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September 29, 2000

PG&E Letter DCL-00-126

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 September 26, 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 1. 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).

If you have any questions regarding this response, please contact Tom Grozan at (805) 545-4231.

Sincerely,

D. Mikkelsen for D.H. Oatley

David H. Oatley

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

A-001

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 B. Miklush, of lawful age, first being duly sworn upon oath says that he is Manager-Engineering Services 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.

D. Miklush

David B. Miklush
Manager-Engineering Services

Subscribed and sworn to before me this 29th day of September, 2000.

Chuck Mackey

Notary Public
State of California
Country of San Luis Obispo

CHUCK MACKEY
Commission # 1204640
Notary Public - California
San Luis Obispo County
My Comm. Expires Dec 12, 2002

PG&E Response to Request for Additional Information For Diablo Canyon Power Plant (DCPP) Unit 1 Power Uprate

Introduction

The various piping and component evaluations performed for the Unit 1 uprate, and discussed within this Request for Additional Information (RAI) response, all concluded that there was a minimal change in the operating parameters such that the existing analyses remained conservatively bounding.

The following tables provide a summary of the key design and operating data comparisons which were used to evaluate the Unit 1 uprate. Table 1 provides a comparison of the key Unit 1 design parameters based on the current core rating of 3338 MWt with the revised design parameters for the uprated Unit 1 core power of 3411 MWt. Table 2 provides a comparison of the revised design parameters for the uprated Unit 1 core power of 3411 MWt with the current Unit 2 design parameters, which are already based on a core rating of 3411 MWt. Table 3 provides a comparison of recent Unit 1 Cycle 10 operating data at the current core rating of 3338 MWt with the best estimate projections for Unit 1 Cycle 11 operating parameters at the uprated core power level of 3411 MWt. The evaluations of the Unit 1 uprate parameters are discussed in more detail in the following RAI responses which support the following overall conclusions:

1. The Unit 1 uprate design parameters have not significantly changed, and they remain bounded by the current Unit 2 design parameters since both units have almost identical systems and components which are already designed and analyzed to meet the more limiting Unit 2 parameters.

2. There have been no significant changes in the projected Unit 1 uprate best estimate operating parameters; accordingly there is no impact on the analyses and calculations which were performed and based on nominal Unit 1 operating parameters.

Table 1
Comparison of Key Unit 1 Current vs. Uprate Design Parameters

Parameter [1]	Unit 1 Current Design Data	Unit 1 Uprate Design Data	Uprate Minus Current Design Difference
NSSS Thermal Power (MWt)	3350	3425	75
Core Thermal Power (MWt)	3338	3411	73
RCS Loop Flow (gpm/ loop)	87,700	87,700	0
Minimum Measured Flow RCS Flow (gpm)	359,200	359,200	0
Vessel Outlet - RCS T_{hot} (°F)	608.8	610.1	1.3
Vessel Average - RCS T_{avg} (°F)	576.6	577.3	0.7
Steam Generator Outlet - RCS T_{cold} (°F)	544.2	544.2	0
RCS Loop ΔT (°F)	64.6	65.9	1.3
Zero Load RCS T_{avg}(°F)	547.0	547.0	0
RCS Pressure (psia)	2250	2250	0
Steam Pressure (psia)	805	800	-5
Steam Temperature (°F)	519	518.2	-0.8
Total Steam Flow (Mlb/hr)	14.52	14.91	0.39
Feedwater Temperature (°F)	432.1	435.0	2.9
Steam Generator Tube Plugging (%)	0	0	0

Note 1: All values are 100% Full Power values unless indicated otherwise.

Table 2
Comparison of Unit 1 Uprate Design
Vs Unit 2 Current Design Parameters

Parameter [1]	Unit 1 Uprate Design Data	Unit 2 Current Design Data	Unit 2 - Unit 1 Difference
NSSS Thermal Power (MWt)	3425	3425	0
Core Thermal Power (MWt)	3411	3411	0
RCS Loop Flow (gpm/ loop)	87,700	88,500	800
Minimum Measured Flow RCS Flow (gpm)	359,200	362,500	3,300
Vessel Outlet - RCS T_{hot} (°F)	610.1	610.1	0
Vessel Average - RCS T_{avg} (°F)	577.3	577.6	0.3
Steam Generator Outlet - RCS Tcold (°F)	544.2	544.8	0.6
RCS Loop ΔT (°F)	65.9	65.3	-0.6
Zero Load RCS T_{avg} (°F)	547.0	547.0	0
RCS Pressure (psia)	2250	2250	0
Steam Pressure (psia)	800	804	4
Steam Temperature (°F)	518.2	518.8	0.6
Total Steam Flow (Mlb/hr)	14.91	14.91	0
Feedwater Temperature (°F)	435.0	435.0	0
Steam Generator Tube Plugging (%)	0	0	0

Note 1: All values are 100% Full Power values unless indicated otherwise.

Table 3
Comparison of Best Estimate Operating Conditions
Unit 1 Cycle 10 Current versus Projected Unit 1 Uprate

Parameter[1]	Cycle 10 Current Operating Data	Projected Uprate Operating Data	Projected Uprate Minus Current Operating Delta
Full Power RCS T_{ref} (°F)	572.76	573.49	0.73
NSSS Thermal Power (MWt)	3350	3425	75
Core Thermal Power (MWt)	3338	3411	73
RCS Loop Flow (gpm / loop)	91550	91550	0
RCS T_{hot} (°F)	602.8	604.2	1.4
RCS T_{avg} (°F)	571.3	572.0	0.7
RCS Tcold (°F)	539.8	539.8	0
RCS Loop ΔT (°F)	63.0	64.5	1.5
Steam Pressure (psia)	798.5	795.2	-3.3
Steam Temperature (°F)	518.0	517.5	-0.5
Total Steam Flow (Mlb/hr)	14.45	14.84	0.39
Feedwater Temperature (°F)	429.7	432.0	2.3
Steam Generator Tube Plugging (%)	3.4	3.4	0

Note 1:All values are 100% Full Power values unless indicated otherwise.

Note 2: Due to RCS T_{avg} control system auctioneer high circuit and associated control deadband, the actual average RCS T_{avg} is always maintained less than Tref.

Question 1

In regard to Section 5.1.1 of Attachment B to your December 31, 1999 submittal, provide a comparison of the design parameters (i.e., steam pressure, temperature, primary-to-secondary pressure differential) and transients for the steam generators (SG) Model 51 against the power uprate condition. Also, provide the maximum calculated stress and cumulative fatigue usage factor (CUF) for the critical locations (such as the vessel shell, secondary manway bolts, Tubes, and nozzles), the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

PG&E Response to Question 1

The evaluations of the DCPP Model 51 SGs considered the design parameters listed in Table 1-1 as well as the best estimate operating values listed in Table 1-1. As summarized in Section 2.2 of WCAP-14819 (Enclosure B of LAR 99-03), it was determined that the existing nuclear steam supply system (NSSS) design transients (i.e., temperatures, pressures, and power levels) remained applicable and bounding for the Unit 1 uprated conditions. The design transients generate pressure and temperature excursions which are based on conservative plant transient assumptions, and are analyzed to bound a wide range of plant operating conditions. Consequently, the design transients, and in particular, the peak transient values are not significantly impacted by the small changes in the initial plant operating parameters associated with the Unit 1 uprate. In addition, the primary-to-secondary pressure differential has not changed for the uprating. Therefore, the design transients are unaffected by the uprating. The revised maximum calculated stresses for the critical components are provided in Table 1-2, and the calculated CUF for the critical locations are provided in Table 1-3, respectively, for the components impacted by the initial condition changes associated with the uprating. Any SG components not listed in Table 1-2 are not impacted and the results remain unchanged due to the uprating.

The evaluation used the original code of record, Section III of the ASME Boiler and Pressure Vessel Code 1965 Edition, Winter 1965 Addenda. That code of record has not changed.

Table 1-1
Operating Parameters/Characteristics for Diablo Canyon Unit 1

	Current Rating			Uprated Power			
	Design	Best Estimate	Current Oper.	Design	----- Best Estimate -----		
					Design T_{avg}	Current T_{hot}	Reduced T_{avg}
Operating Parameters							
Power - %	100	100	100	102.2	102.2	102.2	102.2
Power - MWt/SG	837.5	837.5	837.5	856.3	856.3	856.3	856.3
Primary Temps. - °F							
T_{hot}	608.8	606.6	601.3	610.1	607.9	601.3	602.7
T_{cold}	544.2	546.4	540.7	544.2	546.5	539.3	540.7
T_{avg}	576.6	576.5	571	577.3	577.2	570.3	571.7
Primary Flow - gpm	87700	94500	94500	87700	94500	94500	94500
Steam Flow - Mlbm/hr	3.62	3.628	3.62	3.72	3.72	3.72	3.72
Blowdown Flow - Mlbm/hr		0.02	0.02		0.02	0.02	0.02
Feed Temp. - °F	432	430	430	435	435	435	435
Steam Press. - psia	805	850	808	756	848	795	805
Fouling - h-ft^2-EF/Btu 10^6	180	40	40	180	40	40	40
Plugging - %	0	1.7	1.7	15	1.7	1.7	1.7
Operating Characteristics							
Steam Flow - Mlbm/hr	3.623	3.628	3.622	3.714	3.724	3.72	3.72
Steam Temperature - °F	519	525	519	510	525	517	519
Steam Press. - psia	805	850	808	756	848	795	805
Circulation Ratio	5.33	5.32	5.33	5.19	5.18	5.19	5.19
Bundle Liquid Flow-Mlbm/h	19.31	19.30	19.31	19.28	19.29	19.31	19.31
Damping Factor - hr^{-1}	-432	-422	-429	-444	-424	-434	-432
S/G Mass - lbm	113650	115013	114046	111551	114160	112910	113171
Average Heat Flux	55489	56449	56449	66741	57714	57710	57713
Peak Heat Flux	97882	107793	107978	112713	109677	110602	109982
Total Secondary ΔP - psi	20.3	20.2	20.3	21.6	21.2	21.5	21.4
Separator Parameter	7.40	7.01	7.37	8.31	7.40	7.91	7.80

Table 1-2
Comparison of Stress Limits for Diablo Canyon Unit 1 Uprating Evaluation

Component	Stress/Limit	
	Pre-Uprating Design Basis	Revised Uprating Calculation
Tubesheet	0.83	0.89
Divider Plate	(1)	(1)
Tubes	0.79	0.81

(1) Exceeded 3Sm limit. Therefore, a simplified elastic-plastic based fatigue analysis was performed as shown in Table 1-3 .

Table 1-3
Summary of Fatigue Usage in Steam Generator Components for Diablo Canyon Unit 1 Uprating Conditions

Load Condition	Component	Usage Factor	
		Pre-Uprating Design Basis	Uprating with Pressure Reduction
Normal and Upset	Tubesheet	.0143	0.144
	Tubes	(1)	(1)
	Divider Plate	0.791	0.866
	Secondary Manway		
	Shell Penetration	0.17	0.0575
	Bolts	(2)	(2)
	Steam Nozzle	0.59	0.733

(1) The fatigue usage and stresses are well below the acceptable limits. As such no analyses were performed.

(2) The stresses used for the analysis of the bolts are taken from another model steam generator, with appropriate scale factors to account for geometry and load variations in Series 51 generator. The bolts are qualified for 34 year operation for the original reference conditions and 31 years for the pressure reduction conditions.

Question 2

In Section 5.1.2, you stated that the replacement interval for the SG manway is reduced from 34 years to 31 years for operation at the uprated conditions. Please confirm that the plant procedures will be revised to incorporate the 31-year replacement interval for the SG manway bolts prior to the implementation of the power uprate. Also, provide an evaluation of the flow-induced vibration of the SG U-bends tubes and moisture carryover due to power UPRATE regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated elastic-fluid instability ratio.

PG&E Response to Question 2

SG Secondary Side Manway Bolts

Maintenance Procedure MP M-7.25, "Removal and Installation of Steam Generator Manway Covers," has been revised to reflect that the SG secondary side manway bolts service life replacement interval is 31 years.

Moisture Carryover

The DCPP Unit 1 moisture separator performance is established based on field data. The operating parameters which can have an effect on moisture separator performance are steam flow (power), steam pressure and water level. Using the current design as a reference, the best estimate case at the current rating indicates that the steam pressure is significantly higher as a result of the large reduction in fouling and the still nominal plugging level. The current operating steam pressure is well below the best estimate value. This is because the case was run with a primary T_{avg} 5.5°F below the design T_{avg}, typical of the way the plant is currently operated. The increase in steam pressure, which would result from the lower more realistic fouling value, is offset by the decrease in primary temperature. Since the power is the same, the feed flow is unchanged and the separator parameter variation is governed by the changes in steam specific volume. A high steam pressure results in low specific volume and a lower value of separator parameter.

The design case for the uprated power exhibits a reduced steam pressure, due mostly to the high values of fouling and plugging which have been assumed. The increase in feed flow, coupled with the reduction of steam pressure, results in a significantly higher value of separator parameter. The best estimate case for uprated power, using the same design T_{avg} along with the current fouling and plugging, yields a higher steam pressure and correspondingly lower value of separator parameter. This value is near the value for the current operating condition; the higher steam flow balanced the higher steam pressure. If the plant is operated at a reduced T_{avg} or if the current T_{hot} is maintained, intermediate values of separator parameter will apply.

Figure 2-1 plots the trend of moisture carryover versus separator parameter for two plants with the same separator package as DCPP Unit 1. This trend is followed by all Model 51 SGs with this package, although the scatter of individual moisture measurements is significant. Also plotted on this figure is the single available moisture measurement made at DCPP Unit 1. This value is seen to be close to the trend line plotted. The vertical line, identified as the current rating at a steam pressure of 808 psia, represents the present full power operating point in terms of the separator parameter. The expected and measured moisture carryover are near 0.1 percent.

To determine the expected values of moisture carryover at the uprated power, vertical lines are drawn on Figure 2-1 corresponding to the operating assumptions, which determine the value of steam pressure. The line on the far right corresponds to the uprated design condition with a low steam pressure resulting from assumptions of high fouling and plugging levels. The moisture carryover for this case will significantly exceed 0.25 percent. On the far left, the uprated operating point corresponds to best estimate assumptions with regard to fouling and plugging. The high steam pressure causes the separator parameter to remain near current levels even with of the increase in power. The moisture will also remain near the current value of 0.1 percent. If the plant is operated at a primary T_{avg} reduced by 5.5°F, a steam pressure of 805 psia is expected and the separator parameter will take on an intermediate value; the moisture level could be expected to be near 0.25 percent. Increasing power but maintaining T_{hot} at the current operating value will result in a slightly lower steam pressure, 795 psia. At this pressure, the moisture could exceed 0.25 percent, but only slightly.

The analysis has indicated that the moisture separator performance will be a function of the steam pressure at which the plant is operated. As shown in Table 3 of the introduction, the best estimate projected steam pressure for the planned Unit 1 uprated full power programmed reactor coolant system (RCS) T_{avg} = 573.49°F is about 795.2 psia which should maintain the SG moisture carryover near a value of 0.25 percent. However, DCPP will be performing a test after Unit 1 achieves steady state conditions at the new uprated full power to measure the moisture carryover. If it is determined that the moisture carryover exceeds the desired limit of 0.25 percent in one or more SGs, DCPP has several options which can be pursued. These include increasing the full power RCS T_{avg} and associated steam pressure and/or performing separator repairs as necessary during the next refueling outage. In addition, Westinghouse has developed modifications for the separator systems which have been field tested over the complete range of operating conditions considered in this analysis and could deliver a moisture carryover of less than 0.15 percent over the range of all Unit 1 uprate conditions.

U-Bend FIV

The effect of uprate on U-bend flow-induced vibration has focused on the most limiting aspects of the resulting tube response. This includes the effect of potential increased

tube wear at the Anti-Vibration Bar (AVB) locations, and the effect of decreased steam pressure resulting in increased fatigue usage of certain unsupported U-bend tubes.

U-bend wear evaluations have been performed for various SG models. These evaluations were made to assess the effect of power increases and/or steam pressure reductions on U-bend wear at the AVBs. The steam pressure reductions were a result of assumed tube plugging or primary temperature reductions. Table 2-1 shows the results of these evaluations in terms of the increase in percentage and number of tubes subject to plugging due to wear, as a result of the operating condition changes. Evaluations were also made for SG Models 51, F and D3.

In the table the Model 51 shows a potential increase in plugging of only one tube per SG. The Model 51 uprating included a 5 percent increase in power and a 110 psi reduction in steam pressure, changes that are significantly greater than those for the Unit 1 uprating. These results are consistent with operating experience; the Model 51 SGs have shown minimal susceptibility to U-bend wear. Therefore, the uprating is anticipated to have no significant effect on U-bend wear for Unit 1.

With respect to the potential for the uprated operating condition to increase the fatigue usage accumulation of unsupported U-bend tubes, analyses have been completed that address these effects. The original evaluation performed in 1988 contains specific details regarding the methodology, models, computer codes and assumptions used in the analysis of unsupported U-bend tubes at DCPP. These documents contain specific information regarding the acceptability of unsupported SG tubes at DCPP in a nonuprated condition. The original evaluation also addressed the effects of various historical changes in operating condition that occurred in the plant. The methods used to demonstrate acceptability of unsupported Unit 1 U-bend tubes in an uprated condition are the same as the methods used to document acceptability of the original operating conditions including the historical changes. The general analysis methods included determining the necessary SG secondary side operating conditions using the ATHOS computer code. These values were defined on a tube-by-tube basis and included: secondary side fluid velocity, density, and void fraction. Computer codes FASTVIB and PLOTVIB were then used to determine the fluid elastic stability ratio and stress ratio associated with each of the tubes and included the effects of localized flow peaking as a result of nonuniform AVB insertions. The effects of past operation and future operation (at a given condition, such as uprate) were then used to determine the fatigue usage associated with the limiting tubes. Tubes found with a fatigue usage greater than 1.0 were then recommended for preventative action.

Table 2-2 provides a summary of the general changes in secondary side thermal hydraulic conditions that could be expected as a result of the Unit 1 uprate. These values are generalized U-bend values developed on a consistent basis to define the general thermal/hydraulic characteristics of the U-bend. A one dimensional relative stability ratio (1D-RSR) multiplier was developed using this data that defines the

relative change in stability ratio that could be expected at a revised operating condition as documented in WCAP-12064, "Diablo Canyon 1 and 2 for Tube Vibration Induced Fatigue Evaluation", submitted in PG&E letter DCL-88-290, dated November 30, 1988. This 1D-RSR multiplier was then used to define changes in any given tube as a result of the revised condition. (Note that each tube was originally evaluated on an individual basis using tube specific velocity, density and void fraction profiles.) Table 2-2 shows that while some revised uprate conditions would tend to increase the tube stability ratio, others would tend to decrease the tube stability ratio. Parameters that would increase the stability ratio include increased fluid velocity and increased void fraction. Although void fraction is not used directly in the calculation of fluid elastic instability, the effect of decreased damping, which is related to increased void fraction, has a significant effect. It has been noted that a decrease in density would result in a lower stability ratio.

It has been determined that any changes in frequency due to the uprate are relatively small. The decrease in secondary side density would tend to increase the frequency of the tube, which would then tend to decrease the potential for fluid elastic excitation. However, these changes are so small that the effects of frequency changes have been conservatively omitted from the calculation.

The net impact of these effects produces a condition where an increase in the fluidelastic stability ratio would be expected. The analysis of the U-bend tubes in the uprated condition was then performed by modifying the original analysis to account for the increased potential for fluid elastic instability. Note that an evaluation showing that the Diablo Canyon Unit 1 steam generator tubing will not rupture by fatigue in the manner of North Anna Unit 1 can only be done by an assessment relative to the Row 9 Column 51 tube of Steam Generator C, North Anna Unit 1, for the following reasons:

1) methods for direct analytical prediction of actual stability ratios incorporate greater uncertainties than a relative ratio, and

2) the stress amplitude (or displacement) associated with a specific value of stability ratio can only be estimated by the analysis of North Anna Unit 1.

For these reasons, the evaluation was done on a relative basis to Row 9, Column 51 as documented in WCAP-12064. The specific criteria associated with the uprated condition that were used to demonstrate acceptability of the tubes included: (1) demonstrating that the stress ratio is less than 1.0, and (2) the total fatigue usage (considering all prior operating experience, and future operation at a given condition) is also less than 1.0.

Based upon this evaluation, it was determined that both the stress ratio and the fatigue usage in the U-bend region is acceptable for a steam pressure reduction of 760 psia or higher for all tubes except one Unit 1 tube. This tube is located in S/G 1 Row 8 column 30 and has been removed from service. A Framatome cable tube damper was installed

and the tube was plugged during 1R8 in May 1997. The next tube having the highest stability ratio and stress ratio has been determined to be tube R10C60 located in SG 4. This tube was found to be acceptable with a stress ratio of 0.77 in an uprated condition (vs. 0.50 in a non-uprated condition). In addition, a total fatigue usage of 0.22 was calculated which includes the effects of previous operation at prior operating conditions, and future operation at an uprated condition with steam pressures as low as 760 psia. All other tubes were determined to be less limiting than this tube and have been found to have a total fatigue usage less then the value calculated for R10C60.

As discussed previously, the projected Unit 1 full power programmed RCS T_{avg} of 573.49°F should provide a steam pressure of about 795.2 psia, which is well above the lower evaluation value of 760 psia. DCPP has established the 760 psia SG pressure as a design basis limit. Unit 1 would have to experience a significant increase in SG tube plugging and/or fouling to cause steam pressure to reduce to this lower limit. In addition, it is not expected that Unit 1 would ever operate with such a low SG pressure, since this would result in significant thermal efficiency and MWe output losses due to choking conditions in the high pressure turbine.

Table 2-1
Increased Tube Plugging Due to AVB Wear Results for Uprating And Steam Pressure Reduction

	Uprating	Steam Pressure Decrease	Increase in Plugging	
Model /Operational Change	(%)	(psi)	% of Total	Number per SG
Model 51, Uprating and 20% Tube Plugging	5.0	110	0.03	1
Model F, Uprating	4.5	47	0.08	5
Model F, Uprating and T_{hot} Reduction	4.5	85	0.15	8
Model D3, 18% Tube Plugging	--------	105	None	None

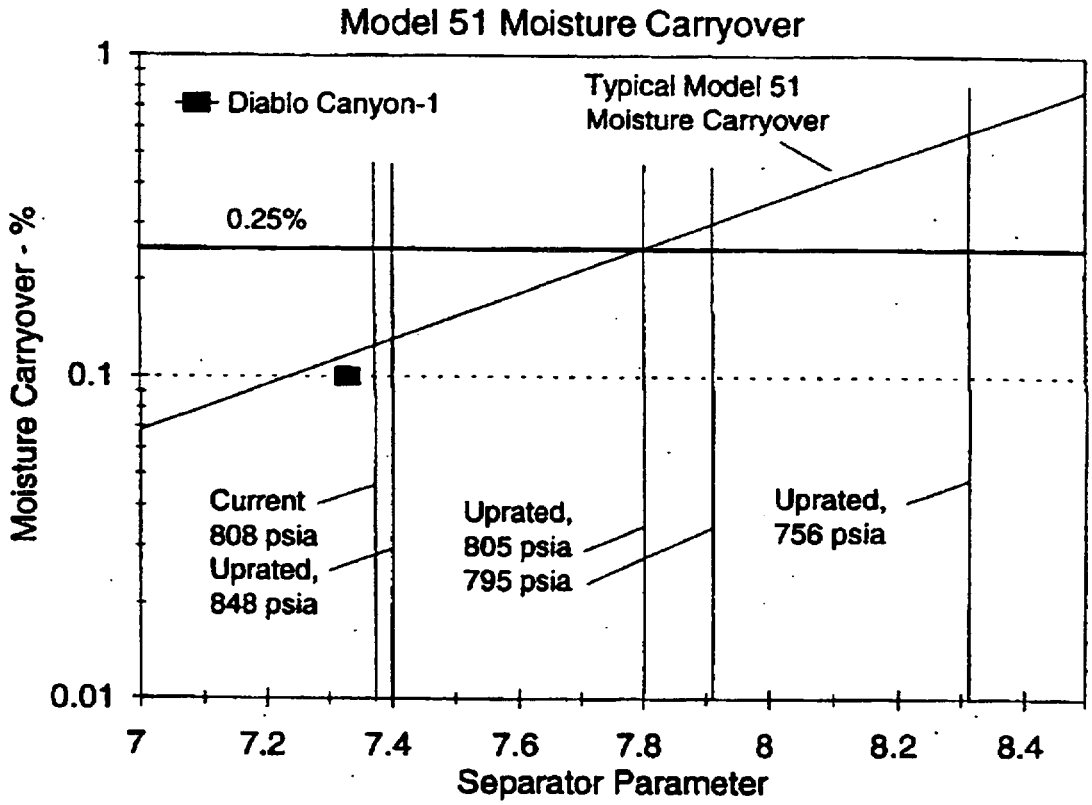
Table 2-2

Summary of Generalized Secondary Side Thermal Hydraulic Conditions in the SG U-bend

Condition	Density (lb/ft^3)	Velocity (Ft/Sec)	Slip Flow Void Fraction	Damping
Original	11.83	8.59	0.7818	1.238
Uprated	11.54	8.95	0.7880	1.205

Note: The values listed above are generalized values developed to define the 1D-RSR multipliers used to modify the original analysis.

Figure 2-1
Model 51 Steam Generator Moisture Carryover



Question 3

In Section 5.2, you stated that the pressurizer structural evaluation was performed by modifying the existing analysis of record based on the NSSS performance parameters provided in Table 2.1-1 and that the results indicated that the Diablo Canyon Power Plant (DCPP) Unit 1 and 2 pressurizer components meet the stress/fatigue analysis requirements of the ASME Code for the 3425 Mwt NSSS uprating parameters and transients. Provide the maximum calculated stress and CUF at the critical locations (such as surge nozzle, skirt support, spray nozzle, safety and relief nozzle, upper head/upper shell and instrument nozzle) of the pressurizer, the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

PG&E Response to Question 3

The evaluation of the pressurizer components at the uprated design conditions as listed in Table 2.1-1 of LAR 99-03, Enclosure B, determined that the original maximum calculated stresses remained bounding.

The evaluation determined that the uprate results in a small increase in the RCS T_{hot} from 608.8°F to 610.1°F, which decreases the temperature difference between the pressurizer (653°F) and any insurge of T_{hot} fluid which flows through the surge nozzle. This will actually reduce the stresses slightly in the surge nozzle and the other lower head components. The RCS T_{cold} for the Unit 1 uprate remains unchanged at 542.2 °F. Thus the existing evaluations of the spray nozzle and upper shell components are unaffected by the uprate.

Table 3-1 lists the applicable code of record, the allowable code limit, the maximum calculated stress, and the CUF for each pressurizer component.

These stress values all remain unchanged from the original stress report. However, the fatigue usage factors for some of the components have been previously changed (unrelated to the uprate) to account for various changes in the loadings or off-normal transients since the original stress report was generated.

Table 3-1
Summary of Pressurizer Structural and Fatigue Analysis Results

System or Component	Applicable Code Edition and Addenda	Allowable Code Limit	Maximum Calculated Stress	Cumulative Fatigue Usage Factor
Upper Head and Upper Shell	ASME Sect. III 1965, Summer '66 Addenda	90.00 ksi	67.01 ksi	<0.83
Spray Nozzle	ASME Sect. III 1965, Summer '66 Addenda	39.60 ksi	21.73(1) ksi	0.78
Safety and Relief Nozzle	ASME Sect. III 1965, Summer '66 Addenda	38.88 ksi	17.97 ksi	<0.24
Manway	ASME Sect. III 1965, Summer '66 Addenda	40.05 ksi	34.54 ksi	0.00
Valve Support Bracket	ASME Sect. III 1965, Summer '66 Addenda	54.00 ksi	40.22 ksi	0.01
Seismic Support Lug	ASME Sect. III 1965, Summer '66 Addenda	90.00 ksi	51.25 ksi	<0.05
Lower Head (Cast/Fabricated)	ASME Sect. III 1965, Summer '66 Addenda	58.20 / 90.00 ksi	34.15 / 37.45 ksi	<0.02
Heater Well	ASME Sect. III 1965, Summer '66 Addenda	69.90 ksi	27.30 ksi	<0.07
Immersion Heater	ASME Sect. III 1965, Summer '66 Addenda	49.20 ksi	38.23 ksi	<0.01
Surge Nozzle	ASME Sect. III 1965, Summer '66 Addenda	58.16 ksi	42.66 ksi	<0.28
Instrument Nozzle (Cast/Fabricated) Head)	ASME Sect. III 1965, Summer '66 Addenda	57.84 / 90.00 ksi	55.78 / 81.64 ksi	<0.16
Support Skirt	ASME Sect. III 1965, Summer '66 Addenda	69.30 ksi	68.60 ksi	<0.02

(1) Thermal bending stresses removed as part of simplified elastic-plastic analysis

Question 4

In regard to Section 5.4, provide the maximum calculated stress and CUF at the critical locations of the RPV and internals (nozzles, lower and core plates, core barrel, baffle/barrel, thermal shield supports, control rod drive mechanism, and fuel assembly). Also, provide the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification.

PG&E Response to Question 4

Reactor Vessel

The previous reactor vessel analyses were determined to remain applicable for the Unit 1 uprate parameters, and the maximum stress intensity ranges and cumulative usage factors are unchanged. The basis for the reactor vessel information is the Enhanced Load Follow Study which was performed in 1986, which contains tabulations of the maximum ranges of stress intensity and the maximum cumulative usage factors at the various critical location in the reactor vessel. These results are compared to the allowable Code limits, and are all certified to the requirements of the 1965 Edition of Section III of the ASME Boiler and Pressure Vessel Code through the Winter 1966 Addenda. This code edition and addenda comprise the code of record for the Unit 1 reactor vessel. Table 4-1 summarizes the stress intensity and CUF results for the various critical locations in the Unit 1 reactor vessel.

Reactor Vessel Internals

The reactor internals components were designed and fabricated before the existence of Subsection NG (core structures) of the ASME code and therefore, no ASME code design or stress report exists for those reactor internals. Instead, analyses were performed on an item by item basis for all the reactor internals components using sound engineering practice at the time of construction, and since then have been updated by analyses for specific application or points of interest (such as power uprate, fuel upgrades, etc.).

Baffle-Barrel Region (Core barrel, baffle plates, bolting and former plates)

No new CUF calculations were performed for the Unit 1 uprate. As discussed in Response 1, the existing design transients remain valid for the power uprate. The heat generation rates seen by the baffle-barrel region for the power uprate are bounded by the existing analysis. The current analysis of record remains applicable for the proposed power uprate conditions.

Upper Core Plate

No new CUF calculations were performed since the existing design transients remained valid for the power uprate. The effect of heat generation is negligible on the upper core plate, due to the distance between the active fuel and the upper core plate. The current analysis of record remains applicable for the power uprate conditions.

Thermal Shield Support System

No new CUF calculations were performed since the existing design transients remain valid for the power uprate. The effect of heat generation is negligible on the thermal shield support system. The current analysis of record remains applicable for the power uprate conditions.

Lower Core Plate

New CUF calculations were performed for the power uprate conditions as summarized below, due to the increase in heat generation seen by the lower core plate. The existing design transients remain valid for the power uprate.

SI = 43.31 KSI $3S_m$ = 48.3 KSI Margin = $(3S_m/S) - 1 = 0.12$

CUF = 0.52 (Allowable Limit = 1.0)

The applicable code and additions are ASME B&PV Code Section III, Division I, Appendices 1989, 1990 Edition and 1989 Edition Appendices.

The control rod drive mechanisms (CRDMs) and fuel assemblies are noncritical components in that their design bases remain unchanged after consideration of the uprating, as stated in Sections 5.5 and 6.4 of WCAP-14819.

TABLE 4-1
Summary of Reactor Vessel Structural and Fatigue Analysis Results

Vessel Location	**Applicable Code Edition and Addenda**	**Allowable Code Limit for $P_L + P_b + Q$ Range**	**Maximum Range of Primary plus Secondary Stress Intensity**	**Cumulative Fatigue Usage Factor**
Flange Region				
1. Closure Head Flange	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 80.1 ksi	41.4 ksi	---
2. Vessel Flange	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 80.1 ksi	44.6 ksi	---
3.. Closure Studs	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 99.0 ksi	93.1 ksi	0.9842 < 1.0
Outlet Nozzle	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 80.1 ksi	52.59 ksi	0.243 < 1.0
Instrumentation Tubes	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 69.9 ksi	42.94 ksi	0.11 < 1.0
Core Support Blocks (Pads)	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 69.9 ksi	46.22 ksi	0.257 < 1.0
Inlet Nozzle	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 80.1 ksi	48.0 ksi	0.064 < 1.0
Vessel Wall Transition	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 80.1 ksi	31.9 ksi	0.020 < 1.0
Bottom Head to Shell Juncture	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 80.1 ksi	34.09 ksi	0.008 < 1.0
CRDM Housing	ASME Sect. III, 1965, Winter '66 Addenda	$3S_m$ = 69.9 ksi	57.69 ksi	0.013 < 1.0

Question 5

In regard to Section 5.4.2, provide in detail a quantitative assessment of flow-induced vibration of the reactor internal components due to power uprate.

PG&E Response to Question 5

The flow-induced vibrations of pressurized water reactor internals have been studied at Westinghouse for a number of years. The objective of these studies was to assure the structural integrity and reliability of the reactor internals components. These efforts have included in-plant tests, scale model tests, tests in fabricators' shops, bench tests of components, and analytical investigations. The evaluations performed to determine any impact on the structural integrity of the reactor internals with regard to flow-induced vibrations for the Unit 1 uprating parameters were based on these study results. These evaluations are summarized below.

Lower Internals Response

The primary cause of lower internals excitations is flow turbulence generated by the expansion and turning of the flow at the transition from the inlet nozzle to the barrel-vessel annulus, and wall turbulence generated in the downcomer. For the uprating program, the response of the Unit 1 lower internals could be affected by changes in design performance parameters, i.e.:

a) Mechanical design flow rates
b) Operating temperatures
c) System modes of vibration

Since the mechanical design, thermal design flows, and the reactor internals system frequencies are unchanged with the uprating program, and the core inlet temperature is only slightly changed (from 544.4°F to 544.5°F per Table 2.1-1 of LAR 99-03, Enclosure B), the power uprating will have no significant impact on the lower internals vibrational response.

Upper Internals Response

The significant flow-induced forces on the upper internals are due to random turbulence generated by the cross flows that converge on the outlet nozzles. Therefore, the guide tubes and the upper support columns, which lie in the vicinity of the outlet nozzle, will experience the maximum flow-induced forces. The magnitude of these forces (fluctuation as well as steady state drag) is proportional to the square of the fluid velocity at the outlet nozzles. Evaluations have been performed which show that these

forces, due to flow-induced vibration on the guide tubes and the upper support columns, remain essentially unchanged for the uprated conditions.

Summary

The results of this assessment indicate that there is no adverse impact on the flow-induced vibrational response of the reactor internals for the Unit 1 uprated conditions.

Question 6

In reference to Section 5.7, list the ASME Class 1 branch piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating NSSS piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Provide the calculated maximum stresses for the critical piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

PG&E Response to Question 6

NSSS Piping (Reactor Coolant Loops)

The methodology and assumptions used for evaluating the effects on the DCPP reactor coolant loop (RCL) piping, component nozzles and pipe supports are summarized below.

The RCL piping and supports are analyzed for thermal effects, seismic effects, Loss-of-Coolant Accident (LOCA) effects, and effects from pipe breaks. The only impact to the piping and supports due to the uprating would be because of temperature changes to the hot leg piping, the crossover leg piping, and the cold leg piping as shown in Table 1-1. The change in hot leg temperature is determined from the vessel outlet temperature, which changes by only 1.3°F. The change in crossover leg temperature is determined from the SG outlet temperature, which does not change. The change in cold leg temperature is determined from the vessel inlet temperature, which changes by 0.1°F. These temperature changes are very small and are not significant for the evaluation of the RCL piping.

The seismic analysis results are not impacted by thermal effects associated with temperature changes. The LOCA and pipe break analyses are not affected since there are no changes to any LOCA forces which are imposed on the piping and equipment, and there are no changes to the postulated breaks at secondary side pipe nozzles. DCPP has implemented the leak-before-break (LBB) methodology as detailed in WCAP-13039, "Technical Justification for Eliminating Large Loop Rupture as a Structural Design Basis for Diablo Canyon Units 1 and 2 Nuclear Power Plants," submitted in PG&E letter DCL-92-059, dated March 16, 1992. However, the limiting pipe stress values for the RCL piping as shown in Tables 6-1 and 6-2 are conservatively based on the large break LOCA forces.

The faulted stress ratios were relative comparisons of axial stress (Sa) to yield and stress intensity (SI) to yield ratios. These original design stress ratios were limited to a maximum value of 3.0. Subsequently, as reported in the DCPP Final Safety Analysis

Report (FSAR), the allowable was changed to 3.6 Sh, which is equal to 3.0 Sy (the original limit). Both of the values 3.6 Sh and 3.0 Sy are equal to 61.38 ksi, which is the applicable faulted limit for the ratios shown in Table 6-2.

The RCL piping will experience no additional displacements due to any loading conditions associated with the uprated parameters. Therefore, the component nozzles, branch line nozzles, pipe supports, and class 1 branch piping systems will not see any change in loads or stresses from those contained in the original design basis analysis. These original component loads and stresses along with the allowable code limits are listed in Tables 6-1 and 6-2. The code of record and code edition as documented in the original design basis analysis remain unchanged. The applicable code of record is the ANSI B31.1 Power Piping Code, 1967; where the stress requirements are not provided in the 1967 edition, the stress equations of ANSI B31.1 Summer 1973 are used.

NSSS Branch Systems

The Unit 1 uprated conditions were also evaluated for any impact on the stress analyses for the NSSS branch piping, branch systems and components including the RCL branch piping (surge and spray lines), the chemical and volume control system (CVCS), the residual heat removal (RHR) system, and the safety injection (SI) system.

As stated above, the uprate results in a T_{hot} temperature of 610.1 °F, an increase of 1.3 °F over the current T_{hot} value. The RCS pressure does not change with the uprate. The temperature increase impacts only the surge line's current pipe stress analysis since this analysis assumes a temperature of 610 °F. The 0.1 °F difference between the current pipe stress analysis input temperature and the uprated T_{hot} temperature was determined to have a negligible impact on the RCS surge line pipe stress analysis. Tables 6-3 and 6-4 show a summary of the pressurizer surge line's current pipe stress analysis results. The margin between the maximum calculated stress levels and allowable stress levels, as well as the small temperature increase involved, establish that there is a negligible impact on the surge line stress results

For the remaining NSSS branch systems, the uprated pressure and temperature values did not increase above the values assumed in the current pipe stress analysis.

Therefore, the Unit 1 uprate did not require any revised calculations or stress analysis revision for the NSSS branch piping and associated components and supports.

The PG&E design class and their relationship to the applicable codes of record for the NSSS and Balance of Plant (BOP) piping systems are summarized in Attachment A.

Table 6-1
Summary of RCL Stress Analysis Results

Piping Section	Dead-wt. + Pressure Stress	Allowable Stress	Thermal Expansion Stress	Allowable Stress	Upset Stress	Allowable Stress
Hot Leg	6,735 psi	17,050 psi	10,500 psi	27,700 psi	11,635 psi	20,460 psi
Crossover Leg	6,771 psi	17,050 psi	4,800 psi	27,700 psi	11,371 psi	20,460 psi
Cold Leg	6,700 psi	17,050 psi	2,500 psi	27,700 psi	15,300 psi	20,460 psi

Table 6-2
Summary of RCL Faulted Condition Piping Stress

	Piping Stress (ksi)			Stress Ratios	
	Sl (max)	**Sy**	**Sa**	**Sa/Sy**	**Sl/Sy**
Hot Leg	17.8	20.46	6.6	0.32	0.87
Crossover Leg	27.9	20.46	5.9	0.29	1.36
Cold Leg	45.7	20.46	11.4	0.56	2.23

Table 6-3
Summary of B31.1 Piping Stresses - RCS Surge Line (without thermal stratification effects)

Design Condition	Node	Max Stress (psi)	Allowable Stress (psi)		Ratio to Allowable
Thermal Expansion Eqn. 14 (11)	5020	24303	$S_A + S_h =$	44000	0.55

Note: S_h = Basic Allowable Stress at maximum (hot) temperature (16600 psi)

S_A = Allowable Stress Range for Expansion Stresses (27400 psi)

Table 6-4
Summary of the four highest ASME III NB-3600 Code Eqn. 12 Stresses including Thermal Stratification effects

Component	Maximum Stress (ksi)	Allowable Stress 3.0 Sm (ksi)	Ratio to Allowable
Hot Leg Nozzle	55.7	57.9	0.96
Pressurizer Nozzle to Pipe Weld	40.7	54.4	0.75
5-D Bend	44.5	57.9	0.77
Long-Radius Elbow	50.2	54.4	0.92

Fatigue Usage Factor		
Location	Max Usage Factor	Allowable
RCL Nozzle Safe End	0.97	1.0

Question 7

Discuss the functionality of safety-related mechanical components (i.e., all safety related valves and pumps, including power-operated relief valves) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) in your Generic Letter (GL) 89-10 MOV program at DCPP will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Also, provide evaluation of the effects of the proposed power uprate on the pressure locking and thermal binding of safety-related power-operated gate valves for GL 95-07 and on the evaluation of overpressurization of isolated piping segment for GL 96-06. Identify mechanical components for which functionality at the uprated power level could not be confirmed.

PG&E Response to Question 7

Safety-related Mechanical Components

There is no change or impact on any safely-related equipment functionality or performance criteria due to the Unit 1 power uprate. As discussed in Section D of Enclosure B (LAR 99-03), the majority of the DCPP safety analyses including the current Environmental Qualification (EQ), mass and energy, containment integrity, steam generator tube rupture (SGTR), and most non-LOCA analyses assumed the higher Unit 2 core power of 3411 MWt, combined with a lower Unit 1 RCS flow rate in order to bound both Units with one analysis. The large break LOCA and small break LOCA analyses are evaluated separately due to the differences in the Unit 1 and Unit 2 reactor vessel designs as discussed in RAI Response 6 in PG&E letter DCL 00-098, dated July 7, 2000. The emergency core cooling systems (ECCS) performance criteria are the same for both Units. The Unit 1 performance criteria and safety-related mechanical components (including ECCS valves and pumps, power-operated relief valves, and safety relief valves) are identical to Unit 2 which is already analyzed and licensed for a core thermal power of 3411 MWt.

With regard to other NSSS systems, it is noted that the uprated Unit 1 RCS hot leg temperature increased by less than 2°F, and the increases in T_{avg} and T_{cold} were less than 1°F. The temperature of the SG outlet (crossover leg), to which the CVCS letdown line is connected, was unchanged. Other auxiliary system pressures, temperatures, and flow rates either remain unchanged or are bounded by the Unit 2 analyses. While updated RHR cooldown times have been determined for Unit 1 at the uprated conditions, the system pressure, temperature, and flow rate remain unchanged.

WCAP-14819 contains the following conclusions concerning the functionality of the DCPP Unit 1 auxiliary equipment at the uprated conditions:

- The DCPP Unit 1 uprating design parameters have no effect on the qualification of the auxiliary pumps, auxiliary heat exchangers, auxiliary tanks, and auxiliary valves.

- There are no new limitations associated with the auxiliary pumps, auxiliary heat exchangers, auxiliary tanks, and auxiliary valves due to the implementation of the Unit 1 uprating program.

Therefore, the performance requirements of the Unit 1 safety-related NSSS pumps and valves remain unchanged by the uprated plant conditions. Furthermore, these conditions are now comparable to those of Unit 2, which were evaluated previously.

GL 89-10 Evaluation

The Unit 1 power uprate has not impacted any parameters or performance criteria used to establish the functionality of safety related MOVs in the DCPP response to GL 89-10 (PG&E letter DCL89-324, dated December 27, 1989). The pressure, temperatures, and flows used in the GL 89-10 evaluations were based on the allowable system condition limits established in the Technical Specifications (TS), alarm setpoints, the ECCS performance criteria, and the postaccident conditions which could exist when the MOV is required to function. A common set of GL 89-10 MOV evaluations were performed to bound both Unit 1 and Unit 2 MOVs such that the uprate remains bounded by the existing evaluations as discussed below.

There is no change in any plant alarm setpoint including the maximum RCS letdown temperature and the maximum excess letdown temperature. The only change to the TSs are the core thermal limits and the maximum RCS programmed temperature associated with Mode 1 full power operation. There is no change in any TS definition or temperature limit related to any of the lower modes of operation which are applicable to the post-accident conditions established for the GL 89-10 evaluations. In addition, there is no change in the pressurizer power-operated relief valve or the pressurizer safety valve setpoints which establish the maximum RCS pressure conditions assumed for the GL 89-10 evaluations.

As discussed previously, the Unit 1 uprate does not impact any ECCS performance criteria as these systems are identical for both units and their performance has always met the Unit 2 design criteria for a core thermal power of 3411 MWt. There is no change in the maximum assumed pump flow, pump head, or the differential pressure related to the performance of ECCS MOVs. Similarly, the postaccident conditions assumed for GL 89-10 were based on analysis input assumptions which are common to

both Unit 1 and Unit 2 (e.g., refueling water storage tank (RWST) temperature), and the more limiting postaccident LOCA conditions (e.g., containment and recirculation sump conditions) which are already bounding for the Unit 2 core thermal power of 3411 MWt. In summary, the Unit 1 uprate does not impact the acceptable postaccident function of any MOVs as established in the GL 89-10 evaluations.

GL 95-07 and GL 96-06

The postaccident conditions and equipment functions evaluated for the gate valves in GL 95-07 and penetration piping in GL 96-06 are common and bounding for both Units 1 and 2. These evaluations are not impacted by the Unit 1 uprate.

PG&E's response to GL 95-07 was documented in PG&E letter DCL-95-205, dated October 11, 1995. The evaluation established that neither Unit 1 nor Unit 2 had any gate valve designs susceptible to thermal binding and documented the evaluations or physical modifications which were performed to ensure that the pressure locking of gate valves was not a concern at DCPP. The PG&E response to GL 96-06 was documented in PG&E letter DCL-012, dated January 28, 1997. This letter discussed the evaluations performed to ensure that during postaccident conditions, there would be no adverse two phase flow conditions within the component cooling water system (CCWS) or the auxiliary saltwater system (ASW), and there would be no over pressurization of any isolated piping segments which penetrate the containment.

The Unit 1 uprate does not involve any physical plant modifications or changes in equipment functions. Enclosure B, Sections 4.2.1 and 4.2.2 of LAR 99-03 document that the Unit 1 uprate has a minimal impact on normal operating conditions of the CCWS and ASW system. The performance requirements for these systems remain bounding for the Unit 1 uprate, since the applicable post-accident conditions assumed for these evaluations are already bounding for the more limiting Unit 2 core thermal power of 3411 MWt.

In summary, the DCPP evaluations for GL 95-07 and GL 96-06 remain valid and bounding for the Unit 1 uprate conditions.

Question 8

In reference to Section 7, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate. Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports. Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences. Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

PG&E Response to Question 8

The following BOP systems were evaluated for the uprate from a thermal hydraulic performance perspective.

- main steam
- auxiliary steam
- extraction steam and heater drip
- circulating water
- condensate
- main feedwater
- auxiliary feedwater
- service cooling water
- main lube oil and electro hydraulic (EH)

This group of systems (with the exceptions noted below) were further evaluated to determine if the uprate would impact the current pipe stress analysis. Service cooling water, circulating water, and main lube oil and EH were excluded from this evaluation since only the PG&E Design Class I systems or Design Class II systems with operating temperatures greater than 200 °F (for carbon steel pipe) are required to have thermal pipe stress analysis.

To assess what impact the new uprated conditions would have on the pipe stress analyses for the remaining systems of this group, PG&E established that the current Unit 2 secondary heat balance (FSAR Fig. 10.1-4) was applicable to both units since they have essentially identical secondary components, similar piping layouts, and (after the uprate) will have the same thermal rating . With the exception of main feedwater (MFW), applying the Unit 2 heat balance temperature and pressure parameters to the Unit 1 BOP systems did not impact any pipe stress analyses since the current Unit 1

pipe stress analysis temperature and pressure inputs bound or are considered equivalent to the associated Unit 1 uprated BOP values.

Since it was determined that no temperature or pressure inputs to the Unit 1 pipe stress analyses have changed (with the exception of MFW), no further analyses were performed for the BOP piping, components, pipe supports, nozzles, penetrations, guides, valves, pumps heat exchangers, or anchorage for pipe supports.

The Unit 1 uprate MFW maximum operating temperature increased to 432 °F, which is 1 °F above the temperature value of 431 °F used in the current pipe stress analysis. PG&E determined that this temperature increase would have a negligible impact on the pipe stress analysis. Also note that for this case, the pressure stress component of the current MFW pipe stress analysis is based on a pressure of 1415 psi which is the maximum pipe specification design pressure rating. This margin between the stress analysis design input pressure (1415 psi) and the maximum operating pressure (1100 psi) and the slight temperature increase involved (431 °F to 432 °F) establishes the bases that no revision to the existing MFW pipe stress analysis is required.

PG&E performed a supplementary quantitative evaluation of several critical piping segments within the MFW system which is summarized in Table 8-1. As Table 8-1 indicates, the effect on piping stresses as a result of the small temperature change (from 431°F to 432°F) is negligible and a detailed reanalysis is not warranted. Furthermore, the impact on the stress results due to an assumed increase in MFW temperature from 431°F to 435°F was also examined for reasons discussed below, and the changes were found to be negligible as well. The resulting margin in these supplemental pipe stress evaluations confirm the PG&E determination that the Unit 1 uprate did not require any reanalysis of the MFW system pipe stress.

As stated above, the new MFW system maximum operating temperature comes from the Unit 2 100 percent full power secondary heat balance (FSAR Figure 10.1-4). Note that in Table 2 of the introduction, the Unit 1 uprated design MFW temperature is listed as 435 °F. This maximum design MFW temperature is based on a conservative primary side heat balance which incorporates various limiting factors such as minimum RCS flow capability and maximum degraded SG heat transfer capability. This design primary heat balance does not predict the actual performance of the secondary system at 100 percent full licensed power, and as such, is not considered an appropriate design input for secondary system pipe stress analysis. However, the supplementary MFW pipe stress analysis was also performed with this design maximum temperature of 435 °F and still resulted in allowable stress ratios below 1.0 as shown in Table 8-1.

For all of the BOP systems, the code of record and edition have not changed from that used in the original design basis analysis. The codes of record for the various PG&E BOP piping systems are shown in Attachment A.

A proprietary piping analysis computer program developed by Bechtel Corporation was used in the original design basis pipe stress analysis for PG&E analyzed Design Class I piping systems. This same code was used to generate the supplementary quantitative evaluation results summarized in Table 8-1. No new computer codes have been employed for any piping or component stress analyses related to the Unit 1uprate.

Table 8-1
Summary of Supplementary Quantitative MFW Stress Analysis Evaluation

Pipe Description	Loading Condition	Location of Maximum Stress	Pre-Unit Uprate Stress (psi)	Post-Unit Uprate Stress (psi)		Allowable Stress (psi)	Pre-Unit Uprate Ratio to Allowable	Post-Unit Uprate Ratio to Allowable		Stress Increase (psi)	
				432°F	435°F			432°F	435°F	432°F	435°F
SG 1 FW Supply (inside ctmt)	Primary+Sec (Eqn 14)	430	33188	33233	33380	37500	0.885	0.886	0.890	45	192
SG 2 FW Supply (inside ctmt)	Primary+Sec (Eqn 14)	60	31596	31639	31780	37500	0.843	0.844	0.847	43	184
SG 3 FW Supply (inside ctmt)	Primary+Sec (Eqn 14)	55	36657	36708	36874	37500	0.978	0.979	0.983	51	217
SG 4 FW Supply (inside ctmt)	Primary+Sec (Eqn 14)	430	37128	37180	37350	37500	0.990	0.991	0.996	52	222
FW Supply Outside Ctmt	Primary+Sec (Eqn 14)	430	32565	32620	32803	37500	0.868	0.870	0.875	55	238

Question 9

Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of evaluation for power uprate effects on the high energy line break analysis, jet impingement and pipewhip loads for the power uprate condition.

PG&E Response to Question 9

Flow-induced Vibration

The Unit 1 uprate does not introduce any potential for flow-induced vibration on any heat exchangers based on the extensive operating history of comparable flow conditions and components which exist on Unit 2. The Unit 1 feedwater heaters, gland steam condenser, steam jet air ejector condenser, and main condenser are essentially identical to the Unit 2 components. The Unit 1 secondary flows after the power uprate will be comparable to the current Unit 2 flows. The Unit 2 heat exchangers have not experienced any indication of component degradation due to flow vibration effects and the same performance is expected for the identical Unit 1 heat exchangers after the uprate.

HELB/MELB and Pipe Whip

The following discussion provides a summary of the evaluation for determining the potential impact of the uprate conditions on the high energy line break (HELB) analysis, including any affects to jet impingement and pipewhip loads.

Inside Containment

The impact on the analyses of dynamic effects (e.g., jet impingement, pipe whip) of HELB inside containment was reviewed. The HELB line temperatures and pressure conditions for lines inside containment currently used as input for the jet impingement analysis are equal to or bound the new uprate conditions. Therefore, there is no adverse impact to HELB jet impingement or pipe whip effects inside containment.

Westinghouse evaluated the Unit 1 uprate conditions and determined there was no impact on the leak-before-break (LBB) margins, such that the LBB conclusions remain unchanged as detailed in WCAP 13039, "Technical Justification for Eliminating Large Loop Rupture as a Structural Design Basis for Diablo Canyon Units 1 and 2 Nuclear Power Plants," submitted in PG&E letter DCL-92-059, dated March 16, 1992. Thus, there is no impact to current HELB considerations.

The Unit 1 uprate conditions remain bounded by the liquid inventory assumptions which establish the maximum containment flood levels. Therefore, the containment flood level analyses are not adversely impacted by the Unit 1 uprate.

The Unit 1 uprate does not impact any input assumptions related to the generation of missiles and has no adverse impact on the DCPP missile analyses.

Outside Containment

The Unit 1 uprate conditions were also reviewed for any potential impact on the dynamic effects (e.g., jet impingement, pipe whip) associated with a HELB outside containment. The temperature and pressure conditions assumed for the HELB lines outside containment bound the new uprate conditions, including the major secondary cycle systems in the turbine building. Therefore, the existing analyses for these effects remain valid. Also, the analyses for HELB crack breaks and moderate energy line break (MELB) crack breaks for secondary cycle systems are not adversely impacted by the Unit 1 uprate conditions.

The temperature, pressure, and humidity conditions considered for postaccident environmental conditions for several areas outside containment are based on bounding main steam line breaks (MSLB) postulated between the containment penetrations, and the turbine stop valves in the turbine building. Because the temperature and pressure conditions in this section of the main steam supply are equal to or less than that previously considered, the existing analyses for establishing environmental conditions resulting from MSLB remain bounding or unchanged.

Based on the above discussions, the Unit 1 uprate conditions have no adverse impact on HELB, jet impingement, pipe whip loads, or MELB evaluations.

Question 10

Do you project modifications to piping or equipment supports for the proposed power uprate? If any, provide examples of pipe supports requiring modification and discuss the nature of these modifications. Did you follow WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," for evaluating the DCPP power uprate? If not, discuss the differences between the current power uprate analysis and the WCAP-10263 methodology.

PG&E Response to Question 10

As stated in LAR 99-03, Section B, "Background" the Unit 1 design criteria was established based on an ultimate expected thermal power of 3488 MWt. Therefore, increasing the Unit 1 core thermal power to 3411 MWt is well within the original plant design criteria and does not require any physical modifications to implement.

The Unit 1 uprate evaluations were performed consistent with the methods outlined in WCAP-10263.

Attachment A

PG&E RELATIONSHIP OF DESIGN AND QUALITY GROUP CLASSIFICATIONS

(UFSAR 3.2.3, Reference 8,"Classification of Structures, Systems, and Components for Diablo Canyon Power Plant Units 1 and 2 (Q-List), PG&E," Table 1)

TABLE 1

RELATIONSHIP OF DESIGN AND QUALITY GROUP CLASSIFICATIONS

TYPE OF EQUIPMENT	SEISMIC DESIGN CLASSIFICATION (a)	QUALITY GROUP CLASSIFICATION							REMARKS
	NRC REG. GUIDE 1.29	PG&E ENGINEERING		QA	ANSI N18.2 SAFETY CLASS (f)	NRC REG GUIDE 1.26 QUALITY GROUP (f)	IEEE-308-1971	PG&E PIPING CODE CLASS	(u) OTHER CODES AND STANDARDS
		DESIGN CLASS	QUALITY/CODE OR INSTR CLASS/CAT.						
MECHANICAL	SEISMIC CATEGORY I	I	I	Q	1	A		NONE	(b) ASA B31.1-1955; ASME B&PV Code Section III-1971
								A	(h) ANSI B31.1-1967; B31.7-1969 with 1970 Addenda, Class I
			II		2	B		B	ANSI B31.7-1969 with 1970 Addenda, Class II
								@	(g) ANSI B31.1-1967; ASME B&PV Code Section I-1968; Section III-1968
			III		3	C		C,D	(i) ANSI B31.7-1969 with 1970 Addenda, Class III
		II							
				G(d)				G	(j) ANSI B31.1-1967 and NFPA Standards
		I	III	Q				J	(k) ANSI B31.1-1967
ELECTRICAL							CLASS 1E		
INSTRUMENT & CONTROL			INSTR CLASS IA	Q					
			INSTR CLASS IB — CAT. 1	Q			Standby Power Highly Reliable		Reg Guide 1.97, Revision 3
			INSTR CLASS IB — CAT. 2	Q/T(o)					
			INSTR CLASS IC	Q(q)			NON-CLASS 1E		
			INSTR CLASS ID(s)						
STRUCTURAL				Q					
STRUCTURAL	NON-SEISMIC CATEGORY I	II(l)							
MECHANICAL					NNS			@(m),E	ANSI B31.1-1967
				R	NNS	D		F	(e) ANSI B31.1-1967; Reg Guide 1.143
				G(d)				G1	NFPA STANDARDS
								G1(v)	NFPA STANDARDS
				R	NNS	D		H	ANSI B31.1-1967; REG GUIDE 1.143
INSTRUMENT & CONTROL			INSTR CLASS II — CAT. 3	T(o)			NON-CLASS 1E		Reg Guide 1.97, Revision 3
			INSTR CLASS II	(d)					APPLICABLE INDUSTRY CODES AND STANDARDS
ELECTRICAL									(NON-1E)
STRUCTURAL AND MECHANICAL		III(l)							APPLICABLE INDUSTRY CODES AND STANDARDS

TABLE 1 NOTES

(a) General Design Criterion 2 (1967), Appendix A to 10 CFR 50 and Appendix A to 10 CFR 100.

(b) Reactor coolant loop and pressurizer surge line piping. Design to ASA B31.1-1955 using Nuclear (N) Code Cases N-7, N-9, and N-10. Fabrication, erection, and inspection to ASME B&PV Code Section III-1971.

(c) DELETED

(d) QA Classification G is required for those "Design Class II" fire protection components which require application of Quality Program as described in Appendix A to NRC Branch Technical Position 9.5.1 as defined in IDAP OM8.ID1.

(e) Seismic qualification requires Design Earthquake analysis.

(f) Regulatory Guide 1.26 (formerly AEC Safety Guide 26) establishes quality group classifications for Reactor Coolant pressure boundary and remaining safety-related components containing radioactive material, water, or steam. Other systems not covered by this guide include instrument and service air, diesel engine and its generators and auxiliary support systems, diesel fuel, emergency and normal ventilation, fuel handling, and radioactive waste management systems.

The Code Class I classification generally includes the fluid systems and components identified as Safety Class 1 in ANSI N18.2 and Quality Group A in AEC Safety Guide 26. However, the classification and quality standards for Diablo Canyon fluid systems and components were established prior to the existence of these documents and, therefore, do not always fall within their strict definitions. All Code Class I fluid systems and components are in accordance with the accepted industry codes and standards that were in effect during the design and construction of Diablo Canyon. If fluid systems and components were designed and constructed to codes and standards outside of the requirements of the above-mentioned documents, additional quality standards have normally been applied, so that their intent has been met.

The Code Class II classification generally includes the fluid systems and components identified as Safety Class 2a in ANSI N18.2 and Quality Group B in AEC Safety Guide 26. However, the classification and quality standards for Diablo Canyon fluid systems and components were established prior to the existence of these documents and, therefore, do not always fall within their strict definitions. All Code Class II fluid systems and components are in accordance with the accepted industry codes and standards that were in effect during the design and construction of Diablo Canyon. If fluid systems and components were designed and constructed to codes and standards outside of the requirements of the above-mentioned documents, additional quality standards have normally been applied, so that their intent has been met.

The Code Class III classification generally includes the fluid systems and components identified as Safety Classes 2b and 3 in ANSI N18.2 and Quality Group C in AEC Safety Guide 26. However, the classification and quality standards for Diablo Canyon fluid systems and components were established prior to the existence of these documents and, therefore, do not always fall within their strict definitions. All Code Class III fluid systems and components are in accordance with the accepted industry codes and

standards that were in effect during the design and construction of Diablo Canyon. If fluid systems and components were designed and constructed to codes and standards outside of the requirements of the above-mentioned documents, additional quality standards have normally been applied, so that their intent has been met.

(g) Feedwater piping from the (final) main feedwater check valve to the steam generator; auxiliary feedwater from the main feedwater line back to the second check valve; main steam piping from the steam generator to the main steam isolation valve; steam generator blowdown piping from the steam generator to the first valve outside containment; design to ANSI B31.1-1967; fabrication, erection, and inspection to ASME B&PV Code Section I-1968. Requirements for the main steam safety valves are in accordance with ASME B&PV Code Section III-1968.

(h) Design to ANSI B31.1-1967. Fabrication, erection, and inspection to ANSI B31.7-1969 with 1970 Addenda, Class 1.

(i) For Code Class D piping, B31.7 applies for new work. Original work was to B31.1-1967. Refer to component drawings for codes and standards on Mechanical Component.

(j) Piping is seismically qualified for the Hosgri event.

(k) Piping originally was installed as Design Class II and has been qualified seismically for the Hosgri event, but is Design Class I for repair, replacement, and new construction.

(l) Certain Design Class II and III SSCs have seismic qualification requirements and may be identified as Seismic Category I; they are designated as QA Class S in the Q-List and are controlled by IDAP CF3.ID11.

(m) @ also covers portions of the Design Class II Auxiliary Steam System (Boilers); see Section III.E.4 and Notes C-4c, C-7d, and C-7e in Table 2.

(n) DELETED

(o) QA in accordance with IDAP CF3.ID12.

(p) DELETED

(q) 'Q' applies to the portions of the component required to maintain system pressure boundary integrity (PBI) and structural integrity. Instrument Class IC devices require seismic qualification and mounting to assure PBI and structural integrity, but not functionality. This includes component supports and all tubing required for maintenance of system pressure boundary (see Instrument Schematics for IC classification boundaries).

(r) DELETED

(s) Instrument Class ID instruments also exist. The specific requirements for the Instrument Class ID instruments are given on a case-by-case basis in the respective system DCMs and/or Instrument Schematics.

(t) DELETED

(u) Refer to PG&E Drawings 102028 (Unit 1) and 104628 (Unit 2) for ASME Code Boundaries for Inservice Inspection.

(v) This line entry on the table is for the South Site Fire Water System (SSFWS).The SSFWS was not installed under the fire water graded QA program.