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PG&E Letter DCL-02-079

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
License Amendment Request 02-04,
Revision of Technical Specification 5.6.6 - Reactor Coolant System Pressure and Temperature Limits Report

Dear Commissioners and Staff:

In accordance with 10 CFR 50.90, enclosed is an application for amendment to Facility Operating License Nos. DPR-80 and DPR-82 for Units 1 and 2 of the Diablo Canyon Power Plant (DCPP), respectively. This License Amendment Request (LAR) proposes to revise Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to obtain approval of WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for use at DCPP to allow changes to the PTLR without prior NRC approval. This LAR are also incorporates NRC approved Industry/TSTF Standard Technical Specification Change Traveler No. 419.

PG&E initially submitted the DCPP PTLR for NRC review in PG&E Letter DCL-99-146 dated November 24, 1999. Following submittal, the NRC staff requested additional information, due in part, to the diverse references PG&E used to document its PTLR methodology. After discussing the questions with the NRC staff, PG&E decided to perform a new analysis consolidating the supporting calculations into a more cohesive and comprehensive calculation and to resubmit the PTLR. In the meantime, PG&E stated it would continue to operate DCPP within the existing pressure/temperature (P/T) and low-temperature overpressure protection (LTOP) limits approved in License Amendment Nos. 133 and 131, dated May 3, 1999, for DCPP Units 1 and 2, respectively. (Reference PG&E Letter DCL-00-070 dated April 26, 2000).

The changes proposed by this LAR are not required to address an immediate safety concern. New PTLR P/T and LTOP curves will be needed for the cycle 13

A001

reload cores for both Unit 1 and Unit 2. The cycle 13 core for Unit 1, the lead unit, will be installed during Refueling Outage No. 12, scheduled for February 2004. To provide sufficient lead time to prepare new curves and implement procedure changes, PG&E requests that the NRC staff complete its review and approval of this LAR by August 1, 2003. PG&E requests that the proposed TS change become effective immediately, to be implemented within 30 days from the date of issuance.

Sincerely,

Gregory M. Rueger

cc: Edgar Bailey, DHS
Ellis W. Merschoff
David L. Proulx
Girija S. Shukla
Diablo Distribution

Enclosures
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY	) Docket No. 50-275) Facility Operating License) No. DPR-80
Diablo Canyon Power Plant Units 1 and 2	) Docket No. 50-323) Facility Operating License) No. DPR-82

AFFIDAVIT

Gregory M. Rueger, of lawful age, first being duly sworn upon oath says that he is Senior Vice President - Generation and Chief Nuclear Officer of Pacific Gas and Electric Company; that he has executed LAR 02-04 on behalf of said company with full power and authority to do so; that he is familiar with the content thereof; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

Gregory M. Rueger
Senior Vice President – Generation and Chief Nuclear Officer

Subscribed and sworn to before me this 30th day of July, 2002

Notary Public
County of San Francisco
State of California

AMY EMIKO DONG
Commission # 1206749
Notary Public - California
San Francisco County
My Comm. Expires Jan 2, 2003

LICENSE AMENDMENT REQUEST FOR DIABLO CANYON POWER PLANT REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE LIMITS REPORT

1.0 DESCRIPTION

This License Amendment Request (LAR) proposes to revise Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," to obtain approval of WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," for use at Diablo Canyon Power Plant (DCPP) to allow changes to the PTLR without prior NRC approval. The PTLR is submitted in accordance with the guidance of Generic Letter 96-03 for NRC staff review to allow the plant-specific application of the WCAP-14040-NP-A PTLR methodology to calculate new plant pressure/temperature (P/T) and low-temperature overpressure protection (LTOP) limits in the future, without prior NRC staff approval.

2.0 PROPOSED CHANGE

The proposed change would revise TS 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)," by adding the phrase "and LTOP" to TS 5.6.6.b, and replacing the references under TS 5.6.6.b with WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RSC Heatup and Cooldown Limit Curves." In addition, the definition of PTLR in TS 1.1, "Definitions," would be revised to delete the reference to TS containing the limits specified in the PTLR. The proposed TS changes to reference the topical report WCAP-14040-NP-A by number and title, and to revise the PTLR definition, are consistent with NRC approved Industry/TSTF Standard Technical Specification Change Traveler No. 419 (TSTF-419).

Specifically, the proposed changes are:

TS 1.1 - PTLR definition - Delete the last sentence which states in part; "Plant operation within...(LTOP) System."

TS 5.6.6.b - Add "and LTOP" after "temperature" and replace the PTLR methodology references in subparagraphs 1, 2 and 3 with the following subparagraph:

1. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RSC Heatup and Cooldown Limit Curves."

The proposed TS changes are noted on the marked up TS pages in Enclosure 2. The proposed TS pages are provided in Enclosure 3. The proposed PTLR is provided in Enclosure 4. PG&E Calculation STA-138, which contains the revised LTOP analysis to support the PTLR, is submitted for information in Enclosure 5. Supporting references are included in Enclosure 6 for information.

There are no changes required to the TS Bases.

3.0 BACKGROUND

NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996, provides guidance for relocating P/T limit curves and LTOP system limits from TS to a PTLR or a similar document. GL 96-03 states that this alternative was based on a change included in the improved standard technical specifications (STS) to remove the P/T limit curves and LTOP system limits from the TS and relocate them to a PTLR or a similar document to reduce the number of amendment requests associated with changes to these curves and limits.

GL 96-03 states that since an LAR must be submitted whenever a change is made to the TS, the relocation of the P/T curves and LTOP system limits will result in a resource savings for the licensees and the NRC by eliminating unnecessary license amendment requests for changes to the P/T limit curves and LTOP system limits in TS.

In NRC letter to PG&E, "Conversion to Improved Technical Specifications for Diablo Canyon Power Plant, Units 1 and 2 – Amendment No. 135 to Facility Operating License Nos. DPR-80 and DPR-82," License Amendment (LA) 135/135) dated May 28, 1999, the NRC staff approved conversion of the DCPP TS to the Improved TS. As part of this conversion, the RCS P/T and LTOP limits were relocated from the TS to the DCPP PTLR. The safety evaluation (SE) for LA 135/135 stated that the limits addressed in the PTLR for TS 5.6.6 are the limits that the NRC staff previously approved in LA 133/131, dated May 13, 1999. These LAs approved P/T limit curves that are valid for 16 effective full power years (EFPY). The SE for LA 135/135 further stated that the NRC staff will review PG&E's plant-specific application of the PTLR methodology to allow PG&E to use the PTLR methodology in the future to calculate new P/T and LTOP limits without prior NRC approval.

In PG&E Letter to NRC, DCL-99-146, "Request for NRC Approval of Diablo Canyon Methodology for Establishing Pressure/ Temperature and Low Temperature Overpressure Protection Limits Using WCAP-14040-NP-A in Accordance with Generic Letter 96-03," dated November 24, 1999, PG&E initially submitted its plant specific PTLR methodology for approval. Following submittal, the NRC staff requested additional information due, in part, to the diverse references PG&E used to document its PTLR methodology. After discussing the questions with the NRC staff, PG&E decided (Reference PG&E letter to NRC, DCL-00-070, "Supplement to Reactor Coolant System Pressure and Temperature Limits Report," dated April 26, 2000) to perform a new analysis consolidating the supporting calculations into a more cohesive and comprehensive calculation, and to resubmit the PTLR methodology for approval. In the meantime, PG&E stated it would operate DCPP within the existing RCS P/T limit curves approved in LA 133/131.

TSTF-419 allows revising the TS definition of PTLR. The NRC safety evaluation that approved TSTF-419 (Reference 14) states that the definition of PTLR identifies the TS in which the P/T limits are addressed. However, TS 5.6.6.a requires that the individual TS that address RCS P/T limits be referenced. The proposed change to the definition eliminates the duplication between the definition of PTLR and TS 5.6.6.

TSTF-419 also allows topical reports identified in TS 5.6.6 to be identified by number and title. Reference 14 states this will allow licensees to use current topical reports to support limits in the PTLR without having to submit an LAR every time the topical report is revised. The PTLR would provide the specific information identifying the particular approved topical report used to determine the P/T limits or LTOP system limits.

4.0 TECHNICAL ANALYSIS

The analysis below explains how PG&E utilizes the guidance of GL 96-03 and the methodology of WCAP-14040-NP-A, Revision 2, dated January 1996, with minor variations to establish the DCPP PTLR methodology. The following discussion includes; (1) differences between the DCPP PTLR methodology and WCAP-14040-NP-A; (2) changes between the PTLR that PG&E submitted initially (Reference 7) and the PTLR submitted with this LAR; and (3) a summary of the PTLR methodology proposed for approval.

4.1 Differences from WCAP-14040-NP-A

In a follow up to its initial PTLR methodology submittal (Reference 7), PG&E submitted a response (Reference 8) to an NRC staff request for additional information that identified several areas where there were differences between PG&E's determination of LTOP setpoints and WCAP-14040-NP-A.

These differences have been resolved and the DCPP PTLR methodology is now completely consistent with the WCAP-14040-NP-A methodology, but also includes the following programmatic enhancements which are not explicitly discussed in the WCAP.

1. The WCAP-14040-NP-A methodology evaluates the RCS pressure overshoot for starting a reactor coolant pump (RCP) with a maximum RCS/steam generator (SG) temperature difference of 50°F and the RCS in a water solid condition. DCPP TSs 3.4.6 and 3.4.7 allow starting a RCP without any RCS/SG temperature restrictions when the pressurizer level indicates less than or equal to 50 percent. PG&E has added to the methodology an additional heat injection analysis with the pressurizer partially full to reiterate the bases for this TS limiting condition for operation (LCO) allowance. The heat injection evaluation demonstrated that the reduced pressurizer level provides enough additional expansion volume to ensure that the maximum RCS/SG temperature difference allowed within the operating procedures guidelines does not challenge the ASME Section III Appendix G P/T limits.

2. The stress correction factors, M_m and M_b, used in WCAP 14040-NP-A are taken from graphs within the WCAP and reference Welding Research Council Bulletin No. 175. The DCPP PTLR stress correction factors are graphically presented in Figures A-3300-3 and A-3300-5 of ASME Code Section XI Appendix A. The equation for the flaw shape factor, Q, is from EPRI NP-1181, "Computational Method to Perform the Flaw Evaluation Procedure as Specified in ASME Code, Section XI, Appendix A," dated September 1979. The use of ASME Section XI Appendix A is approved and discussed in LA 133/131 SE, section 3.2.2. As stated in the SE, PG&E opted to use the technical methods provided in nonmandatory Appendix A to Section XI of the ASME Code as the methodology for generating the DCPP P/T limit curves. To test the validity of PG&E's proposed curves, the staff performed an independent assessment of the PG&E's submittal. PG&E's proposed P/T limit curves for normal operating and pressure testing conditions, effective to 16 effective full power years, were found to be slightly more conservative than the P/T limit curves generated by the staff in accordance with the methods of Appendix G to the Code.

3. The WCAP-14040-P-A methodology evaluates a variable cold overpressure mitigation system (COMS) power operated relief valve (PORV) setpoint that decreases with RCS temperature and the associated Appendix G P/T limit. The DCPP LTOP system requires only a single constant lift setpoint and administrative controls on mass injection capability and RCP operation to ensure acceptable margin as the RCS temperature decreases.

4. The structural integrity of the PORV and the associated discharge piping are assured throughout the LTOP range based on safety and relief valve testing performed in accordance with NUREG-0737. The testing demonstrated the ability of the PORVs to mitigate cold overpressurization events (Ref. NRC Safety Evaluation Report: TMI Action NUREG-0737 (II.D.1), "Relief and Safety Valve Testing for Diablo Canyon Units 1 and 2," dated January 27, 1986). This is different from the approach specified in WCAP 14040-NP-A which was based on a generic study by Westinghouse for COMS (variable pressure setpoint system) using a type of PORV which would cause maximum back pressure in the piping during an overpressure transient. DCPP LTOP system is not a COMS system and the DCPP PORVs are not of this limiting type.

5. The neutron fluence calculated at the ¼ t and ¾ t locations by PG&E is in accordance with the guidance given in Regulatory Guide (RG) 1.99, Rev. 2. That is, the fluence attenuation is from the wetted vessel surface, versus from the clad/base metal interface specified by WCAP-14040-NP-A.

6. The existing heatup/cooldown curves are based on the "best-estimate" methodology of WCAP-14040-NP-A. However, the ongoing in-vessel and ex-vessel reactor vessel (RV) radiation surveillance programs, as well as the generation of future heatup/cooldown curves (after 16 EFPY) will utilize projected fluences based on pure transport theory rather than the "best-estimate" methodology of WCAP-14040-NP-A. These pure transport theory projected fluences will be compared to plant specific measured dosimetry results for validation only. Measured results will not be used to modify future transport theory projections without prior NRC approval.

4.2 Changes from Previous PTLR Submittal

There are three significant changes between the previously submitted PTLR (Reference 7) and the PTLR submitted with this LAR.

1. The previous DCPP LTOP methodology was based on a Westinghouse parametric study of DCPP mass injection cases with the LOFTRAN computer code, while the heat injection and RCS undershoot results were based on DCPP evaluations of Westinghouse generic results. The proposed PTLR is based on using the approved RETRAN computer code for the complete spectrum of mass injection, heat injection, and RCS undershoot cases consistent with the WCAP-14040-NP-A methodology. PTLR Section 2.2, "Low Temperature Overpressure Protection (LTOP)

Setpoints (LCO 3.4.12)," provides a detailed discussion of the DCPP LTOP methodology.

2. The LTOP administrative temperature restrictions in PTLR Table 2.2-2 have changed slightly due to minor variations in the latest RCS overpressure results obtained with the DCPP RETRAN model as compared to the original Westinghouse LOFTRAN results. In particular, the RETRAN model uses revised emergency core cooling system (ECCS) and charging injection profiles which are explicitly based on DCPP pump characteristics, and which has resulted in reduced temperature restrictions related to securing the safety injection (SI) and charging system flow paths. Also, the RETRAN model generates a more conservative estimate of the dynamic head effects on the RCS wide range pressure transmitters, which has resulted in slightly more restrictive (greater) temperature limits for RCP operation and establishing the RCS vent.

3. The bolt up temperature, based on ASME Appendix G and 10 CFR 50 Appendix G, Table 1, is required to be the initial nil-ductility temperature (RT_{NDT}) of the flange plus any irradiation effects. The highest initial RT_{NDT} of the vessel and closure head flange materials is 53°F. The flange area is sufficiently distant from the fuel region, that the fluence has negligible affect on the RT_{NDT} of the materials in this area. Currently, the bolt up temperature of 70°F is based on the value given in the original Combustion Engineering (CE) instruction manual for the RV. The proposed curves set the temperature at 60°F based on the Westinghouse WCAP-14040-NP-A position of Section 2.7 and correspondence from CE that upgraded the original instruction manual in conformance with ASME Code requirements (Reference 10). A copy of Reference 10 is included in Enclosure 6 for information.

In addition, to provide the option in the future, separate curves could be developed, if warranted, for Unit 1 and Unit 2. Currently, the composite curves are common to both units as the most limiting adjusted reference temperature (ART) between the units at the ¼ t and ¾ t locations is utilized for the stress intensity factor, K_{Ia}, calculation (K_{Ia} is defined under the response to Provision 5, below). Should the ART difference between the units justify separate P/T curves, those curves will be generated using the same methodologies described herein.

4.3 Methodology and PTLR

To relocate the P/T curves and LTOP system limits from the TS, GL 96-03 requires the licensee to; (1) reference a methodology approved by the NRC for deriving the parameters used for constructing the curves and setpoints, (2) develop a PTLR or a similar document, and (3) make appropriate changes to the applicable sections of the TS.

The first two of the three GL 96-03 requirements for relocating the P/T curves and LTOP system limits are to use an NRC-approved methodology, and to develop a PTLR. GL 96-03, Attachment 1 contains seven provisions for PTLR methodology from the administrative controls section of the STS. The following information explains how PG&E meets each of these seven provisions.

Provision 1: "The methodology shall describe how the neutron fluence is calculated (reference new regulatory guide when it is issued)."

The existing heatup/cooldown curves for DCPP Units 1 and 2 are based on the Westinghouse "best-estimate" methodology for calculation of neutron fluence, whereby pure transport fluences are modified with both in-vessel and ex-vessel measured dosimetry results. This methodology is as described in WCAP-14040-NP-A, and as implemented in WCAP-14284, Revision 0, "Pacific Gas and Electric Company Reactor Cavity Neutron Measurement Program for Diablo Canyon Unit 1 - Cycles 1 through 6," dated January 1995, and WCAP-14350, Revision 0, "Pacific Gas and Electric Company Reactor Cavity Neutron Measurement Program for Diablo Canyon Unit 2 - Cycles 1 through 6," dated November 1995.

The neutron fluence calculations for DCPP were carried out using forward and adjoint formulations in r, θ geometry of the two-dimensional Discrete Ordinates Transport (DORT) code. The anisotropic scattering was treated with a P_3 expansion of the scattering cross section and the angular discretization was modeled with an S_8 order of angular quadrature. The core power distribution and neutron source distribution were estimated conservatively, accounting for spectral changes due to plutonium accumulation. The methodology used the BUGLE-93 cross section library which is based on the data set of the Evaluated Nuclear Data File/B-VI (ENDF/B-VI).

The fast neutron fluence is calculated for any depth into the vessel wall in conformance with RG 1.99, Revision 2, as follows:

$$f = f_{surface} [\exp(-0.24x)]$$

where $f_{surface}$ is the calculated value of the neutron fluence at the inner wetted surface of the vessel at the location of the postulated defect and x (in inches) is the depth into the vessel wall measured from this surface.

WCAP 14040-NP-A methodology differs from this approach as it calculates fluence attenuation from the clad/base metal interface. Also, Westinghouse typically only provides a clad/base metal interface fluence ($f_{c/bm}$) in their surveillance capsule and cavity dosimetry reports. To resolve this difference, DCPP vessel wetted surface fluence, f_S, is calculated per Reference 11 as follows:

$$f_S = 1.029 \ f_{c/bm}$$

A copy of Reference 11 is included in Enclosure 6 for information.

As discussed above, the existing heatup/cooldown curves are based on the "best-estimate" methodology of WCAP-14040-NP-A. However, the ongoing in-vessel and ex-vessel RV radiation surveillance programs, as well as the generation of future heatup/cooldown curves (after 16 EFPY) will utilize projected fluences based on pure transport theory rather than the "best-estimate" methodology of WCAP-14040-NP-A. These pure transport theory projected fluences will be compared to plant specific measured dosimetry results for validation only. Measured results will not be used to modify future transport theory projections without prior NRC approval. The latest pure transport theory methodology for DCPP, as described in WCAP-15423, Revision 0 (Unit 2), WCAP-15780, Revision 0 (Unit 1), and WCAP-15782, Revision 0 (Unit 2), follows the methods outlined in RG 1.190. All DCPP transport calculations are currently carried out using DORT Version 3.1 and the BUGLE-96 cross-section library. In these calculations, anisotropic scattering is treated with a P_5 legendre expansion and the angular discretization is modeled with an S_{16} order of angular quadrature.

Provision 2 "The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves."

The RV surveillance program is designed to monitor radiation effects on RV materials under actual operating conditions. The radiation effects are determined from changes in fracture toughness of the material obtained by pre- and post-irradiation testing of vessel material specimens removed from the surveillance capsules. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," requires that the surveillance program satisfy ASTM Standard E-185 which specifies material selection, material testing, specimen sizes and specimen quantities.

The DCPP Units 1 and 2 RV surveillance programs are in compliance with 10 CFR Part 50, Appendix H. The material test requirement and the acceptance standard utilize the nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the RV steel is developed in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Protection Against Non-Ductile Failure." The surveillance capsule removal schedule is presented in Final Safety Analysis Update (FSARU) Table 5.2-22 and meets the requirements of ASTM E185-70 (Unit 1) and E185-73 (Unit 2).

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

The Unit 1 surveillance program is described in WCAP-8465, "Pacific Gas and Electric Co. Diablo Canyon Unit No. 1 Reactor Vessel Radiation Surveillance Program," and WCAP-13440, "Supplemental Reactor Vessel Radiation Surveillance Program for the [PG&E Co.] Diablo Canyon Unit No. 1." The Unit 2 surveillance program is described in WCAP-8783, "[PG&E Co.] Diablo Canyon Unit No. 2 Reactor Vessel Radiation Surveillance Program." The withdrawal schedules are in accordance with ASTM E185, and have been reviewed and accepted by the NRC (References 12 and 13).

The current surveillance capsule reports are as follows:

1. WCAP-14284, "Pacific Gas and Electric Company Reactor Cavity Neutron Measurement Program for Diablo Canyon Unit 1 – Cycles 1 through 6," January, 1995.
    a. WCAP-11567, "Pacific Gas and Electric Company Analysis of Capsule S From Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," December, 1987.
    b. WCAP-13750, "Pacific Gas and Electric Company Analysis of Capsule Y From Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," July, 1993.
2. WCAP-14350, "Pacific Gas and Electric Company Reactor Cavity Neutron Measurement Program for Diablo Canyon Unit 2 – Cycles 1 through 6," November, 1995.
    a. WCAP-11851, "Pacific Gas and Electric Company Analysis of Capsule U From Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," May, 1988.

b. WCAP-12811, "Pacific Gas and Electric Company Analysis of Capsule X From Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," December, 1990.

c. WCAP-14363, "Pacific Gas and Electric Company Analysis of Capsule Y From Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," August, 1995.

PG&E letter DCL-01-004, "Reactor Vessel Material Surveillance Program Capsule V Technical Report," dated January 12, 2001, transmitted Westinghouse technical report, WCAP-15423, Revision 0, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," dated September 2000. PG&E stated in DCL-01-004 that: (1) the results of the Capsule V specimen testing show that the limiting vessel beltline plate and weld material are behaving in accordance with previous predictions; (2) the results do not indicate any changes needed to the LTOP setpoints or P/T curves currently approved to 16 EFPY; and (3) that the PTLR will be updated with the Capsule V data upon approval of the PTLR methodology.

Provision 3: "Low temperature overpressure protection (LTOP) system limits developed using NRC-approved methodologies may be included in the PTLR."

The DCPP LTOP system limits are determined in Calculation STA-138, "RETRAN Evaluation of DCPP LTOP Parameters," which is included in Enclosure 5 for information, and summarized in the following discussion.

The DCPP LTOP system protects the RCS from overpressure transients that could occur at low operating temperatures such as during plant startups and shutdowns. The LTOP system consists of two mutually redundant and independent systems, which can each open a PORV as described in FSARU Section 5.2.2.4. The LTOP system will maintain the RCS pressure below the ASME Section III, Appendix G isothermal pressure-temperature limit curve in accordance with ASME Code Case N-514. The LTOP system is manually enabled by reactor operators via a switch on the main control board prior to decreasing the RCS temperature below the predetermined enable temperature. The LTOP system then automatically engages the PORV LTOP lift setpoint when the RCS temperature reaches the enable temperature. The LTOP lift setpoint and enable temperature are established using the COMS methodology in the Westinghouse WCAP-14040-NP-A. This COMS methodology has been approved by the NRC and was evaluated in the applicable safety evaluation report dated October 16, 1995, to satisfy the requirements of Standard Review Plan (SRP) Section 5.2.2 and Branch

Technical Position RSB 5-2. The LTOP enable temperature of 270°F was calculated using ASME Section XI, Code Case N-514 and found to ensure LTOP would be operational at the proper water temperature, which, consistent with Code Case N-514, is the greater of: (1) that associated with the vessel metal temperature at least 50°F above the RT_{NDT} at the limiting beltline location; or (2) 200°F. The use of ASME Code Case N-514 is described in WCAP-14040-NP-A and was approved for DCPP in LA 133/131. As discussed in Section 4.1.3, the DCPP LTOP system requires only one constant lift setpoint as compared to the variable COMS setpoints discussed in WCAP-14040-NP-A.

The design basis of the DCPP LTOP system includes both mass injection and heat injection events as documented in Calculation STA-138. These events are assumed to occur with the RCS in water solid conditions and letdown isolated, so that the RCS pressure rapidly increases to the PORV actuation setpoint. The RCS pressure continues increasing even after the lift setpoint is reached and until the PORV has sufficiently opened so that the relief capacity equals the RCS 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 delay time, opening time and flow characteristics. The DCPP LTOP setpoints incorporate an appropriate conservative instrumentation delay and PORV opening time to ensure the peak RCS overshoot does not exceed the applicable Appendix G limit as documented in Calculation STA-138.

The design basis mass injection event is defined as the 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 temperature range is established by the TS restriction to secure the SI pumps, one centrifugal charging pump and isolating all SI accumulators prior to entering the LTOP mode of operation. As the RCS temperature decreases, successive administrative controls are implemented to prevent the ECCS charging injection discharge valves from opening, and to limit the number of charging pumps that can be operated to ensure that the maximum mass injection capability remains bounded by the LTOP design basis. As the RCS temperature continues to decrease, the number of operating RCPs is administratively limited to decrease the dynamic pressure of the system as discussed in Calculation STA-138, and included in the PTLR as referenced in the Bases of TS 3.4.12. The LTOP design basis includes an administrative temperature limit for establishing an RCS vent based on determining the temperature at which the reduced Appendix G P/T limit no longer has sufficient margin to accommodate the mass injection RCS overshoot associated with the PORV response time. Conversely, during a RCS heatup, the administrative restrictions are removed as the RCS temperature increases. The LTOP administrative controls are implemented

within the Operating Procedures (OP) L-1, "Plant Heatup From Cold Shutdown to Hot Standby, OP L-5, "Plant Cooldown From Minimum Load to Cold Shutdown, and OP L-6, " Refueling."

The design basis heat injection event is described in Calculation STA-138. It is defined as the starting of the first RCP during water solid conditions with a maximum allowable temperature difference of 50°F between the RCS and the SGs as allowed by TSs 3.4.6 and 3.4.7, while the residual heat removal system is isolated. This results in a sudden heat input from the SGs and causes a pressure increase transient due to the thermal expansion of the water solid RCS. Since the residual heat removal (RHR) system is isolated, the RHR relief valves are not available. 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.

In addition to the spectrum of heat injection cases specified in WCAP-14040-NP-A, the DCPP LTOP design basis also includes an analysis which establishes that there are no RCS/SG temperature restrictions for starting a RCP when the pressurizer level indicates less than or equal to 50 percent.

This additional heat injection case evaluated the additional expansion volume provided by the reduced pressurizer level with respect to the maximum RCS/SG temperature difference allowed within the operating procedures guidelines. The SGs temperature can only exceed the RCS during a plant shutdown when the steam generator heat removal lags behind that of the RCS. The steam generators must remain slightly cooler than the RCS since they are the primary source for RCS heat removal down to a temperature of 350°F, at which time the RHR system could be placed into service. The DCPP operating procedures and chemistry guidelines then maintain at least one RCP operating for complete mixing and a uniform cooldown of the RCS and steam generators until the temperature has decreased below 160°F. At this time, all RCPs could be secured and additional RCS cooling could be performed with only RHR flow. Depending on the RHR flow conditions, the main RCS vessel and loop volumes could be cooled down while the steam generator tube bundle secondary liquid volumes remain at the higher temperature.

A worst case operational scenario for LTOP consideration assumes that the steam generator liquid volumes remain at 160°F, while the RCS has been cooled down to the minimum temperature at which an RCS vent must be established. A assumed RCS vent temperature of 50°F is well below the lowest expected value, and creates a potential RCS/SG temperature differential of 110°F. Assuming the maximum temperature measurement

uncertainties of 15°F, starting an RCP at this time would represent a LTOP heat injection event with a net RCS/SG temperature differential of 140°F. Since heat injection events become more severe with increasing RCS temperature, a LTOP heat injection analysis was performed assuming the RCS temperature was at the maximum LTOP value of 270°F, the SG temperature was 150°F greater, and the pressurizer was at 50 percent indicated narrow range level. The heat injection analysis showed that the additional expansion volume provided by the reduced pressurizer level ensures that even these bounding LTOP RCS/SG conditions will not challenge the Appendix G P/T limits.

The major function of the LTOP system is to protect the structural integrity of the reactor pressure vessel from brittle fracture at low temperatures. In order to achieve this purpose, P/T limits established in accordance with the requirements of Appendix G to 10 CFR 50 and ASME Code Case N-514 are considered as the upper limits for the RCS during postulated transient conditions. However, since the overpressure events most likely occur at isothermal conditions in the RCS, the steady state Appendix G limits are used for the design of the LTOP system.

The structural integrity of the PORV and the associated discharge piping are assured throughout the LTOP range based on safety and relief valve testing performed in accordance with NUREG-0737. The testing demonstrated the ability of the PORVs to mitigate cold overpressurization events (Reference: NRC Safety Evaluation Report: TMI Action NUREG-0737 (II.D.1), "Relief and Safety Valve Testing for Diablo Canyon Units 1 and 2," dated January 27, 1986). This is different from the approach specified in WCAP 14040-NP-A, which was based on a generic study by Westinghouse for COMS (variable pressure setpoint system) using a type of PORV, which would cause maximum back pressure in the piping during an overpressure transient. DCPP LTOP system is not a COMS system and the DCPP PORVs are not of this limiting type.

The DCPP LTOP setpoints are established in accordance with the ASME Code Case N-514 as provided in LAs 133/131 for Unit 1 and Unit 2, respectively. The LTOP methodology provides adequate protection for the RV integrity and maintains proper operating margins. In establishing the LTOP setpoints, the DCPP specific plant parameters and transient conditions listed in Section 3.2.1 of WCAP-14040-NP-A are considered. This includes the initial volume and fluid conditions for the RCS and SGs, the PORV opening and relief characteristics, the mass input and heat input to the RCS, and the pressure limits which protect the DCPP RV. The LTOP setpoint evaluation also includes the RCP startup dynamics, and the dynamic and static pressure difference between the limiting RV weld elevation and the LTOP pressure transmitters. These LTOP analysis assumptions, including

those listed in the PTLR, may be revised based on changes to the applicable P/T limits, changes in plant equipment, or changes in operating strategy as long as they remain consistent with the approved LTOP methodology.

A RETRAN computer model of DCPP is used to evaluate the DCPP plant response during the various LTOP transient conditions. The RETRAN code is a thermal hydraulic computer code developed by the Electric Power Research Institute (EPRI) for analyzing plant transients. PG&E previously received NRC approval (Reference LAs 108/107 dated October 1, 1995) for use of the RETRAN code in analyzing the loss of load event which was submitted as part of LAR 95-06 (Reference PG&E Letter DCL-95-220, dated September 30, 1995) to revise the main steam safety valve setpoint tolerance. The DCPP LTOP analysis model used in Calculation STA-138 is developed directly from this NRC approved RETRAN model of DCPP. In addition, this RETRAN analysis of the LTOP event has been verified to be consistent with the applicable restrictions and conditions of the latest RETRAN SER which includes the original Mod 4 SER condition responses provided by DCPP in Attachment E.1 to LAR 95-06.

The RETRAN computer code model of DCPP is benchmarked to verify it generates comparable results to the specialized LOFTRAN model used in WCAP-14040-NP-A. The two LTOP PORVs have the same lift pressure setpoint, such that if one fails the other is individually capable of mitigating an overpressure event. The opening of both PORVs simultaneously is also considered for verifying that an excessive pressure undershoot condition would not challenge the RCP number one seal performance criteria. Consistent with WCAP-14040 Section 3.2.2, when there is insufficient range between the upper (ASME overpressure) and lower (RCP seal undershoot) pressure limits, the DCPP LTOP methodology establishes the precedence for selecting the LTOP lift setpoint to provide protection against the upper limit.

The uncertainties in the pressure and temperature instrumentation used by the LTOP system are explicitly accounted for in the determination of the LTOP setpoints and operating margins. These instrumentation uncertainties are determined using the process described in ISA Standard S67.04.01-2000.

Provision 4: "The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for irradiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2."

As described in DCPP FSARU, Section 5.2.4.1, due to the fabrication dates of the DCPP RVs, the Charpy impact test orientation for the vessel plate material was in the longitudinal direction. However, full Charpy test curves in

the transverse direction were subsequently obtained for the intermediate and lower shell course plates of both vessels. The transverse C_v data given in FSARU Tables 5.2-17A & 5.2-17B are either the results of the available data, or estimated by adding 20°F to the longitudinal C_v data, per the guidance provided in Branch Technical Position MTEB 5-2. The initial RT_{NDT} is then taken to be the higher of either the measured T_{NDT} or (T_{Cv} - 60°F), per the guidance of ASME Code, Section III, NB-2331 and WCAP-14040-NP-A.

As identified in FSARU Tables 5.2-21A & 5.2-21B, the initial RT_{NDT} was not determined by testing for all beltline weld material. Where not available, a value of –56°F is used per the guidance provided in 10 CFR 50.61. The use of generic values is allowed by RG 1.99 and the specific value has been reviewed and approved by the NRC in NRC Letter dated June 28, 1996, "Diablo Canyon 1: Assessment Of Diablo Canyon Surveillance Material For Issuance Of Revision 1 Of The Reactor Vessel Integrity Database."

The method for calculating ART is performed as described in WCAP-14040-NP-A, which conforms with RG 1.99, Revision 2. The ART is calculated by adding the initial nil-ductility transition reference temperature of the unirradiated material (IRT_{NDT}), the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin to account for uncertainties as follows:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + margin$$

The determination of ΔRT_{NDT} due to irradiation conforms to RG 1.99, Revision 2, and is calculated as follows:

$$\Delta RT_{NDT} = CF \times f^{**}(0.28 - 0.10 \log f)$$

where CF is the chemistry factor and f is the neutron fluence at a specific depth calculated as described in the Provision 1 discussion. The CF is taken from RG 1.99, Revision 2, based on the copper and nickel content of the vessel material. Alternatively, the CF is calculated using credible surveillance data.

The margin included in the ART calculation, in conformance with RG 1.99, Revision 2, is included to account for uncertainties in the values of IRT_{NDT}, copper and nickel contents, fluence and the calculation procedures. The margin is calculated as follows:

$$Margin = 2 \times [\sigma_i^{**}2 + \sigma_\Delta^{**}2]^{**}1/2$$

where σ_i is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_i is estimated from the precision of the test method. For generic mean values, σ_i is the standard deviation

from the set of data used to establish the mean. The ΔRT_{NDT} standard deviation, σ_Δ, is 28°F for welds and 17°F for base metal in accordance with RG 1.99, Revision 2, except that σ_Δ need not exceed half the mean value of ΔRT_{NDT}. σ_Δ is reduced by half when credible surveillance data are used.

Provision 5: "The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800, SRP Section 5.3.2, Pressure-Temperature Limits."

PG&E used linear elastic fracture mechanics from ASME Section III, Appendix G, and ASME Section XI, Appendix G, for calculating the P/T limits. The method is based on restricting the stress intensity factor of the postulated defect to be less than the reference stress intensity factor of the RV material, K_{IR}. This factor is denoted as K_{Ia} in ASME Section XI equations and in WCAP-14040–NP–A and will be used going forward. The K_{Ia} is determined by the metal temperature and RT_{NDT} at the tip of the postulated flaw. The flaw is assumed to have a depth of one-fourth of the beltline thickness and a length of 1.5 times the beltline thickness. The K_{Ia} curve in the ASME Code is given as follows:

$$K_{Ia} = 26.78 + 1.223 \times \exp[0.0145(T - RT_{NDT} + 160)]$$

where T is the metal temperature and RT_{NDT} is the ART value of the limiting vessel material at the ¼ t and ¾ t locations of the vessel wall. The stress intensity factor caused by the postulated crack is limited to the reference stress intensity factor of the vessel material as follows:

$$C \times K_{IM} + K_{IT} < K_{Ia}$$

where K_{IM} is the stress intensity factor caused by pressure (membrane) stress, K_{IT} is the stress intensity factor caused by the thermal stress and C is a safety factor that is 2 for the heatup and cooldown, and 1.5 for the hydrostatic and leak test conditions when the reactor is not critical.

Equations used in determination of K_{IT} are in accordance with Westinghouse Electric Corporation Proprietary computer program "OPERLIM" which was verified by Westinghouse in the requests for additional information contained in WCAP-14040-NP-A. The K_{IT} solution is used in combination with K_{Ia} determined above to solve for the limiting K_{IM} as follows:

$$K_{IM}(max) = (K_{Ia} - K_{IT})/2$$

Using the methodology from ASME Section XI, Appendix A Article A-3000, K_{IM} calculated above is related to the specific vessel geometry and the postulated flaw size by the following equation:

$$K_{IM} = (\sigma_m M_m + \sigma_b M_b)\sqrt{\pi a/Q}$$

where σ_m is the membrane stress and σ_b is the bending stress in psi, "a" is 0.5 times the axis of the elliptical flaw, Q is the flaw shape parameter, M_m and M_b are the correction factor for membrane stress and bending stress, respectively, from Section XI Appendix A Figures A-3300-3 and 3300-5. The equation for the flaw shape factor, Q, is from EPRI NP-1181 "Computational Method to Perform the Flaw Evaluation Procedure as Specified in ASME Code, Section XI, Appendix A", September, 1979, and for a/t = 0.25 and a/l = 0.167 (l is the major axis of the flaw) is as follows:

$$Q = 1.2404 - 0.212\frac{(|\sigma_m| + |\sigma_b|)^2}{\sigma_{ys}^{\ 2}}$$

where σ_{ys} is the minimum specified yield strength for ASME SA-533, the vessel beltline material.

This ASME Section XI Appendix A and EPRI NP-1181 based methodology for determining the flaw shape factor is different from that used in WCAP-14040-NP-A, Revision 2, and SRP Section 5.3.2. This difference is approved and discussed in LAs 133/131 SE, Section 3.2.2.

The general equation for hoop stress due to internal pressure in a thick wall pressure vessel:

P = σ [(Ro**2 - Ri**2)/(Ri**2 + Ro**2)]

is applied in an iterative process to calculate the allowable pressure. The steady state, cooldown and heatup P/T curves are determined using this process. For steady state, K_{IT} is zero and K_{Ia} is determined at the ¼ t location. For cooldown, K_{IT} and K_{Ia} are determined at the ¼ t location. The P/T curve at ¼ t is compared with the steady-state curve. The allowable pressure for cooldown is determined by the lesser of the two values, and the resulting curve is the composite cooldown limit curve. For heatup, K_{IT} and K_{Ia} are determined at the ¼ t and ¾ t locations. The P/T curves at ¼ t, ¾ t, and steady-state are compared. The lowest of the three for each heatup rate is used to generate the composite heatup limit curve. The composite cooldown limit curve and composite heatup limit curve provide the allowable operating range for operation. The composite curves are common to both units as the most limiting ART_{NDT} between the Units at the ¼ t and ¾ t locations is utilized for the K_{Ia} calculation. Should the ART_{NDT} difference between the units justify separate P/T curves, those curves will be generated using the same methodologies described herein. This composite curve construction is in accordance with The WCAP-14040-NP-A methodology.

Provision 6: "The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature curves."

10 CFR Part 50, Appendix G, imposes a minimum temperature at the closure flange based on the reference temperature for the flange material. With the core not critical; (1) when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. This "flange notch" has been incorporated into the P/T curves; (2) When the pressure is less than 20 percent of the pre-service system hydrotest pressure, the temperature of the closure flange region's highly stressed by the bolt preload must be greater than the reference temperature for the flange material. This value therefore becomes the minimum bolt up temperature

The bolt up temperature is based on ASME Appendix G, and RG 1.99 that require the bolt up temperature to be the initial RT_{NDT} of the flange plus any irradiation effects. The flux exposed in the RV flange and RV head flange result in negligible RT_{NDT} shift, and, thus minimum bolt up temperature does not change with time. The boltup temperature of 70°F in the previously provided PTLR is based on the value given in the original CE Instruction Manual for the Reactor Vessel. The highest flange RT_{NDT}, calculated in accordance with Branch Technical Position MTEB 5-2, between DCPP Unit 1 and 2 is 53°F (Unit 1 RV closure head flange). The proposed curves set the temperature at 60°F based on the Westinghouse WCAP 14040-NP-A position of Section 2.7 and correspondence from CE (Reference 10). Between the minimum bolt up temperature and the minimum LTOP operating temperature (72°F), a 2.07 square inch opening is relied on for RCS venting when the RV head is bolted up.

When the core is critical the minimum temperature for the RV, in accordance with 10 CFR 50, Appendix G, is the larger of (1) the minimum permissible temperature for the in-service hydrostatic pressure test, or (2) the limiting flange RT_{NDT} +160°F. These minimum temperature requirements are reflected in the DCPP P/T heatup and cooldown curves.

Provision 7: "Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value (2 σ_Δ) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + 2 σ_Δ), the

licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology."

Multiple surveillance capsules have been removed and evaluated for both units in compliance with 10 CFR Part 50, Appendix H. The PTLR presents the measured and predicted 30 ft-lb transition temperature shift (ΔRT_{NDT}) for the plate and weld surveillance materials for all capsules, both credible and non-credible. Currently, the PTLR surveillance capsule data meets the required $2\sigma_\Delta$ between measured and predicted ΔRT_{NDT} as required by this provision for credible surveillance data.

The credibility of each capsule's surveillance data has been evaluated in accordance with the five criteria of RG 1.99, Revision 2, Section B. Unit 1 does not have two or more sets of either credible surveillance plate or weld data at this time. Therefore, the requirement of this provision currently does not apply to Unit 1. For Unit 2, three sets of credible data are presented in the PTLR for both the surveillance plate and weld materials. The absolute difference between the measured and predicted ΔRT_{NDT} was less than the plate $2\sigma_\Delta$ (34°F) for each of the plate samples and less than the weld $2\sigma_\Delta$ (56°F) for each of the three surveillance weld samples. Therefore, measured surveillance data is currently consistent with the proposed methodology.

<u>Technical Specifications</u>

GL 96-03 requires three separate actions to modify the plant TS:

- *Action (1) "Definitions" - the addition of the definition of a named formal report (PTLR or a similar document) that would contain the explanations, figures, values, and parameters derived in accordance with an NRC-approved methodology and consistent with all of the design assumptions and stress limits for cyclic operation;*

  DCPP Unit 1 and Unit 2 TS contain the following definition in Section 1.1, "Definitions:"

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

This definition meets the requirements of GL 96-03 and is being revised in accordance with TSTF-419, to delete the last sentence which states in part; "Plant operation within...(LTOP) System."

- *Action (2) LCOs - the addition of references to the PTLR noting that the P/T limits shall be maintained within the limits specified in the PTLR,*

  DCPP Unit 1 and Unit 2 TS contain the following:

  LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

  LCO 3.4.12 An LTOP System shall be OPERABLE with no safety injection pumps and a maximum of one centrifugal charging pump capable of injecting into the RCS and the accumulators isolated and one of the following pressure relief capabilities:

  a. Two Class I power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

  b. The RCS depressurized and an RCS vent of ≥ 2.07 square inches.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, Low Temperature Overpressure Protection (LTOP) arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and

2. Specification 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

These TS LCOs/requirements meet the requirements of GL 96-03.

- *Action (3) "Administrative Controls" - the addition of a reporting requirement to submit the PTLR to the NRC, when it is issued, for each reactor vessel fluence period.*

  TS 5.6.6.c states "The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto."

  This TS requirement meets the requirements of GL 96-03. However changes to the references to TS 5.6.6.b, as proposed under Section 2.0 of this enclosure, are needed to complete relocation of the P/T limit curves and LTOP system limits from TS to the PTLR in accordance with GL 96-03.

Conclusion

The proposed methodology, and PTLR, and the existing TS as modified by the proposed changes, meet the provisions required by GL 96-03, for relocation of the P/T limit curves and LTOP system limits from TS to the PTLR. The proposed changes also meet the provisions of TSTF-419.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

1. *Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?*

Response: No

The proposed Technical Specification (TS) changes provide the reference for the NRC approved methodology for the Diablo Canyon Power Plant (DCPP) Pressure And Temperature Limits Report (PTLR). The TS and PTLR were developed using the guidance of NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996, which provides guidance on relocating reactor coolant system (RCS) pressure/temperature (P/T) limit curves and low-temperature overpressure (LTOP) system limits from TS to a PTLR. NRC approval of the DCPP specific application of the PTLR methodology will allow PG&E to use the approved PTLR methodology in the future to calculate new P/T and LTOP limits without prior NRC staff approval.

The proposed PTLR was developed using methodology previously approved by the NRC, primarily WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," dated January 1996. PG&E has evaluated this methodology and concludes it is applicable for use at DCPP. As a result, use of this methodology does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?*

Response: No

The proposed change completes relocation of the RCS P/T and LTOP limits from the TS to the PTLR. The DCPP PTLR submitted with this amendment has been developed primarily using the NRC-approved methodology of WCAP-14040-NP-A, Revision 2.

The proposed change makes no changes to plant equipment, and does not physically alter or change the function of any structures, systems or components that could initiate an accident. Through the PTLR, it provides operational controls to assure that current RCS P/T and LTOP limits are not violated. It provides for use of NRC-approved methodology for changing the RCS P/T and LTOP limits in the future without requiring prior NRC approval. As a result, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. *Do the proposed changes involve a significant reduction in a margin of safety?*

Response: No

The proposed change completes relocation of the RCS P/T and LTOP limits from the TS to the PTLR, and submits the DCPP PTLR methodology for NRC approval. The DCPP PTLR submitted with this amendment has been developed using the methodology of WCAP-14040-NP-A, Revision 2, which has previously been approved by the NRC.

The proposed change makes no changes to plant equipment, and does not physically alter or change the function of any structures, systems or components that could affect any margin of safety. Through the PTLR, it provides operational controls to assure that current RCS P/T and LTOP limits are not violated. It provides for use of NRC approved methodology for changing the RCS P/T and LTOP limits in the future without requiring prior NRC approval. As a result, the proposed change has no affect on any margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluations, PG&E concludes that the activities associated with the above described change present no significant hazards consideration under the standards set forth in 10 CFR 50.92 and accordingly, a finding by the NRC of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

In LA 135/135 for both DCPP Units 1 and 2, the NRC staff approved conversion of the DCPP TS to Improved TS. As part of this conversion, the RCS P/T and LTOP system limits were relocated from the TS to the DCPP PTLR. The SE for LA 135/135 stated that the limits addressed in the PTLR of TS 5.6.6 are the limits that the NRC staff previously approved in LA 133/131 dated May 3, 1999. These LAs approved P/T limit curves that are valid for 16 EFPY. The SE for LA 135/135 further stated that the NRC staff will review PG&E's plant-specific application of the PTLR methodology to allow PG&E to use the PTLR methodology in the future to calculate new P/T and LTOP limits without prior NRC approval.

The application of the PTLR methodology submitted for approval by this LAR was developed in accordance with the requirements of GL 96-03. The technical analysis presented in Section 4.0 above describes how the PTLR and associated TS meet the provisions of GL 96-03, including use of an NRC-approved methodology, specifically WCAP-14040-NP-A, Revision 2.

Submittal of this LAR completes a commitment made in PG&E letter DCL-00-070 in which PG&E stated it would resubmit the PTLR after performing a new analysis consolidating the supporting calculations into a more cohesive and comprehensive calculation. PG&E stated, In the meantime, it would operate DCPP within the existing RCS P/T limit curves approved in LA 133/131.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

PG&E has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the

proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7.0 REFERENCES

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

2. NRC Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

3. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements."

4. Code of Federal Regulations, Title 10, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

5. NRC letter to PG&E, "Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit 1 and Unit 2," dated May 3, 1999. (License Amendments 133/131)

6. NRC letter to PG&E, "Conversion to Improved Technical Specifications for Diablo Canyon Power Plant, Units 1 and 2 – Amendment No. 135 to Facility Operating License Nos. DPR-80 and DPR-82," dated May 28, 1999.

7. PG&E Letter to NRC, DCL-99-146, "Request for NRC Approval of Diablo Canyon Methodology for Establishing Pressure/ Temperature and Low Temperature Overpressure Protection Limits Using WCAP-14040-NP-A in Accordance with Generic Letter 96-03," dated November 24, 1999.

8. PG&E Letter to NRC, DCL-00-039, "Supplement to PG&E's Request for NRC Approval of Diablo Canyon Methodology for Establishing Pressure/ Temperature and Low Temperature Overpressure Protection Limits Using WCAP-14040-NP-A in Accordance with Generic Letter 96-03," dated March 16, 2000.

9. PG&E letter to NRC, DCL-00-070, "Supplement to Reactor Coolant System Pressure and Temperature Limits Report," dated April 26, 2000.

10. CE Power Systems letter to Westinghouse Nuclear Energy Systems dated September 12, 1979, transmitted to PG&E by Westinghouse letter

PGE-4083 dated July 21, 1980 as amended by Westinghouse letter PGE-6352 dated December 20, 1984.

11. Westinghouse letter PGE-88-765 dated December 14, 1988.

12. NRC Letter dated September 4, 1992, "Evaluation of Diablo Canyon Unit 1 Supplemental Reactor Vessel Radiation Surveillance Program (TAC No. M83285)."

13. NRC Letter dated February 10, 1998, "Pacific Gas & Electric Company's Revision to the Reactor Vessel Surveillance Capsule Withdrawal Schedule for Diablo Canyon Unit No. 2 (TAC No. M99917)."

14. NRC Letter from William D. Beckner to Anthony R. Pietrangelo, NEI, dated March 21, 2002.

Proposed Technical Specification Changes

Remove	Insert
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1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. ~~Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt for each unit.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature and LTOP limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Insert 1

~~1. NRC Letter from NRC to Gregory M. Rueger dated May 28, 1999~~

~~2. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with:~~

~~10 CFR 50, Appendix G and H~~
~~Regulatory Guide 1.99, Revision 2~~
~~NUREG-0800, Standard Review Plan Section 5.3.2~~
~~Branch Technical Position MTEB 5-2~~
~~ASME B&PV Code Section III, Appendix G~~
~~ASME B&PV Code, Section XI, Appendix A~~
~~WCAP-14040-NP-A, Section 2.2~~

~~3. LTOP limits (Power Operated Relief Valves (PORV) pressure relief setpoint and LTOP enable temperature) were developed in accordance with:~~

~~NUREG-0800, Standard Review Plan Section 5.2.2~~
~~Branch Technical Position RSB 5-2~~
~~10 CFR 50, Appendix G and H~~
~~Regulatory Guide 1.99, Revision 2~~
~~Branch Technical Position MTEB 5-2~~
~~WCAP-14040-NP-A, Section 2.2~~

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not Used

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not Used

~~135~~
~~135~~

(continued)

Insert 1 for TS 5.6.6

1. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

Revised Technical Specification Pages

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt for each unit.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

b. The analytical methods used to determine the RCS pressure and temperature and LTOP limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP 14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not Used

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not Used

(continued)

Diablo Canyon PTLR

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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

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. A maximum heatup of 60°F in any 1-hour period.

b. A maximum cooldown of 100°F in any 1-hour period.

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

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

2.1.1 RCS P/T Limits:

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

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

10CFR50 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 70deg to 550deg range for various heat up and cool down rates. The stress intensities are dependent on geometry and temperature change rate and are not affected by embrittlement. Thus, the Westinghouse provided values remain valid throughout Plant life.

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

2.1.2 Reactor Vessel Bolt-up Temperature:

Operating Restrictions illustrated on the P-T curve also include Reactor Flange Boltup Temperature. This is based on ASME Appendix G and 10CFR50 Appendix G that require the Bolt Up temperature to be the initial RTndt of the flange plus any irradiation effects. The flux exposed in the R.V. Flange and R.V. Head Flange result in negligible RTndt shift, and, thus minimum Bolt Up Temperature does not change with time. The highest flange RTndt between DCPP Unit 1 and 2 is 53deg F (Unit 1 R.V. closure head). The curves conservatively set the temperature at 60 deg F based on WCAP 14040-NP-A minimum temperature. Between the minimum bolt up temperature and the minimum LTOP operating temperature (72 deg F), a 2.07 sq. in. opening is relied on for RCS venting.

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

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

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

2.2.1 LTOP Enable Setpoints:

The LTOP lift setpoint and arming temperature are based on the methodology established in the Westinghouse WCAP - 14040 - NP - A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996. The lift setpoint is 435 psig based on limiting the maximum RCS pressure overshoot to a value below the Appendix G P/T curve and limiting the minimum RCS undershoot to maintain a nominal operating pressure drop across the number one RCP seal. The arming temperature setpoint is 200°F or RT_{NDT} + 50°F which ever is greater in accordance with ASME Code Case N-514.

2.2.2 RCS Pressure Overshoot:

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

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

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

2.2.3 LTOP Mass Injection Case:

The LTOP mass injection analysis is based on an inadvertent initiation of the maximum injection flow capability for the applicable Mode of operation into a water solid RCS with letdown isolated. The initial mass injection capability within the LTOP range is established by Tech Spec. 3.4.12 restriction to secure the safety injection (SI) pumps and one centrifugal charging pump (CCP), and isolate all SI Accumulators 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 CCP injecting through the SI injection flowpath and the positive displacement pump (PDP) injecting through the normal and the alternate charging flowpaths simultaneously. The administrative temperature limit for operating with a maximum of one charging pump is based on a mass injection case with one CCP 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.

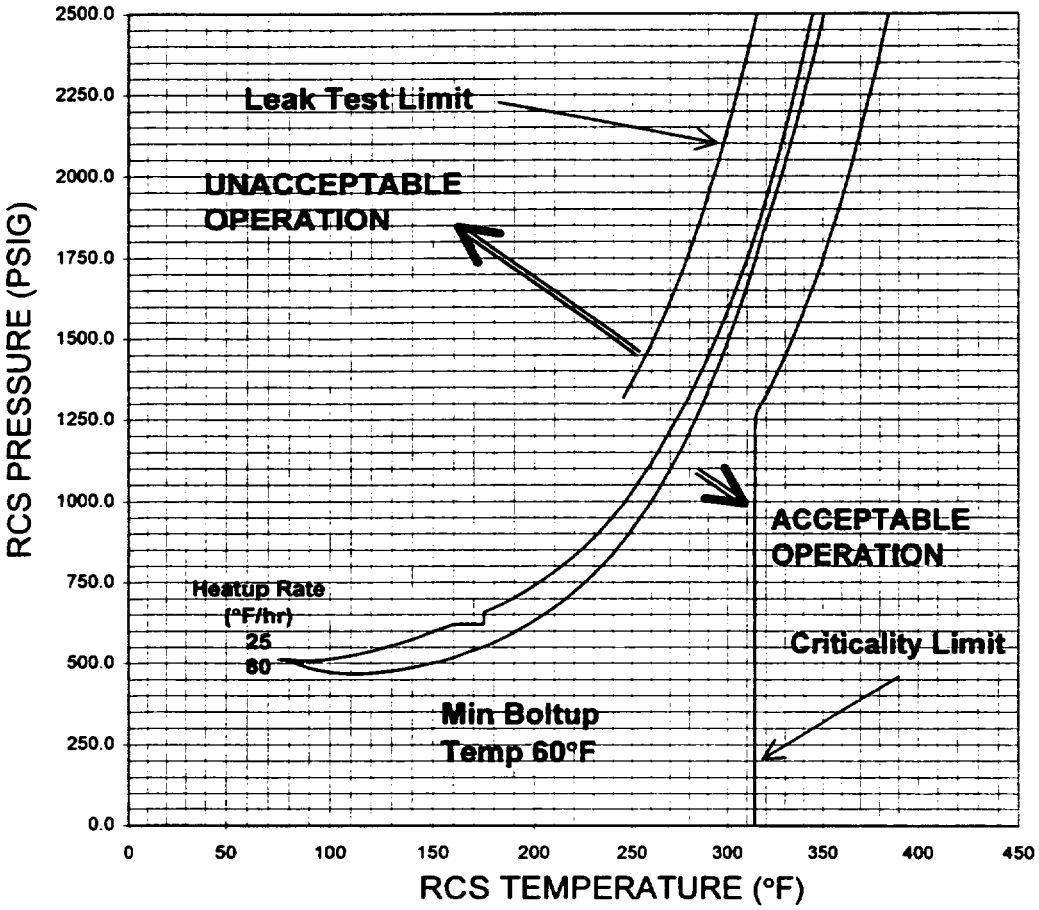

FIGURE 2.1-1 Diablo Canyon Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable to 16 EFPY (Without Margins for Instrumentation Errors)

TABLE 2.1-1

Diablo Canyon Heatup Data at 16 EFPY
Without 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	510.15	75	510.15				
80	513.50	80	513.50				
85	517.11	85	517.11				
90	520.98	90	514.36				
95	525.15	95	506.57				
100	529.63	100	500.99				
105	534.45	105	497.82				
110	539.63	110	496.27				
115	545.19	115	496.41				
120	551.18	120	497.84				
125	557.61	125	500.63				
130	564.53	130	504.51				
135	571.97	135	509.52				
140	579.96	140	515.53				
145	588.56	145	522.59				
150	597.80	150	530.56				
155	607.73	155	539.56				
160	618.40	160	549.54				
161.1	621.0						
165	621.0	165	560.56				
170	621.0	170	572.59				
173	621.0						
173	650.2						
175	655.48	175	585.74				
180	669.74	180	600.01				
185	685.07	185	615.52				
190	701.54	190	632.26				
195	719.25	195	650.37				
200	738.28	200	669.91				
205	758.73	205	690.99				
210	780.71	210	713.68				
215	804.34	215	738.13				
220	829.73	220	764.42				
225	857.01	225	792.71				
230	886.33	230	823.13				
235	917.83	235	855.82				

TABLE 2.1-1

Diablo Canyon Heatup Data at 16 EFPY
Without Margins for Instrumentation Errors

25° F/hr		60°F/hr		60°F/hr Crit. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
240	951.68	240	890.92				
245	988.04	245	928.66			245	1313.55
250	1027.10	250	969.17			250	1365.16
255	1069.05	255	1012.68			255	1420.55
260	1114.11	260	1059.36			260	1479.99
265	1162.49	265	1109.48			265	1543.76
270	1214.44	270	1163.26			270	1612.16
275	1266.63	275	1220.93	315	1220.93	275	1685.50
280	1321.05	280	1282.77	320	1282.77	280	1764.12
285	1379.42	285	1349.08	325	1349.08	285	1848.36
290	1442.01	290	1420.15	330	1420.15	290	1938.58
295	1509.11	295	1484.66	335	1484.66	295	2035.17
300	1581.04	300	1547.80	340	1547.80	300	2138.51
305	1658.10	305	1615.38	345	1615.38	305	2249.01
310	1740.65	310	1687.76	350	1687.76	310	2367.09
315	1829.05	315	1765.22	355	1765.22	315	2493.16
320	1923.67	320	1848.14	360	1848.14	320	2627.63
325	2024.86	325	1936.80	365	1936.80	325	2770.93
330	2133.08	330	2031.65	370	2031.65	330	2923.46
335	2248.68	335	2132.94	375	2132.94	335	3085.60
340	2372.13	340	2241.22	380	2241.22		
345	2503.81	345	2356.72	385	2356.72		
350	2644.18	350	2479.95	390	2479.95		
355	2793.59	355	2611.26	395	2611.26		
360	2952.49	360	2751.09	400	2751.09		
365	3121.20	365	2899.78	405	2899.78		
370	3300.09	370	3057.72	410	3057.72		
375	3489.35	375	3225.27	415	3225.27		

Calc. N-NCM-97010

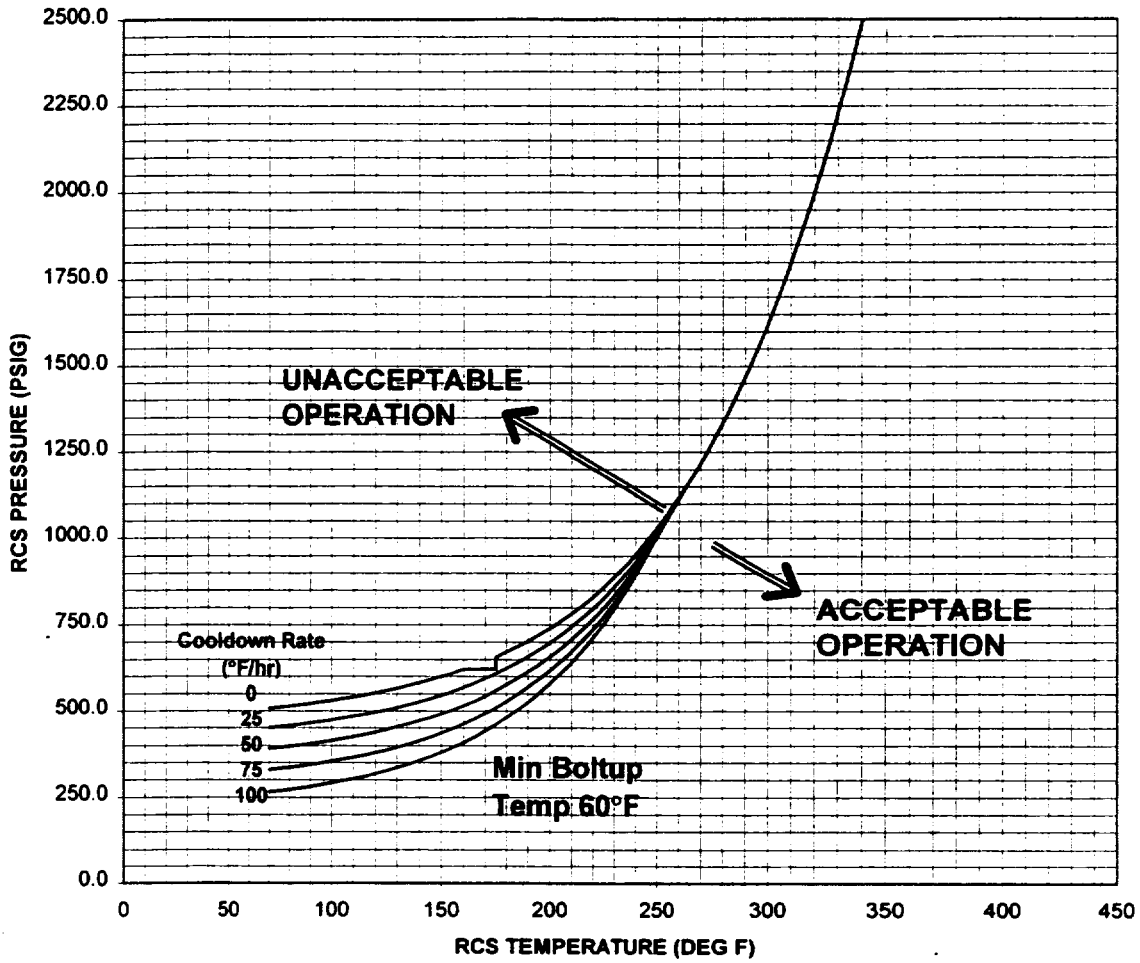

FIGURE 2.1-2 Diablo Canyon Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50, 75 and 100°F/hr) Applicable to 16 EFPY (Without Margins for Instrumentation Errors)

TABLE 2.1-2

Diablo Canyon Cooldown Data at 16 EFPY
Without Margins for Instrumentation Errors

Steady State		25°F/hr		50°F/hr		75°F/hr		100°F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
350	2787.30	350	2787.30	350	2787.30	350	2787.30	350	2787.30
345	2633.00	345	2633.00	345	2633.00	345	2633.00	345	2633.00
340	2488.11	340	2488.11	340	2488.11	340	2488.11	340	2488.11
335	2352.19	335	2352.19	335	2352.19	335	2352.19	335	2352.19
330	2224.83	330	2224.83	330	2224.83	330	2224.83	330	2224.83
325	2105.58	325	2105.58	325	2105.58	325	2105.58	325	2105.58
320	1994.03	320	1994.03	320	1994.03	320	1994.03	320	1994.03
315	1889.74	315	1889.74	315	1889.74	315	1889.74	315	1889.74
310	1792.30	310	1792.30	310	1792.30	310	1792.30	310	1792.30
305	1701.30	305	1701.30	305	1701.30	305	1701.30	305	1701.30
300	1616.37	300	1616.37	300	1616.37	300	1616.37	300	1616.37
295	1537.13	295	1537.13	295	1537.13	295	1537.13	295	1537.13
290	1463.22	290	1463.22	290	1463.22	290	1463.22	290	1463.22
285	1394.31	285	1394.31	285	1394.31	285	1394.31	285	1394.31
280	1330.07	280	1330.07	280	1330.07	280	1330.07	280	1330.07
275	1270.22	275	1270.22	275	1270.22	275	1270.22	275	1270.22
270	1214.44	270	1214.44	270	1214.44	270	1214.44	270	1214.44
265	1162.49	265	1162.20	265	1162.49	265	1162.49	265	1162.49
260	1114.11	260	1109.14	260	1109.76	260	1114.11	260	1114.11
255	1069.05	255	1058.79	255	1054.90	255	1057.27	255	1067.57
250	1027.10	250	1012.89	250	1003.87	250	1000.67	250	1004.66
245	988.04	245	970.00	245	956.45	245	948.10	245	946.25
240	951.68	240	930.26	240	912.34	240	899.24	240	891.96
235	917.83	235	892.57	235	871.38	235	853.90	235	841.61
230	886.33	230	858.23	230	833.29	230	811.77	230	794.83
225	857.01	225	826.13	225	797.94	225	772.69	225	751.48
220	829.73	220	796.36	220	765.07	220	736.39	220	711.22
215	804.34	215	768.60	215	734.58	215	702.74	215	673.93
210	780.71	210	742.65	210	706.23	210	671.49	210	639.32
205	758.73	205	718.65	205	679.95	205	642.55	205	607.29
200	738.28	200	696.51	200	655.52	200	615.67	200	577.57
195	719.25	195	675.93	195	632.88	195	590.79	195	550.08
190	701.54	190	656.26	190	611.84	190	567.69	190	524.59
185	685.07	185	638.52	185	592.35	185	546.33	185	501.04

TABLE 2.1-2

Diablo Canyon Cooldown Data at 16 EFPY
Without Margins for Instrumentation Errors

Steady State		25°F/hr		50°F/hr		75°F/hr		100°F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
180	669.74	180	622.04	180	574.25	180	526.51	180	479.21
175	655.48	175	606.73	175	557.48	175	508.18	175	459.06
173	650.18								
173	621.00								
170	621.00	170	592.34	170	541.91	170	491.18	170	440.38
165	621.00	165	578.88	165	527.50	165	475.48	165	423.16
160	618.40	160	566.64	160	514.13	160	460.92	160	407.21
155	607.73	155	555.11	155	501.76	155	447.49	155	392.53
150	597.80	150	544.57	150	490.27	150	435.04	150	378.93
145	588.56	145	534.59	145	479.67	145	423.56	145	366.43
140	579.96	140	525.35	140	469.82	140	412.93	140	354.87
135	571.97	135	516.95	135	460.73	135	403.15	135	344.25
130	564.53	130	509.15	130	452.30	130	394.09	130	334.44
125	557.61	125	501.75	125	444.53	125	385.77	125	325.46
120	551.18	120	494.92	120	437.32	120	378.06	120	317.16
115	545.19	115	488.72	115	430.69	115	371.00	115	309.57
110	539.63	110	482.81	110	424.54	110	364.46	110	302.57
105	534.45	105	477.50	105	418.88	105	358.49	105	296.20
100	529.63	100	472.58	100	413.64	100	352.96	100	290.32
95	525.15	95	467.76	95	408.83	95	347.92	95	284.99
90	520.98	90	463.56	90	404.38	90	343.27	90	280.09
85	517.11	85	459.68	85	400.30	85	339.04	85	275.65
80	513.50	80	455.91	80	396.53	80	335.14	80	271.59
75	510.15	75	452.53	75	393.09	75	331.61	75	267.92
70	507.03	70	449.30	70	389.90	70	328.36	70	264.57

Calc. N-NCM-97010

Table 2.2-1

Low Temperature Over-Pressure (LTOP)

System Setpoints

Function	Setpoint
PORV Arming Temperature(1)	270 °F
PORV Pressure Setpoint(2)	435 psig

(1) Calc. N-NCM-97011, Rev. 0

(2) STA-138, Rev. 0

Table 2.2-2

Low Temperature Over-Pressure (LTOP)

Temperature Restrictions

Restriction	Setpoint
SI Pumps Secured, 1 CCP Secured, SI Accumulators Isolated	≤ 270 °F
Safety Injection Flowpath Blocked, and SI Blocked	≤ 153 °F
2 of 3 Charging Pumps Secured	≤ 139 °F
1 of 4 RCPs Secured	≤ 131 °F
2 of 4 RCPs Secured	≤ 115 °F
3 of 4 RCPs Secured	≤ 101 °F
4 of 4 RCPs Secured	≤ 91 °F
RCS Vent Path of 2.07 in^2 Established	≤ 72 °F

Calc. STA-138, Rev. 0

Assumptions: 1) PORV Stroke Time of 2.9 seconds.

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

3. ADDITIONAL CONSIDERATIONS

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

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

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

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

4. REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

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

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

The programs are described in the following:

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

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

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

The surveillance capsule reports are as follows:

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

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

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

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

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

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

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

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

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

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

5. REACTOR VESSEL SURVEILLANCE DATA CREDIBILITY

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

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

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

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

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

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

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

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

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

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

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

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

The Diablo Canyon Unit 1 Surveillance Capsule S for the Intermediate Shell Plate B4106-3 and Surveillance Weld Heat 27204 both show scatter in excess of the Criterion 3 allowable values. The Diablo Canyon limiting CF values are based upon the CF Tables 1 and 2 of 10CFR50.61 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 may be utilized.

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

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

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

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

Table 5.0-1
Diablo Canyon Unit 1 Surveillance Capsule Data

Material	Capsule	CF[a]	FF	Best Fit ΔRT_{NDT}[b]	Measured ΔRT_{NDT}[c]	Scatter in ΔRT_{NDT}
Inter Shell Plate B4106-3	S[d]	32.2	0.675	21.7	-2.0	23.7
Inter Shell Plate B4106-3	Y		1.006	32.4	46.9	-14.5
Surveillance Weld Heat 27204	S[d]	211.2	0.675	142.6	110.0	32.6
Surveillance Weld Heat 27204	Y		1.006	212.5	234.3	-21.8

WCAP 13771

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

(b) Best fit ΔRT_{NDT} = CF * FF.

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

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

Table 5.0-2
Diablo Canyon Unit 2 Surveillance Capsule Data

Material	Capsule	CF[a]	FF	Best Fit ΔRT_{NDT}[b]	Measured ΔRT_{NDT}[c]	Scatter in ΔRT_{NDT}
Inter Shell Plate B5454-1 (Long)	U	102.5	0.716	73.4	65.9	7.5
Inter Shell Plate B5454-1 (Long)	X		0.960	98.4	101.0	-2.6
Inter Shell Plate B5454-1 (Long)	Y		1.08	110.7	113.0	-2.3
Inter Shell Plate B5454-1 (Trans)	U	102.5	0.716	73.4	72.3	1.1
Inter Shell Plate B5454-1 (Trans)	X		0.960	98.4	98.9	-0.5
Inter Shell Plate B5454-1 (Trans)	Y		1.08	110.7	110.7	0.0
Surveillance Weld	U	211.7	0.716	151.6	173.8	-22.2
Surveillance Weld	X		0.960	203.2	204.2	-1.0
Surveillance Weld	Y		1.08	228.6	212.5	16.1

WCAP-14364

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

(b) Best fit ΔRT_{NDT} = CF * FF.

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

6. SUPPLEMENTAL DATA TABLES

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

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

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

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

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

Table 6.0-6 DCPP-1 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t and ¾t Locations at 16 EFPY

Table 6.0-7 DCPP-2 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t and ¾t Locations at 16 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 16 EFPY

Table 6.0-9 Diablo Canyon Unit 2 Adjusted Reference Temperatures (ARTs) for the Reactor Vessel Beltline Materials at the ¼t and ¾t Locations for 16 EFPY

Table 6.0-10 Calculation of Adjusted Reference Temperature at 16 EFPY for the Limiting Diablo Canyon Reactor Vessel Materials

7. PRESSURIZED THERMAL SHOCK (PTS) SCREENING

10 CFR 50.61 requires that RT_{PTS} be determined for each of the vessel beltline materials. The RT_{PTS} is required to meet the PTS screening criterion of 270°F for plates, forgings, and axial weld material, and 300°F for circumferential weld material. If the screening criterion is not met, specific actions taken to either meet the screening criterion or prevent potential reactor vessel failure as a result of PTS require review and approval of the NRC. The maximum projected RT_{PTS} for Units 1 and 2 is 259°F (Unit 1 Weld 3-442c), therefore, at a projected 32 EFPY at EOL, the PTS screening criteria is met. The PTS evaluations are described in the following reports:

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

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

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.

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 ($X 10^{19}$ n/cm^2)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted (°F) [a]	Measured (°F) [b]	Predicted (%) [a]	Measured (%) [b] [c]
Plate B4106-3	S	0.305	35	-2	14	0
	Y	1.02	52	47	19	3 (10)
Surveillance Weld Metal	S	0.305	150	110	26	11
	Y	1.02	224	234	34	33 (39)
Heat Affected Zone Metal	S	0.305	--	77	--	15
	Y	1.02	--	84	--	26 (26)
Correlation Monitor Plate HSST 02	S	0.305	68	66	18	2
	Y	1.02	102	112	23	2 (10)

WCAP-13750

[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 by EPRI Hyperbolic Tangent Curve Fitting Routine, Revision 2.0.

[c] Values in parenthesis are based on the definition of upper shelf energy given in ASTM E185-82.

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 ($X 10^{19}$ 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	0.357	74.0	65.9	18	14.4
	X	0.866	99.2	101.0	22	20.7
	Y	1.320	111.3	113.0	24	18.4
Plate B5454-1 (Transverse)	U	0.357	74.0	72.3	18	0.4
	X	0.866	99.2	98.9	22	10.3
	Y	1.320	111.3	110.7	24	6.6
Surveillance Weld Metal	U	0.357	150.9	173.8	28	29.7
	X	0.866	202.3	204.2	34	38.5
	Y	1.320	226.9	212.5	38	36.1
Heat Affected Zone Metal	U	0.357	--	234.2	--	40.3
	X	0.866	--	253.5	--	31.4
	Y	1.320	--	255.3	--	37.4

WCAP-14363

[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 by EPRI Hyperbolic Tangent Curve Fitting Routine, Revision 2.0.

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

Unit 1 - Material	Capsule	F[a]	FF[b]	Measured ΔRT_{NDT}[d]	$FF \times \Delta RT_{NDT}$	FF^2
Intermediate Shell Plate B4106-3	S[c]	0.305	0.675	-2	0	0.456
	Y	1.020	1.006	46.9	47.2	1.012
				SUM	47.2	1.468
	$CF_{Plate} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (47.2°F) \div (1.468) = 32.2°F$ [c]					
Weld Metal	S[c]	0.305	0.675	110	74.3	0.456
	Y	1.020	1.006	234.3	235.7	1.012
				SUM	310.0	1.468
	$CF_{weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (310.0) \div (1.468) = 211.2°F$ [c]					
Unit 2 - Material	**Capsule**	**F[a]**	**FF[b]**	**Measured ΔRT_{NDT}[d]**	**$FF \times \Delta RT_{NDT}$**	**FF^2**
Intermediate Shell Plate B5454-1 (Long)	U	0.357	0.716	65.9	47.2	0.513
	X	0.866	0.960	101.0	97.0	0.922
	Y	1.320	1.080	113.0	121.7	1.160
Intermediate Shell Plate B5454-1 (Transverse)	U	0.357	0.716	72.3	51.8	0.513
	X	0.866	0.960	98.9	94.9	0.922
	Y	1.320	1.080	110.7	119.2	1.160
				SUM	531.8	5.190
	$CF_{Plate} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (531.8°F) \div (5.19) = 102.5°F$					
Weld Metal	U	0.357	0.716	173.8	124.4	0.513
	X	0.866	0.960	204.2	196.0	0.922
	Y	1.320	1.080	212.5	228.9	1.160
				SUM	549.3	2.600
	$CF_{Weld} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2) = (549.3°F) \div (2.600) = 211.7°F$					

WCAP-13771 (Unit 1) WCAP-14364 (Unit 2)

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

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

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

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

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

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

Calc N-NCM-97009

[a] Per CE NPSD-1039, Rev 2

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

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

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

Calc N-NCM-97009

[a] Per CE NSPD-1039, Rev. 2

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

TABLE 6.0-6

DCPP-1 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t, and ¾t Locations at 16 EFPY

Material	Fluence f_s	Fluence $f_{c/bm}$	Fluence $f_{¼t}$	Fluence $f_{¾t}$
Upper Shell Plate[a]	1.59 E + 17	1.54 E + 17	9.00 E + 16	3.20 E + 16
B4105-1	1.59 E + 17	1.54 E + 17	9.00 E + 16	3.20 E + 16
B4105-2	1.59 E + 17	1.54 E + 17	9.00 E + 16	3.20 E + 16
B4105-3				
Inter Shell Plate				
B4106-1	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18
B4106-2	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18
B4106-3	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18
Lower Shell Plate				
B4107-1	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18
B4107-2	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18
B4107-3	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18
Upper Shell Long[a] Welds 1-442 A,B,C	1.59 E + 17	1.54 E + 17	9.00 E + 16	3.20 E + 16
Upper Shell to Inter Shell Weld 8-442[a]	1.59 E + 17	1.54 E + 17	9.00 E + 16	3.20 E + 16
Inter Shell Long				
Welds 2-442 A,B	5.31 E + 18	5.16 E + 18	3.00 E + 18	1.07 E + 18
Weld 2-442 C	2.74 E + 18	2.66 E + 18	1.55 E + 18	5.50 E + 17
Inter Shell to Lower Shell Weld 9-442	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18
Lower Shell Long				
Welds 3-442 A,B	4.24 E + 18	4.12 E + 18	2.40 E + 18	8.50 E + 17
Weld 3-442 C	7.68 E + 18	7.46 E + 18	4.34 E + 18	1.54 E + 18

Calc N-NCM-97009, Calc. 921130-0

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

TABLE 6.0-7

DCPP-2 Summary of the Projected Peak Pressure Vessel Neutron Fluence Values at the Vessel Surface, Clad to Base Metal Interface, ¼t and ¾t Locations at 16 EFPY

Material	Fluence f_s	Fluence $f_{c/bm}$	Fluence $f_{¼t}$	Fluence $f_{¾t}$
Upper Shell Plate[a]				
B5453-1	1.57 E + 17	1.53 E + 17	8.90 E + 16	3.10 E + 16
B5453-3	1.57 E + 17	1.53 E + 17	8.90 E + 16	3.10 E + 16
B5011-1R	1.57 E + 17	1.53 E + 17	8.90 E + 16	3.10 E + 16
Inter Shell Plate				
B5454-1	7.58 E + 18	7.37 E + 18	4.29 E + 18	1.52 E + 18
B5454-2	7.58 E + 18	7.37 E + 18	4.29 E + 18	1.52 E + 18
B5454-3	7.58 E + 18	7.37 E + 18	4.29 E + 18	1.52 E + 18
Lower Shell Plate				
B5455-1	7.58 E + 18	7.37 E + 18	4.29 E + 18	1.52 E + 18
B5455-2	7.58 E + 18	7.37 E + 18	4.29 E + 18	1.52 E + 18
B5455-3	7.58 E + 18	7.37 E + 18	4.29 E + 18	1.52 E + 18
Upper Shell Long[a] Welds 1-201 A,B,C	1.57 E + 17	1.53 E + 17	8.90 E + 16	3.10 E + 16
Upper Shell to Inter Shell Weld 8-201[a]	1.57 E + 17	1.53 E + 17	8.90 E + 16	3.10 E + 16
Inter Shell Long				
Weld 2-201 A	3.81 E + 18	3.70 E + 18	2.15 E + 18	7.70 E + 17
Welds[b] 2-201 B, C	7.79 E + 18	7.57 E + 18	4.40 E + 18	1.56 E + 18
Inter Shell to Lower Shell Weld 9-201	7.58 E + 18	7.37 E + 18	4.29 E + 18	1.52 E + 18
Lower Shell Long				
Welds[b] 3-201 A,C	7.79 E + 18	7.57 E + 18	4.40 E + 18	1.56 E + 18
Weld 3-201 B	3.81 E + 18	3.70 E + 18	2.15 E + 18	7.70 E + 17

Calc N-NCM-97009, Calc. 921130-0

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

[b] Fluence unreduced by neutron pads.

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 16 EFPY

Material	16 EFPY ART[a]		
	RG 1.99 Rev. 2 Method	¼t (°F)	¾t (°F)
Upper Shell Plate[d]			
B4105-1	Position 1.1	71.4	66.2
B4105-2	Position 1.1	52.4	47.2
B4105-3	Position 1.1	59.5	53.0
Inter Shell Plate			
B4106-1	Position 1.1	89.5	67.4
B4106-2	Position 1.1	93.2	72.2
B4106-3	Position 1.1	120.5	106.2
Lower Shell Plate			
B4107-1	Position 1.1	118.0	94.7
B4107-2	Position 1.1	117.1	95.8
B4107-3	Position 1.1	74.5	53.4
Upper Shell Long[d] Welds 1-442 A,B,C	Position 1.1	24.0	0.7
Upper Shell to Inter[d] Shell Weld 8-442	Position 1.1	3.7	-11.3
Inter Shell Long			
Welds 2-442 A,B	Position 1.1	161.5	107.2
Weld 2-442 C	Position 1.1	125.2	79.4
Inter Shell to Lower Shell Weld 9-442	Position 1.1	141.8	97.1
Lower Shell Long			
Welds 3-442 A,B	Position 1.1	148.7	96.9
Weld 3-442 C[c]	Position 1.1	183.7[b]	124.9

Calc N-NCM-97009 & Calc. N-282

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

(b) This ART value is used to generate the heatup and cooldown curves.

(c) DCPP-1 Surveillance Capsule S was not judged "credible" per 10CFR50.61. The higher chemistry values of CE NPSD-1039, Rev 2 for this heat are used to generate the heatup and cooldown Appendix G curves.

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

TABLE 6.0-9
Diablo Canyon Unit 2 Adjusted Reference Temperatures (ARTs) for the Reactor Vessel Beltline Materials at the ¼t and ¾t Locations for 16 EFPY

Material	16 EFPY ART[a]		
	RG 1.99 Rev. 2 Method	¼t (°F)	¾t (°F)
Upper Shell Plate[c]			
B5453-1	Position 1.1	43.1	35.0
B5453-3	Position 1.1	47.3	42.6
B5011-1R	Position 1.1	42.4	37.7
Inter Shell Plate			
B5454-1	Position 2.1	147.4	120.8
B5454-2	Position 1.1	177.2	151.4[b]
B5454-3	Position 1.1	151.5	122.9
Lower Shell Plate			
B5455-1	Position 1.1	94.1	68.7
B5455-2	Position 1.1	109.1	83.7
B5455-3	Position 1.1	98.9	82.0
Upper Shell Long[c]			
Welds 1-201 A,B,C	Position 2.1	-7.2	-30.1
Upper Shell to Inter[c]			
Shell Weld 8-201	Position 1.1	-0.4	-13.0
Inter Shell Long			
Weld 2-201 A	Position 2.1	102.2	55.6
Welds 2-201 B, C	Position 2.1	141.4	86.3
Inter Shell to Lower Shell Weld 9-201	Position 1.1	12.8	-0.7
Lower Shell Long			
Welds 3-201 A,C	Position 1.1	107.0	74.1
Weld 3-201 B	Position 1.1	83.6	55.8

Calc N-NCM-97009 & Calc N-282

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

(b) This ART value is used to generate the heatup and cooldown curves.

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

TABLE 6.0-10
Calculation of Adjusted Reference Temperature at 16 EFPY for the Limiting Diablo Canyon Reactor Vessel Materials

Parameter	ART Value	
Location	¼t[d]	¾t[e]
Chemistry Factor, CF (°F)	226.8[f]	99.6
Fluence ÷ 10^{19} n/cm^2 (E > 1.0 MeV), f[a]	0.434	0.152
Fluence Factor, FF[b]	0.768	0.5058
ΔRT_{NDT} = CF x FF, (°F)	174.2[f]	50.4
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	183.7[f]	151.4

Calc N-NCM-97009

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

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

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

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

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

(f) DCPP-1 Surveillance Capsule S was not judged "credible" per 10CFR50.61. The higher chemistry value of CE NPSD-1039, Rev 2 for this heat are used to generate the heatup and cooldown Appendix G curves.

Calculation STA-138

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1. PURPOSE

This calculation uses the RETRAN-02 Mod4 (RETRAN) computer code (Ref. 17) model of DCPP to evaluate the acceptability of the current LTOP setpoints for the applicable range of heat input and mass input cases consistent with the NRC approved LTOP methodology established in WCAP 14040 (Ref. 1). This calculation will establish the bases for the DCPP LTOP setpoint methodology in the Pressure Temperature Limits Report (PTLR) submittal to the NRC.

2. BACKGROUND

In the process of answering NRC questions on the proposed LTOP evaluation methodology for the PTLR it became evident that DCPP needed to update several LTOP related evaluations to ensure consistency with the applicable Reference 1 Westinghouse methodology in WCAP 14040.

This calcnote forms the technical basis for the DCPP LTOP methodology within the PTLR license amendment. Therefore, it has been structured such that the main body contains the information and level of detail appropriate for direct compilation into a LAR, while the explicit details and documentation necessary for an independent technical verification have been provided in STA-145 (Ref. 23).

The RETRAN code is a versatile thermal hydraulic computer code developed by EPRI for the purpose of analyzing various PWR and BWR transients. The NRC has reviewed the RETRAN code and issued an SER approving it for analyzing certain transients as delineated in NRC regulations. DCPP has already received NRC approval for use of the RETRAN code in analyzing the Loss of Load (LOL) event (calcnote N-098, Ref. 2) which was submitted as part of the License Amendment Request (LAR) 95-06 (Ref. 3) to revise the Main Steam Safety Valve (MSSV) setpoint tolerance. This analysis uses the same DCPP RETRAN model and code version which has already received NRC acceptance for use in analyzing the Loss of External Electrical Load and/or Turbine Trip (LOL/TT) event (calcnote N-098, Ref. 2). In addition, this RETRAN analysis of the LTOP event has been verified to be consistent with the applicable restrictions and conditions of the latest RETRAN SER which includes the original Mod 4 SER condition responses provided by DCPP in Attachment E.1 to LAR 95-06 (Ref. 3) This calculation will establish the bases for the DCPP LTOP setpoint methodology in the Pressure Temperature Limits Report (PTLR) submittal to the NRC.

3. ASSUMPTIONS

To ensure conservative results consistent with the WCAP 14040 (Ref. 1) methodology, the following conservative assumptions are used in this calculation.

3.1. General LTOP Methodology

1. The RCS including the pressurizer is initially at steady state water solid conditions.
2. There is no credit or modeling of any RCS metal expansion or heat transfer during the LTOP pressure transient.
3. Since the water inertia reduces the flow at the beginning of the mass injection LTOP event, it is conservative to neglect the inertia effect. The charging flow is determined by steady state flow calculation.

4. The pressurizer PORV flow is minimized by assuming the Pressurizer Relief Tank is at the maximum design discharge pressure of 50 psig per section 4.3.3.5.c of Ref. 22.

5. The mass injection and heat injection cases are all evaluated assuming that two RHR pumps are operating with a conservative maximum flow of 5000 gpm each. The RHR system is modeled such that there is no net effect on RCS mass injection or heat removal but the RHR flow does maximize the dynamic pressure drop across the RCS. This is conservatively bounding since the WCAP 14040 methodology assumes that the RHR system is isolated such that RHR heat removal and relief capability is not available.

6. The maximum RCS pressure overshoot is determined based on a limiting single failure for one of the two LTOP system PORVs fails to open.

7. Since both LTOP PORVs have the same nominal lift and reset setpoints, the minimum RCS pressure undershoot is determined based on both LTOP system PORVS opening and closing at the same time.

8. The PORV LTOP setpoint pressure uncertainty is assumed to be 32 psi to bound Design Input 5.

9. The LTOP RCS wide range temperature uncertainty is assumed equal to Design Input 6 (15 °F).

10. The SG secondary temperature uncertainty is assumed 15 °F to bound Design Input 9.

11. The pressurizer level uncertainty is assumed to be 15% to bound design input 12.

12. The maximum normal injection fluid temperature is assumed to be 100 °F consistent with the Westinghouse LTOP analyses for DCPP in Ref. 9.

13. The minimum acceptable RCS undershoot is verified by ensuring the RCP volume pressure does not decrease below 235 psig. This is based on the minimum 200 psid required across the number one RCP seal (Ref. 6) and a VCT backpressure of 35 psig which is the maximum of the normal operating range of 15-35 psig. (Ref. 6).

14. The RCP seal pressure is best represented as the average of the RCP volume and the RCP discharge or Cold Leg Volume pressures calculated by RETRAN. This is based on the RCP seal balancing chamber pressure increase during RCP flow conditions as established in Reference 21.

3.2. Westinghouse Benchmark

1. None.

3.3. Mass Input Evaluation

1. The initial RCS pressurizer pressure is assumed to be between 300 psig and 400 psig as necessary to ensure subcooled liquid conditions at the RCS temperature and prevent cavitation code errors due to the conservatively fast start time of the RCP. The initial RCS pressure does not affect the transient since there is sufficient time to evaluate the transient response prior to reaching the PORV lift setpoint of 435 psig.

3.4. *Heat Input Evaluation*

1. The RCS is assumed to have been cooled down to a steady state condition via RHR flow such that the RCS liquid in the SG tubes, the SG tube and shell, and the SG secondary liquid are all at a higher temperature than the remainder of the RCS.
2. The temperature difference between the RCS and the SG secondary side is assumed to be at the Tech Spec maximum plus uncertainty when an RCP is started in one loop.
3. The specific heat capacity of the SG tubes is increased by a factor of 1.5 to conservatively maximize the amount of metal heat transferred to the RCS.
4. The thermal conductivity of the SG tubes is increased by a factor of 10 to conservatively maximize the effective secondary side heat transfer efficiency to the RCS.
5. The SG secondary side temperature is manually measured using a WAHL Model 392 Digital Thermometer or an instrument of equivalent accuracy.

4. DESIGN INPUT

1. The water properties are from ASME Steam Tables (Ref. 16).

2. The PORV LTOP setpoint is 435 psig per the DCPP PTLR (Ref. 7)

3. The maximum PORV stroke time is 2.9 seconds per DCPP PTLR (Ref. 7)

4. The total PORV LTOP actuation time delay is 1.5 seconds, which consists of a 1.05 seconds electronic delay, and a 0.45 second process and pneumatic delay. The process and pneumatic delay are part of the PORV stroke time acceptance criteria in Design Input 3. (Ref. 19)

5. The PORV LTOP Pressure uncertainty is 31.7 psi (Ref. 13)

6. The LTOP RCS wide range temperature uncertainty is 15 °F (Ref. 14)

7. The charging injection flow versus RCS pressure bounds a C_v of 26 for both FCV-128 and HCV-142 as established in STA-143 Rev. 0 (Ref. 5)

8. SG Wide Range Pressure Uncertainty = 33.7 psi (Ref. 12)

9. WAHL Model 392 Digital Thermometer Accuracy at 250 °F is ± 3.4 °F and at 400 °F is ± 5 °F (Ref. 18).

10. The ASME Appendix G steady state pressure limit (including a 10% limit relaxation per ASME Code Case 514) as established in PTLR Table 2.1-2 (Ref. 7). Here after in this report, the term Appendix G P/T curve limit is meant to include the 10% relaxation per ASME Case 514.

11. Not Used.

12. The pressurizer level indication uncertainty is ± 6.1% (Ref.. 20)

13. The 95% probability value for both PORVs drifting low simultaneously is 12.4 psi as established in STA-145. (Ref. 23)

5. METHODOLOGY

5.1. Develop RETRAN LTOP Model

The DCPP LTOP analysis model is developed directly from the RETRAN-02 Mod3 (RETRAN) model of DCPP which analyzed the Loss of Load (LOL) event in calcnote N-098 (Ref. 2) and was submitted as part of the License Amendment Request (LAR) 95-06 (Ref. 3) to revise the Main Steam Safety Valve (MSSV) setpoint tolerance. This RETRAN model was demonstrated to accurately model the DCPP primary to secondary thermal hydraulic behavior in benchmark comparisons to DCPP start up tests, particularly the turbine trip from full power. This RETRAN model and the associated analysis results were accepted by the NRC for licensing applications and were incorporated into the DCPP Unit 1 and Unit 2 design basis per the License Amendments (LAs) 108 and 107, respectively.

The vast majority of the Ref. 2 RETRAN LOL Model has not been changed. The changes implemented to model the LTOP analyses are considered superficial in that they do not impact the basic thermal hydraulic performance of the RETRAN LOL model. These changes can be classified into three types. The first type involves simplifying changes to eliminate unnecessary modeling options and functions which are not needed for the LTOP analysis. These include eliminating the core neutronics and heat conductors models, all reactor protection and ECCS trip functions, the non-equilibrium pressurizer model, and eliminating all secondary side volumes and components except for the steam generators. The second type of changes involve expanding the RETRAN DCPP model into four individual RCS loops and adding the components specifically required to evaluate the asymmetric LTOP mass input and heat input scenarios. The third type involves minor adjustments to the RETRAN DCPP model for benchmarking to generic Westinghouse LOFTRAN code results and for establishing the conservatively bounding analysis assumptions specific to the DCPP LTOP methodology.

5.2. RETRAN LTOP Model Benchmark

The DCPP RETRAN model is adjusted as necessary to provide a direct comparison of LTOP analysis results to comparable results obtained with the Westinghouse LOFTRAN model. The RETRAN model is used to generate RCS pressure overshoot and undershoot results for heat input and mass input scenarios. The comparisons cover a wide range of RCS conditions, heat addition rates, mass addition rates, and PORV actuation parameters. The benchmark results establish that the RETRAN model generates consistent thermal hydraulic results and is acceptable for use in evaluating the DCPP LTOP setpoints per the NRC approved methodology established in WCAP 14040 (Ref. 1).

5.3. RETRAN Evaluation of DCPP LTOP Setpoints

The RETRAN model is used to evaluate the RCS response to an appropriate range of mass input and heat events using the applicable DCPP LTOP setpoints. The mass addition scenarios assume that

charging flow is injecting into a water solid RCS when the letdown flow is inadvertently isolated. The RETRAN model is used to determine the RCS overshoot for a range of mass addition rates and for the various dynamic pressure drop effects associated with RCP and RHR pump operation.

The heat addition scenarios assume that an RCP is started in one loop, and the water solid RCS is 80 °F cooler than the steam generators (SGs) and RCS fluid within the tube bundle region. The RETRAN model is used to determine the RCS overpressure which occurs due to thermal expansion of the RCS fluid as it is heated by the SGs. Since the thermal expansion properties of water vary significantly with temperature, the heat addition scenario is evaluated over the full range of LTOP applicable temperatures. The LTOP PORV parameters evaluated for the impact on RCS overshoot include the lift setpoint, valve stroke time, flow capacity, instrument uncertainty, and electronic delays.

The RETRAN LTOP model is used to evaluate the RCS undershoot which could occur during a mass injection or heat injection event based on the DCPP PORV closing characteristics. This RCS undershoot section evaluates the minimum RCS pressure during a LTOP event with respect to maintaining an adequate operational pressure drop across the +number one RCP seal.

The RETRAN LTOP model is also used to establish that with the pressurizer level at 50% or less, an RCP may be started without temperature restrictions, since even the most limiting heat input transient will not challenge the Appendix G P/T curve limit.

5.4. LTOP Administrative Limits

These LTOP analysis results establish that the DCPP LTOP setpoints in the PTLR ensure the maximum RCS pressure overshoot for the applicable range of mass input and heat events remains below the Appendix G steady state Pressure limits as allowed per the ASME Code Case N-514. These results also establish the appropriate administrative controls to ensure that the actual DCPP operating conditions remain bounded by the range of RCS conditions, injection flow capability, and RCP operation assumed in the LTOP analyses.

6. ACCEPTANCE CRITERIA

6.1. RETRAN LTOP Model Benchmark

There are no explicit or numerical acceptance criteria for the RETRAN benchmark evaluation. However, a qualitative review shall conclude that the RETRAN DCPP model generates thermal hydraulic results consistent in both characteristic trend and magnitude compared to the available Westinghouse LOFTRAN data.

6.2. RETRAN Evaluation of DCPP LTOP Setpoints

1. The maximum RCS pressure (overshoot) shall not exceed the ASME Appendix G steady state pressure limit (including a 10% limit relaxation per ASME Code Case 514) as established in PTLR Table 2.1-2 (Ref. 7) for the applicable range of LTOP RCS temperatures, when accounting for instrument uncertainty.

2. In order to maintain appropriate operation of the number one RCP seal, the minimum RCS pressure at the RCP (undershoot) should not decrease below 235 psig as identified in assumption 13.

3. With the pressurizer level less than or equal to 50% narrow range, the worst case heat input due to the start of an RCP in one loop will not challenge the Appendix G P/T curve limit.

6.3. DCPP LTOP Administrative Limits

1. The Administrative Controls shall be implemented as necessary on RCS injection capability, RCS pressure relief capacity, and RCP operation to ensure that the actual plant configuration remains conservatively bounded with respect to the plant performance assumed in the LTOP analyses.

7. CALCULATION

7.1. Develop RETRAN LTOP Model

7.1.1. Base RETRAN LTOP Model for DCPP

NRC Approved DCPP RETRAN LOL Model

The DCPP LTOP analysis model is developed directly from the RETRAN model of DCPP, which analyzed the Loss of Load (LOL) event in calcnote N-098 (Ref. 2) and was submitted and approved per LAR 95-06 (Ref. 3) to revise the MSSV setpoint tolerance. The Ref. 2 model consisted of a single RCS loop and corresponding steam generator (SG), while the other three RCS loops were combined into one lumped RCS loop and combined SG. This methodology of using a lumped RCS loop is common in transient analysis when evaluating symmetric events. By scaling the appropriate physical and hydraulic parameters, the computer model is simplified to run faster, while preserving the accuracy of the overall thermal hydraulic plant response.

The conservative LTOP events as analyzed per WCAP-14040 (Ref. 1) involve subcooled water solid, and essentially isothermal shutdown conditions. The LTOP event is a much less complex thermal hydraulic event than the design basis LOL transient which occurs at dynamic full power conditions. Therefore, the LTOP model can be made much simpler and does not require modeling numerous options and functions which are necessary for the LOL analysis model. Unnecessary model options which were eliminated for the LTOP model include the core neutronic and heat transfer models, the two region non-equilibrium pressurizer model, all reactor protection trips and ESF actuation functions, and all plant control system models. In addition, since the primary to secondary heat transfer is either ignored or conservatively controlled in the LTOP analysis, all of the secondary volumes except the SG themselves were eliminated from the LTOP model (i.e., steam lines, main steam isolation valves, condenser, steam dumps, etc.). This base LTOP model was originally documented in PG&E calculation STA-121 (Ref. 8).

Reactor Coolant System

The LTOP analysis is not a symmetric event, and requires evaluating any combination of RCP flow in each of the four RCS loops. Therefore, the lumped RCS loop of the Ref. 2 model was expanded into three individual loops, each with a corresponding steam generator and RCP. This expansion in the LTOP base model was documented in STA-121 (Ref. 8) and is considered cosmetic since all four RCS loops have the exact same dimensions and hydraulic properties as the existing single RCS loop in the LOL model. The LTOP model also uses the exact same nodalization, physical dimensions, and hydraulic parameters for the reactor vessel volumes as in the LOL model. Figure 6-1 shows the nodalization diagram of the DCPP LTOP four loop model. Tables 6-1 and 6-2 summarize the key RCS volume and junction parameters, respectively.

Reactor Coolant Pumps

The input data tables used for modeling the RCPs include the same pump head and torque curve data for the Westinghouse Model 93A1 RCP that was used in the Ref. 2 LOL model. However, the LTOP model also included a minor change to allow initializing the model with no RCPs running and then start them as needed for the various cases analyzed. Each RCP is assigned a trip which allows it to be started as needed. An additional RETRAN pump curve data table is provided to establish the pump rpm as a function of time during the startup sequence. As discussed in the evaluation sections, an appropriately conservative pump startup time is used for the heat addition evaluations. The pump startup time does not impact the mass input evaluation results since this LTOP transient is initiated from steady state RCS flow conditions with the designated number of RCPs already running.

Pressurizer Model

As discussed earlier, the limiting LTOP events are analyzed assuming a water solid and subcooled RCS conditions such that the two region non-equilibrium RETRAN pressurizer model is not needed. In addition, the pressurizer heater and spray models including associated junctions and control systems are not needed, and were removed from the DCPP LOL RETRAN model. The LTOP model did require some additional PORV modeling options specific to the LTOP analysis. This included modifying the nominal trip setpoint, modifying the junction flow loss coefficient to generate the appropriate PORV liquid relief capacity. In addition, general data tables were added to model the PORV valve area as a function of the opening and closing times.

RHR , Charging, and RCP Seal Injection

The LTOP model includes five additional RETRAN junctions to model the RHR letdown from the Loop 4 hot leg and RHR injection to each of the four individual loop cold legs, respectively. Normal RHR injection is actually only provided to the Loops 1 and 2 cold legs. However, this nodalization allows for future model flexibility in modeling ECCS cold leg injection flow if desired. Since there is no credit for any RHR heat removal in the LTOP analysis, the RHR system is modeled to inject liquid at the same enthalpy as is removed from the letdown volume. The only significant effect related to modeling the RHR flow is the impact on the total RCS flow through the core. This establishes the maximum associated dynamic pressure drop between the LTOP pressure sensor and the peak pressure location at the bottom of the RCS vessel. The total RCS pressure drop is not

sensitive to which cold legs receive RHR injection, but is only the total flow rate. The LTOP analysis assumes a conservatively bounding RHR flow rate of 5000 gpm per pump.

The LTOP model also includes four specific RETRAN junctions for modeling the combination of normal charging, alternate charging, and RCP seal injection flow to the four RCP loop cold legs. These junctions are also used to model the charging injection flow though the ECCS injection path.

The flow rate into the RCS through the various injection junctions is established by using fill tables which specify the injection flow rate as a function of RCS pressure. The injection flow rates that are used are specifically identified and discussed for each appropriate analysis section.

Steam Generator

The only other significant RETRAN model change for the LTOP analysis is the simplification of the multiple volume two phase steam generator model used for the LOL analysis, into a homogenous single volume steam generator with a bubble rise model. As stated earlier, the LTOP analysis does not credit any primary to secondary heat transfer in the mass input cases and the heat input cases only model reverse heat transfer from the SG to the RCS during shutdown conditions. The single volume SG model and the appropriately conservative RETRAN heat transfer characteristics are consistent with the Westinghouse LTOP methodology in WCAP 14040, as established in the benchmark studies of Section 6.2.

The RETRAN nodalization and input parameters used to model the SG heat conductors are the same as was used in the Ref. 2 LOL model. Table 6-3 lists the key RETRAN input data for the SG heat conductors, which are calculated from the geometry data for the Series 51 SGs at DCPP. The Series 51 SGs have 3388 tubes each with an inner diameter of 0.775 inches and an outer tube diameter of 0.875 inches. The total conductor left and right side heat transfer surface areas and the total conductor volume are also provided in Table 6-3. These LTOP heat conductors use the Inconel 600 thermal conductivity and specific heat properties as established in the Ref. 2 LOL deck.

7.2. RETRAN LTOP Model Benchmark

The DCPP RETRAN LTOP model is benchmarked based on generating thermal hydraulic results, which are comparable to the West. LOFTRAN model used for the NRC approved methodology established in WCAP 14040 (Ref. 1).

7.2.1. Benchmark RETRAN Mass Input

7.2.1.1.RETRAN Mass Input Benchmark Model

The Ref. 9 report (PGE-88-642 dated July 7, 1988) documents an extensive Westinghouse LOFTRAN parametric study of PORV setpoints, PORV stroke times, and mass injection rates, which establish the licensing basis for the current DCPP LTOP setpoints. Table 6-4 summarizes the key LTOP input parameters used by Westinghouse to evaluate the RCS overshoot and undershoot for various mass input events in the Ref. 9 report. The DCPP LTOP model developed in the previous section was modified slightly to more closely match the Westinghouse LOFTRAN model and the

range of input assumptions such that the results of the two models can be directly compared. These input changes are summarized below.

RCS Volume

As identified in Table 6-4, Westinghouse assumed an RCS volume of 12732 ft^3 for both DCPP Units 1 and 2. However, the RETRAN LTOP model is based on the more conservative LOL model which used an RCS volume of 11700 ft^3. Since the RCS pressure response is sensitive to the total RCS volume, the RETRAN benchmark model had the RCS volume increased by about 700 ft^3 to more closely match the Westinghouse LOFTRAN model in Ref. 9. The key hydraulic parameters associated with the pressurizer and surge line have not been changed and are not impacted by this increase in volume. Since the mass injection event occurs at water solid, isothermal conditions, the RCS response is essentially a function of the total RCS volume, mass injection rate, and the PORV relief characteristics. It should be noted that a smaller RCS volume results in more limiting RCS pressure overshoot and undershoot results. Therefore, the original RETRAN RCS volume from the LOL analysis is conservatively bounding and is used for evaluating the DCPP LTOP setpoints in Section 6-4.

Pressurizer PORV

The Ref. 9 Westinghouse LTOP analyses assume that the PORV valve position changes linearly as a function of stroke time for both the open and close cycles. Combining the linear valve stroke with the valve flow coefficient (C_V) as a function of valve position, results in the normalized PORV C_V value versus valve position data used in the Westinghouse LTOP analyses and shown in Table 6-5. The RETRAN model uses these PORV flow and stroke characteristics for the benchmark comparisons. The general data tables number one and two were used in the RETRAN model to implement the PORV valve area versus time for the opening and closing sequence, respectively. The RETRAN trip numbers 8 and 9 were used to open the two PORVs, respectively, based on the specified pressure setpoint as measured in the hot leg volume 306. These trips were assigned a delay time of 1.1 seconds to match the West. analyses. It should be noted that since the Westinghouse LTOP analyses assume a limiting single failure of one PORV to open, one of the RETRAN PORV trips is assigned a very long delay time which prevents it from opening during the transient. The Westinghouse analyses assumed a constant PORV closure time of 2.0 seconds, as did the RETRAN benchmark model.

RCS Flow

West. does not model the dynamic pressure drop effects due to RCS flow in the overshoot and undershoot cases in Ref. 9 (DCPP evaluated these effects separately in a subsequent calculation). Therefore, the RCP and RHR pump trips were set to 1.0E6 seconds to ensure they did not run during the mass injection benchmark cases.

Initial RCS Conditions

The RETRAN RCS and SG volumes were set to isothermal conditions of 200 psig and 100 °F to match the Westinghouse LOFTRAN model.

7.2.2. Benchmark RETRAN Heat Input

7.2.2.1.RETRAN Heat Input Benchmark Model

The Westinghouse heat input LTOP analyses, which establish the current DCPP license basis, are based on a generic LOFTRAN plant model that has been evaluated to be conservatively bounding for the DCPP parameters. These Westinghouse heat input results are documented in the Reference 10 report. Table 6-7 summarizes the key LOFTRAN model input parameters which Westinghouse used to generate the heat input results in Ref. 10. In order to perform an appropriate benchmark comparison, the DCPP RETRAN LTOP model is modified slightly to match the generic Westinghouse generic model as summarized below.

Initial RCS Conditions

The Westinghouse heat input methodology and model are based on the starting of one RCP with the RCS liquid volume at a lower temperature than the SGs. The specific assumption is that the SG secondary liquid, SG tubes and shell, and RCS liquid inside the tubes are all at a higher temperature than the rest of the RCS. In order to model this conservative but physically unrealistic assumption, the RETRAN heat input model was modified to have a valve on the inlet and outlet of each SG tube bundle. The RETRAN model was then initialized with these valves closed such that the RCS liquid temperature within the SG tubes can be set to a value the same as the SG tubes and secondary liquid, and independently higher than the rest of the RCS. Once the RCP is started, the valves are very quickly opened (in less than 0.1 seconds) so that there is no delay in the RCS mass flow or the heat transfer to the RCS.

RCS Volume

As listed in Table 6-7, the Westinghouse generic heat input model used an RCS volume of 13,000 ft^3. The DCPP RETRAN model has an RCS volume of 11700 ft^3. Since the RCS pressure increase response is sensitive to the total RCS volume, an additional 1300 ft^3 was added to the DCPP RETRAN model. This additional volume was included in the reactor head volume. Since very little mixing occurs in this volume, there is a minimal impact on the calculated heat conduction and volumetric expansion within the RCS loops and steam generator tube volumes. In addition, the Ref. 10 report indicates that the Westinghouse study used an artificially small pressurizer volume of only 100 ft^3 to conservatively offset the LOFTRAN limitations related to modeling the pressurizer with subcooled liquid conditions. Although, RETRAN has no such limitation, the RETRAN heat input bench mark model also used a pressurizer volume of 100 ft^3 to maintain consistency with the Westinghouse model. As in the Westinghouse model, the 1700 ft^3 (1800 – 100) removed from the pressurizer volume was relocated to an inactive RCS vessel volume to minimize any impact on the RCS / SG heat transfer rate. Thus, the RETRAN upper head volume was increased by a total of 3000 ft^3 to account for these two differences.

SG Heat Transfer

Table 6-7 shows that the Westinghouse generic heat input model used a total steam generator secondary side heat transfer surface of 58,000 ft^2. As shown in Table 6-3, the DCPP RETRAN model is based on a Series 51 SG with an effective area of about 51, 500 ft^2. Therefore, the RETRAN SG heat conductor areas and volume were increased by the appropriate ratio to be

equivalent to the Westinghouse model. The LOFTRAN model assumed that the SG liquid inventory was at a conservatively maximum level and that all secondary liquid (not just the liquid near the tube bundles) was directly available for heat transfer to the RCS. The single volume SG in the RETRAN was similarly modeled. The original Westinghouse generic heat input analysis calculated a conservative RCS to SG heat transfer rate based only on the secondary free convection heat transfer coefficient in the SG. Westinghouse did not model the thermal conductance through the SG tubes, or the RCS forced convection on the inside of the tubes. The DCPP RETRAN model explicitly models all of the physical SG heat transfer phenomena. In order to more closely match the Westinghouse generic model for the benchmark study, the RETRAN SG heat transfer properties were adjusted to provide the appropriate overall heat transfer of energy from the SG into the RCS. This included increasing the volumetric heat capacity of the Inconel SG tubes by a factor of 1.5 and increasing the thermal conductivity by a factor of ten. These adjustments to the RETRAN heat transfer model effectively reproduce the Westinghouse results and conservatively bound the maximum potential secondary to RCS heat transfer. These conservative adjustments to the RETRAN heat transfer model are incorporated into the DCPP LTOP model.

Pressurizer PORV Model

The RETRAN PORV model was revised to match the Westinghouse reference PORV model parameters as shown in Table 6-7. This reference pressurizer PORV model has a valve coefficient of $C_V = 50$ gpm/ psi$^{1/2}$ which varies linearly with the valve stroke position and time. This reference PORV has 3 second opening time which consists of a 0.6 second delay and a 2.4 second valve stroke time. (Ref. 10)

7.3. RETRAN Evaluation of DCPP LTOP Setpoints

The DCPP RETRAN model has been demonstrated to effectively model both the mass input and heat input analyses consistent with the WCAP 14040 methodology. This section summarizes the DCPP RETRAN evaluation of the DCPP LTOP setpoints established in the PTLR, which ensures that they provide adequate protection for the range of mass injection and heat injection scenarios consistent with the WCAP 14040 methodology. Table 6-9 lists the current DCPP LTOP setpoints and the other key LTOP related parameters, which are used in the evaluation. As discussed in Assumption 2.1.5, all of the mass injection and heat injection cases are evaluated assuming that two RHR pumps are operating at 5000 gpm each. This conservatively maximizes the RCS dynamic pressure drop across the RCS and the resultant effect on the peak pressure at the bottom of the RCS vessel. The PORV opening and closing sequence including delays is shown in Figure 6-3.

7.3.1. RETRAN Evaluation of LTOP Mass Input Overshoot

As identified in the DCPP PTLR, the Safety Injection (SI) pumps and one Centrifugal Charging Pump (CCP) are secured prior to entering the LTOP range. However, the SI signal and charging ECCS valves are still in service. Therefore, the most limiting mass injection case in the LTOP range assumes that one CCP and the Positive Displacement Charging Pump (PDP) are injecting through the ECCS flow path and through the RCP seal injection flow to all four RCPs. These injection assumptions are conservative since a Safety Injection signal would isolate the normal charging path except for the RCP seal injection. The total mass injection flow rates as a function of the RCS pressure for this ECCS injection case are listed in Table 6-10. These flow rates were calculated in

Ref. 5 based on a conservatively bounding combination of maximum pump performance curves and minimum system line resistances. The ECCS flow and RCP seal injection flow is evenly distributed to each of the four RCS loops. The peak pressure results from this ECCS injection case establish the minimum administrative RCS temperature limit at which the charging system CCS flow path must be blocked.

Once the charging system CCS flow path is blocked, the most limiting mass injection case is one CCP and the PDP injecting simultaneously through both the normal and the alternate charging paths. This is conservative since DCPP plant procedures specify that only one charging injection path is used at a time. Table 6-11 summarizes the mass injection flow rates versus RCS pressure for the CCP and the PDP. These flow rates were calculated in Ref. 5 based on a bounding combination of maximum pump performance curves and minimum system line resistances. The RCP seal injection flow is evenly distributed to each of the four RCS loops. The alternate charging path into RCS Loop 3 has slightly less system resistance and slightly more charging flow than the normal injection path into RCS loop 4. The peak pressure results from this CCP/PDP charging injection case establish the minimum administrative RCS temperature limit at which only one CCP or one PDP at a time (but not both) can be available for RCS injection.

With only one charging pump allowed for injection, the most limiting mass injection case is one CCP injecting through the normal and the alternate charging path simultaneously. This is conservative since DCPP plant procedures specify that only one charging injection path is used at a time. Table 6-12 summarizes the mass injection flow rates versus RCS pressure for one CCP with the flow rates based on a bounding combination of a maximum CCP performance curve and minimum system line resistances. The RCP seal injection flow is evenly distributed to each of the four RCS loops. The CCP injection flow for the alternate charging path enters RCS Loop 3, while flow through the normal charging path enters RCS loop 4. The peak pressure results from this CCP charging injection case establish the minimum administrative RCS temperature limit at which all four RCPs may be operated.

The number of operating RCPs determine the dynamic pressure drop between the peak pressure which occurs at the bottom of the RCS vessel (RETRAN volume 3) and the RCS hot leg (RETRAN volume 306) where the LTOP pressure transmitter is located. Since the DCPP PORV actuation parameters (setpoint, delay time, and stroke time) are all constant throughout the LTOP range, the dynamic pressure drop translates into a direct increase in the RCS peak pressure overshoot. The mass injection flow capability versus RCS pressure remains the same as listed previously in Table 6-12. However, as the RCS temperature and the corresponding Appendix G P/T limit continue to decrease, the number of operating RCPs must be restricted to ensure the LTOP PORV parameters adequately protect the resulting peak pressure at the bottom of the RCS vessel. The next four mass injection cases evaluate the peak pressure results for one CCP charging injection with 3,2, and 1 RCPs operating, respectively. These cases then determine the minimum administrative RCS temperature limits for operating 3,2, and 1 RCPs , respectively.

Once all of the RCPs are secured, the next mass injection case evaluated is for the CCP charging injection through the normal and alternate paths with no RCS flow except that from the two RHR pumps operating. The mass injection flow capability versus RCS pressure remains the same as listed previously in Table 6-12. This case determines the minimum RCS peak pressure overshoot for the mass injection capability and LTOP actuation parameters established in the PTLR. The results for

this case then determine the minimum RCS temperature at which the LTOP PORV parameters can still maintain the peak pressure overshoot below the Appendix G P/T limit curve and subsequently the minimum administrative RCS temperature limit at which an RCS vent must be established.

The last mass injection case is evaluated to establish that with the RCS vent open, there is no credible LTOP which could challenge the Appendix G pressure limit. This case evaluates the maximum possible ECCS mass injection as assumed in Case 1 combined at the lowest credible RCS temperature of 55 °F.

7.3.2. RETRAN Evaluation of LTOP Heat Input Overshoot

The heat input cases evaluate the startup of an RCP in one loop assuming that there is a maximum allowable temperature difference between the RCS and the SG secondary side. These heat input cases are evaluated over the range of applicable RCS temperatures to ensure that the DCPP LTOP parameters established in the PTLR maintain the resulting peak pressure overshoot within the Appendix G P/T curve limits. Table 6-9 lists the current DCPP LTOP setpoints and the other key LTOP related parameters which are being used in both the mass injection and heat injection evaluations. As discussed in Section 6.2.2, the DCPP LTOP model uses conservative adjustments to the RETRAN heat transfer model to bound the maximum potential SG to RCS heat transfer capability. These adjustments include increasing the volumetric heat capacity of the Inconel SG tubes by a factor of 1.5 and increasing the thermal conductivity by a factor of ten.

As established in WCAP 14040, the variation in the volumetric expansion of water versus temperature causes the heat input peak pressure results to increase significantly at higher RCS temperatures. Therefore, the heat input cases are evaluated at the minimum and maximum RCS temperatures which bound the applicable LTOP range. The heat input cases are also evaluated at appropriate intervals of RCS temperatures which adequately define the variation in peak pressure results throughout the LTOP temperature range. In particular, the heat input cases are evaluated near each of the minimum administrative RCS temperature limits established by the mass injection results of the previous section to identify whether the mass input or the heat input case is more limiting.

7.3.3. RETRAN Evaluation of LTOP RCS Undershoot

The RCS undershoot evaluation is performed to determine if the LTOP PORV closes before the RCS pressure in the cold legs decreases below the minimum value needed to maintain operability of the number one RCP seal. Since the RCP seals are located between the impeller and the diffuser, the RCP seal pressure is best represented as the average of the RCP volume and the RCP discharge or Cold leg volume. The minimum RCS undershoot is determined based on the two LTOP PORVs opening and closing simultaneously assuming that both the lift and reset setpoints have drifted 13 psi low. This drift value represents a 95% probability that both PORVs would not have simultaneously drifted lower than a setpoint value of 422 psig as established in design input 3.13. The PORV performance characteristics (lift setpoint, stroke time and delays) are constant throughout the complete LTOP range at DCPP. Therefore, the most limiting RCS undershoot would occur for the LTOP event with the minimum RCS mass increase and/or thermal expansion, since by definition, these effects tend to offset the pressure relief capabilities of the LTOP PORVs. The WCAP 14040 methodology for RCS undershoot can be implemented by evaluating the limiting RCS undershoot

cases based on the least severe mass input, and the least severe heat input RCS overshoot event which causes the LTOP PORV to actuate, respectively.

The least severe mass input RCS overshoot in the DCPP LTOP range was established in Section 6.3.1 based on a single injecting through the normal and alternate flow paths with no RCS flow except that from the two RHR pumps operating. This represents the smallest mass input flow rate, which could be expected to result in an actuation of the LTOP circuitry. Since the RCS flow and the RCP developed head change significantly as additional RCPs operate, an additional RCS undershoot case is evaluated for all four RCPs operating. These two RCS undershoot evaluations bound any potential combination of RCP operating conditions.

The limiting RCS undershoot evaluation for the heat input case is performed for the start of one RCP, since by definition, there is no potential RCS/SG temperature mismatch if one or more RCPs are already operating. Table 6-14 shows that the least severe heat input case which would still cause the LTOP PORV to lift at the 95% minimum setpoint of 436.7 psig is the case with a RCS/SG temperature difference of 120/200 °F. In order to evaluate the sensitivity of the RCS undershoot due to the relative heat input, a case is evaluated with the RCS/SG temperature difference at 180/260 °F.

The results of these evaluations determine if the DCPP LTOP setpoints are adequate or whether additional administrative controls are necessary to demonstrate the RCP seal are not adversely impacted during an LTOP event.

7.3.4. RETRAN Evaluation of Heat Input at 50% Pressurizer Level

The DCPP RETRAN LTOP model was used to verify that the Tech Spec LCO 3.4.6.2 restriction on RCP operation is consistent with and remains bounded by the DCPP LTOP analysis. The Tech Spec LCO 3.4.6.2 restriction specifies that an RCP can not be started with the RCS/SG temperature difference greater than 50 °F unless the pressurizer level is less than 50%. The reduced pressurizer level provides additional margin for RCS fluid expansion to ensure that a worst case SG heat transfer to the RCS could not increase the RCS pressure above the Appendix G P/T limit. This case was evaluated to bound any potential RCS/SG temperature difference within the LTOP range based on assuming the RCS was at 270 °F and the SG secondary liquid (and RCS tube volume) was at 420 °F. This RCS/SG temperature difference conservatively bounds the 100 °F mismatch which Westinghouse established as the maximum physically credible difference in Ref. 10. This heat input case was then evaluated with the initial pressurizer liquid level set to a value of 67% of the total pressurizer volume of 1800 ft^3 which conservatively bounds the pressurizer level uncertainty of ±6.1% (Design Input 12, Reference 20).

This case models the pressurizer partially filled with liquid and the LTOP PORV will relieve an air and steam mixture upon opening. Therefore, the PORV junction area and loss coefficient were reset to their original values as established in the Ref. 2 LOL analysis.

7.4. LTOP Administrative Limits

7.4.1. Statistical Treatment of Measurement Uncertainties

This section summarizes the methodology for determining the appropriate App. G P/T curve limit value which bounds the mass input and heat input peak pressure results including measurement uncertainties.

7.4.1.1. Mass Input Measurement Uncertainties

The uncertainty terms for the mass input analysis are the uncertainty associated with the pressurizer PORV LTOP pressure actuation setpoint (±32 psi) and the uncertainty associated with measuring the RCS wide range temperature (± 15 °F). Since the measurement uncertainties for the RCS pressure and the RCS temperature are independent they may be statistically combined using the sum of the squares methodology. However, while the pressure error has a constant effect on the P/T curve limit, the 15 °F temperature error has a significantly greater affect on the Appendix G P/T limit curve as temperature increases. Therefore, the following process is used to determine the equivalent pressure error with respect to the Appendix G P/T limit curve which conservatively bounds the mass input peak pressure at the specified RCS temperature.

1. Determine P/T curve "analysis temperature limit" at the "analysis peak pressure"
2. Determine new P/T curve pressure limit at "analysis temperature limit" plus measurement uncertainty
3. Calculate the change in the P/T pressure limit ("P/T Temperature error") between temperature limits 1 and 2
4. Calculate equivalent "P/T pressure error" due to "P/T temperature error" and pressure measurement error
5. Add equivalent "P/T pressure error" to "analysis peak pressure" to obtain "peak pressure limit"
6. Determine final "P/T temperature limit" corresponding to "peak pressure limit"

It should be noted that as discussed in Section 6.3.1, the mass input analyses were performed with the 32 psi pressure measurement uncertainty already applied to the PORV LTOP lift setpoint to ensure that the analysis bounded any dynamic peak pressure affects due to a delayed PORV actuation. Therefore, the original 32 psi pressure error is subtracted from the equivalent "P/T pressure error", and only the net "P/T temperature error" is added to the "analysis peak pressure" which already includes the pressure error. The corresponding P/T limits including the measurement uncertainties for the mass input analysis cases are summarized in Table 6-16.

7.4.1.2. Heat Input Measurement Uncertainties

The heat input analysis must address the additional measurement uncertainty for the Steam Generator or secondary water temperature, which is used to establish the initial RCS/SG temperature difference prior to starting an RCP. Feedwater temperature is not an accurate indicator of the overall SG liquid temperature. If the SG pressure is above atmospheric conditions, the indicated SG saturation pressure can be used to determine the corresponding liquid saturation temperature. However, as identified in Design Input 8 and Ref. 12, the SG pressure control room indication has a 34 psi

uncertainty. As shown in Table 6.4.1.2-1 below, this pressure error translates into very large temperature errors as the SG pressure decreases to atmospheric conditions.

Table 6.4.1.2-1 - SG Temperature Error vs. SG Pressure Indication

Saturation Temperature	Saturation Pressure	Average Error	Temperature Indication Error
°F	Press	F/psia	F/(34 psi)
212	14.70		
220	17.19	3.21	109.2
240	24.97	2.57	87.4
260	35.43	1.91	65.0
280	49.20	1.45	49.4

In order to eliminate the potential for such a large error when determining the maximum RCS/SG temperature differential is ≤ 50 °F, the SG temperature is determined by use of a digital thermometer used to measure the secondary side SG metal temperature. As identified in Design Input 9, Ref. 18, the digital thermometer accuracy at 250 °F is ± 3.4 °F and at 400 °F is ± 5 °F. Therefore, the heat input analysis assumes a secondary side temperature error of ± 15 °F to conservatively bound the digital thermometer measurement process and the potential for minor variations between the SG liquid and metal temperatures. This SG temperature uncertainty is conservative since the WCAP 14040 methodology defines the LTOP events as occurring at essentially steady state thermal conditions and the SG liquid and metal temperatures would be very close.

The resulting RCS/SG temperature uncertainty is based on assuming a ± 15 °F uncertainty for both the RCS and SG temperature measurements, respectively. Although these uncertainties are physically independent and may be statistically combined, the heat input analysis conservatively evaluates the worst case bounding values for the RCS/SG temperature difference. The Table 6.4.1.2-2 below shows the eight possible combinations of RCS and SG measurement errors and their impact on the actual RCS/SG temperature difference for a measured 50 °F RCS/SG limit. The RCS peak pressure results tend to become more severe as the RCS temperature increases, and as the RCS/SG temperature difference increases. The Table shows that due to the offsetting impact of the RCS and SG errors on these two effects, it is not possible to select one uncertainty combination that conservatively bounds all of the other cases. Therefore, the LTOP heat input analysis evaluates the maximum possible RCS/SG temperature difference of 80 °F for all RCS temperatures within the LTOP range, including at the maximum RCS temperature plus uncertainty of 285 °F. The peak RCS pressure for each heat input case is then evaluated with respect to the lower Appendix G P/T curve limit corresponding to the measured RCS temperature value without uncertainty. As an example, the RCS/SG temperature difference of 285/365 °F, is evaluated with respect to the Appendix P/T curve limit for an RCS temperature of 270 °F. This evaluation methodology ensures that the heat input peak pressure results conservatively bound any physical possible combination of RCS/SG temperature values and their associated measurement uncertainties.

Table 6.4.1.2-2 – RCS./SG Temperature Error Combinations

Case	Measured RCS /SG Temperature Difference	RCS Temperature Error	SG Temperature Error	Actual RCS /SG Temperature Difference
	°F			°F
1	270/320	High	High	285/335
2	270/320	High	Low	285/305
3	270/320	Low	High	255/335
4	270/320	Low	Low	255/305
5	270/320	None	High	270/335
6	270/320	None	Low	270/305
7	270/320	High	None	285/320
8	270/320	Low	None	255/320

The heat input peak pressure results for each case are then evaluated for the impact of the RCS temperature and RCS pressure measurement uncertainties on the Appendix G P/T curve limit identical to that discussed previously for the mass input analysis results. Table 6-17 provides a summary of the heat input peak pressure results and compares them to the corresponding Appendix G P/T limits when including the measurement uncertainties.

8. RESULTS

8.1. RETRAN LTOP Model Benchmark

8.1.1. RETRAN Mass Input Benchmark

Table 6-6 summarizes the comparison of the RETRAN mass input results with those of the Westinghouse LOFTRAN model established in Ref. 9. The RETRAN LTOP model is evaluated over an applicable range of mass injection flow rates, PORV lift setpoints, and PORV opening times established by the Westinghouse LOFTRAN results. The RETRAN LTOP model generates RCS pressure overshoot and undershoot results which are comparable to the Westinghouse LOFTRAN model.

8.1.2. RETRAN Heat Input Benchmark

Table 6-8 provides a parametric summary of the various heat input cases for which the DCPP RETRAN LTOP model was compared to the generic Westinghouse results. Figure 6-2 compares the RETRAN heat input results to the LOFTRAN results for three different PORV actuation setpoints. The RETRAN model accurately models both the magnitude and characteristic thermal response of the heat input transient over an appropriate range of RCS conditions.

8.2. *RETRAN Evaluation of DCPP LTOP Setpoints*

8.2.1. Mass Input RCS Overshoot

A summary of the mass input cases evaluated and the peak RCS pressure results are provided in Table 6-13. Figure 6-4 compares the mass input peak pressure results while Figure 6-5 plots the peak pressure values versus the corresponding administrative temperature limit to demonstrate how they remain bounded by the Appendix G P/T curve limit.

8.2.2. Heat Input RCS Overshoot

Table 6-14 lists the heat input cases and the peak RCS pressure results. Figure 6-6 and Figure 6-7 compare the RCS pressure and temperature responses versus time, respectively for the various heat input cases. The heat input peak pressure results are also plotted in Figure 6-8 to demonstrate how they remain bounded by the Appendix G P/T curve limit.

8.2.3. LTOP RCS Undershoot

The RCS undershoot results are summarized in Table 6-15. This Table lists the PORV setpoints, the time of the minimum undershoot, and the relative RCS pressures including the minimum effective value at the RCP seal. Figures 6-8 and 6-9 plot the RCS pressure versus time for the mass input cases with one RCP and four RCPs operating, respectively. These results show that the RCS pressure in the RCP volume remains above 235 psig (250 psia). The limiting mass input RCS undershoot case does not challenge the operation of the number one RCP seals. The results indicate that as more than one RCPs operates, the relative RCP suction pressure increases and the RCS undershoot becomes less limiting.

The heat input undershoot results are also shown in Table 6-15 while the relative RCS volume pressures for the two cases evaluated are plotted in Figure 6-11 and 6-12, respectively. The results show that the RCS undershoot for the heat input case is more severe than for the mass input cases, since the RCS pressurization is significantly less, and there is no net RCS mass addition to offset the release through the PORVs. As expected, the undershoot becomes less severe as the heat input pressure transient increases. Figure 6-11 shows that following the initial RCS pressure increase, there are two brief periods where the RCS pressure experiences an additional decrease due to the mixing effects of the initially heated RCS fluid with that which is still relatively cool. After the first closure of the PORVS, this mixing effect causes the RCS pressure to continue to drop below the 250 psia value for a brief period of about one minute. After a couple of minutes, the RCS volume is thoroughly mixed such that the SG heat transfer becomes the only significant effect. As discussed in Reference 6, the RCP seals can operate for up to two minutes without any seal injection flow. Based on the conservative conditions established in this RCS undershoot evaluation, the small and brief pressure decrease below 250 psia is not considered to adversely impact the RCP seal performance.

8.2.4. LTOP Pressurizer Level LCO

Figure 6-13 plots the RCS pressure and pressurizer liquid level versus time for the heat input case with the initial pressurizer level at 67%. Due to the additional expansion volume within the partially

filled pressurizer, the PORV lift pressure is not reached until a time of 128 seconds. The RCS pressure and pressurizer level continue to increase until about 480 seconds when the conservatively low PORV relief capacity is able to offset the thermal expansion effects. The RCS pressure increases to a peak value of 622 psia which remains bounded by the comparable water solid heat input case already evaluated. The heat input transient and the subsequent peak pressures become less limiting as the RCS and SG temperatures decrease. This confirms that as long as the pressurizer level indicates less than or equal to 50%, an RCP started with even the maximum credible RCS/SG temperature differential remains bounded by the heat input cases and does not challenge the Appendix G P/T limit.

8.3. LTOP Administrative Temperature Limits

Table 6-18 summarizes the DCPP LTOP administrative temperature limits based on the most limiting results obtained from the mass injection and heat injection evaluations including the appropriate measurement uncertainty. It should be noted that for the DCPP LTOP parameters established in the PTLR, the mass injection results establish the administrative temperature limits since they generate more limiting peak pressure results than the heat injection cases over the range of LTOP applicability. These administrative limits ensure that the mass injection capability and the dynamic pressure drop across the RCS remain bounded by the LTOP analysis assumptions.

9. CONCLUSIONS

9.1. RETRAN LTOP Model Benchmark

The RETRAN LTOP model generates RCS pressure overshoot and undershoot results, which are comparable to the Westinghouse LOFTRAN model. Therefore, the DCPP RETRAN model is appropriate for evaluating the LTOP setpoints per the WCAP 14040 methodology.

9.2. RETRAN Evaluation of DCPP LTOP Setpoints

The DCPP LTOP setpoints as established in the PTLR (Ref. 7) ensure that the maximum RCS overshoot results remain bounded by the Appendix G P/T curve limit, including the appropriate measurement uncertainty. The DCPP LTOP setpoints ensure that with a minimum RCS vent available that is equal to the pressurizer PORV area of 2.07 in^2 no credible LTOP event can challenge the Appendix G pressure limit. The DCPP LTOP setpoints also ensure that the minimum RCS undershoot results do not adversely impact the operation of the number one RCP seal during an LTOP event. The Tech Spec LCO 3.4.6.2 restriction on RCP operation and pressurizer level remains bounded by the DCPP LTOP analysis.

9.3. LTOP Administrative Temperature Limits

The LTOP administrative limits established in Table 6-18 ensure that the mass injection capability and the dynamic pressure drop across the RCS remain bounded by the LTOP analysis assumptions.

10. IMPACT EVALUATION

These evaluation results will be used as the technical basis for a License Amendment Request to the DCPP PTLR, and will become the licensing basis for the DCPP LTOP setpoints, upon NRC approval.

11. REFERENCES

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14. PG&E Calculation PAM-0 7-413, Rev. 7 "Post Accident Monitoring Indication , RCS Cold Leg Water Temperature and Hot Leg Water Temperature",
15. Crane Manual, Technical paper No. 410,"Flow of fluids Through Valves, Fittings and Pipe", Twenty Fifth Printing-1991.
16. ASME Steam Table Fifth Edition.

Tables

Table 6-1: RETRAN LTOP Model Volume Summary

Volume Number	Volume Description	Volume	Height	Flow Length	Flow Area	Hydraulic Diameter	Lowest Elevation
		(ft3)	(ft)	(ft)	(ft2)	(ft)	ELEV
1	Upper Downcomer	280.032	9	3.085	90.831	1.531	103.917
2	Lower Downcomer	408.593	15.292	0	26.72	0.629	88.625
3	Lower Plenum	1023.73	9.95	0	128.497	3.774	79.51
4	Lower Core Section	218	4.458	0	51.013	0.0362	89.4583
6	Middle Core Section	218	4.458	0	51.013	0.0362	93.9163
7	Upper Core Section	218	4.458	0	51.013	0.0362	98.3743
8	Core Bypass Volume	71.32	13.375	0	5.33	0.0375	89.4583
9	Upper Plenum	873.15	12.67	12.67	72.26	0.962	102.8323
10	Reactor Vessel Head	485.3	7.14	11.23	89.44	1.82	112.917
101, 201, 301, 501	Reactor Coolant Pump	79	8.43	0	13.6	N/A	101.1875
102, 202, 302, 502	RCS Cold Leg 1	76.696	2.292	0	4.126	2.292	105.854
103, 203, 303, 503	RCS Cold Leg 2	26.888	2.612	0	4.778	2.344	105.854
105, 205, 305, 505	RCS Hot Leg 1	15.957	2.932	0	5.09	2.388	105.534
106, 206, 306, 506	RCS Hot Leg 2	60.324	2.42	0	4.6	2.42	105.79
107, 207, 307, 507	Steam Generator HL Inlet	19.459	3.648	0	4.914	2.5	105.79
108, 208, 308, 508	SG HL Plenum	158.771	5.234	0	30.335	9.022	108.689
111, 211, 311, 511	SG CL Plenum	158.771	5.234	0	30.335	9.022	108.689
112, 212, 312, 512	SG CL Outlet	16.464	3.142	0	5.24	2.583	105.72
113, 213, 313, 513	RCS CL Cross Under 1	23.727	4.53	0	5.24	2.583	101.19
114, 214, 314, 514	RCS CL Cross Under 2	37.04	5.794	0	5.24	2.583	95.396
115, 215, 315, 515	RCS CL Cross Under 3	18.445	2.583	0	5.24	2.583	95.396
116, 216, 316, 516	RCP Suction	37.04	5.794	0	5.24	2.583	95.396
401	Pressurizer Surge Line	45.93	6.883	0	0.6829	0.933	106.534
402	Pressurizer	1800	52.8	0	38.485	7	113.16
141, 241, 341, 541	RCS HL SG Tube Bundle	61.635	5.553	0	11.099	0.0646	119.476
142, 242, 342, 542	RCS HL SG Tube Bundle	143.872	12.963	0	11.099	0.0646	132.439
143, 243, 343, 543	RCS HL SG Tube Bundle	143.872	12.963	0	11.099	0.0646	145.402
144, 244, 344, 544	RCS SG U-Tube Bundle	77	4.987	0	11.099	0.0646	132.439
145, 245, 345, 545	RCS CL SG Tube Bundle	143.872	12.963	0	11.099	0.0646	119.476
146, 246, 346, 546	RCS CL SG Tube Bundle	143.872	12.963	0	11.099	0.0646	113.923
149, 249, 349, 549	RCS CL SG Tube Bundle	61.635	5.553	0	11.099	0.0646	113.923

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Table 6-2: RETRAN LTOP Model Junction Summary, 1 of 2

Junction No.	Description	From Volume	To Volume	Initial Flow Rate	Junction Area	Junction Elevation	Forward Loss Coefficient	Reverse Loss Coefficient	Hydraulic Diameter
				(lb/sec)	(ft2)	(ft2)			(ft)
101, 201, 301, 501	RCP Discharge to RCS CL 1	101	102	0	4.125	107	0	0	2.292
102, 202, 302, 502	RCS CL 1 to RCS CL 2	102	103	0	4.125	107	0	0.036	2.292
104, 204, 304, 504	RCS CL Nozzle	103	1	0	6.752	107	1.413	0.394	2.292
1	Upper Downcomer to Lower Downcomer	1	2	0	26.72	103.917	0.686	0	0
2	Lower Downcomer to Lower Plenum	2	3	0	26.72	88.625	1.03	0	0
3	Lower Plenum to Lower Core Inlet	3	5	0	51.013	89.4583	8.06	0	0
5	Lower Core to Middle Core	5	6	0	51.013	93.9163	9.33	0	0
6	Middle Core to Upper Core	6	7	0	51.013	98.3743	9.33	0	0
7	Upper Core to Upper Internals	7	9	0	51.013	102.8323	4.41	0	0
8	Lower Plenum to Core Bypass Inlet	3	8	0	5.33	89.4583	30.976	0	0
9	Core Bypass to Upper Internals	8	9	0	5.33	102.8333	0	0	0
10	Upper Downocmer Bypass to Reactor Head	1	10	0	0.0167	112.917	0	0	0.0364
11	Reactor Head to Upper Internals	10	9	0	0.9235	115.502	0	0	0.139
105, 205, 305, 505	RCS HL Nozzle	9	105	0	6.752	107	1.06	1.07	2.292
106, 206, 306, 506	RCS HL 1 to RCS HL 2	105	106	0	4.6	107	0	0.45	2.42
107, 207, 307, 507	RCS HL 2 to SG HL Inlet	106	107	0	4.6	107	0.189	0.071	2.42
108, 208, 308, 508	SG HL Inlet to SG HL Plenum	107	108	0	5.23	108.689	1.003	0.5	2.583
140	SG HL Plenum to HL Tube Bundle 1	108	141	0	11.099	113.923	0.66	1	0.0646
141, 241,341, 541	HL Tube Bundle 1 to HL Tube Bundle 2	141	142	0	11.099	119.476	0	0	0.0646
142, 242,342, 542	HL Tube Bundle 2 to HL Tube Bundle 3	142	143	0	11.099	132.439	0	0	0.0646
143, 243,343, 543	HL Tube Bundle 3 to U-Tube Bundle	143	144	0	11.099	145.402	1.729	1.31	0.0646
144, 244,344, 544	U-Tube Bundle to CL Tube Bundle 1	144	145	0	11.099	145.402	1.729	1.31	0.0646
145, 245,345, 545	U Tube Bundle 1 to CL Tube Bundle 2	145	146	0	11.099	132.439	0	0	0.0646
146, 246,346, 546	CL Tube Bundle 3 to CL Tube Bundle 3	146	147	0	11.099	119.476	0	0	0.0646

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Table 6-2: RETRAN LTOP Model Junction Summary , 2 of 2

Junction No.	Description	From Volume	To Volume	Initial Flow Rate	Junction Area	Junction Elevation	Forward Loss Coefficient	Reverse Loss Coefficient	Hydraulic Diameter
147, 247,347, 547	SG CL Tube Bundle 3 to CL Plenum	147	111	0	11.099	113.923	1.32	0.5	0.0646
112, 212, 312, 512	SG CL Plenum to SG CL Outlet	111	112	0	5.24	108.689	0.153	0.818	2.583
113, 213, 313, 513	SG CL Outlet to RCS CL Cross Under 1	112	113	0	5.24	105.72	0	0.058	2.583
114, 214, 314, 514	RCS CL Cross Under 1 to Cross Under 2	113	114	0	5.24	101.19	0.3	0.113	2.583
115, 215, 315, 515	RCS CL Cross Under 2 to Cross Under 3	114	115	0	5.24	96.688	0	0.114	2.583
116, 216, 316, 516	RCS CL Cross Under 3 to Cross Under 4	115	116	0	5.24	96.688	0.3	0.227	2.583
117, 217, 317, 517	RCS CL Cross Under 4to RCP Suction	116	101	0	5.24	101.1875	0	0	2.583
461	RCP Seal Injection to Loop 1	0	102	0	1	0	0	0	0
462	RCP Seal Injection to Loop 2	0	202	0	1	0	0	0	0
463	Alternate Charging & RCP Seal Injection to Loop 3	0	302	0	1	0	0	0	0
464	Normal Charging & RCP Seal Injection to Loop 4	0	502	0	1	0	0	0	0
401	RCS HL 1 to Pressurizer Surge line	206	401	0	0.6829	107	2.308	0.308	0.933
402	Pressurizer Surge Line to Pressurizer	401	402	0	0.6829	113.167	1.273	0.801	0.933
		PORV	MODEL						
411	Pressurizer PORV # 1 to Containment	402	900	0	0.00824	165.96	0.0727	0	0.116
412	Pressurizer PORV # 2 to Containment	402	900	0	0.00824	165.96	0.0727	0	0.116
900	RHR Suction from RCS Loop 4	0	506	0	0	107	0	0	0
901	RHR Injection to RCS CL Loop 1	0	103	0	0	107	0	0	0
902	RHR Injection to RCS CL Loop 2	0	203	0	0	107	0	0	0
903	RHR Injection to RCS CL Loop 3	0	303	0	0	107	0	0	0
904	RHR Injection to RCS CL Loop 4	0	503	0	0	107	0	0	0

Table 6-3: RETRAN LTOP Model SG Heat Conductor Summary

Heat Conductor Number	Left Hand Side Volume	Right Hand Side Volume	Conductor Center Elevation (ft)	Left Side Heat Transfer Surface Area (ft2)	Right Side Heat Transfer Surface Area (ft2)	Heat Conductor Volume (ft3)	Right Side Heated Diameter	Left Side Channel Length (ft)	Right Side Channel Length (ft)
141, 241,341, 541	141	172	1.901	3817.4	2951.783	16.932	0.14277	67.945	4.25
142, 242,342, 542	142	172	6.481	8910.852	10060.64	39.524	0.14277	67.945	4.25
143, 243,343, 543	143	172	6.481	8910.852	10060.64	39.524	0.14277	67.945	4.25
144, 244,344, 544	144	172	2.578	4772.181	5387.946	21.166	0.14277	67.945	4.25
145, 245,345, 545	145	172	6.481	8910.852	10060.64	39.524	0.14277	67.945	4.25
146, 246,346, 546	146	172	6.481	8910.852	10060.64	39.524	0.14277	67.945	4.25
147, 247,347, 547	147	172	1.901	3817.4	2951.783	16.932	0.14277	67.945	4.25
Total Per SG				48050.4	51534.1	213.1			

Table 6-4 : Westinghouse Benchmark LTOP Mass Input Parameters

RCS Volume (ft^3)	12372	
RCS Temperature (Deg. F)	100	
PORV Opening Characteristics	Per Table 6-5	
PORV Opening Setpoint (psig)	350, 400, 450, 500	
PORV Opening Times (sec)	2.0, 4.0 , 6.0	
PORV Closing Time (sec)	2.0	
PORV Flow Rate C_V ($gpm/(psi)^{1/2}$)	46	
PORV Electronic Delay (sec)	1.1	
Mass Injection Flow Rate (gpm)	200, 300,400	

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Table 6-5 : Westinghouse Benchmark PORV C_V vs. Stroke

Normalized Time	Normalized Flow Cv
	0
0.000	0
0.100	0.033
0.200	0.055
0.300	0.08
0.400	0.115
0.500	0.165
0.600	0.235
0.700	0.335
0.800	0.48
0.900	0.7
1.000	1

Table 6-6 : Benchmark Comparison of RETRAN vs LOFTRAN Results
Mass Input

RCS Overshoot					RCS Under Shoot		
RCS temp (F)	PORV Setpt (psig)	PORV Stroke time (sec)	Mass Injection Flow Rate (gpm)	RETRAN Over shoot (psi)	West Over Shoot (psi)	RETRAN Under Shoot (psi)	West. Under shoot (psi)
100	450	4	200	31.8	32	75.9	71
100	450	4	300	52.3	54	64.2	63
100	450	4	400	76.5	78	54.6	59
100	350	4	300	59.7	56	55.1	55
100	400	4	300	55.9	55	59.9	59
100	450	4	300	52.3	54	64.2	63
100	500	4	300	50.9	53	69	67
100	450	2	300	37.3	36	64.4	63
100	450	4	300	52.3	54	64.2	63
100	450	6	300	68.4	72	64.7	64

Table 6-7 : Westinghouse Benchmark LTOP Heat Input Parameters

RCS Volume (ft^3)	13000	
RCS Temperature (Deg. F)	100	
SG Temperature (Deg. F)	100	
PORV Opening Characteristics	Linear vs. time	
PORV Opening Setpoint (psig)	350, 400, 450, 500	
PORV Opening Times (sec)	3.0	
PORV Closing Time (sec)	3.0	
PORV Flow Rate C_V (gpm/(psi)$^{1/2}$)	50	
PORV Electronic Delay (sec)	0.6	
RCS/ SG Temperature Delta (F))	50, 100	
RCP Flow Rate (gpm)	95,000	
RCP Start Time (sec)	10	

Table 6-8 : Westinghouse Benchmark LTOP Heat Input Results

RCS / SG Temperatures (F)	PORV setpoint (pisg)	LOFTRAN Peak RCS Pressure (psia)	RETRAN Peak RCS Pressure (psia)
140/240	400	592	595
140/240	500	682	675
140/240	600	790	787

Table 6-9 : DCPP LTOP Parameters

RCS Volume (ft^3)	11716	
Pressurizer Volume (ft3)	1800	
RCS Flow Rate per Loop (gpm)	95,000	
RCP Start Time (sec)	10	
RHR Flow per Pump (gpm)	5000	
PORV Opening Characteristics	Figure 6-10	
PORV Opening Setpoint (psig)	435	
PORV Reseat Setpoint (psig)	415	
PORV Stroke Time (sec)	2.9	
PORV Actuation Delay Time (sec)	1.05	
PORV Flow Rate C_V (gpm/(psi)$^{1/2}$)	46	
PORV Vent Area (in^2)	2.07	
RCS Temperature Uncertainty (°F)	15	
SG Temperature Uncertainty (°F)	15	
RCS Pressure Uncertainty (psi)	32	
Maximum Measured RCS/SG Temperature Difference (°F)	50	
Maximum LTOP RCS Temperature (°F)	285	
Minimum LTOP RCS Temperature (°F)	55	

Table 6-10 : DCPP CCP Injection (ECCS) with PDP

RCS Pressure (psia)	ECCS/ PDP Flow (gpm)
14.7	595.2
114.7	582.8
214.7	571.2
314.7	559.4
414.7	547.6
514.7	535.7
614.7	523.9
714.7	512
814.7	500.1
914.7	486.1
1014.7	472.2

Table 6-11 : DCPP CCP Charging Injection with PDP

RCS Pressure (psia)	RCP Seal Injection to Each Loop (gpm)	Total Alternate Charging Flow to Loop-3 (gpm)	Total Normal Charging Flow to Loop-4 (gpm)	Total CCP/PDP Injection Flow to RCS (gpm)
14.7	21.75	223.75	205.75	473
114.7	21.25	219.25	201.25	463
214.7	20.75	214.75	197.75	454
314.7	20.50	211.50	193.50	446
414.7	20.00	207.00	190.00	437
514.7	19.75	202.75	185.75	428
614.7	19.25	198.25	182.25	419
714.7	18.75	193.75	177.75	409
814.7	18.50	189.50	174.50	401
914.7	18.00	185.00	170.00	391
1014.7	17.50	180.50	165.50	381

Table 6-12 : DCPP CCP Charging Injection

RCS Pressure (psia)	RCP Seal Injection to Each Loop (gpm)	Total Alternate Charging Flow to Loop-3 (gpm)	Total Normal Charging Flow to Loop-4 (gpm)	Total CCP Injection Flow to RCS (gpm)
14.7	19.00	198.00	182.00	418
114.7	18.50	193.50	177.50	408
214.7	18.25	189.25	174.25	400
314.7	17.75	184.75	169.75	390
414.7	17.50	180.50	165.50	381
514.7	17.00	176.00	162.00	372
614.7	16.50	171.50	157.50	362
714.7	16.00	167.00	153.00	352
814.7	15.75	162.75	148.75	343
914.7	15.25	157.25	144.25	332
1014.7	14.75	152.75	139.75	322

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Table 6-13 : DCPP LTOP Mass Input Peak Pressure Results

Mass Input Case	Num. RCPs	Time of Peak (sec)	Pressurizer Pressure Volume 402 (PSIA)	Vessel Pressure Volume 3 (PSIA)	Hot Leg Pressure Volume 306 (PSIA)	RCS Pressure Overshoot (PSID)
CCP/ PDP (ECCS)	4	37.9	587.2	648.4	579.1	198.4
CCP and PDP	4	38.9	564.2	625.1	565.5	175.1
CCP - 4 RCPs	4	39.5	553.3	614.0	545.4	164
CCP - 3 RCPs	3	39.5	544.7	593.8	546.3	143.8
CCP - 2 RCPs	2	39.5	543.7	579.1	546.6	129.1
CCP - 1 RCPs	1	39.5	543.1	570.7	546.6	120.7
CCP - 0 RCPs	0	39.2	531.9	557.0	547.3	107
CCP/ PDP (ECCS) w/vent	0	350	172.6	196.5	186.7	N/A

Table 6-14 : DCPP LTOP Heat Input Peak Pressure Results

RCS / SG Temperatures (°F)	RCS Pressure (psia)		Time (sec)	Pressurizer Volume 402 (psia)	Vessel Volume 3 (psia)	Hot Leg Volume 306 (psia)
100/180	300		10.6	276.1	301.9	278.2
120/200	300		75	470.8	496.4	472.7
135/215	300		12.1	478.9	504.5	480.9
150/230	300		12.4	548.4	574.2	550.7
180/260	300		12.4	621.9	648.1	624.8
200/280	350		11.9	665.4	692.6	669.4
230/310	350		11.9	669.5	697.6	686.2
270/350	400		11.2	669.7	698.3	687
285/365	400		12.1	671.1	698.1	675.8

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Table 6-15 : DCPP LTOP RCS Undershoot Results

Description	PORV 1 Lift	PORV 1 Reset	PORV 2 Lift	PORV 2 Reset	Time of Under Shoot (sec)	PRES 402 (PSIA)	PRES 506 (PSIA)	PRES 502 (PSIA)	PRES 501 (PSIA)	PRES 516 (PSIA)	RCP Seal Pressure (psia)
Mass Input 1 RCP	436.7	416.7	436.7	416.7	40.3	275.1	279.0	282.2	252.2	194.3	267.2
Mass Input 4 RCP	436.7	416.7	436.7	416.7	41.1	289.7	283.5	340.6	291.4	222.8	316.0
Heat Input RC/SG 120/200	436.7	416.7	436.7	416.7	282	231.44	232.96	238.6	209.64	152.95	224.1
Heat Input RC/SG 180/260	436.7	416.7	436.7	416.7	16.7	247.2	248.3	253.9	225.8	169.9	239.9

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Table 6-16 : DCPP LTOP Mass Input P/T Limit with Uncertainty

MASS INPUT CASE	RCS Press. Uncert. (psi).	RCS Temp. Uncert. (°F)	Analysis Peak Pressure Overshoot (psia)	LTOP Temp. Limit @ Analysis Overshoot (°F)	Revised Temp. Limit with Temp. Uncert. Added (°F)	Revised Press. Limit at Temp. Uncert. Added (psia)	Pressure Error for Temp. Uncert (psi)	Total Press. Error Sum of Squares (psi)	Analysis Over shoot Plus Total Press. Error (psia)	Temp. Limit with Total Press. Error (°F)
CCP/ PDP (ECCS)	32	15	648.4	145.5	160.5	681.5	33.1	46.0	662.4	152.2
CCP and PDP 4 RCPs	32	15	625.1	132.5	147.5	652.5	27.4	42.1	635.2	138.5
CCP - 4 RCPs	32	15	614	125.4	140.4	638.7	24.7	40.5	622.5	130.9
CCP - 3 RCPs	32	15	593.8	110.2	125.2	613.6	19.8	37.6	599.4	114.8
CCP - 2 RCPs	32	15	579.1	96.5	111.5	595.4	16.3	35.9	583.0	100.4
CCP - 1 RCPs	32	15	570.7	87.2	102.2	584.9	14.2	35.0	573.7	90.7
CCP - 0 RCPs	32	15	557	68.9	83.9	568.0	11.0	33.8	558.8	71.6

Table 6-17 : DCPP LTOP Heat Input P/T Limit with Uncertainty

Heat Input Case	RCS Press. Uncert. (psi).	RCS Temp. Uncert. (°F)	Analysis Peak Pressure Overshoot (psia)	LTOP Temp. Limit @ Analysis Overshoot (°F)	Revised Temp. Limit with Temp. Uncert. Added (°F)	Revised Press. Limit at Temp. Uncert. Added (psia)	Pressure Error for Temp. Uncert (psi)	Total Press. Error Sum of Squares (psi)	Analysis Over shoot Plus Total Press. Error (psia)	Temp. Limit with Total Press. Error (°F)
100/180	32	15	301.9	60.0	75.0	561.2	3.4	32.2	302.1	60.0
120/200	32	15	496.4	60.0	75.0	561.2	3.4	32.2	496.6	60.0
135/215	32	15	504.3	60.0	75.0	561.2	3.4	32.2	504.5	60.0
150/230	32	15	574.2	91.2	106.2	589.3	15.1	35.4	577.6	94.9
180/260	32	15	648.1	145.3	160.3	681.1	33.0	46.0	662.1	152.0
200/280	32	15	692.6	164.9	179.9	736.4	43.8	54.2	714.8	172.9
230/310	32	15	697.6	166.7	181.7	742.6	45.0	55.2	720.8	174.9
270/350	32	15	698.3	167.0	182.0	743.5	45.2	55.3	721.6	175.2
285/365	32	15	698.1	166.9	181.9	743.2	45.1	55.3	721.4	175.1

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Table 6-18 : DCPP LTOP Administrative Temperature Limits

LTOP Administrative Action	RETRAN PTLR Code Case N-514 Temp. Limit
LTOP Enable - Disable one CCP	270
Block ECCS Flow Path	153
Disable Second Charging Pump	139
Stop 1 of 4 RCPs	131
Stop 2 of 4 RCPs	115
Stop 3 of 4 RCPs	101
Stop 4 of 4 RCPs	91
Establish RCS Vent	72

Figures

Figure 6-1 : DCPP RETRAN LTOP Evaluation Model

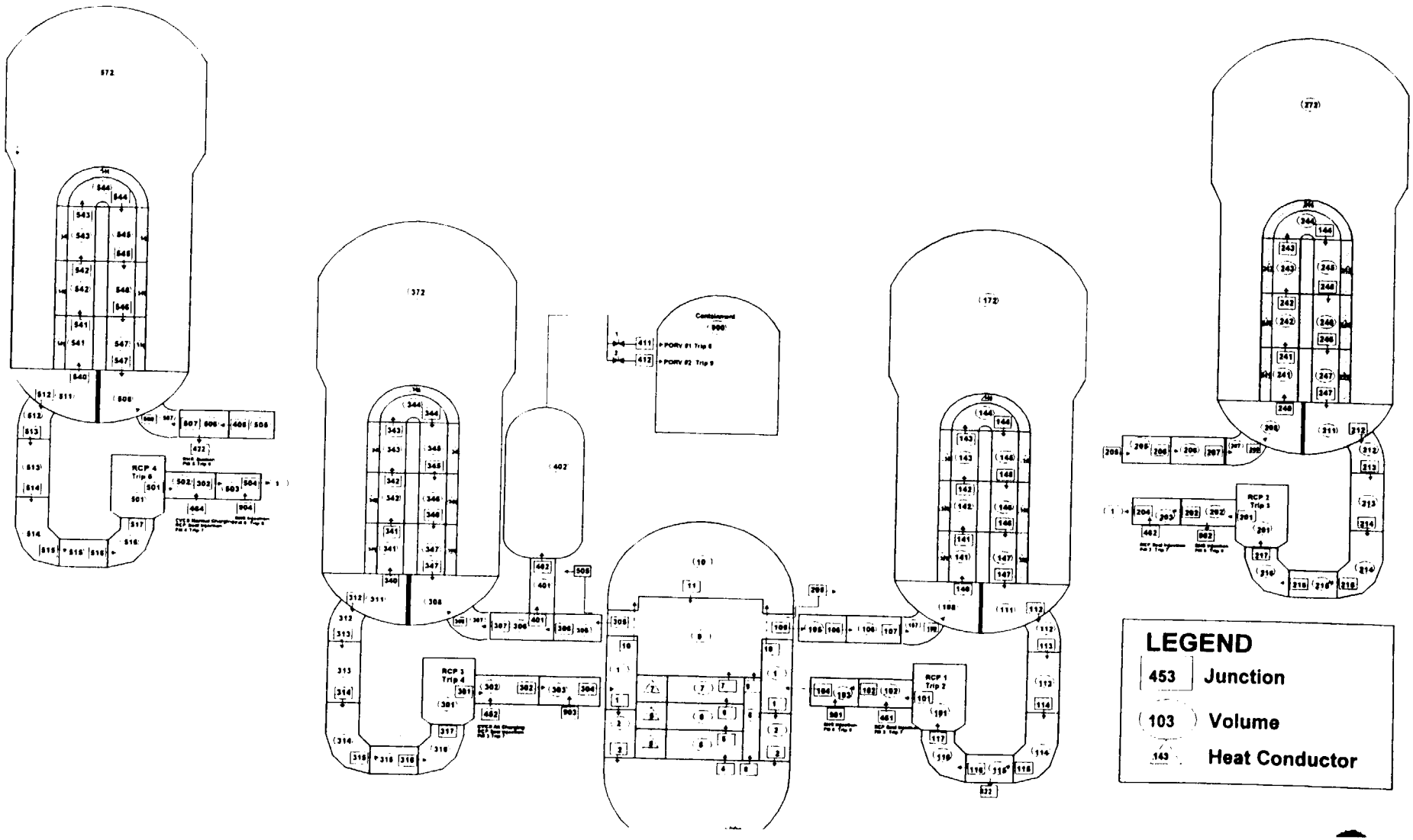

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Figure 6-2 : RETRAN vs LOFTRAN LTOP Heat Input Results

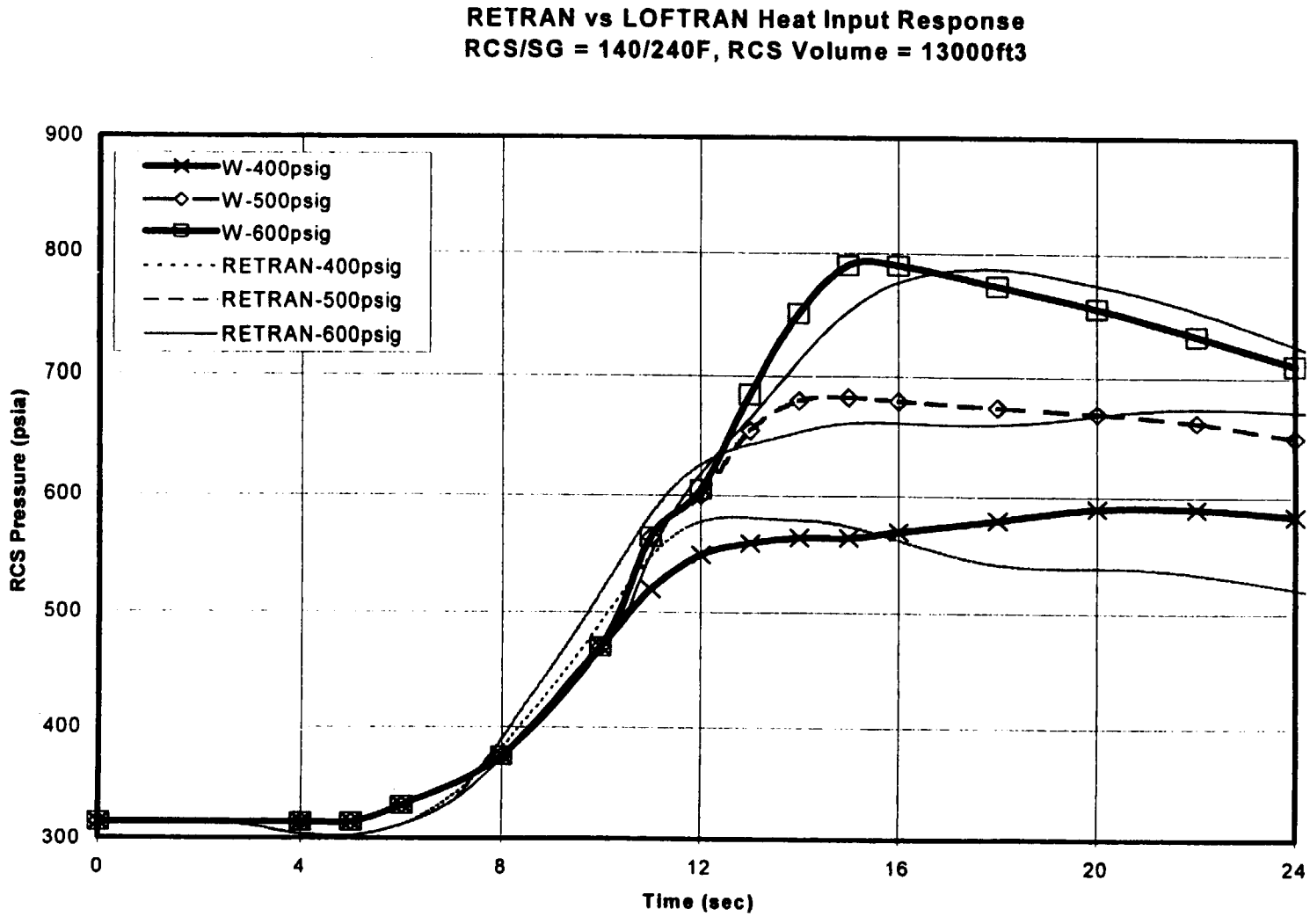


Figure 6-3 : DCPP PORV Normalized Valve Cv vs Position

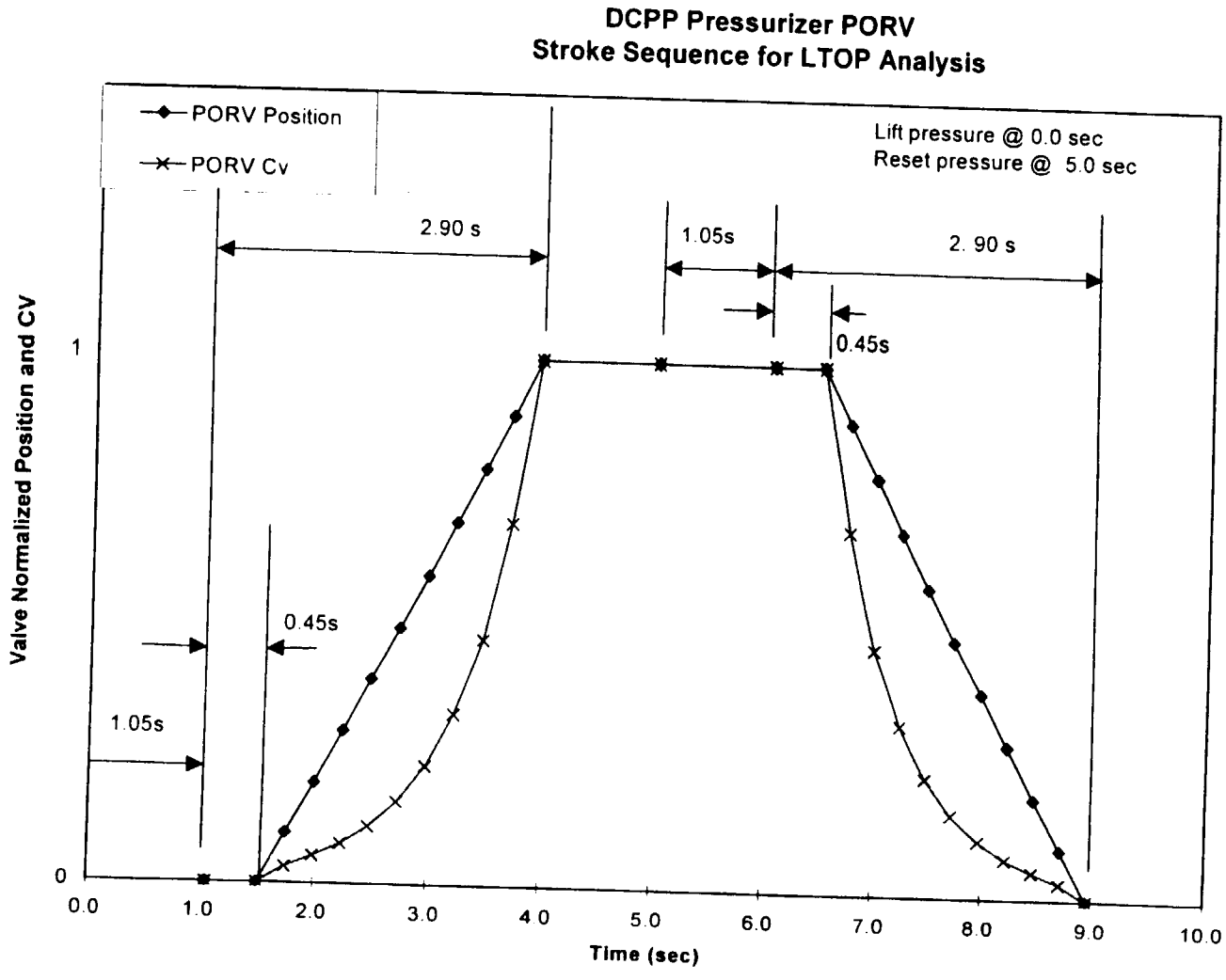


Figure 6- 4 : DCPP RETRAN LTOP Mass Input Pressure Results

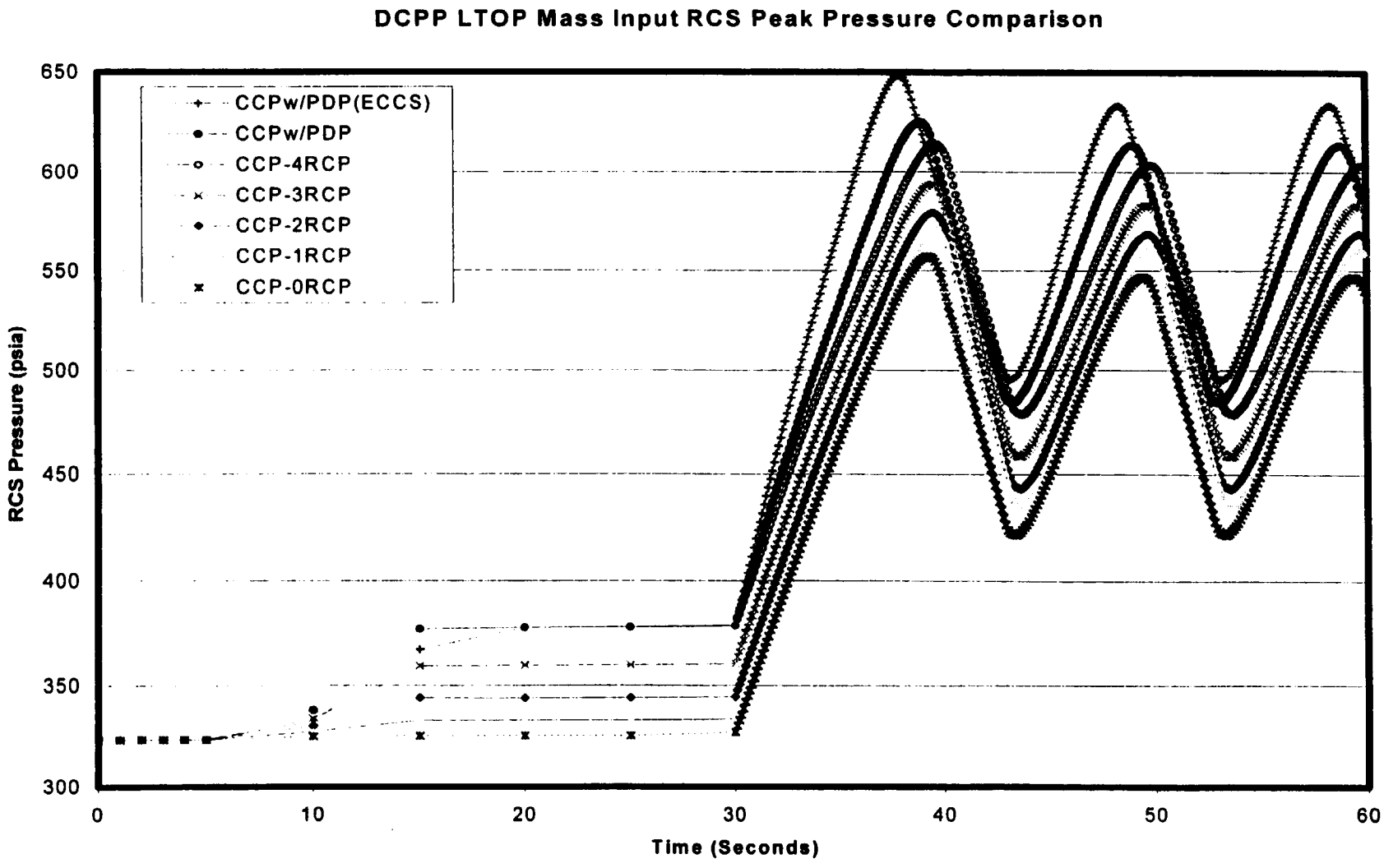
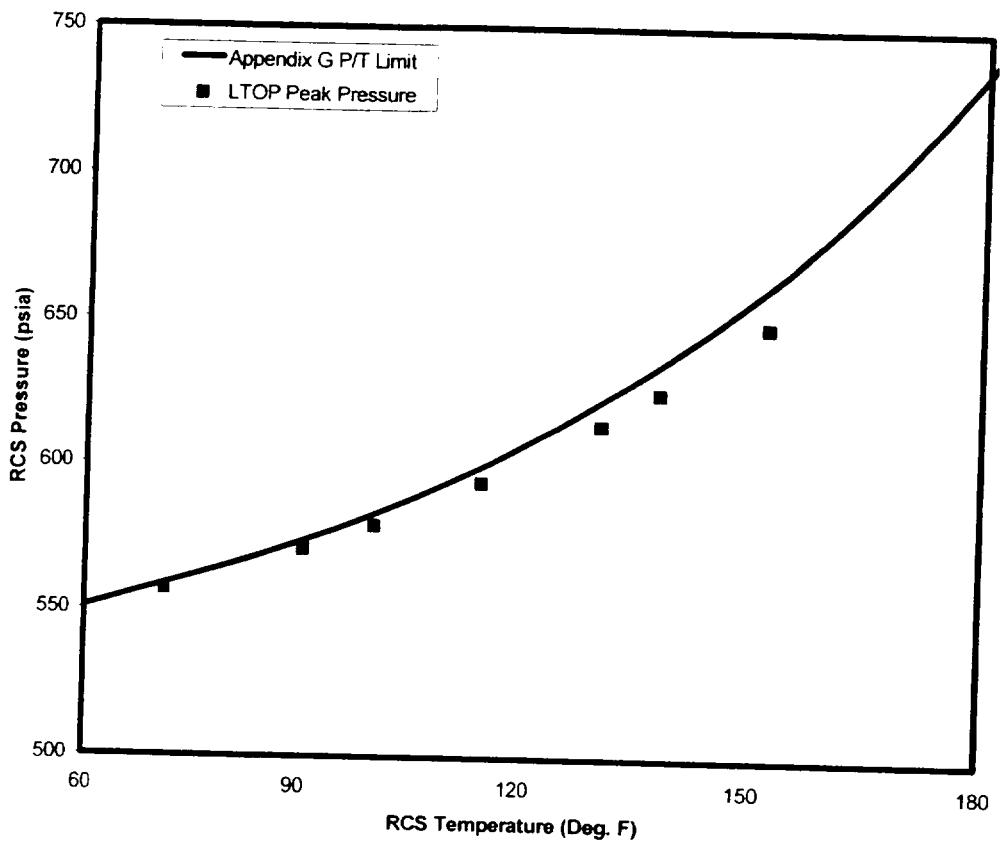


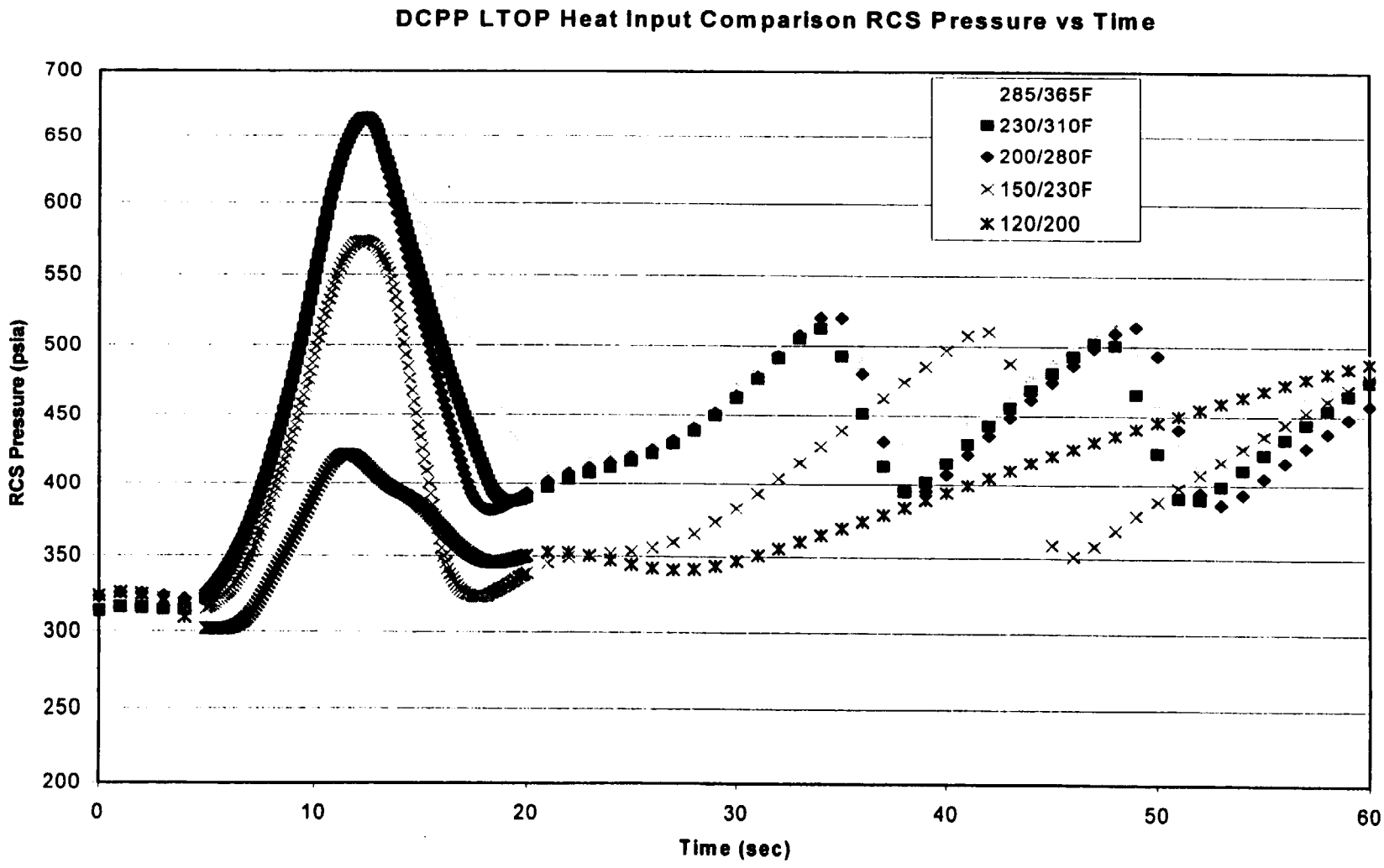
Figure 6- 5 : DCPP RETRAN LTOP Mass Input P/T Summary

Mass Injection Peak Pressure vs P/T Limit (with Uncertainties)



SUBJECT RETRAN Evaluation of DCPP LTOP Parameters
MADE BY Jerry E. Ballard DATE 11/ 26 /01 CHK'D BY Dixon Yee DATE 11/ 26 /01

Figure 6- 6 : DCPP RETRAN LTOP Heat Input Pressure Results



SUBJECT RETRAN Evaluation of DCPP LTOP Parameters
MADE BY Jerry E. Ballard DATE 11/ 26 /01 CHK'D BY Dixon Yee DATE 11/ 26 /01

Figure 6- 7 : DCPP RETRAN LTOP Heat Input Temperature Results

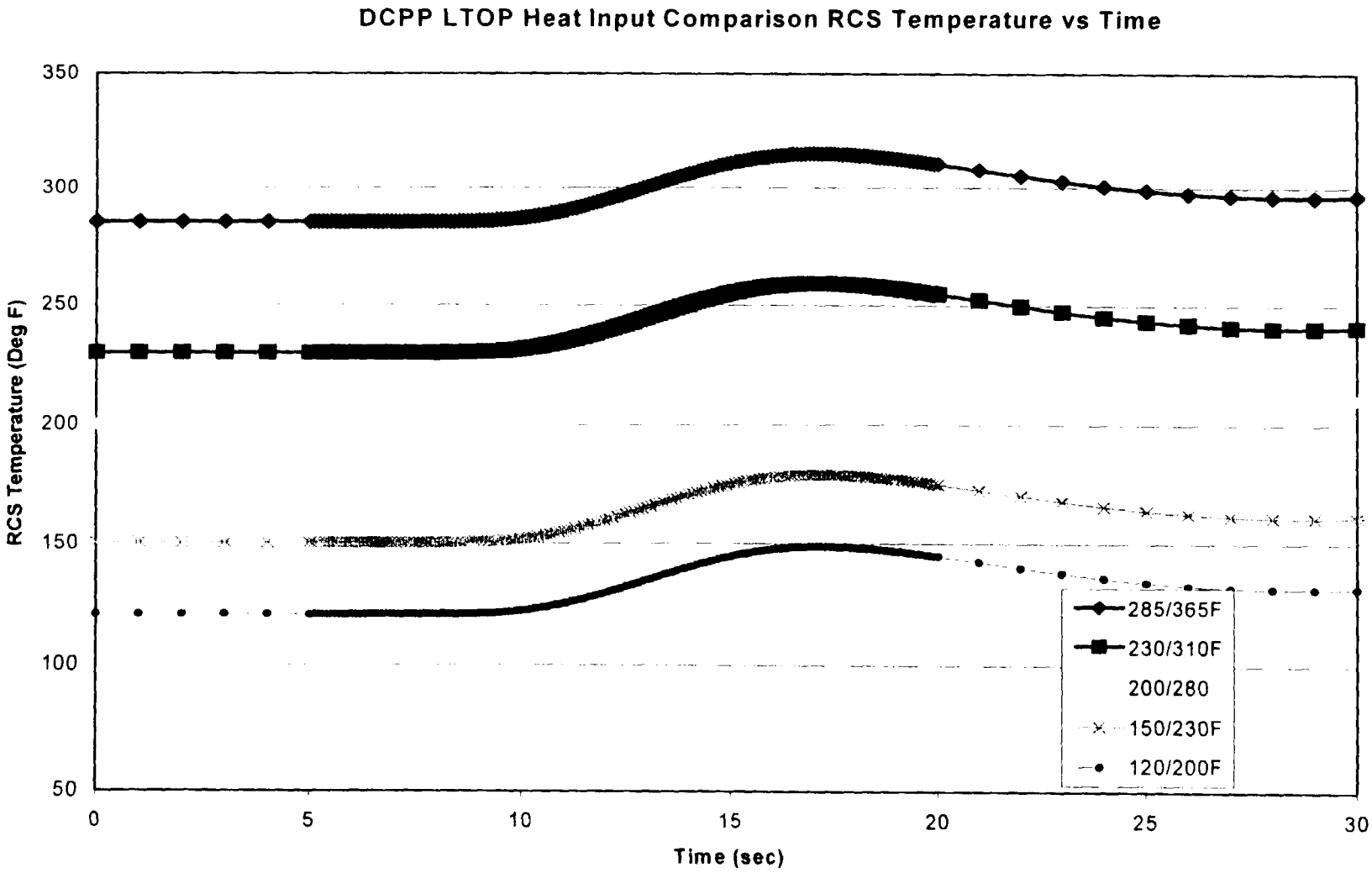


Figure 6- 8 : DCPP RETRAN LTOP Heat Input P/T Summary

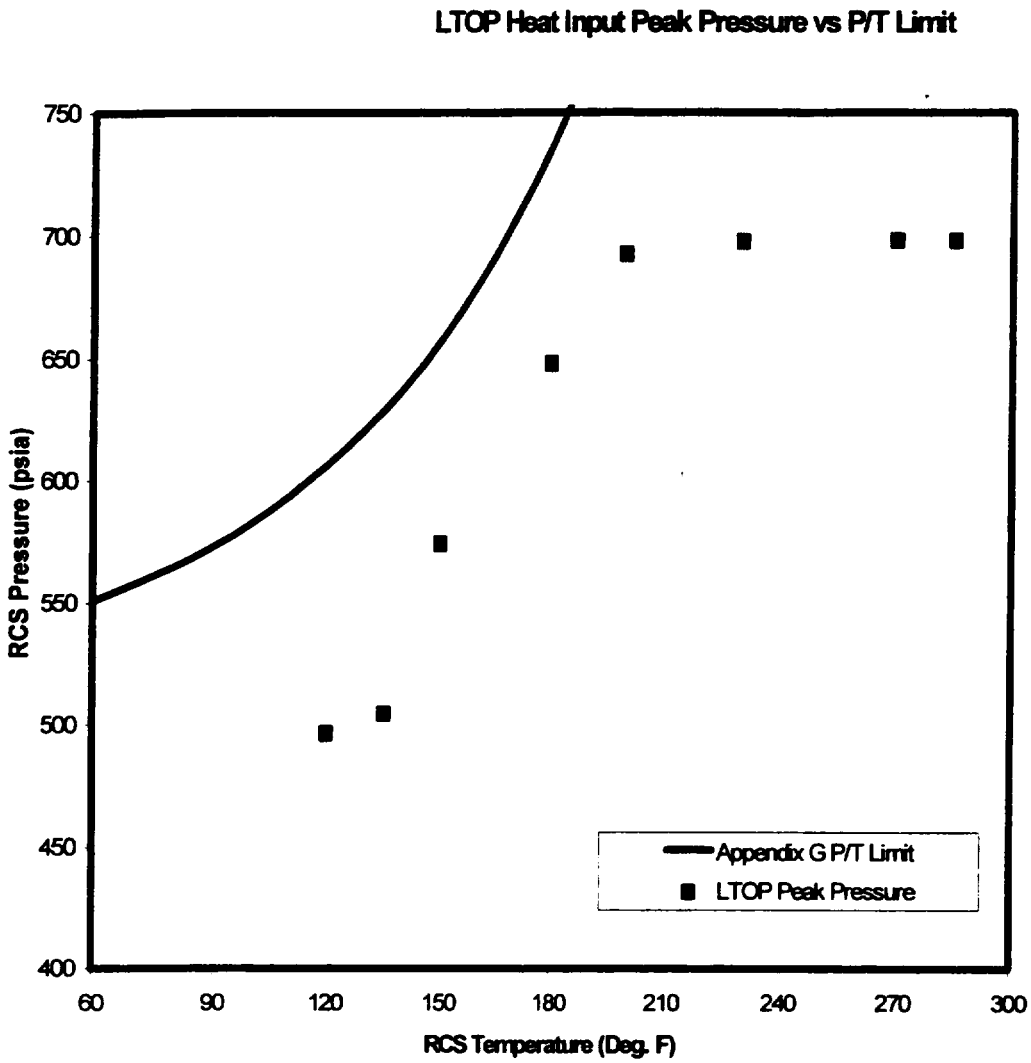
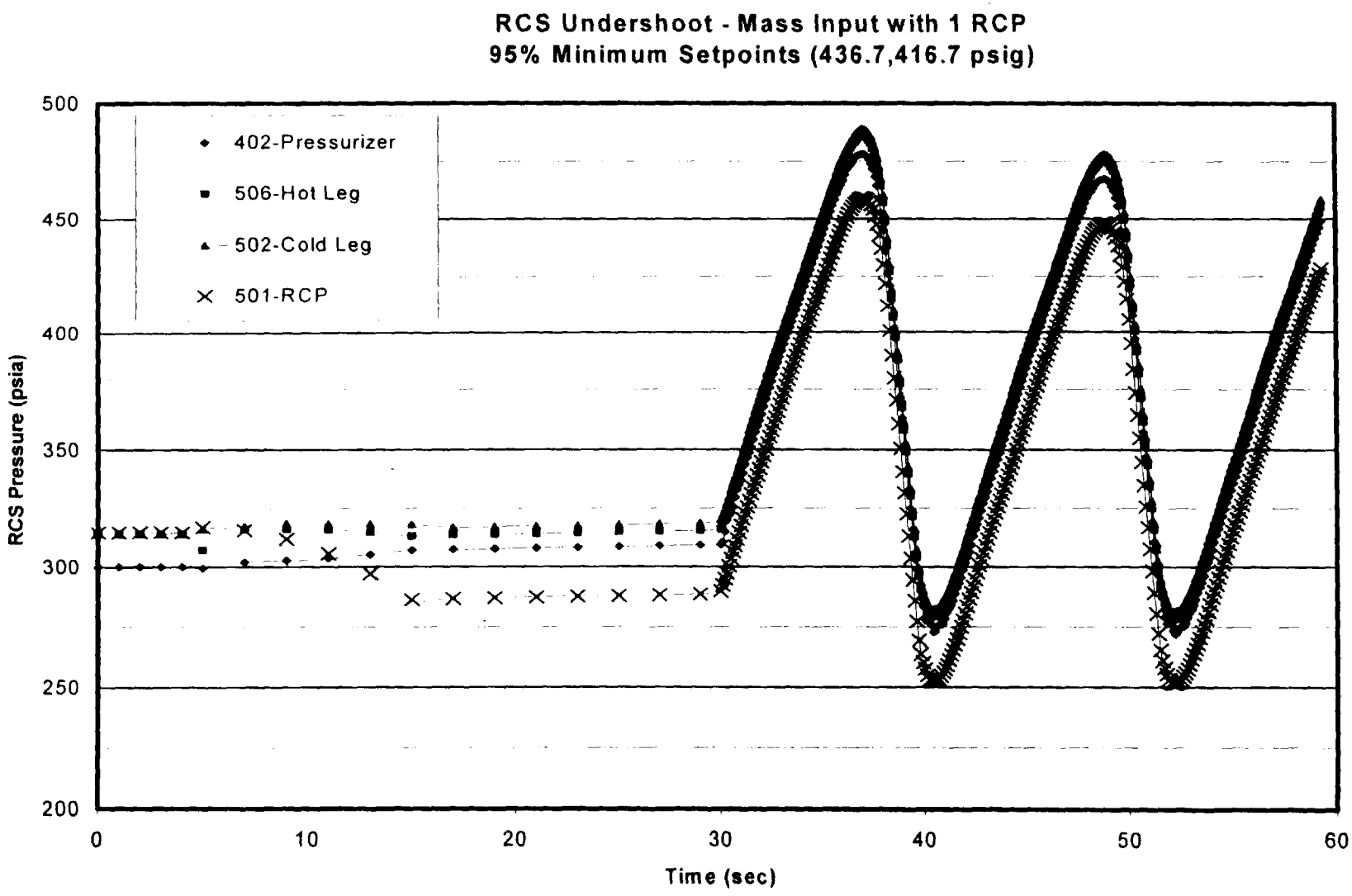


Figure 6- 9 : DCPP RETRAN LTOP 1 RCP Mass Input RCS Undershoot



SUBJECT RETRAN Evaluation of DCPP LTOP Parameters
MADE BY Jerry E. Ballard DATE 11/ 26 /01 CHK'D BY Dixon Yee DATE 11/ 26 /01

Figure 6- 10 : DCPP RETRAN LTOP 4 RCP Mass Input RCS Undershoot

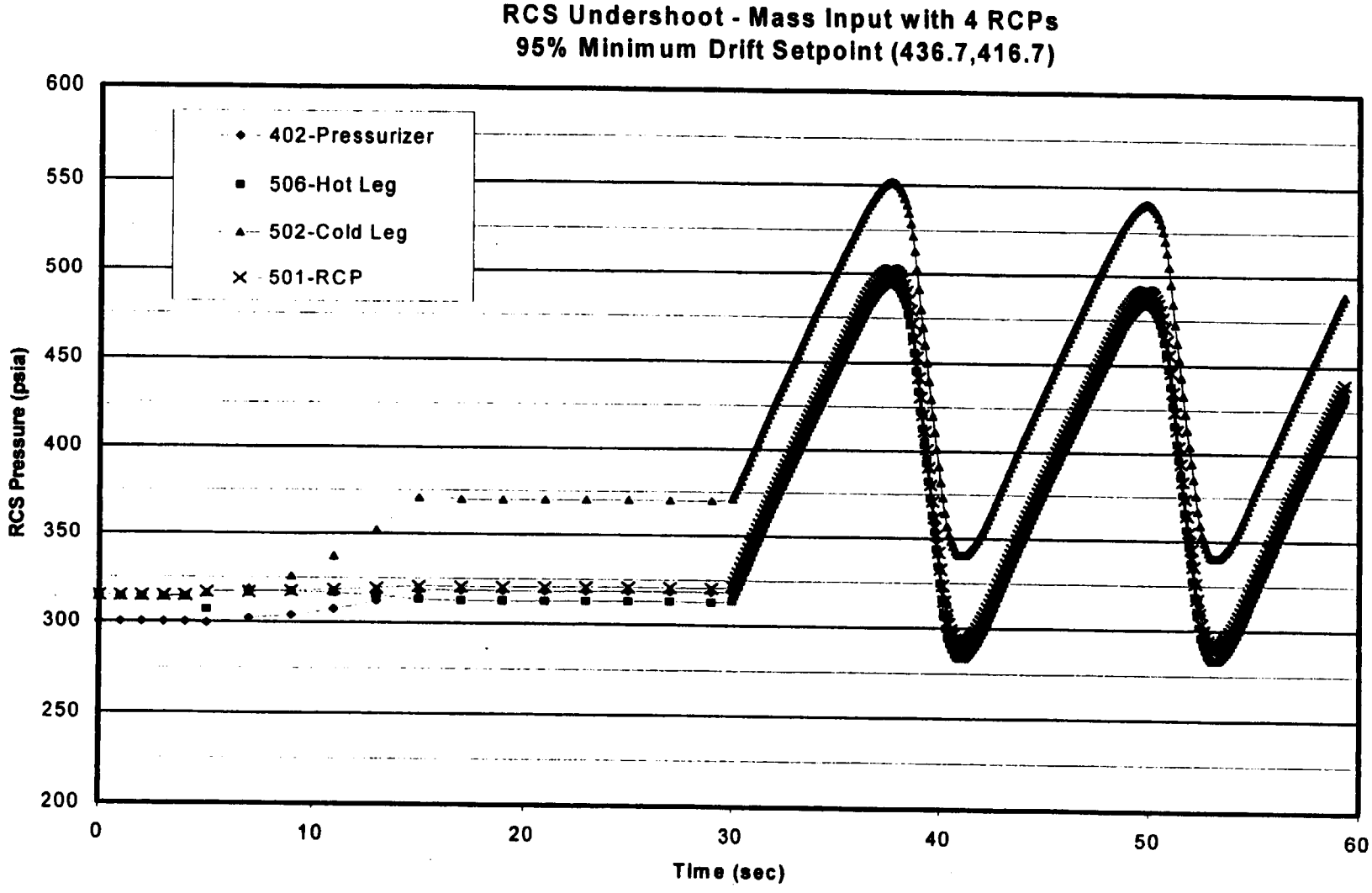
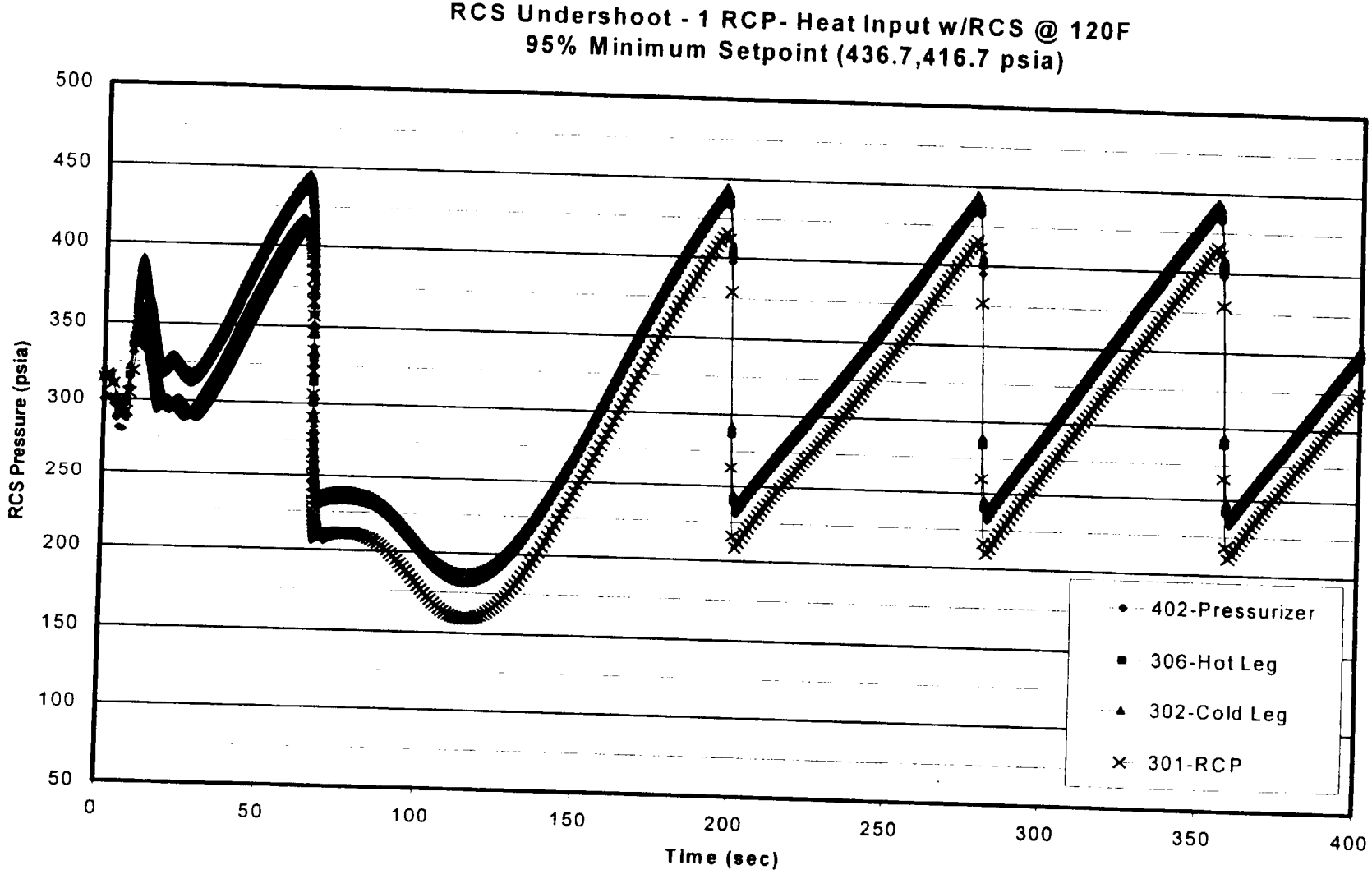


Figure 6- 11 : DCPP RETRAN LTOP Heat Input RCS Undershoot @ 120 F



SUBJECT RETRAN Evaluation of DCPP LTOP Parameters
MADE BY Jerry E. Ballard DATE 11/ 26 /01 CHK'D BY Dixon Yee DATE 11/ 26 /01

Figure 6- 12 : DCPP RETRAN LTOP Heat Input RCS Undershoot @ 180 F

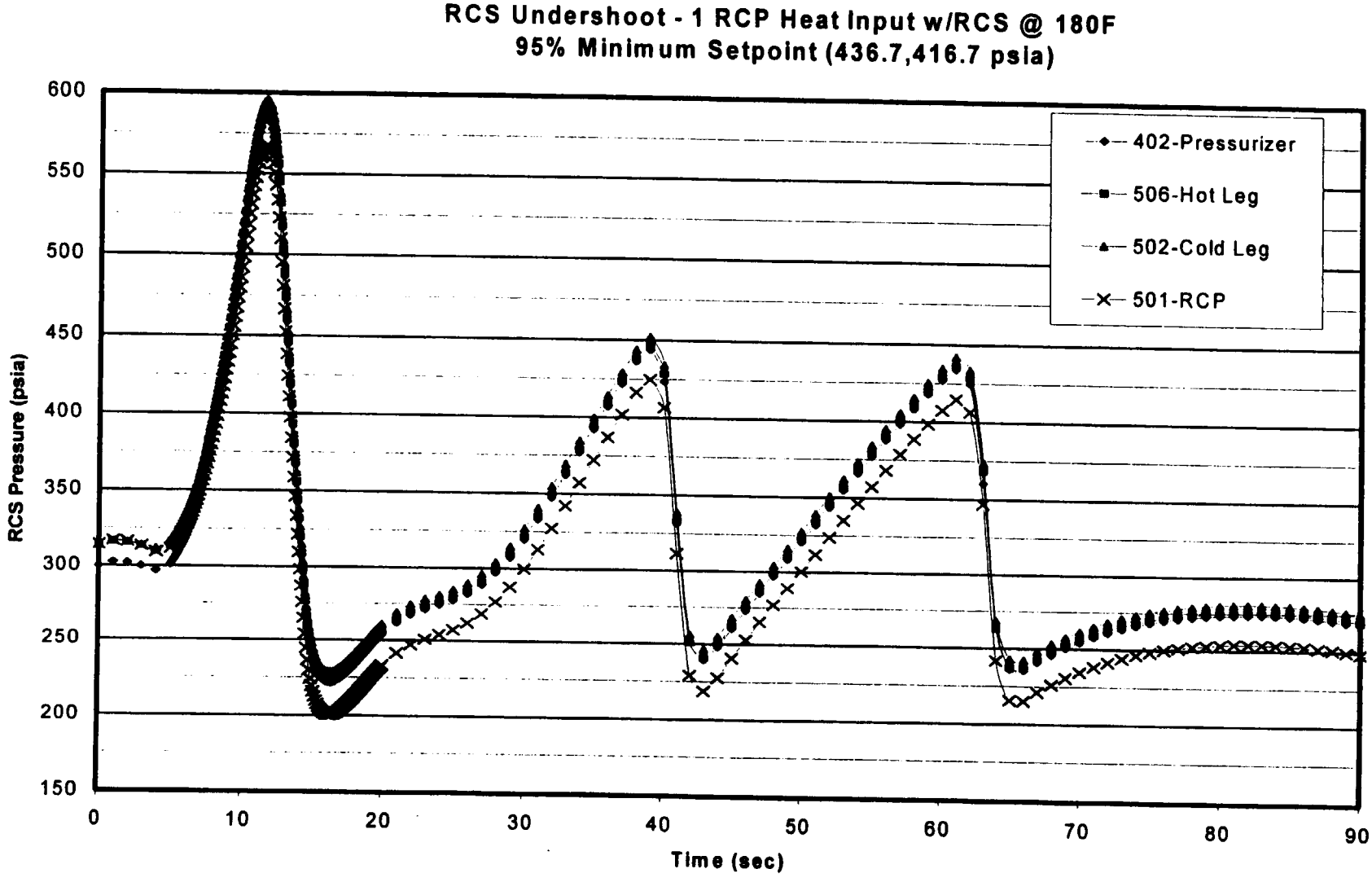
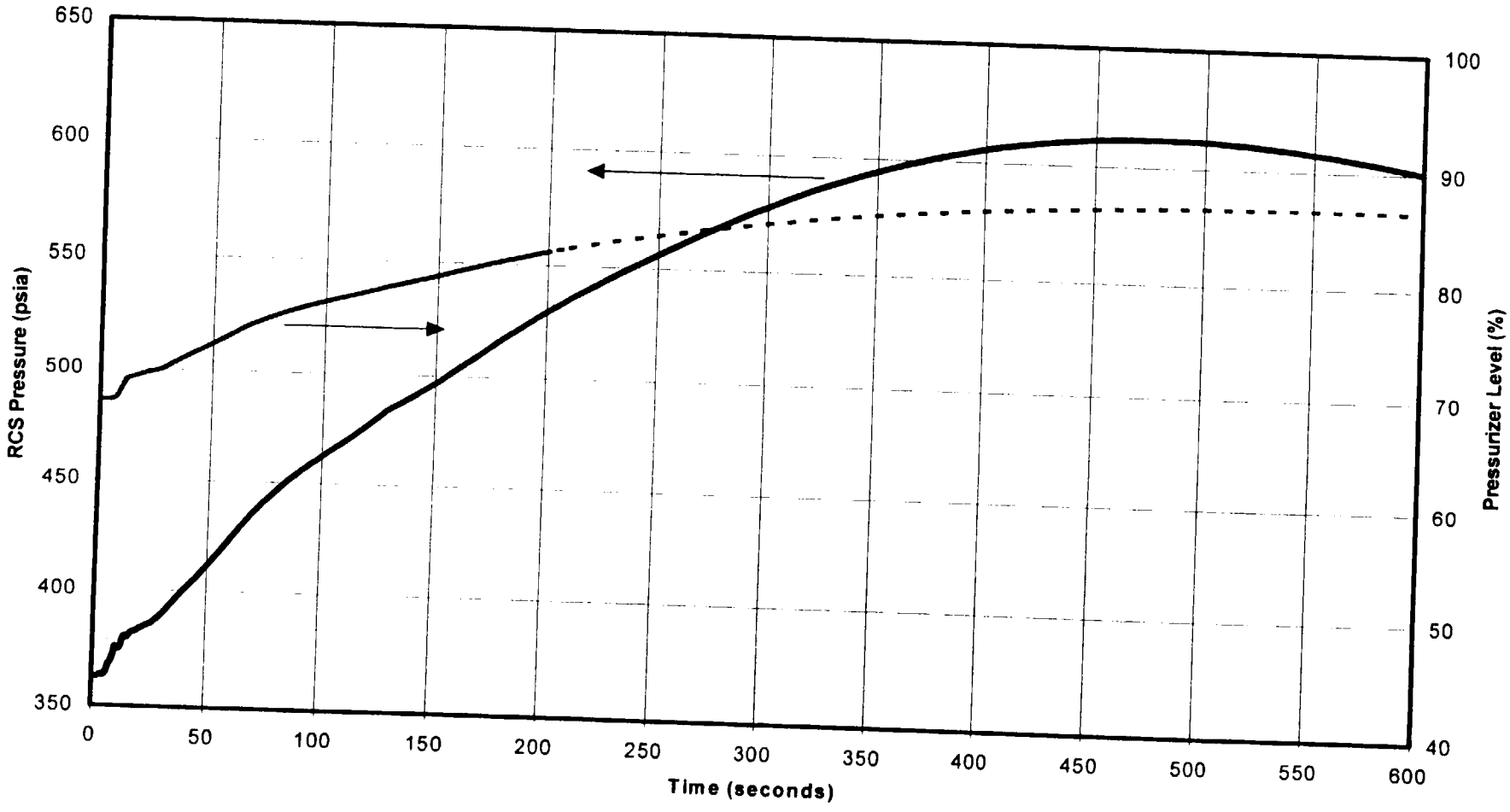


Figure 6- 13 DCPP RETRAN LTOP Heat Input Pzr @ 50%

RETRAN Heat Input Response RCS/SG = 270/420F, Pressurizer Level @ 67%



Supporting References for DCPP PTLR

This enclosure contains:

1. CE Power Systems letter to Westinghouse Nuclear Energy Systems dated September 12, 1979, transmitted to PG&E by Westinghouse letter PGE-4083 dated July 21, 1980 as amended by Westinghouse letter PGE-6352 dated December 20, 1984.

2. Westinghouse letter PGE-88-765 dated December 14, 1988.

C63645

PGE-6352

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Operations Division

Box 355
Pittsburgh Pennsylvania 15230

December 20, 1984

Ref: PGE-4083

J. V. Rocca
Chief Mechanical Engineer
Pacific Gas & Electric Company
c/o Bechtel Power Corporation
Diablo Canyon Project
45 Fremont Street, 10th Floor, Room D28
San Francisco, CA 94602

Attention: J. J. McCracken

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR PLANT, DIABLO CANYON UNITS 1 & 2
Reactor Vessel Minimum Bolt-Up Temperatures

Dear Mr. Rocca:

In the referenced letter, Westinghouse forwarded for inclusion in the reactor vessel manual a Combustion Engineering letter stating that the minimum bolt-up temperature is RTNDT per the ASME Code requirements. This RTNDT per the ASME Code is that of the affected areas, the upper shell and vessel flange. The referenced letter itself stated that even if RTNDT was below 60°F a minimum bolt-up temperature of 60°F should be used.

Since the RTNDT's for the affected areas given in the Diablo Canyon Unit 1 Tech Specs and the Draft Diablo Canyon Unit 2 Tech Specs transmitted to PGandE by Westinghouse are below 60°F, a minimum bolt-up temperature of 60°F is applicable to both units. This bolt-up temperature is applicable for the life of the vessels.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

John C. Hoebel, Manager
Pacific Gas and Electric Project

CWVernon/rcc/2479d

cc: J. V. Rocca 1L
J. E. Murphy (W San Francisco Office) 1L
J. B. Hoch 1L
W. Vogsland 1L
R. C. Thornberry 1L

RECEIVED
DEC 31, 1984
DIABLO CANYON 15320

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PGE-4083
LETTER NO.

DIABLO CANYON
Unit #1 & #2
☑ Original
File: 140.010

☐ Copy For:
☐ Eng. File
☐ File 1.13
☐ RML
☐ JJM
☐ JDS
☐

663201

Westinghouse Electric Corporation — Water Reactor Divisions

Nuclear Commercial Operations Division
Box 355
Pittsburgh Pennsylvania 15230

Mr. D. V. Kelly
Chief Mechanical Engineer
PACIFIC GAS AND ELECTRIC COMPANY
77 Beale Street
San Francisco, California 94106

Date July 21, 1980
S.O. No. PEG/PGE-105
Engr. Ltr. EP/SA-2696

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR PLANT, DIABLO CANYON SITE
REVISION TO REACTOR VESSEL INSTRUCTION MANUAL

The following documents are transmitted herewith for your use. The status of each docum is one of the following as noted below:

- Preliminary (PRE)
- Approved For Layout (AFL)
- Certified For Construction (CFC)
- Certified For Construction With Comments (CC
- As Manufactured (ASM)
- Approved (APP)

SPIN No.	Document No.	Sht.	Rev.	Status	Unit	Document Title
RCPCRV	C.E. Letter Sept. 12, 1979	-	-	APP	1	Revision to Reactor Vessel Instruction Manual
RCPCRV	C.E. Letter Sept. 12, 1979	-	-	APP	2	Revision to Reactor Vessel Instruction Manual

Comment: The attached letter relays the minimum bolt-up temperature for the reactor ves: studs. Please insert this letter into your Reactor Vessel Instuction Manuals modify your bolt-up procedures appropriately. However, Westinghouse adds an a tional requirement that the bolt-up temperature shall not be below 60°F.

WESTINGHOUSE NUCLEAR ENERGY SYSTEMS

W. C. Gangloff, Project Manager
Pacific Gas and Electric Project

J. F. Duran
AAG/YH/mm

D. V. Kelly 5L, 17A
R. W. Beckwith 1L
(W San Francisco)

0002210652

C-E Power Systems
Combustion Engineering, Inc.
911 W. Main Street
Chattanooga, Tennessee 37402

Tel. 615/265-4631

POWER SYSTEMS

September 12, 1979

Westinghouse Electric Corporation
Nuclear Energy Systems
Northern Pike Road
Monroeville, Pennsylvania 15146

Attention: Mr. N. A. Ditillo
Subject: Reactor Vessel Bolt-up and Hydrotest Temperature Requirements
Reference: Contract PGE/67757/23066

Gentlemen:

The instruction manual for the reference contract is hereby upgraded in conformance with ASME Code Section III, Appendix G, Paragraph 2222(c), Summer 1976 Addenda. This applies only to bolt-up and hydrotest pressure requirement.

Bolt-up temperature shall not be lower than RT_{NDT} at a pressure not to exceed 600 psig.

Reactor Vessel pressurization above 600 psig will be at a temperature not lower than RT_{NDT} +60°F.

Yours very truly,

COMBUSTION ENGINEERING, INC.

Jack L. Fyle
Contract Administrator

JLP/jpm

cc - Mr. J. A. Abel, Jr.
Mr. A. B. Harper
Mr. J. S. Neek