Pacific Gas and Electric Company

Hossein G. Hamzehee
Manager
Regulatory Services

Diablo Canyon Power Plant
Mail Code 104/5/535
P.O. Box 56
Avila Beach, CA 93424

805.545.4720
Internal: 691.4720
Fax: 805.545.3459

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U.S. Nuclear Regulatory Commission
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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, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," Pacific Gas and Electric Company (PG&E) is submitting the enclosed Revision 16 of the PTLR for Units 1 and 2, dated July 31, 2017.

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

dqmg/6192/50943731
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

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
DIABLO CANYON POWER PLANT
PRESSURE AND TEMPERATURE LIMITS REPORT
(PTLR)-1
REVISION 16
EFFECTIVE DATE: July 31, 2017

TITLE: PTLR for Diablo Canyon

07/31/17
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 35 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.

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 2 intermediate shell plate B5454-2 based on an ART of 197.8°F. For this pre-service hydro test, with no fuel in the vessel, the minimum RCS temperature for all pressures is 257.8°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 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 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 (86 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 327.5°F. Thus, the minimum temperature at which 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 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.

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.

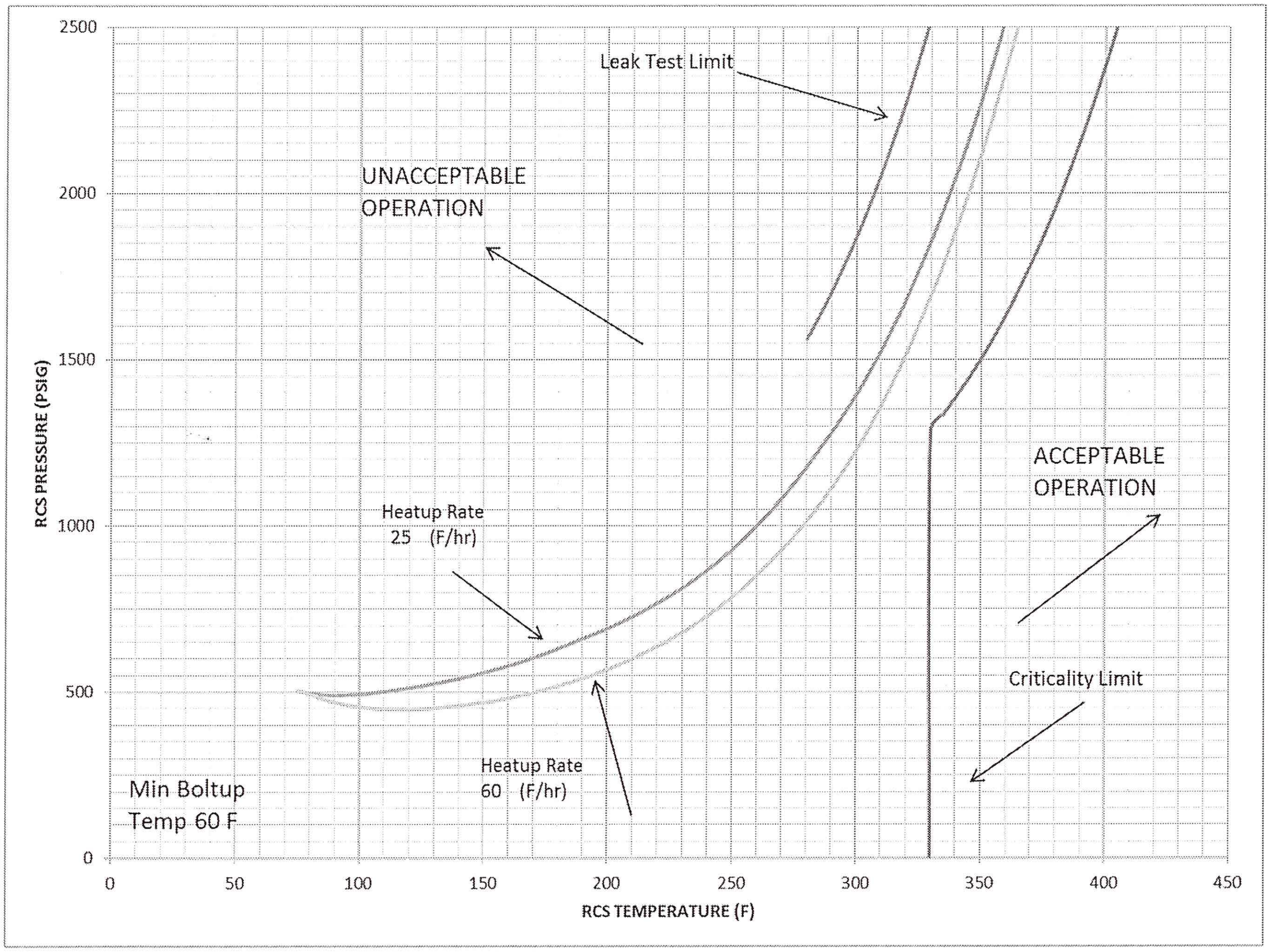

FIGURE 2.1-1: Diablo Canyon Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable to 35 EFPY (Unit 1 and Unit 2) (Without Margins for Instrumentation Errors)

TABLE 2.1-1

**Diablo Canyon Heatup Data at 35 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	471.9	75	470.3				
80	464.3	80	462.4				
85	459.1	85	443.7				
90	458.4	90	424.3				
95	459.7	95	409.9				
100	461.8	100	407.7				
105	465.1	105	410.3				
110	469.0	110	412.3				
115	473.7	115	414.3				
120	478.9	120	416.0				
125	484.8	125	417.9				
130	491.2	130	419.9				
135	498.2	135	422.4				
140	505.7	140	425.5				
145	513.8	145	428.9				
150	522.5	150	433.1				
155	531.9	155	437.0				
160	541.9	160	442.7				
165	552.7	165	449.1				
170	564.2	170	455.5				
175	576.5	175	462.0				
180	589.8	180	470.4				
185	604.0	185	479.7				
190	618.7	190	489.1				
195	633.8	195	499.3				
200	648.9	200	510.3				
205	664.3	205	522.1				
210	680.8	210	535.0				
215	698.4	215	548.7				
220	717.4	220	563.5				
225	737.7	225	579.2				
230	759.4	230	596.4				
235	782.7	235	614.8				
240	807.6	240	634.3				

TABLE 2.1-1

**Diablo Canyon Heatup Data at 35 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	834.4	245	655.5				
250	863.1	250	678.2				
255	893.9	255	702.7				
260	926.9	260	728.9				
265	962.3	265	757.0				
270	1000.3	270	786.7				
275	1041.0	275	819.1				
280	1084.7	280	853.9				
285	1131.6	285	891.2				
290	1181.7	290	931.3			285.0	1505.9
295	1234.6	295	974.2			290.0	1571.9
300	1287.7	300	1020.4			295.0	1642.8
305	1344.1	305	1069.6			300.0	1718.7
310	1404.6	310	1121.8			305.0	1800.1
315	1469.4	315	1178.6	355.0	1449.4	310.0	1887.2
320	1538.8	320	1239.5	360.0	1510.5	315.0	1980.4
325	1613.3	325	1304.8	365.0	1575.9	320.0	2080.2
330	1693.0	330	1374.9	370.0	1645.5	325.0	2187.0
335	1778.2	335	1449.9	375.0	1720.2	330.0	2301.1
340	1869.6	340	1530.2	380.0	1800.2	335.0	2422.9
345	1967.2	345	1616.3	385.0	1885.7	340.0	2552.9
350	2071.6	350	1708.0	390.0	1977.1	345.0	2691.5
355	2183.3	355	1805.3	395.0	2074.9	350.0	2839.1
360	2302.5	360	1911.8	400.0	2179.2	355.0	2996.2
365	2429.6	365	2022.5	405.0	2290.7	360.0	3163.0
370	2565.3	370	2142.6	410.0	2409.6	365.0	3339.9
375	2709.9	375	2270.8	415.0	2536.4	370.0	3527.2
380	2863.7	380	2405.1	420.0	2671.5	375.0	3725.2
385	3026.9	385	2537.5	425.0	2815.2	380.0	3933.9
390	3200.4	390	2672.5	430.0	2967.9	385.0	4153.4
395	3383.8	395	2816.1	435.0	3130.1	390.0	4383.6
400	3578.2	400	2968.8	440.0	3302.0	395.0	4624.2

Ref. Calc. N-291

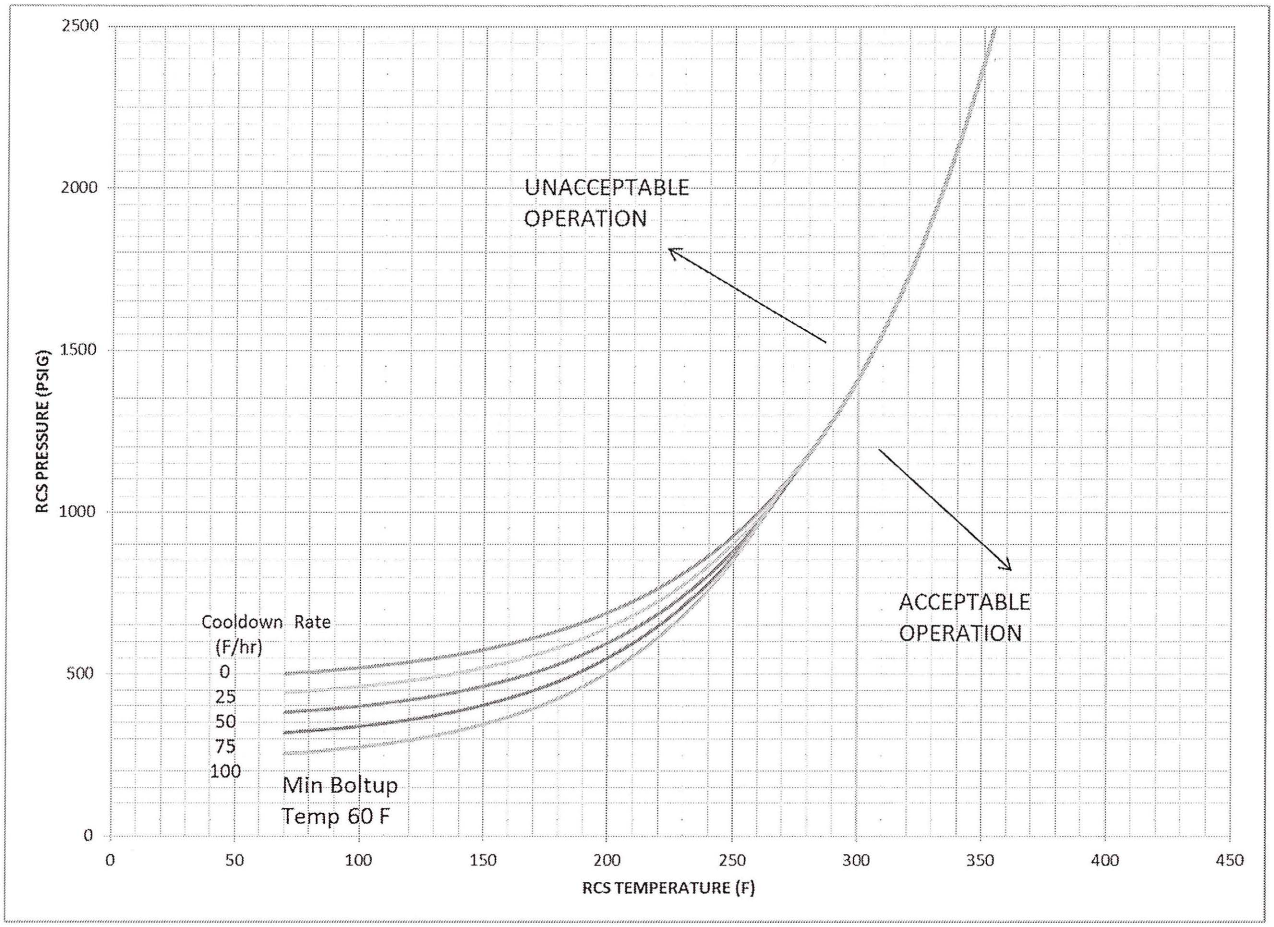

FIGURE 2.1-2: Diablo Canyon Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50, 75 and 100°F/hr) Applicable to 35 EFPY (Unit 1 and Unit 2) (Without Margins for Instrumentation Errors)

TABLE 2.1-2									
Diablo Canyon Cooldown Data at 35 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	3398.7	390	3398.7	390.0	3398.7	390.0	3398.7	390.0	3398.7
385	3208.9	385	3208.9	385.0	3208.9	385.0	3208.9	385.0	3208.9
380	3029.6	380	3029.6	380.0	3029.6	380.0	3029.6	380.0	3029.6
375	2860.8	375	2860.8	375.0	2860.8	375.0	2860.8	375.0	2860.8
370	2701.9	370	2701.9	370.0	2701.9	370.0	2701.9	370.0	2701.9
365	2552.5	365	2552.5	365.0	2552.5	365.0	2552.5	365.0	2552.5
360	2412.4	360	2412.4	360.0	2412.4	360.0	2412.4	360.0	2412.4
355	2281.0	355	2281.0	355.0	2281.0	355.0	2281.0	355.0	2281.0
350	2157.9	350	2157.9	350.0	2157.9	350.0	2157.9	350.0	2157.9
345	2042.7	345	2042.7	345.0	2042.7	345.0	2042.7	345.0	2042.7
340	1935.0	340	1935.0	340.0	1935.0	340.0	1935.0	340.0	1935.0
335	1834.3	335	1834.3	335.0	1834.3	335.0	1834.3	335.0	1834.3
330	1740.2	330	1740.2	330.0	1740.2	330.0	1740.2	330.0	1740.2
325	1652.4	325	1652.4	325.0	1652.4	325.0	1652.4	325.0	1652.4
320	1570.3	320	1570.3	320.0	1570.3	320.0	1570.3	320.0	1570.3
315	1493.8	315	1493.8	315.0	1493.8	315.0	1493.8	315.0	1493.8
310	1422.4	310	1422.4	310.0	1422.4	310.0	1422.4	310.0	1422.4
305	1355.8	305	1355.8	305.0	1355.8	305.0	1355.8	305.0	1355.8
300	1293.7	300	1293.7	300.0	1293.7	300.0	1293.7	300.0	1293.7
295	1235.9	295	1235.9	295.0	1235.9	295.0	1235.9	295.0	1235.9
290	1181.9	290	1180.7	290.0	1181.9	290.0	1181.9	290.0	1181.9
285	1131.6	285	1127.9	285.0	1129.8	285.0	1131.6	285.0	1131.6
280	1084.7	280	1076.8	280.0	1074.8	280.0	1078.0	280.0	1084.7
275	1041.0	275	1028.9	275.0	1022.0	275.0	1021.2	275.0	1027.0
270	1000.3	270	984.0	270.0	972.5	270.0	966.7	270.0	967.6
265	962.3	265	942.3	265.0	926.5	265.0	915.7	265.0	911.1
260	926.9	260	903.5	260.0	883.7	260.0	868.3	260.0	858.5
255	893.9	255	867.3	255.0	843.8	255.0	824.2	255.0	809.6
250	863.1	250	833.6	250.0	806.7	250.0	783.3	250.0	764.2
245	834.4	245	802.2	245.0	772.2	245.0	745.2	245.0	722.0
240	807.6	240	772.9	240.0	740.1	240.0	709.8	240.0	682.8
235	782.7	235	745.7	235.0	710.3	235.0	676.9	235.0	646.4
230	759.4	230	720.3	230.0	682.5	230.0	646.3	230.0	612.6

TABLE 2.1-2

Diablo Canyon Cooldown Data at 35 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	737.7	225	696.7	225.0	656.7	225.0	617.9	225.0	581.2
220	717.4	220	674.6	220.0	632.6	220.0	591.5	220.0	552.1
215	698.4	215	654.1	215.0	610.2	215.0	567.0	215.0	525.1
210	680.8	210	635.0	210.0	589.4	210.0	544.2	210.0	500.0
205	664.3	205	617.1	205.0	570.0	205.0	523.0	205.0	476.7
200	648.9	200	600.5	200.0	551.9	200.0	503.3	200.0	455.0
195	634.6	195	585.0	195.0	535.1	195.0	485.0	195.0	435.0
190	621.1	190	570.5	190.0	519.5	190.0	468.0	190.0	416.4
185	608.6	185	557.1	185.0	504.9	185.0	452.2	185.0	399.1
180	596.9	180	544.5	180.0	491.4	180.0	437.5	180.0	383.0
175	586.0	175	532.8	175.0	478.8	175.0	423.8	175.0	368.2
170	575.8	170	521.9	170.0	467.0	170.0	411.2	170.0	354.4
165	566.3	165	511.8	165.0	456.2	165.0	399.4	165.0	341.7
160	557.4	160	502.3	160.0	446.0	160.0	388.5	160.0	329.8
155	549.1	155	493.5	155.0	436.6	155.0	378.4	155.0	318.9
150	541.4	150	485.3	150.0	427.9	150.0	369.0	150.0	308.8
145	534.2	145	477.7	145.0	419.8	145.0	360.4	145.0	299.4
140	527.5	140	470.6	140.0	412.2	140.0	352.3	140.0	290.8
135	521.2	135	464.0	135.0	405.3	135.0	344.9	135.0	282.8
130	515.3	130	457.9	130.0	398.8	130.0	338.0	130.0	275.4
125	509.9	125	452.2	125.0	392.8	125.0	331.7	125.0	268.7
120	504.8	120	446.9	120.0	387.3	120.0	325.8	120.0	262.4
115	500.0	115	442.0	115.0	382.1	115.0	320.4	115.0	256.7
110	495.6	110	437.4	110.0	377.4	110.0	315.5	110.0	251.5
105	491.5	105	433.2	105.0	373.0	105.0	310.9	105.0	246.6
100	487.6	100	429.2	100.0	369.0	100.0	306.7	100.0	242.2
95	484.1	95	425.6	95.0	365.2	95.0	302.8	95.0	238.3
90	480.7	90	422.2	90.0	361.8	90.0	299.3	90.0	234.6
85	477.6	85	419.1	85.0	358.7	85.0	296.2	85.0	231.3
80	474.7	80	416.2	80.0	355.7	80.0	293.2	80.0	228.3
75	472.0	75	413.5	75.0	353.0	75.0	290.7	75.0	225.5
70	469.3	70	410.8	70.0	350.4	70.0	287.8	70.0	222.9

Calc. N-291

Table 2.2-1

Low Temperature Over-Pressure (LTOP)

System Setpoints

Function	Setpoint
PORV Arming Temperature[1]	≥ 273 °F
PORV Pressure Setpoint[2]	435 psig

(1) Calc. N-298, Rev 4. Valid to 35 EFPY

(2) STA-249, Rev 4

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	≤ 273 °F
Safety Injection Flowpath Blocked, and SI Blocked	≤ 164 °F
2 of 3 Charging Pumps Secured	≤ 151 °F
1 of 4 RCPs Secured	≤ 143 °F
2 of 4 RCPs Secured	≤ 127 °F
3 of 4 RCPs Secured	≤ 113 °F
4 of 4 RCPs Secured	≤ 104 °F
RCS Vent Path of 2.07 in^2 Established	≤ 86 °F

(1) Calc. STA-249, Rev 4

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.

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 the Intermediate Shell Plate B5454-2 in the Unit 2 reactor vessel and the most limiting ¾t location is found in the Intermediate Shell Plate B5454-2 in the Unit 2 reactor vessel.

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, 5.0-2, 5.0-3, and 5.0-4 present the Surveillance Capsule Data for Diablo Canyon Units 1 and 2, and sister plants. 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. However, when combined with the surveillance data for Palisades Weld Heat 27204 (WCAP 17315 NP Rev 0, Table A.1-8), the combined data is deemed credible per Criterion 3.

Per WCAP 17315 NP Rev 0, Table A.2-2, data for U2 Intermediate Shell Longitudinal Weld Metal Heat 21935/12008 indicates that three of the four surveillance data points fall within the 28°F scatter band for surveillance weld materials; therefore, the weld material (Heat 21935/12008) is deemed credible per Criterion 3.

Per WCAP 17315 NP Rev 0, Section A.2, data for U2 Intermediate Shell Longitudinal Weld Metal Heat 33A277 is not contained in the Diablo Canyon Unit 2 surveillance program. However, it is contained in the Farley Unit 1 and Calvert Cliffs Unit 1 surveillance programs and most closely represents the situation for Diablo Canyon Unit 2 weld Heat 33A277. WCAP 17315 NP Rev 0, Table A.2-10, indicates that all eight surveillance data points using Farley Unit 1 and Calvert Cliffs Unit 1 Data fall within the 28°F scatter band for surveillance weld materials; therefore, the weld material (Heat 33A277) is deemed credible per Criterion 3.

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	32.2	0.655	21.07	-0.38	21.5
Inter Shell Plate B4106-3	Y		1.014	32.59	48.26	15.7
Inter Shell Plate B4106-3	V		1.085	34.9	33.22	1.7
Surveillance Weld DCPP Heat 27204	S	199.6	0.655	130.79	112.19	18.6
Surveillance Weld DCPP Heat 27204	Y		1.014	202.31	232.19	29.9
Surveillance Weld DCPP Heat 27204	V		1.085	216.64	199.97	16.7

Source: WCAP-17315-NP Table A.1-4

(a) CF is calculated from surveillance data using Reg. Guide 1.99 Regulatory Position 2.1 (see WCAP-17315-NP Table A.1-3).

(b) Best fit ΔRT_{NDT} = CF * FF.

(c) Measured ΔRT_{NDT} values are derived from the measured 30 ft-lb shift values from Table 4.1-1 (see WCAP-17315-NP). These measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Reg. Guide 1.99, Rev. 2, Position 2.1 since this calculation is based on the actual surveillance weld metal shift values. Therefore, as shown in Table A.1-1 (see WCAP-17315-NP), the Diablo Canyon Unit 1 surveillance capsules are adjusted by the temperature adjustment values summarized in Table A.1-2 (see WCAP-17315-NP).

Table 5.0-2[d]
Diablo Canyon Unit 1 & Palisades Unit 1 Surveillance Capsule Data

Material	Capsule	CF[a]	FF	Best Fit ΔRT_{NDT}[b]	Measured ΔRT_{NDT}[c]	Scatter in ΔRT_{NDT}
Surveillance Weld DCPP Heat 27204	S	210.4	0.655	137.88	116.24	21.6
Surveillance Weld DCPP Heat 27204	Y		1.014	213.27	237.44	24.2
Surveillance Weld DCPP Heat 27204	V		1.085	228.38	204.9	23.5
Surveillance Weld Palisades Heat 27204	SA-60-21	210.4	1.112	234.02	245.92	11.9
Surveillance Weld Palisades Heat 27204	SA-240-1		1.234	259.6	261.16	1.6

Source: WCAP-17315-NP Table A.1-8

(a) CF is calculated from surveillance data using Reg. Guide 1.99 Regulatory Position 2.1 (see WCAP-17315-NP Table A.1-7).

(b) Best fit ΔRT_{NDT} = CF * FF.

(c) Measured ΔRT_{NDT} values are derived from the measured 30 ft-lb shift values from Tables 4.1-1 and 4.1-2 (see WCAP-17315-NP) for Diablo Canyon Unit 1 and Palisades, respectfully. These ΔRT_{NDT} values for the surveillance weld data are adjusted first by the difference in operating temperature then using the ratio procedure to account for difference in the surveillance weld chemistry and the beltline weld chemistry. The temperature adjustments are shown in Table A.1-6 (see WCAP-17315-NP). The ratios applied are 1.01 for Diablo Canyon Unit 1 and 0.99 for Palisades.

(d) As established in WCAP-17315-NP Appendix A.1, specifically NRC Case 1 and Case 4 guidelines, the combined surveillance data from Diablo Canyon Unit 1 and Palisades may be applied to the Diablo Canyon Unit 1 reactor vessel weld Heat #27204.

Table 5.0-3[d]
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	99.0	0.695	68.80	65.4	3.4
Inter Shell Plate B5454-1 (Long)	X		0.972	96.25	100.1	3.9
Inter Shell Plate B5454-1 (Long)	Y		1.118	110.63	111.6	1.0
Inter Shell Plate B5454-1 (Long)	V		1.234	122.14	123.4	1.3
Inter Shell Plate B5454-1 (Trans)	U	99.0	0.695	68.80	73.3	4.5
Inter Shell Plate B5454-1 (Trans)	X		0.972	96.25	99.5	3.3
Inter Shell Plate B5454-1 (Trans)	Y		1.118	110.63	111.6	1.0
Inter Shell Plate B5454-1 (Trans)	V		1.234	122.14	112.9	9.2
Surveillance Weld	U	197.9	0.695	137.53	173	35.5
Surveillance Weld	X		0.972	192.40	203.2	10.8
Surveillance Weld	Y		1.118	221.16	211.4	9.8
Surveillance Weld	V		1.234	244.15	224.5	19.7

Source: WCAP-17315-NP Table A.2-2

(a) CF is calculated from surveillance data using Reg. Guide 1.99 Regulatory Position 2.1 (see WCAP-17315-NP Table A.2-1).

(b) Best fit ΔRT_{NDT} = CF * FF.

(c) Measured ΔRT_{NDT} values are derived from the measure 30 ft-lb shift values from Table 4.2-1 (see WCAP-17315-NP). These measured ΔRT_{NDT} values for the surveillance weld metal do not include the adjustment ratio procedure of Reg. Guide 1.99, Revision 2, Position 2.1, since this calculation is based on the actual surveillance weld metal measured shift values. In addition, all of the Diablo Canyon Unit 2 surveillance capsules were irradiated at the same temperature; therefore, no temperature adjustments are required.

(d) As established in WCAP-17315-NP Appendix A.2, Diablo Canyon Unit 2 surveillance and weld metal (Heat#21935/12008) were evaluated using Diablo Canyon Unit 2 Data and following NRC Case 1 guidelines.

Table 5.0-4[d]
Farley Unit 1 and Calvert Cliffs Unit 1 Surveillance Capsule Data

Material	Capsule	CF[a]	FF	Best Fit ΔRT_{NDT}[b]	Measured ΔRT_{NDT}[c]	Scatter in ΔRT_{NDT}
Surveillance Weld Heat #33A277 (Farley)	Y	79.2	0.862	68.27	79.2	10.9
Surveillance Weld Heat #33A277 (Farley)	U		1.151	91.09	84.4	6.7
Surveillance Weld Heat #33A277 (Farley)	X		1.295	102.54	99.5	3.1
Surveillance Weld Heat #33A277 (Farley)	W		1.392	110.20	113.3	3.1
Surveillance Weld Heat #33A277 (Farley)	V		1.466	116.05	135.7	19.6
Surveillance Weld Heat #33A277 (Farley)	Z		1.492	118.09	130.7	12.6
Surveillance Weld Heat #33A277 (Calvert Cliffs)	263°	79.2	0.866	68.56	48.5	20.1
Surveillance Weld Heat #33A277 (Calvert Cliffs)	79.2°		1.26	99.72	74.3	25.4

Source: WCAP-17315-NP Table A.2-10

(a) CF is calculated from surveillance data using Reg. Guide 1.99 Regulatory Position 2.1 (see WCAP-17315-NP Table A.2-9).

(b) Best fit ΔRT_{NDT} = CF * FF.

(c) Measured ΔRT_{NDT} values are derived from the measured 30 ft-lb shift values from Table 4.2-2 (see WCAP-17315-NP). These ΔRT_{NDT} values for the surveillance weld data are adjusted by the difference in operating temperature then using the ratio procedure to account for difference in the surveillance weld chemistry and the beltlines weld chemistry. The temperature adjustments are shown in Table A.2-8 (see WCAP-17315-NP). The ratios applied are 1.17 for Farley Unit 1 and Calvert Cliffs Unit 1, respectfully.

(d) As established in WCAP-17315-NP Appendix A.2, Farley Unit 1 and Calvert Cliffs Unit 1 were evaluated using their respective surveillance data and following NRC Case 5 guidelines.

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 243°F (Unit 1 Weld 3-442C), at 54 EFPY (EOL). Therefore at 35 EFPY 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 STA-197, superseded by STA-249

8.8 PG&E Calculation N-291, Rev 5, "Pressure-Temperature Limits for Heatup & Cooldown"

8.9 PG&E Calculation N-298, Rev 4, "LTOP Enable Temperature for 35 EFPY"

8.10 PG&E Calculation STA-249 Rev 4, "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] ($\times 10^{19}$ 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 6.0-3A Calculation of Diablo Canyon Unit 1 Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f[a] ($\times 10^{19}$ n/cm^2, E > 1.0 MeV)	FF[b]	ΔRT_{NDT}[c] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
IS Plate B4106-3 (Longitudinal)	S	0.283	0.655	6.00 (0[d])	3.93	0.429
	Y	1.05	1.014	52.86 (48.66)	53.58	1.027
	V	1.36	1.085	37.82 (34.32)	41.05	1.178
				SUM:	98.56	2.635
	$CF_{IS\ Plate\ B4106\text{-}3} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (98.56) \div (2.635) = \mathbf{37.4°F}$					
Weld Metal Heat # 27204 (Diablo Canyon Unit 1 data)	S	0.283	0.655	119.13 (110.79)	78.06	0.429
	Y	1.05	1.014	241.53 (232.59)	244.82	1.027
	V	1.36	1.085	208.66 (201.07)	226.49	1.178
Weld Metal Heat # 27204 (Palisades data)	SA-60-1	1.50	1.112	250.10 (253.1)	278.18	1.237
	SA-240-1	2.38	1.234	265.50 (267.8)	327.59	1.522
				SUM:	1155.14	5.395
	$CF_{Heat\ \#\ 27204} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1155.14) \div (5.395) = \mathbf{214.1°F}$					

(a) f = fluence.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All Diablo Canyon Unit 1 values are taken from Table 4.1-1 of WCAP-17315-NP. The Diablo Canyon Unit 1 ΔRT_{NDT} values have been adjusted according to the temperature adjustments summarized in Table 4.1-1 of WCAP-17315-NP. Then, the Diablo Canyon Unit 1 ΔRT_{NDT} values for the surveillance weld data are adjusted by a ratio of 1.02 (pre-adjusted values are listed in parentheses). Ratio = $CF_{Vessel\ Weld}/CF_{Surv.\ Weld}$ = 226.8°F/222.3°F = 1.02.

All Palisades values are taken from Table 4.1-2 of WCAP-17315-NP. The Palisades surveillance weld ΔRT_{NDT} values have been adjusted according to the temperature adjustments summarized in Table 4.1-2 (pre-adjusted values are listed in parentheses). No ratio is applied since the ratio was calculated to be 1.00. Ratio = $CF_{Vessel\ Weld}/CF_{Surv.\ Weld}$ = 226.8°F/227.8°F = 1.00.

(d) Measured ΔRT_{NDT} value was determined to be negative, but physically a reduction should not occur. Therefore, a conservative value of zero will be used.

Table 6.0-3B Calculation of Diablo Canyon Unit 2 Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f(a) (x10^{19} n/cm^2, E > 1.0 MeV)	FF(b)	ΔRT_{NDT}(c) (°F)	FF*ΔRT_{NDT} (°F)	FF^2
IS Plate B5454-1 (Longitudinal)	U	0.330	0.695	72.4 (65.4)	50.32	0.483
	X	0.906	0.972	107.1 (100.1)	104.14	0.945
	Y	1.53	1.118	118.6 (111.6)	132.55	1.249
	V	2.38	1.234	130.4 (123.4)	160.89	1.522
IS Plate B5454-1 (Transverse)	U	0.330	0.695	80.3 (73.3)	55.81	0.483
	X	0.906	0.972	106.5 (99.5)	103.55	0.945
	Y	1.53	1.118	118.6 (111.6)	132.55	1.249
	V	2.38	1.234	119.9 (112.9)	147.94	1.522
				SUM:	887.76	8.400
	$CF_{IS\ Plate\ B5454-1} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (887.76) \div (8.400) = \mathbf{105.7°F}$					
Diablo Canyon Unit 2 Weld Metal (Heat # 21935/12008)	U	0.330	0.695	180.0 (173.0)	125.10	0.483
	X	0.906	0.972	210.2 (203.2)	204.38	0.945
	Y	1.53	1.118	218.4 (211.4)	244.09	1.249
	V	2.38	1.234	231.5 (224.5)	285.64	1.522
				SUM:	859.22	4.200
	$CF_{Heat\ \#\ 21935/12008} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (859.22) \div (4.200) = \mathbf{204.6°F}$					

(a) f = fluence.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All values are taken from Table 4.2-1 of WCAP-17315-NP. The Diablo Canyon Unit 2 ΔRT_{NDT} values have been adjusted according to the temperature adjustments summarized in Table 4.2-1 of WCAP-17315-NP (pre-adjusted values are listed in parentheses). No ratio is applied to the ΔRT_{NDT} values for the surveillance weld data since the beltline weld and surveillance weld chemistry factors are identical.

Table 6.0-3C Calculation of Diablo Canyon Unit 2 Weld Heat # 33A277 Chemistry Factors Using Surveillance Capsule Data from Farley Unit 1 and Calvert Cliffs Unit 1

Material	Capsule	Capsule f[(a)] ($x10^{19}$ n/cm^2, E > 1.0 MeV)	FF[(b)]	ΔRT_{NDT}[(c)] (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Weld Metal Heat # 33A277 (Farley Unit 1 data)	Y	0.612	0.862	118.1 (66.9)	101.86	0.744
	U	1.73	1.151	125.3 (75.1)	144.20	1.324
	X	3.06	1.295	146.2 (87.4)	189.42	1.678
	W	4.75	1.392	165.3 (98.3)	230.15	1.938
	V	7.14	1.466	196.4 (117.5)	287.90	2.149
	Z	8.47	1.492	189.4 (113.5)	282.59	2.225
Weld Metal Heat # 33A277 (Calvert Cliffs Unit 1 data)	263°	0.62	0.866	73.1 (59)	63.35	0.750
	97°	2.64	1.260	109.2 (93)	137.54	1.587
				SUM:	1436.99	12.396
	$CF_{Heat\ \#\ 33A277} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (1436.99) \div (12.396) =$ **115.9°F**					

(a) f = fluence.

(b) FF = fluence factor = $f^{(0.28 - 0.10*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values. All values are taken from Table 4.2-2 of WCAP-17315-NP. ΔRT_{NDT} values for the surveillance weld data are adjusted first by the difference in operating temperature then using the ratio procedure to account for differences in the surveillance weld chemistry and the beltline weld chemistry (pre-adjusted values are listed in parentheses). The temperature adjustments and ratios applied are as follows:

Farley Unit 1:

Temperature adjustment per Table 4.2-2 (on a capsule-by-capsule basis) Ratio = $CF_{Vessel\ Weld}/CF_{Surv.\ Weld}$ = 126.3°F/78.1°F = 1.62

Calvert Cliffs Unit 1:

Temperature adjustment per Table 4.2-2 (+10.00°F for each capsule) Ratio = $CF_{Vessel\ Weld}/CF_{Surv.\ Weld}$ = 126.3°F/119.4°F = 1.06

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 35 EFPY

Material [(a)]	Fluence $f_{1/4t}$	Fluence $f_{3/4t}$
Inter Shell Plate		
B4106-1	7.98 E + 18	2.83 E + 18
B4106-2	7.98 E + 18	2.83 E + 18
B4106-3	7.98 E + 18	2.83 E + 18
Lower Shell Plate		
B4107-1	7.98 E + 18	2.83 E + 18
B4107-2	7.98 E + 18	2.83 E + 18
B4107-3	7.98 E + 18	2.83 E + 18
Inter Shell Long		
Welds 2-442 A,B	5.89 E + 18	2.09 E + 18
Weld 2-442 C	3.07 E + 18	1.09 E + 18
Inter Shell to Lower		
Shell Weld 9-442	7.98 E + 18	2.83 E + 18
Lower Shell Long		
Welds 3-442 A,B	4.87 E + 18	1.73 E + 18
Weld 3-442 C	7.98 E + 18	2.83 E + 18

Calc N-288 Rev 4

(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 35 EFPY

Material [(a)]	Fluence $f_{1/4t}$	Fluence $f_{3/4t}$
Inter Shell Plate		
B5454-1	9.04 E + 18	3.21 E + 18
B5454-2	9.04 E + 18	3.21 E + 18
B5454-3	9.04 E + 18	3.21 E + 18
Lower Shell Plate		
B5455-1	9.04 E + 18	3.21 E + 18
B5455-2	9.04 E + 18	3.21 E + 18
B5455-3	9.04 E + 18	3.21 E + 18
Inter Shell Long		
Weld 2-201 A	5.01 E + 18	1.78 E + 18
Weld 2-201 B	6.06 E + 18	2.15 E + 18
Weld 2-201 C	5.16 E + 18	1.83 E + 18
Inter Shell to Lower Shell Weld 9-201	9.04 E + 18	3.21 E + 18
Lower Shell Long		
Weld 3-201 A	5.16 E + 18	1.83 E + 18
Weld 3-201 B	5.01 E + 18	1.78 E + 18
Weld 3-201 C	6.06 E + 18	2.15 E + 18

Calc N-288 Rev 4

[(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 35 EFPY

Material	35 EFPY ART[a]		
	RG 1.99 Rev 2 Method	¼t (°F)	¾t (°F)
Inter Shell Plate			
B4106-1	Position 1.1	103.9	79.9
B4106-2	Position 1.1	106.9	84.1
B4106-3	Position 1.1	129.8	114.3
Lower Shell Plate			
B4107-1	Position 1.1	133.1	107.9
B4107-2	Position 1.1	131.0	107.9
B4107-3	Position 1.1	88.2	65.4
Inter Shell Long			
Welds 2-442 A,B	Position 2.1	170.4	112.3
Weld 2-442 C	Position 2.1	132.8	81.1
Inter Shell to Lower Shell Weld 9-442	Position 1.1	170.8	122.4
Lower Shell Long			
Welds 3-442 A,B	Position 2.1	159.2	102.7
Weld 3-442 C	Position 2.1	188.6	128.4

Calc N-288 Rev 4

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

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 35 EFPY

Material	35 EFPY ART[a]		
	RG 1.99 Rev 2 Method	¼t (°F)	¾t (°F)
Inter Shell Plate			
B5454-1	Position 2.1	171.7	141.7
B5454-2	Position 1.1	197.8[b]	169.5[b]
B5454-3	Position 1.1	174.4	143.0
Lower Shell Plate			
B5455-1	Position 1.1	114.4	86.5
B5455-2	Position 1.1	129.4	101.5
B5455-3	Position 1.1	112.4	93.8
Inter Shell Long			
Weld 2-201 A	Position 2.1	143.1	88.9
Weld 2-201 B	Position 2.1	153.9	98.1
Weld 2-201 C	Position 2.1	144.8	90.3
Inter Shell to Lower Shell Weld 9-201	Position 1.1	24.4	8.7
Lower Shell Long			
Weld 3-201 A	Position 2.1	82.5	51.7
Weld 3-201 B	Position 2.1	81.6	50.8
Weld 3-201 C	Position 2.1	87.7	56.1

Calc N-288 Rev 4

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

(b) These limiting ART values are used to generate heatup and cooldown curves (based on 35 EFPY).

TABLE 6.0-10 Calculation of Adjusted Reference Temperature at 35 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)	99.6[f]	99.6[f]
Fluence ÷ 10^{19} n/cm^2 (E > 1.0 MeV), f[a]	0.904	0.321
Fluence Factor, FF[b]	0.9717	0.6878
ΔRT_{NDT} = CF x FF, (°F)	96.8	68.5
Initial RT_{NDT}, I (°F)	67	67
Margin, M (°F)[c]	34	34
ART = I + (CF x FF) + M (°F) per Regulatory Guide 1.99, Rev 2	197.8[f]	169.5[f]

Calc N-288 Rev 4

(a) Fluence, f, is based upon $f_{¼t}$ and $f_{¾t}$ from Table 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-2 intermediate shell plate B5454-2 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 ARTs used to generate the heatup and cooldown curves are bounding based on 35 EFPY values of 197.8°F for ¼t and 169.5°F for ¾t.