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PG&E Letter DCL-16-117

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Reactor Coolant System Pressure and Temperature Limits Report for Units 1 and 2

Dear Commissioners and Staff:

In accordance with Diablo Canyon Power Plant Technical Specification 5.6.6.c, Pacific Gas and Electric Company (PG&E) is submitting the enclosed Revision 15 of the Pressure and Temperature Limits Report (PTLR) for Units 1 and 2, dated August 15, 2016.

The PTLR was revised to reflect that the limiting adjusted reference temperature values, reactor coolant system heatup and cooldown curves, and low temperature overpressure protection setpoint and arming temperature are valid for an increased vessel burnup limit of 27.85 effective full-power years (EFPY) instead of the 27 EFPY limit given in PTLR Revision 14.

PG&E makes no new or revised regulatory commitments in this submittal (as defined by NEI 99-04).

If there are any questions regarding the PTLR, please contact Ms. Candice Chou at (805) 545-6164.

Sincerely,

Hossein G. Hamzehee

jmsp/4927/50875871
Enclosure

cc:	Diablo Distribution
cc/enc:	Kriss M. Kennedy, NRC Region IV Administrator
	Christopher W. Newport, NRC Senior Resident Inspector
	Balwant K. Singal, NRC Senior Project Manager

Enclosure
PG&E Letter DCL-16-117

DIABLO CANYON POWER PLANT
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)-1
REVISION 15
EFFECTIVE DATE: August 15, 2016

TITLE: PTLR for Diablo Canyon

08/15/16
EFFECTIVE DATE

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 27.85 EFPY on Unit 1 and Unit 2. This increase in the applicable burnup limit is based on N-288 Rev 4 which has incorporated new surveillance data from "sister plants" that is applicable to DCPP and demonstrates that the current DCPP limiting ART values established for 27 EFPY in N-288 Rev 3 are valid to a vessel burnup of ≤ 27.85 EFPY.

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.

As documented in the Reference 8.12 evaluation, the RCS pressure and temperature conditions implemented during the Vacuum Refill process per procedure OP A-2:IX (Ref. 8.11) remain bounded by the RCS P/T limits as shown in Figure 2.1-1 and Figure 2.1-2, and the LTOP P/T limits established in Section 2. The RCS Vacuum Refill restricts RCS pressure criteria to values above 0 psia to ensure RHR system operability.

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 assumptions 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. 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 35°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 35°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 125°F (RT_{NDT} + 90°F). For Condition 1.c, the limiting material is Unit 1 lower shell weld 3-442 C based on an ART of 207.8°F. For this pre-service hydro test, with no fuel in the vessel, the minimum RCS temperature for all pressures is 267.8°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 270°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 35 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 (96 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 155°F (RT_{NDT} of 35°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 equal to the minimum temperature for the in-service system hydrostatic pressure of 2459 psig, which is 337.3°F. Thus, the minimum temperature at which the core may be critical is 340°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 setting and an arming temperature in accordance with Table 2.2-1.

Operation of plant equipment shall comply with the temperature 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 whichever 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 to ensure that the LTOP setpoints and temperature restrictions are acceptable as documented in the calculation STA-249 (Ref. 8.10) 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. Per Note 2 on Table 2.2-2, an administrative exception has been established for the RCS vent temperature restriction when performing the RCS vacuum refill per procedure OP A-2:IX. Calculation STA-298 documents that when the RCS level is maintained at an elevation of less than 123', there is more than adequate time for operators to take action and preclude any credible water solid challenge to the LTOP system.

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.

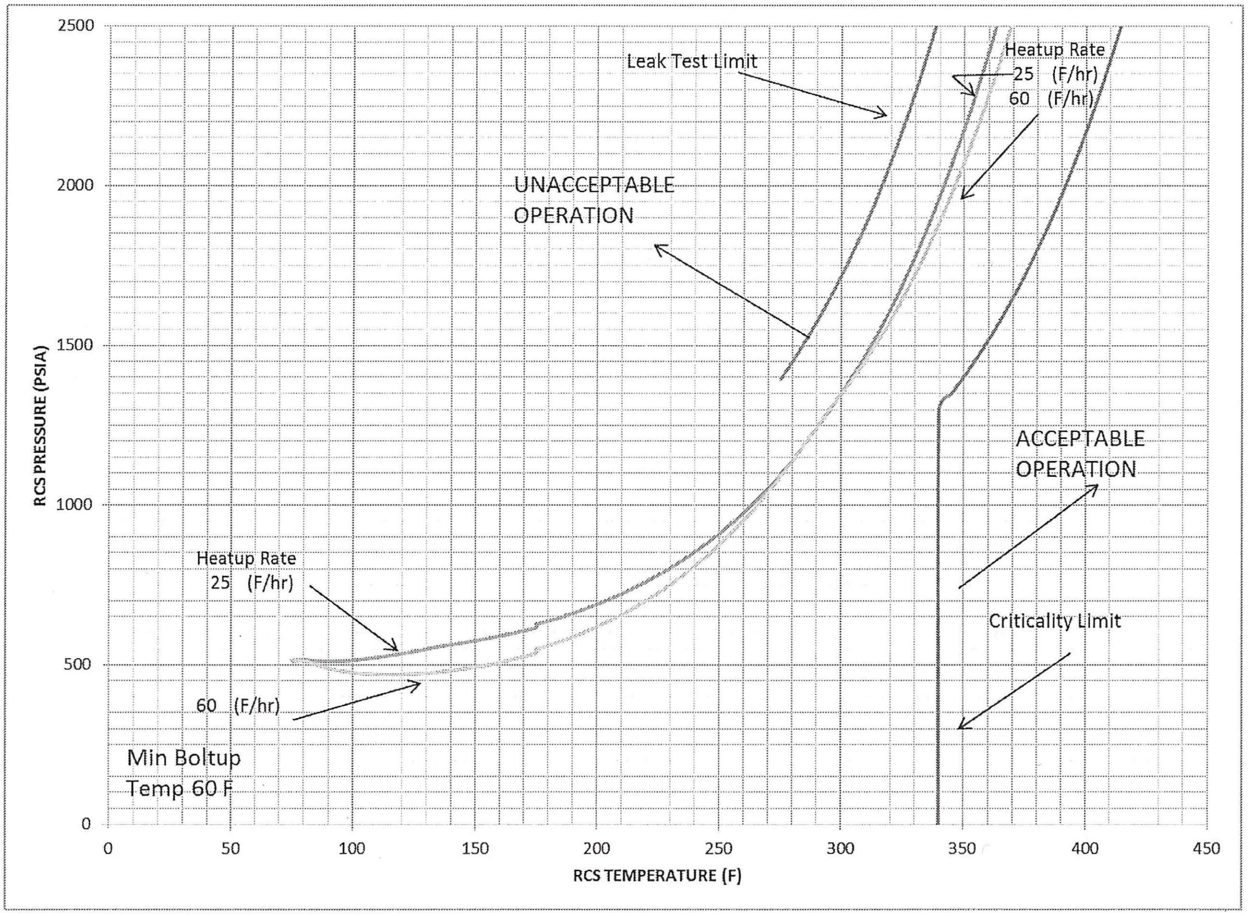

FIGURE 2.1-1: Diablo Canyon Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable to 27.85 EFPY (Unit 1 and Unit 2) (Without Margins for Instrumentation Errors)

TABLE 2.1-1							
Diablo Canyon Heatup Data at 27.85 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	467.0	75	465.4				
80	469.3	80	466.1				
85	465.8	85	452.5				
90	463.9	90	435.9				
95	465.6	95	421.5				
100	468.0	100	413.7				
105	471.6	105	416.6				
110	475.9	110	418.9				
115	481.1	115	421.1				
120	486.9	120	423.0				
125	493.3	125	425.1				
130	500.3	130	427.4				
135	507.9	135	430.2				
140	514.5	140	433.6				
145	521.1	145	437.5				
150	527.5	150	442.0				
155	534.2	155	446.4				
160	541.4	160	452.6				
165	549.1	165	459.6				
170	557.4	170	466.6				
175	566.3	175	473.6				
180	575.8	180	482.8				
185	586.0	185	492.9				
190	596.9	190	503.3				
195	608.6	195	514.4				
200	621.1	200	526.3				
205	634.6	205	539.2				
210	648.9	210	553.3				
215	664.3	215	568.2				
220	680.8	220	584.4				
225	698.4	225	601.6				
230	717.4	230	620.3				
235	737.7	235	640.3				
240	759.4	240	661.6				

TABLE 2.1-1							
Diablo Canyon Heatup Data at 27.85 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)
245	782.7	245	684.8				
250	807.6	250	709.6				
255	834.4	255	736.3				
260	863.1	260	765.0				
265	893.9	265	795.6				
270	926.9	270	828.1				
275	962.3	275	863.5				
280	1000.3	280	901.4				
285	1041.0	285	942.2			285	1386.7
290	1084.7	290	985.8			290	1444.2
295	1131.6	295	1032.6			295	1505.9
300	1181.7	300	1082.7			300	1571.9
305	1234.6	305	1134.2			305	1642.8
310	1287.8	310	1184.2			310	1718.7
315	1344.1	315	1237.9	355	1339.6	315	1800.1
320	1404.6	320	1292.9	360	1392.8	320	1887.2
325	1469.4	325	1342.3	365	1449.8	325	1980.4
330	1538.8	330	1395.3	370	1510.4	330	2080.2
335	1613.1	335	1452.2	375	1575.5	335	2187.0
340	1692.8	340	1512.6	380	1645.3	340	2301.1
345	1778.0	345	1577.6	385	1719.9	345	2422.9
350	1869.2	350	1647.2	390	1799.8	350	2552.9
355	1966.8	355	1721.7	395	1885.1	355	2691.5
360	2071.3	360	1801.4	400	1976.4	360	2839.1
365	2182.7	365	1886.7	405	2074.0	365	2996.2
370	2301.9	370	1977.9	410	2178.3	370	3163.0
375	2429.1	375	2075.4	415	2289.6	375	3339.9
380	2564.7	380	2179.6	420	2408.3	380	3527.2
385	2709.0	385	2290.8	425	2534.9	385	3725.2
390	2862.7	390	2409.4	430	2669.7	390	3933.9
395	3025.8	395	2536.0	435	2813.2	395	4153.4
400	3199.2	400	2670.7	440	2965.8	400	4383.6

Ref. Calc. N-291

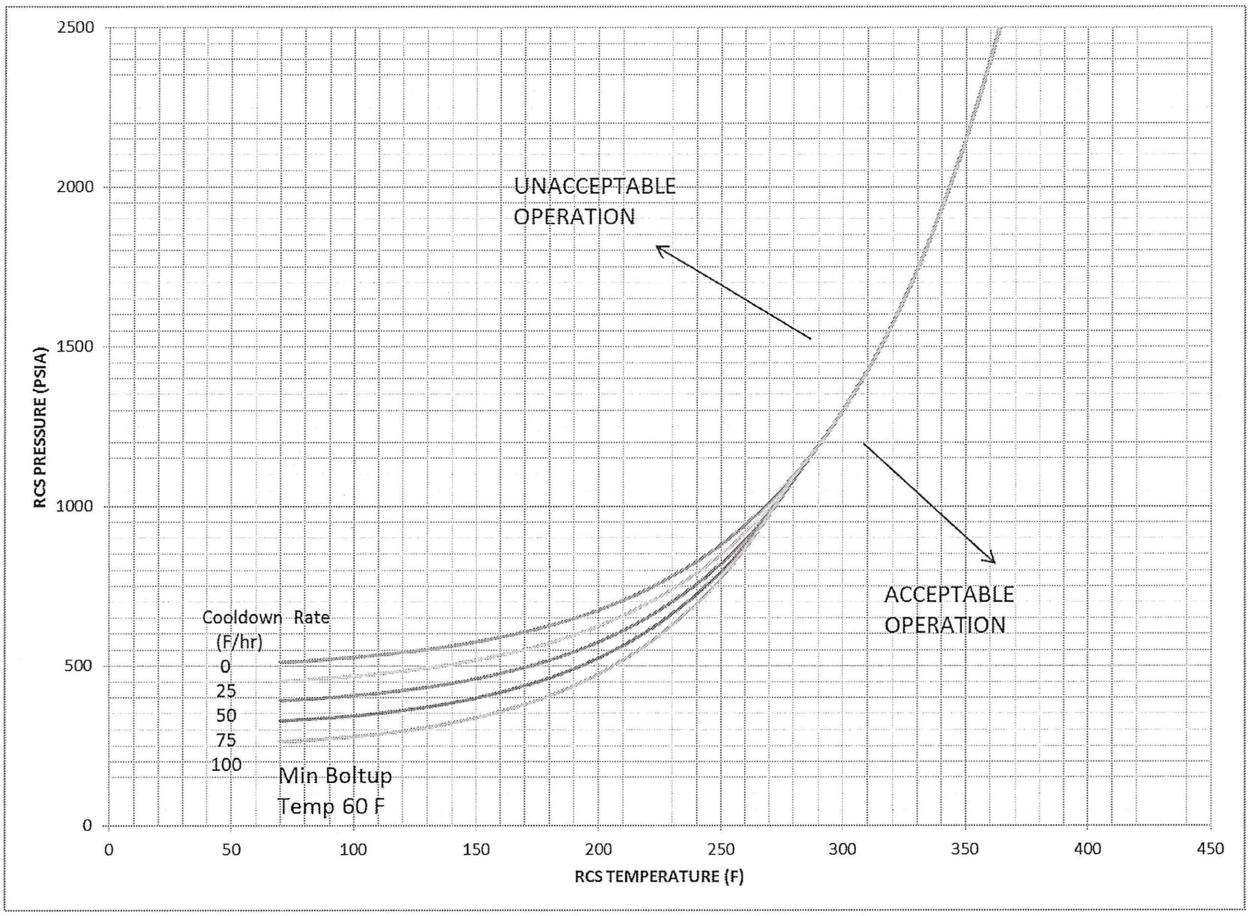

FIGURE 2.1-2: Diablo Canyon Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50, 75 and 100°F/hr) Applicable to 27.85 EFPY (Unit 1 and Unit 2) (Without Margins for Instrumentation Errors)

TABLE 2.1-2									
Diablo Canyon Cooldown Data at 27.85 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)
390	3029.6	390	3029.6	390	3029.6	390	3029.6	390	3029.6
385	2860.8	385	2860.8	385	2860.8	385	2860.8	385	2860.8
380	2701.9	380	2701.9	380	2701.9	380	2701.9	380	2701.9
375	2552.5	375	2552.5	375	2552.5	375	2552.5	375	2552.5
370	2412.4	370	2412.4	370	2412.4	370	2412.4	370	2412.4
365	2281.0	365	2281.0	365	2281.0	365	2281.0	365	2281.0
360	2157.9	360	2157.9	360	2157.9	360	2157.9	360	2157.9
355	2042.7	355	2042.7	355	2042.7	355	2042.7	355	2042.7
350	1935.0	350	1935.0	350	1935.0	350	1935.0	350	1935.0
345	1834.3	345	1834.3	345	1834.3	345	1834.3	345	1834.3
340	1740.2	340	1740.2	340	1740.2	340	1740.2	340	1740.2
335	1652.4	335	1652.4	335	1652.4	335	1652.4	335	1652.4
330	1570.3	330	1570.3	330	1570.3	330	1570.3	330	1570.3
325	1493.8	325	1493.8	325	1493.8	325	1493.8	325	1493.8
320	1422.4	320	1422.4	320	1422.4	320	1422.4	320	1422.4
315	1355.8	315	1355.8	315	1355.8	315	1355.8	315	1355.8
310	1293.7	310	1293.7	310	1293.7	310	1293.7	310	1293.7
305	1235.9	305	1235.9	305	1235.9	305	1235.9	305	1235.9
300	1181.9	300	1181.9	300	1181.9	300	1181.9	300	1181.9
295	1131.6	295	1128.1	295	1129.7	295	1131.6	295	1131.6
290	1084.7	290	1075.3	290	1074.6	290	1077.8	290	1084.7
285	1041.0	285	1028.3	285	1021.7	285	1020.9	285	1026.9
280	1000.3	280	983.7	280	972.2	280	966.3	280	967.3
275	962.3	275	942.1	275	926.1	275	915.2	275	910.7
270	926.9	270	903.2	270	883.2	270	867.7	270	857.9
265	893.9	265	867.0	265	843.3	265	823.5	265	808.9
260	863.1	260	833.3	260	806.2	260	782.5	260	763.4
255	834.4	255	801.9	255	771.6	255	744.3	255	721.1
250	807.6	250	772.6	250	739.5	250	708.8	250	681.8
245	782.7	245	745.3	245	709.6	245	675.9	245	645.3
240	759.4	240	719.9	240	681.8	240	645.3	240	611.4
235	737.7	235	696.3	235	655.9	235	616.8	235	579.9
230	717.4	230	674.2	230	631.8	230	590.3	230	550.7

TABLE 2.1-2									
Diablo Canyon Cooldown Data at 27.85 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)
225	698.4	225	653.7	225	609.4	225	565.8	225	523.5
220	680.8	220	634.5	220	588.5	220	542.9	220	498.4
215	664.3	215	616.7	215	569.1	215	521.7	215	475.0
210	648.9	210	600.0	210	551.0	210	501.9	210	453.3
205	634.6	205	584.5	205	534.2	205	483.6	205	433.2
200	621.1	200	570.1	200	518.5	200	466.5	200	414.5
195	608.6	195	556.6	195	504.0	195	450.7	195	397.2
190	596.9	190	544.0	190	490.4	190	436.0	190	381.1
185	586.0	185	532.3	185	477.8	185	422.4	185	366.3
180	575.8	180	521.4	180	466.1	180	409.7	180	352.4
175	566.3	175	511.3	175	455.2	175	397.9	175	339.6
170	557.4	170	501.8	170	445.0	170	387.0	170	327.8
165	549.1	165	493.0	165	435.6	165	376.8	165	316.8
160	541.4	160	484.8	160	426.8	160	367.4	160	306.6
155	534.2	155	477.2	155	418.7	155	358.8	155	297.3
150	527.5	150	470.1	150	411.2	150	350.7	150	288.6
145	521.2	145	463.5	145	404.2	145	343.3	145	280.6
140	515.3	140	457.3	140	397.7	140	336.4	140	273.2
135	509.9	135	451.6	135	391.8	135	330.0	135	266.4
130	504.8	130	446.3	130	386.2	130	324.2	130	260.2
125	500.0	125	441.4	125	381.0	125	318.8	125	254.5
120	495.6	120	436.8	120	376.3	120	313.8	120	249.2
115	491.5	115	432.6	115	371.9	115	309.2	115	244.4
110	487.6	110	428.7	110	367.8	110	305.0	110	240.0
105	484.1	105	425.0	105	364.2	105	301.2	105	236.0
100	480.7	100	421.7	100	360.7	100	297.6	100	232.3
95	477.6	95	418.6	95	357.5	95	294.4	95	229.0
90	474.7	90	415.7	90	354.6	90	291.5	90	226.0
85	472.0	85	413.0	85	352.0	85	288.8	85	223.2
80	469.5	80	410.5	80	349.5	80	286.4	80	220.8
75	467.1	75	408.2	75	347.2	75	284.3	75	218.5
70	464.7	70	405.9	70	345.0	70	281.8	70	216.3

Calc. N-291

Table 2.2-1

Low Temperature Over-Pressure (LTOP)

System Setpoints

Function	Setpoint
PORV Arming Temperature(1)	≥ 283 °F
PORV Pressure Setpoint(2)	435 psig

(1) Calc. N-298, Rev 3-1. Valid to 27.85 EFPY

(2) STA-249, Rev 3-1

Table 2.2-2

Low Temperature Over-Pressure (LTOP)

Temperature Restrictions

Restriction	Setpoint
	RSGs(1)
SI Pumps Secured, CCP 1 or CCP 2 Secured, SI Accumulators Isolated, CCP 3 aligned for LTOP operation	≤ 283 °F
Safety Injection Flowpath Blocked, and SI Blocked	≤ 174 °F
2 of 3 Charging Pumps Secured	≤ 161 °F
1 of 4 RCPs Secured	≤ 153 °F
2 of 4 RCPs Secured	≤ 137 °F
3 of 4 RCPs Secured	≤ 123 °F
4 of 4 RCPs Secured	≤ 114 °F
RCS Vent Path of 2.07 in^2 Established	≤ 96 °F(2)

(1) Calc. STA-249, Rev 3-1

Assumptions: 1) PORV Stroke Time of 2.9 seconds.

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

(2) Calculation STA-298 establishes an exception for vacuum refill that an RCS vent is not required as long as the RCS temperature is greater than 91 °F and when the RCS level < 123'.

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.

4.15 WCAP-17472-NP Rev 1, Ex-Vessel Neutron Dosimetry Program for Diablo Canyon Unit 1 Cycle 16, October 2011.

4.16 WCAP-17528-NP Rev 0, Ex-Vessel Neutron Dosimetry Program for Diablo Canyon Unit 2 Cycle 16, February 2012.

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 data sets 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.

Per Calculation N-288 Rev 3, data for U2 Intermediate Shell Longitudinal Weld Metal Heat 21935/12008 also shows scatter in excess of Criterion 3 allowable values.

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	37.4	0.655	24.51	6.00	-18.51
Inter Shell Plate B4106-3	Y		1.014	37.92	52.86	14.94
Inter Shell Plate B4106-3	V		1.085	40.60	37.82	-2.78
Surveillance Weld Heat 27204	S	208.5	0.655	136.62	119.13	-17.49
Surveillance Weld Heat 27204	Y		1.014	211.33	241.53	30.20
Surveillance Weld Heat 27204	V		1.085	226.30	208.66	-17.64

Calculation N-288 Rev 3, Table 1

(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, and adjusted for the temperature difference between RV inlet temperature during capsule irradiation and 538°F.

Table 5.0-2
Diablo Canyon Unit 2 Surveillance Capsule Data

Material	Capsule	$CF^{(a)}$	FF	Best Fit $\Delta RT_{NDT}^{(b)}$	Measured $\Delta RT_{NDT}^{(c)}$	Scatter in ΔRT_{NDT}
Inter Shell Plate B5454-1 (Trans)	U	105.7	0.695	73.45	80.30	6.85
Inter Shell Plate B5454-1 (Trans)	X		0.972	102.76	106.50	3.74
Inter Shell Plate B5454-1 (Trans)	Y		1.118	118.12	118.60	0.48
Inter Shell Plate B5454-1 (Trans)	V		1.234	130.40	119.90	-10.50
Inter Shell Plate B5454-1 (Long)	U	105.7	0.695	73.45	72.40	-1.05
Inter Shell Plate B5454-1 (Long)	X		0.972	102.76	107.10	4.34
Inter Shell Plate B5454-1 (Long)	Y		1.118	118.12	118.60	0.48
Inter Shell Plate B5454-1 (Long)	V		1.234	130.40	130.40	0.00
Surveillance Weld	U	204.6	0.695	142.19	180.00	37.81
Surveillance Weld	X		0.972	198.92	210.20	11.28
Surveillance Weld	Y		1.118	228.64	218.40	-10.24
Surveillance Weld	V		1.234	252.42	231.50	-20.92

Calculation N-288 Rev 3, Table 2

(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, and adjusted for the temperature difference between RV inlet temperature during capsule irradiation and 538°F.

6. SUPPLEMENTAL DATA TABLES

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 249°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 report:

7.1 WCAP-17315-NP, Rev. 0, "Diablo Canyon Units 1 and 2 Pressurized Thermal Shock and Upper-Shelf Energy Evaluations", July 2011.

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 N-288, Rev 3, "Adjusted RT-NDT Versus EFPY"

8.8 PG&E Calculation N-291, Rev 4, "Pressure-Temperature Limits for Heatup & Cooldown"

8.9 PG&E Calculation N-298, Rev 3, "LTOP Enable Temperature for 27.85 EFPY"

8.10 PG&E Calculation STA-249 Rev 3, "RSG - LTOP Analysis"

8.11 Operating Procedure OP A-2:IX, "Reactor Vessel - Vacuum Refill of the RCS"

8.12 Westinghouse Letter PGE -14-12, "Applicability of the Pressure-Temperature Limit Curves During Vacuum Refill of the RCS in Mode 5", February 21, 2014

8.13 PG&E Calculation N-288, Rev 4, "Reactor Vessel Adjusted RT-NDT Versus EFPY"

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 WCAP-15958 calculated fluence values given here are slightly higher than the more recent WCAP-17315-NP Rev 0 values.

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 WCAP-15958 calculated fluence values given here are slightly higher than the more recent WCAP-17315-NP Rev 0 values.

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

Unit 1 - Material	Capsule	$F^{(a)}$	$FF^{(b)}$	Measured $\Delta RT_{NDT}{}^{(c)}$ °F	$FF \times \Delta RT_{NDT}$ °F	FF^2
Intermediate Shell Plate B4106-3	S	0.283	0.655	6.00	3.93	0.429
	Y	1.050	1.014	52.86	53.58	1.027
	V	1.360	1.085	37.82	41.05	1.178
				SUM	98.56	2.635
	$CF_{Plate} = \Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (98.56°F) \div (2.635) = 37.4°F$					
Weld Metal	S	0.283	0.655	119.13	78.07	0.429
	Y	1.050	1.014	241.53	244.83	1.027
	V	1.36	1.085	208.66	226.49	1.178
				SUM	549.38	2.635
	$CF_{weld} = \Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (549.38) \div (2.635) = 208.5°F$					
Unit 2 - Material	**Capsule**	$F^{(a)}$	$FF^{(b)}$	**Measured $\Delta RT_{NDT}{}^{(c)}$ °F**	$FF \times \Delta RT_{NDT}$ °F	FF^2
Intermediate Shell Plate B5454-1 (Long)	U	0.330	0.695	72.40	50.32	0.483
	X	0.906	0.972	107.10	104.14	0.945
	Y	1.530	1.118	118.60	132.55	1.249
	V	2.380	1.234	130.40	160.89	1.522
Intermediate Shell Plate B5454-1 (Transverse)	U	0.330	0.695	80.30	55.81	0.483
	X	0.906	0.972	106.50	103.55	0.945
	Y	1.530	1.118	118.60	132.55	1.249
	V	2.380	1.234	119.90	147.94	1.522
				SUM	887.76	8.400
	$CF_{Plate} = \Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (887.76°F) \div (8.400) = 105.7°F$					
Weld Metal	U	0.330	0.695	180.00	125.10	0.483
	X	0.906	0.972	210.20	204.38	0.945
	Y	1.530	1.118	218.40	244.09	1.249
	V	2.380	1.234	231.50	285.64	1.522
				SUM	859.22	4.200
	$CF_{Weld} = \Sigma(FF^* \Delta RT_{NDT}) \div \Sigma(FF^2) = (859.22°F) \div (4.200) = 204.6°F$					

Calculation N 288 Rev 3, Table 1 (Unit 1) and Table 2 (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) Calculated using Charpy data plotted by EPRI Hyperbolic Tangent Curve Fitting Routine, Revision 2.0, and adjusted for the temperature difference between RV inlet temperature during capsule irradiation and 538°F.

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 ¼t, and ¾t Locations at 27.85 EFPY

Material [(a)]	Fluence $f_{1/4t}$	Fluence $f_{3/4t}$
Inter Shell Plate		
B4106-1	6.19 E + 18	2.20 E + 18
B4106-2	6.19 E + 18	2.20 E + 18
B4106-3	6.19 E + 18	2.20 E + 18
Lower Shell Plate		
B4107-1	6.19 E + 18	2.20 E + 18
B4107-2	6.19 E + 18	2.20 E + 18
B4107-3	6.19 E + 18	2.20 E + 18
Inter Shell Long		
Welds 2-442 A,B	4.55 E + 18	1.62 E + 18
Weld 2-442 C	2.35 E + 18	8.34 E + 17
Inter Shell to Lower Shell Weld 9-442	6.19 E + 18	2.20 E + 18
Lower Shell Long		
Welds 3-442 A,B	3.71 E + 18	1.32 E + 18
Weld 3-442 C	6.19 E + 18	2.20 E + 18

Calc N-288 Rev 3

(a) Only beltline materials are included. WCAP-17315-NP demonstrates that extended beltline materials are not limiting through at least 54 EFPY.

TABLE 6.0-7

DCPP-2 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the ¼t and ¾t Locations at 27.85 EFPY

Material [(a)]	Fluence $f_{1/4t}$	Fluence $f_{3/4t}$
Inter Shell Plate		
B5454-1	6.88 E + 18	2.44 E + 18
B5454-2	6.88 E + 18	2.44 E + 18
B5454-3	6.88 E + 18	2.44 E + 18
Lower Shell Plate		
B5455-1	6.88 E + 18	2.44 E + 18
B5455-2	6.88 E + 18	2.44 E + 18
B5455-3	6.88 E + 18	2.44 E + 18
Inter Shell Long		
Weld 2-201 A	3.82 E + 18	1.36 E + 18
Weld 2-201 B	4.67 E + 18	1.66 E + 18
Weld 2-201 C	3.98 E + 18	1.41 E + 18
Inter Shell to Lower Shell Weld 9-201	6.88 E + 18	2.44 E + 18
Lower Shell Long		
Weld 3-201 A	3.98 E + 18	1.41 E + 18
Weld 3-201 B	3.82 E + 18	1.36 E + 18
Weld 3-201 C	4.67 E + 18	1.66 E + 18

Calc N-288 Rev 3

(a) Only beltline materials are included. WCAP-17315-NP demonstrates that extended beltline materials are not limiting through at least 54 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 27.85 EFPY

Material	27.85 EFPY ART[a]		
	RG 1.99 Rev 2 Method	¼t (°F)	¾t (°F)
Inter Shell Plate			
B4106-1	Position 1.1	97.9	74.5
B4106-2	Position 1.1	101.1	79.0
B4106-3	Position 1.1	125.9	109.9
Lower Shell Plate			
B4107-1	Position 1.1	126.7	102.2
B4107-2	Position 1.1	125.2	102.7
B4107-3	Position 1.1	82.5	60.2
Inter Shell Long			
Welds 2-442 A,B	Position 1.1	186.6	127.4
Weld 2-442 C	Position 1.1	147.5	96.0
Inter Shell to Lower Shell Weld 9-442	Position 1.1	158.6	111.5
Lower Shell Long			
Welds 3-442 A,B	Position 1.1	174.2	117.2
Weld 3-442 C[c]	Position 1.1	205.9[b]	143.9

Calc N-288 Rev 3

(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 (207.8°F, based on 28 EFPY).

(c) DCPP-1 Surveillance Capsule data were not judged "credible" per 10 CFR 50.61.

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 27.85 EFPY

Material	27.85 EFPY ART[(a)]		
	RG 1.99 Rev 2 Method	¼t (°F)	¾t (°F)
Inter Shell Plate			
B5454-1	Position 2.1	163.6	134.4
B5454-2	Position 1.1	190.2	162.6[(b)]
B5454-3	Position 1.1	165.9	135.3
Lower Shell Plate			
B5455-1	Position 1.1	106.9	79.7
B5455-2	Position 1.1	121.9	94.7
B5455-3	Position 1.1	107.4	89.3
Inter Shell Long			
Weld 2-201 A	Position 1.1	160.9	107.5
Weld 2-201 B	Position 1.1	172.4	117.0
Weld 2-201 C	Position 1.1	163.2	109.4
Inter Shell to Lower Shell Weld 9-201	Position 1.1	20.1	5.0
Lower Shell Long			
Weld 3-201 A	Position 1.1	103.5	71.3
Weld 3-201 B	Position 1.1	102.2	70.2
Weld 3-201 C	Position 1.1	109.0	75.9

Calc N-288 Rev 3

(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 (163.4°F, based on 28 EFPY).

TABLE 6.0-10 Calculation of Adjusted Reference Temperature at 27.85 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/cm² (E > 1.0 MeV), f(a)	0.619	0.244
Fluence Factor, FF(b)	0.8658	0.6183
ΔRT_{NDT} = CF x FF, (°F)	196.4(f)	61.6
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	205.9(f)	162.6(f)

Calc N-288 Rev 3

(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) The higher CF based on CE NPSD-1039, Rev 2 for these limiting materials is used to generate the heatup and cooldown Appendix G curves. The ART's used to generate the heatup and cooldown curves are bounding based on 28 EFPY values of 207.8°F for 1/4t and 163.4°F for 3/4t.