Pacific Gas and Electric Company

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January 18, 2000

PG&E Letter DCL-00-007

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

Docket No. 50-275, OL-DPR-80
Diablo Canyon Unit 1
Supplement to License Amendment Request 99-03, Unit 1 Reactor Core Thermal Power Uprate

Dear Commissioners and Staff:

By letter dated December 31, 1999 (PG&E Letter DCL-99-170, "License Amendment Request 99-03, Unit 1 Reactor Core Thermal Power Uprate"), PG&E submitted license amendment request (LAR) 99-03 to amend the facility operating license for Diablo Canyon Power Plant (DCPP) Unit 1 to increase the reactor core power level to 3411 megawatts thermal (100 percent rated power).

LAR 99-03 also includes proposed changes to the DCPP Final Safety Analysis Report Update (Enclosure G to PG&E Letter DCL-99-170), and proposed changes to the DCPP Precautions, Limitations, and Setpoints document (Enclosure H to PG&E Letter DCL-99-170). Certain pages in those enclosures are incorrectly marked "Westinghouse Proprietary Class 2C." The information contained on those pages is nonproprietary, and need not be withheld from public disclosure. Attached are revised enclosures containing corrected pages with the proprietary marking removed. These enclosures supersede the same lettered enclosures included in PG&E Letter DCL-99-170.

The changes proposed in this supplement do not impact the safety evaluation or the no significant hazards consideration determination provided in LAR 99-03.

Sincerely,

DH Oatley

David H. Oatley

A001 1/1

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

Enclosures

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of PACIFIC GAS AND ELECTRIC COMPANY Diablo Canyon Power Plant Unit 1	Docket No. 50-275 Facility Operating License No. DPR-80

AFFIDAVIT

David H. Oatley, of lawful age, first being duly sworn upon oath says that he is Vice President - Diablo Canyon Operations and Plant Manager of Pacific Gas and Electric Company; that he is familiar with the content thereof; that he has executed this Supplement to License Amendment Request 99-03 on behalf of said company with full power and authority to do so; and that the facts stated therein are true and correct to the best of his knowledge, information, and belief.

D. H. Oatley

David H. Oatley
Vice President - Diablo Canyon Operations
and Plant Manager

Subscribed and sworn to before me this 18th day of January, 2000.

Amy J. Calloway

Notary Public
State of California
County of San Luis Obispo

AMY J. CALLOWAY
COMM. #1096602
NOTARY PUBLIC CALIFORNIA
SAN LUIS OBISPO COUNTY
My Comm. Expires April 28, 2000

**ENCLOSURE G OF
PG&E LETTER DCL-99-170**

MARK-UP OF FINAL SAFETY ANALYSIS REPORT UPDATE

Final Safety Analysis Report (FSAR) Update proposed changes related to the Unit 1 uprate:

Chapter 1	Changes which reflect overall plant description
Chapter 4	Changes that relate to fuel design. Pages shown are the current draft and may be further modified following a Westinghouse review. The intent is to reduce references to LOPAR fuel and update the Unit 1 values to reflect the uprated condition.
Chapter 5	Changes which reflect the revised residual heat removal (RHR) cooldown calculation. These changes include more conservative inputs and a specification of the design criteria, rather than a particular analysis result. This is not a reflection of reduced capability or greater load on the RHR system. Both the prior and new RHR cooldown calculations assume a 3411 MWt licensed core power.
Chapter 6	Changes which reflect the revised hydrogen generation calculation were placed into the FSAR Update in Revision 12, September 1998, and are not reproduced here.
Chapter 10	Changes in electric generator performance requirements.
Chapter 15.1	Changes which eliminate the need for describing Unit 1 and Unit 2 power differences, and which update references.
Chapter 15.2	Changes which relate to the new OTΔT/OPΔT setpoint calculations and accidental reactor coolant system depressurization.
Chapter 15.3	Changes related to the new small break loss-of-coolant accident (LOCA) analysis. (Note: Though included here, these changes are not contingent upon this license amendment request, but rather upon approval of PG&E's request in letter DCL-99-099, "Supplement to License Amendment Request 98-09," to use the COSI methodology of WCAP-10054-P-A, Addendum 2, Revision 1. Those changes were approved in License Amendments 136 and 136, for Units 1 and 2, respectively, dated November 15, 1999.)

Chapter 15.4 Changes which reflect the revised large break LOCA were placed into the FSAR Update in Revision 12, September 1998, and are not reproduced here.

CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

The Final Safety Analysis Report (FSAR) Update for the Diablo Canyon Power Plant (DCPP) is submitted in accordance with the requirements of 10 CFR 50.71(e) and contains all the changes necessary to reflect information and analyses submitted to the U.S. Nuclear Regulatory Commission (NRC) by Pacific Gas and Electric Company (PG&E) or prepared by PG&E pursuant to NRC requirements since the submittal of the original FSAR. The original FSAR was submitted in support of applications for permits to operate two substantially identical nuclear power units (Unit 1 and Unit 2) at the DCPP site. The DCPP site is located on the central California coast in San Luis Obispo County, approximately 12 miles west southwest of the city of San Luis Obispo.

The Construction Permit for Unit 1 (CPPR-39) was issued April 23, 1968, in response to PG&E's application dated January 16, 1967 (USAEC, Docket No. 50-275). The Construction Permit for Unit 2 (CPPR-69) was issued on December 9, 1970; the application was made on June 28, 1968 (USAEC, Docket No. 50-323).

Westinghouse Electric Corporation and PG&E jointly participated in the design and construction of each unit. The plant is operated by PG&E. Each unit employs a pressurized water reactor (PWR) nuclear steam supply system (NSSS) furnished by Westinghouse Electric Corporation and similar in design concept to several projects licensed by the NRC. Certain components of the auxiliary systems are shared by the two units, but in no case does such sharing compromise or impair the safe and continued operation of either unit. Those systems and components that are shared are identified and the effects of the sharing are discussed in the chapters in which they are described. The NSSS for each unit is contained within a steel-lined reinforced concrete structure that is capable of withstanding the pressure that might be developed as a result of the most severe postulated loss-of-coolant (LOCA) accident. The containment structure was designed by PG&E to meet the requirements specified by Westinghouse Electric Corporation.

While the reactors, structures, and all auxiliary equipment are substantially identical for the two units, there is a difference in the reactor internal flow path that results in a lower coolant flow rate for Unit 1. Consequently, the original license application reactor ratings were~~are~~ 3338 MWt for Unit 1 and 3411 MWt for Unit 2. The corresponding ~~estimated~~ net electrical outputs were approximately~~are~~ 1084 MWe and 1106 MWe, respectively.

During the design phase, the~~The~~ expected ultimate output of the Unit 1 reactor was ~~is~~ 3488 MWt; the expected ultimate output of the Unit 2 reactor was~~is~~ 3568 MWt. The corresponding NSSS outputs were~~are~~ 3500 MWt and 3580 MWt. (The difference of 12 MWt is due to the net contribution of heat to the reactor coolant system from nonreactor sources, primarily pump heat.) The corresponding estimated ultimate net electrical outputs wereare 1131 MWe for Unit 1 and 1156 MWe for Unit 2.

The NRC issued a low power operating license for Unit 1 on September 22, 1981. PG&E voluntarily postponed fuel loading due to the discovery of design errors in the annulus region of the containment structure. Subsequently, the NRC revoked the low power operating license on November 19, 1981, pending completion of redesign and construction activities.

After completion of redesign and construction activities in November 1983, the NRC reinstated the fuel load portion of the Unit 1 low power operating license. On April 19, 1984, the NRC fully reinstated the low power operating license, which included low power testing. The Unit 1 full power operating license was issued on November 2, 1984. Commercial operation for Unit 1 began on May 7, 1985, with a license expiration date of April 23, 2008.

The NRC issued a low power operating license for Unit 2 on April 26, 1985. Unit 2 fuel loading was completed on May 15, 1985. A full power operating license for Unit 2 was issued on August 26, 1985. Unit 2 commercial operation began on March 13, 1986, with a license expiration date of December 9, 2010.

In March 1996, the NRC approved license amendments extending the operating license for Unit 1 until September 22, 2021, and for Unit 2 until April 26, 2025.

In ______ 2000, the NRC approved a license amendment for Unit 1 to increase its rated thermal power from the original licensed value of 3338 MWt to 3411 MWt to increase electric production and be consistent with Unit 2.

Chapter 4

REACTOR

This chapter describes the design for the reactors at Diablo Canyon Power Plant (DCPP) Units 1 and 2, and evaluates their capability to function safely under all operating modes expected during their lifetimes.

4.1 SUMMARY DESCRIPTION

This chapter describes the following subjects: (a) the mechanical components of the reactor and reactor core, including the fuel rods and fuel assemblies, reactor internals, and the control rod drive mechanisms, (b) the nuclear design, and (c) the thermal-hydraulic design.

The~~Beginning with Cycle 6, the~~ reactor core of each unit typically consists of VANTAGE 5 fuel assemblies, instead of the low parasitic (LOPAR) fuel previously used. ~~On occasion, one or more used LOPAR fuel assemblies may be reinserted in the reactor, if warranted, following the normal reload analysis process.~~ Some of the current Chapter 15 accident analyses, including the large break and small break loss of coolant accidents, assume an all Vantage 5 core. Therefore, it is not expected that LOPAR fuel will be used without further analysis. Nevertheless, this section addresses both LOPAR fuel assemblies and Vantage 5 arranged in a low leakage core loading pattern.~~The reference design described herein consists of LOPAR fuel assemblies and VANTAGE 5 fuel assemblies arranged in a low leakage core loading pattern.~~

The significant mechanical design features of the VANTAGE 5 design, as defined in Reference 1, relative to the LOPAR fuel design may include the following:

- Integral Fuel Burnable Absorber (IFBA)
- Intermediate Flow Mixer (IFM) Grids
- Reconstitutable Top Nozzle (RTN)
- Slightly longer fuel rods and thinner top and bottom nozzle end plates to accommodate extended burnup
- Axial Blanket (typically six inches of natural or slightly enriched UO_2 at both ends of fuel stack
- Replacement of six intermediate Inconel grids with zirconium alloy grids
- Reduction in fuel rod, guide thimble and instrumentation tube diameter

4.3.1.2.2 Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption effects (Doppler) associated with changing fuel temperature, and the spectrum effect resulting from changing moderator density. These basic physics characteristics are often identified by reactivity coefficients. The use of slightly enriched uranium ensures that the Doppler coefficient of reactivity, which provides the most rapid reactivity compensation, is negative. The core is also designed to have an overall negative MTC of reactivity at full power so that average coolant temperature or void content provides another, slower, compensatory effect. A small positive MTC is allowed at low power. The negative MTC at full power can be achieved through use of fixed burnable absorbers and/or boron coated fuel pellets and/or control rods by limiting the reactivity held down by soluble boron.

Burnable absorber content (quantity and distribution) is not stated as a design basis other than as it relates to achieving a nonpositive MTC at power operating conditions, as discussed above.

4.3.1.3 Control of Power Distribution

4.3.1.3.1 Basis

The nuclear design basis, with at least a 95 percent confidence level, is as follows:

(1) The fuel will not be operated at greater than ~~13.3 kW/ft (Unit 1) or~~ 13.6 kW/ft ~~(Unit 2)~~ under normal operating conditions, including an allowance of 2 percent for calorimetric error and densification effects.

(2) Under abnormal conditions, including the maximum overpower condition, the fuel peak power will not cause melting as defined in Section 4.4.1.2.

(3) The fuel will not operate with a power distribution that violates the departure from nucleate boiling (DNB) design basis (i.e., the departure from nucleate boiling ratio (DNBR) shall not be less than the design limit DNBR, as discussed in Section 4.4.1) under Conditions I and II events, including the maximum overpower condition.

(4) Fuel management will be such as to produce fuel rod powers and burnups consistent with the assumptions in the fuel rod mechanical integrity analysis of Section 4.2.

The above basis meets GDC 10.

Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements. For example, Figure 4.3-24 illustrates BOL, MOL, and EOL steady state conditions.

Finally, this upper bound envelope is based on operation within an allowed range of axial flux steady state conditions. These limits are detailed in the Core Operating Limits Reports and rely only on excore surveillance supplemented by the required normal monthly full core map. If the axial flux difference exceeds the allowable range, an alarm is actuated.

Allowing for fuel densification, the average linear ~~power at 3338 MWt is 5.33 kW/ft for Unit 1, and~~ power is 5.44 kW/ft for both units~~Unit 2~~ at 3411 MWt. From Figure 4.3-23, the conservative upper bound value of normalized local power density, including uncertainty allowances, is 2.45, corresponding to a peak linear power of ~~13.3 kW/ft and~~ 13.6 kW/ft at 102 percent power ~~for Units 1 and 2, respectively~~.

To determine reactor protection system setpoints, with respect to power distributions, three categories of events are considered: rod control equipment malfunctions, operator errors of commission, and operator errors of omission. In evaluating these three categories, the core is assumed to be operating within the four constraints described above.

The first category is uncontrolled rod withdrawal (with rods moving in the normal bank sequence). Also included are motions of the banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions were calculated, assuming short-term corrective action. That is, no transient xenon effects were considered to result from the malfunction. The event was assumed to occur from typical normal operating situations, which include normal xenon transients. It was also assumed that the total power level would be limited by the reactor trip to below 118 percent. Results are given in Figure 4.3-21 in units of kW/ft. The peak power density which can occur in such events, assuming reactor trip at or below 118 percent, is less than that required for fuel centerline melt, including uncertainties and densification effects (Figure 4.3-20).

The second category, also appearing in Figure 4.3-21, assumes that the operator mispositions the rod bank in violation of insertion limits and creates short-term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a flux difference violation. The results shown in Figure 4.3-22 are F_Q^T multiplied by 102 percent power, including an allowance for calorimetric error. The peak linear power does not exceed 21.1 kW/ft, provided the operator's error does not continue for a period which is long compared to the xenon time constant. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error. Additional detailed discussion of these analyses is presented in Reference 23.

4.4.2.2.6 Fuel Cladding Temperatures

The fuel rod outer surface at the hot spot operates at a temperature of approximately 660°F for steady state operation at rated power throughout core life, due to the onset of nucleate boiling. At beginning of life (BOL), this temperature is that of the cladding metal outer surface.

During operation over the life of the core, the buildup of oxides and crud on the fuel rod cladding outer surface causes the cladding surface temperature to increase. Allowance is made in the fuel center melt evaluation for this temperature rise. The thermal-hydraulic DNB limits ensure that adequate heat transfer is provided between the fuel cladding and the reactor coolant so that cladding temperature does not limit core thermal output. Figure 4.4-4 shows the axial variation of average cladding temperature for the average power rod both at beginning and end of life (EOL).

4.4.2.2.7 Treatment of Peaking Factors

The total heat flux hot channel factor, F_Q^T, is defined by the ratio of the maximum to core average heat flux. The design value of F_Q^T for normal operation is 2.45 including fuel densification effects as shown in Table 4.3-1. This results in a peak local linear power density of ~~13.06 and~~ 13.34 kW/ft at full power~~ for Units 1 and 2, respectively~~. The corresponding peak local power at the maximum overpower trip point is 18 kW/ft. Centerline temperature at this kW/ft must be below the UO_2 melt temperature over the lifetime of the rod including allowances for uncertainties. From Figure 4.4-2, the centerline temperature at the maximum overpower trip point is well below that required to produce melting. Fuel centerline and average temperature at rated (100 percent) power and at the maximum overpower trip point for Units 1 and 2 are presented in Table 4.1-1.

4.4.2.3 Departure from Nucleate Boiling Ratio

The minimum DNBRs for the rated power, and anticipated transient conditions are given in Table 4.1-1 for Units 1 and 2. The minimum DNBR in the limiting flow channel will occur downstream of the peak heat flux location (hot spot) due to the increased downstream enthalpy rise.

DNBRs are calculated by using the correlation and definitions described in Section 4.4.2.3.1. The THINC-IV[47] computer code (discussed in Section 4.4.3.4.1) determines the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The use of hot channel factors is discussed in Section 4.4.3.2.1 (nuclear hot channel factors) and in Section 4.4.2.3.4 (engineering hot channel factors).

	LOPAR	VANTAGE 5
Design Limit		
Typical Cell	1.38	1.34~~3~~
Thimble Cell	1.34	1.32~~1~~
Safety Limit		
Typical Cell	1.48	1.71
Thimble Cell	1.44	1.68

The maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU based on Reference 88. At burnups greater than 24,000 MWD/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown. Due to the decrease in fissionable isotopes and the buildup of fission product inventory, no additional rod bow penalty is required.

4.4.2.3.6 Transition Core

The Westinghouse transition core DNB methodology is given in References 89 and 90 and has been approved by the NRC via Reference 91. Using this methodology, transition cores are analyzed as if they were full cores of one assembly type (full LOPAR or full VANTAGE 5), applying the applicable transition core penalties. This penalty was~~will be~~ included in the safety analysis limit DNBRs such that sufficient margin over the design limit DNBR existeds to accommodate the transition core penalty and the appropriate rod bow DNBR penalty. However, since the transition to a full VANTAGE 5 core has been completed, various analyses, such as large break and small break loss of coolant accident analysis, have assumed a full VANTAGE 5 core and no longer assume a transition core penalty.

The LOPAR and VANTAGE 5 designs have been shown to be hydraulically compatible in Reference 85.

4.4.2.4 Flux Tilt Considerations

Significant quadrant power tilts are not anticipated during normal operation since this phenomenon is caused by asymmetric perturbations. A dropped or misaligned RCCA could cause changes in hot channel factors. These events are analyzed separately in Chapter 15.

Other possible causes for quadrant power tilts include X-Y xenon transients, inlet temperature mismatches, enrichment variations within tolerances, and so forth.

In addition to unanticipated quadrant power tilts, other readily explainable asymmetries may be observed during calibration of the excore detector quadrant power tilt alarm. During operation, at least one incore map is taken per effective-full-power month; additional maps are obtained periodically for calibration purposes. Each of these maps is reviewed for deviations movement of the fuel rods relative to the grids. Thermal expansion of fuel rods is considered in the grid design so that axial loads imposed on the fuel rods during a thermal transient will not result in excessively bowed fuel rods (see Section 4.2.1.2.2).

4.4.3.8 Energy Release During Fuel Element Burnout

As discussed in Section 4.4.3.3, the core is protected from going through DNB over the full range of possible operating conditions. At full power operation, the minimum DNBR was found to be ~~2.35 (LOPAR) and~~ 2.53 [THIS VALUE WILL BE FURTHER UPDATED WITH INPUT FROM WESTINGHOUSE] (VANTAGE 5) for Unit 1 and ~~2.29 (LOPAR) and~~ 2.47 (VANTAGE 5) for Unit 2. This means that, for these conditions, the probability of a rod going through DNB is less than 0.1 percent at 95 percent confidence level based on the statistics of the ~~WRB-1 and~~ WRB-2 correlations[84,85]. In the extremely unlikely event that DNB should occur, cladding temperature will rise due to steam blanketing the rod surface and the consequent degradation in heat transfer. During this time a potential for a chemical reaction between the cladding and the coolant exists. Because of the relatively good film boiling heat transfer following DNB, the energy release from this reaction is insignificant compared to the power produced by the fuel. These results have been confirmed in DNB tests conducted by Westinghouse[66,78].

4.4.3.9 Energy Release During Rupture of Waterlogged Fuel Elements

A full discussion of waterlogging including energy release is contained in Section 4.4.3.6.

4.4.3.10 Fuel Rod Behavior Effects from Coolant Flow Blockage

Coolant flow blockage can occur within the coolant channels of a fuel assembly or external to the reactor core. The effect of coolant flow blockage within the fuel assembly on fuel rod behavior is more pronounced than external blockages of the same magnitude. In both cases, the flow blockages cause local reductions in coolant flow. The amount of local flow reduction, its location in the reactor, and how far downstream does the reduction persist, are considerations that influence fuel rod behavior. Coolant flow blockage effects in terms of maintaining rated core performance are determined both by analytical and experimental methods. The experimental data are usually used to augment analytical tools such as the THINC-IV program. Inspection of the DNB correlation (Section 4.4.2.3) shows that the predicted DNBR depends on local values of quality and mass velocity.

The THINC-IV code can predict the effects of local flow blockages on DNBR within the fuel assembly on a subchannel basis, regardless of where the flow blockage occurs. THINC-IV accurately predicts the flow distribution within the fuel assembly when the inlet nozzle is completely blocked (Reference 59). For the DCPP reactors operating at nominal full power conditions as specified in Table 4.1-1, the effects of an increase in enthalpy and decrease in mass velocity in the lower portion of the fuel assembly would not result in the reactor reaching the safety limit DNBR.

The analyses, which assume fully developed flow along the full channel length, show that a reduction in local mass velocity greater than ~~75 percent (LOPAR) and~~ 56 percent [THIS VALUE WILL BE FURTHER UPDATED WITH INPUT FROM WESTINGHOUSE](VANTAGE 5) for Unit 1 and ~~72 percent (LOPAR) and~~ 53 percent (VANTAGE 5) for Unit 2 would be required to reduce the DNBRs from the DNBRs at the nominal conditions shown in
Table 4.4-1 to the safety limit DNBRs. In reality, a local flow blockage is expected to promote turbulence and thus would likely not effect DNBR.

Coolant flow blockages induce local cross flows as well as promoting turbulence. Fuel rod vibration could occur, caused by this cross flow component, through vortex shedding or turbulent mechanisms. If the cross flow velocity exceeds the limit established for fluid elastic stability, large amplitude whirling will result in, and can lead to, mechanical wear of the fuel rods at the grid support locations. The limits for a controlled vibration mechanism are established from studies of vortex shedding and turbulent pressure fluctuations. Fuel rod wear due to flow-induced vibration is considered in the fuel rod fretting evaluation (Section 4.2).

4.4.3.11 Pressurization Analyses for Shutdown Conditions

The objective of these analyses is to evaluate, for low-to-high decay heat shutdown conditions, the thermal hydraulic response, particularly the maximum RCS pressure limits, if no operator recovery actions were taken to limit or prevent boiling in the RCS (References 97 and 98). The results of these analyses are used to determine acceptable RCS vent path configurations used during outage conditions as a contingency to mitigate RCS pressurization upon a postulated loss of residual heat removal (RHR). Typical RCS vent path openings capable of use include the reactor vessel head flange, one or more pressurizer safety valves, steam generator primary hot leg manways, or combinations of these openings depending on the decay heat load.

4.4.4 TESTING AND VERIFICATION

4.4.4.1 Testing Prior to Initial Criticality

Reactor coolant flow tests, as noted in Tests 3.9 and 3.10 of Table 14.1-2, are performed following fuel loading, but prior to initial criticality. Coolant loop pressure drop data are obtained in this test. These data, in conjunction with coolant pump performance information, allow determination of the coolant flowrates at reactor operating conditions. This test verifies that proper coolant flowrates have been used in the core thermal and hydraulic analysis.

4.4.4.2 Initial Power Plant Operation

Core power distribution measurements are made at several core power levels (see Section 4.3.2.2.7) during startup and initial power operation. These tests are used to verify

REACTOR DESIGN COMPARISON

		Unit 1	Unit 2
Thermal and Hydraulic Design Parameters (Using ITDP)[a]			
Reactor Core Heat Output, MWt		3411~~3,338~~	3,411
Reactor Core Heat Output, 10^6 Btu/hr		11,641.7~~11,392.6~~	11,641.7
Heat Generated in Fuel, %		97.4	97.4
Core Pressure, Nominal, psia[b]		2,280	2,280
Core Pressure, Min Steady State[b] psia		2,250	2,250
Fuel Type~~Minimum DNBR at nominal Conditions[c]~~		Vantage 5	Vantage 5
Minimum DNBR at nominal Conditions[c]	~~(LOPAR)~~	~~2.50~~	~~2.44~~
~~Typical Flow Channel~~			
Typical Flow Channel	~~(V-5)~~	2.63[1]~~9~~	2.63
~~Thimble (Cold Wall) Flow Channel~~	~~(LOPAR)~~	~~2.35~~	~~2.29~~
Thimble (Cold Wall) Flow Channel	~~(V-5)~~	2.47[1] ~~53~~	2.47
Limit DNBR for Design Transients			
~~Typical Flow Channel~~	~~(LOPAR)~~	~~1.48~~	~~1.48~~
Typical Flow Channel	~~(V-5)~~	1.71	1.71
~~Thimble (Cold Wall) Flow Channel~~	~~(LOPAR)~~	~~1.44~~	~~1.44~~
Thimble (Cold Wall) Flow Channel	~~(V-5)~~	1.68	1.68
~~DNB Correlation~~	~~(LOPAR)~~	~~WRB-1~~	~~WRB-1~~
DNB Correlation	~~(V-5)~~	WRB-2	WRB-2

[1] Values need review by Westinghouse

		Unit 1	Unit 2
HFP Nominal Coolant Conditions[d]			
Vessel Minimum Measured Flow[e] Rate (including Bypass)			
10^6 lbm/hr		135.4	136.6
gpm		359,200	362,500
Vessel Thermal Design Flow[e] Rate (including Bypass)			
10^6 lbm/hr		132.2	133.4
gpm		350,800	354,000
Core Flow Rate (excluding Bypass, based on TDF)			
10^6 lbm/hr		122.3	123.4
gpm		324,490	327,450
Effective Flow Area[f] for Heat Transfer, ft^2	~~(LOPAR)~~	~~51.08~~	~~51.08~~
	~~(V-5)~~	54.13	54.13
Average Velocity along Fuel[f,k] Rods, ft/sec (Based on TDF)	~~(LOPAR)~~	~~14.8~~	~~15.1~~
	~~(V-5)~~	14.0	14.2
Core Inlet Mass Velocity,[f] 10^6 lbm/hr-ft (Based on TDF)	~~(LOPAR)~~	~~2.39~~	~~2.42~~
	(V-5)	2.26	2.28

		Unit 1	Unit 2
Thermal and Hydraulic Design Parameters (Based on Thermal Design Flow)			
Nominal Vessel/Core Inlet Temperature, °F		544.54[e]	545.1[e]
Vessel Average Temperature, °F		577.3~~576.6~~	577.6
Core Average Temperature, °F		581.5~~580.7~~	581.8
Vessel Outlet Temperature, °F		610.1~~608.8~~	610.1
Average Temperature Rise in Vessel, °F		65.6~~64.4~~	65.0
Average Temperature Rise in Core, °F		70.3~~69.1~~	69.7
Heat Transfer			
Active Heat Transfer Surface Area,[f] ft^2	~~(LOPAR)~~	~~59,742~~	~~59,742~~
	~~(V-5)~~	57,505	57,505
Average Heat Flux, Btu/hr-ft^2	~~(LOPAR)~~	~~185,740585,740~~	~~189,800~~
	~~(V-5)~~	197,180~~192,960~~	197,180
Maximum Heat Flux for Normal[h] Operation, Btu/hr-ft^2	~~(LOPAR)~~	~~455,070~~	~~465,010~~
	~~(V-5)~~	483,100~~472,760~~	483,100
Average Linear Power, kW/ft		5.44~~5.33~~	5.44
Peak Linear Power for Normal Operation,[h] kW/ft		13.34~~13.06~~	13.34
Peak Linear Power for Determination of Protection Setpoints, kW/ft		21.1[i]	21.1[i]
Pressure Drop[j] Across Core, psi	~~(LOPAR)~~	~~22.6 + 2.3~~	~~23.2 + 2.3~~
	~~(V-5)~~	24.9 + 2.5	25.8 + 2.6
Across Vessel,[m] including nozzle, psi	~~(LOPAR)~~	~~53.5 + 5.4~~	~~48.9 + 4.9~~
	~~(V-5)~~	53.3 + 5.3	48.7 + 4.9
Thermal and Hydraulic Design Parameters			
Heat Flux Hot Channel Factor, F_Q^T		2.45	2.45
Temperature at Peak Linear Power for Prevention of Centerline Melt, °F		4700	4700
Fuel Central Temperature, °F			

TABLE 4.1-1

Peak at 100% power	<3230[1] ~~3170~~	<3230
Peak at maximum thermal output for maximum overpower DT trip point	<4080[1]	<4080

[1] Value needs review by Westinghouse

(a) Includes the effect of fuel densification

(b) Values used for thermal hydraulic core analysis

(c) Based on T_{in} = 545.1°F (Unit 1) and T_{in} = 545.7°F (Unit 2) corresponding ~~in~~ to Minimum Measured Flow of each unit

(d) Based on Safety Analysis T_{in} = 548.4°F and Pressure = 2280 psia

(e) Includes 15 percent steam generator tube plugging

(f) Assumes all ~~LOPAR or~~ VANTAGE 5 core

(g) Safety Analysis T_{in} = 548.4°F for both units

(h) This limit is associated with the value of $F_Q^T = 2.45$

(i) See Section 4.3.2.2.6

(j) Based on best estimate reactor flow rate, Section 5.1

(k) At core average temperature

(l) Enrichments for subsequent regions can be found in the Nuclear Design Report issued each cycle

(m) Assuming mechanical design flow

A separate residual heat removal (RHR) system is provided for each unit. This section describes one system with the second being identical unless otherwise noted.

The RHR system transfers heat from the RCS to the component cooling water system (CCWS) to reduce reactor coolant temperature to the cold shutdown temperature at a controlled rate during the latter part of normal plant cooldown, and maintains this temperature until the plant is started up again.

As a secondary function, the RHR system also serves as part of the ECCS during the injection and recirculation phases of a LOCA.

The RHR system can also be used to transfer refueling water between the refueling water storage tank and the refueling cavity before and after the refueling operations.

5.5.6.1 Design Bases

RHR system design parameters are listed in Table 5.5-8. A schematic diagram of the RHR system is shown in Figure 3.2-10.

The RHR system is designed to remove heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the RCS is reduced by transferring heat from the RCS to the steam and power conversion system (SPCS) via the steam generators.

The RHR system is placed in operation ~~approximately 4 hours after reactor shutdown,~~ when the nominal temperature and pressure of the RCS are ≤ 350°F and ≤ 390 psig, respectively. The cooldown calculation of Reference 12 assumes the RHR is placed in service no sooner than 4 hours after reactor shutdown. Assuming that two RHR heat exchangers and two RHR pumps are in service and that each heat exchanger is supplied with component cooling water at design flow and temperature, the analysis shows that the RHR system design is capable of reducing~~is designed to reduce~~ the temperature of the reactor coolant ~~from 350~~ to 140°F ~~within~~ in less than 20~~10~~ hours after reactor shutdown. The heat load handled by the RHR system during the cooldown transient includes sensible and decay heat from the core and RCP heat. ~~The design heat load is based on the decay heat fraction that exists at 20 hours following reactor shutdown from an extended run at full power.~~

5.5.6.2 System Description

The RHR system consists of two RHR heat exchangers, two RHR pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet line to the RHR system is connected to the hot leg of reactor coolant loop 4, while the return lines are connected to the cold legs of each of the reactor coolant loops. These normal return lines are also the ECCS low-head injection lines (see Figure 6.3-4).

When the reactor coolant nominal temperature and pressure are reduced to ≤ 350°F and ≤ 390 psig, respectively, ~~approximately 4 hours after reactor shutdown,~~ the second phase of cooldown starts with the RHR system being placed in operation. Data and procedure reviews indicate it will require more than 4 hours after reactor shutdown to initiate RHR cooldown (Ref. 12).

Startup of the RHR system includes a warmup period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the RHR heat exchangers. By adjusting the control valves downstream of the RHR heat exchangers, the mixed mean temperature of the return flows is controlled. Coincident with the manual adjustment, the heat exchanger bypass valve contained in the common bypass line is regulated to give the required total flow.

The reactor cooldown rate is limited by RCS equipment cooling rates based on allowable stress limits, as well as the operating temperature limits of the CCWS. As the reactor coolant temperature decreases, the reactor coolant flow through the RHR heat exchangers is increased.

As cooldown continues, the pressurizer is filled with water and the RCS is operated in the water-solid condition.

At this stage, pressure is controlled by regulating the charging flow rate and the alternate letdown rate to the CVCS from the RHR system.

After the reactor coolant pressure is reduced and the temperature is 140°F or lower, the RCS may be opened for refueling or maintenance.

5.5.6.2.2.4 Refueling

Several systems may be used during refueling to provide borated water from the refueling water storage tank to the refueling cavity. These include the RHR system, containment spray system, safety injection system, refueling water purification system, and the charging system (which includes the LHUTs). During this operation, the isolation valves to the refueling water storage tank are opened.

The reactor vessel head is removed. The refueling water is then pumped into the reactor vessel and into the refueling cavity through the open reactor vessel.

After the water level reaches the desired level, the refueling water storage tank supply valves are closed, and RHR operation continues.

During refueling, the RHR system is maintained in service w th the number of pumps and heat exchangers in operation as required by the heat load.

11. ~~11.~~Tube Structural Evaluation for Diablo Canyon Units 1 and 2 Under Packed Conditions, NSD-E-SGDA-98-334/SG-98-10-003, Westinghouse Electric Company, November 1998.
12. Westinghouse Calculation SE/FSE-C-PGE-0013, "RHRS Cooldown Performance at Uprated Conditions," Rev. 0, June 5, 1996.

TABLE 5.5-8

DESIGN BASES FOR RESIDUAL HEAT REMOVAL SYSTEM OPERATION (BOTH UNITS)

Residual heat removal system startup	No sooner than 4 hours after reactor shutdown
Number of Trains in Operation	2
Reactor coolant system initial pressure, psig	390
Reactor coolant system initial temperature, °F	350
Component cooling water design temperature, °F	95
Cooldown time, hours after reactor shutdown~~initiation of RHRS operation~~	≤20~~10~~
Reactor coolant system temperature at end of cooldown, °F	140
Decay heat generation used in cooldown analysis~~at 20 hours after shutdown~~, Btu/hr	75.5×10^6 ~~70.6×10^6 (Unit 1)~~ ~~72.1×10^6 (Unit 2)~~

10.2 TURBINE-GENERATOR

The basic function of the turbine-generator is to convert thermal energy initially to mechanical energy and finally to electrical energy. The turbine-generator receives saturated steam from the four steam generators through the main steam system. Steam is exhausted from the turbine-generator to the main condenser.

More detailed information, including design features and the safety evaluation of the turbine-generator and associated systems, is presented in the following sections.

10.2.1 DESIGN BASES

The design bases for the turbine-generator include performance requirements, operating characteristics, functional limitations, and code requirements.

10.2.1.1 Performance Requirements

The main turbine-generators and their auxiliary systems are designed for steam flow corresponding to 3500 MWt and 3580 MWt, which in turn correspond to the maximum calculated thermal performance data of the Units 1 and 2 nuclear steam supply systems (NSSS), respectively, at the original design ultimate expected thermal power. The Unit 2 turbine-generator has a higher power rating because of subsequent uprating of the Unit 2 NSSS. The intended mode of operation of both units is base loaded at levels limited to the ~~much~~ lower licensed reactor levels of ~~3338 MWt for Unit 1, and~~ 3411 MWt ~~for Unit 2~~ (see Table 15.1-1).

10.2.1.2 Operating Characteristics

The steam generator characteristic pressure curves (Figure 10.2-1) are the bases for design of the turbine. The pressure at the turbine main steam valves does not exceed the pressure shown on the steam characteristic pressure curve for the corresponding turbine load. With a pressurized water reactor, it is recognized that the pressure at the turbine steam valves rises as the load on the turbine is reduced below rated load. During abnormal conditions at any given load, the pressure may exceed the pressure on the steam generator characteristic pressure curve by 30 percent on a momentary basis, but the total aggregate duration of such momentary swings above characteristic pressure over the whole turbine load range does not exceed a total of 12 hours per 12-month operating period.

The turbine inlet pressure is not directly controlled. A load index from the turbine first-stage pressure is compared to the reactor coolant T_{avg}; the control rods are then positioned accordingly.

15.1.2.1 Power Rating

Table 15.1-1 lists the principal power rating values that are assumed in analyses performed in this section. Two ratings are given:

(1) The guaranteed nuclear steam supply system (NSSS) thermal power output. This power output includes the thermal power generated by the reactor coolant pumps.

(2) The engineered safety features (ESF) design rating. The Westinghouse-supplied ESFs are designed for a thermal power higher than the guaranteed value in order not to preclude realization of future potential power capability. This higher thermal power value is designated as the ESF design rating. This power output includes the thermal power generated by the reactor coolant pumps.

Where initial power operating conditions are assumed in accident analyses, the guaranteed NSSS thermal power output (plus allowance for errors in steady state power determination for some accidents) is assumed. Where demonstration of the adequacy of the ESF is concerned, the ESF design rating plus allowance for error is assumed. The thermal power values for each transient analyzed are given in Table 15.1-4.

15.1.2.2 Initial Conditions

~~With the exceptions noted below, the accident evaluations are based on the design parameters appropriate to Unit 2. As demonstrated in Table 4.4-1, Unit 2 is more limiting with respect to power capability than is Unit 1.~~ For most accidents which are DNB limited, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure are determined on a statistical basis and are included in the limit DNBR, as described in Reference 3. This procedure is known as the "Improved Thermal Design Procedure" (ITDP) and these accidents utilize the WRB-1 and WRB-2 DNB correlations (References 4 and 5). ITDP allowances may be more restrictive than non-ITDP allowances. The initial conditions for other key parameters are selected in such a manner to maximize the impact on DNBR. Minimum measured flow is used in all ITDP transients. The allowances on power, temperature, pressure, and flow that were evaluated for their effect on the ITDP analyses for a 24-month fuel cycle are reported in Reference 22.

For accident evaluations that are not DNB limited, or for which the Improved Thermal Design Procedure is not employed, the initial conditions are obtained by adding maximum steady state errors to rated values. The following steady state errors are considered:

(1)	Core power	±2% allowance calorimetric error
(2)	Average RCS temperature	±4.7°F allowance for deadband and measurement error

(3) Pressurizer pressure — ±38 psi or ±60 psi allowance for steady state fluctuations and measurement error (see Note)

Note: Pressurizer pressure uncertainty is ±38 psi in analyses performed prior to 1993; however, NSAL 92-005 (Reference 17) indicates ±60 psi is ~~the correct~~a conservative value for future analyses. Reference 18 evaluates the acceptability of existing analyses, which use ±38 psi.

For some accident evaluations, an additional ~~2.0°F~~allowance has been conservatively added to the measurement error for the average RCS temperatures to account for steam generator fouling. Generic accident analyses also consider T_{avg}/power coastdown as an initial condition for accidents, limited to full power T_{avg} of 565°F and steam generator pressure of 750 psia.

15.1.2.3 Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distribution through the placement of fuel assemblies, control rods, and by operation instructions. The power distribution may be characterized by the radial peaking factor $F_{\Delta H}$ and the total peaking factor F_q. The peaking factor limits are given in the Technical Specifications.

For transients that may be DNB-limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.1-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications. typ.

The axial power shape used in the DNB calculation is discussed in Section 4.4.3.

For transients that may be overpower-limited, the total peaking factor F_q is of importance. The value of F_q may increase with decreasing power level so that the full power hot spot heat flux is not exceeded, i.e., F_q x Power = design hot spot heat flux. All transients that may be overpower-limited are assumed to begin with a value of F_q consistent with the initial power level as defined in the Technical Specifications.

The value of peak kW/ft can be directly related to fuel temperature as illustrated in Figures 4.4-1 and 4.4-2. For transients that are slow with respect to the fuel rod thermal time constant (approximately 5 seconds), the fuel temperatures are illustrated in Figures 4.4-1 and 4.4-2. For transients that are fast with respect to the fuel rod thermal time constant, (for example, rod ejection), a detailed heat transfer calculation is made.

15.1.3 TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanism to release the rod cluster control assemblies (RCCAs) which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 15.1-2. Reference is made in that table to the overtemperature and overpower ΔT trip shown in Figure 15.1-1. This figure presents the allowable reactor coolant loop average temperature and ΔT for the design flow and the NSSS Design Thermal Power distribution as a function of primary coolant pressure. The boundaries of operation defined by the Overpower ΔT trip and the Overtemperature ΔT trip are represented as "protection lines" on this diagram. The protection lines are drawn to include all adverse instrumentation and setpoint errors so that under nominal conditions trip would occur well within the area bounded by these lines. The utility of this diagram is in the fact that the limit imposed by any given DNBR can be represented as a line. The DNB lines represent the locus of conditions for which the DNBR equals the safety analysis limit values (~~(1.44 and 1.48 for Standard thimble cell and typical cells, respectively;~~ 1.68 and 1.71 for V-5 thimble cell and typical cells, respectively) for ITDP accidents. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit values. The diagram shows that DNB is prevented for all cases if the area enclosed with the maximum protection lines is not traversed by the applicable DNBR line at any point.

The area of permissible operation (power, pressure and temperature) is bounded by the combination of reactor trips: ~~high neutron flux (fixed setpoint);~~ high pressurizer pressure (fixed setpoint); low pressurizer pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints)<u>; and by a line defining conditions at which the steam generator safety valves open.</u>~~.~~

The limit values, which were used as the DNBR limits for all accidents analyzed with the Improved Thermal Design Procedure are conservative compared to the actual design DNBR values required to meet the DNB design basis.

The difference between the limiting trip point assumed for the analysis and the normal trip point represents an allowance for instrumentation channel error and setpoint error. During startup tests, it is demonstrated that actual instrument errors and time delays are equal to or less than the assumed values.

15.1.9.5 TWINKLE

The TWINKLE[16] program is a multidimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one-, two-, and three-dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-cladding-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points and performs its own steady state initialization. Aside from basic cross section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits provide channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, fuel temperatures, and so on.

The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.
TWINKLE is further described in Reference 16.

15.1.9.6 THINC

The THINC code is described in Section 4.4.3.

15.1.9.7 RETRAN-02

The RETRAN-02 program is used to perform the best-estimate thermal-hydraulic analysis of operational and accident transients for light water reactor systems. The program is constructed with a highly flexible modeling technique that provides the RETRAN-02 program the capability to model the actual performance of the plant systems and equipment.

The main features of the RETRAN-02 program are:

(1) A one-dimensional, homogeneous equilibrium mixture thermal-hydraulic model for the reactor cooling system

(2) A point neutron kinetics model for the reactor core

(3) Special auxiliary or component models (such as non-equilibrium pressurizer temperature transport delay)

(4) Control system models

(5) A consistent steady state initialization technique

The RETRAN-02 program is further discussed in Reference 21.

15.1.10 REFERENCES

1. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended.

2. M. Ko, Setpoint Study for PG&E Diablo Canyon Units 1 and 2, WCAP 8320, June 1974.

3. H. Chelmer, et al., Improved Thermal Design Procedure, WCAP-8567-P-A (Proprietary) and WCAP-8568-A (Non-Proprietary), ~~July 1975~~February 1989.

4. F. E. Motley, et al., New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids, WCAP-8762-P-A and WCAP-8763-A, July 1984.

5. S. L. Davidson, and W. R. Kramer; (Ed.) Reference Core Report VANTAGE 5 Fuel Assembly, Appendix A.2.0, September 1985.

6. K. Shure, Fission Product Decay Energy in Bettis Technical Review, WAPD-BT-24, December 1961, pp. 1-17.

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12. ~~C. Hunin~~H.G.Hargrove, FACTRAN, a Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod, WCAP-7908-A, December 1989~~, June 1972~~.

13. T. W. T. Burnett et al, LOFTRAN Code Description, WCAP-7907-A, April 1984.

Figures 15.2.11-5 through 15.2.11-8 illustrate the transient assuming the reactor is in the automatic control mode. Both the BOL minimum and EOL maximum moderator feedback cases show that core power increases, thereby reducing the rate of decrease in coolant average temperature and pressurizer pressure. For both of these cases, the minimum DNBR remains above the limit value.

For all cases, the plant rapidly reaches a stabilized condition at the higher power level. Normal plant operating procedures would then be followed to reduce power.

The excessive load increase incident is an overpower transient for which the fuel temperatures will rise. Reactor trip does not occur for any of the cases analyzed, and the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in steam flow. Since DNB does not occur at any time during the excessive load increase transients, the ability of the primary coolant to remove heat from the fuel rod is not reduced. Thus, the fuel cladding temperature does not rise significantly above its initial value during the transient.

15.2.12.4 Conclusions

The analysis presented above shows that for a 10 percent step load increase, the DNBR remains above the safety analysis limit values, thereby precluding fuel or cladding damage. The plant reaches a stabilized condition rapidly, following the load increase.

15.2.13 ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

15.2.13.1 Identification of Causes and Accident Description

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. Since a safety valve is sized to relieve approximately twice the steam flowrate of a relief valve, and will therefore allow a much more rapid depressurization upon opening, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. Initially, the event results in a rapidly decreasing RCS pressure ~~until this pressure reaches a value corresponding to~~which could reach the hot leg saturation pressure if a reactor trip does not occur. ~~At that time, the pressure decrease is slowed considerably.~~ The pressure continues to decrease~~, however,~~ throughout the transient. The effect of the pressure decrease ~~would be~~is to decrease the neutron flux via the moderator density feedback, but the reactor control system (if in the automatic mode) functions to maintain the power and average coolant temperature essentially constant ~~throughout the initial stage of the transient~~until reactor trip occurs. Pressurizer level increases initially due to expansion caused by depressurization and then decreases following reactor trip.

The reactor will be tripped by the following reactor protection system signals:

(1) Pressurizer low pressure

(2) Overtemperature ΔT

15.2.13.2 Analysis of Effects and Consequences

The accidental depressurization transient is analyzed with the LOFTRAN code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level. This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 5.

In calculating the DNBR the following conservative assumptions are made:

(1) Plant characteristics and initial conditions are discussed in Section 15.1. Uncertainties and initial conditions are included in the limit DNBR as described in Reference 5.

(2) A positive moderator temperature coefficient of reactivity (+7 pcm/°F) for BOL operation in order to provide a conservatively high amount of positive reactivity feedback due to changes in moderator temperature. The spatial effect of voids due to local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape. These voids would tend to flatten the core power distribution.

(3) A low (absolute value) Doppler coefficient of reactivity such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

15.2.13.3 Results

Figure 15.2.12-1 illustrates the flux transient following the RCS depressurization accident. The flux increases until the time reactor trip occurs on ~~Low Pressurizer Pressure~~Overtemperature ΔT, thus resulting in a rapid decrease in the nuclear flux. The time of reactor trip is shown in Table 15.2-1. The pressure decay transient following the accident is given in Figure 15.2-.12-2. The resulting DNBR never goes below the safety analysis limit value as shown in Figure 15.2.12-1.

15.2.13.4 Conclusions

The pressurizer low pressure and the overtemperature ΔT reactor protection system signals provide adequate protection against this accident, and the minimum DNBR remains in excess of the safety analysis limit value.

(6) Turbine Load

Turbine load was assumed constant until the electrohydraulic governor drives the throttle valve wide open. Then turbine load drops as steam pressure drops.

(7) Reactor Trip

Reactor trip was initiated by low pressure. The trip was conservatively assumed to be delayed until the pressure reached 1860 psia.

15.2.15.3 Results

The transient response for the minimum feedback case is shown in Figures 15.2.14-1 through 15.2.14-2. Nuclear power starts decreasing immediately due to boron injection, but steam flow does not decrease until 25 seconds into the transient when the turbine throttle valve goes wide open. The mismatch between load and nuclear power causes T_{avg}, pressurizer water level, and pressurizer pressure to drop. The low-pressure trip setpoint is reached at 23 seconds and rods start moving into the core at 25 seconds.

After trip, pressures and temperatures slowly rise since the turbine is tripped and the reactor is producing some power due to delayed neutron fissions and decay heat.

15.2.15.4 Conclusions

Results of the analysis show that spurious safety injection with or without immediate reactor trip presents no hazard to the integrity of the RCS.

DNBR is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the reactor coolant system.

If the reactor does not trip immediately, the low-pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident

15.2.16 REFERENCES

1. W. C. Gangloff, An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors, WCAP-7486, May 1971.

2. D. H. Risher, Jr. and R. F. Barry, TWINKLE-A Multi-Dimensional Neutron Kinetics Computer Code, WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary), January 1975.

3. ~~C. Hunin~~H.G.Hargrove, FACTRAN, A Fortran IV Code for Thermal Transients in UO_2 Fuel Rod, WCAP-7908-A, December 1989~~, June 1972~~.

4. T. W. T. Burnett, et al., LOFTRAN Code Description, WCAP-7907-A, ~~June 1972~~April 1984.

5. H. Chelemer, et al., Improved Thermal Design Procedure, WCAP-8567-P-A (Proprietary) and WCAP-8568-A (Non-Proprietary), February 1989~~July 1975~~.

6. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended.

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12. Westinghouse letter PGE-96-565, Diablo Canyon Units 1 & 2 Spurious Safety Injection/Pressurizer Safety Valve Water Relief Final Results, May 31, 1996.

13. PG&E Calculation STA-035, "LOFTRAN 10.01 Simulation of Inadvertent SF with one PORV available," February 29, 1996.

TABLE 15.1-1

NUCLEAR STEAM SUPPLY SYSTEM POWER RATINGS

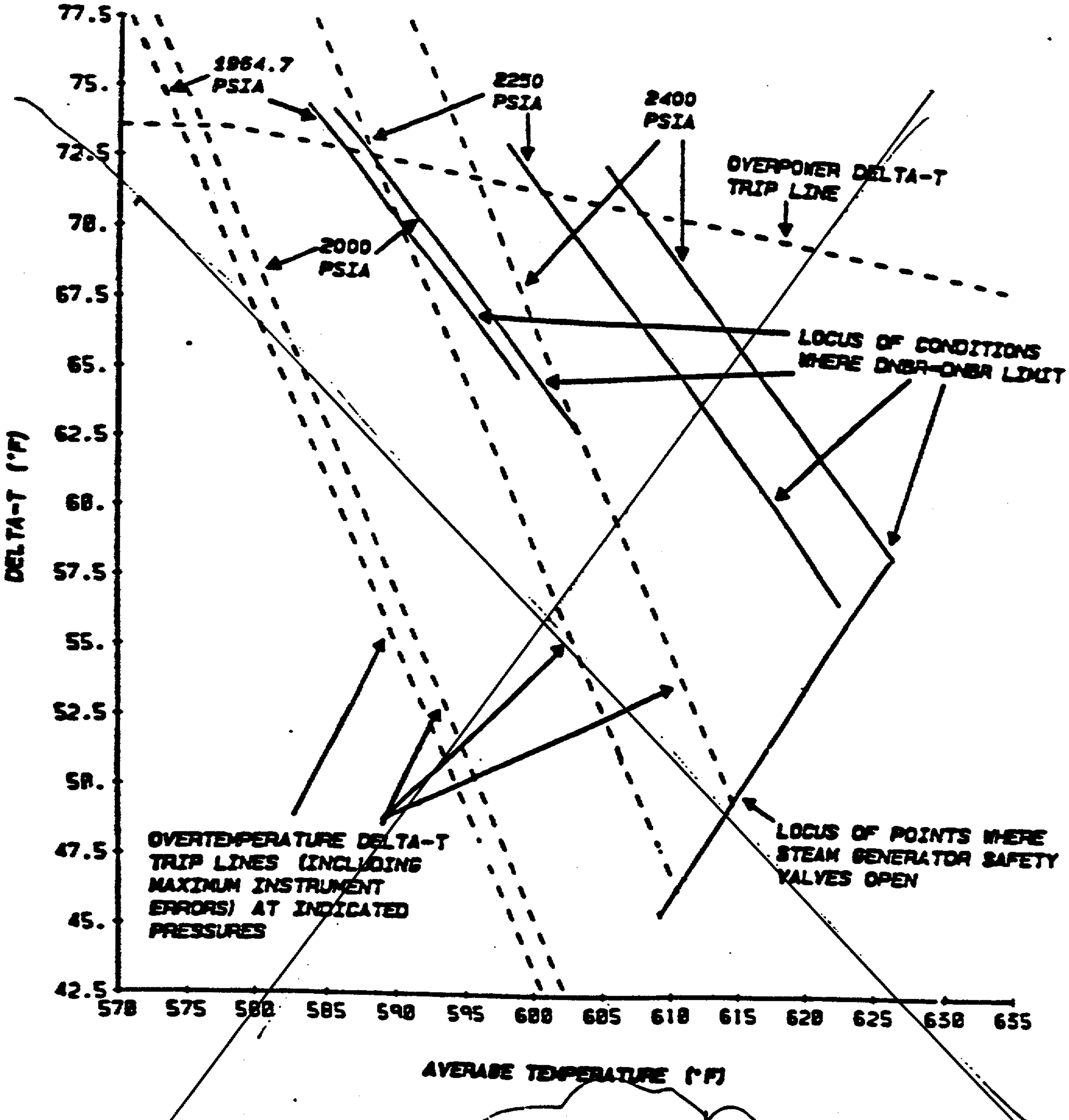
	~~Unit 1~~	~~Unit 2~~
Guaranteed core thermal power (license level)	~~3338~~	3411
Thermal power generated by the reactor coolant pumps minus heat losses to containment and letdown system[(b)]	~~14~~	14
Guaranteed nuclear steam supply system thermal power output[(b)]	~~3352~~	3425
The engineered safety features design rating (maximum calculated turbine rating)[(a)]	~~3570~~	3570

(a) The units will not be operated at this rating because it exceeds the license ratings.

(b) As noted on Table 15.1-4, some analyses assumed a full-power NSSS thermal power output of 3423 MWt, based on the previous net reactor coolant pump heat of 12 MWt. An evaluation concludes that the effect of an additional 2 MWt for NSSS is negligible such that analyses based on 3423 MWt remain valid.

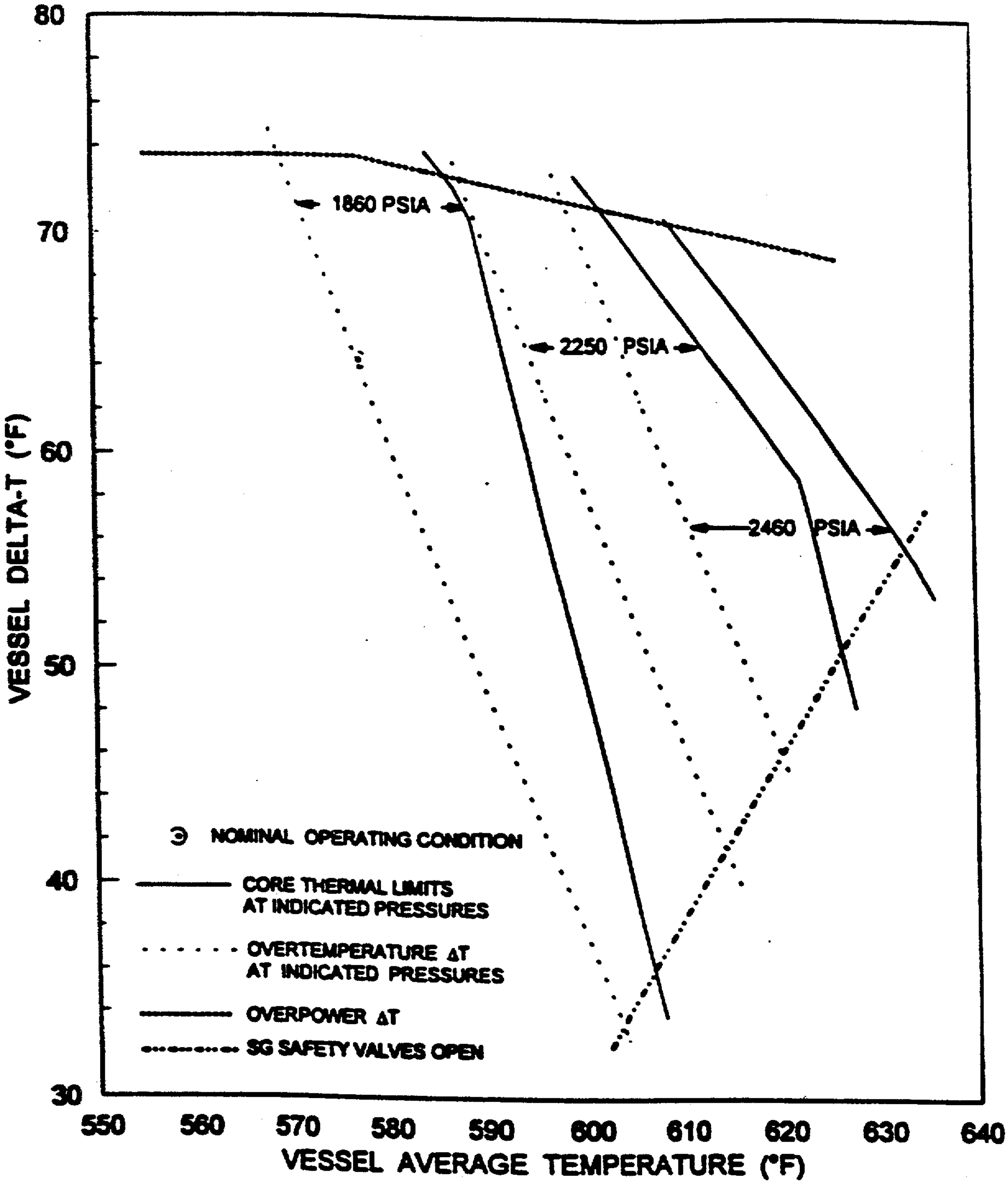
Faults	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp[a], pcm/°F[d]	Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II (Cont'd)					
Loss of offsite power to the plant auxiliaries	LOFTRAN	+8	-	Upper	3431
Excessive heat removal due to feedwater system malfunctions	LOFTRAN	-	0.43	Lower	0 and 3423
Excessive load increase	LOFTRAN	-	0 and 0.43	Lower and Upper	3423
Accidental depressurization of the reactor coolant system	LOFTRAN	+57	-	Lower	34235
Accidental depressurization of the main steam system	LOFTRAN	-	Function of the moderator density. See Sec. 15.2.13 (Figure 15.2.13-1)	See Figure 15.4.2-1	0 (Subcritical)
Inadvertent operation of ECCS during power operation	LOFTRAN	+5	0.43	Lower and Upper	3423
CONDITION III					
Loss of reactor coolant from small ruptured pipes or from cracks in large pipe which actuate emergency core cooling	NOTRUMP SBLOCTA	-	-	-	3479

..ccident	Event	Time, sec
Excessive Feedwater at Full Load	One main feedwater control valve fails fully open	0.0
	Minimum DNBR occurs	45.5
	Feedwater flow isolated due to high-high steam generator level	51.0
Excessive Load Increase		
1. Manual reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	240
2. Manual reactor control (EOL maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	64
Automatic reactor control (BOL minimum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	150
4. Automatic reactor control (EOL maximum moderator feedback)	10% step load increase	0.0
	Equilibrium conditions reached (approximate times only)	150
Accidental Depressurization of the Reactor Coolant System	Inadvertent opening of one ~~RCS~~ pressurizer safety valve	0.0
	~~Low pressurizer pressure~~Overtemperature ΔT reactor trip setpoint reached	~~39.8~~27.5
	Rods begin to drop	~~41.8~~29.5
	Minimum DNBR occurs	~~42.2~~29.8



Remove and Replace with the Following

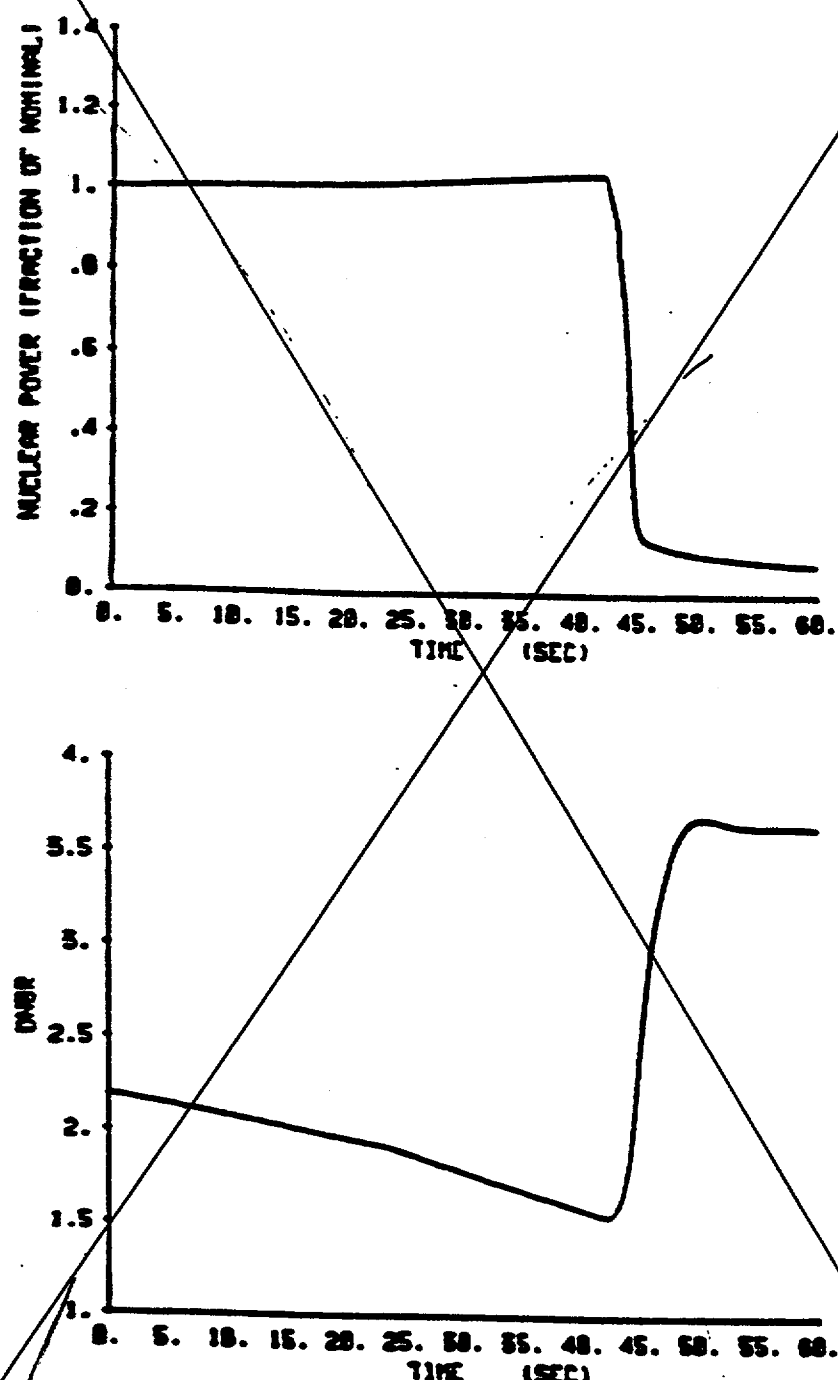
DIABLO CANYON UNITS 1 AND 2
FIGURE 15.1-1 OVERTEMPERATURE AND OVERPOWER DELTA-T PROTECTION



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-1

ILLUSTRATION OF OVERPOWER AND OVERTEMPERATURE ΔT PROTECTION



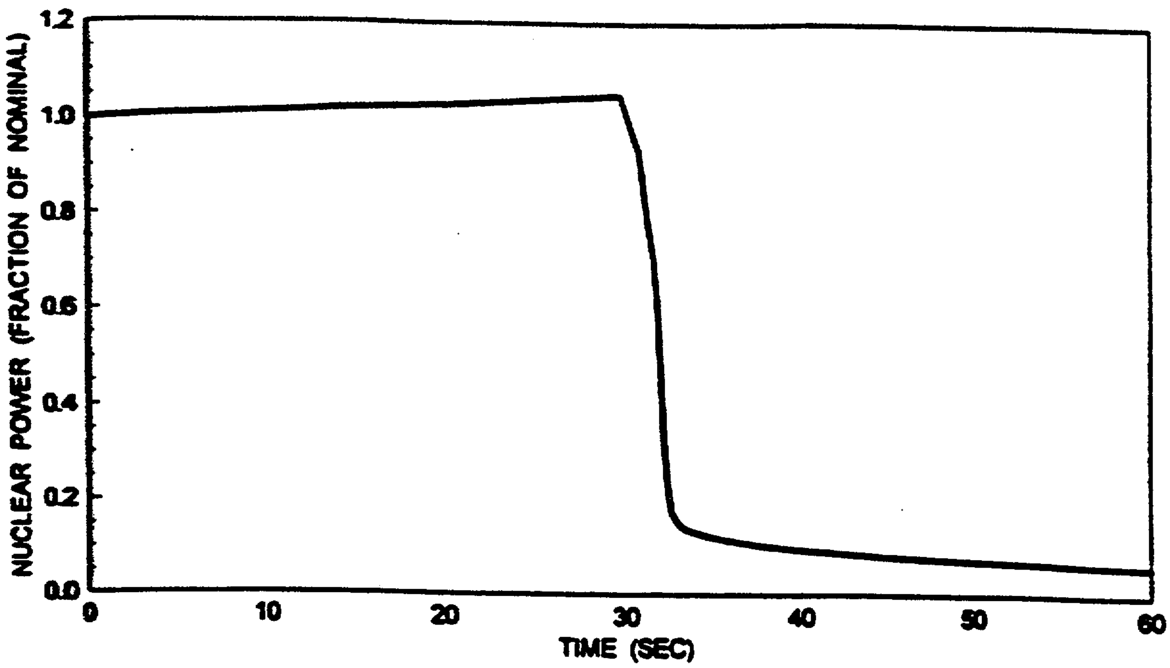
Replace with the Following

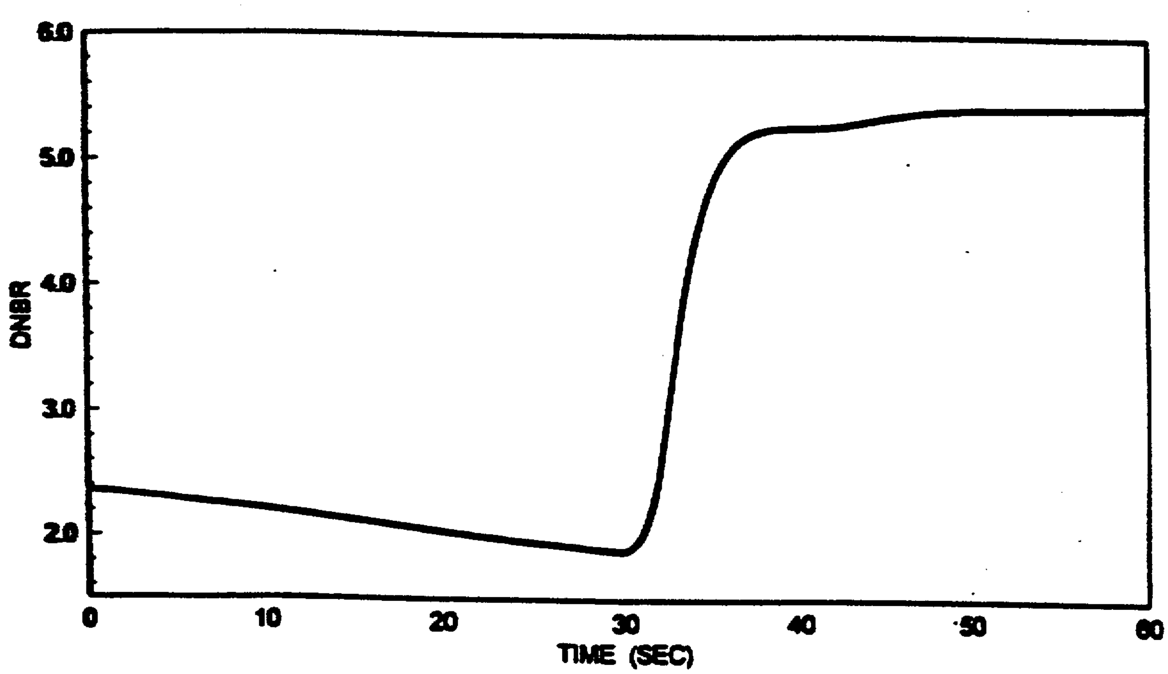
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.12-1

NUCLEAR POWER AND DNBR TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF REACTOR COOLANT SYSTEM

Revision 11 November 1996

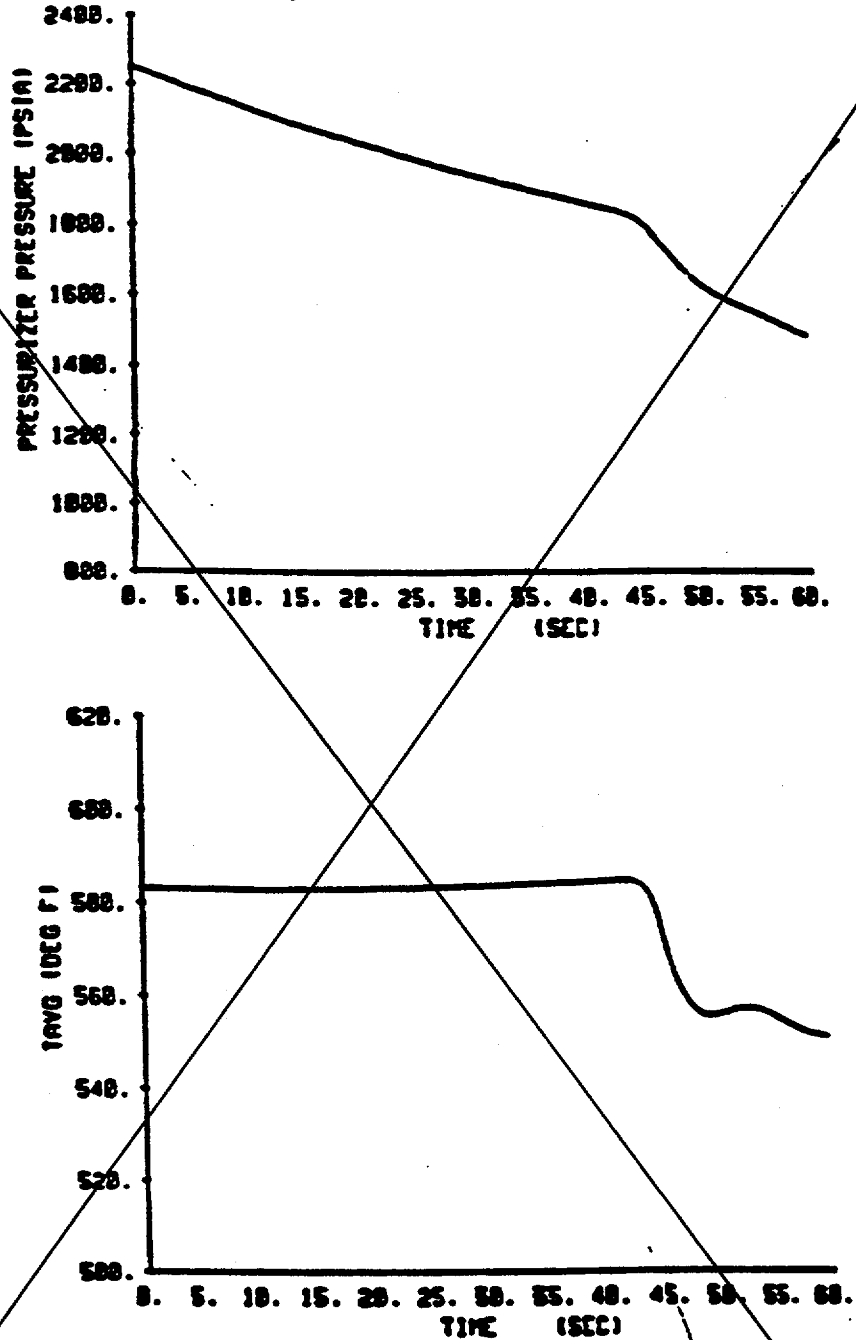




DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.12-1

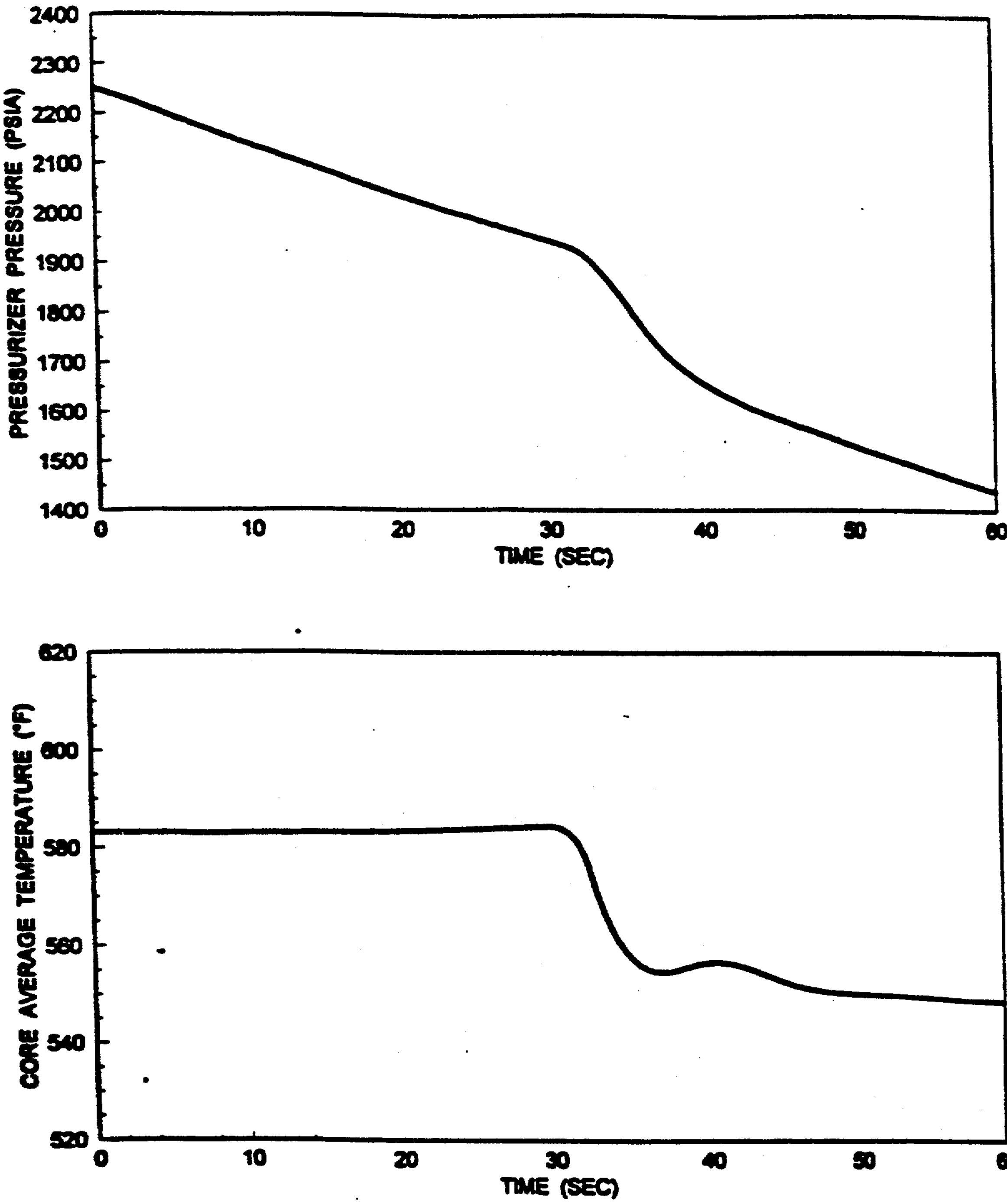
NUCLEAR POWER AND DNBR TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM



Replace with the Following

DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.12-2
PRESSURIZER PRESSURE AND VESSEL AVERAGE TEMPERATURE TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF REACTOR COOLANT SYSTEM



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.12-2

PRESSURIZER PRESSURE AND CORE AVERAGE TEMPERATURE TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF THE REACTOR COOLANT SYSTEM flow by starting AFW pumps. The secondary flow aids in the reduction of RCS pressure. When the RCS depressurizes to below approximately 600 psia, the accumulators begin to inject water into the reactor coolant loops. The reactor coolant pumps are assumed to be tripped at the beginning of the accident and the effects of pump coastdown are included in the blowdown analyses.

15.3.1.2 Analysis of Effects and Consequences

For loss-of-coolant accidents due to small breaks less than 1 square foot, the NOTRUMP[12] computer code is used to calculate the transient depressurization of the RCS as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP computer code is a state-of-the-art one-dimensional general network code with a number of advanced features. Among these features are the calculation of thermal nonequilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly, with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A detailed description of the NOTRUMP code is provided in References 12 and 13.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with the associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Safety injection flowrate to the RCS as a function of the system pressure is used as part of the input. The SIS was assumed to be delivering water to the RCS 27 seconds after the generation of a safety injection signal.

For the analysis, the SIS delivery considers pumped injection flow that is depicted in Figure 15.3-1 as a function of RCS pressure. This figure represents injection flow from the SIS pumps based on performance curves degraded 5 percent from the design head. The 27-second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency buses. The effect of residual heat removal (RHR) pump flow is not considered here since their shutoff head is lower than RCS pressure during the time portion of the transient considered here. Also, minimum safeguards ECCS capability and operability have been assumed in these analyses.

Peak cladding temperature analyses are performed with the LOCTA IV[4] code that determines the RCS pressure, fuel rod power history, steam flow past the uncovered part to the core, and mixture height history.

15.3.1.3 Results

15.3.1.3.1 Reactor Coolant System Pipe Breaks

This section presents the results of a spectrum of small break sizes analyzed for both DCPP Unit 1 and DCPP Unit 2. The small break analysis was performed at 102% of the Rated Core Power (3411 MWt), a Peak Linear Power of 15.00 kW/ft, a Total Peaking Factor (F_Q^T) of 2.70, a Thermal Design Flow of 85,000 gpm/loop and a steam generator tube plugging level of 15%.

The worst break size (small break) for both Units was shown to be a 3-inch diameter break in the cold leg. In the analysis of this limiting break, a Reactor Coolant System Tavg window of 572.0°F, +10.3°F, -12.0°F was considered. For both Units , the High Tavg cases were shown to be more limiting than the Low Tavg cases and therefore are the subject of the remaining discussion. The time sequence of events and the fuel cladding results for the breaks analyzed are shown in Tables 15.3-1 and 15.3-2.

During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rods and cladding to very near the coolant temperature as long as the core remains covered by a two-phase mixture. This effect is evident in the accompanying figures.

The depressurization transient for the limiting 3-inch breaks are shown in Figures 15.3-2-DCPP1/DCPP2. The extent to which the core is uncovered for these breaks are presented in Figures 15.3-3-DCPP1/DCPP2. The maximum hot spot cladding temperature reached during the transient, including the effects of fuel densification as described in Reference 3, is 1304°F and 1293°F for Units 1 and 2, respectively. The peak cladding temperature transients for the 3-inch breaks are shown in Figures 15.3-4-DCPP1/DCPP2. The top core node vapor temperatures for the 3-inch breaks are shown in Figures 15.3-5-DCPP1/DCPP2. When the mixture level drops below the top of the core, the top core node vapor temperature increases as the steam superheats along the expose portion of the fuel. The rod film coefficients for this phase of the transient are given in Figures 15.3-6-DCPP1/DCPP2. The hot spot fluid temperatures are shown in Figures 15.3-7-DCPP1/DCPP2 and the break mass flows are shown in Figures 15.3-8-DCPP1/DCPP2.

~~This section presents the results of a spectrum of small break sizes analyzed for DCPP Unit 2. The worst break size (small break) for DCPP Unit 2 is a 4 inch diameter break in the cold leg. This limiting break size was also analyzed for DCPP Unit 1 in order to demonstrate that the lower power level for Unit 1 will result in a less severe transient. The time sequence of events and the results for all the breaks analyzed are shown in Tables 15.3-1 and 15.3-2.~~

~~During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rods and cladding to very near the coolant temperature as long as the core remains covered by a two-phase mixture. This effect is evident in the accompanying figures.~~

~~The depressurization transient for the limiting 4-inch break is shown in Figure 15.3-2. The extent to which the core is uncovered for the same break is presented in Figure 15.3-3. The maximum hot spot cladding temperature reached during the transient is 1358°F, including the effects of fuel densification as described in Reference 3. The peak cladding temperature transient for the limiting break size is shown in Figure 15.3-4. The core steam flowrate for the 4-inch break is shown in Figure 15.3-5. When the mixture level drops below the top of the core, the steam flow computed in NOTRUMP provides cooling to the upper portion of the core. The rod film coefficients for this phase of the transient are given in Figure 15.3-6. Also, the hot spot fluid temperature for the worst break is shown in Figure 15.3-7.~~

~~Since a separate analysis was performed for DCPP Unit 1, a set of figures similar to those presented for the Unit 2 limiting break size can be found in Figures 15.3-14a through 15.3-14f.~~

The core power (dimensionless) transient following the accident (relative to reactor scram time) is shown in Figure 15.3-9~~8~~. The reactor shutdown time (4.7 seconds) is equal to the reactor trip signal processing time (2.0 seconds) plus 2.7 seconds for complete rod insertion. During this rod insertion period, the reactor is conservatively assumed to operate at 102% rated power. ~~rated power.~~ The small break analyses considered 17x17 Vantage 5 fuel with IFM's, ZIRLO cladding, and an axial blanket. Fully enriched annular pellets, as part of an axial blanket core design, were modeled explicitly in this analysis. The results when modeling the enriched annular pellets were not significantly different than the results from the solid pellet modeling.

Several figures are also presented for the additional break sizes analyzed. Figures 15.3-10~~9~~-DCPP1/DCPP2 and 15.3-11-DCPP1/DCPP2~~10~~ present the RCS pressure transient for the 2~~3~~-inch and 4~~6~~-inch breaks~~, respectively~~. ~~, and~~ Figures 15.3-12-DCPP1/DCPP2~~11~~ and 15.3-13-DCPP1/DCPP2~~12~~ present the core mixture height plots for both breaks. The peak cladding temperature transients for the 2~~3~~-inch break ~~is~~ ares shown in Figures -15.3-14-DCPP1/DCPP2~~13~~. The peak cladding temperature transients for the 4-inch breaks are ~~plot is~~ shown in Figures 15.3-15-DCPP1/DCPP2~~14 for the 6 inch break~~.

The small break analysis was performed with the Westinghouse ECCS Small Break Evaluation Model[12,4] approved for this use by the NRC in May_-1985. An improved cold leg SI condensation model, COSI[26], was utilized as part of the Evaluation Model.

~~15.3.1.3.2 Effect of Changes to Small Break LOCA Evaluation Model on PCT~~

~~The small break LOCA analysis results of Section 15.3.1.3.1 were calculated for a full core of VANTAGE 5 fuel using the 1985 version of the Westinghouse small break LOCA ECCS evaluation model incorporating the NOTRUMP analysis technology (References 12 and 13). For Diablo Canyon Units 1 and 2, the limiting size small break is a 4 inch equivalent diameter break in the cold leg. The calculated PCT values of analysis of record were 1275F for Unit 1 and 1358F for Unit 2. However, a combination of several different 10 CFR 50.59 and 10 CFR 50.92 safety evaluations and permanent 10 CFR 50.46 ECCS model assessments to the small break LOCA evaluation model and input had to be made after these PCT values were calculated. Consequently, the results of the small break LOCA analysis for Units 1 and 2 were examined to assess the effect of model and assumption changes on PCT results.~~

~~These assessments have resulted in some benefits and penalties to the PCT values. The resultant PCT values for both Units 1 and 2 remain within the PCT limit of 2200F specified in 10 CFR 50.46. Since the PCT assessment process is continuous as issues are identified, the latest PCT values are documented in the most recent PG&E submittal to the NRC. Readers are referred to the most recent PG&E submittal for the latest PCT values and issue descriptions. The following discussions are provided as examples of some of the assessments made and should not be construed as a complete list of PCT assessments to the small break LOCA model.~~

~~The effect of the potentially significant ECCS Evaluation Model modifications, which are discussed in References 14 and 16, on the small break LOCA analyses for Diablo Canyon Units 1 and 2 was conservatively assessed. An increase of 42F to the PCT was estimated as a result of ECCS Evaluation Model changes when determining the available margin to the limits of 10 CFR 50.46.~~

~~The small break LOCA analysis results have been supplemented by a safety evaluation for the effect of purging the steam generator auxiliary feedwater piping of the residual main feedwater during a small break LOCA. As reported in Reference 15, this evaluation determined a maximum increase in the small break LOCA analysis PCT of 111F for each unit.~~

~~Changes to the ECCS flow requirements in the Technical Specifications were made in License Amendments Numbers 65 and 64 for Units 1 and 2, respectively. Because the revised minimum charging and SI pump flows are lower than were assumed in the small break LOCA analysis, a PCT penalty of 58F is incurred. Increased detail in the determination of the accumulator pressure instrument uncertainty was done in 1992. This resulted in larger uncertainties than those used in the original SBLOCA analysis and resulted in PCT penalties of~~

~~14F and 16F for Units 1 and 2, respectively[30]. In addition, there is a 4F penalty assessed for pressurizer pressure control uncertainty.~~

~~A PCT effect of -13F has been assessed for DCPP Units 1 and 2 with respect to NOTRUMP drift flux flow regime map errors. Errors were discovered in both WCAP 10079-P-A and related coding in NOTRUMP SUBROUTINE DECORRS where the improved TRAC-P1 vertical flow regime map is evaluated. These errors have been corrected[33].~~

~~Westinghouse has assessed in their Nuclear Safety Advisory Letter (NSAL) a net PCT effect of 16F for small break LOCA due to the correction of LUCIFER errors (NSAL 94-004). The LUCIFER code is used to generate component databases from raw input data for small and large break LOCA analyses[34].~~

~~Further assessment by Westinghouse (NSAL 94-018R) resulted in a net PCT effect of 18F, due to an error in the steam line isolation logic for the DCPP Units 1 and 2 small break LOCA analyses. The correction of this error consists of two portions; (a) a possible plant specific effect that applies only to analyses that assume main feedwater isolation (FWI) to occur on S-signal, and (b) a generic effect applying to all previous analyses[35].~~

~~Westinghouse has also assessed (NSAL 94-672) a net PCT effect of -319F and -344F, due to error corrections in small break LOCA code SBLOCTA for small break LOCA analyses for DCPP Units 1 and 2, respectively. SBLOCTA is a part of the NOTRUMP and WFLASH small break LOCA ECCS evaluation models. In addition, Westinghouse has assessed in their letter NSAL 94-018R a net PCT effect of -6F due to boiling heat transfer correlation errors for the DCPP Units 1 and 2 small break analyses[36]. The implementation of Westinghouse Eagle-21 upgrade, which replaced the Westinghouse analog process protection equipment with digital equipment, has effected a net PCT change of 18F for Units 1 and 2[36].~~

~~The individual PCT assessments discussed above were conservatively determined by Westinghouse. Westinghouse has reasonable assurance that the arithmetic summation of these individual assessments is conservative, and bounds any synergistic effects that may occur when the model changes are collectively considered. This assurance is based upon Westinghouse's knowledge of the physics of the LOCA phenomena and upon known evaluation model sensitivities.~~

15.3.1.4 Conclusions

Analyses presented in this section show that the high-head portion of the ECCS, together with the accumulators, provides sufficient core flooding to keep the calculated peak cladding temperatures below required limits of 10 CFR 50.46. Hence adequate protection is afforded by the ECCS in the event of a small break LOCA.

14. Deleted~~10 CFR 50.46 Annual Notification for 1989 of Modifications in the Westinghouse ECCS Evaluation Models, Letter from W.J. Johnson (Westinghouse) to T.E. Murley (NRC), NS-NRC-69-3463, October 5, 1989.~~

15. Deleted~~Disposition of LOCA Related PIs for Diablo Canyon Unit 1 (PG&E) Cycle 4 Reload, NS-SAT-SAI-89-415, September 11, 1989.~~

16. Deleted~~Correction of Errors and Modifications to the NOTRUMP Code in the Westinghouse Small Break LOCA ECCS Evaluation Model Which Are Potentially Significant. Letter from W. J. Johnson (Westinghouse) to T. E. Murley (NRC), NS-NRC-69-3464, October 5, 1989.~~

17. Deleted in Revision 12.

18. Deleted in Revision 12.

19. Deleted in Revision 12.

20. Deleted~~Accumulator Pressure Setpoint, Letter from S. A. McHugh (Westinghouse) to M. R. Tresler (PG&E), PGE-92-641, August 17, 1992.~~

21. Deleted in Revision 12.

22. Deleted in Revision 12.

23. Deleted~~10 CFR 50.46 30 Day Notification Report of Significant ECCS Evaluation Model Changes That Affect Peak Cladding Temperature, PG&E submittal to the NRC, November 5, 1993, DCL-93-259.~~

24. Deleted~~10 CFR 50.46 Annual Report of Emergency Core Cooling System Evaluation Model Changes, PG&E submittal to the NRC, April 19, 1994, DCL-94-079.~~

25. ~~25.~~ Deleted~~10 CFR 50.46 30 Day Report of Emergency Core Cooling System Evaluation Model Changes, PG&E submittal to the NRC, December 1, 1994, DCL-94-268.~~

26. WCAP-10054-P, Addendum 2, Revision 1, "NOTRUMP SBLOCA Using the COSI Steam Condensation Model", October, 1995.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS ~~FOR EACH~~ SMALL BREAK LOCA ANALYSIS

	UNIT 1		
	2-inch	3-inch	4-inch
Break Occurs (sec)	0.0	0.0	0.0
Reactor Trip Signal (sec)	48.7	19.6	11.1
Safety Injection Signal (sec)	60.7	28.2	18.6
Top of Core Uncovered (sec)	1781	995	605
Accumulator Injection Begins (sec)	N/A[1]	1845	852
Peak Clad Temperature Occurs (sec)	4250	1852	928
Top of Core Covered (sec)	N/A[2]	3160	1571

	UNIT 2		
	2-inch	3-inch	4-inch
Break Occurs (sec)	0.0	0.0	0.0
Reactor Trip Signal (sec)	49.2	19.5	11.1
Safety Injection Signal (sec)	61.2	28.2	18.5
Top of Core Uncovered (sec)	1750	1066	607
Accumulator Injection Begins (sec)	N/A[1]	2250	857
Peak Clad Temperature Occurs (sec)	4371	1948	937
Top of Core Covered (sec)	N/A[2]	3176	1628

1 Transient determined to be over prior to Accumulator injection
2 Transient determined to be over prior to complete core recovery

	~~Unit 2~~			~~Unit 1~~
	~~Equivalent Break Size~~			
~~Event~~	~~3 in.~~	~~4 in.~~	~~6 in.~~	~~4 in.~~
	~~Time, sec.~~			
~~Start~~	~~0.0~~	~~0.0~~	~~0.0~~	~~0.0~~
~~Reactor trip signal~~	~~7.74~~	~~4.47~~	~~2.30~~	~~4.47~~
~~Top of core uncovered (approx.)~~	~~1375~~	~~650~~	~~136~~	~~660~~
~~Accumulator injection begins~~	~~2350~~	~~894~~	~~378~~	~~900~~
~~PCT occurs~~	~~1868~~	~~959~~	~~172~~	~~948~~
~~Top of core covered (approx.)~~	~~2133~~	~~1195~~	~~413~~	~~1117~~

TABLE 15.3-2

FUEL CLADDING RESULTS - SMALL BREAK LOCA ANALYSIS~~SMALL COLD LEG BREAK CLADDING PARAMETERS AND CALCULATION ASSUMPTIONS~~

	UNIT 1 2-inch	UNIT 1 3-inch	UNIT 1 4-inch
Peak Cladding Temperature (°F)	956	1304	1264
Peak Cladding Temperature Location (ft)[1]	10.75	11.25	11.00
Peak Cladding Temperature Time (sec)	4250	1852	928
Local Zr/H_2O Reaction, Max (%)	0.03	0.20	0.09
Local Zr/H_2O Reaction Location (ft)[1]	11.00	11.25	11.00
Total Zr/H_2O Reaction (%)	$\leq$1.0	$\leq$1.0	$\leq$1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A

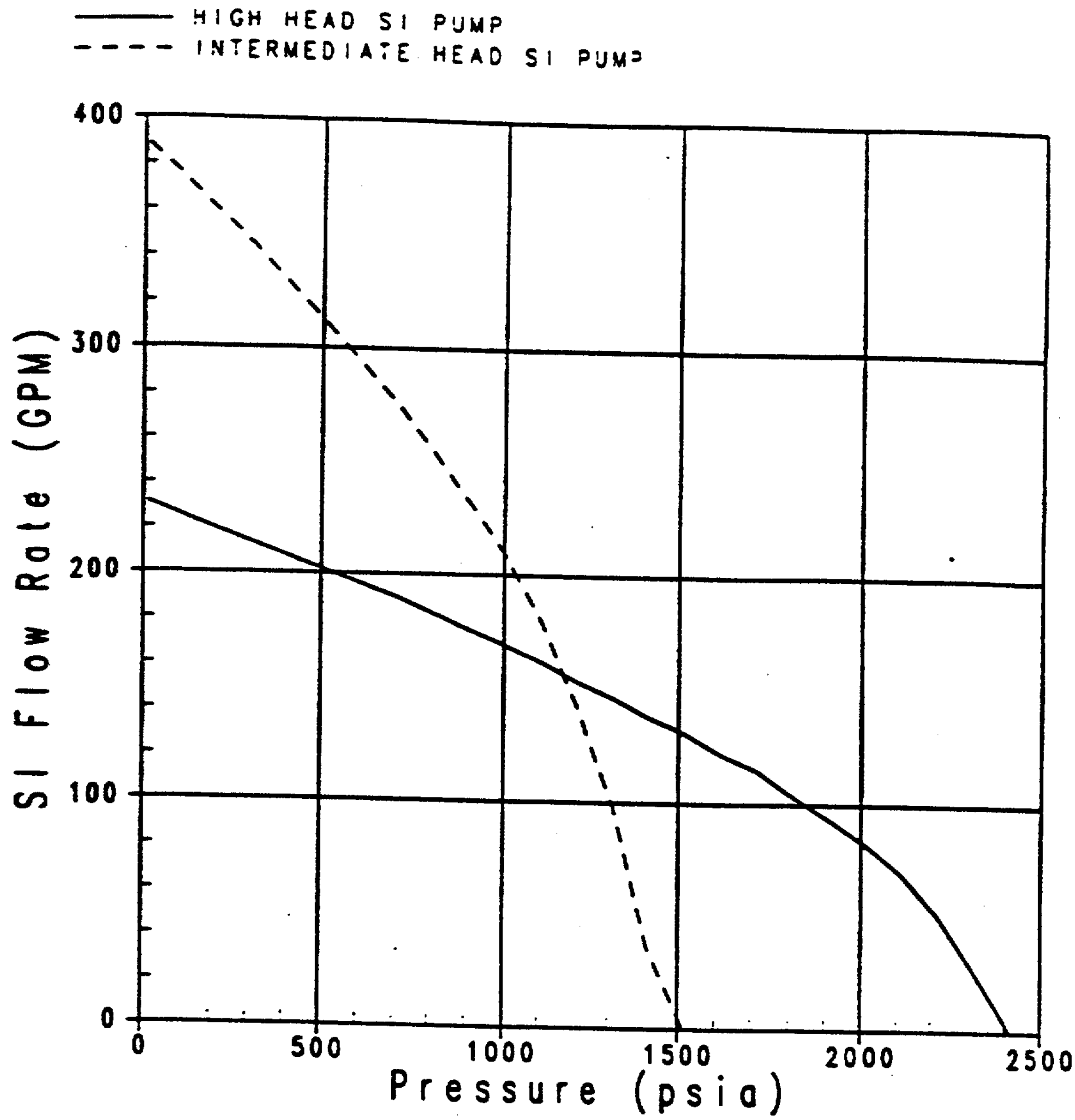
	UNIT 2 2-inch	UNIT 2 3-inch	UNIT 2 4-inch
Peak Cladding Temperature (°F)	955	1293	1225
Peak Cladding Temperature Location (ft)[1]	11.00	11.25	11.00
Peak Cladding Temperature Time (sec)	4371	1948	937
Local Zr/H_2O Reaction, Max (%)	0.03	0.25	0.07
Local Zr/H_2O Reaction Location (ft)[1]	11.00	11.25	11.00
Total Zr/H_2O Reaction (%)	$\leq$1.0	$\leq$1.0	$\leq$1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A

[1] From bottom of active fuel

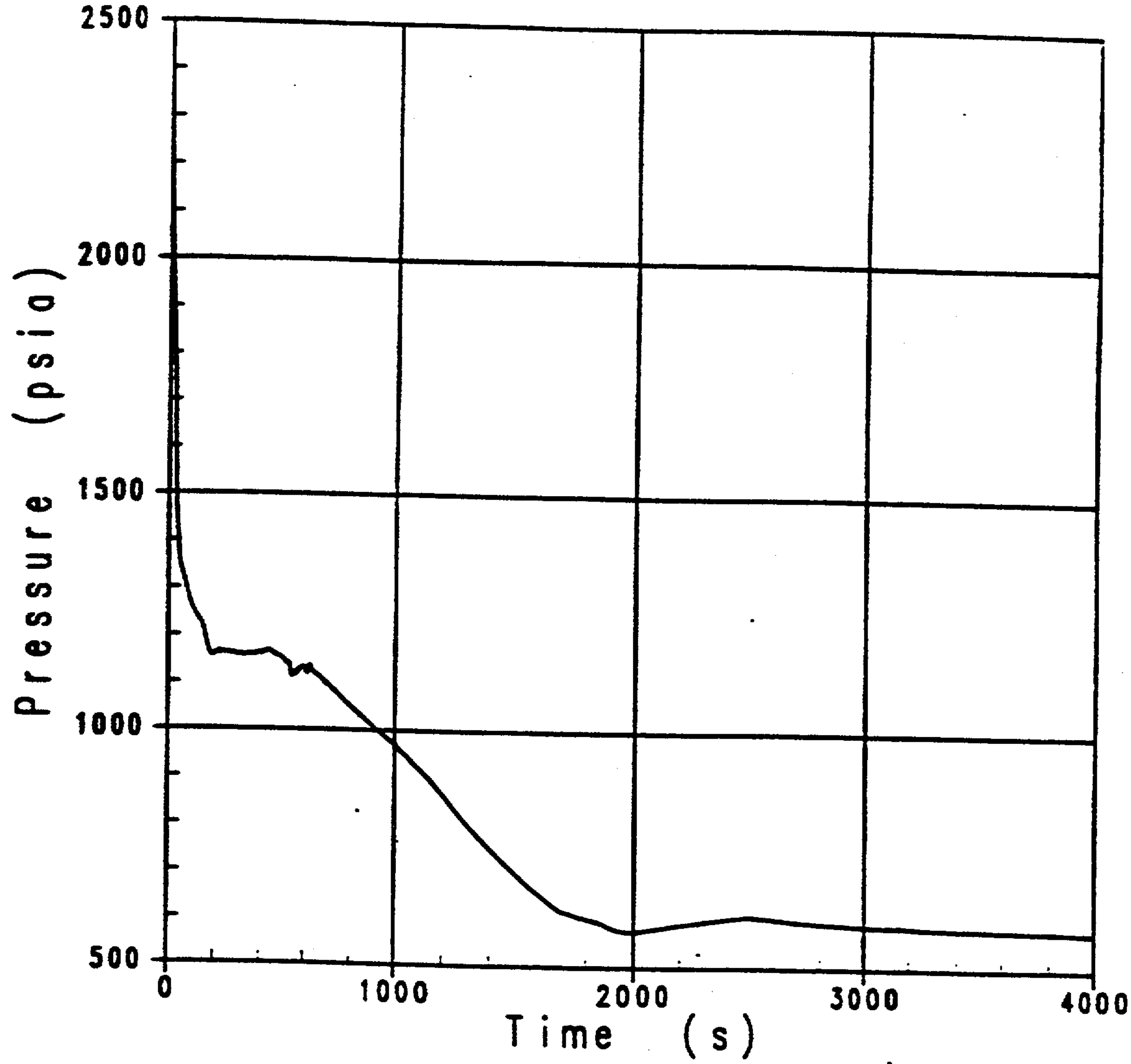
	~~Unit 2 Equivalent Break Size 3 in.~~	~~Unit 2 Equivalent Break Size 4 in.~~	~~Unit 2 Equivalent Break Size 6 in.~~	~~Unit 1 4 in.~~
~~Results~~				
~~Peak cladding temperature, °F~~	~~1023~~	~~1358~~	~~1099~~	~~1275~~
~~Peak cladding location, ft~~	~~12.0~~	~~12.0~~	~~12.0~~	~~12.0~~
~~Local Zr/H_2O reaction (max), %~~	~~0.076~~	~~0.193~~	~~0.073~~	~~0.133~~
~~Local Zr/H_2O location, ft~~	~~12.0~~	~~12.0~~	~~12.0~~	~~12.0~~
~~Local Zr/H_2O reaction, %~~	~~<0.3~~	~~<0.3~~	~~<0.3~~	~~<0.3~~
~~Hot rod burst time, sec~~	~~No burst~~	~~No burst~~	~~No burst~~	~~No burst~~
~~Hot rod burst location, ft~~	—	—	—	—

~~Calculation Assumptions~~	~~Unit 2~~	~~Unit 1~~

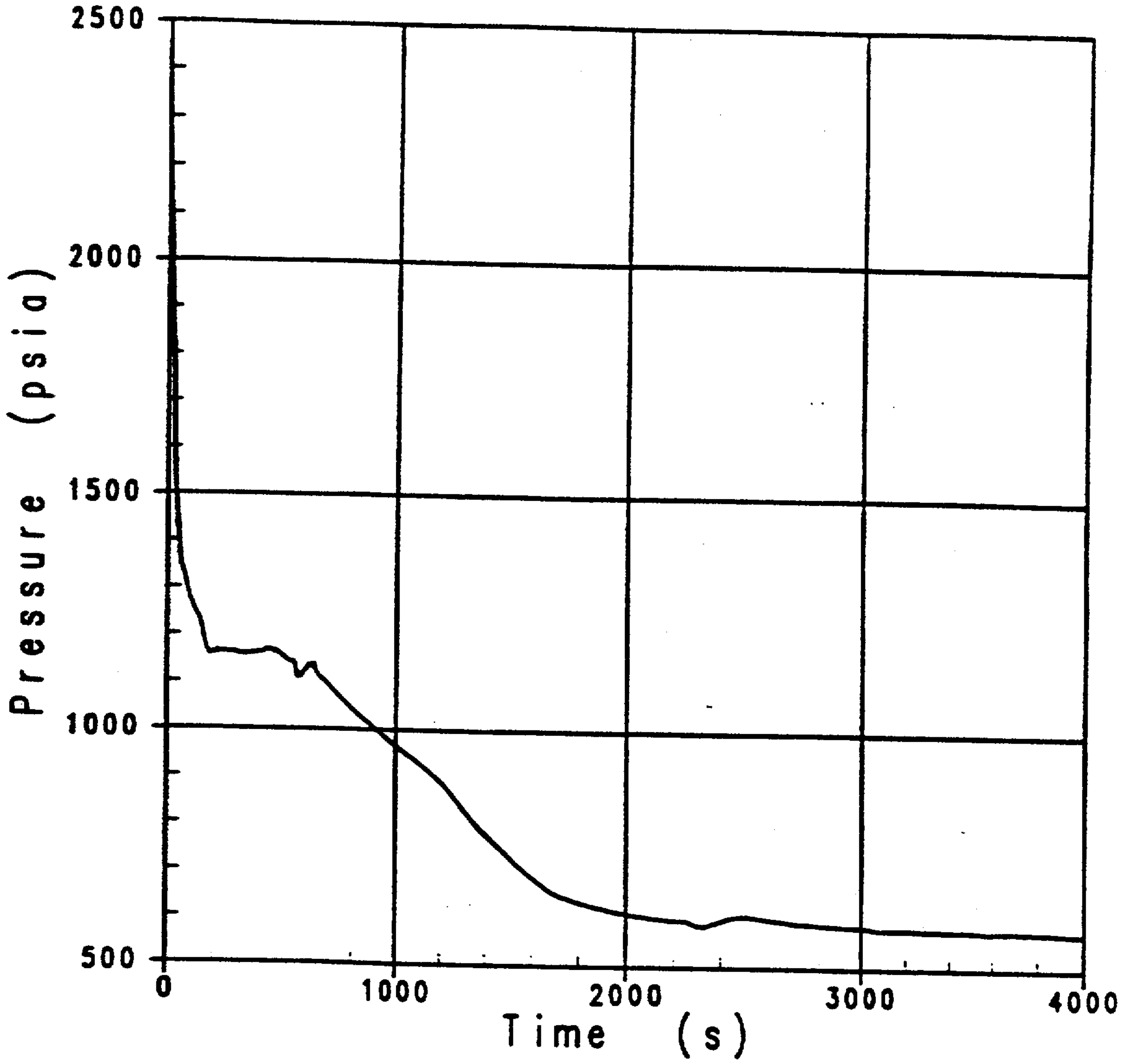
Remove SBLOCA Figures 15.3-1 through 15.3-14f
Replace with following Figures 15.3-1 through 15.3-15-DCPP2



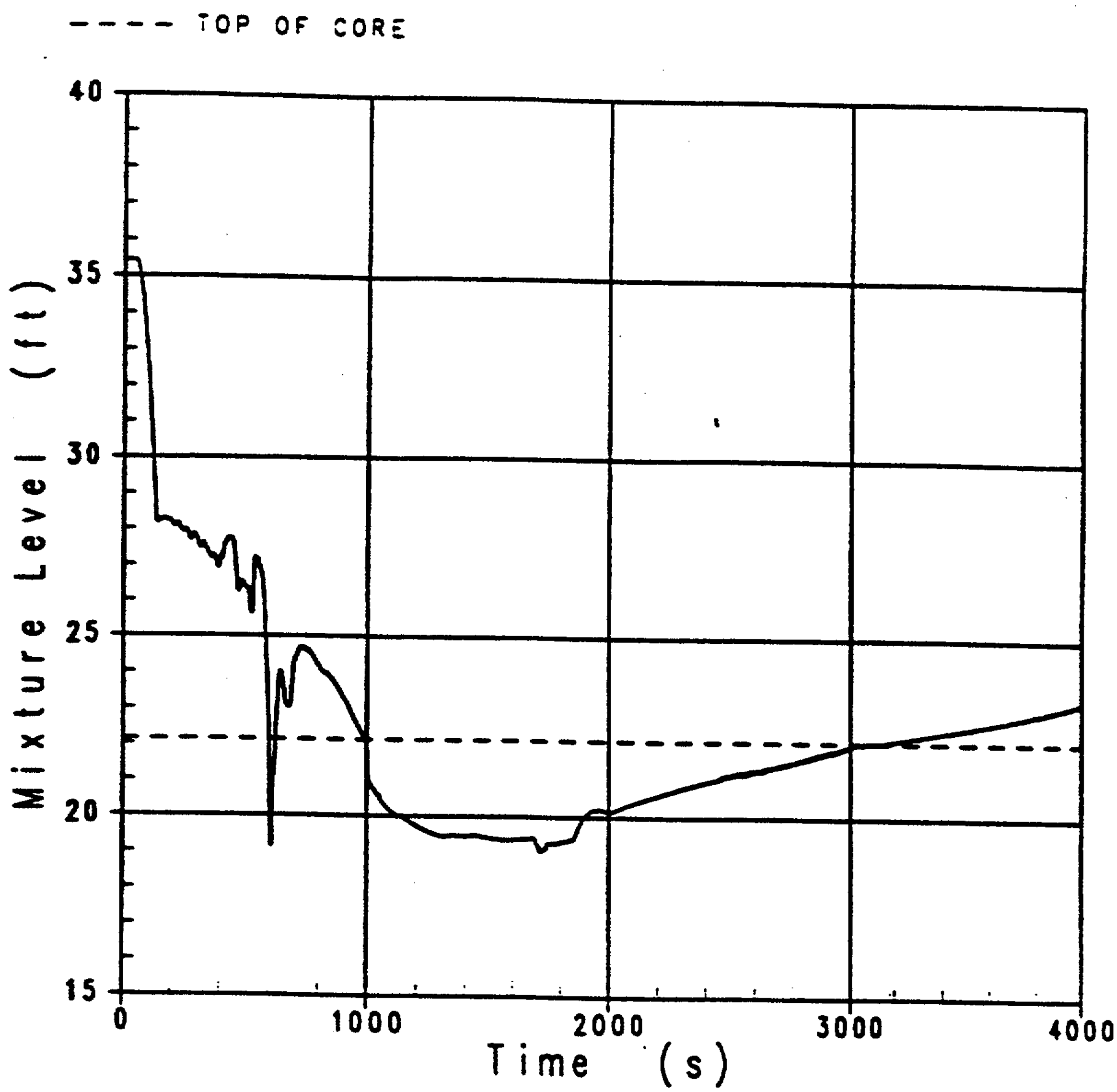
Safety Injection Flowrate for Small Break LOCA	DIABLO CANYON UNITS 1 and 2
	Figure 15.3-1



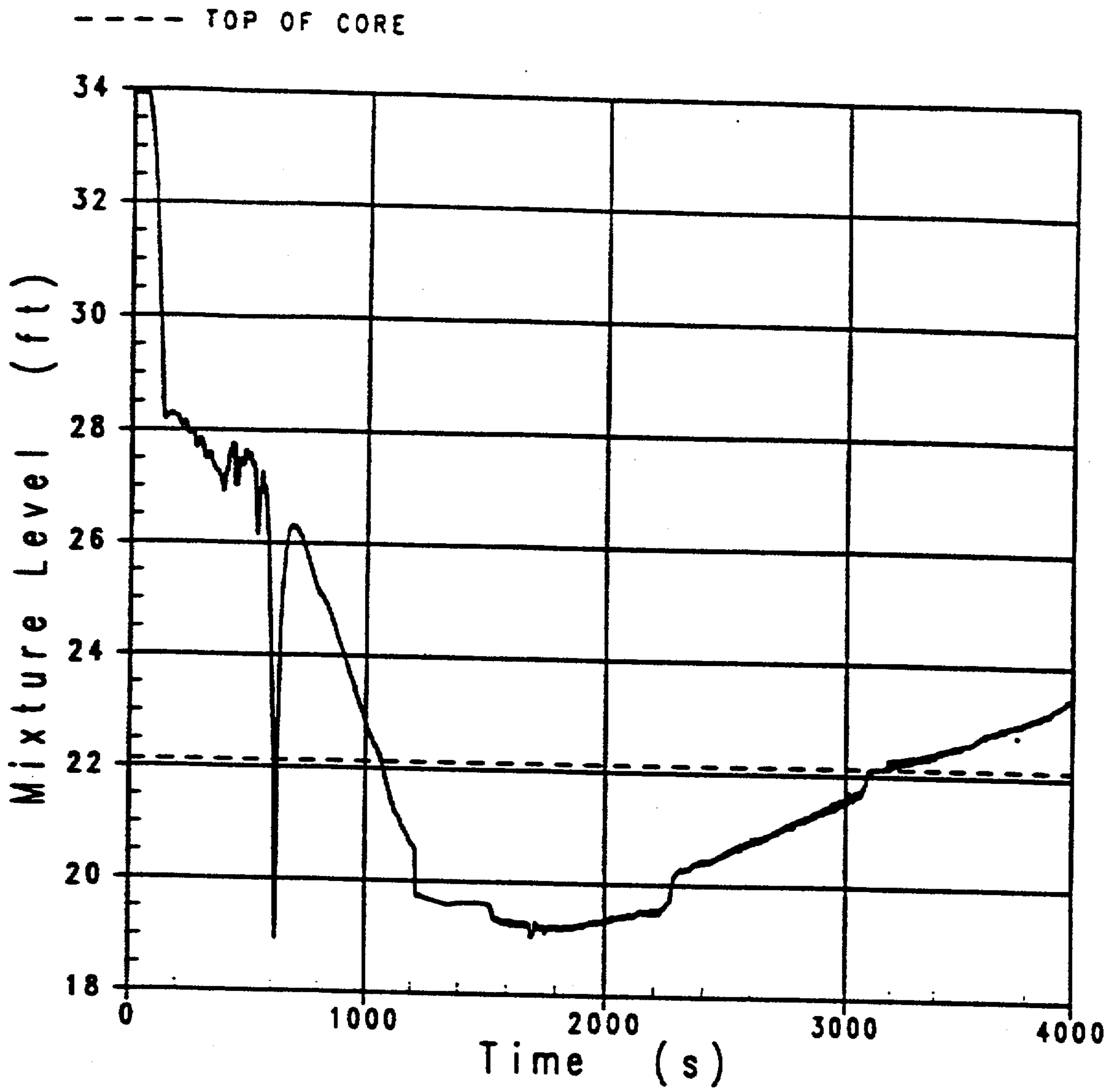
RCS Depressurization **3-inch Cold Leg Break**	**DIABLO CANYON UNIT 1** **Figure 15.3-2-DCPP1**



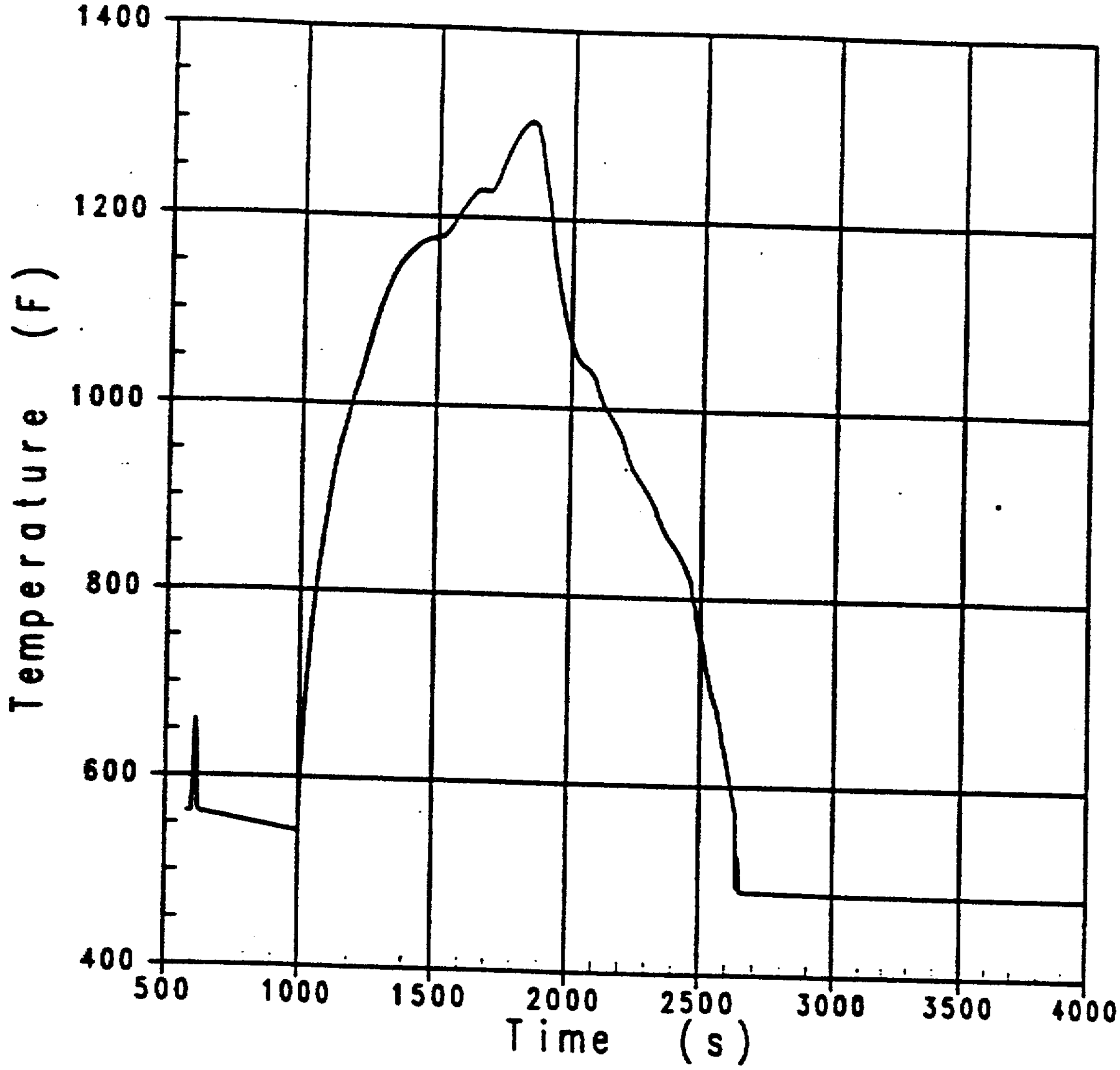
RCS Depressurization 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-2-DCPP2



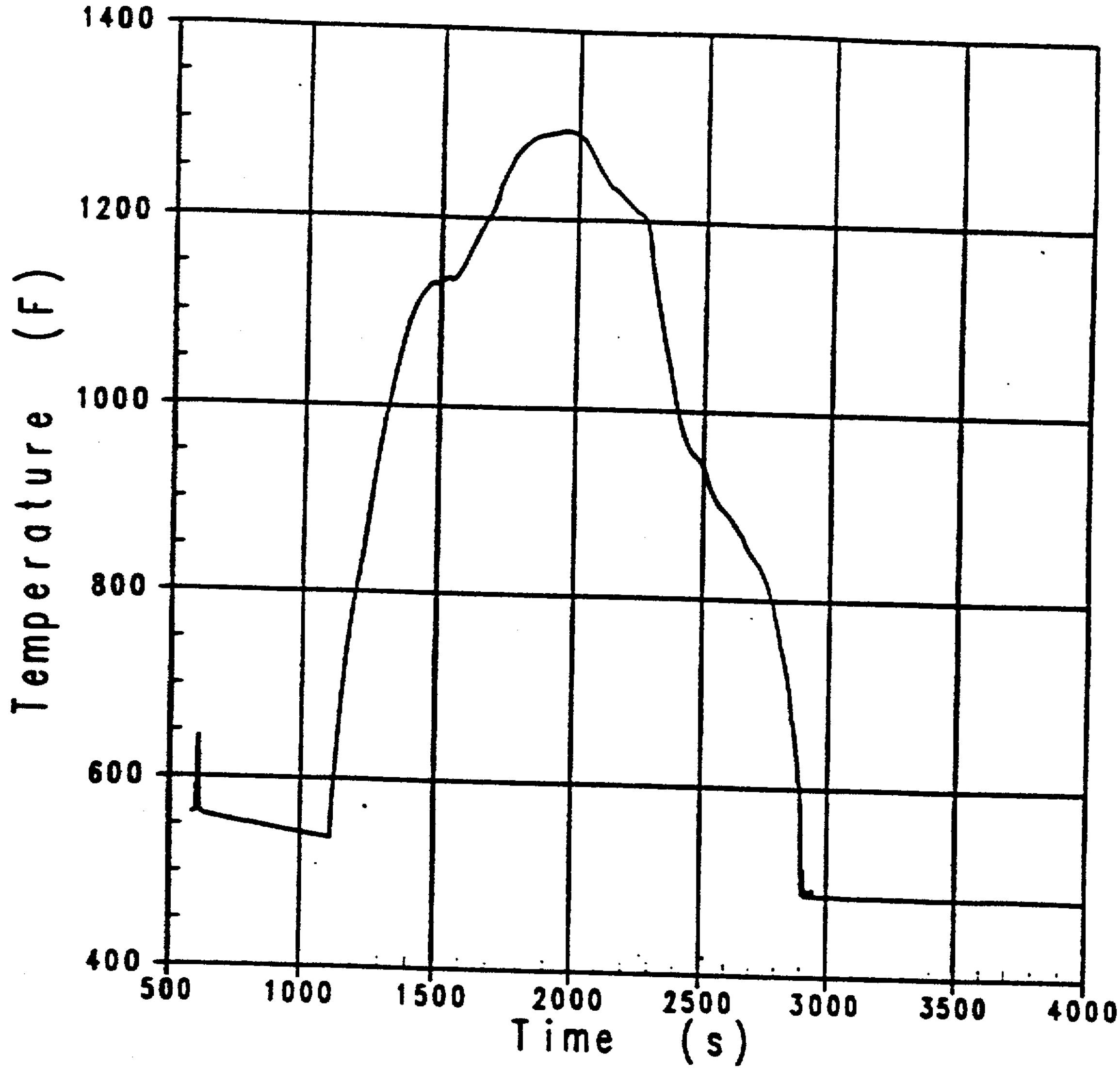
Core Mixture Elevation **3-inch Cold Leg Break**	**DIABLO CANYON UNIT 1**
	Figure 15.3-3-DCPP1



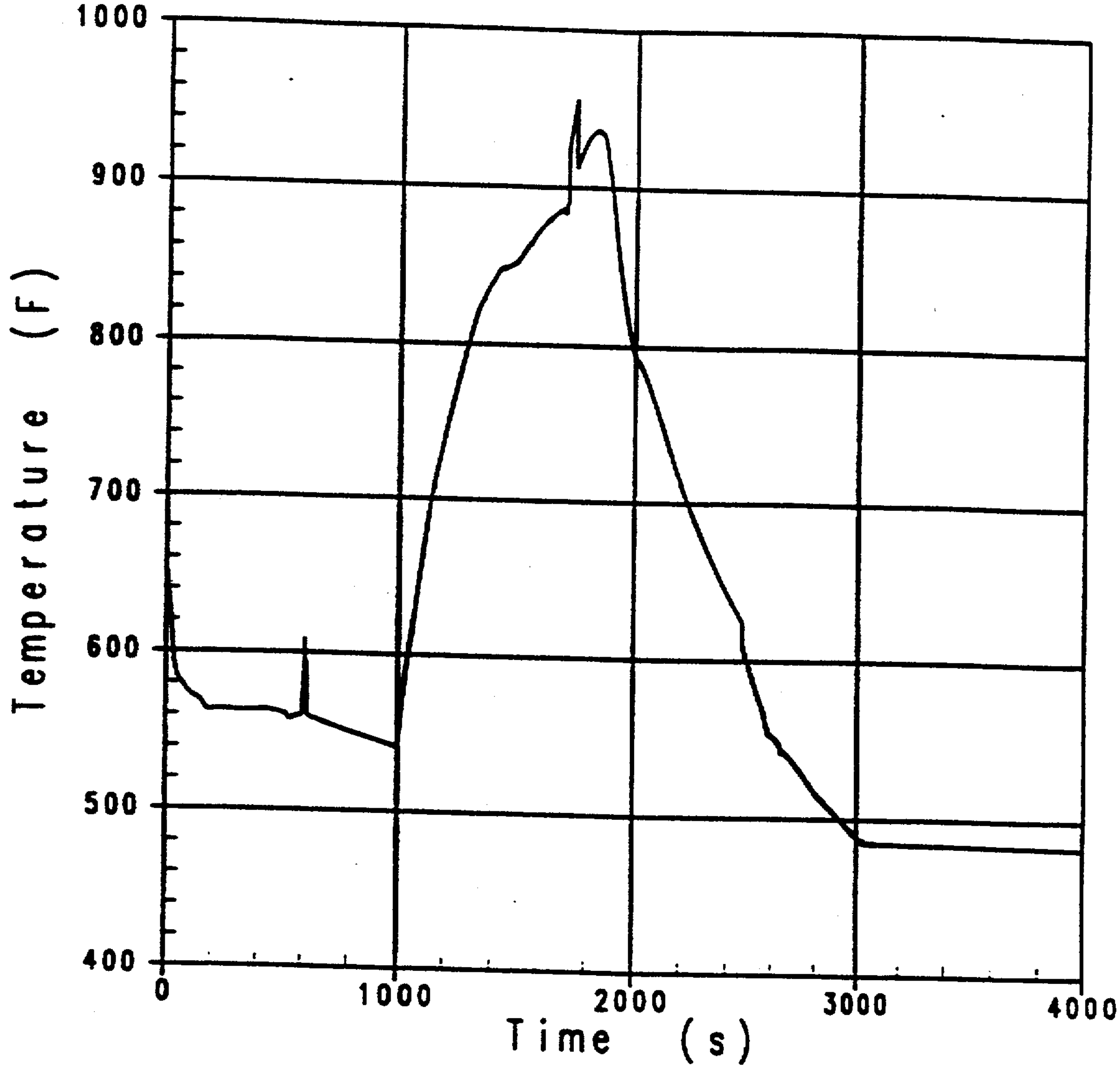
Core Mixture Elevation 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-3-DCPP2



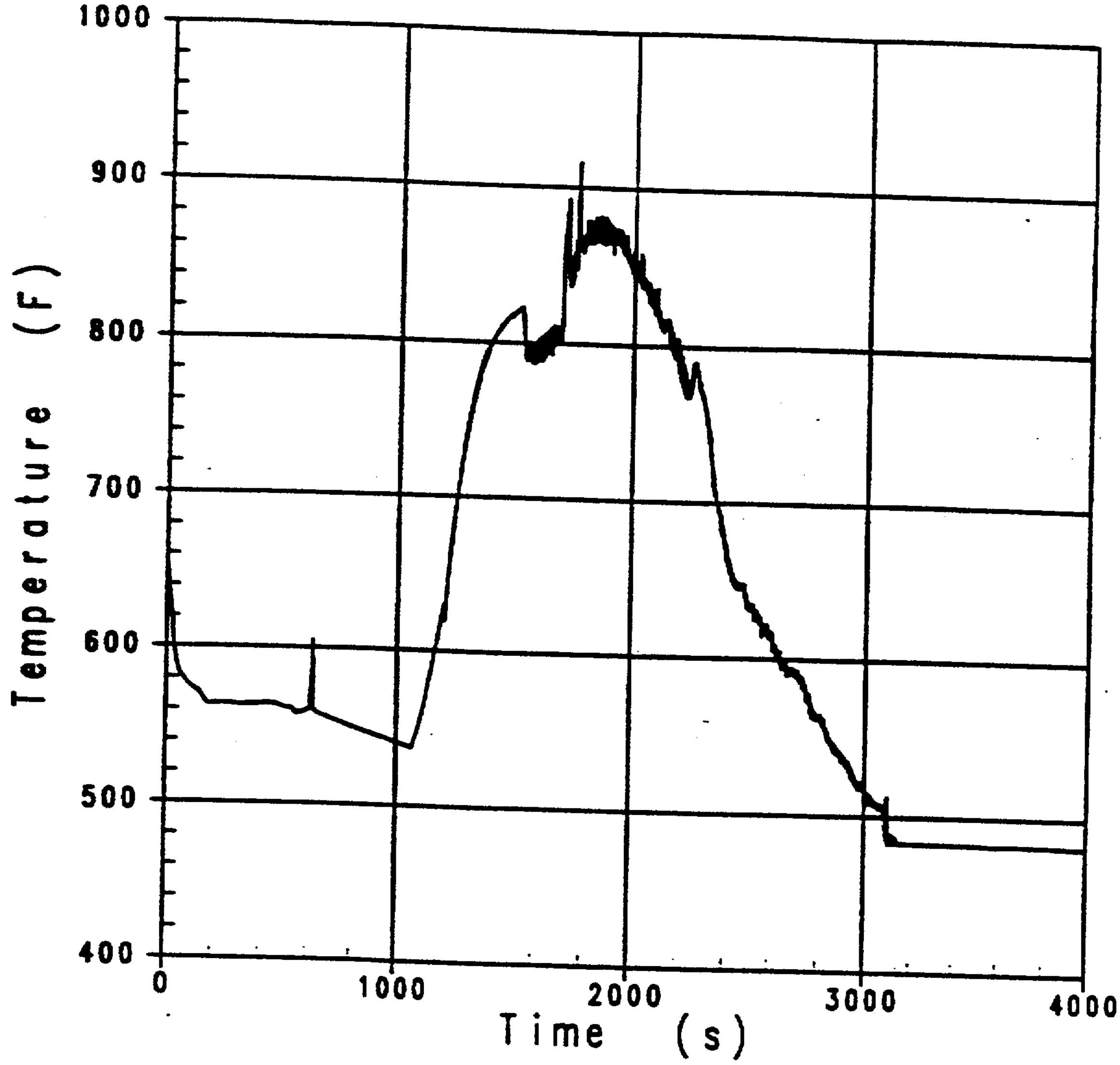
Clad Temperature Transient 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-4-DCPP1



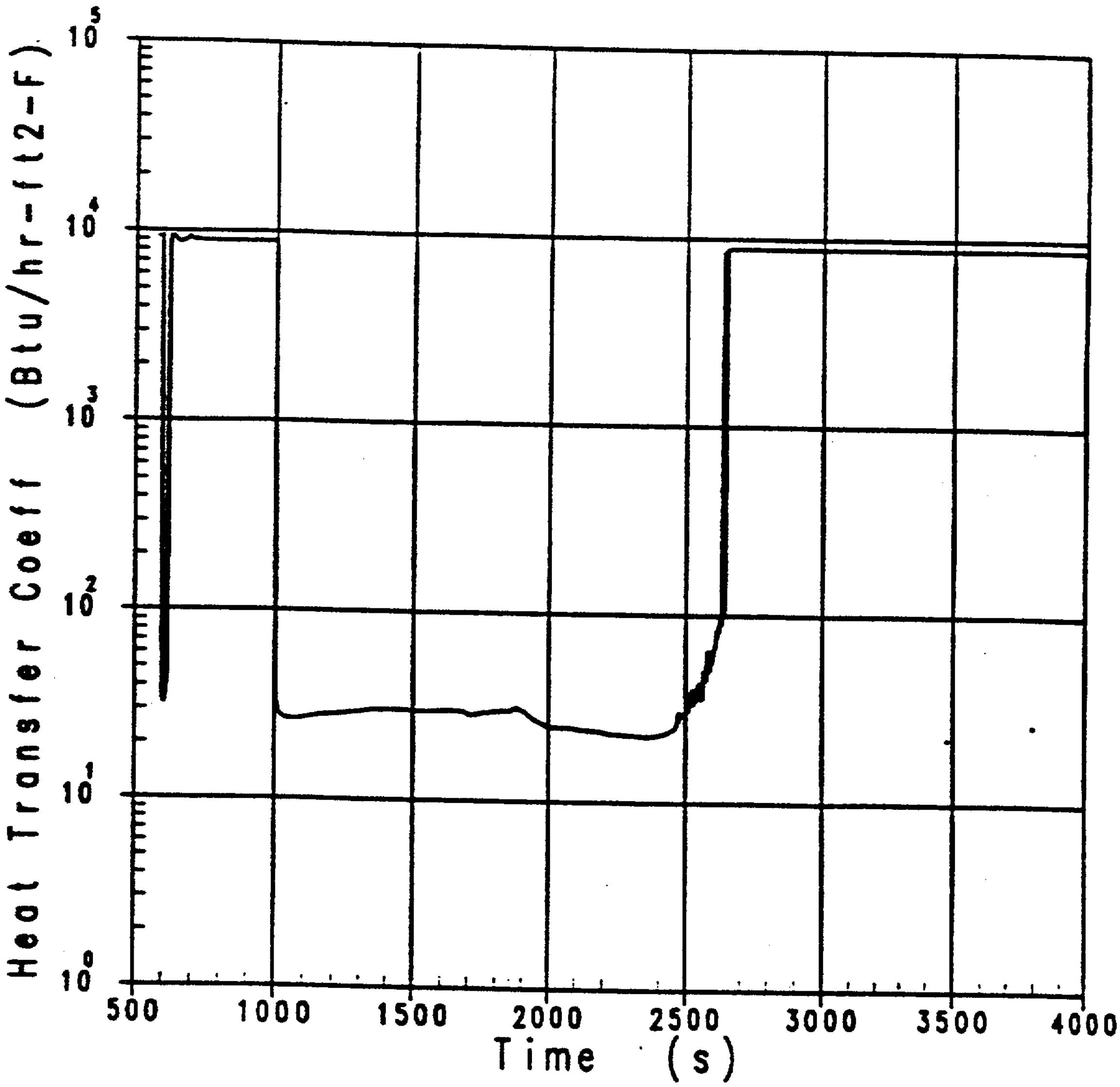
Clad Temperature Transient 3-inch Cold Leg Break	DIABLO CANYON UNIT 2 Figure 15.3-4-DCPP2



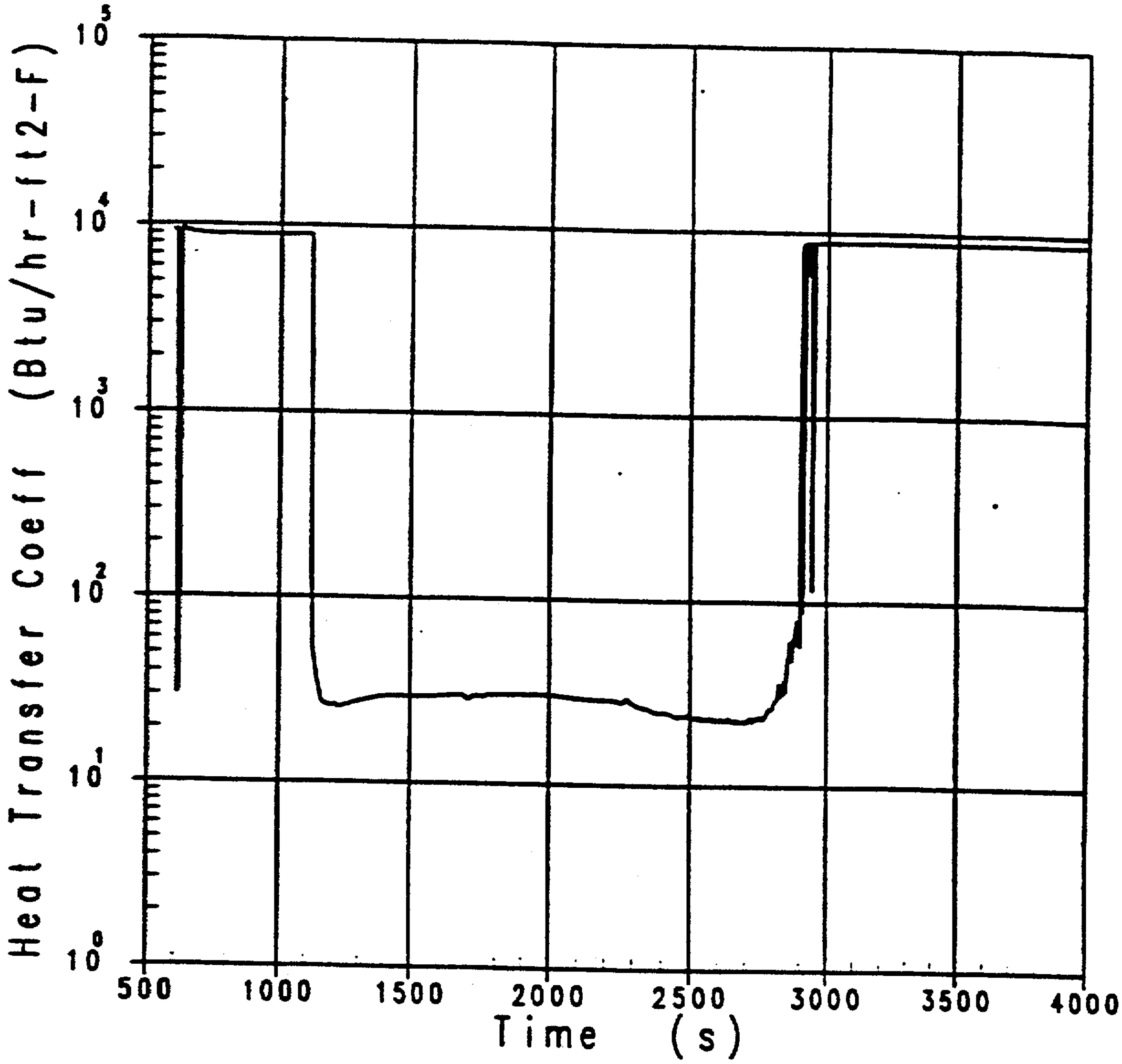
Top Core Node Vapor Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 1 Figure 15.3-5-DCPP1



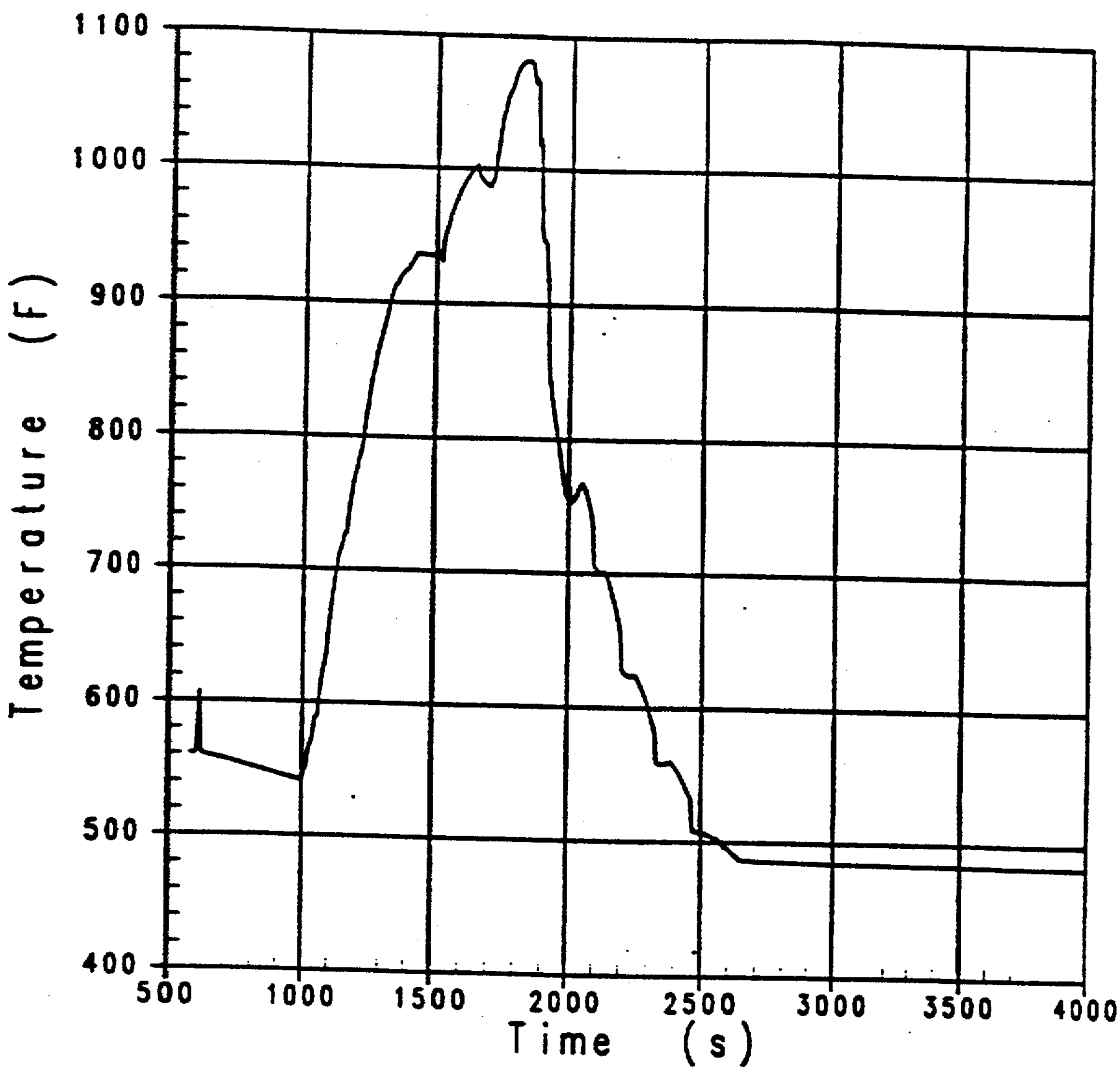
Top Core Node Vapor Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 2 Figure 15.3-5-DCPP2



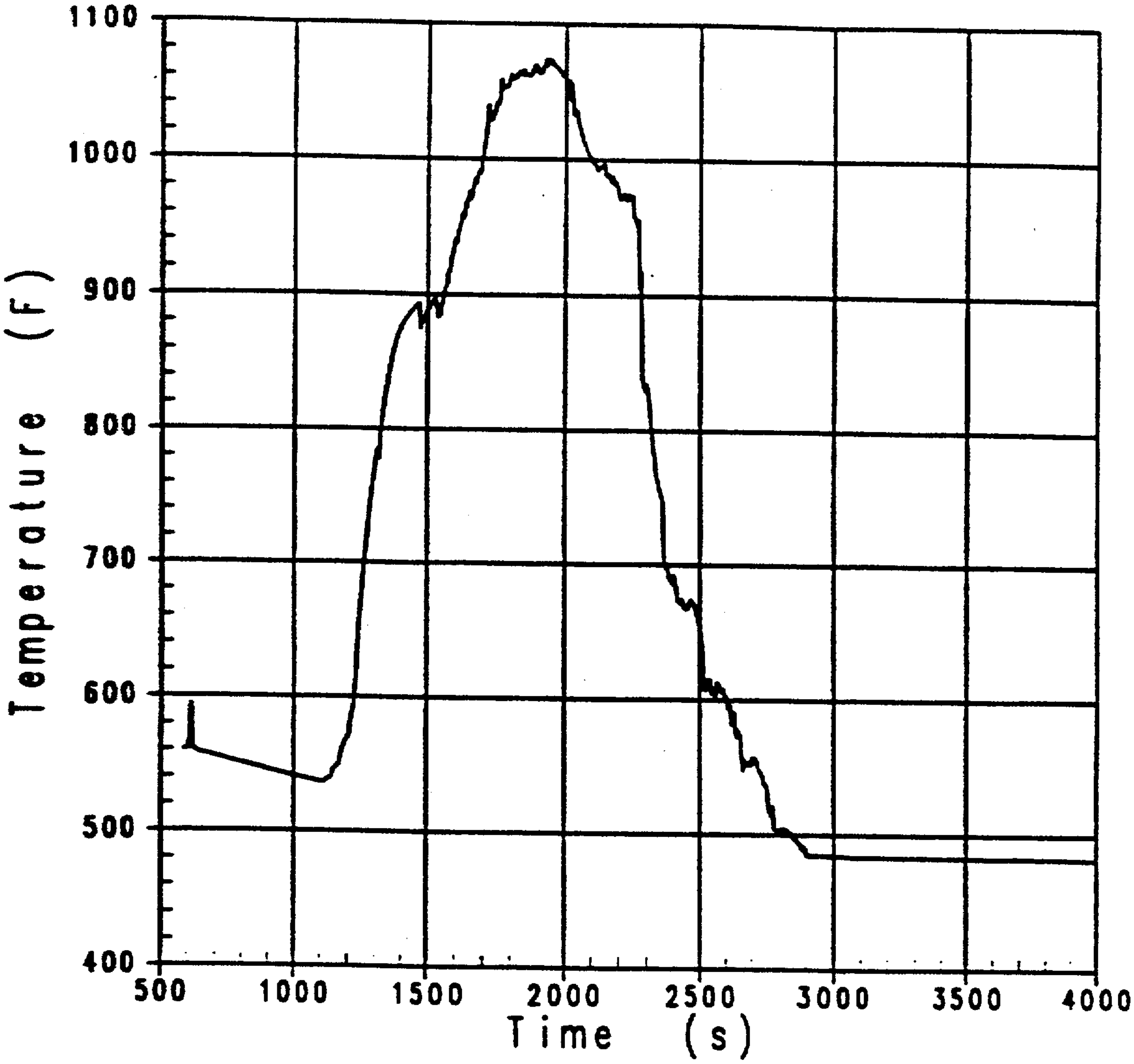
Rod Film Coefficient **3-inch Cold Leg Break**	**DIABLO CANYON UNIT 1**
	Figure 15.3-6-DCPP1



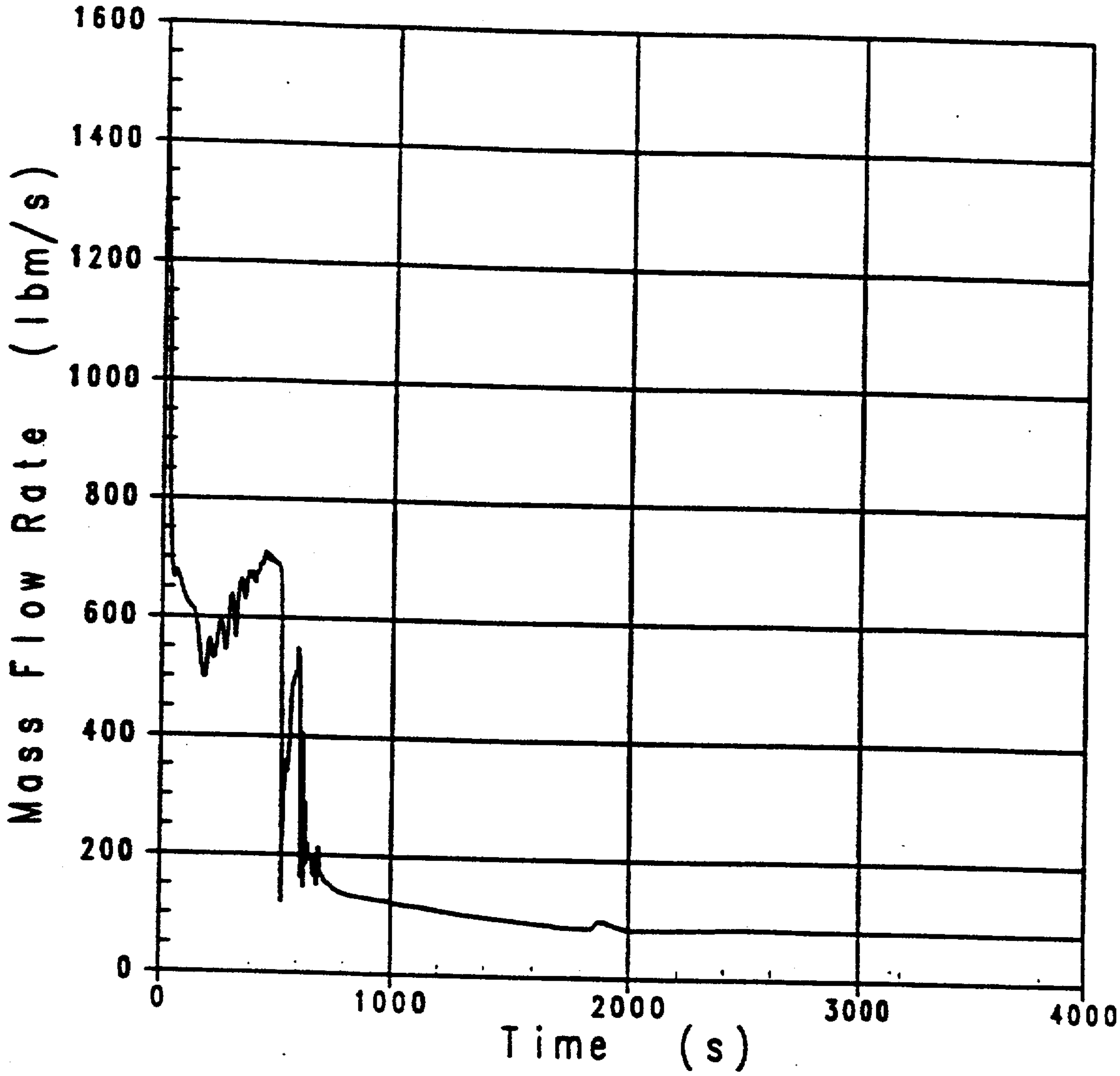
Rod Film Coefficient 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-6-DCPP2



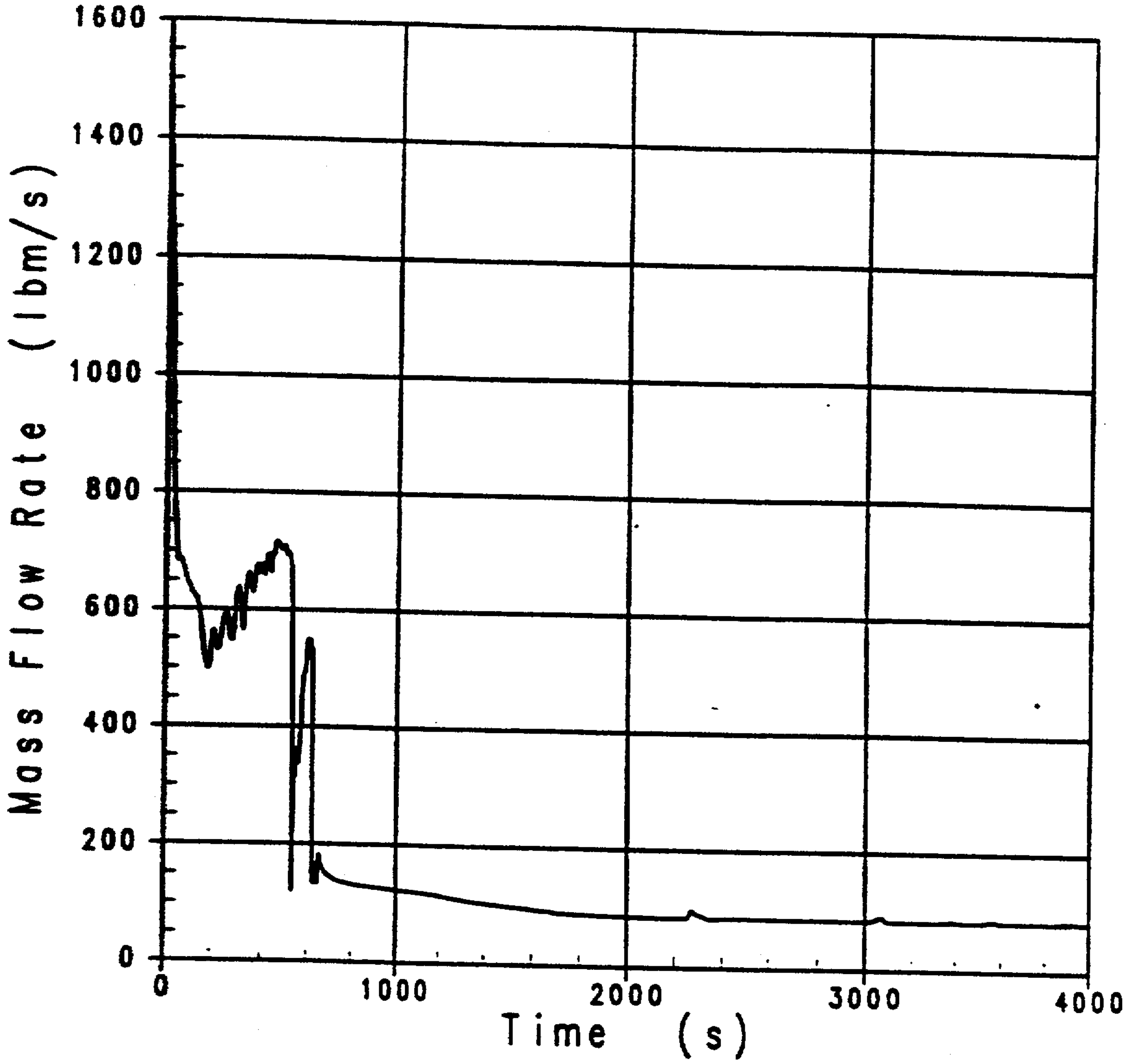
Hot Spot Fluid Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 1 Figure 15.3-7-DCPP1



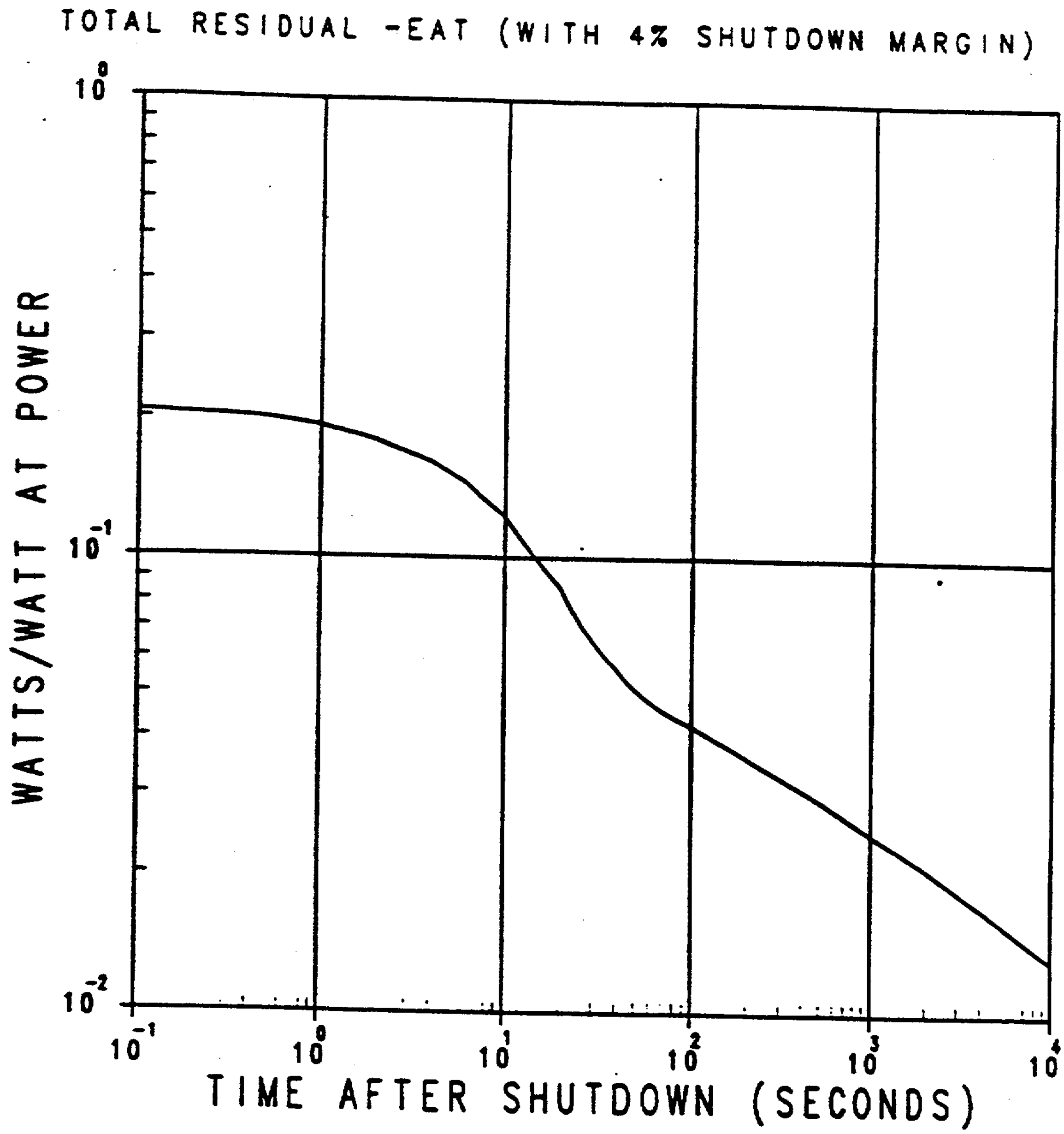
Hot Spot Fluid Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-7-DCPP2



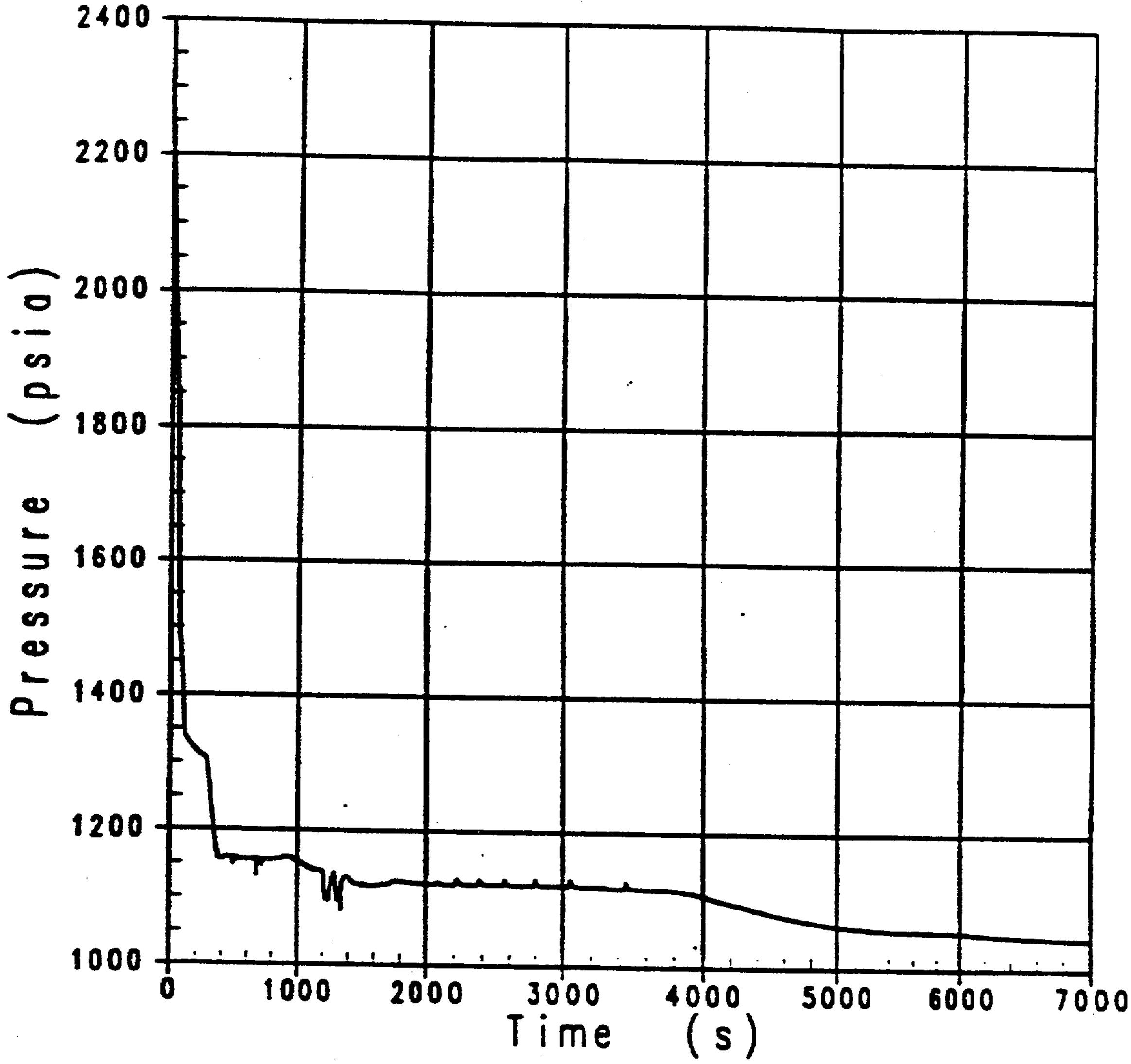
Break Mass Flow 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-8-DCPP1



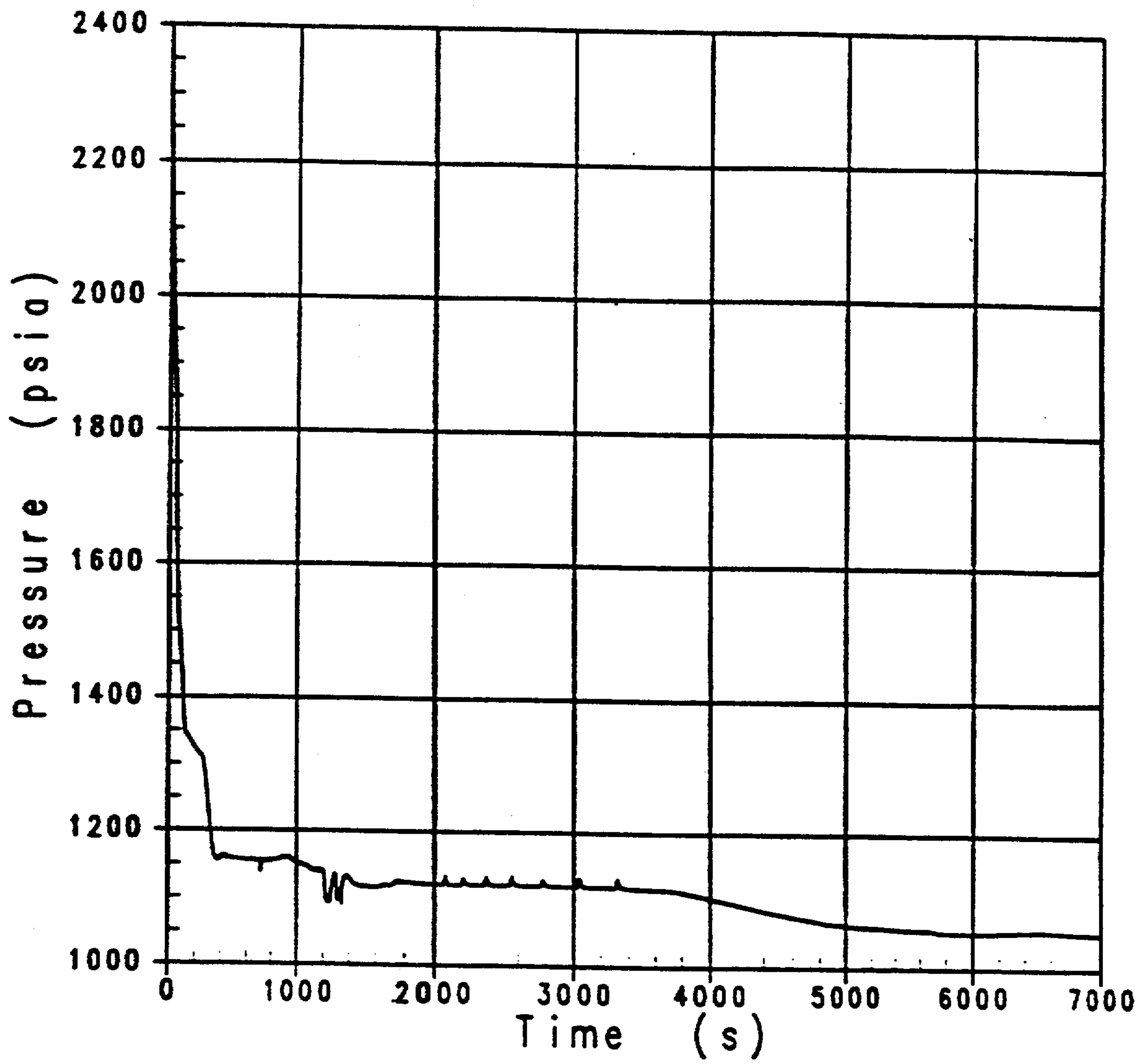
Break Mass Flow 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-8-DCPP2



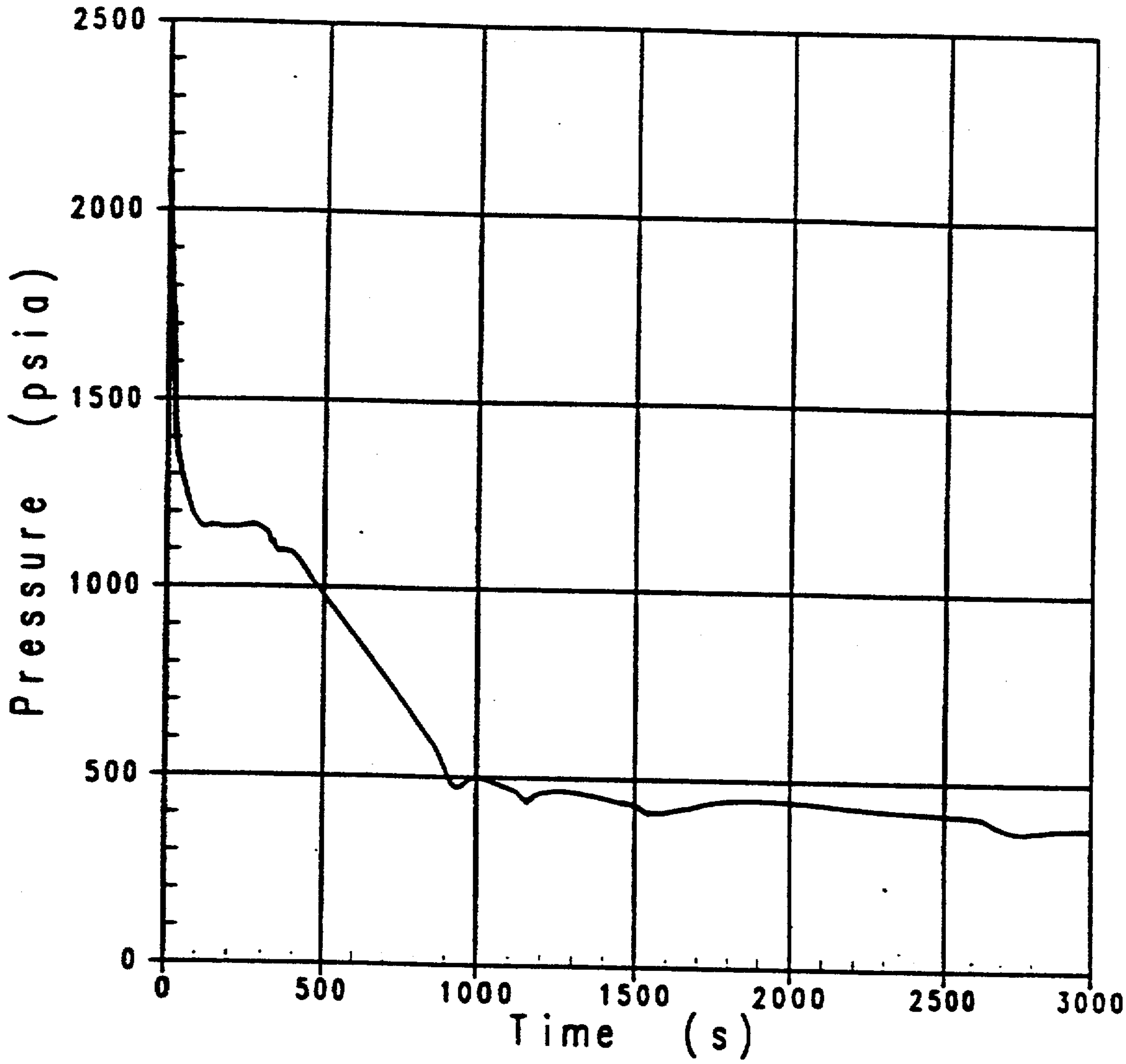
LOCA Core Power Transient	DIABLO CANYON UNITs 1&2
	Figure 15.3-9



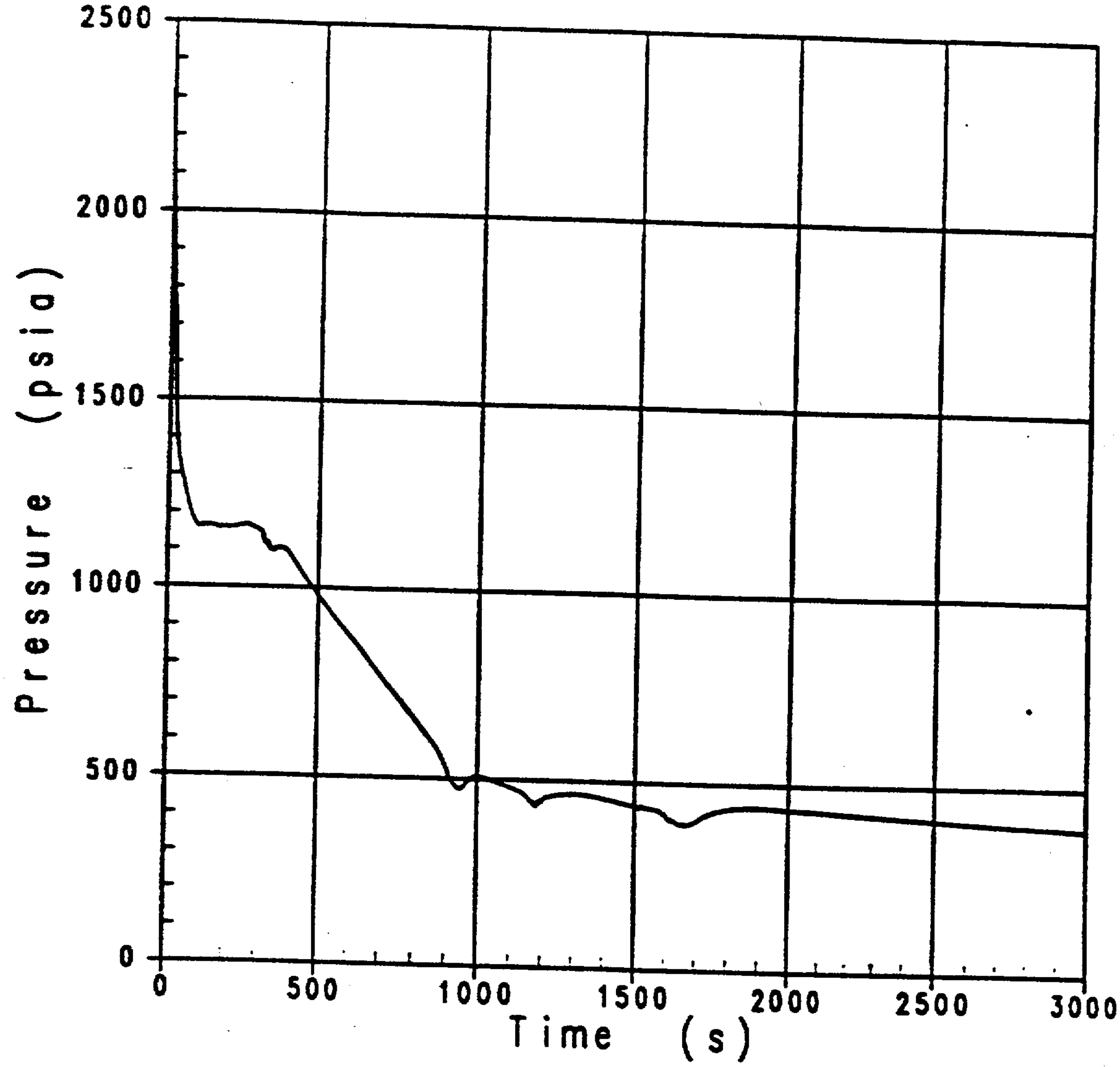
RCS Depressurization **2-inch Cold Leg Break**	**DIABLO CANYON UNIT 1**
	Figure 15.3-10-DCPP1



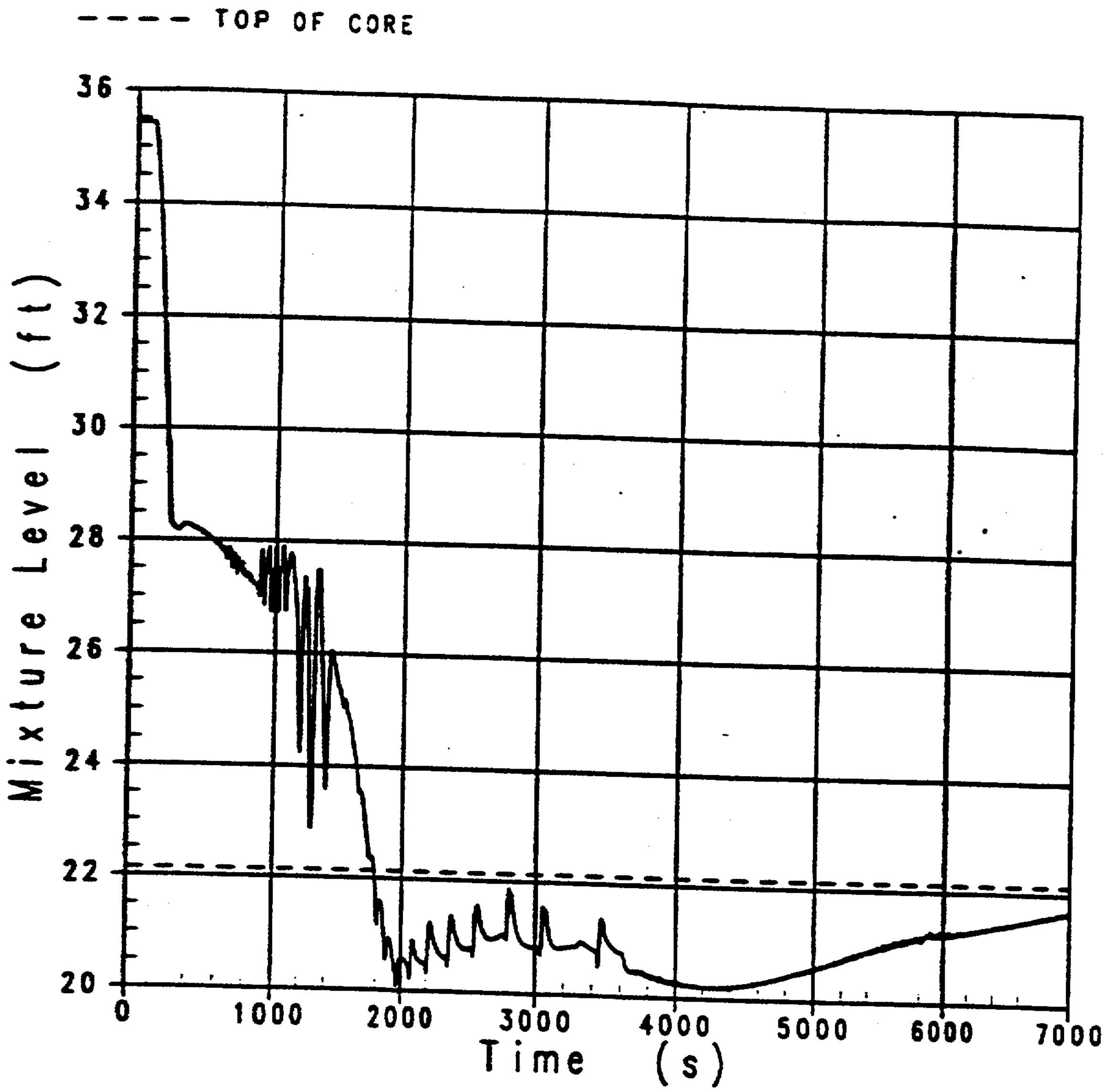
RCS Depressurization 2-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-10-DCPP2



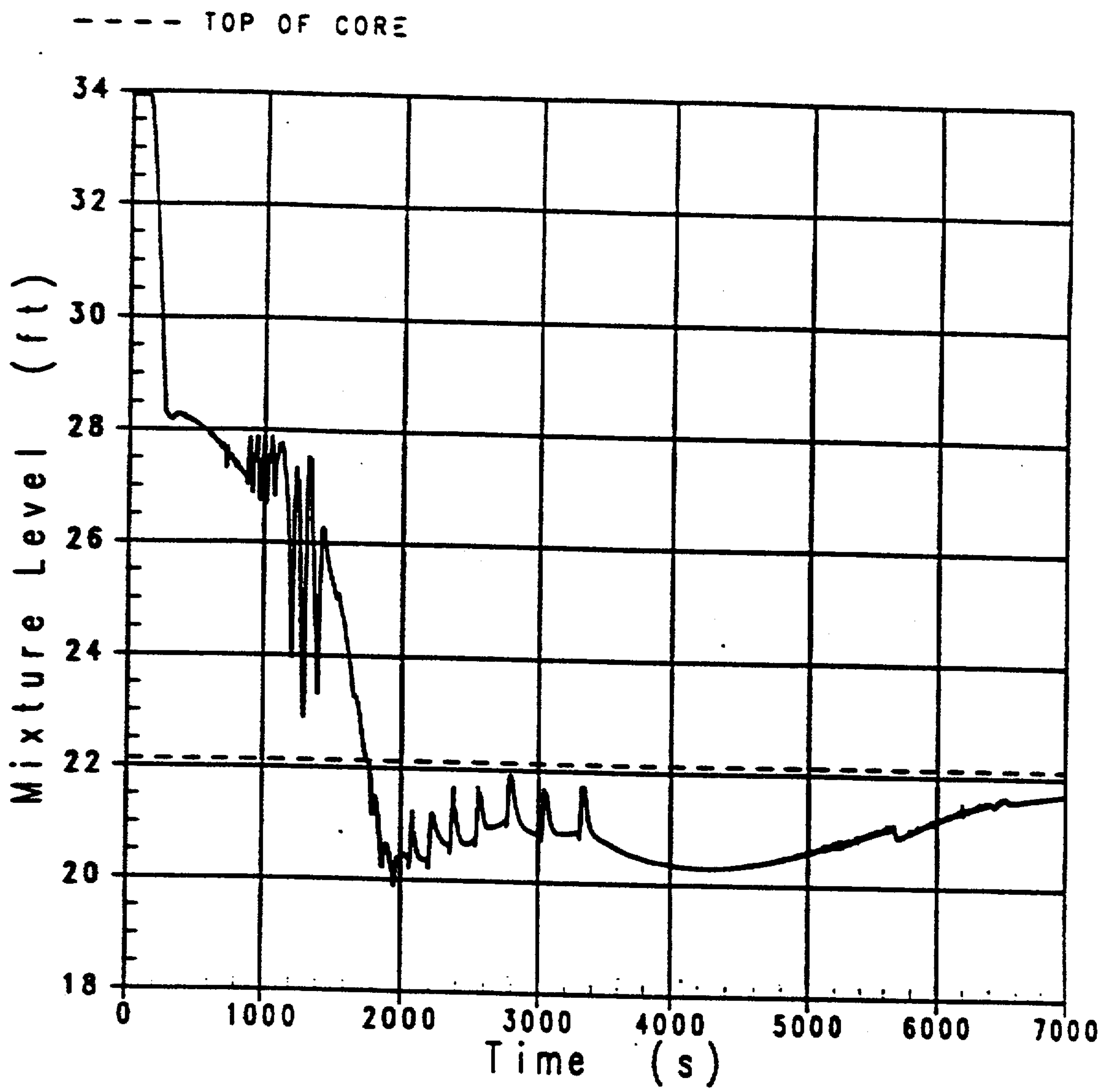
RCS Depressurization 4-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-11-DCPP1



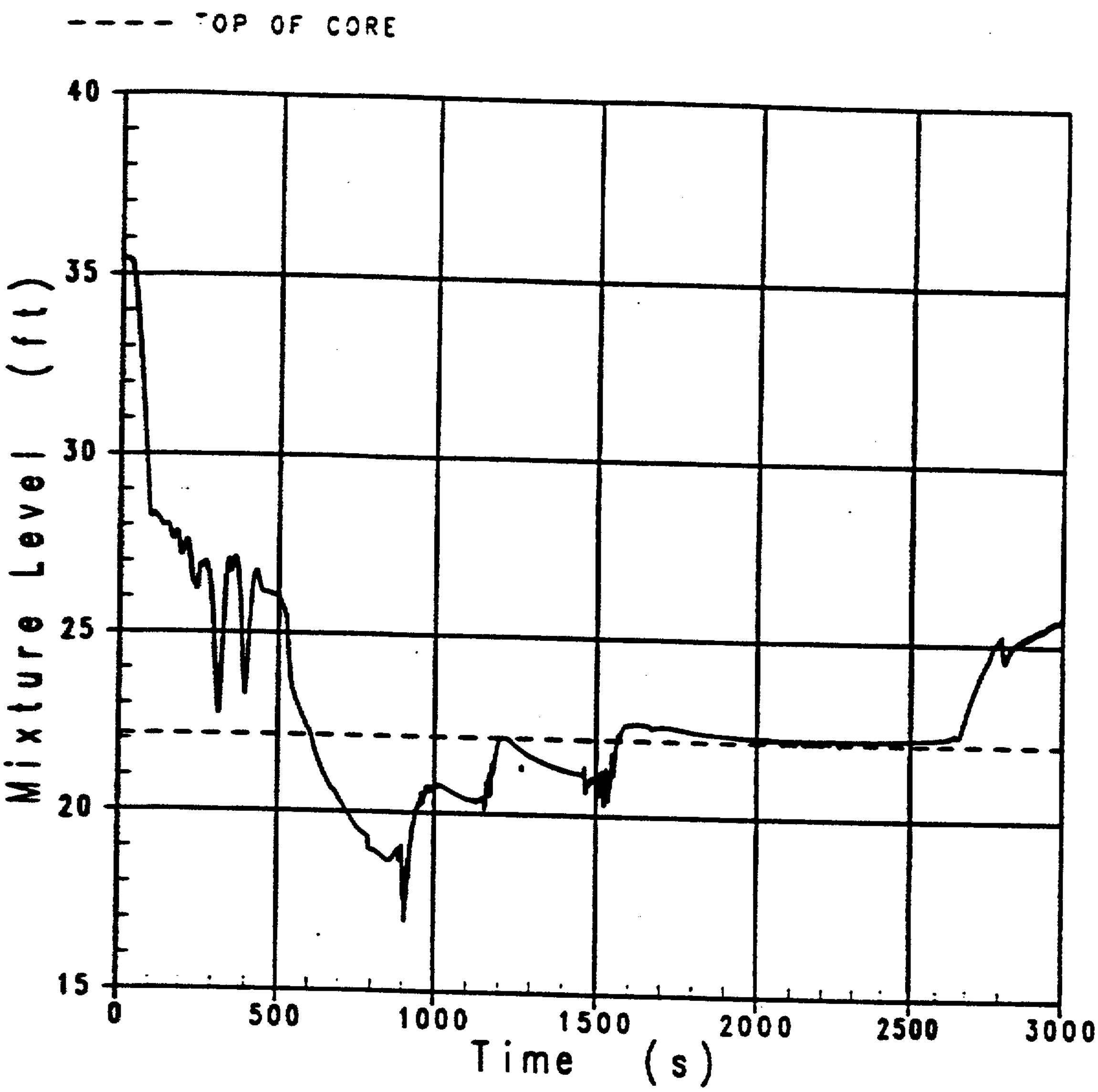
RCS Depressurization 4-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-11-DCPP2



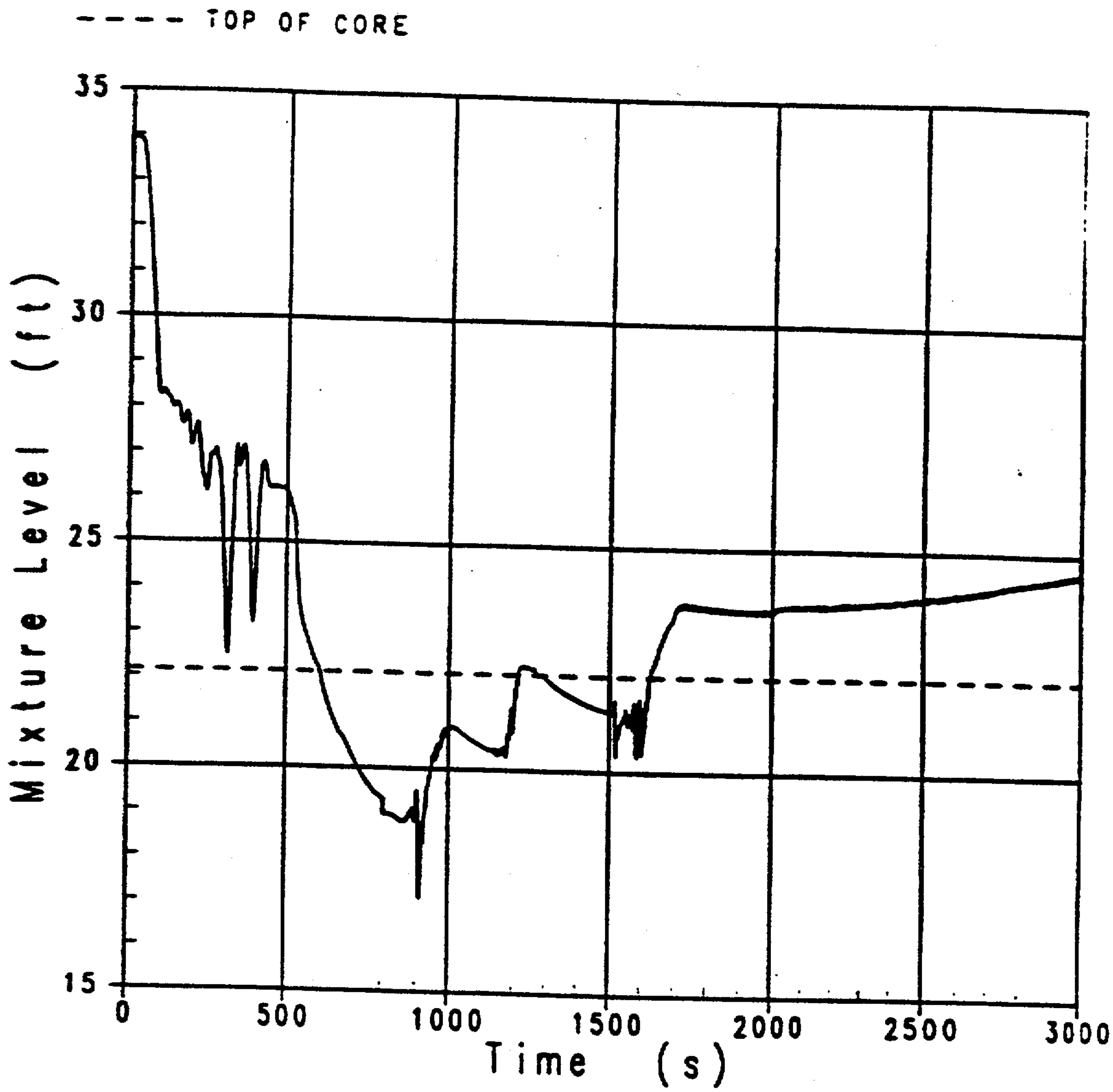
Core Mixture Elevation 2-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-12-DCPP1



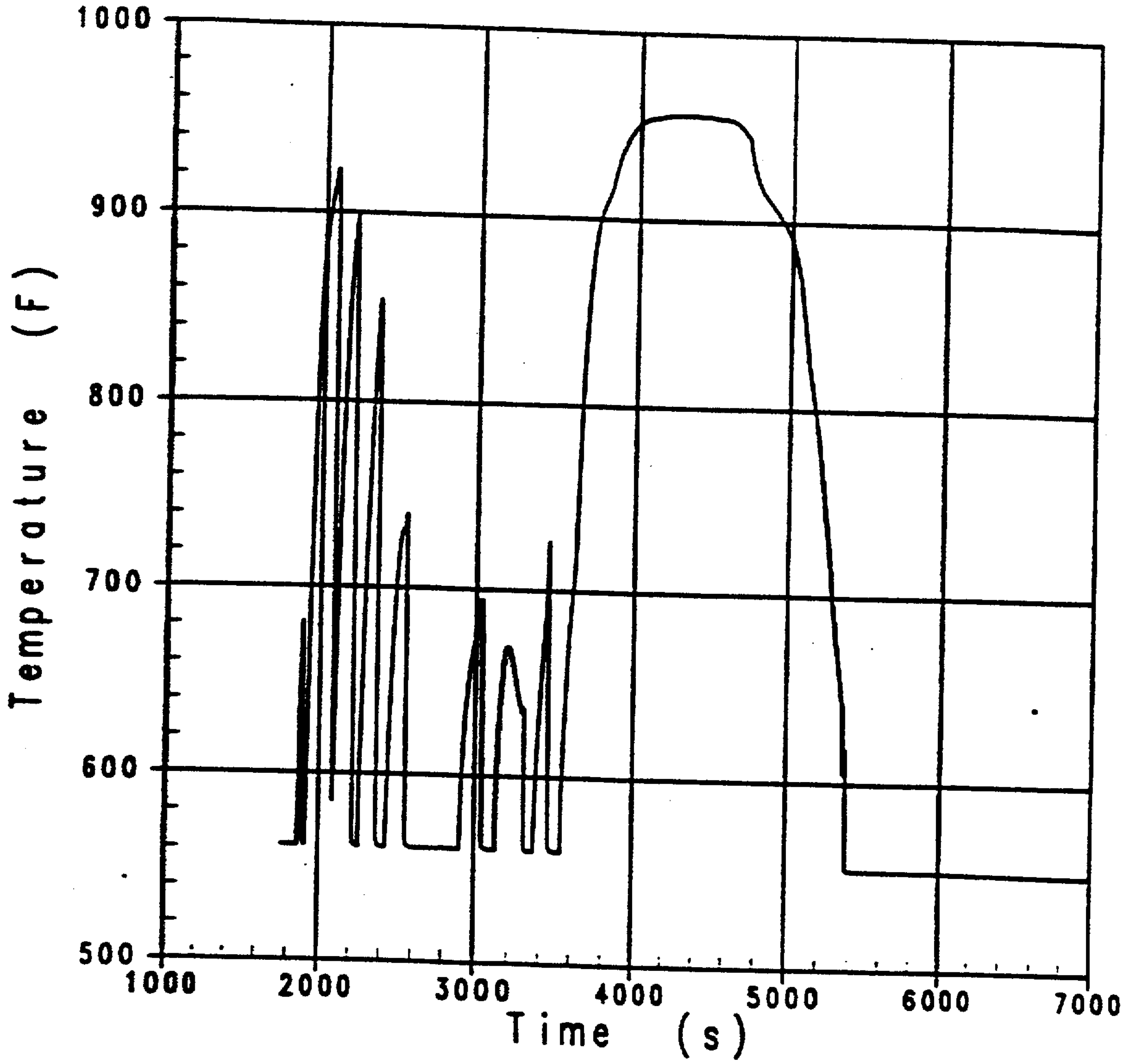
Core Mixture Elevation 2-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-12-DCPP2



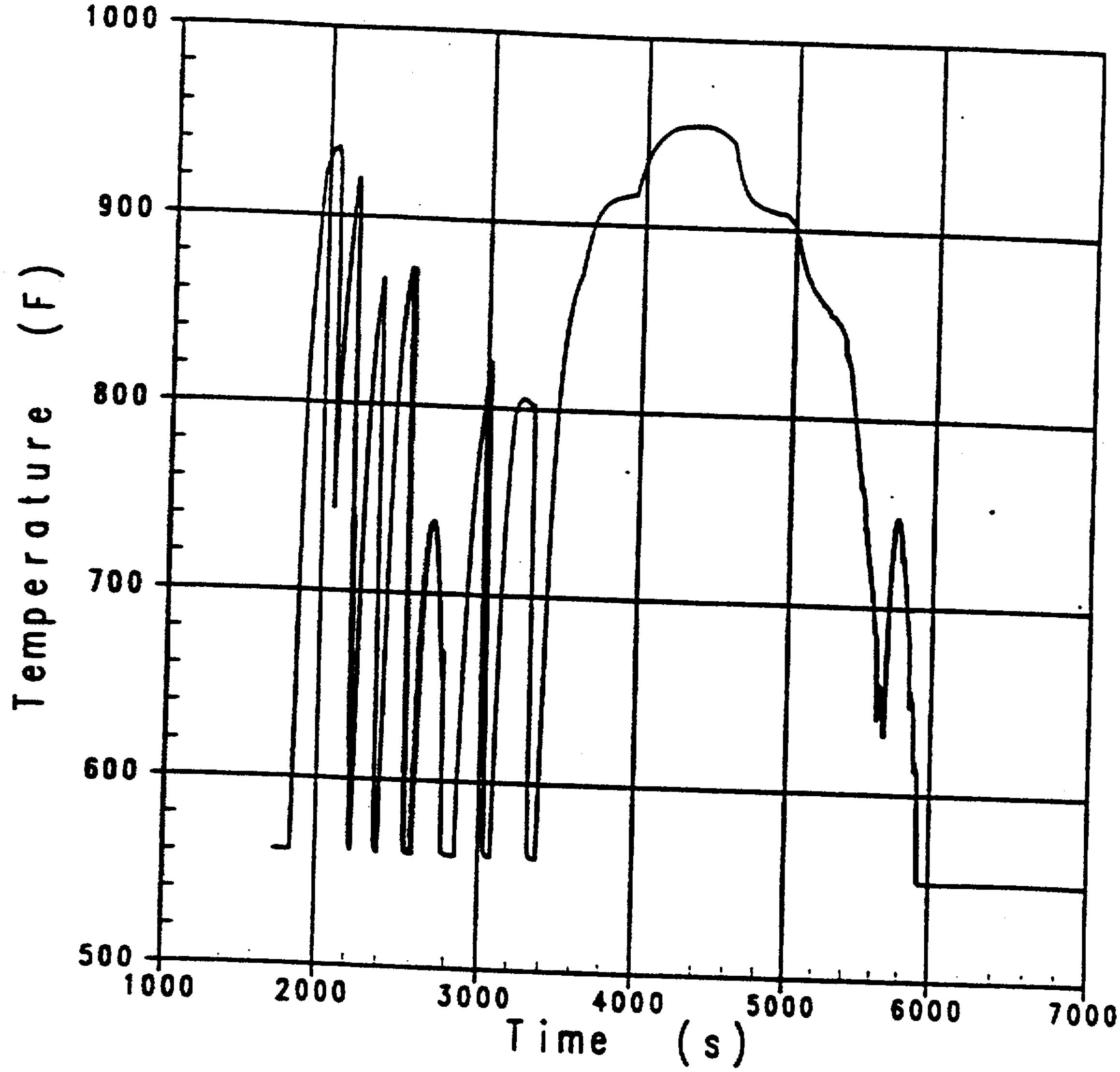
Core Mixture Elevation 4-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-13-DCPP1



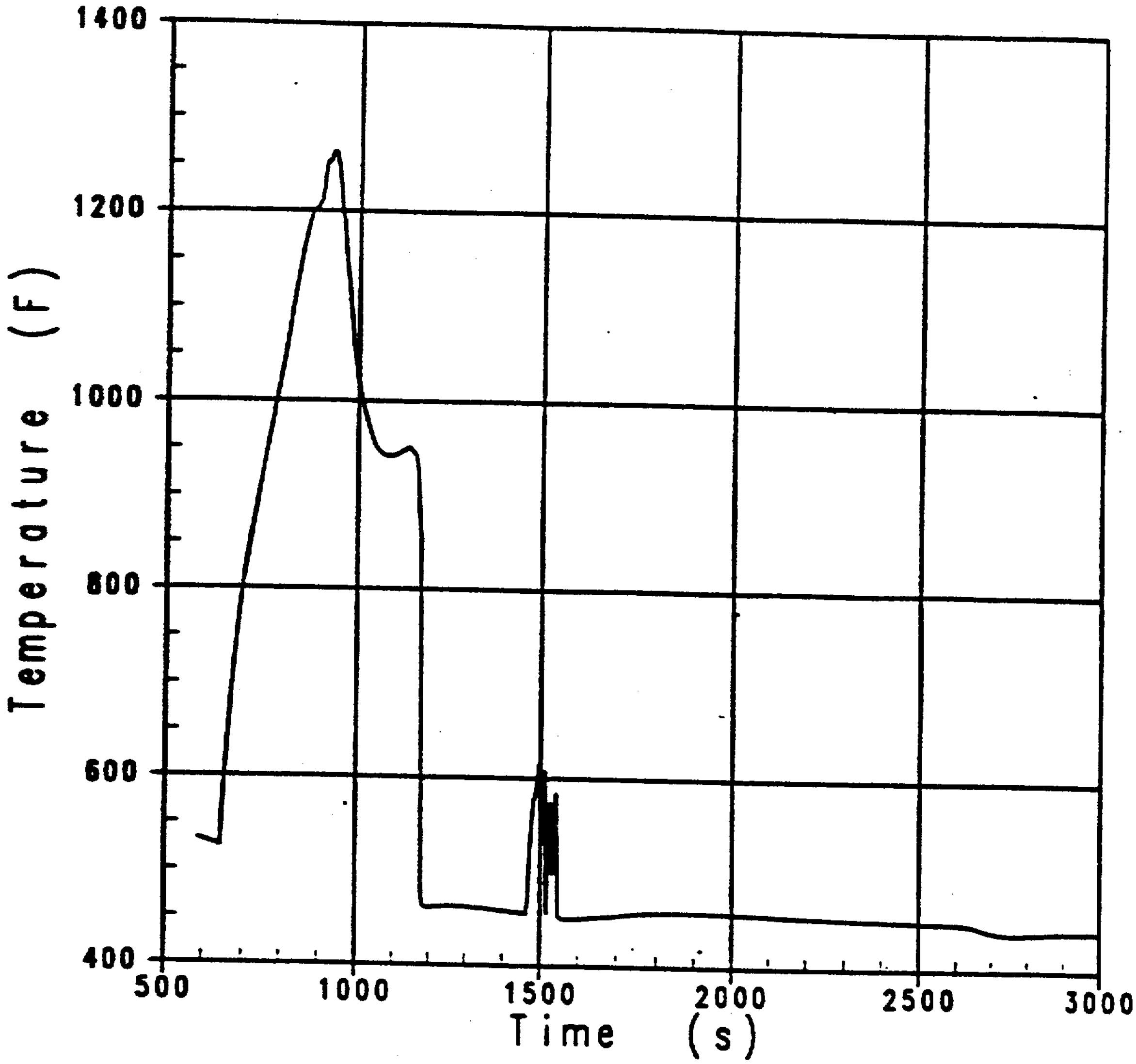
Core Mixture Elevation 4-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-13-DCPP2



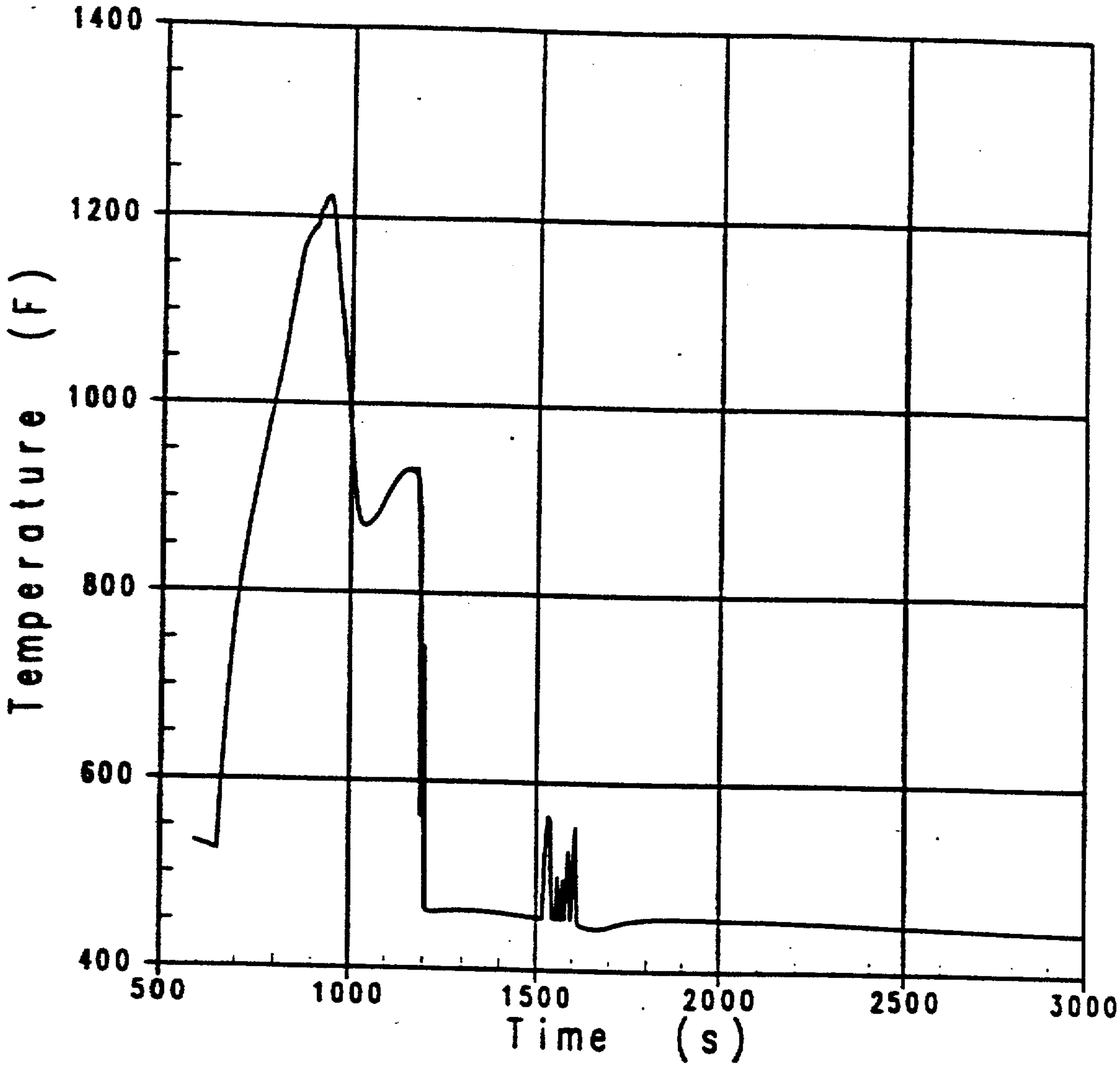
Cladding Temperature Transient	**DIABLO CANYON UNIT 1**
2-inch Cold Leg Break	**Figure 15.3-14-DCPP1**



Cladding Temperature Transient 2-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-14-DCPP2



Cladding Temperature Transient 4-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-15-DCPP1



Cladding Temperature Transient **4-inch Cold Leg Break**	**DIABLO CANYON** **UNIT 2**
	Figure 15.3-15-DCPP2

ENCLOSURE H OF
PG&E LETTER DCL-99-170

MARK-UP OF PRECAUTIONS, LIMITATIONS AND SETPOINT PAGES

B-3

3. Overpower ΔT trip
(TC-411G, TC-421G, TC-431G, TC-441G) **4

ΔT reactor trip setpoint:

37 4

$$\Delta T \left(\frac{1+T_4S}{1+T_5S}\right) = \Delta To \left[K4 - K5\left(\frac{T_3S}{1+T_3S}\right)T - K6\ (T - T') - f_2(\Delta q)\right]$$

where,

ΔT = Measured loop differential temperature (TH-TC)

ΔTo = Indicated ΔT at rated thermal power

T = Average temperature, °F

T' = Indicated average temperature at nominal conditions and rated power, for the channel being calibrated (for plant startup, assume T' = 576.6°F for Unit 1 and 577.6°F for Unit 2)

**577.3

K_4 = (see parameter list below)
K_5 = (see parameter list below)
T_3 = (see parameter list below)
K_6 = (see parameter list below)
$f_2(\Delta q)$ = 0 for all Δq
ΔT = Measured loop differential temperature (TH-TC)
ΔTo = Indicated ΔT at rated thermal power

37 4

<u>Parameter</u>

K_4 = 1.072 (Units 1 and 2 Cycle 4 and after);

K_5 = 0.0174/°F for increasing average temperature and 0 for decreasing average temperature;

K_6 = 0.00145/°F for T > T ; K_6 = 0 for T≤ T ; (Units 1 and 2 Cycle 4 and after)

37 4

T_3	10 seconds
T_4	0 seconds
T_5	0 seconds

1. Impulse unit time constant (PM-506C) — 140 sec.

2. C-7A load loss setpoint (PC-506C) — Pressure equivalent to 10% of full power

3. C-7B load loss setpoint (PC-506D) — Pressure equivalent to 50% of full power

P. C-9 (signals indicating that condenser is not available for steam dump)

(By others)

Q. C-11 (rod withdrawal block when Control Bank D is above withdrawal limit)

(DC-442D) [YC-422D] — 220 steps

III. <u>Control Systems</u>

1. <u>Reactor Control</u>

A. Coolant average temperature (program)

		Setpoint for full load	Setpoint for full load Tavg = 568.0°F
1.	High limit (TC-505, TC-505A)	** 577.3 (Unit 1) 576.6°F(4) (Unit 2) 577.6°F(4)	568.0
2.	Low limit (TC-505, TC-505A)	547°F (4)	547°F
3.	Full power temperature	** 577.3 (Unit 1) 576.6°F(1) (Unit 2) 577.6°F(1)	568°F
4.	Hot shutdown	547°F (1)	547°F
5.	Temperature gain	** 0.303 (Unit 1) 0.296°F/% power (1) (Unit 2) 0.306°F/% power	0.21°F/% power
6.	Lag time constant (TM-505C)	28 seconds (4)	

**5173

3. for full load T_{avg}* = 576.6°F (Unit 1)
   = 577.6°F (Unit 2)

fouled SG tubes)

3a. high limit — 59.8% of level span[4] (Unit 1); 61.1% of level span[4] (Unit 2)

3b. low limit — 22.3% of level span[4]
(program is linear from 547°F to full load T_{avg})

* See the Note in 1.A above.

B. Low-Low Level Heater Cutout (letdown line isolation) (LC-459C and LC-460C) — 17 percent of level span

C. Level Controller (LC-459D)

1. proportional gain — 7.95 (CALCULATED)
2. rate time constant — OFF[1]
3. reset time constant — 1540 seconds[1]

D. Hi Level Deviation Heaters On (LC-459E) — 5 percent of span above level program

5. <u>Feedwater Control</u>

A. Low T_{avg}, Reactor Trip Override for Feedwater Valve Closure (TC-412G, TC-422G, TC-432G, TC-442G) valve closure on low T_{avg} — 554°F[1]

B. Level Control

~~(LC-505)~~ 32

NOTE: FOR FEEDWATER CONTROL PARAMETERS, REFER TO:

DC 6010364-112 (UNIT-1)
DC 6010364-111 (UNIT-2)