Pacific Gas and Electric Company

David H. Oatley
Vice President–Diablo Canyon
Operations and Plant Manager

Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

805.545.6000

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PG&E Letter DCL-99-170

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
License Amendment Request 99-03, Unit 1 Reactor Core Thermal Power Uprate

Dear Commissioners and Staff:

Enclosed is a license amendment request (LAR) to Facility Operating License No. DPR-80 for Diablo Canyon Power Plant (DCPP) Unit 1. This LAR would revise facility operating license section 2.C.(1) to authorize operation of Unit 1 at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power). Unit 2 is already authorized to operate at that power level. This LAR would also revise the Technical Specification (TS) 1.1 definition of rated thermal power to reflect Unit 1 operation at the uprated reactor core power level, change the reactor core safety limits in TS Figure 2.1.1-1 to reflect the current fuel type and provide additional margin for OTΔT and OPΔT setpoint calculations, and change the nominal full power T_{avg} in the OTΔT and OPΔT function in notes 1 and 2 to TS table 3.3.3-1.

A description of the proposed changes and the basis for them are provided in Enclosures A, B, and C. The proposed change to the facility operating license is included in a marked-up page in Enclosure D. Changes to the TS are noted in the marked-up Improved Technical Specification (ITS) pages provided in Enclosure E. The proposed ITS pages are provided in Enclosure F. Proposed changes to the Final Safety Analysis Report and the Precautions, Limitations, and Setpoints document are included in Enclosures G and H, respectively. The changes do not involve a significant hazards consideration, as defined in 10 CFR 50.92, or an unreviewed environmental question. Further, there is reasonable assurance that the proposed changes will not adversely affect the health and safety of the public.

The proposed changes are not required to address an immediate safety concern. Therefore, PG&E requests that the NRC review this LAR on a medium priority, and approve it prior to the Unit 1, cycle 10 refueling outage (1R10), currently scheduled

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to begin October 2000. PG&E also requests that the TS changes requested in this LAR be effective upon issuance of the license amendment, to be implemented upon completion of the 1R10 refueling outage.

Sincerely,

David H. Oatley

cc: Edgar Bailey, DHS
Steven D. Bloom
Ellis W. Merschoff
David Proulx
Diablo Distribution

Enclosures

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Unit 1	Docket No. 50-275 Facility Operating License No. DPR-80

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath says that he is Vice President - Diablo Canyon Operations and Plant Manager of Pacific Gas and Electric Company; that he is familiar with the content thereof; that he has executed License Amendment Request 99-03 on behalf of said company with full power and authority to do so; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

David H. Oatley
Vice President - Diablo Canyon Operations
and Plant Manager

Subscribed and sworn to before me this 31st day of December, 1999.

Notary Public
State of California
County of San Luis Obispo

AMY J. CALLOWAY
COMM. #1096602
NOTARY PUBLIC CALIFORNIA
SAN LUIS OBISPO COUNTY
My Comm. Expires April 28, 2000

UNIT 1 REACTOR CORE THERMAL POWER UPRATE

A. DESCRIPTION OF AMENDMENT REQUEST

This license amendment request (LAR) would revise Facility Operating License No. DPR -80, section 2.C.(1), to authorize operation of Unit 1 at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power). Unit 2 is already authorized to operate at that power level. Specifically, section 2.C.(1) would be revised to read:

> "Maximum Power Level
>
> The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3411 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein."

A mark-up of the proposed facility operating license change is presented in Enclosure D.

This LAR would also revise the following Improved Technical Specifications (ITS) issued in License Amendment (LA)135:

- TS 1.1, "RATED THERMAL POWER (RTP)" would be revised to read: "RTP shall be a total reactor core heat transfer rate for the reactor coolant of 3411 MWt for both units."
- ITS Figure 2.1.1-1, "Reactor Core Safety Limits," would be revised to reflect the current fuel type and provide additional margin for OTΔT and OPΔT setpoint calculations.
- ITS Table 3.3.3-1, "Reactor Trip System Instrumentation," Note 1, "Overtemperature ΔT," would be revised to note that the Unit 1 nominal full power T_{avg} is now 577.3° F instead of the current value of 576.6° F.
- ITS Table 3.3.3-1, "Reactor Trip System Instrumentation," Note 2, "Overpower ΔT," would be revised to note that the Unit 1 nominal full power T_{avg} is now 577.3° F instead of the current value of 576.6° F.

Changes to the Technical Specifications (TS) are noted in the marked-up ITS pages provided in Enclosure E. The proposed ITS pages are provided in Enclosure F.

B. BACKGROUND

During the design of Diablo Canyon Power Plant (DCPP) Units 1 and 2, the nuclear steam supply system (NSSS) vendor evolved its reactor internals design to reduce the flow resistance in the reactor coolant system (RCS). Although the reactors, structures, and all auxiliary equipment are substantially identical for the two units, this difference in the reactor internal design resulted in a lower coolant flow rate for Unit 1. The RCS minimum thermal design flow for Unit 1 is 359,200 gpm as compared to the Unit 2 value of 362,500 gpm. Consequently, the license application reactor ratings were 3338 MWt for Unit 1 and 3411 MWt for Unit 2. These power levels included inherent margins since the design criteria and expected ultimate reactor core power was 3488 MWt for Unit 1 and 3568 MWt for Unit 2. These expected ultimate reactor core powers are identified in Section 1.1 of Final Safety Analysis Report (FSAR) and subsequent FSAR Updates.

This LAR would increase the Unit 1 reactor core thermal power by 2.2 percent to the higher thermal power level of 3411 MWt permitted for Unit 2. This change will result in identical power ratings for both DCPP units. The revised reactor core thermal power level is within the initial design rating of Unit 1, and does not require physical modifications to Unit 1.

C. JUSTIFICATION

The proposed increase in licensed reactor core thermal power is being made to increase electric energy production from DCPP Unit 1, and simplify operation of Units 1 and 2 by making their thermal power levels identical. Operating Units 1 and 2 at the same reactor core thermal power will allow greater standardization between the units in terms of analysis, documentation, and procedures. This request involves a power uprate similar to that granted in 1986 in License Amendment No. 71 to Facility Operating License No. DPR-70 for the Salem Unit 1 facility, where the power level was increased from 3338 MWt to 3411 MWt to be consistent with Salem Unit 2.

D. SAFETY EVALUATION

The evaluation of operating DCPP Unit 1 at the increased reactor core thermal power of 3411 MWt is included in Enclosures B and C. Enclosure B contains the joint Westinghouse-PG&E Licensing Report WCAP-14819, "Pacific Gas and Electric Company Diablo Canyon Power Plant, Unit 1 3425 MWt Uprating Program Licensing Report." Enclosure C contains an addendum to the Licensing Report describing changes in the plant since the report was written, and providing additional detail relating to issues identified during NRC reviews of other plant uprates.

A summary of the report and addendum are included below:

Most safety-related analyses, such as containment integrity, environmental qualification, dose assessment, hydrogen generation, and steam generator tube rupture, and most non-loss-of-coolant accident (non-LOCA) analyses, were previously performed assuming the higher Unit 2 core power level of 3411 MWt and the lower Unit 1 RCS flow rate to bound both units with a single analysis. These analyses did not need to be modified to accommodate the proposed change. The analyses that did require modification are the large break loss-of-coolant accident (LOCA), the small break LOCA, the over temperature and over pressure ΔT (OTΔT/OPΔT) setpoints calculation, and the accidental RCS depressurization event. The residual heat removal (RHR) cooldown calculation was also reanalyzed as part of the uprate project.

Large Break LOCA Analysis

The large break LOCA analysis has recently been updated to reflect a power level of 3411 MWt for both units. The large break LOCA analysis utilized the Westinghouse Best Estimate methodology of WCAP-12945-P-A. The analysis is documented in WCAP-14775, "Diablo Canyon Power Plant Units 1 and 2 BE LOCA Analysis," and was reviewed and approved by the NRC in 1998 LAs 121 and 119 for Units 1 and 2, respectively).

Using the best estimate methodology, the Unit 1 and Unit 2 resultant peak clad temperature (PCT) was revised from 2042°F and 2169°F, as reported in PG&E Letter DCL-97-124, respectively, to a value of 2043°F for both units, as reported in PG&E Letter DCL-99-096. The improved best estimate methodology consolidated the numerous outstanding PCT evaluations on both units and while Unit 2 gained significant margin, the Unit 1 uprated power level was accommodated with only a very small net PCT increase.

Small Break LOCA Analysis

The results of the small break LOCA reanalysis were submitted to the NRC in 1998 by PG&E Letter DCL-98-183, "License Amendment Request 98-09, Revision of TS 6.9.1.8 to Allow Use of NRC Approved Addenda to WCAP-0054-P-A to Determine Core Operating Limits: Small Break Loss-of-Coolant Accident Reanalysis." In that letter, PG&E requested allowance to use any applicable NRC approved addenda to WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code." to determine core operating limits. At the NRC Staff's request, in PG&E Letter DCL-99-099, "Supplement to License Amendment Request 98-09," PG&E limited the requested change to the use of WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model," July 1997, only. In LAs 136 (Unit 1) and 136 (Unit 2), dated November 15, 1999, the NRC staff found the use of WCAP-10054-P-A, Addendum 2, Revision 1, acceptable for use in DCPP licensing applications, including reference in TS 6.9.1.8 and the Core Operating Limits Report (COLR).

Using the COSI methodology, the Unit 1 and Unit 2 Analysis of Record results for PCT change from 1275° F and 1358° F, respectively, to 1304° F and 1293° F, respectively. The Unit 2 PCT decreased due to the benefit of the COSI methodology and the specification of an all Vantage 5 core. The Unit 1 PCT increased due to the higher power level, but the new PCT is still below the design parameters assumed for fuel design. The new Unit 1 PCT is below the previously calculated Unit 2 PCT because of the benefit of the COSI methodology and the specification of an all Vantage 5 core.

OTΔT/OPΔT Setpoint Calculation

The OTΔT/OPΔT setpoints are functions of both reactor core power level and RCS flow rate, and are independently calculated for Units 1 and 2. The Unit 2 analysis does not bound Unit 1 because of Unit 1's lower RCS flow rate. The purpose of the setpoint calculation is to demonstrate that the setpoints adequately protect against exceeding the temperature versus power core safety limits as specified in ITS Figure 2.1.1-1 for design basis accidents. Other than analysis input changes, the calculations performed for Unit 1 at the uprated conditions use the same NRC approved methodology (WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions", September 1986) as previous OTΔT/OPΔT setpoint calculations. Changes to the inputs include a revision to ITS Figure 2.1.1-1 and a higher nominal T_{avg} as discussed below.

The reactor core safety limits specified in the existing ITS Figure 2.1.1-1 bound reactor cores that utilize both 17x17 Vantage 5 fuel and 17x17 standard fuel, while the latter fuel type is no longer utilized at DCPP. The proposed change to Figure 2.1.1-1 revises the reactor core safety limits to be bounding for reactor cores using Vantage 5 fuel only. With this revision, the current OTΔT/OPΔT setpoints remain bounding for Unit 1 in the uprated condition. No other fuel related design limitations are required by this figure change since the large and small break LOCAs already bound exclusive use of Vantage 5 fuel.

The OTΔT/OPΔT setpoint calculations use the assumed parameters associated with the uprated condition. This includes the slightly higher RCS T_{avg} of 577.3° F instead of the previous value of 576.6° F. Since the revised value was used in the calculations, the reference to the upper T_{avg} nominal value in Notes 1 and 2 to ITS Table 3.3.3-1 is revised.

Accidental Depressurization of the RCS

The accidental depressurization of the RCS analyzed in the FSAR Update assumes that a pressurizer safety relief valve fails open. This is a Condition 2 event discussed in FSAR Update Section 15.2.13. The current calculation considers each unit separately. The Unit 2 analysis assumes a core power of 3411 MWt, but it credits the higher Unit 2 RCS flow rate and thus does not bound the Unit 1 uprated condition. Therefore the Unit 1 accidental depressurization of the RCS accident was reanalyzed using the identical previous methodology but with the Unit 1 uprated power conditions. This methodology includes the LOFTRAN computer code and Improved Thermal Design Procedure. This analysis also assumes use of Vantage 5 fuel and a conservatively large positive moderator temperature coefficient of +7 pcm/°F. The use of the larger positive moderator temperature coefficient adds conservatism and margin for future possible core designs, but other design basis accident analyses still use +5 pcm/°F which is consistent with ITS Figure 3.1.3-1. The results of the analysis demonstrate that the departure from nucleate boiling ratio remains above the appropriate limit value such that no fuel or clad damage is predicted for this accident.

RHR Cooldown

The RHR cooldown calculation demonstrates the ability of the RHR system to cool the core to Mode 6 conditions (140° F) within 20 hours if both RHR trains are available, or to cool the core to cold shutdown conditions (200° F) within 36 hours if only one train is available. There is no difference between design inputs for Unit 1 and 2 for this RHR cooldown calculation. The uprate project provided an opportunity for replacing the previous cooldown calculation with one employing more conservative assumptions, particularly with regard to component cooling water system assumptions. The new calculation bounds both units and demonstrates the ability to cool the core to Mode 6 conditions (140° F) within 17.4 hours if both RHR trains are available, or to cool the core to cold shutdown conditions (200° F) within 29.2 hours if only one train is available.

Other Impacts

In addition to the above described safety analyses, the Unit 1 uprated power conditions were evaluated to ensure there was no impact on any other existing analyses due to any Unit 1 and Unit 2 design differences. These include component analysis, hydraulic forces, balance of plant performance, and normal operating transient analysis. In a few cases, most notably the main generator cooling system, the margin between the Unit 1 operating conditions and design limits is less than Unit 2; however these cases have been thoroughly reviewed and still found to be acceptable. As noted in the Background section, the original design power level of Unit 1 was 3488 MWt which is higher than the proposed uprated condition. This original design power level was the one used as the plant design criteria during procurement and construction. In fact, for all safety-related systems, the original design criteria was a reactor power rating of 3568 MWt so that a single set of evaluations would envelope both Unit 1 and Unit 2. The original plant FSAR of 1973 states in section 1.1, "The containment system and engineered safety features are designed and evaluated for operation at the reactor power rating of 3568 MWt." This analysis included flow vibration, component stress, and all applicable regulatory requirements at the time. Since the plant was designed for a potentially higher power level, it is consistent with expectations that this review concludes that the Unit 1 design is adequate for operation at 3411 MWt.

The impact of higher neutron fluence on the reactor pressure vessel due to the uprated power level was also examined. The flux projection for the uprated 21-month fuel cycle core is less than that used in the current Unit 1 vessel analysis. Therefore the higher fluence does not impact the current vessel integrity calculations or pressure-temperature limitations.

The impact of the uprate on electric grid stability was reviewed. The proposed Unit 1 uprate represents only a 1 percent increase in the total plant load to the grid. This is evaluated by PG&E's distribution system engineers to have a negligible impact on grid stability.

The impact on steam generator moisture carryover was reviewed. Moisture carryover will increase with higher steam flow and/or lower steam pressure. While the uprate will cause a small immediate change in these parameters, the long term effects of plant aging are still considered the dominant factor . Nevertheless, the Unit 1 uprate will be monitored closely for any potential loss of efficiency or increased high pressure turbine blade wear. In the future, PG&E may perform steam generator modifications to offset any additional moisture carryover. However, this is not considered to be a safety-related issue.

Turbine missile generation probability was calculated and compared to annual frequency criteria in WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," June 1987. Since that time, DCPP low pressure (LP) turbine rotors have been refurbished and the missile generation probability has been greatly decreased. The uprate will cause slightly higher temperatures in the LP rotors (less than an additional 1.5°F out of about 500°F from the main steam reheater). Higher temperature contributes to stress corrosion cracking, but Westinghouse reviewed this concern and concluded that the rotor refurbishment more than offsets the small effect of the uprate. The net effect is still a reduction in missile generation probability compared to previously documented values.

Loading on the steam generator manway closure bolts will be increased due to slightly increased thermal stresses. Therefore, the bolt replacement schedule will be reduced from 34 years to 31 years to maintain the same safe bolt strength margin.

Human performance factors were also considered. Operation of Unit 1 at a 2 percent higher level power level will have only minor impacts on Operations and Maintenance. Beneficial aspects include more consistent procedures and a training simulator that will now match the power level at both units.

The current probabilistic risk assessment (PRA) model for DCPP is based on the higher rated Unit 2 power level. The Unit 1 uprate will not impact the current PRA or individual plant examination (IPE) submittal for DCPP.

In summary, the operation impacts of the proposed power increase were reviewed against the unit design capability, and it was determined that no system, structure, or component would experience an adverse reduction in operating margin to the design conditions or loads. Details of this review are contained in Enclosures B and C. As a result, it was determined that there would be no impact on any estimated component reliabilities, and therefore no impact on the resultant PRA core damage frequency.

Based on the above, PG&E believes there is reasonable assurance that the health and safety of the public will not be adversely affected by the proposed change.

E. NO SIGNIFICANT HAZARDS EVALUATION

PG&E has evaluated the no significant hazards considerations (NSHC) involved with the proposed amendment, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

"The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or a testing facility involves no significant hazards considerations, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety."

The following evaluation is provided for the NSHC standards:

1. *Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?*

   All previously evaluated accidents have been reviewed for the proposed increase in Unit 1 power rating, and these reviews are summarized in WCAP-14819, "Pacific Gas and Electric Company Diablo Canyon Power Plant, Unit 1 3425 MWt Uprating Program Licensing Report." The majority of the Diablo Canyon Power Plant (DCPP) accident analyses already bound the higher power rating of Unit 2 combined with the lower reactor coolant system (RCS) flow rate of Unit 1. Hence, the uprate has no impact on these previously evaluated accidents. This is also true of dose assessment, which remains based on the original 3568 MWt core source terms and is not impacted by the uprate.

   The previously evaluated accidents that are impacted by the uprate are large break loss-of-coolant accident (LOCA), small break LOCA, the OTΔT/OPΔT setpoint calculations, and accidental depressurization of the RCS. The large break LOCA was reanalyzed for uprated conditions using best estimate methodology. The reanalysis demonstrated no increase in consequence and was approved by the NRC in License Amendments 121 (Unit 1) and 119 (Unit 2). The small break LOCA was also reanalyzed, and continues to demonstrate a large margin to peak clad temperature limits. The current OTΔT/OPΔT setpoints are bounding for the Unit 1 uprated power conditions based on revising the reactor core safety limits in TS Figure 2.1.1-1 to credit the exclusive use of Vantage 5 fuel. The accidental RCS depressurization reanalysis shows that the departure from nucleate boiling ratio remains above the applicable limit value. In summary, no design or analysis acceptance criteria will be exceeded, the functional integrity of all plant systems are unaffected, and there is no impact on the integrity of the fission product barriers or assumed dose source terms. Therefore, the consequences of all previous evaluated accidents are not substantially increased.

It was determined that there would be no impact on any component reliabilities assumed in the PRA model, and therefore no impact on the resultant core damage frequency. The PRA model envelopes both units, based on using the originally higher rated Unit 2 power level.

The operation impacts of the proposed power increase were reviewed against the unit design capability, and it was determined that no system, structure, or component would exceed design conditions or loads. While the low pressure turbines see a small (less than 1.5°F) increase in temperature, the effect on missile generation probability is not significant. There is no significant increase in the probability of component failure, offsite power loss, or any other accident initiator. Therefore, the probability of all previously evaluated accidents is not substantially increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?*

Normal operation will not be substantially impacted by increasing the Unit 1 licensed power rating to match Unit 2. Procedures will be essentially unchanged, or where changes are required, they will be made to more closely resemble those in effect at Unit 2. Training will communicate all impacts to personnel and the plant simulator will be updated to match the power level of both Units 1 and 2. There is, therefore, no possibility of a new or different kind of accident related to human performance.

Plant systems, structures, and components have been evaluated for the proposed uprate. Most have identical counterparts in operation at Unit 2 at this higher power level. A few are slightly different, such as the generator cooling system, and for these the design margins have been reviewed and found to be acceptable. Therefore, there is no possibility of a new or different kind of accident related to system, structure, or component performance.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does the change involve a significant reduction in a margin of safety?*

The proposed changes do not involve a significant reduction in a margin of safety because the margin of safety associated with plant parameters as verified by the results of the accident analyses are within acceptable limits. As mentioned, most analyses demonstrating adequate margins of safety already assume the higher thermal power rating of Unit 2 and bound Unit 1 at the uprated thermal power conditions. The few transients that are reanalyzed meet the applicable acceptance criteria.

The reactor core safety limits specified in TS Figure 2.1.1-1 envelope operation with both 17x17 standard and 17x17 Vantage 5 fuel. The proposed change revises the reactor core safety limits in Figure 2.1.1-1 to credit the exclusive use of Vantage 5 fuel. These revised safety limits will continue to satisfy fuel design criteria. The current OTΔT and OPΔT setpoints provide adequate margin to the revised reactor core safety limits at the uprated Unit 1 conditions, which include a slightly higher nominal full power T_{avg} in Notes 1 and 2 to ITS Table 3.3.3-1.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the above, PG&E concludes that the change proposed by this LAR satisfies the NSHC standards of 10 CFR 50.92(c), and accordingly a no significant hazards finding is justified.

G. ENVIRONMENTAL EVALUATION

PG&E has evaluated the proposed change and determined the change does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.