December 19, 2003

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

SUBJECT: REVIEW OF DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 2 – 2003 REFUELING OUTAGE 11 STEAM GENERATOR INSPECTIONS 90-DAY REPORT (TAC NO. MB9969)

Dear Mr. Rueger:

By letter dated June 23, 2003, and supplemental letters dated September 30 and October 31, 2003, Pacific Gas and Electric Company submitted a report summarizing the steam generator tube inspections performed during the 2003 Diablo Canyon Power Plant Unit 2 eleventh refueling outage (2R11).

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

Sincerely,

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

Docket No. 50-323

Enclosure: Safety Evaluation

cc w/encl: See next page

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*Memo dated

Accession No: ML033570331 **NRR-106**

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE STEAM GENERATOR 90-DAY REPORT

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON UNIT 2

DOCKET NO. 50-323

1.0 INTRODUCTION

By letter dated June 23, 2003, and supplemental letters dated September 30 and October 31, 2003, Pacific Gas and Electric Company (PG&E or licensee) submitted a report summarizing the steam generator tube inspections performed during the 2003 Diablo Canyon Power Plant Unit 2 eleventh refueling outage (2R11).

2.0 STAFF EVALUATION

The scope and results of the licensee's inspections are contained in the documents referenced above. Based on a review of the above documents, the staff concludes that the licensee provided the information required by their technical specifications. In addition, the staff did not identify any technical issues that warranted follow-up action at this time. However, the staff's observations regarding the licensee's inspection and assessments are given below:

1. The staff made several observations following its review of the Unit 1 2002 inspection summary reports, which are documented in an NRC letter dated November 20, 2003 (ADAMS Accession No. ML33250133). These observations are also applicable to Unit 2 and, therefore, not repeated here.

2. In response to question 3 of the staff's requests for additional information (RAI) on the W* alternate repair criteria (ARC), the licensee provided a rationale for why no adjustment for crack growth is required in the flexible W* criterion for unflawed tubes. In their response, the licensee indicated that new indications are not anticipated to have sufficient through-wall depth to significantly decrease the contact pressure between the tube and tubesheet hole. The licensee, however, did not provide any data supporting their assertion that these newly initiated flaws (either axial or circumferential) do not significantly decrease the contact pressure. In future condition monitoring assessments, the licensee should consider providing the supporting data which confirms that the depths representative of newly initiated flaws in the W* region do not significantly decrease the contact pressure. This issue is more important as the potential for flaws below the W* region increases since any degradation below the W* region when combined with the newly initiated degradation in the W* region may result in a longer length of tubing being required to reduce the potential for tube pullout.

3. In response to questions 5 and 6 of the staff's RAI on the W* ARC, the licensee discussed laboratory test data supporting a conclusion that a higher differential pressure across a tube wall will result in increased leakage for indications in the tubesheet region (which would result in conservative estimates of the leakage). In their response, however, the licensee did not discuss whether the existing leakage models also exhibited a similar trend. In approving the W* repair criteria, the NRC safety evaluation discussed some limitations in the leakage model. Nevertheless, the staff concluded the leakage model was acceptable for several cycles, in part, because the number of steam generator tubes affected by primary water stress corrosion cracking should remain low for the period of time for which the ARC was approved. Assuming that it may be desirable to use the W* ARC for longer period of times (i.e., beyond the period it is currently approved for), it appears that this issue would need to be addressed in a future technical specification amendment request. In addition, the effect of temperature changes during the postulated accident conditions may also need to be addressed in future amendment requests.

Principal Contributor: Ken Karwoski

Date: December 19, 2003