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U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Power Plant Units 1 and 2
Pressure and Temperature Limits Report, Revision 10

Dear Commissioners and Staff:

In accordance with Diablo Canyon Power Plant Technical Specification 5.6.6.c, enclosed is the Pressure and Temperature Limits Report (PTLR) for Units 1 and 2. It was issued as PTLR-1, Revision 10, effective March 23, 2009. Due to an oversight, this was not submitted within 30 days.

As provided under 10 CFR 50.59, the PTLR changes were made without prior NRC approval. The PTLR continues to meet the requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. This change removed references to the Original Steam Generators due to completion of the replacement of steam generator project. The changes are denoted with revision bars in the right margin.

There are no new or revised regulatory commitments in this report. If there are any questions regarding this submittal, contact Mr. Steve Hamilton at (805) 545-3449.

Sincerely,

James R. Becker

ddm/469/50271149
Enclosure
cc: Diablo Distribution
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NRR

PRESSURE AND TEMPERATURE LIMITS REPORT
DIABLO CANYON POWER PLANT, UNITS 1 AND 2
PTLR-1, REVISION 10 (31 Pages)
EFFECTIVE DATE MARCH 23, 2009

TITLE: PTLR for Diablo Canyon

1 AND 2

03/23/09
EFFECTIVE DATE

PROCEDURE CLASSIFICATION: QUALITY RELATED

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1. REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

This PTLR for Diablo Canyon has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.6. The TS addressed in this report are listed below:

- LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits
- LCO 3.4.12 Low Temperature Overpressure Protection (LTOP) Systems

The limits provided in this report remain valid until 23 EFPY on Unit 1 and Unit 2.

2. OPERATING LIMITS

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

The RCS temperature rate-of-change limits are:

- A maximum heatup of 60°F in any 1-hour period.
- A maximum cooldown of 100°F in any 1-hour period.
- A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Tables 2.1-1 and 2.1-2.

2.1.1 RCS P/T Limits:

The parameter limits for the specifications listed in section 1. are presented in the following subsections. The limits were developed using a methodology that is in accordance with the NRC approved methodology provided in WCAP 14040-NP-A (Ref. 8.4). The analysis methods implemented per ASME B&PV Code Section III Appendix G utilize linear elastic fracture mechanics, determine the maximum permissible stress intensity correlated to the reference stress intensity (K_{IR}) as a function of vessel metal temperature, define the size of the assumed flaw, and apply specified safety factors.

The reference stress intensity (K_{IR}) is the combined thermal and pressure stress intensity limit at a given temperature. The assumed crack has a radial depth of ¼ of the reactor vessel wall thickness and an axial length of 1.5 times wall thickness and is elliptically shaped.

10 CFR 50 Appendix G and Reg. Guide 1.99 provide guidelines for determining the maximum permissible (allowable) stress intensity, based on nil-ductility of the reactor vessel metals during the operational life of the reactor. The transition temperature at which the metal becomes acceptably ductile is affected by neutron radiation embrittlement over the course of reactor operation. Appendix G and Reg. Guide 1.99 provide formulas which are used to calculate this Adjusted Reference Temperature based on fluence and vessel material chemistry. The shift in nil-ductility resulting from the fluence effect is added to the unirradiated nil-ductility transition temperature and, with Reg. Guide 1.99 defined margins included, the Adjusted Referenced Temperature (ART) is established for a specified neutron fluence.

The allowable stress intensity is determined from ASME Code formula and is based on the difference between any given vessel metal temperature and the ART.

The thermal stress intensities were provided by Westinghouse (Appendix A to PG&E Technical & Ecological Services – TES - Letter file no. 89000571 - Chron. no. 126962 – RLOC 04014-1712) over the 70 deg to 550 deg range for various heat up and cool down rates. The stress intensities are dependent on geometry and temperature change rate and are not affected by embrittlement. Thus, the Westinghouse provided values remain valid throughout Plant life.

The membrane (pressure induced) stress can then be determined as a function of the allowable stress intensity reduced by thermal stress intensity and that difference divided by 2 as specified in ASME Section III Appendix G. Several safety factors and conservative assumption are incorporated into the calculation process for determining the remaining allowable pressure stress. The RCS pressure that imposes this Pressure Stress can then be determined at the various temperatures. Note that during heatup the Thermal Stress can be offset by the pressure stress on an internal crack and conversely during cooldown, the thermal stress can offset the pressure stress on an external crack during heatup. The heat up and cooldown curves extract the values that are based on the highest magnitude combined stress at either the 1/4t or 3/4t location.

2.1.2 RCS Pressure Test Limits:

10 CFR 50, Appendix G establishes the pressure and temperature requirements for pre-service hydrostatic test (no fuel) and hydro test and leak tests performed with fuel in the core.

To meet Condition 1.a of 10 CFR Appendix G, Table 1, the limiting temperature for the closure flange is the Unit 1 head flange that has an RT_{NDT} of 53°F. The 20% of pre-service system hydrostatic test pressure is 621 psig. Thus, the minimum RCS temperature for the hydro tests and leak tests with fuel in the vessel and core not critical that do not exceed 621 psig pressure is 53°F. For Condition 1.b, the minimum RCS temperature for the hydro tests and leak tests with fuel in the vessel and core not critical that do exceed 621 psig pressure is 143°F (RT_{NDT} + 90°F). For Condition 1.c, the limiting material is Unit 1 lower shell weld 3-442 C based an ART of 198.3°F. For this pre-service hydro test, with no fuel in the vessel, the minimum RCS temperature for all pressures is 258.3°F (RT_{NDT} + 60°F). The limiting temperature for all these conditions is for Condition 1.c. Thus, the pressure temperature limits for leak testing are imposed starting with a minimum temperature of 260°F.

2.1.3 Reactor Vessel Bolt-up and Criticality Temperature Limits:

Operating restrictions illustrated on the P-T curve also include reactor flange bolt up temperature. This is based on ASME Appendix G and 10 CFR 50 Appendix G that require the bolt-up temperature to be the initial RT_{NDT} of the flange plus any irradiation effects. The flux exposed in the R.V. Flange and R.V. Head Flange result in negligible RT_{NDT} shift, and, thus minimum Bolt Up Temperature does not change with time. The highest flange RT_{NDT} between DCPP Unit 1 and 2 is 53 deg F (Unit 1 R.V. closure head). The curves conservatively set the temperature at 60 deg F based on WCAP 14040-NP-A minimum temperature. Between the minimum bolt up temperature and the minimum LTOP operating temperature (72 deg F), a 2.07 sq. in. opening is relied on for RCS venting. This satisfies Condition 2.a of the 10 CFR Appendix G, Table 1.

To comply with Condition 2.b of 10 CFR Appendix G, Table 1, the pressure temperature limits impose a minimum temperature of 173°F (RT_{NDT} of 53°F + 120°F) at pressures not exceeding the 20% hydro test pressure or 621 psig. These portions of the Figures 2.1-1 and 2.1-2 curves are graphically bounded by the heatup and cooldown curves and are not visible.

When the core is critical, the 10 CFR Appendix G, Table 1 Conditions 2.c and 2.d require that the temperature be at least 40°F greater than the corresponding ASME Appendix G limit. The minimum temperature for criticality is a minimum temperature for the In-service system hydrostatic pressure temperature, which is 2459 psig. The corresponding temperature for a hydrostatic test at 2459 psig is 327.9°F. Thus, the minimum temperature at with the core may be critical is 330°F.

2.2 Low Temperature Overpressure Protection (LTOP) System Setpoints (LCO 3.4.12)

The power-operated relief valves (PORVs) shall each have a lift settings and an arming temperature in accordance with Table 2.2-1.

Plant equipment shall be operated in accordance with the restrictions of Table 2.2-2.

2.2.1 LTOP Enable Setpoints:

The LTOP lift setpoint and arming temperature are based on the methodology established in the Westinghouse WCAP - 14040 - NP - A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996. The lift setpoint is 435 psig based on limiting the maximum RCS pressure overshoot to a value below the Appendix G P/T curve and limiting the minimum RCS undershoot to maintain a nominal operating pressure drop across the number one RCP seal. The arming temperature setpoint is 200°F or RT_{NDT} + 50°F which ever is greater in accordance with ASME Code Case N-514. The RETRAN-02 Mod3 computer code (Ref. 8.6) was used to perform the thermal hydraulic analysis and verify that the LTOP setpoints and temperature restrictions are acceptable as documented in the calculation STA-249 (Ref. 8.11) with input from STA-197 (Ref. 8.7) for Unit 1 and Unit 2 w/Replacement Steam Generators (RSG's).

2.2.2 RCS Pressure Overshoot:

The mass injection and heat injection events are assumed to occur with the RCS in water solid conditions and letdown isolated, so the RCS pressure rapidly increases to the PORV actuation setpoint. The RCS pressure continues increasing even after the PORV setpoint is reached until the PORV has sufficiently opened so that the relief capacity equals the RCS mass increase or volumetric expansion. The magnitude of the RCS pressure overshoot above the PORV setpoint is dependent on the mass injection and heat injection rates, and the associated PORV electronic delay time and valve opening time. The LTOP analysis assumes a conservative PORV lift setpoint, PORV opening time, and also includes appropriate instrumentation delays. Even considering the limiting single failure of one pressurizer PORV to open, there is still a qualified PORV available to adequately relieve the RCS system pressure.

The RCS peak system pressure occurs at the bottom of the reactor vessel requiring that the elevation head be accounted for between this peak location and the RCS wide range pressure transmitters that generate the PORV open signal. In addition, the RHR pump and RCP flow impacts the PORV setpoint by generating a dynamic pressure drop across the reactor vessel which increases the difference between the RCS wide range pressure transmitters and the bottom of the reactor vessel. The magnitude of the total pressure drop determines the limiting RCS pressure at the bottom of the vessel for a given RCS overshoot case. An appropriate range of mass injection and heat injection cases are evaluated to ensure they conservatively bound the dynamic pressure drop effects due to the RCS flow conditions.

The administrative temperature restrictions in Table 2.2-2 are established based on the most limiting RCS overshoot results obtained from the spectrum of mass injection and heat injection cases evaluated at the specified RCS conditions.

2.2.3 LTOP Mass Injection Case:

The LTOP mass injection analysis is based on an inadvertent initiation of the maximum injection flow capability for the applicable Mode of operation into a water solid RCS with letdown isolated. The initial mass injection capability within the LTOP range is established by Tech Spec. 3.4.12 restriction to secure the safety injection (SI) pumps and one ECCS centrifugal charging pump (CCP), isolate all SI Accumulators, and align CCP 3 for LTOP operation prior to entering the LTOP mode of operation. The administrative temperature limit for blocking the SI signal is based on a mass injection case with one ECCS CCP and CCP 3 aligned for LTOP operation injecting through the SI injection flowpath. The administrative temperature limit for operating with a maximum of one charging pump is based on a mass injection case with one ECCS CCP (which bounds operation with CCP 3 aligned for LTOP operation) injecting through the normal and the alternate charging flowpaths. The administrative temperature limits for starting and stopping RCPs are based on limiting the dynamic pressure drop increase on the RCS overshoot for a mass injection case with one CCP injecting through the normal and alternate charging flowpaths. The administrative temperature limit for establishing an RCS vent is based on determining the temperature at which the reduced Appendix G P/T limit no longer has additional margin to accommodate the mass injection RCS overshoot associated with the PORV response time. All mass injection cases account for a conservative RCP seal injection flow into the RCS and the dynamic effects of both RHR pumps running.

2.2.4 LTOP Heat Injection Case:

The heat injection cases are based on starting an RCP in one loop with a maximum allowable measured temperature difference of 50 °F between the RCS and the Steam Generators (SGs). The heat injection cases are evaluated at various RCS temperature conditions which bound the potential volumetric expansion effects of water on the RCS overshoot within the LTOP range. The heat injection RCS overshoot cases were determined to remain below the Appendix G P/T curve and are conservatively bounded by the mass injection overshoot results throughout the LTOP temperature range. The heat injection cases establish that there are no LTOP administrative RCS temperature restrictions for starting an RCP when the measured SG temperature does not exceed the RCS by more than 50 °F. A bounding heat injection case was also evaluated to establish that if the pressurizer level indicates less than or equal to 50%, there are no RCS/SG temperature restrictions for starting an RCP, since even the maximum credible RCS/SG temperature differential will not challenge the Appendix G P/T limit in the LTOP range.

2.2.5 RCS Pressure Undershoot:

Once an LTOP PORV has opened to mitigate the pressure transient due to a mass injection or heat injection case, the RCS pressure continues decreasing even after the close setpoint has been reached and until the PORV has fully closed. The limiting RCS undershoot case is based on the maximum RCS pressure relief capacity associated with both LTOP PORVs opening and closing simultaneously during the least severe mass injection and heat injection overshoot case, respectively. The RCS undershoot evaluation is based on maintaining the RCS pressure above the minimum value which is considered acceptable for the number one RCP seal operating conditions. The PORV lift setpoint in Table 2.2-1 was evaluated to adequately limit the RCS undershoot to an acceptable value for the applicable mass injection and heat injection cases within the LTOP range.

Where there is insufficient range between the upper and lower pressure limits to select a PORV setpoint to provide protection against violation of both limits, setpoint selection to provide protection against the upper pressure limit violation shall take precedence.

2.2.6 Measurement Uncertainties:

The LTOP mass injection and heat injection overshoot analyses incorporate the appropriate measurement uncertainties associated with the RCS wide range pressure transmitters and the RCS wide range RTDs. Since these two measurement processes are independent of each other, they are statistically combined into one equivalent pressure error term with respect to the Appendix G P/T curve that is added onto the calculated peak pressure. This bounding peak pressure is then used to determine the corresponding temperature limit which ensures compliance with the applicable Appendix G P/T curve.

The heat injection case overshoot analysis also incorporates the measurement uncertainty associated with establishing the SG secondary temperature prior to starting an RCP. The RCS and SG measurement uncertainties are then assumed to be in the worst case opposite direction to establish a conservatively bounding RCS/SG temperature difference for the heat injection analysis.

The LTOP mass injection and heat injection undershoot analyses incorporate the appropriate measurement uncertainty for the RCS wide range pressure transmitters associated with both PORVs opening and closing simultaneously. Since each PORV has a normal and independent setpoint uncertainty distribution, they are statistically combined into a value which represents the lowest simultaneous drift setpoint with a 95% probability.

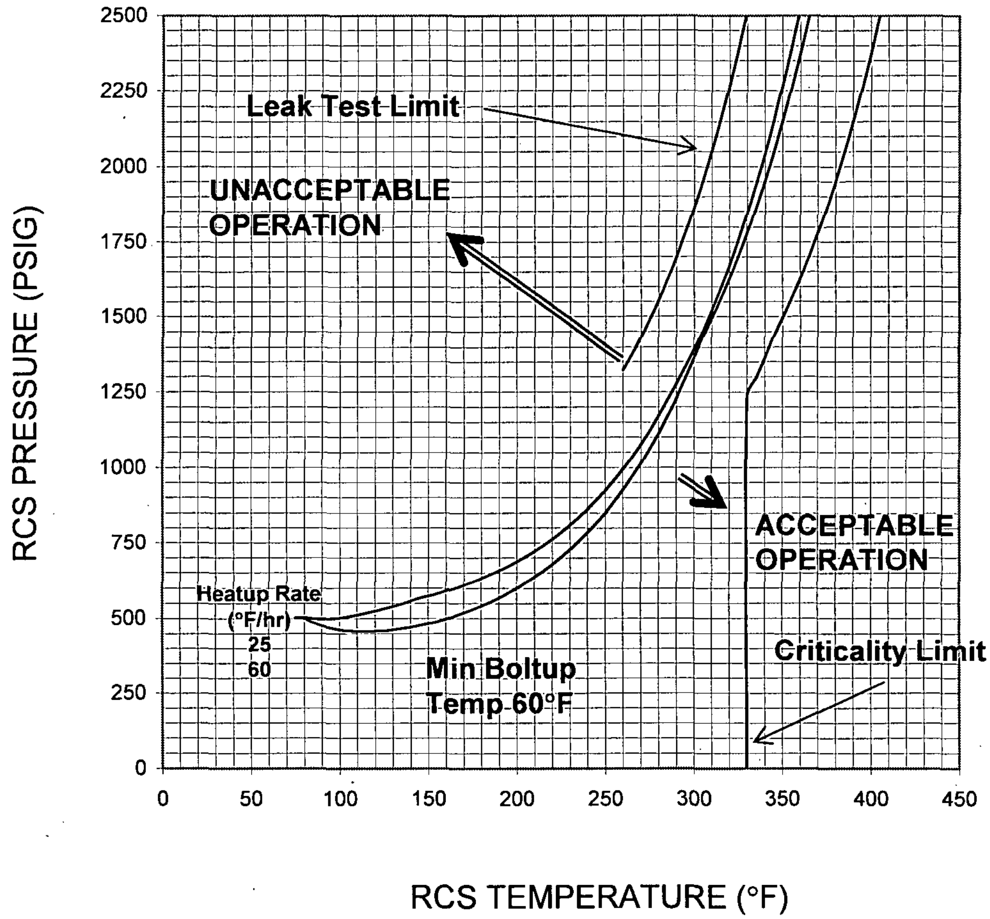

FIGURE 2.1-1: Diablo Canyon Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable to 23 EFPY (Unit 1 and Unit 2) (Without Margins for Instrumentation Errors)

TABLE 2.1-1							
Diablo Canyon Heatup Data at 23 EFPY (Unit 1 and Unit 2) With Margins for Instrumentation Errors							
25°F/hr		60°F/hr		60°F/hr Crit. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
75	469.7	75	469.7				
80	471.0	80	468.9				
85	468.0	85	453.2				
90	466.3	90	435.8				
95	467.9	95	424.3				
100	470.9	100	424.0				
105	474.1	105	424.2				
110	478.0	110	424.5				
115	482.8	115	425.4				
120	488.2	120	426.6				
125	494.4	125	428.5				
130	501.1	130	431.0				
135	508.5	135	434.3				
140	516.5	140	438.2				
145	524.8	145	442.8				
150	533.0	150	448.2				
155	541.7	155	453.4				
160	550.8	160	460.6				
165	559.0	165	468.7				
170	567.8	170	475.9				
175	577.1	175	485.3				
180	587.1	180	496.0				
185	597.8	185	506.8				
190	609.2	190	518.4				
195	621.5	195	531.0				
200	634.6	200	544.5				
205	648.6	205	559.1				
210	663.6	210	574.7				
215	679.6	215	591.6				
220	696.8	220	609.7				
225	715.2	225	629.1				
230	734.9	230	650.0				
235	756.0	235	672.4				

TABLE 2.1-1							
Diablo Canyon Heatup Data at 23 EFPY (Unit 1 and Unit 2) With Margins for Instrumentation Errors							
25° F/hr		60°F/hr		60°F/hr Crit. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
240	778.6	240	696.5				
245	802.9	245	722.3				
250	829.0	250	750.0				
255	856.9	255	779.7				
260	886.8	260	811.6			260	1182.4
265	919.0	265	845.8			265	1224.8
270	953.5	270	882.5			270	1270.3
275	990.5	275	921.9			275	1319.1
280	1030.2	280	964.1			280	1371.4
285	1072.9	285	1009.5			285	1427.6
290	1118.6	290	1058.1	330.0	1058.1	290	1487.8
295	1167.6	295	1110.2	335.0	1110.2	295	1552.4
300	1220.2	300	1166.1	340.0	1166.1	300	1621.6
305	1272.7	305	1226.0	345.0	1226.0	305	1695.9
310	1327.8	310	1290.0	350.0	1290.0	310	1775.4
315	1386.9	315	1358.6	355.0	1358.6	315	1860.7
320	1450.3	320	1432.1	360.0	1432.1	320	1951.9
325	1518.2	325	1492.6	365.0	1492.6	325	2049.6
330	1590.9	330	1556.7	370.0	1556.7	330	2154.1
335	1668.9	335	1624.7	375.0	1624.7	335	2265.9
340	1752.3	340	1697.7	380.0	1697.7	340	2385.2
345	1841.7	345	1776.0	385.0	1776.0	345	2512.7
350	1937.3	350	1859.7	390.0	1859.7	350	2648.6
355	2039.5	355	1949.2	395.0	1949.2	355	2793.4
360	2148.8	360	2044.9	400.0	2044.9	360	2947.4
365	2265.6	365	2147.1	405.0	2147.1		
370	2390.3	370	2256.3	410.0	2256.3		
375	2523.3	375	2372.9	415.0	2372.9		
380	2665.0	380	2497.1	420.0	2497.1		
385	2815.9	385	2629.6	425.0	2629.6		
390	2976.2	390	2770.6	430.0	2770.6		
395	3146.5	395	2920.5	435.0	2920.5		
400	3326.9	400	3079.7	440.0	3079.7		

Ref. Calc. N-291

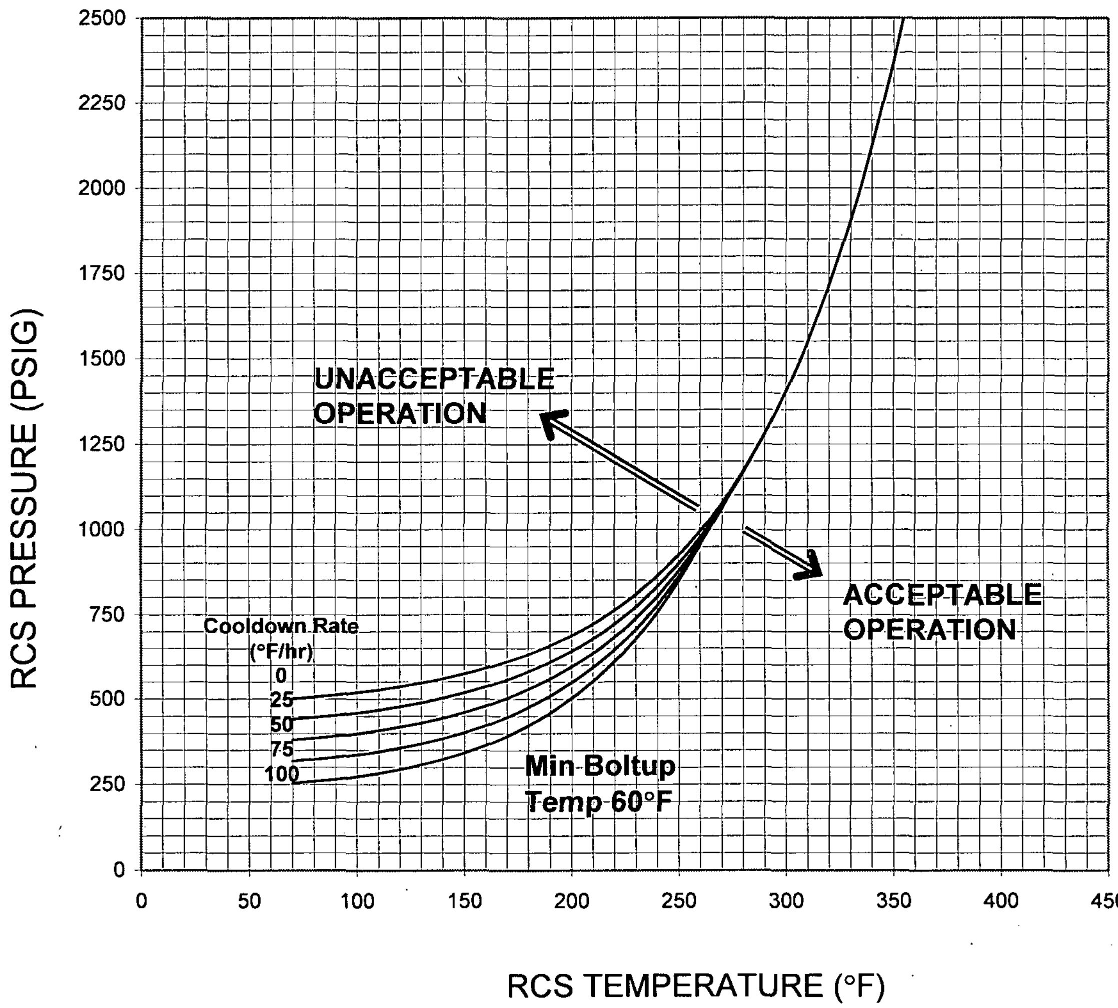

FIGURE 2.1-2: Diablo Canyon Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50, 75 and 100°F/hr) Applicable to 23 EFPY (Unit 1 and Unit 2) (Without Margins for Instrumentation Errors)

TABLE 2.1-2

**Diablo Canyon Cooldown Data at 23 EFPY (Unit 1 and Unit 2)
With Margins for Instrumentation Errors**

Steady State		25°F/hr		50°F/hr		75°F/hr		100°F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
350	2009.6	350	2009.6	350	2009.6	350	2009.6	350	2009.6
345	1904.1	345	1904.1	345	1904.1	345	1904.1	345	1904.1
340	1805.6	340	1805.6	340	1805.6	340	1805.6	340	1805.6
335	1713.5	335	1713.5	335	1713.5	335	1713.5	335	1713.5
330	1627.5	330	1627.5	330	1627.5	330	1627.5	330	1627.5
325	1547.3	325	1547.3	325	1547.3	325	1547.3	325	1547.3
320	1472.5	320	1472.5	320	1472.5	320	1472.5	320	1472.5
315	1402.7	315	1402.7	315	1402.7	315	1402.7	315	1402.7
310	1337.7	310	1337.7	310	1337.7	310	1337.7	310	1337.7
305	1277.0	305	1277.0	305	1277.0	305	1277.0	305	1277.0
300	1220.4	300	1220.4	300	1220.4	300	1220.4	300	1220.4
295	1167.8	295	1165.7	295	1167.8	295	1167.8	295	1167.8
290	1118.6	290	1113.8	290	1115.3	290	1118.6	290	1118.6
285	1072.9	285	1063.5	285	1059.7	285	1062.7	285	1072.9
280	1030.2	280	1016.6	280	1008.0	280	1005.4	280	1010.1
275	990.5	275	972.9	275	959.9	275	952.2	275	951.0
270	953.5	270	932.3	270	915.0	270	902.6	270	896.2
265	919.0	265	894.4	265	873.2	265	856.3	265	844.8
260	886.8	260	859.1	260	834.4	260	813.4	260	797.2
255	856.9	255	826.3	255	798.3	255	773.4	255	752.9
250	829.0	250	795.8	250	764.7	250	736.4	250	711.8
245	802.9	245	767.3	245	733.5	245	701.9	245	673.6
240	778.6	240	740.8	240	704.4	240	669.9	240	638.2
235	756.0	235	716.1	235	677.4	235	640.2	235	605.3
230	734.9	230	693.2	230	652.3	230	612.6	230	574.7
225	715.2	225	671.7	225	628.9	225	586.9	225	546.5
220	696.8	220	651.8	220	607.2	220	563.1	220	520.3
215	679.6	215	633.2	215	586.9	215	541.0	215	495.9
210	663.6	210	615.9	210	568.1	210	520.4	210	473.3
205	648.6	205	599.7	205	550.6	205	501.3	205	452.3
200	634.6	200	584.6	200	534.2	200	483.5	200	432.9
195	621.5	195	570.5	195	519.0	195	467.0	195	414.8
190	609.2	190	557.4	190	504.9	190	451.7	190	398.1
185	597.8	185	545.2	185	491.7	185	437.4	185	382.5

TABLE 2.1-2

Diablo Canyon Cooldown Data at 23 EFPY (Unit 1 and Unit 2)
With Margins for Instrumentation Errors

Steady State		25°F/hr		50°F/hr		75°F/hr		100°F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
180	587.1	180	533.7	180	479.4	180	424.1	180	368.1
175	577.1	175	523.1	175	468.0	175	411.8	175	354.7
170	567.8	170	513.1	170	457.3	170	400.3	170	342.3
165	559.0	165	503.8	165	447.4	165	389.7	165	330.7
160	550.8	160	495.1	160	438.1	160	379.8	160	320.0
155	543.1	155	487.0	155	429.5	155	370.5	155	310.1
150	535.9	150	479.4	150	421.5	150	362.0	150	300.9
145	529.2	145	472.3	145	414.0	145	354.0	145	292.4
140	522.9	140	465.7	140	407.0	140	346.6	140	284.5
135	517.0	135	459.6	135	400.6	135	339.8	135	277.2
130	511.5	130	453.8	130	394.5	130	333.4	130	270.4
125	506.4	125	448.5	125	388.9	125	327.5	125	264.2
120	501.5	120	443.5	120	383.7	120	322.1	120	258.4
115	497.0	115	438.9	115	378.9	115	317.0	115	253.1
110	492.8	110	434.5	110	374.4	110	312.4	110	248.2
105	488.9	105	430.5	105	370.3	105	308.1	105	243.7
100	485.2	100	426.8	100	366.5	100	304.1	100	239.6
95	481.8	95	423.3	95	362.9	95	300.5	95	235.8
90	478.6	90	420.1	90	359.6	90	297.2	90	232.3
85	475.6	85	417.1	85	356.6	85	294.1	85	229.2
80	472.7	80	414.1	80	353.7	80	291.1	80	226.1
75	469.9	75	411.4	75	350.9	75	288.5	75	223.3
70	467.2	70	408.7	70	348.3	70	285.7	70	220.7

Calc. N-291

Table 2.2-1

Low Temperature Over-Pressure (LTOP)

System Setpoints

Function	Setpoint
PORV Arming Temperature(1)	≥ 280 °F
PORV Pressure Setpoint(2)	435 psig

(1) Calc. N-298, Rev. 0. Valid to 23 EFPY

(2) STA-197, Rev. 0

Table 2.2-2

Low Temperature Over-Pressure (LTOP)

Temperature Restrictions

Restriction	Setpoint
	RSGs(1,2)
SI Pumps Secured, CCP 1 or CCP 2 Secured, SI Accumulators Isolated, CCP 3 aligned for LTOP operation	≤ 280 °F
Safety Injection Flowpath Blocked, and SI Blocked	≤ 169 °F
2 of 3 Charging Pumps Secured	≤ 156 °F
1 of 4 RCPs Secured	≤ 148 °F
2 of 4 RCPs Secured	≤ 132 °F
3 of 4 RCPs Secured	≤ 117 °F
4 of 4 RCPs Secured	≤ 108 °F
RCS Vent Path of 2.07 in^2 Established	≤ 90 °F

(1) Calc. STA-197, Rev. 0

(2) Calc. STA-249, Rev. 0

Assumptions: 1) PORV Stroke Time of 2.9 seconds.

2) Apply 10 % per Code Case N-514.

3. ADDITIONAL CONSIDERATIONS

Revisions to the PTLR or its supporting analyses should include the following considerations to ensure that the assumptions are still valid:

3.1 The PORV piping qualification under LTOP conditions is bounded by testing performed in accordance with NUREG 0737.

3.2 At the LTOP setpoints, there is no credible way to challenge RCP number 1 seal operation.

3.3 LTOP heat injection case is bounded by the mass injections case throughout the current range of operation.

4. REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The reactor vessel material surveillance program is in compliance with Appendix H to 10 CFR 50, entitled "Reactor Vessel Material Surveillance Program Requirements" and Section 5.2.4.4 of the Final Safety Analysis Report (FSAR). The withdrawal schedule is presented in FSAR Table 5.2-22.

Diablo Canyon Units 1 & 2 each have their own independent material surveillance program allowing each to have its own unit specific heat up and cooldown curves and LTOP setpoints. Both units are currently operated using the same limitations resulting from the most conservative limitations in either unit.

The programs are described in the following:

4.1 WCAP-8465, PG&E Diablo Canyon Unit 1 Reactor Vessel Surveillance Program, January, 1975.

4.2 WCAP-13440, Supplemental Reactor Vessel Radiation Surveillance Program for PG&E Diablo Canyon Unit 1, December, 1992.

4.3 WCAP-8783, PG&E Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program, December, 1976.

The surveillance capsule reports are as follows:

4.4 WCAP-11567, Analysis of Capsule S from Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, December, 1987.

4.5 WCAP-13750, Analysis of Capsule Y from Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, July, 1993.

4.6 WCAP-15958, Analysis of Capsule V from Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program, January 2003.

4.7 WCAP-11851, Analysis of Capsule U from Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program, May, 1988.

4.8 WCAP-12811, Analysis of Capsule X from Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program, December, 1990.

4.9 WCAP-14363, Analysis of Capsule Y from Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program, August, 1995.

4.10 WCAP-15423, Analysis of Capsule V from Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program, September 2000.

Diablo Canyon Units 1 and 2 also have Reactor Cavity Neutron Measurement Programs described in:

4.11 WCAP-14284, Reactor Cavity Neutron Measurement Program for Diablo Canyon Unit 1 – cycles 1 through 6, January, 1995.

4.12 WCAP-15780, Fast Neutron Fluence and Neutron Dosimetry Evaluations for the Diablo Canyon Unit 1 Reactor Pressure Vessel, December, 2001.

4.13 WCAP-14350, Reactor Cavity Neutron Measurement Program for Diablo Canyon Unit 2 – cycles 1 through 6, November, 1995.

4.14 WCAP-15782, Fast Neutron Fluence and Neutron Dosimetry Evaluations for the Diablo Canyon Unit 2 Reactor Pressure Vessel, December, 2001.

5. REACTOR VESSEL SURVEILLANCE DATA CREDIBILITY

Regulatory Guide 1.99, Revision 2, describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there have been three surveillance capsules removed and analyzed from the Diablo Canyon Unit 1 reactor vessel and four from the Diablo Canyon Unit 2 reactor vessel. They must be shown to be credible in order to use these surveillance data sets. There are five requirements that must be met for the surveillance data to be judged credible in accordance with Regulatory Guide 1.99, Revision 2.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the Diablo Canyon reactor vessel surveillance data.

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," as follows:

> "The reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

The Diablo Canyon pressure and temperature limits are derived using the most limiting locations of both units to create a single set of limiting parameters. The most limiting ¼t location is found in Seam Weld 3-442 C in the Unit 1 reactor vessel while the most limiting ¾t location is found in the Intermediate Shell Plate B5454-2 in the Unit 2 reactor vessel. The Unit 1 Weld Surveillance Capsules are fabricated from a weld manufactured using the same weld wire heat number (Heat 27204).

The Unit 2 Base Metal Surveillance Capsules are made using material from Intermediate Shell Plate B5454-1. This material is the same type of material as the controlling material (B5454-2) and has nearly identical properties (Cu content is identical and Ni content is 0.06% higher than the controlling material). The Diablo Canyon Surveillance Program meets the intent of this criterion.

Criterion 2: Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper shelf energy unambiguously.

The Charpy energy versus temperature curves (irradiated and unirradiated) for the surveillance materials show reasonable scatter and allow determination of the RT_{NDT} at 30 ft-lb and upper shelf energy.

Criterion 3: Where there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

Tables 5.0-1 and 5.0-2 present the Surveillance Capsule Data for Diablo Canyon Units 1 and 2. The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 should be less than 1 σ (standard deviation) of 17°F for base metal and 28°F for weld material.

The Diablo Canyon Unit 1 Surveillance Capsule S for the Intermediate Shell Plate B4106-3 and Surveillance Weld Heat 27204 both show scatter in excess of the Criterion 3 allowable values. The Diablo Canyon limiting CF values are based upon the CF Tables 1 and 2 of 10 CFR 50.61 and the chemistry values provided by CE Report CE NPSD-1039, Rev 2. Should the credibility criteria be met upon future surveillance capsule withdrawal and evaluation, then Reg. Guide 1.99, Rev. 2, Position C.2 shall be utilized.

Criterion 4: The irradiation temperature of the Charpy specimens in the capsule should match the vessel wall temperature at the cladding/base metal interface within +/- 25°F.

The capsule specimens are located in the reactor between the thermal shield (Unit 1) or neutron pads (Unit 2) and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the thermal shield (Unit 1) or neutron pads (Unit 2). The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence this criteria is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the data base for that material.

The surveillance data for the correlation monitor material in the capsules fall within the scatter band for this (Correlation Monitor Material Heavy Section Steel Technology Plate 02) material.

Table 5.0-1
Diablo Canyon Unit 1 Surveillance Capsule Data

Material	Capsule	CF(a)	FF	Best Fit ΔRT_{NDT}(b)	Measured ΔRT_{NDT}(c)	Scatter in ΔRT_{NDT}
Inter Shell Plate B4106-3	S(d)	32.8	0.656	21.52	-1.78	-23.3
Inter Shell Plate B4106-3	Y		1.014	33.26	48.66	15.4
Inter Shell Plate B4106-3	V		1.087	35.65	34.32	-1.33
Surveillance Weld Heat 27204	S(d)	199.7	0.656	131.00	110.79	-20.21
Surveillance Weld Heat 27204	Y		1.014	202.50	232.59	30.09
Surveillance Weld Heat 27204	V		1.087	217.07	201.07	-16.0

WCAP 15958

(a) CF is calculated from surveillance data using Reg. Guide 1.99 Regulatory Position 2.1 (see Table 6.0-3).

(b) Best fit ΔRT_{NDT} = CF * FF.

(c) Calculated using measured Charpy data plotted by EPRI Hyperbolic Tangent Curve Fitting Routine, Revision 2.0.

(d) Diablo Canyon Surveillance Capsule S is currently not judged Credible per Reg. Guide 1.99, Rev 2, Position 2.1.

Table 5.0-2
Diablo Canyon Unit 2 Surveillance Capsule Data

Material	Capsule	CF[(a)]	FF	Best Fit ΔRT_{NDT}[(b)]	Measured ΔRT_{NDT}[(c)]	Scatter in ΔRT_{NDT}
Inter Shell Plate B5454-1 (Long)	U	98.6	0.701	69.1	65.4	-3.7
Inter Shell Plate B5454-1 (Long)	X		0.976	96.2	100.1	3.9
Inter Shell Plate B5454-1 (Long)	Y		1.121	110.5	111.6	1.1
Inter Shell Plate B5454-1 (Long)	V		1.237	122.0	123.4	1.4
Inter Shell Plate B5454-1 (Trans)	U	98.6	0.701	69.1	73.3	4.2
Inter Shell Plate B5454-1 (Trans)	X		0.976	96.2	99.5	3.3
Inter Shell Plate B5454-1 (Trans)	Y		1.121	110.5	111.6	1.1
Inter Shell Plate B5454-1 (Trans)	V		1.237	122.0	112.9	-9.1
Surveillance Weld	U	197.2	0.701	138.2	173.0	34.8
Surveillance Weld	X		0.976	192.5	203.2	10.7
Surveillance Weld	Y		1.121	221.1	211.4	-9.7
Surveillance Weld	V		1.237	243.9	224.5	-19.4

WCAP-15423

(a) CF is calculated from surveillance data using Reg. Guide 1.99 Regulatory Position 2.1 (see Table 6.0-3).

(b) Best fit ΔRT_{NDT} = CF * FF.

(c) Calculated using measured Charpy data plotted by EPRI Hyperbolic Tangent Curve Fitting Routine, Revision 2.0.

6. SUPPLEMENTAL DATA TABLES

Table 6.0-1 Comparison of Diablo Canyon Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Table 6.0-2 Comparison of Diablo Canyon Unit 2 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Table 6.0-3 Calculation of Chemistry Factors Using Surveillance Capsule Data

Table 6.0-4 DCPP-1 Reactor Vessel Beltline Material, Chemistry, and Unirradiated Toughness Data

Table 6.0-5 DCPP-2 Reactor Vessel Beltline Material, Chemistry, and Unirradiated Toughness Data

Table 6.0-6 DCPP-1 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t and ¾t Locations at 23 EFPY

Table 6.0-7 DCPP-2 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t and ¾t Locations at 23 EFPY

Table 6.0-8 Diablo Canyon Unit 1 Adjusted Reference Temperatures (ARTs) for the Reactor Vessel Beltline Materials at the ¼t and ¾t Locations for 23 EFPY

Table 6.0-9 Diablo Canyon Unit 2 Adjusted Reference Temperatures (ARTs) for the Reactor Vessel Beltline Materials at the ¼t and ¾t Locations for 23 EFPY

Table 6.0-10 Calculation of Adjusted Reference Temperature at 23 EFPY (Unit 1 and Unit 2) for the Limiting Diablo Canyon Reactor Vessel Materials

7. PRESSURIZED THERMAL SHOCK (PTS) SCREENING

10 CFR 50.61 requires that RT_{PTS} be determined for each of the vessel beltline materials. The RT_{PTS} is required to meet the PTS screening criterion of 270°F for plates, forgings, and axial weld material, and 300°F for circumferential weld material. If the screening criterion is not met, specific actions taken to either meet the screening criterion or prevent potential reactor vessel failure as a result of PTS require review and approval of the NRC. The maximum projected RT_{PTS} for Units 1 and 2 is 250.9°F (Unit 1 Weld 3-442c), therefore, at a projected 32 EFPY at EOL, the PTS screening criteria is met. The PTS evaluations are described in the following reports:

7.1 WCAP-13771, Evaluation of Pressurized Thermal Shock for Diablo Canyon Unit 1, July, 1993.

7.2 WCAP-14364, Evaluation of Pressurized Thermal Shock for the Diablo Canyon Unit 2 Reactor Vessel, August, 1995.

7.3 PG&E Calculation N-287 (Unit 1)

7.4 PG&E Calculation N-272 (Unit 2)

8. REFERENCES

8.1 Technical Specification 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)"

8.2 License Amendment No. 135 (U1)/135 (U2), dated May 28, 1999

8.3 License Amendment No. 133 (U1)/131 (U2), dated May 3, 1999

8.4 WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Revision 2," January 1996

8.5 PG&E letter DCL-00-070, Supplement to Reactor Coolant System Pressure and Temperature Limits Report

8.6 "RETRAN-02 A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems", EPRI NP-1850-CCM-A, Project 889-3, December, 1996

8.7 PG&E Calculation STA-197 Rev. 0, "LTOP Temperature Limits for 23 EFPY"

8.8 PG&E Calculation N-288, Rev. 0, "Adjusted RT-NDT Versus EFPY"

8.9 PG&E Calculation N-291, Rev. 1, "Pressure-Temperature Limits for Heatup & Cooldown"

8.10 PG&E Calculation N-298, Rev. 0. "LTOP Enable Temperature for 23 EFPY"

8.11 PG&E Calculation STA-249 Rev. 0, "RSG – LTOP Analysis"

Table 6.0-1
Comparison of Diablo Canyon Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Materials	Capsule	Fluence [d] (X 10^{19} n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%) [c]
Plate B4106-3	S	0.284	36.2	-1.78	14	0
	Y	1.05	56.0	48.66	19	6.8
	V	1.37	60.0	34.32	20	0
Surveillance Weld Metal	S	0.284	145.8	110.79	25.5	11
	Y	1.05	225.4	232.59	34.5	34.1
	V	1.37	241.6	201.07	36.5	27.5
Heat Affected Zone Metal	S	0.284	--	72.31	--	8.1
	Y	1.05	--	79.77	--	19.9
	V	1.37	--	110.90	--	14.7
Correlation Monitor Plate HSST 02	S	0.284	73.01	65.62	--	2.4
	Y	1.05	112.9	115.79	--	8.9
	V	1.37	121.0	116.61	--	4.9

WCAP-15958

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1.

(c) Values are based on the definition of upper shelf energy given in ASTM E185-82.

(d) The fluence values given here are the calculated fluence values, not the best estimate.

Table 6.0-2
Comparison of Diablo Canyon Unit 2 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decreases with Regulatory Guide 1.99, Revision 2, Predictions

Materials	Capsule	Fluence [c] (X 10^{19} n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%) [b]
Plate B5454-1 (Longitudinal)	U	0.338	71.0	65.4	18	11
	X	0.919	98.9	100.1	22	20
	Y	1.55	113.6	111.6	25	18
	V	2.41	125.3	123.4	28	24
Plate B5454-1 (Transverse)	U	0.338	71.0	73.3	18	0
	X	0.919	98.9	99.5	22	12
	Y	1.55	113.6	111.6	25	7
	V	2.41	125.3	112.9	28	6
Surveillance Weld Metal	U	0.338	148.1	173.0	28	31
	X	0.919	206.1	203.2	35	38
	Y	1.55	236.8	211.4	40	40
	V	2.41	261.3	224.5	44	40
Heat Affected Zone Metal	U	0.338	--	234.4	--	41
	X	0.919	--	253.5	--	31
	Y	1.55	--	257.7	--	40
	V	2.41	--	291.5	--	52

WCAP-15423

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using the mean weight percent values of copper and nickel of the surveillance material.

(b) Calculated using measured Charpy data plotted using CVGRAPH, Version 4.1.

(c) The fluence values presented here are calculated fluence values, not the best estimate.

Table 6.0-3
Calculation of Chemistry Factors Using Surveillance Capsule Data

Unit 1 - Material	Capsule	F[(a)]	FF[(b)]	Measured ΔRT_{NDT}[(d)] °F	FF×ΔRT_{NDT}°F	FF^2
Intermediate Shell Plate B4106-3	S[(c)]	0.284	0.656	-1.78	0	0.430
	Y	1.050	1.014	48.66	49.34	1.028
	V	1.37	1.087	34.32	37.31	1.182
				SUM	86.65	2.64
	CF Plate = $\Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (86.65°F) \div (2.64) = 32.8°F$ [(c)]					
Weld Metal	S[(c)]	0.284	0.656	110.79	72.68	0.430
	Y	1.050	1.014	232.59	235.85	1.028
	V	1.37	1.087	201.07	218.56	1.182
				SUM	527.09	2.64
	CF weld = $\Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (527.09) \div (2.64) = 199.7°F$ [(c)]					
Unit 2 - Material	**Capsule**	**F[(a)]**	**FF[(b)]**	**Measured ΔRT_{NDT}[(d)] °F**	**FF×ΔRT_{NDT}°F**	**FF^2**
Intermediate Shell Plate B5454-1 (Long)	U	0.338	0.701	65.39	45.84	0.491
	X	0.919	0.976	100.06	97.67	0.953
	Y	1.55	1.121	111.58	125.08	1.257
	V	2.41	1.237	123.43	152.68	1.530
Intermediate Shell Plate B5454-1 (Transverse)	U	0.338	0.701	73.30	51.38	0.491
	X	0.919	0.976	99.52	97.13	0.953
	Y	1.55	1.121	111.59	125.09	1.257
	V	2.41	1.237	112.90	139.66	1.530
				SUM	834.53	8.462
	CF Plate = $\Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (834.53°F) \div (8.462) = 98.6°F$					
Weld Metal	U	0.338	0.701	172.99	121.27	0.491
	X	0.919	0.976	203.23	198.35	0.953
	Y	1.55	1.121	211.39	236.97	1.257
	V	2.41	1.237	224.47	277.67	1.530
				SUM	834.26	4.231
	CF Weld = $\Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (834.26°F) \div (4.231) = 197.2°F$					

WCAP-15958 (Unit 1) WCAP-15423 (Unit 2)

(a) F = Calculated Fluence (10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = Fluence Factor = $F^{(0.28 - 0.1 * \log F)}$

(c) Unit 1 Capsule S is not currently judged "credible" per RG 1.99, Rev 2. All other capsules are "credible" per RG 1.99, Position C.2.

(d) Calculated using Charpy data plotted by EPRI Hyperbolic Tangent Curve Fitting Routine, Revision 2.0.

TABLE 6.0-4
DCPP-1 Reactor Vessel Beltline Material, Chemistry, and Unirradiated Toughness Data

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT} (°F)
Upper Shell Plate [b]			
B4105-1	0.12	0.56	28
B4105-2	0.12	0.57	9
B4105-3	0.14	0.56	14
Inter Shell Plate			
B4106-1	0.125	0.53	-10
B4106-2	0.12	0.50	-3
B4106-3	0.086	0.476	30
Lower Shell Plate			
B4107-1	0.13	0.56	15
B4107-2	0.12	0.56	20
B4107-3	0.12	0.52	-22
Upper Shell Long [b] Welds 1-442 A,B,C	0.19	0.97	-20
Upper Shell to Inter Shell Weld 8-442[b]	0.25	0.73	-56
Inter Shell Long Welds 2-442 A,B,C	0.203[a]	1.018[a]	-56
Inter Shell to Lower Shell Weld 9-442	0.183[a]	0.704[a]	-56
Lower Shell Long Welds 3-442 A,B,C	0.203[a]	1.018[a]	-56

Calc N-NCM-97009

(a) Per CE NPSD-1039, Rev 2

(b) Upper shell materials are included for completeness since EOL exposure is expected to exceed 1.0E + 17.

TABLE 6.0-5
DCPP-2 Reactor Vessel Beltline Material, and Chemistry, and Unirradiated Toughness Data

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT} (°F)
Upper Shell Plate [b]			
B5453-1	0.11	0.60	28
B5453-3	0.11	0.60	5
B5011-1R	0.11	0.65	0
Inter Shell Plate			
B5454-1	0.14	0.65	52
B5454-2	0.14	0.59	67
B5454-3	0.15	0.62	33
Lower Shell Plate			
B5455-1	0.14	0.56	-15
B5455-2	0.14	0.56	0
B5455-3	0.10	0.62	15
Upper Shell Long[b] Welds 1-201 A,B,C	0.22	0.87	-50
Upper Shell to Inter Shell Weld 8-201[b]	0.183[a]	0.704[a]	-56
Inter Shell Long Welds 2-201 A,B,C	0.22	0.87	-50
Inter Shell to Lower Shell Weld 9-201	0.046[a]	0.082[a]	-56
Lower Shell Long Welds 3-201 A,B,C	0.258[a]	0.165[a]	-56

Calc N-NCM-97009

(a) Per CE NSPD-1039, Rev. 2

(b) Upper shell materials are included for completeness since EOL exposure is expected to exceed 1.0E + 17.

TABLE 6.0-6
DCPP-1 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t, and ¾t Locations at 23 EFPY

Material	Fluence f_s	Fluence $f_{c/bm}$	Fluence $f_{1/4t}$	Fluence $f_{3/4t}$
Upper Shell Plate[(a)]				
B4105-1	2.01 E + 17	1.96 E + 17	1.14 E + 17	4.04 E + 16
B4105-2	2.01 E + 17	1.96 E + 17	1.14 E + 17	4.04 E + 16
B4105-3	2.01 E + 17	1.96 E + 17	1.14 E + 17	4.04 E + 16
Inter Shell Plate				
B4106-1	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18
B4106-2	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18
B4106-3	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18
Lower Shell Plate				
B4107-1	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18
B4107-2	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18
B4107-3	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18
Upper Shell Long[(a)] Welds 1-442 A,B,C	2.01 E + 17	1.96 E + 17	1.14 E + 17	4.04 E + 16
Upper Shell to Inter Shell Weld 8-442[(a)]	2.01 E + 17	1.96 E + 17	1.14 E + 17	4.04 E + 16
Inter Shell Long				
Welds 2-442 A,B	7.32 E + 18	7.12 E + 18	4.14 E + 18	1.47 E + 18
Weld 2-442 C	3.66 E + 18	3.56 E + 18	2.07 E + 18	7.36 E + 17
Inter Shell to Lower Shell Weld 9-442	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18
Lower Shell Long				
Welds 3-442 A,B	6.01 E + 18	5.84 E + 18	3.40 E + 18	1.21 E + 18
Weld 3-442 C	9.71 E + 18	9.44 E + 18	5.49 E + 18	1.95 E + 18

Calc N-288, WCAP-15958

(a) Upper shell materials are included for completeness since EOL exposure is expected to exceed 1.0E + 17.

TABLE 6.0-7
DCPP-2 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t and ¾t Locations at 23 EFPY

Material	Fluence f_s	Fluence $f_{c/bm}$	Fluence $f_{1/4t}$	Fluence $f_{3/4t}$
Upper Shell Plate[a]				
B5453-1	2.25 E + 17	2.19 E + 17	1.27 E + 17	4.52 E + 16
B5453-3	2.25 E + 17	2.19 E + 17	1.27 E + 17	4.52 E + 16
B5011-1R	2.25 E + 17	2.19 E + 17	1.27 E + 17	4.52 E + 16
Inter Shell Plate				
B5454-1	1.09 E + 19	1.06 E + 19	6.15 E + 18	2.19 E + 18
B5454-2	1.09 E + 19	1.06 E + 19	6.15 E + 18	2.19 E + 18
B5454-3	1.09 E + 19	1.06 E + 19	6.15 E + 18	2.19 E + 18
Lower Shell Plate				
B5455-1	1.09 E + 19	1.06 E + 19	6.15 E + 18	2.19 E + 18
B5455-2	1.09 E + 19	1.06 E + 19	6.15 E + 18	2.19 E + 18
B5455-3	1.09 E + 19	1.06 E + 19	6.15 E + 18	2.19 E + 18
Upper Shell Long[a] Welds 1-201 A,B,C	2.25 E + 17	2.19 E + 17	1.27 E + 17	4.52 E + 16
Upper Shell to Inter Shell Weld 8-201[a]	2.25 E + 17	2.19 E + 17	1.27 E + 17	4.52 E + 16
Inter Shell Long				
Weld 2-201 A	5.98 E + 18	5.81 E + 18	3.38 E + 18	1.20 E + 18
Weld 2-201 B	5.62 E + 18	5.47 E + 18	3.18 E + 18	1.13 E + 18
Weld 2-201 C	4.72 E + 18	4.59 E + 18	2.67 E + 18	9.48 E + 17
Inter Shell to Lower Shell Weld 9-201	1.09 E + 19	1.06 E + 19	6.15 E + 18	2.19 E + 18
Lower Shell Long				
Weld 3-201 A	4.72 E + 18	4.59 E + 18	2.67 E + 18	9.48 E + 17
Weld 3-201 B	5.98 E + 18	5.81 E + 18	3.38 E + 18	1.20 E + 18
Weld 3-201 C	5.62 E + 18	5.47 E + 18	3.18 E + 18	1.13 E + 18

Calc N-288, WCAP-15423 Rev. 0.

(a) Upper shell materials are included for completeness since EOL exposure is expected to exceed 1.0E + 17.

TABLE 6.0-8
Diablo Canyon Unit 1 Adjusted Reference Temperatures (ARTs) for the Reactor Vessel Beltline Materials at the ¼t and ¾t Locations for 23 EFPY

Material	23 EFPY ART[a]		
	RG 1.99 Rev. 2 Method	¼t (°F)	¾t (°F)
Upper Shell Plate[d]			
B4105-1	Position 1.1	73.2	67.0
B4105-2	Position 1.1	54.2	48.0
B4105-3	Position 1.1	61.7	54.1
Inter Shell Plate			
B4106-1	Position 1.1	95.0	72.1
B4106-2	Position 1.1	98.4	76.6
B4106-3	Position 1.1	124.0	107.2
Lower Shell Plate			
B4107-1	Position 1.1	123.7	99.6
B4107-2	Position 1.1	122.4	100.3
B4107-3	Position 1.1	79.7	57.9
Upper Shell Long[d] Welds 1-442 A,B,C	Position 1.1	31.6	4.6
Upper Shell to Inter[d] Shell Weld 8-442	Position 1.1	9.0	-8.9
Inter Shell Long			
Welds 2-442 A,B	Position 1.1	180.8	122.5
Weld 2-442 C	Position 1.1	140.6	90.8
Inter Shell to Lower Shell Weld 9-442	Position 1.1	152.8	106.5
Lower Shell Long			
Welds 3-442 A,B	Position 1.1	168.9	112.9
Weld 3-442 C[c]	Position 1.1	198.3[b]	137.3

Calc N-288 & WCAP-15958

(a) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

(b) This limiting ART value is bounded by that used to generate heatup and cooldown curves (198.3°F).

(c) DCPP-1 Surveillance Capsule S was not judged "credible" per 10 CFR 50.61. The higher chemistry values of CE NPSD-1039, Rev 2 for this heat are used to generate the heatup and cooldown Appendix G curves.

(d) Upper shell materials are included for completeness since EOL exposure is expected to exceed 1.0E + 17.

TABLE 6.0-9
Diablo Canyon Unit 2 Adjusted Reference Temperatures (ARTs) for the Reactor Vessel Beltline Materials at the ¼t and ¾t Locations for 23 EFPY

Material	23 EFPY ART[a]		
	RG 1.99 Rev. 2 Method	¼t (°F)	¾t (°F)
Upper Shell Plate[c]			
B5453-1	Position 1.1	47.1	37.2
B5453-3	Position 1.1	49.9	43.9
B5011-1R	Position 1.1	45.0	39.0
Inter Shell Plate			
B5454-1	Position 2.1	154.2	127.3
B5454-2	Position 1.1	187.1	159.8[b]
B5454-3	Position 1.1	162.5	132.3
Lower Shell Plate			
B5455-1	Position 1.1	103.8	77.0
B5455-2	Position 1.1	118.8	92.0
B5455-3	Position 1.1	105.3	87.5
Upper Shell Long[c] Welds 1-201 A,B,C	Position 1.1	4.5	-23.7
Upper Shell to Inter[c] Shell Weld 8-201	Position 1.1	5.8	-10.0
Inter Shell Long			
Weld 2-201 A	Position 1.1	154.2	102.0
Weld 2-201 B	Position 1.1	150.8	99.3
Weld 2-201 C	Position 1.1	141.3	91.8
Inter Shell to Lower Shell Weld 9-201	Position 1.1	8.4	-2.6
Lower Shell Long			
Weld 3-201 A	Position 1.1	92.3	59.0
Weld 3-201 B	Position 1.1	100.2	68.3
Weld 3-201 C	Position 1.1	98.1	66.6

Calc N-288 & WCAP-15423

(a) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

(b) This limiting ART value is bounded by that used to generate heatup and cooldown curves (159.8°F).

(c) Upper shell materials are included for completeness since EOL exposure is expected to exceed 1.0E + 17.

TABLE 6.0-10
Calculation of Adjusted Reference Temperature at 23 EFPY (Unit 1 and Unit 2) for the Limiting Diablo Canyon Reactor Vessel Materials

Parameter	ART Value	
Location	**¼t[d]**	**¾t[e]**
Chemistry Factor, CF (°F)	226.8[f]	99.6
Fluence ÷ 10^{19} n/cm^2 (E > 1.0 MeV), f[a]	0.549	0.219
Fluence Factor, FF[b]	0.8323	0.5908
ΔRT_{NDT} = CF x FF, (°F)	188.8[f]	58.8
Initial RT_{NDT}, I (°F)	-56	67
Margin, M (°F)[c]	65.5	34
ART = I + (CF x FF) + M (°F) per Regulatory Guide 1.99, Rev. 2	198.3[f]	159.8[f]

Calc N-288

(a) Fluence, f, is based upon $f_{¼t}$ and $f_{¾t}$ from Tables 6.0-6 and 6.0-7. The Diablo Canyon reactor vessel wall thickness is 8.625 inches at the beltline region.

(b) Fluence Factor (FF) per Regulatory Guide 1.99, Revision 2, is defined as FF = $f^{(0.28 - 0.10 \log f)}$.

(c) Margin is calculated as $M = 2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$. The standard deviation for the initial RT_{NDT} margin term σ_I, is 0°F for plate since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT} term σ_Δ, is 17°F for the plate, except that σ_Δ need not exceed the 0.5 times the mean value of ΔRT_{NDT}.

(d) DCPP-1 lower shell longitudinal weld 3-442 C is limiting for the heatup and cooldown Appendix G curves at ¼t.

(e) DCPP-2 intermediate shell plate B5454-2 is limiting for the heatup and cooldown Appendix G curves at ¾t.

(f) DCPP-1 Surveillance Capsule S was not judged "credible" per 10 CFR 50.61. The higher chemistry value of CE NPSD-1039, Rev 2 for this heat are used to generate the heatup and cooldown Appendix G curves. The calculated ART's are used to generate the heatup and cooldown curves (198.3°F for 1/4t and 159.8°F for 3/4t).