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PG&E Letter DCL-00-098

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

Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Response to NRC Request for Additional Information Regarding License Amendment Request (LAR) 99-03, Unit 1 Reactor Core Thermal Power Uprate

Dear Commissioners and Staff:

In a letter dated June 21, 2000, the NRC staff identified additional technical information required in order for them to complete their evaluation associated with the Diablo Canyon Power Plant Unit 1 reactor core thermal power uprate. PG&E's response to the request for additional information is included in Enclosure A. This additional information does not affect the results of the safety evaluation performed for LAR 99-03 (PG&E Letter DCL-99-170, dated December 31, 1999).

Also enclosed are WCAP-14775, "Best Estimate Analysis of the Large Break Loss-of-Coolant Accident for Diablo Canyon Power Plant Units 1 & 2 to Support 24-Month Cycles and Unit 1 Uprating," dated January 1997 (Enclosure B); WCAP-13907, "Analysis of Containment Response Following Loss-of-Coolant Accidents for Diablo Canyon Units 1 and 2," dated December 1993 (Enclosure C), including addendum; and WCAP-13908, "Analysis of Containment Response Following Steamline Break Accidents for Diablo Canyon Units 1 and 2," dated December 1993 (Enclosure D). These documents are being provided in response to requests from the NRC staff.

WCAP-14775 contains information proprietary to Westinghouse Electric Company. Therefore, Enclosure E contains a Westinghouse Application for Withholding Proprietary Information from Public Disclosure, a Proprietary Information Notice, and accompanying Affidavit CAW-00-1403 signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from the public disclosure by the Commission, and it addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.790 of the Commission's regulations.

AP01

PG&E requests that the Westinghouse proprietary information be withheld from public disclosure in accordance with 10 CFR 2.790.

Correspondence with respect to the Westinghouse report listed above, the Copyright Notice, or the supporting Westinghouse affidavit should reference CAW-00-1403 and should be addressed to H. A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Sincerely,

David H. Oatley

cc: Edgar Bailey, DHS (Enclosure A only)
Steven D. Bloom
Ellis W. Merschoff (Enclosure A only)
David Proulx (Enclosure A only)
Diablo Distribution (Enclosure A only)

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 this response to an NRC request for additional information regarding 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 7th day of July, 2000.

Notary Public
State of California
Country of San Luis Obispo

LISA M. NELSON
Commission # 1258488
Notary Public - California
San Luis Obispo County
My Comm. Expires Mar 31, 2004

PG&E Responses to Draft Request for Additional Information For DCPP-1 Power Uprate (PG&E Letter DCL-99-170)

Annular Fuel Pellets Blankets:

Section 3.1.2 of WCAP-14819 (Enclosure B) states that the Diablo Canyon 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 DCPP-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 DCPP-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).

Question 1:

If the fully enriched annular fuel pellet blanket fuel design to be used for uprate reload is described in a separate report, provide 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 DCPP-1 Uprating Program.

PG&E Response to Question 1:

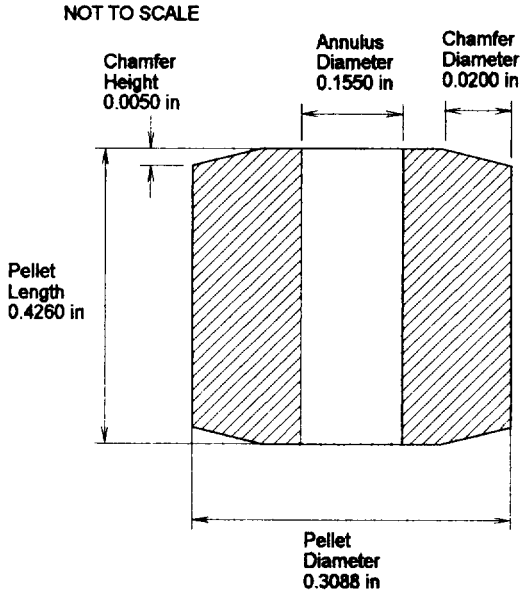
The fully enriched annular fuel pellet blanket design does not represent a new fuel design feature but has been in use at DCPP-1 since Cycle 9 (April 1997). The statement on Page 3-6 of WCAP-14819 (Enclosure B) refers to the fact that the annular fuel pellets are now being explicitly modeled in the small break loss of coolant accident (SBLOCA) analysis. Prior to this latest analysis, the potential effects of annular pellets on the SBLOCA results were conservatively bounded by applying a 10°F penalty to the calculated peak clad temperature (PCT). This statement should not be interpreted to indicate that the use of annular fuel pellets is being introduced for the first time in conjunction with the Unit 1 power uprate.

Axial blankets represent the top and bottom six inches of the active fuel region which usually

have a lower enrichment than nominal in order to optimize the fuel utilization in these outer regions of the core where the flux is less. The annular pellet blanket incorporates a cylindrical opening in the center of the pellet to increase the available volume inside the fuel rod to gain margin for the internal pressurization that occurs as a function of fuel burnup. The introduction of annular pellet blankets was provided as a supplement to WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995, Section J, which has been reviewed and accepted by the NRC. Annular pellet blankets are an integrated feature of the VANTAGE+ fuel design and are compatible with the cladding (currently ZIRLO) and range of pre-pressurization values associated with this fuel design. Annular pellet blankets may be designed with natural uranium (0.711 w/o), or uranium enrichment ranging from 2.5 w/o up to the fully enriched value of 5 w/o. The annular pellet blanket lengths remain the same, six inches at the top and six inches at the bottom as solid pellet blankets. The annular pellet dimensions are provided in the following, Figure 1-1.

The annular pellet blanket design feature was evaluated in accordance with 10 CFR 50.59 for the Cycle 9 core reload design using the NRC accepted reload safety analysis methodology in WCAP-9272-P-A, "Reload Safety Evaluation Methodology," July 1985. The evaluation concluded that the annular pellet blankets did not impact any non-LOCA, thermal hydraulic, or large break LOCA analysis results. These conclusions were based on fact that the analysis results were not affected by the explicit internal design of the fuel pellet, but used limiting heat flux and power distribution assumptions that were bounding for the calculated core design values. The evaluation did assign a 10 °F PCT penalty to the existing SBLOCA results to conservatively bound any potential impact of enriched annular pellet blankets until the Westinghouse model was revised to explicitly model this design feature. As discussed in more detail in Responses 14 and 9, Westinghouse and PG&E jointly generate and review the Reload Safety Analysis Checklist (RSAC) to verify that all applicable non-LOCA and LOCA reload safety limits remain bounding for the calculated core design values and including the effects of enriched annular pellet blankets. In summary, a topical report was not submitted to the NRC and the Unit 1 uprate license amendment request does not include a safety evaluation specifically for this fuel design feature, since it was already incorporated into the licensing basis under 10 CFR 50.59.

Figure 1-1: Annular Fuel Pellet Dimensions



Question 2:

Section 6 of WCAP-14819 states that the core design, thermal and hydraulic evaluations are evaluated for DCPP-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?

PG&E Response to Question 2:

As discussed in Response 1, the required cycle specific core design and thermal-hydraulic evaluations were done for DCPP-1, Cycles 9 and 10. These reload design evaluations addressed the use of annular fuel pellets in the axial blanket region. The evaluation determined that using enriched annular blankets did not impact any nuclear design or thermal hydraulic design performance criteria, and did not constitute an unreviewed safety question per 10 CFR 50.59.

Question 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 DCPP-1 power uprate be limited to the ZIRLO cladding?

PG&E Response to Question 3

It should be noted that the original PG&E reference to "Vantage 5+" fuel in the License Amendment Request (LAR) Enclosure C, Item 4 was incorrect. The appropriate title is "Vantage+" fuel which represents the Westinghouse marketing name for "Vantage 5" fuel which incorporates several enhanced design features as presented in WCAP-12610-P-A, the most notable being ZIRLO cladding. The original "Vantage 5" fuel design was first introduced into the Unit 1 and Unit 2 Cycle 4 core reloads respectively, while the "Vantage+" fuel design was introduced during the Unit 1 and Unit 2 Cycle 9 core reloads, respectively. In summary, the appropriate terminology for the current DCPP fuel design with respect to this LAR is "Vantage+", while "Vantage 5" is applicable for historical discussions related to core reloads prior to Cycle 9.

The Unit 1 uprate evaluations are applicable to both Zirc-4 and ZIRLO fuel cladding designs. PG&E anticipates that most future cores will consist exclusively of ZIRLO clad fuel, so the Westinghouse uprate evaluation was based on an all ZIRLO core. However, DCPP currently has fuel with Zirc-4 cladding that could be used in the core, and therefore, the option to use it following a cycle-specific evaluation is maintained.

During each reload cycle design, all fuel regions are evaluated for the projected maximum cycle burnup to ensure that all fuel rod design criteria are met. The paragraph in Section 6.3.2 is stating that if Zirc-4 clad fuel is used in a cycle design, a cycle-specific analysis is required to ensure the cladding corrosion, as predicted by the now-implemented Zirc-4 cladding corrosion model, meets all corrosion-related design limits. As discussed by Westinghouse in WCAP-14819, the fuel cladding corrosion model evaluation is essentially a function of fuel burnup and operating temperature. This evaluation requirement for Zirc-4 fuel is not uprate related, and would have to be performed for any future use of Zirc-4 clad fuel.

Since PG&E currently plans to load 17 fuel assemblies with Zirc-4 cladding into Unit 1 Cycle 11 (from Regions 8A, 9A, and 10A), these assemblies will be evaluated as outlined in WCAP-14819. Also, in future cycles beyond Cycle 11, it is possible that a limited number of fuel assemblies with Zirc-4 cladding will be reinserted as needed. Therefore, cycle-specific evaluations of ZIRLO and Zirc-4 clad fuel will be performed to ensure that all fuel rod design criteria are met for the Unit 1 uprate conditions.

Question 4:

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

PG&E Response to Question 4:

The fuel description contained in the technical specification (TS) Design Features section was added by License Amendments 104 and 103, for Units 1 and 2 respectively, and is consistent with the standard TS language in NUREG-1431. The standard language does not discuss fuel-specific fuel design features but states that "Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods, and shown by tests or analysis to comply with all fuel safety design bases." TS 4.2.1, "Fuel Assemblies," was not revised to reflect the use of annular pellet blankets since this represents a level of detail not included in the standard TS language.

LBLOCA Analysis:

Question 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 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 amendment 121 and 119 the review and approval of WCAP-14775 is described.

PG&E Response to Question 5:

The Best Estimate Loss of Coolant Accident (BELOCA) methodology was accepted for use at DCPP by the NRC in license amendments (LAs) 121 and 119, issued February 13, 1998, and was incorporated into the DCPP licensing basis prior to the submittal of the Unit 1 uprate license amendment. Enclosure C was included to provide a convenient technical summary of the recent and related BELOCA amendment. The 1976°F PCT value as discussed in Enclosure C is the original "analysis of record" 95 percent probability PCT value for Unit 1 as submitted to the NRC in the BELOCA LAR 97-08 (PG&E letter DCL-97-030 dated May 14, 1997). Enclosure A was prepared to provide a summary of the uprate LA including the latest related analysis results such as those for BELOCA. Therefore, Enclosure A identifies that the latest "net" Unit 1 PCT, as reported per 10 CFR 50.46, is 2043°F. This value includes a 67°F PCT penalty as documented in the 1998 annual 10 CFR 50.46 report to the NRC (PG&E letter DCL-99-096, dated July 26, 1999).

It should be noted that DCPP recently provided a 10 CFR 50.46 30-Day report to the NRC (PG&E letter DCL-00-051, dated April 5, 2000) based on a Westinghouse revision to the BELOCA PCT reporting methodology. The new methodology now reports a separate BELOCA PCT value for both the Reflood 1 and the Reflood 2 analysis periods. In the process of calculating these separate reflood PCT values, Westinghouse evaluated the penalties separately for each period. Therefore, the latest BELOCA values for Unit 1 as reported in DCL-00-051 are 2009°F for the Reflood 1 PCT, and 1964°F for the Reflood 2 PCT.

LAs 121 and 119 allow DCPP to use the NRC accepted BELOCA methodology as established in WCAP-12945. WCAP-14775, "Best Estimate Analysis of the Large Break Loss of Coolant Accident for Diablo Canyon Power Plant Units 1 & 2 to Support 24-Month Fuel Cycles and Unit 1 Uprating," January 1997, documents the DCPP plant specific results of using this NRC accepted BELOCA methodology for DCPP Unit 1 and Unit 2. While the LAR 97-08 (PG&E letter DCL-97-030) included a summary of the major input assumptions and the results for the DCPP BELOCA analysis, WCAP-14775 was not submitted to the NRC. PG&E has provided revisions to the appropriate pages of Enclosures A and C to DCL-99-170 to clarify the NRC review of the BELOCA methodology. PG&E has agreed to submit WCAP-14775 as part of the Unit 1 uprate LAR 99-03.

Question 6:

Enclosure A states that the difference in the reactor internal design between DCPP Units 1 and 2 resulted in lower 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?

PG&E Response to Question 6:

The following physical differences exist between the Unit 1 and Unit 2 reactor vessel internals.

DCPP Unit 1	DCPP Unit 2
"Top Hat" Upper Support Plate	Flat Upper Support Plate
Dome Lower Support Plate	Flat Lower Support Plate
Thermal Shield	Neutron Pads
Diffuser Plate	No Diffuser Plate

Due to the physical differences and their respective flow characteristics, Westinghouse developed separate models for the Unit 1 and Unit 2 reactor vessels in order to evaluate which design was the most limiting for the BELOCA analysis. WCAP-14775, Section 3-2-1, provides a detailed discussion of the two vessel models. In WCAP-14775 Section 4-4-9, Westinghouse summarized the comparison of the base model PCT results for both. While the Unit 2 model produced a slightly greater PCT for the initial blowdown period, the Unit 1 PCT was consistently greater for the most limiting reflood period. Consequently, the Unit 1 reactor vessel model was determined to be the most limiting, and was used for all subsequent calculations to determine the final 95th percentile PCT. Since the current limiting 95th percentile PCT is based on the Unit 1 reactor vessel, and includes a 3411 MWt core power rating, the BELOCA analysis results are conservatively bounding for Unit 1 at the uprated power conditions.

Question 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?

PG&E Response to Question 7:

PG&E letter DCL-96-163, dated August 6, 1996, notified the NRC (as part of a 10 CFR 50.46 30-Day Report), of the 2°F PCT penalty for LBLOCA for Unit 1. At the time the BELOCA analysis input assumptions were being developed, the Unit 1 core contained both fuel with

Zirc-4 and fuel with ZIRLO cladding. Consequently, the discussion in Enclosure C, Item 4, summarizes the basis for performing the BELOCA analysis for a core containing all ZIRLO cladding fuel, since this cladding had produced more limiting PCT results for the existing LBLOCA analysis methodology. Since the current BELOCA 95th percentile PCT is based on a core containing all ZIRLO clad fuel, the latest PCT results as reported in PG&E letter DCL-00-051, dated April 5, 2000, no longer reflect a PCT penalty associated with ZIRLO cladding fuel.

PG&E letter DCL-96-163, dated August 6, 1996, also notified the NRC of the 2°F PCT penalty for the SBLOCA results for Unit 1. As discussed in Response 12, SBLOCA reanalysis results were then submitted to the NRC in PG&E letter DCL-98-183, dated December 29, 1998, which also submitted LAR 98-09, requesting revision of the TS to allow use of NRC accepted addenda to WCAP-10054-P-A, "Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (specifically Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break LOCA ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model," July 1997) to determine core operating limits. LAR 98-09 was approved by the NRC in License Amendments 136 and 136, issued November 15, 1999, for Unit 1 and Unit 2, respectively. PG&E has updated the Final Safety Analysis Report (FSAR) to reflect the reanalysis. Since the current SBLOCA analysis is based on a core containing all ZIRLO clad fuel, the latest SBLOCA PCT as reported in PG&E letter DCL-00-051, dated April 5, 2000, no longer reflects a PCT penalty associated with ZIRLO cladding fuel.

Question 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?

PG&E Response to Question 8:

Annular pellets are not explicitly modeled in the LBLOCA transient analyses performed using accepted Westinghouse methods (BASH EM and BELOCA with WCOBRA/TRAC). The methodology is based on analyzing a bounding range of flux profiles and power distributions. Since the annular pellets have lower nominal enrichments than the solid pellets and are located only at the top and bottom six inches of the fuel rod, the limiting peak core power density does not occur in these blanket fuel regions. Explicitly modeling this small reduction in initial fuel stored energy due to annular pellets is expected to result in a small LBLOCA PCT benefit. The impact of annular pellets on the BELOCA analysis is bounded by ensuring that conservative flux shapes and relative power distributions modeled in the analysis remain bounding. Therefore, the core reload safety evaluation performed in accordance with the NRC accepted methodology from WCAP-9272-P-A, verifies that the actual calculated flux profiles, which include the effects of the annular pellet blankets, remain bounded by the conservative flux profiles in the BELOCA analysis.

Question 9:

Was the bounding BE LOCA analysis performed based on the power peaking factors specified in technical specifications (or 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?

PG&E Response to Question 9:

The BELOCA analysis was performed with more limiting peaking factors than currently licensed in the TS. The TS peaking factor limits are $F_Q \leq 2.45$ (COLR 2.5.1, TS 3.2.1) and $F_{\Delta H} \leq 1.59$ (COLR 2.6 ,TS 3.2.2). The peaking factors used in the BELOCA analysis were F_Q = 2.7 and $F_{\Delta H}$ = 1.7 as listed in Table 11-1 of WCAP-14775. This was done to establish available analysis margin for PCT evaluations, and to support the possibility of a future LAR to increase peaking factors. The Reload Safety Analysis Checklist (RSAC) is used to document the review of the applicable reload safety analysis parameters per the accepted WCAP-9272-P-A methodology. The RSAC Section 3.0 specifically lists the key LOCA related reload safety parameters including the associated core peaking factors assumed in the analysis. Westinghouse and DCPP jointly review the RSAC to ensure that the applicable licensing and analysis peaking factor limits are met for each core reload.

SBLOCA Analysis:

Question 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?

PG&E Response to Question 10:

Improved Technical Specification (ITS) surveillance requirement SR 3.5.2.4 states " Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head." ITS Bases 3.5.2 discusses the emergency core cooling system (ECCS) pump performance requirements with respect to their credit in the accident analysis, including SBLOCA, and identifies the following performance criteria: "...ECCS pumps are required to develop the indicated differential pressure when tested on recirculation flow:

CCP $\geq$ 2400 psid

SI pump $\geq$ 1455 psid

RHR pump $\geq$ 165 psid"

The ECCS pump curves used to calculate the minimum injection flow per Figure 3.1.2-2 of WCAP-14819 assume a conservatively degraded pump head capability compared to the ITS Bases specified values. Therefore, when an ECCS pump meets the minimum ITS differential pressure requirement, its actual flow capacity exceeds that assumed in the safety analysis.

In summary, the high head safety injection (HHSI) and intermediate head safety injection (IHSI) flow versus pressure values listed in Figure 3.1.2-2 of WCAP-14819 were calculated using pump performance curves which conservatively bound the minimum ECCS performance requirements specified in ITS Bases, Section 3.5.

Question 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?

PG&E Response to Question 11:

WCAP-14819, Sections 3.1.4, "Hot Leg Switchover (HLSO)," and 3.1.5, "Post LOCA Long Term Core Cooling (LTCC)," discuss how the Unit 1 uprated power level remains bounded by the existing analyses. The HLSO analysis determines the minimum time to initiate hot leg recirculation to prevent boron precipitation in the core, and also ensures that there is adequate ECCS cooling flow to remove the decay heat existing at that time. The core power rating only affects the ultimate core decay heat which must be removed as part of the long term core cooling (LTCC) requirements. Therefore, as indicated in Section 3.1.4, the Unit 1 uprate core power of 3411 MWt is already bounded by the hot leg switchover analysis which is based on the current Unit 2 power level of 3411 MWt.

In addition, the LTCC analysis establishes that the post LOCA recirculation sump pH remains within the required limits to prevent stress corrosion cracking of the stainless steel reactor coolant system (RCS), and to ensure that the iodine activity in the sump remains in solution and bounded by the offsite dose analysis assumptions. The LTCC analysis also ensures that the post LOCA sump recirculation boron concentration remains great enough to maintain the core subcritical. The Unit 1 core power uprate can only impact the LTCC analysis if the maximum core reactivity and associated RCS boron concentration must be increased. Since the LTCC analysis already bounds the maximum RCS boron concentration and core reactivity associated with the current Unit 2 core power of 3411 MWt, the Unit 1 power uprate to 3411 MWt is also bounded.

The term "RSAC" refers to the Westinghouse Reload Safety Analysis Checklist which is used to document the review of the applicable reload safety analysis parameters per the accepted WCAP-9272-P-A methodology. The "RSAC generation" represents the compilation and review of the calculated core reload design values and their comparison to the appropriate safety analysis limiting values to verify they remain conservatively bounding for the core reload. As discussed above, the post LOCA boron source data, which includes the maximum post LOCA critical boron value and the maximum RCS boron concentration, are specific acceptance criteria within the RSAC LOCA Section 3.0 that are verified for each core reload.

Question 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.

PG&E Response to Question 12:

As discussed in Enclosure C, Item 3, the SBLOCA was reanalyzed with the NRC accepted methodology, including the COSI condensation model as established in WCAP-10054-P-A, Addendum 2, Revision 1. The results of the SBLOCA reanalysis were submitted to the NRC in PG&E letter DCL-98-183, dated December 29, 1998 which also submitted LAR 98-09, requesting revision of the TS to allow use of NRC accepted addenda to WCAP-10054-P-A (specifically Addendum 2, Revision 1) to determine core operating limits. LAR 98-09 was accepted by the NRC in LAs 136 and 136, issued November 15, 1999, for Units 1 and 2, respectively. PG&E has updated the FSAR to reflect the reanalysis.

The differences between the previous SBLOCA analysis, which established the 4 inch case as limiting, and the current SBLOCA analysis, which established the 3 inch case, are discussed in the following paragraphs.

Table 12-1 compares the major input parameters between the previous SBLOCA analysis to the current analysis. In addition to increasing core power from 3338 MWt to 3411 MWt, the current SBLOCA analysis also assumed more limiting peaking factors. These changes resulted in a significantly larger peak linear heat rate of 15.0 kW/ft when compared to the previous value of 12.9 kW/ft. The other significant input assumption change is the revised ECCS injection flow versus RCS pressure curve as shown in WCAP-14819, Figure 3.1.2-2. Compared to the previous injection curve, the current ECCS injection profile provides slightly more flow at RCS pressures above 1400 psia, and considerably less flow at lower RCS pressures including runout conditions.

The methodology/model differences between the previous and the current SBLOCA analyses are the use of the new COSI condensation and broken loop safety injection (SI) models as

discussed in WCAP-14819, Section 3.1.2, and the direct incorporation of the existing PCT penalties into the analysis of record. The last 10 CFR 50.46 report summarizing the previous Unit 1 SBLOCA (PG&E letter DCL-98-101, dated July 24, 1998) summarized a total of 1152°F in various PCT assessments and evaluations resulting in a net PCT penalty of 68°F, increasing the Unit 1 PCT from 1275°F to 1343°F. These PCT penalties included several code and modeling errors and a number of evaluations for revised input assumptions. As discussed in WCAP-14819, the current SBLOCA analysis was performed to fulfill an NRC commitment to recalculate the analysis of record PCT. Consequently, directly incorporating the various code error corrections and the revised assumptions into the calculated analysis of record PCT was implemented through appropriate changes to the SBLOCA ECCS model.

Tables 12-2 and 12-3 provide a direct comparison of the calculated results and sequences of events for the previous and current 4-inch SBLOCA cases for Unit 1. Since a previous 3-inch case does not exist for Unit 1, the relative comparison of the calculated results and sequence of events for the previous and current 3-inch cases is provided for Unit 2 in Tables 12-2 and 12-3, respectively.

The combination of the methodology/model changes, ECCS injection flow, peaking factors, and uprate power level represent a substantial change in the Unit 1 SBLOCA model. As discussed in WCAP-14819, the competing effects and the complex nature of the SBLOCA physical phenomena make it very difficult to predict what establishes the limiting PCT case based solely on relative comparisons in input assumptions. The SBLOCA methodology basis for ensuring that the limiting PCT case has been appropriately selected is by analyzing the next larger and the next smaller size break.

As discussed in WCAP-14819, the 4-inch and 3-inch cases have always generated similar PCT results. Not unexpectedly, a review of the input assumptions, sequences of events and fuel cladding results for the relative 3-inch and 4-inch cases does not clearly establish any physical basis for one case becoming slightly more limiting than the other. However, the following general conclusions can be made with respect to the limiting RCS break size when considering SBLOCA phenomena separately.

1. Modeling the broken loop SI increases the subcooled liquid inventory in the RCS cold leg. This effect delays the time at which the RCS loop seal is cleared and results in increased total break flow and decreased RCS mass at the time this occurs. As the RCS break size decreases, the broken loop SI model predicts an increased cold leg liquid mass, a longer delay time until the loop seal is cleared, and for otherwise comparable transients would result in a greater PCT.

2. The application of the COSI condensation model is limited by the NRC safety evaluation report (SER) to RCS pressures between 550 and 1200 psia. This model generally provides a benefit to the SBLOCA analysis by increasing the condensation cooling effects of the cold leg vapor space that is generated by the injecting SI flow stream. Larger break sizes depressurize the RCS more rapidly and therefore receive the COSI condensation benefit earlier in the transient. This effect would tend to generate less limiting PCT values for larger RCS break sizes when compared to an equivalent case with a smaller break size.

3. The SBLOCA analysis models two paths for energy removal from the core. Heat transfer across the steam generator (SG) tubes is the less significant of the two paths, and may change direction during the transient. RCS break flow is the major heat removal mechanism, and a larger break size would generate a greater mass flow rate throughout the transient. Any increase in core power would represent more of a PCT penalty to a smaller RCS break size due to the more limited energy transfer capability associated with the smaller break flow.

Given the substantial differences between the previous and the current SBLOCA models, it is not considered significant that the limiting RCS break size has changed from the previous 4-inch diameter to the current 3-inch diameter. The increase in the Unit 1 uprate power level is not considered a significant factor on the resultant PCT when compared to the other analysis revisions which have been incorporated.

Table 12-1

Comparison of Previous and Current Unit 1 SBLOCA Input Parameters

Parameter	Previous	Current
Reactor core rated thermal power, (MWt)[1,2]	3338	3411
Peak linear power, (kW/ft)[2]	12.90	15.00
Total peaking factor (F_Q^T) at peak[2]	2.50	2.70
Power shape[2]	Scaled to 2-line K(z)	Figure 3.1.2-1
$F_{\Delta H}$	1.65	1.70
Fuel[3]	17x17V5	17x17V5
Accumulator water volume, nominal (ft^3/acc.)	850	850
Accumulator tank volume, nominal (ft^3/acc.)	1350	1350
Accumulator gas pressure, minimum (psia)	600	594
Pumped safety injection flow	FSAR Rev. 12 Figure 15.3-1	Figure 3.1.2-2
Steam generator tube plugging level (%)[4]	15	15
Thermal Design Flow/loop, (gpm)	85,900	85,000
Vessel average temperature w/uncertainties, (°F)	581.7	582.3
Reactor coolant pressure w/uncertainties, (psia)	2280	2310
Aux. feedwater flow rate/SG, (gpm)	320	205

1. Two percent is added to this power to account for calorimetric error. Reactor coolant pump heat is not modeled in the small break LOCA analyses.
2. This represents a power shape corresponding to a one-line segment peaking factor envelope, K(z), based on maximum F_Q^T.
3. Annular pellet blankets were explicitly modeled.
4. Maximum plugging level in any one or all steam generators.

Table 12-2
Comparison of Previous and Current SBLOCA Fuel Cladding Results
DCPP Unit 1/ DCPP Unit 2

	Unit 2 Previous 3-inch	Unit 2 Current 3-inch		Unit 1 Previous 4-inch	Unit 1 Current 4-inch
Peak Cladding Temp (°F)	1023	1293		1275	1264
Peak Cladding Temp Location (ft) [1]	12.0	11.25		12.0	11.00
Peak Cladding Temp Time (sec)	1868	1948		948	928
Local Zr/H_20 Reaction, Max (%)	0.076	0.25		0.133	0.09
Local Zr/H_20 Reaction Location(ft) [1]	12.0	11.25		12.0	11.00
Total Zr/H_20 Reaction (%)	<0.3	< 1.0		<0.3	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst		No Burst	No Burst

1. From bottom of active fuel.

Table 12-3
Comparison of Previous and Current SBLOCA Time Sequence of Events
DCPP Unit 1/ DCPP Unit 2

	Unit 2 Previous 3-inch	Unit 2 Current 3-inch		Unit 1 Previous 4-inch	Unit 1 Current 4-inch
Break Occurs	0.0	0.0		0.0	0.0
Reactor Trip Signal	7.74	19.5		4.47	11.1
Safety Injection Signal	15.02	28.2		7.39	18.6
Top of Core Uncovered	1375	1066		660	605
Accumulator Injection Begins	2350	2250		900	852
Peak Clad Temperature Occurs	1868	1948		948	928
Top of Core Covered	2133	3176		1117	1571

Question 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 is based on a lower nominal T_{avg} is acceptable for supporting the uprate.

PG&E Response to Question 13:

In several instances the Unit 1 uprate LA indicates that the RCS T_{avg} is being revised from 576.6 °F to 577.3 °F. Any reference in the LAR documentation to these RCS T_{avg} values should be interpreted as "maximum nominal" RCS T_{avg} values. These T_{avg} values represent the maximum full power T_{avg} for which the RCS T_{avg} control system can be programmed. Since the actual full power RCS T_{avg} varies from the T_{avg} control system programmed value based on control system response characteristics and temperature uncertainties, the programmed value is considered a nominal value. This is also the basis for incorporating conservative assumptions into the safety analyses to bound any uncertainty between the actual RCS T_{avg} and the maximum nominal RCS T_{avg}.

DCPP has historically operated with an actual nominal RCS T_{avg} several degrees below the maximum nominal value in order to minimize the T_{hot} effects associated with long term SG tube integrity. As discussed in the LAR 99-03, Enclosure C, Item 14.1, the Unit 1 uprate will be accomplished with a minimal change to T_{hot} based on an expected increase in the actual nominal RCS T_{avg} of about 0.7 °F. Operation with a lower nominal RCS T_{avg} is conservative since it remains bounded by the maximum nominal T_{avg} value.

Operation with a lower nominal RCS Tavg is also consistent with the current terminology in ITS Table 3.3.3-1 Notes 1 and 2, which specify that the OTΔT term T' and the OPΔT term T":

"... is the nominal loop specific indicated T_{avg} at RTP , ≤ 576.6 (Unit 1)..."

Similarly, the revised ITS Table 3.3.3-1 Notes 1 and 2, will specify respectively that the OTΔT term T' and the OPΔT term T":

"... is the nominal loop specific indicated T_{avg} at RTP , ≤ 577.3 (Unit 1)..."

In summary, revised LAR pages have been provided to clarify that the Unit 1 maximum nominal RCS T_{avg} is being increased from 576.6 °F to 577.3 °F as part of the uprate LAR. The licensing basis provides the flexibility to operate with an actual nominal RCS T_{avg} value below the maximum value and DCPP will implement a value that optimizes the opposing effects of an increase in RCS T_{hot} on plant thermal efficiency and long term SG tube integrity. The actual programmed nominal RCS T_{avg} value may be revised accordingly as plant thermal performance and/or SG tube conditions change over time.

WCAP-14819, Section 3.1.2, discusses the range of RCS T_{avg} values which were analyzed for the SBLOCA event to ensure that the limiting T_{avg} case was appropriately established. This section also explains that the competing effects and complex nature of the SBLOCA transient have not always resulted in the expected trend of a reduced limiting PCT value for a reduced RCS T_{avg}. The range of RCS Tavg values analyzed consisted of a High T_{avg} case, a low Tavg, case, and a nominal Tavg case which used a value of 572.0 °F since this was representative of the Unit 1 nominal RCS Tavg at that time (and is still representative). The section indicates that the High T_{avg} case generated the limiting PCT results. The High T_{avg} case is based on a value of 572.0 + 10.3 °F which is bounding for the Unit 1 uprate maximum nominal T_{avg} value of 577.3 °F including uncertainties. In summary, the SBLOCA analysis is conservatively bounding for the Unit 1 uprate based on establishing that a lower nominal RCS T_{avg} generates less limiting PCT results, and analyzing a range of RCS Tavg values including a High T_{avg} case that bounds the maximum value.

Question 14:

Show that the large- and small-break LOCA analyses methodologies referenced in WCAP-14819 apply to DCPP 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.

PG&E Response to Question 14:

PG&E procedure TS6.NE2 "Reload Core Development Process" establishes the roles, responsibilities (including fuel vendor interface), analyses, and documentation which are required to implement the core reload safety evaluation methodology established in WCAP-9272-P-A. In addition to providing logistical and economic guidelines, the process ensures that the actual plant operating parameters for each cycle remain bounded by the range of key operating parameters assumed in the safety analyses. The following discussion briefly summarizes four key documents related to this safety analysis verification which are generated as part of the core reload procedural process.

For every core reload, PG&E and Westinghouse conduct a Reload Design Initialization (RDI) Meeting which documents the key operating conditions for the next cycle including core power and burnup, RCS Tavg, RCS ΔT, RCS Flow, SG pressure, and SG tube plugging. The RDI meeting also documents any plant changes which may impact the safety analyses including control system and/or protection system setpoints, changes in ECCS and/or ESF system performance, and changes in secondary system performance. The Reload Safety and Licensing Checklist is then generated to verify that the latest plant licensing basis supports the proposed reload core performance characteristics and plant system performance for the upcoming cycle of operation. After the reload core loading pattern has been designed, the RSAC is generated and jointly reviewed by PG&E and Westinghouse to verify that the calculated core reload values (kinetics, reactivity, peaking factors, etc.) remain bounded by the

assumptions in the safety analyses. Finally, the Reload Safety Evaluation integrates and summarizes all of the reload related analyses and establishes the 10CFR50.59 Safety Evaluation which supports the formal design change process (per procedure CF3.ID9) for implementing the new reactor core, without having to obtain prior NRC approval.

NON-LOCA SAFETY ANALYSES:

Question 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 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.

PG&E Response to Question 15:

The initial Nuclear Steam Supply System (NSSS) power assumed for each of the non-LOCA events is listed in the FSAR Update Table 15.1-4. Table 15.1-4 identifies all the events analyzed with the NSSS power rating of 3423 MWt. In WCAP-14819 Westinghouse evaluated an increase in NSSS power due to an improved calculation of the minimum RCP heat input. This evaluation concluded that the increase in the total NSSS power remained bounded by the existing analyses, ensuring that the licensed core power of 3411 MWt was not exceeded. The Westinghouse evaluation is based on engineering judgment, and is supported by sensitivity studies which are applicable to DCPP.

It should be noted that many safety analysis events are not affected by NSSS power, but rather net core power. For example, although the FSAR Update Table 15.1-4 lists 3423 MWt as an NSSS power assumed for Rod Ejection (15.4.6), this is only used to establish the initial RCS conditions. The NSSS system is not explicitly modeled in the TWINKLE/FACTRAN heat flux analysis which is based on a core power of 3411MWt. The core power is not affected by the small NSSS power increase. Similarly, other analyses may assume an NSSS power, but the key analysis results are affected by the assumed core power. For example, Loss of Flow is analyzed with the LOFTRAN, FACTRAN, and THINC codes. While the LOFTRAN input includes a value for NSSS power, LOFTRAN is only used to calculate the flow and core nuclear power transients as a fraction of nominal. The slight change in NSSS power of 2 MWt does not affect the normalized flow and core power transients results. These LOFTRAN

results are then used in FACTRAN to calculate a heat flux transient. THINC uses the core flow and heat flux transients to calculate a Departure from Nucleate Boiling Ratio (DNBR). Since the THINC calculation only models the detailed core power level and flow, a 2 MWt increase in NSSS power with no change in core power has an insignificant effect on the DNBR results.

The only non-LOCA events that are affected by an increase in NSSS power without a subsequent core power increase are those events with results based on overall gross system parameters, such as RCS pressure, pressurizer water volume, etc. However, a 2 MWt increase in NSSS power from 3423 to 3425 MWt represents an increase of less than 0.06 percent. This increase is considered insignificant and would not result in any substantial change to the calculated safety analysis results. In addition, some of these events (such as Loss of Normal Feedwater/Loss of Offsite Power and Feedline Break) have already used a more bounding NSSS power as noted in the FSAR Update Table 15.1-4.

Even though the increased NSSS power is a result of an increased minimum Reactor Coolant Pump (RCP) heat input calculation, it should be noted that this does not affect the conservative pump heat assumed in safety analysis events for which Auxiliary Feedwater (AFW) heat removal (core and pumps) is critical. These events such as Loss of Normal Feedwater and Feed Line Break have already conservatively assumed 5 MWt per pump, or 20 MWt total pump heat.

Question 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.

PG&E Response to Question 16:

The non-LOCA analyses do not explicitly model annular pellet blankets since they have been determined not to directly impact any fuel performance design feature assumed in these analyses .

The first introduction of annular blankets was provided as a supplement to WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995, Section J, which has been reviewed and accepted by the NRC. While the WCAP provides detail on the fuel mechanical design and related fuel temperatures, it does not discuss the annular blanket feature relative to

applicability of DNBR correlations and safety analysis methodology. The use of annular blankets does not affect the applicability of the DNBR correlation, design limit values, core thermal limits, or the OTΔT and OPΔT protection setpoints. The DNBR methodology does not model, and is not directly impacted by, the internal fuel rod design detail (solid vs. annular pellets, etc.) since it only models the heat flux exiting the fuel rod into the coolant channel.

As discussed in Response 1, the Reload Safety Evaluation (RSE) for Diablo Canyon Unit 1 Cycle 9 included an evaluation to include enriched annular pellet blankets in the VANTAGE+ fuel assembly design. The potential impact of this fuel feature on the non-LOCA safety analyses results was evaluated with respect to effects on the axial and radial power distributions within the core. Since blankets can impact the relative core power shape they must be evaluated for effects on the axial power distribution along the active fuel region as well as the radial power distribution between the blanketed and any unblanketed fuel assemblies. Westinghouse originally indicated expectations that the impact of the enriched annular axial blankets would be bounded by the power shapes observed in reload cores using natural uranium axial blankets and those with no axial blankets. This is reasonable since the fissile content, and thus the fission rate in the enriched annular axial blanket lies between that found in the natural uranium blanket and the fully enriched uranium blanket (i.e., fuel without blankets).

Subsequent Westinghouse analyses have confirmed that the range of power shapes associated with the enriched annular axial blankets is no worse than power shapes previously generated for other reload cores. However, as part of the normal reload safety analysis process, power shape sensitivities will continue to be performed. Note that this is not an additional requirement due to the use of the enriched annular blankets, but is performed as part of the accepted methodology in WCAP-9272-P-A.

In summary, the annular pellet blankets do not directly impact the licensing basis non-LOCA safety analyses, and the normal reload safety analysis process will confirm the acceptability of the reload power shapes for those cores utilizing this fuel feature.

Accidental Depressurization of RCS:

Question 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.

PG&E Response to Question 17:

The key analysis input assumptions for the accidental RCS depressurization are those included in Section 3.2.3.2. A revised page clarifying this is attached.

Question 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.

PG&E Response to Question 18:

Revised pages referring to and listing the appropriate references are attached.

Question 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?

PG&E Response to Question 19:

The response to question 16 establishes that the use of annular pellet blankets is bounded, and requires no explicit modeling or revisions with respect to the current DNBR methodology.

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.

Question 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 judgment. 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.

PG&E Response to Question 20:

The summary provided in WCAP-14819 (April 1998) describes a new analysis for the steamline break core response at power event which was completed specifically for the Unit 1 power uprate. The steam line break at power analysis is performed in two stages. The LOFTRAN code is used to model the overall plant behavior and calculate the nuclear power, core heat flux, and RCS temperatures and pressures which result from the cooldown following the steam line break. Next, the core radial and axial peaking factors are determined using the thermal-hydraulic conditions from the transient analysis as input to the nuclear core models. The THINC code is then used to calculate the minimum DNBR for the limiting time during the transient, with the acceptance criterion that this value remains above the applicable safety analysis DNBR limit. This accident is analyzed using the Improved Thermal Design Procedure as described in WCAP-8567-P-A, "Improved Thermal Design Procedure," February 1989. As discussed in Response 16, the DNBR methodology is bounding for the VANTAGE+ fuel design and is not based on any specific fuel pellet physical design.

The following key analysis input assumptions and methodology are used in order to calculate conservative DNBR results:

1. The initial core power, reactor coolant temperature, and RCS pressure are assumed to be at their nominal full power values. Full power is considered to be more limiting than part power with respect to DNBR. The uncertainties in the initial conditions are included in the limiting DNBR as described in WCAP-8567-P-A.

2. A spectrum of break sizes is analyzed to determine the most limiting conditions based on which reactor trip or safety injection signal is actuated.

3. The steam flow out the pipe break is calculated using the Moody curve for an fL/D value of 0.

4. The analysis assumes a maximum end of life moderator reactivity feedback and a minimum Doppler-only power reactivity feedback in order to maximize the power increase during the transient.

5. The analysis only models those reactor protection system features which would be credited for at power conditions and up to the time a reactor trip is initiated. The FSAR Update Section 15.4.2.1 presents the analysis of the bounding transient following reactor trip, where engineered safety features are actuated to mitigate the effects of a steam line break.

6. The results of the analysis would be made less severe as a result of control system actuation, therefore, the mitigation effects of control systems have been ignored in the analysis.

The spectrum of break sizes analyzed for the Unit 1 uprate conditions determined that the 0.53 ft^2 case established the limiting DNBR since it was the largest steamline break core response at power case which did not generate a low steam pressure safety injection signal. This case was determined to be equivalent to an excessive load increase in that the core reaches a new equilibrium power equivalent to the increased steam flow. The minimum DNBR values for the 0.53 ft^2 case were calculated to be 2.062 and 2.143 for the Unit 1 Cycle 10 thimble and typical cells, respectively. These calculated values are acceptable since they remain above the applicable safety analysis DNBR limits of 1.68 and 1.71 for thimble and typical cells, respectively. For larger break sizes, a low steam pressure safety injection signal is generated within a few seconds and the power increase and minimum DNBR are less limiting. After a safety injection signal is generated, the remainder of the event is then bounded by the maximum cooldown of the hot zero power case in the FSAR.

The same key core kinetics reload parameters (i.e., moderator temperature coefficient, Doppler coefficient, delayed neutron fraction) and power distribution limits (i.e., $F_{\Delta H}$) establish the bounding analysis limits for both the at power and the hot zero power steam line break cases. The acceptability of the calculated core design values with respect to these analysis limits will be verified for each core reload as part of the reload safety evaluation process of WCAP-9272-P-A.

Question 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)?

PG&E Response to Question 21:

PG&E will include a discussion in the FSAR Update summarizing how the steamline break at power core response has been analyzed to verify that the minimum DNBR which occurs prior to a reactor trip or safety injection remains within the appropriate limit.

Steam Generator Tube Rupture

Question 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? Provide a copy of the report and the NRC acceptance.

PG&E Response to Question 22:

The steam generator tube rupture (SGTR) overfill reanalysis as documented in Westinghouse letter PGE-92-685 was evaluated and incorporated into the DCPP design basis per a 10 CFR 50.59 screen (FSAR Update Revision 9, November 1993). Therefore, the reanalysis was not submitted to the NRC for review.

The original SGTR overfill analysis for DCPP, documented in WCAP-11723, was approved by the NRC in April 1991 for close out of the SGTR analysis issue for DCPP. The SGTR reanalysis documented in PGE-92-685 incorporated revised input assumptions related to AFW flow rates and SG power operated relief valve (PORV) relief capacity. The methodology used was the same as that used in the original SGTR overfill analysis. The enhanced modeling of the turbine driven AFW pump and the AFW control valves as a function of SG level resulted in a net gain in overfill margin from the original 61 ft^3 reported in WCAP-11723 to a value of 109 ft^3. The SGTR overfill analysis results per PGE-92-685 are discussed in FSAR Update Chapter 15, Section 15.4.3.

Question 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 DCPP 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."

PG&E Response to Question 23:

The current SGTR analysis is based on the Unit 2 setpoint OTΔT setpoint constants. Using an initial RCS Tavg = 577.3 °F vs. 577.6 °F, would provide positive margin with respect to the OTΔT trip setpoint, and would require slightly longer to generate a reactor trip signal when the SGTR occurs and RCS pressure begins decreasing. Explicitly modeling the Unit 1 lower T_{avg}, lower RCS flow, and OTΔT setpoint constants (T' = Tref = 577.3 °F) would be expected to generate a similar reactor trip time as the Unit 2 analysis.

Westinghouse established in the original SGTR analysis methodology (Supplement 1 to WCAP-10698) that an earlier reactor trip was more conservative with respect to maximizing the calculated offsite dose release. This conclusion is based on three effects. The first effect is when the initial offsite dose release begins relative to the fixed operator action times assumed for the SGTR mitigation. As discussed in FSAR Update Section 15.4.3, the initial SGTR offsite dose release begins after the reactor trip when the steam pressure in the steam

generators increases to the PORV and/or safety valve lift setpoints. The steam mass release versus time for the ruptured and the intact SGs during the SGTR offsite dose analysis are plotted in FSAR Update Figures 15.4-108 and 15.4-109, respectively. Since the operator action times and intervals are essentially fixed assumptions, an earlier reactor trip can only increase the duration time of the steam release until the subsequent operator actions are completed.

The second effect involves the relative iodine concentration in the RCS and associated break flow when the steam release occurs. Since the preaccident iodine spike and concurrent iodine spike are assumed to occur at time zero, an earlier reactor trip would allow for less iodine decay, and greater RCS activity when the initial steam release begins. This also applies to the ruptured SG PORV which is assumed to fail open just after the ruptured SG isolation action is completed. As listed in FSAR Update Table 15.4-12, the SGTR analysis assumes that the ruptured SG is isolated at ten minutes after the rupture occurs, or when the ruptured SG level reaches 27 percent narrow range, whichever time is longer. As shown in the sequence of events summary in FSAR Update Table 15.4-13, the ruptured SG is isolated at 636 seconds based on the more limiting time when the ruptured SG level was calculated to reach 27 percent narrow range. Consequently, an earlier reactor trip would essentially shift the whole SGTR analysis sequence of events including SG isolation forward in time. This is because the time from reactor trip until the combination of AFW and break flow fills the ruptured SG narrow range to 27 percent is essentially independent of when the trip occurs. Therefore, an earlier reactor trip results in the ruptured SG PORV failing open earlier (although not before 600 seconds).

The third effect and the most significant contribution to the offsite dose release is based on the fraction of the RCS break flow which directly flashes to vapor as it enters the lower pressure steam generator. The SGTR analysis conservatively assumes that all of the flashed vapor and associated activity are directly released to the environment with no credit for iodine scrubbing effects. The SGTR analysis calculates the flashing fraction based on the break flow liquid expansion from the RCS pressure to the steam generator pressure. Therefore, an earlier reactor trip will result in a slightly greater RCS pressure and associated flashing fraction when the offsite dose release begins.

These three effects established in the SGTR analysis methodology lead to increased offsite dose release values if the reactor trip occurs earlier.

In summary, the evaluation in WCAP 14819 concluded that the Unit 1 uprate parameters (RCS T_{avg}, RCS Flow, and core power) remain bounded by the current SGTR analysis which is based on the Unit 2 parameters.

Question 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?

PG&E Response to Question 24:

The use of enriched annular fuel pellet blankets does not impact the bounding calculated core source term. As discussed in FSAR Update Section 11.1, the core isotopic inventory is conservatively calculated based on a fixed fission reaction rate per power generation rate, conservative isotopic yield fractions for both uranium and plutonium, and standard decay constants. FSAR Update Table 11.1-2 indicates that the design basis core isotopic inventory is calculated for a 3568 MWt core, operating with a one year equilibrium cycle length and an 80 percent capacity factor. The design basis core source term (isotopic inventory) is based on a composite source of 3.5 w/o and 4.5 w/o fuel enrichments and fuel burnup ranges from 1,000 MWD/MTU to 50,000 MWD/MTU. The FSAR Update states that this calculated isotopic inventory is conservatively bounding for the Unit 2 power rating of 3411 MWt, a 5 w/o enrichment, and the current 21-month cycle length and associated maximum cycle burnup of 50,000 MWD/MTU.

In summary, the use of annular fuel pellets does not impact the bounding design basis core source term since it is not based on any specific fuel pellet design, and is strictly a function of the assumed core power level and fuel burnup.

Residual Heat Removal System:

Question 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 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."

PG&E Response to Question 25:

RHR cooldown calculations are performed to demonstrate that Westinghouse internal RHR system design criteria related to the sizing of the RHR heat exchanger (Hx) can be met. These system design criteria assume with the maximum RCS cooldown rate limited to no more than 50 °F per hour that: (1) two RHR trains can cool the RCS to 140 °F within 20 hours after shutdown, and (2) one RHR train can cool the RCS to 200 °F within 36 hours after shutdown.

Two RHR cooldown calculations have been performed. The first calculation was performed in 1970, and was reflected in the original FSAR (i.e., cooldown from 350°F to 140° F within 10 hours). The second calculation was performed for WCAP-14819, and is reflected in the FSAR mark-ups (i.e., cooldown from 350°F to 140°F within 17.4 hours). The analysis presented in WCAP-14819 represents the current RHR cooldown verification calculation for DCPP, and is bounding for the Unit 1 uprate conditions.

The difference in the input assumptions between the two calculations was to update the original 1970 Westinghouse calculation from a set of generic assumptions to the most recent DCPP analysis data which considered various system conditions, flow alignments, and additional conservatism. The major differences between the 1970 Westinghouse calculation and the current analysis presented in WCAP-14819 are:

1. The CCW heat exchanger heat transfer UA is reduced from a Westinghouse generic value of 4.85 MBtu/hr-F to 4.559 MBtu/hr-F to provide conservative margin to DCPP specific data.

2. The original 1970 generic Westinghouse ASW flow of 5.417 Mlb/hr (i.e. 10,950 gpm) per Component Cooling Water (CCW) Hx is reduced in WCAP-14819 analysis to a flow of 5.15 Mlb/hr (i.e., 10,300 gpm) to provide additional conservatism.

3. The Auxiliary Salt Water (ASW) temperature was reduced from 70 °F to 64 F to reflect a more appropriate maximum ocean temperature.

4. The initial (i.e., at 4 hours after shutdown) auxiliary heat loads (e.g., letdown Hx, Spent Fuel Pit (SFP) Hx, Containment Fan Cooler Unit (CFCU) coolers & motors, boric acid evaporator, condenser, etc.) were increased from 40.35 MBtu/hr to 44.3 MBtu/hr to conservatively bound worst case assumptions.

A core power of 3411 MW thermal was assumed in both calculations. The analysis documented in WCAP-14819 demonstrates that the DCPP RHR system continues to meet the Westinghouse system design criteria and applicable regulatory criteria for the Unit 1 uprate power level.

Typos Related to OTΔT and OPΔT Trip Setpoint Calculations:

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

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

PG&E Response:

The following corrected pages are attached:

Enclosure A:

Page 1
Page 3, 3A

Enclosure B, WCAP 14819 :

Page 3-35
Page 3-36
Page 3-42

Enclosure C:

Page 1
Page 2
Page 3, 3A

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 maximum 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 maximum 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.

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 power 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

A summary of the inputs and results for the large break LOCA reanalysis were submitted to the NRC in May of 1997 by PG&E Letter DCL-97-030, "Licensee Amendment Request 97-08 Revision of Technical Specifications to Apply Westinghouse Generic Best Estimate LOCA Analysis Methodology". In that letter, PG&E requested allowance to use the approved WCAP-12945-P-A, "Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis." to determine core operating limits. In LAs 121 (Unit 1) and 119 (Unit 2), dated February 13, 1998, the NRC staff found the use of WCAP-12945-P-A, acceptable for use in DCPP licensing applications.

The BELOCA analysis for DCPP is documented in WCAP-14775 and revised the Unit 1 and Unit 2 resultant peak clad temperature (PCT) values from 2042°F and 2169°F, as reported in PG&E Letter DCL-97-124, respectively, to a value of 1976°F for both units. The current BELOCA PCT is 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,

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.4 Single Reactor Coolant Pump Locked Rotor

15.4.6 RCCA Ejection (full power cases)

The currently applicable analyses or calculations for the following setpoints or transients were not performed in a bounding manner. Rather, separate analyses were performed for each Diablo Canyon unit.

15.1.3 Overtemperature and Overpower ΔT Reactor Trip Setpoint Calculations

15.2.13 Accidental Depressurization of the Reactor Coolant System

In order to support the Unit 1 uprated conditions these items have been reanalyzed, as described in the sections below.

3.2.2 Overtemperature and Overpower ΔT Reactor Trip Setpoint Calculations

The Diablo Canyon units both currently use the same OTΔT/OPΔT trip setpoint constants. However, calculations to confirm the acceptability of these setpoints are performed separately for the specific plant operating conditions of each unit, using the methodology of Reference 1. The currently applicable setpoint calculations are based on reactor core thermal limits for 17x17 standard fuel, which is limiting with respect to the 17x17 VANTAGE 5 fuel type currently used at Diablo Canyon. There is insufficient DNB margin available to support the current setpoints assuming 17x17 standard fuel for the Unit 1 uprated conditions. Therefore, revised core thermal limits were developed based on the uprated Unit 1 power and flow parameters which assume 17x17 VANTAGE 5 fuel only. Setpoint calculations were performed which verify that the present Technical Specification OTΔT/OPΔT trip constants and the associated f(ΔI) penalty function provide adequate protection for the revised core limits at the uprated Unit 1 power conditions.

Note, the above evaluation and results are applicable to VANTAGE 5 fuel with either ZIRLO™ or standard zircaloy.

3.2.3 Accidental Depressurization of the Reactor Coolant System

The currently applicable analysis considered each Diablo Canyon unit separately. The limiting Unit 2 analysis is presented in the updated UFSAR section 15.2.13. Since this analysis credited the higher RCS flow of Unit 2, it does not bound the uprated Unit 1 plant conditions. A new analysis was performed using conservative assumptions that bound both units. The transient results are similar to those presented in the FSAR, except for the specific DNBR calculation which now assumes the VANTAGE 5 fuel type instead of the limiting standard fuel which is no longer used in the Diablo Canyon cores. The DNBR remains above the applicable limit value, and the conclusions of the UFSAR remain valid.

3.2.3.1 Identification of Causes and Accident Description

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure which could reach the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature essentially constant until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor will be tripped by the following reactor protection system signals:

1. Pressurizer low pressure
2. Overtemperature ΔT

An accidental depressurization of the RCS is classified as an ANS Condition 2 event.

3.2.3.2 Analysis of Effects and Consequences

The accidental depressurization transient is analyzed with the LOFTRAN code (Reference 3). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 4.

In order to give conservative results in calculating the DNBR during the transient, the following key assumptions are made:

1. Initial reactor power, pressure, and reactor coolant system temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 4.

2. A positive moderator temperature coefficient of reactivity for BOL (+7 pcm/EF) is assumed in order to provide a conservatively high amount of positive reactivity feedback due to changes in moderator temperature. The spatial effect of voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity

3.2.4 Steam Line Break at Full Power

The steam line break analysis documented in the updated FSAR section 15.4.2.1 assumes zero power initial conditions, and demonstrates that the DNB design basis is met for this accident following a reactor trip. The steam line break at full power initial conditions analysis to demonstrate core integrity is not explicitly documented in the updated FSAR. An analysis of this event was performed to support an assumed increase in the OPΔT reactor trip response time for the RTD Bypass Elimination modification that was performed in conjunction with the Eagle 21 process protection system upgrade (Reference 2). The currently applicable analysis of this event is performed separately for each Diablo Canyon unit, and as such does not bound the uprated Unit 1 plant conditions. A new analysis was performed using conservative assumptions that bound both units. The transient results are less limiting than the previous analyses, due to the use of a higher setpoint for the low steam line pressure safety injection actuation which results in an early reactor trip for a larger range of break sizes. Previously, a very conservatively low setpoint (14.7 psia) was assumed in order to allow flexibility to potentially revise this setpoint at the plant, which never occurred. Use of a higher but still conservative setpoint (459 psia) reduces the size of the largest break that will not trip on low steam pressure SI actuation. This in turn reduces the peak core power that is achieved for the worst case, which will result in a higher minimum DNBR than in previous analyses. Based on a comparison of the transient results as described above it is concluded that the DNB design basis is met. The DNBR is confirmed for this event using cycle-specific core parameters as part of the reload safety evaluation.

3.2.5 Non-LOCA Conclusions

Based on the evaluations and analyses described above, it is concluded that all applicable safety criteria are met and the conclusions of the Diablo Canyon updated FSAR remain valid for the Unit 1 non-LOCA events for the uprated power conditions.

3.2.6 References

1. Ellenberger, S. L., et al., "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8745-P-A (Proprietary) and WCAP-8746-A (Non-Proprietary), September 1986.

2. "Summary Report, Eagle 21 Process Protection System Upgrade for Diablo Canyon Power Plant Units 1 and 2," WCAP-13615-R2, June 1993.

3. Burnett, T.W. T., et. Al., "LOFTRAN Code Description", WCAP-7907-P-A (Proprietary) , WCAP-7907-A (Non-proprietary) , April 1984

4. Chelemer, H., Boman, L. H., and Sharp, D. R. , "Improved Thermal Design Procedure ", WCAP-8567-P-A (Proprietary) , WCAP-8568-A (Non-proprietary) , February 1989

Addendum to WCAP- 14819, "Pacific Gas and Electric Company Diablo Canyon Power Plant, Unit 1 3425 MWt Uprating Program Licensing Report."

Since much of the work that is summarized in WCAP-14819, "Pacific Gas and Electric Company Diablo Canyon Power Plant, Unit 1, 3425 MWt Uprating Program Licensing Report," was completed in 1997, a review was performed to identify the changes to the plant since then that could affect the WCAP. This addendum addresses changes in the Diablo Canyon licensing basis since WCAP-14819 was written, and provides greater detail about certain aspects of the uprating evaluation that were the subject of NRC requests for additional information in the review of uprate requests at other facilities.

1. Codes and Methodologies used: A complete loss-of-coolant accident (LOCA) re-analysis was performed which used codes not previously applied to the Diablo Canyon Power Plant (DCPP). Specifically, the large break LOCA uses best estimate methodology and the small break LOCA analysis used the COSI condensation model. Due to these methodology changes, and due to a commitment made to the NRC to update the DCPP LOCA analyses, the large break and small break LOCAs have been submitted separately to the NRC. The large break LOCA reanalysis has been approved in License Amendments (LAs) 121 and 119.

   The remaining evaluations for the Unit 1 uprate did not require new methodology or codes. Most current licensing basis analyses are common analyses that envelope both Unit 1 and 2. These generally assumed the lower Unit 1 reactor coolant system (RCS) flow rate in combination with the higher Unit 2 power. Thus most already envelope Unit 1 at the same power level as Unit 2. The only two evaluations requiring further analytical work for the uprate were the overtemperature and overpower ΔT (OTΔT/OPΔT) reactor trip setpoint calculations and accidental depressurization of the RCS. These particular analyses were previously performed at unit specific conditions, i.e., the higher Unit 2 power analysis credited the higher Unit 2 flow rate.

   Calculations were performed to confirm the adequacy of the current Unit 1 OTΔT and OPΔT setpoints. The calculations are based on exclusive use of 17x17 Vantage 5 fuel since DCPP has no expectation of using 17x17 standard fuel. This allows slightly higher reactor core safety limits as shown in Technical Specification (TS) Figure 2.1.1-1. The setpoint calculations also assumed the uprated Unit 1 maximum nominal full power T_{avg}, which was increased from 576.6°F to 577.3°F. With these changes, the results show that the current TS OTΔT/OPΔT setpoints and f(ΔI) penalty function are adequate to bound Unit 1 at the uprated conditions. These

setpoint calculations used the previously approved methodology of WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986, and no new codes or methods were required.

- Accidental depressurization of the RCS is assumed to be the result of a failed open pressurizer safety valve. It is analyzed using the LOFTRAN code and the improved thermal design procedure as described in the Final Safety Analysis Review (FSAR) Update Section 15.2.13. The current analysis is performed for each Unit separately, so a new reanalysis was performed which conservatively bounds both units. As in the case of the OTΔT and OPΔT setpoints, the new analysis used revised input assumptions including the exclusive use of 17x17 Vantage 5 fuel, but no new codes or methods were required.

In addition, the residual heat removal (RHR) cooldown calculation was reperformed and documented in WCAP-14819; however, the analysis was redone mostly to add margin for issues related to the component cooling water (CCW) system rather than in response to the uprate. The RHR cooldown calculation is performed to demonstrate that the system meets 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. This is not a design basis accident analysis and there are no safety-related consequences should the cooldown exceed the time specified. The calculation is identical for both units since the RHR system and CCW system are the same, and a bounding decay heat is assumed. 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. As a result, the new RHR cooldown calculation indicates a longer required time to perform the cooldown. This longer cooldown time is a consequence of the more conservative assumptions, not the uprate, since the assumed decay heat has always enveloped the 3411 MWt of Unit 2. Although the analysis assumed more conservative analysis inputs, the RHR cooldown calculation involved no new codes or methodologies.

2. The large break LOCA analysis results were submitted to the NRC in May of 1997 by PG&E Letter DCL-97-030. In that submittal, PG&E requested a revision to TS 6.9.1.8, "Core Operating Limits Report," to allow use of the generic approved BELOCA methodology per WCAP-12945-P-A to determine the core operating limits. The BELOCA methodology utilizes the best estimate of certain key parameters with parameter ranges specified to envelope all expected values. The analysis uses a Monte Carlo process to determine the 95 percent confidence limit for peak clad temperature (PCT) to satisfy 10CFR50 Appendix K

analysis requirements. In LAs 121 (Unit 1) and 119 (Unit 2), dated February 13, 1998, the NRC staff found the use of WCAP-12945-P-A, acceptable for use in DCPP licensing applications. The previous large break LOCA PCT analyses of record indicated PCTs of 2042°F at Unit 1 and 2071°F at Unit 2. The BELOCA analysis of record predicts the single bounding value of 1976°F for both units while the current BELOCA PCT is 2043°F for both units, as reported in PG&E Letter DCL-99-096.

3. The small break LOCA analysis results were submitted to the NRC in December of 1998. The analysis utilizes the COSI Condensation Model from an addendum to WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code." In that submittal, PG&E requested a revision to TS 6.9.1.8, "Core Operating Limits Report," to allow use of any applicable NRC approved addenda to WCAP-10054-P-A to determine the core operating limits. At the NRC Staff's request, in PG&E Letter DCL-99-099, "Supplement to License Amendment Request 98-09," dated July 30, 1999, PG&E limited the requested change to just 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. 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). The small break LOCA analysis predicts a PCT of 1304°F for Unit 1 and 1293 °F for Unit 2.

4. The fuel is assumed to have all Zirlo cladding. This is consistent with Vantage 5+ fuel. The calculation used to generate the analysis inputs, PG&E Calculation STA-031, "Input Data for Unit 1 Uprate Project and Units 1 & 2 Loss of Coolant Accidents," page 14, states, "Future fuel assemblies will have a change in material that caused us to question whether there would be possible effects (Zirlo will be used rather than Zirc-4). Westinghouse states in the 24 month Cycle Safety Evaluation (in draft form in PGE-95-611) that the Zirlo is a small 2 F PCT penalty for Unit 1. It was decided that Zirlo would be modeled both because of this Unit 1 penalty, and because eventually the core will be all Zirlo." The large break LOCA and small break LOCA analysis results which incorporate these fuel cladding impacts have been submitted to the NRC separate from this uprate license amendment request (LAR).

5. The impact of the uprate on electric grid stability is not addressed in the WCAP 14819 licensing report. The Unit 1 uprate will increase the total plant power output to the grid by only 1.1 percent. PG&E engineers have reviewed the uprate and determined that it will have no significant impact on grid stability.

6. The impact of the uprate on RCS hot leg thermal streaming is not addressed in the licensing report. The measured RCS flow rate is compared to the TS required minimum measured flow by performing a precision flow calorimetric test. Hot leg streaming can potentially increase the inaccuracy in the hot leg