Pacific Gas and
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January 13, 2006

PG&E Letter DCL-06-006

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

Docket No. 50-323, OL-DPR-82
Diablo Canyon Unit 2
License Amendment Request 06-02
Revision to Technical Specification 5.6.5, "Core Operating Limits Report (COLR)"

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License No. DPR-82 for Unit 2 of the Diablo Canyon Power Plant. The enclosed license amendment request (LAR) proposes to revise Technical Specification (TS) 5.6.5, "Core Operating Limits Report (COLR)," by adding WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," dated January 2005, as an approved analytical method for determining core operating limits for Unit 2.

Pacific Gas and Electric Company (PG&E) submitted LAR 05-07, PG&E Letter DCL-05-146, dated December 16, 2005, to apply a plant-specific best-estimate loss-of-coolant accident analysis for Unit 1, using a methodology that is statistically different than that presented in WCAP-16009-P-A. Thus, this LAR applies only to Unit 2. Since both LARs request a change to the same TS page, the license amendment that is issued second will have to reflect the changes on the first.

Enclosure 1 contains a description of the proposed changes, the supporting technical analyses, and the no significant hazards consideration determination. Enclosures 2 and 3 contain marked-up and retyped (clean) TS pages, respectively. Enclosure 4 contains the Unit 2 large-break loss-of-coolant accident analysis peak clad temperature results.

PG&E has determined that this LAR does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

A001

The changes in this LAR are not required to address an immediate safety concern. PG&E requests approval of this LAR no later than January 13, 2007. PG&E requests the license amendment be made effective upon NRC issuance, to be implemented within 90 days from the date of issuance.

This communication contains no new or revised commitments.

If you have any questions or require additional information, please contact Mr. Stan Ketelsen at 805-545-4720.

Sincerely,

David H. Oatley
Vice President and General Manager

mjrm/4557
Enclosures
cc: Edgar Bailey, DHS
Terry W. Jackson
Bruce S. Mallett
Diablo Distribution
cc/enc: Alan B. Wang

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Unit 2	Docket No. 50-323 Facility Operating License No. DPR-82

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath states that he is Vice President and General Manager – Diablo Canyon of Pacific Gas and Electric Company; that he has executed license amendment request 06-02 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

David H. Oatley
Vice President and General Manager

Subscribed and sworn to before me this 13th day of January, 2006, by David H. Oatley, personally known to me or proved to me on the basis of satisfactory evidence to be the person who appeared before me.

Notary Public
County of San Luis Obispo
State of California

CHUCK MACKEY
Commission # 1397547
Notary Public - California
San Luis Obispo County
My Comm. Expires Feb 1, 2007

EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Facility Operating License DPR-82 for Unit 2 of the Diablo Canyon Power Plant (DCPP).

This License Amendment Request (LAR) proposes to revise Technical Specification (TS) 5.6.5, "Core Operating Limits Report (COLR)," by adding WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," dated January 2005, as an approved analytical method for determining core operating limits for Unit 2.

The proposed amendment will revise the analysis method used for the large-break loss-of-coolant accident (LBLOCA) by incorporating the use of a new approach, ASTRUM, for the treatment of parameter uncertainties.

2.0 PROPOSED CHANGES

TS 5.6.5 would be revised by adding the following referenced document to 5.6.5.b:

7. WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only).

TS 5.6.5.b.6 would also be revised to reflect the addition of TS 5.6.5.b.7. This change is editorial in nature only.

PG&E submitted LAR 05-07, PG&E Letter DCL-05-146, dated December 16, 2005, to apply a plant-specific best estimate loss-of-coolant accident analysis for Unit 1, using a methodology that is statistically different than that presented in WCAP-16009-P-A. Thus, this LAR applies only to Unit 2. Since both LARs request a change to the same TS page, the license amendment that is issued second will have to reflect the changes on the first.

The proposed TS changes are noted on the marked-up TS page provided in Enclosure 2. The proposed retyped TS page is provided in Enclosure 3.

3.0 BACKGROUND

TS 5.6.5.a states that core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle. TS 5.6.5.a requires the core operating limits to be documented in the COLR for the items listed in TS 5.6.5.a.1 through TS 5.6.5.a.8.

TS 5.6.5.b states that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, and specifically lists the analytical methods that may be used.

The regulations specified in 10 CFR 50.46(a)(1) identify calculation methodology requirements for nuclear power plant loss-of-coolant accident (LOCA) methodologies. Federal Regulation 10 CFR 50.46(c) identifies the types of processes which are required to assure that LOCA analyses performed for a given plant actually represent the plant. Section 50.46(a)(3)(i and ii) specify criteria to be applied and actions to be taken when significant changes or errors in parts of the plant-specific LOCA methodology, defined in accordance with 10 CFR 50.46(a)(1) and (c), are found to have accumulated.

When the licensee makes changes to its plant input model or finds significant errors in parts of the LOCA methodology covered by 10 CFR 50.46(a)(1) and (c), the licensee must reanalyze the plant's LOCA response. This is usually done by repeating the plant's LOCA analyses (reanalyzing) using a LOCA methodology approved for the plant, with changes and errors updated if the base LOCA methodology remains the same.

In PG&E Letter DCL-98-101, "10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes," dated July 24, 1998, PG&E provided new peak cladding temperature (PCT) results for a LBLOCA. The new values included a 67 degree PCT penalty. A schedule for reanalysis was not proposed at that time.

In PG&E Letter DCL-00-134, "Revised Schedule for Large Break Loss-of-Coolant Accident Reanalysis," dated October 19, 2000, PG&E committed to perform a new LBLOCA reanalysis due to the PCT penalty.

In PG&E Letter DCL-03-091, "10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes," dated July 24, 2003, PG&E provided the LBLOCA reanalysis results for Unit 1 as the "pending analysis of record." In addition, PG&E stated the following with respect to Unit 2:

> *It should also be noted that during the BELOCA reanalysis, Westinghouse identified that due to ECCS model changes, the Unit 2 PCT exceeded that of Unit 1 for several comparative cases. The current BELOCA analysis of*

record is based on a bounding plant methodology that established Unit 1 as the limiting plant, and the Unit 1 PCT results as bounding when applied to Unit 2.

Based on the reanalysis results with several comparative cases showing Unit 2 PCTs exceeding those of Unit 1, PG&E has determined that the bounding plant methodology is no longer appropriate for establishing the Unit 2 BELOCA analysis of record. Therefore, PG&E will perform a plant-specific BELOCA analysis for Unit 2 using the accepted methodology established in WCAP 12945 P-A, "Code Qualification Document for Best Estimate LOCA Analysis," Bajorek, S. M. et. al., 1998.

The Unit 2 BELOCA analysis will be completed in support of design changes to be implemented during the Unit 2 twelfth refueling outage. This outage is currently scheduled for the fall of 2004. These Unit 2 design changes include modifying the reactor vessel internals to provide baffle region core bypass flow in the upward direction instead of the current downward direction, and reducing the reactor coolant temperature in the upper head region.

In PG&E Letter DCL-04-017, "Supplement to 2003 10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes, Unit 2 BELOCA Analysis," dated March 2, 2004, PG&E stated that the design modifications had been deferred until the Unit 2 thirteenth refueling outage (2R13). As a result, PG&E committed to instead perform the plant-specific best-estimate LOCA (BELOCA) analysis for Unit 2 prior to 2R13, which is currently scheduled to begin in April 2006.

The NRC approved Revision 0, to WCAP-16009-P-A, by letter dated November 5, 2004, "Final Safety Evaluation for WCAP-16009-P, Revision 0, 'Realistic Large Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)' (TAC NO. MB9483)."

4.0 TECHNICAL ANALYSIS

This section summarizes the application of the Westinghouse ASTRUM BELOCA evaluation model to DCPP Unit 2 for LBLOCAs. Westinghouse analyzed the DCPP Unit 2 LBLOCA using the approved ASTRUM methodology. The analysis was performed in compliance with all the NRC safety evaluation conditions and limitations identified in NRC letter dated November 5, 2004. The WCOBRA/TRAC model used in the ASTRUM analysis was developed to include the upflow and upper head temperature reduction design modifications being implemented in 2R13. The model also incorporates the design for replacement steam generators (scheduled for the Unit 2 fourteenth refueling outage (2R14)) which bounds the current Model 51 steam generators.

WCAP-16009-P-A describes a realistic emergency core cooling system evaluation model for demonstrating plant compliance with 10 CFR 50.46 for postulated plant specific LBLOCA transients. WCAP-16009-P-A uses a revised statistical approach used for developing the PCT, local maximum oxidation (LMO), and core wide oxidation (CWO) results at the 95th percentile. The method is still based on the Code Qualification Document (CQD) methodology and follows the steps in Code Scaling, Applicability, and Uncertainty (CSAU) evaluation methodology. However, the uncertainty analysis (Element 3 in CSAU) is replaced by a technique based on order statistics. The ASTRUM evaluation model is documented in WCAP-16009-P-A.

The ASTRUM methodology requires the execution of 124 calculations to simultaneously bound the 95th percentile of the PCT, LMO, and CWO parameters with a 95 percent confidence level. These parameters are needed to satisfy 10 CFR 50.46 criteria.

Table 1 lists the major plant parameter assumptions used in the BELOCA analysis for DCPP Unit 2 and Table 2 summarizes the results of the ASTRUM analysis.

From the 124 calculations, run 69 proved to be the limiting PCT case, run 40 proved to be the limiting LMO case, and run 113 proved to be the limiting CWO case.

Figure 1 provides the DCPP Unit 2 BELOCA analysis power shape operating space envelope. The scatter plot presented in Figure 2 shows the influence of the effective break area on the final PCT. The effective break area is calculated by multiplying the discharge coefficient (C_D) with the sample value of the break area, normalized to the cold-leg cross sectional area. Figure 2 is provided to illustrate that the break area is a significant contributor to the variation in PCT.

Figures 3, 4, and 5 are presented to compare the reference transient case with the final ASTRUM limiting cladding transients for PCT, LMO, and CWO, respectively. The reference transient case represents the result of the limiting plant evaluations and confirmatory calculations, and forms the basis for the ASTRUM uncertainty analysis to establish the PCT at the 95th percentile.

Based on the results presented in Table 2, PG&E concludes that DCPP Unit 2 continues to maintain a margin of safety to the limits prescribed by 10 CFR 50.46.

Implementation of the approved LBLOCA methodology will result in a change to T.S. 5.6.5.b, to add a reference to WCAP-16009-P-A, as an approved LBLOCA analysis methodology (for Unit 2 only) as shown in the marked-up and retyped TS pages in Enclosures 2 and 3. This LAR does not require a change to the

COLR document, since using the methodology does not result in any new operating limits. The Unit 2 LBLOCA analysis PCT results are summarized in Enclosure 4.

Table 1
Major Plant Parameter Assumptions Used in the BELOCA Analysis

Parameter	Value
Plant Physical Description	
• SG Tube Plugging	≤ 15%
Plant Initial Operating Conditions	
• Reactor Power	≤ 100.3% of 3468 MWt
Fluid Conditions	
• Tavg	560.0 °F ≤ T_{ave} ≤ 582.6 °F
• Pressurizer Pressure	2190 psia ≤ P_{RCS} ≤ 2310 psia
• Reactor Coolant flow	≥ 85,000 gpm/loop
• Accumulator Temperature	85 °F ≤ accumulator temperature ≤ 120 °F
• Accumulator Pressure	579 psia ≤ P_{ACC} ≤ 664 psia
• Accumulator Water Volume	814 ft^3 ≤ V_{acc} ≤ 886 ft^3
• Minimum accumulator boron	≥ 2200 ppm
Accident Boundary Conditions	
• Safety injection flow	Minimum
• Safety injection temperature	46 °F ≤ SI Temperature ≤ 90 °F
• Safety Injection Delay Time	≤ 17 seconds (with offsite power) ≤ 27 seconds (with LOOP)
• Containment pressure	Bounded (minimum)
• Single failure	Loss of one ECCS train
• Control rod drop time	N/A

Table 2
Best Estimate Large Break LOCA Results

10 CFR 50.46 Requirement	Value	10 CFR 50.46 Criterion
95/95 PCT (°F)	1,872	< 2,200
95/95 LMO (%) [1]	1.64	< 17
95/95 CWO (%) [2]	0.17	< 1

1. Local Maximum Oxidation
2. Core Wide Oxidation

Figure 1
DCPP Unit 2 PBOT/PMID Analysis and Operating Limits

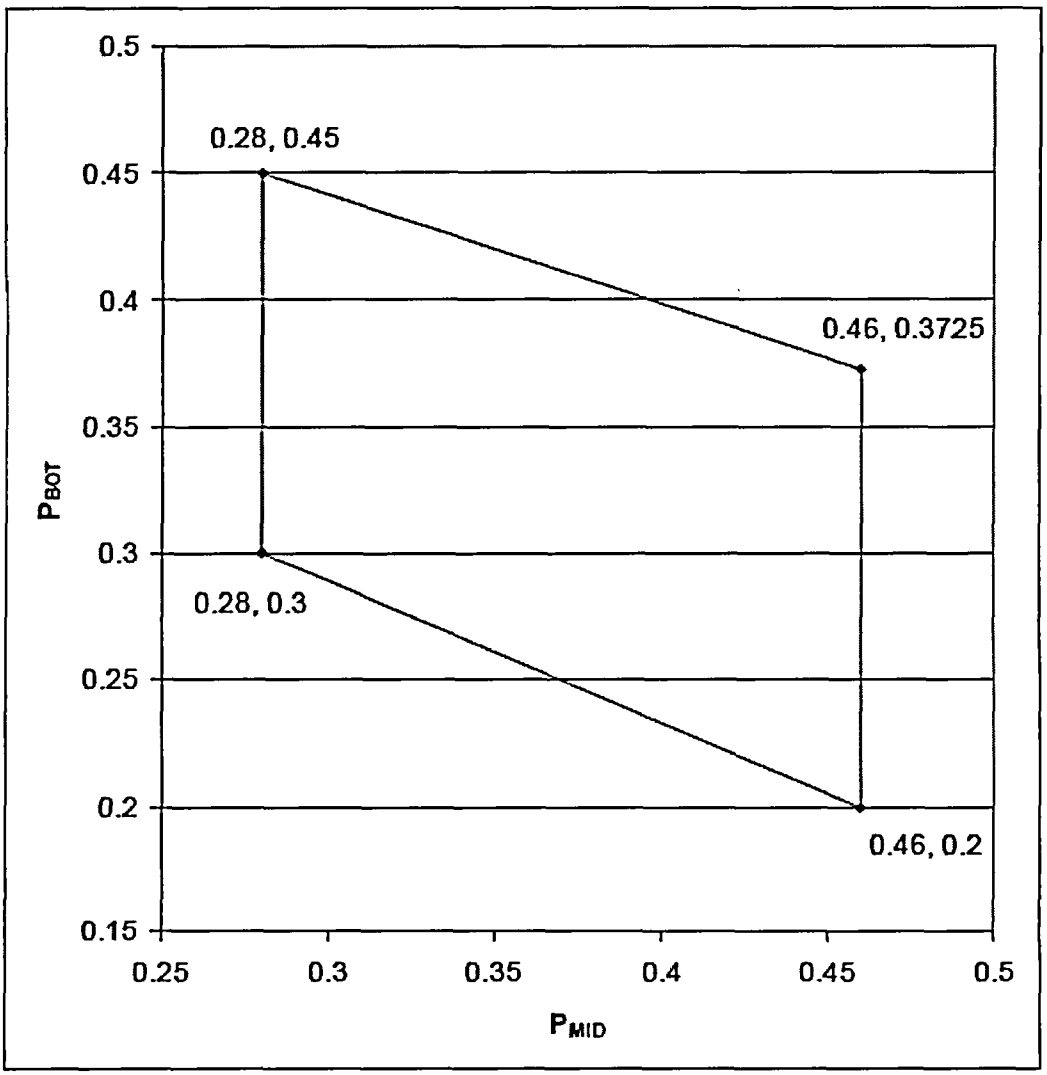


Figure 2
Diablo Canyon Power Plant Unit 2 HOTSPOT Peak Cladding Temperature (Double Ended Guillotine Cold Leg Break and Split Break) Versus Effective Break Area Scatter Plot

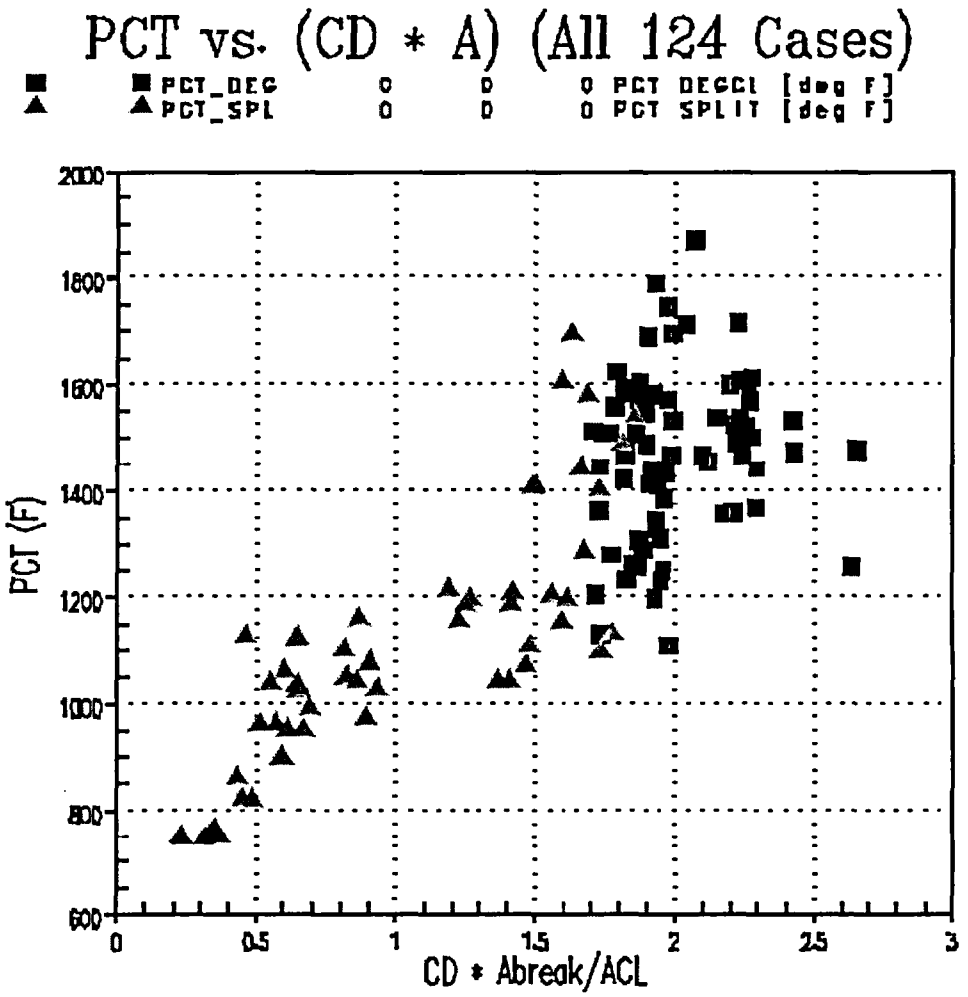


Figure 3
Diablo Canyon Power Plant Unit 2 WCOBRA/TRAC Peak Cladding Temperature for ASTRUM Run 69 and Reference Transient

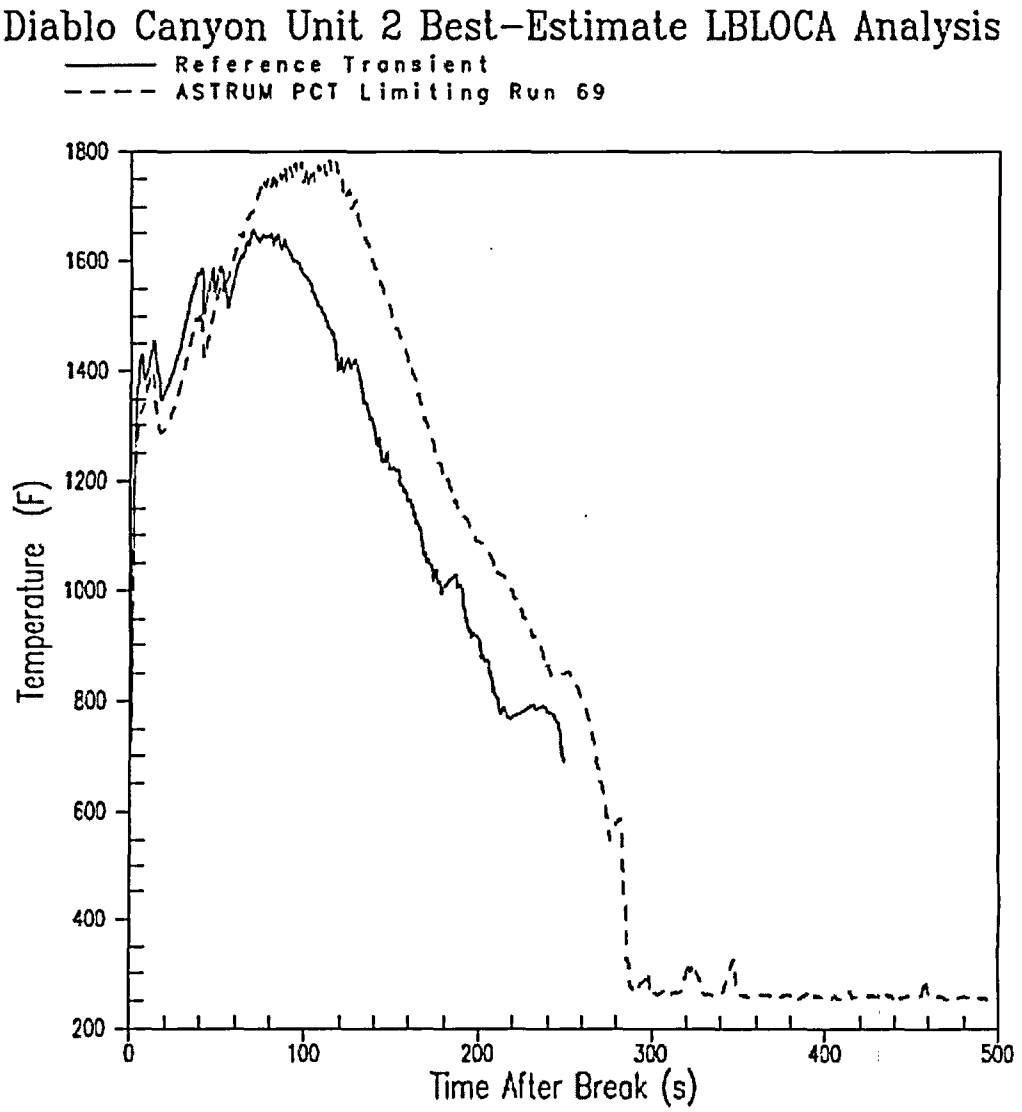


Figure 4
Diablo Canyon Power Plant Unit 2 WCOBRA/TRAC Peak Cladding Temperature for ASTRUM Run 40 and Reference Transient

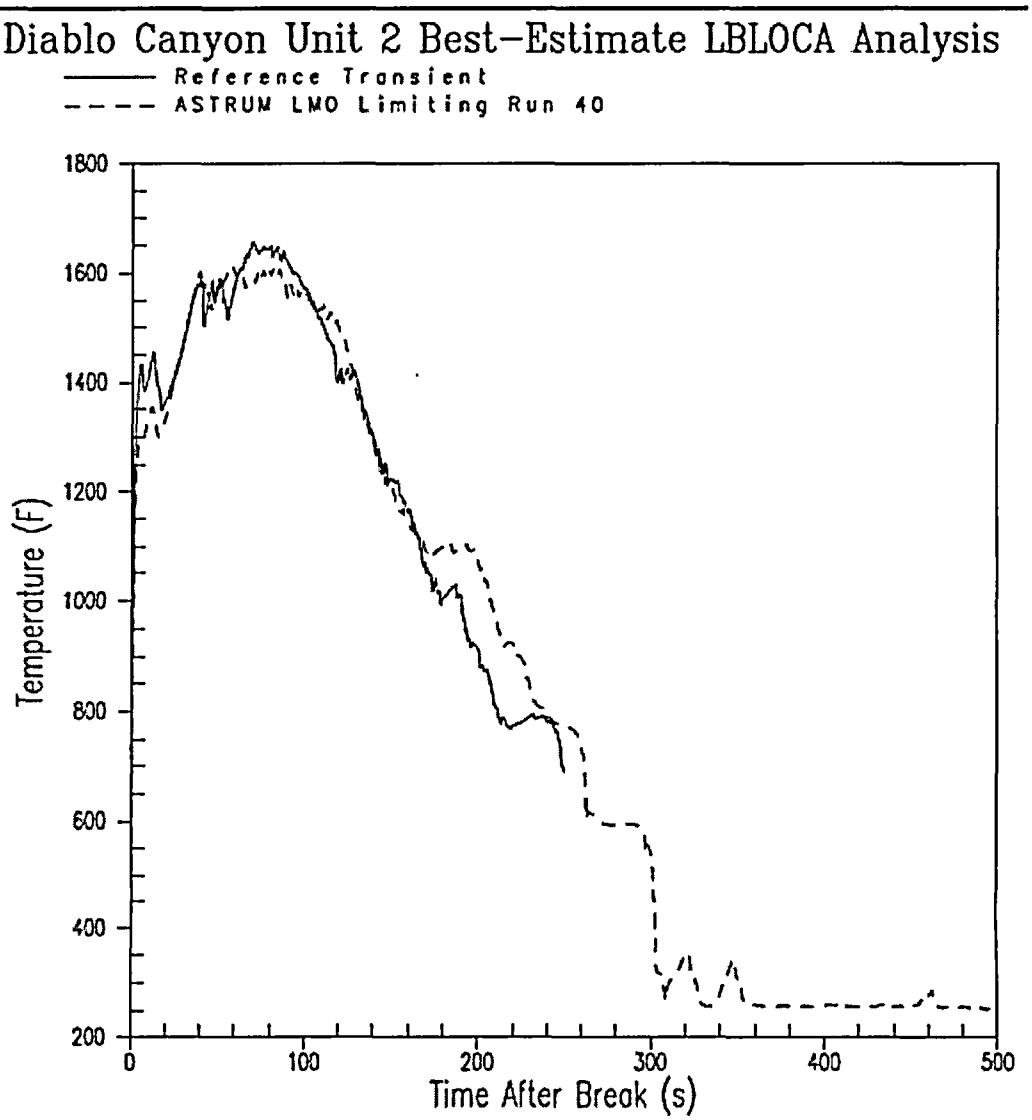
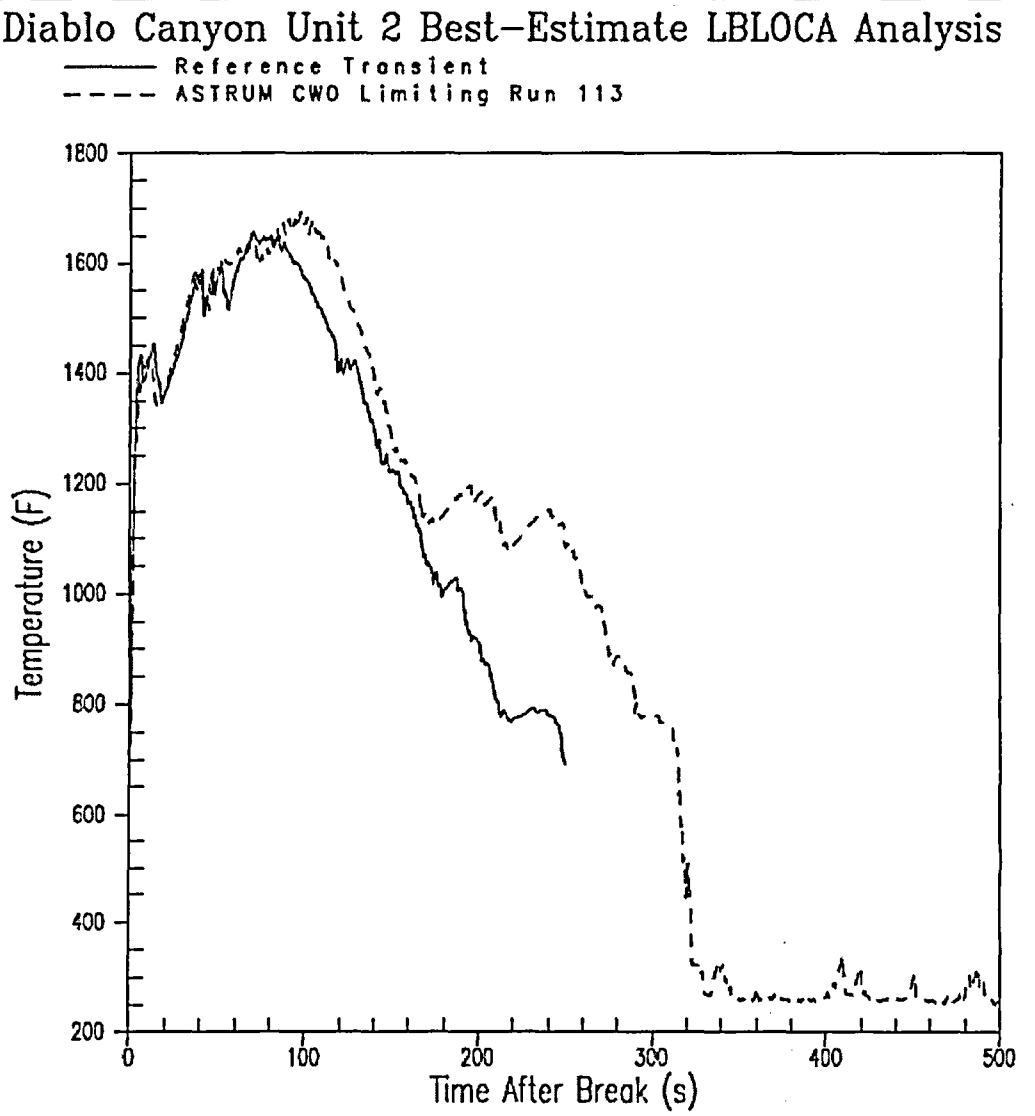


Figure 5
Diablo Canyon Power Plant Unit 2 WCOBRA/TRAC Peak Cladding Temperature for ASTRUM Run 113 and Reference Transient



5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No.

The proposed change to allow the use of the best estimate loss-of-coolant accident (LOCA) analysis methodology using the automated statistical treatment of uncertainty methodology (ASTRUM) does not involve a physical alteration of any plant equipment or change operating practice at Unit 2 of Diablo Canyon Power Plant (DCPP). Therefore, there will be no increase in the probability of a LOCA. The consequences of a LOCA are not being increased.

The plant conditions assumed in the analysis are bounded by the design conditions for all equipment in Unit 2. That is, it is shown that the emergency core cooling system is designed so that its calculated cooling performance conforms to the criteria contained in 10 CFR 50.46, paragraph b. No other accident is potentially affected by this change.

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

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

Response: No.

The proposed change would not result in any physical alteration to any Unit 2 system, and there would not be a change in the method by which any safety related system performs its function. Analyses of transient events have confirmed that no transient event results in a new sequence of events that could lead to a new accident scenario. The parameters assumed in the analysis are within the design limits of existing plant equipment.

In addition, employing the ASTRUM methodology does not create any new failure modes that could lead to a different kind of accident. The design of all systems remains unchanged and no changes are being made to any reactor protection system or engineered safeguard features actuation setpoints.

Based on this review, it is concluded that no new accident scenarios, failure mechanisms or limiting single failures are introduced as a result of the proposed changes.

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

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

Response: No.

It has been shown that the analytic technique used in the analysis realistically describes the expected behavior of the DCPP Unit 2 reactor system during a postulated LOCA. Uncertainties have been accounted for as required by 10 CFR 50.46. A sufficient number of LOCAs with different break sizes, different locations, and other variations in properties have been analyzed to provide assurance that the most severe postulated LOCAs were analyzed. The analysis has demonstrated that all acceptance criteria contained in 10 CFR 50.46 paragraph b continue to be satisfied.

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

Based on the above evaluation, PG&E concludes that the proposed change presents a no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 10 CFR 50.46 Evaluation

In accordance with 10 CFR 50.46, the conclusions of the best estimate large break LOCA analysis show that there is a high level of probability the following criteria are met:

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature, PCT) will not exceed 2,200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

5. After successful initial operation of the emergency core cooling system, the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed amendment and has determined that the proposed amendment 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 amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 References

1. WCAP-16009-P-A, Revision 0, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," dated January 2005

2. NRC Letter, "Final Safety Evaluation for WCAP-16009-P, Revision 0, 'Realistic Large-Break LOCA Evaluation Methodology Using Automated Statistical Treatment of Uncertainty Method (ASTRUM)', (TAC No. MB9483)", dated November 5, 2004

3. PG&E Letter DCL-98-101, "10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes," dated July 24, 1998

4. PG&E Letter DCL-00-134, "Revised Schedule for Large Break Loss-of-Coolant Accident Reanalysis," dated October 19, 2000

5. PG&E Letter DCL-03-091, "10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes," dated July 24, 2003

6. PG&E Letter DCL-04-017, "Supplement to 2003 10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes, Unit 2 BELOCA Analysis," dated March 2, 2004

7.2 Precedent

A similar license amendment request was submitted by Southern Company for their Joseph M. Farley Nuclear Plant, Units 1 and 2, "Technical Specification Amendment Request to Incorporate Best Estimate LOCA Analysis Using ASTRUM," by letter dated October 6, 2005.

Entergy made a similar request for Indian Point, Unit 2, "Proposed Change to Indian Point 2 Technical Specifications Regarding LBLOCA Analysis Methodology," by letter dated September 26, 2005.

Proposed Technical Specification Changes (mark-up)

INSERT 1

7. WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only).

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, February 1994 (Westinghouse Proprietary),

2. WCAP-9272-P-A, Westinghouse Reload Safety Evaluation Methodology, July 1985 (Westinghouse Proprietary),

3. WCAP-8385, Power Distribution Control and Load Following Procedures, September 1974 (Westinghouse Proprietary),

4. WCAP-10054-P-A, Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code, August 1985. (Westinghouse Proprietary), and

5. 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 (Westinghouse Proprietary), and

, and

INSERT 1

6. WCAP-12945-P-A, Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis, June 1996. (Westinghouse Proprietary).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

Proposed Technical Specification Changes (retyped)

Remove Page	Insert Page
5.0-27	**5.0-27**

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-10216-P-A, Revision 1A, Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification, February 1994 (Westinghouse Proprietary),

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6. WCAP-12945-P-A, Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis, June 1996. (Westinghouse Proprietary), and

7. WCAP-16009-P-A, Revision 0, Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM), January 2005. (Westinghouse Proprietary) (Unit 2 Only).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

Initial Unit 2 Large-Break Loss-of-Coolant Accident (LBLOCA) Analysis PCT Results

Pending Analysis of Record
DCPP UNIT 2 PEAK CLADDING TEMPERATURE MARGIN UTILIZATION

BEST ESTIMATE LARGE-BREAK LOCA		**PG&E Letter[1]**
A. ANALYSIS OF RECORD	1872°F	Reference 1
	ΔPCT	
B. PERMANENT 10 CFR 50.46 ECCS MODEL ASSESSMENTS[2]		
1. None[3]	0°F	
C. 10 CFR 50.59 AND 10 CFR 50.92 SAFETY EVALUATIONS		
1. None	0°F	
D. OTHER MARGIN ALLOCATIONS		
1. None	0°F	
LICENSING BASIS PCT + MARGIN ALLOCATION PCT	1872°F	

Reference 1: Westinghouse Letter PGE-05-100, "Pacific Gas and Electric Company, Diablo Canyon Units 1 and 2, BELOCA ASTRUM Analysis - Transmittal of Final WCAP-16443-P," November 17, 2005

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1 For those issues that have been previously reported under 10 CFR 50.46, a PG&E letter number is listed.

2 Only permanent assessments of peak cladding temperature (PCT) margin are included. Temporary PCT allocations that address current loss-of-coolant accident (LOCA) model issues are not considered with respect to 10 CFR 50.46 reporting requirements.

3 The Analysis of Record incorporates the design of the replacement steam generators scheduled for 2R14 which bounds the current Model 51 steam generators.