

June 21, 2000

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

Dear Mr. Rueger:

In a letter dated December 31, 1999, Pacific Gas and Electric Company submitted a request for approval of an increase in reactor core thermal power uprate to 3411 megawatts thermal for the Diablo Canyon Nuclear Power Plant, Unit 1. The NRC staff has reviewed this submittal and determined that additional information is required in order to determine the acceptability of your request. The request for additional information is enclosed.

The enclosed request was discussed with Mr. Pat Nugent of your staff on June 13, 2000. A mutually agreeable target date of July 7, 2000, for your response was established. If circumstances result in the need to revise the target date, please call me at your earliest opportunity. 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

Enclosure: Request for Additional Information

cc w/encl: See next page

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Diablo Canyon Power Plant, Units 1 and 2

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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

Annular Fuel Pellets Blankets

Section 3.1.2 of WCAP-14819 (Enclosure B) states that the Diablo Canyon Nuclear Power Plant (DCPP) uprating program included the introduction of a reload with fully enriched annular fuel pellet blankets at the top and bottom of the core (p. 3.6), and the annular pellet blankets were explicitly modeled for the SBLOCA analysis (Table 3.1.2-1).

Section 6.0, Fuel Design, states that the [annular pellet blankets fuel design] for DCP-1 was evaluated under the Uprating Program in the areas of fuel rod and fuel assembly structural integrity for the uprating conditions. However, the report does not provide the design description, evaluation, or reference of this fuel design, except for such statements as "rod internal pressure analyses performed for DCP-1 Uprating Program indicates that the rod internal pressure criterion will be satisfied for the uprated condition in Table 6.1-1" (which indicates the fuel design considered to be ZIRLO clad, 1.5xIFBA, 100 psi backfill, annular blankets).

1. If the fully enriched annular fuel pellet blanket fuel design to be used for uprate reload is described in a separate report, provide a reference to the topical report including the NRC safety evaluation.

Otherwise, provide: (1) a detailed description of the fully enriched annular fuel pellet blankets fuel design, including the lengths, diameters, and enrichment of the annular fuel blankets, the dimensions of the annular pellet blanket fuel and cladding, cladding material (ZIRLO or Zirc-4), and pre-pressurization, etc., and (2) the evaluation of this fuel design relative to Standard Review Plan (SRP) Section 4.2, including the evaluations performed under the DCP-1 Uprating Program.

2. Section 6 of WCAP-14819 states that the core design, thermal and hydraulic evaluations are evaluated for DCP-1 on a cycle specific basis. Other than the SBLOCA analysis and the fuel structural evaluation described in the text, have analyses with respect to nuclear design and thermal hydraulic design of this fuel design been performed? What are the results?
3. Section 6.3.2 of WCAP-14819 states that the use of Zirc-4 clad fuel will require cycle-specific analysis to confirm its compliance to the new cladding corrosion model currently under development; whereas Enclosure C, Item 4, states that the fuel is assumed to

have all ZIRLO cladding, which is consistent with Vantage 5+ fuel. Should the evaluation related to the DCP-1 power uprate be limited to the ZIRLO cladding?

4. Would TS 4.2.1, "Fuel Assemblies," be revised to reflect the use of the annular pellet blankets?

LBLOCA Analysis

5. It is stated in Enclosure A to PG&E letter DCL-99-170 that the LBLOCA analysis is documented in WCAP-14775, and was reviewed and approved by the NRC in 1998 as license amendments 121 and 119 for Units 1 and 2, respectively, and that using the best estimate methodology, the Units 1 and 2 PCT was revised to a value of 2043°F, as reported in PG&E letter DCL-99-096. However, WCAP-14819 states that the bounding BE LBLOCA analysis for both units has resulted in a PCT at 95% probability of 1976°F. The same was stated in Enclosure C, which also indicated that the LBLOCA analysis was approved by the NRC in 1998 in license amendments 121 and 119. Please clarify the discrepancy in the PCTs discussed above, and identify the PCT of the analysis of record. Please discuss where in the NRC safety evaluation for amendments 121 and 119, the review and approval of WCAP-14775 is described.
6. Enclosure A states that the difference in the reactor internal design between DCP-1 Units 1 and 2 resulted in lower reactor coolant system (RCS) minimum design flow for Unit 1 (359,200 gpm vs. 362,500 gpm for Unit 2). Describe the differences between the two units in the reactor internal design. Was the bounding BE LBLOCA analysis described in WCAP-14775 based on the reactor internals of Unit 1 or Unit 2? What are the bases to conclude a "bounding" analysis is applicable for both units in the BE LOCA analysis in light of the reactor internals differences?
7. Enclosure C, Item 4, states that the LBLOCA and SBLOCA analysis results which incorporate these fuel cladding impacts (i.e., 2°F PCT penalty for ZIRLO fuel cladding) have been submitted to the NRC separate from this uprate license amendment request. What are these submittals?
8. Was the bounding BE LBLOCA analysis based on the annular fuel pellet blankets design and Unit 1 power uprate conditions? If not, what is the basis for its applicability to the power uprate with this new fuel design?
9. Was the bounding BE LOCA analysis performed based on the power peaking factors specified in the technical specifications (or the Core Operating Limits Report)? Where are they documented so that they can be used to confirm the validity of the analysis for any operating cycle.

SBLOCA Analysis

10. Figure 3.1.2-2 in WCAP-14819 provides the degraded HHSI and IHSI pump flows versus pressure curve modeled in the small break LOCA analysis. Explain how these degraded curves are related to technical specification surveillance?

11. Page 3-5 states that the long term core cooling considerations of 10 CFR 50.46 acceptance criteria are not directly applicable to the SBLOCA transient, but are assessed elsewhere as part of the evaluation of ECCS performance. Discuss where it is assessed.

Section 3.1.5 states that if the boron sources are affected by the power uprating, the LTCC calculation will be affected. This calculation is performed on a cycle-specific basis and will be reviewed at the time of the RSAC generation. Please clarify RSAC generation and is this a condition for acceptance?

12. The SBLOCA analysis documented in WCAP-14819 determined that the limiting break for both units to be a 3-inch diameter cold leg break, which is a change from the previous analysis that found the 4-inch break to be the limiting break. Describe the differences (in terms of assumptions, important parameters, modeling and correlations) in the new and the previous analyses that result in different limiting break size.
13. The proposed technical specification changes includes the change in Table 3.3.3-1 of nominal T_{avg} from 576.6°F to 577.3°F. The small break LOCA analysis in Section 3.1.2 of WCAP-14819 is based on the nominal T_{avg} of 572.0°F. Please clarify the value of nominal RCS average temperature T_{avg} and justify why the SBLOCA analysis based on a lower nominal T_{avg} is acceptable for supporting the power uprate.
14. Show that the large- and small-break LOCA analyses methodologies referenced in WCAP-14819 apply to DCPD Units 1 and 2 by confirming that PG&E and Westinghouse (LOCA analysis vendor) have ongoing processes to assure that the values of peak cladding temperature (PCT) sensitive parameters input to the LOCA analyses bound (bounding distribution for BELOCA) the as-operated plant values for those parameters.

NON-LOCA SAFETY ANALYSES

15. Page 3-13 of WCAP-14819 lists the non-LOCA events for which the current at-power safety analyses assume the lower design RCS flow rates associated with Unit 1, the higher licensed core power, NSSS power and coolant average temperature of Unit 2; therefore the safety analyses associated with Unit 1 uprate power remain bounding. However, it also states that several of the analyses assume the previous Unit 2 NSSS power of 3423 MWt (i.e., RC pump heat input of 12 MWt rather than 14 MWt), which is lower than the new nominal NSSS power of 3425 MWt for both units. It is said that this 2 MWt increase is very small and has been evaluated to have a negligible effect on the results of the affected safety analyses.

Identify which events were analyzed with 3423 MWt and provide the evaluation that has been made to conclude that the 2 MWt increase has a negligible effect on the results of the affected safety analyses.

16. Were the current non-LOCA safety analyses performed with the annular pellet blankets fuel design?

If not, what kind of fuel design was analyzed for the safety analyses? What is the basis to conclude the analysis results are applicable to the reload with the annular pellet blankets fuel design and power uprate conditions? The consideration should include the differences in the fuel design, the applicability of the CHF correlation and DNBR limit, the uncertainties of the parameters involved in the ITDP and the design DNBR limit if the ITDP procedure was used, the power distribution of the reload cores with the new fuel, and possible effects on the overtemperature and overpower ΔT reactor trip setpoints.

Accidental Depressurization of RCS

17. Section 3.2.3.2 of WCAP-14819 states that some key analysis input assumptions (for the accidental RCS depressurization) are identified in Appendix A. However, there is no Appendix A in WCAP-14819. Please clarify.
18. It also references Ref. 2 (LOFTRAN) and Ref. 3 (ITDP). However, there are no references 2 and 3 in the text. A revision to the WCAP is needed.
19. For the analysis of the transient with VANTAGE 5 fuel with annular pellet blankets, what CHF correlation is used, and what is the design safety DNBR limit for the ITDP? Is the CHF correlation applicable to the VANTAGE 5 with annular pellet blankets?

Steam Line Break at Full Power

Section 3.2.4 of WCAP-14819 concludes that the DNB design basis is met for the steam line break at full power initial condition based on (1) a previous SLB analysis result documented in WCAP-13615-R2, and (2) a new transient result being less limiting due to the use of a higher low steam line pressure safety injection actuation set point. (Higher SI setpoint results in an earlier reactor trip for a larger range of break sizes, reduces the size of the largest break that will not trip on low steam pressure SI actuation, and, in turn, reduces the peak core power that is achieved for the worst case, which will result in a higher minimum DNBR) It also states that the DNBR is confirmed for this event using cycle-specific core parameters as part of the reload safety evaluation.

20. It is not clear whether a new SLB analysis other than the one described in WCAP-13615, has been done, or the results are simply an engineering judgement. Because the core design and thermal and hydraulic evaluations are performed on a cycle-specific basis (Section 6.0), clarify the statement that the DNBR criterion will be confirmed for this event using cycle-specific core parameters as part of the reload safety evaluation.

If a new SLB analysis has been done, provide the details of the analysis, including computer code used, fuel design, input assumptions including whether new reload cycle with power uprate conditions, and results.

21. It is stated that the SLB at full power initial conditions analysis to demonstrate core integrity is not explicitly documented in the UFSAR. Will the licensee commit to document the analysis of the SLB at full power event in the Updated Final Safety Analysis Report (UFSAR)?

Steam Generator Tube Rupture

22. Section 3.4 of WCAP-14819 states that a reanalysis of the margin to SG overfill for revised auxiliary feedwater and PORV flow rates is presented in PGE-92-685, "SGTR Margin to Overfill Re-Analysis," October 13, 1992. Has this been reviewed and accepted by the NRC? Please provide a copy of the report and the NRC acceptance.
23. Section 3.5.1 states that the bounding SGTR analysis in WCAP-10713 was performed with an RCS average temperature of 577.6°F, compared to 577.3°F for the Unit 1 power uprate.

Explain the statement in Section 3.5.1 that "the difference in RCS average temperature of 0.3°F between DCPD Units 1 and 2 would slightly delay the reactor trip time. Earlier reactor trip results in earlier steam releases to the environment for the offsite radiological dose case. Therefore, the use of the Unit 2 RCS average temperature is conservative and bounds the Unit 1 uprating parameters."

24. Section 3.5.2 states that because the current source term (in the UFSAR) is based on a reactor power level of 105% of Unit 2 rated thermal power, a power uprate of Unit 1 to the Unit 2 rated power has no impact on radiological source terms for the design basis accidents of normal plant operation. Would the use of the annular pellet blankets fuel design have any significant effect on the source terms?

Residual Heat Removal System

25. Section 4.1.2 of WCAP-14819 describes the analysis of RHR system cooling capability and concludes that based on uprated conditions, the analysis results indicates that RCS cooldown to 140°F using two cooling trains is achieved at 17.4 hours. The analysis also indicates that cooldown to 200°F using one cooling train is achieved at 29.2 hours after shutdown. These meet the RHR design criteria of cooling down to 140°F when both trains are available in 20 hours, and to 200°F with one train in 36 hours. On the other hand, Enclosure C, Item 1 states that (1) the RHR cooldown calculation was reperformed and documented in WCAP-14819, however, the analysis was redone mostly to add margin for issues related to the CCW system rather than in response to the uprate; and (2) the reanalysis used more conservative assumptions than the previous analysis including higher heat loads and lower flow rates to bound a larger spectrum of operating conditions; and as a result, the new RHR cooldown calculation indicates a longer required time to perform the cooldown.

The statement that the analysis was redone and documented in WCAP-14819 appears to mean the WCAP-14819 analysis is the "reanalysis," whereas the results as stated appear to mean the WCAP-14819 is the "previous analysis" and the "reanalysis" is one other than that described in WCAP-14819.

Clarify the confusion as to what or where the "previous analysis" and "reanalysis" are, and whether the analysis of WCAP-14819 was performed for power uprate. Provide the differences in the input assumptions and results between the "previous analysis" and "reanalysis."

Typos Related to OT Δ T and OP Δ T Trip Setpoint Calculations

Enclosure A, P. 3, 10th line: "over pressure Δ T" should be "overpower Δ T."

Enclosure B, p. 3-35, Item 15.1.3 "Overtemperature and Overpower AT" should be "... Δ T."