May 18, 2005

Mr. Gregory M. Rueger
Senior Vice President, Generation and
Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P. O. Box 3
Avila Beach, CA 93424

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NO. 1 - REVIEW OF STEAM GENERATOR TUBE INSPECTION REPORTS FOR THE 2004 REFUELING OUTAGE (TAC NO. MC4433)

Dear Mr. Rueger:

By letters dated May 10, 2004 (available in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML041400030), September 7, 2004 (ADAMS Accession No. ML042580372), and March 10, 2005 (ADAMS Accession No. ML050750021), Pacific Gas and Electric Company (PG&E) submitted reports summarizing the results of inspections of the Diablo Canyon Power Plant (DCPP), Unit No. 1 steam generator tubes performed during the 2004 twelfth refueling outage (1R12). Additional information concerning these inspections, that was discussed in a conference call with PG&E on April 14, 2004, was summarized by the Nuclear Regulatory Commission (NRC) staff in a letter dated July 7, 2004 (ADAMS Accession No. ML041900024).

As discussed in the enclosed safety evaluation, the staff concludes that you have provided the information required by their technical specifications. In addition, the staff did not identify any technical issues that warrant follow-up action at this time.

If you have any questions regarding this matter, please contact me at (301) 415-8439.

Sincerely,

/RA/
Girija S. Shukla, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-275

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE STEAM GENERATOR INSPECTION REPORTS

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON POWER PLANT, UNIT NO. 1

DOCKET NO. 50-275

1.0 INTRODUCTION

By letters dated May 10, 2004 (available in the Agencywide Documents Access and Management System (ADAMS) under Accession No. ML041400030), September 7, 2004 (ADAMS Accession No. ML042580372), and March 10, 2005 (ADAMS Accession No. ML050750021), Pacific Gas and Electric Company (PG&E or the licensee) submitted reports summarizing the results of inspections of the Diablo Canyon Power Plant (DCPP), Unit No. 1 steam generator (SG) tubes performed during the 2004 twelfth refueling outage (1R12). Additional information concerning these inspections, that was discussed in a conference call with PG&E on April 14, 2004, was summarized by the NRC staff in a letter dated July 7, 2004 (ADAMS Accession No. ML041900024).

DCCP Unit 1 has four Westinghouse model 51 SGs. Each SG contains approximately 3400 mill annealed Alloy 600 tubes. Each tube has a nominal outside diameter of 0.875-inch and a nominal wall thickness of 0.050-inch. The tubes were explosively expanded (WEXTEX) at both ends for the full length of the tubesheet and are supported by a number of carbon steel tube supports with round shaped holes. The hot-leg temperature is approximately 604EF.

2.0 STAFF EVALUATION

The licensee provided the scope, extent, methods, and results of their SG tube inspections in the documents referenced above. The licensee also described corrective actions (i.e., tube plugging or repair) taken in response to the inspection findings. The Nuclear Regulatory Commission (NRC) staff made several observations as a result of the review of the 2004 inspection reports. Some of these observations were previously provided during the review of the DCPP Unit 2 2003 inspection summary report, provided in a letter dated December 19, 2003 (ADAMS Accession No. ML033570331), and are therefore not repeated below:

1. During the 2004 outage (1R12), three axial primary water stress corrosion cracks were detected at cold-leg tube support 7C. No primary water stress corrosion crack indications were found at several other hotter tube support elevations (e.g., 7H, 6H). Since the licensee’s dent inspection plan generally assumes that degradation will be successively observed as a function of temperature (i.e., at the hottest tube support elevation followed by the next highest temperature tube support elevation (and so on)), the licensee plans to augment the 1R13 rotating probe dent inspection plan by inspecting 100 percent of the greater than 2 volt hot- and cold-leg dents that have never

been previously inspected with a rotating probe, regardless of the tube support plate elevation. A similar augmented inspection was performed during 2R12 and no primary water stress corrosion cracking or circumferential indications were detected.

2. A circumferential indication was detected at a dent whose magnitude was 0.51 volts. This dent resulted in the expansion of the rotating probe examinations at the tube support plate in which this indication was detected and the next highest tube support plate (in the affected SG). The indication detected was considered small. In response to an NRC request for additional information about the implications of this indication, the licensee indicated that additional inspections were not warranted because circumferential indications at dented tube support plates are expected to be small in size as confirmed by the detected indications. Given that in the future larger circumferential cracks may occur and that cracks may occur at cooler tube support plate elevations before being detected at hotter tube support elevations, additional inspections may be necessary in the future to ensure tube integrity.

3. The bobbin overcall rate for primary water stress corrosion cracking indications at tube support plate elevations in SG 1-1 was lower than that in the other SGs. A low overcall rate could imply that the analysts are not flagging as many indications as they had in the past (i.e., a less conservative screening of the data). This lower overcall rate was attributed to finding 5 of the 6 new axial primary water stress corrosion cracks in less than 2 volt dents in SG 1-1. In the future, the licensee expects the number of bobbin identified indications in the less than 2 volt dents to be about the same in SGs 1 and 2.

4. A number of indications were found in tubes that were previously plugged and later returned to service. These indications were located in the region where the plug had been expanded into the parent tube (i.e., the cracking was limited to the portion of the tube where the plug was expanded). This expansion process occurs within the shop hard roll region of the tube. The licensee indicated these indications could be a result of high residual stresses caused by the plug expansion process or sensitization of the tube material from the tungsten inert gas process used to remove the plugs.

5. The leak rate associated with primary water stress corrosion cracking indications at the tube support plate elevations was under predicted for 1R12. The under prediction, in this case, was not significant. Similarly, the burst pressure for several indications was under predicted. In one instance, the cause of the under prediction was the statistical consequence of assessing the burst pressure at a 95 percent probability level. As a preventive measure, the licensee plugged two indications with the lowest 1R13 projected burst pressure.

6. Chemical cleaning was performed during 1R12. In SGs 1-1 and 1-2, the chemical cleaning was performed prior to the eddy current inspections. In SGs 1-3 and 1-4, the chemical cleaning was performed after the eddy current inspections.

Based on a review of the information provided, the staff concludes that the licensee has provided the information required by their technical specifications. In addition, the staff concludes that there are no technical issues that warrant followup action at this time since (1) the inspections appear to be consistent with the objective of detecting potential tube degradation and (2) the inspection results appear to be consistent with industry operating experience at similarly designed and operated units.

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Date: May 18, 2005

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