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

September 29, 2010

Mr. John T. Conway
Senior Vice President – Energy Supply
and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
77 Beale Street, Mail Code B32
San Francisco, CA 94105

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: REVISION TO FINAL SAFETY ANALYSIS REPORT UPDATE SECTION 3.7.1.3, "CRITICAL DAMPING VALUES" (TAC NOS. ME4056 AND ME4057)

Dear Mr. Conway:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 208 to Facility Operating License No. DPR-80 and Amendment No. 210 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant, Unit Nos. 1 and 2 (DCPP), respectively. The amendments consist of changes to the Final Safety Analysis Report Update (FSARU) Section 3.7.1.3, "Critical Damping Values," in response to your application dated June 14, 2010, as supplemented by letter dated August 9, 2010.

The amendments approve the Pacific Gas and Electric Company's (the licensee's) request to incorporate a revision to the FSARU Section 3.7.1.3, "Critical Damping Values," to include critical damping values for the seismic design and analysis of the integrated head assembly (IHA) that are consistent with the recommendations of NRC Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1. In addition, the RG 1.61, Revision 1, Table 1 note, allowing the use of a "weighted average" for design-basis safe-shutdown earthquake damping values applicable to steel structures of different connection types, will also be applied to determine the IHA design-basis operating-basis earthquake damping values listed in Table 2 of RG 1.61, Revision 1.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

Alan Wang

Alan Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 208 to DPR-80
2. Amendment No. 210 to DPR-82
3. Safety Evaluation

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

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-275

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. DPR-80

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated June 14, 2010, as supplemented by letter dated August 9, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the plant licensing basis as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-80 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 208, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

    In addition, the license is amended to authorize changes to the Final Safety Analysis Report Updated (FSARU) Section 3.7.1.3, "Critical Damping Values," to include damping values for the seismic design and analysis of the integrated head assembly (IHA) that are consistent with the recommendations of NRC Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1. In addition, the RG 1.61, Revision 1, Table 1 note, allowing the use of a "weighted average" for design-basis safe-shutdown earthquake damping values applicable to steel structures of different connection types, will also be applied to determine the IHA design-basis operating-basis earthquake damping values listed in Table 2 of RG 1.61, Revision 1. The changes to the FSARU shall be as set forth in the enclosure of the licensee's application dated June 14, 2010, as supplemented by letter dated August 9, 2010.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance. The licensee shall submit the changes authorized by this amendment with the next update of the FSARU in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: September 28, 2010

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

PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-323

DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 210
License No. DPR-82

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Pacific Gas and Electric Company (the licensee), dated June 14, 2010, as supplemented by letter dated August 9, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. DPR-82 is hereby amended to read as follows:

    (2) Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan

    The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 210, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

   In addition, the license is amended to authorize changes to the Final Safety Analysis Report Updated (FSARU) Section 3.7.1.3, "Critical Damping Values," to include damping values for the seismic design and analysis of the integrated head assembly (IHA) that are consistent with the recommendations of NRC Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1. In addition, the RG 1.61, Revision 1, Table 1 note, allowing the use of a "weighted average" for design-basis safe-shutdown earthquake damping values applicable to steel structures of different connection types, will also be applied to determine the IHA design-basis operating-basis earthquake damping values listed in Table 2 of RG 1.61, Revision 1. The changes to the FSARU shall be as set forth in the enclosure of the licensee's application dated June 14, 2010, as supplemented on August 9, 2010.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance. The licensee shall submit the changes authorized by this amendment with the next update of the FSARU in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Date of Issuance: September 29, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 208

TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 210 TO FACILITY OPERATING LICENSE NO. DPR-82

DOCKET NOS. 50-275 AND 50-323

Replace the following pages of the Facility Operating License Nos. DPR-80 and DPR-82 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License Nos. DPR-80

REMOVE	INSERT
-3-	-3-

Facility Operating License Nos. DPR-82

REMOVE	INSERT
-3-	-3-

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 208, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Initial Test Program

The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) Technical Specifications (SSER 32, Section 8)* and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 210, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) Initial Test Program (SSER 31, Section 4.4.1)

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 210 TO FACILITY OPERATING LICENSE NO. DPR-82

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-275 AND 50-323

1.0 INTRODUCTION

By application dated June 14, 2010, as supplemented by letter dated August 9, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML101660039 and ML102230075, respectively), Pacific Gas and Electric Company (PG&E, the licensee) requested changes to the Final Safety Analysis Report Update (FSARU) Section 3.7.1.3, "Critical Damping Values," for the Diablo Canyon Power Plant (DCPP), Unit Nos. 1 and 2.

The proposed amendments would allow revision of its licensing basis, as described in the FSARU, to include critical damping values for the seismic design and analysis of the integrated head assembly (IHA) that are consistent with the recommendations of NRC Regulatory Guide (RG) 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1. In addition, the RG 1.61, Revision 1, Table 1 note, allowing the use of a "weighted average" for design-basis safe-shutdown earthquake (SSE) damping values applicable to steel structures of different connection types, will also be applied to determine the IHA design-basis operating-basis earthquake (OBE) damping values listed in Table 2 of RG 1.61, Revision 1.

The supplemental letter dated August 9, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 27, 2010 (75 FR 44025).

2.0 REGULATORY EVALUATION

Pursuant to Section 90 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.90), the licensee, PG&E, submitted a license amendment request (LAR) to allow the use of RG 1.61, Revision 1, critical damping values for the IHA analysis. The licensee requested NRC approval to revise the FSARU Section 3.7.1.3, "Critical Damping Values," to include critical damping values of 4.9 percent for the Design Earthquake (DE), 6.85 percent for the Double Design Earthquake (DDE), and 6.85 percent for the Hosgri Earthquake (HE), respectively, for use in the IHA analysis. Damping values for the IHA are based on the recommendations in RG 1.61, Revision 1, Tables 1 and 2, using a weighted average for "Welded Steel or Bolted Steel with Friction Connections" and "Bolted Steel with Bearing Connections."

General Design Criterion (GDC) 2, "Design bases for protection against natural phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50 requires, in part, that structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. Such SSCs must also be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation and postulated accidents. Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," of 10 CFR Part 50 specifies the requirements for the implementation of GDC 2 with respect to earthquakes.

The DE and DDE for DCPP, Units 1 and 2, correspond to the OBE and SSE, respectively, as described in Appendix A to 10 CFR 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." The HE is also considered by the NRC to be the SSE for DCPP. These seismic events are discussed in FSARU, Section 3.7, and in Section 2.5.2 of NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Supplement No. 7, dated May 26, 1978.

The IHA consists of design Class 1 and design Class 2 components. These components are evaluated using the acceptance criteria of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Division 1 - Subsection NF, "Component Supports," for Classes 1 and 2.

NRC RG 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants," dated March 2007 (ADAMS Accession No. ML070260029) provides damping values acceptable to the staff for the seismic design of nuclear power plants. However, it does not specifically address the damping values applicable to each and every type of component. The licensee requests approval for the use of the calculated weighted average damping values of 4.9 percent for DE and 6.85 percent for both DDE and HE for the IHA analysis. Section 3.7.1 of NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," states that damping values in accordance with those addressed in RG 1.61 are acceptable.

3.0 TECHNICAL EVALUATION

3.1 Items for Which Damping Values are Requested

The licensee requests the use of damping values based on RG 1.61, Revision 1, for the structural dynamic qualification of the IHA for DE, DDE, and HE seismic events.

3.2 Code Evaluation

The components of the IHA consist of design Classes 1 and 2 components. The licensee evaluated these components using the acceptance criteria in Section III, Division 1 of the ASME B&PV Code, Subsection NF, "Component Supports," for Classes 1 and 2.

3.3 Proposed Damping Values

The licensee requests approval for the use of the calculated weighted average damping values of 4.9 percent for DE, and 6.85 percent for both DDE and HE for the structural dynamic analysis of the IHA.

3.4 Basis for Damping Values

The damping values for the seismic design and analysis of the IHA proposed by the licensee are consistent with the recommendations of RG 1.61, Revision 1.

3.5 NRC Staff Evaluation

3.5.1 Damping

The licensee requested approval for the use of the calculated weighted average damping values of 4.9 percent for DE, and 6.85 percent for both DDE and HE for the structural dynamic analysis of the IHA. These damping values are consistent with the recommendations of RG 1.61, Revision 1. Also, the licensee requests the use of the weighted average approach for OBE.

RG 1.61, Revision 0, dated October 1973 (ADAMS Accession No. ML003740213) addresses some broad categories of structures and components such as piping, welded and bolted steel structures, and concrete structures and provides recommended damping values that are acceptable to the NRC. The original damping values in RG 1.61, Revision 0, were based on limited data, and information available in 1973. Since that time, the NRC and industry have been involved in various studies, research work, and testing to predict and estimate damping values of SSCs. As a result, the NRC updated and revised the damping values to reflect more realistic values in 2007. Revision 1 of RG 1.61, issued in March 2007, includes these revised or increased damping values and includes damping values for some additional components such as electrical distribution systems, heating-ventilation-and air conditioning duct systems, and mechanical and electrical components. The damping values in RG 1.61, in both revisions, are acceptable to the NRC staff for use in elastic design of nuclear power plants.

The NRC staff reviewed the LAR and its enclosures and noted the following key design features of the IHA. The IHA is a replacement structure for the existing reactor vessel head service structure. The IHA design integrates all the removable upper reactor vessel head components into one removable structure. The IHA is primarily a bolted steel structure with some welded connections consisting of various components designed to provide cooling for the control rod drive mechanisms (CRDMs), radiation shielding for workers performing activities near the replacement reactor vessel closure head (RRVCH), seismic support for the CRDMs and other IHA components, and to facilitate lifting of the IHA and the RRVCH during refueling outages. As provided in the licensee's letter dated August 9, 2010, in response to the NRC staff's request for additional information (RAI) dated July 28, 2010 (ADAMS Accession No. ML102090540), the materials for the IHA components are carbon, low alloy, and stainless steels, in accordance with the American Society for Testing and Materials (ASTM) specifications. Although the IHA is a new structure that does not have an existing equivalent, the IHA incorporates the functions of the former CRDM seismic support structure, the CRDM ventilation cooling system, and the vessel head lift rig. In response to the staff's RAI, the licensee responded that IHAs of similar design to DCPP Units 1 and 2 are currently installed at other domestic plants, such as Salem Nuclear Generating Station, Unit Nos. 1 and 2, and Turkey Point Nuclear Generating, Unit Nos. 3 and 4. The IHA is a four-story high, approximately 43 feet tall steel structure consisting of more than 10,000 parts assembled together by bolted and welded connections. The IHA has a greater number of bolted connections as compared to the welded connections. In addition, all bolted connections in the IHA are bearing connections and not friction-type connections. The bolted and welded connections are critical for load transfer and energy dissipation during a seismic event.

In Enclosure 2 of the LAR, the licensee included a detailed table that contains the number of connections and the connection types, namely the bolted (bearing or pinned) type or welded type. For the IHA, the licensee listed different connections that transfer loads during a seismic event which amount to a conservative total of 185 connections of which 9 are of the welded type and 176 are of the bolted bearing type. In its RAI dated July 28, 2010, the NRC staff requested that the licensee explain the number of bolted connections for connection #43 in the table, as it was not provided in the LAR. In its response, the licensee stated that it is conservatively accounted for with connection #25, as explained by Note 5 of the table. The staff concludes the clarification by the licensee is acceptable because the connection types are properly and conservatively accounted for in the total number and utilized in the weighted average damping value calculation.

RG 1.61, Revision 1, differentiates between welded steel or bolted steel with friction connections and bolted steel with bearing connection based on the differences in their energy absorbing capabilities. According to RG 1.61, Revision 1, the damping values for welded steel or bolted steel with friction connections are 3.0 percent for the OBE and 4.0 percent for the SSE. The damping values for bolted steel with bearing connections are 5.0 percent for the OBE and 7.0 percent for the SSE.

Table 1 of RG 1.61, Revision 1, for SSE damping values, has a note stating that "for steel structures with a combination of different connection types, use the lowest specified damping value, or as an alternative, use a 'weighted average' damping value based on the number of each type present in the structure." The licensee chose the alternative to use the weighted average damping value, which is acceptable to the NRC staff. However, the same note regarding the weighted average is not specifically listed in RG 1.61, Revision 1, Table 2 for OBE damping values, and the licensee requested this for its use. The staff sought additional information to clarify whether any inelastic analysis methods were utilized in the IHA analysis. In its response dated August 9, 2010, PG&E provided a table of damping values and analysis methods used for horizontal and vertical direction excitations from DE, DDE, HE, and loss-of-coolant accident (LOCA) events, as summarized in Table A, below:

Table A: IHA Seismic and LOCA Load Analysis Method Values Summary

	DE		DDE		HE		LOCA	
	Horizontal	**Vertical**	**Horizontal**	**Vertical**	**Horizontal**	**Vertical**	**Horizontal**	**Vertical**
Analysis Method	USM[1] Response Spectra	USM[1] Response Spectra	USM[1] Response Spectra and Linear Elastic Time History[4]	USM[1] Response Spectra	USM[1] Response Spectra	USM[1] Response Spectra	Response Spectra[2]	Response Spectra[2]
Damping	4.9%	4.9%	6.85%	6.85%	6.85%	6.85%	6.85%[3]	6.85%[3]

Notes:

1 USM Uniform Support Motion (enveloped spectra) was used as input to the response spectra analysis.

2 LOCA response spectra inputs were applied where the IHA is attached to the reactor head.

3 The LOCA computer analysis was performed using 7 percent damping. At locations with the largest stress/load interaction ratios, the LOCA computer analysis results are adjusted upward by a scale factor so the results represent a 6.85 percent damping value. The scale factor is based on the maximum difference between the 6.85 percent and 7 percent damping input response spectra.

4 The time history analysis method was used to determine DDE loads for the structural qualification of connection SCN-24.

Based on a review of the summary information in Table A, the NRC staff determined that no inelastic analysis method was used. The damping values provided in RG 1.61 are for elastic dynamic analysis and design of SSCs. Although the note regarding weighted average is not specifically included in RG 1.61 for OBE damping values, the staff considers it is reasonable to utilize the weighted average approach for OBE damping as long as (i) inelastic analysis methods are not used, and (ii) the number of connections of welded type and bolted bearing type are properly and realistically accounted for in the weighted average calculation for damping. The staff's review determined that the licensee complied with both of the items mentioned above, and, therefore, the use of weighted average for the OBE damping value calculation is acceptable. Utilizing the weighted average approach for OBE and SSE, the licensee computed damping values of 4.9 percent for OBE, and 6.85 percent for SSE for use in the analysis of the IHA.

The NRC staff notes that the use of damping values for steel structures as described in RG 1.61, Revision 1, is acceptable for the IHA, since the IHA is primarily a tall steel structure with different connection types. Based on a review of the LAR, the RAI responses, and above evaluation, the staff concludes that the calculated damping values of 4.9 percent for OBE (DE), and 6.85 percent for SSE (DDE and HE) for the analysis of the IHA at DCPP are reasonable and acceptable because they are based on a weighted average approach, as described in RG 1.61, Revision 1. The staff also concludes that the licensee has appropriately considered the number and type of bolted, bearing, and welded connection types in the IHA steel structure for weighted average damping calculation.

In its RAI dated July 28, 2010, the NRC staff requested additional information regarding the non-traditional cross sections of the duct work of IHA structure. In its response dated August 9, 2010, the licensee provided two detailed figures showing the shapes and cross sections of the duct work. The staff requested information on the materials used for the various components of the IHA. In its response, the licensee provided a list of the materials for design Class I components (plates and bars, shapes and tube steel) and design Class II/I components (plates, bars, wires, and forgings, shapes and tube steel, and bolts), which are acceptable, since they are all carbon, low alloy, and stainless steels, that meet the ASTM specifications.

3.5.2 IHA Analysis

The IHA is a vertically standing steel structure bolted to three support lugs on the RRVCH and pinned to three lift lugs on the reactor vessel head. In addition to these six attachment points on the RRVCH, there are eight seismic tie rods pinned to the IHA seismic ring beam at the refuel floor elevation. On the cavity wall side, these tie rods are pinned to wall lugs that are an integral part of plates that are bolted to the containment wall. All eight seismic tie-rod connections on both ends of the tie rods are pin connections. A finite element model (FEM) was created by the licensee for the IHA. The licensee performed a structural dynamic finite element analysis of the IHA using the FEM and the critical damping values that were computed and were determined to be consistent with RG 1.61, Revision 1, as summarized in Table A, above.

The IHA was evaluated for the DE, DDE, and HE seismic events. The results from the seismic analyses, that is, the loads, stresses, and displacements, were combined using the appropriate load combinations with those from the other applicable loads, such as deadweight and pressure to determine the total loads, stresses, and displacements for the various components of the IHA. The resulting IHA loads and stresses were evaluated by the licensee for acceptance using the ASME B&PV Code, Section III, Division 1 - Subsection NF, "Component Supports," 2001 Edition through 2003 Addenda, and stress ratios were computed. The stress ratio is the calculated load or stress divided by the allowable value. A value of a stress ratio equal to 1.0 represents an acceptable load or a stress with the required margin per the applicable Code. Stress ratios below 1.0 represent additional margin beyond that required by the applicable Code.

The NRC staff reviewed the stress ratio summary tables containing the component, controlling load combination, and stress ratio for design Class I linear components, design Class I plate components, design Class II/I linear components, design Class II/I plate components, and connections between design Class I components. The staff requested clarification on the symbols used in the stress ratio summary tables. In its response, the licensee enhanced the stress ratio tables by including a description of the symbols used.

The components with the highest stress ratios are listed below, and the NRC staff concludes they are acceptable because the stress ratios are less than 1.0. This indicates that they meet the ASME B&PV Code, Subsection NF, acceptance criteria limits with margin, for all five groups of components, as listed in Table B, below:

Table B: Maximum Stress Ratio for IHA Components

Component Description	Controlling Load Combination	Stress Ratio	Comment
(Design Class I Linear Components) Lift-Rods	DL+P+/- SRSS (DDE+MI)	0.96	Acceptable
(Design Class I Plate Components) Cable bridge vertical support plates in stationary section	DL+P+/- (DE)	0.95	Acceptable
(Design Class I I/I Linear Components) Monorail	DL+P+ML	0.88	Acceptable
(Design Class I I/I Plate Components) Baffle cover	DL+P+/- HE	0.51	Acceptable
(Connections between Design Class I Components) Connection of cable bridge support to Walkway: Bolts	DL+P+T+/- DE	0.97	Acceptable

Symbols:

DL:	Dead Load	LOCA:	Loss of Coolant Accident Load
DE:	Design Earthquake Load	T:	Temperature Load
P:	Pressure Load	MI:	Missile Impact Load
DDE:	Double Design Earthquake Load	SRSS:	Square-Root-of-the-Sum-of-the-Squares Load Combination Method
HE:	Hosgri Earthquake Load		

The NRC staff also requested information on the design class of the various IHA components in relation to ASME B&PV Code Section III, Division 1 - Subsection NF Class. In its response dated August 9, 2010, the licensee provided the following clarification:

> All systems and components of DCPP are classified according to their importance in the prevention and mitigation of accidents. Those items vital to safe shutdown and isolation of the reactor, or whose failure might cause or increase the severity of a LOCA, or result in an uncontrolled release of excessive amounts of radioactivity, are designated as Design Class I. Those items important to the reactor operation, but not essential for safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity, are designated as Design Class II.
>
> For the design and analysis of the IHA, Design Class I corresponds to a safety-related component and Design Class II corresponds to a non-safety related component. Design Class II components are seismically qualified to preclude adverse effects on Design Class I components per DCPP FSARU Section 3.7.3.13.

Design Class I components were evaluated using acceptance criteria corresponding to the ASME B&PV Code, Section III, Division 1 - Subsection NF, "Component Supports," 2001 Edition through 2003 Addenda, Class 1 (ASME NF Class 1). For Design Class II plate and shell components, the loads and stresses were evaluated using the acceptance criteria corresponding to ASME NF Class 2. Design Class II linear component loads and stresses were evaluated using the acceptance criteria corresponding to ASME NF Class 1, since design by analysis for Class 2 linear supports uses the same acceptance criteria as Class 1 per ASME NF paragraph 3350. The staff reviewed the licensee's clarification on design classes of the various IHA components and concludes it is adequate.

The licensee combined the seismic results from the IHA analysis with other applicable stresses to obtain the total stress for each applicable load combination. The combined stresses in IHA were determined at each critical location. The licensee evaluated these stresses in accordance with the Subsection NF rules of the ASME B&PV Code, Section III, Division 1, 2001 Edition through 2003 Addenda, and demonstrated their acceptability.

The licensee has adequately demonstrated, by finite element analysis with the IHA using damping values consistent with RG 1.61, Revision 1, that the stress ratios for the various IHA components meet the acceptance criteria of ASME B&PV Code, Section III, Division 1 - Subsection NF.

The structural dynamic analysis performed by the licensee confirmed the IHA's ability to function under a postulated seismic disturbance, combined with other applicable loadings, while maintaining resulting stresses under the ASME B&PV Code Section III allowable values. The NRC staff reviewed the results of the IHA analysis, summarized in Table B above, and concludes the results are acceptable because they demonstrate margin compared to the applicable ASME Code allowable values as indicated by the stress ratios being less than one.

Based on the above, the NRC staff concludes that the licensee has provided an adequate technical basis and has demonstrated, by the analysis results of the IHA, that the ASME B&PV Code limits will be met. Therefore, authorizing this request for the use of RG 1.61, Revision 1, weighted average damping values for steel structures with different connection types for the IHA analysis will not adversely impact the health and safety of the public.

The NRC staff concludes that compliance with the requirements of ASME B&PV Code, Section III, Division 1 - Subsection NF Classes 1 and 2, along with the use of the weighted average damping values of 4.9 percent for OBE (DE) and 6.85 percent for SSE (DDE and HE) in the structural design and analysis qualification of the IHA, provides reasonable assurance of maintaining an acceptable level of quality and safety and is, therefore, acceptable. Based on the staff's evaluation, as described above, the use of the RG 1.61, Revision 1 damping values for the DCPP IHA is acceptable for DE, DDE, and HE events.

The NRC staff has reviewed the licensee's supporting technical information and the available margins in the results of the IHA structural dynamic analysis, as provided in the LAR and supplemented in the responses to the RAI for the proposed LAR, pertaining to the critical damping values for the IHA. The NRC staff concludes that the proposed critical damping values for the IHA analysis are in accordance with RG 1.61, Revision 1. The licensee's proposed damping values for the IHA analysis provides reasonable assurance that the IHA, as designed and constructed, will perform its intended safety functions. Therefore, the requested critical damping values for the IHA, based on the RG 1.61, Revision 1 weighted average approach, for steel structures with different connection types, are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the California State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on July 27, 2010 (75 FR 44025). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Basavaraju

Date: September 29, 2010

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Alan Wang, Project Manager
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:
1. Amendment No. 208 to DPR-80
2. Amendment No. 210 to DPR-82
3. Safety Evaluation

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