceeded those used to establish the SSE design basis and the NRC approved safety analysis, they were bound by the HE and LTSP.

Alternate View: The prevailing staff view is contrary to the NRC "operability" policy. To be considered "operable," a reasonable assurance must be demonstrated that nonconforming SSC are capable of the performing the safety function(s) specified by the design and within the required range of design physical conditions defined in the CLB, including the design bases. Neither the HE nor the LSTP contain design bases limits, conditions, or assumptions used in the bounding SSE safety analysis. Comparison of the new ground motions only against the HE and LSTP failed to demonstrate that all plant technical specification required SSCs are capable of meeting the specified safety functions established at the higher ground motions:

- Neither the HE nor the LTSP methods are approved for use in the Diablo Canyon SSE design basis or safety analysis. The CLB defined the HE as an exception to the SSE and was only approved for evaluating the Hosgri fault. The LTSP is not part of the seismic design basis or safety analysis.

- Use of the HE and LTSP over-predicts SSC performance when compared to the CLB SSE methods. Neither the HE nor the LTSP are bounding for SSC seismic qualification at Diablo Canyon. Comparisons limited to only ground motion are meaningless for "operability." These comparisons omit other relative CLB requirements including the methods, assumptions, initial conditions, and acceptance criteria applicable to each evaluation.

- Comparison of the new information only to the HE and LTSP failed to demonstrate that the requirements of the American Society of Mechanical Engineers' (ASME) Boiler and Pressure Vessel Code are met at the higher ground motions. "Operability" requires that the Code acceptance criteria are met for key plant components, including the reactor coolant pressure boundary.

Recommended Action: The NRC to initiate enforcement action to ensure PG&E complies with plant technical specification required actions to shutdown the Diablo Canyon reactors. The reactors should remain shut down pending demonstration that SSC safety functions can be meet at the higher seismic stress levels or until the NRC approves necessary dispensation and/or exemptions from the applicable regulatory and Operating License requirements.

Assessment of the Consequences if submitter's position is not adopted by the Agency: The new seismic information resulted in a condition outside of the bounds of the existing Diablo Canyon design basis and safety analysis. Continued reactor operation outside the bounds of the NRC approved safety analyses challenges the presumption of nuclear safety.

The prevailing staff view that "operability" may be demonstrated independent of existing facility design bases and safety analyses requirements establishes a new industry precedent. Power reactor licensees may apply this precedent to other nonconforming and unanalyzed conditions.

2.0 Introduction

The Atomic Energy Act of 1954, as amended, establishes "adequate protection" as the standard of safety on which NRC regulation is based. In the context of NRC regulation, safety means avoiding undue risk or providing reasonable assurance of adequate protection for the public. Safety is the fundamental regulatory objective, and compliance with NRC requirements plays a fundamental role in providing confidence that safety is maintained. NRC requirements have been designed to ensure adequate protection, which in turn, corresponds to "no undue risk to public health and safety." This goal is met through acceptable design and quality assurance measures. In the context of risk-informed regulation, compliance plays a very important role in ensuring that key assumptions used in underlying risk and engineering analyses remain valid.[6]

Adequate protection is presumptively assured by compliance with NRC requirements. These requirements limit plant operation within the design bases. These regulations also required that licensees establish, maintain, and operate within the boundaries of the NRC approved safety analyses. Operation within the bounds of the safety analysis provides confidence that the plant response to accidents and events will be consistent with the design bases.

At Diablo Canyon, the licensee developed new information that revealed that an unforeseen hazard exists. This new information concluded that three local faults are capable of producing earthquakes greater than those bound by the Diablo Canyon safe shutdown earthquake (SSE) design basis. The presumption of nuclear safety is challenged because plant operation is no longer within the bounds of the design basis and quality assurance measures the NRC used to license the facility.

A nonconforming condition exists when the plant safety analysis no longer meets NRC design bases and regulatory requirements. An unanalyzed condition exists when reactor operation occurs outside of the limiting bounds established in the NRC approved safety analysis. The Diablo Canyon seismic information resulted in both nonconforming and unanalyzed conditions. NRC quality assurance requirements required PG&E to implement prompt corrective actions to either restore the plant configuration within the bounds of the safety analysis or request NRC approval to revise the plant Operating License to accommodate the new information. The NRC has not enforced these regulatory requirements to correct the deficient seismic safety analysis at Diablo Canyon.

The NRC staff has discussed Diablo Canyon seismic issues for the past several years. Several staff members viewed the new PG&E seismic information as beyond the existing regulatory framework. These staff members proposed new regulatory processes to review and disposition this information. These recommendations were similar to those proposed for the resolution to Generic Issue 199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," and provided by the Fukishima Near-Term Task Force. These approaches request licensees compare the results of newly developed probabilistic ground motions models against the existing deterministic SSE. Subsequent Regulatory decisions are made based on the risk insights gained from these comparisons.

The updated Diablo Canyon seismic information was unique because PG&E included detailed deterministic evaluations of the local geology. These deterministic evaluations provided a one-to-one correspondence to seismic evaluations included in the CLB. Comparing this new information with the CLB indicated that the plant was operating outside the bounds of the existing safety analysis. This called into question if the plant design bases requirements could still be met following an earthquake. From an inspection point of view, the regulatory framework for addressing nonconforming safety analyses and unanalyzed conditions are familiar. The PG&E case was different because these conditions were specifically related to the seismic design basis, an area rarely touched by the Inspection Program prior to the Fukishima accident.

The integrity of key assumptions used in the safety analyses are maintained by requiring licensees to comply with the plant technical specifications. Technical specifications require plant operators to implement time dependent actions, including shutting down the reactors, when prescribed SSCs are no longer “operable.” Following identification of nonconforming or unanalyzed conditions, the “operability” process provides assurance that the plant is safe to continue to operate during the corrective action period. To be considered “operable,” plant SSCs must be capable of performing the safety functions described in the CLB, including the FSARU safety analyses. These safety functions include the capability to prevent or mitigate accidents and events following the vibratory motion (shaking) associated with the SSE. The staff concluded that all Diablo Canyon SSCs were “operable” using an alternative basis. However, the “operability” process did not provide the staff the flexibility to use this alternate approach. While the NRC has statutory authority to amend the facility Operating License to allow use of these alternate bases or exempt PG&E from regulatory requirements, the staff did not implement either of these processes to waive the Diablo Canyon CLB requirements.

3.0 Diablo Canyon Current Licensing Basis (CLB)

NRC regulations use the terms safety analysis, design bases, and nonconforming condition within the context of the CLB. A clear understanding how the NRC defined these terms and the specific Diablo Canyon License requirements are needed before the seismic corrective actions and “operability” can be assessed. The CLB includes the set of NRC requirements applicable to nuclear power plant license plus the docketed and currently effective written commitments for ensuring compliance with these NRC requirements and the plant-specific design basis.[7] For Diablo Canyon, seismic CLB explicitly includes:

- NRC regulations in 10 CFR Parts 2, 50, 100 (including Appendixes)
- Plant-specific design basis information, as defined in 10 CFR 50.2, and documented FSARU as required by 10 CFR 34 and 50.71(e)
- Plant technical specifications

Design Bases

Title 10 of the Code of Federal Regulations, Part 50.2, defines “design bases” as that information which identified the specific functions to be performed by plant SSCs and the specific values or ranges of values chosen for controlling parameters as reference bounds for the design. The NRC endorsed an expanded definition of “design bases” in NEI 97-04, “Guidance and Examples for Identifying 10 CFR 50.2 Design Bases,” Appendix B.[8] This expanded definition of design bases included:

- **Design Bases Functions:** Functional requirements derived from the principal design criteria used for Diablo Canyon. These establish the minimum standards set by 10 CFR Part 50, Appendix A, General Design Criteria (GDC), and other NRC regulations imposing functional requirements or limits on the plant design. For plant SSCs, design bases function include those:

(1) required by, or otherwise necessary to comply with, regulations, license conditions, orders or technical specifications, or

(2) credited in licensee safety analyses to meet NRC requirements.

For seismic qualification, the design basis functional requirements are established by 10 CFR 50, GDC 2, and 10 CFR 100, Appendix A.[9]

- **Design Bases Values:** Values or ranges of values used for the controlling parameters establishing the reference bounds for the design and to meet the design bases functional requirements. These values may be:

  (1) established by NRC requirement,

  (2) derived from or confirmed by safety analyses, or

  (3) chosen by the licensee from an applicable code, standard or guidance document.

  Design bases values include the bounding conditions under which SSCs must perform the design bases functions for normal operation or following accidents or events. Plant specified events include those specified in the regulations, including the SSE.

  Design Bases Controlling Parameters: Values chosen as reference bounds for the design. For example, for the seismic design basis, the SSE ground motion spectra are a design bases controlling parameter.[10]

The CLB also includes supporting design information. While supporting design information is not explicitly part of the design bases, this information includes assumptions and inputs used in the safety analysis and by the NRC to verify design basis acceptance limits are met. For seismic qualification, examples of supporting design information include:

- Commitment to NRC Safety Guide 29 (Regulatory Guide 1.29), "Seismic Design Classification." Safety Guide 29 provides an NRC approved list of plant SSCs that are required to be qualified for the SSE.

- Methods used in the safety analysis to establish the SSE response spectra.

- Seismic damping values used in the structural dynamic analysis

The facility design bases are a subset of the CLB and are required to be included in the FSARU by 10 CFR 50.34 and 10 CFR 50.71(e).

Regulations Establishing the Seismic Design Bases

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, General Design Criteria (GDC) *2*,[11] "Design Bases for Protection against Natural Phenomena," established the design basis requirements for seismic qualification. SSCs important to safety must be capable of withstanding the effects of earthquakes without loss of capability to perform their safety functions. GDC 2 requires:

- Appropriate consideration of the most severe natural phenomena that has been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period that historical data was accumulated;

- Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

- The importance of the safety functions to be performed.

Title 10, Code of Federal Regulations, Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," implements the GDC 2 requirements for seismic design. SSCs important to safety must be capable of withstanding the effects of the SSE without loss of capability to perform their safety functions. Appendix A defines the SSE as the "*maximum earthquake potential"* considering the regional and local geology and seismology and specific characteristics of local subsurface material. Appendix A applies to those important to safety SSCs necessary to assure:

- The integrity of the reactor coolant pressure boundary,

- The capability to shut down the reactor and maintain it in a safe shutdown condition,

- The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Safety Analysis: Demonstrates that the facility meets the design bases, the capability to withstand or respond to postulated events, and that NRC acceptance criteria are met:[12,13,14]

Seismic Qualification Process

Pacific Gas and Electric seismically qualified plant SSCs (listed in Table 1) that are required to remain functional following the SSE. The seismic qualification process was generally preformed in three steps:

a. Evaluation of the local geology (FSARU Section 2.5)

   This evaluation examined the local geology and deterministically identified the "maximum earthquake potential" that could affect important to safety plant equipment. The safety analysis used NRC approved ground motion and attenuation methods and assumptions to establish the maximum vibratory ground motion for the site. At Diablo Canyon, the maximum ground motion was called the double design earthquake (DDE) and is equivalent to the SSE defined in 10 CFR 100, Appendix A.

b. Attenuation of seismic energy to important to safety SSC (FSARU Section 3.7)

   This evaluation established how much seismic energy, or shaking, each important to safety SSC would be exposed to following the SSE/DDE. The analysis used NRC approved attenuation models and design basis inputs to propagate the seismic energy through plant structures, equipment, and piping systems. These models and inputs are part of the facility CLB.

c. SSC Seismic qualification (FSARU Sections 3.2, 3.8, 3.9, 3.10, & 5.2)

PG&E seismically qualified the plant SSCs listed in Table 1 to ensure they would remain functional at the level of shaking that was determined to occur at that plant location following the SSE/DDE. This qualification was performed by a combination of testing and analyses. The functionality of some plant SSCs were demonstrated by use of a "shaker table" test. Other SSCs were qualified by NRC approved analysis. For example, the reactor coolant pressure boundary, piping systems, and the containment structure were qualified by ensuring that the seismically induced stress would not exceed acceptance levels established by the ASME and other codes.

Table 1 – Plant SSCs Qualified to SSE/DDE

Diablo Canyon Plant Structures, and Systems Required to be Qualified to the SSE/DDE[15]	**Technical Specification Required SSCs**
1. The reactor coolant pressure boundary.	Yes
2. The reactor core and reactor vessel internals.	Yes
3. Systems required for	
- Emergency core cooling system	Yes
- Containment heat removal,	Yes
- Shutdown the reactor shutdown,	Yes
- Remove residual heat	Yes
- Cooling the spent fuel storage pool,	No
4. Steam and feedwater systems up to and including the outermost containment isolation valves.	Yes
5. Cooling water that are required for:	
- Emergency core cooling,	Yes
- Post-accident containment heat removal	Yes
- Residual heat removal from the reactor, or	Yes
- Cooling the spent fuel storage pool.	No
6. Cooling and seal water systems required for functioning of reactor coolant system components important to safety (reactor coolant pumps).	No
7. Systems or portions of systems that are required to supply fuel for emergency equipment.	Yes
8. All electric and mechanical devices and circuitry between the process and the input terminals of the actuator systems involved in generating signals that initiate protective action	Yes
9. Systems or portions of systems required for monitoring of systems important to safety and actuation of systems important to safety.	Yes
10. The spent fuel	No
11. The spent fuel storage pool structure, including the fuel racks.	No
12. The reactivity control systems, control rods, control rod drives and boron injection system.	Yes
13. The control room, including its associated equipment and all equipment needed to maintain the control room within safe habitability limits.	Yes
14. Primary and secondary reactor containment.	Yes
15. Systems, other than radioactive waste management systems, (not covered above) that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (using approved dose methods).	No
16. The Class 1E electric systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for functioning of plant engineered safety features.	Yes
17. Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature included above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure.	May affect TS
18. Seismic Category I design requirements should extend to the first seismic restraint beyond the defined boundaries.	Must meet applicable Code requirements

Diablo Canyon FSARU

The FSARU described the Diablo Canyon seismic design bases and safety analyses results, including assumptions and bounding conditions. This information was used to by the NRC to approve and maintain the facility Operating License.

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 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.

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 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.

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:

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

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

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

The LTSP was completed in 1988.

The LTSP did not address or alter the plant CLB.

The LTSP was not included in the FSARU because the information is not part of the seismic design basis or supporting safety analysis.

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

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.

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.

Hosgri Evaluation (HE)

The Hosgri fault was discovered a few miles off shore during plant construction by oil company geoscientists. During the Diablo Canyon licensing reviews, PG&E argued that the Hosgri was not a "capable," fault as defined in 10 CFR 100, Appendix A, and was not required to be considered for the plant SSE. The NRC argued that the Hosgri fault should be included in the safety analysis for establishing the "maximum earthquake" for the site. The resulting compromise is reflected in the CLB. PG&E provided report separate from the FSAR to address the NRC's question concerning the capability of the plant to "safely shutdown following a 7.5 magnitude earthquake on the Hosgri fault.[16] This report detailed the methods, assumptions and acceptance criteria to support the conclusion that the plant could "safety shutdown" following a Hosgri earthquake. The NRC agreed to PG&E's request to use different methodologies, assumptions, and acceptance criteria for the HE. In most cases, these methods and assumptions were less conservative than those approved for the SSE/DDE. The end result was that the Hosgri fault was excluded (exempted) from the GDC 2 SSE design basis.

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 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 Endnote 11 and the Appendix to this DPO).

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, th operating basis earthquake (OBE) of Appendix A to 10 CFR 100 is equivalent to the DCPP DE.

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

DCPP's capability to withstand a postulated Richter magnitude 7.5 earthquake centerec 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 event of an SSE is provided in SG 29. These plant features, including their foundatior and supports, are designated as Seismic Category I in SG 29. DCPP SSCs, and thei 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 the SSE of SG 29), and/or the postulated Hosgri earthquake (HE). Design Class I pla 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.

PG&E committed to Safety Guide 29, "Seismic Design Classification," (Regulatory Guide (RG) 1.29), to determine the set of SSCs required to be seismically qualified for the SSE/DDE. RG 1.29 provided an NRC acceptable method for this determination. The licensee could have proposed a different set of SSCs, subject to NRC approval.

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 3.2-1 Sheet 2 of 3

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.

Defines the plant quality, seismic, and design classifications.

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."

LTSP did not alter or change the Diablo Canyon design bases. Seismic qualification is 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. 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.

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.

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

3.7.1.3 Critical Damping Values

The specific percentages of critical damping used for Design Class I SSCs, and the Design Class II turbine building and intake structure are listed in the following table:

Type of Structure	% of Critical Damping DE	DDE	HE
Containment structures and all internal concrete structures	2.0	5.0	7.0
Other conventionally reinforced concrete structures above ground, such as shear walls or rigid frames	5.0	5.0	7.0
Welded structural steel assemblies	1.0	1.0	4.0
Bolted or riveted steel assemblies	2.0	2.0	7.0
Mechanical components (PG&E purchased)	2.0	2.0	4.0
Vital piping systems (except reactor coolant loop)[a]	0.5	0.5	3.0[b]

Type of Structure	% of Critical Damping DE	DDE	HE
Reactor coolant loop[a][c]	1.0	1.0	4.0
Replacement Steam Generators[f]	2.0	4.0	4.0
Integrated Head Assembly[g]	4.9	6.85	6.85
CRDMs[h]	5.0	5.0	5.0
Foundation rocking (containment structure only)[d]	5.0	5.0	NA[e]

Damping Values

Damping values (design basis supporting information) are used in the safety analysis and the HE to calculate how seismic energy attenuates through plant structures and components. Generally, the lower the damping value assumed, the larger amount of seismic stress attenuated through the plant. These damping values are part of the CLB.

NRC approval of the damping values used in the analysis was part of the licensing process. The NRC provided acceptable damping values in Regulatory Guide 1.161, "Damping Values for Seismic Design of Nuclear Power Plants." Licensees may use previously NRC approved damping values, for a given material and application, or request approval for alternate values through the license amendment process.

Diablo Canyon Seismic Qualification is Not Limited by the HE

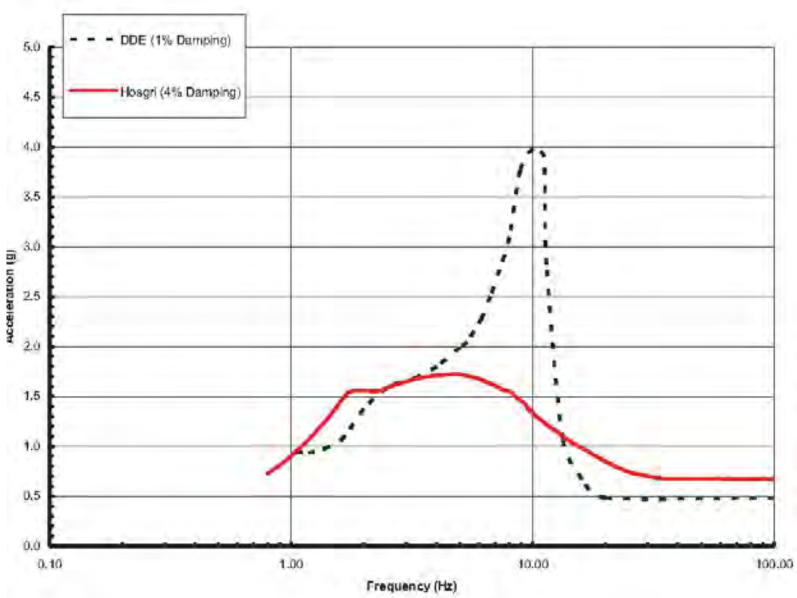

Figure 1, Comparison of the DDE/SSE and the HE Floor Response Spectrum, Containment Elevation 88'

Figure 1 illustrates the results of the different methods and assumptions use in DDE/SSE safety analysis and the HE. This figure compares acceleration levels (shaking) in the reactor containment building.

Plant SSCs are most affected in the 3 to 8.5 Hz frequency range.

Note that the level of "shaking" is significantly greater for the SSE/DDE than for the HE at this plant location. This may seem counterintuitive since the HE is a much larger earthquake. However, as this figure illustrates, comparing ground motion alone is not sufficient to evaluate seismic qualification. Methods, assumptions, initial conditions, and acceptance criteria used in the analyses are just as important as ground motion.

The qualification process used information, such as shown in Figure 1, to establish the amount of seismic stress SSCs may be exposed to during the SSE. A component located at this location would be qualified for the SSE/DDE. If the SSC was also credited for HE safe shutdown, no additional qualification would be required. At this plant location, the seismic stress is dominated by the SSE/DDE. Qualification to the SSE/DDE would envelope the seismic stress generated by the HE.

5.2.1.15.2 Steam Generator Evaluation

The seismic spectra at the elevations of the steam generator upper support and vertical support were used as the seismic input. The horizontal spectra at the upper support and the vertical spectra at the vertical support were used as input. The model was used to evaluate the shell, tube bundles, upper and lower internals, and other pressure boundary components.

The nozzles and support feet of the steam generator were analyzed using static stress analysis methods with externally applied design loads. Loadings on the inlet and outlet nozzles of the steam generator for the Hosgri earthquake were calculated as part of the reactor coolant loop piping analysis. The loadings calculated by this analysis were compared with previous faulted condition loads. The new loads were shown to be lower than the loads that were used initially to evaluate the nozzles. Therefore, the stresses caused by the Hosgri spectra are within the design basis of these nozzles.

The loads on the steam generator support feet and upper seismic support were supplied for the Hosgri evaluation by the reactor coolant loop analysis. These loadings are below the loading originally calculated for the DDE analysis.

The FSARU includes many examples where SSC seismic qualification was more limiting by the SSE/DDE than for HE. In these cases, the SSE/DDE predicts greater seismic stress (shaking) at these plant locations.

Steam generator nozzles

5.2.1.15.3 Reactor Coolant Pump Evaluation

The seismic analyses of the reactor coolant pump were performed using dynamic modal methods with a finite element computer program. The seismic response spectra corresponding to the elevation of the reactor coolant pump support structure were used.

The nozzles and support feet of the reactor coolant pump were analyzed by static stress analysis methods with externally applied design loads. For the Hosgri spectra the external loads applied to the inlet and outlet nozzles of the reactor coolant pump by the reactor coolant loop piping are all below the load for which the nozzles previously were shown acceptable. No further analysis was necessary for the nozzles.

The loads resulting from piping reactions for the Hosgri spectra were lower than the DDE loads for which the reactor coolant pump support feet were analyzed. No further analysis was necessary for the support feet.

Reactor coolant pumps

5.2.1.15.4 Reactor Vessel Evaluation

Several portions of the reactor vessel were evaluated using static stress analysis methods with externally applied design loads. The control rod drive mechanism head adapter, closure head flange, vessel flange, closure studs, inlet nozzle, outlet nozzle, vessel support, vessel wall transition, core barrel support pads, bottom head shell juncture and bottom head instrumentation penetrations were analyzed by this method. The design loads for all areas evaluated except the inlet and outlet nozzles and vesse supports were chosen to be more conservative than any actual load the component would ever experience. The design loads for the inlet and outlet nozzles and vessel supports were umbrellas of loads experienced by past plants. In cases where the actual plant loads exceed the design loads, separate analyses were performed to assure adequacy. All stresses and fatigue usage factors were found to be acceptable

The Hosgri loads calculated by the reactor coolant loop analysis were compared with the DDE seismic loads and are lower. Thus, the previous reactor vessel analysis ensures adequacy for the Hosgri seismic event.

Replacement reactor head

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Requirements are Not Limited by the HE

Title 10 of Code of Federal Regulations, Part 50.55a, “Codes and Standards,” requires important to safety pressure vessels (including the reactor coolant pressure boundary), system piping, and pipe supports to meet the ASME Boiler and Pressure Vessel Code requirements. Section (iii) of the Rule, “Seismic Design of Piping,” provides for use of Code Subarticles NB–3200, NB–3600, NC–3600, and ND–3600. These subparts required SSE/DDE seismic loads to be included when verifying plant SSCs meet the Code acceptance criteria. The Code provides assurance that these SSCs important to safety will remain intact following postulated accidents and events, including the SSE/DDE.

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	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$
Normal	Deadweight + Pressure + Thermal	$P_L + P_b + P_e + Q \leq 3\ S_m$ [b]
Upset - 1	Deadweight + Pressure + Thermal + DE	$U_T \leq 1.0$ [b] $P_L + P_b + P_e + Q \leq 3\ S_m$
	Deadweight + Pressure + Thermal	$U_T \leq 1.0$ [b] $P_L + P_b + P_e + Q \leq 3\ S_m$
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 + Hoagn	Table 5.2-7
Faulted - 4	Deadweight + Pressure + Other Pipe Rupture [f]	Table 5.2-7

(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_T = Cumulative usage factor.

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

HE load combinations and limits were negotiated.

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 5.2-5

STRESS LIMITS FOR CLASS A COMPONENTS

Loading Combinations	Piping[a]	Valves
1. Normal	$P \leq S_h$	See Section 3.9.2
2. Upset (Normal + DE loads)	$P \leq 1.2\ S_h$	See Section 3.9.2
3. Faulted (Normal + DDE loads)	$P \leq 1.8\ S_h$	See Section 3.9.2
4. Faulted (Normal + Hosgri)	$P \leq 2.4\ S_h$	See Section 3.9.2

(a) S_h = allowable stress from USAS B31.1 Code for power piping
P = piping stress calculated per USAS B31.1 Code requirements.

HE load combinations and some limits were negotiated.

The HE stress limits were relaxed for some Class A components

The Code methodology adds seismic loading, generated by either the SSE/DDE safety analysis or the HE, to other non-seismic loads affecting the component. The resulting SSE/DDE stress is significantly greater than for the HE in many loading cases. Again, this may sound counterintuitive since the HE is based on a much larger earthquake. These differences in component stress reflect the differences in the methods, assumptions, load combinations, and initial conditions used in each seismic analysis. For example, Figures 2 and 3 compare the Code bending moments calculated for the control rod drive mechanisms used to support the replacement reactor head modification. As seen in these figures, the bending moments (seismic stress) were much greater for SSE/DDE case than for the HE.

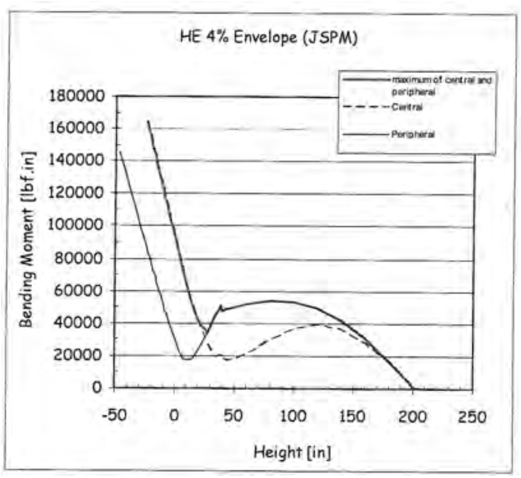

Figure 2
HE Maximum CRDM Bending Moments[17]

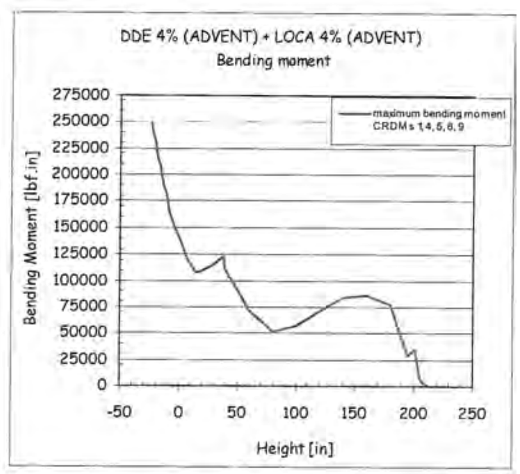

Figure 3
SSE/DDE Maximum CRDM Bending Moments[18]

3.0 Concept of Operability

The Diablo Canyon Technical Specifications are an attachment to the facility Operating License.[19] The technical specifications include a set of *limiting conditions for operation (LCOs)* for key plant SSCs. These LCOs are the lowest functional capability or equipment performance level required to ensure safe operation of the facility. When a limiting condition for operation is not met, PG&E is required to shut down the reactor or follow any prescribed remedial actions until the condition can be met. Compliance with technical specification LCOs provide confidence that plant operation is within the boundary of key assumptions used in the safety analysis and preserve the validity of the design bases.

For example, the plant design bases require two redundant trains of emergency core cooling equipment. The safety analysis concluded that either train is capable of successfully mitigating a loss of coolant accident. However, the plant design bases also assume that one train will fail to perform the safety function. Technical Specification LCO 3.5.2 (below) preserves the integrity of these assumptions by ensuring at least one emergency core cooling train will always be available for accident mitigation during plant operation. This LCO limits reactor operation to 72 hours when one emergency core cooling train is "inoperable" and for 6 hours when both trains are "inoperable."

To be considered "fully qualified,"[20] the emergency core cooling system must conform to all aspects of the CLB, including all applicable codes and standards, design criteria, safety analyses assumptions, specifications, and licensing commitments. In contrast, the system is considered "degraded" or "nonconforming" when it fails to conform to one or more aspect of the CLB.

An unanalyzed condition exists when the licensee identifies that the plant may be operating outside the bounding conditions assumed in the approved safety analysis.

Power reactor licensees sometimes identify degraded, nonconforming, or unanalyzed conditions that call in to question the capability of plant SSCs to perform the safety functions described in the CLB. When this occurs, licensees are expected to immediately evaluate the "operability" of the affected SSCs.

To be considered "operable", plant SSC must be capable of performing the safety functions specified by the design, within the required range of design physical conditions, initiation times, and mission times." For "operability" determination

ECCS – Operating
3.5.2

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

------------------------------NOTE------------------------------
In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valve(s) for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status	72 hours -----------NOTE---------- The Required Action A.1 Completion Time is to be used for planned maintenance or inspections. The Completion Times of Required Actions A.2.1, A.2.2, and A.2.3 are for unplanned corrective maintenance or inspections. ------------------------------
	OR	
	A.2.1 Verify only one subsystem in one ECCS train is inoperable	72 hours
	AND	
	A.2.2 Determine there is no common cause failure in the same subsystem in the OPERABLE ECCS train	72 hours
	AND	
	A.2.3 Restore train to OPERABLE status	14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4	12 hours

DIABLO CANYON - UNITS 1 & 2 3.5-3 Unit 1 – Amendment No. ~~135~~, ~~159~~, 202
Unit 2 – Amendment No. ~~135~~, ~~146~~, ~~160~~, 203

purposes, the mission time is the duration of SSC operation that is credited in the design basis.[21]

While this determination may be based on limited information, the information is required to be sufficient to conclude a “reasonable expectation” that the SSC is “operable.” If unable to conclude this, the licensee is required to declare the SSC “inoperable” and apply the technical specification required actions. If the available information is incomplete, the licensee is required to promptly collect any additional information that is material to the determination and promptly make an “operability” determination based on the complete set of information. If, at any time, information is obtained that negates a previous determination that the SSC is “operable,” then the licensee is required to immediately declare the SSC “inoperable.”

For example, a licensee may identify that an incorrect heat transfer coefficient was used in an emergency core cooling performance calculation. This would be considered a nonconforming condition because NRC regulations require that the design basis be correctly translated into supporting design calculations. An “operability” determination is required because the error calls into question the capability of the system to remove the post-accident heat assumed in the design bases. The licensee would be required to either demonstrate that the “specified safety function” for the system could still be met, accounting for the effect of the incorrect coefficient, or apply the actions specified in Technical Speciation LCO 3.5.2.

The NRC defines “specified safety functions” as those safety function(s) described in the CLB for the facility.[22] In addition to providing the “specified safety function,” a system is expected to perform as designed, tested and maintained. When plant SSC capability is degraded to a point where it cannot perform, with “reasonable expectation,” or reliability, plant operators are required to consider the SSC “inoperable,” even if at this instantaneous point in time the system could provide the specified safety function.

The NRC requires the resident inspector to review between 19 and 25 “operability” evaluations each year at Diablo Canyon.[23] The inspector is asked to verify that degraded or nonconforming SSCs, or compensatory measures taken, does not result in conditions outside of the design basis or inconsistent with safety analyses assumptions.

Summary

a. The plant design bases includes the functions that SSCs are:

    (1) required to comply with, including regulations, and license conditions, and
    (2) credited in the safety analysis to meet NRC requirements.

b. The design base includes the bounding conditions under which SSCs must operate following any accident or event specifically addressed in the CLB.

c. At Diablo Canyon, the SSE/DDE implements the design bases requirements specified in GDC 2 and Part 100, Appendix A. This design basis requires certain SSCs to remain functional following the earthquake which produces the “maximum vibratory ground motion” for the site, considering the regional and local geology and seismology. These SSCs are those 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,
(3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures (10 CFR 50.34 and 10 CFR 100)

d. SSE/DDE ground motion for the is defined as a design basis controlling parameter.

e. An earthquake on the Hosgri fault was an NRC approved exception to SSE/DDE design basis. While the Hosgri earthquake ground motions exceed those developed for the DDE, PG&E was not required to include the Hosgri fault in the safety analysis for determining the Part 100, Appendix A, "maximum earthquake potential" for the site.

f. The licensee developed the HE using different methodologies, assumptions, initial conditions, and acceptance criteria, than those approved for the SSE/DDE design basis. These methods were not included in the FSARU because they were not part of the safety analysis supporting the seismic design basis. Even though the HE represents a larger ground motion, the evaluation is not bounding for Diablo Canyon seismic qualification. In many cases, plant seismic qualification was more limited by the SSE/DDE.

g. The safety analysis demonstrates that SSCs important to safety (listed in RG 1.29 & Table 1) are capable of performing the specified safety functions and meeting the SSE/DDE design basis. Meeting ASME and other Code acceptance limits provides assurance that pressure retaining systems, including the reactor coolant pressure boundary and containment, will remained intact following a SSE/DDE.

4.0 Chronology

Discovery of new Seismic Information

November 2008: Pacific Gas and Electric notified the NRC[24] of discovery of a previously unknown "zone of seismicity" located about a mile offshore from the Diablo Canyon facility. The licensee stated that an initial assessment indicated that the ground motion from the "potential fault" was expected to be bounded by the LTSP spectrum." The licensee concluded an "operability" evaluation was not required because the new information was bound by the LTSP design basis.[25]

Initial NRC Review of the Shoreline Fault

April 8, 2009: The NRC issued Research Information Letter 09-001, "Preliminary Deterministic Analysis of Seismic Hazard at Diablo Canyon NPP from Newly Identified 'Shoreline Fault'" to the public.[26] The Research Information Letter included a confirmatory analysis concluding that potential ground motion from the Shoreline fault was bound by the LTSP spectrum. The Research Information Letter did not draw any conclusions related to the Shoreline fault ground motion being within Diablo Canyon CLB. However, the Office of Nuclear Reactor Regulation (NRR) transmittal letter included the following statements:

> *"PG&E informed the NRC staff that it had performed an initial evaluation of the potential ground motion levels at the DCPP from the hypothesized fault which concluded that these motions would be bounded by the ground motion levels previously determined for the current licensing basis."*

"Based on the NRC staff review of the preliminary geophysical data provided by PG&E in preparation for the call and the license's' preliminary analysis provided during the conference call, the NRC staff concluded that the current licensing basis is bounding and continues to support safe operation of the DCPP. "

"Therefore, based on the currently available information, the NRC staff concludes that the design and licensing basis evaluations of the DCPP structures, systems, and components are not expected to be adversely affected and the current licensing basis remains valid and supports continued operability of the DCPP site."

December 15, 2009: Pacific Gas and Electric determined that that the Shoreline Fault was only 300 meters from the plant inlet (location of SSCs important to safety). PG&E again concluded that a nonconforming condition did not exist because the results were still bounded by the LTSP.[27]

NRC Discovery of Nonconforming/Unanalyzed Condition

September 14, 2010: The resident inspectors identified that postulated Shoreline fault ground motions were greater than those assumed in the DDE safety analysis.[28] The inspectors questioned SSC "operability" because the DDE was identified as the facility SSE in FSARU Sections 2.5 and 3.7. The inspectors also identified that the LTSP was not part of the seismic design basis.

September 28, 2010: The resident inspectors identified and communicated to PG&E that the Shoreline Fault was a condition outside the bounds of the FSARU seismic safety analysis and was required to be evaluated for "operability" as defined in station procedures. PG&E did not take any corrective actions.

October 4, 2010: The resident inspectors recommended an unresolved item be included in Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2010004 and 05000323/2010004, to document concern that an earthquake produced on the Shoreline fault could produce ground motions greater than those described in the SSE/DDE safety analysis. Region IV disapproved the resident inspectors' recommendation.

October 5, 2010: The resident inspectors briefed the Office of NRR Project Manager and Branch Chief on the Shoreline fault findings.

October 10, 2010: Pacific Gas and Electric reviewed the inspectors' "operability" concerns prior to releasing Unit 1 for restart following refueling. Pacific Gas and Electric again concluded that a nonconforming condition did not exist because predicted ground motions were within the LTSP spectrum.[29]

October 14, 2010: The resident inspectors briefed the Region IV Regional Administrator on the Shoreline Fault findings.

Pacific Gas and Electric's Failure to "Assess Operability"

October 19, 2010: The resident inspectors met with the PG&E engineering vice president and discussed seismic "operability" concerns. The engineering vice president stated that the problem was related to an incomplete plant licensing docket. The vice president argued that past agreements made with the NRC to only use the LTSP to evaluate new seismic information were inadvertently omitted from docketed correspondence and the FSARU. The vice president also stated that no additional action was required because the Shoreline fault spectrum was bound by the LTSP.

November 30, 2010: The resident inspectors provided a detailed briefing of the Shoreline fault findings to the Region IV, Reactor Projects Division Director. At this meeting, the Reactor Projects Deputy Division Director took the action to request the PG&E engineering vice president to enter the Shoreline fault into the corrective action program and assess the effect of the higher ground motions on plant SSC (perform an “operability evaluation).

December 16, 2010: Pacific Gas and Electric again declined to evaluate operability of plant SSCs. PG&E engineering and regulatory assurance staff indicated that the Shoreline fault ground motions were too high to successfully demonstrate SSCs “operability” using the SSE/DDE methods specified in the CLB. In response to the Deputy Division Director’s request, PG&E updated the condition report to include a justification for not evaluating the “operability” of technical specification required SSCs.[30] This justification included a summary of the April 8, 2009 NRC NRR letter:

> *“Therefore, based on the currently available information, the NRC staff concludes that the design and licensing basis evaluations of the DCPP structures, systems, and components are not expected to be adversely affected and the current licensing basis remains valid and supports continued operability of the DCPP site.”*

January 2011: PG&E submitted a report to the NRC updating the local geology.[31] This report included detailed deterministic evaluations of the San Luis Bay, Los Osos and Shoreline faults. The report concluded that each of these faults are capable of producing significantly greater vibratory ground motion than assumed in the SSE/DDE safety analysis (Table 2). The inspectors concluded that this information resulted in an unanalyzed condition because the new predicted ground motions where greater that those used as bounds for the existing SSE/DDE safety analysis and seismic qualification basis. The inspector again recommended that Region IV initiate enforcement action because PG&E had failed to demonstrate that technical specification required SSCs were capable of performing the required safety functions.[32] The inspector included a second enforcement recommendation to address the incomplete and inaccurate information PG&E provided the NRC related to the seismic design basis. This incomplete and inaccurate information lead to the incorrect conclusions stated in the April 8, 2009 NRC NRR letter.

Table 2
Comparison of Reanalysis to Diablo Canyon SSE

Local Earthquake Fault[33]	Peak Ground Acceleration[34]
SSE/DDE Design Basis	0.40 g
Shoreline Faults	0.62 g
Los Osos	0.60 g
San Luis Bay	0.70 g
Hosgri (HE)	0.75 g

Note: Peak ground acceleration is anchored at 100 Hz and only used as a bases for comparison

NRC Initial Response to Seismic "Operability"

April 2011: The resident inspector met with the NRR Project Manager, NRR Branch Chief and the Region IV, Reactor Projects Division Director. The inspector again recommended that the NRC initiate enforcement action against PG&E. Enforcement action was required because the licensee continued to operate the plant outside the bounds of the safety analysis. The licensee had refused to demonstrate SSC "operability" at the higher ground motions or shutdown the reactors in accordance with technical specifications. At the meeting, Reactor Projects Division Director stated that initiating enforcement action would reverse the previous NRC conclusion described in the April 8, 2009 NRR letter, that the new seismic information was within the facility design basis. The Division Director requested that NRR formally concur on this reversal of position prior to the agency initiating action. At the Division Director's request, the inspector initiated a Task Interface Agreement to document NRR concurrence on the new position.

May 2011: The NRC opened Unresolved Item: 05000275; 323/2011002-03, "Requirement to Perform an Operability Evaluation Following Receipt of New Seismic Information."[35] This Unresolved Item identified NRC concerns that PG&E had failed to evaluate the effect the new seismic information had on capability of plant SSC to perform the requires safety functions at the higher seismic stress:

> *"The inspectors were unable to confirm the licensee's statements that new seismic information was only required to be evaluated under the LTSP deterministic margin analysis (which is a margin analysis to the Hosgri Event) based on a review of docketed information and the plant safety analysis. The LTSP margin analysis only demonstrated that the new seismic information was bound by the Hosgri Event design basis earthquake, not the Design or Double Design Earthquakes."*

August 2011: The NRC issued Task Interface Agreement (TIA) 2011-010, "Concurrence on Diablo Canyon Seismic Qualification Current Licensing and Design Basis."[36] This TIA documented the agency position that new seismic information developed by the licensee was required to be evaluated against the design earthquake (DE), the DDE, and HE, including the assumptions used in the supporting safety analyses as described in the FSARU. The staff concluded that comparison only against the LTSP (a margin analysis to the HE) was not sufficient to meet this requirement.

October 2011:

- Pacific Gas and Electric completed an "operability" evaluation of the effect of the new seismic information. The licensee concluded that all plant technical specification SSCs were "operable" because the new ground motions were less than those assumed in the HE. The licensee stated that based on "engineering judgment," the HE was sufficient to satisfy SSE/DDE design basis requirements for "operability."

- Pacific Gas and Electric requested NRC approval to change the Diablo Canyon SSE design basis from the DDE to the HE (License Amendment Request 11-05).[37] The licensee submitted the amendment request following several NRC meetings at which various approaches for incorporating the new seismic information into the CLB were discussed.

December 2011: Pacific Gas and Electric submitted Letter DCL-1 1-124, “Standard Review Plan Comparison Tables for License Amendment Request 11-05, Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake,” to the NRC.[38] This letter included 66 attachments (320 pages) detailing the deviations and exceptions between the HE methodology and the NRC SSE review standards (NUREG 800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition). The NRC had requested this information to aid in the acceptance review of License Amendment Request 11-05.

January 2012: The resident inspector concluded that the PG&E October 2011 “operability” determination failed to meet NRC inspection standards. The inspector based this conclusion on:

- The “operability” determination failed to demonstrate that all ASME Code requirements were met for the higher ground motions. The licensee’s failure to demonstrate Code compliance called in to question the integrity of the reactor coolant pressure boundary following an earthquake on the Los Osos, San Luis Bay or Shoreline faults.

- The “operability” determination failed to demonstrate that all plant SSCs credited in the in the SSE design basis would remain functional at the higher stress levels represented by the new ground motions. The licensee’s comparison of the new ground motions only against the HE was not adequate to demonstrate that SSE/DDE CLB requirements were satisfied.

The inspector again recommended that the agency initiate enforcement action against PG&E based on the licensee’s failure to demonstrate that technical specification required equipment would remain function at the higher ground motions. The agency disagreed with the inspector’s recommendations (documented in non-concurrence NCP-2012).[39] The staff stated that the license’s comparison of the new seismic information against the HE was adequate to demonstrate “initial operability.” The staff also stated that additional review of Licensee Amendment Request 11-05 was needed before the agency had enough information to complete an “operability” determination.

February 2012:

- The NRC issued non-cited violation, 05000275; 323/2011005-02, “Failure to Perform an Operability Determination for New Seismic Information.”[40] This violation addressed the failure of PG&E to initially perform an “operability” determination following development of the new seismic information back in January 2011.

- The NRC closed Unresolved Item: 05000275; 323/2011002-03.[41] The staff concluded that PG&E corrective actions were adequate to conclude all Diablo Canyon SSCs were “operable:”

> *“The staff concluded that 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. In order to complete a comprehensive evaluation, the licensee needed NRC approval of the methodology to be used to complete this evaluation.”*

September 2012: The resident inspector was reassigned from Diablo Canyon

Subsequent NRC Actions to Address New Seismic Information

October 2012:

- The NRC completed an evaluation of the Shoreline fault. The staff concluded that the Shoreline scenario should be considered as a lesser included case under the HE.[42] The NRC stated:

  "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. 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)."

- At the NRC's request, PG&E withdrew License Amendment Request 11-05, "Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake."[43] The license amendment request had not met the NRC's acceptance review standard.

November 2012: The NRC revised Task Interface Agreement (TIA 2011-010) "Diablo Canyon Seismic Qualification Current Licensing and Design Basis."[44] The revised TIA stated:

"…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 Update, as necessary, to include the Shoreline scenario in accordance with the requirements of 10 CFR 50.71(e)."

*"The NRC's letter dated October 12, 2012, and the request for information dated March 12, 2012, (*50.54(f)) *provide guidance for assessing new seismic information and what PG&E is expected to do in the event that it becomes apparent that the new seismic information will lead to a GMRS that is higher than the DDE."*

5.0 NRC Corrective Actions to Address Deficient Seismic Safety Analysis were Inadequate

The Staff Proposed FSARU Update Requires an Amendment to the Diablo Canyon Operating License

The staff recommended that PG&E update the FSARU to include the Shoreline scenario as a lesser included case of the HE.[45] This change exempts the Shoreline fault from the existing SSE/DDE design basis requirements. PG&E is required to review proposed FSARU updates under the provisions of 10 CFR 50.59, "Changes, Tests and Experiments."[46,47] This review determines if the proposed change will require an NRC approved amendment to the Operating License prior to implementation. 10 CFR 50.59 states a license amendment is required for changes that:

"Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the FSARU, or

"Result in a departure from a method of evaluation described in the FSARU used in establishing the design bases or in the safety analyses"

Title 10, Code of the Federal Regulations, Part 50.59, includes the following definitions:

- Change: *"A modification or addition to, or removal from, the facility or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished."*

- Departure from a method of evaluation described in the FSARU used in establishing the design bases or in the safety analyses:

  "Changing any of the elements of the method described in the FSARU unless the results of the analysis are conservative or essentially the same;" or

  "Changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application."

- Facility as described in the FSARU:

  "The structures, systems, and components that are described in the FSARU,"

  "The design and performance requirements for such SSCs described in the FSARU," and

  "The evaluations or methods of evaluation included in the FSARU for such SSCs which demonstrate that their intended function(s) will be accomplished."

- Tests or experiments not described in the FSARU means any activity where any SSC is utilized or controlled in a manner which is either:

  "Outside the reference bounds of the design bases as described in the FSARU" or

  "Inconsistent with the analyses or descriptions in the FSARU."

The 50.59 requirements are expanded in the NRC endorsed guidance contained in Nuclear Energy Institute, NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1:[48,49] Adding the Shoreline scenario to the FSARU HE analysis would result in more than a minimal increase in the likelihood of a malfunction of plant SSC because the change departs from the design basis requirements established by GDC-2. NEI 96-07 states:

"Section 4.3.2 Does the Activity Result in More than a Minimal Increase in the Likelihood of Occurrence of a Malfunction of an SSC Important to Safety?"

"The term "malfunction of an SSC important to safety" refers to the failure of SSC to perform their intended design functions-including both non-safety-related and safety-related SSCs. The cause and mode of a malfunction should be considered in determining whether there is a change in the likelihood of a malfunction."

"In determining whether there is more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC to perform its design function as described in the UFSAR, the first step is to determine what SSCs are affected by the proposed activity. Next, the effects of the proposed activity on the affected SSCs should be determined. This evaluation should include both direct and indirect effects."

"Changes in design requirements for earthquakes, tornadoes, and other natural phenomena should be treated as potentially affecting the likelihood of malfunction."

"Although this criterion allows minimal increases, licensees must still meet applicable regulatory requirements and other acceptance criteria to which they are committed (such as contained in Regulatory Guides and nationally recognized industry consensus standards, e.g., the ASME B&PV Code and IEEE standards). Further, departures from the design, fabrication, construction, testing, and performance standards as outlined in the General Design Criteria (Appendix A to Part 50) are not compatible with a "no more than minimal increase" standard."

The Shoreline Scenario results in SSC seismic stress beyond the plant SSE qualification basis. Exposure to higher levels of stress results in an increases likelihood of a malfunction of these SSCs. The change also increases the likelihood of a malfunction of SSCs important to safety because removing the Shoreline scenario from the SSE/DDE departs from applicable regulatory requirements and other acceptance criteria the PG&E had committed to for the SSE/DDE.

The staff proposed FSARU update also requires a licensee amendment because applying the HE methodology to Shoreline fault changes the methods described in the FSARU for establishing the SSE design basis. NEI 96-07 states:

"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?"

"The UFSAR contains design and licensing basis information for a nuclear power facility, including description on how regulatory requirements for design are met and how the facility responds to various design basis accidents and events. Analytical methods are a fundamental part of demonstrating how the design meets regulatory requirements and why the facility's response to accidents and events is acceptable. As such, in cases where the analytical methodology was considered to be an important part of the conclusion that the facility met the required design bases, these analytical methods were described in the UFSAR and received varying levels of NRC review and approval during licensing."

"As discussed further below, for purposes of evaluations under this criterion, the following changes are considered a departure from a method of evaluation described in the UFSAR:"

- *Changes to any element of analysis methodology that yield results that are non-conservative or not essentially the same as the results from the analyses of record.*

- *Use of new or different methods of evaluation that are not approved by NRC for the intended application.*

As described in the FSAR Section 2.5, the seismic SSE/DDE design basis includes the shoreline scenario because the fault is located within 75 miles of plant site. The HE was an exception to this design basis. To change the plant safety analyses to also exclude the Shoreline scenario from the seismic design basis results in a "departure from a method described in the FSARU" that was used to establish the SSE/DDE design basis. NRC approval, in the form a license amendment, is required before the HE methods, including assumptions, initial conditions, etc., can be applied to other local seismic features.

The licensee previously requested that the NRC approve the new information as part of the HE (License Amendment Request 11-05).[50] However, the NRC did not accept the license amendment request for review. The NRC standard for acceptance review required that the license amendment request demonstrate that the proposed change would not impose a “significant hazard.”

The NRC corrective action was also inadequate because the disposition of the San Luis Bay and Los Osos faults was omitted. PG&E had determined that these faults also had significant impact on plant equipment. The FSARU SSE safety analysis is also nonconforming with respect to the deterministic evaluations of the San Luis Bay and Los Osos faults.

Existing Regulatory Framework

Title 10 of the Code of Federal Regulations, Parts 50.34 and 50.71(e), required PG&E to include information in the FSARU that describes the facility, presents the design bases and the limits on its operation, and present a safety analysis of the SSCs and of the facility as a whole. These regulations define safety analyses as analyses performed pursuant to NRC requirement to demonstrate:

(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 that could result in potential offsite exposures comparable to the guidelines in 10 CFR 50.34(a)(1).

The safety analysis is required to demonstrate that acceptance criteria for the facility's capability to withstand or respond to postulated events are met. Supporting FSARU analyses are required to demonstrate that SSC design functions will be accomplished as credited in the accident analyses of events that the facility is required to withstand such as earthquakes and accidents. As previously discussed, the new seismic information resulted in the existing FSARU safety analysis nonconforming with the design basis and Parts 50.34 and 50.71.

Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion III, “Design Control” required PG&E to maintain the plant configuration consistent with regulatory requirements and the design basis:

> “*Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled.”*

A violation of Criterion III occurred after PG&E concluded that the new seismic information would produce greater ground motion that bound by the plant SSE safety analysis and design bases (established by GDC 2 and Part 100). Design measures no longer provided assurance that the important to safety SSCs are capable of performing the required safety functions at the higher ground motions.

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," required PG&E to implement prompt corrective action to restore the plant "as described" in the safety analysis and design basis:

> *"Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management."*

A violation of Criterion XVI occurred after PG&E failed to take prompt corrective actions to correct deficiencies in the plant safety analysis, as required by 10 CFR 50.34 and 50.71(e) and to restore plant SSCs within the capability of meeting the seismic design basis as required by Appendix B, Criterion III .

No Viable Corrective Action Path

This regulatory framework ensures that licensees promptly restore plant operation within the boundary of the design basis and NRC approved safety analysis. Changing the local seismology to meet the CLB is beyond the licensee's control. Adapting plant SSCs to meet the current design basis requirements, if even possible, would require extensive seismic retrofits. Modifying the design basis and safety analysis to accommodate the new information would require an amendment to the Operating License. However, the NRC was not willing to accept the amendment request for review. The end result is the licensee is without a viable corrective action path to deal with the current nonconforming and unanalyzed conditions. The lack of a clear corrective path does not waive the NRC's responsibility to enforce current regulatory requirements for prompt corrective actions and to ensure plant operation is maintained within the boundaries of the approved safety analysis.

Fukishima Near-Term Task Force 10 CFR 50.54(f) Requested Information is not Applicable to the Current Diablo Canyon Nonconforming and Unanalyzed Conditions

In March 2012, the NRC requested information related to the reevaluation of seismic hazards at all power reactor facilities.[51] This request was in response to recommendations from the NRC Near-Term Task Force review of the Fukishima accident. The NRC requested that PG&E develop new probabilistic ground motion models and compare the results of these models to the existing deterministic SSE/DDE. This comparison will provide risk information related to the local geology. The agency will use this risk based information to make future licensing decisions.

The requested information is probabilistic in nature. The Diablo Canyon design bases are deterministic in nature, assuming that the event occurs and requirement specific acceptance criteria are met. While the requested 50.54(f) information will provide risk insights to earthquake hazards affecting the plant, this information is not directly relevant to the CLB. In contrast, the new deterministic information developed by PG&E for the San Luis Bay, Los Osos, and Shoreline faults was directly comparable to the existing facility design bases and Operating License. This new information was sufficient to conclude that the plant is operating outside of the NRC approved safety analysis and the design bases. The current regulatory framework requires these nonconforming and unanalyzed conditions to be promptly disposition within the context of the CLB. These actions are required independent of information developed in response to the 50.54(f) request.

Summary

Pacific Gas and Electric submitted to the NRC information concluding that three local earthquake faults are capable of producing greater ground motion than bounded by the NRC approved safety analysis and the design basis. This condition rendered the plant seismic safety analysis nonconforming with NRC regulations. The NRC has failed to enforce quality requirements (Part 50, Appendix B) that required the licensee to take prompt action to correct the nonconforming safety analysis.

The Staff recommended that PG&E updated the FSARU to include one of these faults as a lesser case under the HE. This action bypassed the regulatory processes (50.2 & 50.90) design to ensure that these changes would not result in a significant hazard. NRC regulations (50.59) require that the licensee first obtain a license amendment before updating the FSARU with this information. A license amendment is required because this change attaches the same regulatory dispensation approved for the Hosgri to the Shoreline fault. The staff's conclusion that "***reasonable assurance of safety"*** is not an adequate basis to bypass the regulatory requirements to amend the facility Operating License.

The licensee previously submitted a license amendment request to redefine the HE as the SSE for the facility. However, this request did not meet the NRC's minimum standards for acceptance into the review process. As a result, the Staff requested that PG&E withdraw the request.

Deferral of corrective action pending completion of the Fukishima Near-Term Task Force seismic reviews is inconsistent with the current regulatory framework. The new seismic information generated by the licensee was sufficient to conclude that the facility is currently operating outside of the current safety analysis and design basis.

The staff's corrective action was also deficient because the reevaluation of the San Luis Bay and Los Osos faults was omitted. While these faults were initially evaluated in the LTSP, the licensee had not deposition the effect of the higher ground motions on the SSE/DDE safety analysis as required by NRC quality regulations. The SSE/DDE safety analysis is also nonconforming due to the higher ground motions associates with these faults.

6.0 The NRC has not Verified Plant Technical Specification Required SSCs are "Operable"

Plant operators are required to demonstrate that all affected technical specification required SSCs are "operable" following identification of nonconforming or unanalyzed conditions. The "operability" processes provide a basis that the reactors can be operated safely during the corrective action period.

Applicability of "Operability" Process

A nonconforming condition exists because the Diablo Canyon FSARU safety analysis is no longer compliant with the regulatory requirements of GCD-2 for earthquakes. NRC "operability" policy states:[52]

Failure to meet a GDC in the CLB should be treated as a degraded or nonconforming condition and, therefore, the technical guidance in this document is applicable.

Also, this was an unanalyzed condition because the new information indicated that the ground motions assumed in the SSE/DDE safety analysis (earthquakes B & D) were no longer bounding for the plant seismic qualification basis. Nonconforming or unanalyzed conditions that call into question the capability of technical specification required SSCs to perform the specified safety functions are required to be evaluated for "operability."[53]

Description of NRC "Operability" Process

The applicable CLB requirements for seismic qualification must be identified before "operability" can be evaluated. The new deterministic ground motions were applicable to the SSE/DDE safety analysis, as described in FSARU Section 2.5 and 3.7, because:

- The new seismic information was identified on earthquake faults within 75 miles from the plant.
- The new seismic information was not associated with the Hosgri fault (the NRC approved exception).
- The SSE/DDE safety analysis implemented the plant seismic design basis, and License and regulatory requirements.

Engineering Margins

The "operability" process allows licensees to use engineering margins. Engineering margins include the difference between actual SSC capability and the performance requirements specified in the CLB. To illustrate this concept, consider the emergency core cooling system example discussed in Sections 2.0 and 3.0. This system has motor operated valves and instruments located around the 88 foot elevation level in the containment building. The seismic stress used to develop the original qualification of these SSCs was shown in Figure 1. The new seismic information calls into question the "operability" of these SSCs because an earthquake on the San Luis Bay fault would result in much higher vibratory motions at this plant location than considered in the SSE/DDE safety analysis. The design basis remains unchanged; these SSCs still are required to remain functional following the

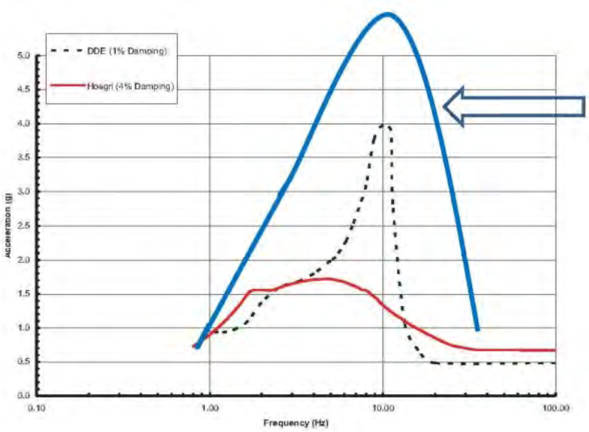

Figure 4
Comparison of the DDE/SSE and the HE Floor Response Spectrum, Containment Elevation 88'

A comparison of the new seismic information against the existing SSE/DDE safety analysis would yield seismic stress greater than the values used during the original SSC qualification. However, in many cases, the actual SSC qualification tests were performed at higher levels than required to meet the design basis. These higher qualification levels provide engineering margin that may be recovered for "operability."

The "operability" process does not require that the new ground motions be reviewed against the HE (red line). As described in the CLB, the HE is limited to an earthquake on the Hosgri fault. Also, at this plant location, seismic qualification would likely be bound by the DDE rather than HE.

"maximum earthquake." The vibratory motions associated with the "maximum earthquake" have changed.

Plant components were generally qualified at higher stress levels (shaking) than the limits specified in the design and engineering specifications. The difference between the reevaluated stress and the actual stress levels used to qualify theses SSCs provides engineering margin. Figure 4 compares the postulated increase in vibratory motions from the San Luis Bay fault against the original DDE qualification levels. The SSCs could be considered "operable," if the original qualification was bound at the new stress levels.

"Operability" also provides for the use of "alternate methods." The license may present an alternate method that demonstrates that the SSC will remain functional beyond the qualified level of "shaking." The NRC standard is a "reasonable assurance" that the SSC will be capable of performing the required safety functions, as described in the CLB, at the higher vibratory motions. For example, the licensee could provide alternate testing data that demonstrates the SSC would remain functional at the higher vibratory motions.

Use of Code Margins

Engineering margin in the ASME Code calculations may be similarly credited for "operability." For example, again consider the emergency core cooling system example. To be considered "operable," the Code acceptance limits must be met at the higher stress levels for the system piping and pipe hangers. Plant operators may credit the margin between the actual pipe stress and Code acceptance limits. For example, the original DDE calculation may have determined that an emergency core cooling pipe weld had bending moment of 120,000 lbf-in with a Code acceptance limit of 200,000 lbf-in. The original calculation provided 80,000 lbf-in of margin. This margin may be used for "operability" when the bending moment is recalculated at the higher seismic stress. The component would be considered "operable" provided the new bending moment is still less than the Code acceptance limits.

Use of Safety Analysis Margins

Methods and "supporting design information," used in the safety analysis also provide margins that may be recovered in the "operability" process. For example, consider the affect damping values have on seismic qualification. Energy dissipation within a structure during an earthquake depends on a number of factors, including the types of joints or connections used within the structure, the structural material, and the magnitude of deformations experienced. In a dynamic elastic analysis, this energy dissipation is usually accounted for by specifying an amount of viscous damping. The damping value affects the energy dissipation in the analytical model. Figure 5 shows the relationship between acceleration and velocity as a function of damping.[54] This relationship determines the level of SSC vibratory motion for seismic qualification. Figure 6 illustrates the relationship between the damping value and the predicted attenuation of seismic energy. Generally, the higher the assumed damping value, for a given spectra, the lower the resulting vibratory motion transmitted to the SSC.

FSARU Section 3.7.1.3, "Critical Damping Values," specified the damping values used in the SSE/DDE safety analysis. NRC approval of the SSE/DDE safety analysis included comparing these damping values against NRC review criteria. However, these damping values may contain margin that could be recovered in the "operability" process. The NRC "operability" policy allows use of "engineering judgment." Use of higher damping values would reduce the amount of seismic stress assumed to attenuate to the plant SSCs. Use of "engineering judgment" is subject to a couple of tests. [55,56]

"In such instances, the application of the alternative analysis must be consistent with the technical specifications, license condition, or regulation"

"If the analytic method in question is described in the CLB, the licensee should evaluate the situation-specific application of this method, including the differences between the CLB-described analyses and the proposed application in support of the operability determination process."

"Occasionally, a regulation or license condition may specify the name of the analytic method for a particular application. In such instances, the application of the alternative analysis must be consistent with the technical specifications, license condition, or regulation."

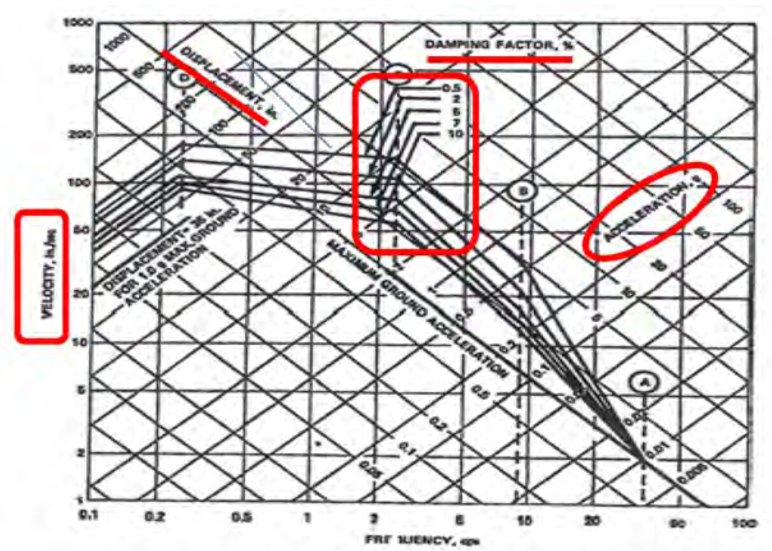

Figure 5
Relationship between Acceleration, Velocity as a Function of Damping[57]

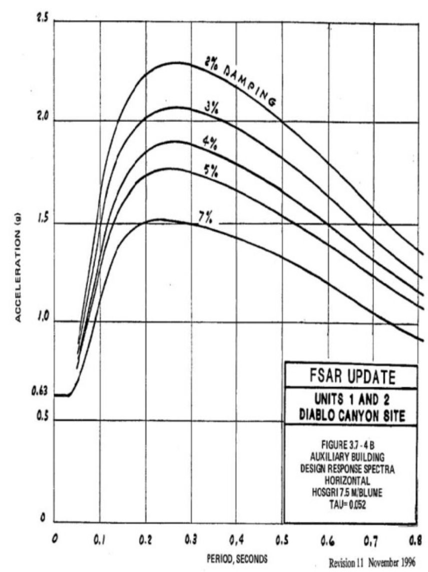

Figure 6
Relationship between Damping & Propagation of Seismic Energy

Higher damping values may be used for "operability," provided that these values are appropriate to the application, as defined in the CLB. For example, the damping values specified for the SSE in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants,[58]" may be used. Also, damping values higher than presented in Regulatory Guide 1.61, may also be used provided that they have been NRC approved for the specific application and material.

Engineering Margins were Insufficient to Demonstrate "Operability"

These NRC principles were not practical for determined SSC "operability" for the new seismic information. The new vibratory motions are much greater than those bound by the existing SSE/DDE CLB. This combined with very little engineering margin available in the original SSE/DDE safety analysis would likely result in the CLB acceptance criteria to be exceeded.

NRC Conclusion all Diablo Canyon Seismically Qualified Equipment were "Operable"

The NRC concluded that all Diablo Canyon technical specification required SSCs were "operable" after performing a review of new earthquake potential.[59] The staff stated that NRC "operability" requirements were satisfied because the new ground motions were bound by those assumed in the HE and LTSP. During this review, the staff also stated:

- *"The NRC will not ask the licensee to use the new ground motion input data in the DE or DDE 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 ground motion data and the calculation method, including damping values, are correlated parameters. They must be based on the same assumptions for the calculation to have validity."*
- *"It is appropriate for the licensee to use the available new ground motion data in the HE analysis because the new ground motion data is consistent with that evaluation."*

"Operability" was not Evaluated Against the Current Design and Licensing Bases

The NRC failed to assess "operability" against the CLB. The staff's approach to exclude the SSE/DDE design basis and safety analysis for the seismic "operability" determination was not support by NRC "operability" policy. "Operability required that SSC performance be compared against CLB requirements.[60]

> *"In order to be considered operable, an SSC must be capable of performing the safety functions specified by its design, within the required range of design physical conditions"*

The CLB includes the SSE/DDE safety analysis. This safety analysis implements the plant seismic design basis and demonstrates specific regulatory requirements are met. The staff's argument for not using the SSE/DDE for "operability" was that the new seismic loads were beyond the capability and limitations of the safety analysis. In other words, the NRC acceptance criteria cannot be demonstrated when the new ground motions are compared against the plant SSE design basis. When the "operability" determination fails to demonstrate these specified safety functions can be met, then the system should be considered "inoperable."[61]

> *"The specified function(s) of the system, subsystem, train, component or device (hereafter referred to as system) is that specified safety function(s) in the CLB for the facility….When system capability is degraded to a point where it cannot perform with reasonable expectation or reliability, the system should be judged inoperable."*

The staff's argument is correct that the HE, including assumptions, initial conditions, and acceptance criteria, is more consistent with the new ground motions. The HE methodology may be adapted by the staff as a basis for a licensing action. However, the HE may not be used as a standard for "operability" because the methodology was not approved for the SSE as described in the CLB. As such, the HE cannot be the basis to conclude SSCs are "operable" for the SSE design basis.

While the HE damping values and other inputs are correlated parameters, the CLB restricts the use of these values to analysis of the Hosgri fault (FSARU Section 3.7). The CLB prescribes the damping values and other inputs to be used for the SSE. Substitution of HE damping and other inputs for "operability," based solely on the magnitude of the new ground motions, is inappropriate. Use of higher damping values is permitted provided the NRC has approved those values for same application (for the SSE and specified materials). The NRC "Operability" process requires these input values be consistent with those used in the SSE CLB.

As described in Section 4.0, "Chronology," the licensee had requested NRC approval to use the HE methodology for SSE applications (License Amendment Request 11-05).[62] PG&E Letter DCL-1 1-124, described the considerable departure between the HE methodology and the NRC's SSE approval standards.[63] The end result was that the NRC did not accept the licensee's request for review. The licensee was unable to demonstrate that use of the HE for SSE applications met the "no significant hazards consideration" standard.[64,65]

While not appropriate for "operability," use of the HE analysis, and correlated input parameters, may use as a basis for NRC approval of an amendment to the facility Operating License or waving regulatory (50.2, 50.55a) or technical specification requirements.

The NRC "Operability" Method Over-Predicted SSC Performance when Compared to the CLB

NRC policy allows use of alternative analytical methods when performing "operability" determinations. However, these methods are required to be consistent with the methods used in the CLB and not over-predict the capability of plant SSC.[66]

> *"If the analytic method is not currently described in the CLB, the models employed must be capable of properly characterizing the SSC's performance. This includes modeling of the effect of the degraded or nonconforming condition."*
>
> *"Acceptable alternative methods such as the use of "best estimate" codes, methods, and techniques. In these cases, the evaluation should ensure that the SSC's performance is not over-predicted by performing a benchmark comparison of the non-CLB analysis methods to the applicable CLB analysis methods".*

Comparing the new information solely against the HE attaches all of the HE methods and assumptions to the new information. These methods and assumptions result in significantly underestimating the resulting seismic stress that plant SSCs would be exposed when compared to the SSE/DDE methods described in the CLB. As a result, use of the HE over-predicts SSC seismic performance when compared to the SSE/DDE CLB methods.

As discussed in Section 3.0, the SSE/DDE safety analysis predicated greater stress (shaking) and was more limiting for the seismic qualification of some plant SSCs than for the HE. As demonstrated in these examples, ground motion taken alone is not a meaningful representation of the seismic design bases.[67] Considered the control rod drive mechanism bending moment example discussed in Section 3.0, "Diablo Canyon Current Licensing Bases." Appling the HE methods to the San Luis Bay ground motions would result in less stress than shown in Figure 2. This is because the San Luis Bay fault spectrum is slightly lower than the HE. However, applying SSE/DDE methods to San Luis Bay fault would result in significantly larger stresses than shown in Figure 3. HE methods are not appropriate for "operability" because these method significantly over-predict the capability of plant SSCs when compared to the CLB method (SSE/DDE).

NRC "Operability" Review Failed to Demonstrate ASME Code Requirements were Met

Title 10, Code of Federal Regulations, Part 50.55a, Codes and "Standards," requires the licensee to meet "the ASME Boiler and Pressure Vessel Code requirements. The Code requires the SSE "maximum earthquake" dynamic loading to be included when demonstrating the acceptance limits are met for Class1 systems. The new information concluded that higher vibratory motions could affect plant Code components that were used in the original SSE/DDE calculations. The HE cannot be used for SSE Code compliance because the HE (along with the methods, assumptions, etc.) was not identified as the SSE in the CLB. This new loading calls into question if Code limits can still be met given the potential for a much larger "maximum earthquake." "Operability" requires certain plant SSCs either meet the ASME Code acceptance criteria or provisions in an NRC approved Code Case.[68]

> *"When ASME Class 1 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."*

> *"Structures may be required to be operable by the Technical Specifications, or they may be related support functions for SSCs in the Technical Specifications…..As long as the identified degradation does not result in exceeding acceptance limits specified in applicable design codes and standards referenced in the design basis documents, the affected structure is either operable or functional."*

> *"When a degradation or nonconformance associated with piping or pipe supports is discovered, the licensee should use the criteria in Appendix F of Section III of the ASME Boiler and Pressure Vessel Code for operability determinations. The licensee should continue to use these criteria until CLB criteria can be satisfied (normally the next refueling outage). For SSCs that do not meet the above criteria but are otherwise determined to be operable, licensees should treat the SSCs as if inoperable until NRC approval is obtained to use any additional criteria or evaluation methods to determine operability. Where a piping support is determined to be inoperable, the licensee should determine the operability of the associated piping system."*

The NRC Inappropriately Deferred "Operability" Pending License Amendment Request Approval

The NRC stated:[69]

> *"The staff concluded that 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. In order to complete a comprehensive evaluation, the licensee needed NRC approval of the methodology to be used to complete this evaluation."*

NRC "operability" does not provide for an indeterminate state.[70] Plant SSCs are either "operable" or "inoperable." The "operability" process also does not include "initial basis" for "Operability." NRC policy only provides for immediate and prompt "operability" determinations. Prompt "operability" determinations should be completed within the technical specification out-of-service times.[71] For the seismic issues, this would be about 24 hours. Operability is assessed against the CLB, not against a pending license amendment request. Plant SSCs should be immediately considered "inoperable" when inadequate margin is available, as described in the CLB, to ensure the components are capable of performing the CLB specified safety functions. The staff's deferral of "comprehensive evaluation" for "operability" was inconsistent with the current regulatory framework and Diablo Canyon Operating License.

Research Information Letter 12-01, "Confirmatory Analysis of Seismic Hazard at the Diablo Canyon Power Plant from the Shoreline Fault Zone"

In October 2012, the NRC released Research Information Letter 12-01.[72,73] This Letter included the results of a conformational analysis of potential ground motions that could be produced by the Shoreline fault. The Letter did not address the seismic qualification of plant SSCs, ASME Code requirements, or "operability." However, the Letter stated:

> *"It should be reiterated that the NRC staff has concluded 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 HE ground motion and the 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 existing design basis for the plant already is sufficient to withstand those ground motions."*

- The staff's conclusion of "***reasonable assurance of safety"*** is not applicable to either resolving the noncompliant safety analysis or determining "operability." This information may be useful input for regulatory decisions, such as approval of license amendments or exemptions from existing regulations. However, the current regularly framework and facility Operating License requirements are still required to be satisfied. Continued operation of Diablo Canyon is dependent on successful demonstration of SSC "operability." Since "operability" is evaluated against the CLB, this demonstration may require amendment of the Operating License and/or waving current regulatory requirements. The staff's conclusion of "***reasonable assurance of safety"*** may be used to support justification for these regulatory actions.

- The current regulatory framework does not provide for deferral of the "operability" evaluation until development of new probabilistic ground motions models, such as those requested by the Fukishima Near-Term Task Force. Sufficient information is currently available to assess "operability." Because the facility design bases is deterministic in nature, the NRC "operability" policy specifically excludes use of probabilistic information:[74]

  > *"Probabilistic risk assessment is a valuable tool for evaluating accident scenarios because it can consider the probabilities of occurrence of accidents or external events. Nevertheless, the definition of operability is that the SSC must be capable of performing its specified safety function or functions, which inherently assumes that the event occurs and that the safety function or functions can be performed. Therefore, the use of PRA or probabilities of occurrence of accidents or external events is not consistent with the assumption that the event occurs, and is not acceptable for making operability decisions."*

Summary

The staff failed to enforce plant technical specification requirements to shut down the Diablo Canyon reactors. Continued reactor operation was dependent on the licensee's demonstration that technical specification required SSCs were "operability" following discovery of nonconforming and unanalyzed conditions associated with the new seismic information. The failure to demonstrate "operability," required the licensee to take the prescribed technical specification actions for the "inoperable" equipment, including shutdown the reactors. The "operability" determination method used by PG&E was inadequate because:

- Neither the HE nor the LTSP methods were approved by the NRC to be used for the Diablo Canyon SSE design basis. The CLB defined the HE as an exception to the SSE and was only approved for evaluating the Hosgri fault. The LTSP is not part of the seismic design basis.

- Use of the HE and LTSP over-predicts SSC performance when compared to the CLB methods used for the SSE/DDE. Neither the HE nor the LTSP are bounding for SSC seismic qualification at Diablo Canyon. Comparisons limited to only ground motion are meaningless for "operability." These comparisons omit other relative CLB requirements including the methods, assumptions, initial conditions, and acceptance criteria applicable to each evaluation.

- Comparison of the new information only to the HE and LTSP failed to demonstrate that the requirements of the American Society of Mechanical Engineers' (ASME) Boiler and Pressure Vessel Code are met at the higher ground motions. "Operability" requires that the Code acceptance criteria are met for key plant components, including the reactor coolant pressure boundary.

The staff's conclusion in Research Information Letter 12-01 that "***reasonable assurance of safety"*** exists does not provide an adequate basis for concluding "operability." A "***reasonable assurance of safety"*** does not satisfy the requirement that plant SSCs are capable of meeting the specific safety functions described in the SSE/DDE safety analysis and design basis.

7.0 Previous Attempts for Resolution

a. The author of the DPO discussed these issues with senior Region IV management, including the region administrator, and NRR Division of Operating Reactor Licensing staff between the fall 2010 and the fall of 2012 (see Section 4.0, "Chronology").

b. The author of the DPO was not provided an opportunity to review or supply input to either the October 2012 NRC letter[75] or the revised TIA 11-05.[76]

c. The author of the DPO provided written recommendations for regulatory action in January 2011.[77]

d. The author of the DPO discussed the definition of "deign basis" and applicability of 10 CFR 50.59 to the NRC recommend FSARU changes with the Region IV, Division of Reactor Projects, Chief of Reactor Projects Branch B, on June 27, 2013.

e. The author of the DPO non-concurred on Diablo Canyon Power Plant Inspection report 050000275/323-2011005, ML120450843 NCP-2012-001[78]

Appendix – Comparison of 1967 GDC 2 with 10 CFR 50, Appendix A, GDC 2

1967 GDC Criterion 2, 1967 - Performance Standards (Category A)	Appendix A to Part 50, General Design Criteria for Nuclear Power Plants, *Criterion 2—Design bases for protection against natural phenomena*.
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 (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.	Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Applicability of 10 CFR 50, Appendix A, GDC 2 to Diablo Canyon

PG&E committed to address any exceptions taken to Appendix A to Part 50, General Design Criteria, during the original Diablo Canyon licensing process.[79] Prior to the NRC issuing the Operating License, PG&E stated that the Diablo Canyon conforms to 10 CFR 50, Appendix A, GDC 2, (without exception).[80] The NRC recently issued Notice of Violation (VIO 05000275;323/2012-004-01, “Failure to Incorporate Required Information in the Final Safety Analysis Report Update”)[81] associated with the failure of PG&E to include this information in the FSARU.

End Notes:

[1] Report on the Analysis of the Shoreline Fault Zone, Central Coast California to the USNRC, PG&E , January 2011, Figure 6-19, page 6-51, ADAMS ML110140400

[2] Diablo Canyon Power Plant, Unit Nos. 1 and 2 -NRC Review of Shoreline Fault (TAC NOS. ME5306 and ME5307) October 12, 2012 ML120730106

[3] Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011005 and 05000323/2011005 (ML 120450843), Section 1R15, Operability Evaluations, February 14, 2012

[4] Non-Concurrence, NCP-2012-001, Diablo Canyon Power Plant Inspection report 050000275/323-2011005, ML120450843

[5] Diablo Canyon Power Plant, Unit Nos. 1 And 2 -NRC Review of Shoreline Fault (TAC Nos. ME5306 and ME5307) October 12, 2012, ML120730106

[6] Operation – Safety and Compliance, Part 9900: Technical Guidance http://www.nrc.gov/reading-rm/doc-collections/insp-manual/technical-guidance/tg-operation-safety.pdf

[7] RIS 2005-20, NRC Inspection Manual, PART 9900: Technical Guidance, Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety, Attachment, Section 3.1, Current Licensing Basis (http://pbadupws.nrc.gov/docs/ML0735/ML073531346.pdf)

[8] Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," (http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/rg/division-1/division-1-181.html), 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"

[9] NEI 97.04, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases, Appendix B, page B21, "Seismic Topical Design Bases." ML003678532, (https://adamsxt.nrc.gov/WorkplaceXT/IBMgetContent?vsId={D8B2D4B4-E3BD-4488-B75B-E832F3B33F5D}&objectType=document&id={862C2A33-9C8C-44D8-833F-E54E3D7F44A6}&objectStoreName=Main.__.Library)

[10] Ibid 9, Page B21

[11] PG&E stated that Diablo Canyon conforms to 10 CFR 50, App A, GDC 2, Letter to FJ Miraglia, NRC, Division of Licensing, from PA Crane, PG&E, September 10, 1981

[12] 10 CFR 50.34 "Contents of Applications; Technical Information

[13] 10 CFR 50.71, "Maintenance of Records, Making of Reports,"

[14] Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e), ML003740112; endorsed use of NEI 98-03, Revision 1,Guidelines For Updating Final Safety Analysis Reports (http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/rg/01-181/)

[15] The Diablo Canyon CLB designated the following SSCs as Seismic Category I. SSCs listed per RG 1.29

[16] Seismic Evaluation for Postulated 7.5M Hosgri Earthquake, DCPP Units 1&2, PG&E

[17] Areva Replacement reactor head, Calculation 6 CS 20327, Appendix 2, revision A, "Primary Stress Evaluations, Design Conditions DE 3%, DDE 4% + LOCA, HE 4% + Displacement

[18] Ibid 17

[19] Diablo Canyon, Unit 1, Current Facility Operating License DPR-80, Tech Specs, ML09181008 (http://adamswebsearch2.nrc.gov/webSearch2/doccontent.jsp?doc={B9458677-D714-43C8-A0C0-12DFC3A173EF})

[20] Regulatory Issue Summary 2005-20, "Revision to NRC Inspection Manual Part 9900 Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety". (**http://pbadupws.nrc.gov/docs/ML0734/ML073440103.pdf)**

[21] Ibid 7, Section 3.8, Operability

[22] Ibid 7, Section 3.10, Specified Safety Function

[23] Inspection Procedure 71111.15, Operability Determinations and Functionality Assessments, for a two unit site (http://pbadupws.nrc.gov/docs/ML1120/ML112010663.pdf)

[24] Event Number 44675, Offsite Notification and Media Briefing due to Potential Discovery of Off Shore Fault near Plant, November 21, 2008

[25] Notification 50086062, "LTCA-Ident of Seis Lineament Offsiter," November 14, 2008

[26] Diablo Canyon Power Plant, Unit Nos. 1 and 2 – NRC Preliminary Review of Potential Shoreline Fault, April 8, 2009

[27] Notification 50086062, "LTCA-Ident of Seis Lineament Offsiter," November 14, 2008

[28] Notification 50341463, NRC SRI Question on the Shoreline Fault Study, September 14, 2010

[29] Notification 50086062, "LTCA-Ident of Seis Lineament Offsiter," Task 24, October 10, 2010

[30] Notification 50086062, "LTCA-Ident of Seis Lineament Offsiter," Task 30, December 16, 2010

[31] PG&E submitted to the NRC "Report on the Analysis of the Shoreline Fault, Central Coast California, January, 7, 2011, ML 110140400

[32] E-Mail and Attachment, from Michael Peck to Geoffrey Miller and et al, Subject: ACT: Diablo Canyon - Recommendation for Regulatory Disposition, Attachments: Diablo Canyon Seismic White Paper.docx, February 3, 2011

[33] Report on the Analysis of the Shoreline Fault Zone, Central Coast California to the USNRC, PG&E , January 2011

[34] From Figure 6-19, Report on the Analysis of the Shoreline Fault Zone, Central Coast California to the USNRC, PG&E , January 2011

[35] Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011002 and 05000323/2011002, May 11, 2011, (http://adamswebsearch.nrc.gov/idmws/ViewDocByAccession.asp?AccessionNumber=ML111310608)

[36] Task Interface Agreement (TIA) – Concurrence on Diablo Canyon Seismic Qualification Current Licensing and Design Basis (TIA 2011-010), August 1, 2011, ML112130665

[37] PG&E Letter DCL-1 1-097, License Amendment Request 11-05, "Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake," ADAMS ML 11312A166
[38] PG&E submitted Letter DCL-1 1-124, “Standard Review Plan Comparison Tables for License Amendment Request 11-05, Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake, December 6, 2011, ML 11342A238
[39] Non-Concurrence, NCP-2012-001, Diablo Canyon Power Plant Inspection report 050000275/323-2011005, ML120450843
[40] Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011005 and 05000323/2011005, Section 1R15, February 14, 2012 (http://adamswebsearch.nrc.gov/webSearch2/doccontent.jsp?doc={D8DD93EB-2036-4A68-8ADC-39F302FFEAEE})
[41] Ibid 40
[42] Diablo Canyon Power Plant, Unit Nos. 1 And 2 -NRC Review of Shoreline Fault (TAC NOS. ME5306 and ME5307), October 12, 2012, ML120730106
[43] PG&E Letter DCL-12-1 08, Withdrawal of License Amendment Request 11-05, "Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake," October 25, 2012, ML 12300A105
[44] Revised Response To Task Interface Agreement – Diablo Canyon Seismic Qualification Current Licensing and Design Basis, TIA 2011-010 (TIA 2012-012) (TAC NOS. ME9840 and ME9841), February 14, 2012, ML12297A199
[45] Diablo Canyon Power Plant, Unit Nos. 1 and 2 -NRC Review of Shoreline Fault (TAC NOS. ME5306 and ME5307), October 12, 2012, ML120730106
[46] Regulatory Guide 1.181, Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e), (http://pbadupws.nrc.gov/docs/ML0037/ML003740112.pdf)
[47] NEI 98-03, Revision 1, “Guidelines for Updating FSARs, June 1999, ML003779023
[48] Regulatory Guide 1.187, Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments, (http://pbadupws.nrc.gov/docs/ML0037/ML003759710.pdf)
[49] NEI 96-07, “Guidelines for10 CFR 50.59 Evaluations,” ML003636043
[50] PG&E Letter DCL-1 1-097, License Amendment Request 11-05, "Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake," ADAMS ML 11312A166
[51] 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, March 12, 2012, ML12056A046 & ML12053A340
[52] Ibid 7, Section C-1 Relationship Between the General Design Criteria and the Technical Specifications
[53] Ibid 7
[54] Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants, http://pbadupws.nrc.gov/docs/ML1303/ML13038A102.pdf
[55] Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants, October 1973, http://pbadupws.nrc.gov/docs/ML0037/ML003740213.pdf
[56] Ibid 7, Section C.4, Use of Alternative Analytical Methods in Operability Determinations
[57] Ibid 54
[58] Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants, October 1973, http://pbadupws.nrc.gov/docs/ML0037/ML003740213.pdf
[59] Non-Concurrence, NCP-2012-001, Diablo Canyon Power Plant Inspection report 050000275/323-2011005, ML120450843
[60] Ibid 7
[61] Ibid 7, Section 3.10,
[62] PG&E Letter DCL-1 1-097, License Amendment Request 11-05, "Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake," ADAMS ML11312A166
[63] PG&E submitted Letter DCL-1 1-124, “Standard Review Plan Comparison Tables for License Amendment Request 11-05, Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake, December 6, 2011, ML11342A238
[64] As defined in 10 CFR 2.102, 2.107, & 2.108, and NRR Office Instruction LIC-109, Acceptance Review Procedures, Revision 1, ML091810088
[65] Discussion with Diablo Canyon NRR PM, January 2012
[66] Ibid 7, Section C-4, Use of Alternative Analytical Methods in Operability Determinations
[67] Supplement No. 7 to the Safety Evaluation Report By The Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission In The Matter Of Pacific Gas And Electric Company Diablo Canyon Nuclear Power Station, Units 1 And 2 Docket Nos. 50-275 And 50-323, 2.5.2 Seismology
[68] NRC Approved Code Cases (exceptions to Code requirements), Regulatory Guide 1.84, Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, (http://pbadupws.nrc.gov/docs/ML1018/ML101800532.pdf)

[69] Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2011005 and 05000323/2011005, Section 1R15, February 14, 2012 (http://adamswebsearch.nrc.gov/webSearch2/doccontent.jsp?doc={D8DD93EB-2036-4A68-8ADC-39F302FFEAEE})

[70] Ibid 7, Section 3.9, Reasonable Expectation

[71] Ibid 7, Section 4.6.2, Prompt Determinations

[72] Diablo Canyon Power Plant, Unit Nos. 1 And 2 -NRC Review of Shoreline Fault (TAC NOS. ME5306 and ME5307), October 12, 2012, ML120730106

[73] Research Information Letter (RIL) 12-01 "Confirmatory Analysis of Seismic Hazard at the Diablo Canyon Power Plant from the Shoreline Fault Zone" (ADAMS Accession No. ML 121230035).

[74] Ibid 7, Section C.6. Use of Probabilistic Risk Assessment in Operability Decisions

[75] Ibid 70

[76] Revised Response To Task Interface Agreement – Diablo Canyon Seismic Qualification Current Licensing and Design Basis, TIA 2011-010 (TIA 2012-012) (TAC NOS. ME9840 and ME9841), February 14, 2012, ML12297A199

[77] E-Mail and Attachment, from Michael Peck to Geoffrey Miller and et al, Subject: ACT: Diablo Canyon - Recommendation for Regulatory Disposition, Attachments: Diablo Canyon Seismic White Paper.docx, February 3, 2011

[78] Non-Concurrence, NCP-2012-001, Diablo Canyon Power Plant Inspection report 050000275/323-2011005, ML120450843

[79] Letter, from A. Giambusso, Director of Licensing, Atomic Energy Commission (AEC), to F.T. Searls, Pacific Gas and Electric, dated August 13, 1973

[80] 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

[81] Diablo Canyon Power Plant - NRC Integrated Inspection Report 05000275/2012004 and 05000323/2012004, November 13, 2012, ML12318A385

Document 2 – Memo Establishing Panel

September 3, 2013

MEMORANDUM TO: Michael Case - Chair
Britt Hill - Member
Rudolph Bernhard - Member

FROM: Eric J. Leeds, Director ***/RA/***
Office of Nuclear Reactor Regulation

SUBJECT: AD HOC REVIEW PANEL - DIFFERING PROFESSIONAL OPINION INVOLVING SEISMIC ISSUES AT DIABLO CANYON (DPO-2013-002)

In accordance with Management Directive (MD) 10.159, "The NRC Differing Professional Opinions Program," I am appointing you as members of a Differing Professional Opinion (DPO) Ad Hoc Review Panel (DPO Panel) to review a DPO that was forwarded to me to disposition.

The DPO (Enclosure 1) raises concerns on seismic issues at Diablo Canyon.

I have designated Mike Case chairman of this DPO Panel and Britt Hill as a DPO Panel member. Rudolph Bernhard was proposed by the DPO submitter and serves as the third member of the DPO Panel. In accordance with the guidance included in MD 10.159 and consistent with the DPO Program objectives, I task the DPO Panel to do the following:

❑ Review the DPO submittal to determine if sufficient information has been provided to undertake a detailed review of the issue.

❑ Meet with the submitter, as soon as practicable, to ensure that the DPO Panel understands the submitter's concerns and scope of the issues. (Normally within 7 days).

❑ Promptly after the meeting, document the DPO Panel's understanding of the submitter's concerns, provide the Statement of Concerns (SOC) to the submitter, and request that the submitter review and provide comments, if necessary. (Normally within 7 days).

❑ Maintain the scope of the review to not exceed those issues as defined in the original written DPO and confirmed in the SOC.

❑ Consult with me as necessary to discuss schedule-related issues, the need for technical support (if necessary), or the need for administrative support for the DPO Panel's activities.

❑ Perform a detailed review of the issues and conduct any record reviews, interviews, and discussions you deem necessary for a complete, objective, independent, and impartial review. The DPO Panel should re-interview individuals as necessary to clarify information during the review. In particular, the DPO Panel should have periodic discussions with the submitter to provide the submitter the opportunity to further clarify the submitter's views and to facilitate the exchange of information.

- ❑ Provide monthly status updates on your activities via email to Renée Pedersen, Differing Views Program Manager (DVPM) about the last day of the month. This information will be reflected in the Milestones and Timeliness Goals for this DPO. Please provide a copy of email status updates to the submitter and to me.

- ❑ Issue a DPO Panel report, including conclusions and recommendations to me regarding the disposition of the issues presented in the DPO. The report should be a collaborative product and include all DPO Panel members' concurrence. Follow the specific processing instructions for DPO documents.

- ❑ Consult me as soon as you believe that a schedule extension is necessary to disposition the DPO.

- ❑ Recommend whether the DPO submitter should be recognized if the submitter's actions result in significant contributions to the mission of the agency.

Disposition of this DPO should be considered an important and time sensitive activity. The timeliness goal included in the MD for issuing a DPO Decision is 120 calendar days from the day the DPO is accepted for review. The timeliness goal for issuing this DPO Decision is November 29, 2013.

Process Milestones and Timeliness Goals for this DPO are included as Enclosure 2. The timeframes for completing process milestones are identified strictly as <u>goals</u>—a way of working towards reaching the DPO timeliness goal of 120 calendar days. The timeliness goal identified for your DPO task is 70 calendar days.

Although timeliness is an important DPO Program objective, the DPO Program also sets out to ensure that issues receive a thorough and independent review. The overall timeliness goal should be based on the significance and complexity of the issues and the priority of other agency work. Therefore, if you determine that your activity will result in the need for an extension beyond the overall 120-day timeliness goal, please send me an email with the reason for the extension request and a new completion date. I will subsequently forward this request to the DVPM who will forward it to the EDO for approval.

Please ensure that all DPO-related activities are charged to Activity Code ZG0007.

Because this process is not routine, the DVPM will be meeting and communicating with all parties during the process to ensure that everyone understands the process, goals, and responsibilities. The DVPM will be subsequently sending you information intended to aid you in implementing the DPO process.

An important aspect of our internal safety culture includes respect for differing views. As such, you should exercise discretion and treat this matter sensitively. Documents should be distributed on an as-needed basis. In an effort to preserve privacy, minimize the effect on the work unit, and keep the focus on the issues, you should simply refer to the employee as the DPO submitter. Avoid conversations that could be perceived as "hallway talk" on the issue. We need to do everything that we can in order to create an organizational climate that does not chill employees from raising dissenting views.

As a final administrative note, please ensure that all correspondence associated with this case include the DPO number in the subject line, be profiled in accordance with ADAMS template OE-011, be identified as non-public and declared an official agency record *when the correspondence is issued*. Please email the ADAMS accession number for the record to DPOPM.Resource@nrc.gov and the record will be filed in the applicable DPO case file folder (DPO-2013-002) in the ADAMS Main Library. Following this process will ensure that a complete agency record is generated for the disposition of this DPO. If the submitter requests that the documents included in the DPO Case File be made public when the process is complete, you will be provided specific guidance to support a releasability review.

I appreciate your willingness to serve and your dedication to completing an independent and objective review of this DPO. Successful resolution of the issues is important for NRC and its stakeholders. If you have any questions, you may contact me, Trent Wertz, NRR OCWE Champion, or Renée Pedersen, DVPM, at (301) 415-2742 or email Renee.Pedersen@nrc.gov.

I look forward to receiving your independent review results and recommendations.

Enclosures:
1. DPO-2013-002
2. Milestones and Timeliness Goals

cc w/o enclosure: Submitter
DVPM

DPO submitter. Avoid conversations that could be perceived as “hallway talk” on the issue. We need to do everything that we can in order to create an organizational climate that does not chill employees from raising dissenting views.

As a final administrative note, please ensure that all correspondence associated with this case include the DPO number in the subject line, be profiled in accordance with ADAMS template OE-011, be identified as non-public and declared an official agency record *when the correspondence is issued*. Please email the ADAMS accession number for the record to DPOPM.Resource@nrc.gov and the record will be filed in the applicable DPO case file folder (DPO-2013-002) in the ADAMS Main Library. Following this process will ensure that a complete agency record is generated for the disposition of this DPO. If the submitter requests that the documents included in the DPO Case File be made public when the process is complete, you will be provided specific guidance to support a releasability review.

I appreciate your willingness to serve and your dedication to completing an independent and objective review of this DPO. Successful resolution of the issues is important for NRC and its stakeholders. If you have any questions, you may contact me, Trent Wertz, NRR OCWE Champion, or Renée Pedersen, DVPM, at (301) 415-2742 or email Renee.Pedersen@nrc.gov.

I look forward to receiving your independent review results and recommendations.

Enclosures:
1. DPO-2013-002
2. Milestones and Timeliness Goals

cc w/o enclosure: Submitter
DVPM

ADAMS Accession Number: ML13242A305

Office	NRR	NRR
Name	TWertz	ELeeds
Date	09/03/13	09/03/13

OFFICIAL RECORD COPY

Document 3 – Panel Report

Differing Professional Opinion (DPO) Involving Seismic Issues at Diablo Canyon Nuclear Power Plant (DPO-2013-002)

DPO Panel Report

Michael Case, Panel Chair

Brittain Hill, Panel Member

Rudolph Bernhard, Panel Member

DPO Panel Report

1. Introduction

On July 19, 2013, in accordance with Management Directive (MD) 10.159, "The NRC Differing Professional Opinions Program," an individual (the submitter) filed a differing professional opinion (DPO) associated with seismic issues at the Diablo Canyon Power Plant (DPO-2013-002), ADAMS Accession No. ML13268A466). By memorandum dated September 3, 2013, the Director of NRR established an Ad Hoc Review Panel (DPO Panel or Panel) in accordance with MD 10.159. Consistent with DPO program objectives, the Director of NRR directed the DPO Panel to conduct a thorough and independent review of the DPO and to issue a report with its conclusions and recommendations.

The issues raised by the DPO submitter occurred over a period from 2010 to the present but generally focused on the agency's consideration of new seismic information related to potential ground motions from the Shoreline and several other earthquake faults near the Diablo Canyon Power Plant (DCPP). Although the DPO Panel focused its review on this issue, during the course of its review, the Panel needed to understand licensee and staff activities that occurred significantly before and after this timeframe. As an example, as part of their consideration, the Panel needed to research staff activities associated with the initial licensing of Diablo Canyon. The Panel also considered information as current as the agency's response to Fukushima seismic issues. This proved to be an enormous scope of information that ranged across five decades. As appropriate to facilitate understanding of the issue, the DPO Panel has included information it learned about these activities that occurred before and after the specific timeframe associated with the DPO.

2. Background

Diablo Canyon's original seismic evaluations (Design Earthquake [OBE-equivalent for DCPP], and the Double Design Earthquake [SSE-equivalent for DDPP]) were accepted prior to issuing the Unit 1 Construction Permit in 1968. These seismic evaluations were performed under and met the Atomic Energy Commission's requirements at time of the submittal. For simplicity, the level of peak ground motion (i.e., horizontal or vertical acceleration) expected from an earthquake is commonly expressed as a unit of gravitational acceleration (g, or m/s^2). The DE/OBE was accepted as being 0.2 g and was thought to be the largest earthquake that was expected to occur during the lifetime of the plant (a 0.2 g earthquake was estimated to occur once in more than 200 years). The DDE/SSE is simply double the ground motion of the largest expected earthquake (DE/OBE), and is not tied directly to any expected earthquake. The higher ground acceleration of the DDE was used to add safety margin to the evaluations and ensure that safety-related structures, systems, and components needed to safely shut the plant down and maintain it safely would function after the earthquake.

In 1973, the licensee for the DCPP, Pacific Gas and Electric Company (PG&E), became aware of the Hosgri fault, which was discovered offshore from the plant during oil exploration. This fault was previously unknown, and no significant earthquake had previously been attributed to an offshore fault in that area. Based on the timing of this new discovery, the NRC was able to include this information in the approval of operating licenses for DCPP (1984 for Unit 1). As part of this approval, the NRC required PG&E to perform a seismic re-evaluation to include the possible effects of the Hosgri fault using the latest NRC requirements (10 CFR 100 and Regulatory Guide 1.61). The state-of-the-science in seismic evaluation had significantly improved, so the NRC had upgraded its seismic requirements. The NRC obtained assistance in evaluating the Hosgri fault from U.S. Geological Survey (USGS) and other consultants.

When the Hosgri evaluation was completed, the NRC accepted that this fault could possibly produce 0.75 g peak ground acceleration at Diablo Canyon, but such an extreme event was expected to occur once every 2,000 – 25,000 years. Nonetheless, the NRC required PG&E to make substantial plant modifications to be able to withstand 0.75 g and maintain the same level of plant capability as was required under the SSE. The NRC added these site-specific requirements on top of the existing regulatory requirements.

Therefore, DCPP has the following unique licensing aspects:

1. The plant meets NRC's seismic safety requirements through the DE (0.2 g) and DDE (0.4 g) and the Hosgri evaluation (0.75 g).
2. The plant was required and designed to withstand 0.75 g (based on the Hosgri Evaluation) at the same degree of functionality as an SSE.
3. PG&E used two different NRC-approved seismic methodologies that are part of the design and licensing bases for the plant, one for the DE and DDE, and the other for the Hosgri evaluation.
4. The plants were required to have instrumentation installed to cause an automatic reactor trip if onsite seismic sensors register 0.4 g.
5. A license condition was added to require a confirmatory seismic study over the first 10 years of operation using the latest methods to verify that the Hosgri evaluation remained accurate. PG&E completed this one-time action, but has maintained a continuous seismic assessment program, working with USGS and state agencies to maintain state-of-the-science knowledge and further study the region around the plant.
6. PG&E was required to develop a seismic risk assessment.

The operating license included a license condition requiring a confirmatory seismic study over the first 10 years of operation, using the latest methods, to verify that the Hosgri evaluation remained accurate. PG&E completed this one-time action (known as the Long-Term Seismic Program, or LTSP) using both deterministic and probabilistic, state-of-the-science methods. The LTSP evaluations concluded that the reanalyzed ground motions were generally lower than already considered for the Hosgri evaluation, and that slightly higher ground motions at frequencies >15 Hz were well within the safety margins of the plant. The staff extensively reviewed the study and agreed with its conclusions as documented in a Supplemental Safety Evaluation Report (SSER) in 1991.

The first new significant seismic information to be identified after plant licensing was the USGS reassessment of seismic activity and new indications in 2008 that there may be an offshore fault close to the plant. PG&E reported this information to the NRC and performed an intensive study of what became known as the Shoreline fault. Although some of the physical features of an active fault are not present, others are, so PG&E concluded that a fault was present and reported their study results in January 2011. This study also reevaluated potential ground motions from the Hosgri and other nearby faults. PG&E concluded that potential ground motions from the Shoreline and other reanalyzed faults were lower than previously considered in the LTSP. Therefore, PG&E believed the plant was safe and no further action was needed.

The NRC, with contractor support, evaluated potential ground motions from the Shoreline fault based on the licensee's data and independent analyses. The NRC's results for a number of possible cases showed similar but slightly higher results based on some added conservatisms. Based on these detailed reviews, the NRC issued Research Information Letter 2012-01 to report the evaluation results. The associated cover letter documented the NRC's conclusions that the Shoreline fault report should be treated as a lesser included case under the NRC-approved Hosgri evaluation because the assumptions and calculations appropriately correlated to those used in the Hosgri evaluation. Because the Shoreline results were lower than the Hosgri evaluation (HE) results, this action resolved the inspection question about which set of requirements (DE/DDE, or HE) should be used to assess the safety impact to the plant and the impact to plant safety.

2.1. Use of Seismic Ground Motions in Safety Analyses

In simplest terms, earthquakes create waves of energy that travel through the Earth and produce vibratory ground motions at the surface. Like ocean waves, seismic waves can be reflected or refracted as the move away from the source, and can be amplified or muted as they approach the Earth's surface. Seismic ground motions commonly are represented as response spectra, which plot the spectral frequency of vibration against the expected level of ground acceleration (Figure 1).

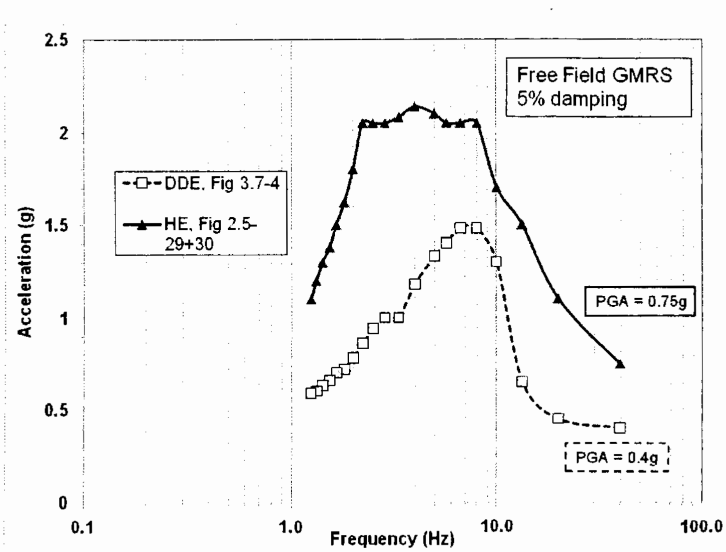

Figure 1. Characteristic response spectra for Diablo Canyon NPP, from FSARU figures indicated in legend.

This relationship between the frequency of a seismic wave (or ground motion) and level of ground acceleration is important. Different frequencies of ground motion have different effects on different structures, systems, and components. Ground motions from different earthquakes often are compared by their "peak ground accelerations," which is the level of acceleration that occurs with spectral frequencies of 100 Hz. Even though we refer to the DDE has having a "peak ground acceleration" of 0.4 g, Figure 1 shows that much larger ground accelerations are considered for spectral frequencies lower than 20 Hz. Many important structures, systems and components at nuclear power plants are sensitive to these lower frequency accelerations.

For most nuclear power plants, the Safe Shutdown Earthquake (SSE) is represented by a single response spectrum, which represents the maximum vibratory ground motion expected for a site. A typical response spectrum (like those in Figure 1) is developed for the free-field surface response, meaning that the calculations assume there are no engineered structures resting on the surface.

The Diablo Canyon NPP was licensed before the SSE concept was established in regulations. Instead of a single SSE, seismic analyses for Diablo evaluated the maximum earthquake potential with two distinct earthquakes: the double design earthquake (DDE) and Hosgri earthquake (HE). The DDE is represented by a free-field response spectrum that was developed using standard methods. However, the HE used several different methods in order to account appropriately for the physical characteristics of larger ground motions. Some of the HE analyses developed free-field response spectra (as shown in Figure 1), whereas other HE analyses accounted for the effects of large structures being present at the site. In addition, the HE analyses also accounted for some non-linear effects in the near-surface site response, which was not a significant consideration for the DDE analyses.

In addition to the different analytical methods for the DDE and HE, different assumptions can be made about the amount of energy lost by the ground vibrations due to friction and heating (e.g., NRC Regulatory Guide 1.61). This energy loss is referred to as "damping," with higher damping values representing larger energy losses (i.e., lower magnitude accelerations). As shown in Regulatory Guide 1.61, larger damping values are acceptable for SSE analyses compared to Operating Basis Earthquake analyses. These higher damping values reflect the larger energy losses expected in larger magnitude ground motions. Typically, a damping value of 5% is used as a reference value, although higher or lower damping values can be used in different safety analyses.

For the Diablo Canyon NPP, higher damping values were used in HE evaluations than for most DDE evaluations. It is important to note that the HE values are consistent with the values recommended for the SSE in Regulatory Guide 1.61. The DDE, which is viewed as equivalent to the SSE, used damping values that generally were lower than allowable for the SSE. In other words, many of the DDE analyses were conservative and allowed for more efficient energy transfer through structures, systems, and components than normally would be assumed in SSE analyses.

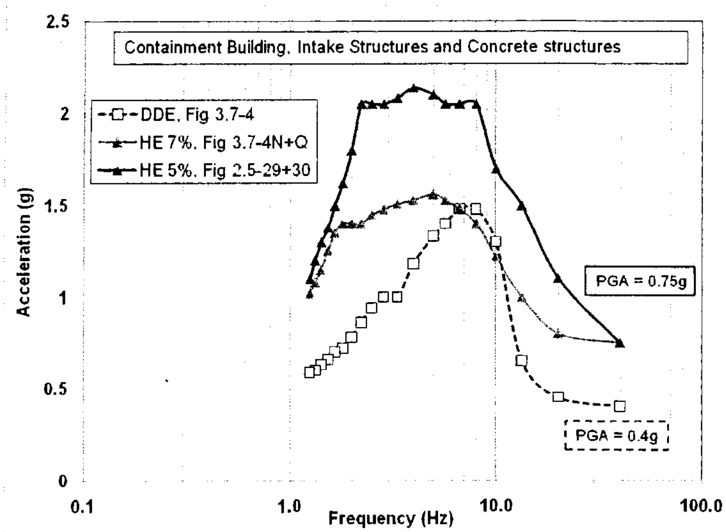

Figure 2. Response spectra for Diablo Canyon NPP containment building and other sturctures, from FSARU figures indicated in legend.

Figure 2 illustrates the importance of these differences in analytical methods and assumptions. Analyses of seismic loads on the Diablo Canyon containment building used 5% damped, *free-field* ground motions for the DDE. However, the HE analyses used 7% damped, *foundation-filtered* ground motions, which are approximately 10% lower than the DDE ground motions for 7–10 Hz frequencies. Load calculations on the containment building are sensitive to this frequency range. Consequently, these different methods and assumptions result in the DDE creating higher calculated loads on the containment building than the HE. For some other important structures, systems, and components, the DDE also represents higher calculated loads than the HE (e.g., reactor coolant pressure boundary components, FSARU section 5.2.1.15).

Because of the complex development of the Diablo Canyon NPP ground motion analyses, there is no single response spectrum that appropriately represents the level of seismic ground motion that was used in the safety analyses. In order to accurately compare ground motions from new information with the current licensing basis, the ground motions in question need to have:

1) Comparable response surfaces (i.e., free-field versus foundation filtered);

2) Comparable approach to modeling nonlinear effects, if any, and;

3) Comparable damping values, which correspond to the damping used in the specific safety analyses.

NRC reviews of the SAR, LTSP, and Shoreline report, along with additional discussions in IPEEE and GI-199 evaluations, clearly show that different analytical methods and assumptions can be used acceptably to derive appropriate response spectra. Regardless of analytical approach used, ground motions in the current licensing basis for the Diablo Canyon NPP have potentially significant differences in surface loading, nonlinearity, and damping that must be recognized to compare modeling results accurately.

3. Statements of Concerns

On October 23, 2013, the DPO Panel met for the first time with the submitter to discuss his DPO submittal and his perspective on the concerns. Prior to the meeting, the DPO Panel reviewed the DPO submittal and identified seven areas that looked like potential concerns. The Panel provided this to the submitter for his consideration. The submitter narrowed his concerns to the following:

- The NRC did not enforce the Diablo Canyon Technical Specifications with respect to this seismic issue, because the new seismic information showed that SSCs could be exposed to greater vibratory motion than previously considered for the SSE

- PG&E's operability evaluation following the development of the new seismic information was inadequate, because the new seismic information was not compared correctly to the plant's licensing basis.

- The NRC failed to enforce 10 CFR 50.59 requirements that PG&E obtain an amendment to their license, because the new seismic information showed that more than a minimal increase would occur in the likelihood of SSC malfunction.

- The NRC failed to adequately address the Los Osos and San Luis Bay faults, which could produce ground motions in excess of the SSE ground motion.

The full statement of DPO concerns is included as Appendix A and was used by the Panel to focus its activity.

4. Evaluation

In support of its independent evaluation of DPO-2013-002, the DPO Panel met and communicated with the DPO submitter initially and throughout the DPO process to obtain his perspectives on the concerns as well as to discuss with him the status and results of its review. The Panel also reviewed the documents, records, and references cited throughout the DPO report and listed in Appendix B, "Records and Documents Reviewed by the DPO Review Panel." The Panel also interviewed other individuals related to DPO issues to obtain additional background information and the processes that were (or are being) followed by the licensee and the staff to address the issue. The Panel members invested a considerable amount of time reading the extensive record associated with this issue. Finally, the Panel members met among themselves to plan their work, to review the issues, and to document their conclusions and recommendations.

4.1. Factors Framing the Evaluation

In order to complete its decisions of the DPO concerns, the Panel needed to weigh and place into a contextual framework a number of issues that relate to the DPO. The Panel sought to develop a basis for a decision on the DPO concerns and not a detailed argument for who is right. Safety concerns were the overriding factor, although other factors contribute significantly to understanding the DPO concerns. These factors and the Panel's underlying understanding of these factors are explained below.

4.1.1. Treatment of New Siting-related Information

As part of completion of the license condition for the Long-Term Seismic Program, PG&E committed to "maintain a strong geosciences and engineering staff to keep abreast of new geological, seismic, and seismic engineering information and evaluate it with respect to its significance to Diablo Canyon,..." (NRC, 1991, p. 2-49). The NRC did not specify exactly how PG&E would evaluate new information. Nevertheless, there was a clear expectation that new seismic information would continue to be evaluated for significance, without the need for the NRC to take additional regulatory actions to initiate such evaluations.

The NRC expects licensees to evaluate new information that has the potential to affect the licensing basis of the plant, based on the applicable regulatory requirement (e.g., operability determinations, 10 CFR 50.59 evaluations, corrective actions under Criterion XVI of 10 CFR 50, Appendix B and other quality assurance programs). However, guidance that specifically addresses this issue could be improved, as discussed in the conclusions.

4.1.2. Unique Diablo Canyon Seismic Design Basis

The seismic design basis for Diablo Canyon is both the Double Design Earthquake and Hosgri Evaluation. Throughout the FSARU, both the Double Design and Hosgri earthquakes are used to design and qualify SSCs that are important to safety. This basis has been well established from the time of the operating license, through the LTSP evaluation until the current time. Nevertheless, applicable regulations and review guidance are designed to evaluate a single seismic design basis.

4.1.3. Ambiguity in the FSARU

For a variety of reasons (such its role of documenting historical information, writing style, complexity of the seismic design basis, lack of guidance on new information), the FSARU is not always as clear as it could be with respect to the seismic licensing basis and how we use that basis to evaluate new seismic information. For example, one FSARU section implies that only certain SSCs were designated to withstand the Hosgri earthquake. In the Prompt Operability Assessment (POA), the licensee clarified that all seismic Category I SSCs were evaluated for the 1977 HE. Consequently, a reasonable person could easily draw different meanings from the seismic information in the FSARU.

4.1.4. Risk Insights

Despite the complexity of the licensing issues, from a risk and safety perspective, the Diablo Canyon NPP is seismically robust. The Diablo Canyon NPP is relatively well-studied from a seismic risk perspective. In 1979, the staff evaluated seismic risk associated with the Diablo Canyon NPP without the "Hosgri fix" and estimated the likelihood of core damage from seismic events to be approximately one chance per 22,000 years. In the 1991 LTSP, a more extensive evaluation was conducted by PG&E and reviewed by the staff. This review included consideration of both the Los Osos and San Luis Bay seismic sources and estimated the core damage frequency from seismic events to be approximately one chance per 27,000 years. These seismic core-damage results are comparable to other nuclear power plants that were evaluated in the Long-term Seismic Program and as part of the Individual Plant Examination of External Events program in the early 1990's (see NUREG-1742).

4.2. Evaluation of Specific DPO Concerns

Concern #1 - The NRC has not enforced Diablo Canyon Technical Specification requirements that key plant safety equipment remain operable during reactor operation. New seismic information developed by Pacific Gas and Electric concluded that Technical Specification required Structures, Systems and Components (SSCs) can be exposed to greater vibratory motion than was used to qualified this equipment for the facility safe shutdown earthquake (SSE) design basis. For Technical Specification required SSCs to be considered operable, the licensee is required to demonstrate a reasonable assurance that this plant equipment would still be capable of performing the safety functions in accordance with the plant design bases and safety analysis.

The Panel believes that the NRC has properly evaluated the licensee's determination of operability as presented in Prompt Operability Assessment (POA) of October 21, 2011, and as guided by NRC Inspection Manual Part 9900. The requirement for this concern is driven by the plant's Technical Specifications which, in many cases, prescribe direct surveillance requirements. In this specific circumstance, there is not a specific surveillance requirement to demonstrate SSC operability for seismic issues. So the situation of new seismic information on SSCs is assessed against the definition of OPERABLE contained in the facility's Technical Specifications. The definition of operability states:

> A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety functions, and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication and other auxiliary equipment that are required for the

system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support functions(s).

This definition prescribes the "requirement" for this particular concern and is basically silent on how to accomplish this evaluation. The NRC and its licensees have a long history of precedents in this area and have also developed guidance for this determination (i.e., IMC Part 9900). Neither, however, is a requirement unto itself.

The Panel examined the licensee POA update of October 21, 2011, and believes it to be technically credible and procedurally consistent with IMC Part 9900 guidance. The underlying licensee logic was to compare the ground motions from the new information to previous ground motions where SSC performance has been shown to be adequate. The licensee examined the effect of new information on the DE, DDE, and HE and used insights from the LTSP evaluation. The DPO submitter had advocated this comparative approach. Although this may not have initially been the case, the updated POA did take all three earthquake levels into account. We agree with the DPO Submitter's approach to the POA, in that examination of all three earthquake levels is appropriate for and consistent with the seismic licensing basis for the plant. The Panel's evaluation of the technical approach to the operability issue is contained in Section 4.2.1.

In its evaluation of the DDE, the licensee recognized that it was inappropriate to analyze the new ground motions with the "old" DDE calculation methodology. The licensee reassesses the DDE performance using an alternate evaluation methodology, which appears consistent with past approaches used in licensing (e.g., DDE versus HE, LTSP methods). In addition, IMC Part 9900 allows the use of alternate evaluation methodologies in Appendix C.4. The NRC found the use of an alternate methodology to be acceptable in the Shoreline analyses, as alternative approaches were used previously in the FSARU and LTSP to analyze potential ground motions. The Panel believes that the use of an alternate methodology is technically acceptable and consistent with the NRC operability guidance.

As discussed in section 4.2.1, in March 2014, PG&E developed additional information to allow direct comparison of the ground motions in the 2011 Shoreline report to those used in the FSARU to design and license the plant. This information confirmed the conclusions of the POA. The POA also recognized that the use of alternate methodologies is only acceptable for operability and not for full compliance with the CLB. This issue is being tracked as a corrective action to close the POA.

Ultimately, the Panel believes that the licensee's expected response to the Fukushima 2.1 seismic issue should provide the appropriate framework for evaluating the potential significance of new seismic information. Re-evaluated ground motions need to be placed into an integrated context with all other seismic and safety information relevant to Diablo Canyon. The Fukushima 2.1 response activity is expected to do that. Although Diablo Canyon has the advantage of a detailed post-licensing evaluation of SSC seismic performance from the LTSP, the methods used in the LTSP are not always current. If the reevaluated seismic hazard for Diablo Canyon turns out to be greater than the plant's design basis, a seismic risk assessment (if warranted) should provide an up-to-date analysis of how SSCs are expected to function at a potentially higher level of seismic hazard.

4.2.1. Technical Assessment of the Potential for Seismic Loads on SSCs to Exceed Previously Analyzed Conditions

The crux of the DPO submitter's concern focuses on a potentially important safety consideration: do new ground motions in the 2011 Shoreline report (PG&E, 2011) exceed the levels of ground motion considered in the FSARU for design and qualification of Category 1 SSCs? The DPO submitter asserts that this exceedance occurs, and that the licensee and NRC should have taken additional actions to ensure SSC operability. However, if the new ground motions were actually lower than those already used in the FSARU to design and license the plant, then further assessments would not be warranted.

For this concern, the Panel determined that the evaluations to-date may not have fully considered the potential significance of the new ground motions on the existing FSARU licensing basis for Diablo Canyon. The DPO submitter identifies some shortcomings in previous evaluations, but also makes incorrect comparisons between the new information and information in the FSARU to reach a conclusion about operability and appropriate licensee and NRC actions. The incorrect comparisons appeared to have occurred, however, because PG&E provided insufficient information in the 2011 Shoreline report to appropriately compare the new ground motions to the range of ground motions actually used in the seismic analyses described in the FSARU. Thus, additional information was needed by NRC, PG&E, and the DPO submitter to make correct comparisons.

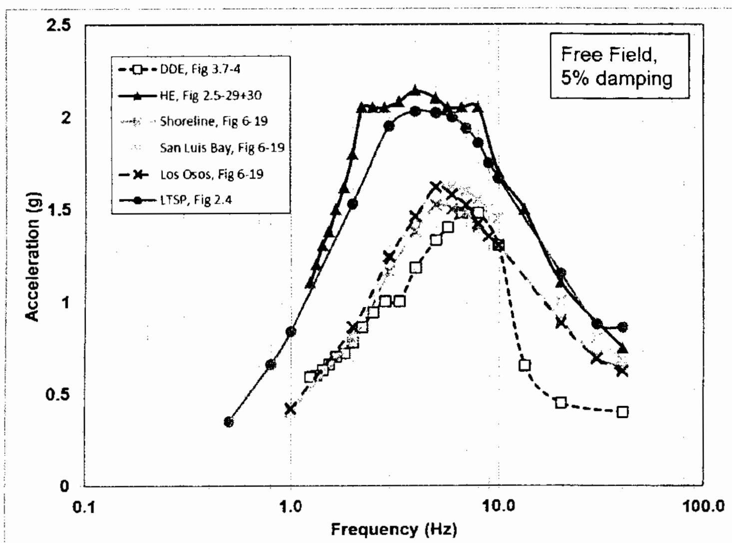

Figure 3 Ground motions from indicated figures in FSARU (DDE, HE), Long-term seismic program (LTSP), and 2011 PG&E Shoreline report (ergodic method).

In the 2011 Shoreline report, PG&E develops new ground motions for the Shoreline, Hosgri, San Luis Bay, and Los Osos faults and compares these ground motions to the original Hosgri evaluation and the Long-Term Seismic Program. This comparison only evaluates results for a standard reference condition, which is a free-field ground motion with 5% damping (Figure 3). As shown in Figure 3, if the original Hosgri evaluation represented the largest ground motions actually considered in the design and qualification of SSCs, then the newer ground motions would clearly be lower than already considered in the original FSARU licensing basis. However, as discussed in the background information (see Figure 2), some SSC analyses used ground motions for the Hosgri evaluation that were effectively lower than DDE ground motions. This situation occurs because most safety analyses were done with ground motions that were different than the 5% damped free-field reference condition.

Returning to the containment building example from Figure 2, seismic loads for the Hosgri evaluation were represented by a 7% damped, foundation-filtered ground motion. In contrast, DDE seismic loads in these analyses were calculated with a 5% damped, free-field ground motion. Thus, the DDE created the highest accelerations (i.e., structural loads) at spectral frequencies of 7 to 11 Hz. If these DDE and HE ground motions were inappropriately compared to the new ground motions shown in the 2011 Shoreline report (Figure 4), it would appear that the new ground motions might exceed the levels previously considered in the FSARU for frequencies of 5 to 30 Hz. Nevertheless, the assumed validity of this incorrect comparison is a fundamental assumption for the logic in the DPO submittal regarding the need for additional actions by both PG&E and NRC.

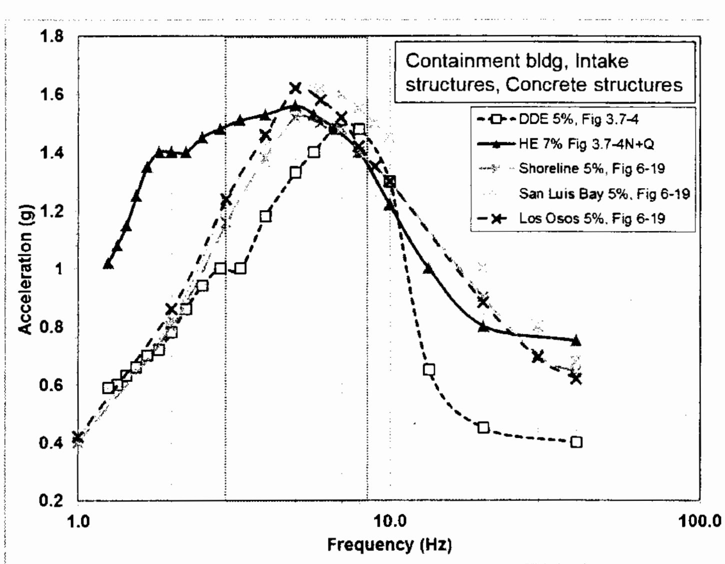

Figure 4. Ground motions from FSARU and 2011 Shoreline Report figures (ergodic method), incorrectly assuming the new ground motions are directly comparable to FSARU inputs for containment building analyses.

During the review of the DPO submittal, the Panel determined that this type of comparison incorrectly assumes the 5% damped free-field ground motions appropriately represent potential ground motions from the Shoreline, Los Osos, and San Luis Bay faults. In order to make an appropriate comparison to the ground motions used to design and license the plant, additional information was needed from PG&E. This information would need to consider if other levels of damping should be used for the new ground motions, such as those corresponding to a Safe Shutdown Earthquake in Regulatory Guide 1.61. PG&E also would need to consider if other potentially significant effects, such as the presence of building foundations or non-linear material responses (e.g., FSARU rev 21, section 2.5.3.10), should be considered for the new ground motions. These considerations are not expressed in the 2011 Shoreline report, or in staff's previous evaluations of the issues surrounding the Shoreline Fault ground motions, or in the DPO submittal.

On 19 December 2013, Panel members discussed this issue of ground-motion comparability with PG&E staff, and outlined the need to compare the new ground motions with the ground motions actually used in the FSARU analyses for design and qualification of safety-related SSCs. PG&E agreed to conduct additional analyses of the new ground motions, so that the results of these analyses would be directly comparable to the inputs used in the FSARU analyses rather than an alternative metric such as the LTSP.

On 5 March 2014, Panel members reviewed additional calculations that were developed by PG&E to allow for direct comparison of potential ground-motions in the 2011 Shoreline report to the ground motions used in the FSARU analyses. PG&E calculated in-structure acceleration response spectra as the basis for comparison, as these spectra already were available for the DDE and HE from FSARU section 3.7 analyses.

To convert the 2011 Shoreline, Los Osos, and San Luis Bay ground-motion spectra to in-structure acceleration response spectra, PG&E developed a scaling relationship from the LTSP analyses that compares the calculated free-surface ground motion to an in-structure response spectrum. This scaling relationship accounts for the effects of processes such as soil-structure interaction and the presence of building foundations. PG&E applied this scaling factor to the 2011 Shoreline, Los Osos, and San Luis Bay ground-motion spectra to calculate in-structure response spectra for 5% damping. PG&E used both ergodic and single-station ground motions from the 2011 Shoreline report.

To account for the different damping values used to analyze the seismic performance of different SSCs (i.e., FSARU rev 21, section 3.7.1.3), PG&E used analytical methods in Pacific Earthquake Engineering Research Center report 2012/01 (Spectral Damping Scaling Factors for Shallow Crustal Earthquakes in Active Tectonic Regions) to develop scaling factors. PG&E applied these scaling factors to the 5% damped in-structure response spectra for the Shoreline, Los Osos, and San Luis Bay faults (SLS), to develop response spectra for the different damping values shown in Table 1. Although most of the damping values used for these faults correspond to SSE values in NRC Regulatory Guide 1.61, PG&E used slightly lower damping values (i.e., more conservative) in several analyses.

Table 1. Scaling factors used in March 2014 PG&E analyses.

Type of SSC	Percentage Damping		
	DDE	HE	SLS
Containment structures	5	7	7
Welded structural steel assemblies	1	4	4
Bolted or riveted steel assemblies	2	7	7
Mechanical components	2	4	3
Vital piping systems (except RCL) >12"	0.5	3	3
Vital piping systems (except RCL) <12"	0.5	2	2
Reactor Coolant Loop	1	4	3
Steam Generators	4	4	3
Integrated Head Assembly	6.85	6.85	6.85
Control Rod Drive Mechanisms	5	5	5

For each of the 10 classes of SSCs (i.e., FSARU rev 21, section 3.7.1.3), PG&E first plotted frequency versus acceleration response for the highest values from either the DDE or HE analyses. PG&E then compared the appropriately scaled Shoreline, Los Osos, and San Luis Bay in-structure response spectra to the DDE+HE spectrum. These comparisons used both the ergodic and single station results from the 2011 Shoreline report.

The in-structure response spectra for the reanalyzed Shoreline, Los Osos, and San Luis Bay faults were all lower than the DDE+HE response spectrum, for both ergodic and single-station results at spectral frequencies of <30 Hz. For several SSCs, the ergodic response spectra met or slightly (<10%) exceeded the DDE+HE spectrum at spectral frequencies of 30–50 Hz. This small high-frequency exceedance would not be expected to significantly affect the performance of these types of SSCs. In addition, most of the slight exceedances occurred for SSCs that PG&E had selected a conservative damping value (i.e., lower than used for HE analyses). All of the reanalyzed single-station response spectra were lower than the DDE+HE response spectrum.

In summary, PG&E reanalyzed the ground motions from the 2011 Shoreline report using the same assumptions as in the FSARU for damping level and foundation filtering. The reanalysis allows for direct comparison of the in-structure responses from potential earthquakes on the Shoreline, Los Osos, and San Luis Bay faults to the in-structure responses that were used to design and license the plant. Nearly all the reanalyzed in-structure response are lower than those used to design and license the plant, with the exception of slight (<10%) exceedances at 30–50 Hz spectral frequencies for several SSCs using ergodic analyses. These slight exceedances arise, in large part, from conservative damping values used by PG&E and are not judged significant for the SSCs being considered. The Panel concludes that these comparisons are appropriate, and that potential ground motions from faults characterized in the 2011 Shoreline report do not exceed the levels of in-structure acceleration already considered in the design and licensing of the plant.

4.2.2. Summary of Concern #1

To summarize the Panel's assessment of Concern #1, the DPO raised an important issue that highlights the complexity of information used to assess the seismic loads on safety related SSCs during the licensing and construction of Diablo Canyon NPP. Nevertheless, the DPO inappropriately compares different types of ground motions to incorrectly conclude that SSC functionality should be re-assessed, and asserts that NRC staff did not respond appropriately to new information. This mis-comparison appears understandable, as appropriate ground-motion information was not available to NRC, PG&E, or the DPO submitter to make a correct comparison.

Previously, NRC and PG&E staffs reached an apparently reasonable conclusion that the new ground motions were bound by existing ground motions (i.e., the Hosgri evaluation). Thus, no further analyses appeared warranted, and staff's approach on additional licensing or enforcement actions appears defensible. Based on the Panel's current understanding, this conclusion only appears supportable when all the ground motions are compared to a common reference condition of 5% damping, free-field response. However, most of the Diablo Canyon safety analyses were not conducted at this reference condition. The FSARU analyses used two different ground motions, each of which used different damping values and, at times, different analytical assumptions, which do not always correspond to the common reference condition used in the 2011 Shoreline report. As a result, only a few of the ground motions in the 2011 Shoreline report are directly comparable to the actual ground motions used in the FSARU safety analyses.

In the previous analyses, neither PG&E nor NRC staff, nor the DPO submitter, appeared to recognize the need to compare the new information more clearly to the licensing basis in the FSARU. The need for this comparison apparently was not identified because of the complex differences between the reference ground motion conditions and the range of conditions actually considered in the FSARU analyses. Nevertheless, the DPO submitter succeeded in raising awareness of these important differences, and illustrating how seemingly reasonable interpretations resulted in different implications for operability and safety. As discussed extensively in section 4.2.1, the Panel concludes that once appropriate comparisons are made, potential ground motions from faults characterized in the 2011 Shoreline report do not exceed the levels of in-structure acceleration already considered in the design and licensing of the Diablo Canyon Nuclear Generating Station.

Concern #2 – Pacific Gas and Electric's operability evaluation following development of the new seismic information was inadequate. Comparison of the new seismic information only against the Hosgri Event (HE) and Long Term Seismic Program (LTSP) ground motions was not adequate to demonstrate Technical Specification required SSCs were operable. Neither the HE not the LTSP methods were approved to be used in SSE safety analysis. The HE and LTSP methods over-predicted SSC performance when

compared to the SSE design basis methods. Even though the HE and LTSP include higher ground motions, neither of these methods were bounding for plant Technical Specification SSCs seismic qualification. Use of the HE and LTSP ground motions failed to demonstrate that the requirements of the American Society of Mechanical Engineers' (ASME) Boiler and Pressure Vessel Code acceptance limits would be met at the higher ground motions. 10 CFR 50.55a required that ASME acceptance limits be met for plant safety Class 1 and 2 following an SSE. Demonstration that the ASME acceptance limits are met provides assurance that the integrity of key plant systems, including the reactor coolant pressure boundary would be maintained following the higher seismic stress levels represented by the new seismic information.

The Panel does not believe that Concern #2 raises new fundamental issues with respect to the seismic safety issue that is not already discussed in the Panel's consideration of Concern #1. However, the concern raises some considerations on evaluation methods and the ASME Code that the Panel addressed below.

During plant operation, conditions or equipment changes that are outside what is considered normal can occur. For failures associated with the Technical Specification's requirements, specific testing to determine equipment operability is often provided and Action Statements are used for the timing of actions due to the condition under construction. For conditions that are not as well defined, equipment inoperability is determined to exist at the time there is sufficient evidence that the equipment is not capable of meeting its design basis function.

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.

Concern #2 suggests that there is only one appropriate evaluation method in this case, which is to substitute new seismic information into the original DDE method. In the Panel's estimation, there were three viable evaluation methods to assess seismic performance of plant equipment in the DPO scenario. The first would be to directly substitute the new information into the calculation construct of the HE and DDE. Although this method would provide the most direct comparison to the FSAR commitments, it would offer very little insight as to how the SSCs would actually perform to seismic loads shown in the new information. This is because the older analytical techniques are overly conservative. Even by 1981 when the staff issued its SSER supplement, the staff allowed the use of more modern insights (e.g., damping values) because the use of these more conservative DDE values was no longer technically justified.

A second evaluation method available to the licensee would be to use completely up-to-date probabilistic methods. This approach would be similar to the approach used for Fukushima 2.1. Although this evaluation method would be the most technically credible, it would take a considerable amount of time to complete. Such an approach would not have been responsive enough for the purpose of an operability evaluation.

The final possible evaluation method is the one used by the licensee. This evaluation method involved comparison against the HE and the LTSP. This evaluation method is attractive because the methods used in the LTSP are improved over those of initial licensing. The LTSP was extensively reviewed by the staff and provides an additional regulatory perspective that the staff agreed with the licensee's conclusions in that report that adequate seismic margin is provided in the Diablo Canyon design of SSCs. The NRC staff also thought the LTSP evaluation bound the seismic hazards for the Diablo Canyon NPP. The shortcoming of this evaluation method is that it does not compare directly with parts of the FSARU licensing basis. The Panel further reviewed this issue as detailed in Concern #1.

As discussed earlier, there is no regulatory requirement known to the Panel that dictates that the only option for evaluating new information is to substitute it into the original licensing basis calculations. Further, the staff's operability guidance specifically allows the use of alternate evaluation methods. Inspection Manual Chapter 0326 provides some insight on operability and functionality in section 03.09 on Reasonable Expectation when it writes:

> The discovery of a degraded or nonconforming condition may call the operability of one or more SSCs into question. A subsequent determination of operability should be based on the licensee's "reasonable expectation," from the evidence collected, that the SSCs are operable and that the operability determination will support that expectation. Reasonable expectation does not mean absolute assurance that the SSCs are operable. The SSCs may be considered operable when there is evidence that the possibility of failure of an SSC has increased, but not to the point of eroding confidence in the reasonable expectation that the SSC remains operable. The supporting basis for the reasonable expectation of SSC operability should provide a high degree of confidence that the SSCs remain operable.

The Panel believes that the licensee's method of evaluation meets this standard.

The Panel also sought pertinent guidance to help it understand the potential weakness in the licensee's evaluation approach (i.e. incomplete mapping to the FSARU methods). Guidance to the staff on the balance between safety and compliance during evaluation of plant operations is covered in the Inspection Manual Technical Guidance section. It also indicates that discretion can be exercised in cases where conditions do not pose undue risk. The guidance states, in part:

> The NRC has the authority to exercise discretion to permit continued operations—despite the existence of a noncompliance—where the noncompliance is not significant from a risk perspective and does not, in the particular circumstances, pose an undue risk to public health and safety. When non-compliances occur, the NRC must evaluate the degree of risk posed by that non-compliance to determine if specific immediate action is required. Where needed to ensure adequate protection of public health and safety, the NRC may demand immediate licensee action, up to and including a shutdown or cessation of licensed activities. In addition, in determining the appropriate action to be taken, the NRC must evaluate the non-compliance both in terms of its direct safety and regulatory significance... Based on the NRC's evaluation, the appropriate action could include refraining from taking any action, taking specific enforcement action, issuing orders, or providing input to other regulatory actions or assessments, such as increased oversight (e.g., increased inspection).
>
> Where requirements exist that the NRC concludes have no safety benefit, the NRC can and should take action, as appropriate, to modify or remove such requirements from the

regulations or licenses. Requirements that are duplicative, unnecessary, or unnecessarily burdensome can actually have a negative safety impact. They also can tend to create an inappropriate NRC and licensee focus on "safety versus compliance" debates. As the Commission states in its principles of Good Regulation, "There should be a clear nexus between regulations and agency goals and objectives, whether explicitly or implicitly stated."

The Panel believes that by linking the evaluation to the LTSP, the licensee established an important insight to a well-studied (by the staff) seismic risk assessment. The seismic risk of core damage in that study was relatively low (3.7×10^{-5}/reactor-year). This level of risk is well below a level that would indicate an immediate safety concern as discussed in LIC-504, "Integrated Risk-Informed Decision Making Process for Emergent Issues." In addition, the letter from the NRC to the licensee on the results of its review of the new seismic information and the staff's 50.54(f) letter on Fukushima 2.1 provide an adequate regulatory footprint to follow up on potential FSARU non-compliances.

Finally, Concern #2 raises issues with respect to 10 CFR 50.55a. The Panel sees no unique issues with PG&E's operability evaluation with respect to 50.55a issues that were not more fully explored in relation to Concern #1. Therefore, the Panel concludes that the associated operability assessment was adequate. 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. Nevertheless, the relatively low level of damping used in the DDE analyses (e.g., Table 1) results in the DDE creating the limiting load for these SSCs, which is not exceeded by the reanalyzed ground motions from the 2011 Shoreline report (see discussion in Concern #1).

The new information by itself did not alter the FSARU approach to maintain both the DDE and HE as faulted conditions with respect seismic component and piping analysis. The Panel's evaluation of Concern #1 concluded that the new information is bounded by the existing DDE and Hosgri evaluations. So the current FSARU conclusions with respect to the ASME acceptance criteria appear valid. In this particular section, the Panel could not see an adequate technical justification to use the new information as a more limiting requirement than those previously identified.

In summary, the Panel believes that in the context of Concern #2, the method used by the licensee was appropriate, with accepted assumptions, to verify that the new information did not indicate the presence of a hazard that would constitute an undue risk to public health and safety. Showing the new postulated hazard is less limiting than already evaluated hazards, using accepted methodology, was one acceptable method for performing the operability evaluation.

Concern #3 – The NRC has failed to enforce the 10 CFR 50.59 requirements that Pacific Gas and Electric obtain an amendment to the Diablo Canyon Operating License prior to incorporating the Shoreline scenario into the FSARU. A license amendment was required because the change resulted in more than a minimal increase in the likelihood of a malfunction of SSCs important to safety than previously evaluated in the FSARU. A license amendment was also required because this change represents a departure from the FSARU method of evaluation used to establish the seismic SSE design basis. The NRC conclusion that a "reasonable assurance of safety" existed was not an adequate

basis to conclude an amendment to the Diablo Canyon Operating License was not required.

The DPO Panel believes that the staff did not fail to enforce 10 CFR 50.59 requirements with respect to a proposed update to FSARU Revision 20, in which PG&E was requested to add information concerning the 2011 Shoreline Fault Report. Analysis of this concern requires an understanding of the context of proposed FSARU change itself, the context of where it was placed, and the context of how it was written.

First, this update to FSARU Revision 20 was specifically requested by the staff in a letter documenting the staff's review of the Shoreline Fault. Specifically the staff's letter states:

> Therefore, the staff has concluded that the Shoreline scenario should be considered 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).

This created a problematic situation for the licensee because NRC guidelines for FSAR updates suggest that an update might not be warranted. The guidance states that for analyses of new safety issues, the evaluations must be reflected in the FSAR only if, on the basis of the results of the requested analyses or evaluation, the licensee determines that the existing design basis, safety analyses or FSAR description are either not accurate or not binding or both. Nevertheless, the Panel believes that the FSAR update was appropriate because of the long and at times complex evolution of seismic information for Diablo Canyon. However, the change was likely not required at all, let alone, something that required a license amendment.

The second contextual factor concerns where the updated information was placed. The FSAR update (Revision 21) was placed in the section of the FSAR (Section 2.5) that discusses geology and seismology. The context of the information is that it factually presents results of seismic and geological information about the site, and provides additional explanations of the historical development of the seismic hazards analyses for Diablo Canyon NPP. The FSARU Revision 21 information on the Shoreline Fault zone now discusses the results of both the PG&E and NRC evaluations. The information presented focuses on conclusions from several seismic and geological investigations, which generated little controversy in the DPO submittal. However, the update did not embellish the description with respect to how the conclusions are used in seismic design, which is an area of DPO controversy. A plain reading of FSAR Revision 21 would indicate that the update has little or no direct 50.59 implications.

Finally, as the DPO submitter suggests, it may be appropriate to consider the implications of the use of this new information with respect to 50.59. As noted earlier, the writing style of FSAR Revision 21 in this section is factual and historical. Although it did include a reference to the NRC letter on the review of the Shoreline Fault, the FSAR update did not include important contextual information from the NRC letter. The first piece of contextual information is that the NRC evaluation was preliminary. Second, PG&E is required to take action consistent with the staff's 10 CFR 50.54(f) letter on Fukushima seismic issues and that "changes to the licensing basis may be appropriate to capture the information developed in response" to the Fukushima seismic issue. Finally, after the Fukushima seismic letter was issued, PG&E committed to providing NRC with an interim evaluation if new information is uncovered that would suggest the Shoreline fault is more capable than currently believed (PG&E, 2012, ADAMS Accession No. ML12300A105). Any such interim evaluation would occur before completion of the evaluations requested in the Fukushima seismic letter. Although the Panel believes that the information requested in the Fukushima seismic letter should provide a comprehensive basis to evaluate potential seismic hazards at the Diablo Canyon NPP, incorporating current information on the Shoreline fault into FSARU Revision 21 appears reasonable given PG&E and NRC communications.

Given that contextual information, the DPO Panel assessed the 50.59 evaluation criteria as described in the staff-endorsed NEI report 96-07. The guidance suggests that changes in design requirements for earthquakes should be best treated as potentially affecting the likelihood of a malfunction rather than frequency of occurrence of an accident. Based on the documented information, both PG&E and NRC had concluded that the ground motions from the 2011 Shoreline report were bounded by existing analyses. Thus, the newer ground motions would not be expected to increase the likelihood of malfunction of SSCs that are important to safety. As discussed in DPO Concern #1, incomplete information was available to make appropriate comparisons between the newer ground motions and the range of ground motions used to assess the safety of the Diablo Canyon NPP. The DPO submitter uses the previously available information to conclude that the newer ground motions exceed the plant's design basis and, thus, indicate an increase in the likelihood of equipment malfunction. For the reasons discussed in DPO Concern #1, these ground motions are not directly comparable. Consequently, there was insufficient basis to conclude that a license amendment was required to address the 2011 Shoreline report, and NRC staff's recommendation for an FSAR update was reasonable.

The DPO Panel evaluated DPO Concerns #1 and #2, and considered the new ground-motion information provided by PG&E to supplement the 2011 Shoreline report. The Panel believes that there is a sufficient basis to conclude that the likelihood of a malfunction has not increased more than minimally (or more specifically as stated in the guidance, "the uncertainties in determining whether a change in likelihood has actually changed (i.e., there is no clear trend towards increasing the likelihood)." Therefore, the Panel believes that an amendment to the operating licensee is not required, and that the FSARU Revision 21 update is an appropriate action in response to the new information.

Concern #4 – The NRC failed to adequately address the Los Osos and San Luis Bay faults. The new seismic information concluded that these faults were also capable of producing ground motions in excess of the current plant SSE design basis.

Although this DPO concern has many important similarities to DPO Concern #1, there is an important distinction that warrants clarification. The focus of the 2011 Shoreline report was on assessing the potential significance of the Shoreline fault, which was a newly characterized fault system. In contrast, both the Los Osos and San Luis Bay faults were recognized previously and evaluated as part of the LTSP. Ground motions for these two faults were simply reevaluated in the 2011 Shoreline report with the same updated methods used to assess the Shoreline fault. As shown in Figure 3 of the Panel report, the reevaluated potential ground motions for the Los Osos and San Luis Bay faults are approximately 10% higher than potential Shoreline fault ground motions.

In the LTSP, ground motions from the Los Osos and San Luis Bay faults were shown to be significantly lower than the ground motions for the Hosgri fault, which was thought to be the bounding ground motion for the Diablo Canyon NPP. Although the Los Osos and San Luis Bay faults were not addressed explicitly in NRC staff's 2009–2011 evaluations, the detailed 2012 NRC Research Information Letter 12-01 (ADAMS Accession No. ML121230035) evaluated these faults in the context of the Shoreline fault system. In addition, staff used information from the LTSP to conclude in RIL 12-01 that these faults were not capable of producing ground motions that challenged the licensing basis of the plant in a deterministic framework (see RIL 12-01, Chapter 5.10). Staff also recognized the non-negligible contribution that these faults might make to a probabilistic assessment (see RIL 12-01, Chapter 6), which would consider both the likelihood and magnitude of potential ground motions.

The Panel agrees with the DPO submitter's concern that NRC staff did not clearly and consistently consider the potential ground motions from the Los Osos and San Luis Bay faults in all reports and actions associated with the 2011 Shoreline Report. From a deterministic perspective, this omission is understandable because the ground motions from these two previously analyzed faults did not increase significantly, and were well within the limits already considered explicitly in LTSP analyses. From a risk perspective, initial analyses showed that individual contributions from these faults to the total seismic hazard were small, and bounded by Hosgri fault ground motions (e.g., RIL 12-01, Chapter 6). Nevertheless, the basis for staff's actions and conclusions sometimes were not clear because the Los Osos and San Luis Bay potential ground motions were not addressed explicitly. The DPO has highlighted the need for more explicit consideration of the Los Osos and San Luis Bay faults in future communications, based on the prominence of these faults in the 2011 Shoreline report.

The remainder of this DPO concern focuses on the same issue of ground-motion comparability that was discussed for DPO Concern #1. As shown in Figure 3, potential ground motions for the Los Osos and San Luis Bay faults can be approximately 10% higher than potential ground motions for the Shoreline fault. Ground motions for these three faults were all calculated with the same methods and assumptions. Thus, equivalent changes in the amount of damping or presence of building foundations should have equivalent changes in the calculated ground motions. In other words, the relative relationships between these three ground motion response spectra should not change significantly. Nevertheless, the Los Osos and San Luis Bay potential ground motions shown in Figure 3 have the same limitation as the Shoreline potential ground motions, in that they are not directly comparable to the full range of ground motions used in the FSARU to license Diablo Canyon. As discussed in Concern #1, these faults were considered explicitly in the March 2014 supplemental analyses by PG&E. The reanalyzed ground motions for the Los Osos and San Luis Bay (and the Shoreline) faults do not exceed the level of ground motion already used to design and license the plant.

5. Conclusions

Based on the preceding evaluation, the DPO Panel concludes:

1) The review of the DPO circumstances and information did not reveal a significant or immediate concern with the current understanding of seismic safety of the Diablo Canyon NPP.

2) The seismic licensing history at the Diablo Canyon NPP is long, complex and unique, and has been thoroughly evaluated by both the staff and licensee. Unlike other operating plants, seismic safety at the Diablo Canyon NPP has been evaluated using large ground motions from two different earthquakes. However, the safety analyses often use different physical conditions and analytical assumptions for each earthquake. As a result of these differences, PG&E and NRC staffs, and the DPO submitter, were unable to make an appropriate range of comparisons between the plant's licensing basis and new seismic information.

3) The DPO submitter was a positive contributor to both the licensee's and the staff's actions on seismic safety at the Diablo Canyon NPP, especially with bringing attention to important

safety relationships that were not always clear in the FSARU or supporting documents. The staff and licensee rationale in this area could have been improved by having a more direct comparison of the new information with the existing seismic licensing basis. As a result of this DPO, additional information was developed by PG&E to clearly demonstrate that potential ground motions from the Shoreline, Los Osos, and San Luis Bay faults would not exceed the levels of ground motion already considered during the design and licensing of the plant.

4) The staff followed its processes for technical specification operability of plant equipment and 10 CFR 50.59 evaluations with a reasonable technical and safety rationale. The staff's Fukushima 2.1 evaluation process is expected to provide an up-to-date assessment of both Diablo Canyon's seismic safety and the staff's evaluations regarding the Shoreline Fault.

5) The lack of a formal regulatory guidance for evaluating new information on natural hazards appears to be a contributing cause in creating many of the differing interpretations for the potential significance of this information.

6. Recommendations

1) Continue the Fukushima 2.1 evaluation process to both confirm the staff's analyses of the Shoreline Fault and assess the continued safe operation of the Diablo Canyon in consideration of the reevaluated seismic hazards at the site.

2) Better define (perhaps through Fukushima 2.2 or other durable regulatory products) the staff position on assessing new information about potential natural hazards at a site.

Appendix A:
Statement of Technical Concerns, Derived from Diablo Canyon DPO-2013-002

1) The NRC has not enforced Diablo Canyon Technical Specification requirements that key plant safety equipment remain operable during reactor operation. New seismic information developed by Pacific Gas and Electric concluded that Technical Specification required Structures, Systems and Components (SSCs) can be exposed to greater vibratory motion than was used to qualified this equipment for the facility safe shutdown earthquake (SSE) design basis. For Technical Specification required SSCs to be considered operable, the licensee is required to demonstrate a reasonable assurance that this plant equipment would still be capable of performing the safety functions in accordance with the plant design bases and safety analysis.

2) Pacific Gas and Electric's operability evaluation following development of the new seismic information was inadequate. Comparison of the new seismic information only against the Hosgri Event (HE) and Long Term Seismic Program (LTSP) ground motions was not adequate to demonstrate Technical Specification required SSCs were operable. Neither the HE nor the LTSP methods were approved to be used in SSE safety analysis. The HE and LTSP methods over-predicted SSC performance when compared to the SSE design basis methods. Even though the HE and LTSP include higher ground motions, neither of these methods were bounding for plant Technical Specification SSCs seismic qualification. Use of the HE and LTSP ground motions failed to demonstrate that that the requirements of the American Society of Mechanical Engineers' (ASME) Boiler and Pressure Vessel Code acceptance limits would be met at the higher ground motions. 10 CFR 50.55a required that ASME acceptance limits be met for plant safety Class 1 and 2 following an SSE. Demonstration that the ASME acceptance limits are met provides assurance that the integrity of key plant systems, including the reactor coolant pressure boundary would be maintained following the higher seismic stress levels represented by the new seismic information.

3) The NRC has failed to enforce the 10 CFR 50.59 requirements that Pacific Gas and Electric obtain an amendment to the Diablo Canyon Operating License prior to incorporating the Shoreline scenario into the FSARU. A license amendment was required because the change resulted in more than a minimal increase in the likelihood of a malfunction of SSCs important to safety than previously evaluated in the FSARU. A license amendment was also required because this change represents a departure from the FSARU method of evaluation used to establish the seismic SSE design basis. The NRC conclusion that a "reasonable assurance of safety" existed was not an adequate basis to conclude an amendment to the Diablo Canyon Operating License was not required.

4) The NRC failed to adequately address the Los Osos and San Luis Bay faults. The new seismic information concluded that these faults were also capable of producing ground motions in excess of the current plant SSE design basis.

Appendix B:

Publically Available Records and Documents Reviewed by DPO Panel

1. Diablo Canyon Power Plant Units 1 & 2 FSAR Update, Revision 21, September 2013. ML13280A390.

2. Criterion 2, Design Basis Protection Against Natural Phenomena, of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the Code of Federal Regulations (10 CFR), Part 50.

3. NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, February 2000. ML003686043.

4. Union of Concerned Scientists, "Seismic Shift – Diablo Canyon Literally and Figuratively on Shaky Ground," November 2013.

5. Rezaeian, S., and others, "Spectral Damping Scaling Factors for Shallow Crustal Earthquakes in Active Tectonic Regions," Pacific Earthquake Engineering Research Center Report 2012/01, July 2012.

6. Licensee Amendment Request 11-05, "Evaluation Process for New Seismic Information and Clarifying the Diablo Canyon Power Plant Safe Shutdown Earthquake," October 2011. ML11298A247.

7. Letter from Barry S. Allen to Nuclear Regulatory Commission, "Withdrawal of License Amendment Request 11-05," October 2012. ML12300A105.

8. Letter from Joseph M. Sebrosky to Edward D. Halpin, "Diablo Canyon Power Plant Units 1 and 2 – Withdrawal of an Amendment Request," October 2012. ML12289A076.

9. Letter from James R. Becker to Nuclear Regulatory Commission, "Standard Review Plan Comparison Tables for License Amendment Request 11-05," December 2011. ML11342A238.

10. Letter from Eric Leeds to All Power Reactor Licensees, "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.

11. Letter from James R. Becker to Nuclear Regulatory Commission, "Report on the Analysis of the Shoreline Fault Zone, Central Coastal California," January 2011. ML110140400.

12. Memorandum from Kriss M. Kennedy to Robert Nelson, "Task Interface Agreement – Concurrence on Diablo Canyon Seismic Qualification Current Licensing and Design Basis," August 2011. ML112130665.

13. Memorandum from Sher Bahadur to Kriss M. Kennedy, "Revised Response to Task Interface Agreement – Diablo Canyon Seismic Qualification Current Licensing and Design Basis, TIA 2011-010 (TIA 202-012)," November 2012. ML12297A199.

14. Letter from Joseph Sebrosky to Edward D. Halpin, "Diablo Canyon Power Plant – NRC Review of Shoreline Fault," October 2012. ML120730106.

15. Research Information Letter 12-01, "Confirmatory Analysis of Seismic Hazard at the Diablo Canyon Power Plant from the Shoreline Fault Zone," September 2012, ML121230035.

16. Memorandum from Brian W. Sheron to Eric J. Leeds, "Research Information Letter 09-001: Preliminary Deterministic Analysis of Seismic Hazard at Diablo Canyon NPP from Newly Identified "Shoreline Fault" April, 2009. ML090330188.

17. Letter from Neil O'Keefe to John T. Conway, "Diablo Canyon Power Plant – NRC Integrated Inspection Report 05000275/2011005 and 05000323/2011005," February 2012. ML120450843.

18. Non-Concurrence Process Record NCP-2012-001, "Diablo Canyon Power Plant – Inspection Report 05000275/323-2011005," June 2012. ML12151A173.

19. Report, "Additional Branch Chief Comments Related to NCP 2012-001 with Annotations," June 2012. ML12284A066.

20. IE Information Notice No. 79-06, "Stress Analysis of Safety Related Piping," March 1979. ML080310608.

21. Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1, March 2007. ML070260029.

22. Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)," September 1999. ML003740112.

23. Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," Revision 0, December 2000. ML003754825.

24. NUREG/CR-1429, "Seismic Review Table," May 1980. ML110880747.

25. NUREG/CR-6919, "Recommendations for Revision of Seismic Damping Values in Regulatory Guide 1.61," November 2006. ML063260342.

26. NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program, vols. 1 and 2," April 2002. ML021270070 and ML021270674.

27. NRC Inspection Manual Part 9900: Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Non-Conforming Conditions Adverse to Quality or Safety." April 2008. ML051520373.

28. NRR Office Instruction LIC-202, "Procedures for Managing Plant-Specific Backfits and 50.54(f) Information Requests," Revision 2, May 2010. ML092010045.

29. NRR Office Instruction LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," Revision 3, April 2010. ML100541776.

Non-Publically Available ADAMS Records and Documents Reviewed by DPO Panel

1. Memorandum from Bradley W. Jones to John A. Grobe, "NRC Sources of Legal Requirements and the Applicability of 10 CFR Part 100 Standards," August 2008, ML082460980 (NONPUBLIC).

2. Memorandum from Michele G. Evans to Eric J. Leeds, "NRC and Licensee Actions in Response to New Information from a Third Party," ML112730055 (NONPUBLIC).

3. NRR Office Instruction LIC-100, "Control of Licensing Bases for Operating Reactors," Revision 1, January 2004. ML033530249 (NONPUBLIC).

4. Memorandum from Michael T. Markley to Neil F. O'Keefe, "Response to Senior Resident Inspector Question Regarding the Diablo Canyon Research Information Letter Associated with the Shoreline Fault," October 2012. ML12213A079 (NONPUBLIC).

5. NUREG-0675 Supplement No. 34, Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant," June 1991. ML093070113 (NONPUBLIC).

Information not Located in ADAMS and Assumed to be Non-Publically Available

1. Meeting Summary: Pre-Licensing Meeting with PG&E on Plans to Submit a License Amendment to Incorporate Management of New Geotechnical Seismic Information Into Its Design and Licensing Basis, January 2011.

2. Meeting Summary: Pre-Licensing Meeting with PG&E on Proposal License Amendment for a New Seismic and Design Evaluation Process, July 2011.

3. Meeting Summary: Pre-Licensing Meeting with PG&E on Responses to Staff Questions From Previous Public Meeting on January 26, 2011, May 2011.

4. DCPP Form 69-20108, "UFSAR Change Request," June 2013.

5. Prompt Operability Update, "DCPP Shoreline Fault POA 10-21-2011."

6. Memorandum from Meena K. Kanna to Michael T. Markley, "Safety Evaluation DCPP Units 1 & 2 License Amendment Request for Damping Values for the Seismic Design and Analysis of the Reactor Vessel Integrated Head Assembly (IHA)."

7. Memorandum from Catherine E. Kanatas to Edward Williamson, "Legal Process Regarding North Anna Restart Decision," November 2, 2011.

8. Report, "Diablo Canyon Seismic Licensing History Briefly Summarized." November 2013.

9. Report, "Timeline of Seismic Issues at Diablo Canyon." August 2013.

10. Report, "Resident Inspectors Recommendation for Regulatory Disposition of the Failure of PG&E to Perform on Operability Evaluation Following Discovery of the Shoreline Fault." February 2011.

11. Letter from John F. Stolz to John C. Morrissey, "Staff Evaluation of Probabilistic Seismic Risk Assessment," November 1978.

12. PG&E Letter DCL-88-192 from D.A. Brand to NRC, "Long Term Seismic Program Completion," July 31, 1988.

13. NUREG-0675 Supplement No. 7, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant," May 1978.

14. NUREG-0675 Supplement No. 8, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant," November 1978.

Document 4 – DPO Decision

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 29, 2014

MEMORANDUM TO: Michael S. Peck, Senior Reactor Technical Instructor
Reactor Technical Training Branch B
Office of the Chief Human Capital Officer

FROM: Eric J. Leeds, Director
Office of Nuclear Reactor Regulation

SUBJECT: DIFFERING PROFESSIONAL OPINION INVOLVING SEISMIC ISSUES AT DIABLO CANYON (DPO-2013-002)

On July 19, 2013, in accordance with Management Directive 10.159, "The NRC Differing Professional Opinions Program", you submitted a differing professional opinion (DPO) concerning seismic issues at the Diablo Canyon Nuclear Power Plant (DCNPP) (DPO-2013-002). Specifically, your DPO states that there is less than adequate corrective actions to incorporate the new seismic information into the current licensing basis and that the licensee has failed to demonstrate that the plant technical specification required structures, systems, and components (SSCs) are operable. The purpose of this memorandum is to respond to your DPO.

On September 3, 2013, I established a DPO Ad Hoc Review Panel (the Panel) and tasked them to meet with you, review your DPO submittal, and issue a DPO report, including conclusions and recommendations to me regarding the disposition of the issues presented in your DPO. The Panel initially met with you on October 23, 2013 to establish a concise statement of your concerns. On April 3, 2014, after reviewing the applicable documents, completing internal interviews of relevant individuals and completing their deliberations, the Panel issued their report to me.

Following receipt of their report, I provided a copy of the report to you for your review. On April 21, 2014, I talked to you by telephone to discuss the Panel's report and to get your insights and comments. You provided me additional insights into your concerns and your thoughts for resolving the issues concerning DCNPP.

In order to make a decision with regard to your DPO, I reviewed the Panel's report, met with the Panel, talked with you, and then re-considered your comments to me.

The Panel concluded that there was not a significant or immediate concern with the current understanding of seismic safety of the DCNPP. The Panel also concluded that the seismic licensing history at DCNPP is extremely complex and that the licensee, the U.S. Nuclear Regulatory Commission (NRC) staff, and you were unable to make an appropriate range of comparisons between the plant's licensing basis and new seismic information. The Panel acknowledged that you were a positive contributor to the staff and the licensee on seismic safety at DCNPP. As a result of your DPO additional information was developed by the licensee to clearly demonstrate that potential ground motions from nearby faults would not exceed the levels of ground motion already considered during the design and licensing of the plant.

In addition, the Panel concluded that the licensee and the staff followed its processes for technical specification operability of plant equipment and Title 10, Code of Federal Regulations 50.59 evaluations with a reasonable technical and safety rationale.

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

After considering all the information, I have concluded that this is not a safety significant issue as independent groups (as documented in Research Information Letter 12-01 and the DPO Panel report) have verified that the new seismic information is bounded within the existing analysis for DCNPP. Further, during our discussion you indicated that you defer judgment of the safety significance of this issue to the NRC technical staff who are trained and qualified to make those judgments. Finally, I agree with the Panel report with respect to its assessment of your technical concerns.

Nevertheless, your examination of this issue in the context of the DCNPP experience has re-enforced to me that the NRC hasn't been definitive in how new information on natural hazards should be considered in the regulatory process. I believe that the work currently underway on the Fukushima Near Term Task Force Recommendations 2.1 (re-evaluation of seismic and flooding hazards) and 2.2 (require licensees to confirm seismic and flooding hazards every 10 years) will address this issue.

A summary of the DPO will be included in the Weekly Information Report (when the case is closed) to advise interested employees of the outcome. Based on your interest in this area, please contact the Office of Nuclear Reactor Regulation Technical Assistant, Trent Wertz (trent.wertz@nrc.gov), if you would like to be placed on distribution for all meeting notices, summaries, and correspondence related to Fukushima Near Term Task Force Recommendations 2.1 and 2.2. A yellow ticket will be issued to keep you and the Office of Enforcement informed of any schedule changes and when the final report(s) are issued.

Thank you for raising your DPO and for your active participation in the DPO process. An open and thorough exploration of how we carry out our regulatory process is essential to keeping these programs effective. I believe that in raising the DCNPP seismic issue, you helped mobilize the Agency to ensure safety at that site. Your willingness to raise concerns with your colleagues and managers and ensure that your concerns are heard and understood is admirable and vital to ensuring a healthy safety culture within the Agency.

Enclosure: DPO Panel report April 3, 2014

cc: D. Skeen, NRR
R. Pedersen, OE
R. Zimmerman, OE
M. Case, OIP
R. Bernhard, RII
M. Dapas, RIV
B. Hill, RES
M. Johnson, EDO
M. Satorius, EDO

Document 5 – DPO Appeal Submission

NRC FORM 690
(11-2002)
NRCMD 10.159

U.S. NUCLEAR REGULATORY COMMISSION

DIFFERING PROFESSIONAL OPINION -- APPEAL

FOR PROCESSING USE ONLY

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

☐ Appeal sustained

☐ Appeal denied (see attached)

Differing Professional Opinion Closed

DATE

NRC FORM 690 (11-2002)

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)."

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.

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)

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.

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)

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?"

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

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.

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)

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.

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

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.

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).

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.

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 is equivalent to the DDE

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.

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 Design 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.

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)

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
@	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

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.

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

DCPP UNITS 1 & 2 FSAR UPDATE

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.

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[39].

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

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

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.

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:

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

(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:

Differing 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.

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.

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 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 issu 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	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$
Normal	Deadweight + Pressure + Thermal	$P_L + P_b + P_e + Q \leq 3\ S_m$[(b)]
Upset - 1	Deadweight + Pressure + Thermal ± DE	$U_T \leq 1.0$[(b)] $P_L + P_b + P_e + Q \leq 3\ S_m$
	Deadweight + Pressure + Thermal	$U_T \leq 1.0$[(b)] $P_L + P_b + P_e + Q \leq 3\ S_m$
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

(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_T = Cumulative usage factor.

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.

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.

Document 6 – Statement of Views

June 27, 2014

MEMORANDUM TO: Mark A. Satorius
Executive Director for Operations

FROM: Eric J. Leeds, Director ***/RA/***
Office of Nuclear Reactor Regulation

SUBJECT: STATEMENT OF VIEWS REGARDING APPEAL OF DIFFERING PROFESSIONAL OPINION CONCERNING DPO 2013-002

On July 19, 2013, in accordance with Management Directive 10.159, "The NRC Differing Professional Opinions Program," a differing professional opinion (DPO) concerning seismic issues at the Diablo Canyon Nuclear Power Plant (DCNPP) (DPO-2013-002) was submitted. On September 3, 2013, I established a DPO Ad Hoc Review Panel (the Panel) and tasked them to meet with the submitter, review the DPO submittal, and issue a DPO report, including conclusions and recommendations, to me regarding the disposition of the issues presented in the DPO.

On April 3, 2014, after reviewing the applicable documents, completing internal reviews of relevant individuals and completing their deliberations, the Panel issued their report to me. On May 29, 2014, I issued a closeout memorandum to the submitter documenting my decision regarding the DPO. On June 23, 2014, the submitter submitted an appeal to you regarding the DPO and my decision. This memorandum is to provide you with my views regarding statements in the appeal.

After reading the appeal, the submitter reiterated his stance on the reasons the DPO was originally submitted. I think it's extremely important to note that the submitter continues to agree with the Panel's conclusion that issues raised in the DPO did NOT result in a significant or immediate safety concern. The safety of the DCNPP is not in question. However, the submitter did not include any new, safety significant or other information that would cause me to alter my disposition of the DPO.

I also think it is important to note that the submitter's DPO illustrates the need for the Agency to generically resolve how changes to external natural hazard parameters are processed by both licensees and the staff. This work is currently underway with regard to seismic and flooding hazards in response to the Fukushima accident. I expect the outcome of the staff's work on seismic and flooding issues will result in a well-defined process for both licensees and the staff to follow in the future, and this will help prevent a recurrence of the issues raised in the submitter's DPO.

Please contact me if you have any questions regarding this Memorandum.

cc: M. Johnson, OEDO
D. Dorman, NRR
R. Pedersen, OE

June 27, 2014

MEMORANDUM TO: Mark A. Satorius
Executive Director for Operations

FROM: Eric J. Leeds, Director ***/RA/***
Office of Nuclear Reactor Regulation

SUBJECT: STATEMENT OF VIEWS REGARDING APPEAL OF DIFFERING PROFESSIONAL OPINION CONCERNING DPO 2013-002

On July 19, 2013, in accordance with Management Directive 10.159, "The NRC Differing Professional Opinions Program," a differing professional opinion (DPO) concerning seismic issues at the Diablo Canyon Nuclear Power Plant (DCNPP) (DPO-2013-002) was submitted. On September 3, 2013, I established a DPO Ad Hoc Review Panel (the Panel) and tasked them to meet with the submitter, review the DPO submittal, and issue a DPO report, including conclusions and recommendations, to me regarding the disposition of the issues presented in the DPO.

On April 3, 2014, after reviewing the applicable documents, completing internal reviews of relevant individuals and completing their deliberations, the Panel issued their report to me. On May 29, 2014, I issued a closeout memorandum to the submitter documenting my decision regarding the DPO. On June 23, 2014, the submitter submitted an appeal to you regarding the DPO and my decision. This memorandum is to provide you with my views regarding statements in the appeal.

After reading the appeal, the submitter reiterated his stance on the reasons the DPO was originally submitted. I think it's extremely important to note that the submitter continues to agree with the Panel's conclusion that issues raised in the DPO did NOT result in a significant or immediate safety concern. The safety of the DCNPP is not in question. However, the submitter did not include any new, safety significant or other information that would cause me to alter my disposition of the DPO.

I also think it is important to note that the submitter's DPO illustrates the need for the Agency to generically resolve how changes to external natural hazard parameters are processed by both licensees and the staff. This work is currently underway with regard to seismic and flooding hazards in response to the Fukushima accident. I expect the outcome of the staff's work on seismic and flooding issues will result in a well-defined process for both licensees and the staff to follow in the future, and this will help prevent a recurrence of the issues raised in the submitter's DPO.

Please contact me if you have any questions regarding this Memorandum.

cc: M. Johnson, OEDO
D. Dorman, NRR
R. Pedersen, OE

ADAMS Accession No. ML14177A613

OFFICE	NRR
NAME	ELeeds
DATE	06/27/14

OFFICIAL RECORD COPY

Document 7 – DPO Submitter's Appeal Presentation

Diablo Canyon Seismic Issues

Appeal of DPO 2013-02 Decision

DPO Issues:

The NRC failed to enforce:

- 10 CFR 50.59 requirement that Pacific Gas and Electric (PG&E) obtain an amendment to the Operating License prior to incorporating new seismic information into the FSARU.
- Plant Technical Specifications following inadequate demonstration of operability:
  - New seismic information resulted in greater stress than plant structure, system and components (SSCs) were qualified.
  - New seismic stresses exceeded the ASME Code acceptance limits for the reactor coolant pressure boundary (RCPB).

Diablo Canyon
Seismic Design and Licensing Bases

General Design Criteria (GDC) 2 & Part 100, Appendix A: Certain SSCs remain functional following the maximum earthquake potential considering the local geology and seismology:

- Integrity of the reactor coolant pressure boundary (ASME Code acceptance limits),
- Capability to shut down the reactor and maintain it in a safe shutdown condition, and
- Capability to prevent or mitigate the consequences of accidents (seismic qualification of plant SSCs).

Diablo Canyon
Design and Licensing Bases

Safe shutdown earthquake (SSE) design bases :

- Safety analysis (10 CFR 50.34) developed the Double Design Earthquake (DDE, 0.4 pga).
- Demonstrated that the GDC 2 functional requirements were satisfied for the maximum ground motion based on the local geology and seismology.
- Important to safety SSCs (RG 1.29, Seismic Design Classification) qualified to the DDE spectra.
- RCPB (Class 1 Systems) qualified to ASME, Section III, for DDE plus accident loads (10 CFR 50.55a).

Diablo Canyon
Design and Licensing Bases

Hosgri Fault discovered during plant construction:

- Licensee concluded that the fault was not “capable” per Part 100, Appendix A, and excluded the ground motion from the SSE safety analysis.
- PG&E prepared the Hosgri Evaluation (HE) in response to an NRC question during plant licensing.
- HE demonstrated that the plant could safety shutdown following 7.5 M on the Hosgri fault (0.75 pga)

Diablo Canyon
Design and Licensing Bases

Hosgri Evaluation (HE):

- Used different assumptions, methodology, load combinations, and acceptance limits than the DDE/SSE.
  - Did not assume coincidental accident or fire.
  - Explicitly excluded RG 1.29 (SSCs qualified for the SSE).
  - Some Code limits exceeded (non-liner effects).
  - Included ASME , Section III, calculations for the RCPB, excluding LOCA loads (no accident).
- For many SSCs, including the RCPB, seismic qualification was more limited by the DDE/SSE (0.4 pga) rather than the HE (0.75 pga).

Diablo Canyon Licensing Bases

Long Term Seismic Program

- License Condition – Reevaluate the local seismologic within 10 years.
- Completed in 1988.
- NRC concluded that PG&E satisfied the License Condition (1991):
  - Did not alter the plant design bases.
  - Seismic qualification basis will continue to be the DD and the DDE (OBE & SSE) design basis plus the HE, along with associated analytical methods, initial conditions, etc.

New Seismic Information

Seismic Reevaluation submitted to the NRC (2011):

- Concluded three local faults were “capable” of generating significantly greater ground motion (0.7 pga) than used to establish the facility SSE.
- PG&E submitted License Amendment Request (LAR) 2011-05 to change the “method of evaluation” used for the facility SSE from the DDE to the HE.
- The NRC concluded that the HE did not meet NRC requirements for the SSE. At the NRC’s request, PG&E withdrew the LAR (2012).

NRR Disposition

NRR PM directed PG&E to add the Shoreline fault to the FSARU as a “lessor case of the HE.”

- The Hosgri ground motions were “previously demonstrated to have reasonable assurance of safety.”
- Deferred further evaluations pending Fukishima Recommendation 2.1.
- Did not address other faults that exceeded the SSE.

DPO Panel Conclusions:

"The new seismic information did not reveal a significant or immediate seismic safety concern"

- However, the DPO did not assert that significant or immediate safety concern existed at Diablo Canyon.
- The DPO was written to draw attention and promote correct actions following the <u>agency's failure to enforce existing regulatory and statutory requirements</u>.
- Adequate protection (nuclear safety) is presumptively assured by compliance with NRC requirements.

DPO Panel Conclusions:

"The staff followed its processes for technical specification operability of plant equipment and 10 CFR 50.59 evaluations."

- However, the Panel's conclusions were based on a different facility design and licensing bases than presented in the DPO and the FSARU.
- The Panel inappropriately considered the HE as a facility SSE.
- The Panel Report did not offer explanation for the deviation.

What Does the Regulations Require?

10 CFR 50.71(e) and 10 CFR 50.59:

- These statutory requirements required PG&E to evaluate the new information against the "***facility as described in the FSARU*.**"
- Ensures fidelity is maintained between the functional GDC requirements, the methods used to demonstrate that the GDCs were met (10 CFR 50.34 safety analysis, as presented in the License Application, as amended), and the plant technical specifications.

What Does the Regulations Require?

10 CFR 50.71(e) required PG&E to updated the FSARU with the new seismic information:

- The new information was developed by PG&E.
- The new ground motions were greater than those used in the safety analysis demonstrating that GDC 2 was satisfied (design basis controlling parameter).
- The new ground motions were also greater than those used to demonstrate that ASME Code requirements were satisfied for the SSE per 10 CFR 50.55a.

10 CFR 50.71 (e)

"…contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement"

"…shall include the effects of all changes made in the facility or procedures ***as described in the FSAR***; all safety analyses and evaluations…"

10 CFR 50.59

- Change: “A modification or addition to, or removal from, the facility or procedures that affects a design function, method of performing or controlling the function, or an evaluation that demonstrates that intended functions will be accomplished.”
- *Facility as described in the FSAR*:
  - “Evaluations or methods of evaluation included in the FSAR for such SSCs which demonstrate that their intended function(s) will be accomplished.”
  - “Submitted in accordance with §50.34, as amended and supplemented, and as updated per the requirements of Sec. 50.71(e).”

10 CFR 50.59

- Departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses:
  - "Changing any of the elements of the method described in the FSAR unless the results of the analysis are conservative or essentially the same;" or
  - "Changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application."

What Does the Regulations Require?

10 CFR 50.59 required PG&E to obtain a an amendment to the Operating License for the FSARU:

- Modification of the design basis controlling parameter (ground motions) to the existing safety analysis resulted in exceeding acceptance limits for ASME Code and current SSC seismic qualification.
- This *"resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety," (*NEI 96-07).

What Does the Regulations Require?

10 CFR 50.59 also required PG&E to obtain an amendment to the Operating License to change the SSE methodology from the DDE to the HE:

- The HE was less conservative than the DDE.
- *“Results in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analysis”* (NEI 96-07).
- Considering the Shoreline fault a “lessor case of the HE” attached the HE method of evaluation to the SSE.

DPO Panel Concluded:

"The new ground motions were bound by the Hosgri and the Long Term Seismic Program."

- The DPO did not dispute that the new ground motions were bound by the HE and the LTSP.
- However, this information was not relative since neither the HE nor the LTSP were part of the facility SSE (GDC 2) design bases.
- The DPO Panel Report did not include the based for using the alternant design and licensing bases.

The HE was not the Facility SSE

The original and current FSARs were clear:

- The DDE, with supporting safety analysis, was used to demonstrate (GDC 2) SSE design basis.
- DDE/SSE design basis was implemented and controlled by the licensee using the facility Q-List.
- The DDE ground motions are used to establish SSC seismic qualification per PG&E's commitment to RG 1.29.
- The Hosgri was treated as a licensing basis commitment (beyond design bases), not attached to GDC 2 (or any other regulatory requirement).

The HE was not the Facility SSE

- Prior to the Panel Report, the DDE/SSE design basis was not in dispute.
- NRR and PGE initially agreed that an amendment to the license was required because the FSAR did not described the HE as the SSE.
- After the staff determined that the HE did not meet NRC requirements for the SSE (non-acceptance of LAR 11-05), NRR implemented a “work around” to the license amendment process.

SSC Seismic Qualification was More Limiting for the DDE

- For a given ground motion, the DDE methodology will <u>always</u> produce greater SSC seismic stress that the HE.
- Ground motion alone does not established seismic design basis (NRC - SSER 7). Equally important are other factors:
  - Methods of analysis
  - Shape of spectra
  - Damping values used
  - Load combinations
  - Initial conditions
  - Acceptance criteria, including allowable stress

Consequence of the Failure to Obtain an License Amendment

The NRR PM’s action subverted the required License Amendment Request public notice and hearing opportunities per 10 CFR 50.91:

- Substantial stake holder interest in Diablo Canyon seismic issues.
- Inadequate NRC review of plant SSC response to the higher ground motions.
- Adversely affects public perception of NRC as regulator.
- Established a new precedent for discovery of conditions outside of the existing design bases.

DPO Panel Concluded:

"The staff followed its processes for technical specification operability of plant equipment."

- However, the Panel incorrectly assumed that the HE satisfied GDC 2 safety analysis (*as described in the FSARU*).
- As a result, the Panel Report did not address the specific technical issues raised in the DPO associated with use of the HE or LTSP as an "alternative analytical method" for determining operability.

What does the License Require?

Plant Technical Specifications required important to safety SSCs to be operable:

- Equipment needed to prevent or mitigate an accident must be capable of performing required safety functions following the SSE.
- The new seismic information called into question if this SSC functional requirement can be still be met at the higher ground motions.
- Applying the new ground motions to the existing SSE safety analysis resulted in stress exceeding the seismic qualification limits of important to safety SSCs.

What dose the License Require?

PG&E did not evaluate the new information against the SSE/DDE:

- Used the HE as an "alternative analytical method."
- Not permitted per IMC 0326 (Appendix C.4):
  - The HE methodology will <u>always</u> over-predict SSC performance when compared to the FSARU SSE methodology.
  - For a given ground motion, the DDE/SSE will <u>always</u> be more limiting for seismic qualification.
  - Successful demonstration of Technical Specification SSC operability is required for continued reactor operation.

DPO Panel Concluded:

"The new information by itself did not alter the FSARU approach to maintain both he DDE and HE as failed conditions with respect to seismic component and piping analysis:"

- The Panel concluded that either the HE or the DDE established the limiting loads for ASME acceptance.
- Since the new ground motions were less than those assumed in the HE, then all ASME Code requirements were satisfied.

What Does the Regulations Require?

10 CFR 50.55a required PG&E to met ASME, Section III, Code requirements for the RCPB (Class 1 Systems):

- SSE plus accident loads must be less than acceptance limits (Service Level D).
- The new seismic information resulted in a greater maximum (credible) earthquake potential than described in the FSARU SSE safety analysis. This rendered the 50.34 safety analysis nonconforming with the requirements of GDC 2, (Criteria III "Design Control," and XVI "Corrective Actions)."
- The nonconforming supporting safety analysis was used as input for satisfying 10 CFR 50.55a.

What Does the Regulations Require?

10 CFR 50.55a required PG&E to meet ASME, Section III, Code requirements:

- Applying the new ground motions (design bases controlling parameter) to the existing SSE safety analysis resulted in exceeding Code acceptance limits.
- The Code did not include provision for substitution of the HE for the SSE for seismic inputs.
- Meeting ASME Code acceptance limits for the RCPB (Class 1 Systems) was required for continued reactor operation (IMC 0326, Appendix C.11).

DPO Panel Concluded:

"The lack of a formal regulatory guidance for new information appeared to contribute creating differing interpretations for the potential significance."

- The DPO limited the potential significance of the new information to the nexus between compliance and safety.
- The Enforcement Manual and Significance Determination Process should have been used to establish the actual safety significance of the issues and ensure adequate corrective actions.

The Current Regulatory Framework Ensures Continuity Between:

- GDC functional requirements and the design bases.
- FSAR safety analysis (10 CFR 50.34) demonstrating that the design bases satisfies the GDC functional requirements.
- Part 50, Appendix B, ensures that this design bases is maintained by the facility safety analysis and design control for individual plant SSCs.
- 10 CFR 50.71(e) ensures new information that affects the design bases/safety analysis is updated in the FSARU.
- 10 CFR 50.59 ensures fidelity is maintained between the design bases and FSAR safety analysis methods (GDCs).

Additional Formal Regulatory Guidance Was Available

Supplemental information reinforced agency regulations to address non-conforming conditions associated new information related to natural phenomena or the failure to meet a GDC as described in the current licensing bases (FSARU):

- Letter (Leeds), supplemental information related Recommendation 2.1
- Regulatory Issue Summary RIS 2013-05,
- IMC 0326, Appendix C1,
- Past enforcement actions (Watts Bar Flooding)

DPO and Nonoccurrence Processes were Ineffective

Non-concurrence NCP 2012-01 addressed PG&E’s inadequate operability evaluation. The Agency did not respond to the technical issues:

- Code compliance,
- Inappropriate use of HE as “alternate analytical method.“

The DPO Panel created a new facility design and license bases. The Agency did adequately address the issues raised in the DPO:

- Specific criteria in 10 CFR 50.59/NEI 96.07

Summary

DPO Panel did not fully consider the original and current facility design and licensing bases (FSAR):

- Led to incorrect conclusions related to compliance with 10 CFR 50.71(e) and 10 CFR 50.59.
- FSARU “ambiguities” require corrective action and do not provide an adequate bases for deferring enforcement action.
- The lack of an immediate or significant safety issue does not provide adequate justification for failing to enforce statutory and license requirements.

Summary

- The NRR PMs bypassed the license amendment process:
  - A “reasonable assurance of safety,” was inconsistent with current regulatory requirements (10 CFR 50.59).
- Agency “work around” of the licensee amendment process created potential for safety significance:
  - Inadequate agency review of the impact of new seismic information on plant SSCs (10CFR 50.59)
  - Subverting the required notice and hearing opportunity (10 CFR 50.91)

Summary

- Continued failure to enforce plant technical specification requirements.
- Improvements are needed to enhance agency accountability for the non-concurrence and DPO process decisions.

Recommend Actions

1. Disapprove the DPO 2013-002 Panel Report decision:
   – The Panel Report applied an incorrect design and licensing bases when addressing the DPO compliance issues.
   – Use of the correct facility design and licensing based substantiates the issues raised in the DPO.
2. Initiate enforcement action to address the ongoing non-compliances with Part 50, Appendix B, 10 CFR 50.59, and plant technical specifications at Diablo Canyon:
   – The facility continues to operate outside the bounds of the current safety analysis and design bases.
   – PG&E has not adequately demonstrated that all technical specification required SSCs are operable.

Recommend Actions

3. Initiate a review to determine why the non-concurrence and the DPO processes were not effective:
   - Region IV response to NCP 2012-01 did not address the technical issues raised.
   - DPO Panel created a new licensing bases to justify past inappropriate agency actions.

Document 8 – DPO Appeal Decision

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 9, 2014

MEMORANDUM TO: Michael S. Peck
Senior Reactor Technical Instructor
Technical Training Center

FROM: Mark A. Satorius
Executive Director for Operations

SUBJECT: DIFFERING PROFESSIONAL OPINIONS APPEAL DECISION INVOLVING SEISMIC ISSUES AT DIABLO CANYON (DPO-2013-002)

The purpose of this memorandum is to inform you of my considerations and conclusions regarding the appeal you submitted on June 23, 2014, on the May 29, 2014, decision issued by the Director, Nuclear Reactor Regulation on the Differing Professional Opinion (DPO) you submitted on July 19, 2013. The DPO Program is addressed in Management Directive 10.159, "The NRC Differing Professional Opinions Program." I appreciate your efforts to provide a thorough description of your concerns and the supporting documentation in both your initial DPO and the subsequent appeal.

BACKGROUND

Your DPO is rooted in the Diablo Canyon Power Plant (DCPP) seismic licensing history and how the licensing basis is applied to the plant today. Although I know you are intimately familiar with this background, it serves as a reference for those less acquainted with the details who may read this response. In 1968, when the DCPP Unit 1 Construction Permit was issued to Pacific Gas and Electric (PG&E), the seismic evaluation had been completed under the Atomic Energy Commission's requirements. Based on the information available at the time, the design earthquake (DE) was defined as having a peak ground acceleration of 0.2 g, and the double design earthquake (DDE) was a doubling of the DE earthquake to ensure safety-related structures, systems, and components would function as expected after the earthquake, 0.4 g. In 1973, PG&E became aware of the Hosgri fault. PG&E evaluated the Hosgri fault using Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973. Though not included in the construction application, NRC reviewed PG&E's evaluation of the Hosgri fault and required PG&E to make plant modifications to be able to withstand the 0.75 g peak ground acceleration associated with the Hosgri fault. The operating license for Unit 1, issued in 1984, was based on review of the Final Safety Analysis Report Update which included two different seismic methodologies, the DDE and the Hosgri evaluation, as documented in NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Power Plant, Units 1 and 2," Supplement No. 7, dated May 1978. Given expected advances in the science of seismic evaluation, the license was also conditioned to require a confirmatory seismic study over the first 10 years of operation, referred to as the Long Term Seismic Program (LTSP). The NRC's review and acceptance of PG&E's report on the LTSP are discussed in NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Supplement No. 34, dated June 1991 (SSER 34), and in NRC letter dated April 14, 1992, "Transmittal of Safety Evaluation Closing Out Diablo Canyon Long-Term Seismic Program (TAC Nos. M80670 and M80671)."

In November 2008, PG&E reported identification of a new offshore fault, subsequently identified as the Shoreline fault zone. The initial licensee evaluation indicated that the newly identified fault was "smaller than the Hosgri fault, which is the current bounding seismic feature for DCPP." In January 2011, PG&E submitted the "Report on the Analysis of the Shoreline Fault Zone, Central Coast California" (Agencywide Documents Access and Management System (ADAMS) Accession Number ML110140431) which documented the investigation of the Shoreline fault zone and its relationship to other seismic sources in proximity to the DCPP. In October 2011, PG&E submitted license amendment request 11-05, ADAMS Accession No. ML11312A166. This amendment request proposed to revise the licensing basis, as described in the Final Safety Analysis Report Update and Technical Specifications (TS), to provide requirements for the actions, evaluations, and reports necessary when PG&E identifies new seismic information relevant to the design and operation of DCPP. PG&E submitted a withdrawal request for license amendment request 11-05, on October 25, 2012, ADAMS Accession Number ML12300A105. The withdrawal request followed the issuance of the NRC letter, "Diablo Canyon Power Plant, Units Nos. 1 and 2 - NRC Review of Shoreline Fault (TAC Nos. ME5306 and ME5307)," dated October 12, 2012 (ADAMS Accession Number ML12073016), and the NRC Research Information Letter 12-01, "Confirmatory Analysis of Seismic Hazard at the Diablo Canyon Power Plant from the Shoreline Fault Zone," dated September 2012 (ADAMS Accession Number ML121230035). The conclusion of the review from the Research Information Letter 12-01, executive summary is excerpted below:

> "In this review of the hazard from the Shoreline fault, the NRC compared the resulting deterministic seismic ground motions to loading levels for which the plant has been previously reviewed, specifically the Hosgri Earthquake (HE) ground motion response spectrum as described in NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Power Plant, Units 1 and 2,"Supplement No. 7 (NRC, 1978), and the LTSP ground motion response spectrum as detailed in NUREG-0675, Supplement No. 34 (NRC, 1991). The results indicate 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 HE ground motion and the 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."

The licensee's withdrawal was based on the NRC staff's affirmation that the DCPP seismic qualification basis was the original design basis plus the Hosgri evaluation basis, along with associated analytical methods, initial conditions, etc. Additionally, on March 12, 2012, the NRC letter, "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" (ADAMS Accession Number ML12053A340) was issued to all power reactors and provided DCPP with specific guidance regarding the use of present day methodologies for evaluating seismic hazards. NRC staff confirmed the withdrawal request by letter dated October 31, 2012, ADAMS Accession Number ML12289A076.

As described in the chronology section of your DPO, beginning in 2010, as the senior resident inspector for Diablo Canyon, you raised concerns with your Region IV management regarding the adequacy of PG&E's evaluation of the Shoreline fault zone. You were engaged in the agency's consideration of this issue and participated in two non-concurrences prior to submitting the DPO. On July 19, 2013, you submitted this DPO documenting your concerns. The statement of concerns from your DPO is summarized as follows:

1. The NRC did not enforce the Diablo Canyon Technical Specifications with respect to this seismic issue because the new seismic information showed that required structures, systems and components could be exposed to greater vibratory motion than previously considered for the safe shutdown earthquake.

2. PG&E's operability evaluation following the development of the new seismic information was inadequate because the new seismic information was not compared correctly to the plant's licensing basis.

3. The NRC failed to enforce 10 CFR 50.59 requirements that PG&E obtain an amendment to its license because the new seismic information showed that more than a minimal increase would occur in the likelihood of malfunction of structures, systems and components important to safety.

4. The NRC failed to adequately address the Los Osos and San Luis Bay faults which could produce ground motions in excess of the safe shutdown earthquake ground motion.

The DPO Ad Hoc Review Panel was established on September 3, 2013. The Ad Hoc Review Panel met with you on October 23, 2013, and then periodically throughout its review to further discuss your concerns. The Ad Hoc Review Panel focused its review on the agreed to statement of concerns, as summarized above. The panels' conclusions are summarized for each concern:

1. The panel noted that your DPO was instrumental in identifying the complexity of the range of conditions considered in the DCPP Final Safety Analysis Report Update seismic evaluation. However, the panel concluded that given appropriate comparisons, the potential ground motions from the Shoreline fault zone do not exceed the levels of acceleration considered in the design and licensing of DCPP for required structures, systems and components.

2. The panel noted that this concern is based upon your conclusion that there is only one appropriate evaluation method for new seismic information, specifically to substitute the new seismic information into the original DDE method. The panel concluded that the licensee's evaluation method was acceptable given that the Final Safety Analysis Report Update identifies both the DDE and the Hosgri evaluation.

3. The panel concluded that an amendment to the license was not required because the Shoreline fault zone ground motions do not exceed the levels evaluated in the DCPP design and licensing. This conclusion relies upon the determination that there is more than one appropriate evaluation method for evaluating new seismic information. The Hosgri evaluation methods for structures used higher damping values than the DDE evaluation. Based on the staff's findings in NUREG-0675, Supplement No. 7, the higher damping values used in the Hosgri evaluation are consistent with Regulatory Guide 1.61

and were realistic and acceptable. The panel concluded that substituting the new seismic information into the calculation construct of the DDE would offer little insight as to how the structures, systems, and components would perform because the older analytical techniques were overly conservative and no longer technically justified. The panel found that the licensee's evaluation (i.e., comparison against the Hosgri Evaluation and the LTSP) was an acceptable method. Because the results of the evaluation show that the Shoreline fault zone is bounded by the licensing basis, there is no potential violation of the 10 CFR 50.59 requirements.

4. The panel agreed with you that the NRC staff did not clearly and explicitly consider the potential ground motions from the Los Osos and San Luis Bay fault in the evaluation of the Shoreline Fault zone (ADAMS Accession Number ML12073016). However, the panel noted that both the Los Osos and San Luis Bay faults had previously been evaluated in the LTSP. The Los Osos and San Luis Bay faults were shown to be lower than the ground motions for the Hosgri fault. Additionally, these faults were evaluated from a risk perspective by NRC staff in Research Information Letter 12-01. The panel concluded that the Los Osos, San Luis Bay, and the Shoreline faults do not exceed the level of ground motion already considered in the design and licensing of DCPP.

On May 29, 2014, the Director, Office of Nuclear Reactor Regulation, provided you with his decision on your DPO. The Office Director agreed with the Ad Hoc Review Panel's report with respect to your specific technical concerns. However, he noted that your DPO highlighted the need for the agency to further evaluate how new information on natural hazards should be considered in the regulatory process and his expectation that the work currently underway on the Fukushima Near Term Task Force Recommendations 2.1 and 2.2 would address this issue.

On June 23, 2014, you filed a DPO appeal. In the appeal, you restated your concern that the facility licensing basis, as described in the Final Safety Analysis Report Update must be used as the basis for review of the Shoreline fault zone. You specifically noted concern with the Ad Hoc Review Panel report's conclusions, noting that they were based on the "inappropriate assumption" that a combination of the DDE safety analysis and Hosgri evaluation could be considered the licensing basis. You asserted that the current licensing basis identifies the DDE as the safe shutdown earthquake and that all new ground motions, such as in the Shoreline fault zone, must be evaluated using the assumptions, methodology, load combinations, and acceptance limits associated with the DDE evaluation. Your concerns include the licensee's failure to perform an analysis of the Shoreline fault zone ground motions using the methodology associated with DDE and subsequently, the agency's failure to take appropriate enforcement action to require the analysis to be performed.

EXECUTIVE DIRECTOR FOR OPERATIONS REVIEW AND DECISION

When I received your appeal, I initiated a review of relevant information related to DPO 2013-002. I reviewed a number of documents including, but not limited to, the DPO you originally submitted, the Ad Hoc Review Panel's report dated April 3, 2014, the Office Director's decision regarding your DPO, your appeal of the Office Director's decision, and the Office Director's statement of views on the contended issues in your DPO appeal. To understand the issues fully, I met with members of the Ad Hoc Review Panel on July 28, 2014. I met with you on July 30, 2014, and listened carefully to your points. I also reviewed the additional information you provided at our meeting. My review was focused on the agreed upon issues that you raised in your DPO submittal.

In the appeal, you noted your agreement with the Ad Hoc Review Panel's conclusion that issues raised in the DPO do not result in a significant or immediate safety concern. You also state agreement that the potential ground motions from the Shoreline fault zone do not exceed the levels considered during licensing of the plant. However, you have narrowly defined the licensing basis and approved methodology for seismic evaluation as being limited to the methodology associated with the DDE from the original license application. Based on your exclusion of the Hosgri evaluation from the licensing basis, your appeal reiterates your belief that a license amendment is required to revise the DDE evaluation to the higher ground motions associated with the new seismic information. Additionally, you recommend the agency initiate enforcement action for the failure to take appropriate actions to address the new seismic information associated with the Shoreline fault zone.

I would like to commend you on a package that was well-researched and insightful. Based on my review, discussion with the Ad Hoc Review Panel, and our interview, I agree that there is no significant or immediate safety concern associated with the issue you have raised. However, you have highlighted the complexity of the Diablo Canyon licensing basis as documented in the Final Safety Analysis Report Update, which is a direct result of how the licensing basis was augmented during the original licensing process, between issuance of the construction permit in 1968 and issuance of the operating licensing in 1984. While I appreciate your concern with the clarity of the Final Safety Analysis Report Update, I am unable to arrive at the same conclusion whereby you exclude the Hosgri evaluation and associated methodologies from the licensing basis.

Nevertheless, your questioning attitude and perseverance were key to ensuring that the licensee and staff fully evaluated the implications of the Shoreline fault zone. You correctly noted that the seismic hazard should be evaluated for not only comparison of the ground motion response spectra, but also the plant's design and construction to ensure continued safe operation. In addition to raising awareness of the complexity of the DCPP seismic licensing basis for this specific issue, you have illustrated the need for the agency to ensure there are clear guidelines for staff and licensees regarding how changes in natural hazards should be evaluated for all licensees. This awareness is particularly timely and important as we move forward with the Fukushima Near Term Task Force Recommendations. Specifically, the NRC staff will be considering such concerns as we continue the implementation of Recommendation 2.3, the reevaluation of seismic and flooding hazards; develop the implementation of Recommendation 2.2, "Periodic Confirmation of External Hazards"; and develop the implementation of the additional issue of evaluating other external natural hazards.

A compelling basis for my conclusion is drawn from our meeting on July 30, 2014, when you and I agreed that there is not now nor has there been an immediate or significant safety concern associated with this DCPP issue. That said, you have raised important concerns with our processes which merit further consideration. We will evaluate our licensing process and the manner in which we resolve new seismic hazard information as we complete the Near Term Task Force Recommendations. Lastly, if there are programmatic changes identified to improve our processes, we will take appropriate actions to ensure they are implemented.

In accordance with Management Directive 10.159, a summary of this DPO appeal decision will be included in the Weekly Information Report posted on the NRC's public web site to advise interested employees and members of the public of the outcome.

I want to thank you for bringing your concerns to my attention. Your DPO was well thought out and researched. As you know, our agency relies on its staff members to raise concerns regarding decisions so that they can be properly considered. Your perseverance in raising these concerns demonstrates your dedication to safety that is the foundation of the agency's excellent staff, and I applaud your efforts in this regard. I take concerns such as the ones you raised very seriously and hope that my interactions with you have demonstrated my efforts to consider and fairly evaluate your concerns in making my decision.

cc: D. Dorman, NRR
P. Holahan, OE
M. Sewell, OE
R. Pedersen, OE
M. Case, RES
B. Hill, NRO
R. Bernhard, RIII
M. Dapas, RIV
K. Kennedy, RIV
W. Walker, RIV
M. Galloway, OEDO