September 26, 2000

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 94177

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) - DIABLO CANYON NUCLEAR POWER PLANT, UNIT NO. 1 REGARDING THE REACTOR CORE THERMAL POWER UPRATE (TAC NO. MA7813)

Dear Mr. Rueger:

By letter dated December 31, 1999, as supplemented by letters dated January 18 and July 7, 2000, Pacific Gas and Electric Company submitted a request for approval of Diablo Canyon Unit 1 reactor core thermal power uprate to 3411 megawatts thermal. The NRC staff has reviewed your submittal pertaining to the structural integrity of several safety-related components. These included pressure-retaining piping, components and their supports, reactor vessel and internals, core support structures, control rod drive system, and safety-related equipment. The NRC staff has identified the need for additional information in order to complete the staff's review. The enclosed RAI describes the specific information needed by the NRC.

The enclosed request was discussed with Mr. Pat Nugent of your staff on September 15, 2000. A mutually agreeable target date of October 1, 2000, for your response was established. Please ensure that you provide your response on or before October 1, 2000, so that we can complete our effort consistent with your proposed schedule. If you have any questions regarding this matter, please contact me at (301) 415-1313.

Sincerely,

/RA/

Steven D. Bloom, Project Manager, Section 2
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-275

Enclosed: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

CONCERNING TECHNICAL SPECIFICATION CHANGES FOR

REACTOR CORE THERMAL POWER UPRATE

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-275

1. In regard to Section 5.1.1 of Attachment B to your December 31, 1999 submittal, provide a comparison of the design parameters (i.e., steam pressure, temperature, primary-to-secondary pressure differential) and transients for the steam generators (SG) Model 51 against the power uprate condition. Also, provide the maximum calculated stress and cumulative fatigue usage factor (CUF) for the critical locations (such as the vessel shell, secondary manway bolts, tubes, and nozzles), the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

2. In Section 5.1.2, it was stated that the replacement interval for the SG manway is reduced from 34 years to 31 years for operation at the uprated conditions. Please confirm that the plant procedures will be revised to incorporate the 31-year replacement interval for the SG manway bolts prior to the implementation of the power uprate. Also, provide an evaluation of the flow-induced vibration of the SG U-bend tubes and moisture carryover due to power uprate regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated elastic-fluid instability ratio.

3. In Section 5.2, it was stated that the pressurizer structural responds evaluation was performed by modifying the existing analysis of record based on the NSSS performance parameters provided in Table 2.1-1 and that the results indicated that the Diablo Canyon Power Plant (DCPP) Units 1 and 2 pressurizer components meet the stress/fatigue analysis requirements of the ASME Code for the 3425 Mwt NSSS uprating parameters and transients. Provide the maximum calculated stress and CUF at the critical locations (such as surge nozzle, skirt support, spray nozzle, safety and relief nozzle, upper head/upper shell and instrument nozzle) of the pressurizer, the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

Enclosure

4. In regard to Section 5.4, provide the maximum calculated stress and CUF at the critical locations of the reactor pressure vessel and internals (nozzles, lower and core plates, core barrel, baffle/barrel, thermal shield supports, control rod drive mechanism, and fuel assembly). Also, provide the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification.

5. In regard to Section 5.4.2, provide in detail a quantitative assessment of flow-induced vibration of the reactor internal components due to power uprate.

6. In regard to Section 5.7, list the ASME Class 1 branch piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating NSSS piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Provide the calculated maximum stresses for the critical piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, provide a justification.

7. Discuss the functionality of safety-related mechanical components (i.e., all safety related valves and pumps, including power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in the Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," MOV program at DCPP will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Also, provide an evaluation of the effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," and on the evaluation of overpressurization of isolated piping segment for GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions." Identify mechanical components for which functionality at the uprated power level could not be confirmed.

8. In regard to Section 7, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, provide a justification. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

9. Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of the evaluation for power uprate effects on the high energy line break analysis, jet impingement and pipewhip loads for the power uprate condition.

10. Do you project modifications to piping or equipment supports for the proposed power uprate? If any, provide examples of pipe supports requiring modification and discuss the nature of these modifications. Did you follow WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," for evaluating the DCPP power uprate? If not, discuss the differences between the current power uprate analysis and the WCAP-10263 methodology.