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

June 30, 1999

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

SUBJECT: CLOSEOUT OF THE RESPONSES TO THE REQUESTS FOR ADDITIONAL INFORMATION TO GENERIC LETTER 92-01, REVISION 1, SUPPLEMENT 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," FOR THE DIABLO CANYON POWER PLANT, UNITS 1 AND 2 (TAC NOS. MA0541 AND MA0542)

Dear Mr. Rueger:

On May 19, 1995, the U.S. Nuclear Regulatory Commission (NRC) issued Generic Letter 92-01, Revision 1, Supplement 1 (GL 92-01, Rev. 1, Supp. 1), "Reactor Vessel Structural Integrity," to holders of nuclear operating licenses. In issuing the GL the staff required addressees of the GL to:

(1) identify, collect and report any new data pertinent to the analysis of structural integrity of the reactor pressure vessels (RPVs) at their nuclear plants, and

(2) to assess the impact of that data on their RPV integrity analyses relative to the requirements of Sections 50.60 and 50.61 to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.60 and 10 CFR 50.61), and to the requirements of Appendices G and H to Part 50 of Title 10 of the *Code of Federal Regulations* (Appendices G and H to 10 CFR Part 50).

On August 16, 1995, you submitted your initial response to GL 92-01, Rev. 1, Supp. 1., and provided the requested information relative to the structural integrity assessments for the Diablo Canyon Nuclear Plant, Units 1 and 2. The staff evaluated your response to GL 92-01, Rev. 1, Supp. 1, and provided its conclusion relative to your response on August 7, 1996. However, since the time of the staff's closure letter, the Combustion Engineering (CE) Owners Group and the Babcock and Wilcox (B&W) Owners Group have each submitted additional data regarding the alloying chemistries of beltline welds in CE and B&W fabricated vessels. The additional alloying data were submitted in Topical Reports CE NPSD-1039, Revision 2, CE NPSD-1119, Revision 1 for CE fabricated RPV welds, and BAW-2325, Revision 1 for B&W fabricated RPV welds.

In addition, Chicago Bridge and Iron (CB&I) BWR data were submitted in Topical Report BWRVIP-46. As a result of the efforts by CE and B&W, the staff determined that additional information was necessary relative to the structural integrity assessments for your plants. On April 6, 1998, the staff issued a request for additional information (RAI) in regard to the alloying chemistries of beltline welds, your assessment of surveillance data for your facility, pressure-temperature (P-T) limits, and pressurized thermal shock (PTS) assessments for the Diablo Canyon Nuclear Plant, Units 1 and 2. In general, with respect to the contents of

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the RAI, the staff requested that you reassess the alloying chemistries for the beltline welds and RPV surveillance welds for the Diablo Canyon units relative to the chemistries provided in Topical Report CE NPSD-1039, Revision 2, and CE NPSD-1119, Revision 1, and provide the impact of any changes to the best-estimate chemistries for your beltline RPV welds on the structural integrity assessments for your facilities relative to the requirements of 10 CFR 50.60, 10 CFR 50.61, and Appendices G and H to 10 CFR Part 50, as applicable to the licensing bases for your plants.

You provided your response to the staff's RAI for Diablo Canyon Nuclear Plant, Units 1 and 2, on July 6, 1998. As a result of the staff's review of your responses to GL 92-01, Revision 1, GL 92-01, Rev. 1, Supp. 1, and the Supp. 1 RAI, the staff has revised the information in the Reactor Vessel Integrity Database (RVID) and is releasing it as RVID Version 2. It should be noted that with respect to the beltline material data for Diablo Canyon Units 1 and 2, there are some variations in the data inputted by the staff, and the corresponding values reported by you in your response to the GL 92-01, Rev. 1, Supp. 1 RAI. The deviations between the data are explained in the reference sections for each Unit, or the individual component screen notes (i.e., each forging, plate, and weld has a specific area for notes which is a new feature of the database).

The variations in the data applicable to the PTS assessment of the beltline materials at Diablo Canyon Unit 2 are not particularly significant. However, the staff has determined that the staff's RT_{PTS} value for the limiting material at Diablo Canyon Unit 1 could differ from your values by as much as +17.2°F. In your response to the RAI dated July 6, 1998, you indicated that the limiting material in the beltline region of the reactor vessel at Diablo Canyon Unit 1 is lower shell axial weld 3-442C, which was fabricated from material heat number 27204. In the response you listed that the limiting end-of-license RT_{PTS} value for this material is 253.4°F if Table 1 in 10 CFR 50.61 is used to establish the chemistry factor (CF), and 241.2°F if the plant-specific surveillance data for the heat are used to establish the CF. In your response to the RAI, you indicated that these values were based on alloying contents of 0.198 wt-% copper and 0.999 wt-% nickel, which were obtained from analytical measurements of surveillance coupons that had been removed from the reactor vessel. You also informed us that the chemistry factor, and therefore the RT_{PTS} value, for the limiting material would be slightly higher if the chemistries for the heat in Topical Report CE NPSD-1039, Revision 2, were applied to the RT_{PTS} assessment (e.g., 0.203 wt-% copper and 1.018 wt-% nickel). In your response to the RAI, you indicated that it was your position that the source of the best-estimate chemistry for the Unit 1 axial welds fabricated from heat 27204 remained the Unit 1 surveillance weld coupons, since the surveillance weld coupons were made at the same time, using the same welding procedure, specification, wire heat and flux, and baseplates as were the corresponding axial welds in the reactor vessel for the Unit.

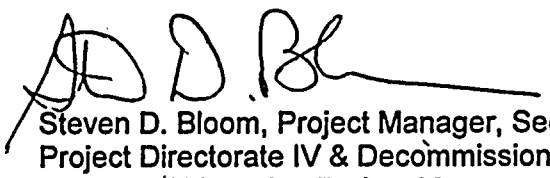
The staff has determined that using chemistry values from the plant-specific surveillance coupons as the sole basis for establishing the CF of a beltline material represented in the surveillance program is not consistent with the criteria of section (c)(ii)(B) of 10 CFR 50.61, which stated the values used for a weld in the beltline should be based on an average of all values used for the heat. To date only two surveillance capsules (Capsules S and Y) have been removed from the reactor vessel at Diablo Canyon Unit 1. The data from these capsules do not meet the credibility criteria specified in 10 CFR 50.61. In our letter of June 28, 1996, we informed you that since the surveillance data are not credible, the values in Table 1 of

10 CFR 50.61 should be used as the basis for calculating the CF and hence the RT_{PTS} values for the beltline welds fabricated from heat number 27204. In addition, the CF for this heat should be based on values of 0.203 wt-% copper and 1.018 wt-% nickel, as listed in Topical Report CE NPSD-1039, Revision 2. These values represent the best-estimate average copper and nickel values for the material. This position is reflected in the PTS Summary Report, and is consistent with our letter of June 28, 1996, and with the approach taken in your latest technical specification amendment submittal for the Diablo Canyon P-T limit curves (PG&E letter DCL-98-121, dated September 3, 1998). Thus, the staff's calculation lists the RT_{PTS} value for limiting, lower shell axial weld 3-442C as 258.3°F.

The new database diskettes for the RVID are posted on the world-wide-web at a location which is linked to the NRC home page (http://www.nrc.gov/NRR/RVID/index.html). We recommend that you review this information. If the staff does not receive comments by September 1, 1999, we will assume that the data entered into the RVID are acceptable for your plants. No additional information is necessary with regard to the structural integrity assessments. Future submittals on P-T limits, PTS, or upper shelf energy (USE) should reference the most current information.

The staff appreciates your efforts in regard to this matter.

Sincerely,



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Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

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Sincerely,

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Docket Nos. 50-275 and 50-323

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