WCAP-14819

"Pacific Gas and Electric Company Diablo Canyon Power Plant, Unit 1 3425 MWt Uprating Program Licensing Report."

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Pacific Gas and Electric Company
Diablo Canyon Power Plant, Unit 1
3425 MWt Uprating Program
Licensing Report

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LIST OF ACRONYMS AND ABBREVIATIONS

AFW	Auxiliary Feedwater System
ANS	American Nuclear Society
APC	Alternate Plugging Criteria
ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
ASW	Auxiliary Saltwater System
BIT	Boron Injection Tank
BOP	Balance of Plant
CCW	Component Cooling Water System
C_D	Discharge Coefficient
C&FS	Condensate and Feedwater System
CHG/SI	Charging/Safety Injection
COLR	Core Operating Limits Report
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
CVCS	Chemical and Volume Control System
DBE	Design Basis Earthquake
DECL	Double-Ended Cold Leg
DEHL	Double-Ended Hot Leg
DEPS	Double-Ended Pump Suction
DER	Double-Ended Rupture
DF	Decontamination Factor
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EAB	Exclusion Area Boundary
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFPM	Effective Full Power Months
EFPY	Effective Full Power Years

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

EOP	Emergency Operating Procedure
ERG	Emergency Response Guidelines
ESF	Engineered Safety Features
ESFAS	Engineered Safety Feature Actuation System
ESW	Essential Service Water
$F\Delta H$	Hot Channel Enthalpy Rise Factor
F_Q	Total Peaking Factor
FHA	Fuel Handling Accident
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GPM	Gallons per Minute
HELB	High Energy Line Break
HFP	Hot Full Power
HHSI	High Head Safety Injection
HTC	Heat Transfer Coefficient
HZP	Hot Zero Power
IFBA	Integral Fuel Burnable Absorbers
IFM	Intermediate Flow Mixing
ISI	In-Service Inspection
ITDP	Improved Thermal Design Procedure
LB	Large Break
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LOL/TT	Loss of Load/Turbine Trip
LOOP	Loss of All AC Power to the Station Auxiliaries
LPZ	Low Population Zone
M/E or M&E	Mass and Energy
MCO	Moisture Carryover
MMF	Minimum Measured Flow
MSLB	Main Steam Line Break

LIST OF ACRONYMS AND ABBREVIATIONS (Continued)

MTC	Moderator Temperature Coefficient
MWt	Megawatt Thermal
NIS	Nuclear Instrumentation System
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OPΔT	Overpower Delta T
OTΔT	Overtemperature Delta T
PCT	Peak Clad Temperature
PG&E	Pacific Gas & Electric
PLOF	Partial Loss of Reactor Coolant Flow
PORV	Power Operated Relief Valve
PRT	Pressurizer Relief Tank
PTS	Pressurized Thermal Shock
PSSM	Power Shape Sensitivity Model
PSV	Pressurizer Safety Valve
PWR	Pressurized Water Reactor
RC	Reactor Coolant
RCCA	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RSE	Reload Safety Evaluation
RSG	Replacement Steam Generator
RSR	Relative Stability Ratio
RTDP	Revised Thermal Design Procedure
RT_{NDT}	Reference Temperature for Nil-Ductility Transition

SUMMARY AND CONCLUSIONS

PROGRAM SUMMARY

The purpose of this document is to provide the safety analysis and evaluation results to support operation of Diablo Canyon Power Plant Unit 1 with an NSSS power level of 3425 MWt, which incorporates 3411 MWt of core power and 14 MWt of reactor coolant pump heat. The analyses and evaluations were performed using the NSSS design parameters which feature a reactor vessel outlet temperature of 610.1°F, a steam generator tube plugging (SGTP) level up to 15%, and a Thermal Design Flow of 87,700 gpm/loop.

The parameters for the increase in NSSS power level and the additional margins listed above will be referred to throughout this report as the "Uprating Program". This report provides the necessary documentation to support the Technical Specification changes associated with the Uprating Program. The topics addressed in this report are as follows:

- Description of License Amendment
- Basis for Evaluations/Analyses Performed
- Loss of Coolant Accident Analyses
- Post-LOCA Hydrogen Production
- Post-LOCA Hot Leg Switchover
- LOCA Hydraulic Forces
- Non-LOCA Analyses
- Containment Integrity Analyses
- Steam Generator Tube Rupture Analyses
- Radiological Dose Analysis
- NSSS Primary Components Evaluations
- NSSS Fluid and Auxiliary Systems Evaluations
- Fuel Structural Evaluation

A brief summary of the results of each analysis and evaluation is provided below.

ACCIDENT ANALYSIS CONCLUSIONS

The results of the accident analyses and evaluations performed for the Uprating Program demonstrate that continued compliance with all industry and regulatory requirements is maintained for Diablo Canyon Power Plant Unit 1. The bases for the evaluations and analyses performed are provided in Section 3.0.

Large Break LOCA (Section 3.1.1)

The current UFSAR large break LOCA analyses were performed using the NRC-approved 1981 ECCS Evaluation Model with BASH.

A bounding Best Estimate LBLOCA analysis has been performed using the Westinghouse BELOCA Evaluation Model with WCOBRA/TRAC. This analysis was performed to bound conditions and parameters associated with operation of 24-month fuel cycles for Diablo Canyon Power Plant Units 1 and 2, including increased peaking factors and an uprating of Unit 1 to a core power of 3411 MWt.

Small Break LOCA (Section 3.1.2)

The small break LOCA analysis was performed with the Westinghouse small break LOCA ECCS Evaluation Model using the NOTRUMP code.

LOCA Hydraulic Forcing Functions (Section 3.1.3)

The Diablo Canyon Power Plant Unit 1 LOCA hydraulic forces were evaluated and it was determined that the RCS parameters used in the existing analysis-of-record bound the conditions for the Uprating Program. Therefore, the existing LOCA forces analyses remain bounding relative to the Uprating Program.

Non-LOCA Analyses (Section 3.2)

The non-LOCA events were addressed by evaluations and analyses for the impact of the increased NSSS power level. The computer codes and methods used for the non-LOCA analyses have been previously approved by the NRC. The non-LOCA safety analyses were reviewed on the basis of both DNB and non-DNB acceptance criteria. All DNB event reanalyses were found to yield a minimum DNBR which remains above the limit value. The analyses demonstrate that all licensing basis criteria continue to be met and the conclusions presented in the FSAR remain valid.

Containment Integrity Analyses (Section 3.3)

The containment integrity analyses have been evaluated for the impact of the increased power level of Diablo Canyon Unit 1 to 3411 MWt core power for the design basis LOCA and MSLB transients. Since the current design basis analyses of record for both LOCA and MSLB analyses were performed at the current power level of Diablo Canyon Unit 2 of 3411 MWt core power, these analysis results are bounding for the uprating of Diablo Canyon Unit 1 to the power level equivalent with Unit 2 (3411 MWt). The General Design Criteria (GDC) of 16, 38, and 50, and 10 CFR 50 Appendix K are satisfied for Diablo Canyon Unit 1.

<u>Steam Generator Tube Rupture</u> **(Section 3.4)**

The SGTR event was analyzed for the impact of the Uprating Program parameters. The SGTR analysis was not impacted by any of the Uprating Program revised operating conditions. A review of the SGTR analysis to support an uprating to 3425 MWt showed that the thermal-hydraulic results remain applicable for the uprating power conditions. Therefore, the conclusions of the FSAR remain valid.

<u>Radiological Doses</u> **(Section 3.5)**

The existing SGTR Accident analysis remains bounding for the Power Uprating Program conditions.

<u>Post-LOCA Hydrogen Generation</u> **(Section 3.6)**

The post-LOCA hydrogen generation was re-analyzed for the Power Uprating Program and acceptable results were obtained.

FLUID AND AUXILIARY SYSTEMS EVALUATION CONCLUSIONS (Section 4.0)

The fluid systems proof of design calculations were reviewed for the Uprating Program conditions. This review demonstrated that the NSSS fluid systems will continue to function as designed for all conditions of the Uprating Program.

In the NSSS/BOP interface area, the results of the evaluation show that the Uprating Program will have no adverse effects on the BOP systems performance (Main Steam System, Condensate and Feedwater System, Steam Dump System, Auxiliary Feedwater System, Steam Generator Blowdown and Sampling System). They will continue to perform acceptably at the conditions associated with Uprating Program.

PRIMARY COMPONENTS EVALUATION CONCLUSIONS (Section 5.0)

<u>Steam Generators</u> **(Section 5.1)**

Structural analyses and evaluations performed for the Diablo Canyon Power Plant Unit 1 steam generators indicate that the steam generator components remain in compliance with the applicable ASME Code requirements under the Uprating Program conditions.

<u>Pressurizer</u> **(Section 5.2)**

A fatigue analysis performed for the Diablo Canyon Power Plant Unit 1 pressurizer, incorporating the most conservative conditions of the Uprating Program, demonstrated that the pressurizer remains in compliance with the applicable ASME Code criteria.

Reactor Vessel (Section 5.3)

The results of the structural evaluations performed for the reactor vessel demonstrate that operation of Diablo Canyon Power Plant Unit 1 at the uprated conditions does not result in stress intensities or fatigue usage factors which exceed the acceptance criteria of the applicable ASME Code versions. The changes in the neutron fluence resulting from the uprated power level were evaluated for the impact on reactor vessel integrity, and determined to remain in compliance with all criteria. The assessment included a review of the upper shelf energy values, current material surveillance capsule withdrawal schedule, heatup and cooldown pressure-temperature limit curves, RT_{PTS} values, and the Emergency Response Guideline (ERG) limits.

Reactor Vessel Internals (Section 5.4)

Results of the thermal-hydraulic analyses performed for the reactor internals indicate that the Uprating Program for Diablo Canyon Power Plant Unit 1 results in continued compliance with established limits and criteria for core bypass flow, pressure drops, component lift forces, and momentum flux values. It was also confirmed that the currently assumed RCCA scram performance remains bounding for the Uprating Program conditions.

From the component stress and fatigue analysis and the flow induced vibration evaluations, it is concluded that the reactor internals remain in compliance with all applicable industry and regulatory requirements at the uprated conditions.

Control Rod Drive Mechanisms (Section 5.5)

The conclusion of structural evaluations performed for the Uprating Program conditions for the CRDMs demonstrate that the operability, service life, and structural integrity of the CRDM latch assembly, drive rod, and coil stack will remain in compliance with all Diablo Canyon Unit 1 applicable industry and regulatory requirements.

Reactor Coolant Pumps and Motors (Section 5.6)

The review performed of the reactor coolant pumps and motors for the Uprating Program conditions demonstrate that the conditions are acceptable for the RCP, and no additional thermal or structural analyses are required to demonstrate compliance with the applicable codes and standards. The RCP motor evaluation revealed that the motors are acceptable for operation at the Uprating Program conditions.

Reactor Coolant Loop Piping and Supports (Section 5.7)

Analyses were reviewed to determine the effects of the Uprating Program conditions on the primary loop piping, primary equipment supports, and the primary equipment nozzles. The analyses demonstrated continued compliance with all Diablo Canyon Unit 1 industry and regulatory requirements and therefore, the adequacy of these components at the Uprating Program conditions.

It was also determined that the Uprating Program conditions will have an insignificant impact on the design basis analysis for the NRC Bulletin 88-08 evaluation of the auxiliary spray piping and the NRC Bulletin 88-11 evaluation of the pressurizer surge line piping.

Auxiliary Components (Section 5.8)

Evaluations were performed for the auxiliary tanks, pumps, valves, and heat exchangers to determine the effects of the revised RCS parameters due to the Uprating Program. The results of these evaluations demonstrated that the Uprating Program parameters will not adversely affect the function or structural integrity of this equipment.

Fuel Evaluation (Section 6.0)

Evaluations were performed for the Diablo Canyon Power Plant Unit 1 Uprating Program in the areas of fuel rod and fuel assembly and structural integrity.

The fuel assembly structural integrity is not affected by the Uprating Program, and the core coolable geometry is maintained for the fuel in the Unit 1 core. The evaluation of the fuel rod structural integrity indicates these conditions will be acceptable.

The results of the core design evaluation and the thermal-hydraulic analyses will be provided by Westinghouse.

Balance of Plant (Section 7.0)

The secondary side of Diablo Canyon Units 1 and 2 are virtually identical. More than ten years of operation of Unit 2 at 3411 MWt demonstrates that Unit 1 can also be safely and reliably operated at this power level. Differences between the two units are limited to different electric generator cooling systems and different predicted steam quality. The Unit 1 electric generator was designed for 105% power, hence its cooling system envelopes this 2.2% uprate, though generator margins will be less for Unit 1 than Unit 2. The steam conditions are predicted to be different because Unit 1 has a lower RCS thermal design flow and could potentially have lower steam pressure. In practice, however, both units have similar RCS flow rates and at the current steam generator tube plugging levels, both units have almost identical predicted steam conditions. Therefore, the ability of the balance of plant to adequately support a core power of 3411 MWt is demonstrated by the years of successful Unit 2 operation.

Turbine Generator Systems & Components Review (Section 8.0)

Evaluations and analyses were performed to predict the performance of the Turbine Generator systems and components at the uprated conditions. It was concluded that the uprating conditions are acceptable with no component replacement required. Turbine missile analyses demonstrated that the original missile calculations remain valid and bounding.

Environmental and Permit Evaluation (Section 9.0)

The uprating of Unit 1 will not require any modification to the Environmental Protection Plan; however, the margin to allowable thermal discharge will be decreased. The temperature of the discharged circulating water must be no higher than 22°F above the temperature of the intake. This temperature differential has never been reached at DCPP. The uprating will increase the temperature differential by about 0.2°F. It should be noted that this is a monitored parameter, and if any violation of the NPDES permit limit is threatened, plant operators can take actions (such as decreasing power) to avoid a violation. Thus, the uprating is consistent with the Environment Protection Plan and all relevant permits.

Security Plan and Emergency Plan (Section 10.0)

The operation of Unit 1 at the existing Unit 2 power level does not involve any additional personnel, significant procedure changes, or impacts on potential radiological releases. Therefore, the Security and Emergency Plans are not impacted by the uprate.

1.0 INTRODUCTION

1.1 LICENSING PERSPECTIVE

Currently, the licensing basis analyses for Diablo Canyon Power Plant Unit 1 are documented in the Updated Final Safety Analysis Report (UFSAR). This amendment request reflects the changes to the safety analysis assumptions and results due to the revised operating conditions resulting from an increased NSSS power level. The primary purpose of the uprating program was to demonstrate continued compliance with all industry and regulatory standards and requirements applicable to the Diablo Canyon Unit 1 NSSS with the thermal power level of the unit increased from the originally licensed core power level of 3338 MWt to the current Unit 2 licensed core power level of 3411 MWt.

The parameters associated with an increased NSSS power of 3425 MWt, both directly and indirectly, are referred to throughout this report as the "Uprating Program".

1.2 PURPOSE AND OBJECTIVES

The purpose of the Uprating Program was to perform the necessary NSSS-related efforts to support an increase in the NSSS power level to 3425 MWt.

1.3 SCOPE SUMMARY

The Westinghouse scope for the Diablo Canyon Power Plant Unit 1 Uprating Program is to perform the NSSS systems, safety, and components analyses and evaluations which demonstrate continued compliance with all Diablo Canyon Unit 1 applicable industry and regulatory standards and requirements to support operation over the range of uprated conditions identified in this report.

PG&E as licensee has the final responsibility for the technical accuracy of this document, hence PG&E has worked with Westinghouse to develop the plant specific input data, perform an independent review of the Westinghouse results, and documentation, and provide analysis in the specific areas of Unit 1 to Unit 2 differences, Environmental Qualification, Dose Assessment, Primary Water Chemistry, Cooling Water Systems, Radioactive Waste, Post Accident Sampling, Ventilation, Reactor Vessel, Balance of Plant, Environmental Protection and Permits, and Security and Emergency Planning.

2.0 BASIS FOR EVALUATIONS/ANALYSES PERFORMED

The purpose of the Uprating Program was to perform the necessary NSSS-related efforts to support an increase in the NSSS power level to 3425 MWt and continue operational flexibility in terms of primary temperature and pressure.

2.1 NSSS PERFORMANCE PARAMETERS

This section describes the parameters which were used as the basis for the evaluations and analyses performed to support the Uprating Program for Diablo Canyon Power Plant Unit 1. The NSSS design parameters incorporate an NSSS power of 3425 MWt, a reactor vessel outlet temperature (T_{HOT}) of 610.1°F, an average steam generator tube plugging (SGTP) level up to 15%, and a Thermal Design Flow of 87,700 gpm per loop. (It should be noted that the Unit 2 Thermal Design Flow is 88,500 gpm per loop. This difference in flow is alluded to in later sections of this report.)

A brief description of each set of parameters is provided below:

Case 1: These are the nominal NSSS performance parameters for Unit 1 prior to the uprating; they are shown for comparison with the revised uprated parameters. These parameters incorporate an NSSS power of 3350 MWt and 0% SGTP.

Case 2: These parameters incorporate the uprated core power level of 3411 MWt, an NSSS power level of 3425 MWt including 14 MWt of reactor coolant pump heat, a steam generator tube plugging level of 0%, Thermal Design Flow of 87,700 gpm per loop, and a reactor vessel outlet temperature of 610.1°F.

Case 3: These parameters incorporate the same primary system parameters as Case 2, with a steam generator tube plugging level of 15%. The increased plugging level results in corresponding reductions in steam temperature, steam pressure, and steam flow.

The Uprating Program NSSS design parameters incorporate the current 17x17 VANTAGE 5 fuel, with a core bypass flow of 7.5%.

Table 2.1-1
Diablo Canyon Power Plant Unit 1 NSSS Performance Parameters for Uprating Program

Parameter	Case 1	Case 2	Case 3
NSSS Power, MWt	3,350	3,425	3,425
Core Power, MWt	3,338	3,411	3,411
RCS Flow, (gpm/loop)[1]	87,700	87,700	87,700
Minimum Measured Flow, (total gpm)[2]	359,200	359,200	359,200
RCS Temperatures, °F			
Core Outlet	613.5	614.8	614.8
Vessel Outlet	608.8	610.1	610.1
Core Average	580.7	581.5	581.5
Vessel Average	576.6	577.3	577.3
Vessel/Core Inlet	544.4	544.5	544.5
Steam Generator Outlet	544.2	544.2	544.2
Zero Load	547.0	547.0	547.0
RCS Pressure, psia	2,250	2,250	2,250
Steam Temperature, °F	519	518.2	511.7
Steam Pressure, psia	805	800	756
Steam Flow, (106 lb/hr. tot.)	14.52	14.91	14.89
Feedwater Temperature, °F	432.1	435.0	435.0
% SG Tube Plugging	0	0	15

[1]RCS Flow (Thermal Design Flow) - The conservatively low flow used for thermal/hydraulic design. The design parameters listed above are based upon this flow.

[2]Minimum Measured Flow - The flow specified in the Technical Specifications which must be confirmed or exceeded by the flow measurements obtained during plant startup and is the flow used in reactor core DNB analyses for plants applying the Revised Thermal Design Procedure. MMF based upon 2.4% flow measurement uncertainty.

2.2 NSSS DESIGN TRANSIENTS

The design transients evaluation for the Diablo Canyon Power Plant Unit 1 Uprating Program was completed and confirmed that the NSSS design transients of record continue to apply to the Diablo Canyon Power Plant Unit 1 at the uprating conditions. The evaluation consisted of a comparison of the NSSS performance parameters for the Uprating Program with the parameters for the transients of record. The comparison concluded that the currently applicable NSSS design transients (i.e., temperatures, pressures, and power levels) remain applicable and bounding for the uprated conditions.

2.3 CONTROL/PROTECTION SYSTEM SETPOINTS

The uprated conditions were evaluated for impact on plant control systems and operability, in accordance with the Diablo Canyon Power Plant Unit 1 Power Uprating Program. Based on the following discussion, it is concluded that there is adequate margin to reactor trip setpoints and that no significant changes to control systems setpoints are required.

1. The Unit 1 uprating incorporates an increase in thermal power from 3350 MW to 3425 MW. The uprated full power design T_{HOT}, T_{COLD}, and T_{AVG} remain within 2°F of the original design values. The revised secondary steam temperature remains within 8°F of the original design value.

2. The Unit 1 uprating parameters and plant configuration are nearly identical to those previously analyzed for Unit 2. Therefore, Unit 2 operability analyses would be bounding for Unit 1.

3. With the following exceptions, the Unit 1 and Unit 2 control systems setpoints are identical. None of the below differences are expected to cause significant difficulty in implementing the uprating:

   3a. Coolant average temperature control system: There is a difference between the Unit 1 and Unit 2 maximum temperature (T_{AVG}) setpoints and control system gains.

   Unit 1: 576.6°F

   Unit 2: 577.6°F

   Temperature Gain Unit 1: 0.296°F/% power

   Temperature Gain Unit 2: 0.306°F/% power

   Due to concerns over steam generator tube plugging, it is currently planned that the T_{AVG} setpoint for Unit 1 will not be increased.

3b. The Steam Dump Control System proportional gain, in percent of available dump capacity per °F, is different in the two units:

Loss of Load Controller: Unit 1 is 4.17%/°F; Unit 2 is 5%/°F

There are several other differences in steam dump control system settings between Unit 1 and Unit 2. The control system lead, lag, deadband for reactor trip, and the control value for the loss of load controller valve trip open are different. Setpoints for the Unit 2 steam dump controller were adjusted during startup tests to enhance plant response during design basis load rejection and reactor trip transients. Based on the discussions in (1) above, it is judged that Unit 1 steam dump controller setpoints are not impacted by uprating. Therefore, no changes to the current control systems setpoints are recommended.

3c. The Pressurizer Level Control Program is slightly different in the two units:

Full Load T_{AVG} for Unit 1: 576.6°F

Full Load T_{AVG} for Unit 2: 577.6°F

High Level Limit Unit 1: 59.8% of span

High Level Limit Unit 2: 61.1% of span

These setpoints will be evaluated in the DCPP Design Change Process (DCP) for implementing the uprate.

3d. The digital feedwater flow control system in each unit has different tuning constants for feedwater control parameters, feedwater level controllers and feedwater pump speed controllers. The Unit 1 control parameters will be evaluated as part of the DCP to implement uprating.

3e. The constants for the control rod insertion alarms in the Insertion Limit Computers are different in the Rod Control Systems for the two units. Westinghouse recommends that the high limit of the Rod Control System (TC-505, TC-505A) for each unit be set equal to the full power T_{AVG}.

4. The Unit 1 protection systems setpoints are identical to those in Unit 2. No changes have been made to the reactor protection systems setpoints. Therefore, Unit 2 margin to trip analyses bound those of Unit 1.

As stated previously, it is concluded that based on the results of the evaluation of the plant control systems and operability, there is adequate margin to reactor trip setpoints and no significant changes to control systems setpoints are required for the Diablo Canyon Power Plant Unit 1 Uprating Program.

3.0 ACCIDENT ANALYSES AND EVALUATIONS

3.1 LOSS OF COOLANT ACCIDENT ANALYSES AND LOCA-RELATED EVENTS

3.1.1 Large Break LOCA

The current UFSAR Large Break (LB) LOCA analyses for Diablo Canyon Power Plant Units 1 and 2 were performed using the NRC-approved 1981 Emergency Core Cooling System (ECCS) Evaluation Model with BASH (Reference 1). These analyses resulted in limiting calculated peak cladding temperatures (PCTs) of 2042°F for Diablo Canyon Power Plant Unit 1 and 2071°F for Unit 2. Additional penalties/benefits for ECCS evaluation model changes and other safety evaluations resulted in current LBLOCA PCTs of 2023°F and 2155°F for Diablo Canyon Power Plant Units 1 and 2, respectively (Reference 2).

A bounding Best Estimate (BE) LBLOCA analysis has been performed for Diablo Canyon Power Plant Units 1 and 2 using the Westinghouse BELOCA Evaluation Model with WCOBRA/TRAC (Reference 3). This analysis was performed to bound conditions and parameters associated with operation of 24-month fuel cycles for Diablo Canyon Power Plant Units 1 and 2, including (increased peaking factors) and an uprating of Unit 1 to 3411 MWt. This bounding BELOCA analysis for Diablo Canyon Power Plant Units 1 and 2 has resulted in a PCT at 95 percent probability of 1976°F.

References:

1. WCAP-10266-P-A, Revision 2, with Addenda (Proprietary), Kabadi, J.N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

2. PGE-96-581, "Pacific Gas and Electric Company Nuclear Plant, Diablo Canyon Units 1 and 2, 10 CFR 50.46 Small Break LOCA Notification and Reporting," July, 1996.

3. Letter, R.C. Jones, Jr. (USNRC) to N.J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945 (P), Westinghouse Code Qualification Document for Best-Estimate Loss of Coolant Analysis," June 1996.

3.1.2 Small Break LOCA

3.1.2.1 Introduction

A small break Loss-of-Coolant Accident (LOCA) analysis was performed in support of the Unit 1 uprating program and to envelope the conditions for Diablo Canyon Units 1 and 2. The purpose of analyzing the small break LOCA is to demonstrate that conformance with the 10 CFR 50.46 (Reference 1) requirements for the conditions associated with the uprating program. Important input assumptions, as well as analytical models and analysis methodology for the small break LOCA, are contained in subsequent sections. Analysis results are provided in the form of tables and figures that provide a detailed description of the limiting transient. It was determined that no design or regulatory limit related to the small break LOCA would be exceeded due to the uprated power and new fuel and plant parameters.

3.1.2.2 Input Parameters and Assumptions

Important plant conditions and features for the SBLOCA analyses are listed in Table 3.1.2-1. Several additional considerations that are not identified in Table 3.1.2-1 are discussed below:

Figure 3.1.2-1 depicts the hot rod axial power shape modeled in the small break LOCA analysis. This shape was chosen because it represents a distribution with power concentrated in the upper regions of the core (the axial offset is + 20%). Such a distribution is limiting for small break LOCA since it minimizes coolant swell while maximizing vapor superheating and fuel rod heat generation at the uncovered elevations. The power shape was conservatively scaled to a flat K(Z) envelope based on the peaking factors given in Table 3.1.2-1

This analysis assumes SI (Safety Injection) flow from one High Head SI (HHSI) pump and one Intermediate Head SI (IHSI) pump. At DCPP, the HHSI and IHSI pumps are referred to as Charging Pumps and Safety Injection Pumps, respectively. This corresponds to the limiting SBLOCA conditions for Diablo Canyon Unit 1. Figure 3.1.2-2 provides the degraded HHSI and IHSI pump flows versus pressure curve modeled in the small break LOCA analysis.

3.1.2.3 Description of Analyses/Evaluations Performed

Analytical Model

For small breaks, the NOTRUMP computer code (References 2 and 3) is used to calculate the transient depressurization of the reactor coolant system (RCS) and to determine the mass and energy release of the fluid flow through the break. The NOTRUMP computer code is a one-dimensional general network code incorporating a number of advanced features, including: calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, regime-dependent drift flux calculations 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 LOCA's, and to address NRC concerns expressed in NUREG-0611 (Reference 4).

The RCS model is nodalized into volumes interconnected by flow paths. The broken loop is modeled explicitly, while the intact loops are lumped together into a second loop. Transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum. The multi-node capability of the program enables explicit, detailed spatial representation of various system components which, among other capabilities, enables a proper calculation of the behavior of the loop seal during a small break LOCA. The reactor core is represented as heated control volumes with associated phase separation models to permit transient mixture height calculations.

Fuel cladding thermal analyses are performed with a version of the LOCTA-IV code (Reference 5) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions (see Figure 3.1.2-3). The version of the LOCTA-IV code used in this analysis also has the capability to explicitly model annular fuel pellets used in the annular pellet blanket feature of core reload designs.

The bounding small break LOCA analysis performed for the Diablo Canyon Units 1 and 2 uprating/fuel program utilized the NRC-approved NOTRUMP Evaluation Model (References 2 and 3). This model includes the pumped SI and accumulator injection into both the broken and intact loops. It also includes an improved condensation model (COSI) for the pumped SI into both the broken and intact loops (Reference 6). This improved condensation model incorporates the restrictions dictated by the NRC in the related Safety Evaluation Report (Reference 7).

Analysis

The most limiting single active failure is that of an SSPS train failure which results in the loss of one complete train of ECCS components. In addition, a Loss-of-Offsite Power (LOOP) is assumed to occur coincident with reactor trip. This means that credit may be taken for at most one HHSI pump, one IHSI pump, and one low head, or residual heat removal (RHR) pump. However, in the analysis of the small break LOCA presented here, only the minimum delivered ECCS flow from the HHSI and IHSI pumps with degraded flow was assumed.

The small break LOCA analysis performed for the Diablo Canyon uprating/fuel program assumes SI is delivered to both the intact and broken loops at the RCS backpressure. The results of Reference 8 demonstrate that the cold leg break location is limiting with respect to postulated cold leg, hot leg and pump suction leg break locations.

Prior to break initiation, the plant is assumed to be at full thermal power (102%) equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. Other initial plant conditions assumed in the analysis are given in Table 3.1.2-1. Subsequent to the break, a period of reactor coolant system blowdown ensues in which the heat from fission product decay, the hot reactor internals, and the reactor vessel continues to be transferred to the RCS fluid. The heat transfer between the RCS and the secondary system may be in either the positive or negative direction and is a function of the relative temperatures of the primary and secondary. In the case of continuous heat addition to the secondary during a period of quasi-equilibrium, an increase in the secondary system pressure results in steam relief via the steam generator safety valves, which were modeled with 3 percent accumulation and 3 percent tolerance.

Should a small break LOCA occur, depressurization of the RCS causes fluid to flow into the RCS loop from the pressurizer resulting in a pressure and level decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure reactor trip setpoint, conservatively modeled as 1860 psia, is reached. LOOP is assumed to occur coincident with reactor trip. A safety injection signal is generated when the pressurizer low-pressure safety injection setpoint, conservatively modeled as 1695 psia, is reached. Safety injection is conservatively assumed to be delayed 27 seconds after the occurrence of the low pressure condition. This delay accounts for signal initiation, diesel generator start up and emergency power bus loading consistent with the assumed loss of offsite power coincident with reactor trip, as well as the time involved in aligning the valves and bringing the SI pumps up to full speed. These countermeasures limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection supplement void formation in causing a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay. No credit is taken in the LOCA analysis for the boron content of the injection water. Credit is taken in the small break LOCA analysis for the insertion of Rod Cluster Control Assemblies (RCCAs) subsequent to the reactor trip signal, while assuming the most reactive RCCA is stuck in the full out position. A rod drop time of 2.7 seconds was assumed with an additional 2 seconds for the signal processing delay time. Therefore, a total delay time of 4.7 seconds from the time of reactor trip signal to full rod insertion was assumed in this small break LOCA analysis.

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive cladding temperatures.

During the earlier part of the small break transient (prior to the assumed LOOP coincident with reactor trip), the loss of flow through the break is not sufficient enough to overcome the positive core flow maintained by the reactor coolant pumps. During this period, upward flow through the core is maintained. However, following the reactor coolant pump trip (due to a LOOP) and subsequent pump coastdown, a period of partial core uncovery occurs. Ultimately, the small break transient analysis is terminated when the ECCS flow provided to the RCS exceeds the break flow rate.

The core heat removal mechanisms associated with the small break transient include not only the break itself and the injected ECCS water, but also that heat transferred from the RCS to the steam generator secondary side. Main Feedwater (MFW) is assumed to be isolated coincident with the safety injection signal, and the MFW pumps coast down to 0% flow in 9 seconds (2 second delay + 7 second coastdown). With an AFW actuation signal, a continuous supply of makeup water is provided to the secondary using the auxiliary feedwater (AFW) system. An AFW actuation signal occurs coincident with the safety injection signal. Full AFW system flow is assumed 60 seconds following the signal. The heat transferred to the secondary side of the steam generator aids in the reduction of the RCS pressure.

Should the RCS depressurize to approximately 579 psig, as in the case of the limiting 3-inch and the 4-inch breaks, the cold leg accumulators begin to inject borated water into the reactor coolant loops. In the case of the 2-inch break however, the vessel mixture level is recovered without the aid of accumulator injection.

3.1.2.4 Acceptance Criteria for Analyses

The Acceptance Criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

1. The calculated maximum fuel element cladding temperature shall not exceed 2200°F,

2. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation,

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react,

4. Calculated changes in core geometry shall be such that the core remains amenable to cooling,

5. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Criteria 1 through 3 are explicitly covered by the subject small break LOCA analysis.

For criterion 4, the appropriate core geometry was modeled in the analysis. The results based on this geometry satisfy the Peak Cladding Temperature (PCT) criterion of 10 CFR 50.46 and consequently, demonstrate the core remains amenable to cooling.

For criterion 5, Long-Term Core Cooling (LTCC) considerations are not directly applicable to the small break LOCA transient, but are assessed elsewhere as part of the evaluation of ECCS performance.

The criteria were established to provide a significant margin in emergency core cooling system (ECCS) performance following a LOCA.

3.1.2.5 Results

Based on the similarity between the Diablo Canyon Units and the slightly higher power rating of Unit 2, the previous small break LOCA analysis for Diablo Canyon analyzed a full spectrum of break sizes for Unit 2 only. Since the current Unit 1 Uprate Program analysis considers the same rated power as for Unit 2 (3411 MWt), and since neither Unit could be established as limiting, a full spectrum breaks sizes was analyzed for each Unit. First the limiting break size was established for each Unit using High T_{AVG} conditions (see discussion below on limiting RCS temperature conditions). Then a sensitivity run was made to verify the non-limiting condition of Low T_{AVG} using the limiting High T_{AVG} break size.

Limiting Break Size

Studies documented in Reference 8 have determined that the limiting small-break transient occurs for breaks with areas of less than 1 ft^2. Based on previous Diablo Canyon small break LOCA analyses, the limiting break was expected to be either a 3-inch or 4-inch diameter break. In order to establish a break size as limiting, incrementally smaller and larger break sizes must be shown to be non-limiting.

The limiting break for both Diablo Canyon Units was found to be a 3-inch diameter cold leg break. This is a change from the previous small break LOCA analysis that found the 4-inch break to be limiting. The shift in limiting break size is not considered significant since that in both the previous and the current small break LOCA analyses, the 3-inch and 4-inch calculated PCT results were similar. Since in this new analysis, the 3-inch break was found to be more limiting than the 4-inch break, a 2-inch break was analyzed to demonstrate that an incrementally smaller break is non-limiting.

The High T_{AVG} 3-inch break PCT's are 1304°F and 1293°F for Units 1 and 2, respectively. The difference in PCT's between Unit 1 and Unit 2 is not significant and is attributed to slight variances in transient behavior. The previous SBLOCA PCT's were 1295°F for Unit 1 and 1356°F for Unit 2.

Limiting RCS Temperature Conditions

Reduced operating temperature typically results in a PCT benefit for the small break LOCA. However, due to competing effects and the complex nature of small break LOCA transients, there have been some instances where more limiting results have been observed for the reduced operating temperature case. For this reason, a study of the small break LOCA transient based on a lower bound RCS T_{AVG} was performed. As expected, the Low T_{AVG} cases produced non-limiting PCT's for both Units.

The High T_{AVG} and Low T_{AVG} values were established to bound an RCS temperature window. This temperature window was based on a nominal vessel average temperature of 572.0°F, +10.3°F, -12.0°F to bound a plant operating range and uncertainties.

Annular Fuel Pellets

The Diablo Canyon uprating program included the introduction of a reload with fully enriched annular fuel pellet blankets at the top and bottom of the core. In this small break LOCA analysis, the annular pellets were explicitly modeled in the cladding heat-up calculations. In order to determine the effect of annular pellet blankets on PCT, fuel cladding heat-up calculations were also performed for non-annular pellet cases. For the limiting break size, the difference on PCT between annular pellet blankets and non-annular pellets blankets was less than ±1°F for both Units.

Results Summary

A summary of the fuel cladding related results is provided in Table 3.1.2-2. A summary of the key small break LOCA transient event times is provided in Table 3.1.2-3.

For each break case Table 3.1.2-3 lists the time at which the small break LOCA transient is considered over. The transient is considered over when there is no longer a concern of violating the 10 CFR 50.46 criteria as described in Section 3.1.2.4 as evidenced by a number of the following conditions:

1. The RCS pressure is gradually decreasing
2. Pumped SI flow exceeds total break flow
3. The core is covered or the core mixture level in increasing
4. Fuel cladding temperatures are decreasing

Plots of the transient response for the limiting High T_{AVG} 3-inch break case for both Units are given in Figures 3.1.2-4 through 3.1.2-12. These figures present the response of the following parameters:

- RCS Pressure Transient,
- Core Mixture Level,
- Peak Cladding Temperature,
- Top Core Node Vapor Temperature,
- Safety Injection Mass Flow Rate for the Intact and Broken Loops,
- Cold Leg Break Mass Flow Rate,
- Hot Spot Rod Surface Heat Transfer Coefficient, and
- Hot Spot Fluid Temperature.

Plots of the transient response for the non-limiting break cases for both Units are shown in Figures 3.1.2-13 through 3.1.2-21. These figures present the response of the following parameters:

1. RCS Pressure Transient,
2. Core Mixture Level, and
3. Peak Cladding Temperature.

3.1.2.6 Conclusions

For each Diablo Canyon Unit, small break LOCA transients were calculated for a break size spectrum using High T_{AVG} RCS conditions. The 3-inch equivalent diameter break was found to be limiting for both Units with calculated peak cladding temperatures of 1304°F and 1293°F for Unit 1 and Unit 2 respectively. Low T_{AVG} RCS conditions were shown to be non-limiting for both Units.

The analyses presented in this section show that the safety injection subsystems of the Emergency Core Cooling System, together with the heat removal capability of the steam generator, provide sufficient core heat removal capability to maintain the calculated peak cladding temperatures below the required limit of 10 CFR 50.46 which is defined in Section 3.1.2.4.

Hence, adequate protection is afforded by the emergency core cooling system in the event of a small break Loss-of-Coolant Accident.

3.1.2.7 References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 1974, as amended in Federal Register, Volume 53, September 1988.

2. Meyer, P. E., "NOTRUMP - A Nodal Transient Small Break and General Network Code," WCAP-10079-P-A, (proprietary) and WCAP-10080-NP-A (non-proprietary), August 1985.

3. Lee, N. et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (proprietary) and WCAP-10081-NP-A (nonproprietary), August 1985.

4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse - Designed Operating Plant," NUREG-0611, January 1980.

5. Bordelon, F. M. et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8301 (proprietary) and WCAP-8305 (non-proprietary), June 1974.

6. Thompson, C. M. et al., "Addendum to the Westinghouse Small Break LOCA Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and the COSI Condensation Model", WCAP-10054-P, Addendum 2 (proprietary) and WCAP-10081-NP (non-proprietary), August 1994.

7. NRC letter to N. J. Liparulo, WCAP-10054-P, Addendum 2, Revision 1 "NOTRUMP SBLOCA Using COSI Steam Condensation Model," (TAC No. M90784), August 12, 1996.

8. Rupprecht, S. D. et al., Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code", WCAP-11145-P-A (proprietary), October 1986.

Table 3.1.2-1
Input Parameters Used in the Small Break LOCA Analysis
DCPP Unit 1 / DCPP Unit 2

Parameter	High T_{AVG} (Low T_{AVG})
Reactor core rated thermal power[1], (MWt)	3411
Peak linear power[1,2], (kW/ft)	15.00
Total peaking factor (F_Q^T) at peak[2]	2.70
Power shape[2]	Figure 3.1.2-1
$F_{\Delta H}$	1.70
Fuel[3]	17x17V5
Accumulator water volume, nominal (ft^3/acc.)	850
Accumulator tank volume, nominal (ft^3/acc.)	1350
Accumulator gas pressure, minimum (psia)	594
Pumped safety injection flow	Figure 3.1.2-2
Steam generator tube plugging level (%)[4]	15
Thermal Design Flow/loop, (gpm)	85,000
Vessel average temperature w/uncertainties, (°F)	582.3 (560.0)
Reactor coolant pressure w/uncertainties, (psia)	2310
Aux. feedwater flow rate/SG, (gpm)	205

1. Two percent is added to this power to account for calorimetric error. Reactor coolant pump heat is not modeled in the small break LOCA analyses.
2. This represents a power shape corresponding to a one-line segment peaking factor envelope, K(z), based on $F_Q^T = 2.70$.
3. Annular pellet blankets were explicitly modeled.
4. Maximum plugging level in any one or all steam generators.

Table 3.1.2-2
Small Break LOCA Analysis Fuel Cladding Results

DCPP Unit 1/ DCPP Unit 2

Unit I Break Cases

	High T_{AVG} 2-inch	High T_{AVG} 3-inch	High T_{AVG} 4-inch	Low T_{AVG} 3-inch
Peak Cladding Temp (°F)	956	1304	1264	1191
Peak Cladding Temp Location (ft)	10.75[1]	11.25[1]	11.00[1]	11.25[1]
Peak Cladding Temp Time (sec)	4250	1852	928	1937
Local Zr/$H_2$0 Reaction, Max (%)	0.03	0.20	0.09	0.11
Local Zr/$H_2$0 Reaction Location(ft)	11.00[1]	11.25[1]	11.00[1]	11.25[1]
Total Zr/$H_2$0 Reaction (%)	< 1.0	< 1.0	< 1.0	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A	N/A

Unit 2 Break Cases

	High T_{AVG} 2-inch	High T_{AVG} 3-inch	High T_{AVG} 4-inch	Low T_{AVG} 3-inch
Peak Cladding Temp (°F)	955	1293	1225	1151
Peak Cladding Temp Location (ft)	11.00[1]	11.25[1]	11.00[1]	11.25[1]
Peak Cladding Temp Time (sec)	4371	1948	937	2005
Local Zr/$H_2$0 Reaction, Max (%)	0.03	0.25	0.07	0.10
Local Zr/$H_2$0 Reaction Location(ft)	11.00[1]	11.25[1]	11.00[1]	11.25[1]
Total Zr/$H_2$0 Reaction (%)	< 1.0	< 1.0	< 1.0	< 1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A	N/A

1. From bottom of active fuel

Table 3.1.2-3
Small Break LOCA Analysis Time Sequence of Events
DCPP Unit 1/ DCPP Unit 2

Unit I Break Cases

	High T_{AVG} 2-inch (sec)	High T_{AVG} 3-inch (sec)	High T_{AVG} 4-inch (sec)	Low T_{AVG} 3-inch (sec)
Break Occurs	0.0	0.0	0.0	0.0
Reactor Trip Signal	48.7	19.6	11.1	16.8
Safety Injection Signal	60.7	28.2	18.6	24.3
Top of Core Uncovered	1781	995	605	1121
Accumulator Injection Begins	N/A[1]	1845	852	2290
Peak Clad Temperature Occurs	4250	1852	928	1937
Top of Core Covered	N/A[2]	3160	1571	3543

Unit 2 Break Cases

	High T_{AVG} 2-inch (sec)	High T_{AVG} 3-inch (sec)	High T_{AVG} 4-inch (sec)	Low T_{AVG} 3-inch (sec)
Break Occurs	0.0	0.0	0.0	0.0
Reactor Trip Signal	49.2	19.5	11.1	16.8
Safety Injection Signal	61.2	28.2	18.5	24.3
Top of Core Uncovered	1750	1066	607	1070
Accumulator Injection Begins	N/A[1]	2250	857	2310
Peak Clad Temperature Occurs	4371	1948	937	2005
Top of Core Covered	N/A[2]	3176	1628	3520

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

3.1.3 LOCA Hydraulic Forces

A LOCA forces evaluation was performed for the Diablo Canyon Power Plant Unit 1 Uprating Program. The Uprating Program and the associated performance parameters, have been reviewed relative to any effect on the reactor vessel and internals LOCA forcing functions. It has been concluded that the uprating conditions have a negligible effect on the reactor vessel, and internals and fuel LOCA forces. Thus, previously applicable LOCA forcing functions remain valid. It should be noted that the current LOCA forcing functions were based on limited displacement primary loop piping breaks and therefore, no reliance on leak-before-break licensing is required.

In addition to the effect on the reactor vessel and internals LOCA forcing functions, the LOCA forces evaluation performed for the Diablo Canyon Power Plant Unit 1 Uprating Program assessed the effect on steam generator and loop LOCA forces. The uprating has a negligible effect on LOCA steam generator and loop forces. Therefore, previously applicable LOCA forcing functions remain bounding.

3.1.4 Hot Leg Switchover

The Hot Leg Switchover (HLSO) analysis of record was evaluated to determine the effect of the Diablo Canyon Power Plant Unit 1 Uprating Program. The HLSO analysis of record was performed at the Diablo Canyon Unit 2 power level of 3411 MWt, and conservatively bounded the lower Unit 1 power level of 3338 MWt. Therefore, the HLSO analysis of record supports the Unit 1 uprating power level of 3411 MWt.

3.1.5 Post-LOCA Long Term Cooling

It is also necessary for the Long Term Core Cooling (LTCC) analysis to support the Diablo Canyon Power Plant Unit 1 Uprating Program. Core power is not a direct input to the LTCC analysis. If the boron sources are affected by the uprating, the LTCC calculation will be affected. This calculation is performed on a cycle-specific basis and will be reviewed at the time of the RSAC generation.

3.2 NON-LOCA SAFETY ANALYSES

This section summarizes the evaluation of the proposed uprating of Diablo Canyon Unit 1 to the same core power as Unit 2 (3411 MWt) relative to the non-LOCA safety analyses.

3.2.1 Evaluation of Events

The majority of the currently applicable non-LOCA safety analyses for the Diablo Canyon plant are performed using bounding assumptions for important plant parameters that envelope both of the two units. The following Updated FSAR non-LOCA events are analyzed at no-load conditions or do not directly assume the specific core power and are thus not affected by an uprated full power condition:

15.2.1 Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition

15.2.4 Uncontrolled Boron Dilution

15.2.10 Excessive Heat Removal due to Feedwater System Malfunctions (zero power cases)

15.2.14 Accidental Depressurization of the Main Steam System

15.4.2.1 Rupture of a Main Steam Line

15.4.6 RCCA Ejection (zero power cases)

The following at-power safety analyses currently assume the lower design RCS flow rates associated with Unit 1 in combination with the higher licensed core and NSSS power and coolant average temperature of Unit 2. For these events the currently applicable analyses remain bounding for the proposed Unit 1 uprated power. It should be noted that several of the analyses assume the previous Unit 2 NSSS power of 3423 MWt. The nominal NSSS power for both Diablo Canyon units will now be 3425 MWt, based on a revised calculation of net reactor coolant pump heat input (minimum 14 MWt instead of assumed value of 12 MWt). This 2 MWt increase in NSSS power is very small and has been evaluated to have a negligible effect on the results of the affected safety analyses.

15.2.2 Uncontrolled RCCA Bank Withdrawal at Power

15.2.3 RCCA Misoperation

15.2.5 Partial Loss of Forced Reactor Coolant Flow

15.2.6 Startup of an Inactive Reactor Coolant Loop

15.2.7 Loss of External Electrical Load/Turbine Trip (DNB analysis; Overpressure analysis addressed in Section 7.15)

15.2.8 Loss of Normal Feedwater

15.2.9 Loss of Offsite Power to the Station Auxiliaries

15.2.10 Excessive Heat Removal due to Feedwater System Malfunctions (full power cases)

15.2.11 Sudden Feedwater Temperature Reduction

15.2.12 Excessive Load Increase

15.2.15 Spurious Operation of the Safety Injection System at Power

15.3.4 Complete Loss of Forced Reactor Coolant Flow

15.3.5 Single RCCA Withdrawal at Full Power

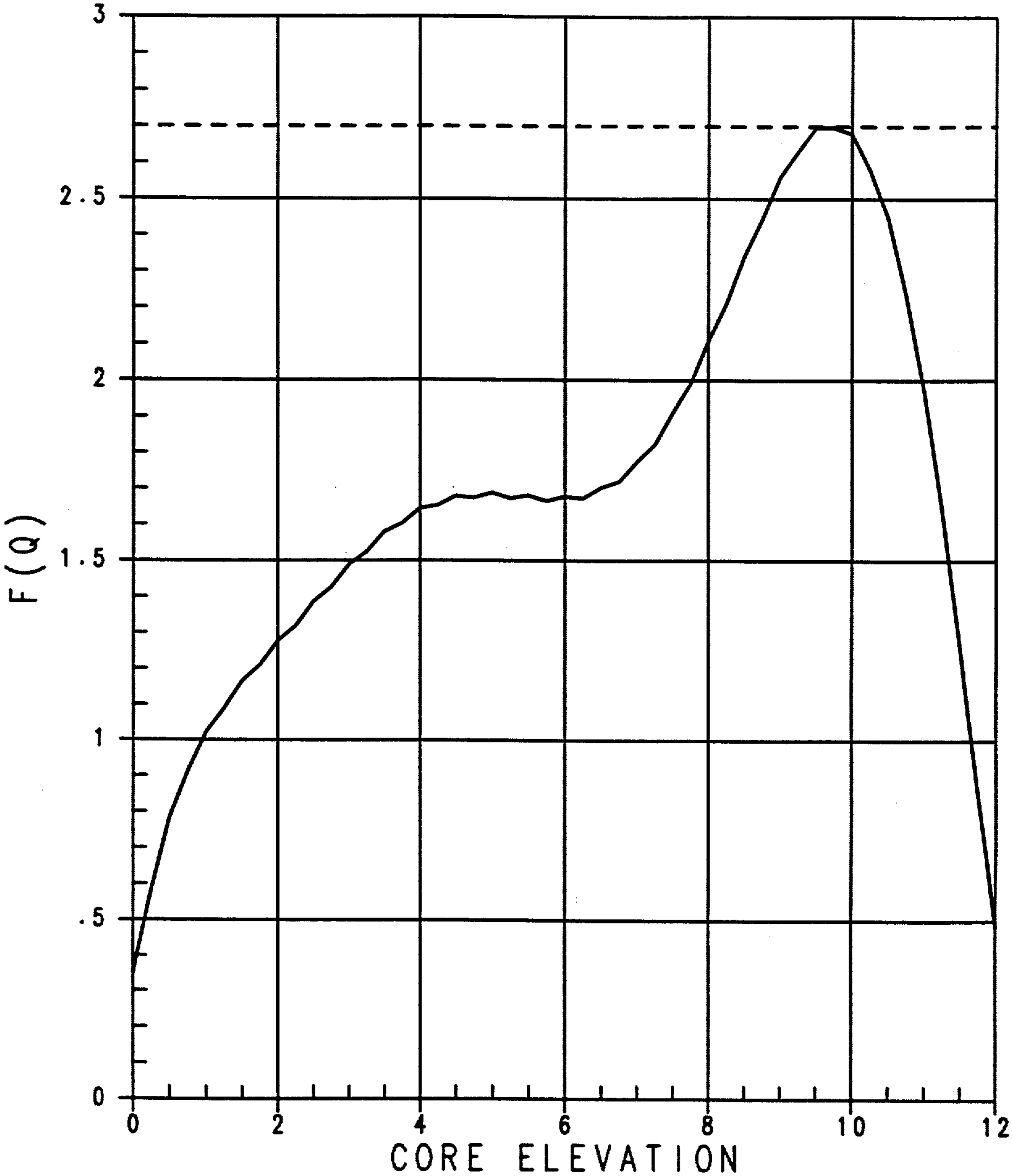


Figure 3.1.2-1 Small Break Hot Rod Power Shape

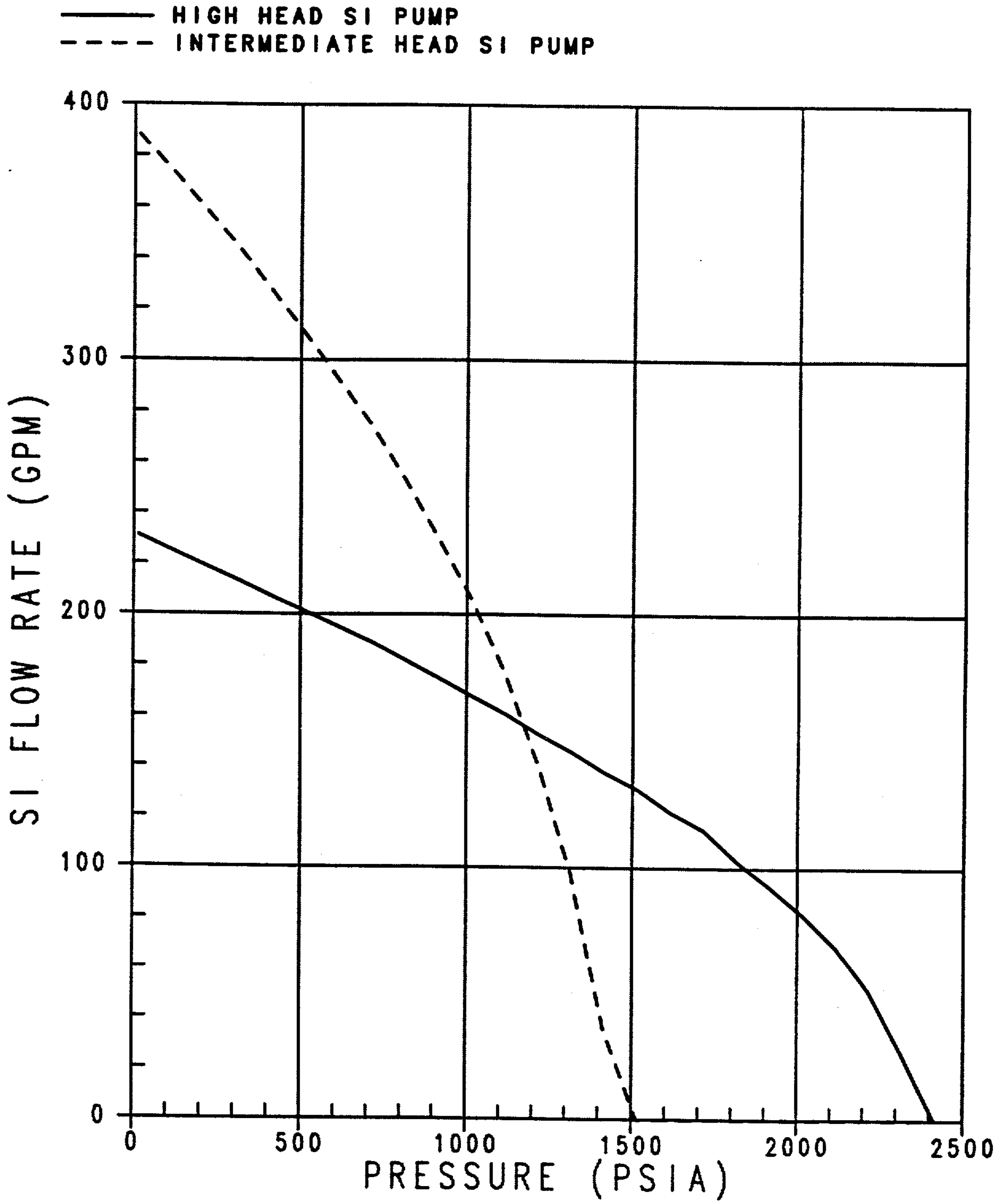


Figure 3.1.2-2 Small Break Pumped SI Flow Rates - IHSI and HHSI Pumps

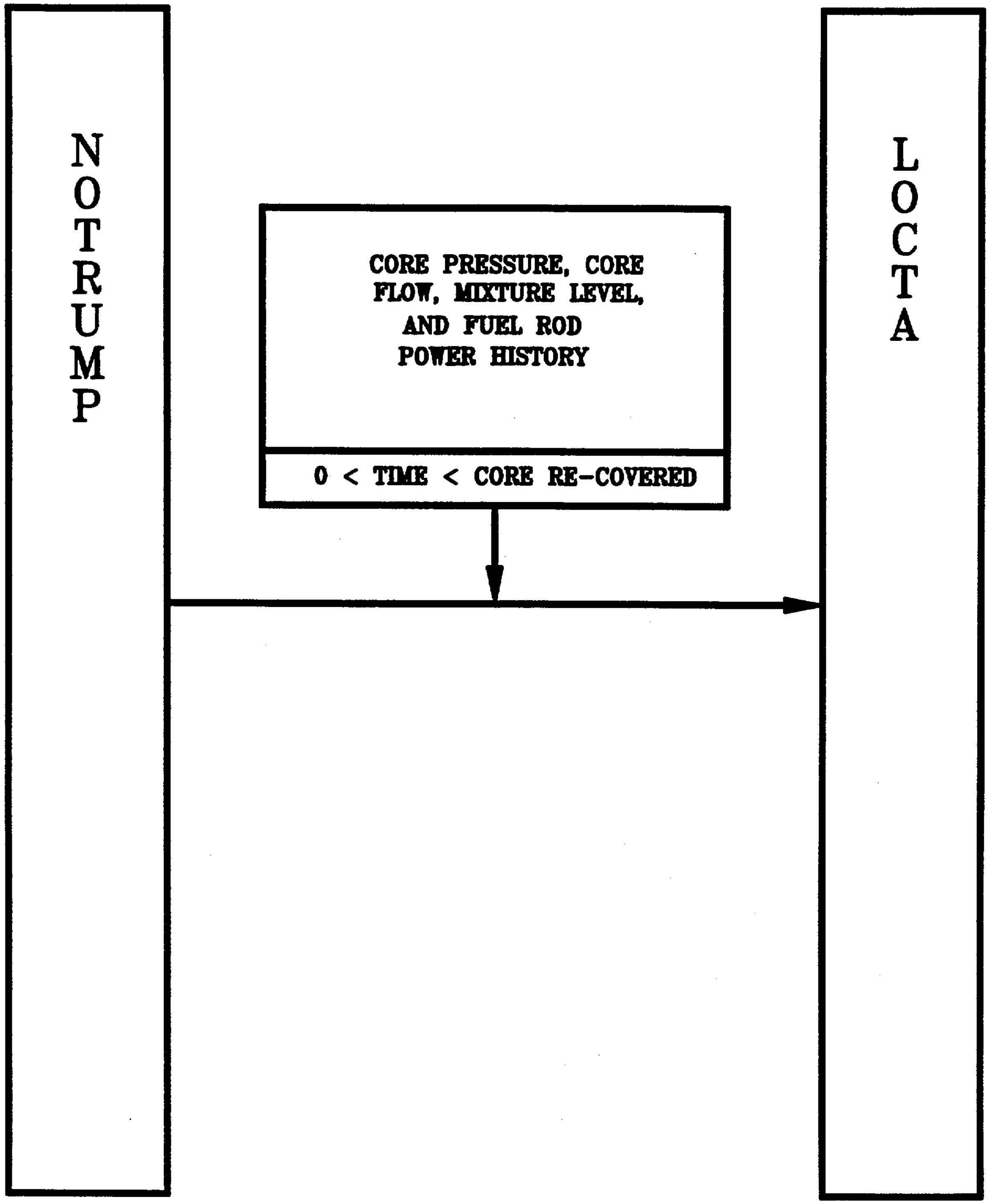
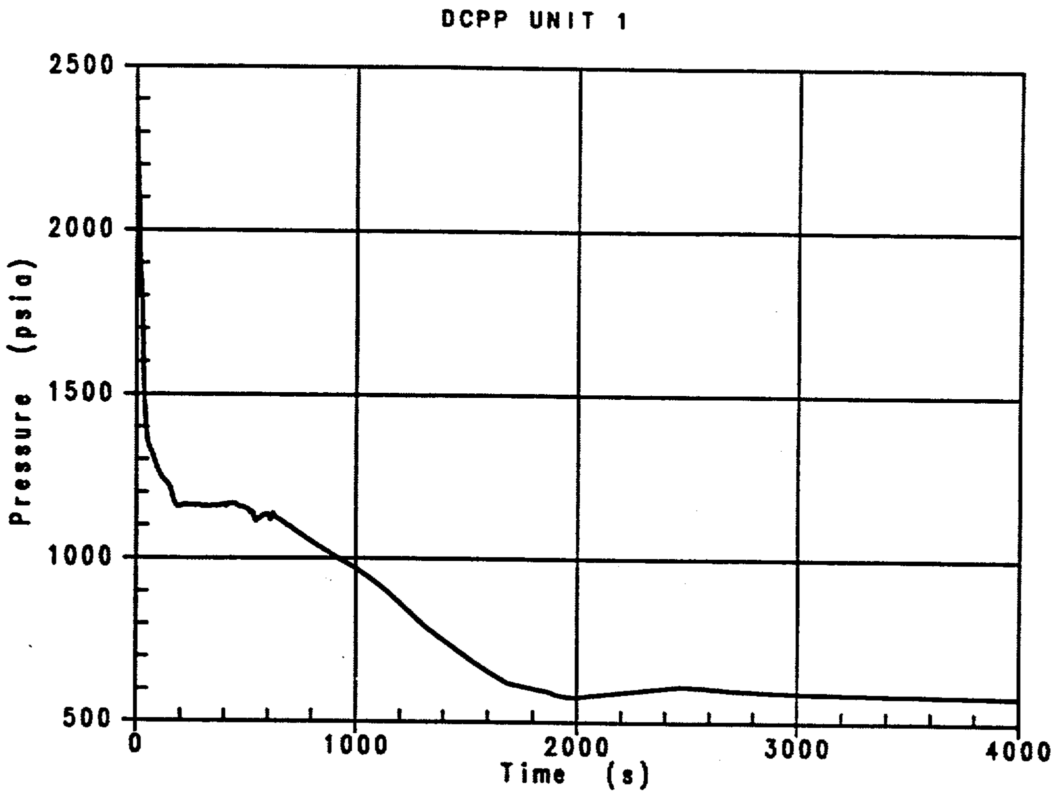


Figure 3.1.2-3 Code Interface Description for the Small Break LOCA Model



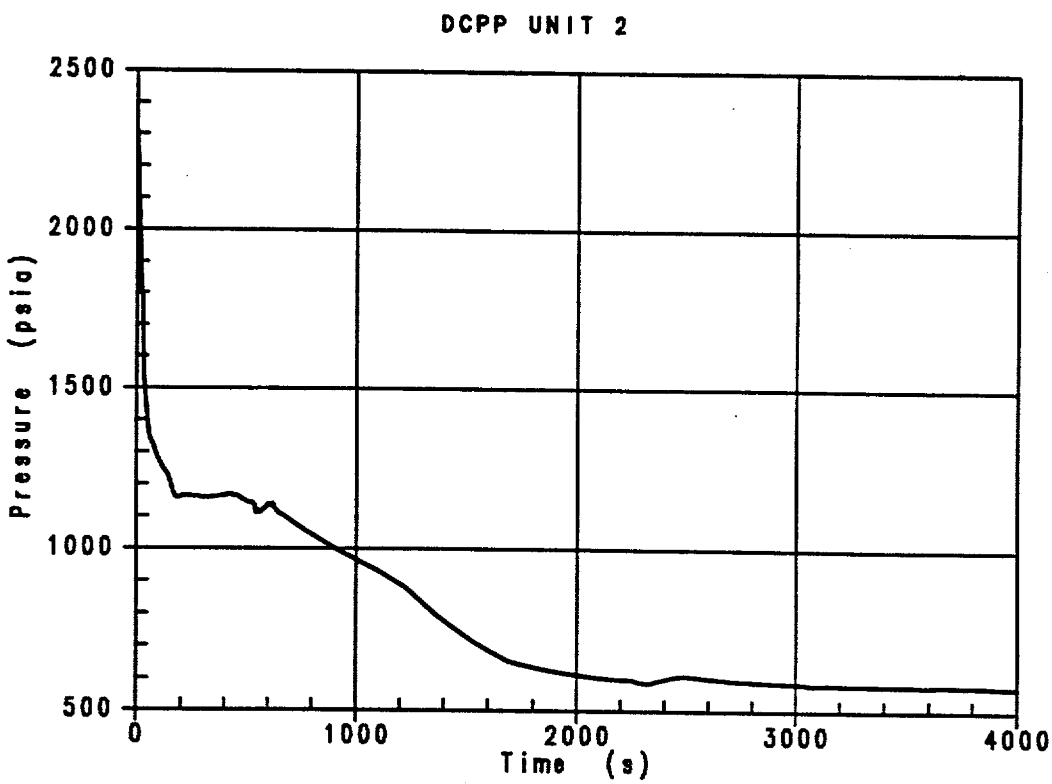


Figure 3.1.2-4 RCS Depressurization Transient, Limiting 3-Inch Break, High T_{AVG}

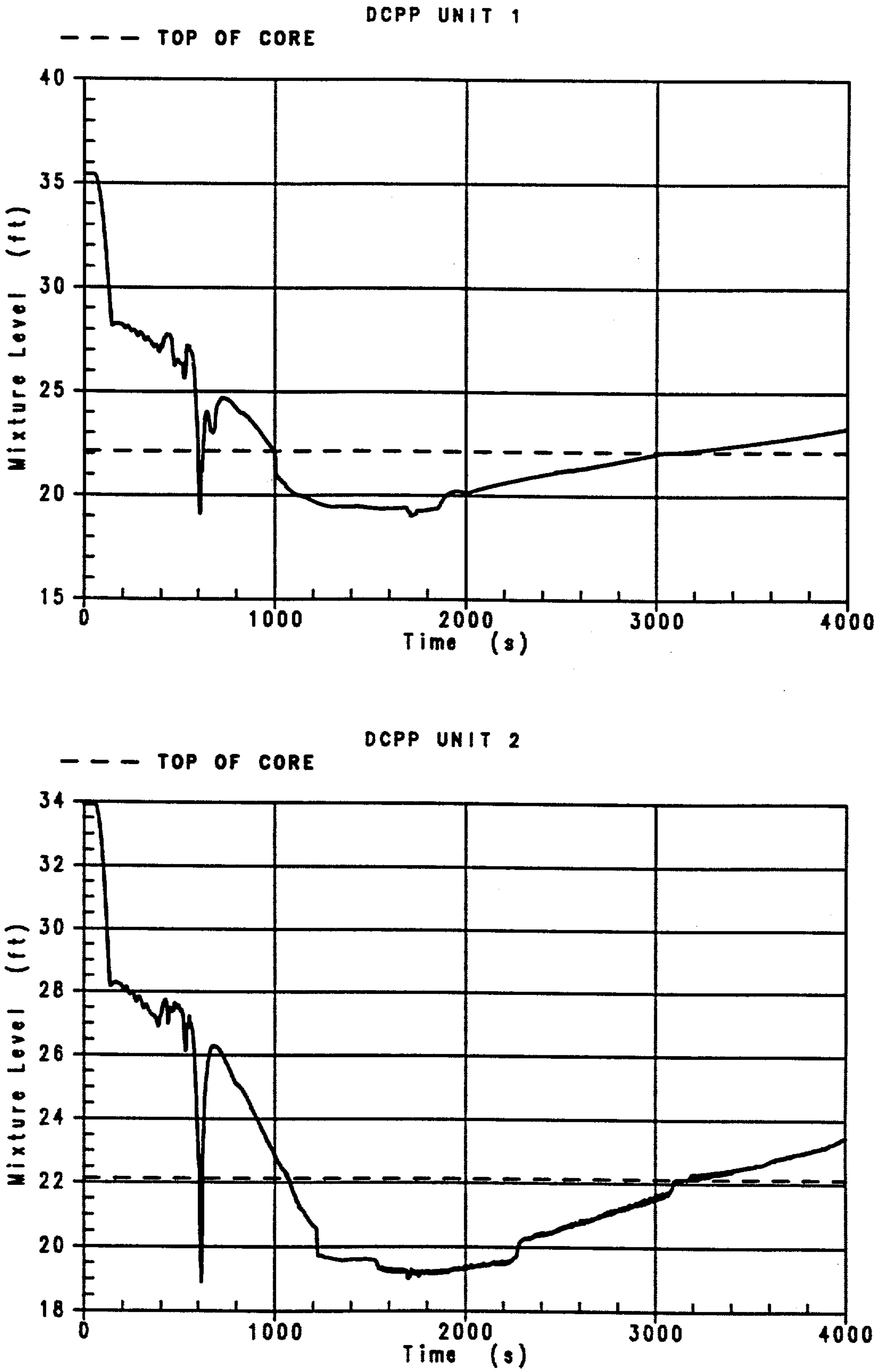
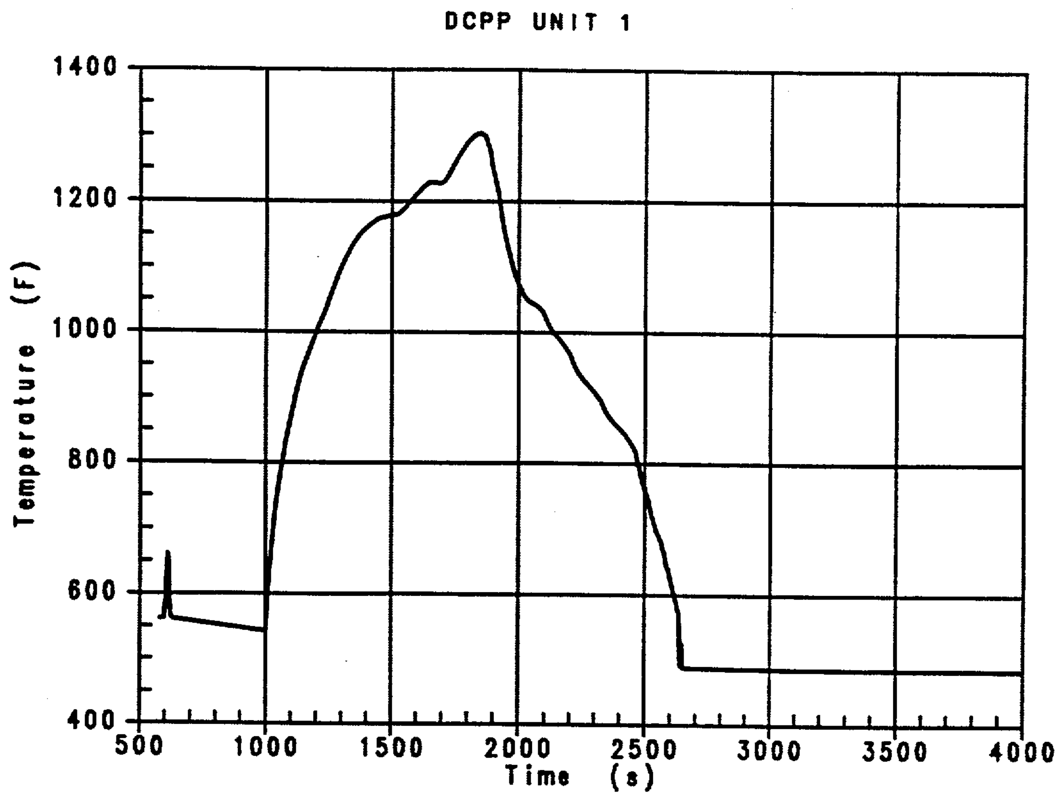


Figure 3.1.2-5 Core Mixture Level, 3-Inch Break, High T_{AVG}



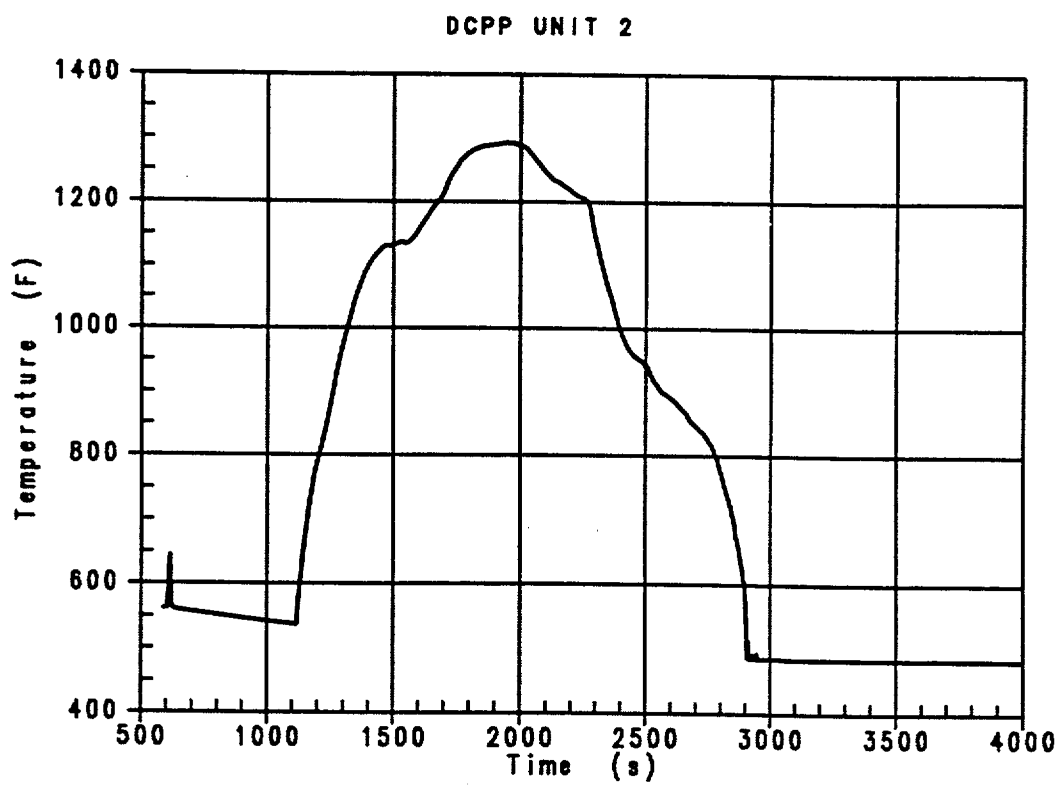
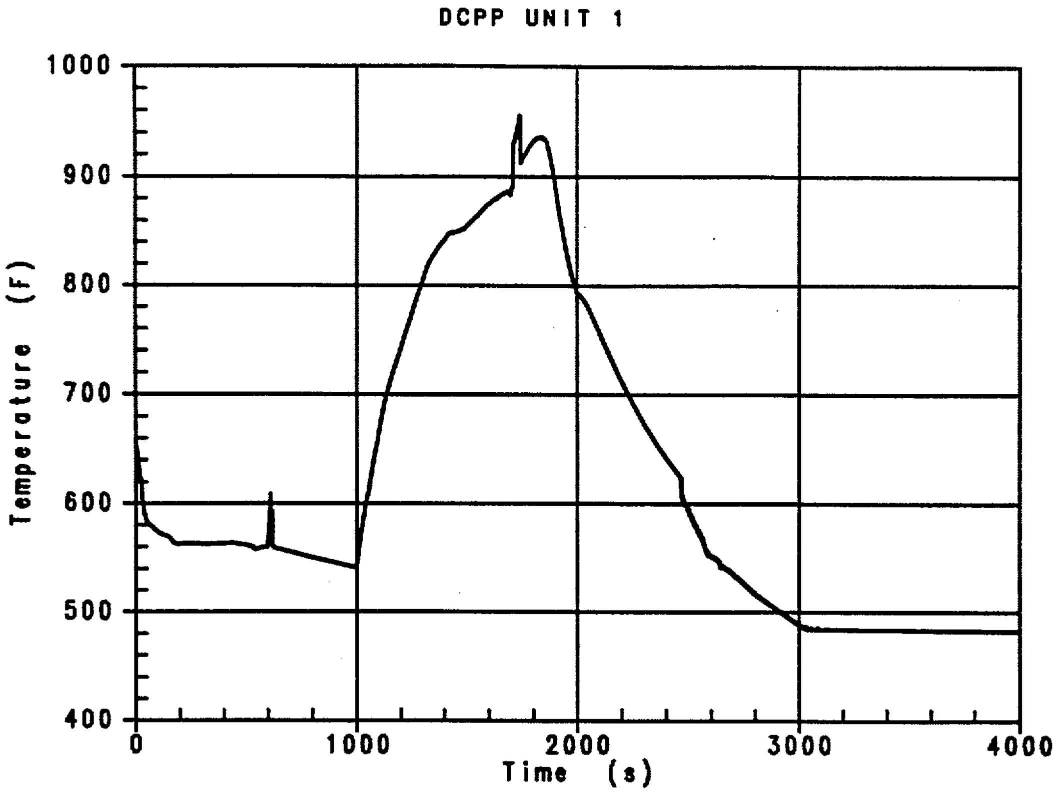


Figure 3.1.2-6 Peak Cladding Temperature - Hot Rod, 3-Inch Break, High T_{AVG}



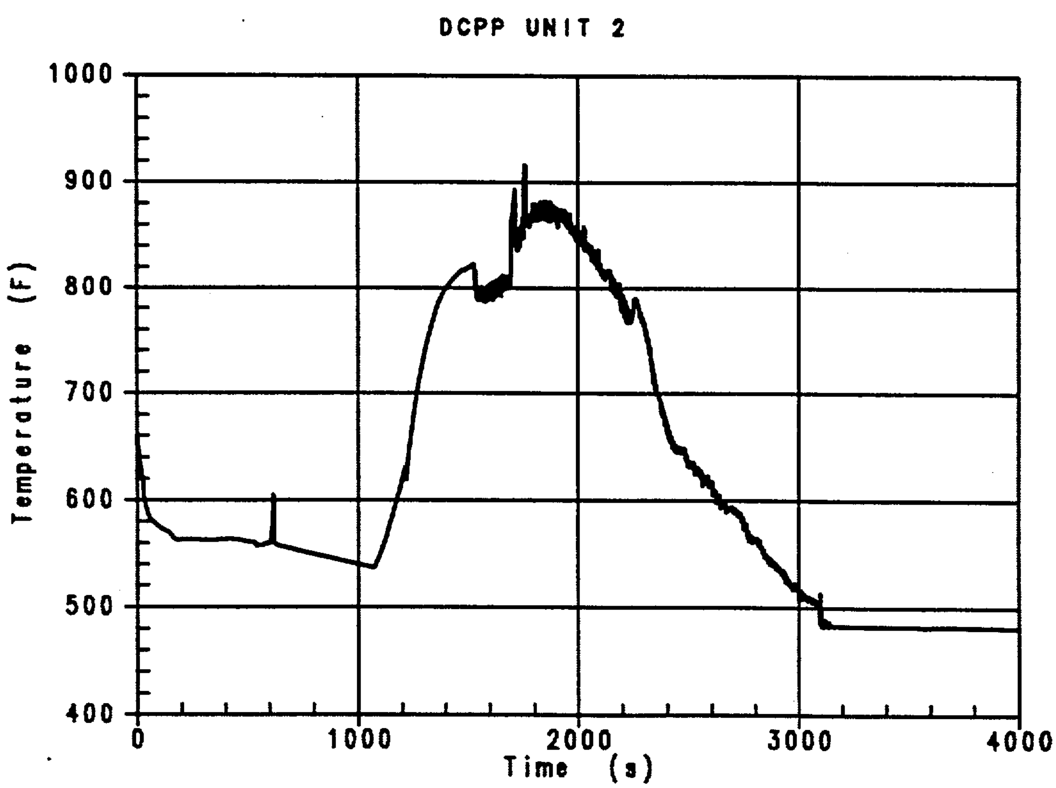
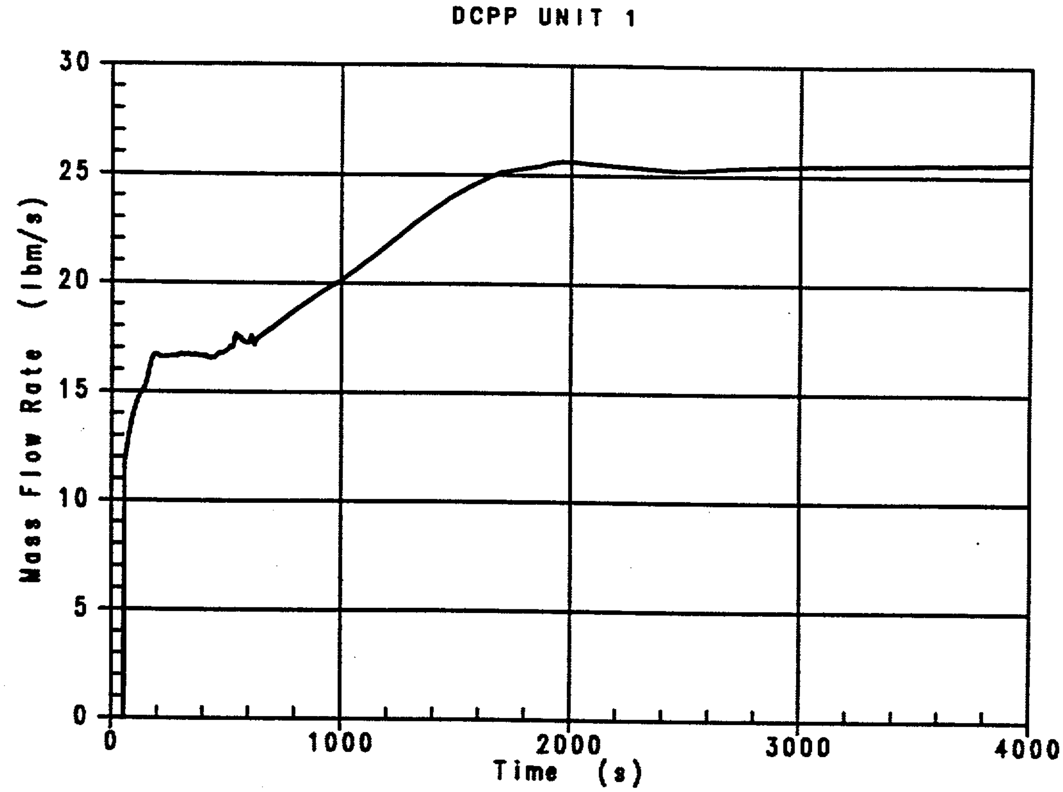


Figure 3.1.2-7 Top Core Node Vapor Temperature, 3-Inch Break, High T_{AVG}



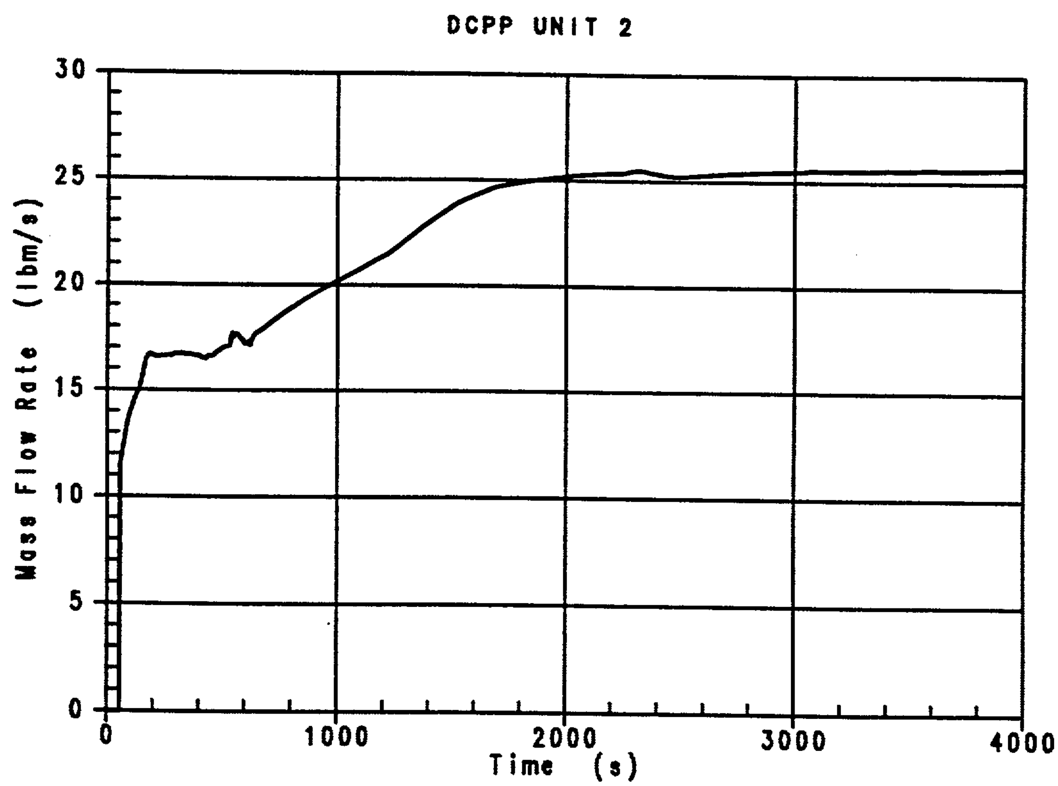
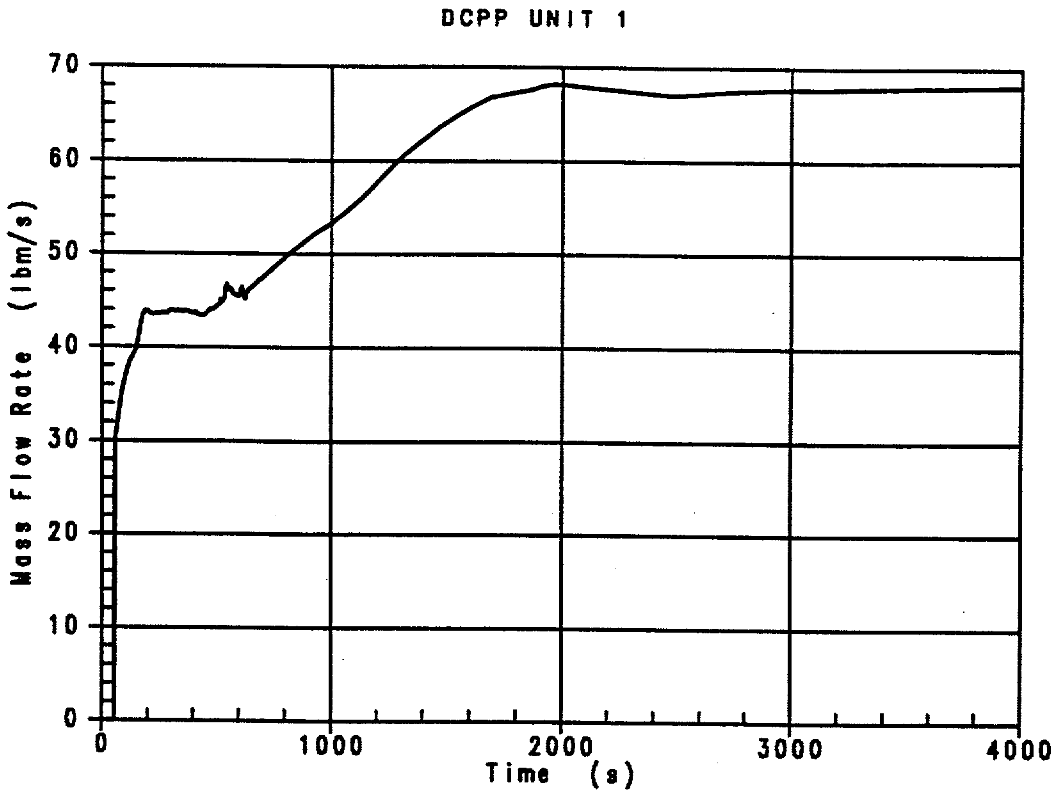


Figure 3.1.2-8 ECCS Pumped SI - Broken Loop, 3-Inch Break, High T_{AVG}



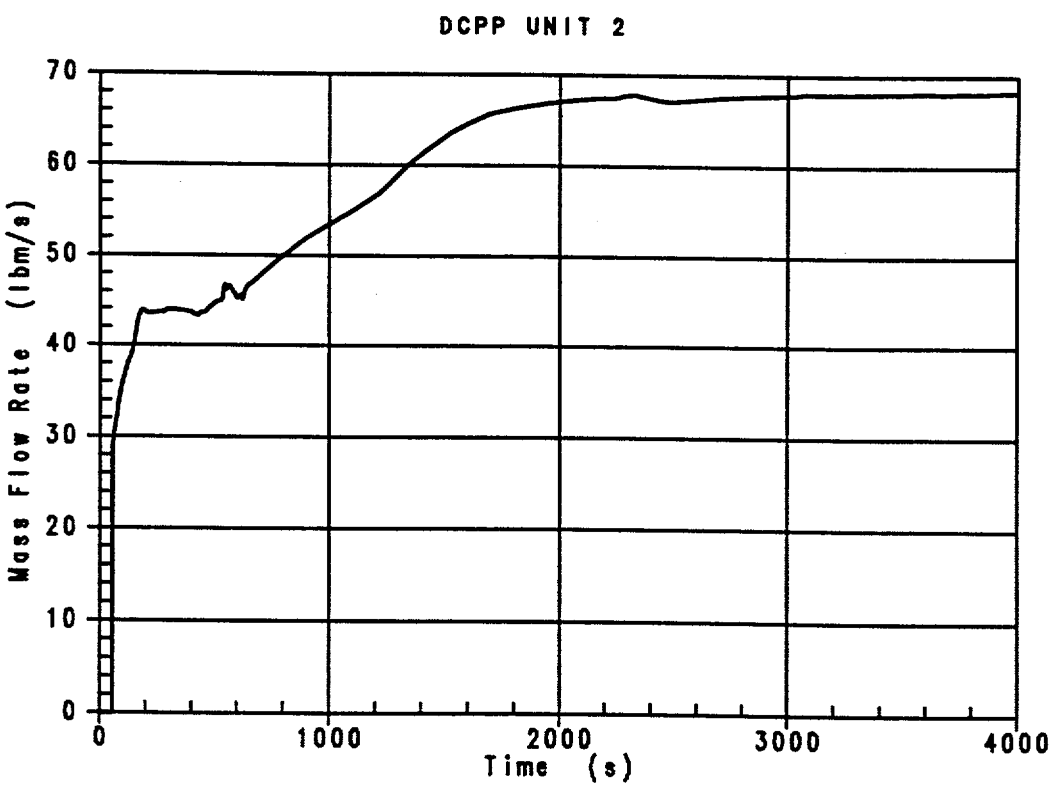
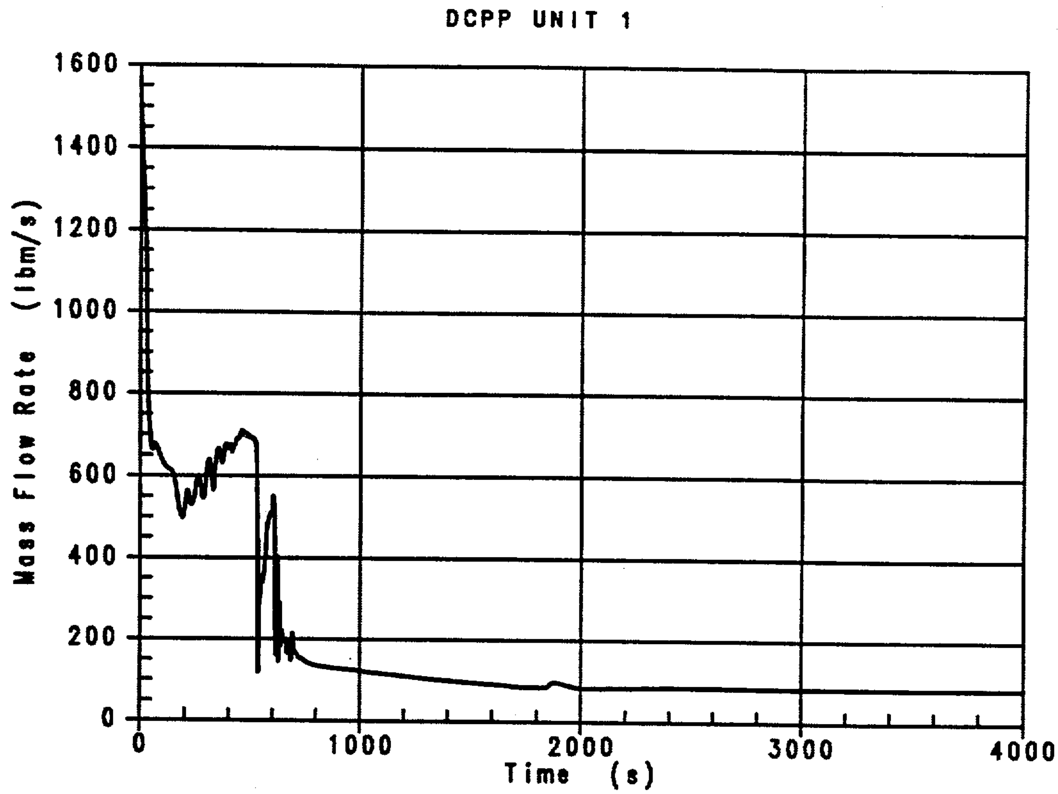


Figure 3.1.2-9 ECCS Pumped SI - Intact Loop, 3-Inch Break, High T_{AVG}



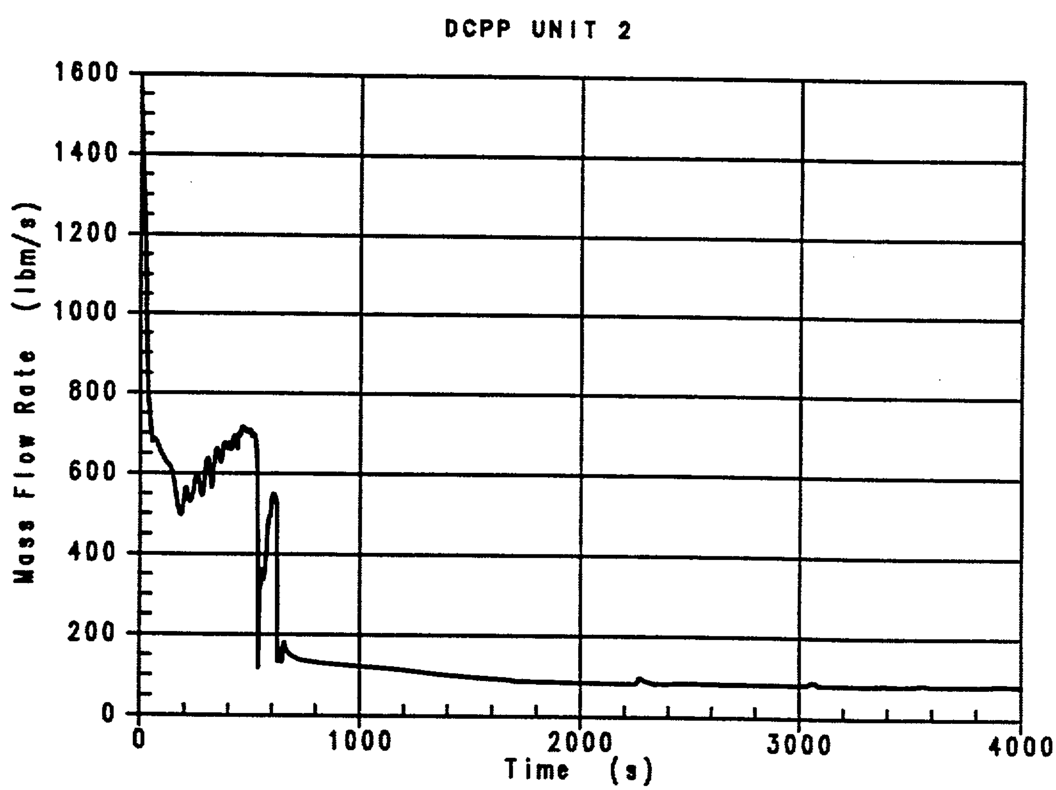
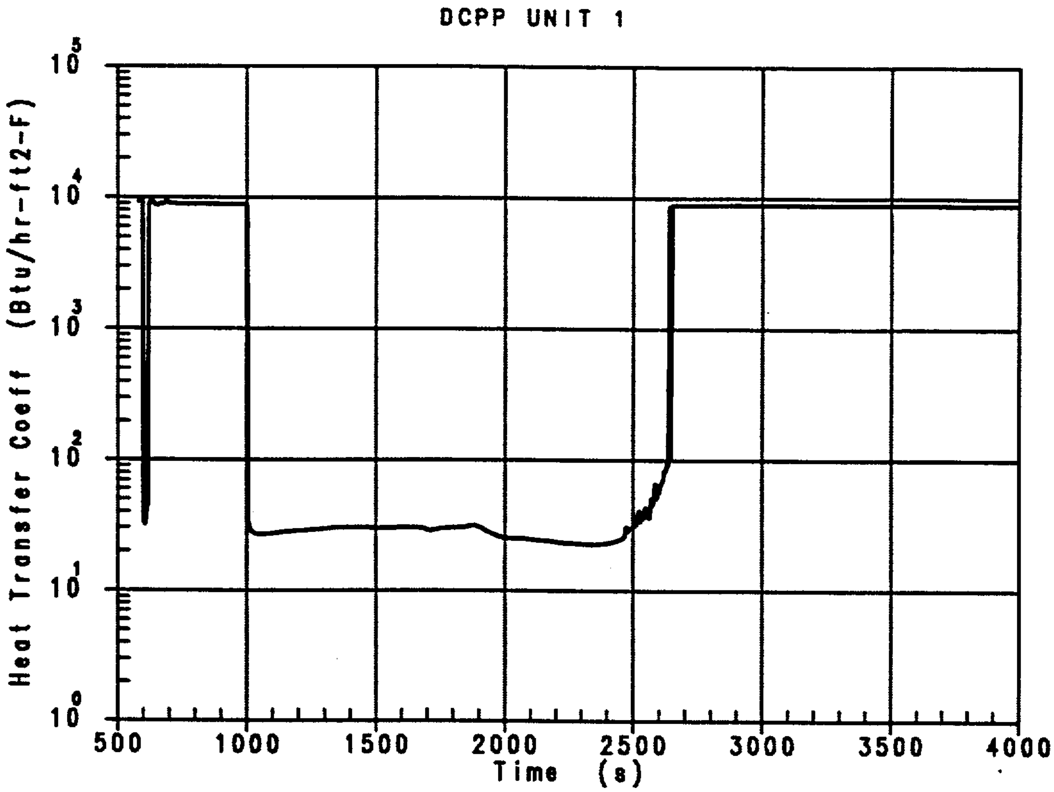


Figure 3.1.2-10 Cold Leg Break Mass Flow, 3-Inch Break High T_{AVG}



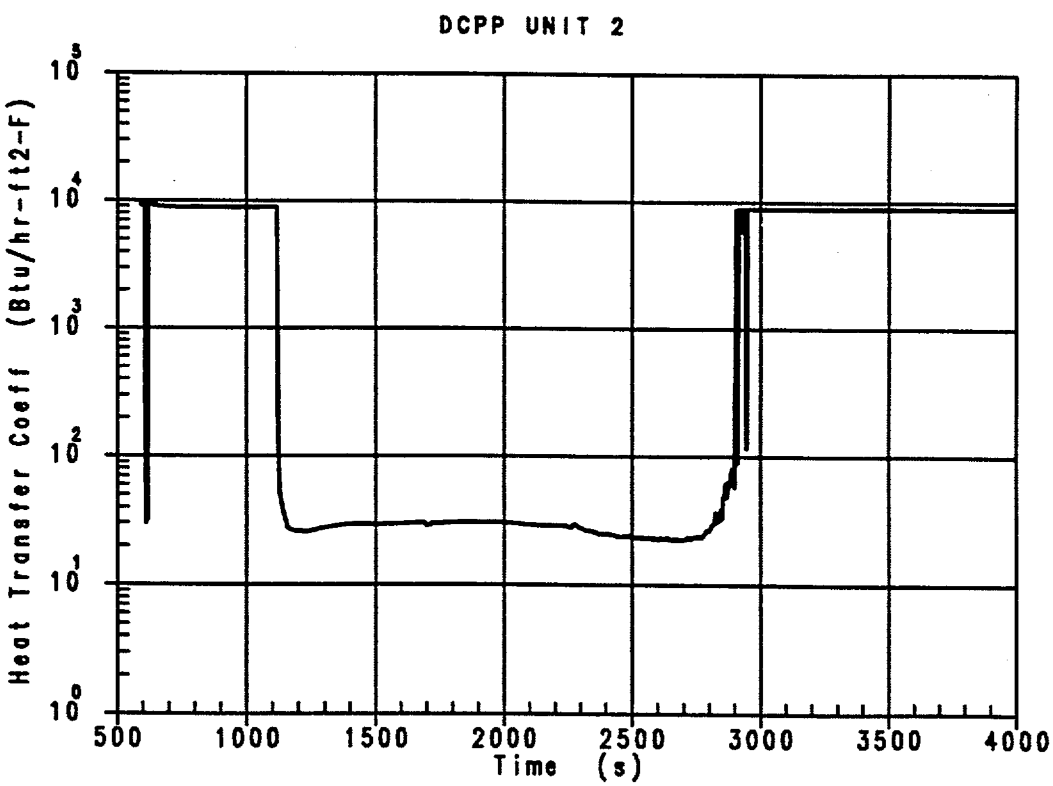
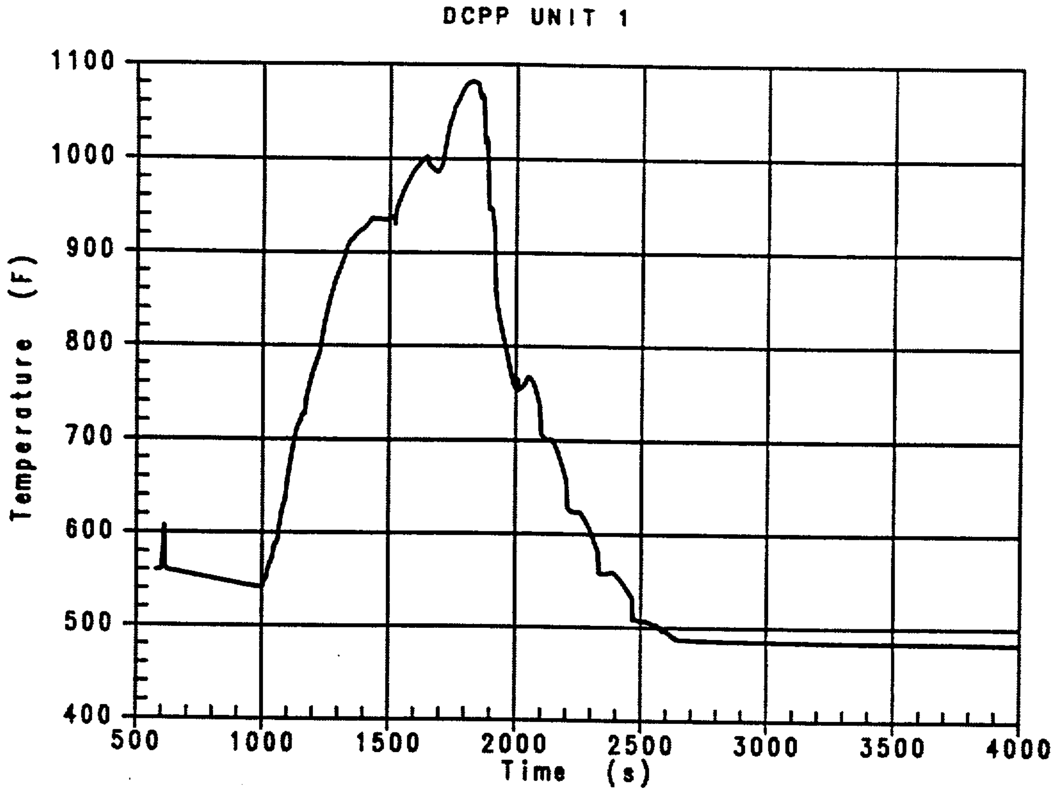


Figure 3.1.2-11 Hot Rod Surface Heat Transfer Coefficient - Hot Spot, 3-Inch Break, High T_{AVG}



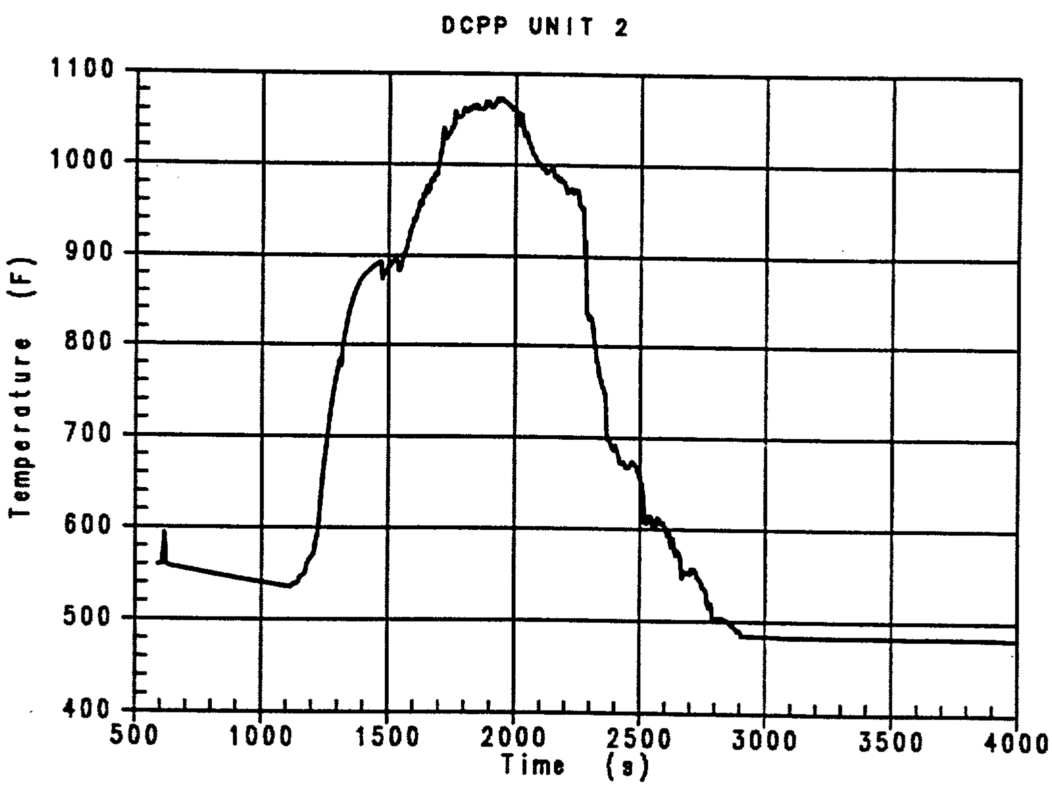
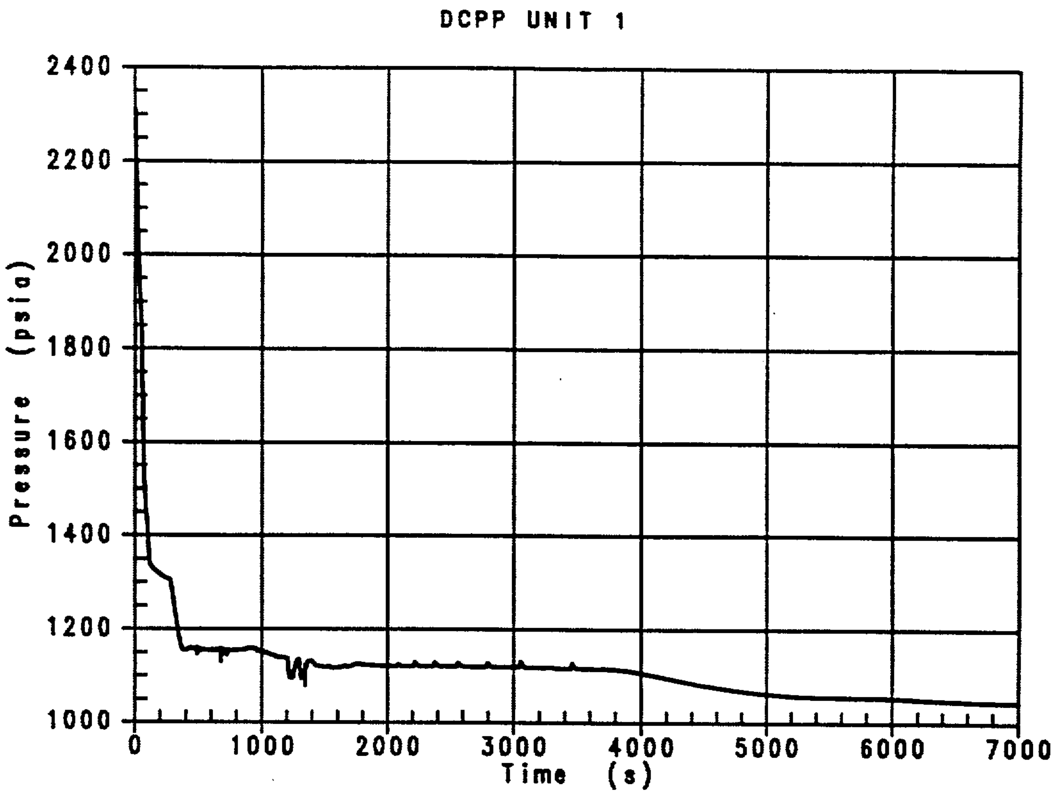


Figure 3.1.2-12 Fluid Temperature - Hot Spot, 3-Inch Break, High T_{AVG}



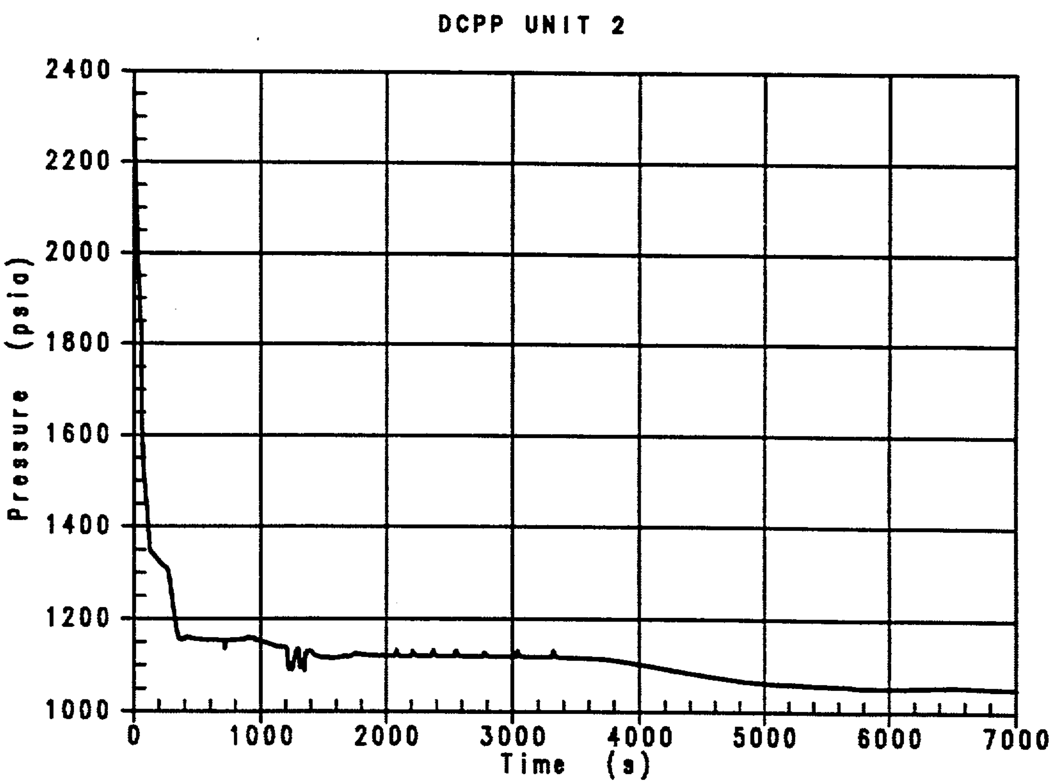
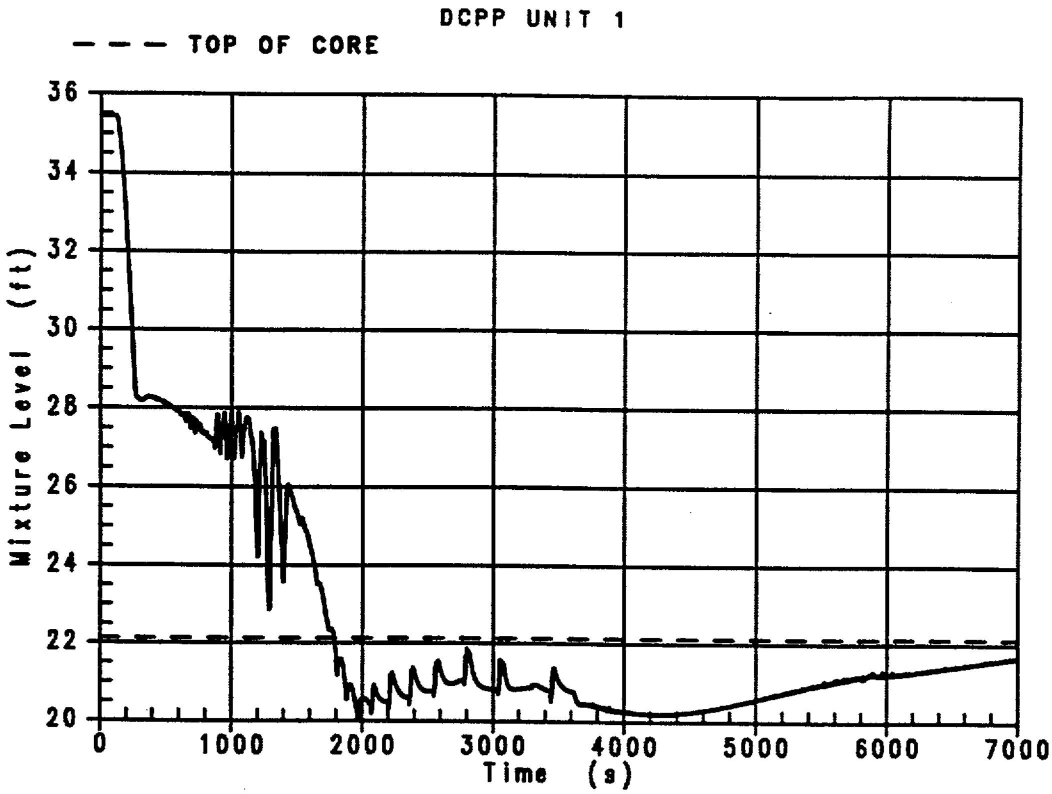


Figure 3.1.2-13 RCS Depressurization Transient, 2-Inch Break, High T_{AVG}



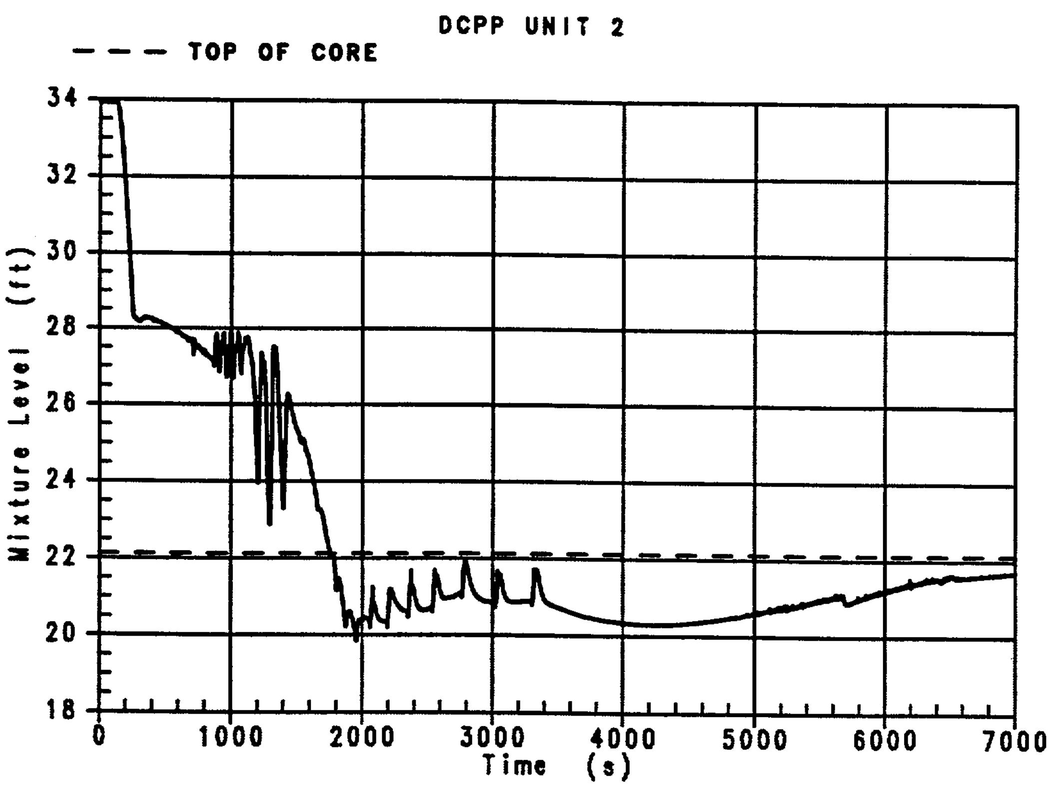


Figure 3.1.2-14 Core Mixture Level, 2-Inch Break, High T_{AVG}

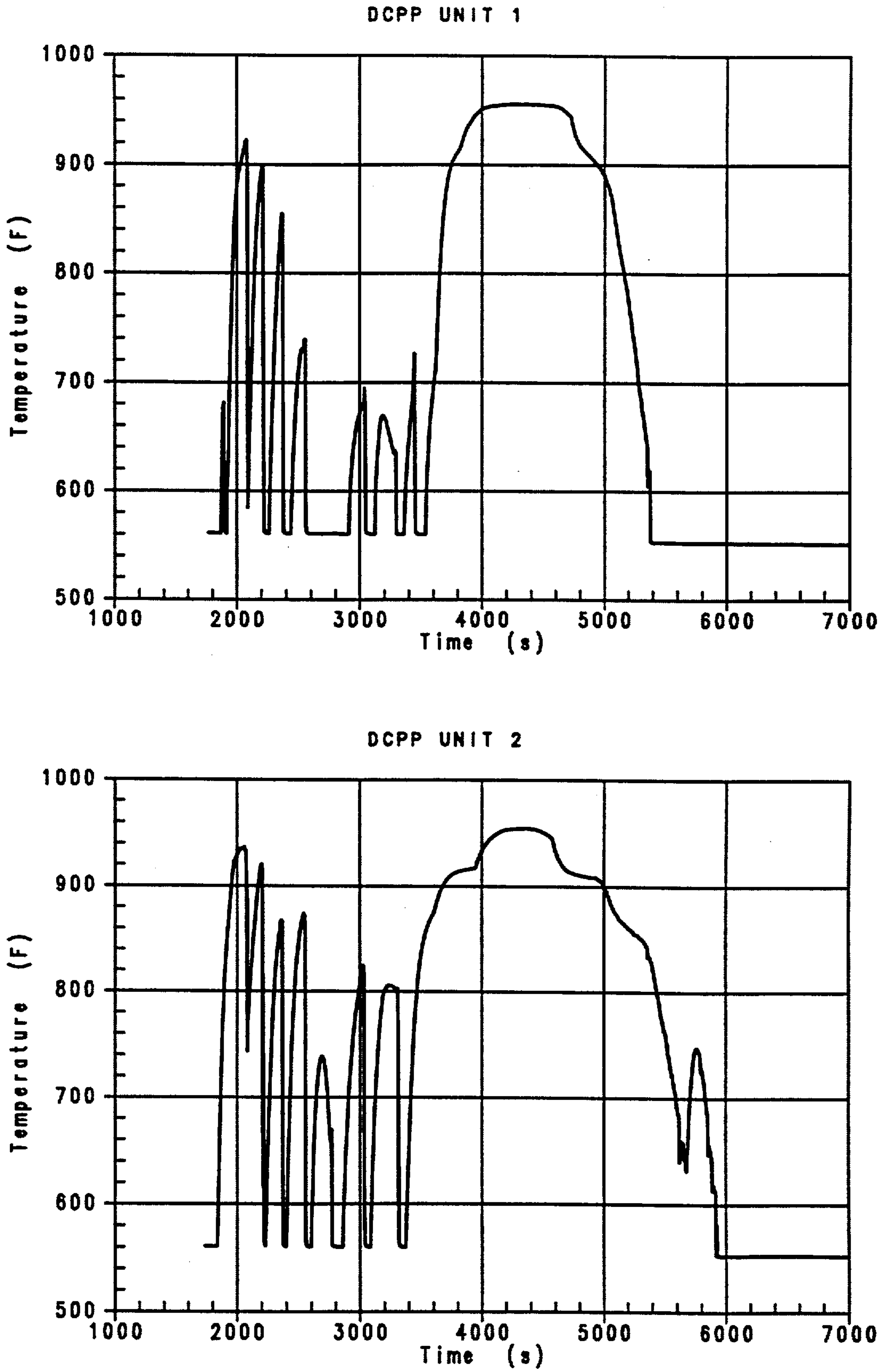


Figure 3.1.2-15 Peak Cladding Temperature - Hot Rod, 2-Inch Break, High T_{AVG}

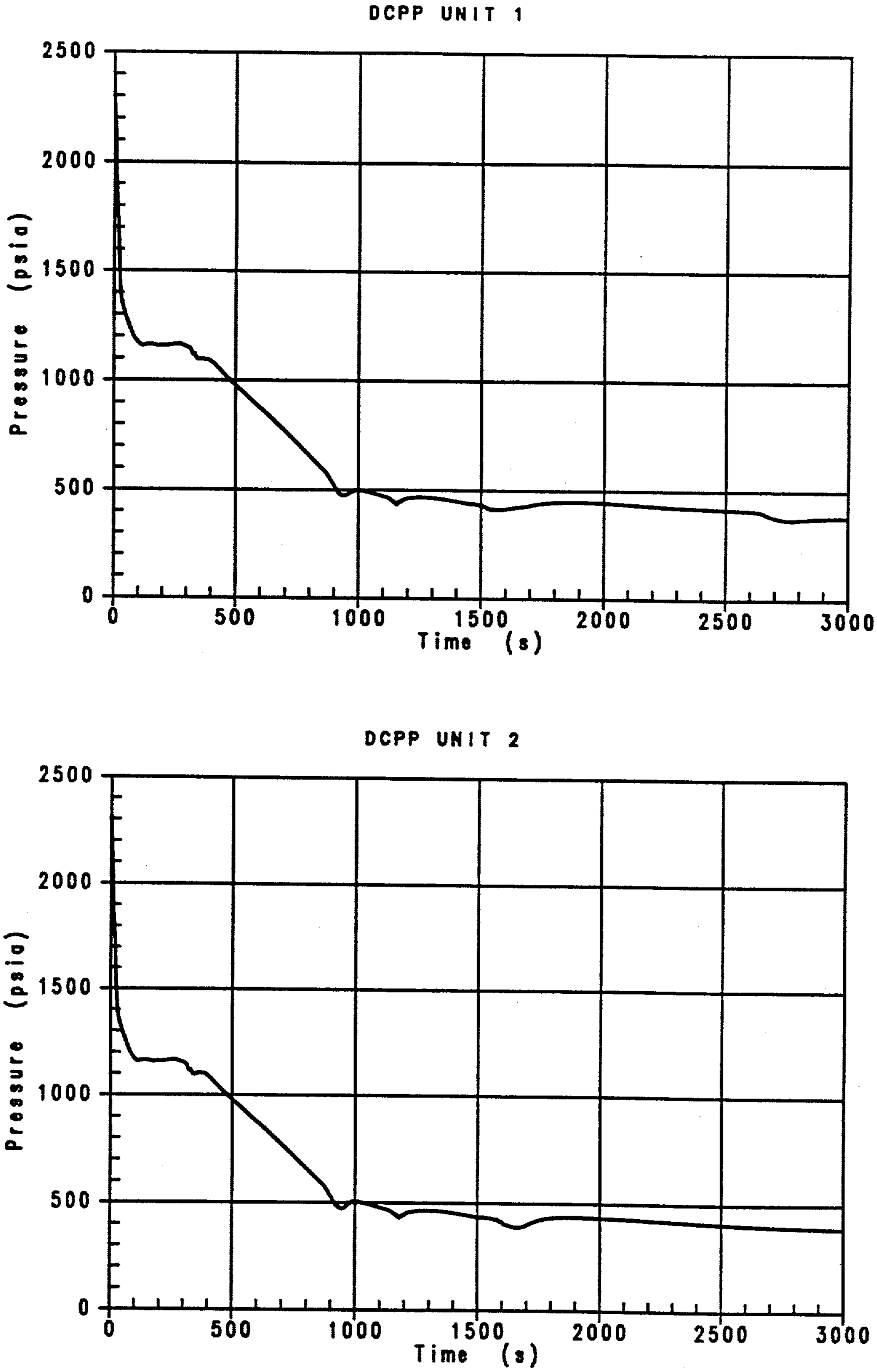
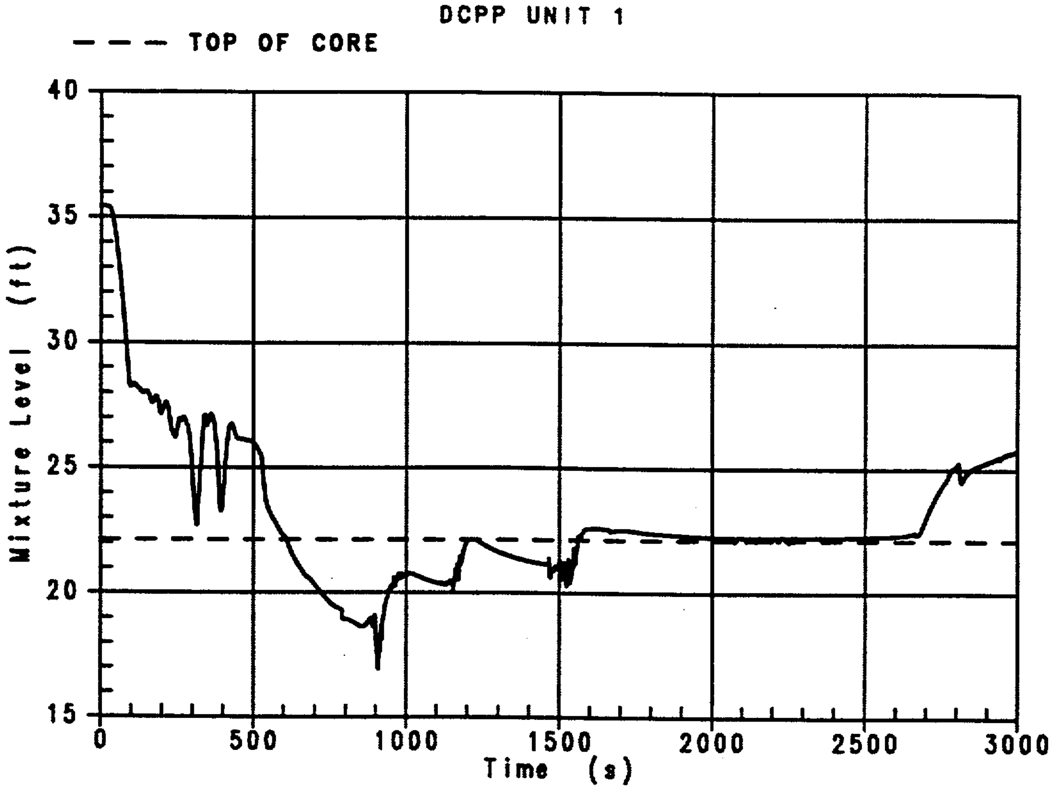


Figure 3.1.2-16 RCS Depressurization Transient, 4-Inch Break, High T_{AVG}



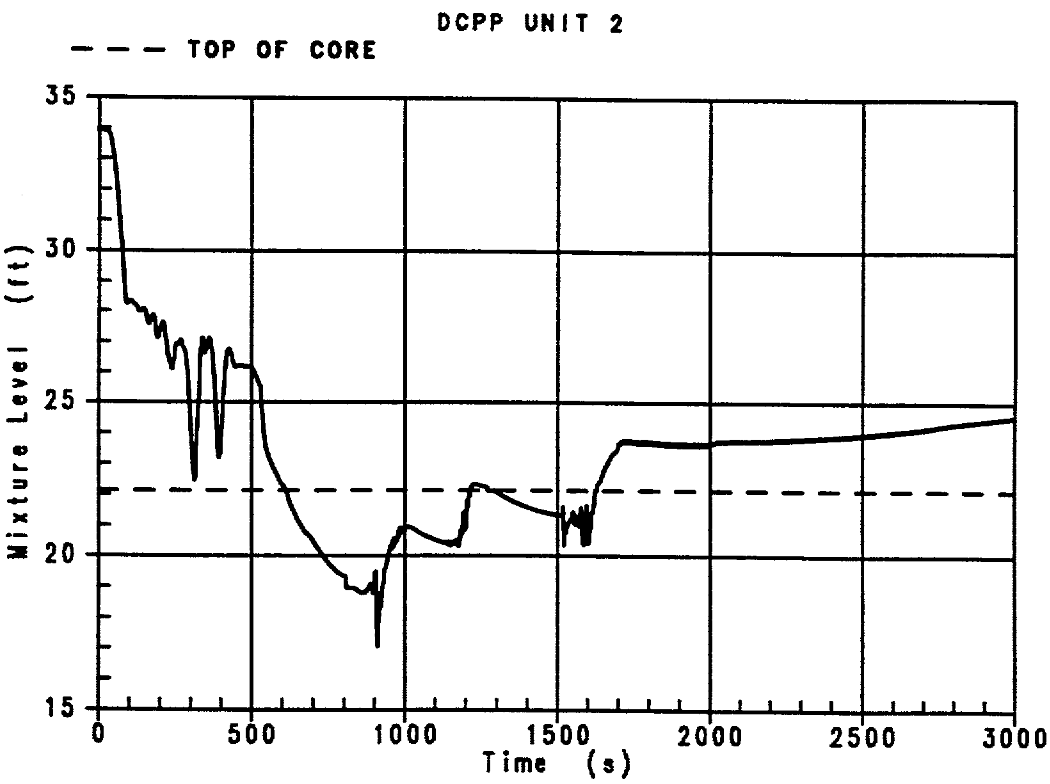


Figure 3.1.2-17 Core Mixture Level, 4-Inch Break, High T_{AVG}

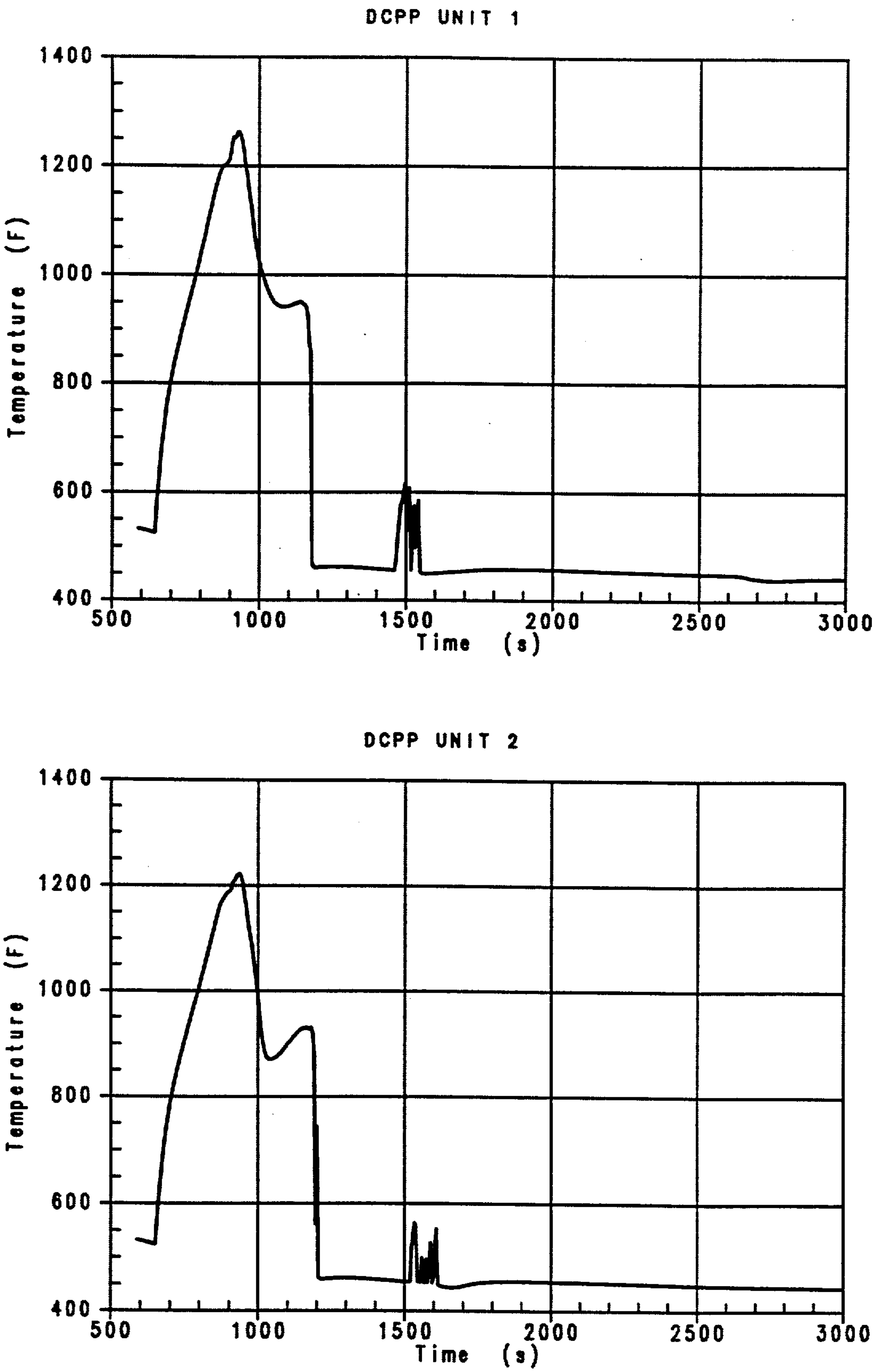


Figure 3.1.2-18 Peak Cladding Temperature - Hot Rod, 4-Inch Break, High T_{AVG}

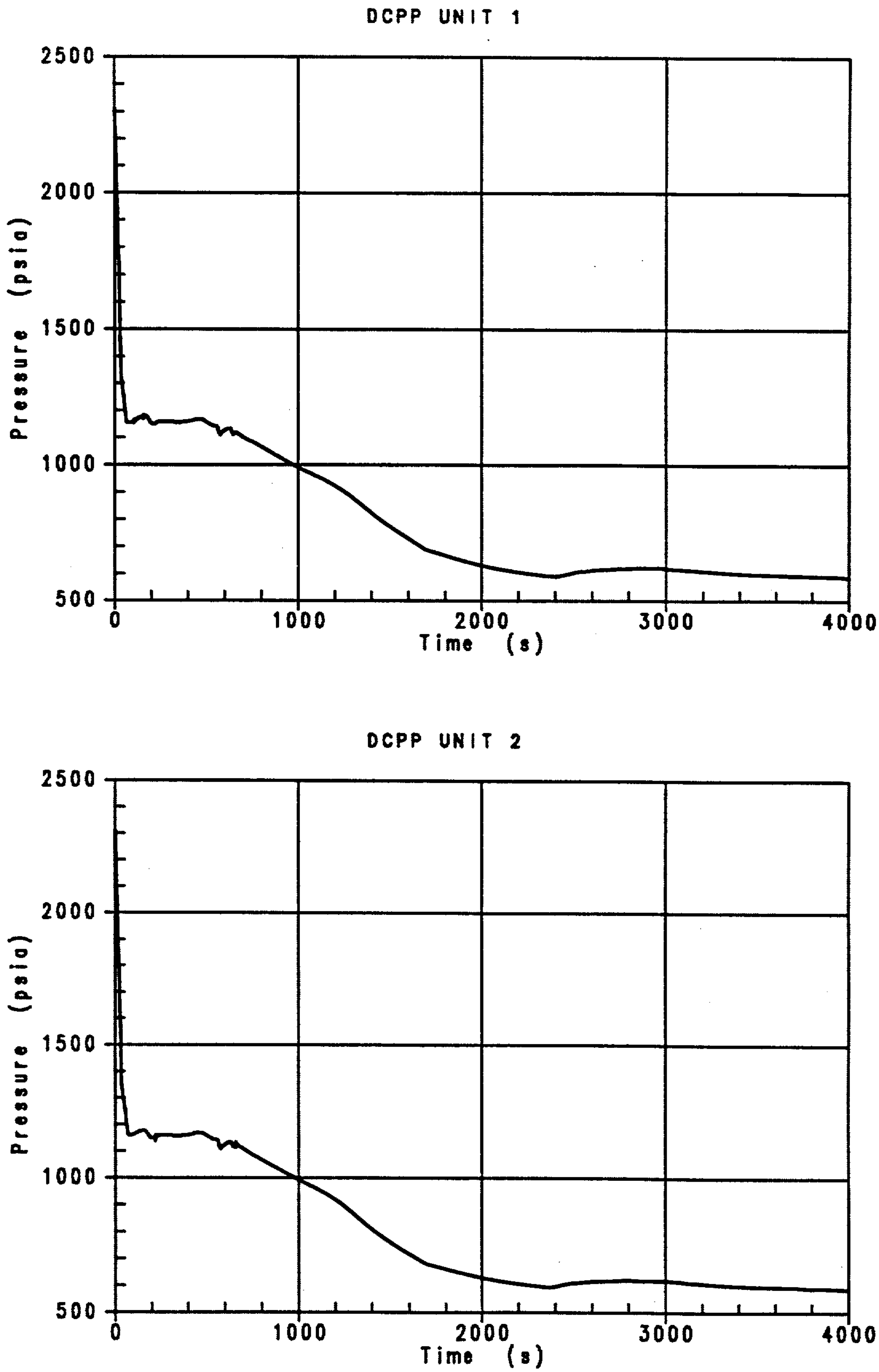
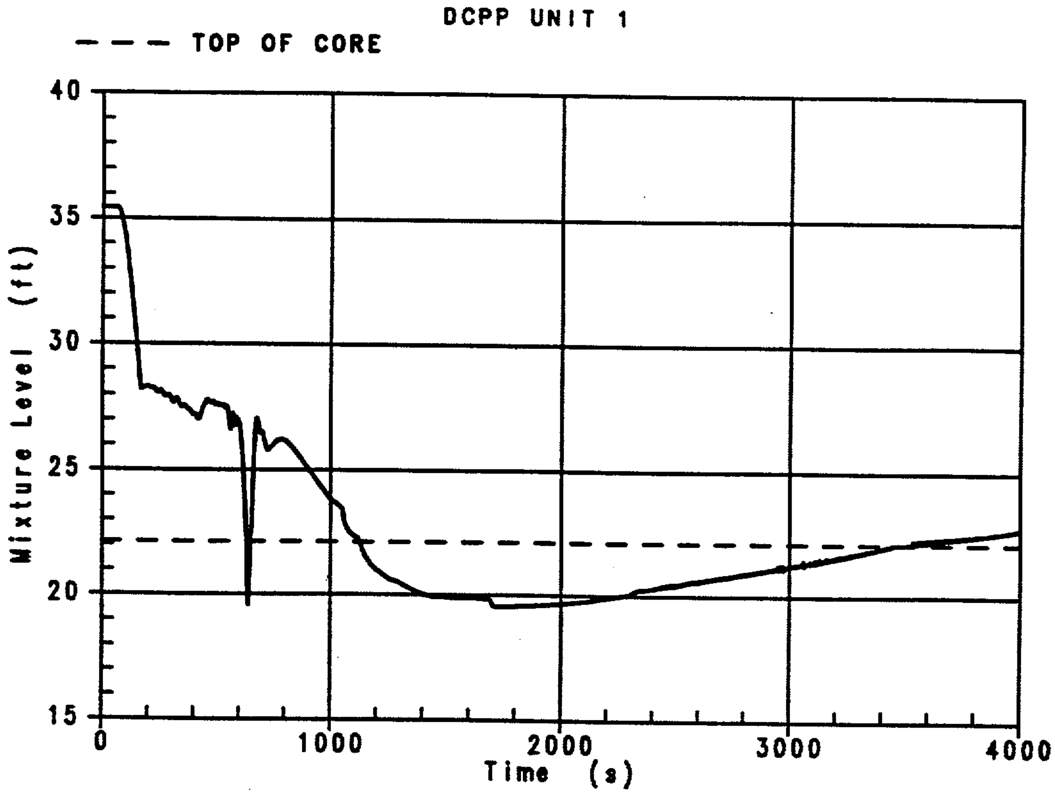


Figure 3.1.2-19 RCS Depressurization Transient, 3-Inch Break, Low T_{AVG}



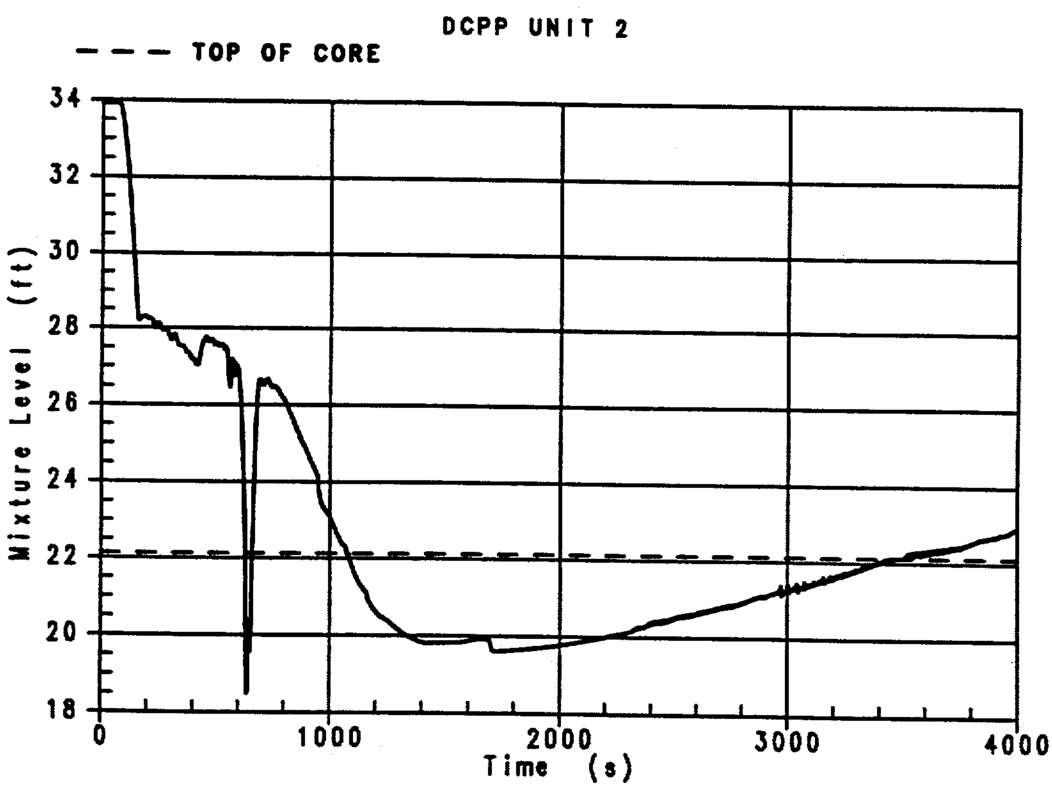


Figure 3.1.2-20 Core Mixture Level, 3-Inch Break, Low T_{AVG}

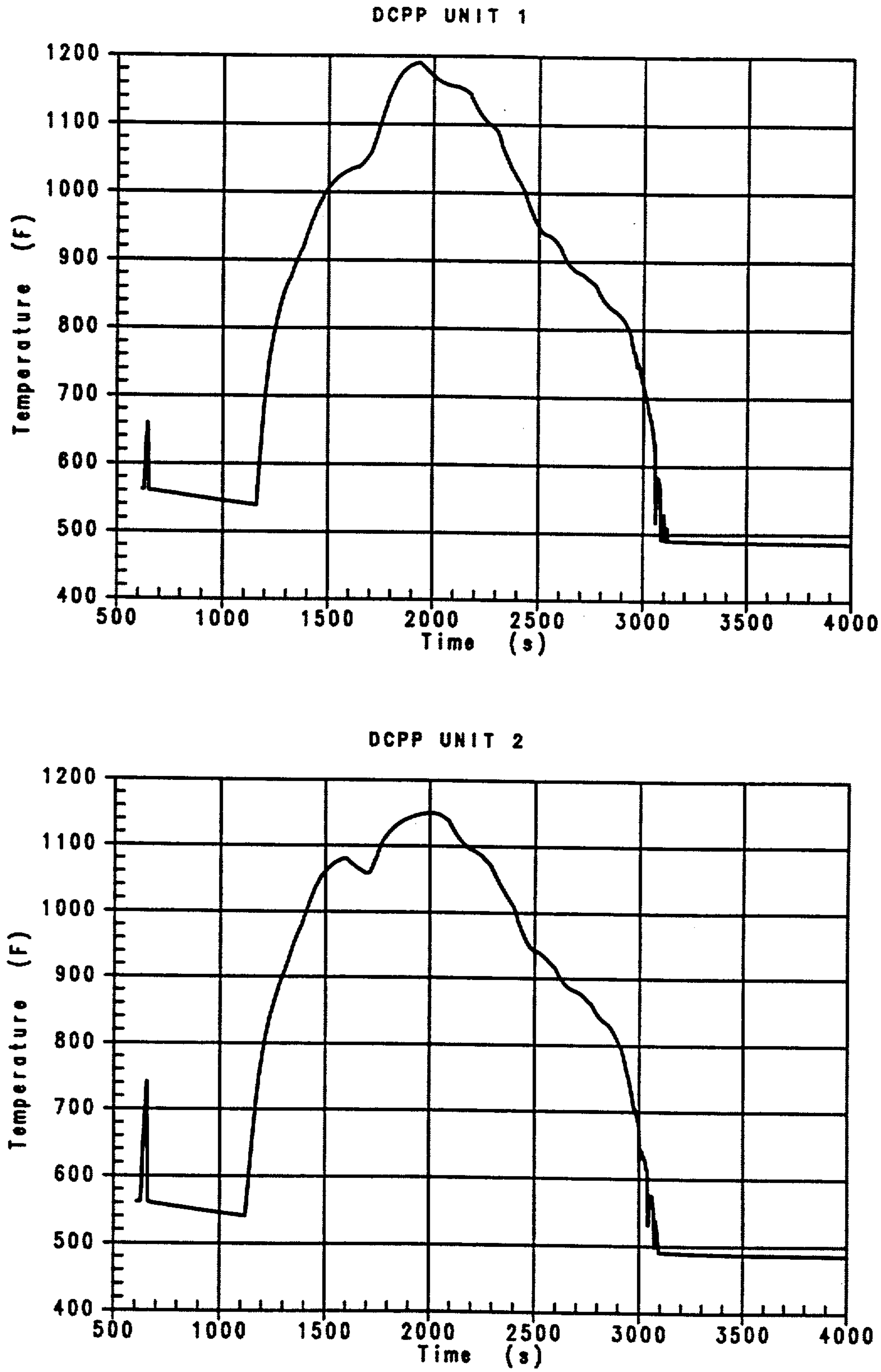


Figure 3.1.2-21 Peak Cladding Temperature, 3-Inch Break, Low T_{AVG}

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.4 Single Reactor Coolant Pump Locked Rotor

15.4.6 RCCA Ejection (full power cases)

The currently applicable analyses or calculations for the following setpoints or transients were not performed in a bounding manner. Rather, separate analyses were performed for each Diablo Canyon unit.

15.1.3 Overtemperature and Overpower AT Reactor Trip Setpoint Calculations

15.2.13 Accidental Depressurization of the Reactor Coolant System

In order to support the Unit 1 uprated conditions these items have been reanalyzed, as described in the sections below.

3.2.2 Overtemperature and Overpower ΔT Reactor Trip Setpoint Calculations

The Diablo Canyon units both currently use the same OTΔT/OPΔT trip setpoint constants. However, calculations to confirm the acceptability of these setpoints are performed separately for the specific plant operating conditions of each unit, using the methodology of Reference 1. The currently applicable setpoint calculations are based on reactor core thermal limits for 17x17 standard fuel, which is limiting with respect to the 17x17 VANTAGE 5 fuel type currently used at Diablo Canyon. There is insufficient DNB margin available to support the current setpoints assuming 17x17 standard fuel for the Unit 1 uprated conditions. Therefore, revised core thermal limits were developed based on the uprated Unit 1 power and flow parameters which assume 17x17 VANTAGE 5 fuel only. Setpoint calculations were performed which verify that the present Technical Specification OTΔT/OPΔT trip constants and the associated f(ΔI) penalty function provide adequate protection for the revised core limits at the uprated Unit 1 power conditions.

Note, the above evaluation and results are applicable to VANTAGE 5 fuel with either ZIRLO™ or standard zircaloy.

3.2.3 Accidental Depressurization of the Reactor Coolant System

The currently applicable analysis considered each Diablo Canyon unit separately. The limiting Unit 2 analysis is presented in the updated UFSAR section 15.2.13. Since this analysis credited the higher RCS flow of Unit 2, it does not bound the uprated Unit 1 plant conditions. A new analysis was performed using conservative assumptions that bound both units. The transient results are similar to those presented in the FSAR, except for the specific DNBR calculation which now assumes the VANTAGE 5 fuel type instead of the limiting standard fuel which is no longer used in the Diablo Canyon cores. The DNBR remains above the applicable limit value, and the conclusions of the UFSAR remain valid.

3.2.3.1 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 flow rate 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 which could reach the hot leg saturation pressure if a reactor trip did not occur. The pressure continues to decrease throughout the transient. The effect of the pressure decrease is to decrease power 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 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

An accidental depressurization of the RCS is classified as an ANS Condition 2 event.

3.2.3.2 Analysis of Effects and Consequences

The accidental depressurization transient is analyzed with the LOFTRAN code (Reference 2). 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 3.

Some key analysis input assumptions are identified in Appendix A. In order to give conservative results in calculating the DNBR during the transient, the following assumptions are made:

1. Initial reactor power, pressure, and reactor coolant system temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 3.

2. A positive moderator temperature coefficient of reactivity for BOL (+7 pcm/°F) is assumed 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-only power coefficient of reactivity is assumed such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the manual mode in order to prevent rod insertion due to an increase in RCS temperature prior to reactor trip. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure will prevent the reactor protection system from functioning properly.

3.2.3.3 Results

The system response to an accidental RCS depressurization is shown in Figures 3.2.3-1 and 3.2.3-2. Nuclear power increases slowly from the initial value until reactor trip occurs on overtemperature)T. The DNBR decreases initially, but increases rapidly following the trip, as shown in Figure 3.2.3-1. The DNBR remains above the safety analysis limit value throughout the transient.

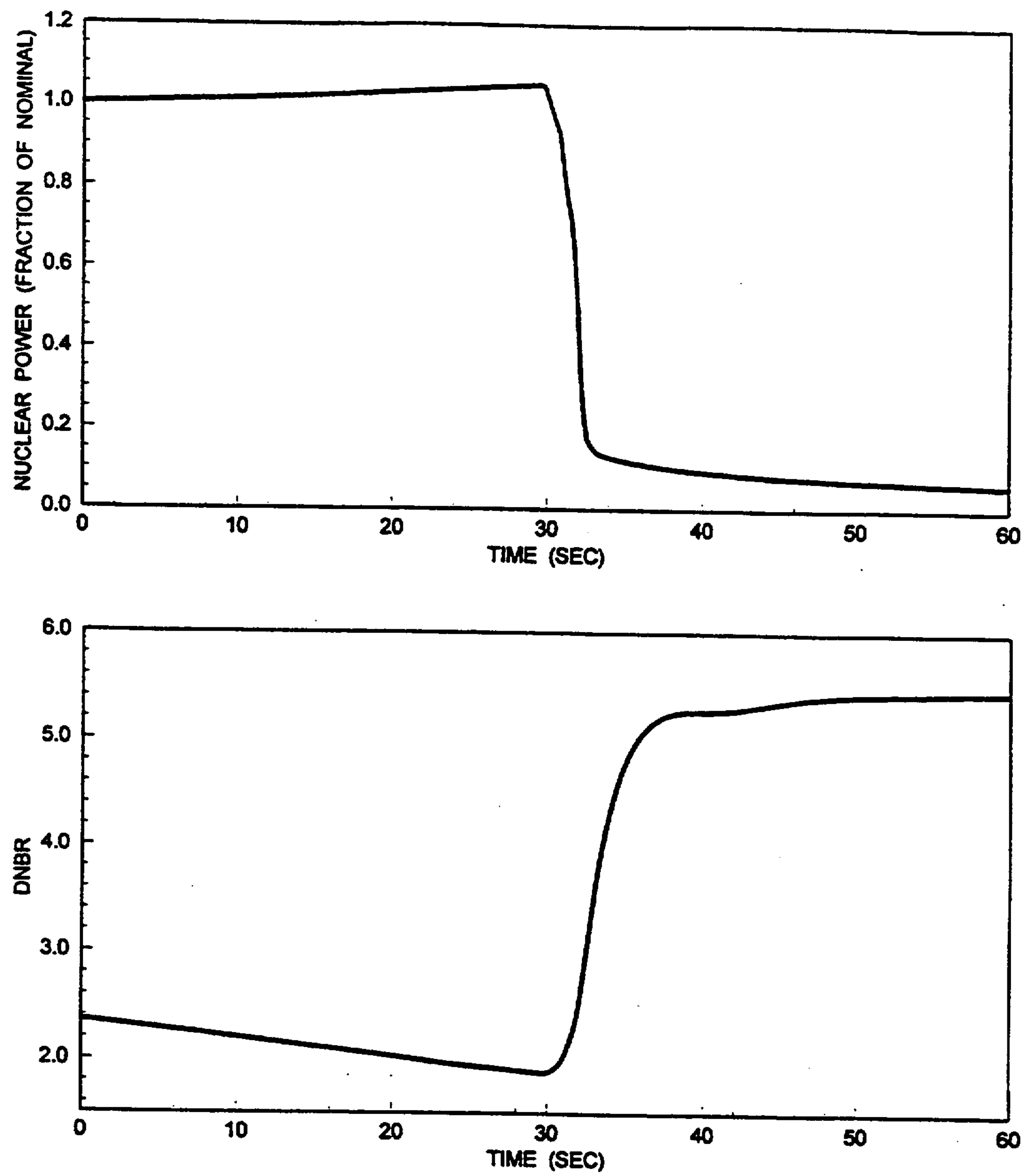
The calculated sequence of events for the accidental depressurization of the RCS incident is shown in Table 3.2-2.

Table 3.2-1	Classification of Non-LOCA Events for Unit 1 Uprating Evaluation			
FSAR Update Section	Non-LOCA Safety Analysis Event/Calculation	A. Zero Power or N/A	B. Uprate Bounded	C. Uprate Not Bounded
15.1.3	OTΔT / OPΔT Reactor Trip Setpoint Calculations			✔
15.2.1	RCCA Bank Withdrawal from a Subcritical Condition	✔		
15.2.2	RCCA Bank Withdrawal at Power		✔	
15.2.3	RCCA Misoperation		✔	
15.2.4	Uncontrolled Boron Dilution	✔		
15.2.5	Partial Loss of Forced Reactor Coolant Flow		✔	
15.2.6	Startup of an Inactive Reactor Coolant Loop		✔	
15.2.7	Loss of External Electrical Load/Turbine Trip *		✔	
15.2.8	Loss of Normal Feedwater		✔	
15.2.9	Loss of Offsite Power to the Station Auxiliaries		✔	
15.2.10	Excessive Heat Removal due to FW System Malfunctions	✔	✔	
15.2.11	Sudden Feedwater Temperature Reduction		✔	
15.2.12	Excessive Load Increase		✔	
15.2.13	Accidental Depressurization of the Reactor Coolant System			✔
15.2.14	Accidental Depressurization of the Main Steam System	✔		
15.2.15	Spurious Operation of the Safety Injection System at Power		✔	
15.3.4	Complete Loss of Forced Reactor Coolant Flow		✔	
15.3.5	Single RCCA Withdrawal at Full Power		✔	
15.4.2.1	Rupture of a Main Steam Line	✔		
15.4.2.2	Major Rupture of a Main Feedwater Pipe		✔	
N/A	Steam Line Break at Full Power (core DNB analysis)			✔
15.4.4	Single Reactor Coolant Pump Locked Rotor		✔	
15.4.6	RCCA Ejection	✔	✔	

* DNB case; the overpressure analysis is PG&E responsibility

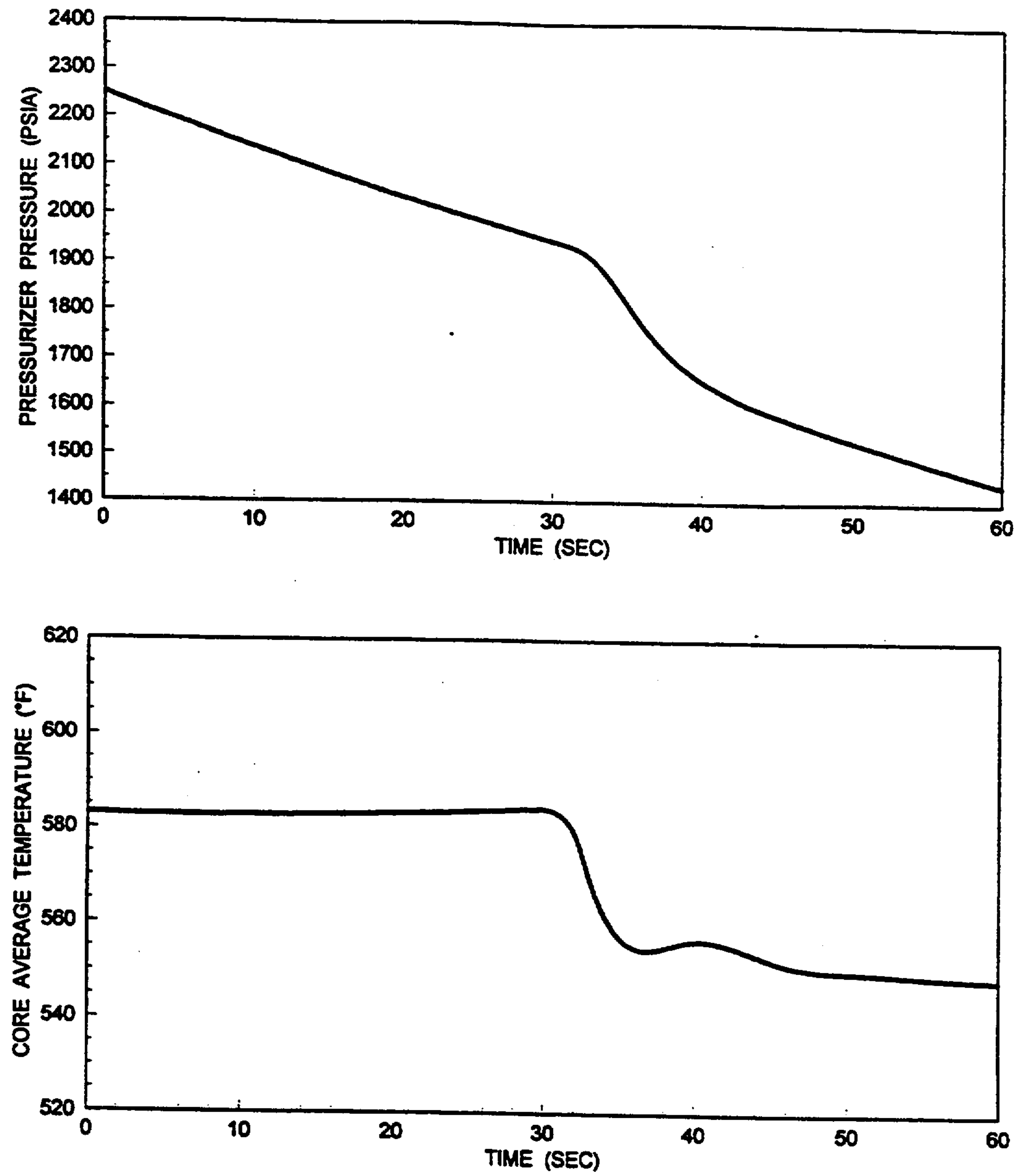
Table 3.2-2 Unit 1 Uprating Non-LOCA Analyses - Time Sequence of Events

Accident	Event	Time (sec)
Accidental Depressurization of the RCS (Section 3.2.3)	Inadvertent opening of a single pressurizer safety valve	0.0
	Overtemperature ΔT reactor trip setpoint reached	27.5
	Rods begin to fall into core	29.5
	Minimum DNBR occurs	29.8
Steam Line Break at Full Power (Section 3.2.4)	Steam line ruptures, 0.53 ft^2	0.0
	Peak core heat flux occurs (minimum DNBR)	59.4



Diablo Canyon Units 1 and 2

Figure 3.2.3-1 **Nuclear Power and DNBR Transients for Accidental Depressurization of the Reactor Coolant System**



Diablo Canyon Units 1 and 2

Figure 3.2.3-2 Pressurizer Pressure and Core Average Temperature Transients for Accidental Depressurization of the Reactor Coolant System

3.2.4 Steam Line Break at Full Power

The steam line break analysis documented in the updated FSAR section 15.4.2.1 assumes zero power initial conditions, and demonstrates that the DNB design basis is met for this accident following a reactor trip. The steam line break at full power initial conditions analysis to demonstrate core integrity is not explicitly documented in the updated FSAR. An analysis of this event was performed to support an assumed increase in the OPΔT reactor trip response time for the RTD Bypass Elimination modification that was performed in conjunction with the Eagle 21 process protection system upgrade (Reference 2). The currently applicable analysis of this event is performed separately for each Diablo Canyon unit, and as such does not bound the uprated Unit 1 plant conditions. A new analysis was performed using conservative assumptions that bound both units. The transient results are less limiting than the previous analyses, due to the use of a higher setpoint for the low steam line pressure safety injection actuation which results in an early reactor trip for a larger range of break sizes. Previously, a very conservatively low setpoint (14.7 psia) was assumed in order to allow flexibility to potentially revise this setpoint at the plant, which never occurred. Use of a higher but still conservative setpoint (459 psia) reduces the size of the largest break that will not trip on low steam pressure SI actuation. This in turn reduces the peak core power that is achieved for the worst case, which will result in a higher minimum DNBR than in previous analyses. Based on a comparison of the transient results as described above it is concluded that the DNB design basis is met. The DNBR is confirmed for this event using cycle-specific core parameters as part of the reload safety evaluation.

3.2.5 Non-LOCA Conclusions

Based on the evaluations and analyses described above, it is concluded that all applicable safety criteria are met and the conclusions of the Diablo Canyon updated FSAR remain valid for the Unit 1 non-LOCA events for the uprated power conditions.

3.2.6 References

1. Ellenberger, S. L., et al., "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8745-P-A (Proprietary) and WCAP-8746-A (Non-Proprietary), September 1986.

2. "Summary Report, Eagle 21 Process Protection System Upgrade for Diablo Canyon Power Plant Units 1 and 2," WCAP-13615-R2, June 1993.

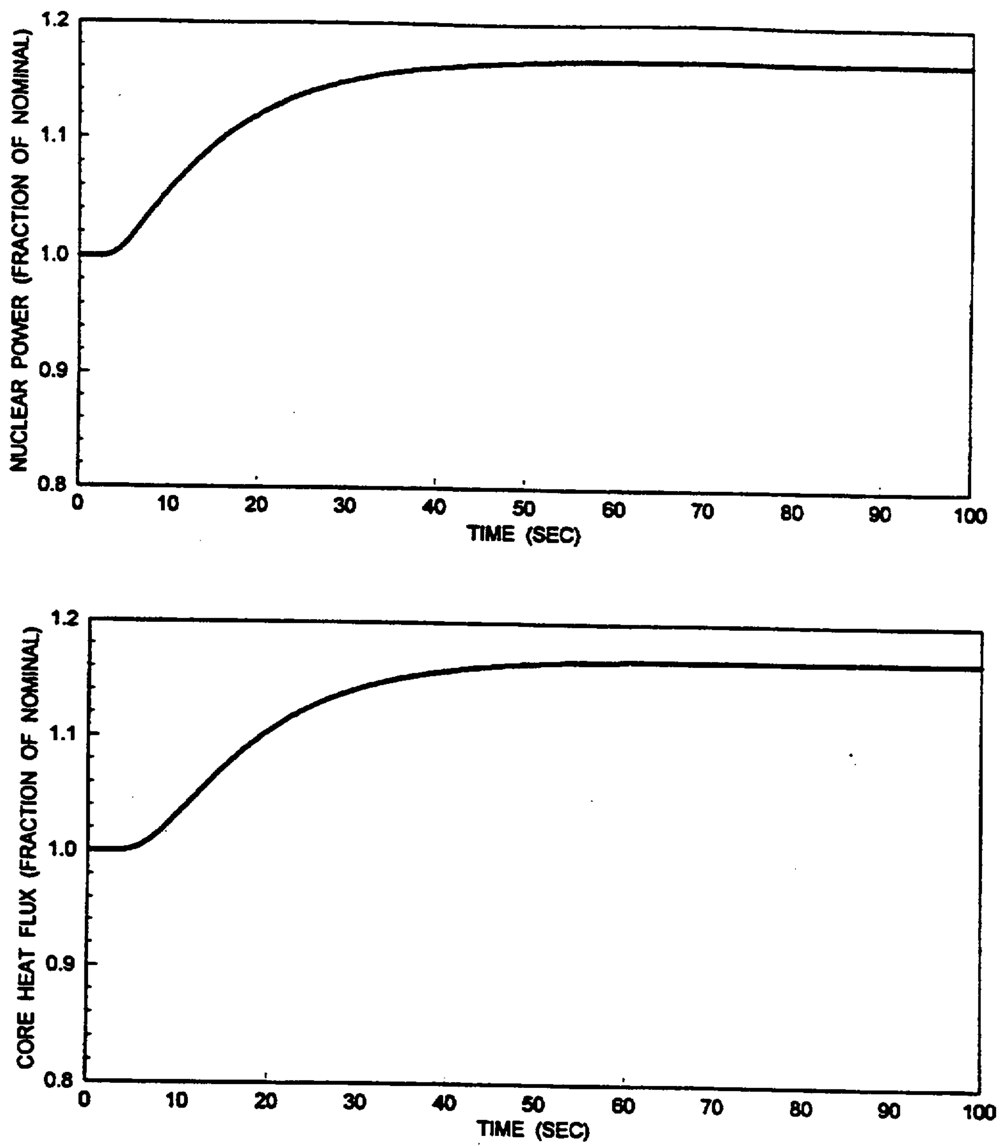


Figure 3.2.4-1 Nuclear Power and Core Heat Flux Transients for Main Steam Line Rupture at Full Power, 0.53 ft² Break

3.3 CONTAINMENT INTEGRITY ANALYSES

3.3.1 Main Steam Line Break (MSLB) Mass and Energy (M&E) Releases

An evaluation has been completed in support of the Power Uprating Program for Diablo Canyon Power Plant Unit 1 to an NSSS power of 3425 MWt. Based on the results of the evaluation, it was concluded that the current safety analyses of record are bounding for the following three licensing basis calculations.

- Main Steamline Break Mass and Energy Releases Inside Containment
- Main Steamline Break Mass and Energy Releases Outside Containment
- Steam Mass Releases for use in a Radiological Consequences Evaluation

Thus, the mass and energy release rates inside containment for the containment integrity evaluation, and outside containment for environmental equipment qualification evaluation remain valid. In addition, the definition of the limiting break size and power level for each set of inside- or outside-containment steamline break analyses also remains valid for the uprating conditions. Similarly, the steam releases calculated following a Loss of Load/Turbine Trip or Steamline Break event for input to a radiological evaluation, remain valid and conservative.

3.3.2 LOCA Mass and Energy (M&E) Releases

The LOCA mass and energy releases for the double-ended hot leg break and the double-ended pump suction break that were presented in Reference 1 were generated at a core power of 3411 MWT plus a two (2) percent calorimetric uncertainty. This was done in order to bound both Units 1 and 2. The LOCA mass and energy releases for the double-ended hot leg break and the double-ended pump suction break that were presented in Reference 1 were generated based on the following limiting conditions in order to bound both Units 1 and 2.

- Loss of Offsite Power Coincident with the Pipe Rupture at t = 0 seconds

- Initial Core Power of 3411 MWt plus 2% Calorimetric Uncertainty
  (the NSSS power is not used because the RCP heat is not pertinent for decay heat calculations)

The performance parameters for the Diablo Canyon Power Plant Unit 1 Uprating Program were reviewed with respect to the performance parameters for Unit 2 with and without the increased pump heat input. The comparison of the Unit 2 parameters without additional pump heat and the uprated Unit 1 parameters shows that the temperatures for Unit 2 are higher, and therefore, more conservative than the temperatures for the uprated Unit 1.

Thus, the LOCA mass and energy releases documented in Reference 1 remain both valid and bounding for use with the containment integrity analyses.

3.3.3 Containment Response

3.3.3.1 Containment Integrity Analyses

The Diablo Canyon Unit 1 containment system is designed such that for all high-energy line break sizes, up to and including the double-ended severance of a reactor coolant pipe or secondary system pipe, the containment peak pressure remains below the design pressure of 47.0 psig with adequate margin. The containment response analysis demonstrates the acceptability of the containment safeguards systems to mitigate the consequences of a high-energy line break inside containment. The impact of hypothetical Main Steam Line Break (MSLB) or Loss-of-Coolant Accident (LOCA) mass and energy releases on the containment pressure is addressed to assure that the containment pressure remains below its design pressure at the uprated core power conditions for 3411 MWt.

The safety systems that are explicitly modeled in the containment integrity analyses as providing a heat removal function or supplying cooling flow include:

a. Safety Injection Pumps
b. Containment Fan Cooler Units
c. Containment Spray Pumps
d. Residual Heat Removal Heat Exchangers
e. Component Cooling Water (CCW) Heat Exchangers
f. Auxiliary Salt Water Pumps
g. CCW Pumps
h. Accumulators
i. Passive Containment Heat Sinks
j. High Pressure Containment Setpoints

It should be noted that the safety systems also include delays due to trip function, signal processing and diesel sequencing.

The current design basis analyses for the LOCA and MSLB containment integrity for Diablo Canyon Units 1 and 2 are documented in References 1 and 2, respectively. All of the cases were performed at the power level of Unit 2 (3411 MWt) because that power level and associated thermal conditions would bound both units. Therefore, the current containment integrity analysis of record is bounding for Diablo Canyon Unit 1 with a core power uprating to 3411 MWt. The peak calculated pressure resulting from the limiting MSLB transient that is documented in Reference 2 is 42.25 psig. The peak calculated pressure documented in Reference 1 for the limiting LOCA case is 41.53 psig. The initial conditions and safety system parameters that support these results are identified in References 1 and 2.

3.3.3.2 Component Cooling Water Heatup Analyses

The CCW Heatup Analyses demonstrate that the CCW system can maintain its intended cooling function during a design basis accident. The current analyses that support the operability of the CCW system during post-accident scenarios are documented in Reference 3. The analyses that were performed for Diablo Canyon Units 1 and 2 are documented in Reference 3, and considered the Unit 2 of core power level of 3411 MWt. Therefore, the CCW heatup analyses in Reference 3 are bounding for the uprating of Diablo Canyon Unit 1 to a core power of 3411 MWt.

3.3.3.3 Subcompartment Analysis

It was determined that the short-term subcompartment analysis would not be affected by the Uprating Program for Diablo Canyon Power Plant Unit 1 because the primary effect for short-term releases is governed by RCS pressure and temperature, not power level. Based on the data presented in Table 2.1-1, the conservative direction for short-term releases is low temperatures and high pressures. The current design basis subcompartment analysis are bounding because the RCS pressure is not increasing and the minor temperature increase actually reduces the releases by an insignificant amount.

3.3.4 Environmental Qualification

Environmental Qualification (EQ) is based on expected temperature and pressures resulting from accident conditions. For example, components within the containment that are relied upon for LOCA or Main Steam Line Break mitigation, must be qualified to perform their function in the hot, moist, and potentially radioactive post-accident atmosphere (up to 47 psig, 100% humidity, and about 250°F for LOCA and 347°F for MSLB).

Currently, the EQ curves of pressure, temperature and radiation levels are identical between Units 1 and 2, and are based on a Unit 2 model. The Unit 2 model is based on the higher power level that the Unit 1 uprate program is seeking to justify. Thus, the EQ curves continue to be applicable to Unit 1 in the uprated condition.

3.3.5 Conclusions

Based on the evaluations that have been performed for the Uprating Program for Diablo Canyon Power Plant Unit 1 from a core power of 3338 MWt (NSSS power = 3350 MWt) to a core power of 3411 MWt (NSSS power = 3425 MWt), the current design basis analyses for LOCA mass and energy releases, MSLB mass and energy releases, steam mass releases for dose considerations, containment integrity, short-term subcompartment analysis, and CCW system overheating remain valid and bounding.

References:

1. WCAP-13907, "Analysis of Containment Response Following Loss-of-Coolant Accidents for Diablo Canyon Units 1 and 2," December 1993.
2. WCAP-13908, "Analysis of Containment Response Following Main Steamline Break Accidents for Diablo Canyon Units 1 and 2," December 1993.
3. WCAP-14282, "Evaluation of Peak CCW Temperature Scenarios for Diablo Canyon Units 1 and 2," March 1995.

3.4 STEAM GENERATOR TUBE RUPTURE

The steam generator tube rupture (SGTR) design basis for Diablo Canyon Power Plant Units 1 and 2 includes an analysis to demonstrate margin to steam generator overfill. A reanalysis of the margin to steam generator overfill for revised auxiliary feedwater and power-operated relief valve (PORV) flow rates is presented in PGE-92-685 (Reference 1).

The SGTR analysis presented in WCAP-11723 (Reference 2) was performed to bound both Diablo Canyon Power Plant Units 1 and 2. Per WCAP-11723, the SGTR evaluation was performed using the limiting parameters for either Unit 1 or Unit 2 such that the analysis was applicable for both units. The analysis was performed at 102% of 3423 MWt with a nominal thermal design flow of 88,500 gpm per loop with up to 15% steam generator tube plugging (SGTP) and a Reactor Coolant System (RCS) average temperature of 577.6°F.

The uprated power level of 3425 MWt and the associated design parameters for Diablo Canyon Power Plant Unit 1 are evaluated below for the SGTR event. Unit 1 will operate at 3425 MWt (NSSS power), a thermal design flow of 87,700 gpm per loop with up to 15% steam generator tube plugging and a RCS average temperature of 577.3°F.

The rated thermal power level of Diablo Canyon Power Plant Unit 2 was used in the licensing basis analysis since this represented a conservatively high power level for the SGTR analysis. The increase of 2 MWt in the pump power was previously evaluated and was determined that the impact is insignificant to the transient. Therefore, the NSSS power for Unit 1 of 3425 is bounded by the SGTR analyses of record in Reference 1.

The difference in the thermal design flows between the Diablo Canyon Power Plant Units 1 and 2 is less than 1%. A small reduction in the thermal design flow assumed in the SGTR analyses would not have a significant impact on the results, since the SGTR is not a DNB related transient. Additionally, the reactor trip time will not significantly change due to this small reduction in thermal design flow. Therefore, the SGTR analysis of record would bound the thermal design flow rate of Unit 1.

The difference in the RCS average temperature of 0.3 F between Diablo Canyon Power Plant Units 1 and 2 would slightly delay the reactor trip time. Since earlier reactor trip is conservative for the SGTR analyses, a higher T_{AVG} is also conservative. Earlier reactor trip results in earlier auxiliary feedwater actuation for the overfill case. Therefore, the use of the Unit 2 RCS average temperature is conservative and bounds the Unit 1 uprating parameters.

References:

1. PGE-92-685, "SGTR Margin to Overfill Re-Analysis," October 13, 1992.

2. WCAP-11723, "LOFTTR2 Analysis for a Steam Generator Tube Rupture for the Diablo Canyon Power Plant Units 1 and 2," February 1988.

3.5 RADIOLOGICAL DOSE ANALYSIS

3.5.1 Steam Generator Tube Rupture

The steam generator tube rupture (SGTR) design basis for Diablo Canyon Power Plant Units 1 and 2 includes an analysis to demonstrate that the calculated offsite radiation doses are less than the allowable guideline values. The offsite radiation dose analysis is documented in WCAP-11723 (Reference 1).

The SGTR analysis presented in WCAP-11723 was performed to bound both Diablo Canyon Power Plant Units 1 and 2. Per WCAP-11723, the SGTR evaluation was performed using the limiting parameters for either Unit 1 or Units 2 such that the analysis is applicable for both units. The analysis was performed at 102% of 3423 MWt with a nominal thermal design flow of 88,500 gpm per loop with up to 15% steam generator tube plugging (SGTP) and a Reactor Coolant System (RCS) average temperature of 577.6°F.

The uprated power level of 3425 MWt and the associated design parameters for Diablo Canyon Power Plant Unit 1 are evaluated below for the SGTR event. Unit 1 will operate at 3425 MWt (NSSS power), a thermal design flow of 87,700 gpm per loop with up to 15% steam generator tube plugging and a RCS average temperature of 577.3°F.

The difference in RCS average temperature of 0.3°F between Diablo Canyon Power Plant Units 1 and 2 would slightly delay the reactor trip time. Earlier reactor trip results in earlier steam releases to the environment for the offsite radiological dose case. Therefore, the use of the Unit 2 RCS average temperature is conservative and bounds the Unit 1 uprating parameters.

The methodology and assumptions employed in the radiological offsite doses for the Diablo Canyon SGTR analysis are documented in WCAP-11723. The core coolant activities used in the radiological analysis were performed for a core power level of 3568 MWt, which bounds the 102% power assumptions in the SGTR analysis. Since there is no change in the thermal and hydraulic analysis results of primary to secondary break flow, flashed break flow, and steam released to the atmosphere, there is no impact on the radiological consequences.

3.5.2 Dose Assessment Source Terms

An evaluation of the radiological source terms was performed to assess the impact of DCPP Unit 1 power uprating to Unit 2 rated power. The current source term is based on a reactor power level of 3568 MW, which is about 105% of Unit 2 rated thermal power (Table 11.1-2 of the DCPP FSAR Update). Therefore, a power uprating of Unit 1 to the Unit 2 power of 3411 MW, has no impact on radiological source terms for either the design basis accidents or normal plant operation. Thus, the Unit 1 uprate will not affect Radiological Dose Analysis.

Reference:

1. WCAP-11723, "LOFTTR2 Analysis for a Steam Generator Tube Rupture for the Diablo Canyon Power Plant Units 1 and 2," February 1988.

3.6 POST-LOCA HYDROGEN GENERATION

The hydrogen analysis for Diablo Canyon Power Plant Unit 1 was updated to reflect the Uprating Program operating conditions. The new hydrogen analysis takes into consideration the following changes as a result of the Uprating Program:

- Power reduction from 3580 MWt to 3425 MWt;
- Containment free volume reduction from 2,680,000 ft^3 to 2,550,000 ft^3;
- Post-accident containment temperature profile identified in WCAP-13907; and
- Percent of zirconium associated with zircaloy-water reaction increase from 1.5% to 5.0%.
- RCS hydrogen concentration increase from 35 cc/kg to 50 cc/kg.

The results of the new hydrogen analysis indicate that a 100 scfm hydrogen recombiner, started when the bulk containment concentration reaches 3.5% by volume (at approximately 4 days), or earlier, will ensure the bulk containment hydrogen concentration will not reach the lower flammability limit of 4% by volume.

4.0 FLUID AND AUXILIARY SYSTEMS

4.1 PRIMARY FLUID SYSTEMS

This section addresses the impact of the Uprating Program for Diablo Canyon Power Plant Unit 1 on the ability of the Reactor Coolant System (RCS) and the auxiliary fluid systems to perform their required functions.

In order to support the operation of Diablo Canyon Power Plant Unit 1 at the Uprating Program conditions, the following systems were evaluated at the new conditions: 1) Reactor Coolant System (RCS), 2) Residual Heat Removal System (RHR), 3) Chemical and Volume Control System (CVCS), and 4) Containment Spray System. A brief description of each system is provided below.

4.1.1 Reactor Coolant System

The following "proof-of-design" calculations for the Reactor Coolant System (RCS) were reviewed to determine that the calculations bound Diablo Canyon Power Plant Unit 1 at the uprated power level.

4.1.1.1 Pressurizer Relief Tank Level Setpoints

The low level alarm and the high level alarm setpoints for the Diablo Canyon Unit 1 Pressurizer Relief Tank (PRT) were previously determined by Westinghouse. The basis of the alarm setpoints is described below.

The PRT low level alarm is based on the minimum volume of 120°F water required to completely quench a design discharge from the pressurizer and maintain the final temperature less than or equal to 200°F. The design discharge to the PRT is 110% of the full power steam volume in the pressurizer.

The PRT high level alarm is based on the maximum volume of water at 120°F that will completely quench a design discharge from the pressurizer while maintaining the pressure in the PRT less than or equal to 50 psig.

The full load pressurizer steam volume is 40% of 1800 ft^3. Since the pressurizer level program has not changed for the uprating conditions in Diablo Canyon Power Plant Units 1 and 2, the full load steam volume, upon which the PRT design discharge is based, is unchanged. Hence, the original calculation for the PRT level setpoints remains applicable for the uprated conditions.

4.1.1.2 Pressurizer Relief Line Pressure Drop

The maximum backpressure that will exist at the outlet of the Diablo Canyon Power Plant Unit 1 pressurizer safety valves when the valves are relieving at their maximum relieving capacity was previously determined by Westinghouse. The backpressure at the safety valve outlet must be limited to a maximum of 500 psia.

The performance of the pressurizer safety valve discharge piping system in limiting the backpressure at the outlet of the safety valves to 500 psia was determined based on the safety valves discharging at the maximum relieving capacity of 420,000 lb/hr per valve. The existing pressurizer safety valves remain adequate for the uprated conditions in Units 1 and 2, and Westinghouse is unaware of any changes in the configuration of the discharge lines that would affect the backpressure at the safety valve outlet at the specified relieving flow rate of the valves. For these reasons, the original safety valve discharge line "proof-of-design" calculation remains applicable to the uprated conditions.

4.1.1.3 Pressurizer Spray Flow Capability

A side stream of reactor coolant from two cold legs is diverted to the top of the pressurizer where it is sprayed into the steam space of the pressurizer, thereby condensing steam and controlling the pressure within the pressurizer. For a given configuration of pressurizer spray line piping and spray flow control valves, the spray flow rate is determined by the pressure drop from the spray flow scoop on the cold leg to the pressurizer surge connection on the hot leg. The design total pressurizer spray flow rate for Diablo Canyon Power Plant Unit 1 is 800 gpm.

A total pressurizer spray flow rate of 800 gpm is achievable for Unit 1 based on realistic RCS flow rates. The limiting analysis for total pressurizer spray flow rate is full load rejection. Therefore, if the flow rate of 800 gpm is not achieved, the ability to load reject is lessened. However, there is no safety impact because (1) full load rejection is not a safety function and (2) load rejection for Diablo Canyon is considered unlikely and unnecessary.

At the uprated conditions for Diablo Canyon Power Plant Unit 1, the best estimate RCS loop flow rates are 94,400 gpm with 0% steam generator tube plugging and 91,500 gpm with 15% tube plugging. Under either of these conditions, the loop pressure drops (and the driving force for pressurizer spray flow) will be greater than those calculated previously, since the best estimate flow rate for the uprated conditions is greater than that previously assumed. Hence, the original pressurizer spray flow "proof-of-design" calculation conservatively bounds the uprated conditions.

4.1.1.4 RCS Loop Pressure Drops

Best estimate RCS loop flow rates are calculated by balancing the head of the reactor coolant pump (RCP) and the system hydraulic losses to determine an operating point on the RCP head curve. The system pressure drops consist of the reactor core delta P, reactor vessel nozzles delta P, the reactor internals delta P, the steam generator primary side delta P, and the RCS loop piping delta P. The reactor core delta P coefficient and steam generator delta P coefficient (depending on the tube plugging level) can be affected by the uprated plant conditions. However, the delta P coefficient of the remainder of the RCS will not be affected by the uprating of Diablo Canyon Power Plant Unit 1.

It was determined that for the range of design conditions considered for the Uprating Program, the estimated flow will continue to meet the technical specification minimum measured flow requirements.

4.1.2 Residual Heat Removal System

The Residual Heat Removal System (RHRS) is designed to remove residual and sensible heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown. The RHRS is placed in service when the temperature of the reactor coolant has been reduced to approximately 350°F by steam generator cooling following a reactor shutdown.

The design of the RHRS includes two residual heat removal pumps and two residual heat exchangers. In a normal plant cooldown, both trains of the RHRS are used to cool the RCS at a rate that is consistent with the cooldown rate permitted by the Diablo Canyon Technical Specifications. However, only one RHR train can be used for plant cooldown, which will extend the time that is required to cool the RCS to 140°F.

The Westinghouse computer code that is used to calculate the primary system cooldown transient integrates the heat transfer capability of both the RHRS and the Component Cooling Water System (CCWS) into a unified model that considers all the heat loads on the CCWS and the temperature of the ultimate heat sink (Auxiliary Saltwater System). The normal two-train and single train cooldown performance of the Diablo Canyon Power Plant Unit 1 RHRS was analyzed using this model to confirm that the actual cooldown times are within the allowable limits based on the higher decay heat load that is associated with the uprated power level.

System and component parameter inputs to the RHR cooldown analysis are at the uprated power level of 3425 MWt. Included among these inputs are the auxiliary heat loads on the CCWS at 4 hours and 20 hours after reactor shutdown. These heat loads were revised as part of the uprating program. These heat loads also form an input to the evaluation of the CCWS for the uprating.

In addition to the auxiliary heat loads described above, the cooldown analysis was based on the major system design parameters that are applicable to the uprated plant conditions. The results this evaluation indicate that RCS cooldown to 140°F using two cooling trains is achieved at 17.4 hours after shutdown. This calculation used the design CCW flow rate to the RHR heat exchanger (5000 gpm). With two cooling trains in service, a minimum CCW flow rate to each RHR heat exchanger of 4438 gpm is required to accomplish cooldown in the allowable time of 20 hours after shutdown.

For cooldown using only one cooling train, RCS cooldown to 200°F is achieved at 29.2 hours after shutdown, based on the design CCW flow rate to the RHR heat exchanger (5000 gpm). With one cooling train in service, a minimum CCW flow rate to the RHR heat exchanger of 4076 gpm is required to accomplish cooldown in the allowable time of 36 hours after shutdown.

From the results of the two-train RHR cooldown calculations, it is concluded that the RHRS remains capable of cooling the RCS to the required final temperature in the allowable length of time at the uprating plant conditions.

4.1.3 Chemical and Volume Control System

4.1.3.1 Heat Exchanger Performance

The Chemical and Volume Control System (CVCS) design bases were reviewed to determine the effect, if any, of NSSS operation at the uprating conditions. The CVCS heat exchanger specification sheets were reviewed and compared to the uprated operating conditions to determine if any CVCS heat exchangers are potentially affected by revised RCS temperatures resulting from plant operation at the uprating plant conditions. The results of the evaluation indicate that the maximum Regenerative Heat Exchanger and Excess Letdown Heat Exchanger inlet temperatures that will occur for the uprated plant conditions are bounded by the original heat exchanger design conditions. Since the temperatures associated with the plant uprating are less than those for which the heat exchangers have been designed and there are no changes in the letdown, charging or excess letdown flow rates, the performance of the Regenerative Heat Exchanger and Excess Letdown Heat Exchanger will remain acceptable under the uprating conditions. Similarly, there will be no effect on the heat loads imposed by these heat exchangers on the Component Cooling Water System under the uprated conditions.

It was also determined that the performance of the Non-Regenerative Heat Exchanger, which is downstream of the Regenerative Heat Exchanger, and the Seal Water Heat Exchanger, which is downstream of the Excess Letdown Heat Exchanger, will also remain acceptable for the uprated conditions. This conclusion is based on the fact that there will be no changes in the letdown flow rate, excess letdown flow rate, component cooling water flow rate or component cooling water supply temperature under the uprated conditions compared to the previously analyzed conditions.

4.1.3.1 Boration for Shutdown

The CVCS design basis includes the function of borating the RCS to attain the required shutdown conditions. The borated water storage requirements and system boration flow rates for meeting shutdown requirements are not directly affected by the core power rating and are evaluated for each cycle as part of the reload safety evaluation. Therefore, since power level does not impact the requirement, there is no system evaluation of this capability performed specifically for the Uprating Program.

4.1.4 Containment Spray System

The Containment Spray System (CSS) is a safeguards system that mitigates the peak pressure inside containment and removes radioactive fission products from the containment atmosphere following design basis accidents (LOCA and MSLB). The performance of the CSS is unaffected by the uprating plant parameters since the system was designed for the (higher) Unit 2 plant power rating. However, PG&E requested that Westinghouse review the Unit 1 and Unit 2 CSS components for similarity. The applicability of the CSS performance analysis of record was determined for both units on the basis of the similarity of major components.

A review of the design parameters of the major CSS components indicates that all components were designed and procured identically for Diablo Canyon Power Plant Units 1 and 2. An investigation of any design modifications, repairs or replacement that may have been implemented on these components after the plant began commercial operation, which could affect the similarity of the components, was beyond the scope of this review. Also, differences between units in the layout and the installation of the equipment in the CSS that could affect system performance, such as differences in elevation or pipe routing, were not examined in this review.

4.1.5 Primary Water Chemistry

The Unit 1 Uprating will result in more energy in the core and potentially higher boron concentrations at the beginning of fuel cycles. The increased boron will be associated with an increase in lithium, which in turn could potentially have a small effect on pressurized water stress corrosion cracking on the primary side of the steam generator tubes; however, the Unit 1 primary chemistry will not be significantly different from that in Unit 2.

The changes in water chemistry due to the uprating are anticipated to be very small and are enveloped by similar consequences for extending the operating cycle to 24 months. The safety evaluation for the extended fuel cycles considered the impact of increased lithium levels and concluded that the increased lithium levels are acceptable. Tube cracking inspections are performed at sufficient intervals such that tube cracking will be discovered prior to leakage.

Primary water chemistry, especially with regard to its effect on steam generator life, is closely monitored and controlled at DCPP. The primary water chemistry for Unit 1 should not be noticeably different from chemistry conditions for Unit 2.

4.1.6 Conclusion

The designs of the NSSS fluid systems were reviewed to confirm their continued ability to meet the applicable design basis functional and performance requirements at the uprated plant conditions. As a result of the review, it was concluded that the designs of the NSSS fluid systems remain adequate for plant operation at the uprated conditions of Diablo Canyon Power Plant Units 1 and 2.

4.2 COOLING WATER SYSTEMS

The increase in thermal power potentially affects systems designed to remove unwanted heat. This section discusses the component cooling water system, the auxiliary salt water system, and the spent fuel pool cooling system. The condenser's Circulating Water System is discussed in Section 7.3.

4.2.1 Component Cooling Water System

The hardware in the component cooling water system is the same in Unit 1 as in Unit 2. The heat load on this system should not be significantly changed during normal operation. Since the reactor coolant pumps and the letdown line takeoff are both located downstream of the steam generators, the load on CCW from the RCP coolers, and the seal injection cooler, and the nonregenerative heat exchanger should be no greater than before the uprating. Since reactor T_{AVG} is at most 1.3°F higher than before the uprating (and will be decreased if T_{HOT} is held constant), the load on CCW from the fan coolers during normal operation should not be changed significantly. The LOCA analysis for DCPP was recently redone; this new LOCA analysis resulted in a reduced peak containment pressure. This analysis will remain bounding for the uprated Unit 1 as discussed in Section 3.3. Hence, it is concluded that the CCW is capable of supporting the Unit 1 uprating.

4.2.2 Auxiliary Salt Water System

As in the case of the CCW system, the hardware in the auxiliary salt water system is the same in Unit 1 as in Unit 2. The heat load on these systems should not be significantly changed during normal operation. Since reactor power is increased just 2.2%, the load on ASW from CCW during operation should not be changed significantly. Ample margin exists in the ASW cooling capability as demonstrated in WCAP 12526, Revision 1, "Auxiliary Salt Water and Component Cooling Water Flow and Temperature Study for Diablo Canyon," dated June 1992. (WCAP-12526 determined that the required ASW flow rate as a function of ocean temperature based on the Unit 2 power level, the same power level to which Unit 1 is being uprated.) The LOCA analysis for DCPP was recently redone and resulted in a reduced peak containment pressure. This analysis will remain bounding for the uprated Unit 1 as discussed in Section 3.3. Hence, the ability of the ASW systems to support the Unit 1 uprating has been demonstrated.

4.2.3 Spent Fuel Pool Cooling System

Studies performed to support License Amendments 104 and 103, which revised Technical Specification 3.9.14.3 to allow storage of 5 weight percent U-235 fuel in the spent fuel pool in anticipation of the 24-month fuel cycle, bound the conditions that will exist with the storage of spent fuel from the uprated Unit 1 reactor. These studies demonstrated that with storage of the specified bounding spent fuel configurations, a critical fuel configuration would not occur, nor would the offsite dose releases from a fuel handling accident exceed our licensing limit. In addition, sufficient margin exists in the spent fuel pool cooling system to adequately remove any slightly greater heat load that might result from the Unit 1 uprating. At DCPP, the spent fuel pool cooling system is not classified as safety related (other than maintaining an RCS boundary), but is still considered to perform an important practical function. Its ability to perform its function will not be challenged by the potentially slightly higher heat load that may eventually result as a consequence of the Unit 1 uprating. In addition, the ability of the Unit 1 fuel pool cooling system to adequately support Unit 1 operation after the uprating is demonstrated by the present ability of the identical fuel pool cooling system of Unit 2 to support Unit 2 operation at the same power level.

4.3 NSSS/BOP INTERFACE SYSTEMS

As part of the Diablo Canyon Unit 1 Uprating Program, the following Balance-of-Plant (BOP) fluid systems were reviewed to assess compliance with Westinghouse Nuclear Steam Supply Systems (NSSS)/BOP interface requirements:

- Auxiliary Feedwater System
- Main Steam System
- Condensate and Feedwater System

The review was performed based on the range of NSSS operating parameters developed to support an NSSS power level of 3425 MWt.

A comparison of the proposed range of NSSS design parameters shown in Table 2.1-1 with the reference operating parameters previously evaluated for systems and components indicates differences that could impact the performance of the above BOP systems. For example, the proposed increase in NSSS power of 2.2 percent (from 3350 MWt to 3425 MWt) would result in about a 2.7 percent increase in steam/feedwater mass flow rates. Additionally, the proposed steam generator tube plugging margin of 15 percent would result in a reduction of full-load steam pressure from 805 psia to 756 psia.

The evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface requirements are delineated below.

4.3.1 Auxiliary Feedwater System

The Auxiliary Feedwater (AFW) System serves as a backup system for supplying feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the heat sink of the steam generators. The system provides an alternate to the Main Feedwater System during startup, hot standby, and cooldown and also functions as an Engineered Safeguards System. In the latter function, the AFW is directly relied upon to prevent core damage and system overpressurization in the event of transients and accidents such as a loss of normal feedwater or a secondary system pipe break. The minimum flow requirements of the AFW are dictated by accident analysis and since the uprating impacts these analyses, evaluations of the limiting transients and accidents are required to determine that the current AFW performance is acceptable at the uprated conditions.

The Westinghouse evaluation reviewed the auxiliary feedwater storage requirements relative to the proposed NSSS operating parameters. The AFW pumps are normally aligned to take suction from the condensate storage tank (CST) and in the longer term can be aligned to the Fire Water Storage Tank (FWST). Sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The total inventory of condensate required to meet a 1 hour hot shutdown period followed by an 8 hour cooldown period was originally determined to be about 222,000 gallons using conservative uprating parameters. This analysis conservatively assumed that the inventory in the steam generators was not only maintained following plant trip but was also restored to the programmed no load level. A new analysis based on the same assumption and the proposed range of NSSS operating parameters at the uprated power level determined that 216,900 gals is required which is bounded by the original analysis requirement of 222,000 gals. Therefore, no change is required to the plant technical specifications which dictates a minimum useable inventory in the FWST and CST of 57,922 and 164,678 gals, respectively, and provide a total useable inventory of 222,600 gals.

4.3.2 Main Steam System

The proposed uprating coupled with the potential reduction in full-load steam pressure to 756 psia impacts main steam line pressure drop. It should be noted that with the current turbine inlet nozzles, the minimum steam generator pressure required to pass 3425 MWt steam through the turbine is 793 psia. This analysis is performed at a steam generator pressure of 756 psia to reflect possible future turbine modifications and provide a conservative assessment of the steam system components. At a steam generator pressure of 756 psia, the full-load steam mass flow rate would increase about 2.5 percent. However, due to the reduced operating pressure and the lower-density steam, the volumetric flow rate would increase by approximately 9.6 percent and steam line pressure drop would increase by approximately 12.4 percent. Note that main steam line pressure drop impacts plant economics, since an increase in pressure drop results in a corresponding increase in plant heat rate over the life of the plant. (An increase in steam line pressure drop of 1 psi is equivalent to an increase of approximately 2 BTU/KW-Hr in plant heat rate.) Initial plant design studies indicated that a pressure drop in the range of 25 to 40 pounds per square inch at rated load provided an acceptable economic balance between the value of a lower heat rate over the life of the plant and the capital cost of larger-bore, longer-length pipes.

Note that the reference NSSS operating parameters for the licensed maximum calculated power (3350 MWt) resulted in a steam line pressure drop of about 40 psi and a pressure of 765 psia at the turbine inlet valves. Based on the range of NSSS operating parameters proposed for the uprating to 3425 MWt, the lowest allowable steam generator pressure would result in a pressure at the turbine inlet valves of approximately 711 psia.

The Westinghouse evaluation reviewed the following major steam system components relative to the proposed NSSS operating parameters:

- Steam Generator Safety Valves
- Steam Generator Power Operated Relief Valves
- Main Steam Isolation Valves and Check Valves

Based on the results of the Westinghouse evaluation, the following conclusions were made relative to the major steam system components listed above.

4.3.2.1 Steam Generator Safety Valves

The Diablo Canyon Power Plant Unit 1 has twenty safety valves with a total capacity of 16,451,144 lb/hr, which provides about 110.3 percent of the maximum calculated steam flow of the 14.91 x 10^6 lb/hr approved for the uprating. Therefore, based on the range of NSSS operating parameters approved for the uprating, the capacity of the installed MSSVs satisfies the Westinghouse sizing criteria.

4.3.2.2 Steam Generator Power Operated Relief Valves

Based on the range of NSSS operating parameters approved for the uprated power level, the installed PORV capacity (1.705 x 10^6 lb/hr at 1020 psia) is about 11.4 percent of the required maximum steam flow (14.91 x 10^6 lbs/hr). Therefore, the PORVs are adequate based on the range of NSSS operating conditions proposed for the Diablo Power Plant Unit 1 Uprating Program.

4.3.2.3 Main Steam Isolation Valves and Check Valves

Rapid closure of the MSIVs following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the main steam system piping and piping supports in the area of the MSIVs. The worst cases for pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables are not impacted by the proposed uprating, the design loads and associated stresses resulting from rapid closure of the MSIVs will not change.

4.3.3 Condensate and Feedwater System

The Condensate and Feedwater System must automatically maintain steam generator water levels during steady-state and transient operations. The proposed range of NSSS operating parameters will result in a required feedwater volumetric flow increase of up to 3 percent during full-power operation. The higher feedwater flow and higher feedwater temperatures will have an impact on system pressure drop, which may increase by as much as 5.7 percent. Also, a comparison of the proposed range of NSSS operating parameters with reference operating parameters indicates that the steam generator full-power operating steam pressure may be decreased by as much as 49 psi (805 psia - 756 psia).

The Westinghouse evaluation reviewed the following major Condensate and Feedwater System components relative to the proposed NSSS operating parameters:

- Feedwater Isolation Valves
- Feedwater Control Valves
- Condensate and Feedwater System Pumps

Based on the results of the Westinghouse evaluation, the following conclusions were made relative to the Condensate and Feedwater System components listed above.

4.3.3.1 Feedwater Isolation Valves

The quick-closure requirements imposed on the Feedwater Isolation Valves (FIVs) and the backup Feedwater Control Valves (FCVs), causes dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam break from no load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the proposed uprating, the design loads and associated stresses resulting from rapid closure of these valves will not change.

4.3.3.2 Feedwater Control Valves

To provide effective control of flow during normal operation, the Feedwater Control Valves (FCVs) are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0-1600 psig). Additionally, rapid closure of the FCVs is required in 7 seconds after receipt of a trip close signal in order to mitigate certain transients and accidents. These requirements are still applicable at the operating conditions for the Diablo Canyon Power Plant Unit 1 Uprating Program.

4.3.3.3 Condensate and Feedwater System Pumps

The hydraulics of the Condensate and Feedwater System in conjunction with the allowable range of feedwater pump speed control should permit operation over the entire range of NSSS operating conditions proposed for uprating. However, to minimize the duty on the feedwater control valves the feedwater pump speed control program will need to be re-set in the event that steam generator full load pressure decreases due to increased steam generator tube plugging.

4.4 RADIOACTIVE WASTE PROCESSING

Based on the performance of the Unit 2 core, it is anticipated that the higher power level in Unit 1 after uprating will not result in a significant change in the quantity of radioactive waste generated in the RCS and removed by the letdown system. A slight increase in the quantity of spent letdown demineralizer resin and depleted letdown filter cartridges, while it should not occur, has no impact on plant safety, and would be acceptable. Similarly, any increase in gaseous radwaste releases should be negligible, and there should be no increase in liquid radwaste. No portions or functions of the radwaste systems are safety-related. Hence, even if there were to be a slight increase in radwaste generation as a result of the uprating, it would have no impact on safe operation of the plant. Other factors not associated with the core power level, such as the quality of fabrication of the fuel assemblies, have a much greater effect on the amount of radioactive waste generated than does the power level at which the core is operated. The impact of the uprating on the radwaste systems will be a secondary effect at most.

4.5 NUCLEAR AND POST ACCIDENT SAMPLING SYSTEM

The Nuclear and Post Accident Sampling Systems provide representative samples of process fluids for radiological and chemical analyses necessary for plant operation, corrosion control monitoring of system equipment and performance, and post-accident assessment. The proposed uprating makes no change in the required sampling or tests. In addition, the systems are essentially identical to the Unit 2 systems that already operate at the uprated power level with the same chemistry.

4.6 CONTAINMENT VENTILATION

The primary function of the Containment Ventilation System during normal operation is to maintain the average containment temperature below the DCPP Technical Specification limit of 120°F. Since the increase in power is being achieved primarily by an increase in steam flow rate, and steam temperature will actually be reduced, and since the average RCS temperature will be increased by no more than 1.3°F, there will not be a significant increase in heat lost to the containment atmosphere as a result of the uprating. The containment fan coolers have more than enough capacity to perform their function, as demonstrated by their present ability to maintain adequate cooling of the Unit 2 containment with only three or four of the five fans normally running. The function of the containment ventilation system during accident conditions is discussed in Section 3.3.

5.0 PRIMARY COMPONENTS

Evaluations and analyses were performed for the NSSS primary and auxiliary components to support the Uprating Program for Diablo Canyon Power Plant Unit 1. The evaluations and analyses were performed for the most limiting cases associated NSSS performance parameters described in Table 2.1-1 for the particular component.

The NSSS components reviewed for the Diablo Canyon Power Plant Unit 1 Uprating Program were as follows:

- Steam Generators
- Pressurizer
- Reactor Vessel
- Reactor Vessel Internals
- Control Rod Drive Mechanisms
- Reactor Coolant Pumps and Motors
- Reactor Coolant Loop Piping and Supports
- Auxiliary Components

5.1 STEAM GENERATORS

The steam generators evaluated are the Model 51 series. Two separate areas of evaluation are addressed for the Model 51 steam generators at the uprated conditions:

- Thermal-hydraulic performance characteristics including moisture separator performance, and
- Structural integrity.

A thermal hydraulic evaluation of the steam generators currently installed in Diablo Canyon Power Plant Unit 1 was performed. The range of operating conditions evaluated included current and uprated power using design and best estimate assumptions. Steam generator thermal/hydraulic operating characteristics at the current and increased thermal rating, with the exception of moisture carryover, have been determined to be acceptable. At the uprated conditions, moisture carryover is projected to exceed 0.25%, but only for the low design steam pressure which results from assuming high fouling and plugging levels. At the current best estimate fouling and plugging levels, moisture carryover is projected to be near or below 0.25%.

Steam generator, thermal-hydraulic operating characteristics were calculated using the GENF Code at the various conditions. Steam pressures and flow rates are used in the calculation of a moisture separator loading parameter. Field data from Diablo Canyon Power Plant Unit 1 and other plants with Model 51 steam generators, are used in conjunction with the loading parameter to project moisture levels at the uprated operating conditions. Values of key characteristics are compared to the design values to demonstrate acceptability. U-bend vibration and wear evaluations were not conducted as part of this uprating evaluation.

Model 51 separator performance is established based on field data. The operating parameters which can have an effect on moisture separator performance are steam flow (power), steam pressure and water level. The projections of this report have shown that the moisture separator performance will be a function of the steam pressure at which the plant is operated. Conditions of high fouling and/or plugging or the need to operate at low primary temperature could cause the moisture to exceed 0.25%. Modifications for the separator systems in Diablo Canyon Power Plant Unit 1 are available. These modifications have been field tested over a range of operating conditions enveloping all the operating conditions considered in this report. The modified separators will deliver a moisture less than 0.15% for all these conditions.

The consequence of excess moisture carryover is turbine blade wear and loss of efficiency. Blade wear is discussed in Section 8.2, Turbine Evaluation.

Several secondary side operating characteristics can be used to assess the acceptability of steam generator operation at uprated conditions. These parameters include circulation ratio, damping factor, secondary mass, heat flux, and secondary side pressure drop. In summary, the thermal-hydraulic operating characteristics of the Diablo Canyon Power Plant Unit 1 steam generators are within acceptable ranges for all anticipated operating conditions for the Uprating Program.

5.1.1 Structural Integrity Evaluation

The bases of the structural and fatigue evaluation of the Diablo Canyon steam generators considers the full duty cycle of events specified in the plant equipment specification as well as the baseline "Steam Pressure Reduction Program" conditions. The critical components of the steam generators that have been evaluated for the uprating design conditions are as follows:

- Tubesheet
- Channel Head and the Divider Plate
- Tubes
- Secondary Side Nozzles
- Secondary Manway

This evaluation assumed that primary side reactor coolant pressure remains unchanged at 2250 psia while steam pressure and steam temperature values decrease to minimum values of 756 psia and 511.7°F, respectively, at 100% thermal power.

In summary, a structural evaluation of the critical components of the steam generator has been performed to demonstrate continued compliance with all applicable regulations for plant operation at the uprated design conditions. The evaluation demonstrates that the critical components of steam generators meet the requirements of ASME Code, Section III, Sub-Section NB at the uprated conditions. For operation at the low pressure conditions, the manway closure bolt replacement interval is reduced from 34 years to 31 years.

5.1.2 Conclusion

The Diablo Canyon Power Plant Unit 1 steam generators have been evaluated for the uprating conditions listed in Table 2.1-1. With the exception of manway closure bolt replacement intervals and the influence of low pressure operation on Moisture Carryover and U-bend vibrations, the steam generators are expected to remain in compliance with the applicable design and analysis criteria at the uprated design conditions identified in Table 2.1-1.

The replacement interval for the steam generator manway closure bolts is reduced from 34 years to 31 years for operation at the uprated conditions. For operation at the uprated power with steam pressures in the range of 805 psia or lower, moisture carryover is expected to exceed 0.25%. For operation with steam pressure less than 760 psia, a review of U-bend vibration is recommended in order to identify tubes which may require preventive action.

5.2 PRESSURIZER

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and pressure and to keep the RCS at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow or outflow to or from the pressurizer as required. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature (T_{SAT}) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool water spray into the steam space at the top of the pressurizer.

The limiting locations from a structural standpoint on the pressurizer are the surge nozzle, the spray nozzle, and the upper shell at the point of spray impingement. The limiting operating condition (relative to the SGTP conditions) of the pressurizer occurs when the RCS pressure is high and the RCS hot leg temperature (T_{HOT}) and cold leg temperature (T_{COLD}) are low. This is explained as follows: Due to inflow and outflow to and from the pressurizer during various transients the surge nozzle alternately sees water at the pressurizer temperature (T_{SAT}) and water from the RCS hot leg at T_{HOT}. If the RCS pressure is high (which means that T_{SAT} is high) and T_{HOT} is low, then the surge nozzle will see maximum thermal gradients and thus experience the maximum thermal stress. Likewise the spray nozzle and upper shell temperatures alternate between steam at T_{SAT} and spray which for many transients is at T_{COLD}. Thus, if RCS pressure is high (T_{SAT} is high) and T_{COLD} is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

The pressurizer analysis performed for the Diablo Canyon Power Plant Unit 1 Uprating Program is based on the NSSS performance parameters provided in Table 2.1-1. The analysis was performed by modifying the previous analysis of record. The models of various components of the pressurizer were subjected to the pressure loads, external loads and the thermal transients for the uprating conditions.

The results of the analysis show that the Diablo Canyon Power Plant Unit 1 and 2 pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III, for the 3425 MWt NSSS Uprating parameters and transients.

5.3 REACTOR VESSEL

An evaluation was performed to assess the impact of the DCPP Unit 1 power uprating on the reactor pressure vessel (RPV). This evaluation is an extension of Reference 1, which evaluated the impact of implementing extended fuel cycle fuel management on RPV integrity and operation. Increasing Unit 1 power from 3338 MWt to 3411 MWt (2.2%) requires a similar neutron flux increase in the reactor core, which will impact the RPV exposure and rate of embrittlement. RPV issues potentially affected by higher embrittlement rates, and requiring evaluation, include RCS Heatup and Cooldown Curve Pressure-Temperature Limits, LTOP setpoints, and Upper Shelf Energy (10CFR50 Appendix G), Pressurized Thermal Shock (10CFR50.61), Surveillance Capsule Withdrawal Schedules (10CFR50 Appendix H), and DCPP Emergency Procedures. The evaluation considered both near term (21-month cycle) and long term (24-month cycle) fuel management, as appropriate, along with higher (2.2%) neutron fast flux levels to account for the impact of uprating Unit 1 from 3338 MWt to 3411 MWt.

5.3.1 Neutron Flux

In Reference 1, a conservative 21-month cycle flux was determined based on the 21Month Cycle Feasibility Study (Reference 2) and the final loading pattern for D2C8 (Reference 3). The peak neutron fast flux calculated at the RPV base-clad interface was 1.45E10 $n/cm^2/s$, occurring at the 45° azimuthal location, and applicable to the RPV plates and circumferential weld. After issuing Reference 1, the DIC9 preliminary core loading patterns (References 4 and 5) were received and used to determine the expected D1C9 (21-month cycle) RPV fast flux, allowing for a power uprate. Both preliminary core loading patterns result in the same peak fast neutron flux value: 1.33E10 $n/cm^2/s$. Since the latest flux projection is less than that used in Reference 1, it is concluded the results from Reference 1 are bounding for the uprated power 21-month cycle case; i.e., implementation of a power uprate will not impact the RPV embrittlement levels with respect to plant operation and regulatory limits in the near-term (Cycles 9-11).

For the long-term, the equilibrium 24 month fuel cycle planned core design (104 feed assembly case) from Reference 6 was evaluated to determine the RPV flux levels (as in Reference 1), which were increased by 2.2% to account for a Unit 1 power uprate. Compared to reference (18-month cycle equilibrium) flux levels, the maximum neutron fast flux would increase 12%. This increase is evaluated below.

5.3.2 RCS Pressure-Temperature Limits

The DCPP Technical Specifications for RCS Heatup/Cooldown Curve pressure-temperature (P-T) limits are valid for 12 EFPY and are based on the maximum expected reactor vessel beltline fluence projected for that period. The 12 EFPY fluences input to the P-T limit analysis (Reference 7) are based on the peak (axial and azimuthal) neutron flux calculated for the Unit 1 and Unit 2 RPV beltline base/clad interfaces, which are 1.45E10 $n/cm^2/s$ for the 1/4 T limiting material (Unit 1 longitudinal weld 3-442C) and 1.48E10 $n/cm^2/s$ for the 3/4 T location limiting material (Unit 2 Intermediate Shell Plate B5454-2), as shown in Reference 8.

The peak RPV fast flux calculated for an uprated Unit 1 with 21-month cycles (1.33E10 $n/cm^2/s$) is less than the value used in developing the P-T limits Technical Specifications (1.45E10 $n/cm^2/s$). Based on this, the current P-T limits would remain valid for Unit 1 operating at 3411 MWt and with 21-month cycles, through 12 EFPY (Cycle 10).

The peak fast flux calculated for an uprated Unit 1 with 24-month cycles (1.56E10 $n/cm^2/s$) is greater than the value used in developing the 12 EFPY P-T limits (1.45E10 $n/cm^2/s$). Based on the higher Unit 1 flux, the P-T limits would need to be revised in order to accommodate uprated, 24-month cycle operation. This is not an issue since the P-T limits will expire (D1C10) prior to implementing 24-month cycles (D1C12). When the P-T limits are recalculated and extended out to 16 EFPY, the higher vessel flux for Unit 1 (uprated, 24-month cycles) will need to be input to the analysis.

The long-term impact of operating at a higher vessel flux on the heatup/cooldown P-T limits is limited by LTOP considerations, and evaluated below.

5.3.3 LTOP Setpoints

Since the Unit 1 uprating to 3411 MWt will not impact the current (12 EFPY) P-T limits as discussed above, and since there are no associated plant hardware changes which would impact the LTOP transient analysis, it follows there is no immediate impact to the current (12 EFPY) LTOP setpoints (enable temperature and PORV pressure setpoint).

Reference 9 evaluated the long-term impact of increasing the Unit 1 RPV fast flux by 12%, on the heatup/cooldown P-T limits and LTOP setpoints. That evaluation is consistent with a Unit 1 power uprate and 24-month cycle operation, since the latter would also result in a 12% flux increase.

The Reference 9 evaluation concluded that the long-term impact of a 12% flux increase was acceptable. The results are summarized here.

Heatup and Cooldown Curves are based on RT_{NDT}, such that a shift increase in RT_{NDT} due to RPV embrittlement results in an equivalent temperature shift in the HU/CD curves; i.e., the curves shift towards the NPSH and T_{SAT} limit curves, making the P-T operating space more restrictive. Accounting for the limiting material in both RPVs, RT_{NDT} is projected to increase 53°F between now and EOL, based on the fluence associated with 24-month cycle fuel management. This would increase the LTOP enable temperature from its current value of 270°F to 323°F which would provide a 77°F margin to the T_{SAT} limit curve. Prior to approval of Reference 8 (which lowered the LTOP enable temperature by crediting Branch Technical Position 5-2), the old (8 EFPY) LTOP enable temperature was 323°F. Based on DCPP's prior operating history with LTOP enable set at 323°F, an EOL LTOP enable temperature of 323°F would provide acceptable operating restrictions for plant heatup and cooldown.

The other LTOP consideration is the PORV setpoint. At EOL, a 53 degree shift in the heatup and cooldown curves would require a reduction in the PORV LTOP pressure setpoint of about 60 psi in order to protect the Appendix G limits. Our current setpoint is at 435 psig and cannot be significantly reduced without risking spurious PORV actuations with LTOP enabled. However, the need to reduce the setpoint can be eliminated by crediting ASME Code Case N-514 (allows use of 110% of the Appendix G limits for establishing LTOP setpoints) currently under review by the NRC. The code case has already been approved by ASME Section XI, and approval by the NRC and incorporation into 10CFR50 is expected within the next two years.

5.3.4 Upper Shelf Energy (USE)

From Reference 10, the Unit 1 Upper Shelf Energy (USE) is currently projected as 52 ft-lb (limiting material: Circumferential Weld 9-442) at EOL. The 12% flux increase associated with the power uprate and 24-month cycle operation would result in a maximum incremental reduction in USE of 1 %, or less than 1 ft-lb. Therefore, Unit 1 USE would remain above 50 ft-lb at EOL, and a power uprate would not impact compliance with the 1 OCFR5O Appendix G regulatory limit.

5.3.5 Pressurized Thermal Shock (PTS)

From Reference 10, the Unit 1 RTpts is currently projected as 216°F (limiting material: Lower Shell Axial Weld 30442C) at EOL. The 12% flux increase associated with a power uprate and 24-month cycle operation would increase the calculated EOL RT_{PTS} to 222°F. This value is well below the 270°F PTS screening criteria. Therefore, a Unit 1 power uprate would have no impact with respect to the 10CFR50.61 regulatory limit.

5.3.6 Surveillance Capsule Withdrawal Schedules

The surveillance capsule withdrawal schedules for DCPP Unit 1 is located in Reference 11, as well as the UFSAR Update. The surveillance capsules are scheduled for removal and evaluation when they achieve a prescribed fluence (e.g., the projected RPV fluence at EOL) in accordance with ASTM El 85-82. Since the surveillance capsules are closer to the core than the vessel is, the capsules are exposed to a higher neutron fast flux than the vessel. the ratio of the flux at a particular capsule location to the peak flux seen in the reactor vessel is defined as the capsule's "Lead Factor". Changes in fuel management can affect a capsule's lead factor if the azimuthal power distribution in the core peripheral assemblies changes from one cycle to another; more specifically, if the power in the assemblies adjacent to a surveillance capsule location changes relative to the power in the peripheral assemblies at the vessel azimuthal peak flux location (45°). Conversely, if a fuel management change does not significantly change the ratio of these powers (local power at surveillance capsule azimuth/local power at vessel 45° azimuth), then the capsules' lead factors will not be impacted and the surveillance capsule withdrawal schedule is not impacted.

At DCPP Unit 1, the surveillance capsule locations are at azimuthal angles 40 and 400 (and symmetric locations), and have lead factors of 1.3 and 3.4, respectively. Reference 1 determined that 21-month and 24-month cycle fuel management would not have a significant impact on the Unit 1 surveillance capsule withdrawal schedules. In addition, a Unit 1 power uprate would be associated with a relatively uniform (global) power increase across the core; i.e., no changes in azimuthal power distribution would be expected. It is therefore concluded that Unit 1 operation with extended cycles and uprated power will not impact the surveillance capsule withdrawal schedule.

5.3.7 Impact to DCPP Emergency Procedures Related to Reactor Vessel Protection

Plant procedures which respond to PTS (EOP FR-P.1 and EOP FR-P.2), as well as related procedures which address a rapid RCS cooldown, and/or LOCA, were reviewed to assess the changes expected in implementing a power uprate at DCPP Unit 1. These procedures attempt to protect the vessel by: 1) limiting cold injection when there is already adequate core cooling (to minimize vessel thermal shock), 2) depressurizing the RCS (to minimize vessel stresses), 3) ensuring the PORV low pressure setpoints are cut-in when RCS pressure drops to 400-425 psi (establish LTOP and prevent system repressurization), and 4) limiting the cooldown rate to -100°F/hr. These actions are consistent with the DCPP Technical Specifications. It was determined above that an uprated power of 3411 MWt for Unit 1 will not create the need to change the Technical Specifications related to vessel integrity: LCO 3.4.9.1 and 3.4.9.3, which establish the Appendix G heatup/cooldown limits (e.g., -100°F/hr max) and LTOP setpoints (e.g., 435 psi), respectively. Based on this, the current DCPP EOPs are consistent with the RPV integrity requirements for Unit 1 operation at 3411 MWt. A similar conclusion was reached in Reference 1 for extended cycle operation.

5.3.8 Reactor Vessel Structural Evaluation

The Diablo Canyon Power Plant Unit 1 Uprating Program design parameters (Table 2.1-1) identify a maximum vessel outlet temperature (T_{HOT}) of 610.1°F and a minimum Vessel inlet temperature (T_{COLD}) of 544.5°F.

The Diablo Canyon Power Plant Unit 1 reactor vessel has previously been analyzed for a maximum vessel outlet temperature (T_{HOT}) of 610.3°F, and a minimum vessel inlet temperature (T_{COLD}) of 534.3°F. Therefore, the normal operating design temperatures for the reactor vessel at the uprated conditions remain within the bounds of the reactor vessel stress report.

The previously applicable NSSS design transients, which are the basis for the previous reactor vessel structural and fatigue analyses, are applicable to the Diablo Canyon Power Plant Unit 1 uprating without modification. Therefore, the previous reactor vessel analyses and evaluations are applicable for the uprated conditions, and the maximum ranges of stress intensity and maximum cumulative fatigue usage factors are unchanged as a result of the uprating.

5.3.9 Conclusion

Based on the above evaluations, it is concluded that implementation of the Unit 1 Power Uprate Program from 3338 MWt to 3411 MWt, along with extended fuel cycle operation, will not impact the DCPP reactor vessel embrittlement levels with respect to plant operation and regulatory limits.

The current Diablo Canyon Power Plant Unit 1 reactor vessel stress report remains valid for the uprated conditions described in Table 2.1-1, and no changes are required.

References:

1. PG&E Chron 229706, "DCPP Extended Cycle Reactor Vessel Evaluation," April 3, 1996.

2. PG&E Chron 227158, Westinghouse Report, "Diablo Canyon Power Plant Unit 2, 21 Month Fuel Cycle Feasibility Study," June 1995.

3. PG&E Chron 227894, Westinghouse Letter 9SPGE-G-0050, "Diablo Canyon Unit 2 Cycle 8 Final Loading Pattern," September 15, 1995.

4. PG&E Chron 229690, Westinghouse Letter 96PGE-0033, "Diablo Canyon Unit 1 Cycle 9 Preliminary Loading Pattern," April 1, 1996.

5. PG&E Chron 229722, Westinghouse Letter 96PGE-G-0035, "Diablo Canyon Unit 1 Cycle 9 Preliminary Loading Pattern," April 4, 1996.

6. WCAP-13968, "Diablo Canyon Power Plant Units 1 and 2, 24 Month Fuel Cycle Feasibility Study," March 1994.

7. PG&E Calculation File 930818-0, "DCPP 1 and 2 Reactor Vessel Fluence Projections for Input to RCS Heatup and Cooldown Curves at 12 EFPY," August 26, 1993.

8. PG&E Chron 222814, "License Amendment Request 94-09," August 17, 1994.

9. PG&E Chron 225683, "Reactor Vessel Embrittlement Management Plan, DCPP 1 and 2," January 20, 1995.

10. NUREG-1511, "Reactor Pressure Vessel Status Report," December 1994.

11. WCAP-13750, "Analysis of Capsule Y from the PG&E Diablo Canyon Unit 1 Reactor Vessel Radiation Surveillance Program," July 1993.

5.4 REACTOR INTERNALS

This section documents the results and conclusions of the evaluations performed to determine the impact of uprating Diablo Canyon Power Plant Unit 1 to an NSSS power level of 3425 MWt on the reactor vessel internals. Evaluations and reanalyses of the reactor vessel internals and associated design features were performed for the design conditions listed in Table 2.1-1. The analyses and evaluations, conducted to demonstrate continued compliance with applicable design and analysis criteria at the uprated conditions, included the following:

- Rod Drop Time
- Flow Induced Vibration
- Baffle-Barrel Region Thermal/Structural
- Upper Core Plate
- Thermal Shield Support System Thermal/Structural

The NSSS design parameters for the reactor internals uprating evaluations and analyses are those listed in Table 2.1-1. The previously applicable NSSS design transients remain applicable at the uprated conditions. Therefore, no reactor internals evaluations are required in support of the Diablo Canyon Power Plant Unit 1 Uprating Program for changes in design transients.

5.4.1 RCCA Scram Performance Evaluation

A rod control cluster assembly (RCCA) drop time assessment was performed for the uprated conditions described in Table 2.1-1.

The purpose of this evaluation was to determine the potential impact of the power uprating at Diablo Canyon Power Plant Unit 1 on RCCA scram characteristics used in the FSAR for accident analyses. This analysis is based on 17x17 VANTAGE 5 Fuel Assemblies with IFM grids.

Calculations resulted in a maximum drop time-to-dashpot entry of about 2.5 seconds, as the most severe case, hence the current RCCA drop time technical specification limit of 2.7 seconds for Diablo Canyon Power Plant Unit 1 remains conservatively applicable for accident analyses.

5.4.2 Flow Induced Vibration

An evaluation was performed to determine the impact of the Diablo Canyon Power Plant Unit 1 parameters, as described in Table 2.1-1 on the structural integrity of the reactor internals with regard to flow induced vibrations. The results of the assessment showed that there is no adverse impact on the vibrational response of the Diablo Canyon Power Plant Unit 1 reactor internals with regard to flow induced vibrations, and that the structural integrity of the components is maintained for the uprating conditions described in Table 2.1-1.

5.4.3 Barrel-Baffle Region

A structural assessment was performed to evaluate the impact of the uprating conditions on the baffle-barrel region components. The baffle-barrel region is sensitive to changes in steady-state and transient temperatures and core power distribution, as it affects heating rates.

This assessment assumed that all the bolts in the baffle-barrel region are intact and functional, and then evaluated the impact of the increase in the heat generated in the baffle-barrel region components, as well as the changes in reactor coolant system temperatures associated with the uprating.

The changes in inlet and outlet baffle-barrel region temperatures due to the uprating were judged to be negligible. A comparison was made of the internal heat generation rates for the pre-uprating conditions to those generated using the EXCEL program, incorporating the uprating conditions. This review indicated that the internals heat generation rate distributions are similar. Therefore, the stresses induced in the baffle-barrel region structures are also similar. In summary, it was judged that the structural integrity and functionality of the baffle-barrel region components will not be adversely affected by the power uprating.

5.4.4 Upper Core Plate

The purpose of this section is to summarize the work performed to assess the impact on the structural integrity of the upper core plate due to a power uprating for Diablo Canyon Power Plant Unit 1. The Unit 1 power uprating will bring this unit to the same power level as Diablo Canyon Power Plant Unit 2; that is, a core power level of 3411 MWt and associated design parameters listed in Table 2.1-1.

The upper core plate functions to position the upper ends of the fuel assemblies and the lower ends of the RCCA control rod guide tubes. The plate also controls the coolant flow as it exits from the fuel assemblies and serves as a boundary between the core and the upper plenum. The plate consists of two distinct regions, i.e., a center (ligament) region perforated with round and square holes and a solid peripheral (rim) region directly above the baffle/barrel region.

For this uprating to a core power level of 3411 MWt, only the thermal loads are affected by the uprating. Moreover, the RCS design transients were unaffected. As a result, since all other loadings on the upper core plate are unchanged and the changes to the thermal loads due to the uprating for the upper core plate were evaluated to be insignificant, it is judged that the structural integrity of the upper core plate for the Diablo Canyon Power Plant Unit 1 will be maintained with the new reactor coolant system conditions described in Table 2.1-1, due to the uprating to a core power level of 3411 MWt.

5.4.5 Thermal Shield Support System

The thermal shield support system for Diablo Canyon Unit 1 was evaluated to assess the impact of the uprating conditions listed in Table 2.1-1 on its component parts. The thermal shield support system was evaluated by comparing the new loadings (due to the uprating) to the current design loadings. Factors which could affect the thermal shield support system are listed below:

1. Thermal transients
2. Heat generation rates
3. RCS performance parameters (Table 2.1-1)
4. LOCA forcing functions
5. Seismic loadings

The uprating program caused changes to the RCS performance parameters and the loadings for the heat generation rates. The remaining factors remained unchanged as a result of uprating and were not evaluated.

5.5 CONTROL ROD DRIVE MECHANISMS

The Control Rod Drive Mechanisms of Diablo Canyon Power Plant Unit 1 were evaluated to determine their continued compliance with applicable design criteria at the uprated conditions listed in Table 2.1-1. Diablo Canyon Power Plant Unit 1 uses model L-106A full-length (F/L) CRDMs manufactured by the Westinghouse Electro-Mechanical Division. There are part-length (P/L) mechanisms manufactured by Royal Industries, which have the control rods removed, but the pressure boundary components are still in place. This section addresses the ASME Code pressure boundary aspects of the new parameters.

The Uprating Program parameters for Diablo Canyon Power Plant Unit 1 as described in Table 2.1-1, reflect a flow rate of 87,700 gpm per loop Thermal Design Flow with Case 1 reflecting 0% steam generator tube plugging (SGTP), and Case 2 reflecting 15% SGTP. The Unit 2 parameters reflect 88,500 gpm per loop Thermal Design Flow with 0% SGTP.

The plant primary system thermal and pressure transients remain unchanged for the uprating. The original system transients are defined in the Equipment Specifications. The CRDM design pressure is 2500 psia and the design temperature is 650°F. The Equipment Specification operating condition is 2250 psia and 550°F.

5.5.1 CRDM Evaluation

The uprating vessel outlet CRDM temperature for the hot loop condition is 610.1°F. The reactor coolant pressure remains at 2250 psia. The full-length CRDM Code pressure boundary component stress analysis used a conservative operating temperature of 650°F, which is actually the design temperature. The part-length CRDM Code pressure boundary component stress analysis also conservatively used 650°F as the operating temperature. Thus, the uprating temperature of 610.1°F is still bounded by the Code analyses.

5.5.2 Conclusion

A review of the structural and thermal code analysis reports for the part length and full length CRDMs shows they conservatively envelope the uprating parameters shown in Table 2.1-1. Since the Equipment Specification and Code pressure boundary reports are unaffected, the proposed uprating for Diablo Canyon Unit 1 full length and part length CRDMs is deemed acceptable.

5.6 REACTOR COOLANT PUMPS AND MOTORS

The Model 93A Reactor Coolant Pumps (RCPs) for the Diablo Canyon Power Plant Unit 1 were evaluated for continued compliance with ASME Boiler and Pressure Vessel Code design criteria at the Uprating Program conditions. The RCP motors were also evaluated for continued compliance with applicable design criteria at the Uprating Program conditions. The Uprating Program efforts included evaluations of the Reactor Coolant Pumps and Motors for the NSSS design parameters listed in Table 2.1-1.

The Diablo Canyon Power Plant Unit 1 RCPs were not ASME Code stamped, but the identical Model 93A RCPs were used in Unit 2 and are Code stamped. Therefore, both unit RCPs are treated identical herein. The uprating parameters for Unit 1 reflect 87,700 gpm per loop Thermal Design Flow with Case 1 reflecting 0% steam generator tube plugging (SGTP), and Case 2 reflecting 15% SGTP. Unit 2 parameters reflect 88,500 gpm per loop Thermal Design Flow with 0% SGTP.

5.6.1 Structural Evaluations

5.6.1.1 RCP Evaluation

The RCP inlet temperature remains at 544.2°F for all cases for Unit 1. For Unit 2, the RCP fluid temperature decreases from 544.9°F to 544.8°F, which is insignificant. The RCP pressure remains at 2250 psia. Note, the Unit 2 pressure boundary stress report used 545.0°F and the generic reports used 550°F (or higher). Thus, the pressure boundary stress report and the generic reports remain applicable for the Diablo Canyon Power Plant Unit 1 uprating parameters listed in Table 2.1-1.

The NSSS design transients remain unchanged. Thus, the Code pressure boundary stress reports, specific reports and generic reports remain applicable. Thus, the Equipment Specification and Code stress requirements are satisfied for the Diablo Canyon Power Plant Unit 1 RCPs.

Since the Equipment Specification and Code pressure boundary structural evaluations are unaffected by the uprating and SGTP parameters listed in Table 2.1-1, the Diablo Canyon Unit 1 Reactor Coolant Pumps continue to comply with applicable design and analysis criteria at the uprated conditions.

5.6.1.2 Reactor Coolant Pump Motor Evaluations

5.6.1.2.a Design Parameter Evaluation

The Diablo Canyon Power Plant Unit 1 Reactor Coolant Pump (RCP) Motors were evaluated for the calculated worst case loads for the motors based on the uprating design parameters. Using the revised loads, the Diablo Canyon RCP motors have been evaluated in the four areas where parameter changes effect performance. This evaluation assumes the condition of the motors is still as designed (or modified) by Westinghouse.

5.6.1.2.b Continuous Operation at Revised Hot Loop Rating

The Reactor Coolant Pump Equipment Specification requires that the motor drive the pump continuously under hot loop conditions without exceeding a stator winding temperature rise of 70°C (corresponding to the NEMA Class B temperature rise limit in a 50°C ambient). Temperature tests performed on a Diablo Canyon motor have shown that the actual temperature rise at the hot loop nameplate rating (6000 HP) is 60.3°C. Therefore, adequate margin exists for continuous operation with loads in excess of the 6000 HP nameplate rating.

5.6.1.2.c Continuous Operation at Revised Cold Loop Rating

The Reactor Coolant Pump (RCP) Equipment Specification requires that the motor drive the pump for up to 50 hours (continuous) under cold loop conditions without exceeding a stator winding temperature rise of 95°C (corresponding to the NEMA Class °F temperature rise limit in a 50°C ambient). Based on the hot loop temperature tests, the estimated temperature rise of the Diablo Canyon motors at the cold loop nameplate rating (7500 HP) is 84.1°C. Therefore, margin exists for continuous operation with loads in excess of the 7500 HP nameplate rating.

5.6.1.2.d Starting

There is no impact on the starting power of the Reactor Coolant Pump Motor at the uprating conditions for Diablo Canyon Power Plant Unit 1.

The Reactor Coolant Pump Equipment Specification requires that the motor start across the line under cold loop conditions, with 80% starting voltage, against the reverse flow from the other pumps running at full speed. The limiting component for this type of starting duty is the rotor cage winding. A conservative all heat stored analysis is used to determine if the case winding temperature exceeds the design limits (300°C on the bars and 50°C on the resistance rings). If the conservative calculation shows unsatisfactory results a more detailed finite element calculation is run.

The starting temperature rise for the rotor bars and resistance rings was calculated. The results show bar temperatures of 294.8°C and ring temperatures 31.3°C. These temperatures do not exceed the design limits. Therefore, the motor can safely accelerate the load under the worst case conditions and the more detailed analysis is not performed.

5.6.1.2.e Loads on Thrust Bearings

Performance of the thrust bearings in an RCP motor can be adversely effected by excessive or inadequate loading. The change in axial down thrust for the revised parameters is insignificant. There will be no impact on thrust bearing performance.

5.6.1.2.f Conclusion

Based on the results of the uprating analyses, the RCP motors at Diablo Canyon Power Plant Unit 1 are considered acceptable for operation under the revised conditions defined by Table 2.1-1.

5.7 REACTOR COOLANT LOOP PIPING AND SUPPORTS

The primary loop piping, primary equipment supports, the primary equipment nozzles, the Westinghouse scope ASME Class 1 auxiliary piping, the ASME Class 1 loop branch nozzles, and the Pressurizer Surge Line for the Diablo Canyon Power Plant Unit 1 were evaluated for operation at the uprating parameter conditions. The evaluation was performed by reviewing the previously completed analysis of the piping, supports, and nozzles, and determining the effects of any differences in input parameters on the analysis results for the Uprating Program conditions.

5.7.1 Piping and Supports

The uprating design parameters do not impact the design basis of the reactor coolant loop piping, primary equipment supports, primary equipment nozzles, the Westinghouse scope ASME Class 1 auxiliary piping, the ASME Class 1 loop branch nozzles, and the Pressurizer Surge Line. Furthermore, the NSSS design transients remain unchanged for the uprated conditions. Therefore, as there is no impact on the existing design basis evaluations and no impact on the T_{AVG} Coastdown evaluations as a result of the uprating, the piping, supports and nozzles continue to comply with the applicable design and analysis criteria at the uprated conditions described in Table 2.1-1.

5.7.2 Leak Before Break

A leak-before-break evaluation was performed for the Diablo Canyon Power Plant Units 1 and 2 primary loops to provide technical justification for eliminating large primary loop pipe rupture as the structural design basis for the Diablo Canyon Power Plant Units.

In order to demonstrate the elimination of RCS primary loop pipe breaks for the Diablo Canyon plants, the following objectives must be achieved.

- Demonstrate that margin exists between the "critical" crack size and a postulated crack which yields a detectable leak rate.

- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability of the Diablo Canyon Power Plant Units 1 and 2.

- Demonstrate margin on applied load.

- Demonstrate that fatigue crack growth is negligible.

These objectives were met in WCAP-13039.

The leak-before-break evaluations include the applied loads as the input. Both normal operating loads and the faulted loads are used as input to the evaluations.

The effect of temperature changes resulting from uprating on the primary loop loads is negligible.

It is further noted that a minor increase in temperature and the corresponding reduction in material properties due to changes in temperature of the magnitude considered for this uprating request would result in no change in LBB margins. Since the magnitude of change in loads and material properties is negligible, the LBB margins previously calculated and documented in WCAP-13039 will remain unchanged.

In summary, an assessment was performed pertaining to the impact of uprating on the LBB conclusions for Diablo Canyon Power Plant Unit 1 primary loops. Based on the assessment, it is judged that the leak-before-break margins will have negligible change and the LBB conclusions of WCAP-13039 will remain unchanged.

5.8 AUXILIARY COMPONENTS

The Diablo Canyon Power Plant Unit 1 NSSS auxiliary components, including tanks, pumps, valves and heat exchangers were reviewed to determine the impact of the uprating NSSS design parameters.

In addition to the auxiliary heat exchangers, pumps and valves, various auxiliary tanks were provided to Diablo Canyon Power Plant Unit 1. The auxiliary tanks have insignificant transients identified in the original design. Hence these vessels are not impacted by the Uprating Program and are not addressed in the remainder of this report.

5.8.1 Discussion of Evaluation

Westinghouse reviewed the original design and qualification requirements for Diablo Canyon Power Plant Unit 1 auxiliary equipment as supplied by Westinghouse. It has been assumed that any equipment replaced or added was designed, procured and installed in accordance with the original Westinghouse quality assurance and technical requirements.

The auxiliary heat exchanger specifications and purchase order documents did not require the seal water heat exchangers and spent fuel pit heat exchangers to be qualified for pressure or temperature transients. The transients were not included in the design because they were expected to have no effect on these components. Therefore, this equipment was designed only for maximum steady state pressures and temperatures. The uprating parameters for the regenerative heat exchangers, non-regenerative (i.e., letdown) heat exchangers, excess letdown heat exchangers, sample heat exchangers, and residual heat exchangers are bounded by the original Diablo Canyon Power Plant Unit 1 design parameters defined by Reference 1.

The governing design transients for Diablo Canyon Power Plant auxiliary pumps and valves are specified in Reference 2. The design transients specified in Reference 2 are still bounding for Diablo Canyon Power Plant Unit 1 at the uprated NSSS operating conditions. Since it has been demonstrated that the original Diablo Canyon Unit 1 transients remain bounding for the uprating conditions as applied to the auxiliary pumps, valves and heat exchangers, there is no effect on qualification of this equipment.

5.8.2 Conclusion

The Diablo Canyon Power Plant Unit 1 Uprating design parameters have no effect on the qualification of the auxiliary pumps, auxiliary heat exchangers, auxiliary tanks and auxiliary valves.

There are no new limitations associated with the auxiliary pumps, auxiliary heat exchangers, auxiliary tanks and auxiliary valves, due to the implementation of the Diablo Canyon Power Plant Unit 1 Uprating Program.

References:

1. Equipment Specification G-676454, Revision 1, "Auxiliary Heat Exchangers General Specification," 10/29/68.

2. Systems Standard Design Criteria 1.3, Revision 1, April 2, 1971.

6.0 FUEL DESIGN

Evaluations were performed of the fuel for Diablo Canyon Power Plant Unit 1 under the Uprating Program in the areas fuel rod and fuel assembly structural integrity for the uprating conditions.

6.1 CORE DESIGN

Core Design is evaluated on a cycle-specific basis for Diablo Canyon Power Plant Unit 1.

6.2 THERMAL & HYDRAULIC EVALUATION

The thermal and hydraulic evaluation is performed for Diablo Canyon Power Plant Unit 1 on a cycle-specific basis.

6.3 FUEL ROD STRUCTURAL INTEGRITY

An evaluation was performed under the Uprating Program of the impact of NSSS performance parameters in Table 2.1-1 on the ability of fuel to satisfy fuel rod design criteria for Diablo Canyon Power Plant Unit 1.

The Uprating Program will have an impact on several key fuel rod design criteria. This section summarizes those design criteria which are typically most affected by the changes in fuel duty. The impacts of each of these parameters on margins to the fuel rod design criteria were evaluated.

6.3.1 Rod Internal Pressure

The rod internal pressure design basis is that the fuel system will not be damaged due to excessive fuel rod internal pressure. The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward clad creep during steady state operation for extensive DNB propagation to occur. NRC-approved Westinghouse PAD3.4 fuel performance models, Reference 1, are used to evaluate rod internal pressure as a function of irradiation time and fuel duty. Margin to the rod internal pressure limit is impacted by changes in the core power rating, since the higher ratings will result in higher fuel operating temperature and higher fission gas release.

Rod internal pressure analyses, performed for the Diablo Canyon Power Plant Unit 1 Uprating Program, indicate that the rod internal pressure criterion will be satisfied for the uprated conditions in Table 6.1-1.

6.3.2 Clad Corrosion

The clad corrosion design basis is that the fuel system will not be damaged due to excessive fuel clad oxidation. The fuel system will be operated to prevent significant degradation of mechanical properties of the clad at low temperatures, as a result of hydrogen embrittlement caused by the formation of zirconium hydride platelets. The calculated clad temperature (metal oxide interface temperature) will be less than accepted limits specified for steady state operation and for Condition 2 events. The hydrogen pickup level in the clad will also be restricted to specified limits predicted for the end of fuel operation. The uprating conditions in Table 6.1-1 will result in increased operating temperatures for the clad due to the increased rod average power rating. Since the corrosion process is a function of clad temperature, the uprating will impact these criteria. Based on corrosion analyses performed for the Diablo Canyon Power Plant Unit 1 Uprating Program, sufficient margin to each of the corrosion-related criteria exists to support the uprated core conditions at the longer cycle lengths with ZIRLO™ clad fuel. The use of Zirc-4 clad fuel will require cycle-specific analysis to confirm its compliance to the new cladding corrosion model currently under development.

6.3.3 Clad Stress

The Clad Stress design basis is that the fuel system will not be damaged due to excessive fuel clad stress. The volume average effective stress calculated with the Von Mises equation considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences, is less than the 0.2% offset yield stress with due consideration to temperature and irradiation effects under Condition 1 and 2 events. While the clad has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design limit. Westinghouse PAD3.4 model, Reference 1, is used to evaluate clad stress limits. The local power duty during Condition 2 events is a key factor in evaluating margin to clad stress limits. The fuel duty at the uprated conditions is expected to be more limiting, which will reduce margins to the clad stress limit. However, evaluations performed for the uprated core conditions indicate that sufficient margin is available to support the uprated conditions in Table 6.1-1.

6.3.4 Summary

The fuel rod design criteria most impacted by a change in core power rating have been reviewed with respect to available margin to support the proposed uprating. It is concluded that although some design criteria are impacted, as stated above, sufficient fuel rod design margin exists to support the uprated conditions listed in Table 6.1-1 for Diablo Canyon Power Plant Unit 1. Cycle specific analysis will be required in determining the acceptability of the Zircaloy-4 clad fuel with respect to meeting the corrosion criteria.

6.4 FUEL ASSEMBLY STRUCTURAL INTEGRITY

6.4.1 LOCA Forces and Core Plate Motions

A key input to the structural analysis of the fuel assembly is the magnitude of core plate motions imparted to the fuel assemblies. The core plate motions are, in turn, strongly influenced by LOCA hydraulic forces generated as a result of primary loop piping breaks.

A LOCA forces evaluation was performed for Diablo Canyon Unit 1 for the uprating conditions to determine their effects on the previously applicable forces. It was concluded that the uprating has a negligible effect on reactor vessel and internals LOCA forces. Therefore, the previously applicable LOCA forcing functions remain applicable at the uprated conditions. The LOCA forcing functions were based on limited displacement primary loop piping breaks and therefore, no reliance on leak-before-break was required.

The current core plate motions analyses remain bounding since the uprating parameters were determined to have negligible effects on the Reactor Vessel and Internals LOCA forces.

6.4.2 Fuel Assembly Structural Evaluation

Fuel assemblies are designed to perform as described in the Technical Specifications. The combined effects of design basis loads are considered in the validation of the fuel assembly and its component design to assure the fuel assembly structural integrity. This is necessary so that the fuel assembly functional requirements are met, the core coolable geometry is maintained, and the reactor core can be shut down safely.

A structural evaluation of the fuel assembly was performed for the Diablo Canyon Power Plant Unit 1 Uprating Program, considering the range of design parameters described in Table 6.1-1. This evaluation assumed 17x17 Vantages fuel for Unit 1.

The NSSS design parameters for the Uprating Program do not impact the resultant effects on core plate motions. Therefore, there is no impact on the fuel assembly seismic/LOCA structural evaluation due to the Uprating Program for Diablo Canyon Power Plant Unit 1.

In conclusion, the Uprating Program for Diablo Canyon Power Plant Unit 1 does not increase the operating and postulated transient loads such that they will adversely affect the fuel assembly functional requirements. The fuel assembly structural integrity is not affected and the core coolable geometry is maintained for the 17x17 Vantage 5 with IFM fuel for Diablo Canyon Power Plant Unit 1.

Reference:

1. WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," August 1988.

Table 6.1-1
Summary of Diablo Canyon Power Plant Unit I Uprating Parameters
Fuel Rod Design Analysis Parameters

Parameter	Units	Cycle Condition	Uprated Condition
Core Power	MWt	3338	up to 3411
Core Inlet Temperature	°F	540.9	544.5
Mass Flow Rate	$x10^6$, lbm/hr-ft^2	2.27	2.27
System Pressure	psia	2250	2250
Cycle Lengths	EFPD	489	570
Fuel Designs Considered	—	Zirc-4 Clad 1.5x IFBA 100 psi backfill Solid Fuel Stack	ZIRLO™ Clad 1.5x IFBA 100 psi backfill Annular Blankets

7.0 BALANCE OF PLANT (BOP) SYSTEMS

Most secondary side systems at Diablo Canyon Units 1 and 2 are identical. However, there are two major differences between the balance of plant systems for Units 1 and 2. The first is the cooling system on the main electrical generator (described in detail in Section 8.0). The second difference is in steam quality at minimum reactor coolant system (RCS) thermal design flow (TDF). Since Unit 1 has a lower RCS thermal design flow rate (87,700 gpm/loop vs. 88,500 gpm/loop), the predicted primary side temperature drop must be greater, T_{AVG} must be less to maintain the same T_{HOT}, and the secondary side temperature and pressure will be a little lower than for Unit 2. Correspondingly, the predicted steam flow rate is slightly greater and the moisture carryover is increased. It should be noted, however, that at this time Unit 2 has a greater percentage of steam generator tubes plugged, and the actual RCS flow rates of Unit 1 and Unit 2 are very close. When RCS flow rates, T_{AVG}, and power level are identical, Units 1 and 2 will have identical predicted steam conditions.

Unit 1 components were designed with the margin for higher power levels. The turbine and generators were sized for 105% power. The design maximum heat balance, PG&E Drawing 330551, assumes a core power of 3483 MWt, which already envelopes the proposed uprate to 3425 MWt.

7.1 STEAM SYSTEM

Typical full power data for Unit 2 is a steam flow of 3.66×10^6 lbm/hr per steam generator and 798.5 psia.

The piping systems stress analysis has been reviewed to assess the impact of the proposed Unit 1 uprating and the associated changes in pressures, temperatures, and flows. The result of the review is that the changes in system operating conditions are sufficiently small that there would be a negligible impact on piping stresses, pipe displacements, pipe support loads, and other piping design considerations. The small increase in flow in the Balance of Plant systems will not have any significant impact on the susceptibility of the piping to erosion-corrosion damage, and in addition, the wear rates are monitored by an inspection program that has ample margin built into the predictions of wear between inspection periods. DCPP has instituting erosion-corrosion programs that replaced the piping posing the highest-susceptibility to erosion-corrosion with resistant materials.

7.2 STEAM GENERATOR BLOWDOWN

This system is used to help maintain proper steam generator water chemistry by bleeding off a portion of the unboiled fraction of the water to prevent the buildup of dissolved solids in the steam generators over time. Although the slightly higher feed rate after the uprating will result in proportionally more solids being fed into the steam generator, this increase can be accommodated by slightly increasing the blowdown flow rate. Other variables in secondary water chemistry have a significantly greater impact on blowdown system operation than the Unit 1 uprating will have; the impact of the uprating on the blowdown system will be a secondary effect at most. The ability of this system to handle the slightly increased flow rates that may be necessary as a result of the Unit 1 uprating is demonstrated by the identical Unit 2 system's ability to do so. The only portions of this system that have an active safety function are the inboard and outboard blowdown containment isolation valves. These valves are stroke tested quarterly, and their ability to function is not impacted by the projected incremental change in blowdown flow rate. Hence, the Unit 1 uprating has no impact on the ability of this system to perform its safety function.

7.3 CIRCULATING WATER

The two large circulating water pumps move sea water through the condenser. Located upstream of the condenser, the unit uprate will be invisible to the pumps and their flow capacity. The ability to carry the heat from Unit 1 is assured by the continuing operation of Unit 2 at this power level. The consequence of the additional heat to the discharged circulating water is discussed in Section 9.0, Environment and Permit Evaluation.

7.4 CONDENSATE SYSTEM

The Condensate System consists of the main condenser and air ejector system, three 50% capacity condensate and condensate booster pumps, the condensate polishing system, and a number of feedwater heaters and banks of feedwater heaters which are heated by various sources. The ability of this system to handle the slightly increased flow rates stemming from the Unit 1 uprating, is demonstrated by the identical Unit 2 system's ability to do so. No portions or functions of this system are safety-related. Hence, the slight increase in condensate flow and feedwater heating associated with the uprating have no impact on safe operation of the plant.

7.5 FEEDWATER SYSTEM

The Feedwater System consists of two 50% capacity turbine driven feedwater pumps, three final stage feedwater heaters in parallel, and four feedwater lines, one to each steam generator. There will be a slightly greater demand for heating steam by the feedwater heaters due to the slightly higher feedwater flow rate but this will be supplied automatically by the present extraction steam system. However, as demonstrated by the Unit 2 feedwater system, the system has the Capacity to supply the additional steam and feedwater heating required. The ability of this system to handle the slightly increased flow rates stemming from the Unit 1 uprating, is demonstrated by the identical Unit 2 system's ability to do so. Each of the four feedwater lines has its own feedwater control valve, feedwater control bypass valve, and feedwater isolation valve. The higher flow rate will be below the full-open rated flow capacity of the feedwater control valves. The only active safety function associated with the feedwater system is closure of its control valves and isolation valves on receipt of a feedwater isolation signal. These valves are stroked tested from the full open position thus, confirming that they can perform their safety function regardless of their position during normal operation. There is no change in the trip signals or tripping capability of the feedwater pump turbines, which is a nonsafety-related but important function. Hence, the Unit 1 uprating has no impact on the ability of this system to perform its safety function.

7.6 AUXILIARY FEEDWATER SYSTEM

The safety function of the Auxiliary Feedwater (AFW) System is to supply adequate makeup flow to the steam generators to enable them to maintain level and remove the decay heat generated from the reactor following reactor shutdown for any reason. The decay heat is a function of the core size and power density prior to shutdown after extended steady state operation at 100% power. Since these parameters will be no greater for Unit 1 following its uprating than they presently are for Unit 2, and since the Unit 1 AFW system has the same capacity as Unit 2, the adequacy of the Unit 1 system for the uprated reactor is demonstrated by the present Unit 2 system.

7.7 BOP ELECTRICAL

In general, the plant electrical load will increase slightly over current loads due to the need to drive more energy through the plant. For example, the Condensate Booster Pumps will need additional energy to supply a higher flow rate. However, the electrical systems are sized for the full loading of plant equipment. Since there is no change in electrical components to support the uprate, and since Unit 1 is electrically equivalent to Unit 2 which already operates at the uprated power level, the capacity of the electrical system to respond to the Unit 1 uprate is assured.

7.8 STARTUP TRANSFORMER

The startup transformer receives power from the offsite power grid and supplies the electrical distribution system for startup and shutdown of the plant. The startup transformer also provides power for vital equipment during transients where the main unit generator is unavailable.

The startup transformer is sized for full loading of the plant equipment. Since no change to the electrical components is required for the Unit 1 uprating, the adequacy of the startup transformer continues to be assured.

7.9 MAIN UNIT TRANSFORMER

Main transformers deliver 96% of the generated power from the plant to the 500kV gird (the additional 4% of the generator output makes up the house loads). In addition, under certain plant configuration, the main transformers are used to "backfeed" power from the 500kV switchyard (offsite power) to the plant to energize electrical busses.

The main transformers at Unit 1 were replaced in 1R7; hence they are essentially new. The Unit 2 transformers are also being replaced in 2R8. The ratings of the new transformers remain unchanged at 1320 MVA. They can supply higher loads up to 1478.4 MVA, though this results in a temperature rise and accelerated aging.

Since the transformers which see 96% of the generator output are rated at 1320 MVA, and the Unit 1 generator is rated at 1300 MVA, the transformer will never be the limiting equipment in an uprating. The proposed Unit 1 uprate will not challenge the transformer capability.

7.10 EMERGENCY DIESEL GENERATORS

Diablo Canyon has three emergency diesel generators at Unit 1 which provide vital power for safe shutdown in the event of a loss of the preferred power source. Any transient that results in the need for emergency diesel generator actuation will also result in a reactor trip. With the reactor tripped, the electrical loads are virtually identical to the non-uprated condition, hence the emergency loads to the diesel generators are not affected by the plant uprating. Similarly, the load shedding and emergency load sequencing are unaffected (note: the studies which verify the appropriateness of the load shedding and emergency load sequencing, e.g., the best estimate LOCA, are performed at the uprated power conditions.) Since there is no change in emergency loading, there is no effect on Diesel Fuel Oil Systems.

7.11 INSTRUMENT AIR / NITROGEN

The Instrument Air System provides clean, oil and moisture free air to the instruments, controls, and other required services throughout the plant. Pneumatic actuators that require air pressure to go to their proper safeguards position during accident conditions have their Instrument Air Supply backed up by a bottled nitrogen gas supply.

The uprate of Unit 1 will not require any change in the pneumatic controls, instruments, or other devices that utilize instrument air. Therefore, the Instrument Air and Nitrogen systems will continue to be capable of performing their function after implementation of the Unit 1 Uprate Program.

7.12 AUXILIARY BUILDING VENTILATION

The purpose of the Auxiliary Building Ventilation System is to maintain the temperatures in the auxiliary building sufficiently low to ensure proper equipment operation. The safety function of the system is to ensure that the temperatures in the rooms containing safety related equipment remain below the environmental qualification temperatures of the equipment under all circumstances, including accident conditions. The Unit 1 uprating does not result in an increase in the heat load in the auxiliary building beyond that of Unit 2. Hence, there is no change in the ability of the Auxiliary Building Ventilation System to perform its normal safety-related functions. See also Section 3.3.4, Environmental Qualification.

7.13 SECONDARY SIDE CHEMISTRY

The secondary side chemistry is controlled to minimize corrosion, reduce ion transport, and to extend the life of the steam generators. The increase in power of Unit 1 will not effect the secondary chemistry or any chemical addition.

7.14 PROCESS SAMPLING SYSTEMS

Process Sampling Systems provide representative samples of non-nuclear process fluids for analyses necessary for plant operation, corrosion control, and monitoring of system equipment and performance.

Plant uprating will slightly change the temperatures of certain fluids being analyzed. Comparison of the heat balances in PGE-96-578, "Unit 1 Uprating Program Systems and Components Report," Westinghouse, June 27, 1996, (Figures 7.2-1 through 7.2-5) to the current heat balance (DWG 330552, 3338 MWt) shows that temperatures on the secondary side are decreasing except in few instances (the condenser is less than a tenth of a degree warmer). These systems have adequate capacity to absorb the power increase as demonstrated by the fact that the design maximum heat balance (DWG 330551) was performed at a core power of 3483 MWt. In addition, the systems are essentially identical to the Unit 2 systems that already operate at the uprated power level.

7.15 PRIMARY/SECONDARY OVERPRESSURE PROTECTION

There are two accident analyses that determine the adequacy of overpressure protection at DCPP. The first is the Condition 2 Loss of Load / Turbine Trip discussed in FSAR Section 15.2.7. The second is the Locked Rotor Event evaluated in FSAR 15.4.4 (applicable to the primary side only).

7.15.1 Loss of Load/Turbine Trip

The FSAR Accident Analysis for Loss of Load, Turbine Trip (LOL/TT) (FSAR Section 15.2.7) is maintained by PG&E using the RETRAN transient analysis code (PG &E Calculation N-158, "Sensitivity Study for FSAR Loss of Load / Turbine Trip Transient"). The purpose of investigating this event is primarily to show the adequacy of the overpressure protection system. LOL/TT is the limiting Condition 2 Event for overpressure protection and the event that most directly challenges the adequacy of the Main Steam Safety Valves and Pressurizer Safety Valves. PG&E took over the maintenance of this accident analysis when obtaining a Licensing Amendment to increased the allowable tolerance on the MSSV setpoints.

The LOL/TT analysis was performed using a detailed model of Unit 2. At the time of the LAR submittal, Unit 2 was clearly the limiting unit due to the higher power level. In investigating the Unit 1 uprate, it is noted that minor changes would be appropriate to model Unit 1; specifically, there is a small change in RCS volume, a lower thermal design flow, and a change in the primary system core flow resistance. These changes have been investigated and they have been determined to have no significant impact. Therefore, primarily because the LOL/TT analysis has already been performed at the uprated power level, the uprating of Unit 1 will not impact the Loss of Load, Turbine Trip Accident Analysis.

7.15.2 Locked Rotor

The Locked Rotor Event of FSAR Section 15.4.4 is modeled by the Westinghouse LOFTRAN computer code, which is less detailed than the RETRAN code used for LOL/TT.

The Locked Rotor analysis contained in the FSAR is performed for the Unit 2 power level, which is the level to which Unit 1 is being uprated. Westinghouse has verified that the existing analysis is valid for the Unit 1 with the uprated conditions.

The MSSVs and PSVs are identical between Units 1 and 2. Since these valves on Unit 2 have been demonstrated to have sufficient capacity for steam relief for the Unit 2 power level, they have also been verified for use at the uprated Unit 1 power level. Therefore, the uprating of Unit 1 will not challenge primary or secondary overpressure protection.

7.16 CONTROL ROOM

The Control Room boards provide a centralized control facility for the plant operators to monitor and control plant operations. Functions such as starting, stopping, tripping, and control of major plant equipment are accomplished from the Main Control Room boards.

No additional equipment requiring control from the control room is required for uprating. The uprating does not involve any new operator action assumptions for accident mitigation. The Unit 1 instruments are the same as Unit 2, and the present scale ranges are adequate and appropriate for Unit 2, 50 these will also be adequate for the uprated Unit 1 conditions. Therefore, the Control Room is not affected by the Unit 1 Uprate Program.

7.17 PROCEDURAL CHANGES

Critical operator actions are proceduralized in series of operating procedures. Very few procedures will need to be changed. Most references to power level utilize normalized power readings, though absolute power levels are specified for certain loop tests, fuel related calculations (defects, burnup or core damage assessment in emergency procedures), and calibrations. Unit 2 procedures demonstrate that procedures can be written and performed for the uprated condition. Section 1.7 identified that Emergency Operating Procedures (EOPs) involving rapid RCS heat up or cooldown stresses, or low temperature overpressure protection, do not need to be modified.

The procedures which specify the current power level of 3338 MWt are Operating Procedure L-4, Emergency Procedure RB-i 4, Surveillance Tests R-14 and PEP R-5, Interdepartmental Administrative Procedure T56.NE1, Loop Tests 19-22D and 8-43 and TAB 41.0. These procedures will need modification should a license request to uprate Unit 1 be approved.

Again, the Diablo Canyon Operators are familiar with operating Unit 2 at 3411 MWt. For most applications, the core thermal power is normalized to a percent of full power value. The absolute core power level is infrequently used in procedures or control room indications. Uprating Unit 1 to this level will present no new challenges for the control room equipment or Operations Personnel.

7.18 SIMULATOR

The DCPP Simulator is a model of the Unit 1 Control Room that responds to hypothesized transients based on a computer model that replicates the behavior of Unit 1 as closely as practical. It has already been assumed that the differences in unit behavior between Units 1 and 2 are not significant enough to invalidate the application of the simulator training to both units (although the training process does stress unit to unit differences wherever they occur).

When the Unit 1 Uprate is implemented, it will be possible to modify the simulator to reflect behavior appropriate to the higher power level.

7.19 CONCLUSIONS

The uprating of DCPP Unit 1 to the same power level as DCPP Unit 2 is greatly simplified by the similarities between the two units. In most systems, the loads and equipment are similar enough to justify the uprating as the basis of the Unit 2 experience. The few differences are primarily related to lower RCS flow (which in turn results in a slightly lower secondary side pressure) and reduced generator rating at Unit 1. These differences are evaluated above and documented to be insignificant in safety consequence. The nonsafety consequences are an increase in moisture carryover and in circulating water discharge temperature. Both consequences will be monitored and, if necessary, further plant modifications or as-needed back-offs in power generation will be utilized to assure continued reliability and compliance with all regulatory limits.

8.0 TURBINE GENERATOR SYSTEMS & COMPONENTS REVIEW

The Diablo Canyon Unit 1 turbine is an 1137 MWe Westinghouse tandem compound turbine generator consisting of one double-flow high pressure turbine, BB96, and three double-flow low pressure turbines, BB81. The generator is rated at 1300 MVA, 0.9 power factor, with a brushless exciter. The unit went into commercial operation in 1984. The original thermal power level was 3350 MWt. In the original design, the turbine was designed for 105% power.

The BB96 high pressure turbine is constructed similarly to Diablo Canyon Unit 2, except for the nozzle blocks, triple pin control stage blades and third stationary reaction row on each end of the double flow element. These rows were originally designed to accommodate the slightly different flow conditions of the two units.

The original BB81 low pressure turbine rotors have been refurbished with upgraded disc design, and have also been modified to allow interchangeable spare rotors to be shared between Unit 1 and 2.

The electrical generator is a hydrogen cooled unit with a water cooled, epoxy Thermalastic type stator winding. Winding maintenance modules were installed in 1989.

Evaluations and analyses were performed to predict the performance of the Turbine Generator systems and components at the uprated conditions described in Table 2.1-1.

The evaluations and analyses were performed to address the following facets of plant performance.

8.1 HEAT BALANCES

A new turbine heat balance was required showing operation of the existing turbine generator at the proposed uprated power level. To evaluate various steam generator tube plugging (SGTP) levels in the steam generator, several additional heat balance cases were performed with varying steam generator pressure. This allowed an estimate to be made of the minimum turbine throttle pressure that would allow the existing unit to pass full steam flow at 3425 MWt, without modification.

Preliminary heat balances were produced for Diablo Canyon Power Plant Unit 1 at the 3425 MWt NSSS uprated power level, and various steam generator outlet pressures. The steam generator outlet pressures were based on assuming a reactor vessel T_{HOT} temperature of 603°F, and the current SGTP level of 1.7%, the steam generator outlet pressure was predicted for best estimate conditions. The steam generator outlet pressure for these conditions is approximately 805 psia. If the steam generator outlet pressure is maintained at 805 psia, and the SGTP level is increased to 15%, the reactor vessel T_{HOT} will have to be increased to approximately 608.4°F.

The estimated minimum steam generator outlet pressure needed to allow the turbine to pass steam flow corresponding to 100% MWt at the uprated conditions is 793 psia. Assuming a reactor vessel T_{HOT} of 603°F, this corresponds to an SGTP level of 6.5% based on best estimate conditions.

Based on the result of the new heat balances, the expected, although not guaranteed, electrical output was projected.

8.2 TURBINE EVALUATION

8.2.1 Structural Integrity

A study was performed to determine the mechanical and thermodynamic adequacy of the turbine for Diablo Canyon Power Plant Unit 1, when operated with throttle steam flow corresponding to a proposed uprated nuclear power of 3425 MWt, versus the original 3350 MWt rating. This represents an uprating of approximately 2.24%. For comparison, the original turbine generator was designed to operate at approximately 105% of rated throttle flow. This is referred to as "maximum calculated flow" and is shown on maximum calculated heat balances in the original thermal kit for Unit 1.

Detailed design calculations were made for components associated with the HP turbine, because of the various throttle pressures considered for this study. Rotating blading, stationary blading, rotor shaft diameters, and the rotor coupling all meet mechanical design requirements at the uprated conditions. However, to maximize reliability, it is recommended that the unit be operated with full arc admission, instead of the original 75% minimum arc of admission. Actual operating experience with triple pin control stage blading has been excellent, with no failures reported.

Detailed design calculations for the Low Pressure turbine components were not performed because of lower flow conditions in the new heat balances, compared to the original design heat balances for the unit. Flows are significantly lower than the original maximum calculated design flows for either Diablo Canyon Unit 1 or 2. All low pressure turbine rotors are considered to be interchangeable between the two units.

8.2.2 Thermal, Hydraulic and Electrical Performance

In order to determine the steam flows, temperatures and pressure associated with the turbine cycle, and to estimate the increase in electrical MWe output, revised full load heat balances were prepared for operation at the new higher MWt, with various turbine throttle pressures. The resulting heat balances were used to provide an estimate of the minimum throttle pressure needed to pass steam flow corresponding to 100% MWt at the uprated power level. The same heat balances were also used for subsequent turbine missile analysis and turbine component evaluations.

The study demonstrates an estimated improvement in gross electrical output of approximately 23 MWe at the new uprated power level of 3425 MWt compared to the original level of 3350 MWt.

8.2.3 Turbine Evaluation Summary

The proposed Uprating Program for the Diablo Canyon Power Plant Unit 1, from 3350 MWt to 3425 MWt, continues to comply with all industry and regulatory requirements, codes and standards based on a mechanical and thermodynamic evaluation of the turbine, with no component replacement required. The 2.24% uprated flow is within the original maximum calculated design conditions for the unit. However, based on evaluation of the existing triple pin control stage blading, using current design methods, it is recommended that the unit be operated with full arc admission in order to maximize reliability.

The increased throttle flow at 3425 MWt is estimated to produce an additional 23 gross MW electric - 1160 MWe versus 1137 MWe.

An estimate was made of the minimum throttle pressure needed to allow the turbine to pass steam flow corresponding to 100% MWt at the new power level. This minimum throttle pressure is approximately 754 psia, with steam generator pressure of 793 psia. The actual minimum pressure level will vary depending on the present condition of the turbine generator and other plant equipment. All turbine heat balance calculations assume "as-new" conditions and 0.25% moisture at the steam generator outlet.

8.3 MOISTURE CONTENT LIMIT

Design guidelines have been established for the design of steam generator moisture separators and high pressure turbine components with respect to moisture content of the steam. The design guidelines are to limit the moisture carryover at the steam generator outlet to less than 0.25 percent. The corresponding moisture content guideline at the entrance to the high pressure turbine is 0.50 percent.

Moisture carryover will increase with uprating as discussed in Reference 1. The consequences of increased moisture carryover is a loss of efficiency and increased turbine blade wear. It is anticipated that at expected best estimate conditions the moisture content at the steam generator will be about 0.25%, and the moisture at the turbine inlet will continue to be 0.5% or less. It has been determined that the moisture content will be acceptable provided that the moisture content at the entrance to the high pressure turbine is 0.5% or less.

Moisture carryover will continue to increase with plant aging due to increased tube plugging, potential sludge build-up in the steam generator dryers, and potential future T_{HOT} reduction. Moisture carryover is a logarithmic function of the steam specific volume and the square of steam flow. Therefore, at the higher moisture, any reduction in steam pressure or increase in steam flow will have increasingly large effects on moisture carryover.

Steam exhausted from the HP turbines is reheated in the Main Steam Reheaters (MSRs) prior to reaching the LP turbines. Westinghouse has concluded that the uprate effects on steam quality will be negligible downstream of the MSRs. Therefore, the Unit 1 LP turbines will see no changes in moisture content.

Reference:

1. PGE-96-578, "Unit 1 Uprating Program Systems and Components Report," Section 5.2, June 27, 1996.

8.4 MISSILE ANALYSIS

The existing turbine missile analysis was reviewed for the Diablo Canyon Power Plant Unit 1 Uprating Program. The review was based on the uprating conditions shown in the heat balances which were generated for the uprating study. This review determined that the current turbine missile analysis is bounding for the uprating conditions, when operating with any of the BB81 low pressure turbine rotors constructed with upgraded disc designs.

Missile generation is an event which may result if stress corrosion cracking of the shrunk-on low pressure discs occurs. Important parameters which affect missile generation probability are material properties, operating stresses, and temperature. The proposed Uprating Program for Diablo Canyon Power Plant Unit 1 does have an impact on missile generation probability. Of the parameters which affect missile generation probability, the uprating affects only temperature. The rate of stress corrosion crack growth is a function of temperature, with crack growth rate increasing as temperature increases. In order to determine the effect of the uprating on missile generation probability, thermodynamic conditions in the LP at the current and uprated conditions were compared. The temperature of discs 2 through 6 at the uprated condition are slightly higher (1.5°F or less) than current conditions. This very small change would result in a slight increase in missile generation probability. Note, for the Diablo Canyon BB81 low pressure turbines, only discs 2 through 6 are considered in the analysis, because disc 1 is contained and would not become a missile at running speed or 120% overspeed.

If the effects of rotor refurbishment and uprating are combined, the large reduction in missile generation probability associated with the rotor refurbishment described above more than offsets the small increase from the uprating. The net effect is a reduction compared to previously documented values. Therefore, the current missile generation probabilities are applicable, and conservative, for the uprating conditions.

8.5 GENERATOR AND EXCITER EVALUATION

The generator for Diablo Canyon Power Plant Unit 1 is a hydrogen cooled turbine generator with a water cooler stator winding. The generator converts the mechanical power from the steam turbine into electrical power. Its capacity is determined by its cooling system, the excitation design, and physical limitations of the windings. The maximum Megawatt-electric rating is a function of the excitation characteristic, described by the Mega VARs generated (overexcited or boosting the system) or consumed (underexcited or bucking the system).

A study of the generator and exciter was performed to determine the impact of the increase in thermal rating from 3350 MWt to 3425 MWt, which corresponds to a maximum 1163.468 MW output electrical. At 1160 MWe, the generator curve envelopes down to 0.9 PF boosting and 0.95 bucking. Actual operation is very close to 1.0 due to the local grid characteristics.

9.0 ENVIRONMENTAL AND PERMIT EVALUATION

The Unit 1 uprate will not result in any change to the DCPP Environmental Protection Plan; however, it will reduce the margin between DCPP performance and the allowable heat rejection to the Pacific Ocean. DCPP is allowed a maximum of 22°F between the circulating water intake and outflow. Because the circulating water intake and outflow mix together between the two units, a 2.2% uprate of Unit 1 will tend to increase the temperature change by 1.1%, or approximately 0.2°F.

It is noted that in the past DCPP temperature changes have come within 0.2°F of the 22°F value, but the frequency of such an impact is unknown. Such an event is unlikely to cause a violation of the NPDES permit limit because the temperature differential between intake and outflow is continuously monitored, and a decrease from uprated power would occur before a violation of the 22°F limit.

10.0 SECURITY PLAN AND EMERGENCY PLAN

A key part of DCPP's licensing are the security plan and emergency plan. All plant changes must be reviewed for potential impacts on the security plan and emergency plan.

Uprating Unit 1 will have minimal impacts on plant operation, as discussed in Section 7.17. Unit 2 is already operating at the higher power level. The physical uprating of the plant will not involve an influx of workers or otherwise cause personnel changes. The additional core energy is not an impact, since the emergency plan already addresses events at the higher power level of Unit 2 (dose assessment source impacts are discussed in Section 3.5). Thus, the Unit 1 Uprating Program will not impact the security plan or emergency plan.

Addendum to WCAP- 14819, "Pacific Gas and Electric Company Diablo Canyon Power Plant, Unit 1 3425 MWt Uprating Program Licensing Report."

Since much of the work that is summarized in WCAP-14819, "Pacific Gas and Electric Company Diablo Canyon Power Plant, Unit 1, 3425 MWt Uprating Program Licensing Report," was completed in 1997, a review was performed to identify the changes to the plant since then that could affect the WCAP. This addendum addresses changes in the Diablo Canyon licensing basis since WCAP-14819 was written, and provides greater detail about certain aspects of the uprating evaluation that were the subject of NRC requests for additional information in the review of uprate requests at other facilities.

1. Codes and Methodologies used: A complete loss-of-coolant accident (LOCA) re-analysis was performed which used codes not previously applied to the Diablo Canyon Power Plant (DCPP). Specifically, the large break LOCA uses best estimate methodology and the small break LOCA analysis used the COSI condensation model. Due to these methodology changes, and due to a commitment made to the NRC to update the DCPP LOCA analyses, the large break and small break LOCAs have been submitted separately to the NRC. The large break LOCA reanalysis has been approved in License Amendments (LAs) 121 and 119.

   The remaining evaluations for the Unit 1 uprate did not require new methodology or codes. Most current licensing basis analyses are common analyses that envelope both Unit 1 and 2. These generally assumed the lower Unit 1 reactor coolant system (RCS) flow rate in combination with the higher Unit 2 power. Thus most already envelope Unit 1 at the same power level as Unit 2. The only two evaluations requiring further analytical work for the uprate were the overtemperature and overpower ΔT (OTΔT/OPΔT) reactor trip setpoint calculations and accidental depressurization of the RCS. These particular analyses were previously performed at unit specific conditions, i.e., the higher Unit 2 power analysis credited the higher Unit 2 flow rate.

   - Calculations were performed to confirm the adequacy of the current Unit 1 OTΔT and OPΔT setpoints. The calculations are based on exclusive use of 17x17 Vantage 5 fuel since DCPP has no expectation of using 17x17 standard fuel. This allows slightly higher reactor core safety limits as shown in Technical Specification (TS) Figure 2.1.1-1. The setpoint calculations also assumed the uprated Unit 1 nominal full power T_{avg}, which was increased from 576.6°F to 577.3°F. With these changes, the results show that the current TS OTΔT/OPΔT setpoints and f(ΔI) penalty function are adequate to bound Unit 1 at the uprated conditions. These setpoint calculations used the previously approved methodology of WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986, and no new codes or methods were required.

- Accidental depressurization of the RCS is assumed to be the result of a failed open pressurizer safety valve. It is analyzed using the LOFTRAN code and the improved thermal design procedure as described in the Final Safety Analysis Review (FSAR) Update Section 15.2.13. The current analysis is performed for each Unit separately, so a new reanalysis was performed which conservatively bounds both units. As in the case of the OTΔT and OPΔT setpoints, the new analysis used revised input assumptions including the exclusive use of 17x17 Vantage 5 fuel, but no new codes or methods were required.

In addition, the residual heat removal (RHR) cooldown calculation was reperformed and documented in WCAP-14819; however, the analysis was redone mostly to add margin for issues related to the component cooling water (CCW) system rather than in response to the uprate. The RHR cooldown calculation is performed to demonstrate that the system meets design criteria of cooling down to 140°F when both trains are available in 20 hours, and to 200°F with one train in 36 hours. This is not a design basis accident analysis and there are no safety-related consequences should the cooldown exceed the time specified. The calculation is identical for both units since the RHR system and CCW system are the same, and a bounding decay heat is assumed. The reanalysis used more conservative assumptions than the previous analysis including higher heat loads and lower flow rates to bound a larger spectrum of operating conditions. As a result, the new RHR cooldown calculation indicates a longer required time to perform the cooldown. This longer cooldown time is a consequence of the more conservative assumptions, not the uprate, since the assumed decay heat has always enveloped the 3411 MWt of Unit 2. Although the analysis assumed more conservative analysis inputs, the RHR cooldown calculation involved no new codes or methodologies.

2. The large break LOCA analysis has been separately submitted and approved. The analysis utilizes Westinghouse best estimate methodology as described in WCAP-12945-P-A. Best estimate methodology utilizes the best estimate of certain key parameters with parameter ranges specified to envelope all expected values. The analysis uses a Monte Carlo process to determine the 95 percent confidence limit for peak clad temperature (PCT) to satisfy 10CFR50 Appendix K analysis requirements. Previous large break LOCA PCT analyses of record indicated PCTs of 2042°F at Unit 1 and 2071°F at Unit 2. The best estimate model analysis of record predicts the single bounding value of 1976°F for both

units. The large break LOCA analysis was approved by the NRC in 1998 in LAs 121 and 119 .

3. The small break LOCA analysis results were submitted to the NRC in December of 1998. The analysis utilizes the COSI Condensation Model from an addendum to WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code." In that submittal, PG&E requested a revision to TS 6.9.1.8, "Core Operating Limits Report," to allow use of any applicable NRC approved addenda to WCAP-10054-P-A to determine the core operating limits. At the NRC Staff's request, in PG&E Letter DCL-99-099, "Supplement to License Amendment Request 98-09," dated July 30, 1999, PG&E limited the requested change to just the use of WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and COSI Condensation Model," July 1997. In LAs 136 (Unit 1) and 136 (Unit 2), dated November 15, 1999, the NRC staff found the use of WCAP-10054-P-A, Addendum 2, Revision 1, acceptable for use in DCPP licensing applications, including reference in TS 6.9.1.8 and the Core Operating Limits Report (COLR). The small break LOCA analysis predicts a PCT of 1304°F for Unit 1 and 1293 °F for Unit 2.

4. The fuel is assumed to have all Zirlo cladding. This is consistent with Vantage 5+ fuel. The calculation used to generate the analysis inputs, PG&E Calculation STA-031, "Input Data for Unit 1 Uprate Project and Units 1 & 2 Loss of Coolant Accidents," page 14, states, "Future fuel assemblies will have a change in material that caused us to question whether there would be possible effects (Zirlo will be used rather than Zirc-4). Westinghouse states in the 24 month Cycle Safety Evaluation (in draft form in PGE-95-611) that the Zirlo is a small 2 F PCT penalty for Unit 1. It was decided that Zirlo would be modeled both because of this Unit 1 penalty, and because eventually the core will be all Zirlo." The large break LOCA and small break LOCA analysis results which incorporate these fuel cladding impacts have been submitted to the NRC separate from this uprate license amendment request (LAR).

5. The impact of the uprate on electric grid stability is not addressed in the WCAP 14819 licensing report. The Unit 1 uprate will increase the total plant power output to the grid by only 1.1 percent. PG&E engineers have reviewed the uprate and determined that it will have no significant impact on grid stability.

6. The impact of the uprate on RCS hot leg thermal streaming is not addressed in the licensing report. The measured RCS flow rate is compared to the TS required minimum measured flow by performing a precision flow calorimetric test. Hot leg streaming can potentially increase the inaccuracy in the hot leg

temperature measurement. PG&E has monitored hot leg streaming and its impact on RCS flow measurement for several fuel cycles. The evidence is that hot leg streaming causes the measured RCS flow rate to appear less than the actual flow rate. This represents a conservative penalty with respect to the tests that demonstrate compliance with the RCS minimum measured flow. The 2.2 percent power uprate is not expected to measurably change any Unit 1 hot leg streaming effects and there should be no significant impact on the associated RCS flow measurement accuracy.

7. The DCPP high energy line break calculations are already consistent with the uprated power level.

8. The WCAP-14819 licensing report was a joint document with some sections written by PG&E and some by Westinghouse. The radiological doses summary on page xvi reflects only the Westinghouse scope of steam generator tube rupture dose assessment, but Section 3.5.2 appropriately summarizes PG&E's review of the FSAR Section 15.5 Dose Assessments. These analyses assume source terms consistent with the original proposed Unit 2 ultimate power level of 3568 MWt, which remain bounding for the Unit 1 uprating.

9. WCAP-14819 Section 2.3, items 3c through 3e, list three control systems that will be evaluated for potential rescaling as part of the Unit 1 uprate. Rescaling any of these control systems will not impact any safety analyses or the licensing basis because they will still remain bounded by the limiting range of values already assumed in safety analyses. The parameters are:

   - pressurizer level, which may change due to the revised nominal full power T_{avg}
   - digital feedwater control system parameters
   - constants in the control rod insertion alarms, which may also change to due to the revised nominal full power T_{avg}.

10. Control room habitability is addressed in the FSAR in Chapter 9 (ventilation and filters), Chapter 12 (shielding), and Chapter 15 (Accident Analysis Section 15.5.17.10). PG&E is actively involved in industry efforts to address recent concerns on this subject and is reviewing the August 1999 draft of NEI 99-03, "Control Room Habitability Assessment Guidance." WCAP-14819, Section 3.5, does not specifically evaluate control room habitability; however, the WCAP conclusions regarding the assumed source terms apply. As shown in FSAR Table 15.5-32, the assumptions used to calculate control room radiological exposures include an assumed power level of 3580 MWt. Since the control room habitability studies already assume a higher power level than the proposed uprate, DCPP will continue to be bounded.

11. Section 3.6, Post-LOCA Hydrogen Generation, in WCAP-14819 was written based on a 1996 calculation documented in Westinghouse REA Calculation REAC-PGE-079, Rev 1. That work was superseded in June of 1997 by a calculation documented in Westinghouse PGE-97-559. The 1997 calculation bounds the Unit 1 uprating. The changes include:

    - conservatively bounding a greater RCS hydrogen concentration (60 cc/kg rather than 50cc/kg).
    - bounding a greater Zinc inventory in containment.
    - limiting the aluminum inventory allowed in containment (3576 lbm versus 4076 lbm).

    The only text revision required in the licensing report due to this revised analysis is to increase the maximum assumed RCS hydrogen concentration from 50 cc/kg to 60 cc/kg as listed on page 3-49. The licensing report conclusions regarding the maximum bulk containment hydrogen concentration are still valid.

12. PG&E has performed an additional evaluation of the containment spray system (CSS) that is subsequent to the uprate conclusions presented in WCAP-14819 Section 4.1.4. In order to ensure seismic integrity, both Unit's CSSs have been modified to include a valve which keeps the spray lines empty until a spray initiation signal is generated. The PG&E evaluation has evaluated this modification and determined that the containment spray pumps can still provide the required flow at the discharge nozzles within the time frame assumed in the accident analyses. Since both Units had CSS flow requirements established based on the original Unit 2 higher power level of 3411 MWt, the Unit 1 uprate does not impact any CSS performance criteria.

13. WCAP-14819 Section 4.2.3 states that the spent fuel pool (SFP) system function is not safety-related. This statement only means that the SFP is not required to perform any active functions during the mitigation of a design basis accident. However, the DCPP SFP cooling system is designed Seismic Category I, and Design Class I, to ensure system pressure boundary integrity. The SFP also has a Design Class I backup makeup water source to maintain adequate water inventory during a postulated loss of pool cooling event. WCAP-14819 appropriately identifies that the Unit 1 SFP cooling system is demonstrated to be adequate for the Unit 1 uprate based on the successful operation of it's identical counterpart for Unit 2's current 3411 MWt rating.

14. WCAP 14819, Section 5.1, describes the steam generator (SG) evaluations performed for the Unit 1 uprated power conditions. DCPP has an aggressive SG strategy plan that closely monitors SG conditions in order to accurately predict

any future degradation. The Unit 1 SG conditions at the uprated power level will not be substantially different from the current Unit 2 conditions. Based on this, the following three NRC questions can be addressed:

14.1. The licensee needs to evaluate the effects of the power uprate on tube degradation mechanisms (present and potential) including wear, and provide a basis for the evaluation results.

The present DCPP Units 1 and 2 tube degradation mechanisms are:

- primary water stress corrosion cracking (PWSCC) at dented tube support plate (TSP) intersections, Rows 1 and 2 U-bend region, and tubesheet region
- outside diameter stress corrosion cracking (ODSCC) at TSP intersections and tubesheet region
- thinning at cold leg TSP intersections
- anti-vibration bar (AVB) wear in the U-Bend region

Free span cracking (other than U-bends) has not been detected, but is a potential degradation mechanism that would be detected by current eddy current inspection techniques.

PG&E has evaluated the effects of power uprate on the present degradation mechanisms.

<u>Stress Corrosion Cracking</u>

The primary tube degradation mechanisms at DCPP are related to stress corrosion cracking (SCC). The dominant factors for SCC are temperature and water chemistry, while thermal power level is not an input to degradation model predictions. This is consistent with past experience since both Unit 1 and 2 operate with similar T_{hot} values and show similar SCC degradation rates despite the different power levels. Primary and secondary water chemistry will not be changed by the uprate. The Unit 1 uprate will be accomplished with minimal changes to T_{hot}. Westinghouse best estimate calculations show that the uprate can be accomplished with T_{hot} near its current temperature of approximately 603°F with an average tube plugging level up to 6.5 percent (currently Unit 1 has 3.4 percent of tubes plugged). Should tube plugging levels increase above ~6.5 percent, the plant will maintain a desirable T_{hot} by reducing power, sleeving SG tubes or performing turbine modifications. In summary, PG&E recognizes the importance of limiting T_{hot} on SG integrity and will maintain it near the current value to ensure the uprate will have no adverse impact on SCC rates.

Cold Leg Thinning

Currently Unit 1 has 39 tubes plugged for cold leg thinning (CLT) and Unit 2 has 35 tubes plugged for this mechanism. The progression of CLT has been similar in both units despite the different power levels. CLT occurs in a limited number of tube locations, confined largely to the peripheral cold leg region at the lower TSP locations. It develops at a slow rate and is limited to the confines of the TSP. Contributory factors are not well understood, however, power level is not considered a factor in degradation rates. Therefore, it is not expected that the power uprating will have any significant effect on the rate of CLT progression in Unit 1.

AVB U-bend Wear

The Unit 1 uprate will proportionally increase the steam flow rate and reduce SG pressure. Westinghouse has performed evaluations to assess the effect of power increases and/or steam pressure reductions on U-bend wear at the AVBs for various SG models, including the DCPP Model 51 design. The Westinghouse evaluation included a 5 percent increase in power and a 110 psi reduction in steam pressure, which are significantly greater than that predicted for the Unit 1 uprating. The Model 51 evaluation results indicate a potential increase of only one additional plugged tube per SG. This result is consistent with operating experience, since the model 51 SGs have shown minimal susceptibility to U-bend wear. Therefore the power uprating is anticipated to have no significant effect on U-bend AVB-wear for DCPP Unit 1.

Tube Fatigue Considerations

A Westinghouse qualitative evaluation of the flow induced U-bend tube vibration for 100 percent power conditions demonstrated that a reduced steam pressure of 760 psia is slightly more limiting than for the current reference conditions. The Unit 1 uprating is not expected to reduce steam pressure below 760 psia since this value is associated with maximum tube plugging conditions. As long as the steam pressure remains above 760 psia, the previous evaluation and recommended actions remain valid. No changes for the uprate are required since PG&E has already identified 760 psia as the minimum analyzed SG pressure.

14.2 Discuss how the SG tube inspection plan will be assessed to monitor potential tube degradation including wear. Will additional inspections be necessary?

In accordance with plant procedures, 100 percent of the DCPP SG tubes are required to be inspected full length with bobbin coil every refueling outage, and rotating coil inspections are required for critical areas of the tube bundle where SCC has been detected. Critical areas are evaluated and defined taking into account the latest inspection results. In accordance with PG&E's commitment to NEI 97-06, condition monitoring is performed to assess tube integrity for the as-found condition of the bundle, and operational assessment is performed to assess tube integrity for the next cycle of operation, taking into account degradation growth rates. Growth rates are evaluated and updated after every inspection.

As per the answer to the previous question, there is no reason to believe that the degradation rates will be affected by the uprating. The impact on AVB wear and CLT is negligible, and maintenance of T_{hot} near its current value should not increase SCC degradation growth rates beyond their present values. The operational assessment will account for any unlikely increase in degradation growth rates. Therefore the existing Unit 1 SG tube eddy current inspection plan will be adequate for the uprated Unit 1. No additional tube inspections are required.

14.3 The licensee needs to evaluate if the TS plugging limit of 40 percent through wall degradation is still adequate and provide a basis for the evaluation results. The licensee needs to evaluate the effects of the power uprate on approved alternate repair criteria (ARC) and provide a basis for evaluation results.

The TS plugging limit of 40 percent is applied to all degradation that is inspected with qualified sizing techniques, such as AVB wear, CLT, and axial PWSCC at dented TSP intersections. NRC-approved ARC are applied to axial PWSCC in the WEXTEX tube sheet region (W* ARC) and axial ODSCC at TSP intersections (GL 95-05, voltage-based ARC). All other degradation is plugged on detection or confirmation using rotating coil probes.

As part of the operational assessment, the 40 percent plugging limit is verified to contain burst margin from the structural limit, taking into account nondestructive examination (NDE) uncertainty and degradation growth. The uprate will have no affect on structural limits for free span degradation (e.g., low row U bend PWSCC and U-bend AVB wear) because these limits are based on conservative 3 dP values (steam pressure as low as 759 psia). The uprate has no affect on NDE uncertainty. As discussed earlier, growth rates should not be affected. Therefore, the 40 percent plugging limit is adequate for the uprated condition.

Similarly, for degradation subject to ARC, the repair limits are assessed prior to each implementation to account for potentially changing growth rates and cycle lengths. For GL 95-05, voltage-based ARC, the upper repair limit is reviewed prior to each inspection and revised as necessary to account for changes in cycle length and voltage growth rates. For W* ARC, crack length growth is added to the as-found upper crack tip to ensure that the crack tip remains below the top of tubesheet during the ensuing cycle. Both of these ARC apply to degradation confined to support structures, so 3 dP criteria are not limiting. Rather, structural limits are based on faulted conditions (1.4 SLB dP for GL 95-05 ARC and 1.4 FLB dP for W* ARC). Therefore, reduction in operating steam pressure is not a factor in these structural limits.

15. The turbine missile generation analysis is described in Section 8.4. In addition, PG&E reviewed WCAP-11525, "Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency," and LAs 42 and 41 which reduced the turbine control valve testing frequency on the basis of a low probability of missile generation. PG&E has committed to review the testing frequency probability analysis every 3 years or any time that major changes are made in the turbine system. The next 3 year review is due in May 2000. The uprate does not impact the WCAP-11525 methodology directly, since the unit rating is not an input to the analysis. Furthermore, since the NRC reviewed the license amendment in 1988, DCPP has replaced its light disc keyway rotors with heavy disk keyplate types. WCAP-11525 states that this is a significant factor in reducing missile generation probability. Therefore, the small increase in missile generation probability related to the estimated 1.5°F temperature rise is more than offset by the rotor improvements.

16. Section 9.0 of WCAP-14819, "Environmental and Permit Evaluation," discusses the thermal discharge only, since all other environmental impacts of the 1.1 percent increase in total site power generation will be negligible. Unlike a fossil fuel burning facility, an increase in nuclear power does not cause a proportional increase in effluent emissions. No additional component operation is required and there will be no increase in sea water flow through the condenser or the required auxiliary salt water system flow. DCPP operates in compliance with National Pollution Discharge Elimination System (NPDES) Permit CA0003751. Although no increase is expected from the uprate, all applicable

effluents are closely monitored to assure compliance with the permit limits. Similarly, there is no reason to assume an increase in the release of radioactive materials.

17. DCPP, like many other plants, reduced the maximum boron concentration in response to GL 85-06 from 12 percent to 4 percent to minimize the potential for boron precipitation. The PG&E LAR was submitted July 5, 1989, in PG&E Letter DCL 89-182, "Reduce Boron Concentration in the Boric Acid System," and approved in LAs 53 and 52. The 4 percent boron concentration is verified to provide adequate shutdown margin and boration capabilities for each core reload as calculated by Westinghouse using the BORDER code. The 4 percent value should bound all Unit 1 uprated core designs since it has already been verified as adequate for the Unit 2 3411 MWt cores. Each uprated Unit 1 core design will have this verification performed as part of the reload process.

18. No specific testing is required following the uprate. There is no change in the design basis performance requirements for any systems or components and the existing surveillance testing remains adequate.

19. Control of Assumptions and Data: The uprate evaluation involved a comprehensive review of the Unit 1 design basis, to ensure compliance with all regulatory and design commitments at the increased power level. PG&E and Westinghouse worked closely throughout the evaluation process, and established the formal control and transmittal of all analysis input information related to the uprate. This was achieved using the PG&E Design Calculation Procedure, IDAP CF3.ID4, "Design Calculations." These analysis inputs are documented in Calculation STA-031, Rev 0, "Input Data for Unit 1 Uprate Project and Units 1 & 2 Loss of Coolant Accidents," dated August 26, 1996.

    A total of 116 input values were generated in STA-031, including several compiled data tables. Each input contains the following:

    - A label and title. For example, "NL9, Pressurizer Pressure Uncertainty," where NL implies this is used in non-LOCA transients.
    - A value or table of values. For example, 60 psi for the pressurizer pressure uncertainty.
    - A direction of conservatism where applicable, identifying the value as either maximum or minimum or best estimate. Pressurizer pressure uncertainty is described as "+", meaning larger values are conservative.
    - A source reference. The pressurizer pressure uncertainty is referenced to a 1993 Westinghouse letter.
    - Informational notes, including the contact engineer responsible for the accuracy of the value.

In addition, each input includes a discussion explaining the origination and basis for the value assumed . For example, the text discussion regarding the pressurizer pressure uncertainty describes an action request concerning the subject, and an explanation of why the assumed channel uncertainty is greater than that required in the TSs. The text notes that the TS shift verification is performed with more accurate control room gauges, as compared to the additional components and uncertainties contained in the pressurizer pressure control circuit. The comment concludes that the larger uncertainty assumed is conservatively bounding and is not an important impact on the final analysis results.

Calculation STA-031 contains 33 pages of tables and text, and includes 95 pages of supporting attachments. This positive method of input data control minimizes the possibility of error and provides a clear description of the thought process behind each value assumed.

MARKED-UP FACILITY OPERATING LICENSE

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This License shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

3411

The Pacific Gas and Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3338 megawatts thermal (100% rated power) in accordance with the conditions specified herein.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 111, are hereby incorporated in the license. Pacific Gas & Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan, except where otherwise stated in specific license conditions.

(3) <u>Initial Test Program</u>

The Pacific Gas and Electric Company shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Pacific Gas and Electric Company's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of PG&E's Final Safety Analysis Report as amended as being essential;

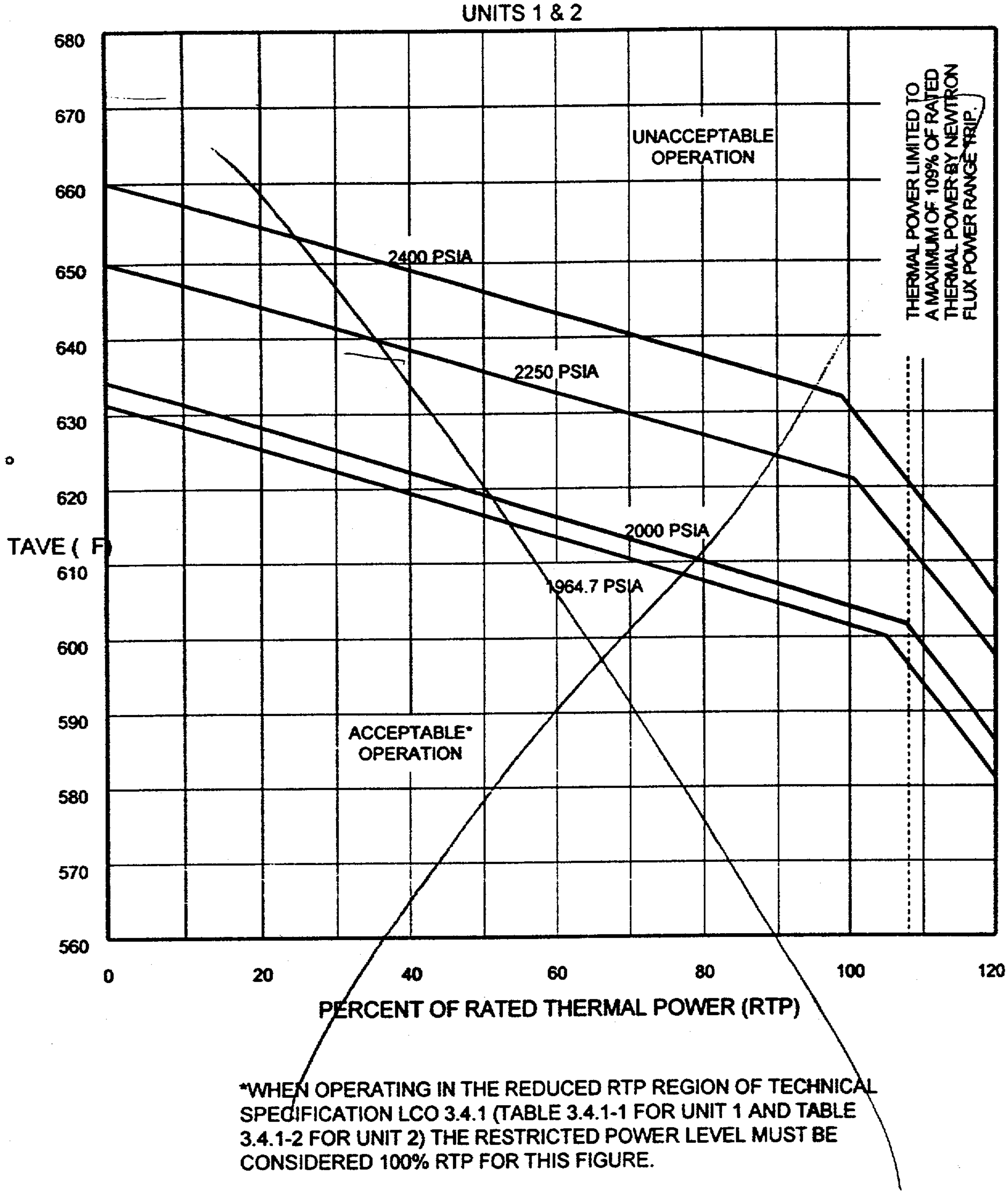
MARKED-UP TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
1.1-5	1.1-5
2.0-2	2.0-2
3.3-17	3.3-17
3.3-18	3.3-18

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve (PORV) lift settings and arming temperature associated with the Low Temperature Overpressurization Protection (LTOP) System, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3338 MWt for Unit 1 and~~ 3411 MWt ~~for Unit 2.~~ for both units.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

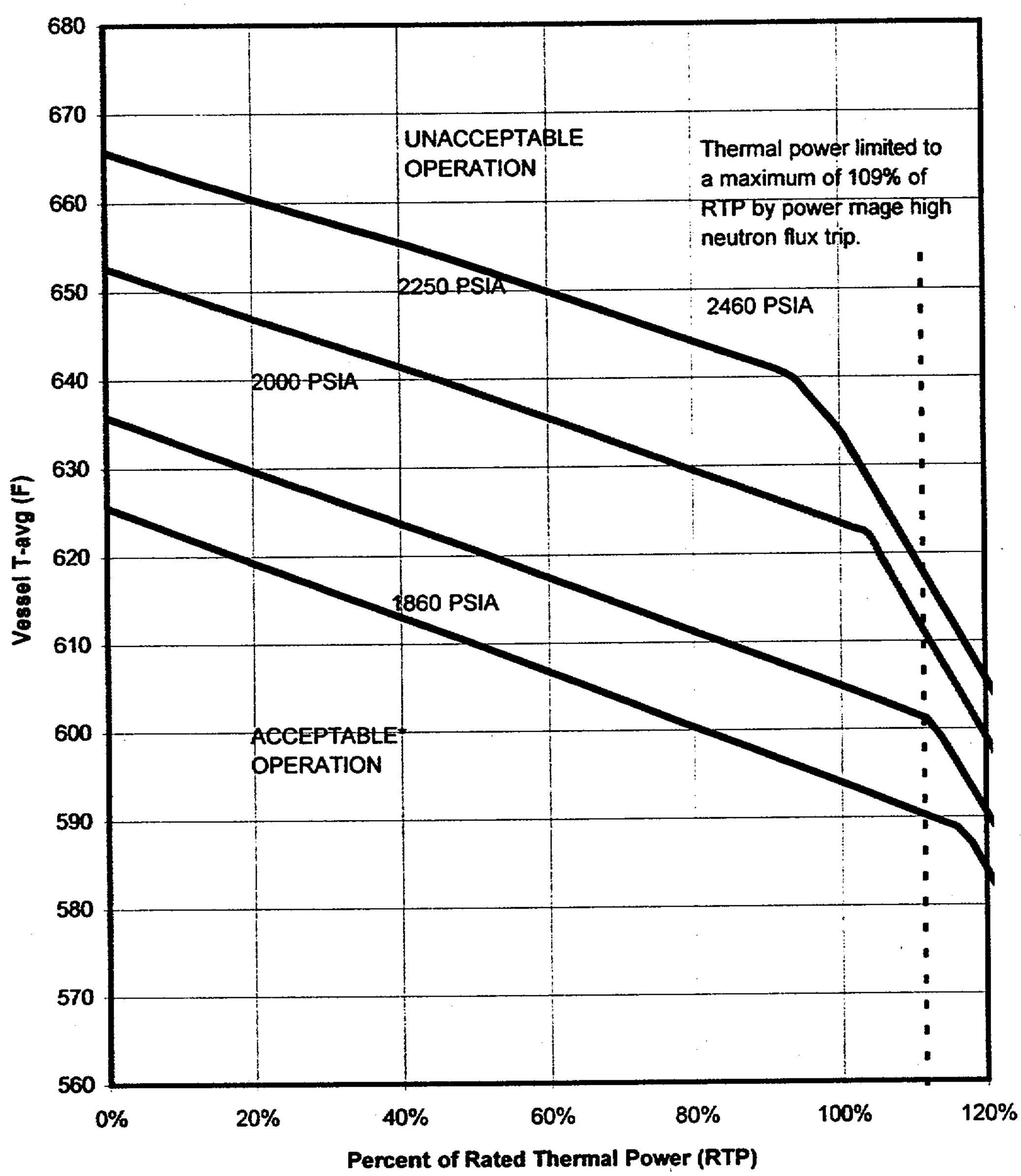
(continued)



*WHEN OPERATING IN THE REDUCED RTP REGION OF TECHNICAL SPECIFICATION LCO 3.4.1 (TABLE 3.4.1-1 FOR UNIT 1 AND TABLE 3.4.1-2 FOR UNIT 2) THE RESTRICTED POWER LEVEL MUST BE CONSIDERED 100% RTP FOR THIS FIGURE.

Figure 2.1.1-1

UNITS 1 & 2



*When operating in the reduced RTP region of Technical Specification LCO 3.4.1 (Table 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2) the restricted power level must be considered 100% for this Figure.

FIGURE 2.1.1-1
REACTOR CORE SAFETY LIMIT

Table 3.3.3-1 (page 6 of 7)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs, 0.14% ΔT span for pressurizer pressure input, 0.19% ΔT span for ΔI inputs.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \le \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[T - T'\right] + K_3(P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T' is the nominal loop specific indicated T_{avg} at RTP, ≤ 577.3~~576.6~~ (Unit 1) & 577.6 (Unit 2)°F.

P is the measured pressurizer pressure, psig

P' is the nominal RCS operating pressure, = 2235 psig

K_1 = 1.20	K_2 = 0.0182/°F	K_3 = 0.000831/psig
τ_1 = 30 sec	τ_2 = 4 sec	
τ_4 = 0 sec	τ_5 = 0 sec	
$f_1(\Delta I)$ =	$-0.0275\{19 + (q_t - q_b)\}$	when $q_t - q_b \le -19\%$ RTP
	0% of RTP	when -19% RTP $< q_t - q_b \le 7\%$ RTP
	$0.0238\{(q_t - q_b) - 7\}$	when $q_t - q_b > 7\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.3-1 (page 7 of 7)

Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than 0.46% of ΔT span for hot leg or cold leg temperature inputs.

$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T^0 \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s} T - K_6 \left[T - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T" is the nominal loop specific indicated T_{avg} at RTP, ≤ 577.3~~576.6~~ (Unit 1) & 577.6 (Unit 2)°F.

K_4 = 1.072

K_5 = 0.0174/°F for increasing T_{avg}; 0/°F for decreasing T_{avg}

K_6 = 0.00145/°F when T > T"; 0/°F when T ≤ T"

τ_3 = 10 sec

τ_4 = 0 sec

τ_5 = 0 sec

$f_2(\Delta I)$ = 0% RTP for all ΔI.

Note 3: Steam Generator Water-Level Low Low Time Delay

The Steam Generator Water Level-Low Low time delay function power allowable value shall not exceed the following trip setpoint power by more than 0.7% RTP.

TD = $B1(P)^3 + B2(P)^2 + B3(P) + B4$

Where: P = RCS Loop ΔT Equivalent to Power (%RTP), P ≤ 50% RTP

TD = Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds).

B1 = -0.007128 sec/$(RTP)^3$

B2 = +0.8099 sec/$(RTP)^2$

B3 = -31.40 sec/(RTP)

B4 = +464.1 sec

PROPOSED TECHNICAL SPECIFICATION PAGES

1.1 Definitions (continued)

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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

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QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt for both units.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

SHUTDOWN MARGIN (SDM)

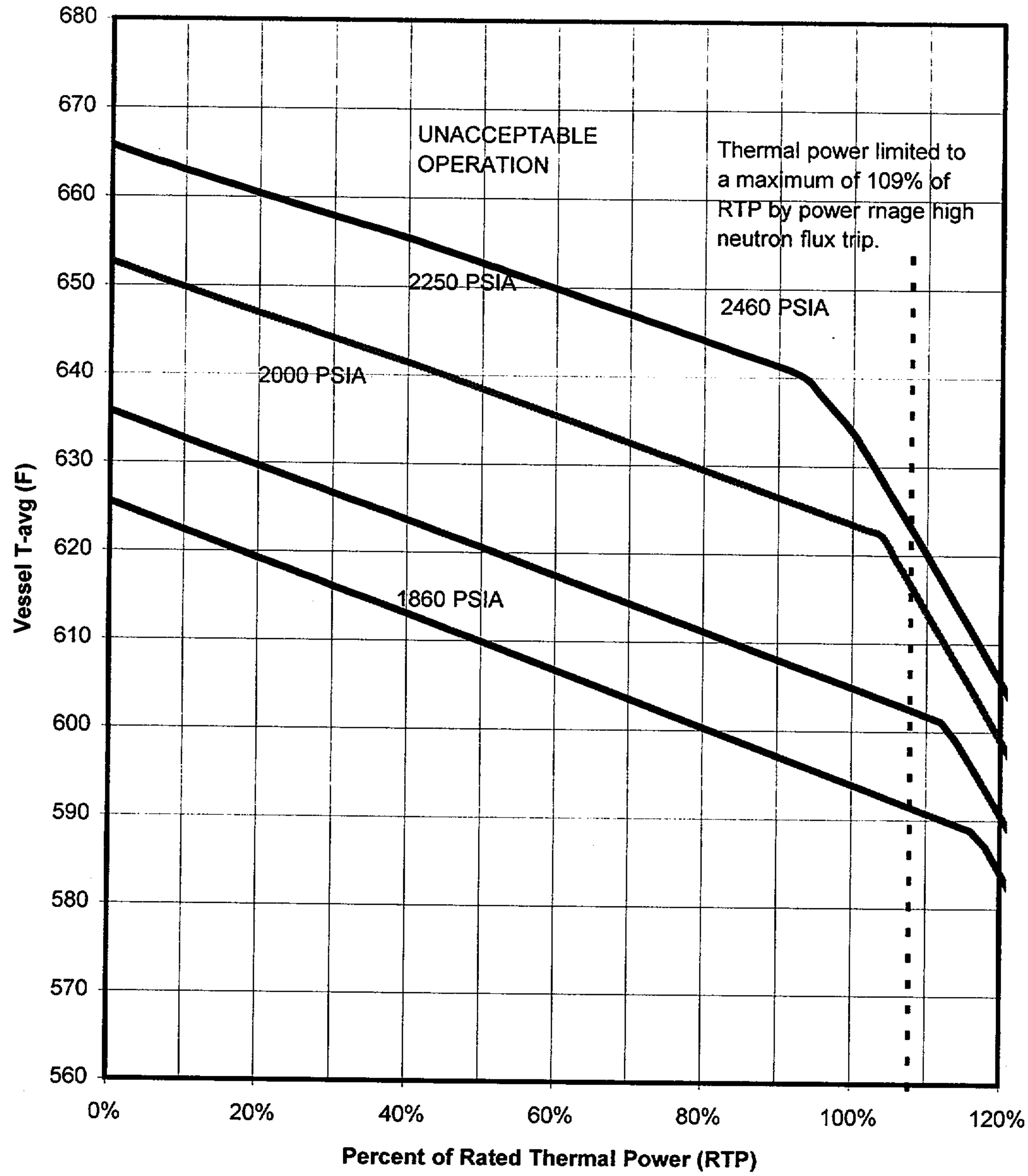
SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.

(continued)

UNITS 1 & 2



When operating in the reduced RTP region of Technical Specification LCO 3.4.1 (Table 3.4.1-1 for Unit 1 and Table 3.4.1-2 for Unit 2) the restricted power level must be considered 100% for this Figure.

FIGURE 2.1.1-1
REACTOR CORE SAFETY LIMIT

Table 3.3.3-1 (page 6 of 7)
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$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T_0 \left\{ K_1 - K_2 \frac{(1+\tau_1 s)}{(1+\tau_2 s)} \left[T - T' \right] + K_3(P - P') - f_1(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

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P is the measured pressurizer pressure, psig

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K_1 = 1.20	K_2 = 0.0182/°F	K_3 = 0.000831/psig
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τ_4 = 0 sec	τ_5 = 0 sec	
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	0% of RTP	when -19% RTP $< q_t - q_b \leq 7\%$ RTP
	$0.0238\{(q_t - q_b) - 7\}$	when $q_t - q_b > 7\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.3-1 (page 7 of 7)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

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$$\Delta T \frac{(1+\tau_4 s)}{(1+\tau_5 s)} \leq \Delta T^0 \left\{ K_4 - K_5 \frac{\tau_3 s}{1+\tau_3 s} T - K_6 \left[T - T'' \right] - f_2(\Delta I) \right\}$$

Where: ΔT is measured RCS ΔT, °F.

ΔT_0 is the loop specific indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.

T" is the nominal loop specific indicated T_{avg} at RTP, ≤ 577.3 (Unit 1) & 577.6 (Unit 2)°F.

K_4 = 1.072	K_5 = 0.0174/°F for increasing T_{avg} 0/°F for decreasing T_{avg}	K_6 = 0.00145/°F when T > T" 0/°F when T ≤ T"
τ_3 = 10 sec	τ_4 = 0 sec	τ_5 = 0 sec

$f_2(\Delta I)$ = 0% RTP for all ΔI.

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TD = Time delay for Steam Generator Water Level Low-Low Reactor Trip (in seconds).

B1 = -0.007128 sec/$(RTP)^3$

B2 = +0.8099 sec/$(RTP)^2$

B3 = -31.40 sec/(RTP)

B4 = +464.1 sec

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 were~~are~~ 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 existed~~s~~ 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

TABLE 4.1-1

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

TABLE 4.1-1

		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[g]	545.1[g]
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] ft²	~~(LOPAR)~~	~~59,742~~	~~59,742~~
	~~(V-5)~~	57,505	57,505
Average Heat Flux, Btu/hr-ft²	~~(LOPAR)~~	~~185,740585, 740~~	~~189,800~~
	~~(V-5)~~	197,180~~192, 960~~	197,180
Maximum Heat Flux for Normal[h] Operation, Btu/hr-ft²	~~(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 with 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 ΔH 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. typo

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); and by a line defining conditions at which the steam generator safety valves open.~~.~~

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

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

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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

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11. Westinghouse letter PGE-96-584, Diablo Canyon Units 1 & 2 Spurious Safety Injection Calculation Note, June 1996.

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

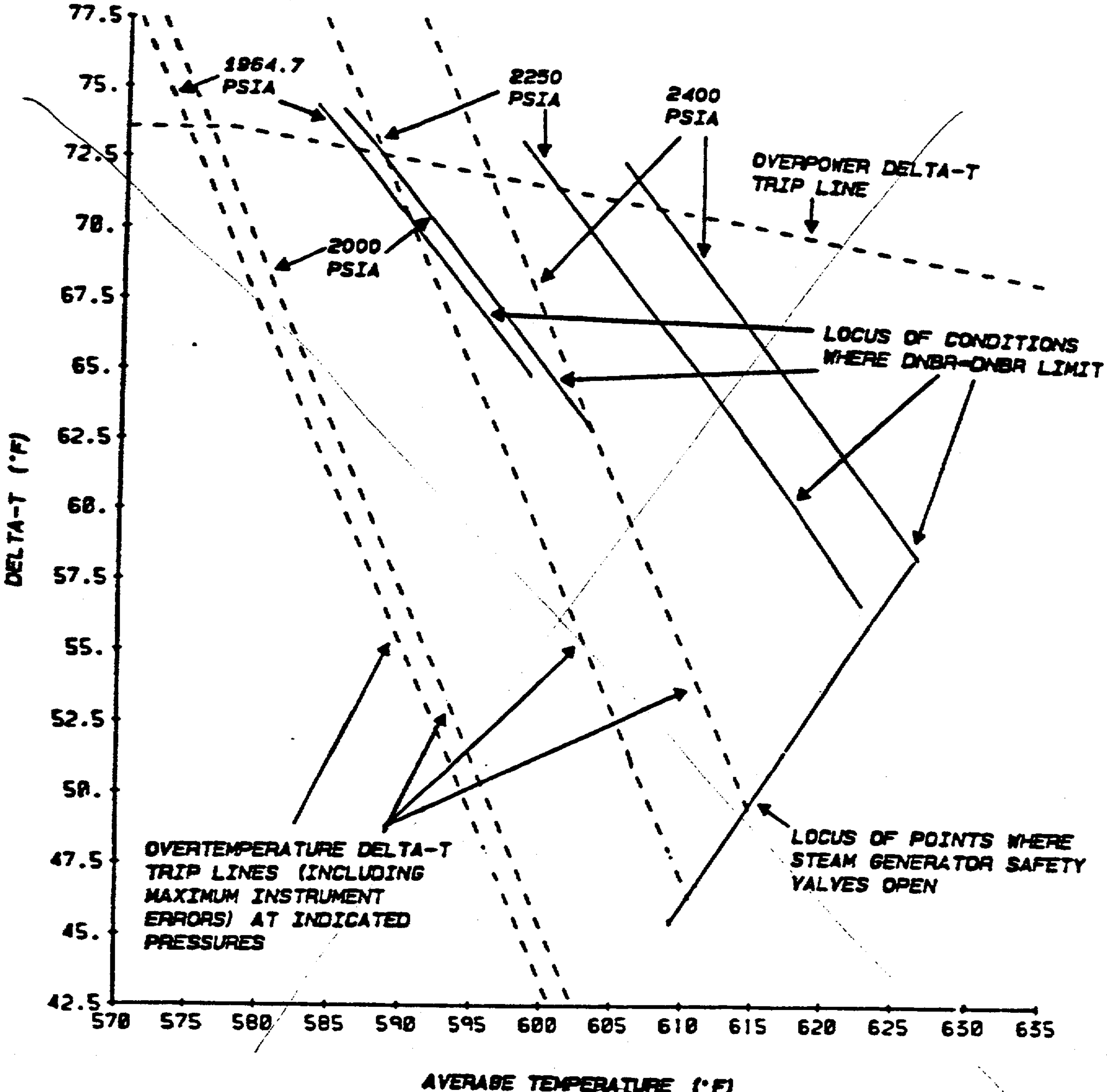
	~~Unit 1~~	~~Unit 2~~
Guaranteed core thermal power (license level)	~~3338~~	3411
Thermal power generated by the reactor coolant pumps minus heat losses to containment and letdown system[(b)]	~~14~~	14
Guaranteed nuclear steam supply system thermal power output[(b)]	~~3352~~	3425
The engineered safety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

TABLE 15.1-4

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp[a], pcm/°F[d]	Assumed Reactivity Coefficients: 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	+~~5~~7	-	Lower	342~~3~~5
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

Accident	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
3. 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

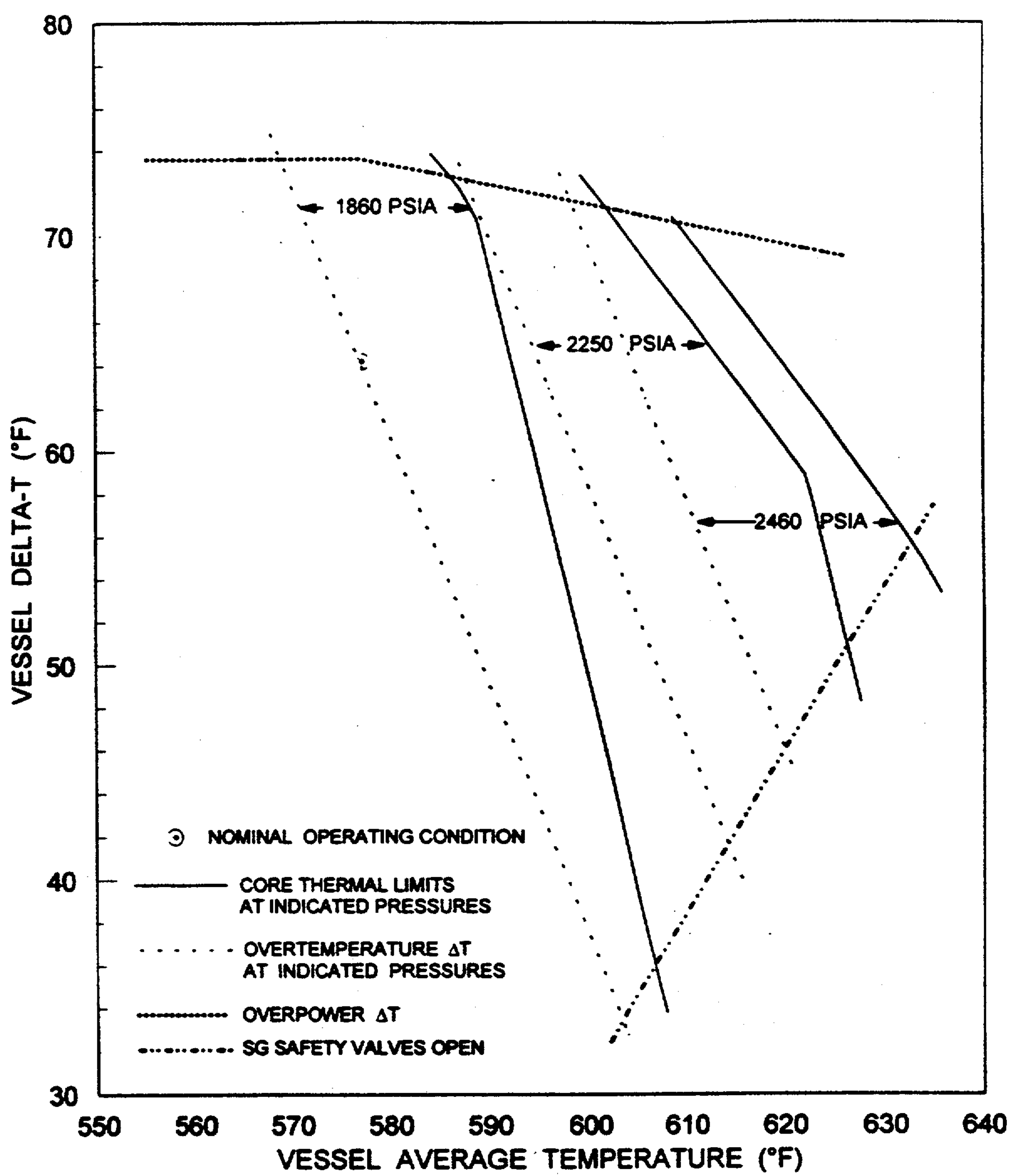


Remove and Replace with the Following

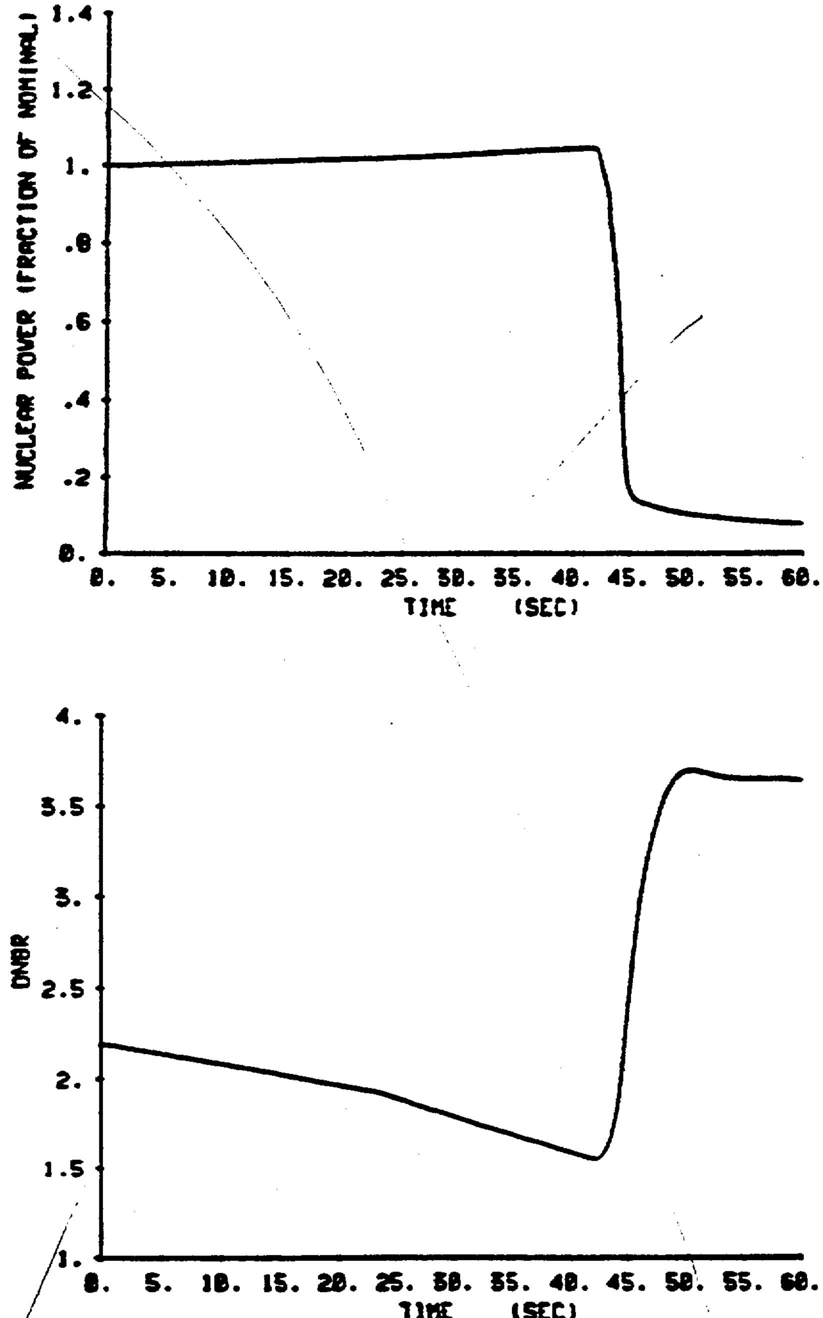
DIABLO CANYON UNITS 1 AND 2

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

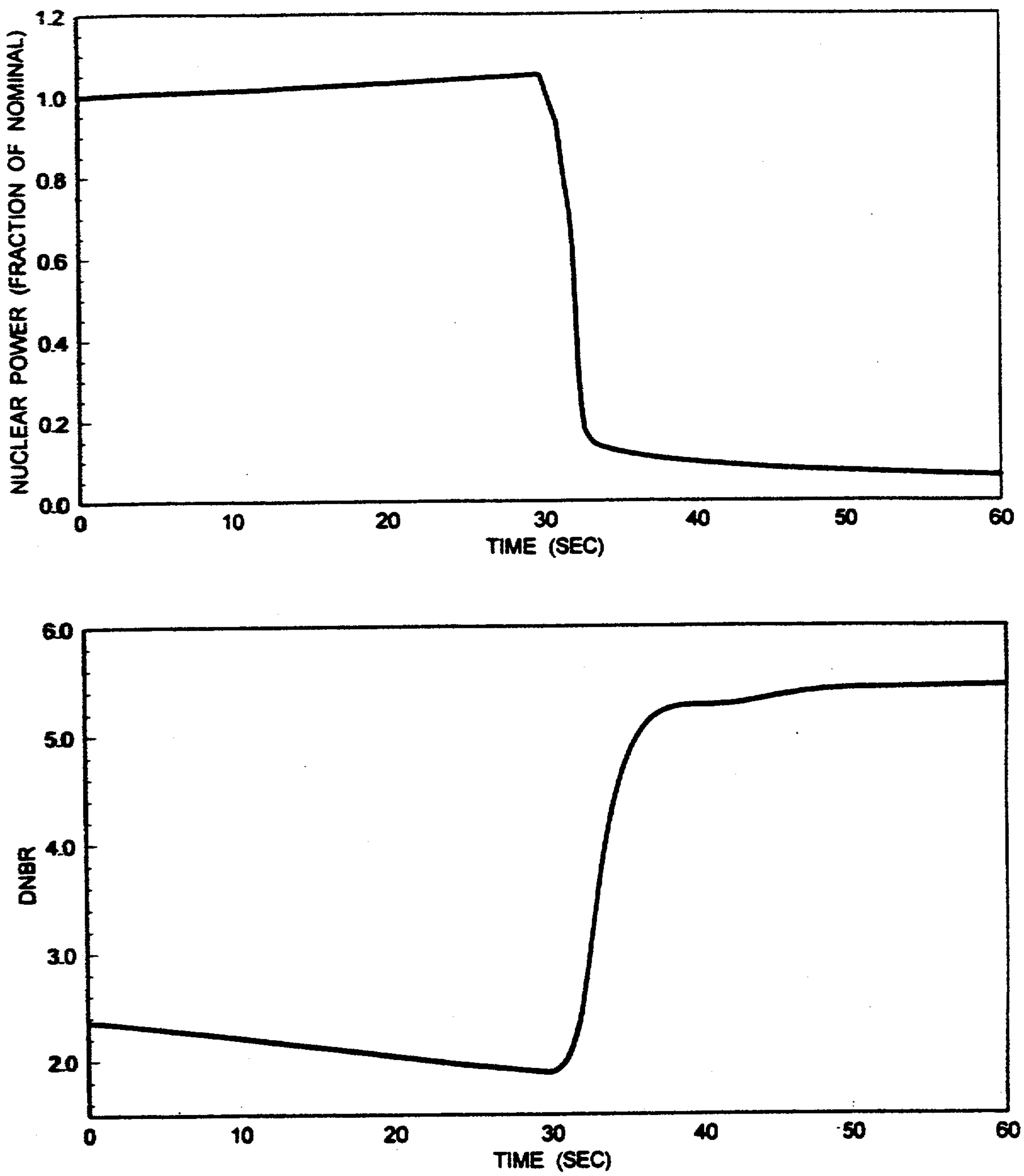


Replace with the Following

DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.12-1

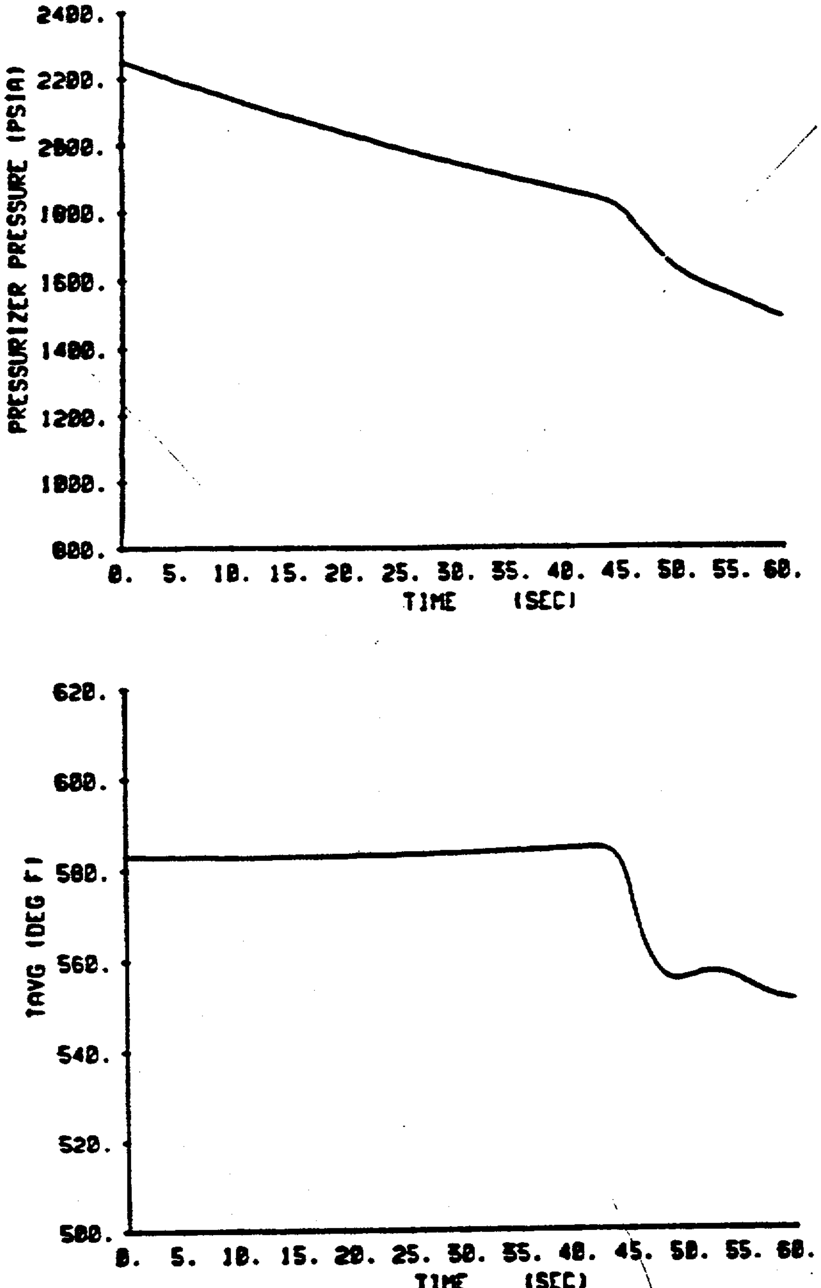
NUCLEAR POWER AND DNBR TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF REACTOR COOLANT SYSTEM



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.12-1

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

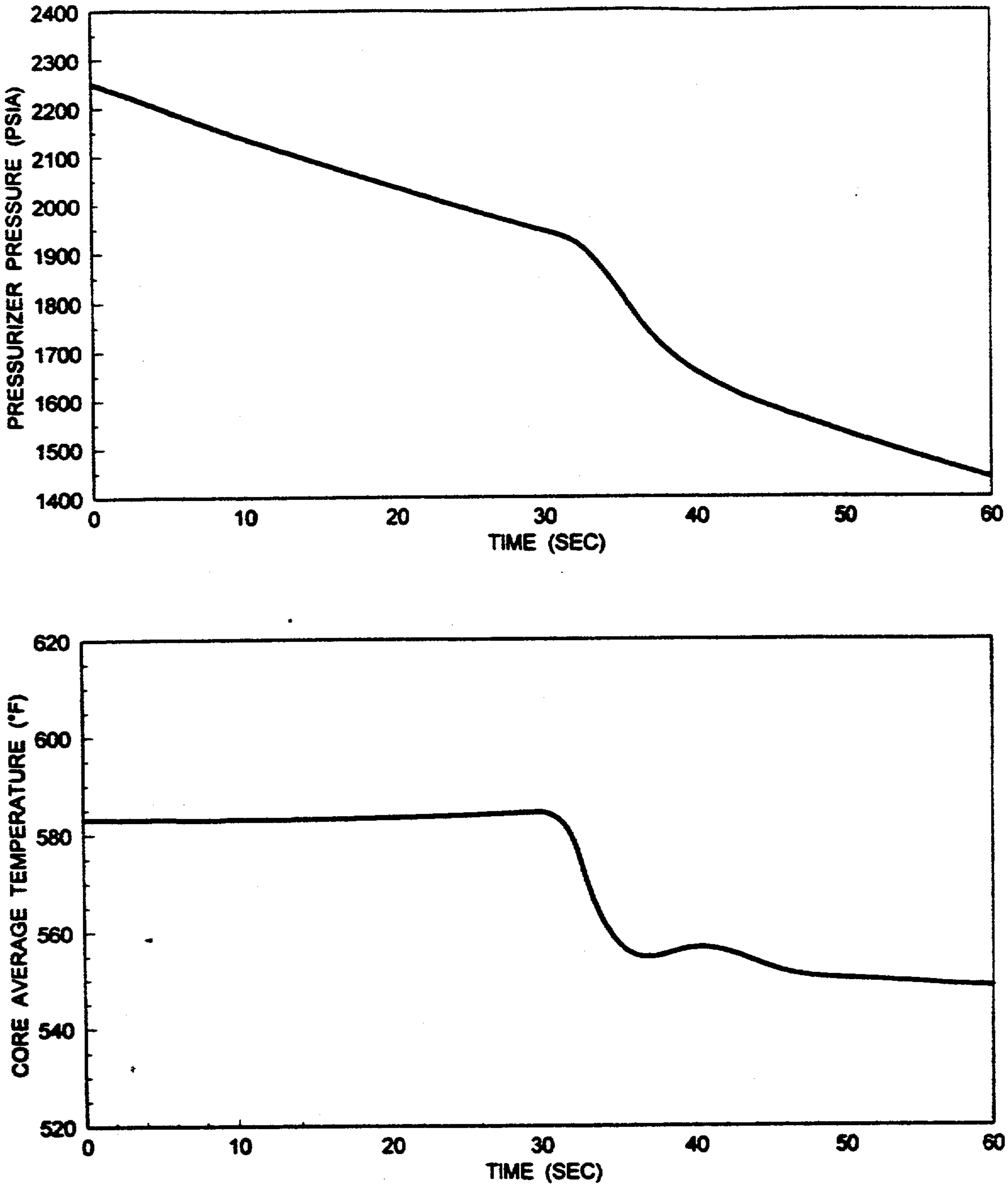


Replace with the Following

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

Revision 11 November 1996



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 ~~i~~are~~s~~ 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[20]. 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[23].~~

~~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[24].~~

~~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[25].~~

~~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[25]. 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[25].~~

~~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~~			
	~~3 in.~~	~~4 in.~~	~~6 in.~~	~~4 in.~~
~~Event~~	~~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 (%)	<1.0	<1.0	<1.0
Hot Rod Burst Time (sec)	No Burst	No Burst	No Burst
Hot Rod Burst Location (ft)	N/A	N/A	N/A

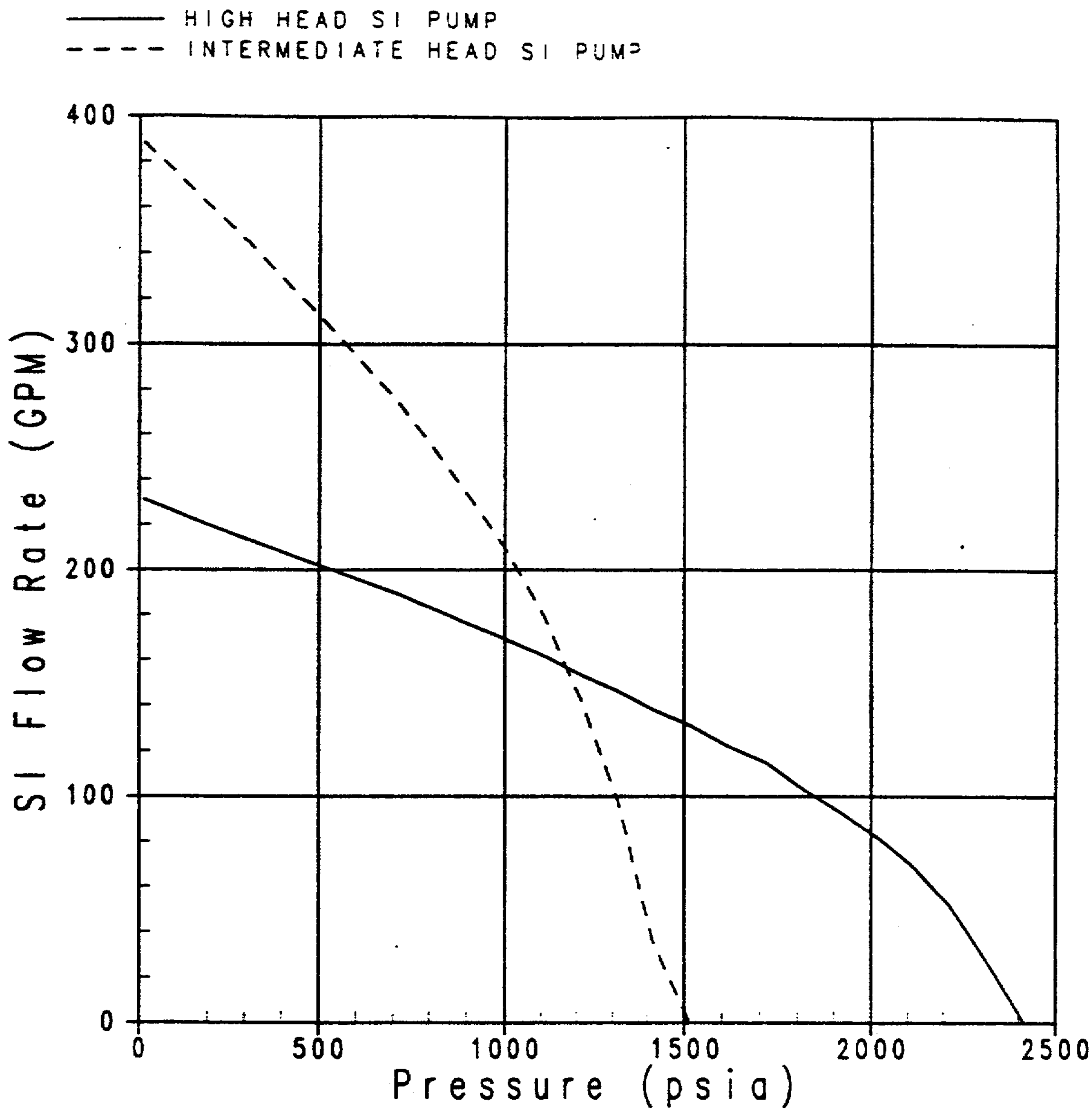
	UNIT 2 2-inch	UNIT 2 3-inch	UNIT 2 4-inch
Peak Cladding Temperature (°F)	955	1293	1225
Peak Cladding Temperature Location (ft)[1]	11.00	11.25	11.00
Peak Cladding Temperature Time (sec)	4371	1948	937
Local Zr/H_2O Reaction, Max (%)	0.03	0.25	0.07
Local Zr/H_2O Reaction Location (ft)[1]	11.00	11.25	11.00
Total Zr/H_2O Reaction (%)	<1.0	<1.0	<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

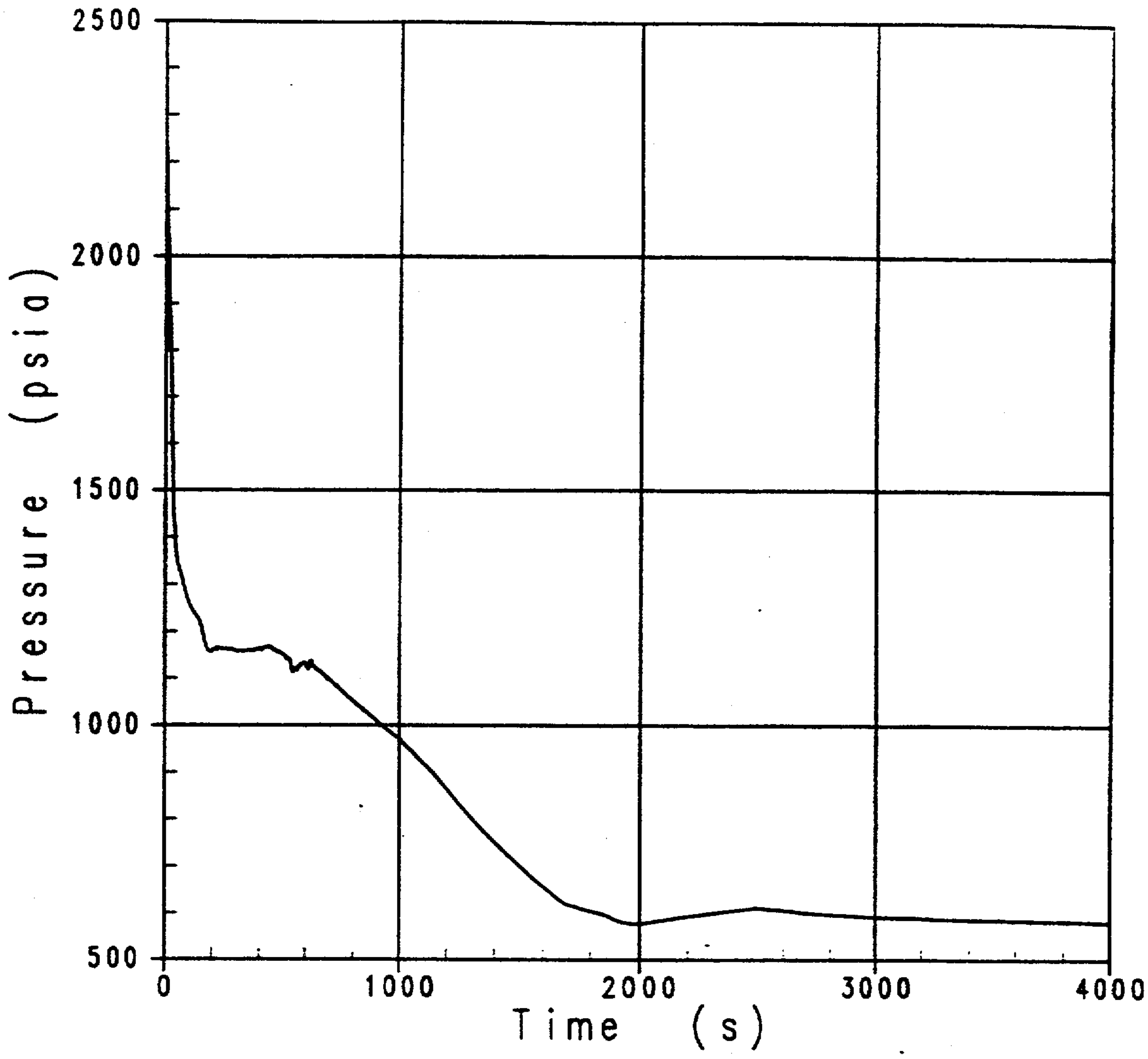
	~~Unit 2 Equivalent Break Size 3 in.~~	~~Unit 2 Equivalent Break Size 4 in.~~	~~Unit 2 Equivalent Break Size 6 in.~~	~~Unit 1 4 in.~~
~~Results~~				
~~Peak cladding temperature, °F~~	~~1023~~	~~1358~~	~~1099~~	~~1275~~
~~Peak cladding location, ft~~	~~12.0~~	~~12.0~~	~~12.0~~	~~12.0~~
~~Local Zr/H_2O 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~~

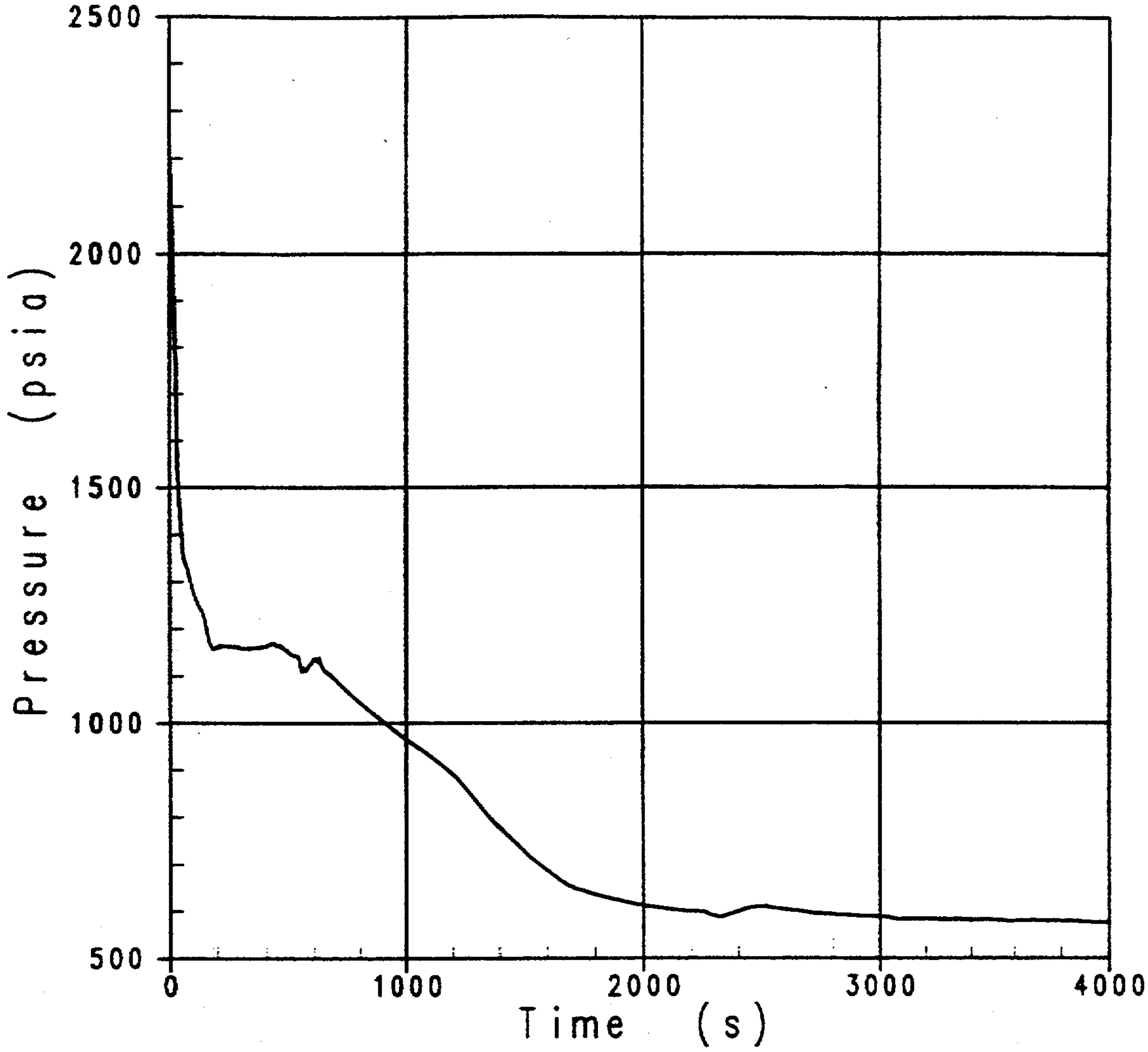
Remove SBLOCA Figures 15.3-1 through 15.3-14f
Replace with following Figures 15.3-1 through 15.3-15-DCPP2



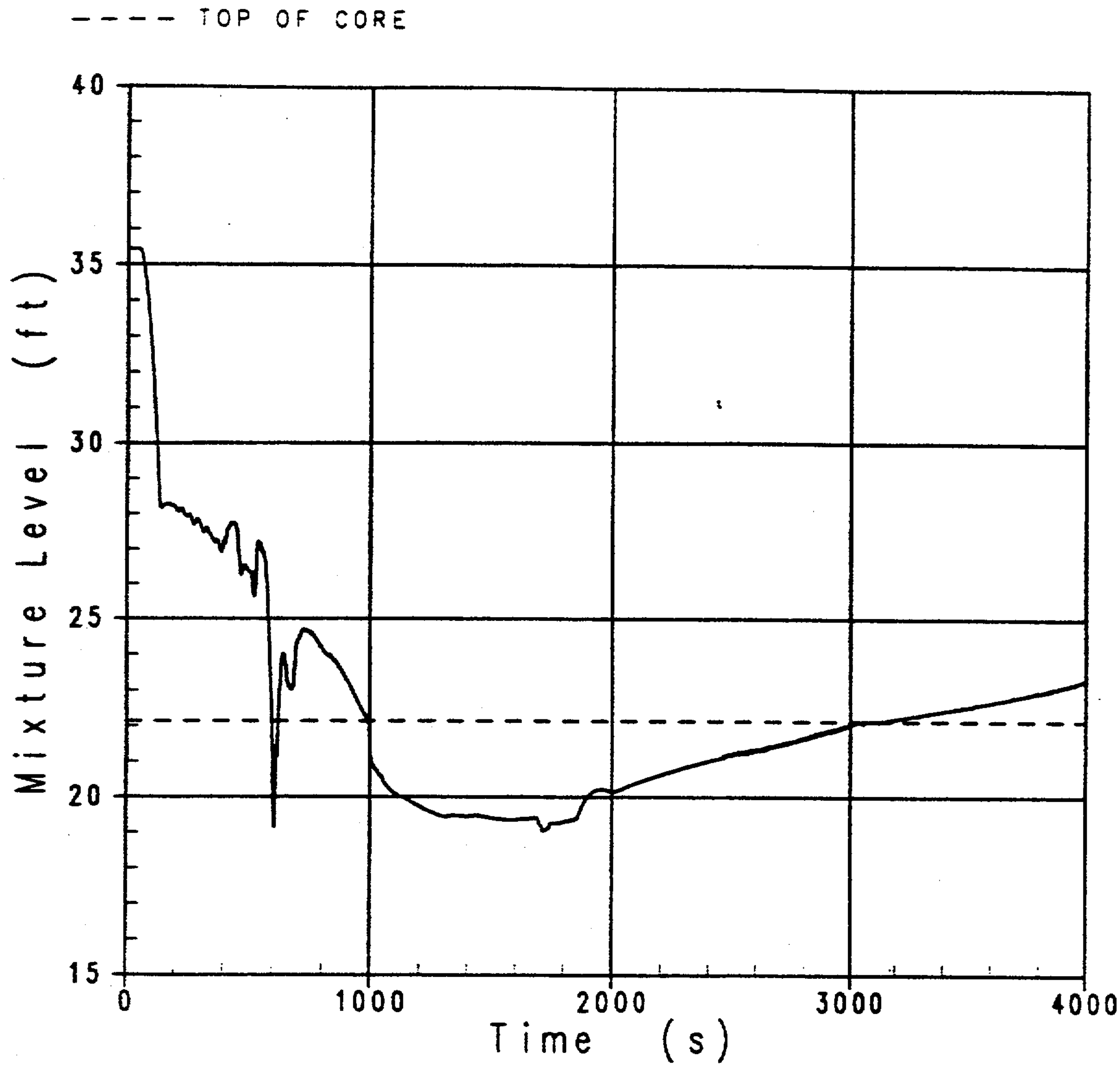
Safety Injection Flowrate for Small Break LOCA	DIABLO CANYON UNITS 1 and 2
	Figure 15.3-1



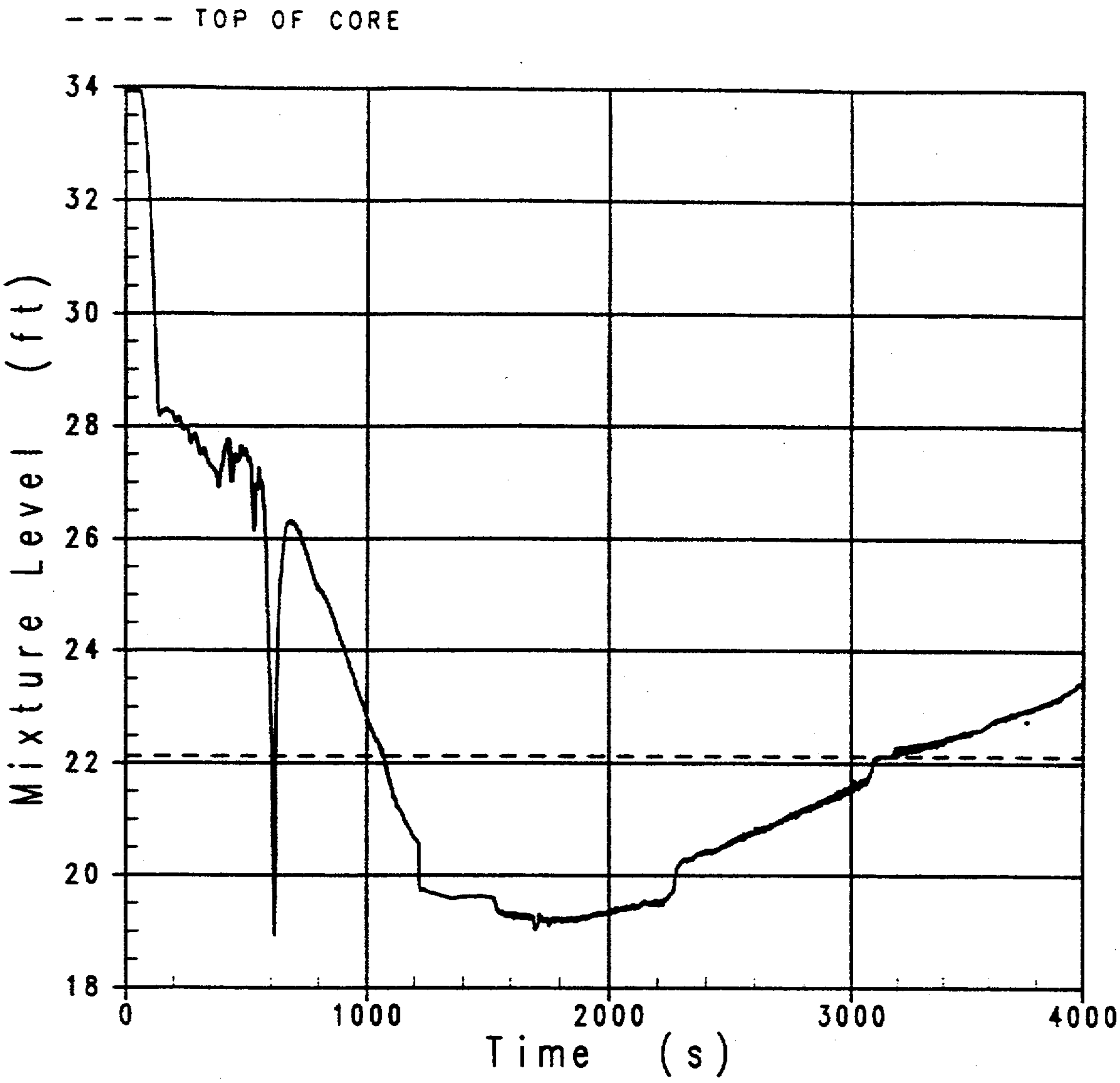
RCS Depressurization 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-2-DCPP1



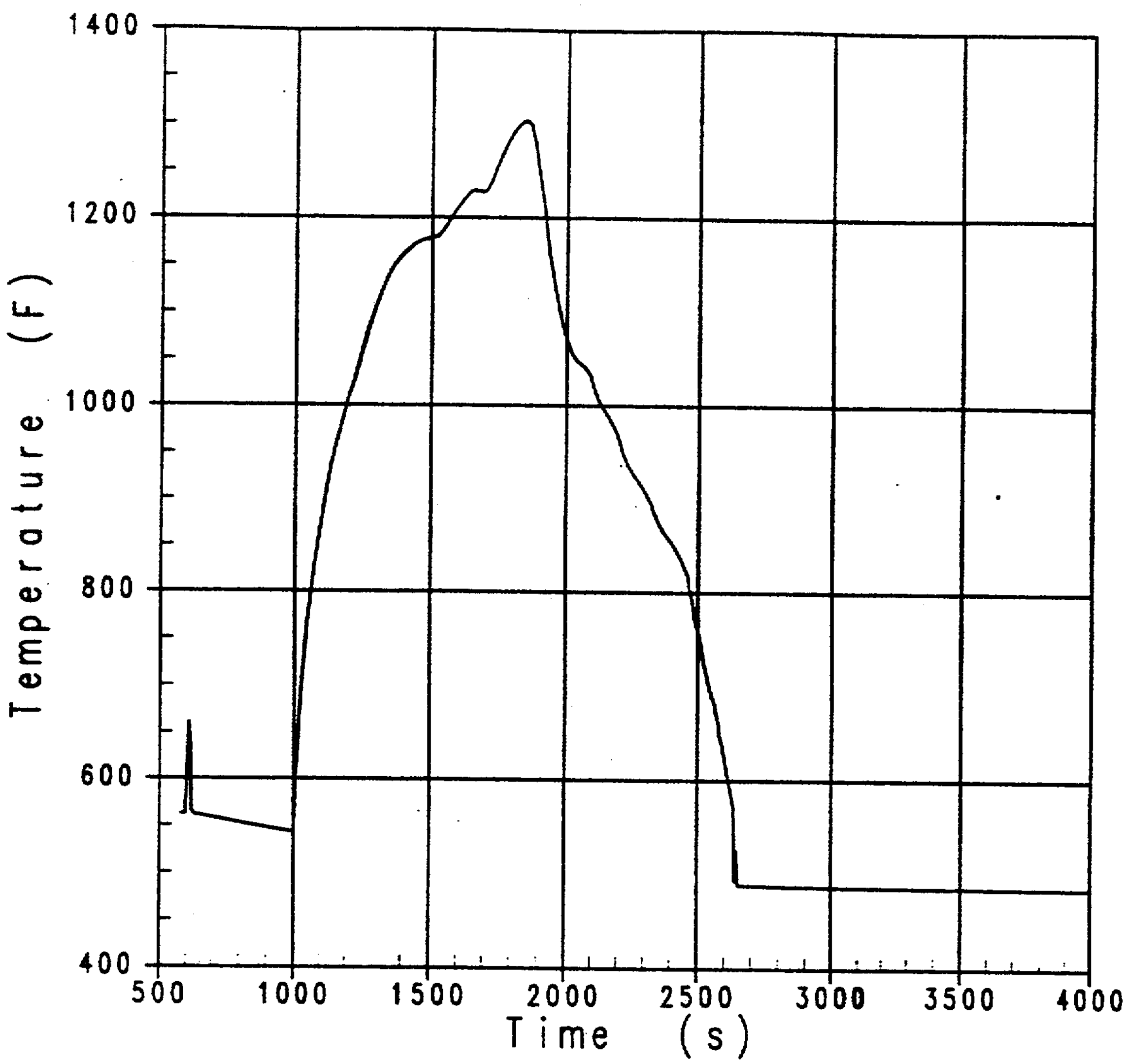
RCS Depressurization 3-inch Cold Leg Break	DIABLO CANYON UNIT 2 Figure 15.3-2-DCPP2



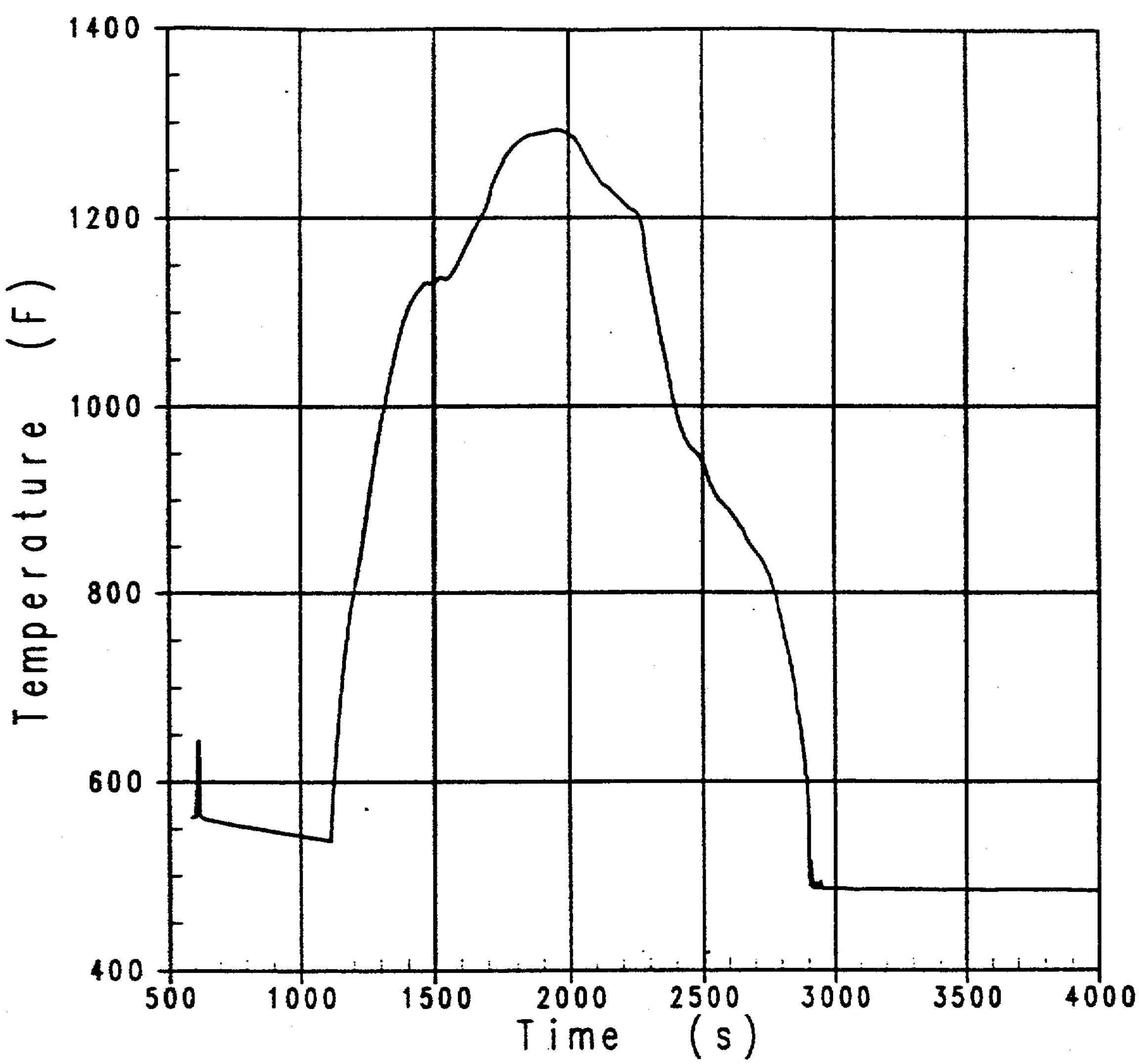
Core Mixture Elevation 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-3-DCPP1



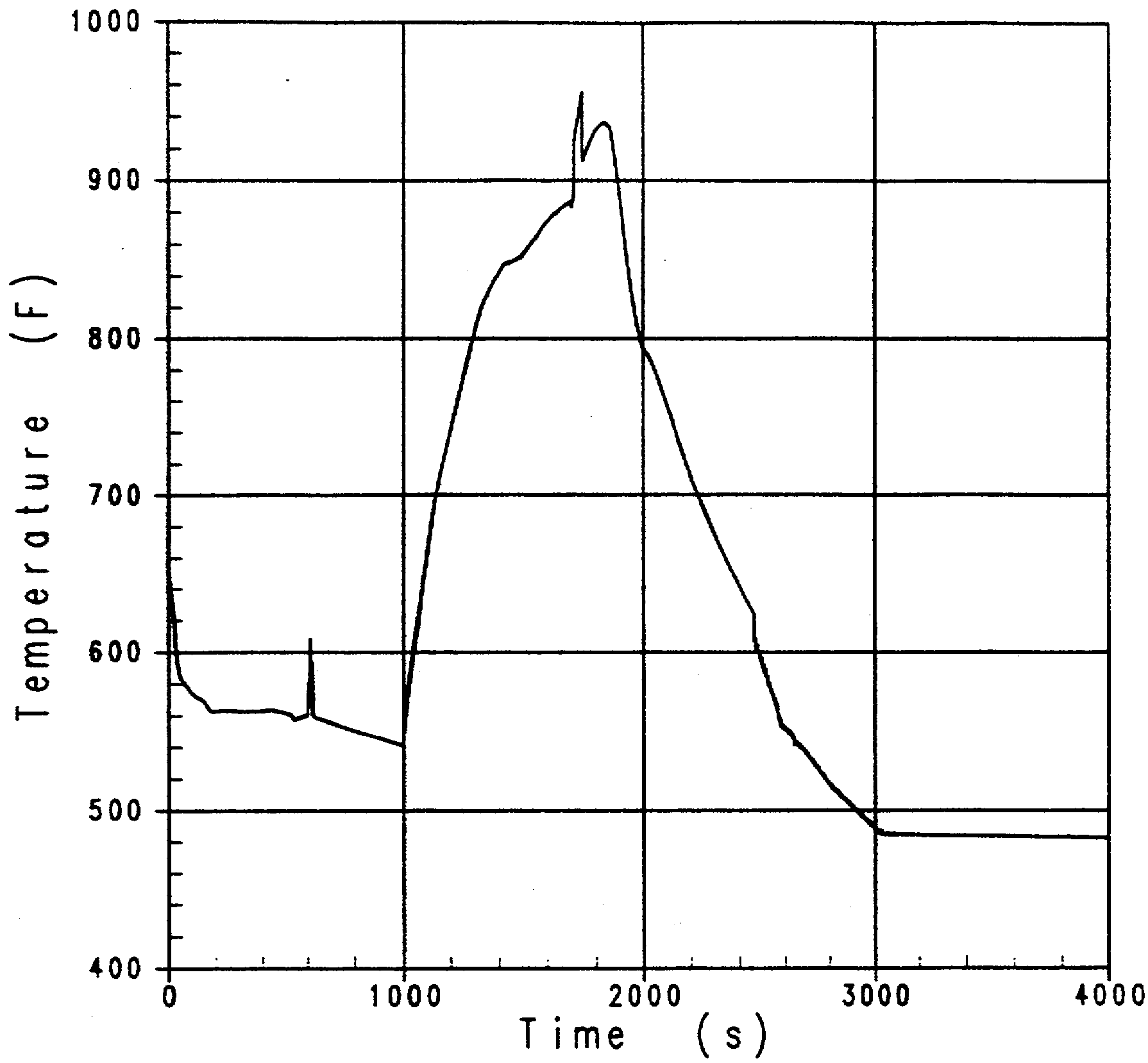
Core Mixture Elevation **3-inch Cold Leg Break**	**DIABLO CANYON UNIT 2**
	Figure 15.3-3-DCPP2



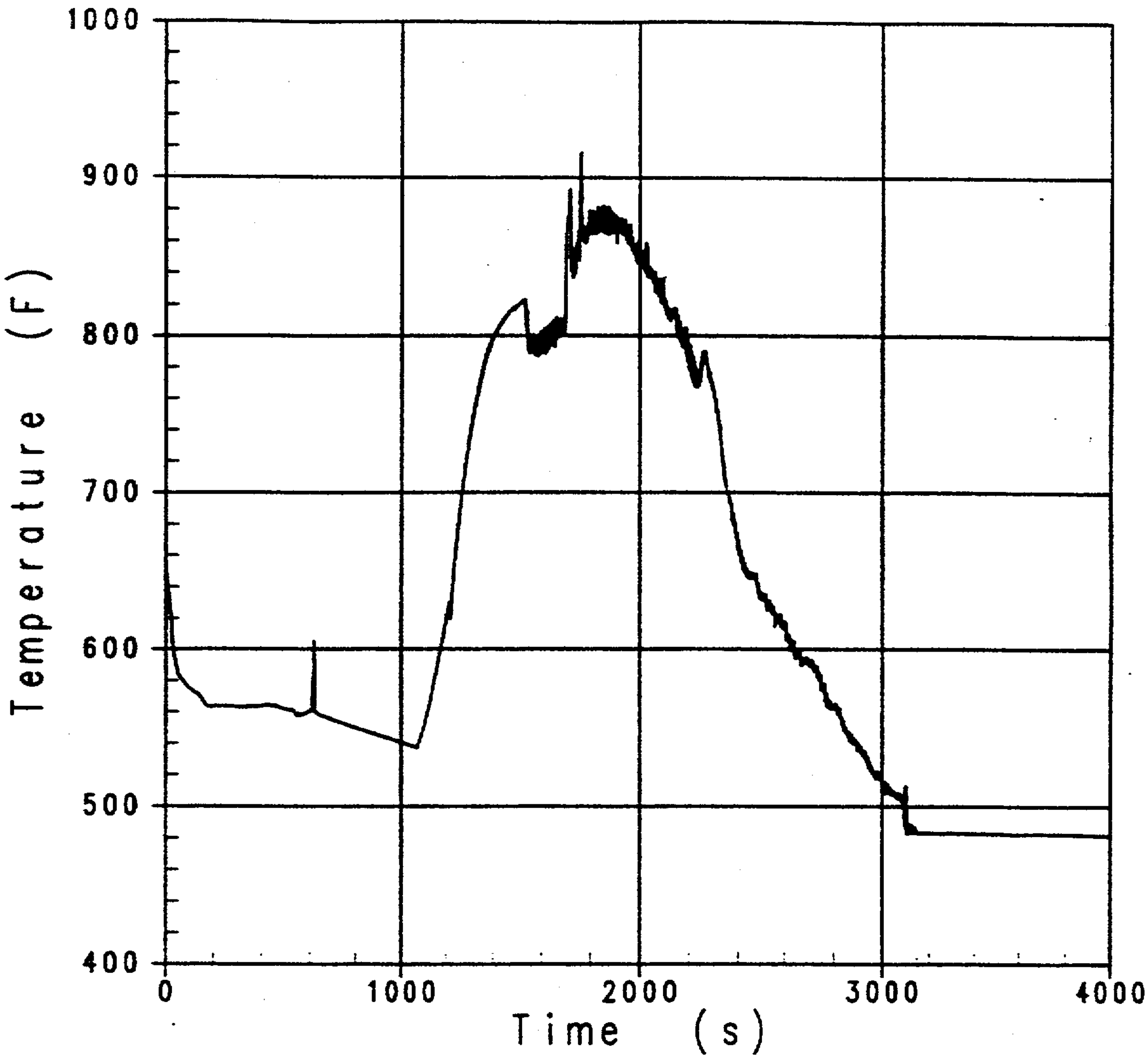
Clad Temperature Transient 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-4-DCPP1



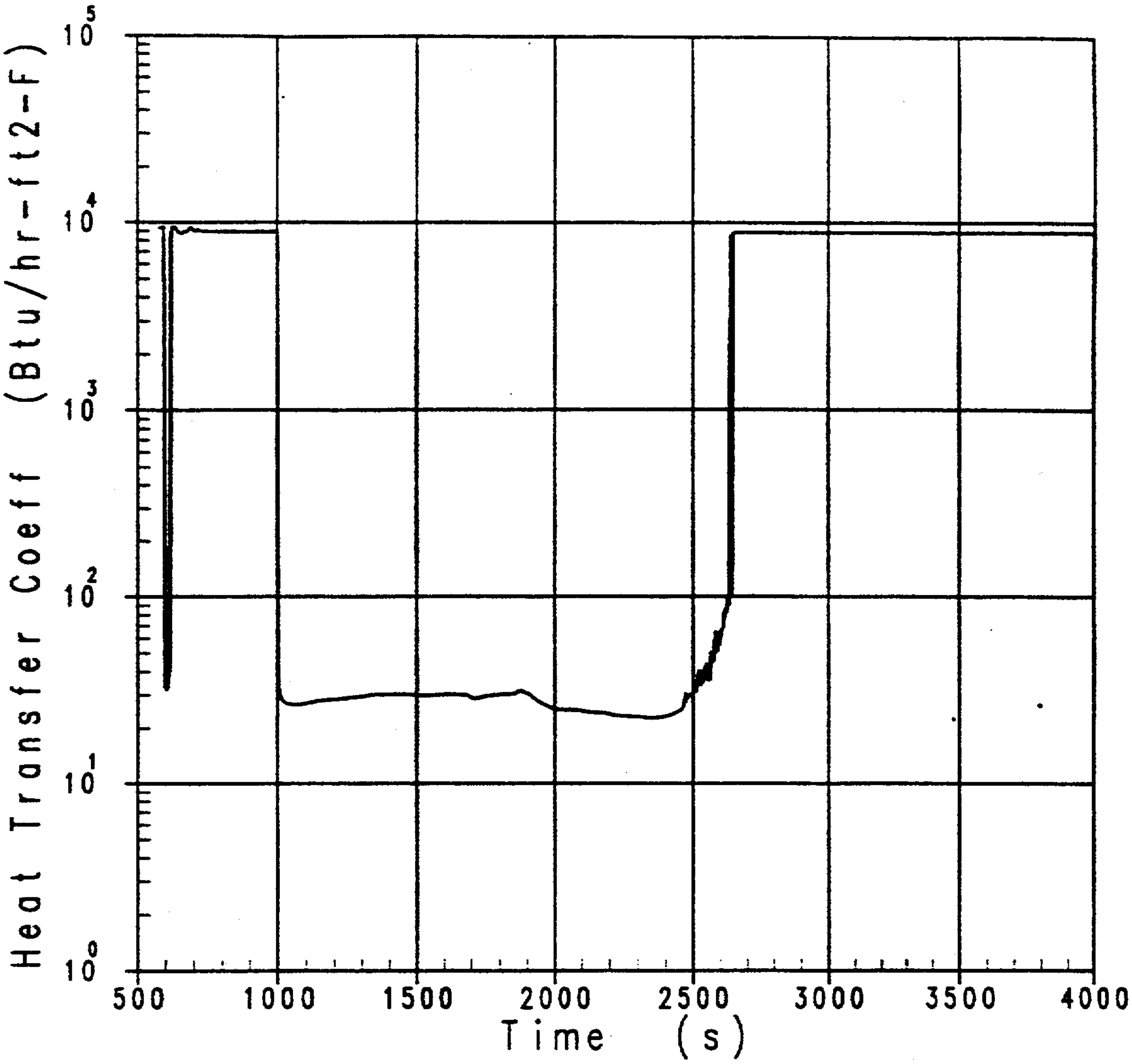
Clad Temperature Transient 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-4-DCPP2



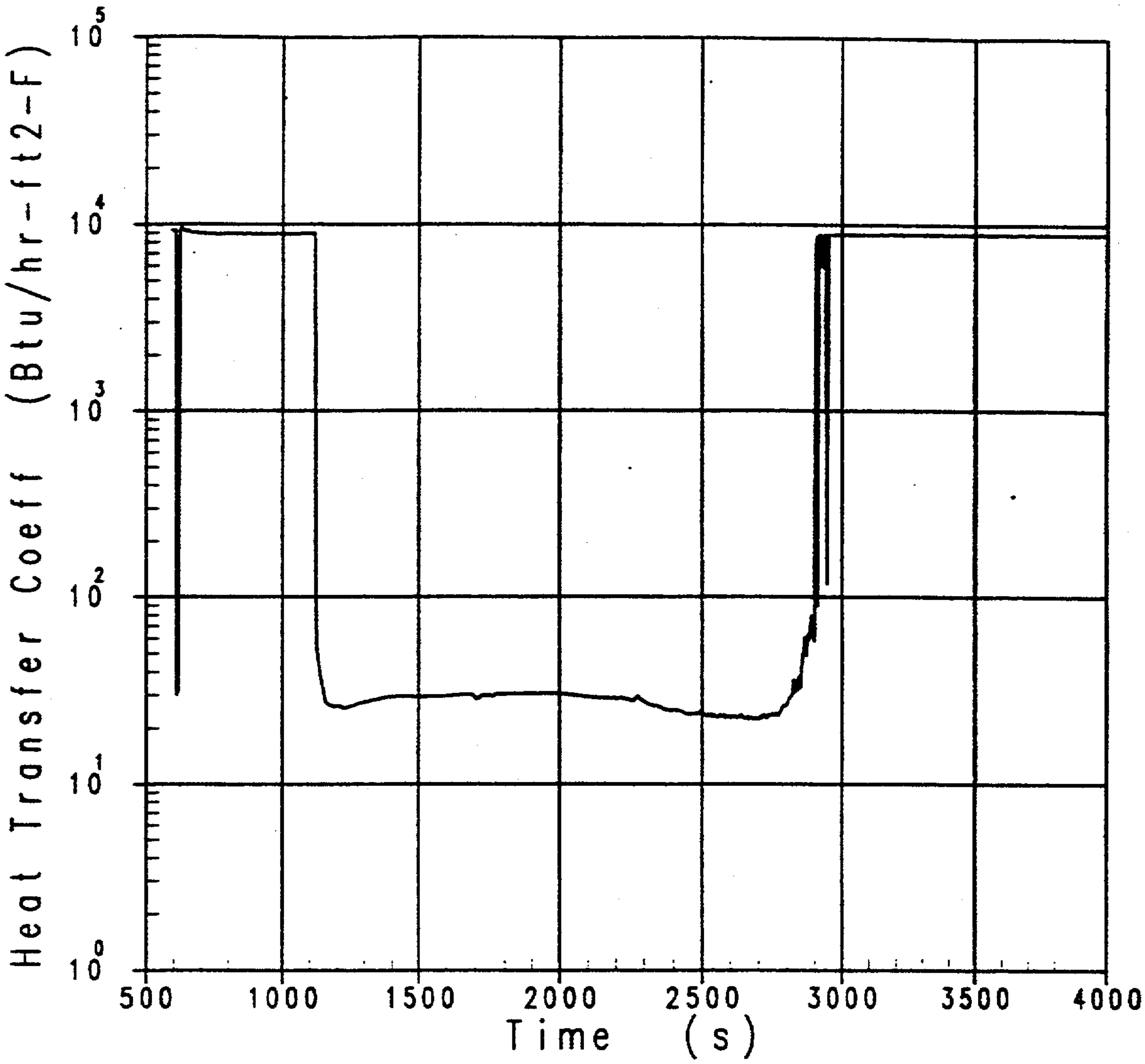
Top Core Node Vapor Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-5-DCPP1



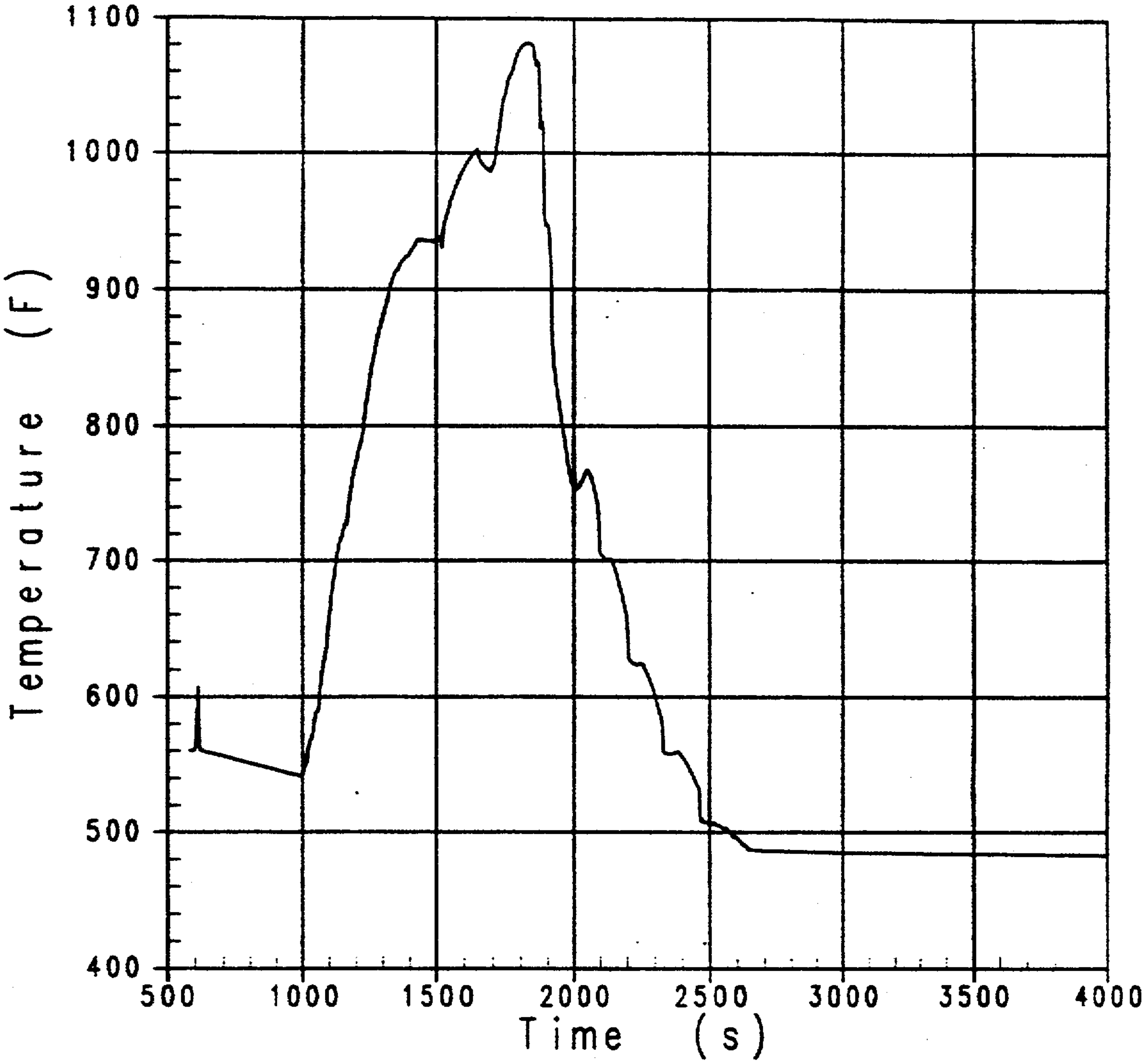
Top Core Node Vapor Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-5-DCPP2



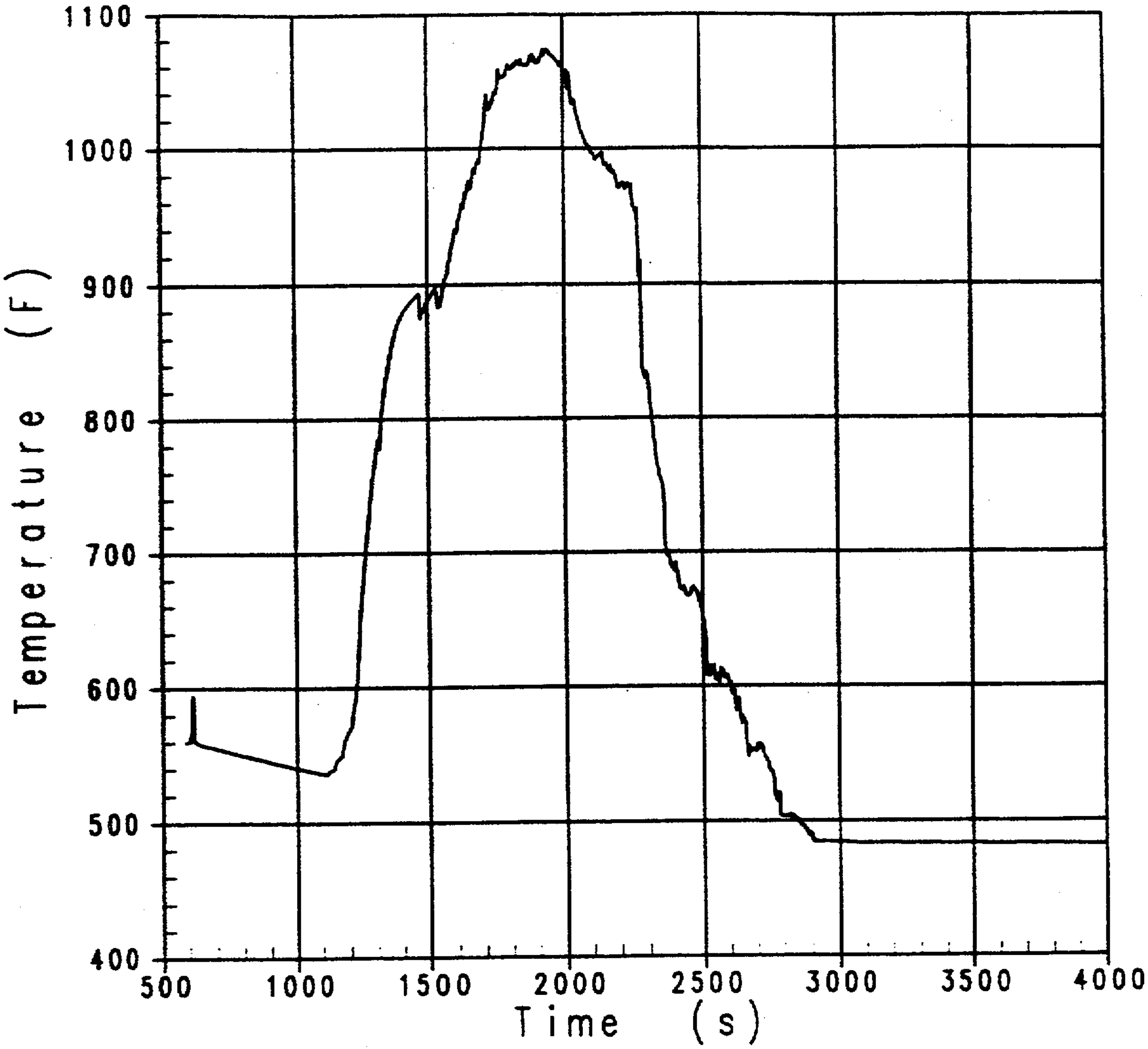
Rod Film Coefficient 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-6-DCPP1



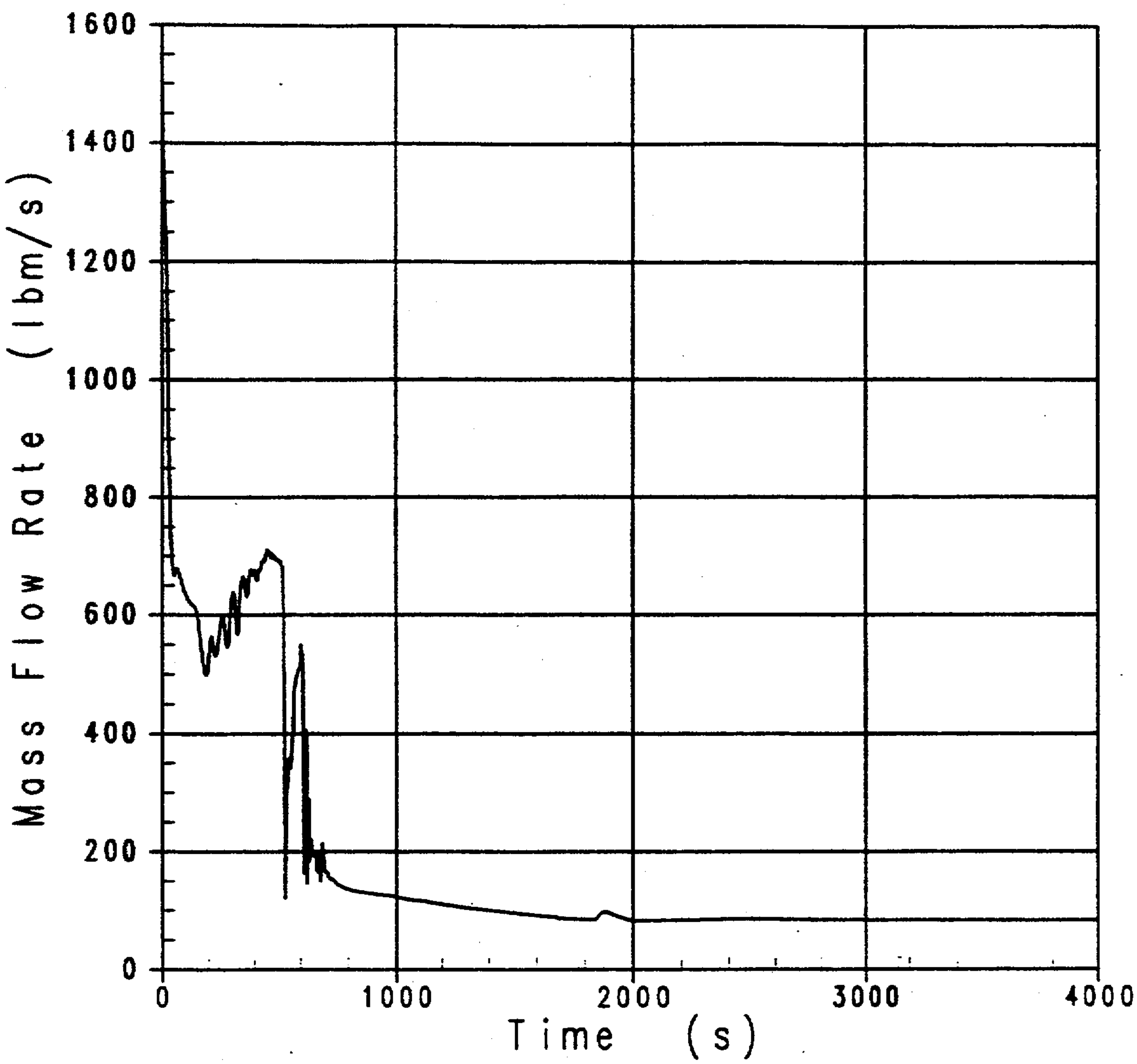
Rod Film Coefficient 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-6-DCPP2



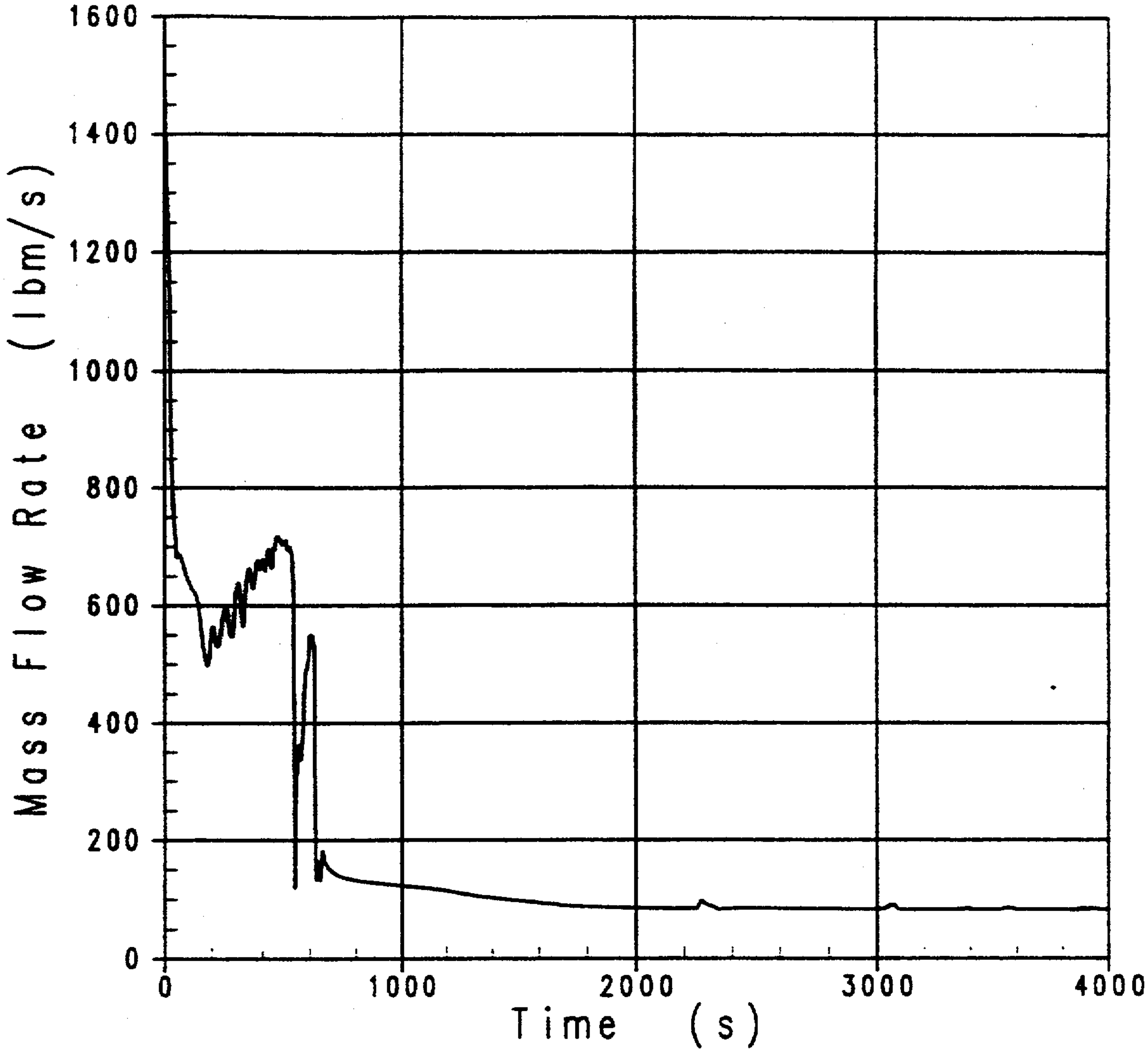
Hot Spot Fluid Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-7-DCPP1



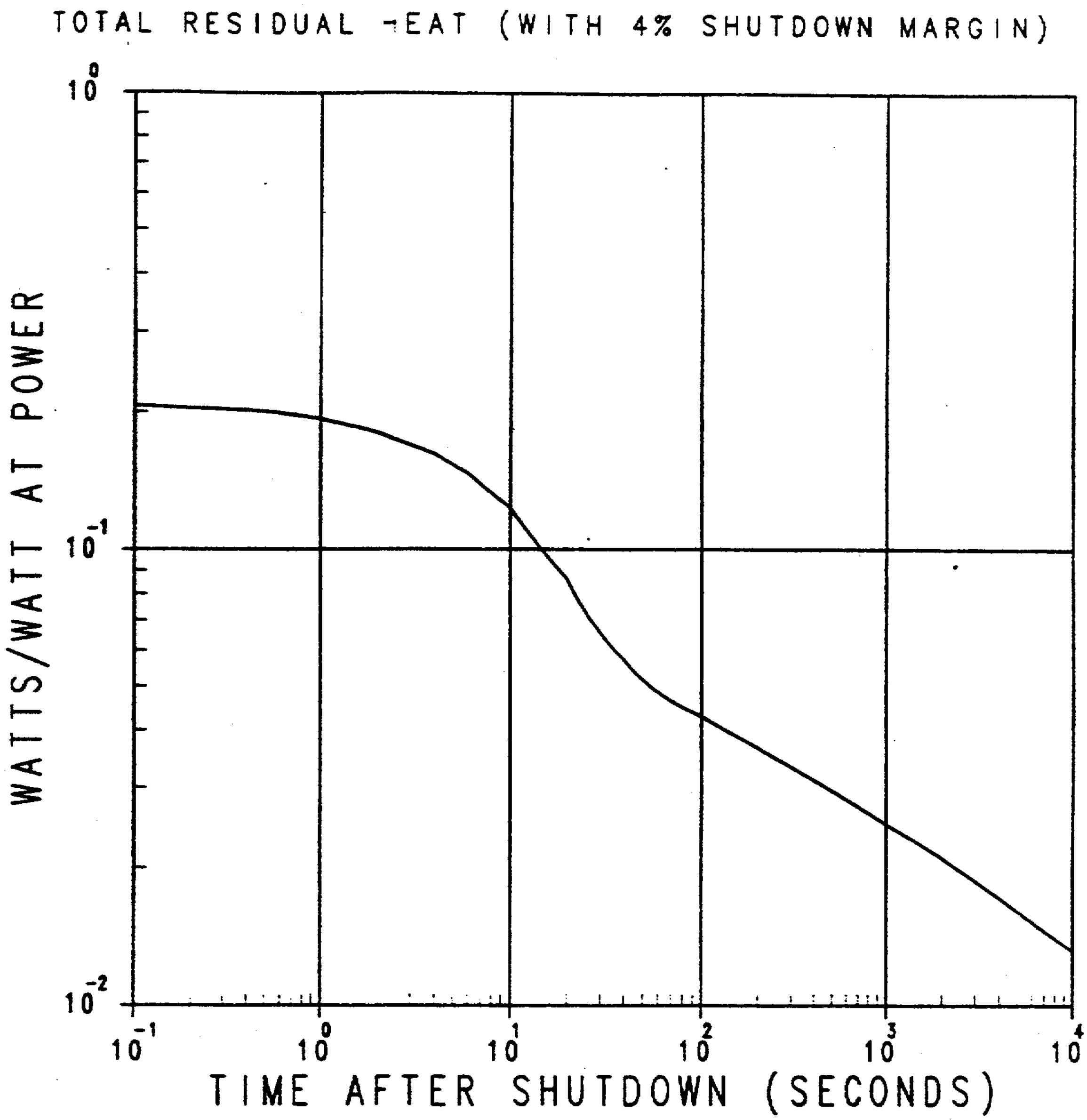
Hot Spot Fluid Temperature 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-7-DCPP2



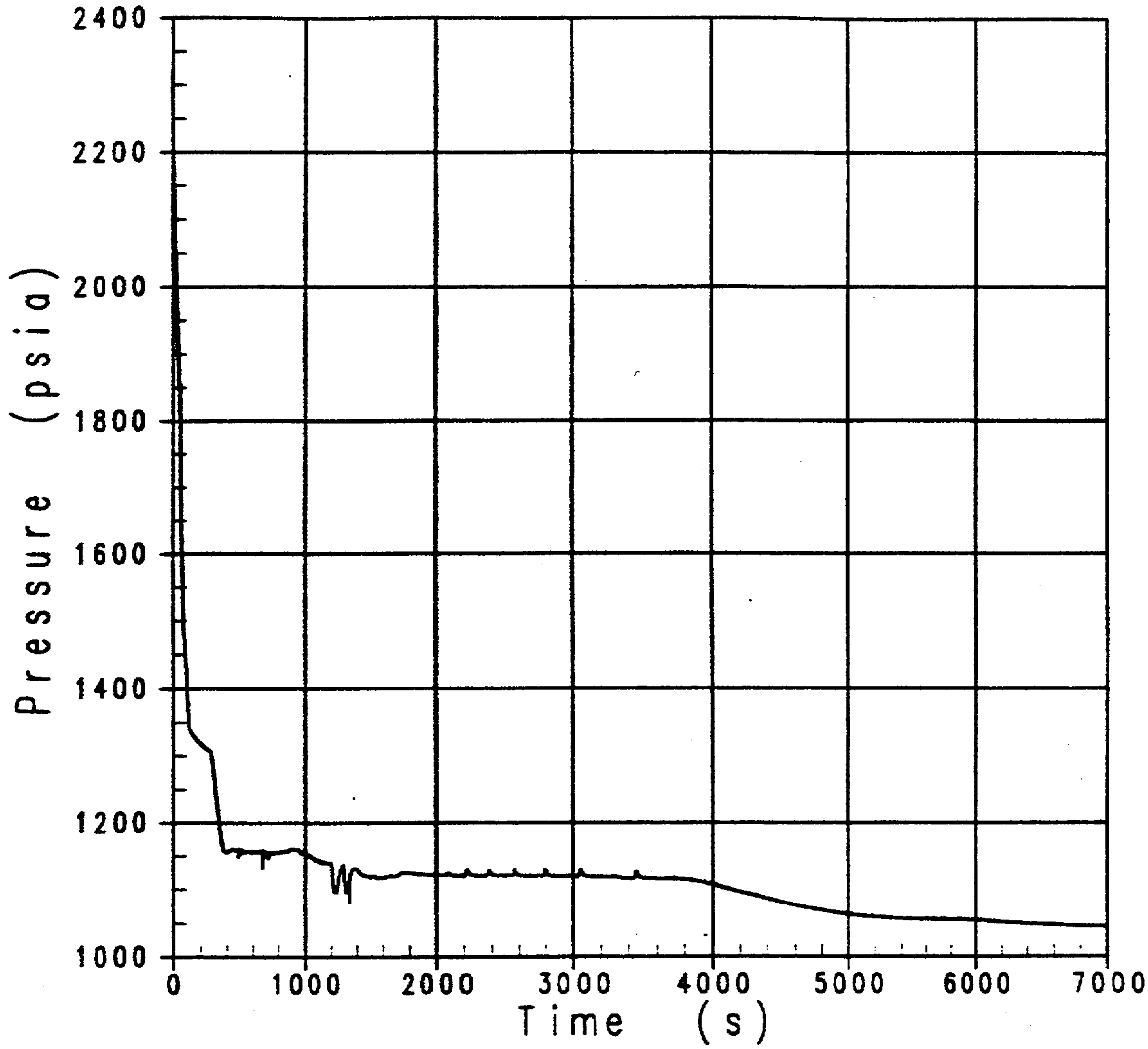
Break Mass Flow **3-inch Cold Leg Break**	**DIABLO CANYON UNIT 1**
	Figure 15.3-8-DCPP1



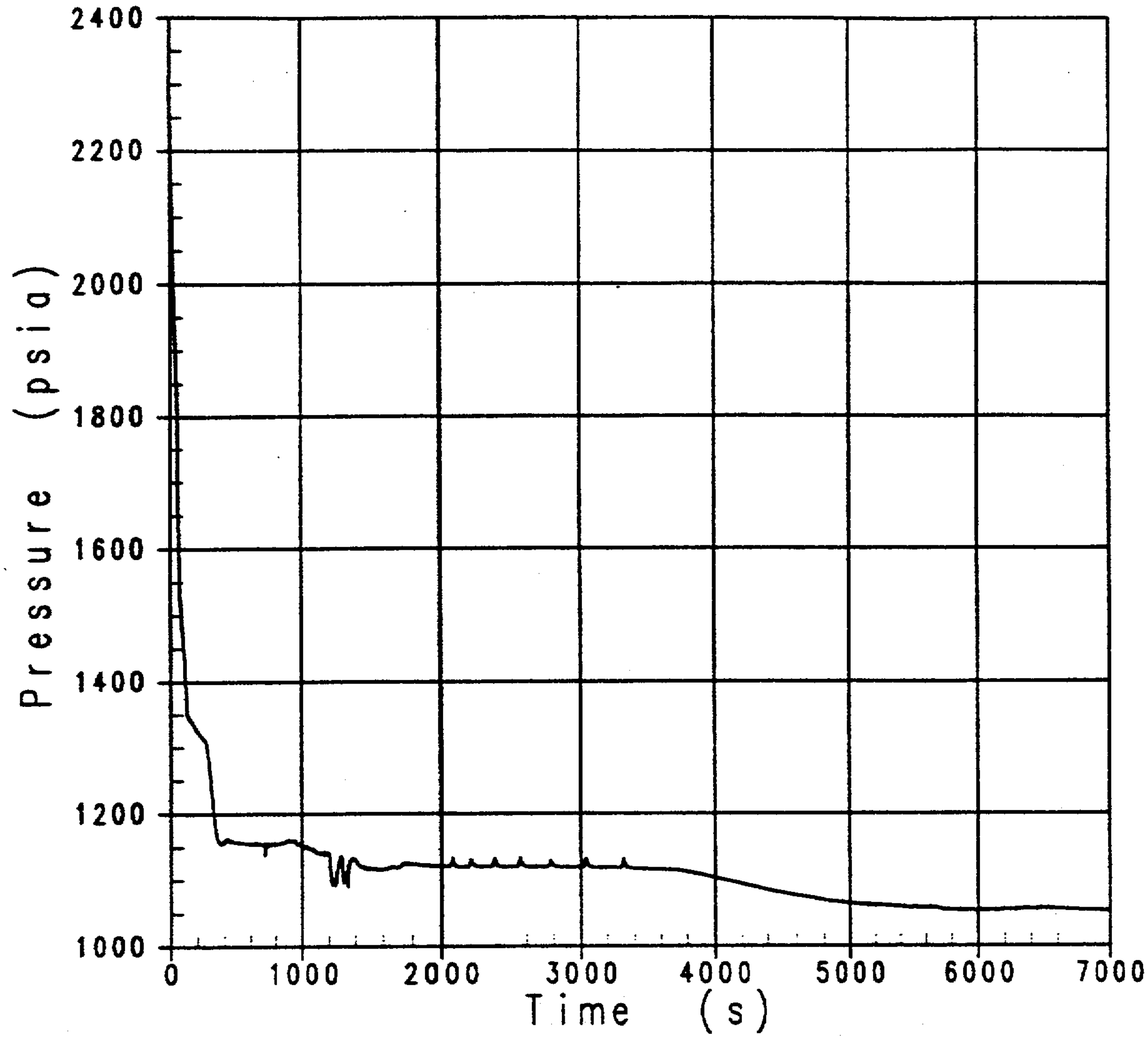
Break Mass Flow 3-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-8-DCPP2



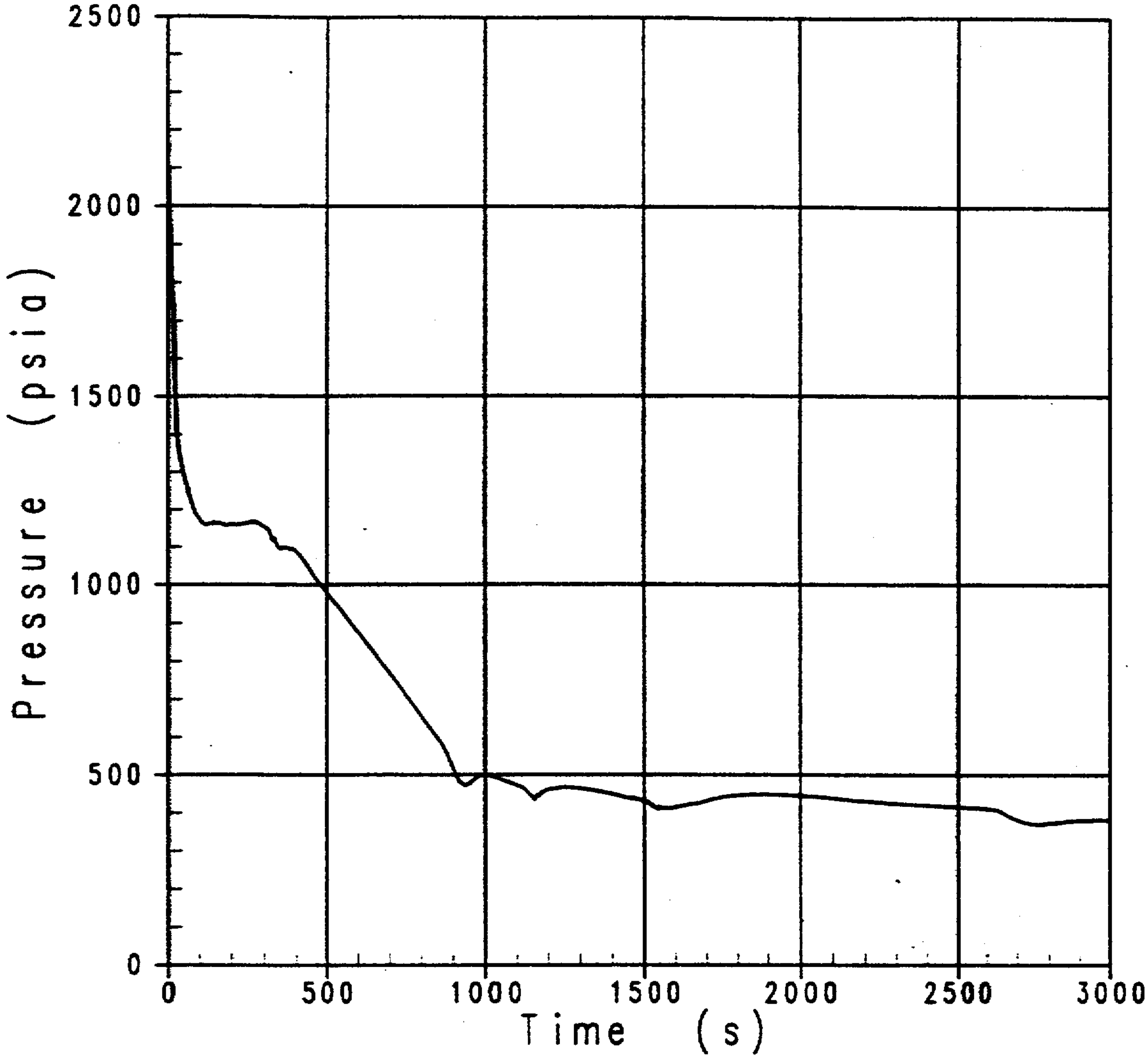
LOCA Core Power Transient	DIABLO CANYON UNITs 1&2
	Figure 15.3-9



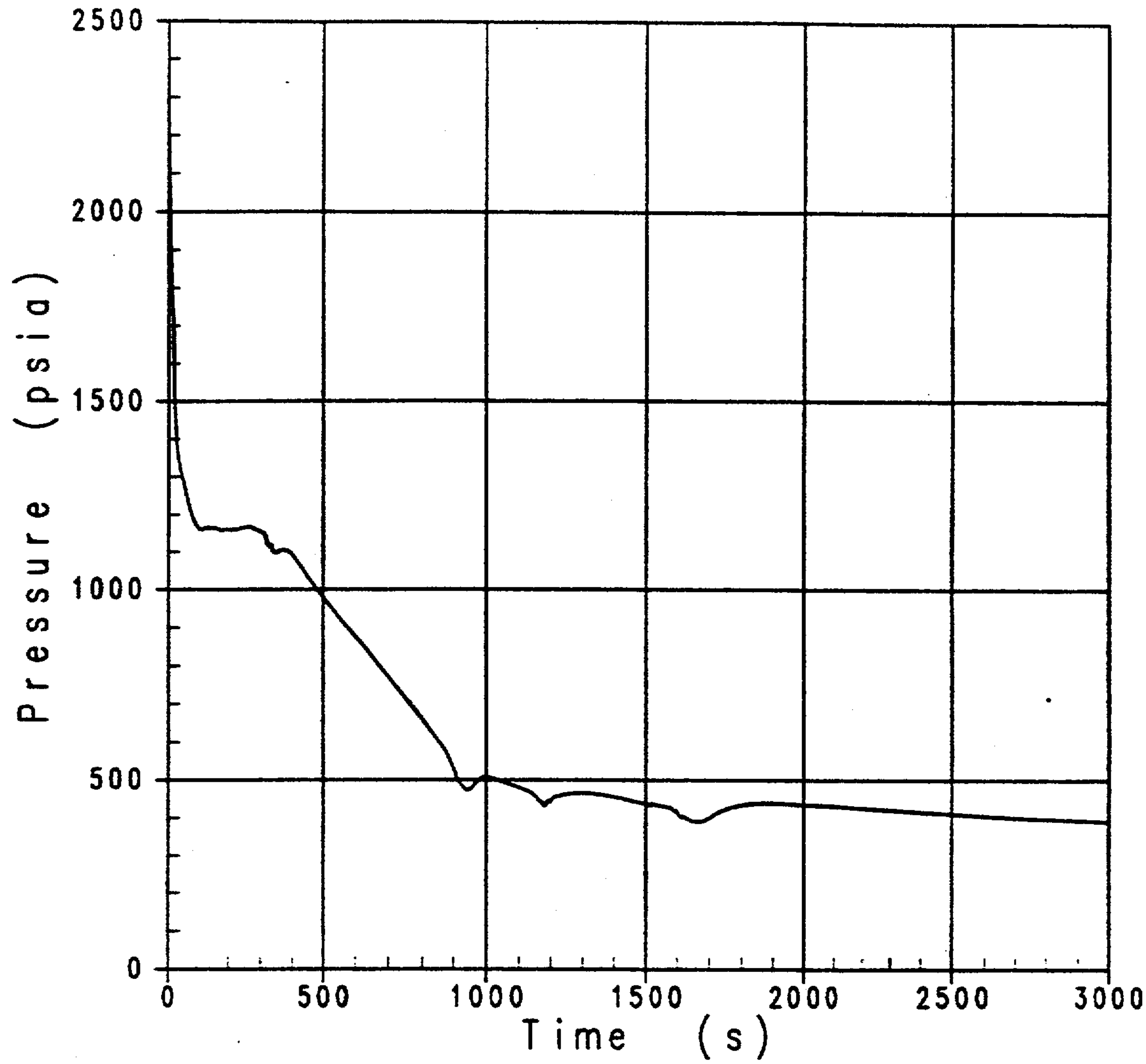
RCS Depressurization **2-inch Cold Leg Break**	**DIABLO CANYON UNIT 1** **Figure 15.3-10-DCPP1**



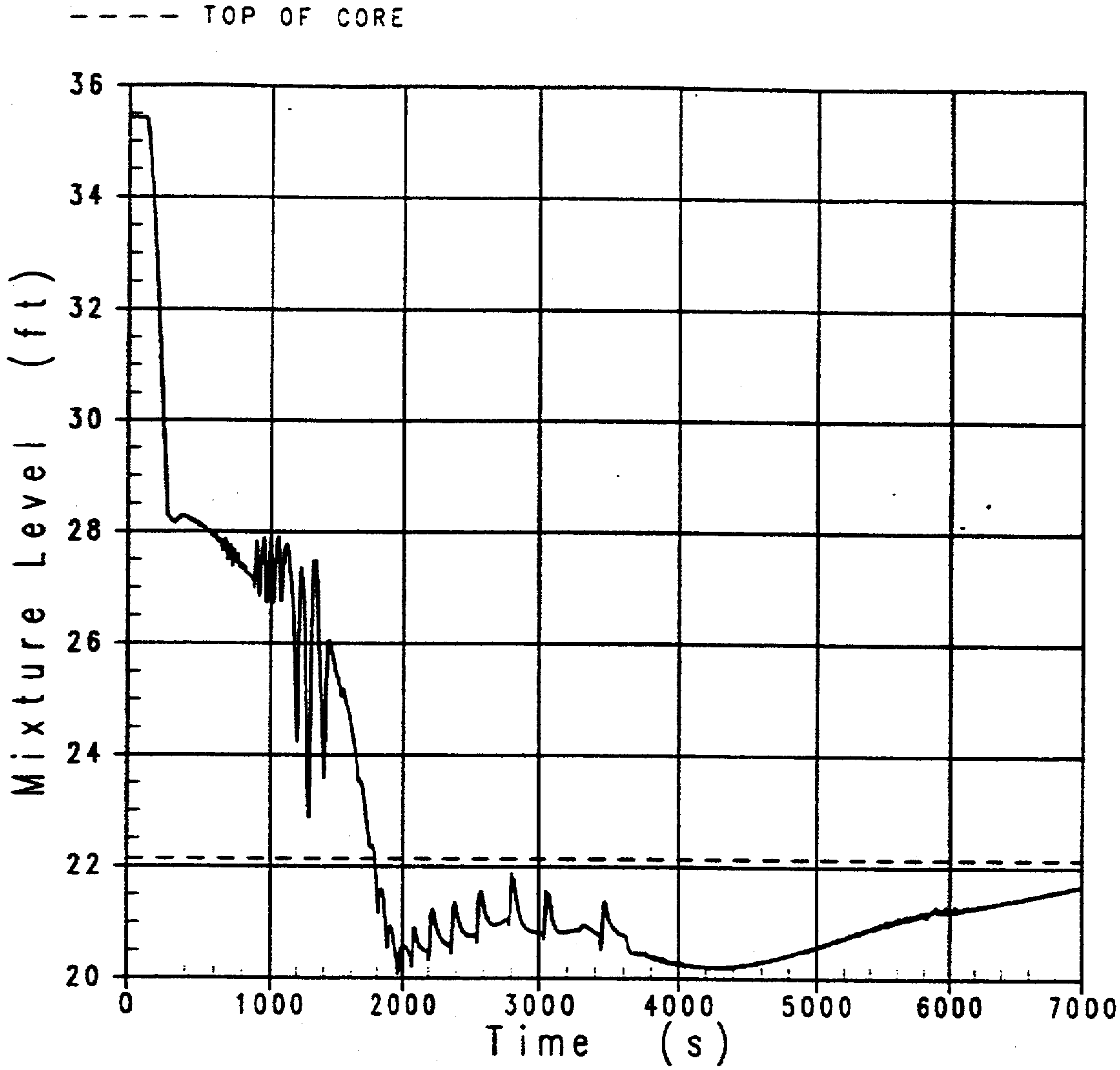
RCS Depressurization 2-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-10-DCPP2



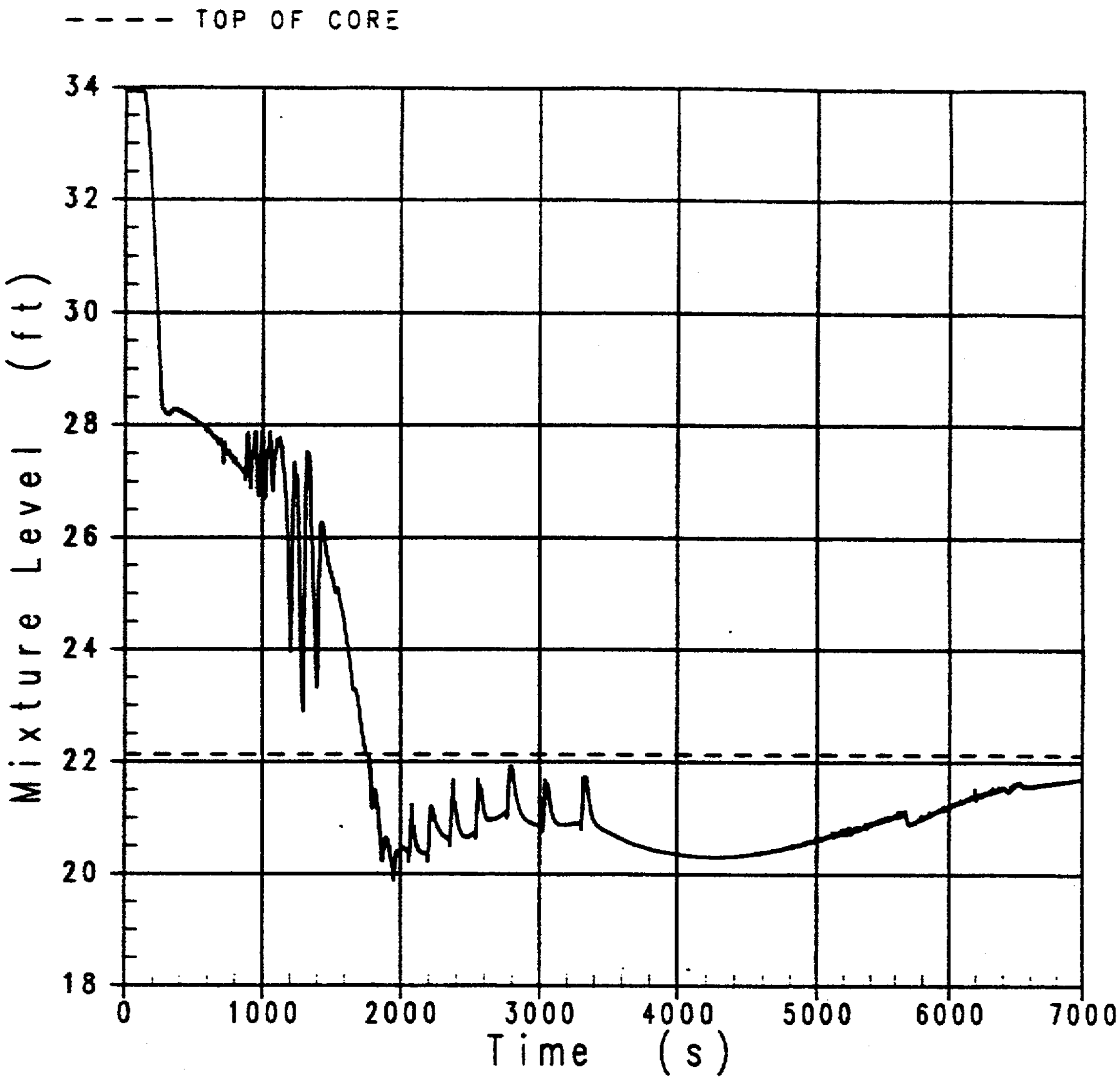
RCS Depressurization 4-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-11-DCPP1



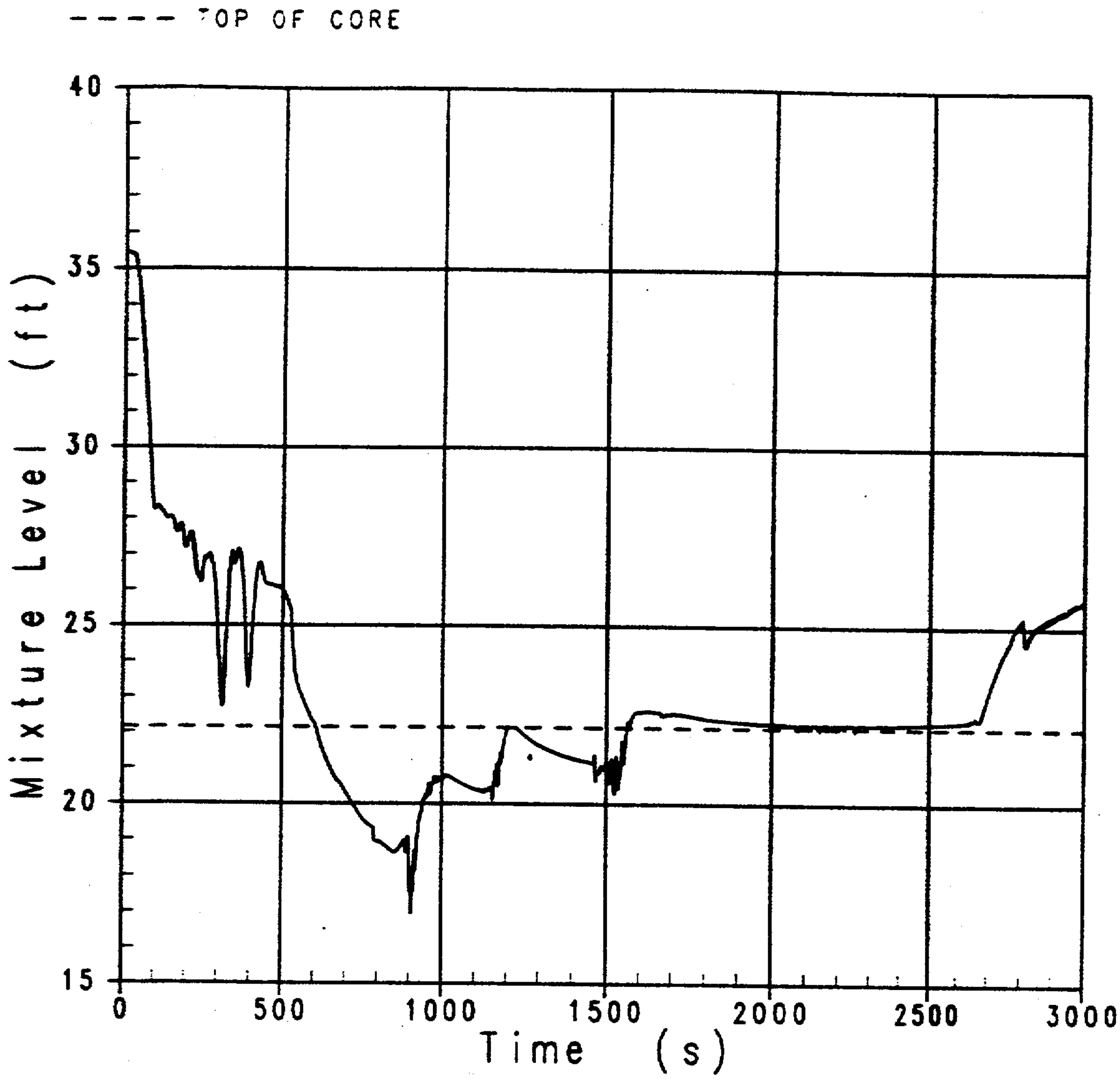
RCS Depressurization 4-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-11-DCPP2



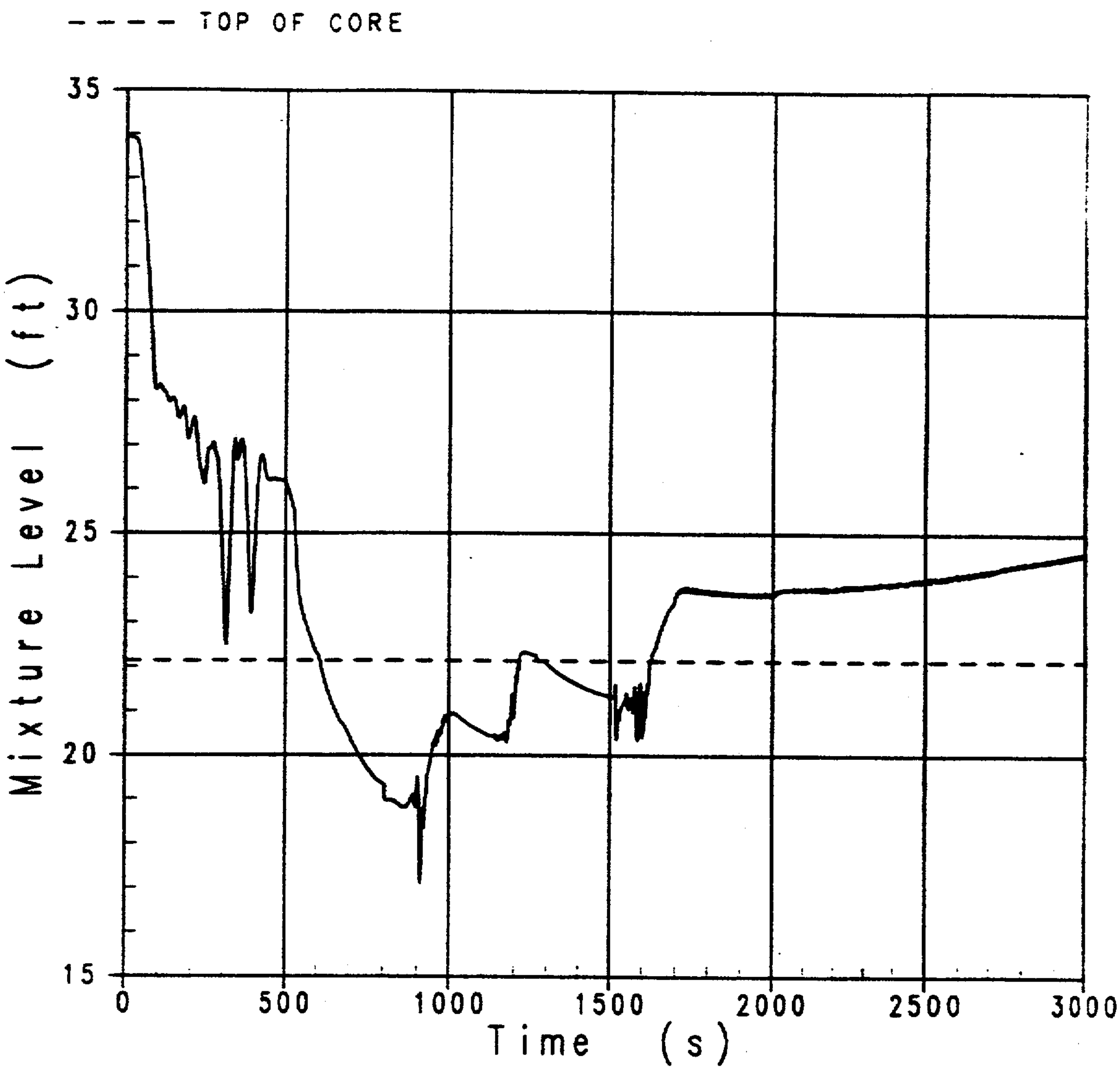
Core Mixture Elevation 2-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-12-DCPP1



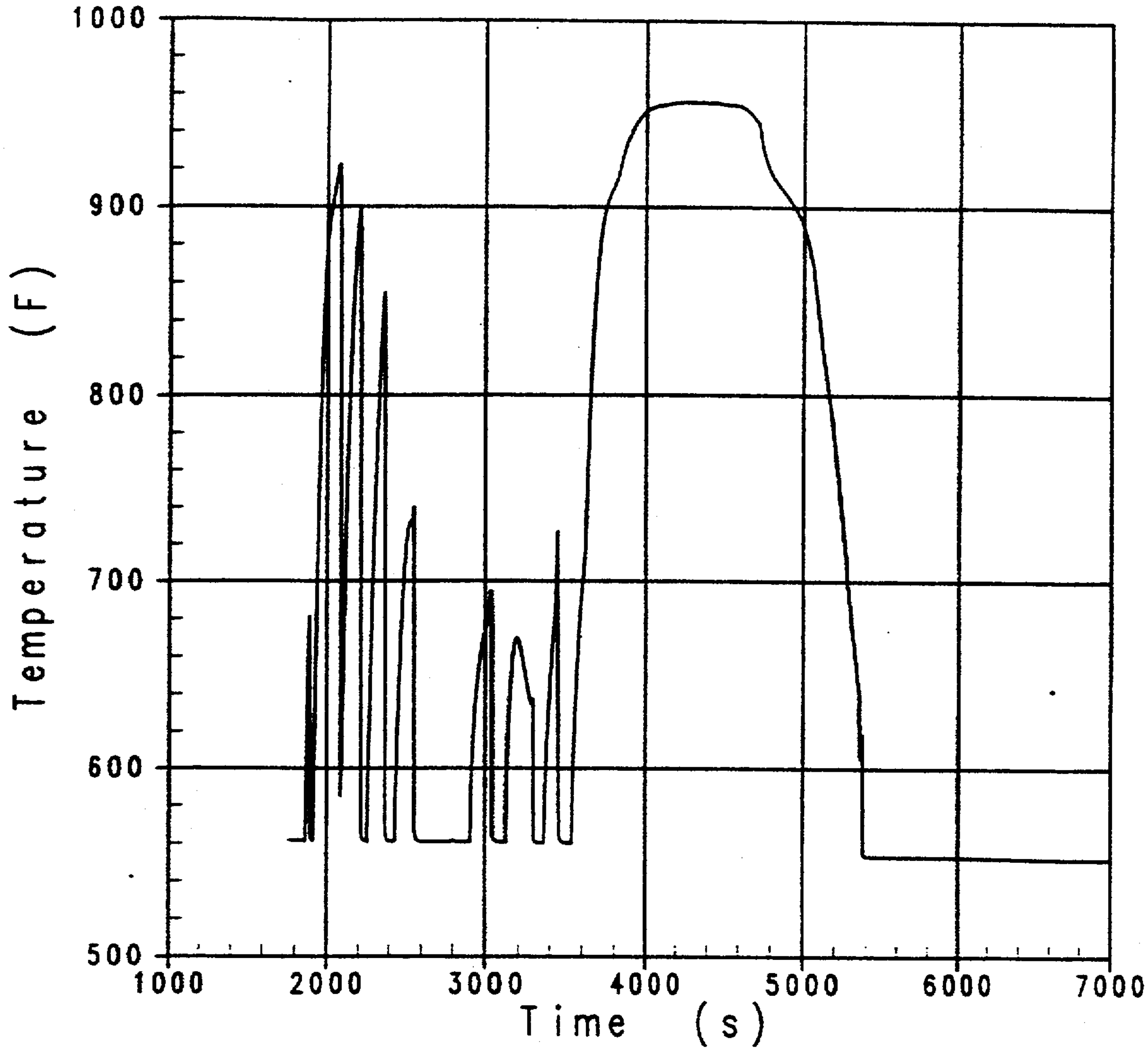
Core Mixture Elevation 2-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-12-DCPP2



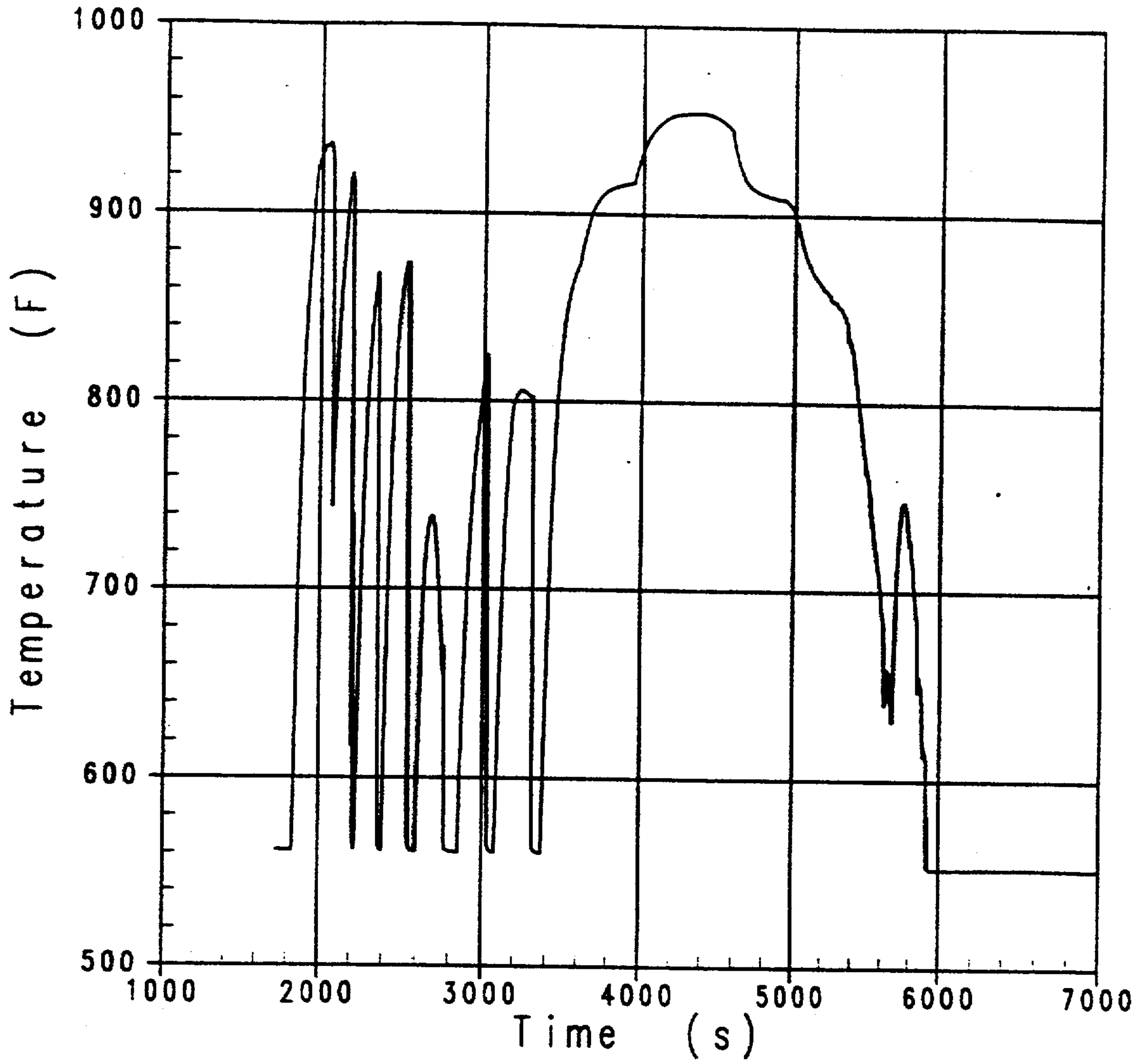
Core Mixture Elevation 4-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-13-DCPP1



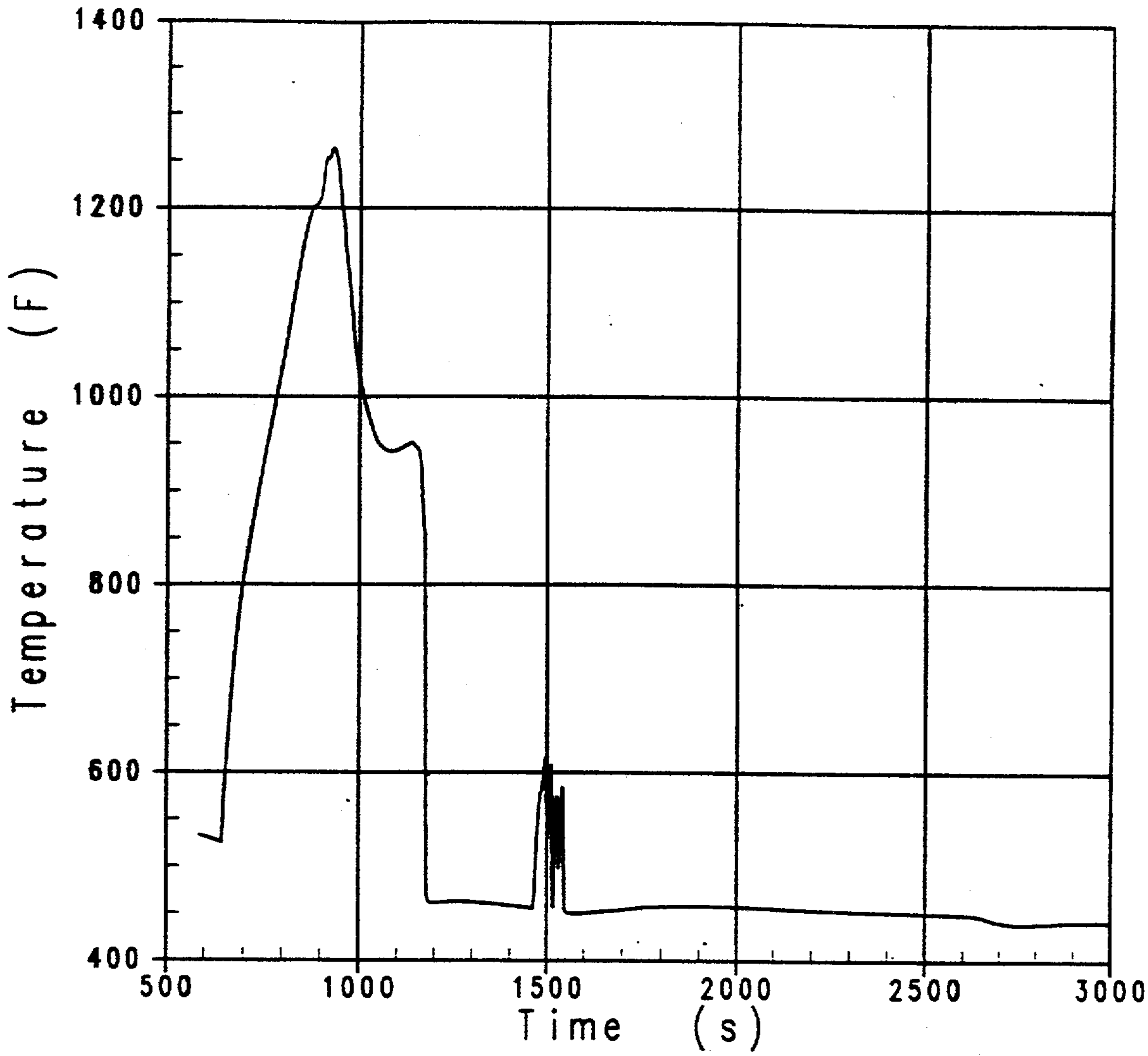
Core Mixture Elevation 4-inch Cold Leg Break	DIABLO CANYON UNIT 2
	Figure 15.3-13-DCPP2



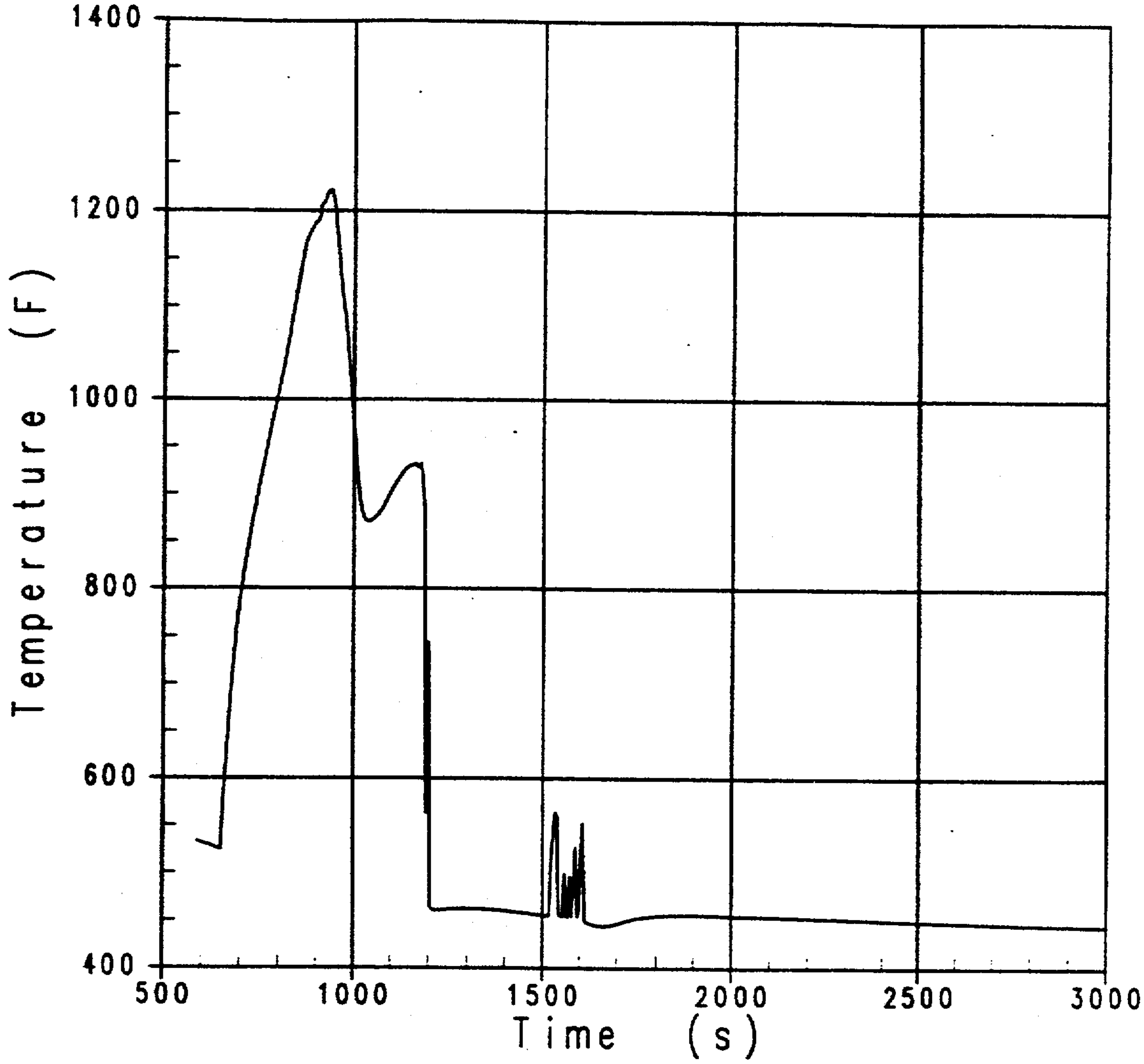
Cladding Temperature Transient 2-inch Cold Leg Break	DIABLO CANYON UNIT 1
	Figure 15.3-14-DCPP1



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

MARK-UP OF PRECAUTIONS, LIMITATIONS AND SETPOINT PAGES

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

ΔT reactor trip setpoint:

4 37 $\Delta T \left(\frac{1+T_4S}{1+T_5S}\right) = \Delta To\ [K4-K5\left(\frac{T_3S}{1+T_3S}\right)T-K6\ (T-T')-f_2(\Delta q)]$

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

4 37

Parameter

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)

4 37
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. Control Systems

1. Reactor Control

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 4 37
2.	Low limit (TC-505, TC-505A)	547°F (4)	547°F
3.	Full power temperature 4 37	**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) 0.298°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)