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 wave 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.

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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.

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(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 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.

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.

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 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 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.

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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.

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

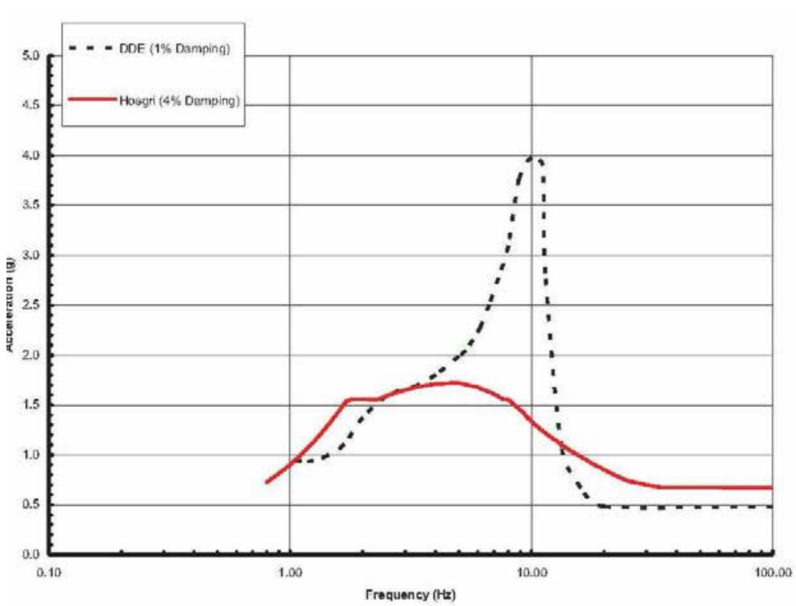

Figure 1, Comparison of the DDE/SSE and the HE Floor Response Spectrum, Containment Elevation 88'

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 ± Hosgri	Table 5.2-7
Faulted - 4	Deadweight + Pressure + Other Pipe Rupture[(f)]	Table 5.2-7

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

HE load combinations and limits were negotiated.

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

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

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TABLE 5.2-5

STRESS LIMITS FOR CLASS A COMPONENTS

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

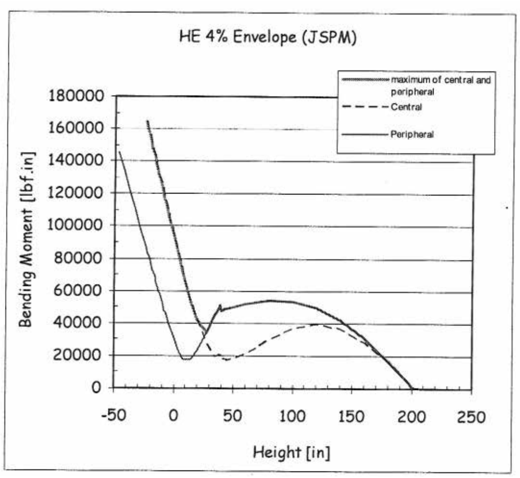

Figure 2
HE Maximum CRDM Bending Moments[17]

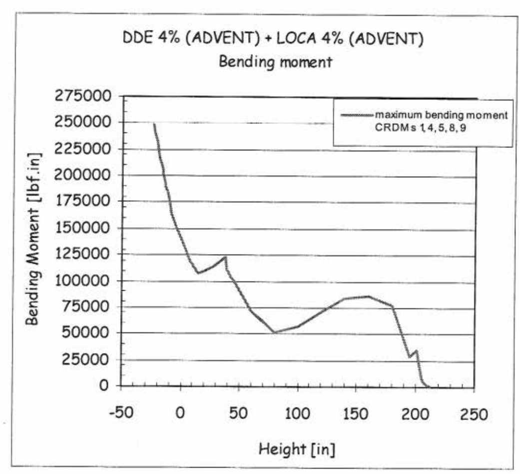

Figure 3
SSE/DDE Maximum CRDM Bending Moments[18]

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.

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

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Unit 2 – Amendment No. ~~135~~, ~~146,~~ ~~160,~~ 203

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 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 wave 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

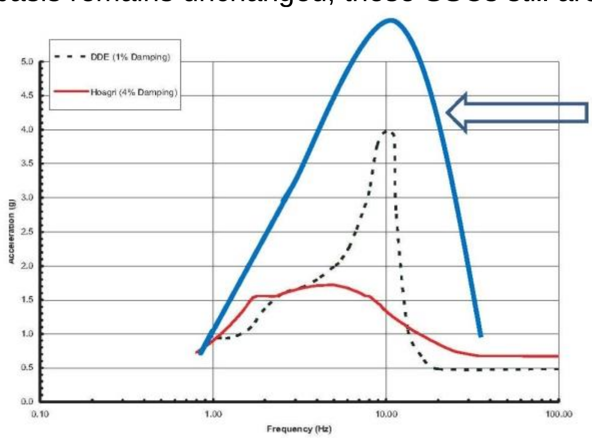

Figure 4
Comparison of the DDE/SSE and the HE Floor Response Spectrum, Containment Elevation 88'

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."*

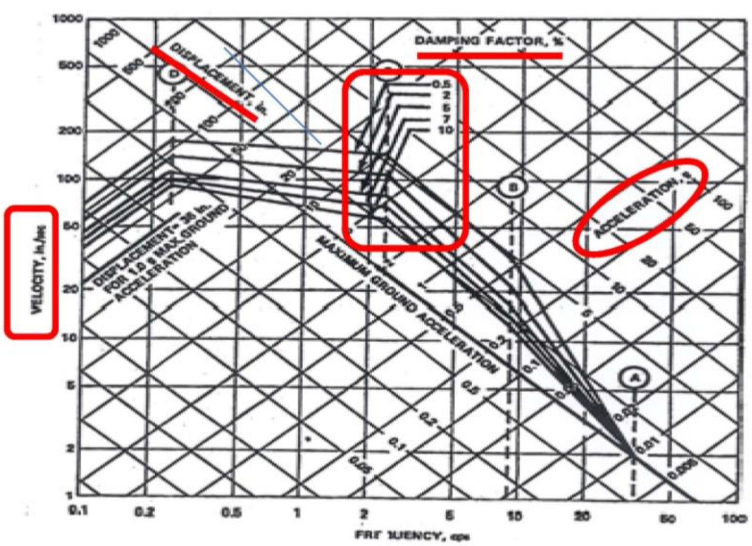

Figure 5
Relationship between Acceleration, Velocity as a Function of Damping[57]

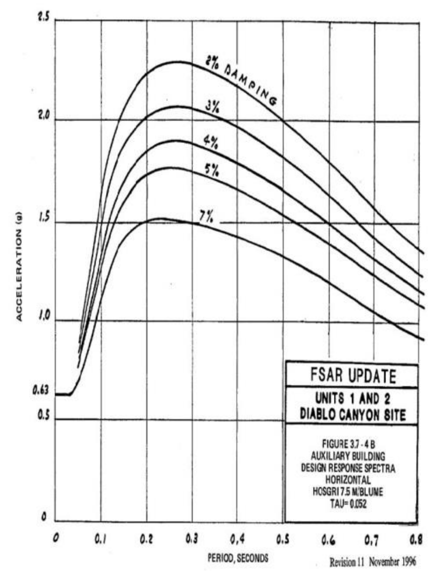

Figure 6
Relationship between Damping & Propagation of Seismic Energy

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