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

July 30, 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: CRITICAL DAMPING VALUES FOR CONTROL ROD DRIVE MECHANISM PRESSURE HOUSINGS (TAC NOS. ME2995 AND ME2996)

Dear Mr. Conway:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 207 to Facility Operating License No. DPR-80 and Amendment No. 209 to Facility Operating License No. DPR-82 for the Diablo Canyon Power Plant, Unit Nos. 1 and 2, 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 December 14, 2009, as supplemented by letters dated April 23, June 11, and July 2, 2010.

The amendments approve the Pacific Gas and Electric Company's request to incorporate a revision to the FSARU Section 3.7.1.3 to allow for the use of a damping value of 5 percent of the critical damping value for the structural dynamic qualification of the control rod drive mechanism pressure housings on the replacement reactor vessel head for the design earthquake, the double design earthquake, the Hosgri earthquake, and the loss-of-coolant accident loading conditions.

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. 207 to DPR-80
2. Amendment No. 209 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. 207
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 December 14, 2009, as supplemented by letters dated April 23, June 11, and July 2, 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 to authorize changes to the Final Safety Analysis Report Update (FSARU) Section 3.7.1.3, "Critical Damping Values," to reflect the adoption of a damping value of 5 percent of the critical damping value for the structural dynamic qualification of the control rod drive mechanism pressure housings on the replacement reactor vessel head for the design earthquake, the double design earthquake, the Hosgri earthquake, and the loss-of-coolant accident loading conditions. The changes to the FSARU shall be as set forth in the enclosure of the licensee's application dated December 14, 2009.

3. The license amendment is effective as of its date of issuance and shall be implemented within 180 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

Attachment:
Changes to the Facility
Operating License No. DPR-80
and Technical Specifications

Date of Issuance: July 30, 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. 209
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 December 14, 2009, as supplemented by letters dated April 23, June 11, and July 2, 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 to authorize changes to the Final Safety Analysis Report Update (FSARU) Section 3.7.1.3, "Critical Damping Values," to reflect the adoption of a damping value of 5 percent of the critical damping value for the structural dynamic qualification of the control rod drive mechanism pressure housings on the replacement reactor vessel head for the design earthquake, the double design earthquake, the Hosgri earthquake, and the loss-of-coolant accident loading conditions. The changes to the FSARU shall be as set forth in the enclosure of the licensee's application dated December 14, 2009.

3. The license amendment is effective as of its date of issuance and shall be implemented within 180 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

Attachment:
Changes to the Facility
Operating License No. DPR-82
and Technical Specifications

Date of Issuance: July 30, 2010

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-80

AND AMENDMENT NO. 209 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 December 14, 2009, as supplemented by letters dated April 23, June 11, and July 2, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML093580092, ML101250466, ML101650084, and ML101940325, 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, Unit Nos. 1 and 2 (DCPP).

Damping is the means by which the response motion of a structural system is reduced as the result of energy losses due to material and structural sources. Damping values are used in the analyses for the structural dynamic qualification of structures. The proposed amendments would approve PGE's request to incorporate a revision to the FSARU Section 3.7.1.3 to allow for the use of a damping value of 5 percent of the critical damping value for the structural dynamic qualification of the control rod drive mechanism (CRDM) pressure housings on the replacement reactor vessel head for the design earthquake (DE), the double design earthquake (DDE), the Hosgri earthquake (HE), and the loss-of-coolant accident (LOCA) loading conditions. The CRDM pressure housings are needed to maintain the integrity of the reactor coolant pressure boundary.

The supplemental letters dated April 23, June 11, and July 2, 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 March 23, 2010 (75 FR 13790).

## 2.0 REGULATORY EVALUATION

Pursuant to Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), PG&E submitted a license amendment request (LAR) to change the critical damping values from 3 percent and 4 percent to 5 percent for evaluation of the structural dynamic qualification of the CRDM pressure housings on the replacement reactor vessel head for the DE, DDE, HE, and the LOCA loading conditions.

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 that Seismic Category I nuclear power plant structures, systems, and components (SSCs) important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, tornados, hurricanes, floods, tsunami, and seiches without lose of capability to perform their safety functions. SSCs shall must also be designed to accommodate appropriate combinations of the effects of normal and accident conditions. The regulations in 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," specify the requirements for the implementation of GDC 2 with respect to earthquakes.

The CRDMs are American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1 components. The regulations in 10 CFR 50.55a, "Codes and standards," require that components which are part of the reactor coolant pressure boundary meet the requirements for Class 1 components in Section III of the ASME Code. Accordingly, the CRDMs are required to be designed and qualified to meet the requirements of the ASME Code, Section III for all design, operating, and accident conditions.

NRC Regulatory Guide (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 NRC staff for the seismic design of nuclear power plants. However, it does not specifically address the damping values applicable to the CRDMs. In its FSARU, the licensee references RG 1.61, Revision 0, dated October 1973 (ADAMS Accession No. ML003740213). The design basis analysis of the CRDMs used 3 percent and 4 percent damping. The licensee requests approval for the use of a 5 percent damping value for the replacement CRDM analysis.

Section 3.7.1 of NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," allows for damping values higher than those in RG 1.61 if a sufficient basis is provided by testing or experiments. RG 1.61 also states that damping values higher than those provided in the regulatory guide may be used if documented test data support the higher values.

As described in the various plant Final Safety Analysis Reports (FSARs), several operating pressurized-water reactor (PWR) plants have used a 5 percent critical damping value in the past for CRDM analysis and qualification.

## 3.0 TECHNICAL EVALUATION

### 3.1 CRDMs Damping Values

The CRDMs are ASME Code Class 1 components. The regulations in 10 CFR 50.55a(c) require that components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Code. The licensee requests a change in its design basis to allow for the use of a damping value of 5 percent of the critical damping value from 3 percent and 4 percent used in the analysis for the original CRDMs for the structural dynamic qualification of the CRDM pressure housings on the replacement reactor vessel head for DE, DDE, HE, earthquakes and LOCA loading conditions.

### 3.2 Licensee Basis for Change

In its letter dated December 14, 2009, the licensee stated in its LAR that a 5 percent damping value can be used for the CRDM pressure housings, based on the following:

(1) No explicit guidance is available on damping values for use in the analysis of CRDMs in RG 1.61, Revision 0 or Revision 1. The RG allows the use of higher damping values, provided test data supports such higher values.

(2) Past precedents: Several plants, such as Watts Bar Nuclear Plant (Watts Bar) and Comanche Peak Nuclear Power Plant (CPNPP), of design and construction similar to DCPP with similar CRDMs have used 5 percent damping in CRDM analyses.

(3) The Westinghouse Electric Company LLC topical report WCAP-7921-AR, "Damping Values of Nuclear Power Plant Components," dated May 1974, that was previously reviewed and approved by the NRC staff in 1974, includes supporting damping values based on test data for the KEP-L-105 model CRDM. The mean value of damping recommended was 8 percent with gap or clearance at the upper supports.

(4) The critical damping values of 3 percent and 4 percent used in the design basis analysis are conservative relative to the actual tests (WCAP-7921 AR).

Therefore, a more realistic critical damping value of 5 percent for the CRDM pressure housings is proposed for use in the CRDM pressure housings analysis associated with the reactor vessel head replacement.

### 3.3 NRC Staff Evaluation

The licensee requested a change from lower damping values of 3 percent and 4 percent of critical damping used in the analyses for the original CRDM to 5 percent of the critical damping ratio for the replacement CRDM pressure housings based on the rationale that: (1) the CRDMs are not specifically addressed in RG 1.61, (2) RG 1.61 allows the use of damping values from

documented test data, and (3) there are precedents for the use of a 5 percent damping value for the CRDMs by other operating nuclear plants.

3.3.1 Past Precedents

Several operating Westinghouse PWRs of design and vintage similar to DCPP, such as CPNPP and Watts Bar, have qualified their CRDMs using the 5 percent damping value on the basis of similarity between the Type L-106A CRDM and the tested CRDM. The NRC staff has concluded that the original CRDM (Type L-106A) at DCPP, as well as the replacement CRDM (AREVA), are very similar to the Westinghouse-tested CRDM (Type L-105). The NRC staff has reviewed the FSARs for operating PWR plants, such as CPNPP, Watts Bar, Callaway Plant, Wolf Creek Generating Station, South Texas Project, and Seabrook, which indicates the prior use of a 5 percent damping value for the CRDM analysis. Therefore, the 5 percent damping value has been used at several other plants with equivalent CRDMs and seismic support structures and is a conservative value based on the testing performed by Westinghouse. Based on the above, the NRC staff concludes that the licensee's use of past precedents to support its application for a license amendment to use a 5 percent damping value for the CRDM analysis is acceptable.

3.3.2 Test Data Basis

The NRC staff reviewed the structural schematic of the CRDM provided in the LAR and notes the following:

1) The CRDM housings consist of a nozzle from the reactor head, latch housing, rod travel housing, control rod drive coil assemblies, digital rod position indicator coil stacks, and seismic restraint plates with gaps or clearances.

2) CRDMs are not typically what would be considered piping systems.

3) RG 1.61, Revision 0, addresses some broad categories of structures and components such as piping, welded and bolted steel structures and concrete structures and provides recommended damping values.

4) RG 1.61, Revision 1 includes revised or increased damping values and includes damping values for some additional components such as electrical distribution systems; heating, ventilation, and air conditioning (HVAC) duct systems; and, mechanical and electrical components.

Both the SRP and RG 1.61 allow the use of damping values from documented test data. The NRC staff agrees that the use of a realistic damping value from documented test data is an acceptable approach. The NRC staff reviewed the licensee's basis for the applicability of the data pertaining to transient testing of CRDMs conducted by Westinghouse as discussed in the topical report, WCAP-7921-AR. The tests were performed on a full-size type KEP-L-105 CRDM with its seismic support system at an internal pressure of 2,250 pounds per square inch and a temperature of 400 degrees Fahrenheit (°F). The report states that the strongest of all effects on damping is the clearance at the supports with higher gap contributing to higher damping.

With a gap or a clearance of 0.06 inches between the seismic support plates, the mean of the measured damping was approximately 8 percent. Based on the above, the NRC staff concludes that the use of 5 percent damping value for the analysis of the CRDM pressure housings at DCPP is reasonable and acceptable as it is bounded by documented test data in WCAP-7921-AR. Even though these components are not explicitly addressed in RG 1.61, the NRC staff concludes that the use of damping values corresponding to piping in RG 1.61 is conservative for CRDMs, based on the WCAP-7921-AR data.

3.3.3 Similarity Evaluation

In the LAR, the licensee provided a similarity evaluation that compared DCPP's original CRDM (Type L-106A) to the AREVA replacement CRDM. The evaluation compared parameters such as gap at the seismic support plates, length, nozzle material, latch housing and rod travel housing, geometric section properties, operating conditions, and appurtenances. Based on a review of the similarity evaluations submitted by the licensee, the NRC staff agrees that the design of the AREVA-supplied CRDMs and supporting structure specified is an equivalent replacement for the original CRDMs (Type L-106A) in fit, form, and function. In comparison with the original CRDMs, the as-built configuration of the AREVA-supplied replacement CRDMs provide the similarity in geometry, seismic support elevation, and physical interfaces required to maintain almost identical structural characteristics. The most significant observation from the CRDM tests was that the seismic support gap at the support plate has the most pronounced influence on the damping with higher gaps contributing to higher damping. The NRC staff notes that the replacement AREVA CRDM design provides a cold gap of 0.070 inches and a hot gap of 0.100 inches between the CRDM seismic support plates. The NRC staff also noted that the damping associated with the replacement AREVA CRDM is expected to be higher than the damping observed in the Westinghouse tests because the seismic support gap, which is the most critical parameter that positively influences damping, is larger than that in the Westinghouse tests.

The NRC staff notes, however, that the similarity evaluation described above in the licensee's LAR was based on a comparison between the original CRDM (Type L-106A) and the replacement CRDM (AREVA CRDM), and not between the CRDM used in tests (Type L-105) (as summarized in WCAP-7921-AR) and the replacement CRDM (AREVA CRDM). Therefore, the NRC staff sought additional information requesting a similarity evaluation comparing the CRDM (Type L-105) tested by Westinghouse and the replacement or AREVA CRDM. In its response dated July 2, 2010, to the NRC staff's request for additional information (RAI), the licensee provided a schematic showing a comparison of lengths, cross-sectional properties, materials, appurtenances, and operating parameters for the CRDM tested by Westinghouse and the replacement CRDM. The NRC staff found very minor changes in cross sectional properties, a difference in the overall length of 232 inches to 255 inches for the replacement CRDM versus 215 inches for the tested CRDM. The differences have negligible effect on the damping except for the support gap. The seismic support gap that has dominant effect on damping, however, is larger in the replacement CRDM with a cold gap of 0.070 inches and a hot gap of 0.100 inches, which is beneficial as compared to 0.060 inches maximum in the Westinghouse-tested CRDM. Based on the above, the NRC staff agrees with the licensee's conclusions that 1) the design differences are insignificant, 2) there are favorable design improvements, such as the use of

higher gaps at the seismic support, 3) the design uses of better materials and better joints, and 4) the differences in design do not affect the damping significantly.

3.3.4 Analysis Results Summary Review

In its letter dated April 23, 2010, in response to the NRC staff's RAI requesting a summary of the results of analysis for the replacement CRDM as well as the original CRDM along with the margins, the licensee submitted tables of calculated stress and fatigue usage factors along with the corresponding allowable values and interaction ratios for various CRDM pressure housing components. This included rod travel housing, latch housing, CRDM nozzle between top of head and dissimilar metal weld, and nozzle J–groove weld. By letter dated June 11, 2010, as supplemented by letter dated July 2, 2010, the licensee provided a revision to its RAI responses because it performed a reanalysis of the CRDMs for the HE horizontal direction loads. The supplements to the RAI responses included a table of the CRDM seismic analysis method and damping values used for the original and replacement CRDMs, as summarized in Table A, provided below. The response to the RAIs also included tables of calculated stresses, allowable values, fatigue usage factors, and interaction ratios for the original and replacement CRDM components, as shown in Table B, also provided below.

**Table A: DCPP 1-2: CRDM Seismic Analysis Method and Damping (% Critical) Values**

| | | DE | | DDE | | HE | |
|---|---|---|---|---|---|---|---|
| | | **Horizontal** | **Vertical** | **Horizontal** | **Vertical** | **Horizontal** | **Vertical** |
| Original CRDM*** | Analysis Method | Non-linear Time History* | Static Equivalent | Non-linear Time History | Static Equivalent | ** | Static Equivalent |
| | Damping | 3% | N/A | 3% | N/A | 4% | N/A |
| Replacement CRDM | Analysis Method | Non-linear Time History | USM Response spectra | Non-linear Time History | USM Response spectra | USM Response spectra | USM Response spectra |
| | Damping | 5% | 5% | 5% | 5% | 5% | 5% |

Notes:
* DDE analysis used for DE using DE allowable
** DDE input response spectra envelopes the HE input response spectra
Static Equivalent method used for vertical direction as the CRDM is rigid in vertical direction.
Damping is not applicable (N/A) to this methodology.
USM: Uniform Support Motion (Enveloped Response Spectra)
*** The analyses for the Original CRDM refers to the current design-basis analyses prior to the replacement of the reactor vessel head.

The LOCA analysis for the existing CRDM nozzle between the top of the head and the dissimilar metal weld is based on an enveloping value of the large-break LOCA moments for two 4-loop plants' moment loading. The LOCA axial and shear loads are based on the worst large-break LOCA and the safe-shutdown earthquake loads from a sampling of Westinghouse plants. This is conservative because only small breaks, based on leak-before-break, are required for DCPP.

The LOCA analysis for the replacement CRDM is based on a 5 percent damping response spectra that were generated for the location where the CRDM attaches to the reactor vessel closure head, based on the results of the LOCA.

**Table B: Maximum Interaction Ratios for Replacement AREVA CRDM & Original CRDM**

| | Load Combination | ASME Classification | Calculated Value | Allowable value | I.R. | Comment |
|---|---|---|---|---|---|---|
| **Replacement AREVA CRDM (5% Damping)** | | | | | | |
| Rod Travel Housing | DE | PL+Pb+Q | 39.74 ksi | 48.6 ksi | 0.82 | Acceptable |
| Rod Travel Housing | DE | CUF | 0.004 | 1.0 | 0.004 | Acceptable |
| Latch Housing | DDE+LOCA | PL+Pb | 54.74 ksi | 58.3 ksi | 0.94 | Acceptable |
| Latch Housing | DE | CUF | 0.033 | 1.0 | 0.033 | Acceptable |
| CRDM Nozzle * | DE | PL+Pb+Q | 66.77 ksi | 69.9 ksi | 0.96 | Acceptable |
| CRDM Nozzle * | DE | CUF | 0.003 | 1.0 | 0.003 | Acceptable |
| CRDM Nozzle ** | DE | PL+Pb+Q | 66.77 ksi | 69.9 ksi | 0.96 | Acceptable |
| CRDM Nozzle ** | DE | CUF | 0.282 | 1.0 | 0.282 | Acceptable |
| **Original CRDM L-106A** | | | | | | |
| Rod Travel Housing | DDE | Moment | 64 in-kip | 73 in-kip | 0.88 | Acceptable |
| Latch Housing | DDE | Moment | 140 in-kip | 670 in-kip | 0.21 | Acceptable |
| CRDM Nozzle * | DE | Pm+Pb | 23.67 ksi | 23.98 ksi | 0.99 | Acceptable |
| CRDM Nozzle * | DE | CUF | 0.002 | 1.0 | 0.002 | Acceptable |
| CRDM Nozzle ** | DE | PL+Pb+Q | 50.21 ksi | 69.9 ksi | 0.72 | Acceptable |
| CRDM Nozzle ** | DE | CUF | 0.191 | 1.0 | 0.191 | Acceptable |

Notes:
Interaction Ratio (I.R.) = Calculated Value / Allowable Value)
I.R. ≤ 1 indicates acceptability
* CRDM Nozzle area between top of head and dissimilar metal weld
** CRDM Nozzle (J-Groove Weld)
CUF: Fatigue Cumulative usage factor for DE load combination
Pm = Primary membrane stress; PL=Local primary membrane stress; Pb=Primary bending stress; Q=Secondary stress

### 3.3.5 NB-3200 Class 1 Analysis and Code Reconciliation

Increasing the damping value reduces the dynamic response of a structure. However, if the higher damping value is justifiable and realistic, then the use of a higher damping value is acceptable. The licensee has adequately demonstrated that the use of a 5 percent critical damping ratio for the ASME NB-3200 Class 1 analysis of the replacement CRDM pressure housings is reasonable based on the following: (a) precedents for use of a 5 percent damping value, (b) a regulatory position as stated in RG 1.61 and the SRP that allows documented test data for the use of a realistic damping value, (c) testing done by Westinghouse for Type L-105 CRDM as reported in WCAP-7921-AR, and (d) comparison and similarity evaluation of Type L-105 CRDM with the AREVA replacement CRDM as well as its seismic support gap, a major parameter that has a predominant effect on damping, and (e) margin in the CRDM analysis results.

The licensee combined the seismic stresses from CRDM analysis with other applicable stresses to obtain the total stress for each applicable load combination. The combined stresses were determined at each critical location along the length of the CRDM assembly. These stresses were then evaluated in accordance with the NB-3200 Class 1 rules of the ASME Code, Section III, 2001 Edition through 2003 Addenda, and showed acceptability.

The structural dynamic analysis performed by the licensee confirmed the CRDMs' ability to function under a postulated seismic disturbance or LOCA while maintaining resulting stresses under the ASME Code, Section III allowable values. The NRC staff concludes this proposed change to the damping value is acceptable based on the following primary considerations: (1) the ASME Code requirements continue to be met, (2) there are precedents for the use of a 5 percent damping value at several operating PWRs, and (3) the support gap is larger than that used in the CRDM tests.

The NRC staff reviewed the summary of results of the CRDM analysis, as summarized in Table B and concludes the results are acceptable because they demonstrate margin compared to the applicable ASME Code allowable values, as indicated by the interaction ratios being less than one. The staff also noted that the licensee satisfactorily reconciled the ASME Code used in the analysis of the replacement AREVA CRDM (2001 Edition through 2003 Addenda) to the ASME Code used in the original CRDM analysis (1965 Edition through Summer 1966 Addenda for Unit 1 and 1968 Edition for Unit 2).

Based on the above, the NRC staff concludes that there is adequate technical justification provided by the licensee and, therefore, increasing the critical damping value from 3 percent and 4 percent to 5 percent will have a minimal impact on the ability of the replacement CRDMs to maintain the reactor coolant pressure boundary. Further, the licensee has provided an adequate technical basis and has demonstrated by analysis of the replacement CRDM, with the analysis results meeting the ASME Code limits. Therefore, authorizing this request for a change or increase in damping value for the replacement CRDMs will not adversely impact the health and safety of the public and is, therefore, acceptable.

The NRC staff concludes that compliance with ASME Code, Section III, Class 1 requirements along with the use of a 5 percent damping value in the structural design and analysis qualification of the CRDM pressure housings provides a reasonable assurance of maintaining an acceptable level of quality and safety. Based on the staff's evaluation, as described above, the use of 5 percent critical damping value for the DCPP replacement CRDMs is acceptable for DE, DDE, HE, and LOCA events.

The NRC staff has reviewed the licensee's supporting technical information and the available margins in the results of the CRDM structural dynamic analysis as provided in the LAR and the responses to the requests for additional information for the proposed LAR, pertaining to a change in the critical damping values from 3 percent and 4 percent to 5 percent for the CRDM pressure housings. The licensee's proposed higher damping for CRDM provides a reasonable assurance that the CRDM pressure housings as designed and constructed, will perform their intended safety functions. Based on the above, the requested increase in critical damping value for CRDM pressure housings is 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 March 23, 2010 (75 FR 13790). 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: July 30, 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

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