Attachment 3

MARKED-UP TECHNICAL SPECIFICATION PAGES

REMOVE	INSERT
iii	iii
v	v
xxii	xxii
2-1	2-1
2-2	2-2
2-2a	2-2a
2-4	2-4
2-7	2-7
2-8	2-8
2-9	2-9
B 2-1	B 2-1
B 2-4	B 2-4
3/4 1-20	3/4 1-20
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-5	3/4 2-5
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
	3/4 2-8a
	3/4 2-8b
	3/4 2-8c
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 2-11	3/4 2-11
	3/4 2-11a
3/4 2-12	3/4 2-12
3/4 2-17	3/4 2-17
3/4 3-23	3/4 3-23
3/4 3-25	3/4 3-25
3/4 3-27	3/4 3-27
3/4 10-2	3/4 10-2
B 3/4 2-4	B 3/4 2-4
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
6-20	6-20

2422S/0065K

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure 2.1-1a for Units 1 and 2, Cycle 4 and after and Figure 2.1-1b for Unit 2, Cycle 3

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.

REPLACE

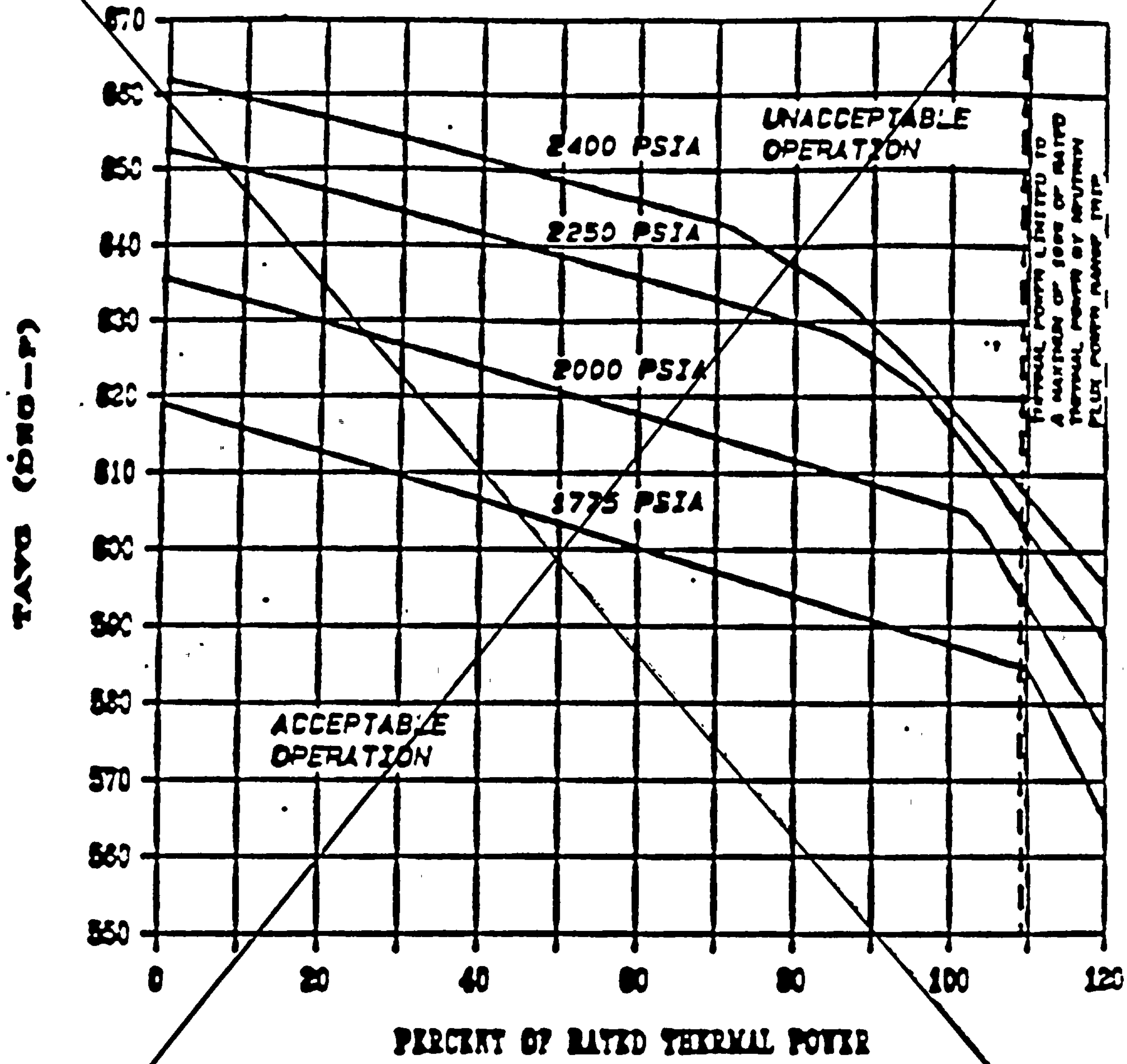


FIGURE 2.1-1a
REACTOR CORE SAFETY LIMIT (UNIT 1)

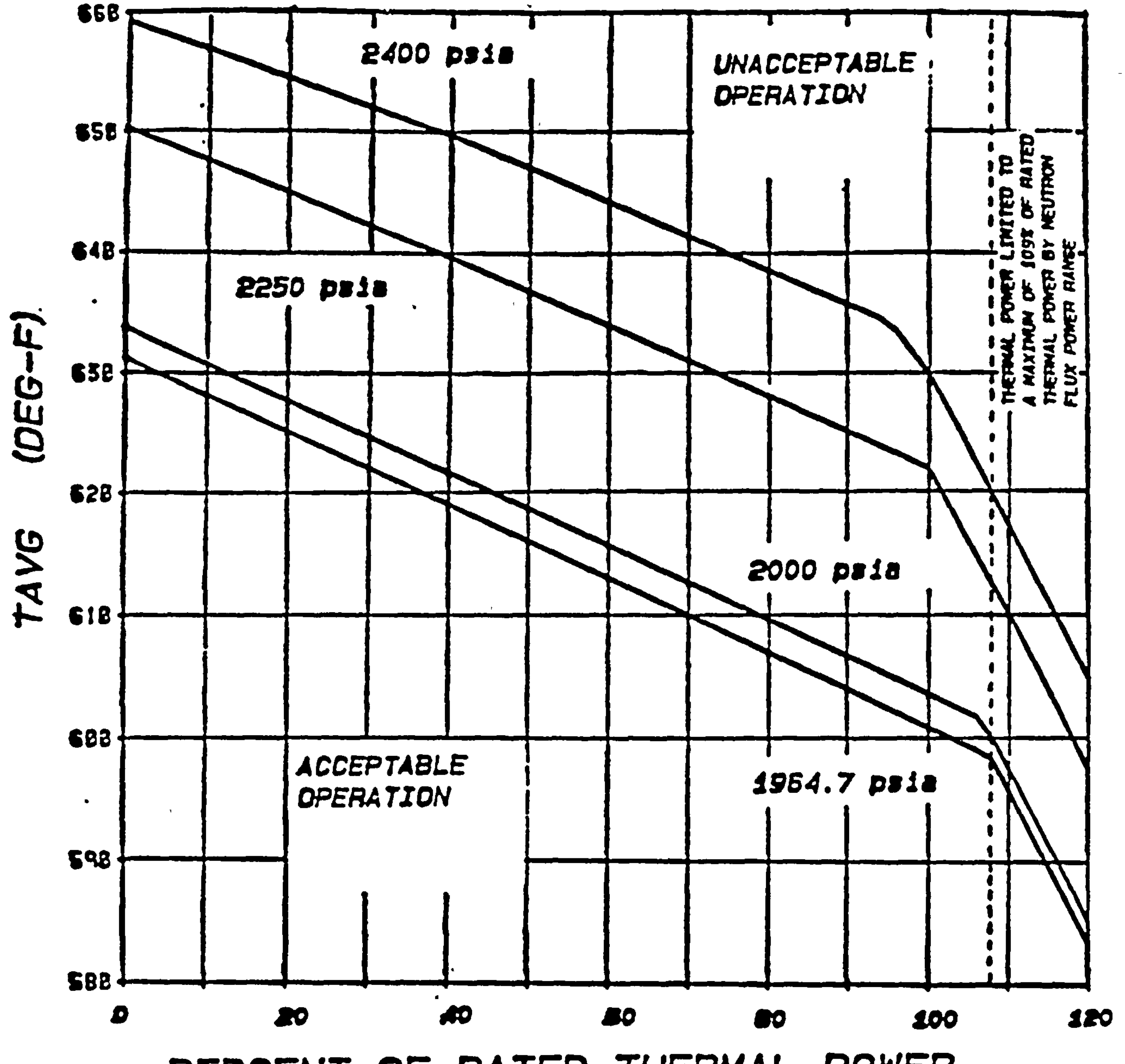


FIGURE 2.1-1a
REACTOR CORE SAFETY LIMIT

DIABLO CANYON - UNITS 1 & 2 Cycle 4 and after

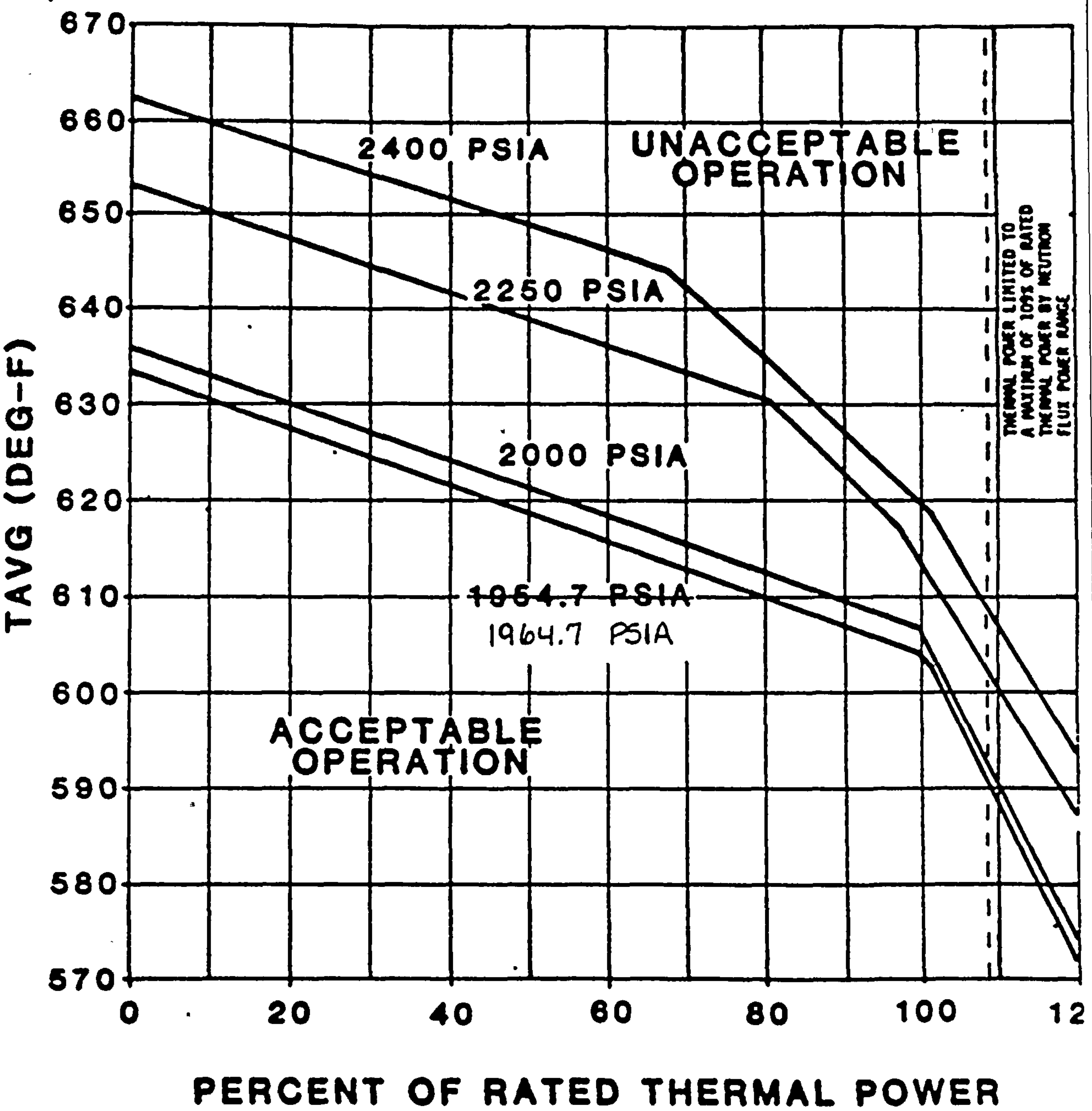


FIGURE 2.1-1b
REACTOR CORE SAFETY LIMIT (UNIT 2, CYCLE 3)

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux a. Low Setpoint b. High Setpoint	 $\leq$ 25% of RATED THERMAL POWER $\leq$ 109% of RATED THERMAL POWER	 $\leq$ 26% of RATED THERMAL POWER $\leq$ 110% of RATED THERMAL POWER
3. Power Range, Neutron Flux High Positive Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
4. Power Range, Neutron Flux High Negative Rate	$\leq$ 5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds	$\leq$ 5.5% of RATED THERMAL POWER with a time constant $\geq$ 2 seconds
5. Intermediate Range, Neutron Flux	$\leq$ 25% of RATED THERMAL POWER	$\leq$ 30% of RATED THERMAL POWER
6. Source Range, Neutron Flux	$\leq 10^5$ counts per second	$\leq 1.3 \times 10^5$ counts per second
7. Overtemperature ΔT	See Note 1	See Note 2
8. Overpower ΔT	See Note 3	See Note 4
9. Pressurizer Pressure-Low	$\geq$ 1950 psig	$\geq$ 1940 psig
10. Pressurizer Pressure-High	$\leq$ 2385 psig	$\leq$ 2395 psig
11. Pressurizer Water Level-High	$\leq$ 92% of instrument span	$\leq$ 93% of instrument span
12. Reactor Coolant Flow-Low	$\geq$ 90% of design flow per loop* for Unit 2 Cycle 3 $\geq$ 90% of minimum measured flow** per loop for Units 1 and 2 Cycle 4 and after	$\geq$ 89% of design flow per loop* for Unit 2 Cycle 3 $\geq$ 88.9% of minimum measured flow** per loop for Units 1 and 2 Cycle 4 and after

*Design flow ~~is 87,700 gpm per loop for Unit 1 and~~ 88,500 gpm per loop for Unit 2.

**Minimum measured flow is 89,800 gpm per loop for Unit 1 and 90,625 gpm per loop for Unit 2.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where: ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

T = Average temperature, °F;

T' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

P = Pressurizer pressure, psig;

P' = 2235 psig (indicated RCS nominal operating pressure);

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 30 s, τ_2 = 4 s;

S = Laplace transform operator, s^{-1};

K_1 = ~~1.174 (Unit 1 Cycle 2);~~ 1.200 (Units 1 and 2 Cycle 4 and after)

K_1 = 1.166 (Unit 2 ~~1~~ Cycle 3 ~~and after, Unit 2~~);

K_2 = ~~0.01358/°F (Unit 1 Cycle 2);~~ 0.01817/°F (Units 1 and 2 Cycle 4 and after)

K_2 = 0.01149/°F (Unit 2 ~~1~~ Cycle 3 ~~and after, Unit 2);~~

K_3 = ~~0.000685/psig (Unit 1 Cycle 2);~~ 0.000831/psig (Units 1 and 2 Cycle 4 and after)

K_3 = 0.000502/psig (Unit 2 ~~1~~ Cycle 3 ~~and after, Unit 2);~~

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 1 (Continued)

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between ~~-32%~~ -19 and ~~+10%~~ +9% ~~(Unit 1 Cycle 2)~~ (Units 1 and 2 Cycle 4 and after, and -32% and +9% (Unit ~~1~~ 2 Cycle 3 ~~and after, Unit 2)~~, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds -19% (Units 1 and 2 Cycle 4 and after) and -32% (Unit 2 Cycle 3) the ΔT Trip Setpoint shall be automatically reduced by ~~2.11%~~ 2.75% (Units 1 and 2 Cycle 4 and after) ~~Cycle 2)~~ and 2.02% (Unit ~~1~~ 2 Cycle 3 ~~and after, Unit 2)~~ of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds ~~+10%~~ +9% (Units 1 and 2 Cycle 4 and after) ~~(Unit 1 Cycle 2)~~ and 9% (Unit ~~1~~ 2 Cycle 3 ~~and after, Unit 2)~~ the ΔT Trip Setpoint shall be automatically reduced by ~~1.45%~~ (Units 1 and 2 Cycle 4 and after) ~~Cycle 2)~~ and 1.454% (Unit ~~1~~ 2 Cycle 3 ~~and after, Unit 2)~~ 1.76% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~4%.~~ 3.2% for Units 1 and 2 Cycle 4 and after, and 4% for Unit 2 Cycle 3.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \leq \Delta T_o [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T-T'') - f_2(\Delta I)]$$

Where: ΔT_o = Indicated ΔT at rated power;

T = Average temperature, °F;

T'' = $\leq$ 576.6°F for Unit 1 and $\leq$ 577.6°F for Unit 2 Reference T_{avg} at RATED THERMAL POWER;

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

K_4 = 1.079 (Unit 2 Cycle 3)

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 $T \leq T''$ (Units 1 and 2 Cycle 4 and after)

K_6 = 0.00121/°F for $T > T''$; $K_6 = 0$ for $T \leq T''$; (Unit 2 Cycle 3)

$\frac{\tau_3 S}{1+\tau_3 S}$ = The function generated by the rate lag controller for T_{avg} dynamic compensation;

τ_3 = Time constant utilized in the rate lag controller for T_{avg} τ_3 = 10 s;

S = Laplace transform operator, s^{-1}; and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3%~~ 2.6% for Units 1 and 2 Cycle 4 and after, and 3% for Unit 2 Cycle 3.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

INSERT 1

Operation above the ~~upper boundary~~ of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the R-Grid correlation. The R-Grid DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1bshow the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$ of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1+ 0.3\ (1-P)]$$

where P is the fraction of RATED THERMAL POWER

The 4% measurement uncertainty is included in the $F^N_{\Delta H}$ value.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trip will reduce the Setpoints to provide protection consistent with core Safety Limits.

INSERT 1

Units 1 and 2 Cycle 4 and after

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during steady-state operation, normal operational transients, and anticipated transients is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 for LOPAR fuel and the WRB-2 for VANTAGE 5 fuel in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with a 95 percent confidence level that DNB will not occur when the minimum DNBR is at or greater than the DNBR limit (1.17 for both the WRB-1 and WRB-2 correlations).

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with a 95 confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. For Diablo Canyon Units, the design DNBR values are 1.33 and 1.37 for thimble and typical cells, respectively, for LOPAR fuel, and 1.30 for thimble and 1.32 for typical cells for the VANTAGE 5 fuel. In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.44 for thimble and 1.48 for typical cells for LOPAR fuel, and 1.68 and 1.71 for thimble and typical cells, respectively, for VANTAGE 5 fuel in performing safety analyses.

The curves in Figure 2.1-1a show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the safety analysis DNBR limits, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$ of 1.56 for LOPAR and 1.59 for VANTAGE 5 fuel, and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expressions:

$$F^N_{\Delta H} = 1.56\ [1 + 0.3(1 - P)] \text{ for LOPAR fuel}$$

$$F^N_{\Delta H} = 1.59\ [1 + 0.3(1 - P)] \text{ for VANTAGE 5 fuel}$$

where P is the fraction of RATED THERMAL POWER

The 4% measurement uncertainty associated with $F^N_{\Delta H}$ is accounted for in the DNBR design limit.

Unit 2 Cycle 3

LIMITING SAFETY SYSTEM SETTINGS

BASES

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate Trip provides protection for control rod drop accidents. At high power, a rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate Trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate Trip for those control rod drop accidents for which the DNBRs will be greater than ~~1.30.~~ or equal to the DNBR limits.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range Neutron Flux trips provide core protection during reactor STARTUP to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Trip System.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity, e.g., no fuel pellet cracking or melting, under all possible overpower conditions, limits the required range for Overtemperature ΔT protection, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied

ROD DROP TIME

27 seconds for Units 1 and 2 Cycle 4 and after, and

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

for Unit 2 Cycle 3

a. T_{avg} greater than or equal to 541°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITIONS FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the allowed operational space defined by Figure 3.2-1a for Units 1 and 2 Cycle 4 and after, and Figure 3.2-1b for Unit 2, Cycle 3. 12&10

APPLICABILITY: MODE 1 ABOVE 50 PERCENT RATED THERMAL POWER*.

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ~~Unit 1~~ Figure 3.2-1a limits or ~~Unit 2~~ Figure 3.2-1b limits, 3&1 12&10

1. Either restore the indicated AFD to within the ~~Unit 1~~ Figure 3.2-1a or ~~Unit 2~~ Figure 3.2-1b limits within 15 minutes, or 3&1 12&10

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55 percent of RATED THERMAL POWER within the next 4 hours. 3&1

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the ~~Unit 1~~ Figure 3.2-1a or ~~Unit 2~~ Figure 3.2-1b limits. 12&10

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 50 percent of RATED THERMAL POWER by: 12&10

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour ~~for the first 24 hours after restoring~~ when the AFD Monitor Alarm ~~to OPERABLE status~~ is inoperable. 3&1

~~b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERNECE shall be assumed to exist during the interval preceding each logging.~~

4.2.1.2 The indicated AFD shall be considered outside of its limits when at least 2 OPERABLE excore channels are indicating the AFD to be outside the limits. 12&10

---

*See Special Test Exceptions Specification 3.10.2

Replace

UNIT 1

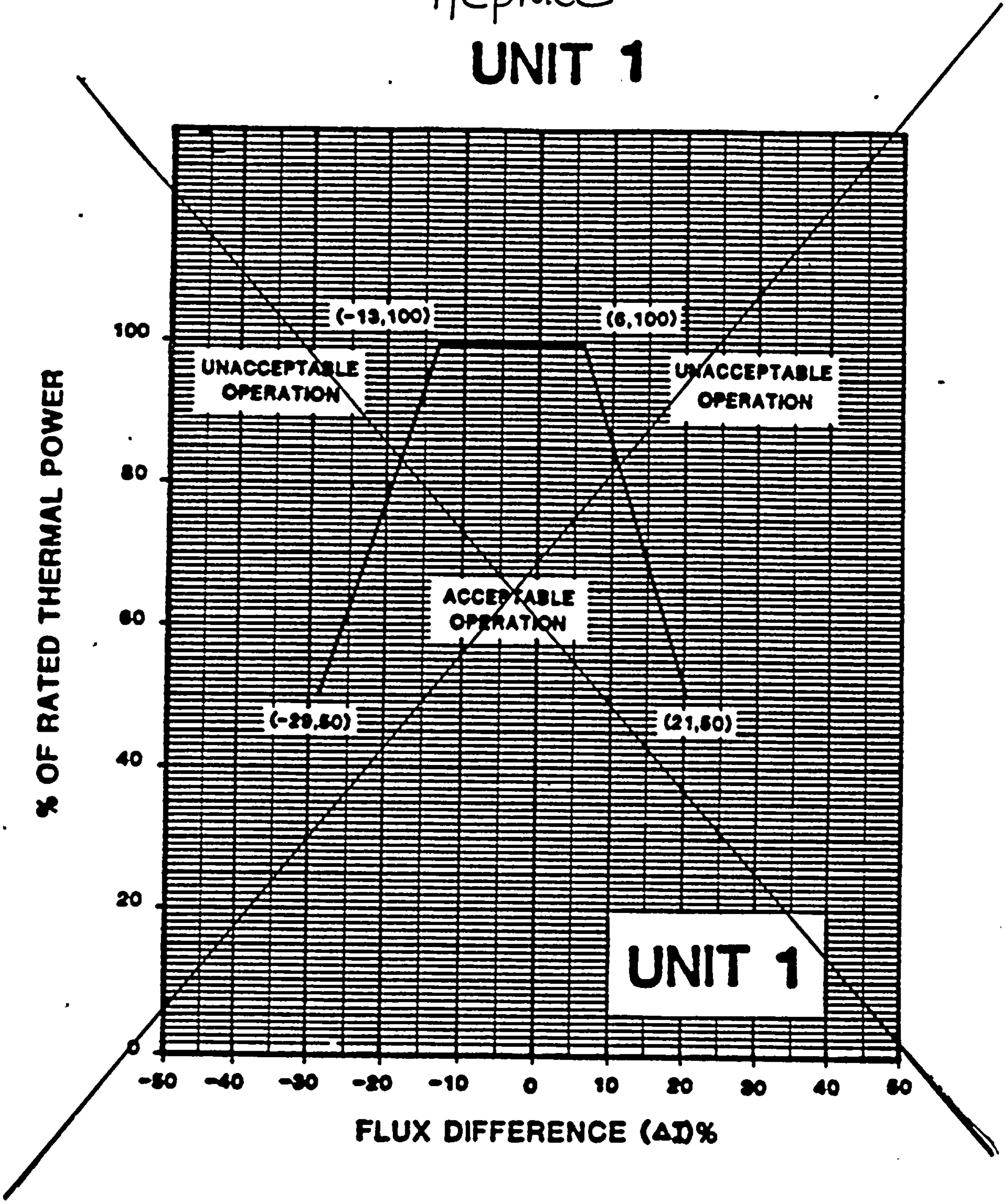


FIGURE 3.2-1a

UNIT 1 AXIAL FLUX DIFFERENCE LIMITS
AS A FUNCTION OF RATED THERMAL POWER

12&10

UNITS 1 & 2 Cycle 4 and after

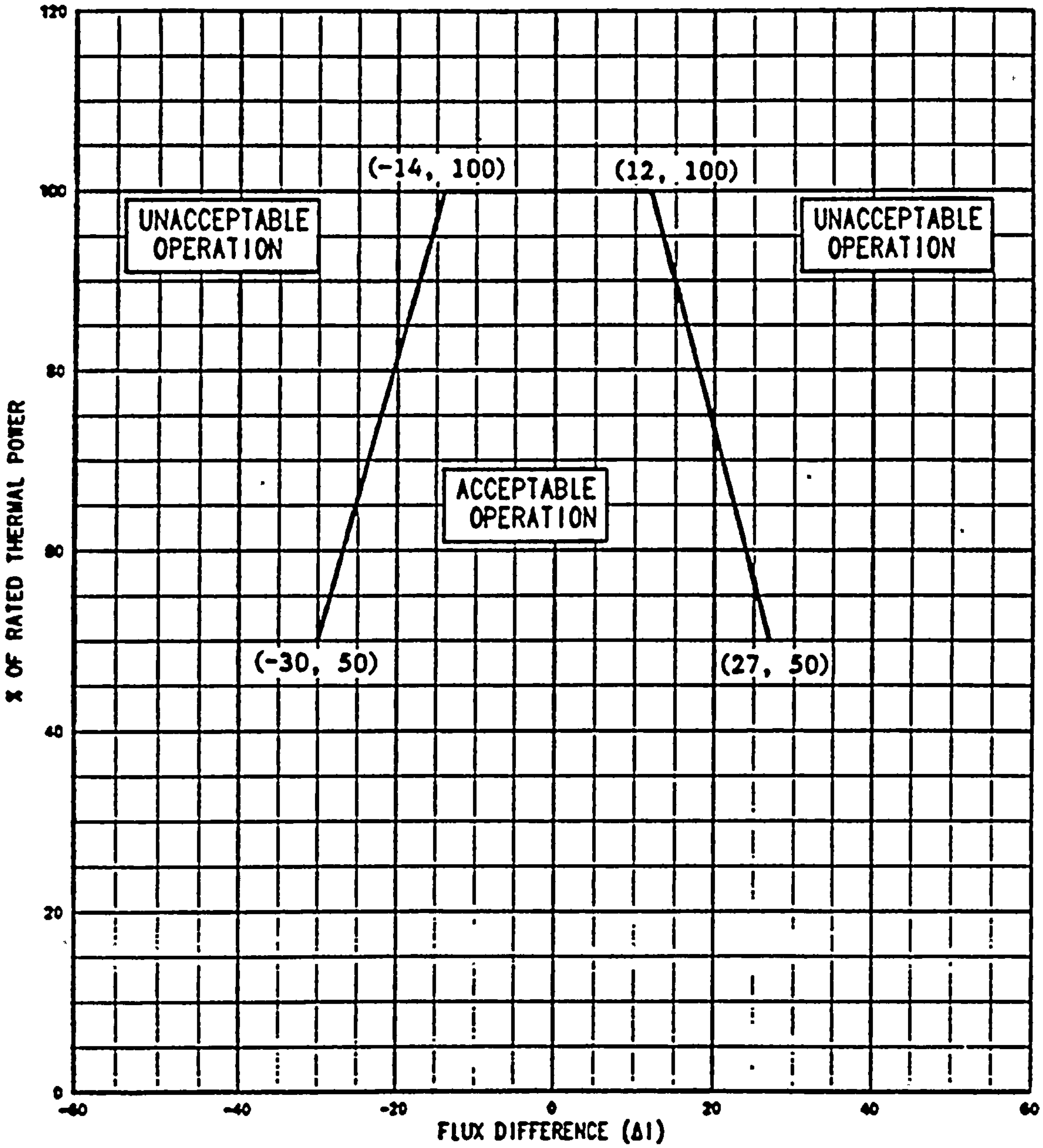


FIGURE 3.2-1A

UNITS 1 and 2 Cycle 4 and after AXIAL FLUX DIFFERENCE LIMITS

AS A FUNCTION OF RATED THERMAL POWER

UNIT 2 CYCLE 3

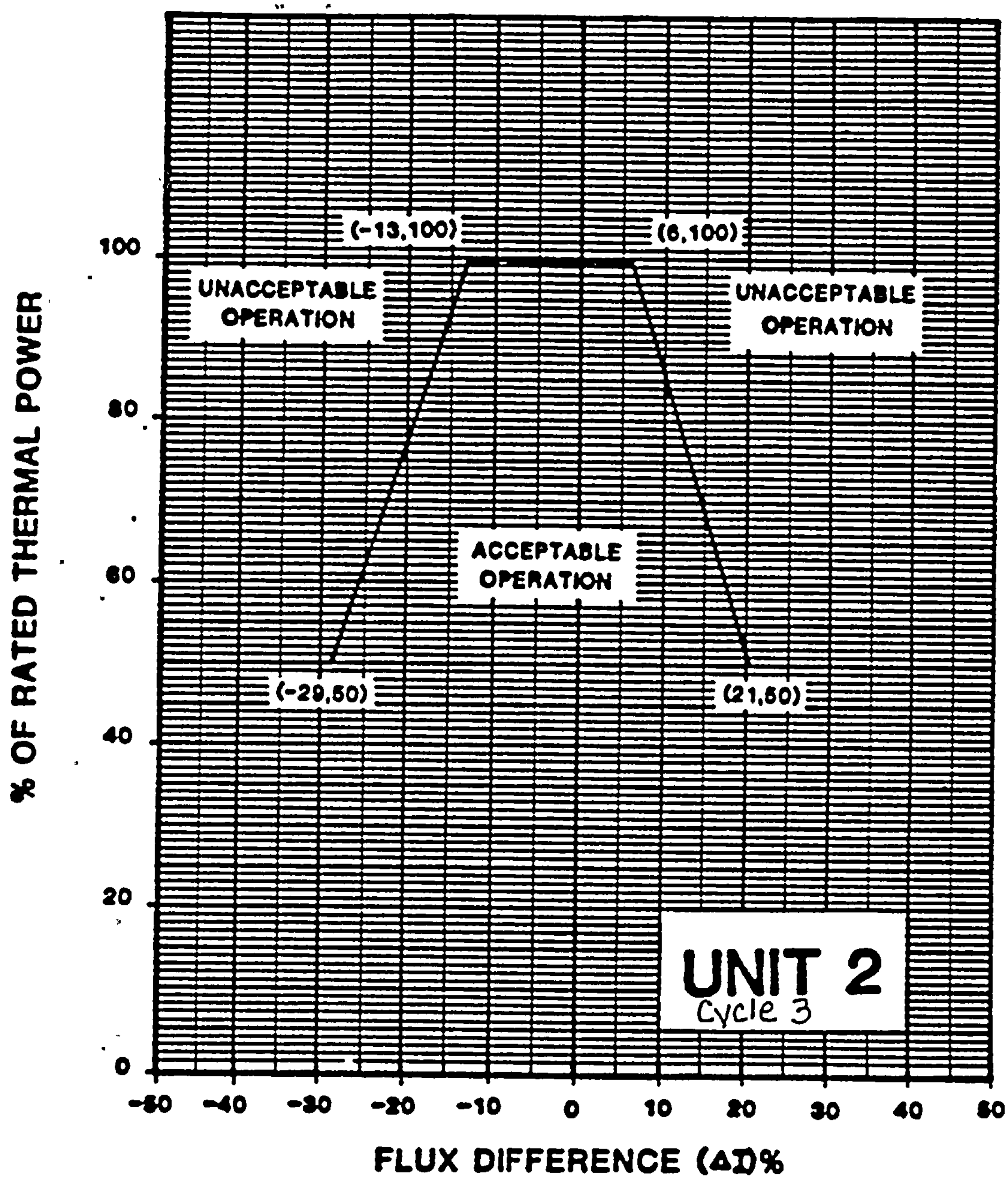


FIGURE 3.2-1b

UNIT 2 Cycle 3 AXIAL FLUX DIFFERENCE LIMITS
AS A FUNCTION OF RATED THERMAL POWER

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2.1 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{\overset{2.45}{\cancel{2.32}}}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq \left[\overset{4.90}{\cancel{4.64}}\right] [K(Z)] \text{ for } P \leq 0.5$$

Where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1* (Units 1 and 2 Cycle 4 and after).

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

INSERT 2

~~4.2.2.2~~ ~~F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:~~

~~a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,~~

~~b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,~~

~~c. Comparing the F_{xy} computed (F^C_{xy}) obtained in Specification 4.2.2.2b., above, to:~~

~~1. The F_{xy} limits for RATED THERMAL POWER (F^{RTP}_{xy}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f. below, and~~

~~2. The relationship:~~

$$\cancel{F^L_{xy} = F^{RTP}_{xy}[1+0.2(1-P)],}$$

~~Where F^L_{xy} is the limit for fractional THERMAL POWER operation expressed as a function of F^{RTP}_{xy} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.~~

~~d. Remeasuring F_{xy} according to the following schedule:~~

~~1. When F^C_{xy} is greater than the F^{RTP}_{xy} limit for the appropriate measured core plane but less than the F^L_{xy} relationship, additional power distribution maps shall be taken and F^C_{xy} compared to F^{RTP}_{xy} and F^L_{xy} either:~~

~~a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F^C_{xy} was last determined, or~~

~~b) At least once per 31 EFPD, whichever occurs first.~~

4.2.2.1.2 $F_Q(z)$ shall be evaluated to determine if $F_Q(z)$ is within its limits by:

a. Using the moveable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{2.45 \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{2.45 \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, 2.45 is the F_Q limit, K(z) is given in Figure 3.2-2, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. This function is given in the Peaking Factor Limit Report as per Specification 6.9.1.8.

d. Measuring $F_Q^M(z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 20% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

e. With measurements indicating

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right)$$

has increased since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

---

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

1) $F_Q^M(z)$ shall be increased by 2% over that specified in Specification 4.2.2.1.2.c, or

2) $F_Q^M(z)$ shall be measured at least once per 7 EFPD until two successive maps indicate that

$$\underset{\text{over } z}{\text{maximum}} \left(\frac{F_Q^M(z)}{K(z)} \right) \text{ is not increasing.}$$

f. With the relationship specified in Specification 4.2.2.1.2.c above not being satisfied:

1) Calculate the percent $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.45}{P} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P \geq 0.5$$

$$\left\{ \left(\underset{\text{over } z}{\text{maximum}} \left[\frac{F_Q^M(z) \times W(z)}{\frac{2.45}{0.5} \times K(z)} \right] \right) - 1 \right\} \times 100 \quad \text{for } P < 0.5$$

2) Either one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in Specification 4.2.2.1.2.c is satisfied. Power level may then be increased provided the AFD limits of Figure 3.2-1a are reduced 1% AFD for each percent $F_Q(z)$ exceeds its limit, or

b) Comply with the requirements of Specification 3.2.2.1 for $F_Q(z)$ exceeding its limit by the percent calculated.

g. The limits specified in Specification 4.2.2.1.2.c, 4.2.2.1.2.e, and 4.2.2.1.2.f above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

SURVEILLANCE REQUIREMENTS (Continued)

~~2. When the F^{C}_{xy} is less than or equal to the F^{RTP}_{xy} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F^{C}_{xy} compared to F^{RTP}_{xy} and F^{L}_{xy} at least once per 31 EFPD.~~

~~e. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.8.~~

~~f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:~~

~~1. Lower core region from 0 to 15%, inclusive,~~

~~2. Upper core region from 85 to 100% inclusive,~~

~~3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and~~

~~4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.~~

~~g. With F^{C}_{xy} exceeding F^{L}_{xy}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.~~

4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR-$F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$$F_Q(Z) \leq \left[\frac{2.32}{P}\right] [K(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) \leq [4.64] [K(Z)] \text{ for } P \leq 0.5$$

Where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1 (Unit 2 Cycle 3).

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. The Overpower ΔT Trip Setpoint reduction shall be performed with the reactor in at least HOT STANDBY.

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

SURVEILLANCE REQUIREMENTS

4.2.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER,

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^C) obtained in Specification 4.2.2.2.2b., above, to:

1. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2.2e. and f. below, and

2. The relationship:

$$F_{xy}^L = F_{xy}^{RTP} [1+0.2(1-P)],$$

Where F_{xy}^L is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1. When F_{xy}^C is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^L relationship, additional power distribution maps shall be taken and F_{xy}^C compared to F_{xy}^{RTP} and F_{xy}^L either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^C was last determined, or

b) At least once per 31 EFPD, whichever occurs first.

SURVEILLANCE REQUIREMENTS (Continued)

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a ~~Radial~~ Peaking Factor Limit Report per Specification 6.9.1.8.

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive,

2. Upper core region from 85 to 100% inclusive,

3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive, and

4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

Cycle 4 and after, Cycle 3, and Figure 3.2-3c for Unit 2 Cycle 4 and after

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3a for Unit 1 ~~and~~ Figure 3.2-3b for Unit 2 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.3\ (1.0 - P)]}$ (Unit 2 Cycle 3)

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

Cycle 4 and after includes measurement uncertainties of 2.4% for flow

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3a for Unit 1 ~~and~~ Figure 3.2-3b for Unit 2 Cycle 3 includes measurement uncertainties of 3.5% for flow ~~and~~ 4% for incore measurement of $F^N_{\Delta H}$. All three Figures account for ~~include~~ measurement uncertainties of

APPLICABILITY: MODE 1.

and Figure 3.2-3c for Unit 2 Cycle 4 and after includes measurement uncertainties of 2.4% for flow.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3a for Unit 1, ~~and~~, Figure 3.2-3b for Unit 2 Cycle 3 and Figure 3.2-3c for Unit 2 Cycle 4 and after Cycle 4 and after:

a. Within 2 hours either:

1. Restore the combination of RCS total flow rate and R to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

$R = \dfrac{F^N_{\Delta H}}{1.56[1.0 + 0.3(1.0-P)]}$ for LOPAR fuel (Units 1 and 2 Cycle 4 and after)

$R = \dfrac{F^N_{\Delta H}}{1.59[1.0 + 0.3(1.0-P)]}$ for VANTAGE 5 fuel (Units 1 and 2 Cycle 4 and after)

REPLACE

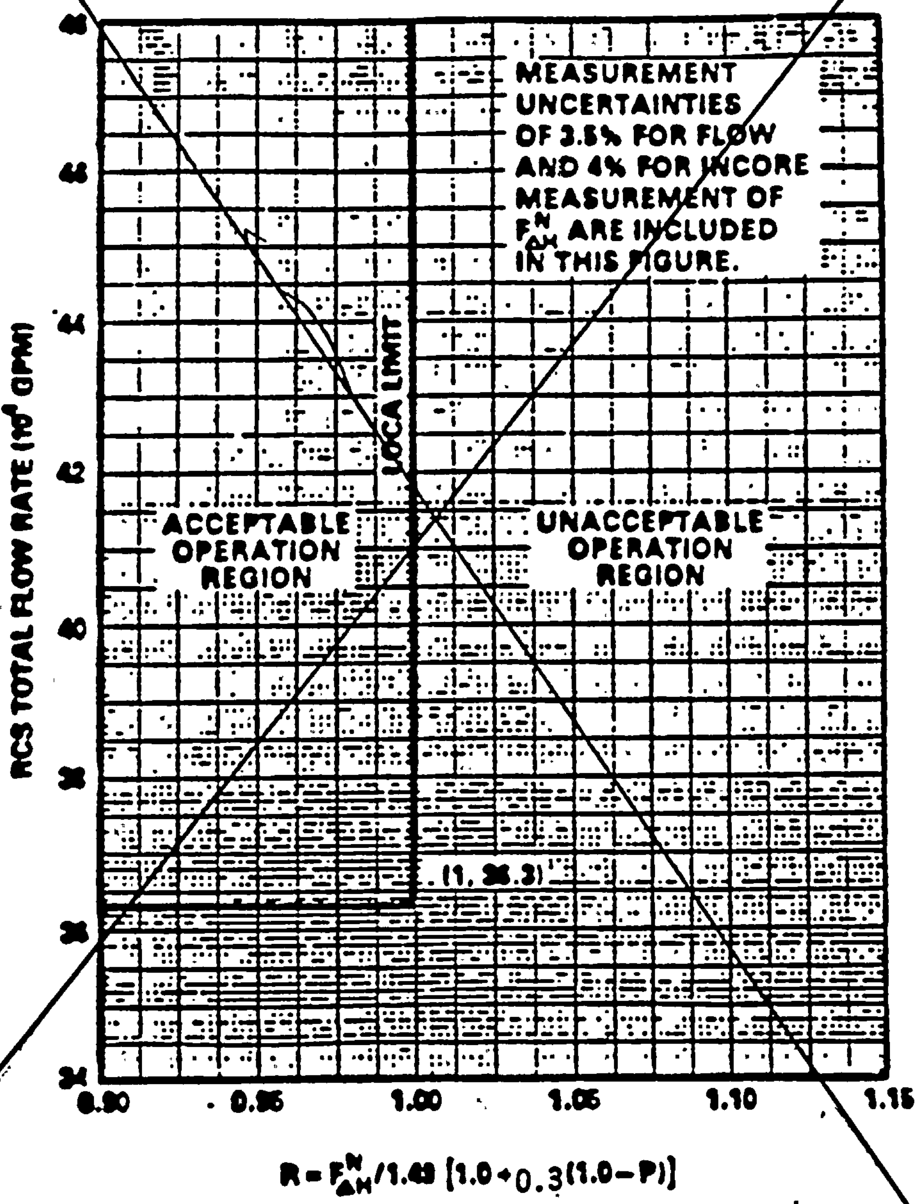
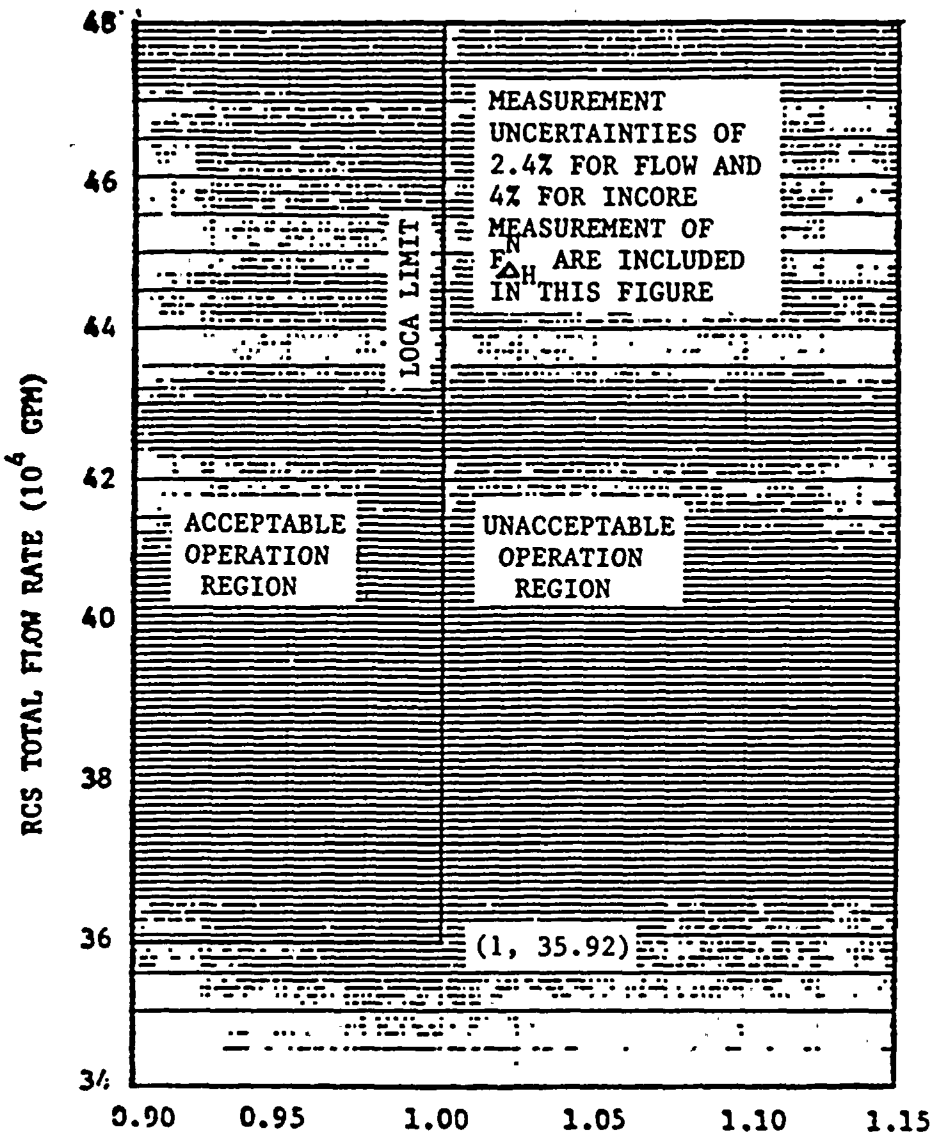


FIGURE 3.2-3a

RCS TOTAL FLOWRATE VERSUS R (UNIT 1)

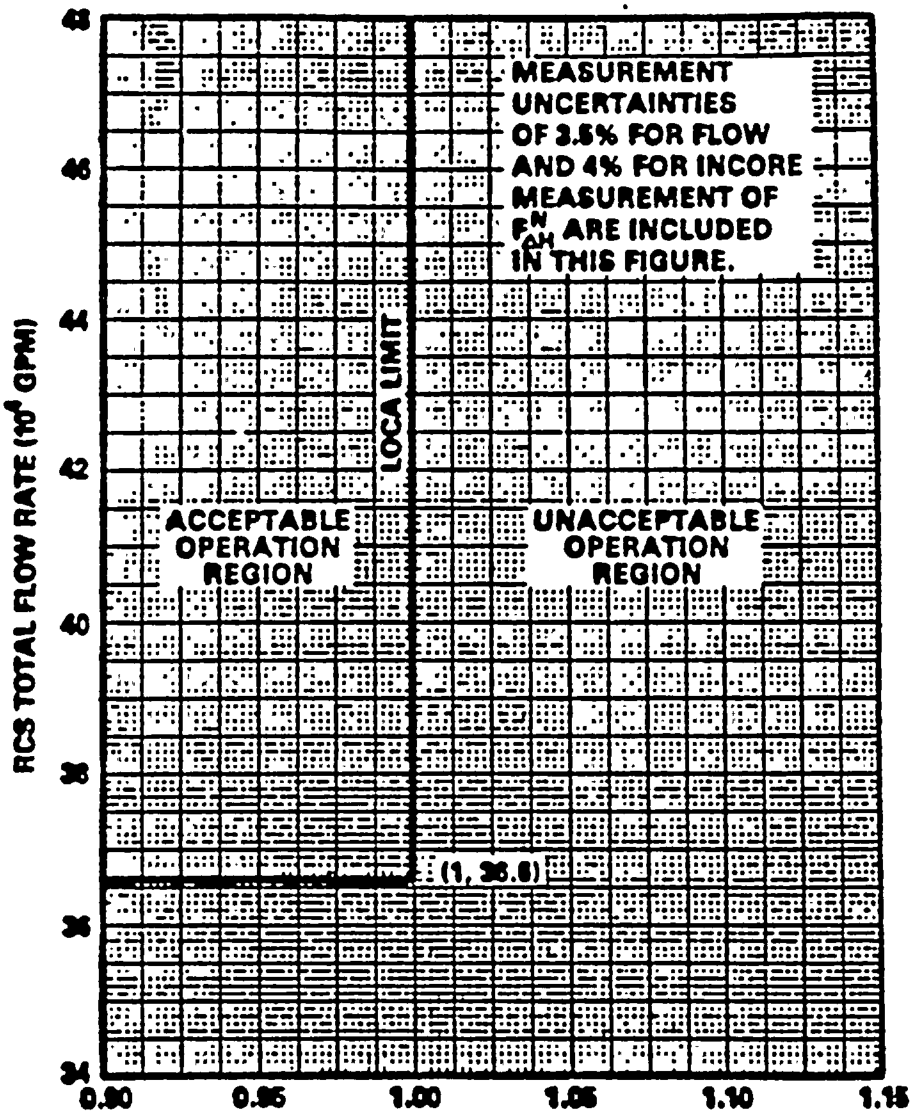


$R = F^N_{\Delta H}/1.56[1.0+0.3(1.0-P)]$ for LOPAR fuel

$R = F^N_{\Delta H}/1.59[1.0+0.3(1.0-P)]$ for VANTAGE 5 fuel

FIGURE 3.2-3a

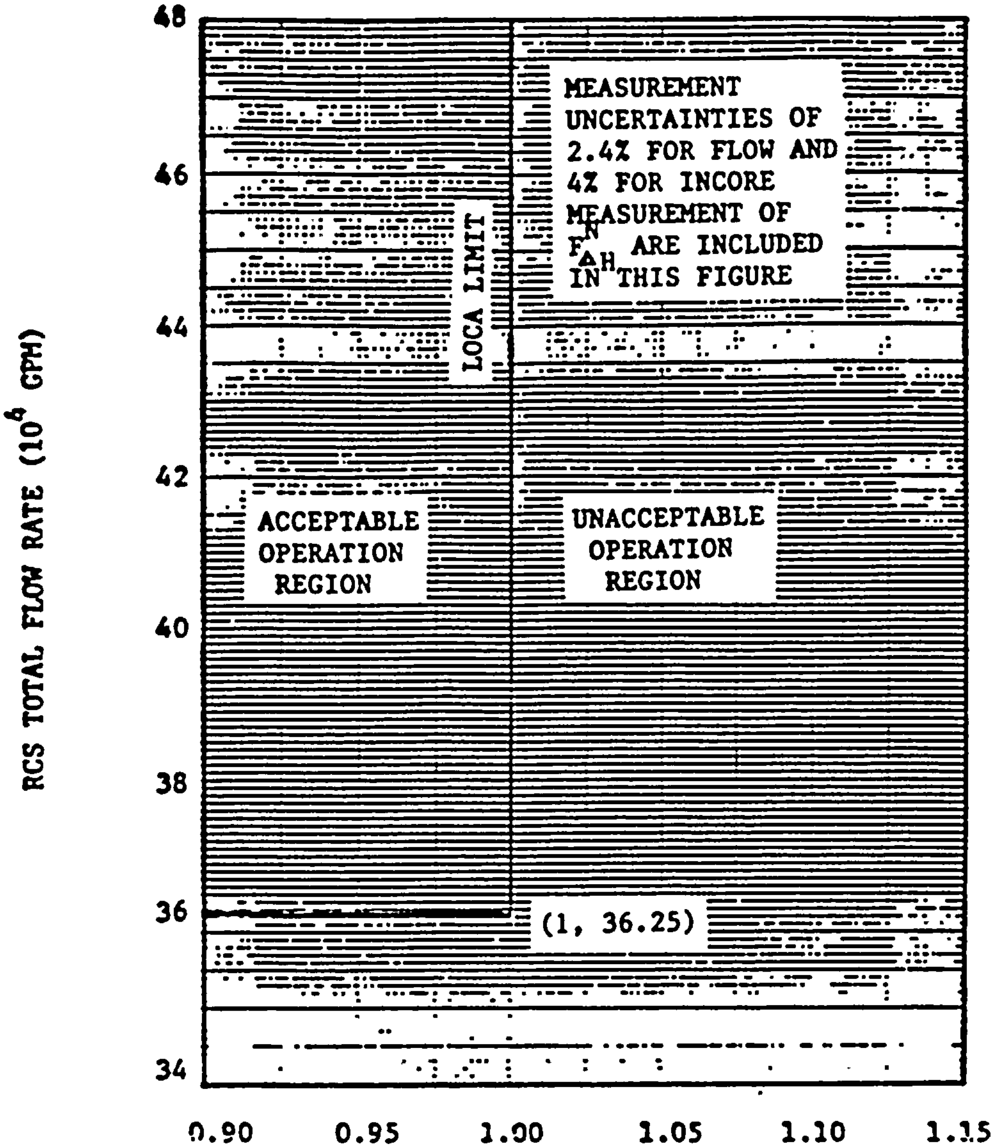
RCS TOTAL FLOW RATE VERSUS R (UNIT 1 cycle 4 and after)



$$R = \frac{F^N_{\Delta H}}{1.49[1.0 + 0.3(1.0-P)]}$$

FIGURE 3.2-3b

RCS TOTAL FLOWRATE VERSUS R (UNIT 2~~)~~ Cycle 3)



$R = F^N_{\Delta H}/1.56[1.0+0.3(1.0-P)]$ for LOPAR fuel

$R = F^N_{\Delta H}/1.59[1.0+0.3(1.0-P)]$ for VANTAGE 5 fuel

FIGURE 3.2-3~~b~~C

RCS TOTAL FLOW RATE VERSUS R (UNIT 2 Cycle 4 and after

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2 and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3a for Unit 1 Cycle 4 and after ~~and~~, Figure 3.2-3b for Unit 2 Cycle 3, and Figure 3.2-3c for Unit 2 Cycle 4 and after prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3a for Unit 1 Cycle 4 and after ~~and~~, Figure 3.2-3b for Unit 2 Cycle 3, and Figure 3.2-3c for Unit 2 Cycle 4 and after:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3a for Unit 1 Cycle 4 and after ~~and~~, Figure 3.2-3b for Unit 2 Cycle 3 and Figure 3.2-3c for Unit 2 Cycle 4 and after at least once per 12 hours when the value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
Actual Reactor Coolant System T_{avg}	$\leq$ ~~581°F~~ 584.3 (Units 1 and 2 Cycle 4 and after) $\leq$ 582°F (Unit 2 Cycle 3)
Actual Pressurizer Pressure	$\geq$ ~~2220~~ 2212 psia* (Units 1 and 2 Cycle 4 and after) $\geq$ 2220 psia* (Unit 2 Cycle 3)

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% RATED THERMAL POWER.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generators, Containment Fan Cooler Units, and Component Cooling Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High	$\leq$ 3 psig	$\leq$ 3.5 psig
d. Pressurizer Pressure-Low	$\geq$ 1850 psig	$\geq$ 1840 psig
e. Differential Pressure Between Steam Lines-High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines-High	$\leq$ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	$\leq$ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.
Coincident With Either		
1) T_{avg}-Low-Low, or	$\geq$ 543°F	$\geq$ 540.2°F (Units 1 and 2 Cycle 4 and after) $\geq$ 541°F (Unit 2 Cycle 3)
2) Steam Line Pressure-Low	$\geq$ 600 psig	$\geq$ 580 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. Containment Isolation (Continued)		
c. Containment Ventilation Isolation		
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
2) Plant Vent Noble Gas Activity-High (RM-14A and 14B)	Per Specification 3.3.3.10.	
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	
4. Steam Line Isolation		
a. Manual	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure-High-High	$\leq$ 22 psig	$\leq$ 24 psig
d. Steam Flow in Two Steam Lines-High	$\leq$ A function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	$\leq$ A function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.
Coincident With Either		
1) T_{avg}-Low-Low, or	$\geq$ 543°F	$\geq$ 540.2°F (Units 1 and 2 Cycle 4 and after) $\geq$ 541°F (Unit 2 Cycle 3)
2) Steam Line Pressure-Low	$\geq$ 600 psig	$\geq$ 580 psig

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TRIP SETPOINT	ALLOWABLE VALUES
7. Loss of Power (4.16 kV Emergency Bus Undervoltage)			
a. First Level			
1) Diesel Start		≥ 0 volts with a ≤ 0.8 second time delay and ≥ 2583 volts with a ≤ 10 second time delay	≥ 0 volts with a ≤ 0.8 second time delay and > 2583 volts with a ≤ 10 second time delay
2) Initiation of Load Shed		One relay ≥ 0 volts with a ≤ 4 second time delay and ≥ 2583 volts with a ≤ 25 second time delay with one relay ≥ 2870 volts, instantaneous	One relay ≥ 0 volts with a ≤ 4 second time delay and > 2583 volts with a ≤ 25 second time delay with one relay ≥ 2870 volts, instantaneous
b. Second Level			
1) Diesel Start		≥ 3600 volts with a ≤ 10 second time delay	≥ 3600 volts with a ≤ 10 second time delay
2) Initiation of Load Shed		≥ 3600 volts with a ≤ 20 second time delay	≥ 3600 volts with a ≤ 20 second time delay
8. Engineered Safety Features Actuation System Interlocks			≤ 545.8°F (Units 1 and 2 Cycle 4 and after) ≥ 540.2°F (Units 1 and 2 Cycle 4 and after)
a. Pressurizer Pressure, P-11		≤ 1915 psig	≤ 1925 psig
b. Low-Low T_{avg}, P-12	increasing decreasing	543°F 543°F	≤ 545°F (Unit 2 Cycle 3) ≥ 541°F (Unit 2 Cycle 3)
c. Reactor Trip, P-4		N.A.	N.A.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2.1 or 3.2.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2.1 or 3.2.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2.1 or 3.2.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.1.2 or 4.2.2.2.2 and 4.2.2.1.3 or 4.2.2.2.3 ~~and 4.2.2.3~~, and

b. Specification 4.2.3.2.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

1. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 12 steps, indicated, from the group demand position,

2. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6,

3. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained, and

4. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions 1. through 4., above, are maintained. ~~As noted on Figure 3.2-3a and Figure 3.2-3b, RCS flow rate and $F^N_{\Delta H}$ may be "traded-off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value.~~ The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

for Unit 2 Cycle 3

and Figure 3.2-3c

R, as calculated per Specification 3.2.3 and used in Figure 3.2-3a, ~~and Figure 3.2-3b~~ accounts for $F^N_{\Delta H}$ less than or equal to 1.49. ese are
Th~~is~~ values ~~is~~ the values used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus are ~~is~~ the maximum "as measured" values allowed. ~~Thus, knowing the "as measured" values of $F^N_{\Delta H}$ and RCS flow allows for "trade offs" in excess of R equal to 1 for the purpose of offsetting the rod bow DNBR penalty.~~

INSERT 3

for Unit 2 Cycle 3

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margin totaling 9.1% DNBR is derived from the difference between the design and required values on the following items: (a) design DNBR limit, (b) grid spacing multiplier, (c) thermal diffusion coefficient, (d) DNBR spacer factor multiplier and (e) pitch reduction. The rod bow penalty is calculated with the method described in WCAP-8691, Revision 1, and is completely compensated by the available margin of 9.1%.

1.56 for LOPAR fuel and 1.59 for VANTAGE 5 fuel for Units 1 and 2 Cycle 4 and after, and

INSERT 3

Margin between the safety analysis limit DNBRs (1.44 and 1.48 for the LOPAR fuel thimble and typical cells, respectively, and 1.68 and 1.71 for the VANTAGE 5 thimble and typical cells) and the design limit DNBRs (1.33 and 1.37 for the LOPAR fuel thimble and typical cells and 1.30 and 1.32 for the VANTAGE 5 fuel thimble and typical cells, respectively) is maintained for Units 1 and 2 Cycle 4 and after. A fraction of this margin is utilized to accommodate the transition core DNBR penalty of maximum 12.5 percent and the appropriate fuel rod bow DNBR penalty (less than 1.5 percent for both fuel types per WCAP-8691, Revision 1) for Units 1 and 2 Cycle 4 and after. The rest of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

POWER DISTRIBUTION LIMITS

BASES

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When RCS flow rate and $F^N_{\Delta H}$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 a, b, and c. Measurement errors of 2.4% for Units 1 and 2 Cycle 4 and after, and 3.5% for Unit 2 Cycle 3 for RCS total flow rate and 4% for $F^N_{\Delta H}$ have been allowed for in determination of the design DNBR values. For Units 1 and 2 Cycle 4 and after, the 4% is applied to the DNBR limit. For Unit 2 Cycle 3, the 4% is applied to $F^N_{\Delta H}$.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figures 3.2-3a ~~and Figure 3.2-3b~~, b, and c.

For Unit 2 Cycle 3, the ~~Radial~~ Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide additional assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F^{RTP}_{xy}) as provided in the ~~Radial~~ Peaking Factor Limit Report per Specification 6.9.1.8 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

INSERT 4

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the power by 3% for each percent of tilt in excess of 1.

INSERT 4

For Units 1 and 2 Cycle 4 and after, the hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to RAOC operation, W(z), to provide assurance that the limit on the hot channel factor $F_Q(z)$ is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The W(z) function for normal operation is provided in the Peaking Factor Limit Report per Specification 6.9.1.8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and safety analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain ~~a minimum DNBR of 1.30~~ throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

the DNBR limits.

MONTHLY OPERATING REPORT

6.9.1.7 Routine reports of operating statistics and shutdown experience, including documentation of all challenges and failures to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

~~RADIAL~~ PEAKING FACTOR LIMIT REPORT

6.9.1.8 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the NRC, in accordance with 10 CFR 50.4 ~~Regional Administrator with a copy to Director of Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555~~, for all core planes containing Bank "D" control rods and all unrodded core planes at least 60 days prior to each cycle initial criticality. In the event that the limit would be submitted at some other time during core life, it will be submitted 60 days prior to the date the limit would become effective unless otherwise exempted by the Commission. This report is not required for the initial cycle.

INSERT 5

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

INSERT 5

The W(z) function for Load Follow operation shall be established for at least each reload core and shall be maintained available at the plant. The limits shall be established and implemented on a time scale consistent with normal procedural changes. The analytical methods used to generate the W(z) function shall be those previously reviewed and approved by the NRC.* If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

A report containing the W(z) function, as a function of core height (and burnup if applicable), shall be provided to the NRC in accordance with 10 CFR 50.4 within 30 days of their implementation.

---

* WCAP-8385 "Power Distribution Control and Load Following Procedures." WCAP-9272-A "Westinghouse Reload Safety Evaluation Methodology," and WCAP-10216-P-A "Relaxation of Constant Axial Offset Control / F_Q Surveillance Technical Specification."

Attachment 4

APPENDICES

2422S/0065K

Appendix A

NON-LOCA ANALYSIS

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont)

TABLE OF CONTENTS (Cont)

TABLE OF CONTENTS (Cont)

TABLE OF CONTENTS (Cont)

TABLES

TABLE OF CONTENTS (Cont)

FIGURES

TABLE OF CONTENTS (Cont)

FIGURES

TABLE OF CONTENTS (Cont)

FIGURES

TABLE OF CONTENTS (Cont)

FIGURES

TABLE OF CONTENTS (Cont)

FIGURES

TABLE OF CONTENTS (Cont)

FIGURES

Chapter 15

ACCIDENT ANALYSES

Since 1970, the ANS classification of plant conditions has been used to divide plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

(1) Condition I: Normal Operation and Operational Transients

(2) Condition II: Faults of Moderate Frequency

(3) Condition III: Infrequent Faults

(4) Condition IV: Limiting Faults.

The basic principle applied in relating design requirements to each of the conditions is that the most frequent occurrences must yield little or no radiological risk to the public, and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safety features functioning is assumed, to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

In the evaluation of the radiological consequences associated with initiation of a spectrum of accident conditions, numerous assumptions must be postulated. In many instances these assumptions are a product of extremely conservative judgments. This is due to the fact that many physical phenomena, in particular fission product transport under accident conditions, are not understood to the extent that accurate predictions can be made. Therefore, the set of assumptions postulated would predominantly determine the accident classification.

The specific accident sequences analyzed in this chapter include those required by Revision 1 of Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, and others considered significant for the Diablo Canyon Power Plant (DCPP). Because the DCPP design differs from other plants, some of the accidents identified in Table 15-1 of Regulatory Guide 1.70, Revision 1, are not applicable to this plant; some comments on these items are as follows:

(Item 10) - There are no pressure regulators or regulating instruments in the Westinghouse pressurized water reactor (PWR) design whose failure could cause heat removal greater than heat generation.

(Item 11) - Reactor coolant flow controller is not a feature of the Westinghouse PWR design. Treatment of the performance of the reactivity controller in a number of accident conditions is offered in this chapter.

(Item 12) - The reactor coolant system (RCS) components whose failure could cause a Condition III or Condition IV loss-of-coolant accident (LOCA) are Design Class I components; that is, they are designed to withstand consequences of the safe shutdown earthquake (SSE) which is equivalent to the double design earthquake (DDE) occurrence. In addition, the analyses of the design LOCA includes the assumption of unavailability of offsite power.

(Item 22) - No instrument lines from the RCS boundary in the DCPP design penetrate the containment[a].

(Item 24) - The analysis of the consequences of such small spills and leaks is included within the cases evaluated in Chapter 11, and larger leaks and spills are analyzed in Section 15.5.

(a) For definition of the RCS boundary, refer to the 1972 issue of ANS N18.2, Nuclear Safety Criteria for the Design of Stationary PWR Plants.

(Item 25) - The radiological consequences of this event are analyzed in Chapter 11, for the case of "Anticipated Operational Occurrences."

(Item 26) - Habitability of the control room following accident conditions is discussed in Chapter 6, and potential radiological exposures are reported in Section 15.5. In addition, Chapter 7 contains an analysis showing that the plant can be brought to, and maintained in, the hot shutdown condition from outside the control room.

(Item 27) - Overpressurization of the residual heat removal system (RHRS) is considered extremely unlikely due to the isolation valve interlocks described in Section 7.6.

(Item 28) - This event is covered by the analyses of Section 15.2.7, Loss of External Electrical Load and/or Turbine Trip.

(Item 29) - Same as Item 28 above.

(Item 30) - Malfunctions of auxiliary saltwater systems and component cooling water systems (CCWS) are discussed in Chapter 9, Sections 9.2.1 and 9.2.2.

(Item 31) - There are no significant safety-related consequences of this event.

(Item 33) - The effects of turbine trip on the RCS are presented in Section 15.2.7.

(Item 34) - Malfunctions of this system are discussed in Section 9.3.2.

(Item 35) - The radiological effects of this event are not significant for PWR plants. Minor leakages are within the scope of the analysis cases presented in Chapter 11.

15.1 CONDITION I - NORMAL OPERATION AND OPERATIONAL TRANSIENTS

Condition I occurrences are those that are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition is generallly based on a conservative set of initial conditions corresponding to the most adverse set of conditions that can occur during Condition I operation.

A typical list of Condition I events is shown below:

(1) Steady state and shutdown operations

Mode 1 - Power operation (> 5% of rated thermal power)

Mode 2 - Startup ($K_{eff} \geq 0.99$, $\leq$ 5% of rated thermal power)

Mode 3 - Hot standby ($K_{eff} < 0.99$, $T_{avg} \geq 350°F$)

Mode 4 - Hot shutdown (subcritical, residual heat removal system in operation, $K_{eff} < 0.99$, $200°F < T_{avg} < 350°F$)

Mode 5 - Cold shutdown (subcritical, residual heat removal system in operation, $K_{eff} < 0.99$, $T_{avg} \leq 200°F$)

Mode 6 - Refueling ($K_{eff} \leq 0.95$, $T_{avg} \leq 140°F$)

(2) Operation with permissible deviations

Various deviations that may occur during continued operation as permitted by the plant Technical Specifications[1] must be considered in conjunction with other operational modes. These include:

(a) Operation with components or systems out of service

(b) Leakage from fuel with cladding defects

(c) Activity in the reactor coolant

1. Fission products

2. Corrosion products

3. Tritium

(d) Operation with steam generator leaks up to the maximum allowed by the Technical Specifications

(3) Operational transients

(a) Plant heatup and cooldown (up to 100°F/hour for the reactor coolant system (RCS); 200°F/hour for the pressurizer)

(b) Step load changes (up to ±10%)

(c) Ramp load changes (up to 5% per minute)

(d) Load rejection up to and including design load rejection transient

15.1.1 Optimization of Control Systems

A setpoint study[2] has been performed in order to simulate performance of the reactor control and protection systems. Emphasis was placed on the development of a control system that will automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both system stability and transient performance.

For each mode of plant operation, a group of optimum controller setpoints is determined. In areas where the resultant setpoints are different, compromises based on the optimum overall performance are made and verified. A consistent set of control system parameters is derived satisfying plant operational requirements throughout the core life and for power levels between 15 and 100%. The study comprises an analysis of the following control systems: rod cluster assembly control, steam dump, steam generator level, pressurizer pressure, and pressurizer level.

15.1.2 Initial Power Conditions Assumed in Accident Analyses

Reactor power-related initial conditions assumed in the accident analyses presented in this chapter are described in this section.

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 capabilty. 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 containment and 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.

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 — $\pm$38 psi allowance for steady state fluctuations and measurement error.

For some accident evaluations, an additional 2.0°F has been conservatively added to the measurement error for the average RCS temperatures to account for steam generator fouling.

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 Technical Specification 3/4.2.

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

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 on 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 on 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 Analysis

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 on 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).

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.

15.1.4 Rod Cluster Control Assembly Insertion Characteristic

The negative reactivity insertion following a reactor trip is a function of the acceleration of the RCCA and the variation in rod worth as a function of rod position.

With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry or approximately 85% of the rod cluster travel. For accident analyses, the insertion time to dashpot entry is conservatively taken as 2.7 seconds. The RCCA position versus time assumed in accident analyses is shown on Figure 15.1-2.

Figure 15.1-3 shows the fraction of total negative reactivity insertion for a core where the axial distribution is skewed to the lower region of the core. This curve is used as input to all point kinetics core models used in transient analyses.

There is inherent conservatism in the use of this curve in that it is based on a skewed axial power distribution that would exist relatively infrequently. For cases other than those associated with xenon oscillations, significant negative reactivity would have been inserted due to the more favorable axial power distribution existing prior to trip.

The normalized RCCA negative reactivity insertion versus time is shown on Figure 15.1-4. The curve shown in this figure was obtained from Figures 15.1-2 and 15.1-3. A total negative reactivity insertion following a trip of 4% Δk is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available as shown in Tables 4.3-2 and 4.3-3.

The normalized RCCA negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 15.1-4) is used in transient analyses.

Where special analyses require the use of three-dimensional or axial one-dimensional core models, the negative reactivity insertion resulting from reactor trip is calculated directly by the reactor kinetic code and is not separable from other reactivity feedback effects. In this case, the RCCA position versus time of Figure 15.1-2 is used as a code input.

15.1.5 Reactivity Coefficients

The transient response of the reactor coolant system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. These reactivity coefficients and their values are discussed in detail in Chapter 4.

In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Some analyses, such as loss of reactor coolant from cracks or ruptures in the RCS, do not depend on reactivity feedback effects. The values used are given in Table 15.1-4; reference is made in that table to Figure 15.1-5 that shows the upper and lower Doppler power coefficients, as a function of power, used in the transient analysis. The justification for use of conservatively large versus small reactivity coefficient values is treated on an event-by-event basis.

To facilitate comparison, individual sections in which justification for the use of large or small reactivity coefficient values is to be found are referenced below:

	Condition II Events	Section
(1)	Uncontrolled RCCA bank withdrawal from a subcritical condition	15.2.1
(2)	Uncontrolled RCCA bank withdrawal at power	15.2.2
(3)	RCCA misoperation	15.2.3
(4)	Uncontrolled boron dilution	15.2.4
(5)	Partial loss of forced reactor coolant flow	15.2.5
(6)	Startup of an inactive reactor coolant loop	15.2.6
(7)	Loss of external electrical load and/or turbine trip	15.2.7
(8)	Loss of all offsite power to the station auxiliaries (station blackout)	15.2.9
(9)	Excessive heat removal due to feedwater system malfunctions	15.2.10
(10)	Excessive load increase incident.	15.2.11

(11)	Accidental depressurization of the RCS	15.2.12
(12)	Accidental depressurization of main steam system.	15.2.13

Condition III events

(1)	Complete loss of forced reactor coolant flow	15.3.4
(2)	Single RCCA withdrawal at full power.	15.3.6

Condition IV Events

(1)	Rupture of a steam pipe	15.4.2
(2)	Single reactor coolant pump locked rotor	15.4.4
(3)	Rupture of a control rod drive mechanism housing (RCCA ejection).	15.4.6

15.1.6 Fission Product Inventories

The fission product inventories existing in the core and fuel rod gaps are described in Section 15.5.2. The description of the models used for calculating fuel gap activities is included in Section 15.5.2.

15.1.7 Residual Decay Heat

Residual heat in a subcritical core consists of:

(1) Fission product decay energy

(2) Decay of neutron capture products

(3) Residual fissions due to the effect of delayed neutrons.

These constituents are discussed separately in the following paragraphs.

15.1.7.1 Fission Product Decay

For short times ($<10^3$ seconds) after shutdown, data on yields of short-half-life isotopes is sparse. Very little experimental data is available for the gamma ray contributions and even less for the beta ray contribution. Several authors have compiled the available data into a conservative estimate of fission product decay energy for short times after shutdown, notably Shure[6], Dudziak[7], and Teage[8]. Of these three selections, Shure's curve is the highest and it is based on the data of Stehn and Clancy[9] and Obenshain and Foderaro[10]. The fission product contribution to decay heat that has been assumed in the LOCA accident analyses is the curve of Shure increased by 20% for conservatism. This curve with the 20% factor included is shown on Figure 15.1-6. For the non-LOCA analyses the 1979 ANS decay heat curve is used.[11] Figure 15.1-7 presents this curve as a function of time after shutdown.

15.1.7.2 Decay of U-238 Capture Products

Betas and gammas from the decay of U-239 (23.5-minute half-life) and Np-239 (2.35-day half-life) contribute significantly to the heat generation after shutdown. The cross sections for production of these isotopes and their decay schemes are relatively well known. For long irradiation times their contribution can be written as:

$$P_1/P_o = \frac{(E_{T1} + E_{\beta 1})c(1+a)}{200\ \text{MeV}} e^{-\lambda_1 t} \text{ watts/watt} \qquad (15.1\text{-}1)$$

$$P_2/P_o = \frac{(E_{T2} + E_{\beta 2})c(1+\alpha)}{200\ \text{MeV}} \left[\frac{\lambda 2}{\lambda 1 - \lambda 2}\left(e^{-\lambda_2 t} - e^{-\lambda_1 t}\right) + e^{-\lambda 2 t}\right] \text{ watts/watt} \qquad (15.1\text{-}2)$$

where:

P_1/P_o is the energy from U-239 decay

P_2/P_o is the energy from Np-239 decay t is the time after shutdown (seconds)

$c(1+\alpha)$ is the ratio of U-238 captures to total fissions = 0.6(1 + 0.2)

λ_1 = the decay constant of U-239 = 4.91×10^{-4} seconds^{-1}

λ_2 = the decay constant of Np-239 decay = 3.41×10^{-6} seconds

E_{T^1} = total T ray energy from U-239 decay = 0.06 MeV

E_{T^2} = total T ray energy from Np-239 decay = 0.30 MeV

E_{β^1} = total β ray energy from U-239 decay = 1/3[a] x 1.18 MeV

E_{β^2} = total β ray energy from Np-239 decay = 1/3[a] x .43 MeV

This expression with a margin of 10% is shown on Figure 15.1-6 as it is used in the LOCA analysis. The 10% margin, compared to 20% for fission product decay, is justified by the availability of the basic data required for this analysis. The decay of other isotopes, produced by neutron reactions other than fission, is neglected. For the non-LOCA analysis the decay of U-238 capture products is included as an integral part of the 1979 decay heat curve presented as Figure 15.1-7.

15.1.7.3 Residual Fissions

The time dependence of residual fission power after shutdown depends on core properties throughout a transient under consideration. Core average conditions are more conservative for the calculation of reactivity and power

(a) Two-thirds of the potential β-energy is assumed to escape by the accompanying neutrinos.

level than actual local conditions as they would exist in hot areas of the core. Thus, unless otherwise stated in the text, static power shapes have been assumed in the analysis and these are factored by the time behavior of core average fission power calculated by a point kinetics model calculation with six delayed neutron groups.

For the purpose of illustration, only one delayed neutron group calculation, with a constant shutdown reactivity of -4% Δk is shown on Figure 15.1-6.

15.1.7.4 Distribution of Decay Heat Following Loss-of-coolant Accident

During a loss-of-coolant accident (LOCA) the core is rapidly shut down by void formation or RCCA insertion, or both, and long-term shutdown is assured by the borated ECCS water. A large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects that are important for the neutron dependent part of the heat generation do not apply to the gamma ray source contribution. The steady state factor of 97.4% that represents the fraction of heat generated within the cladding and pellet drops to 95% for the hot rod in a LOCA.

For example, consider the transient resulting from the postulated double-ended break of the largest RCS pipe; 1/2 second after the rupture about 30% of the heat generated in the fuel rods is from gamma ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10% of the gamma ray contribution or 3% of the total. Since the water density is considerably reduced at this time, an average of 98% of the available heat is deposited in the fuel rods, the remaining 2% being absorbed by water, thimbles, sleeves, and grids. The net effect is a factor of 0.95, rather than 0.974, to be applied to the heat production in the hot rod.

15.1.8 Computer Codes Utilized

Summaries of some of the principal computer codes used in transient analyses are given below. Other codes, in particular, very specialized codes in which the modeling has been developed to simulate one given accident, such as the SATAN-VI code used in the analysis of the RCS pipe rupture (Section 15.4), and which consequently have a direct bearing on the analysis of the accident itself, are summarized in their respective accident analyses sections. The codes used in the analyses of each transient are listed in Table 15.1-4.

15.1.8.1 FACTRAN

FACTRAN[12] calculates the transient temperature distribution in a cross section of a metalclad UO_2 fuel rod (see Figure 15.1-8) and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model that exhibits the following features simultaneously:

(1) A sufficiently large number of finite difference radial space increments to handle fast transients such as rod ejection accidents

(2) Material properties that are functions of temperature and a sophisticated fuel-to-cladding gap heat transfer calculation

(3) The necessary calculations to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials.

The gap heat transfer coefficient is calculated according to an elastic pellet model. The thermal expansion of the pellet is calculated as the sum of the radial (one-dimensional) expansions of the rings. Each ring is assumed to expand freely. The cladding diameter is calculated based on thermal expansion and internal and external pressures.

If the outside radius of the expanded pellet is smaller than the inside radius of the expanded clad, there is no fuel-cladding contact and the gap conductance is calculated on the basis of the thermal conductivity of the gas contained in the gap. If the pellet outside radius so calculated is larger than the cladding inside radius (negative gap), the pellet and the cladding are pictured as exerting upon each other a pressure sufficient to reduce the gap to zero by elastic deformation of both. This contact pressure determines the heat transfer coefficient.

FACTRAN is further discussed in Reference 12.

15.1.8.2 LOFTRAN

The LOFTRAN[13] program is used for studies of transient response of a PWR system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by modeling the reactor core and vessel, hot and cold leg piping, steam generator (tube and shell sides), pressurizer, and reactor coolant pumps, with up to four reactor coolant loops. The pressurizer heaters, spray, relief and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator utilizes a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The reactor protection system is simulated to include reactor trips on neutron flux, overpower and overtemperature reactor coolant ΔT, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated including rod control, steam dump, feedwater control, and pressurizer pressure control. The safety injection system (SIS), including the accumulators, is also modeled.

LOFTRAN is a versatile program that is suited to both accident evaluation and control studies as well as parameter sizing. LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits illustrated on Figure 15.1-1. The core limits represent the minimum value of DNBR as calculated for a typical or thimble cell.

LOFTRAN is further discussed in Reference 13.

15.1.8.3 LEOPARD

The LEOPARD[14] computer program determines fast and thermal spectra using only basic geometry and temperature data. The code optionally computes fuel depletion effects for a dimensionless reactor and recomputes the spectra before each discrete burnup step.

LEOPARD is further discussed in Reference 14.

15.1.8.4 TURTLE

TURTLE[15] is a two-group, two-dimensional neutron diffusion code featuring direct treatment of the nonlinear effects of xenon, enthalpy, and Doppler. Fuel depletion is allowed.

TURTLE was written for the study of azimuthal xenon oscillations, but the code is useful for general analysis. The input is simple, fuel management is handled directly, and a boron criticality search is allowed.

TURTLE is further described in Reference 15.

15.1.8.5 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-clad-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.

15.1.8.6 THINC

The THINC code is described in Section 4.4.3.

15.1.9 REFERENCES

1. Technical Specification, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended to the date of the most recent FSAR Update Revision.

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

3. Chelmer, H., et al., "Improved Thermal Design Procedure," WCAP-8567 (Proprietary) and WCAP-8568 (Non-Proprietary), July 1975.

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

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

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

7. K. Shure and D. J. Dudziak, "Calculating Energy Released by Fission Products," Trans. Am. Nucl. Soc. 4 (1) 30, 1961.

8. U.K.A.E.A. Decay Heat Standard.

9. J. R. Stehn and E. F. Clancy, "Fission-Product Radioactivity and Heat Generation," Proceeding of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, Volume 13, United Nations, Geneva, 1958, pp. 49-54.

10. F. E. Obenshain and A. H. Foderaro, Energy from Fission Product Decay, WAPD-P-652. 1955.

11. ANSI/ANS-5.1-1979, "Decay Heat Power In Light Water Reactors", August 29, 1979.

12. C. Hunin, FACTRAN, a Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod, WCAP-7908, June 1972.

13. T. W. T. Burnett et al, LOFTRAN Code Description, WCAP-7907, June 1972.

14. R. F. Barry, LEOPARD, a Spectrum Dependent Non-Spatial Depletion Code for the IBM-7904, WCAP-3269-26, September 1963.

15. R. F. Altomare and S. Barry, The TURTLE 24.0 Diffusion Depletion Code, WCAP-7213, June 1968, (Westinghouse NES Proprietary); WCAP-7758, September 1971.

16. D. H. Risher, Jr. and R. F. Barry, TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, WCAP-7979, November 1972.

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	12	12
Guaranteed nuclear steam supply system thermal power output	3350	3423
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.

TABLE 15.1-2

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay, sec
Power range high neutron flux, high setting	118%	0.5
Power range high neutron flux, low setting	35%	0.5
Overtemperature ΔT	Variable, see Figure 15.1-1	6[a]
Overpower ΔT	Variable, see Figure 15.1-1	6[a]
High pressurizer pressure	2410 psig	2
Low pressurizer pressure	1845 psig	2
Low reactor coolant flow (from loop flow detectors)	87% loop flow	1
Undervoltage trip	(b)	1.5

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay, sec
Turbine trip	Not applicable	1
Low-low steam generator level	0% of narrow range level span	2
High-high steam generator level trip of the feedwater pumps and closure of feedwater system valves and turbine trips	75% of narrow range level span	2

(a) Total time delay (including RTD bypass loop fluid transport delay, effect bypass loop piping thermal capacity, RTD time response, and trip circuit channel electronics delay) from the time the temperature difference in the coolant loops exceeds the trip setpoint until the rods are free to fall.

(b) A specific undervoltage setpoint was not assumed in the safety analysis.

TABLE 15.1-4

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients: Moderator Temp.[a], pcm/°F[d],	Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II					
Uncontrolled RCCA bank withdrawal from a subcritical condition	THINKLE, THINC, FACTRAN	+5	-	Lower	0
Uncontrolled RCCA bank withdrawal at power	LOFTRAN	+5	0.43	Lower and Upper	3423
RCCA misoperation	THINC, TURTLE LOFTRAN	-	-	Upper	3423
Uncontrolled boron dilution					0 and 3423
Partial loss of forced reactor coolant flow	LOFTRAN THINC, FACTRAN	+5	-	Upper	3423
Startup of an inactive reactor coolant loop	LOFTRAN, FACTRAN, THINC	-	0.43	Lower	2396
Loss of external electrical load and/or turbine trip	LOFTRAN	+5	0.43	Lower and Upper	3423
Loss of normal feedwater	LOFTRAN	+5	-	Upper	3577
Loss of offsite power to the plant auxiliaries (plant blackout)	LOFTRAN	+5	-	Upper	3577

TABLE 15.1-4

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients Moderator Temp.[a], pcm/°F[d]	Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION II (Cont'd)					
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	-	Lower	3423
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	-	-	-	3577

TABLE 15.1-4

Faults	Computer Codes Utilized	Assumed Reactivity Coefficients Moderator Temp.[a], pcm/°F[d],	Moderator Density[a], Δk/gm/cc	Doppler [b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION III (Cont'd)					
Inadvertent loading of a fuel assembly into an improper position	LEOPARD, TURTLE	-	-	-	3483
Complete loss of force reactor coolant flow	LOFTRAN, THINC, FACTRAN	+5	-	Upper	3423
Single RCCA withdrawal at full power	TURTLE, THINC, LEOPARD	-	-	-	3423
Underfrequency accident	LOFTRAN THINC, FACTRAN	+5	-	Upper	3423
CONDITION IV					
Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (loss-of-coolant accident)	SATAN-VI COCO BASH WREFLOOD LOCBART	Function of moderator density. See Sec. 15.4.1	-	Function of fuel temp. See Sec. 15.4.1	3579

TABLE 15.1-4

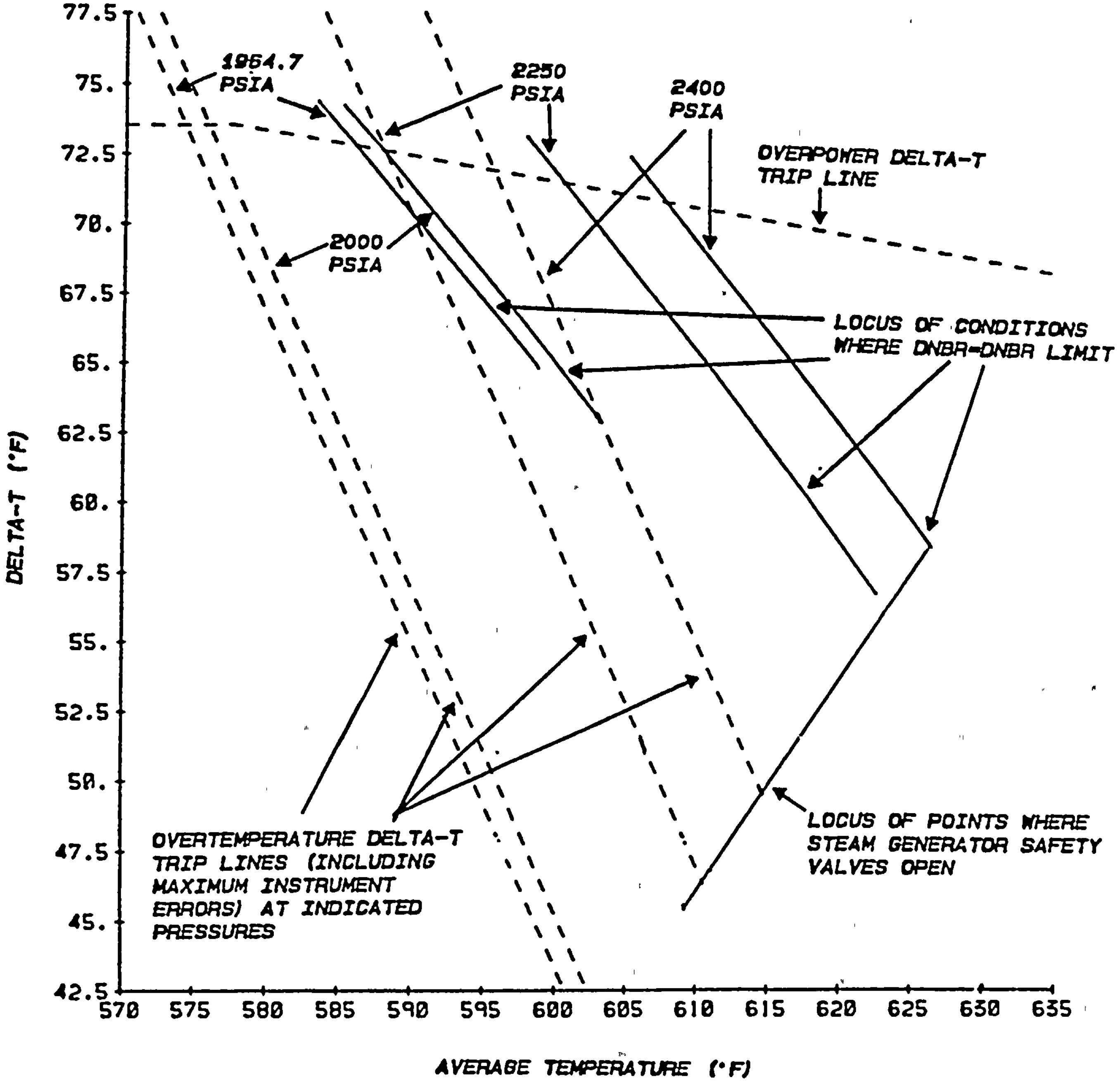
Faults	Computer Codes Utilized	Assumed Reactivity Coefficients Moderator Temp.[a], pcm/°F[d]	Moderator Density[a], Δk/gm/cc	Doppler[b]	Initial NSSS Thermal Power Output Assumed[c], MWt
CONDITION IV (Cont'd)					
Major secondary system pipe rupture up to and including double-ended rupture (rupture of a steam pipe)	LOFTRAN	-	Function of the Moderator Density see Section 15.2.13 (Figure 15.2.13-1)	See Figure 15.4.2-1	0 (Subcritical)
Waste gas decay tank rupture	-	-	-	-	3577
Steam generator tube rupture	-	-	-	-	3577
Single reactor coolant pump locked rotor	LOFTRAN THINC, FACTRAN	+5	-	Upper	3423
Fuel handling accident					3577
Rupture of a control rod mechanism housing (RCCA ejection)	TWINKLE, FACTRAN, LEOPARD	+5.2 BOL -23. EOL	-	Consistent with lower limit shown Fig. 15.1-5	0 and 3423

(a) Only one is used in analysis, i.e., either moderator temperature or moderator density coefficient.

(b) Reference Figure 15.1-5.

(c) Two percent calorimetric error considered where applicable.

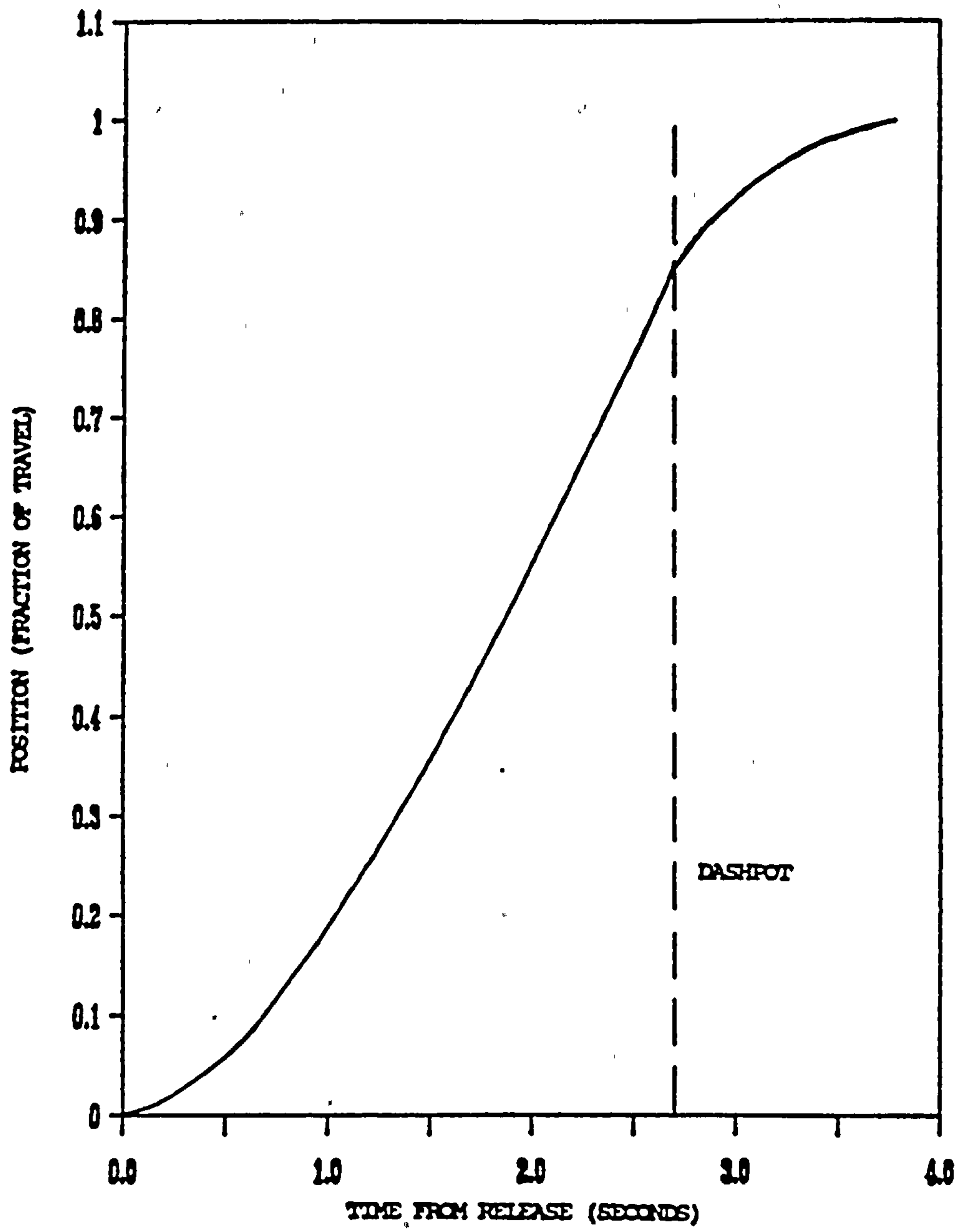
(d) Pcm means percent mille. See footnote Table 4.3-1.



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-1

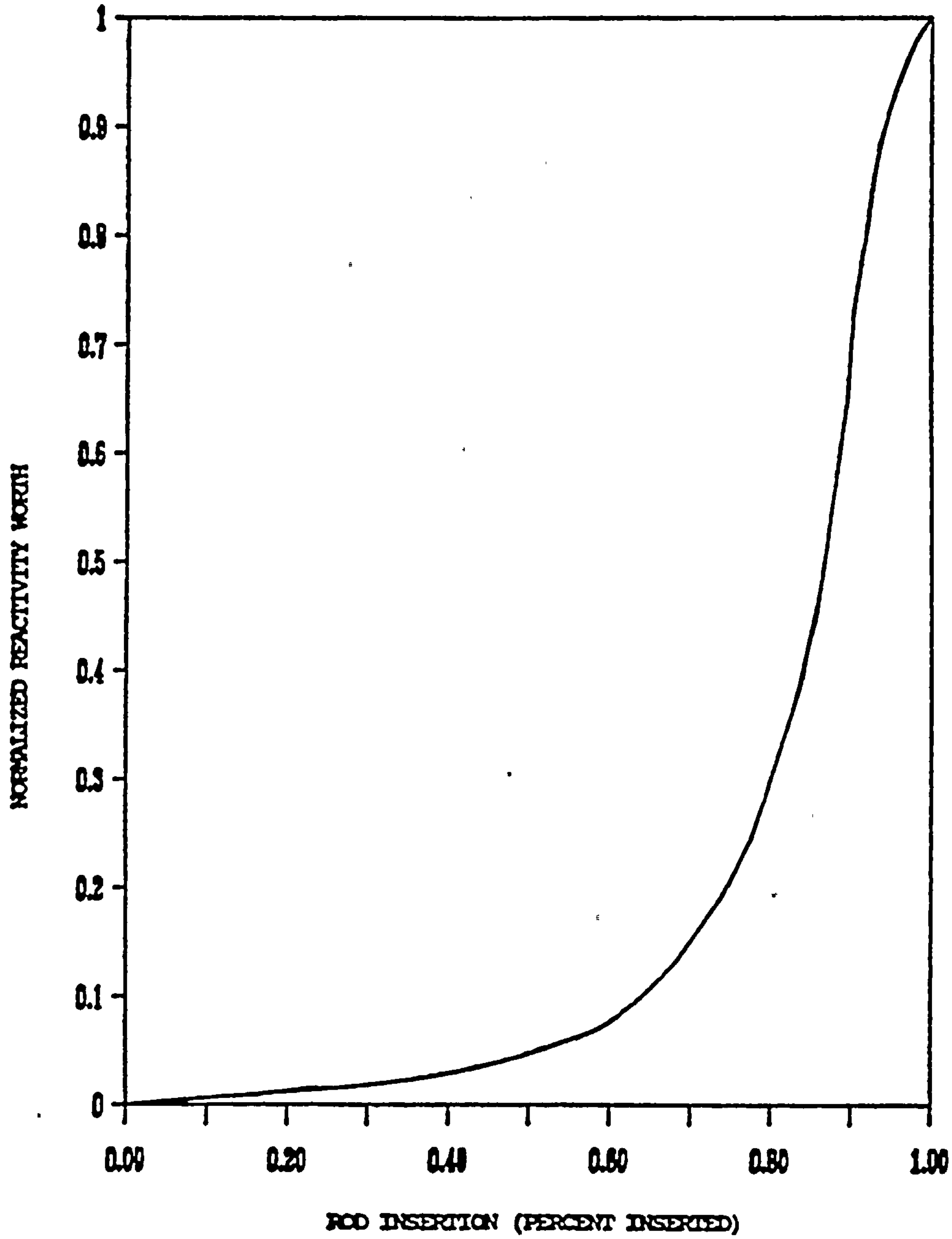
OVERTEMPERATURE AND OVERPOWER DELTA-T PROTECTION



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-2

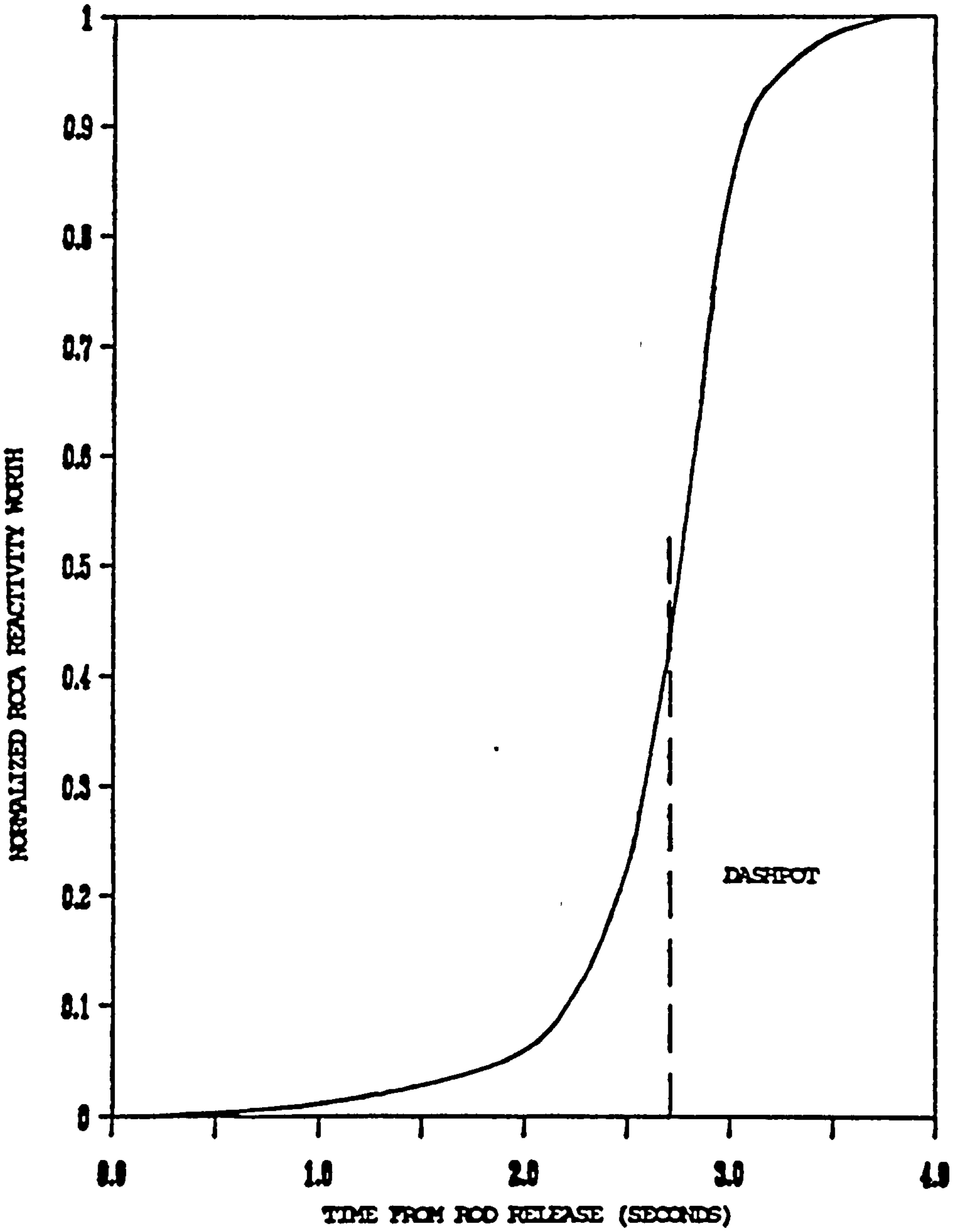
ROD POSITION VERSUS TIME ON REACTOR TRIP



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-3

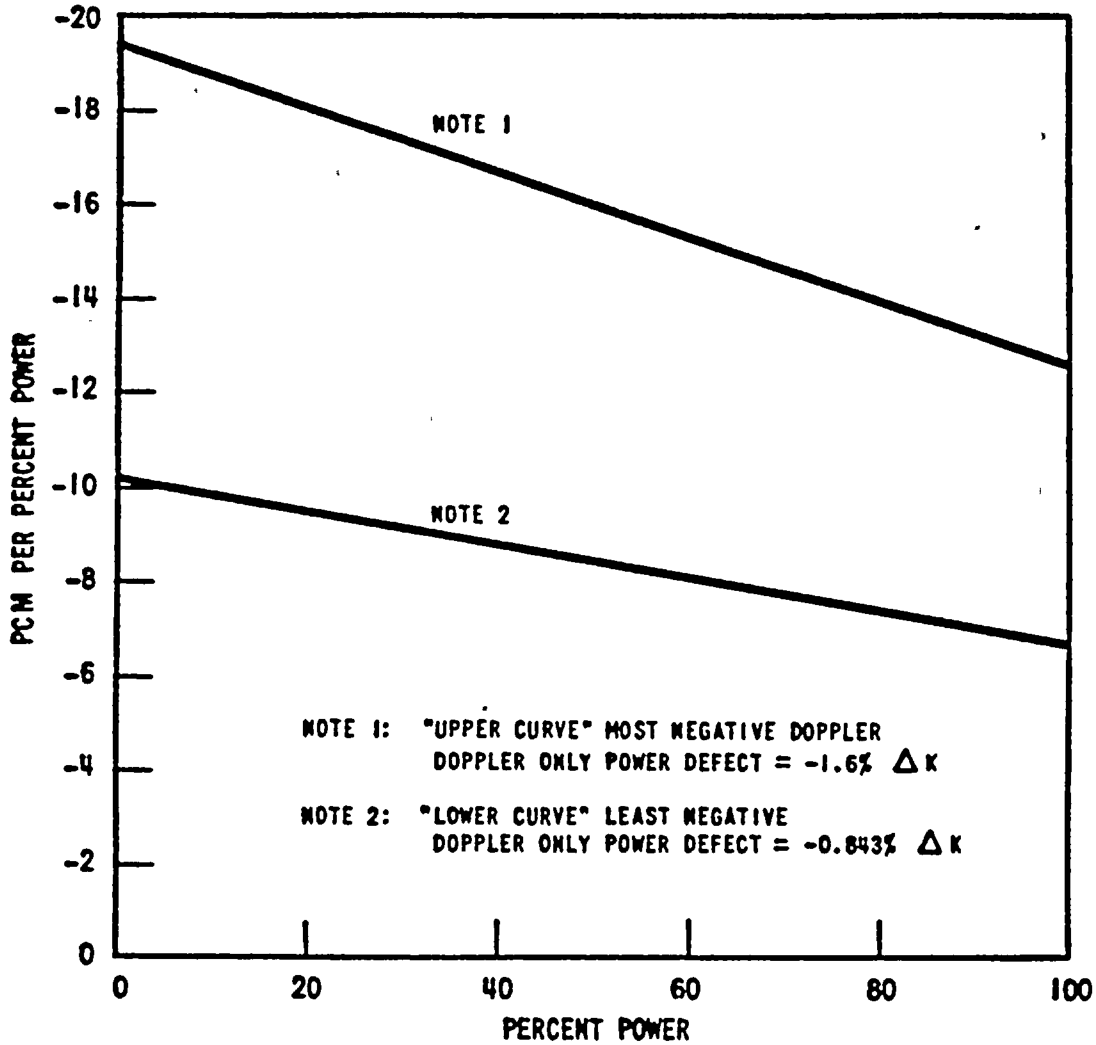
NORMALIZED RCCA REACTIVITY WORTH VERSUS PERCENT INSERTION



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-4

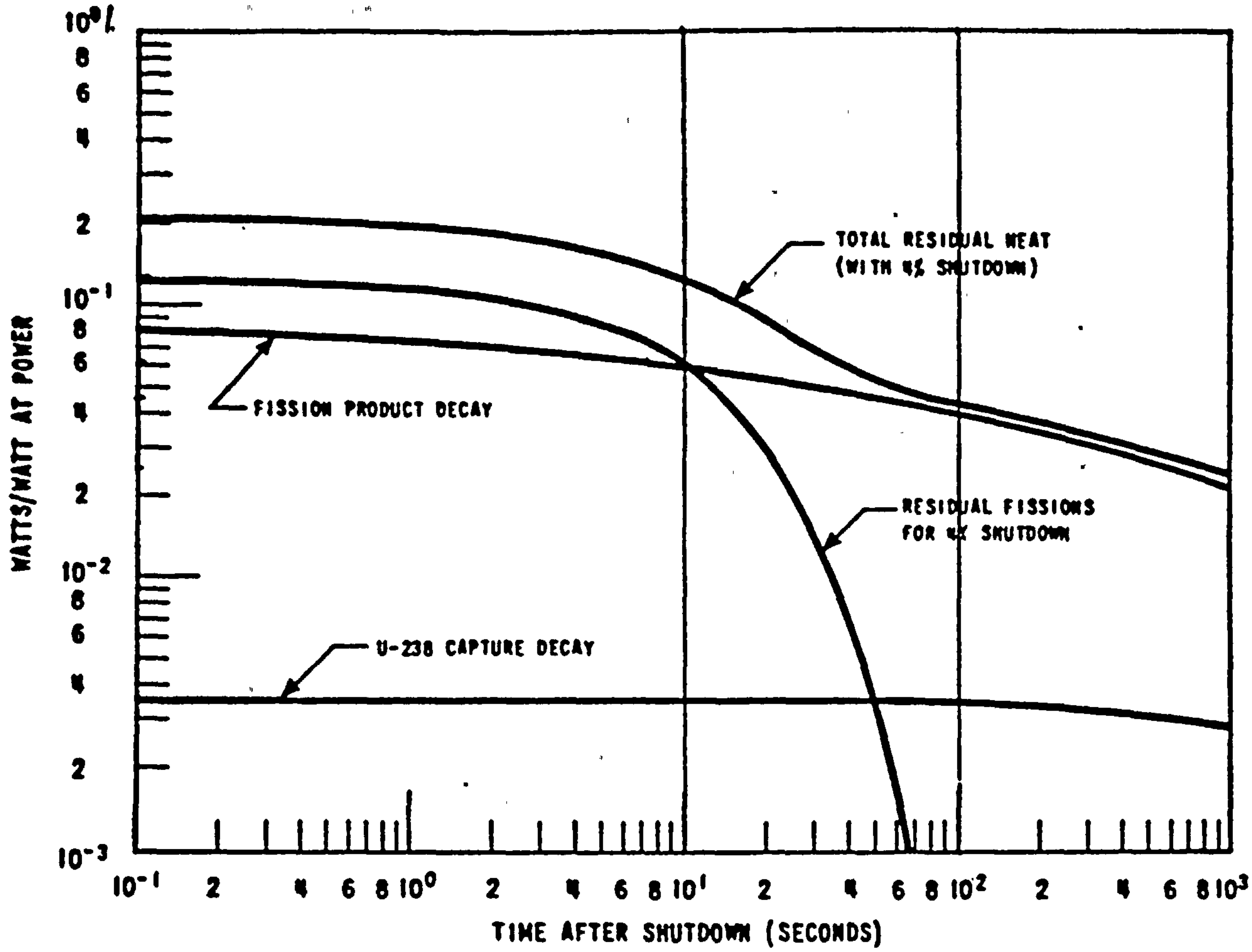
NORMALIZED RCCA BANK REACTIVITY WORTH VERSUS TIME AFTER TRIP



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-5

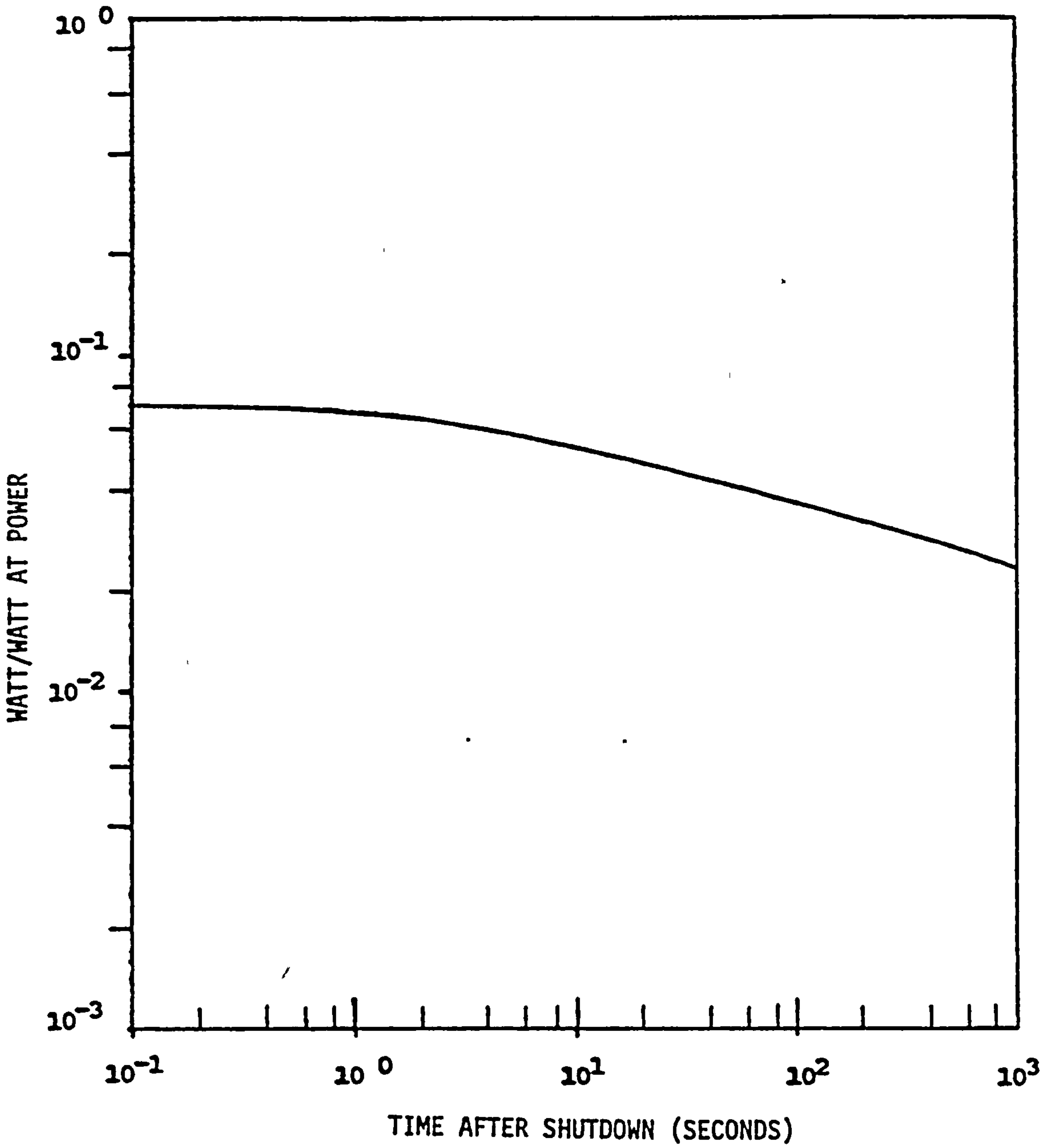
DOPPLER POWER COEFFICIENT
USED IN ACCIDENT ANALYSIS



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-6

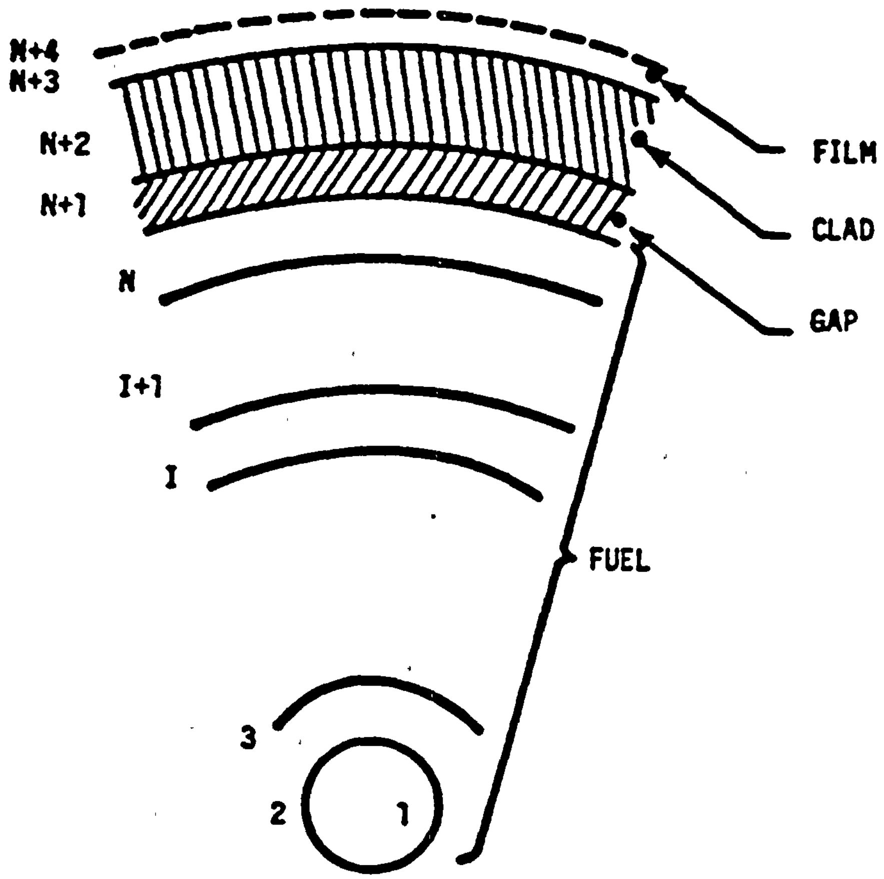
RESIDUAL DECAY
HEAT



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-7

1979 ANS DECAY
HEAT CURVE



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.1-8

FUEL ROD CROSS SECTION

15.2 CONDITION II - FAULTS OF MODERATE FREQUENCY

These faults result at worst in reactor shutdown with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., a Condition III or IV fault. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system (RCS) overpressurization. For the purposes of this report the following faults have been grouped into these categories:

(1) Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical condition

(2) Uncontrolled RCCA bank withdrawal at power

(3) RCCA misoperation

(4) Uncontrolled boron dilution

(5) Partial loss of forced reactor coolant flow

(6) Startup of an inactive reactor coolant loop

(7) Loss of external electrical load and/or turbine trip

(8) Loss of normal feedwater

(9) Loss of offsite power and main generator power to the station auxiliaries (station blackout)

(10) Excessive heat removal due to feedwater system malfunctions

(10A) Sudden feedwater temperature reduction

(11) Excessive load increase

(12) Accidental RCS depressurization

(13) Accidental main steam system depressurization

(14) Spurious operation of safety injection system (SIS) at power.

Each of these faults of moderate frequency are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

An evaluation of the reliability of the reactor protection system actuation following initiation of Condition II events has been completed and is presented in Reference 1 for the relay protection logic. Standard reliability engineering techniques were used to assess the likelihood of the trip failure due to random component failures. Common-mode failures were also qualitatively investigated. It was concluded from the evaluation that the likelihood of no trip following initiation of Condition II events is extremely small (2×10^{-7} derived for random component failures). The reliability of the solid-state protection system has also been evaluated using the same methods. The calculated reliability is of the same order of magnitude as that obtained for the relay protection logic.

Hence, because of the high reliability of the protection system, no special provision is included in the design to cope with the consequences of Condition II events without trip.

The time sequence of events during each Condition II fault is shown in Table 15.2-1.

15.2.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

15.2.1.1 Identification of Causes and Accident Description

An RCCA withdrawal accident is defined as an uncontrolled increase in reactivity in the reactor core caused by withdrawal of RCCAs resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor at either subcritical, hot zero power, or at power. The at-power case is discussed in Section 15.2.2.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 15.2.4).

The RCCA drive mechanisms are wired into preselected bank configurations that are not altered during core reactor life. These circuits prevent the assemblies from being withdrawn in other than their respective banks. Power supplied to the banks is controlled so that no more than two banks can be withdrawn at the same time. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the two control banks having the maximum combined worth at maximum speed.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power burst is of primary importance since it limits the power to a tolerable level during the delay time for protection action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

15.2.1.1.1 Source Range High Neutron Flux Reactor Trip

The source range high neutron flux reactor trip is actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

15.2.1.1.2 Intermediate Range High Neutron Flux Reactor Trip

The intermediate range high neutron flux reactor trip is actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed when two of the four power range channels give readings above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power below this value.

15.2.1.1.3 Power Range High Neutron Flux Reactor Trip (Low Setting)

The power range high neutron flux trip (low setting) is actuated when two-out-of-four power range channels indicate a power level above approximately 25% of full power. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

15.2.1.1.4 Power Range High Neutron Flux Reactor Trip (High Setting)

The power range high neutron flux reactor trip (high setting) is actuated when two-out-of-four power range channels indicate a power level above a preset setpoint. This trip function is always active. In addition, control rod stops on high intermediate range flux level (one-of-two) and high power range flux level (one-out-of-four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

15.2.1.1.5 High Neutron Flux Rate Trip

The high neutron flux rate trip is actuated when the rate of change in power exceeds the positive or negative setpoint in two-out-of-four power range channels. This function is always active.

15.2.1.2 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. The TWINKLE (Ref 2) code is used to calculate the reactivity transient and hence the nuclear power transient. The FACTRAN (Ref 3) code is then used to calculate the thermal heat flux transient based on the nuclear power transient calculated by the TWINKLE Code. FACTRAN also calculates the fuel, cladding, and coolant temperatures. A detailed thermal and hydraulic computer code, THINC (Ref 9) is used to determine if DNB occurs.

In order to give conservative results for a startup accident, the following assumptions are made concerning the initial reactor conditions:

(1) Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservative values (low absolute magnitude) as a function of power are used. See Section 15.1.5 and Table 15.1-4.

(2) Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. The conservative value, given in Table 15.1-4, is used in the analysis to yield the maximum peak heat flux.

(3) The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less

negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect thereby increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1 since this results in maximum neutron flux peaking.

(4) Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and RCCA release, is taken into account. A 10% increase is assumed for the power range flux trip setpoint, raising it from the nominal value of 25 to 35%. Previous results, however, show that the rise in neutron flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth RCCA is stuck in its fully withdrawn position. See Section 15.1.4 for RCCA insertion characteristics.

(5) The maximum positive reactivity insertion rate assumed is greater than that for the simultaneous withdrawal of the combination of the two control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 4.2.3.

(6) The initial power level was assumed to be below the power level expected for any shutdown condition. The combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

15.2.1.3 Results

Figures 15.2.1-1 through 15.2.1-3 show the transient behavior for the indicated reactivity insertion rate with the accident terminated by reactor trip at 35% nominal power. This insertion rate is greater than that for the two highest worth control banks, both assumed to be in their highest incremental worth region.

Figure 15.2.1-1 shows the neutron flux transient. The neutron flux overshoots the full power nominal value but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increase are relatively small. The thermal flux response, of interest for departure from nucleate boiling (DNB) considerations, is shown on Figure 15.2.1-2. The beneficial effect on the inherent thermal lag in the fuel is evidenced by a peak heat flux less than the full power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value and there is extensive subcooling at all times in the core. The minimum DNBR at all times remains above the limiting value.

Figure 15.2.1-3 shows the response of the average fuel, cladding, and coolant temperatures. The average fuel temperature increases to a value lower than the nominal full power value.

15.2.1.4 Conclusions

In the event of an RCCA withdrawal accident from the subcritical condition, the core and the RCS are not adversely affected since the combination of thermal power and the coolant temperature result in a departure from nucleate boiling ratio (DNBR) well above the limiting value.

15.2.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

15.2.2.1 Identification of Causes and Accident Description

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, in order to avert damage to the cladding, the reactor protection system is designed to terminate any such transient before the DNBR falls below the safety analysis limit values.

The automatic features of the reactor protection system that prevent core damage following the postulated accident include the following:

(1) The power range neutron flux instrumentation actuates a reactor trip if two-out-of-four channels exceed a high flux setpoint.

(2) The reactor trip is actuated if any two-out-of-four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature, and pressure to protect against DNB.

(3) The reactor trip is actuated if any two-out-of-four ΔT channels exceed an overpower ΔT setpoint.

(4) A high pressurizer pressure reactor trip actuated from any two-out-of-four pressure channels that are set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

(5) A high pressurizer water level reactor trip actuated from any two-out-of-three level channels that are set at a fixed point.

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

(1) High neutron flux (one-out-of-four)

(2) Overpower ΔT (two-out-of-four)

(3) Overtemperature ΔT (two-out-of-four).

The manner in which the combination of overpower and overtemperature ΔT trips provide protection over the full range of RCS conditions is described in Chapter 7. This includes a plot (also shown as Figure 15.1-1) presenting allowable reactor coolant loop average temperature and ΔT for the design power distribution and flow as a function of primary coolant pressure. The boundaries of operation defined by the overpower ΔT trip and the overtemperature ΔT 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 a 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 a 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 value. All points below and to the left of a DNB line for a given pressure have a DNBR greater than the limit. 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-pressure (fixed setpoint); low-pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints).

15.2.2.2 Analysis of Effects and Consequences

This transient is analyzed by the LOFTRAN (Ref 4) code. This 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. The core limits as illustrated on Figure 15.1-1 are used as input to LOFTRAN to determine the minimum DNBR during the transient.

This accident is analyzed with the Improved Thermal Design Procedure as described in Reference 5. In order to obtain conservative results, the following assumptions are made:

(1) Initial conditions of nominal core power and reactor coolant average temperatures (including 2.5°F for SG fouling) and nominal reactor coolant pressure are assumed. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5.

(2) Reactivity Coefficients - two cases are analyzed:

(a) Minimum reactivity feedback. A positive moderator coefficient of reactivity of +5 pcm/°F is assumed. A variable Doppler power coefficient with core power is used in the analysis. A conservatively small (in absolute magnitude) value is assumed.

(b) Maximum reactivity feedback. A conservatively large positive moderator density coefficient and a large (in absolute magnitude) negative Doppler power coefficient are assumed.

(3) The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118% of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors, while the delays for the trip signal actuation are assumed at their maximum values.

(4) The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

(5) The maximum positive reactivity insertion rate is greater than that which would be obtained from the simultaneous withdrawal of the two control rod banks having the maximum combined worth at maximum speed.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in overtemperature ΔT trip setpoint proportional to a decrease in margin to DNB.

15.2.2.3 Results

Figures 15.2.2-1 and 15.2.2-2 show the response of neutron flux, pressure, average coolant temperature, and DNBR to a rapid RCCA withdrawal starting from full power. Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, small changes in T_{avg} and pressure result and a larger margin to DNB is maintained.

The response of neutron flux, pressure, average coolant temperature, and DNBR for a slow control rod assembly withdrawal from full power is shown on Figures 15.2.2-3 and 15.2.2-4. Reactor trip on overtemperature ΔT occurs after a longer period and the rise in temperature and pressure is consequently larger than for rapid RCCA withdrawal. Again, the minimum DNBR is never less than the safety analysis limit values.

Figure 15.2.2-5 shows the minimum DNBR as a function of reactivity insertion rate from initial full power operation for the minimum and for the maximum reactivity feedbacks. It can be seen that two reactor trip channels provide protection over the whole range of reactivity insertion rates. These are the high neutron flux and overtemperature ΔT trip channels. The minimum DNBR is never less than the safety analysis limit values.

Figures 15.2.2-6 and 15.2.2-7 show the minimum DNBR as a function of reactivity insertion rate for RCCA withdrawal incidents starting at 60 and 10% power, respectively. The results are similar to the 100% power case, except that as the initial power is decreased, the range over which the overtemperature ΔT trip is effective is increased. In neither case does the DNBR fall below the safety analysis limit values.

The shape of the curves of minimum DNB ratio versus reactivity insertion rate in the reference figures is due both to reactor core and coolant system transient response and to protection system action in initiating a reactor trip.

Referring to Figure 15.2.2-7, for example, it is noted that:

1. For reactivity insertion rates ~ above 30 pcm/sec reactor trip is initiated by the high neutron flux trip for the minimum reactivity feedback cases. The neutron flux level in the core rises rapidly for these insertion rates while core heat flux and coolant system temperature lag behind due to the thermal capacity of the fuel and coolant system fluid. Thus, the reactor is tripped prior to significant increase in heat flux or water temperature with resultant high minimum DNB ratios during the transient. As reactivity insertion rate decreases, core heat flux and coolant temperatures can remain more nearly in equilibrium with the neutron flux. Minimum DNBR during the transient thus decreases with decreasing insertion rate.

2. The Overtemperature ΔT reactor trip circuit initiates a reactor trip when measured coolant loop ΔT exceeds a setpoint based on measured Reactor Coolant System average temperature and pressure. It is important to note that the average temperature contribution to the circuit is lead-lag compensated in order to decrease the effect of the thermal capacity of the Reactor Coolant System in response to power increases.

3. For reactivity insertion rate below ~ 30 pcm/sec the Overtemperature ΔT trip terminates the transient.

For reactivity insertion rates between ~ 30 pcm/sec and ~ 7 pcm/sec the effectiveness of the Overtemperature ΔT trip increases (in terms of increased minimum DNBR) due to the fact that with lower insertion rates the power increase rate is slower, the rate of rise of average coolant temperature is slower and the system lags and delays become less significant.

4. For reactivity insertion rates less than ~ 7 pcm/sec, the rise in the reactor coolant temperature is sufficiently high so that the steam generator safety valve setpoint is reached prior to trip. Opening of these valves, which act as an additional heat load on the Reactor Coolant System, sharply decreases the rate of increase of Reactor Coolant System average temperature. This decrease in rate of increase of the average coolant system temperature during the transient is accentuated by the lead-lag compensation causing the Overtemperature ΔT trip setpoint to be reached later with a resulting lower minimum DNBR.

For transients initiated from higher power levels (for example, see Figure 15.2.2-5) the effect described in item 4 above, which results in the sharp peak in minimum DNBR at approximately 7 pcm/sec, does not occur since the steam generator safety valves are not actuated prior to trip.

Figures 15.2.2-5, 15.2.2-6, and 15.2.2-7 illustrate minimum DNBRs calculated for minimum and maximum reactivity feedback.

Since the RCCA withdrawal at power incident is an overpower transient, the fuel temperatures rise during the transient until after reactor trip occurs. For high reactivity insertion rates, the overpower transient is fast with respect to the fuel rod thermal time constant, and the core heat flux lags behind the neutron flux response. Due to this lag, the peak core heat flux does not exceed 118 percent of its nominal value (i.e., the high neutron flux trip setpoint assumed in the analysis). Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will still remain below the fuel melting temperature.

For slow reactivity insertion rates, the core heat flux remains more nearly in equilibrium with the neutron flux. The overpower transient is terminated by the Overtemperature ΔT reactor trip before a DNB condition is reached. The peak heat flux again is maintained below 118 percent of its nominal value. Taking into account the effect of the RCCA withdrawal on the axial core power distribution, the peak fuel centerline temperature will remain below the fuel melting temperature.

Since DNB does not occur at any time during the RCCA withdrawal at power transient, 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.

The calculated sequence of events for this accident is shown on Table 15.2-1. With the reactor tripped, the plant eventually returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

15.2.2.4 Conclusions

The high neutron flux and overtemperature ΔT trip channels provide adequate protection over the entire range of possible reactivity insertion rates; i.e., the minimum value of DNBR is always larger than the safety analysis limit values.

15.2.3 Rod Cluster Control Assembly Misoperation

This section discusses RCCA misoperation that can result either from system malfunction or operator error.

15.2.3.1 Identification of Causes and Accident Description

RCCA misalignment accidents include:

(1) One or more dropped RCCAs within the same group

(2) A dropped RCCA bank

(3) Statically misaligned RCCA.

Each RCCA has a position indicator channel that displays the position of the assembly. The displays of assembly positions are grouped for the operator's convenience. Fully inserted assemblies are further indicated by a rod at bottom signal, which actuates a local alarm and a control room annunciator. Group demand position is also indicated.

RCCAs are always moved in preselected banks, and the banks are always moved in the same preselected sequence. Each bank of RCCAs is divided into two groups. The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation (or deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism) is required to withdraw the RCCA attached to the mechanism. Since the stationary gripper, movable gripper, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure that would cause rod withdrawal would affect a minimum of one group. Mechanical failures are in the direction of insertion, or immobility.

A dropped RCCA, or RCCA bank, is detected by:

(1) A sudden drop in the core power level as seen by the nuclear instrumentation system

(2) Asymmetric power distribution as seen on out-of-core neutron detectors or core-exit thermocouples

(3) Rod at bottom signal

(4) Rod deviation alarm

(5) Rod position indication

(6) Negative neutron flux rate trip circuitry.

Misaligned RCCAs are detected by:

(1) Asymmetric power distribution as seen on out-of-core neutron detectors or core-exit thermocouples

(2) Rod deviation alarm

(3) Rod position indicators.

The deviation alarm alerts the operator whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. If the rod deviation alarm is not operable, the operator is required to take action as required by the Technical Specifications (Ref 6).

If one or more rod position indicator channels should be out of service, detailed operating instructions are followed to ensure the alignment of the nonindicated RCCAs. The operator is also required to take action as required by the Technical Specifications.

15.2.3.2 Analysis of Effects and Consequences

Method of Analysis

(1) One or More Dropped RCCAs from the Same Group

For evaluation of the dropped RCCA event, the transient system response is calculated using 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.

Statepoints are calculated and nuclear models are used to obtain a hot channel factor consistent with the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met using the THINC (Ref 7) code. The transient response, nuclear peaking factor analysis, and DNB design basis confirmation are performed in accordance with the methodology described in Reference 5.

(2) Dropped RCCA Bank

Analysis is not required since the dropped RCCA bank results in a trip.

(3) Statically Misaligned RCCA

Steady state power distributions are analyzed using the computer codes as described in Table 4.1-2. The peaking factors are then used as input to the THINC code to calculate the DNBR.

15.2.3.3 Results

(1) One or More Dropped RCCAs

Single or multiple dropped RCCAs within the same group result in a negative reactivity insertion that may be detected by the power range negative neutron flux rate trip circuitry. If detected, the reactor is tripped within approximately 2.7 seconds following the drop of the RCCAs. The core is not adversely affected during this period since power is decreasing rapidly. Following reactor trip, normal shutdown procedures are followed. The operator may manually retrieve the RCCA by following approved operating procedures.

For those dropped RCCAs that do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern and establishing the automatic rod control mode of operation as the limiting case.

For a dropped RCCA event in the automatic rod control mode, the rod control system detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power. Figures 15.2.3-1 and 15.2.3-2 show a typical transient response to a dropped RCCA (or RCCAs) in automatic control. In all cases, the minimum DNBR remains above the safety analysis limit value.

(2) Dropped RCCA Bank

A dropped RCCA bank typically results in a reactivity insertion of greater than 500 pcm which will be detected by the power range negative neutron flux rate trip circuitry. The reactor is tripped within approximately 2.7 seconds following the drop of a RCCA bank. The core is not adversely affected during this period since power is decreasing rapidly. Following the reactor trip, normal shutdown procedures are followed to further cool down the plant. Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of 10 minutes following the incident.

(3) Statically Misaligned RCCA

The most severe misalignment situations with respect to DNBR at significant power levels arise from cases in which one RCCA is fully inserted, or where Bank D is fully inserted with one RCCA fully withdrawn. Multiple independent alarms, including a bank insertion limit alarm, alert the operator well before the postulated conditions are approached. The bank can be inserted to its insertion limit with any one assembly fully withdrawn without the DNBR falling below the safety analysis limit value.

The insertion limits in the Technical Specifications may vary from time to time depending on a number of limiting criteria. It is preferable, therefore, to analyze the misaligned RCCA case at full power for a position of the control bank as deeply inserted as the criteria on minimum DNBR and power peaking factor will allow. The full power insertion limits on control Bank D must then be chosen to be above that position and will usually be dictated by other criteria. Detailed results will vary from cycle to cycle depending on fuel arrangements.

For this RCCA misalignment, with Bank D inserted to its full power insertion limit and one RCCA fully withdrawn, DNBR does not fall below the safety analysis limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values but with the increased radial peaking factor associated with the misaligned RCCA.

DNB calculations have not been performed specifically for RCCAs missing from other banks; however, power shape calculations have been done as required for the fully withdrawn analysis. Inspection of the power shapes shows that the DNB and peak kW/ft situation is less severe than the Bank D case discussed above assuming insertion limits on the other banks equivalent to a Bank D full-in insertion limit.

For RCCA misalignments with one RCCA fully inserted, the DNBR does not fall below the limit value. This case is analyzed assuming the initial reactor power, pressure, and RCS temperatures are at their nominal values, but with the increased radial peaking factor associated with the misaligned RCCA.

DNB does not occur for the RCCA misalignment incident and thus the ability of the primary coolant to remove heat from the fuel rod is not reduced. The peak fuel temperature corresponds to a linear heat generation rate based on the radial peaking factor penalty associated with the misaligned RCCA and the design axial power distribution. The resulting linear heat generation is well below that which would cause fuel melting.

Following the identification of an RCCA group misalignment condition by the operator, the operator is required to take action as required by the plant Technical Specifications and operating instructions.

15.2.3.4 Conclusions

For all cases of dropped RCCAs or dropped banks, for which the reactor is tripped by the power range negative neutron flux rate trip, there is no reduction in the margin to core thermal limits and, consequently, the DNB design basis is met. It is shown for all cases which do not result in reactor trip that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

For all cases of any RCCA inserted, or Bank D inserted to its rod insertion limits with any single RCCA in that bank fully withdrawn (static misalignment), the DNBR remains greater than the safety analysis limit value.

15.2.4 Uncontrolled Boron Dilution

15.2.4.1 Identification of Causes and Accident Description

Reactivity can be added to the core by feeding unborated water into the RCS via the reactor makeup portion of the chemical and volume control system (CVCS). Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. A boric acid blend system is provided to permit the operator to match the boron concentration of reactor coolant makeup water to that in the RCS during normal makeup injection. The CVCS is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value, which after indication through alarms and instrumentation, provides the operator with sufficient time to correct the situation in a safe and orderly manner.

The opening of the primary water makeup control valves provides makeup to the RCS that can dilute the reactor coolant. Inadvertent dilution from this source can be readily terminated by closing the control valve. In order for makeup water to be added to the RCS at pressure, at least one charging pump must be running in addition to a primary makeup water pump.

The rate of addition of unborated makeup water to the RCS when it is not at pressure is limited by the capacity of the primary water supply pumps. The maximum addition rate in this case is 300 gpm with both pumps running. The 300 gpm reactor makeup water delivery rate is based on a pressure drop calculation comparing the pump curves with the system resistance curve. This is the maximum delivery based on the unit piping layout. Normally, only one charging pump is operating.

The boric acid from the boric acid tank is blended with primary grade water in the blender and the composition is determined by the present flowrates of boric acid and primary grade water on the control board. In order to dilute, two separate operations are required:

(1) The operator must switch from the automatic makeup mode to the dilute mode

(2) The start button must be depressed.

Omitting either step would prevent dilution.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of the pumps in the CVCS. Alarms are actuated to warn the operator if boric acid or demineralized water flowrates deviate from preset values as a result of system malfunction.

15.2.4.2 Analysis of Effects and Consequences

15.2.4.2.1 Method of Analysis

To cover all phases of the plant operation, boron dilution during refueling, startup, and power operation is considered in this analysis. Table 15.2-1 contains the time sequence of events for this accident.

15.2.4.2.2 Dilution During Refueling

During refueling the following conditions exist:

(1) One residual heat removal (RHR) pump is operating to ensure continuous mixing in the reactor vessel.

(2) The seal injection water supply to the reactor coolant pumps is isolated.

(3) The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution.

(4) The boron concentration in the refueling water is approximately ~~2300~~ 2000 ppm, corresponding to a shutdown margin of at least 5% Δk with all RCCAs in; periodic sampling ensures that this concentration is maintained.

(5) Neutron sources are installed in the core and the source range detectors outside the reactor vessel are active and provide an audible count rate. During initial core loading, BF_3 detectors are installed inside the reactor vessel and are connected to instrumentation giving audible count rates to provide direct monitoring of the core.

A minimum water volume in the RCS of 5717 cubic feet is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the RHR loop. A maximum dilution flow of 300 gpm, limited by the capacity of the two primary water makeup pumps, and uniform mixing are assumed.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the control room.

In addition, a high source range flux level is alarmed in the control room. The count rate increase is proportional to the subcritical multiplication factor.

15.2.4.2.3 Dilution During Startup

The RCS is filled with borated (approximately ~~2300~~ 2000 ppm) water from the refueling water storage tank (RWST) prior to startup.

Core monitoring is by external BF_3 detectors. Mixing of the reactor coolant is accomplished by operation of the reactor coolant pumps. High source range flux level and all reactor trip alarms are effective. In the analysis, a maximum dilution flow of 300 gpm limited by the capacity of the two primary water makeup pumps is considered. The volume of the reactor coolant is approximately 9153 cubic feet, which is the active volume of the RCS excluding the pressurizer.

15.2.4.2.4 Dilution at Power

With the unit at power and the RCS at pressure, the dilution rate is limited by the capacity of the charging pumps. The effective reactivity addition rate for the reactor at full power and for a boron dilution flow of 262 gpm is shown as a function of RCS boron concentration on Figure 15.2.4-1. This figure includes the effect of increasing boron worth with dilution. The reactivity rate used in the following evaluation is 1.752×10^{-5} Δk/sec based on a conservatively high value for the expected boron concentration (1600 ppm) at power.

15.2.4.3 Conclusions

For dilution during refueling and startup:

At the beginning of the core life, equilibrium cycle core, the boron concentration must be reduced from 2000 ppm to approximately 1600 ppm before the reactor will go critical. This takes 32 minutes. This is ample time for the operator to recognize a high count rate signal and isolate the reactor makeup water source by closing valves and stopping the primary water supply pumps.

During startup, the minimum time required to reduce the reactor coolant boron concentration to 1600 ppm, where the reactor would go critical with all RCCAs in, is 38 minutes. Once again, this should be more than adequate time for the operator to recognize the high count rate signal and terminate the dilution flow.

For dilution during full power operation:

(1) With the reactor in automatic control at full power, the power and temperature increase from boron dilution results in the insertion of the RCCAs and a decrease in shutdown margin. Continuation of dilution

and RCCA insertion would cause the assemblies to reach the minimum limit of the rod insertion monitor in approximately 4.7 minutes, assuming the RCCAs to be initially at a position providing the maximum operational maneuvering band consistent with maintaining a minimum control band incremental rod worth. Before reaching this point, however, two alarms would be actuated to warn the operator of the accident condition. The first of these, the low insertion limit alarm, alerts the operator to initiate normal boration.

The other, the low-low insertion limit alarm, alerts the operator to follow emergency boration procedures. The low alarm is set sufficiently above the low-low alarm to alarm normal boration without the need for emergency procedures. If dilution continues after reaching the low-low alarm, it takes approximately 15.0 minutes after the low-low alarm before the total shutdown margin (assuming 1.6%) is lost due to dilution. Therefore, adequate time is available following the alarms for the operator to determine the cause, isolate the primary grade water source, and initiate boration.

(2) With the reactor in manual control and if no operator action is taken, the power and temperature rise will cause the reactor to reach the high neutron flux trip setpoint. The boron dilution accident in this case is essentially identical to a RCCA withdrawal accident at power. The maximum reactivity insertion rate for boron dilution is shown on Figure 15.2.4-1 and is seen to be within the range of insertion rates analyzed for a RCCA withdrawal accident. There is ample time available (approximately 14.5 minutes) after a reactor trip for the operator to determine the cause of dilution, isolate the primary grade water sources, and initiate reboration before the reactor can return to criticality assuming a 1.6% shutdown margin at the beginning of dilution.

15.2.5 Partial Loss of Forced Reactor Coolant Flow

15.2.5.1 Identification of Causes and Accident Description

A partial loss of coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in the power supply to the pump. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not tripped promptly.

The necessary protection against a partial loss of coolant flow accident is provided by the low primary coolant flow reactor trip that is actuated by two-out-of-three low flow signals in any reactor coolant loop. Above approximately 35% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10% power (Permissive 7) and the power level corresponding to Permissive 8 low flow in any two loops will actuate a reactor trip. Reactor trip on low flow is blocked below Permissive 7.

A reactor trip signal from the pump breaker position is provided as a backup to the low flow signal. When operating above Permissive 7, a breaker open signal from any two pumps will actuate a reactor trip. Reactor trip on reactor coolant pump breakers open is blocked below Permissive 7.

Normal power for the pumps is supplied through buses connected through transformers to the generator. Two pump buses each supply power to two pumps. When a generator trip occurs, the pumps are automatically transferred to a bus supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip where there are no electrical faults that require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for approximately 30 seconds after the reactor trip before any transfer is made.

15.2.5.2 Analysis of Effects and Consequences

15.2.5.2.1 Method of Analysis

The following case has been analyzed:

All loops operating, two loops coasting down

This transient is analyzed by three digital computer codes. First the LOFTRAN code is used to calculate the loop and core flow during the transient. The LOFTRAN code is also used to calcuate the time of reactor trip, based on the calculated flows and the nuclear power transient following reactor trip. The FACTRAN code is then used to calcuate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC (Ref 7) code is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR transient presented represents the minimum of the typical and thimble cells for Standard and VANTAGE 5 fuel.

15.2.5.2.2 Initial Conditions

The assumed initial operating conditions are the most adverse with respect to the margin to DNB, i.e., nominal steady state power level, nominal steady state pressure, and nominal steady state coolant average temperature (with 2.5°F for steam generator fouling). See Section 15.1.2 for an explanation of initial conditions. The accident is analyzed using the Improved Thermal Design Procedure as described in Reference 5.

15.2.5.2.3 Reactivity Coefficients

A conservatively large absolute value of the Doppler-only power coefficient is used (see Table 15.1-4). The total integrated Doppler reactivity from 0 to 100% power is assumed to be -0.016 Δk.

The most positive moderator temperature coefficient (+5 pcm/°F) is assumed since this results in the maximum hot spot heat flux during the initial part of the transient when the minimum DNBR is reached.

15.2.5.2.4 Flow Coastdown

The flow coastdown analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance, and the pump characteristics and is based on high estimates of system pressure losses.

15.2.5.3 Results

The calculated sequence of events is shown in Table 15.2-1. Figures 15.2.5-1 through 15.2.5-4 show the core flow coastdown, the loop flow coastdown, the nuclear power coastdown, and the heat flux coastdown. The minimum DNBR is not less than the safety analysis limit value. A plot of DNBR vs. time is given in Figure 15.2.5-5 for the most limiting typical or thimble cell for Standard and VANTAGE 5 fuel.

15.2.5.4 Conclusions

The analysis shows that the DNBR will not decrease below the safety analysis limit values at any time during the transient. Thus, no core safety limit is violated.

15.2.6 Startup of an Inactive Reactor Coolant Loop

In accordance with Technical Specification 3/4.4.1, Diablo Canyon Power Plant (DCPP) operation during startup and power operation with less than four loops is not permitted. This analysis is presented for completeness.

15.2.6.1 Identification of Causes and Accident Description

If a plant is operating with one pump out of service, there is reverse flow through the loop due to the pressure difference across the reactor vessel. The cold leg temperature in an inactive loop is identical to the cold leg temperature of the active loops (the reactor core inlet temperature). If the reactor is operated at power, and assuming the secondary side of the steam generator in the inactive loop is not isolated, there is a temperature drop across the steam generator in the inactive loop and, with the reverse flow, the hot leg temperature of the inactive loop is lower than the reactor core inlet temperature.

Administrative procedures require that the unit be brought to a load of less than 25% of full power prior to starting a pump in an inactive loop in order to bring the inactive loop hot leg temperature closer to the core inlet temperature. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core which causes a rapid reactivity insertion and subsequent power increase.

This event is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Section 15.0.

Should the startup of an inactive reactor coolant pump at an incorrect temperature occur, the transient will be terminated automatically by a reactor trip on low coolant loop flow when the power range neutron flux (two out of four channels) exceeds the P-8 setpoint, which has been previously reset for three loop operation.

15.2.6.2 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. The LOFTRAN Code (Ref 4) is used to calculate the loop and core flow, nuclear power and core pressure and temperature transients following the startup of an idle pump. FACTRAN (Ref 3) is used to calculate the core heat flux transient based on core flow and nuclear power from LOFTRAN. The THINC Code (Ref 7) is then used to calculate the DNBR during the transient based on system conditions (pressure, temperature, and flow) calculated by LOFTRAN and heat flux as calculated by FACTRAN.

In order to obtain conservative results for the startup of an inactive pump accident, the following assumptions are made:

(1) Initial conditions of maximum core power and reactor coolant average tempertures and minimum reactor coolant presure resulting in minimum initial margin to DNB. These values are to be consistent with maximum steady state power level allowed with all but one loop in operation including appropriate allowances for calibration and instrument errors. The high initial power gives the greatest temperature difference between the core inlet temperature and the inactive loop hot leg temperature.

(2) Following the start of the idle pump, the inactive loop flow reverses and accelerates to its nominal full flow value.

(3) A conservatively large (absolute value) negative moderator coefficient associated with the end of life.

(4) A conservatively low (absolute value) negative Doppler power coefficient is used.

(5) The initial reactor coolant loop flows are at the appropriate values for one pump out of service.

(6) The reactor trip is assumed to occur on low coolant flow when the power range neutron flux exceeds the P-8 setpoint. The P-8 setpoint is conservatively assumed to be 84 percent of rated power which corresponds to the nominal setpoint plus 9 percent for nuclear instrumentation errors.

15.2.6.3 Results

The results following the startup of an idle pump with the above listed assumptions are shown in Figures 15.2.6-1 through 15.2.6-5. As shown in these curves, during the first part of the transient, the increase in core flow with cooler water results in an increase in nuclear power and a decrease in core average temperature. The minimum DNBR during the transient is considerably greater than the safety analysis limit values.

Reactivity addition for the inactive loop startup accident is due to the decrease in core water temperature. During the transient, this decrease is due both to (1) the increase in reactor coolant flow and, (2) as the inactive loop flow reverses, to the colder water entering the core from the hot leg side (colder temperature side prior to the start of the transient) of the steam generator in the inactive loop. Thus, the reactivity insertion rate for this transient changes with time. The resultant core nuclear power transient, computed with consideration of both moderator and Doppler reactivity feedback effects, is shown on Figure 15.2.6-1.

The calculated sequence of events for this accident is shown in Table 15.2-1. The transient results illustrated in Figures 15.2.6-1 through 15.2.6-5 indicate that a stabilized plant condition, with the reactor tripped, is approached rapidly. Plant cooldown may subsequently be achieved by following normal shutdown procedures.

15.2.6.4 Conclusions

The transient results show that the core is not adversely affected. There is considerable margin to the safety analysis DNBR limit values; thus, no fuel or clad damage is predicted.

15.2.7 Loss of External Electrical Load and/or Turbine Trip

15.2.7.1 Identification of Causes and Accident Description

A major load loss on the plant can result from either a loss of external electrical load or from a turbine trip. For either case, offsite power is available for the continued operation of plant components such as the reactor coolant pumps. The case of loss of all ac power (station blackout) is analyzed in Section 15.2.9.

For a turbine trip, the reactor would be tripped directly (unless it is below approximately 10% power) from a signal derived from the turbine autostop oil pressure and turbine stop valves. The automatic steam dump system accommodates the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if the steam dump system and pressurizer pressure control system are functioning properly. If the turbine condenser were not available, the excess steam generation would be dumped to the atmosphere. Additionally, main feedwater flow would be lost if the turbine condenser were not available. For this situation, steam generator level would be maintained by the auxiliary feedwater system.

For a loss of external electrical load without subsequent turbine trip, no direct reactor trip signal would be generated. Since both units have full load rejection capability, they would be expected to continue operating without a reactor trip. A continued steam load of approximately 5% would exist after total loss of external electrical load because of the electrical demand of plant auxiliaries.

In the event the steam dump valves fail to open following a large loss of load, the steam generator safety valves may lift and the reactor may be tripped by the high pressurizer pressure signal, the high pressurizer water level signal, or the overtemperature ΔT signal. The steam generator shell-side pressure and reactor coolant temperatures will increase rapidly. The pressurizer safety valves and steam generator safety valves are, however, sized to protect the RCS and steam generator against overpressure for all load losses without assuming the operation of the steam dump system, pressurizer spray, pressurizer power-operated relief valves, automatic RCCA control, or direct reactor trip on turbine trip.

The steam generator safety valve capacity is sized to remove the steam flow at the engineered safeguards design rating (105% of steam flow at rated power) from the steam generator without exceeding 110% of the steam system design pressure. The pressurizer safety valve capacity is sized based on a complete loss of heat sink with the plant initially operating at the maximum calculated turbine load along with operation of the steam generator safety valves. The pressurizer safety valves are then able to maintain the RCS pressure within 110% of the RCS design pressure without direct or immediate reactor trip action.

A more complete discussion of overpressure protection can be found in Reference 8.

15.2.7.2 Analysis of Effects and Consequences

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from full power without a direct reactor trip. This is done to show the adequacy of the pressure-relieving devices and to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. The turbine is assumed to trip without actuating all the turbine stop valve limit switches. This assumption delays reactor trip until conditions in the RCS result in a trip due to other signals. Thus, the analysis assumes a worst case transient. In addition, no credit is taken for steam dump. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater (except for long-term recovery) to mitigate the consequences of the transient.

The total loss of load transients are analyzed with the LOFTRAN computer program (see Section 15.1). The program simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

Major assumptions are summarized below:

(1) Initial Operating Conditions

The initial reactor power, RCS pressure, and RCS temperatures are assumed at their nominal values consistent with steady state full power operation.

(2) Moderator and Doppler Coefficients of Reactivity

The turbine trip is analyzed with both maximum and minimum reactivity feedback. The maximum feedback (EOL) cases assume a large negative moderator temperature coefficient and the most negative Doppler power coefficient. The minimum feedback (BOL) cases assume a minimum moderator temperature coefficient and the least negative Doppler coefficient.

(3) Reactor Control

From the standpoint of the maximum pressures attained, it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

(4) Steam Release

No credit is taken for the operation of the steam dump system or steam generator power-operated relief valves. The steam generator pressure rises to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value.

(5) Pressurizer Spray and Power-operated Relief Valves

Two cases for both the BOL and EOL are analyzed:

(a) Full credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are also available.

(b) No credit is taken for the effect of pressurizer spray and power-operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

(6) Feedwater Flow

Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization.

Reactor trip is actuated by the first reactor protection system trip setpoint reached with no credit taken for the direct reactor trip on the turbine trip.

15.2.7.3 Results

The transient responses for a total loss of load from full power operation are shown for four cases; two cases for the BOL and two cases for the EOL on Figures 15.2.7-1 through 15.2.7-8.

Figures 15.2.7-1 and 15.2.7-2 show the transient reponses for the total loss of steam load at BOL assuming full credit for the pressurizer spray and pressurizer power-operated relief valves. No credit is taken for the steam dump. The reactor is tripped by the overtemperature ΔT trip channel. The minimum DNBR is well above the limit value. The pressurizer safety valves are actuated for this case and maintain system pressure below 110 percent of the design value. The steam generator safety valves open and limit the secondary steam pressure increase.

Figures 15.2.7-3 and 15.2.7-4 show the responses for the total loss of load at EOL assuming a large (absolute value) negative moderator temperature coefficient. All other plant parameters are the same as in the above case. As a result of the maximum reactivity feedback at EOL, no reactor protection system trip setpoint is reached. Because main feedwater is assumed to be lost, the reactor is tripped by the low-low steam generator water level trip channel. The DNBR increases throughout the transient and never drops below its initial value. The pressurizer safety valves are not actuated in these transients.

Total loss of load was also studied assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray, pressurizer power-operated relief valves, or steam dump. The reactor is tripped on the high pressurizer pressure signal. Figures 15.2.7-5 and 15.2.7-6 show the BOL transients. The neutron flux remains constant at full power until the reactor is tripped. The DNBR generally increases throughout the transient. In this case the pressurizer safety valves are actuated and maintain the system pressure below 110 percent of the design value.

Figures 15.2.7-7 and 15.2.7-8 show the transients at the EOL with the other assumptions being the same as on Figures 15.2.7-5 and 15.2.7-6. Again, the DNBR increases throughout the transient and the pressurizer safety valves are actuated to limit the primary pressure.

Reference 8 presents additional results for a complete loss of heat sink including loss of main feedwater. This report shows the overpressure protection that is afforded by the pressurizer and steam generator safety valves.

15.2.7.4 Conclusions

Results of the analyses, including those in Reference 8, show that the plant design is such that a total loss of external electrical load without a direct or immediate reactor trip presents no hazard to the integrity of the RCS or the main steam system. Pressure-relieving devices incorporated in the two systems are adequate to limit the maximum pressures to within the design limits.

The integrity of the core is maintained by operation of the reactor protection system; i.e., the DNBR will be maintained above the safety analysis limit values. Thus, no core safety limit will be violated.

15.2.8 Loss of Normal Feedwater

15.2.8.1 Identification of Causes and Accident Description

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of offsite ac power) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core. If the reactor were not tripped during this accident, core damage would possibly occur from a sudden loss of heat sink. If an alternative supply of feedwater were not supplied to the plant, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer would occur. Significant loss of water from the RCS could conceivably lead to core damage. Since the plant is tripped well before the steam generator heat transfer capability is reduced, the primary system variables never approach a DNB condition.

The following provide the necessary protection against a loss of normal feedwater:

(1) Reactor trip on low-low water level in any steam generator

(2) Reactor trip on steam flow-feedwater flow mismatch in coincidence with low steam generator water level

(3) Two motor-driven auxiliary feedwater (AFW) pumps that are started on:

(a) Low-low level in any steam generator

(b) Trip of both main feedwater pumps

(c) Any safety injection signal

(d) Loss of offsite power (automatic transfer to diesel generators)

(e) Manual actuation.

(4) One turbine-driven auxiliary feedwater pump that is started on:

(a) Low-low level in any two steam generators

(b) Undervoltage on both reactor coolant pump buses

(c) Manual actuation.

The motor-driven AFW pumps are connected to vital buses and are supplied by the diesels if a loss of offsite power occurs. The turbine-driven pump utilizes steam from the secondary system and exhausts it to the atmosphere. The controls are designed to start both types of pumps within 1 minute even if a loss of all ac power occurs simultaneously with loss of normal feedwater. The AFW pumps take suction from the condensate storage tank for delivery to the steam generators. Instrumentation is provided in the motor-driven pump discharge to sense low pump discharge pressure indicative of a depressurized steam generator. If low pump discharge pressure should occur, control valves automatically throttle down to prevent pump runout. This automatic action ensures that the required flow is maintained. However, no such instrumentation is provided for the turbine-driven pump and remote-manual action by the plant operator is required to terminate its flow to a depressurized steam generator.

The analysis shows that following a loss of normal feedwater, the AFW system is capable of removing the stored and residual heat thus preventing either overpressurization of the RCS or loss of water from the reactor core.

15.2.8.2 Analysis of Effects and Consequences

A detailed analysis using the LOFTRAN (Ref 4) code is performed in order to determine the plant transient following a loss of normal feedwater. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions are:

(1) Reactor trip occurs on steam generator low-low level at 0.0% of narrow range span.

(2) The plant is initially operating at 102% of the engineered safeguards design rating.

(3) Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 decay heat ANSI 5.1 + 2 SIGMA was used for calculation of residual decay heat levels.

(4) The auxiliary feedwater system is actuated by the low-low steam generator water level signal.

(5) The worst single failure in the auxiliary feedwater system occurs (turbine-driven pump) and one motor-driven pump is assumed to be unavailable. The auxiliary feedwater system is assumed to supply a total of 440 gpm to two steam generators from the available motor-driven pump.

(6) The pressurizer sprays and PORVs are assumed operable. This maximizes the peak transient pressurizer water volume.

(7) Secondary system steam relief is achieved through the self-actuated safety valves. Note that steam relief will, in fact, be through the power-operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, for the sake of analysis these have been assumed unavailable.

(8) The initial reactor coolant average temperature is 6.7°F higher than the nominal value which is comprised of a 4.7°F uncertainty on nominal temperature and 2.0°F to account for steam generator tube fouling. The initial pressurizer pressure uncertainty is 38 psi.

15.2.8.3 Results

Figures 15.2.8-1 to 15.2.8-3 show plant parameters following a loss of normal feedwater. Figure 15.2.8-2 shows the pressurizer pressure as a function of time.

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow through the safety valves continues to dissipate the stored and generated heat. One minute following the initiation of the low-low level trip, the motor-driven AFW pump is automatically started, reducing the rate of water level decrease.

The capacity of the motor-driven AFW pump is such that the water level in the steam generator being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the RCS relief or safety valves.

From Figure 15.2.8-1 it can be seen that at no time is there water relief from the pressurizer. If the auxiliary feed delivered is greater than that of one motor-driven pump, the initial reactor power is less than 102% of the engineered safeguards design rating, or the steam generator water level in one or more steam generators is above the low-low level trip point at the time of trip, then the results for this transient will be less limiting.

The calculated sequence of events for this accident is listed in Table 15.2-1. As shown in Figures 15.2.8-1 through 15.2.8-3, the plant approaches a stabilized condition following reactor trip and auxiliary feedwater initiation. Plant procedures may be followed to further cool down the plant.

15.2.8.4 Conclusions

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the RCS, or the steam system since the AFW capacity is such that the reactor coolant water is not relieved from the pressurizer relief or safety valves.

15.2.9 Loss of OffSite Power to the Station Auxiliaries (Station Blackout)

15.2.9.1 Identification of Causes and Accident Description

During a complete loss of offsite power and a turbine trip there will be loss of power to the plant auxiliaries, i.e., the reactor coolant pumps, condensate pumps, etc.

The events following a loss of ac power with turbine and reactor trip are described in the sequence listed below:

(1) Plant vital instruments are supplied by emergency power sources.

(2) As the steam system pressure rises following the trip, the steam system power-operated relief valves are automatically opened to the atmosphere. Steam dump to the condenser is assumed not to be available. If the power-operated relief valves are not available, the steam generator self-actuated safety valves may lift to dissipate the sensible heat of the fuel and coolant plus the residual heat produced in the reactor.

(3) As the no-load temperature is approached, the steam system power-operated relief valves (or the self-actuated safety valves, if the power-operated relief valves are not available) are used to dissipate the residual heat and to maintain the plant at the hot standby condition.

(4) The emergency diesel generators started on loss of voltage on the plant emergency buses begin to supply plant vital loads.

The AFW system is started automatically as discussed in the loss of normal feedwater analysis. The steam-driven auxiliary feedwater pump (880 gpm delivered) utilizes steam from the secondary system and exhausts to the atmosphere. The motor-driven AFW pumps (440 gpm delivered each) are supplied by power from the diesel generators. The pumps take suction directly from the condensate storage tank for delivery to the steam generators.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops.

15.2.9.2 Analysis of Effects and Consequences

A detailed analysis using the LOFTRAN (Ref 4) code is performed in order to determine the plant transient following a station blackout. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions differing from those in a loss of normal feedwater are:

(1) No credit is taken for immediate response of control rod drive mechanisms caused by a loss of offsite power.

(2) A heat transfer coefficient in the steam generator associated with RCS natural circulation is assumed following the reactor coolant pump coastdown.

(3) The initial reactor coolant temperature evaluated is 4.7°F lower than the nominal value since this results in a greater expansion of RCS water during the transient and, thus, in a higher water level in the pressurizer.

The time sequence of events for the accident is given in Table 15.2-1. The first few seconds after the loss of power to the reactor coolant pumps will closely resemble a simulation of the complete loss of flow incident (see Section 15.3.4); i.e., core damage due to rapidly increasing core temperatures is prevented by promptly tripping the reactor. After the reactor trip, stored and residual heat must be removed to prevent damage to either the RCS or the core. The LOFTRAN code results show that the natural circulation flow available is sufficient to provide adequate core decay heat removal following reactor trip and RCP coastdown.

15.2.9.3 Conclusions

Results of the analysis show that, for the loss of offsite power to the station auxiliaries event, all safety criteria are met. Since the DNBR remains above the safety analysis limit, the core is not adversely affected. AFW capacity is sufficient to prevent water relief through the pressurizer relief and safety valves; this assures that the RCS is not overpressurized.

Analysis of the natural circulation capability of the RCS demonstrates that sufficient long-term heat removal capability exists following reactor coolant pump coastdown to prevent fuel or clad damage.

15.2.10 Excessive Heat Removal Due to Feedwater System Malfunctions

15.2.10.1 Identification of Causes and Accident Description

Reductions in feedwater temperature or excessive feedwater additions are means of increasing core power above full power. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower-overtemperature protection (neutron high flux, overtemperature ΔT, and overpower ΔT trips) prevent any power increase that could lead to a DNBR that is less than the DNBR limit.

One example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the steam generator. With the plant at no-load conditions the addition of cold feedwater may cause a decrease in RCS temperature and thus a reactivity insertion due to the effects of the negative moderator coefficient of reactivity. Continuous excessive feedwater addition is prevented by the steam generator high-high level trip.

15.2.10.2 Analysis of Effects and Consequences

The excessive heat removal due to a feedwater system malfunction transient is analyzed with the LOFTRAN code. This code simulates a multiloop system, neutron kinetics, the 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.

The system is analyzed to evaluate plant behavior in the event of a feedwater system malfunction.

Excessive feedwater addition due to a control system malfunction or operator error that allows a feedwater control valve to open fully is considered. Two cases are analyzed as follows:

(1) Accidental opening of one feedwater control valve with the reactor just critical at zero load conditions assuming a conservatively large moderator density coefficient characteristic of EOL conditions

(2) Accidental opening of one feedwater control valve with the reactor in automatic control at full power.

The reactivity insertion rate following a feedwater system malfunction is calculated with the following assumptions:

(1) For the feedwater control valve accident at full power, one feedwater control valve is assumed to malfunction resulting in a step increase of 250% of nominal feedwater flow to one steam generator.

(2) For the feedwater control valve accident at zero load condition, a feedwater valve malfunction occurs that results in a step increase in flow to one steam generator from zero to the nominal full load value for one steam generator.

(3) For the zero load condition, feedwater temperature is at a conservatively low value of 32°F.

(4) The initial water level in all the steam generators is at a conservatively low level for zero load conditions.

(5) No credit is taken for the heat capacity of the RCS and steam generator thick metal in attenuating the resulting plant cooldown.

(6) No credit is taken for the heat capacity of the steam and water in the unaffected steam generators.

(7) The feedwater flow resulting from a fully open control valve is terminated by the steam generator high-high level signal that closes all feedwater control valves, closes all feedwater bypass valves, trips the main feedwater pumps, and shuts the motor-operated feedwater isolation valves.

15.2.10.3 Results

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power and the above mentioned assumptions, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed in Section 15.2.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition, and its analysis is, therefore, covered by that of the latter. It should be noted that if the incident occurs with the unit just critical at no-load, the reactor may be tripped by the power range high neutron flux trip (low setting) set at approximately 25%.

The full power case (EOL, with control) gives the largest reactivity feedback and results in the greatest power increase. A turbine trip and reactor trip is actuated when the steam generator level reaches the high-high level setpoint.

For all cases of excessive feedwater, continuous addition of cold feedwater is prevented by closure of all feedwater control valves, closure of all feedwater bypass valves, a trip of the feedwater pumps, and closures of the feedwater isolation valves on steam generator high-high level.

Transient results (see Figures 15.2.10-1 and 15.2.10-2) show the core heat flux, pressurizer pressure, T_{avg}, and DNBR, as well as the increase in nuclear power and loop ΔT associated with the increased thermal load on the reactor. Steam generator level rises until the feedwater is terminated as a result of the high-high steam generator level trip. The DNBR does not drop below the limit safety analysis DNBR.

15.2.10.4 Conclusions

The reactivity insertion rate that occurs at no-load following excessive feedwater addition is less than the maximum value considered in the analysis of the rod withdrawal from a subcritical condition. Also, the DNBRs encountered for excessive feedwater addition at power are well above the safety analysis limit DNBR values.

15.2.10A Sudden Feedwater Temperature Reduction

A concern was raised during the Unit 1 power ascension test program that an inadvertent actuation of the load transient bypass relay (LTBR) may initiate a transient that exceeds analyzed reactor operating limits. An evaluation performed (Ref 1) shows that since the expected feedwater temperature decrease due to inadvertent actuation of the LTBR is significantly less than that of the net load trip, the consequences and events of inadvertent actuation of the LTBR are bounded by the feedwater temperature decrease event. A summary of the evaluation is provided below.

15.2.10A.1 Identification of Causes and Accident Description

A reduction in feedwater temperature may be caused by an inadvertant actuation of the LTBR. This would cause the feedwater bypass valve to open, diverting flow around the low pressure feedwater heaters. A consequent reduction in feedwater temperature to the steam generators would occur.

Feedwater temperature may also be reduced during a load rejection trip. The feedwater transient data taken from a 100% net load trip test showed that a maximum feedwater temperature decrease of 230°F occurred over a 400-second time period.

Reductions in temperature of feedwater entering the steam generators result in an increase in core power and create a greater load demand on the RCS. The net effect on the RCS of a reduction in reactor coolant temperature is similar to the effect of increasing secondary steam flow. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The high neutron flux trip, overtemperature delta-T trip, and overpower delta-T trip act to prevent any power increase that could lead to a DNBR less than the limit value. The reactor reaches a new equilibrium condition at a power level corresponding to the new steam generator delta-T. A small temperature reduction results in only a small increase in reactor power and does not result in reactor trip. A larger temperature reduction produces a larger increase in reactor power and may cause a power/temperature mismatch and a reactor trip.

15.2.10A.2 Analysis of Effects and Consequences

15.2.10A.2.1 Temperature Drops Less than 73°F

The protection available to mitigate the consequences of a decrease in feedwater temperature is the same as that for an excessive increase in steam flow event, as discussed in Section 15.2.11. A step load increase of 10% from full load was analyzed, and the minimum DNBR for this event was found to be above the safety analysis limit values.

The increase in heat load resulting from a 10% increase in load is equivalent to a 73°F drop in feedwater temperature at the steam generator inlet. Thus a feedwater temperature transient that results in a feedwater temperature drop of 73°F or less at the steam generator inlets is less severe than the excessive load increase incident presented in Section 15.2.11 and as such does not exceed any safety limits.

15.2.10A.2.2 Temperature Drops Greater than 73°F

To address feedwater temperature reductions that exceed 73°F, calculations were performed assuming instantaneous temperature drops of 150°F and 250°F at the steam generator. The calculations were based on full power and nominal steam flow conditions being maintained until after reactor trip (any reduction in steam flow, such as that which would occur following a load rejection, would only decrease the effect of the temperature reduction). The maximum temperature drop of 250°F was chosen to bound the temperature decrease of 230°F experienced during the net load trip test when the LTBR was actuated in response to a load reduction. In this test, feedwater temperature dropped approximately 230°F over a time period of 400 seconds, which is significantly less severe than the instantaneous drop of 250°F assumed in the analysis.

12.2.10A.3 Results

The results of the analysis for an instantaneous feedwater temperature drop of 150°F show that the reactor would remain in operation. The results of the analysis for an instantaneous feedwater temperature drop of 250°F show that reactor trip occurs on high neutron flux. The minimum DNBRs for both analyses were above the safety analysis limit DNBR values and all acceptance criteria for the event were met.

15.2.10A.4 Conclusions

The performed analyses indicate that no safety limits would be exceeded for instantaneous feedwater temperature reductions of 250°F or less. The feedwater temperature decrease due to the net load trip was approximately 230°F. Since it is expected that the feedwater temperature decrease due to inadvertent actuation of the LTBR would be significantly less than that of the net load trip, the consequences of both discussed events are bounded by the analyses performed.

15.2.11 Excessive Load Increase Incident

15.2.11.1 Identification of Cause and Accident Description

An excessive load increase incident is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. The reactor control system is designed to accommodate a 10% step-load increase or a 5% per minute ramp load increase in the range of 15 to 100% of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the reactor protection system.

This accident could result from either an administrative violation such as excessive loading by the operator or an equipment malfunction in the steam dump control or turbine speed control.

During power operation, steam dump to the condenser is controlled by reactor coolant condition signals; i.e., high reactor coolant temperature indicates a need for steam dump. A single controller malfunction does not cause steam dump; an interlock is provided that blocks the opening of the valves unless a large turbine load decrease or a turbine trip has occurred.

Protection against an excessive load increase accident is provided by the following reactor protection system signals:

(1) Overpower ΔT

(2) Overtemperature ΔT

(3) Power range high neutron flux.

15.2.11.2 Analysis of Effects and Consequences

This accident is analyzed using the LOFTRAN code (Ref 4). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, feedwater system, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

Four cases are analyzed to demonstrate the plant behavior following a 10% step load increase from rated load. These cases are as follows:

(1) Reactor control in manual with BOL minimum moderator reactivity feedback

(2) Reactor control in manual with EOL maximum moderator reactivity feedback

(3) Reactor control in automatic with BOL minimum moderator reactivity feedback

(4) Reactor control in automatic with EOL maximum moderator reactivity feedback

For the BOL minimum moderator feedback cases, the core has the least negative moderator temperature coefficient of reactivity and the least negative Doppler only power coefficient curve; therefore the least inherent transient response capability. For the EOL maximum moderator feedback cases, the moderator temperature coefficient of reactivity has its highest absolute value and the most negative Doppler only power coefficient curve. This results in the largest amount of reactivity feedback due to changes in coolant temperature.

A conservative limit on the turbine valve opening is assumed, and all cases are studied without credit being taken for pressurizer heaters.

This accident is analyzed with the improved thermal design procedure as described in Reference 5. Initial reactor power, RCS pressure and temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR as described in Reference 5.

Plant characteristics and initial conditions are further discussed in Section 15.1.

Normal reactor control systems and engineered safety systems are not required to function. The reactor protection system is assumed to be operable; however, reactor trip is not encountered for most cases due to the error allowances assumed in the setpoints. No single active failure will prevent the reactor protection system from performing its intended function.

The cases which assume automatic rod control are analyzed to ensure that the worst case is presented. The automatic function is not required.

15.2.11.3 Results

The calculated sequence of events for the excessive load increase incident is shown on Table 15.2-1.

Figures 15.2.11-1 through 15.2.11-4 illustrate the transient with the reactor in the manual control mode. As expected, for the BOL minimum moderator feedback case, there is a slight power increase, and the average core temperature shows a large decrease. This results in a DNBR which increases above its initial value. For the EOL maximum moderator feedback manually controlled case, there is a much larger increase in reactor power due to the moderator feedback. A reduction in DNBR is experienced but DNBR remains above the limit value.

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.

15.2.11.4 Conclusions

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

15.2.12 Accidental Depressurization of the Reactor Coolant System

15.2.12.1 Identification of Causes and Accident Description

An accidental depressurization of the Reactor Coolant System 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 the hot leg saturation pressure. 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 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. 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.

15.2.12.2 Analysis of Effects and Consequences

The accidental depressurization transient is analyzed with the LOFTRAN code (Ref 4). 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 anlyzed 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 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.

15.2.12.3 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, 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 on Figure 15.2.12-2. The resulting DNBR never goes below the safety analysis limit value as shown on Figure 15.2.12-1.

15.2.12.4 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.

15.2.13 Accidental Depressurization of the Main Steam System

15.2.13.1 Identification of Causes and Accident Description

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The analyses, assuming a rupture of a main steam pipe, are discussed in Section 15.4.

The steam released as a consequence of this accident results in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin.

The analysis is performed to demonstrate that the following criterion is satisfied: Assuming a stuck RCCA and a single failure in the engineered safety features (ESF) the limit DNBR values will be met after reactor trip for a steam release equivalent to the spurious opening, with failure to close, of the largest of any single steam dump, relief, or safety valve.

The following systems provide the necessary mitigation of an accidental depressurization of the main steam system.

(1) Safety injection system (SIS) actuation from any of the following:

 (a) Two-out-of-four low pressurizer pressure signals

 (b) High differential pressure signals between steam lines.

(2) The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal

(3) Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the backup feedwater isolation valves.

15.2.13.2 Analysis of Effects and Consequences

The following analyses of a secondary system steam release are performed with the LOFTRAN (Ref 4) code:

(1) A full plant simulation to determine RCS temperature and pressure during cooldown

(2) An analysis to ascertain that the reactor does not exceed the limit DNBR values.

The following conditions are assumed to exist at the time of a secondary system break accident.

(1) EOL shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive assembly stuck in its fully withdrawn position. Operation of RCCA banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system break accident will not lead to a more adverse condition than the case analyzed.

(2) A negative moderator coefficient corresponding to the EOL rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure is included. The k_{eff} versus temperature curve at 1150 psia corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown on Figure 15.2.13-1.

(3) Minimum capability for injection of high concentration boric acid solution corresponding to the most restrictive single failure in the safety injection system. The injection curve is shown on Figure 15.2.13-2. This corresponds to the flow delivered by one charging pump delivering its full contents to the cold leg header. No credit has been taken for the low concentration boric acid that must be swept from the safety injection lines downstream of the refueling water storage tank (RWST) isolation valves prior to the delivery of ~~high concentration~~ boric acid (2300 ppm) to the reactor coolant loops. No credit has been taken for the BIT.

(4) The case studied is an initial total steam flow of 228 lb/sec at 1015 psia from one steam generator with offsite power available. This is the maximum capacity of any single steam dump or safety valve. Initial hot shutdown conditions at time zero are assumed since this represents the most pessimistic initial condition.

Should the reactor be just critical or operating at power at the time of a steam release, the reactor will be tripped by the normal overpower protection when power level reaches trip point. Following a trip at power the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel.

Thus, the additional energy stored is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. The additional insertions proceed then in the same manner as in the analysis which assumes no-load condition at time zero. However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are less for steam line breaks occurring at power.

(5) In computing the steam flow, the Moody Curve for fL/D = 0 is used.

(6) Perfect moisture separation in the steam generator is assumed.

15.2.13.3 Results

The results presented are a conservative indication of the events that would occur assuming a secondary system steam release since it is postulated that all of the conditions described above occur simultaneously.

Figure 15.2.13-3 shows the transients arising as the result of a steam release having an initial steam flow of 228 lb/sec at 1015 psia with steam release from one safety valve. The assumed steam release is the maximum capacity of any single steam dump or safety valve. In this case, safety injection is initiated automatically by low pressurizer pressure. Operation of one centrifugal charging pump is considered. Boron solution at 2300 ppm enters the RCS providing sufficient negative reactivity to prevent core damage. The reactivity transient for the cases shown on Figure 15.2.13-3 is more severe than that of a failed steam generator safety or relief valve that is terminated by steam line differential pressure, or a failed condenser dump valve that is terminated by low pressurizer pressure and level. The transient is quite conservative with respect to cooldown since no credit is taken for the energy stored in the system metal other than that of the fuel elements or the energy stored in the other steam generators. Since the transient occurs over a period of about 5 minutes, the neglected stored energy is likely to have a significant effect in slowing the cooldown.

15.2.13.4 Conclusions

The analysis has shown that the criteria stated earlier in this section are satisfied. For an accidental depressurization of the main steam system, the DNB design basis is met. This case is less limiting than the rupture of a main steam pipe case presented in Section 15.4.

15.2.14 Spurious Operation of the Safety Injection System at Power

15.2.14.1 Identification of Causes and Accident Description

Spurious SIS operation at power could be caused by operator error or a false electrical actuating signal. A spurious signal in any of the following channels could cause this accident.

(1) High containment pressure

(2) Low pressurizer pressure

(3) High steam line differential pressure

(4) High steam line flow coincident with low average coolant temperature or low steam line pressure.

Following the actuation signal, the suction of the coolant charging pumps is diverted from the volume control tank to the RWST. The valves isolating the BIT from the charging pumps and the valves isolating the BIT from the injection header open automatically. The charging pumps then force ~~highly concentrated (20,000 ppm)~~ boric acid solution ~~from the BIT~~ through the header and injection line and into the cold legs of each loop. The safety injection pumps also start automatically but provide no flow when the RCS is at normal pressure. The passive injection system and the low-head system also provide no flow at normal RCS pressure.

An SIS signal normally results in a reactor trip followed by a turbine trip. However, it cannot be assumed that any single fault that actuates the SIS will also produce a reactor trip. Therefore, two different courses of events are considered.

Case A: Trip occurs at the same time spurious injection starts

Case B: The reactor protection system produces a trip later in the transient.

For Case A, the operator should determine if the spurious signal was transient or steady state in nature, i.e., an occasional occurrence or a definite fault. The operator will determine this by following approved procedures. In the transient case, the operator would stop the safety injection and bring the plant to the hot shutdown condition. If the SIS must be disabled for repair, boration should continue and the plant brought to cold shutdown.

For Case B, the reactor protection system does not produce an immediate trip and the reactor experiences a negative reactivity excursion causing a decrease in the reactor power. The power unbalance causes a drop in T_{avg} and consequent coolant shrinkage, and pressurizer pressure and level drop. Load will decrease due to the effect of reduced steam pressure on load if the electrohydraulic governor fully opens the turbine throttle valve. If automatic rod control is used, these effects will be lessened until the rods have moved out of the core. The transient is eventually terminated by the reactor protection system low-pressure trip or by manual trip.

The time to trip is affected by initial operating conditions including core burnup history that affects initial boron concentration, rate of change of boron concentration, and Doppler and moderator coefficients.

Recovery from this incident for Case B is in the same manner as for Case A. The only difference is the lower T_{avg} and pressure associated with the power imbalance during this transient. The time at which reactor trip occurs is of no concern for this accident. At lighter loads coolant contraction will be slower resulting in a longer time to trip.

15.2.14.2 Analysis of Effects and Consequences

The spurious operation of the SIS system is analyzed with the LOFTRAN program (Ref 4). The code simulates the neutron kinetics, RCS, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, steam generator safety valves, and the effect of the SIS. The program computes pertinent plant variables including temperatures, pressures, and power level.

Because of the power and temperature reduction during the transient, operating conditions do not approach the core limits. Analyses of several cases show that the results are relatively independent of time to trip.

A typical transient is considered representing conditions at BOL. Results at EOL are similar except that moderator feedback effects result in a slower transient.

The assumptions are:

(1) Initial Operating Conditions

The initial reactor power and RCS temperatures are assumed at their maximum values consistent with steady state full power operation including allowances for calibration and instrument errors.

(2) Moderator and Doppler Coefficients of Reactivity

A positive BOL moderator temperature coefficient was used. A low absolute value Doppler power coefficient was assumed.

(3) Reactor Control

The reactor was assumed to be in manual control.

(4) Pressurizer Heaters

Pressurizer heaters were assumed to be inoperative in order to increase the rate of pressure drop.

(5) Boron Injection

At time zero, two charging pumps begin injection and pump ~~inject 22,500 ppm~~ borated water through the SIS into the cold legs of each loop.

(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.

15.2.14.3 Results

The transient response for the minimum feedback case is shown on 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.

15.2.14.4 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.

15.2.15 References

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

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

3. C. Hunin, FACTRAN, A Fortran N Code for Thermal Transients in UO_2 Fuel Rod, WCAP-7908, June 1972.

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

5. Chelemer, H., et al., "Improved Thermal Design Procedure," WCAP-8567 (Proprietary) and WCAP-8568 (Non-Proprietary), July 1975.

6. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended to the date of the most recent FSAR Update Revision.

7. Chelemer, H. et al, Subchannel Thermal Analysis of Rod Bundle Cores, WCAP-7015, Rev. 1, January 1969.

8. M. A. Mangan, Overpressure Protection for Westinghouse Pressurized Water Reactor, WCAP-7769, October 1971.

9. J. S. Shefcheck, Application of the THINC Program to PWR Design, WCAP-7359-L, August 1969 (Proprietary) and WCAP-7838, January 1972.

TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

Accident	Event	Time, sec
Uncontrolled RCCA Withdrawal from a Subcritical Condition	Initiation of uncontrolled rod withdrawal 7.5×10^{-4} Δk/sec reactivity insertion rate from 10^{-9} of nominal power	0.0
	Power range high neutron flux low setpoint reached	9.6
	Peak nuclear power occurs	9.8
	Rods begin to fall into core	10.1
	Peak heat flux occurs	11.9
	Peak hot spot average fuel temperature occurs	12.4
	Peak hot spot average clad temperature occurs	12.3

Accident	Event	Time, sec
Uncontrolled RCCA Withdrawal at Power		
1. Case A	Initiation of uncontrolled RCCA withdrawal at maximum reactivity insertion rate (7.5×10^{-4} Δk/sec)	0.0
	Power range high neutron flux high trip point reached	1.6
	Rods begin to fall into core	2.1
	Minimum DNBR occurs	3.0
2. Case B	Initiation of uncontrolled RCCA withdrawl at a small reactivity insertion rate (3.0×10^{-5} Δk/sec)	
	Overtemperature ΔT reactor trip signal initiated	30.8
	Rods begin to fall into core	32.8
	Minimum DNBR occurs	33.2

Accident	Event	Time, sec
Uncontrolled Boron Dilution		
1. Dilution during refueling and startup	Dilution begins	0.0
	Operator isolates source of dilution; minimum margin to criticality occurs	~1920 or more
2. Dilution during full power operation		
a. Automatic reactor control	1.6% shutdown margin lost	~1180
b. Manual reactor control	Dilution begins	0.0
	Reactor trip setpoint reached for high neutron flux	40
	Rods begin to fall into core	40.5
	1.6% shutdown is lost (if dilution continues after trip)	~900

Accident	Event	Time, sec
Partial Loss of Forced Reactor Coolant Flow		
All loops operating, two pumps coasting down		
	Coastdown begins	0.0
	Low-flow reactor trip	1.43
	Rods begin to drop	2.43
	Minimum DNBR occurs	3.9

TABLE 15.2-1

Accident	Event	Time, sec
Startup of an Inactive Reactor Coolant Loop	Initiation of pump startup	0.0
	Power reaches high nuclear flux trip	3.2
	Rods begin to drop	3.7
	Minimum DNBR occurs	4

TABLE 15.2-1

Accident	Event	Time, sec
Loss of External Electrical Load		
1. With pressurizer control (BOL)	Loss of electrical load	0.0
	Initiation of steam release from steam generator safety valves	12.5
	Overtemperature ΔT	11.7
	Rods begin to drop	13.7
	Minimum DNBR occurs	15
	Peak pressurizer pressure occurs	14.5
2. With pressurizer control (EOL)	Loss of electrical load	0.0
	Initiation of steam release from steam generator safety valves	12.5
	Low Low steam generator water level reactor trip	57.0
	Rods begin to drop	59.0

Accident	Event	Time, sec
With pressurizer control (EOL) (Cont'd)	Minimum DNBR occurs	(a)
	Peak pressurizer pressure occurs	9.0
3. Without pressurizer control (BOL)	Loss of electrical load	0.0
	Initiation of steam release from steam generator safety valves	12.0
	High pressurizer pressure reactor trip point reached	6.1
	Rods begin to drop	8.1
	Minimum DNBR occurs	(a)
	Peak pressurizer pressure occurs	9.5

Accident	Event	Time, sec
4. Without pressurizer control (EOL)	Loss of electrical load	0.0
	Initiation of steam release from steam generator safety valves	12.5
	High pressurizer pressure reactor trip point reached	6
	Rods begin to drop	8
	Minimum DNBR occurs	(a)
	Peak pressurizer pressure occurs	8.5

Accident	Event	Time, sec	
		W/Power	W/O Power
Loss of Normal Feedwater and Loss of Offsite Power to the Station Auxiliaries (Station Blackout)			
	Low-low steam generator water level reactor trip; reactor coolant pumps begin to coast down	62	62
	Rods begin to drop	64	64
	Two steam generators begin to receive auxiliary feed from one motor-driven auxiliary feedwater pump	122	122
	Peak water level in pressurizer occurs	5120	2244

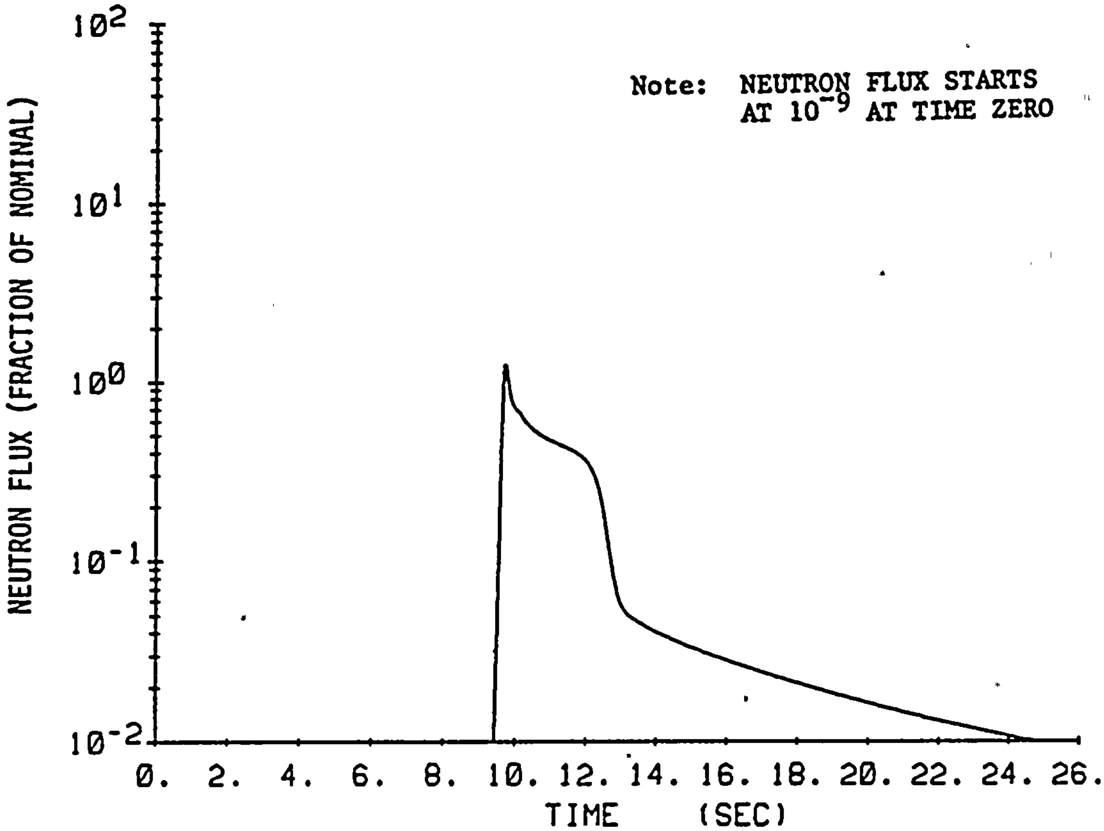
Accident	Event	Time, sec
Excessive Feedwater at Full Load	One main feedwater control valve fails fully open	0.0
	Minimum DNBR occurs	17.5
	Feedwater flow isolated due to high-high steam generator level	30.2

Accident	Event	Time, sec
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

Accident	Event	Time, sec
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 safety valve	0.0
	Low Pressurizer Pressure Reactor Trip Setpoint Reached	39.8
	Rods begin to drop	41.8
	Minimum DNBR occurs	42.2
Accidental Depressurization of the Main Steam System	Inadvertent opening of one main steam safety or relief valve	0.0
	Pressurizer empties	227
	2,300 ppm boron reaches RCS loops	328

Accident	Event	Time, sec
Inadvertent Operation of ECCS During Power Operation	Charging pumps begin injecting borated water	0.0
	Low-pressure trip point reached	23
	Rods begin to drop	25

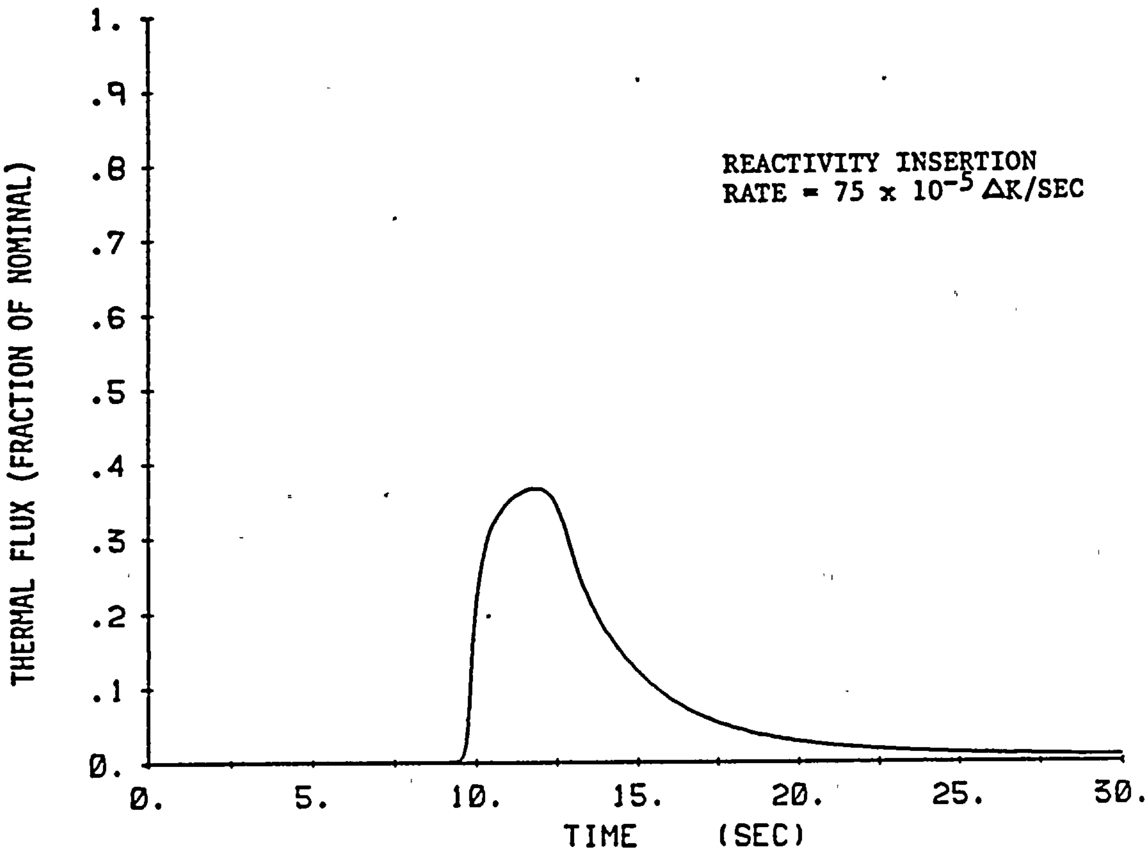
(a) DNBR does not decrease below its initial value.



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.1-1

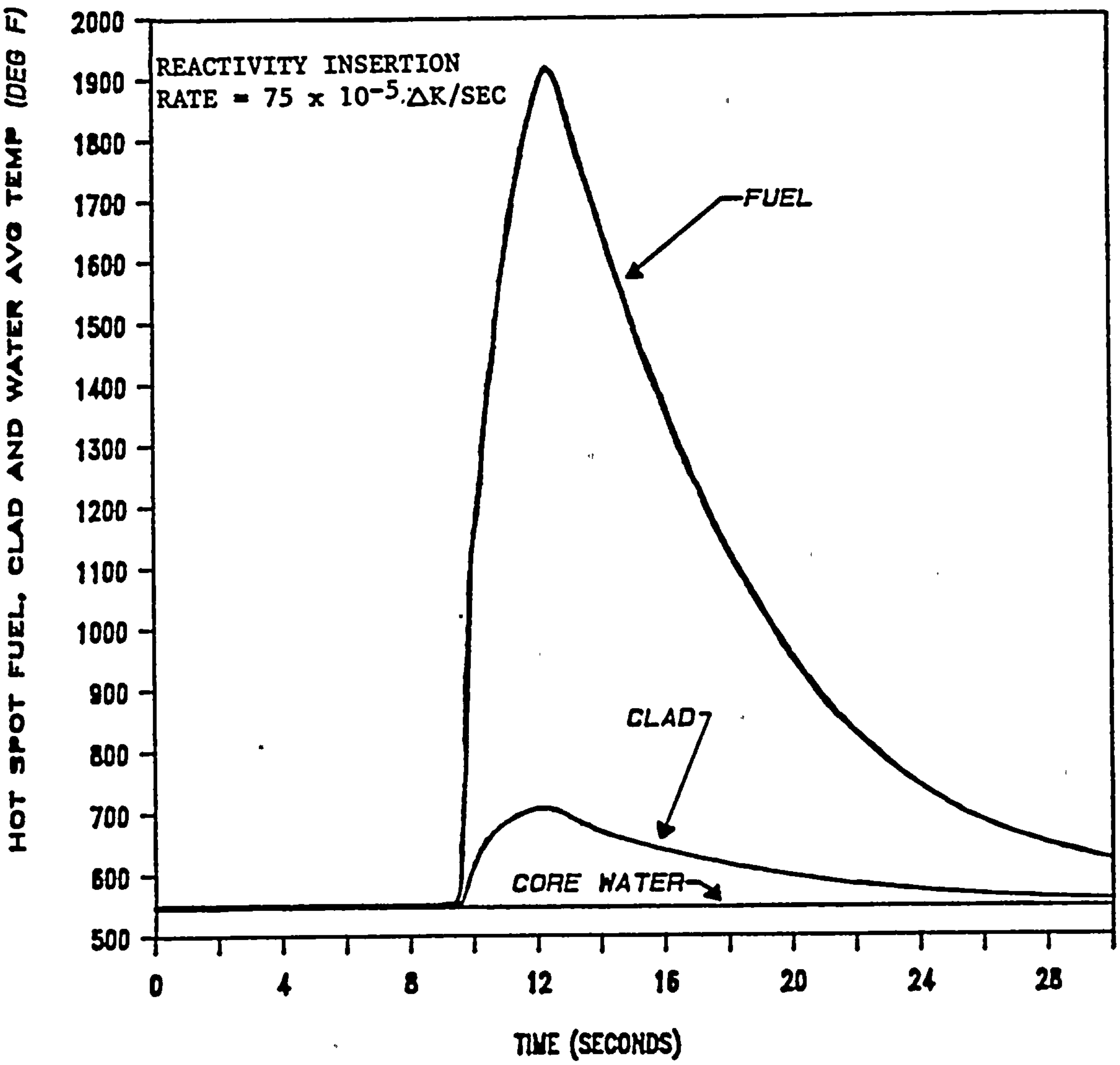
UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION NEUTRON FLUX VERSUS TIME



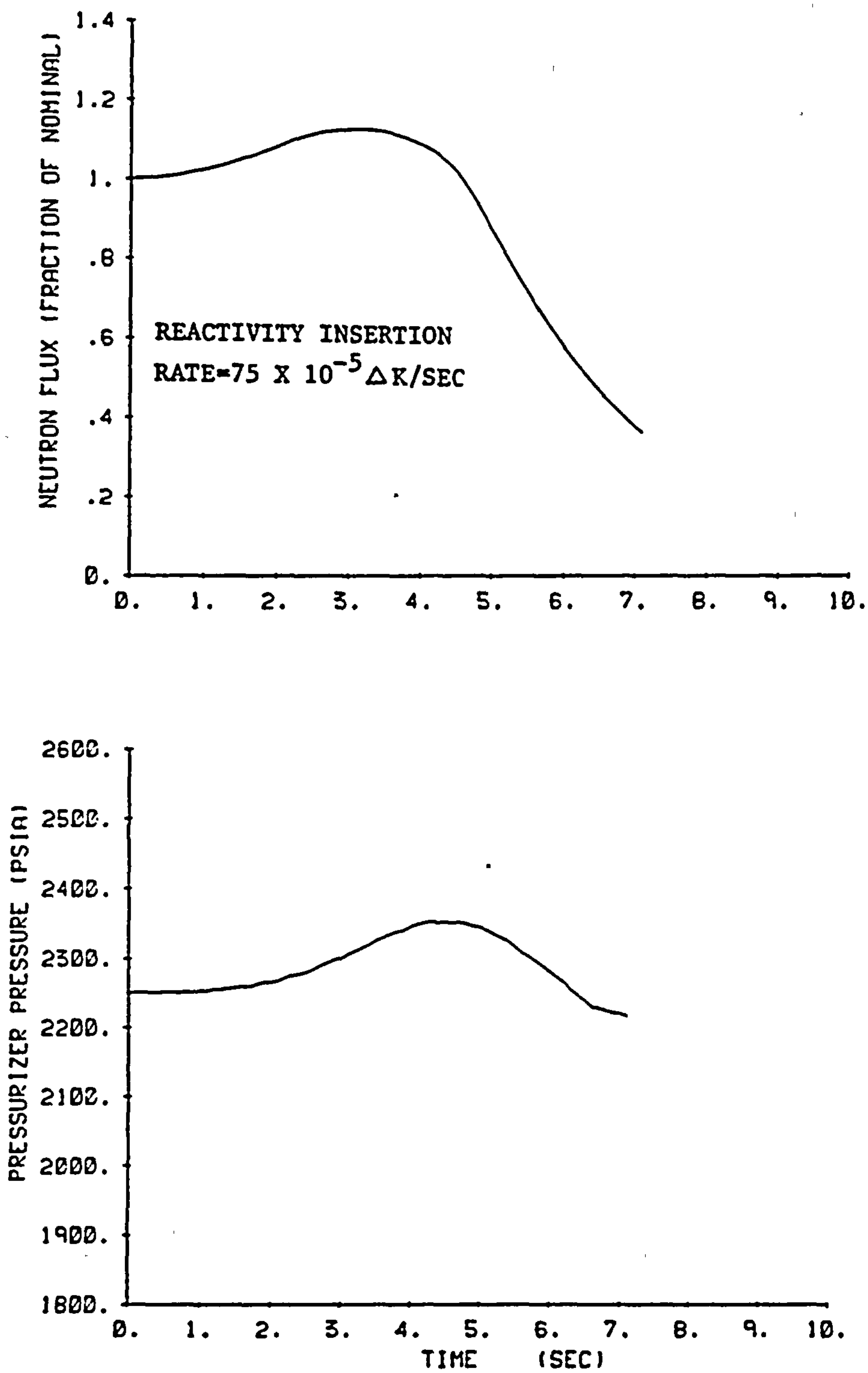
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.1-2

UNCONTROLLED ROD WITHDRAWAL
FROM A SUBCRITICAL CONDITION
THERMAL FLUX VERSUS TIME

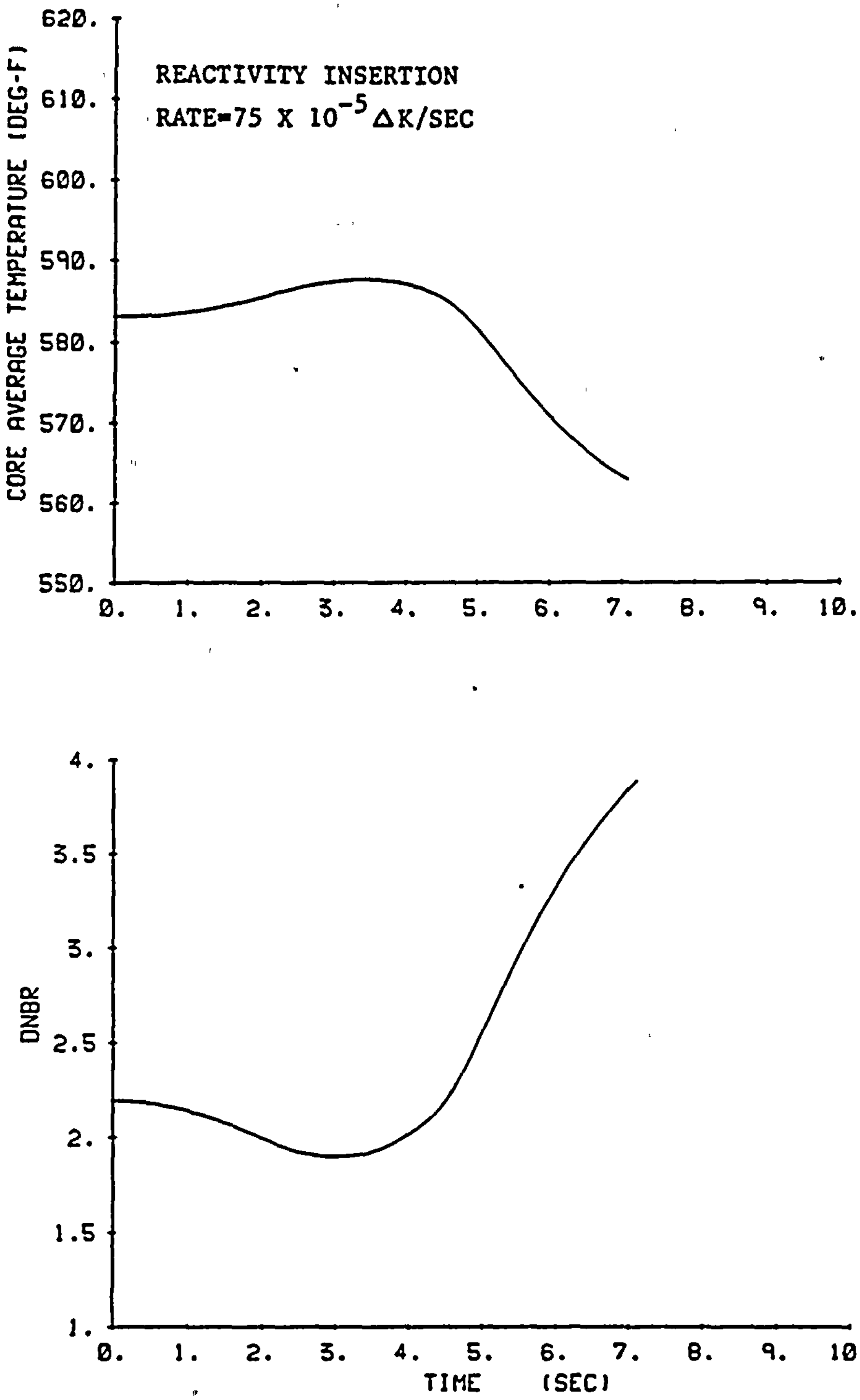


DIABLO CANYON UNITS 1 AND 2
FIGURE 15.2.1-3 UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION. TEMPERATURE VERSUS TIME. REACTIVITY INSERTION RATE 75 X 10^{-6} DELTA K/SEC



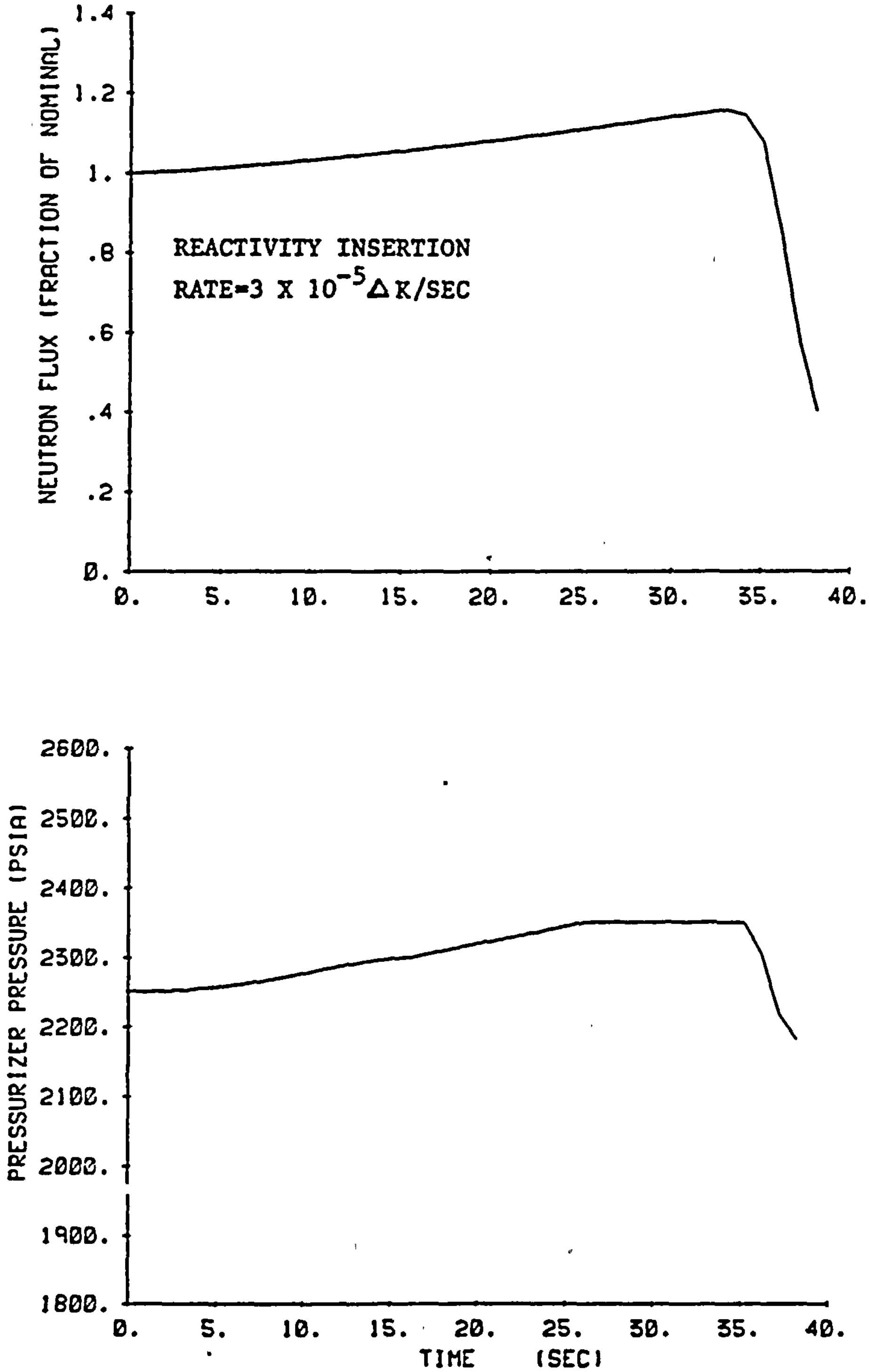
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.2-1
TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY HIGH NEUTRON FLUX TRIP. PRESSURIZER PRESSURE AND NEUTRON FLUX VERSUS TIME



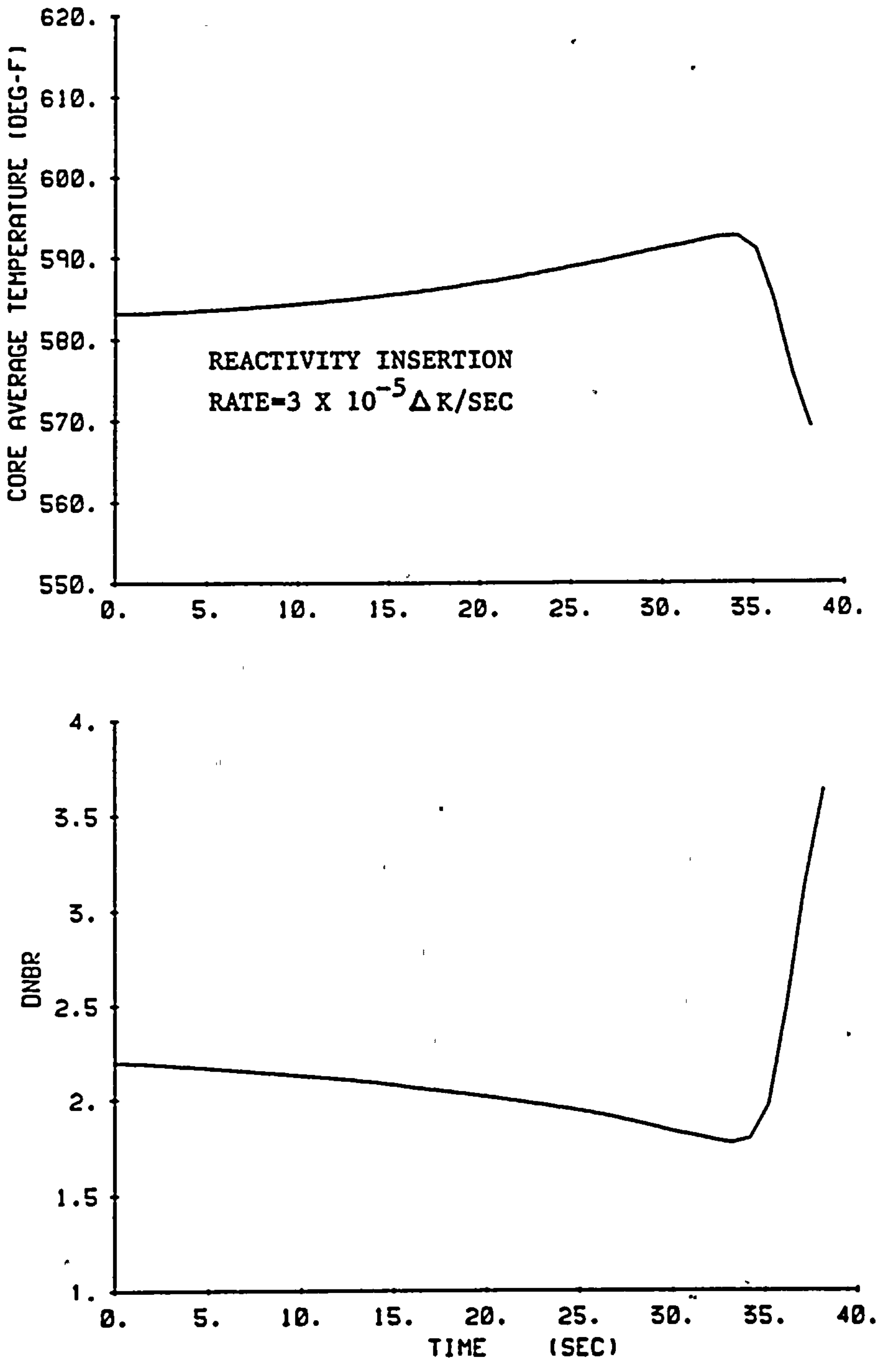
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.2-2
TRANSIENT RESPONSE FOR UNCONTROLLED
ROD WITHDRAWAL FROM FULL POWER
TERMINATED BY HIGH NEUTRON
FLUX TRIP.
DNBR AND TAVG VERSUS TIME



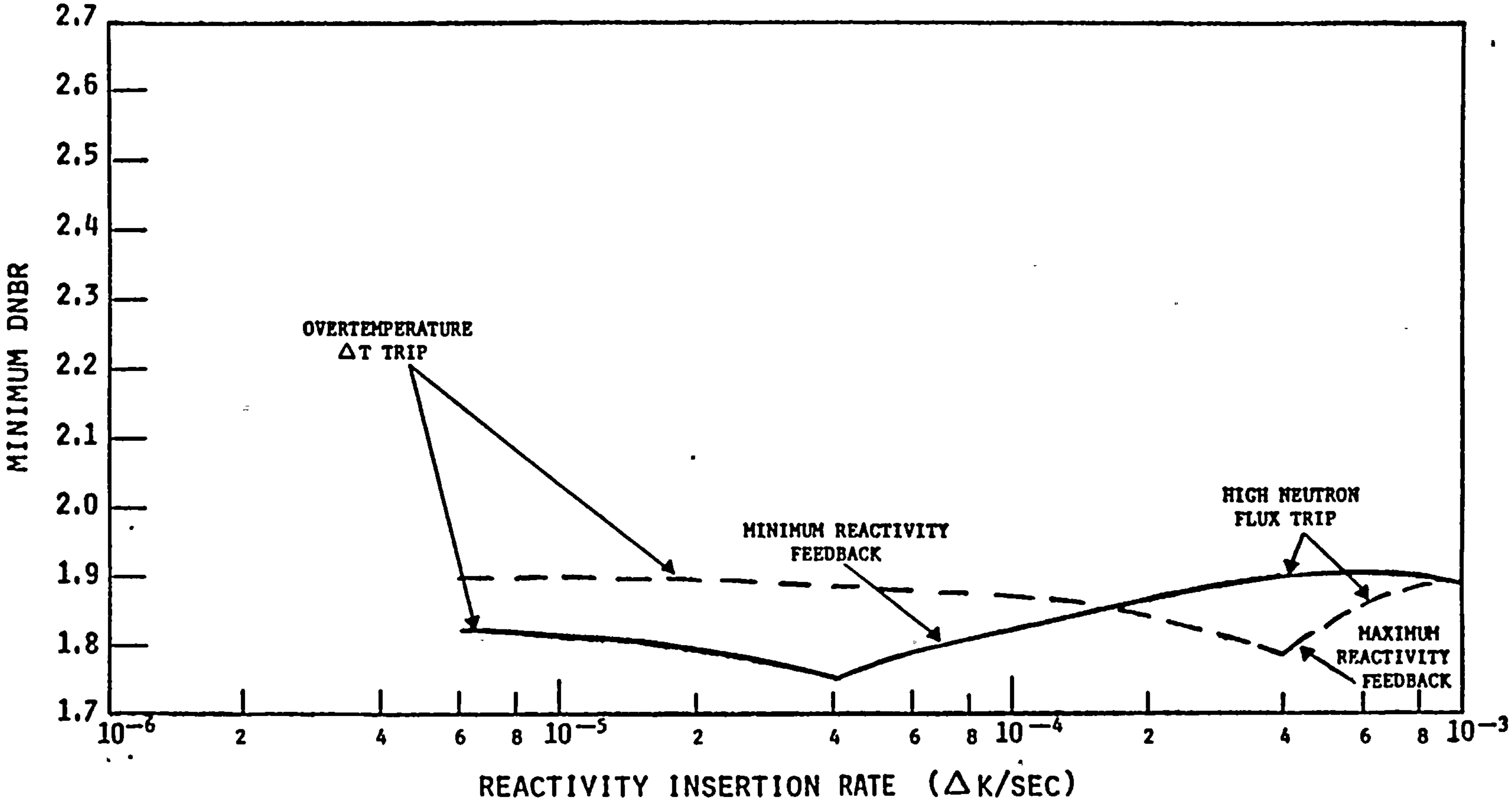
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.2-3
TRANSIENT RESPONSE FOR UNCONTROLLED ROD WITHDRAWAL FROM FULL POWER TERMINATED BY OVERTEMPERATURE DELTA T TRIP. PRESSURIZER PRESSURE AND NEUTRON FLUX VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

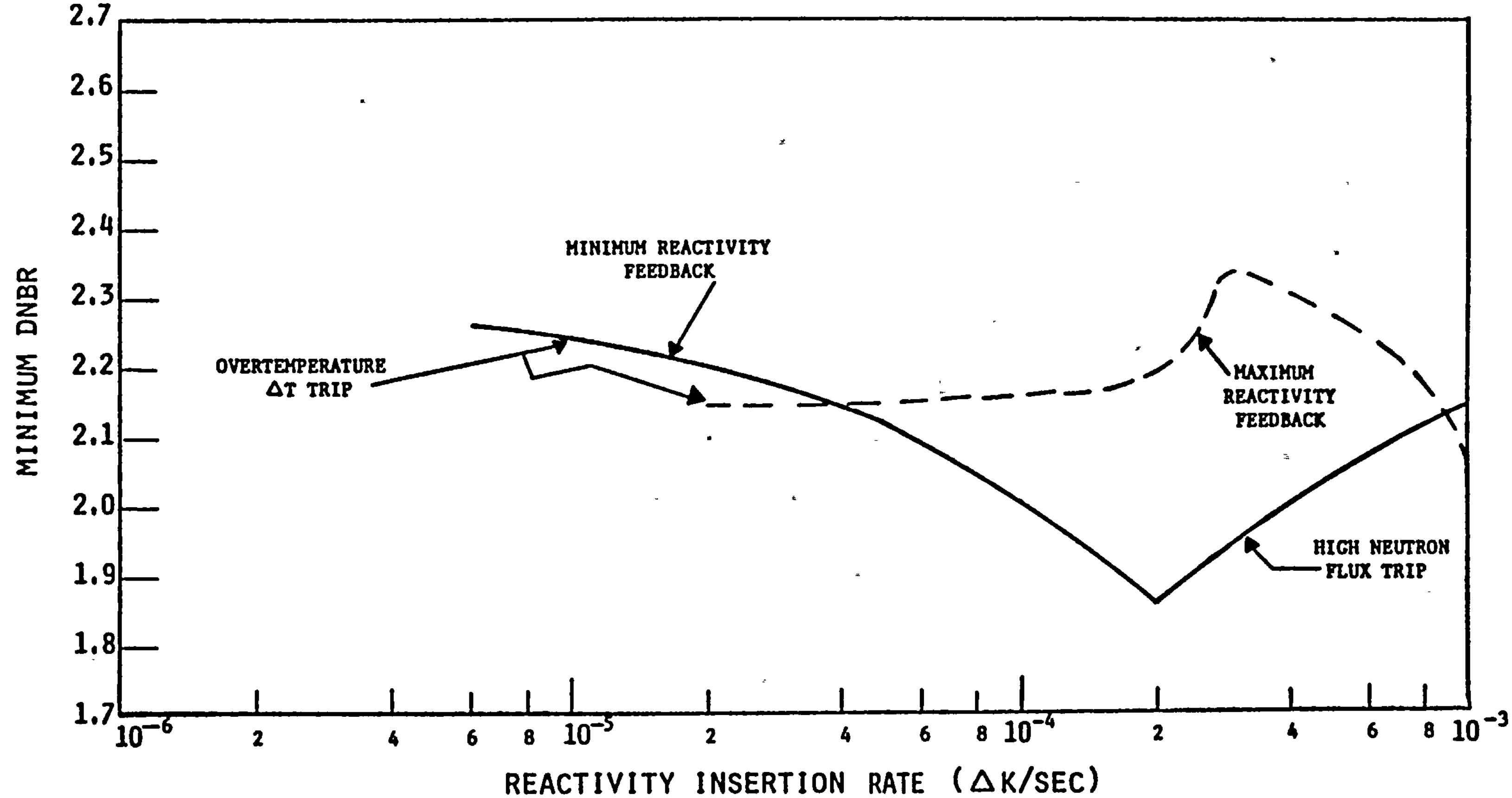
FIGURE 15.2.2-4
TRANSIENT RESPONSE FOR UNCONTROLLED
ROD WITHDRAWAL FROM FULL POWER
TERMINATED BY OVERTEMPERATURE
DELTA T TRIP.
DNBR AND TAVG VERSUS TIME



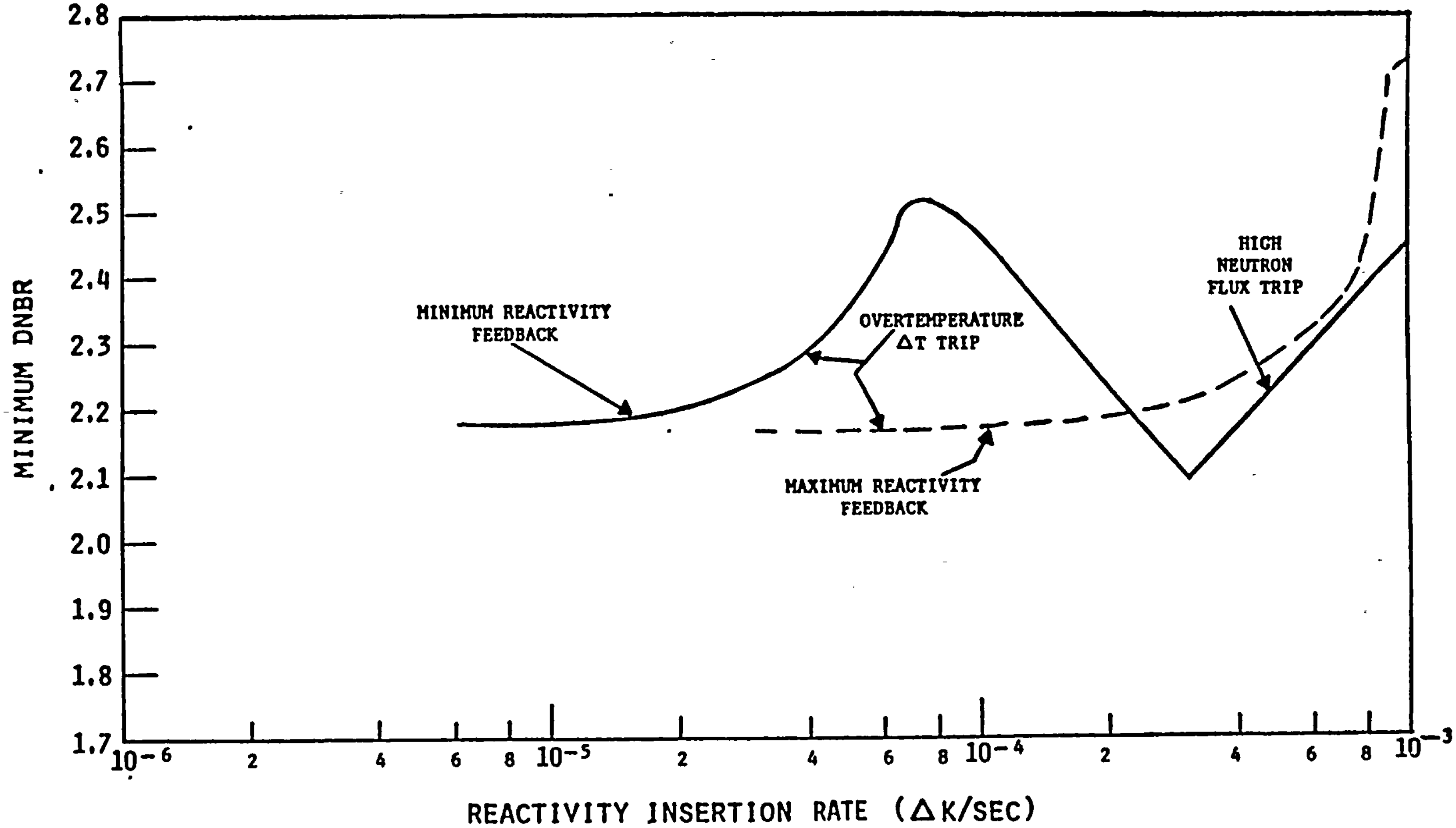
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.2-5

EFFECT OF REACTIVITY INSERTION RATE ON MINIMUM DNBR FOR A ROD WITHDRAWAL ACCIDENT FROM 100% POWER

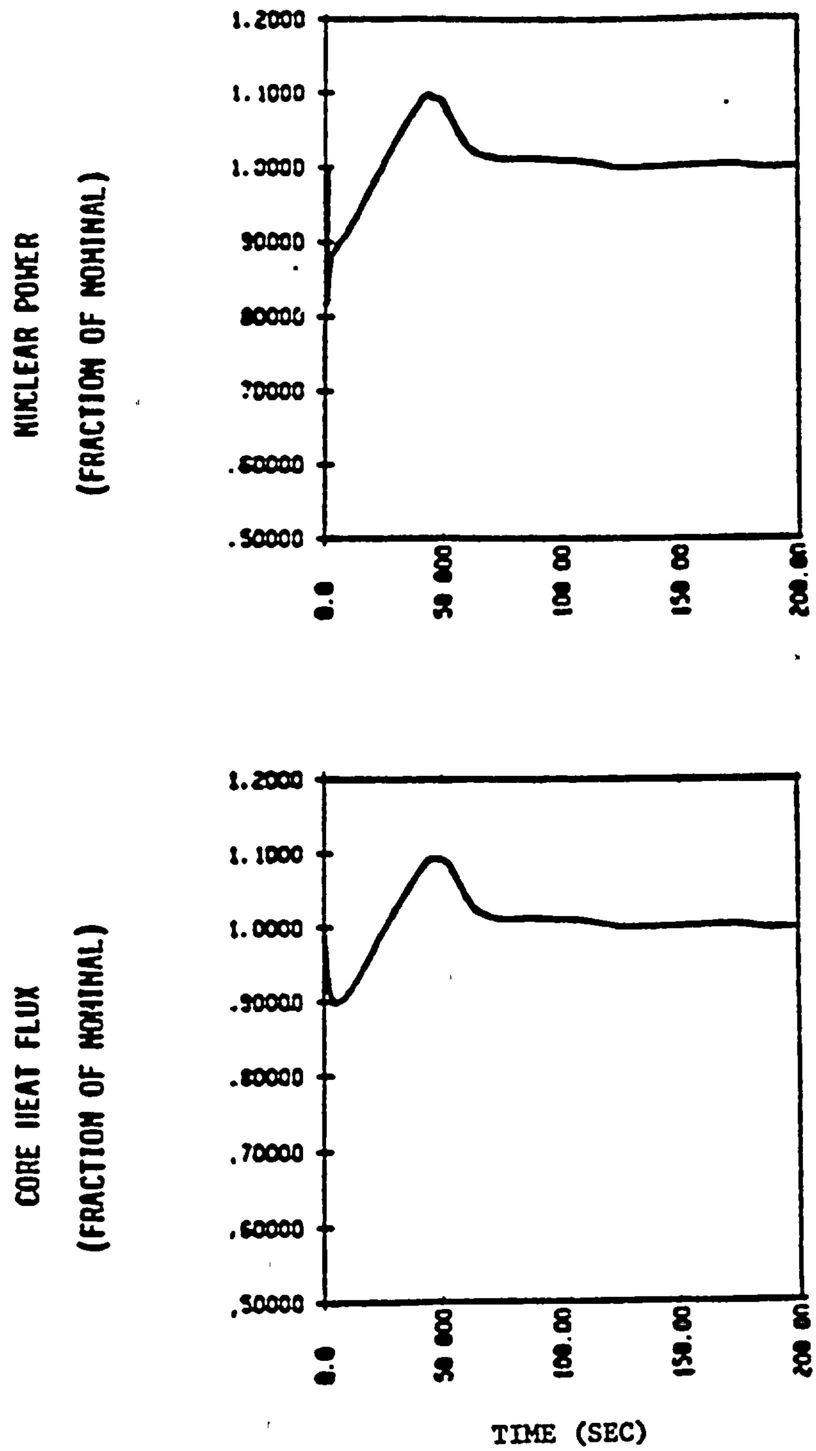


DIABLO CANYON UNITS 1 AND 2
FIGURE 15.2.2-6 EFFECT OF REACTIVITY INSERTION RATE ON MINIMUM DNBR FOR A ROD WITHDRAWAL ACCIDENT FROM 60% POWER



DIABLO CANYON UNITS 1 AND 2

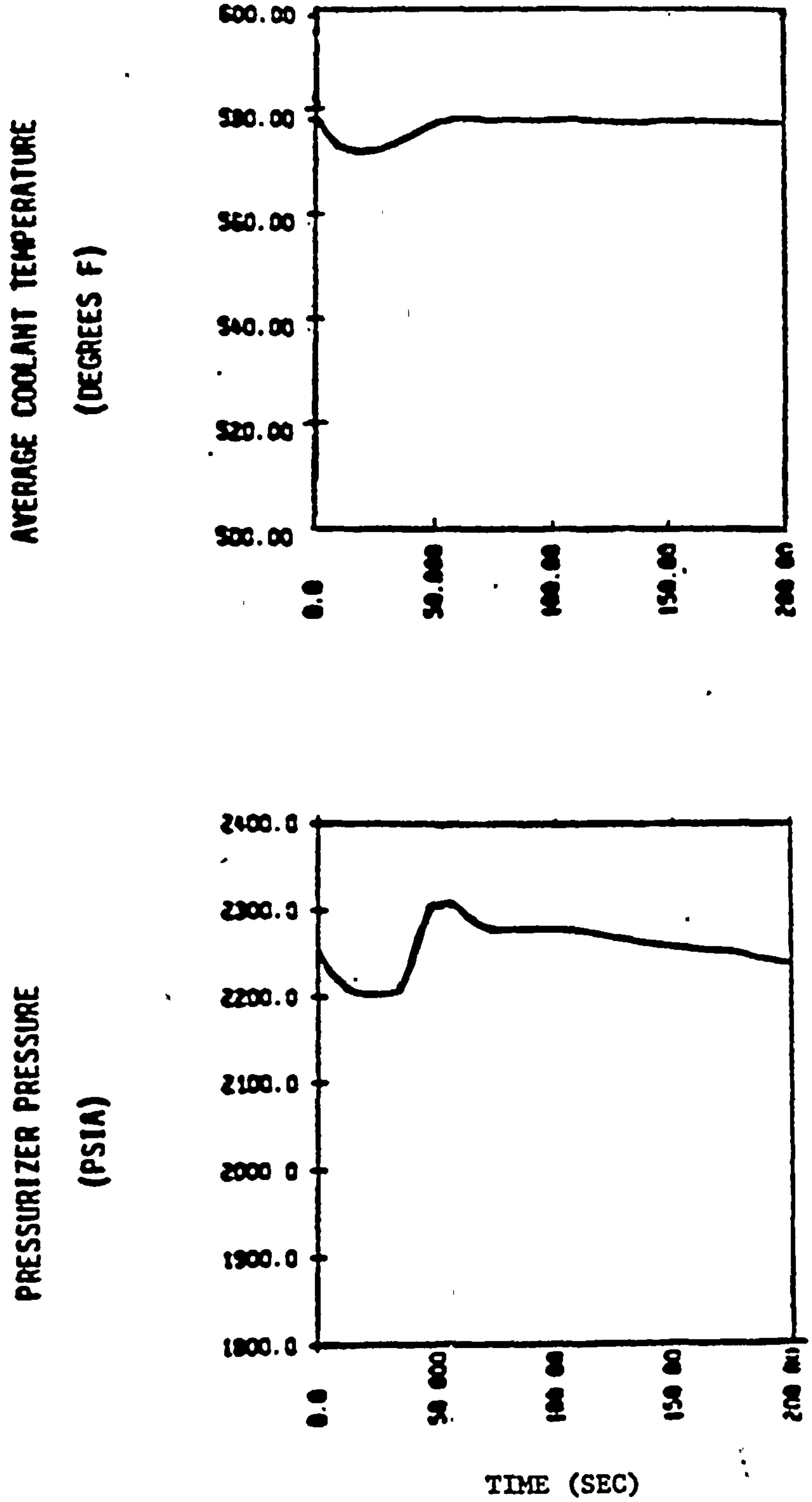
FIGURE 15.2.2-7
EFFECT OF REACTIVITY INSERTION RATE ON MINIMUM DNBR FOR A ROD WITHDRAWAL ACCIDENT FROM 10% POWER



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.3-1

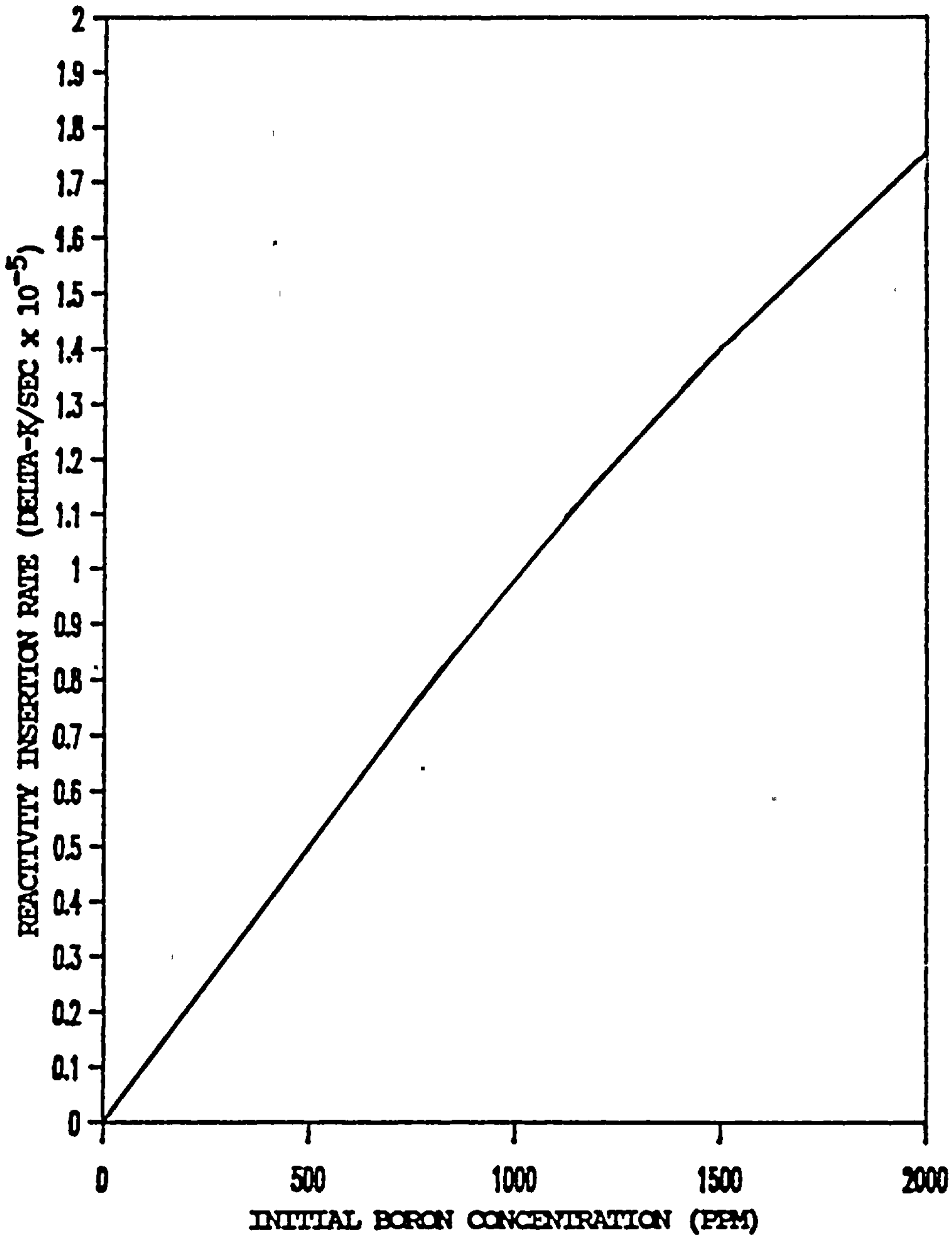
TRANSIENT RESPONSE TO DROPPED ROD CLUSTER CONTROL ASSEMBLY



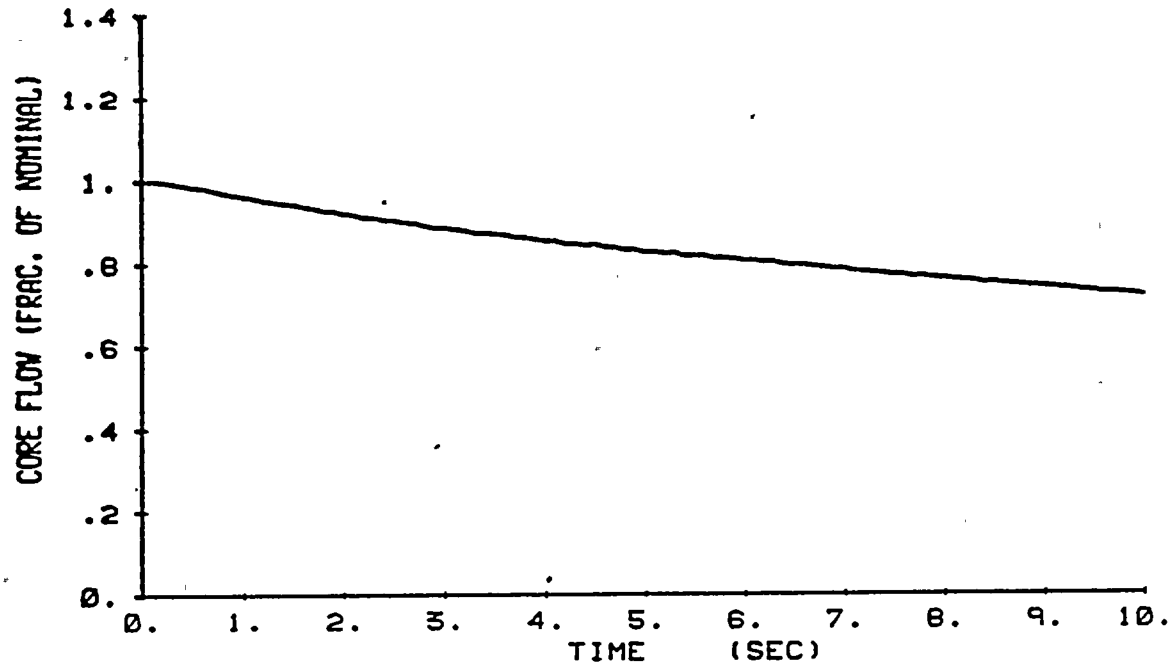
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.3-2

TRANSIENT RESPONSE TO DROPPED ROD CLUSTER CONTROL ASSEMBLY



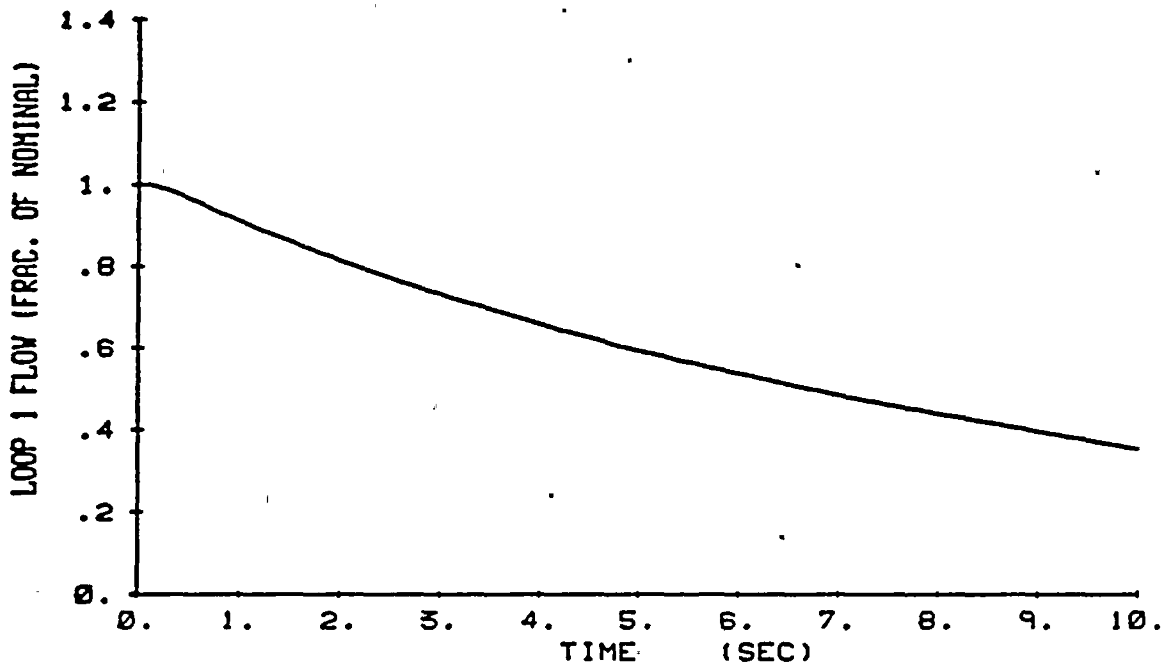
DIABLO CANYON UNITS 1 AND 2
FIGURE 15.2.4-1 VARIATION IN REACTIVITY INSERTION RATE WITH INITIAL BORON CONCENTRATION FOR A DILUTION RATE OF 262 GPM



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.5-1

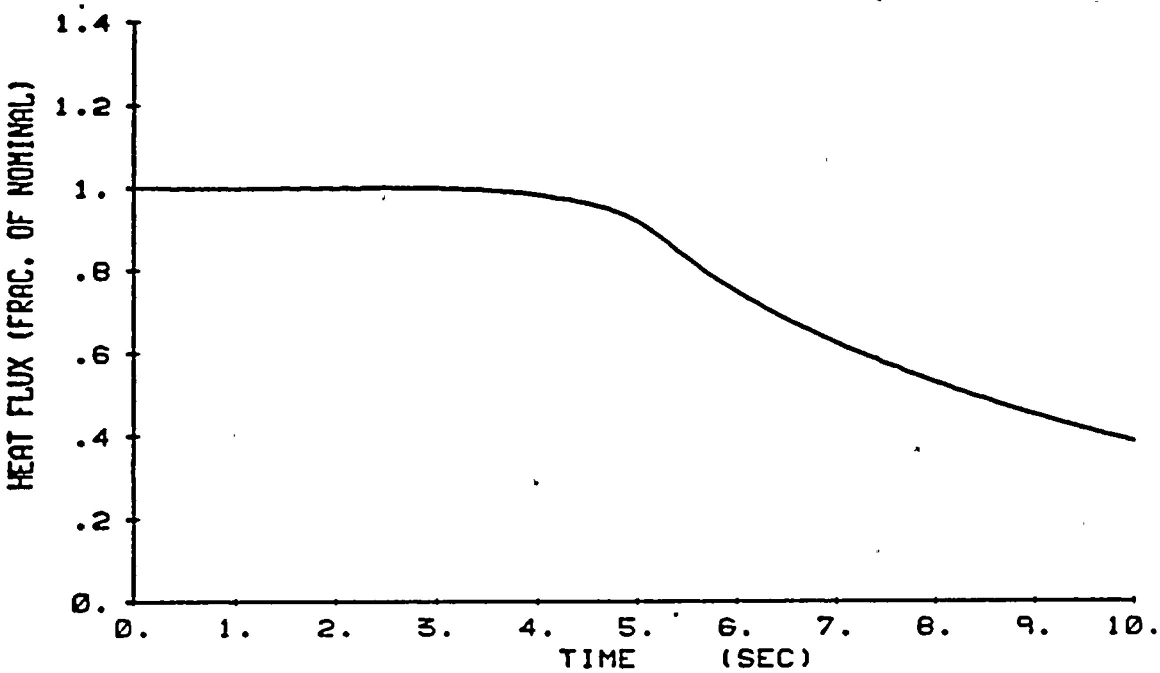
ALL LOOPS OPERATING,
TWO LOOPS COASTING DOWN,
CORE FLOW VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.5-2

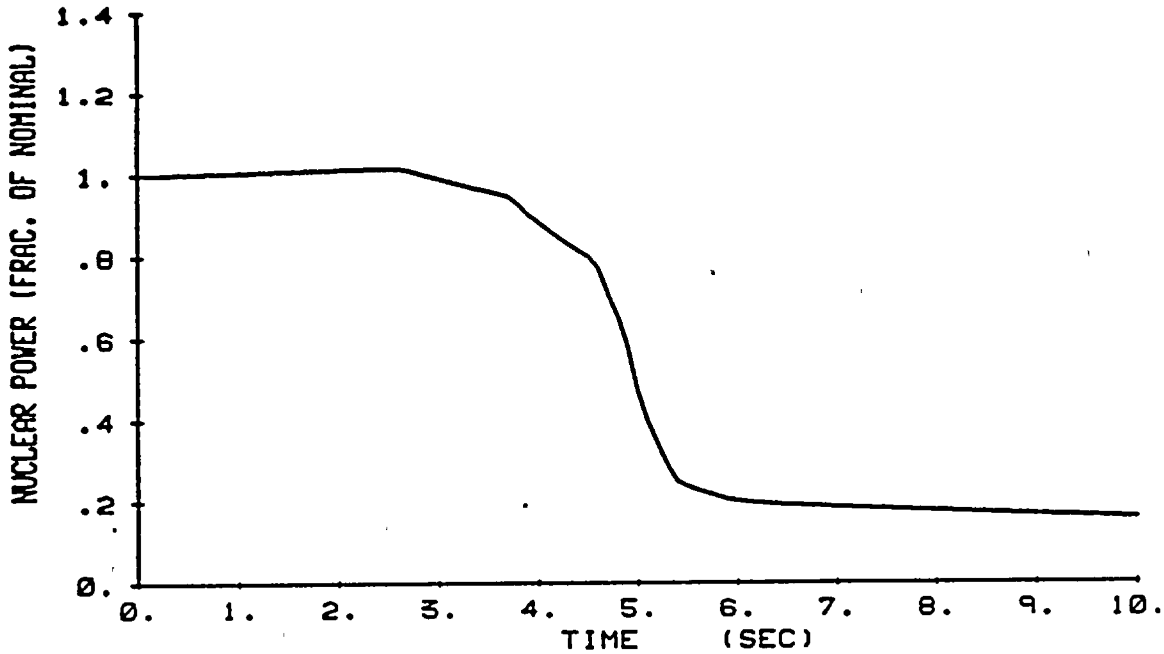
ALL LOOPS OPERATING.
TWO LOOPS COASTING DOWN.
FAULTED LOOP FLOW VERSUS TIME



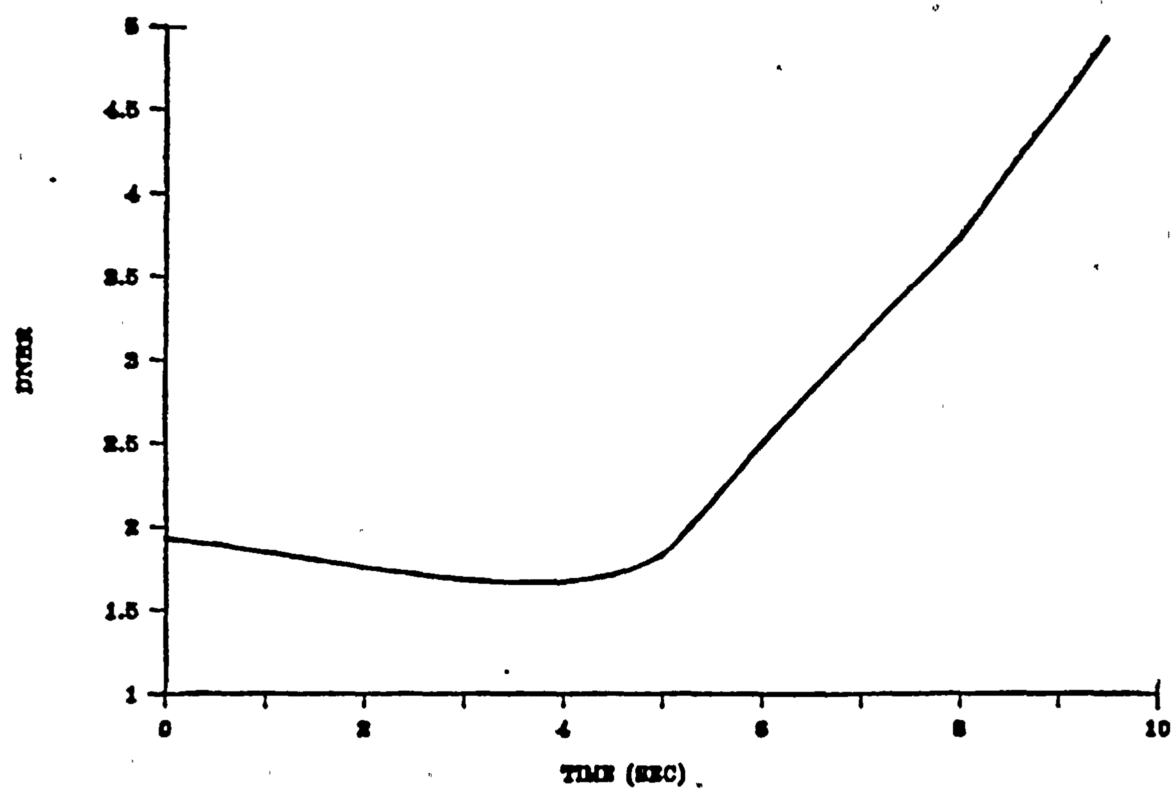
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.5-3

ALL LOOPS OPERATING.
TWO LOOPS COASTING DOWN.
HEAT FLUX VERSUS TIME



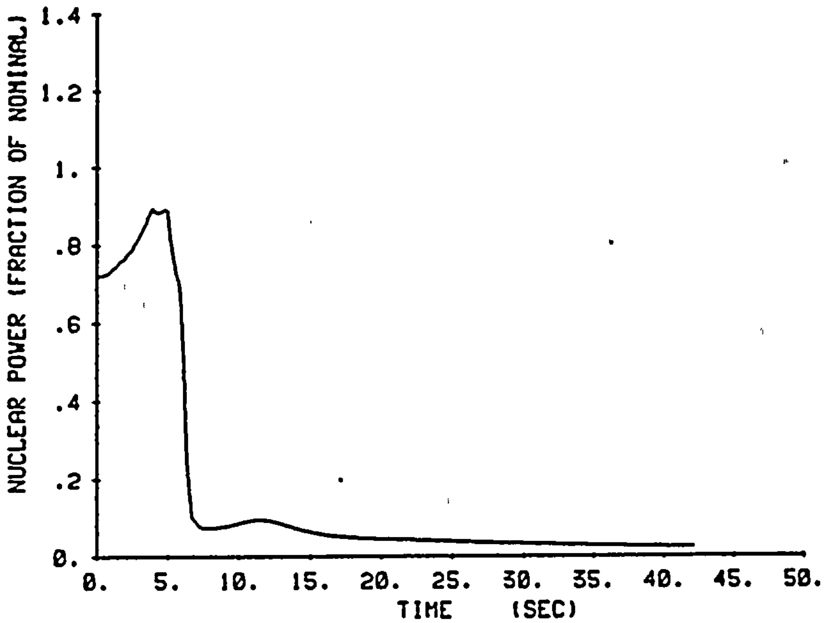
DIABLO CANYON UNITS 1 AND 2
FIGURE 15.2.5-4 ALL LOOPS OPERATING. TWO LOOPS COASTING DOWN. NUCLEAR POWER VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.5-5

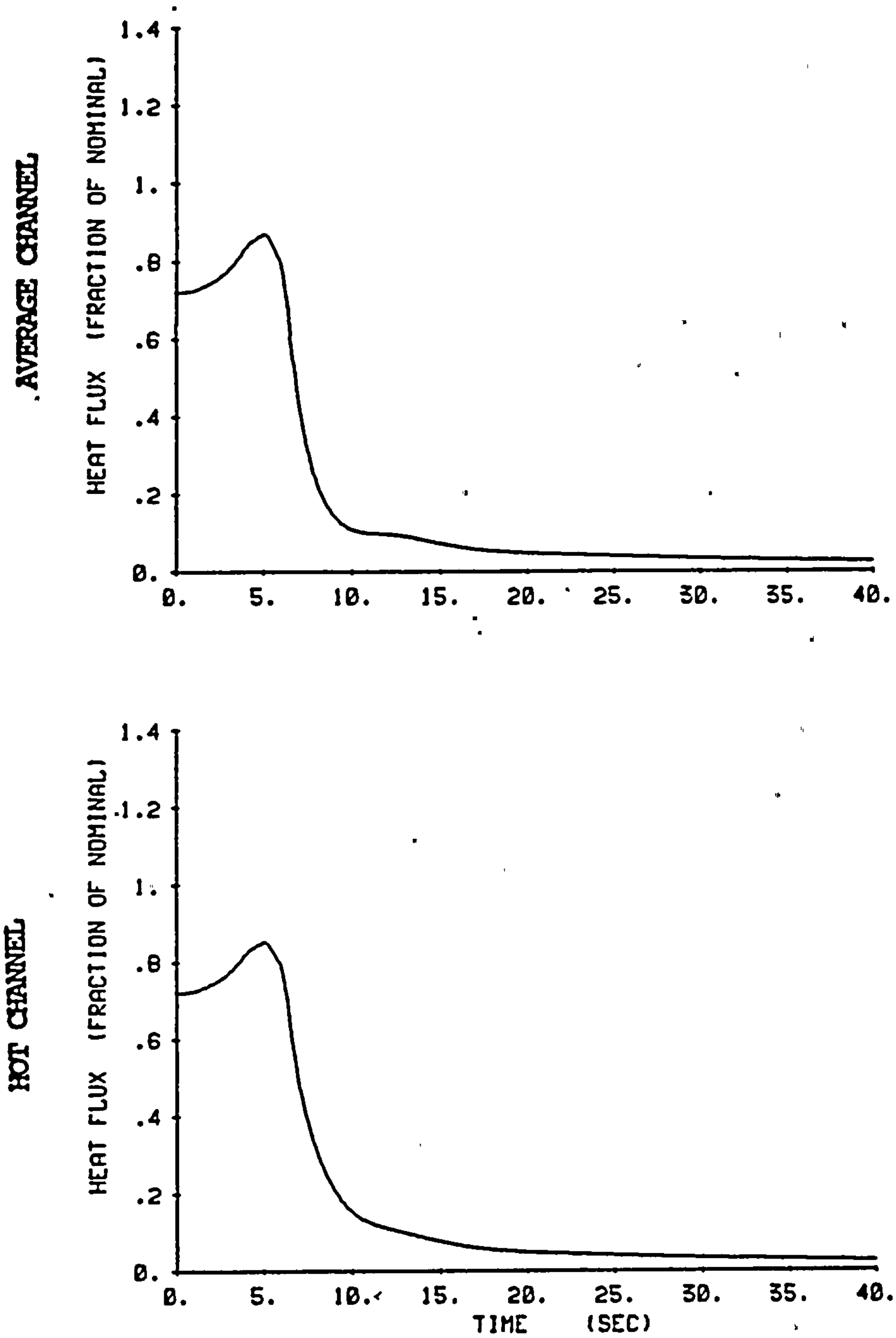
ALL LOOPS OPERATING,
TWO LOOPS COASTING DOWN,
DNBR VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.6-1

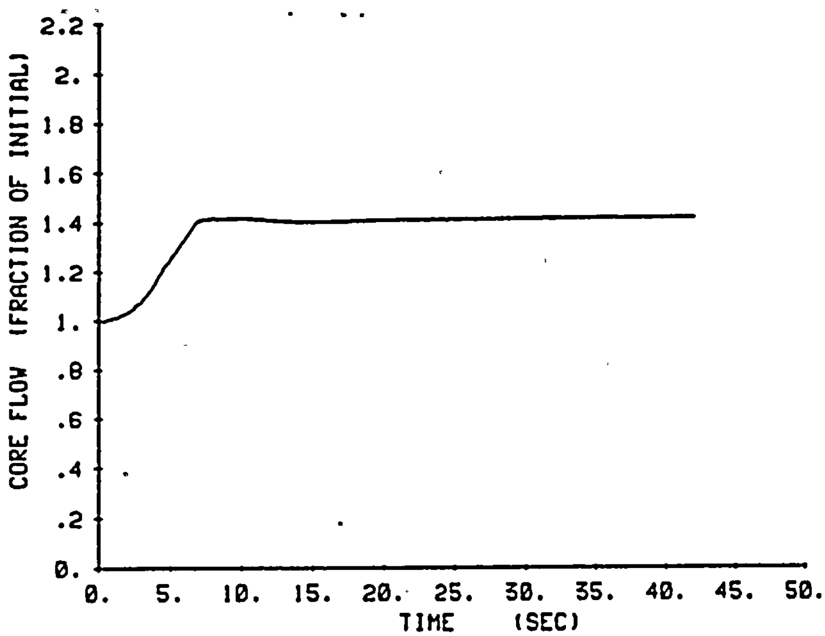
NUCLEAR POWER TRANSIENT DURING STARTUP OF AN INACTIVE LOOP



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.6-2

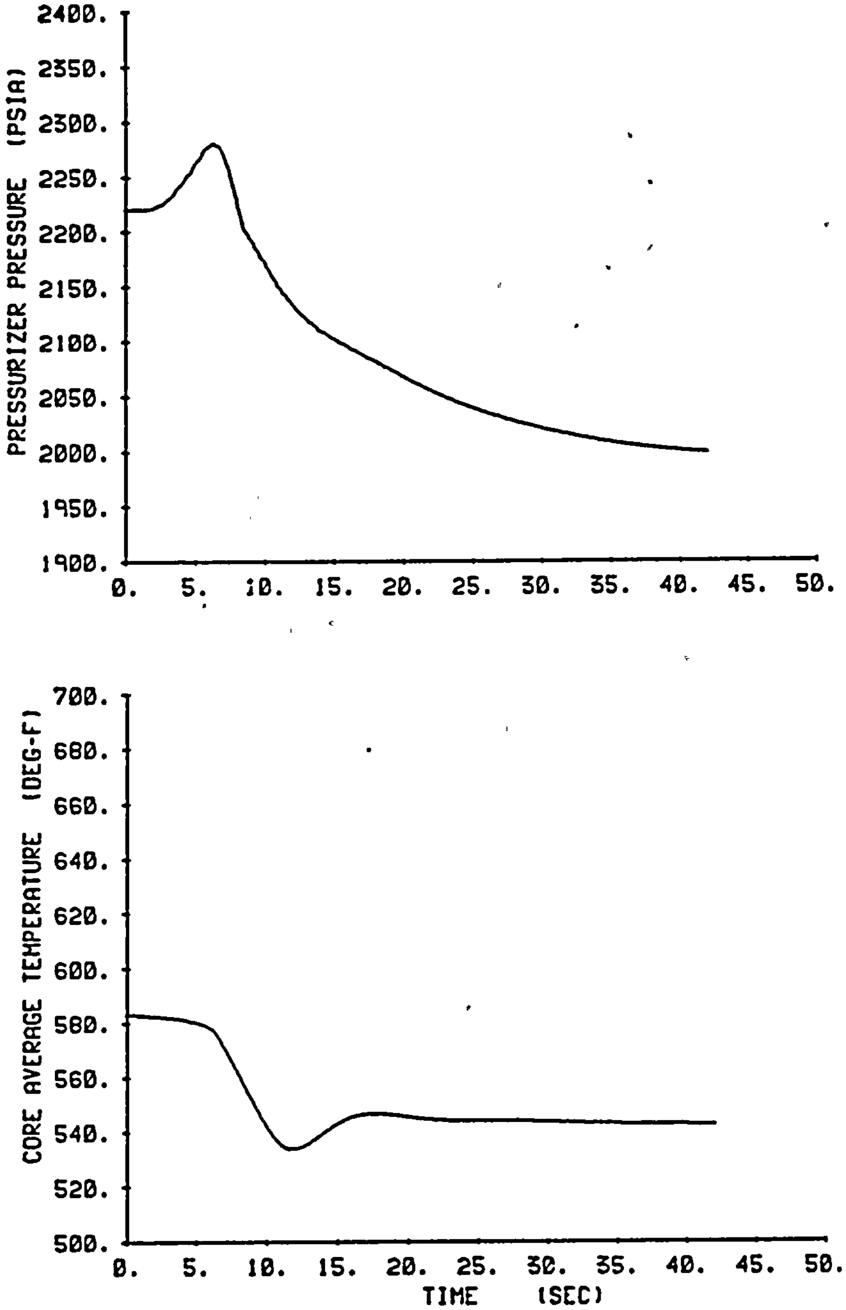
AVERAGE AND HOT CHANNEL HEAT FLUX TRANSIENTS DURING STARTUP OF AN INACTIVE LOOP



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.6-3

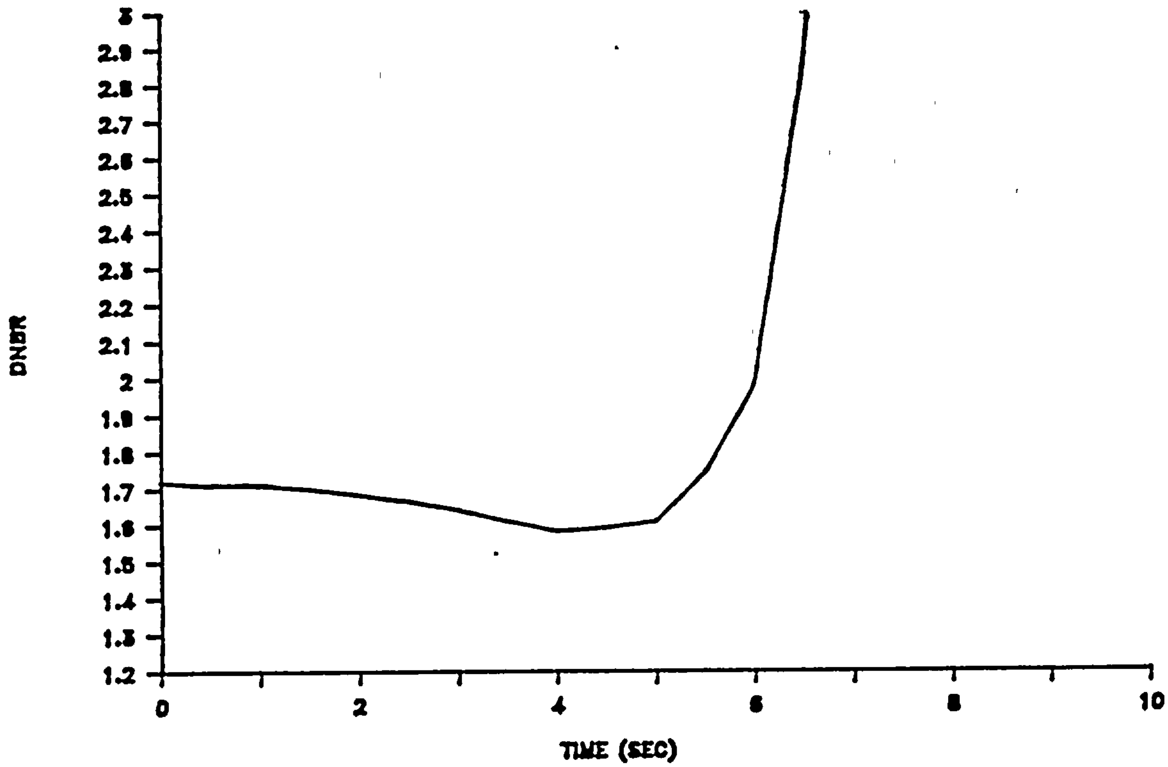
CORE FLOW DURING STARTUP OF AN INACTIVE LOOP



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.6-4

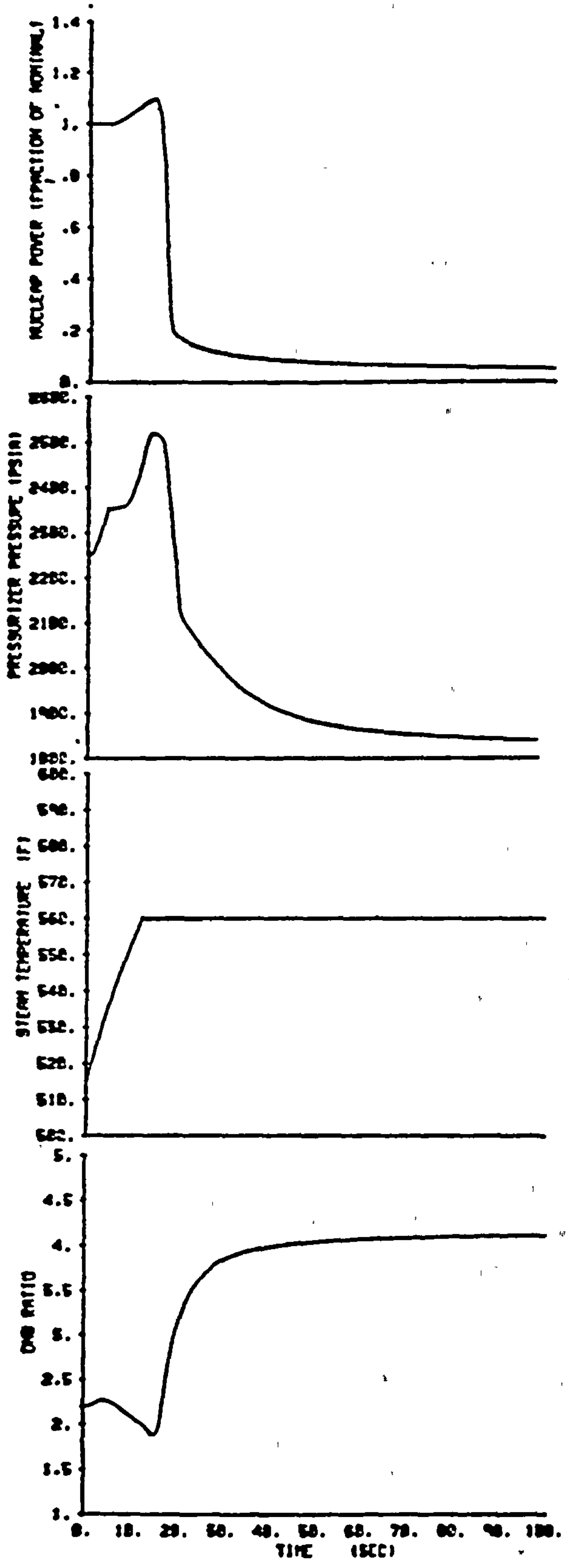
PRESSURIZER PRESSURE TRANSIENT AND CORE AVERAGE TEMPERATURE TRANSIENT DURING STARTUP OF AN INACTIVE LOOP



DIABLO CANYON UNITS 1 AND 2

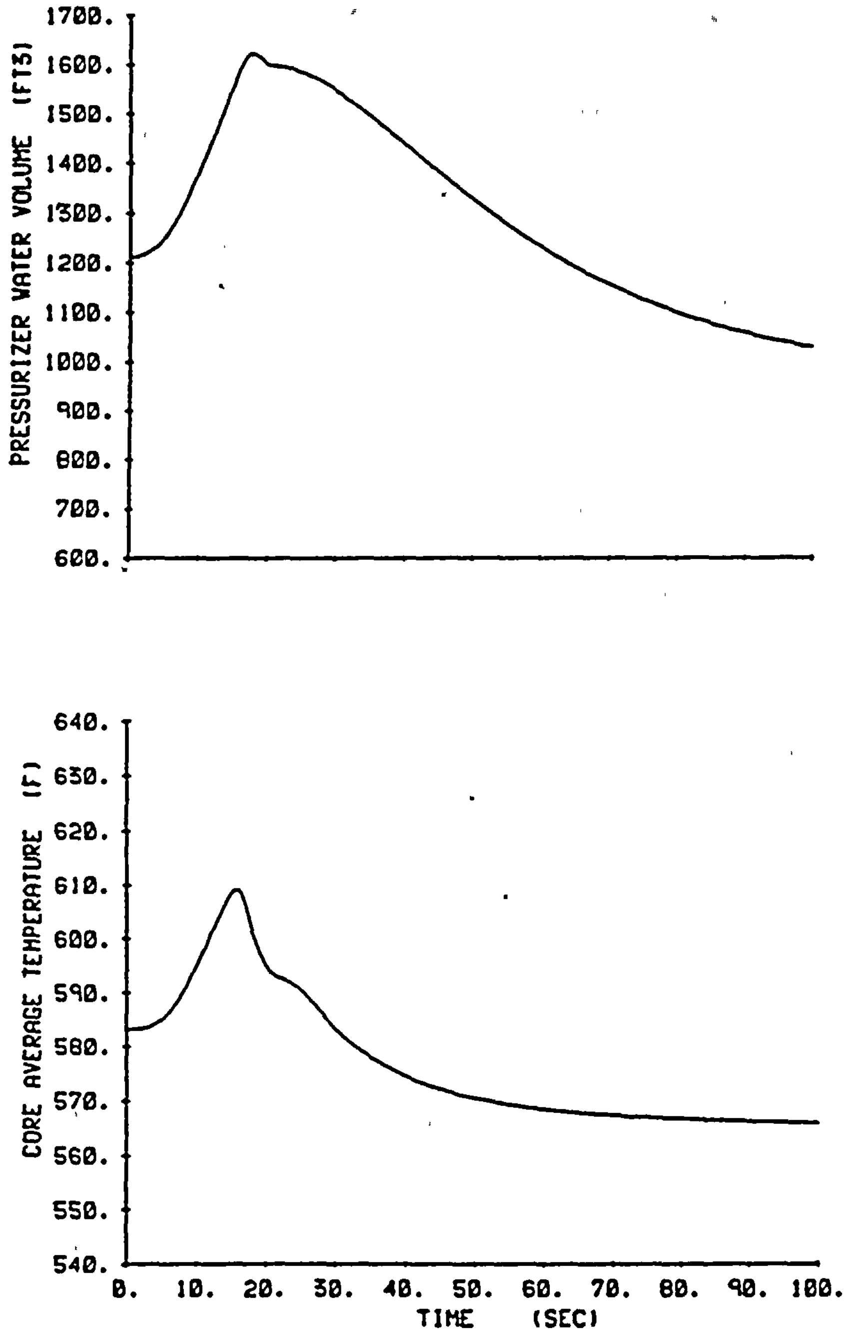
FIGURE 15.2.6-5

DNBR TRANSIENT
DURING STARTUP OF AN INACTIVE LOOP



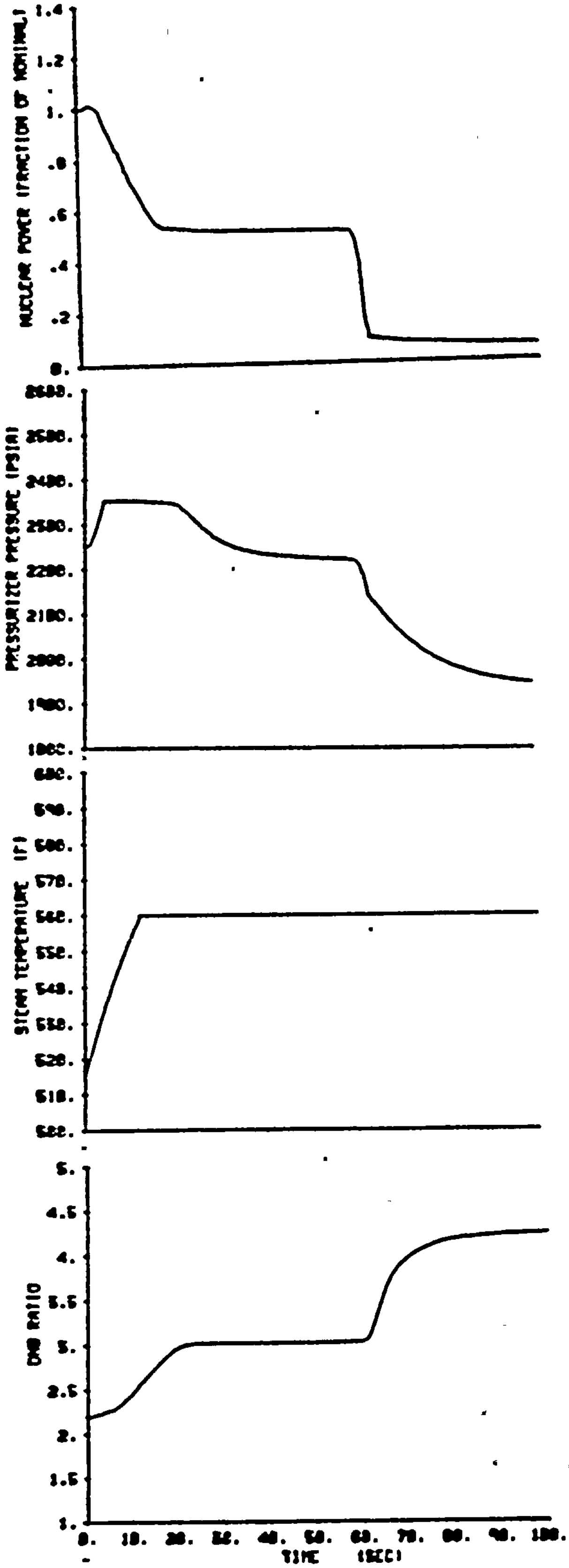
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.7-1
LOSS OF LOAD-WITH PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE. BEGINNING OF LIFE. DNBR, STEAM TEMPERATURE, PRESSURIZER PRESSURE AND NUCLEAR POWER VERSUS TIME



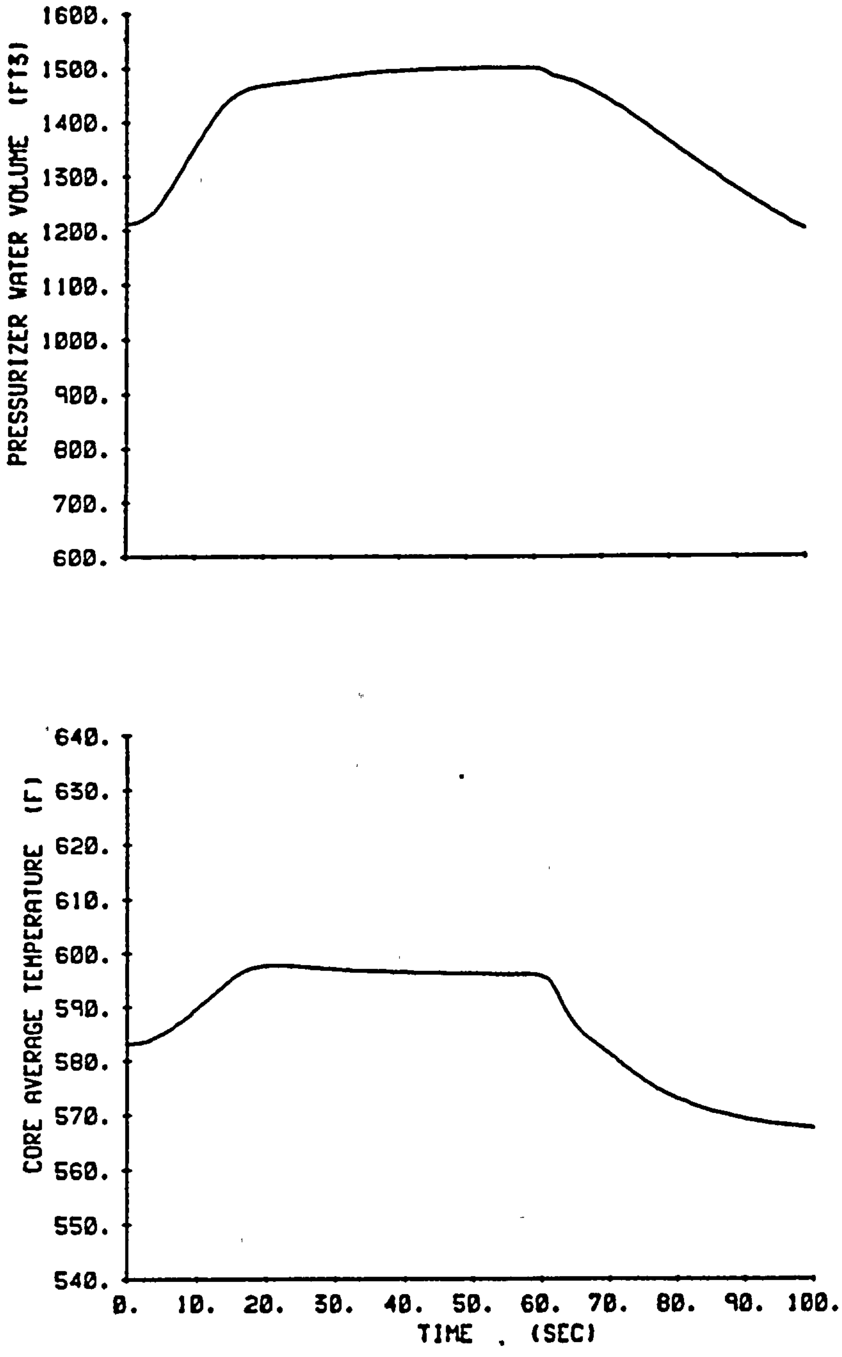
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.7-2
LOSS OF LOAD-WITH PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE, BEGINNING OF LIFE, AVERAGE CORE TEMPERATURE AND PRESSURIZER WATER VOLUME VERSUS TIME



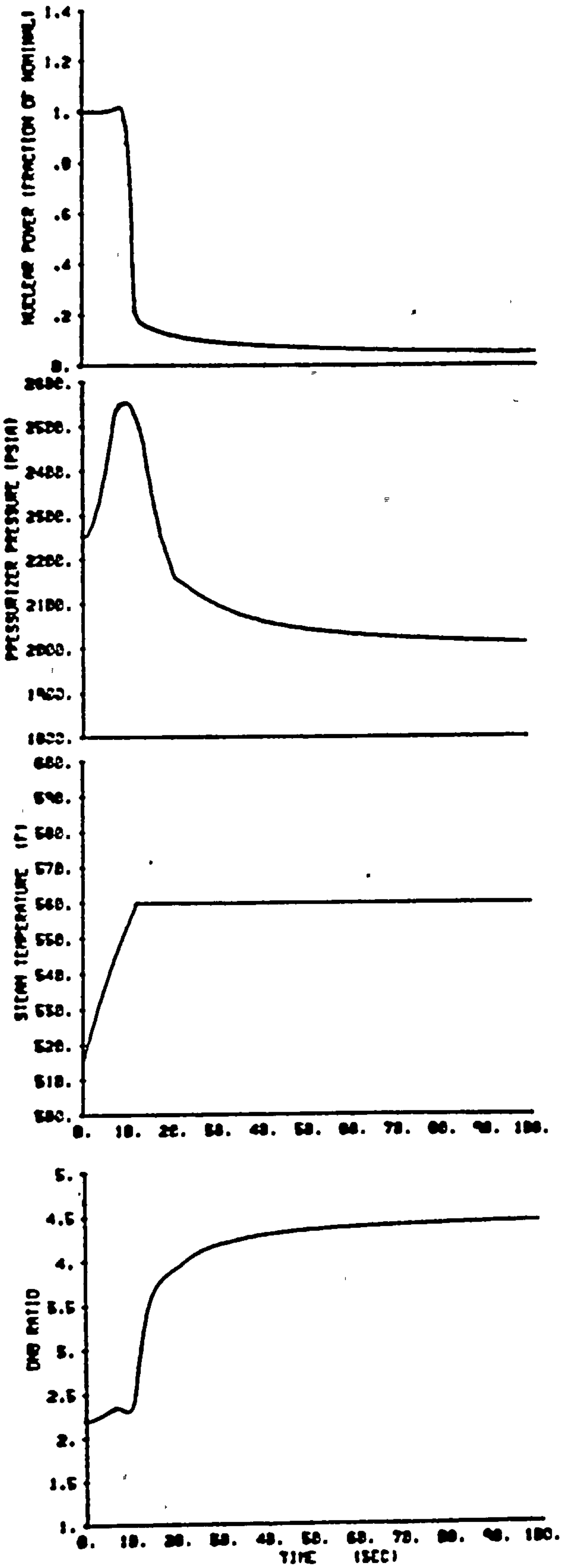
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.7-3
LOSS OF LOAD-WITH PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE. END OF LIFE. DNBR. STEAM TEMPERATURE. PRESSURIZER PRESSURE AND NUCLEAR POWER VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

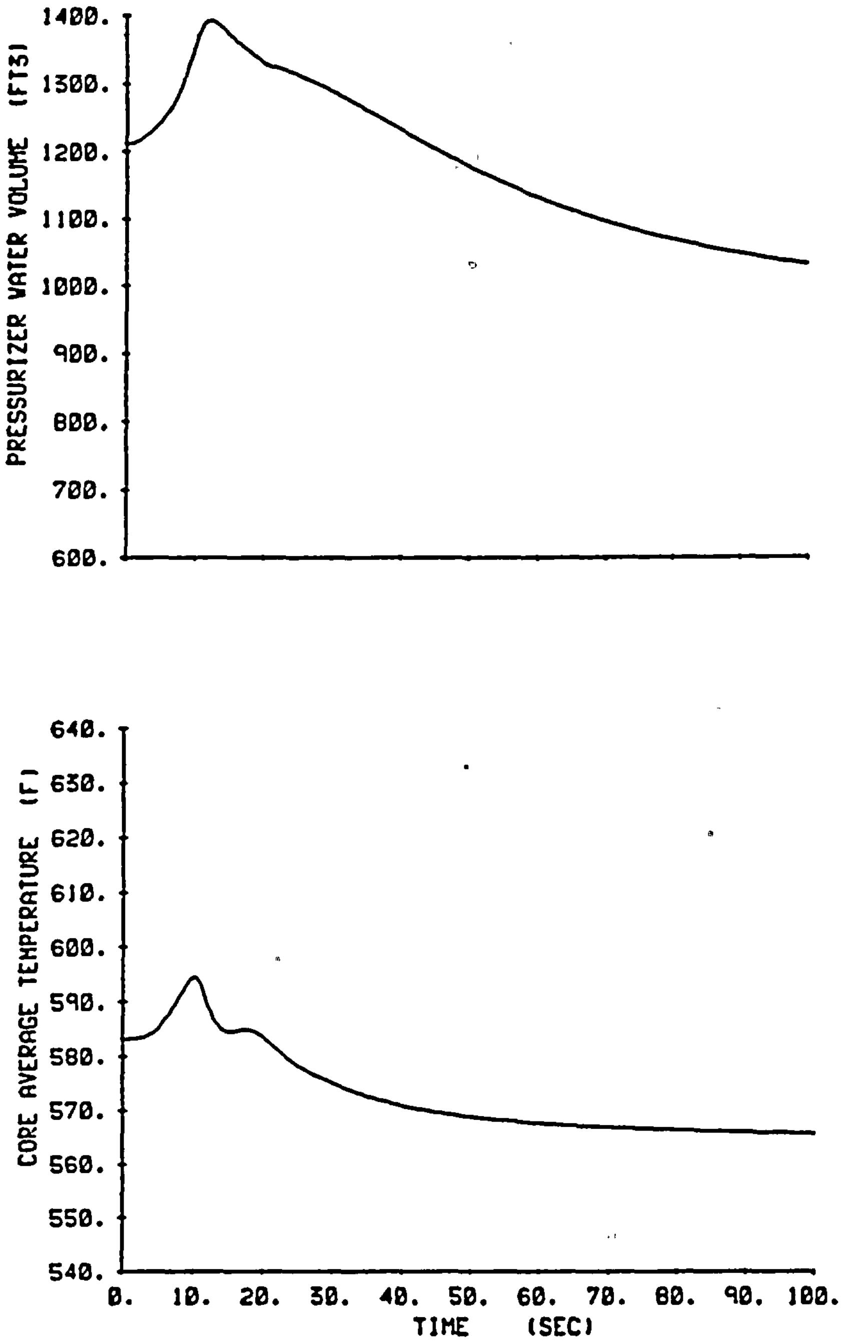
FIGURE 15.2.7-4
LOSS OF LOAD-WITH PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE. END OF LIFE. AVERAGE CORE TEMPERATURE AND PRESSURIZER WATER VOLUME VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

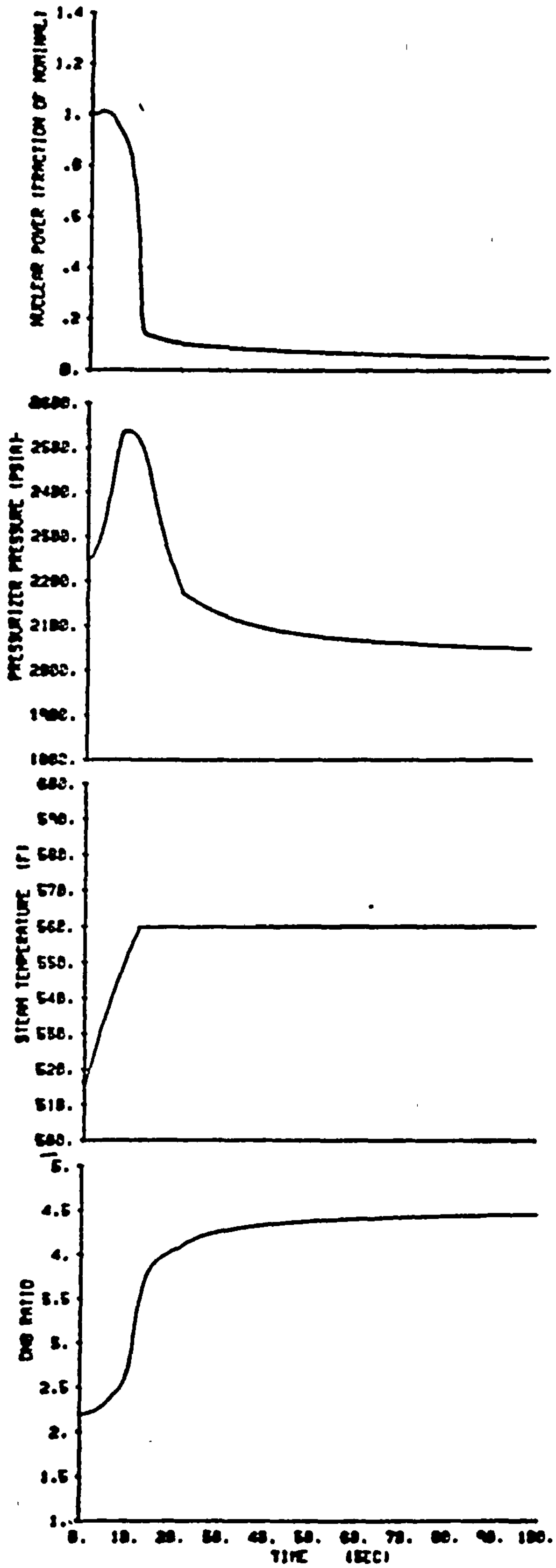
FIGURE 15.2.7-5

LOSS OF LOAD-WITHOUT PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE, BEGINNING OF LIFE, DNBR, STEAM TEMPERATURE, PRESSURIZER PRESSURE AND NUCLEAR POWER VERSUS TIME



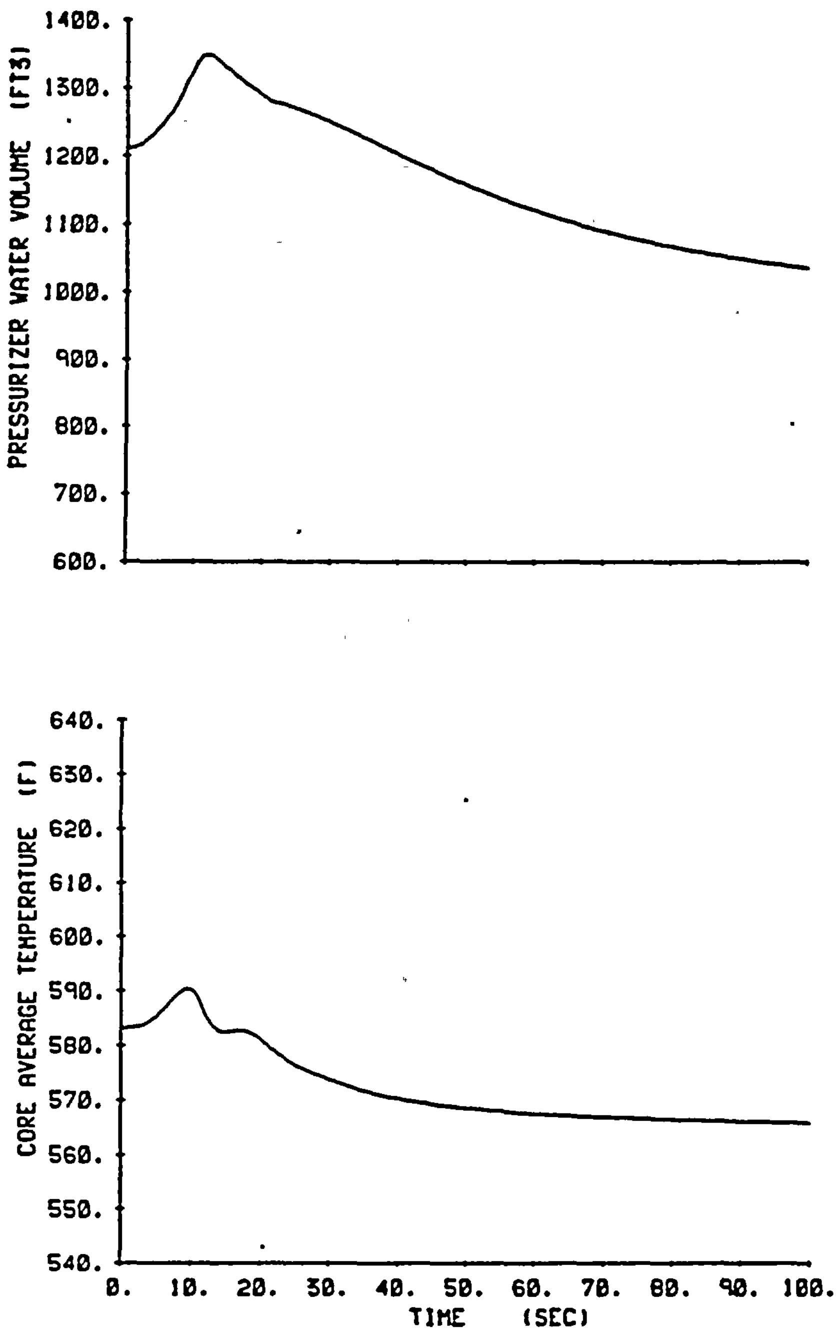
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.7-6
LOSS OF LOAD-WITHOUT PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE, BEGINNING OF LIFE, AVERAGE CORE TEMPERATURE AND PRESSURIZER WATER VOLUME VERSUS TIME



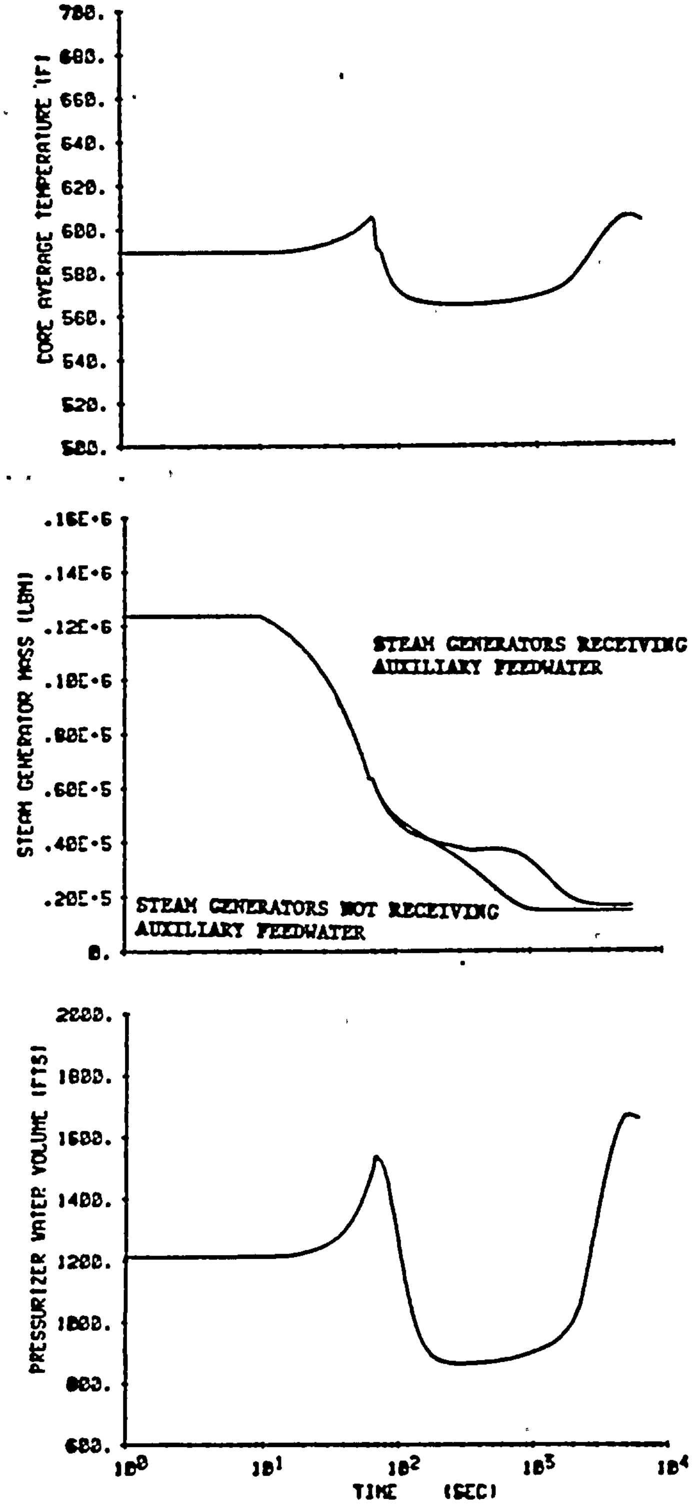
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.7-7
LOSS OF LOAD-WITHOUT PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE, END OF LIFE, DNBR, STEAM TEMPERATURE, PRESSURIZER PRESSURE AND NUCLEAR POWER VERSUS TIME



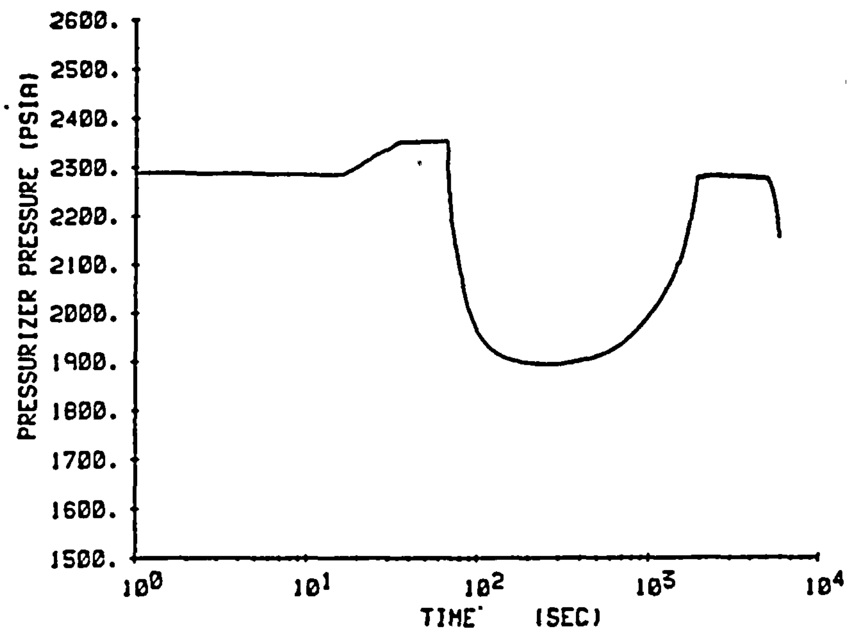
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.7-8
LOSS OF LOAD-WITHOUT PRESSURIZER SPRAY AND POWER OPERATED RELIEF VALVE, END OF LIFE, AVERAGE CORE TEMPERATURE AND PRESSURIZER WATER VOLUME VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

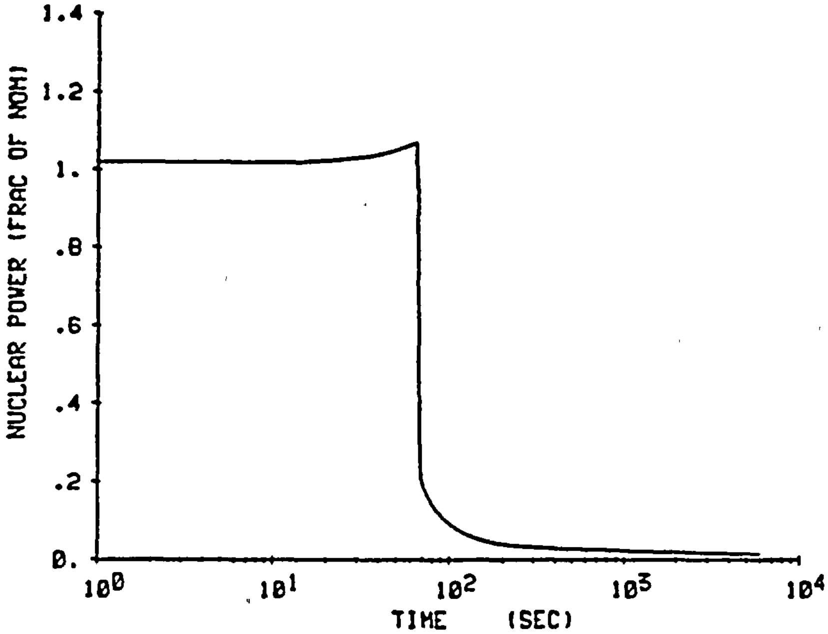
FIGURE 15.2.8-1
LOSS OF NORMAL FEEDWATER ACCIDENT
CORE AVERAGE TEMPERATURE,
STEAM GENERATOR MASS, AND
PRESSURIZER WATER VOLUME
AS A FUNCTION OF TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.8-2

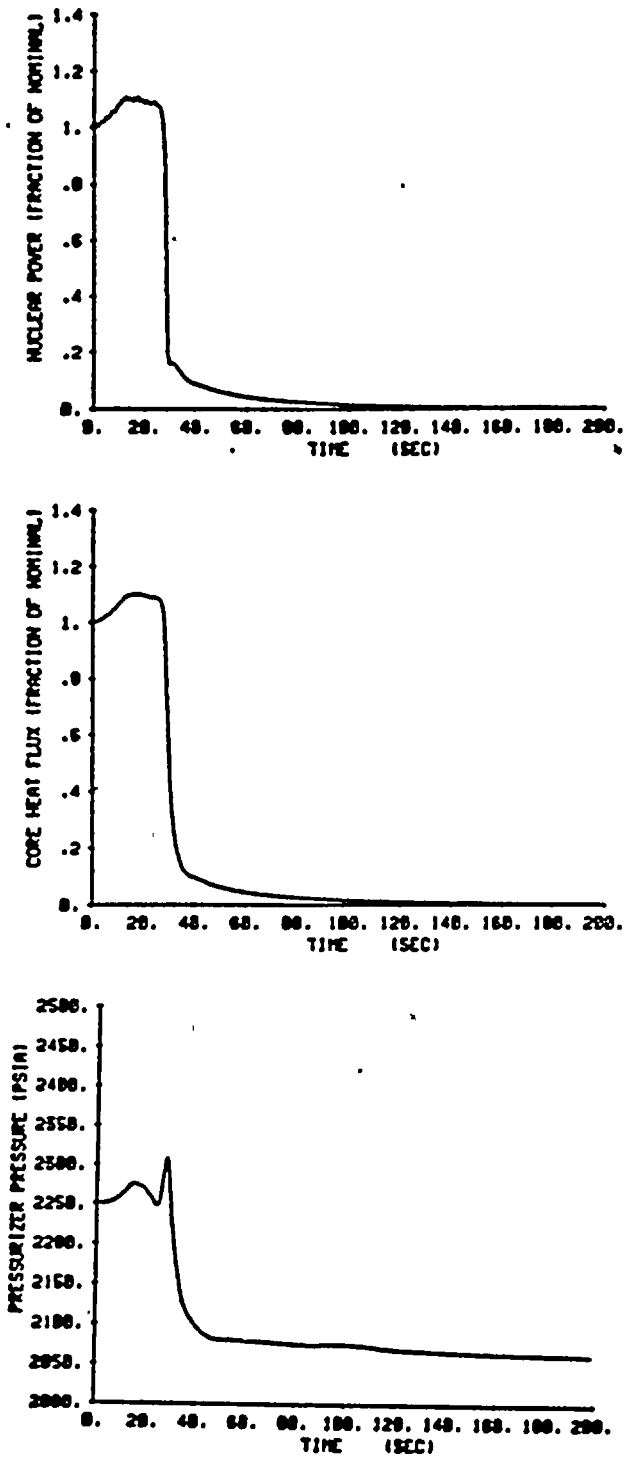
LOSS OF NORMAL FEEDWATER ACCIDENT PRESSURIZER PRESSURE AS A FUNCTION OF TIME



DIABLO CANYON UNITS 1 AND 2

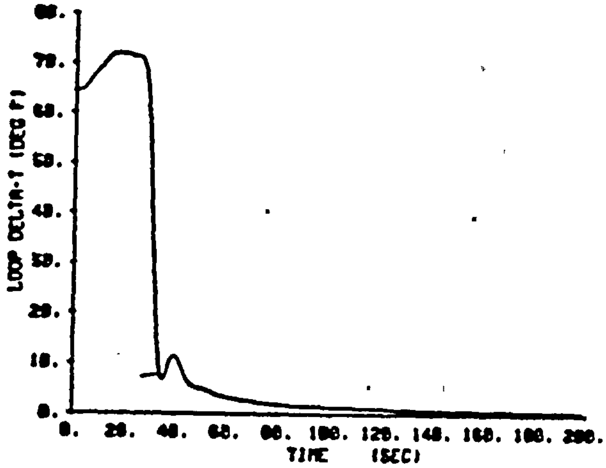
FIGURE 15.2.8-3

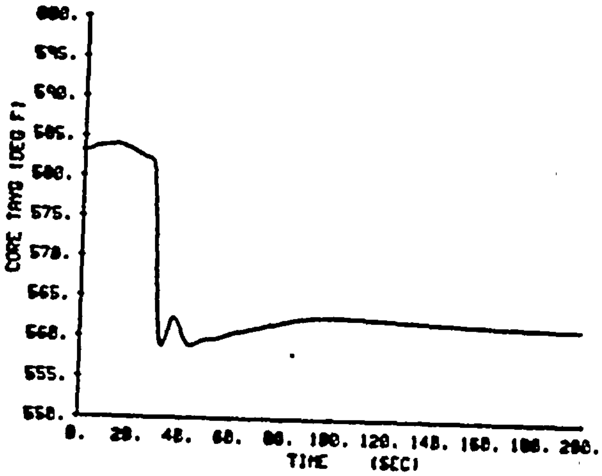
LOSS OF NORMAL FEEDWATER
ACCIDENT NUCLEAR POWER
AS A FUNCTION OF TIME

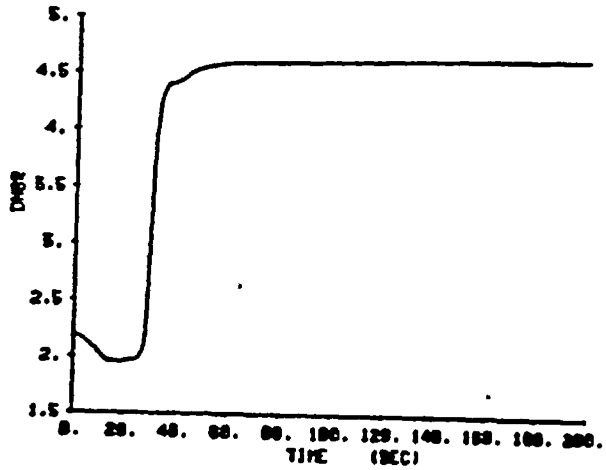


DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.10-1
NUCLEAR POWER, CORE HEAT FLUX
AND PRESSURIZER PRESSURE
TRANSIENTS FOR FEEDWATER
CONTROL VALVE MALFUNCTION.



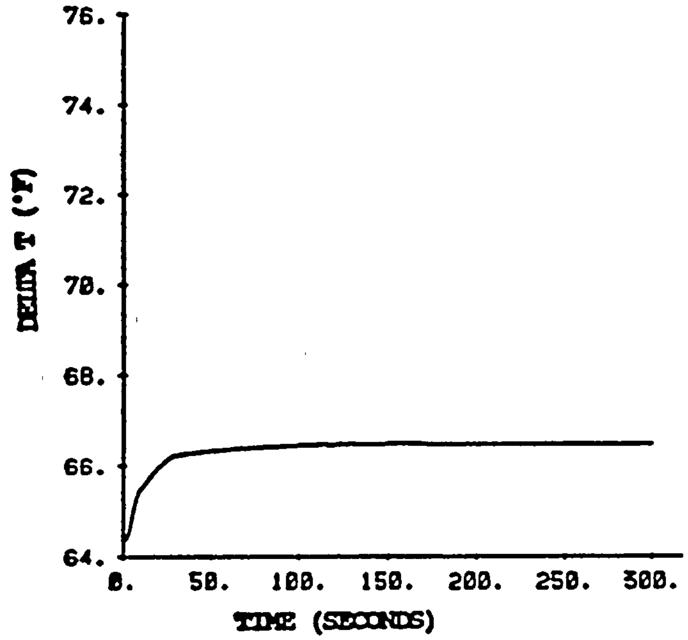


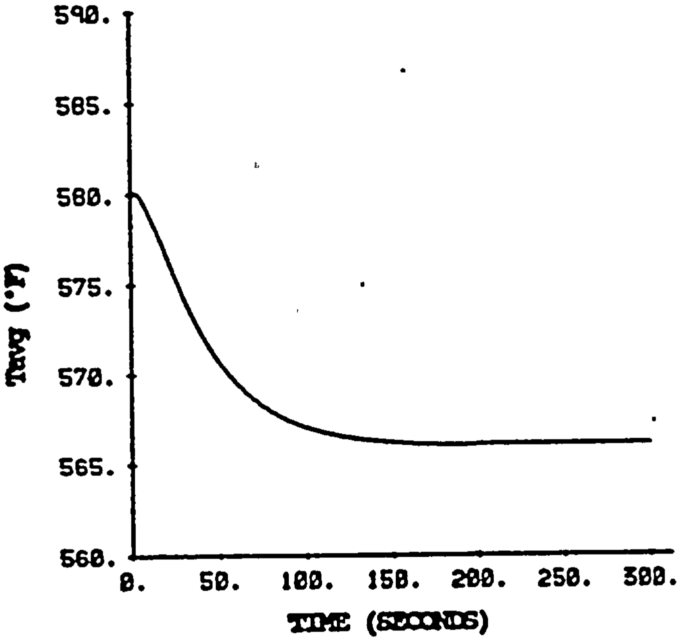


DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.10-2

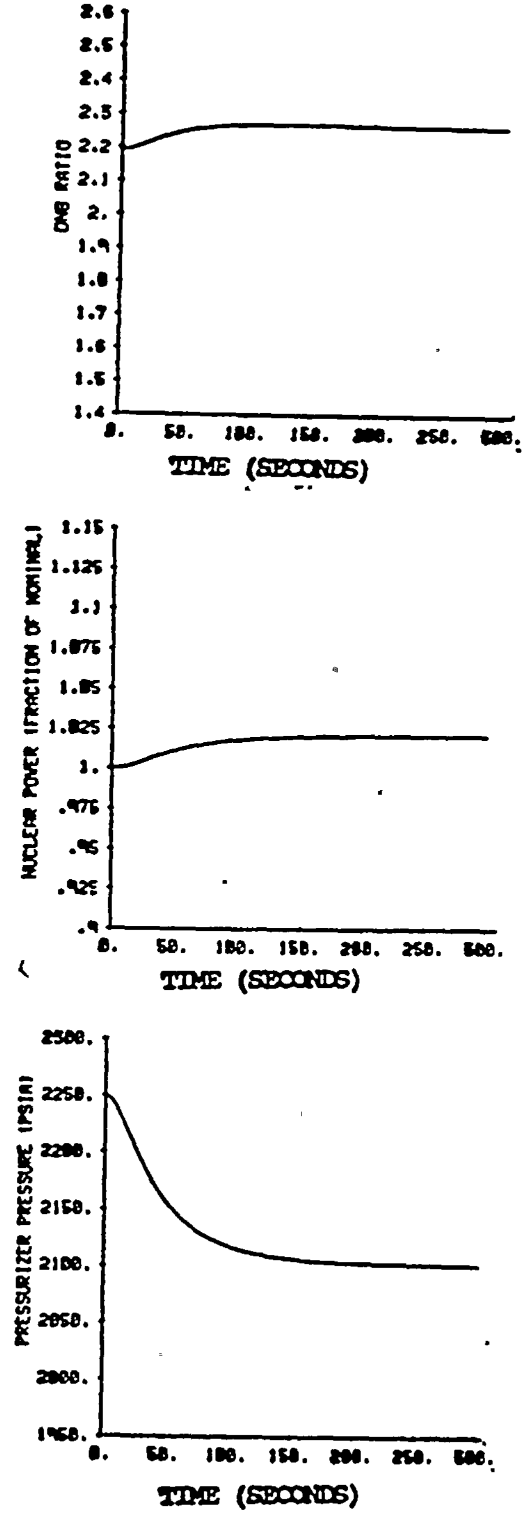
REACTOR COOLANT LOOP DELTA-T, CORE AVERAGE TEMPERATURE AND DNBR TRANSIENTS FOR FEEDWATER CONTROL VALVE MALFUNCTION



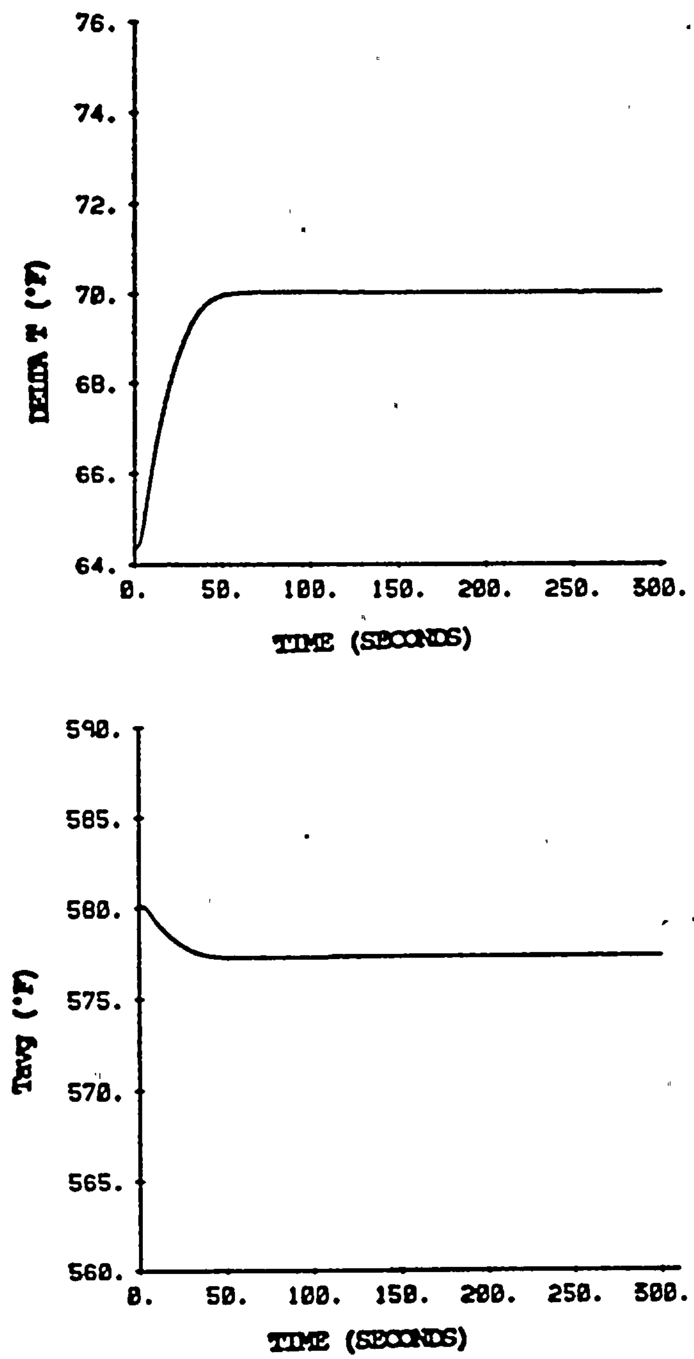


DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.11-1
EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, BEGINNING OF LIFE, (MTC), MINIMUM FEEDBACK, DELTA-T AND TAVG AS A FUNCTION OF TIME

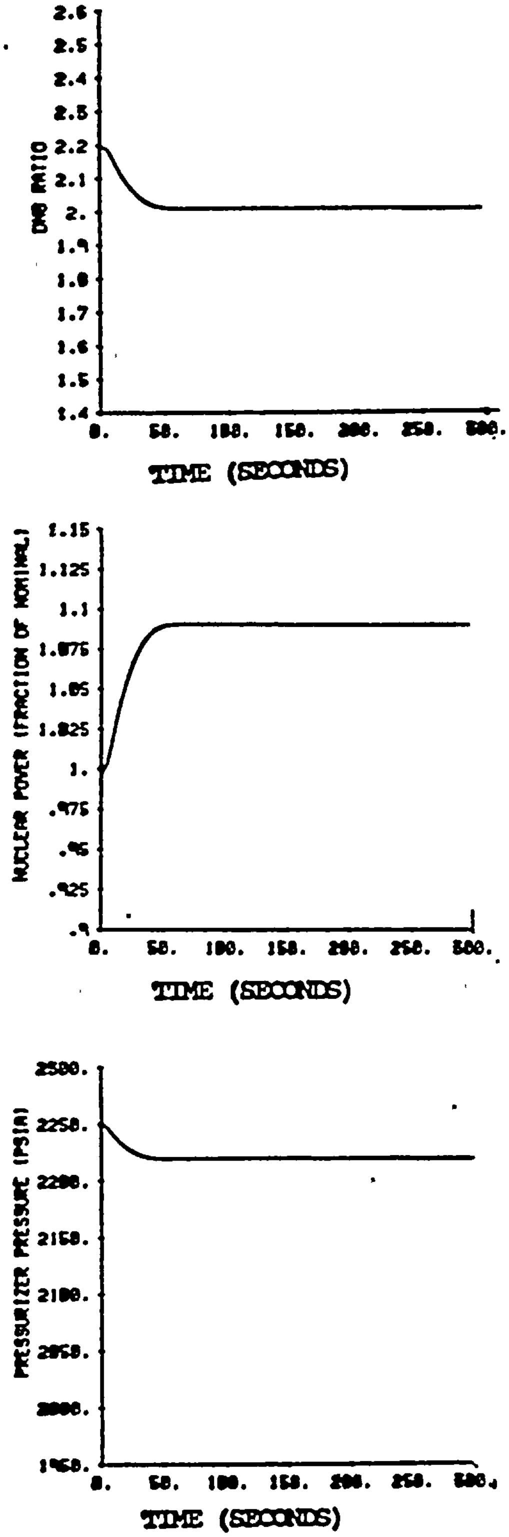


DIABLO CANYON UNITS 1 AND 2
FIGURE 15.2.11-2 EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, BEGINNING OF LIFE, (MTC), MINIMUM FEEDBACK, DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME



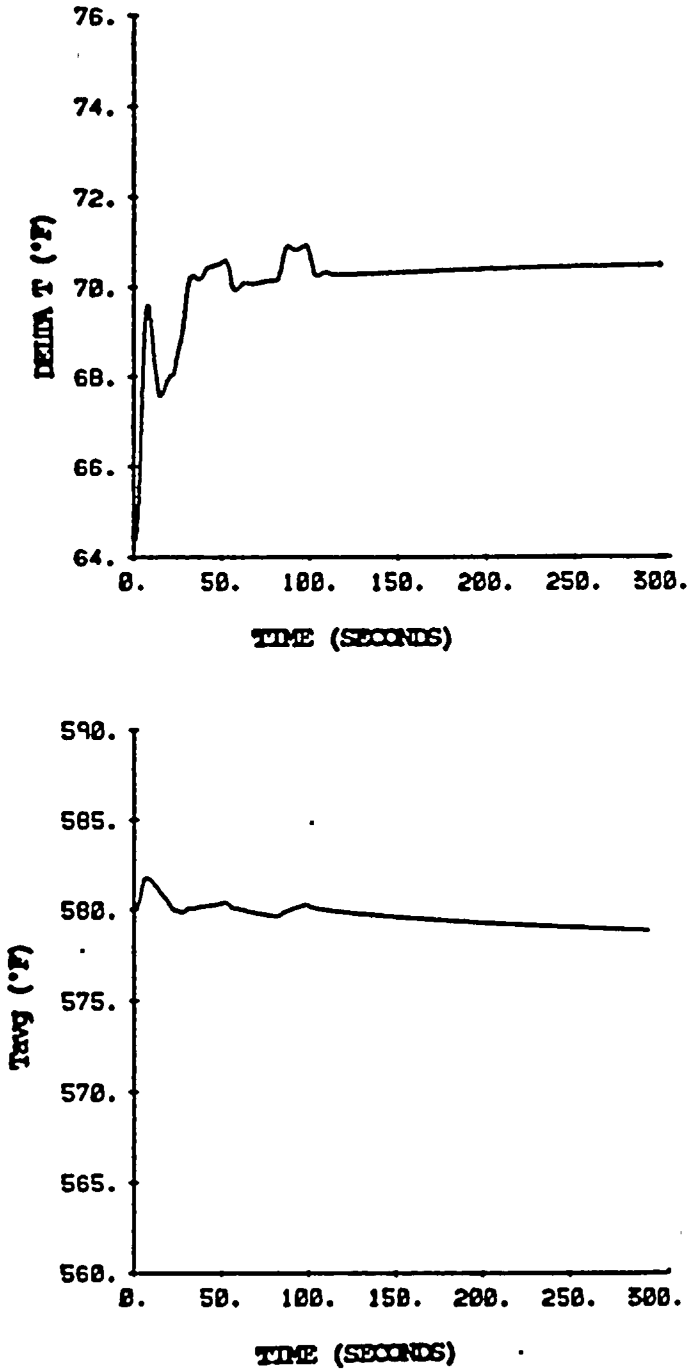
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.11-3
EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, END OF LIFE, (MTC), MAXIMUM FEEDBACK, DELTA-T AND TAVG AS A FUNCTION OF TIME



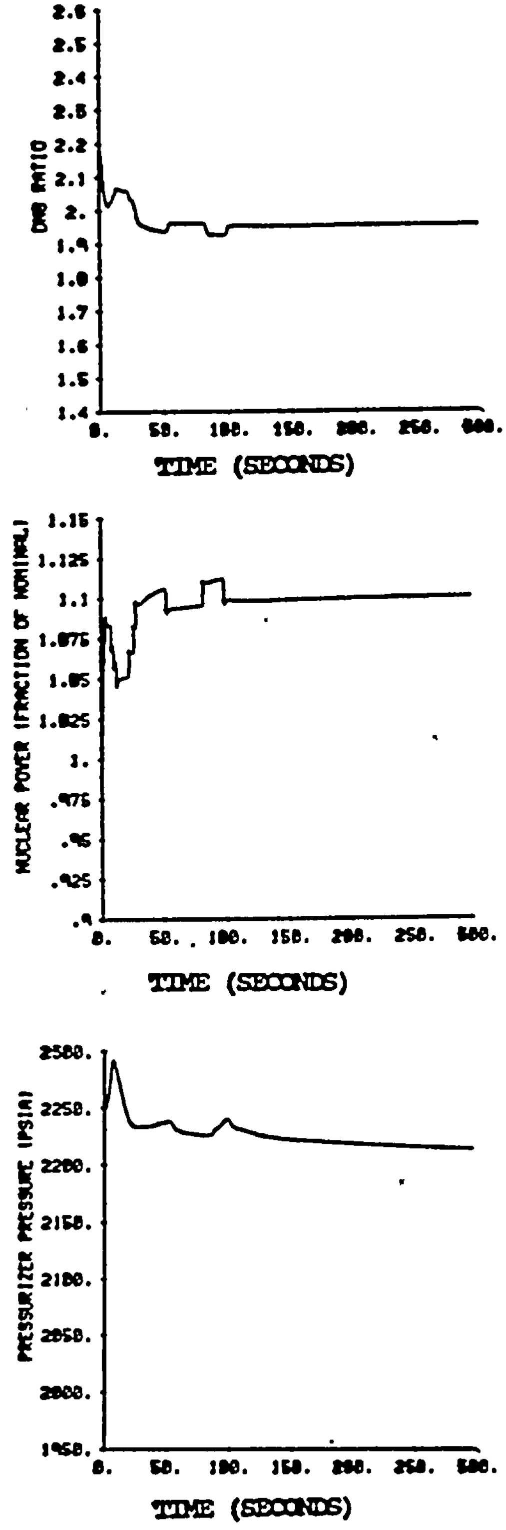
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.11-4
EXCESSIVE LOAD INCREASE WITHOUT CONTROL ACTION, END OF LIFE, (MTC), MAXIMUM FEEDBACK, DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME



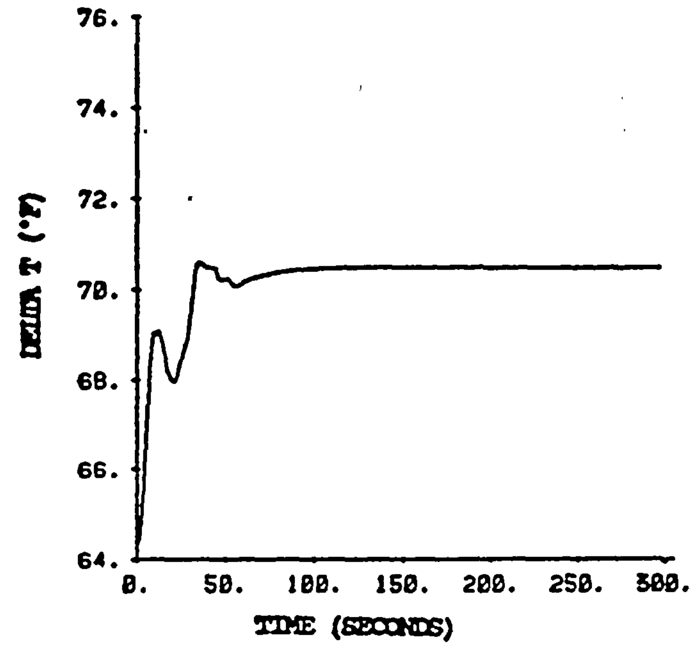
DIABLO CANYON UNITS 1 AND 2

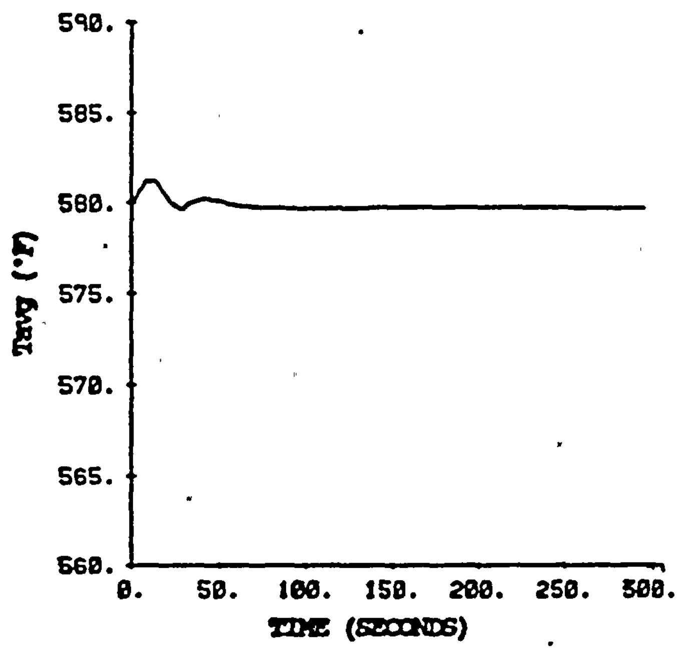
FIGURE 15.2.11-5
EXCESSIVE LOAD INCREASE WITH
REACTOR CONTROL. BEGINNING OF LIFE.
(MTC). MINIMUM FEEDBACK.
DELTA-T AND TAVG
AS A FUNCTION OF TIME



DIABLO CANYON UNITS 1 AND 2

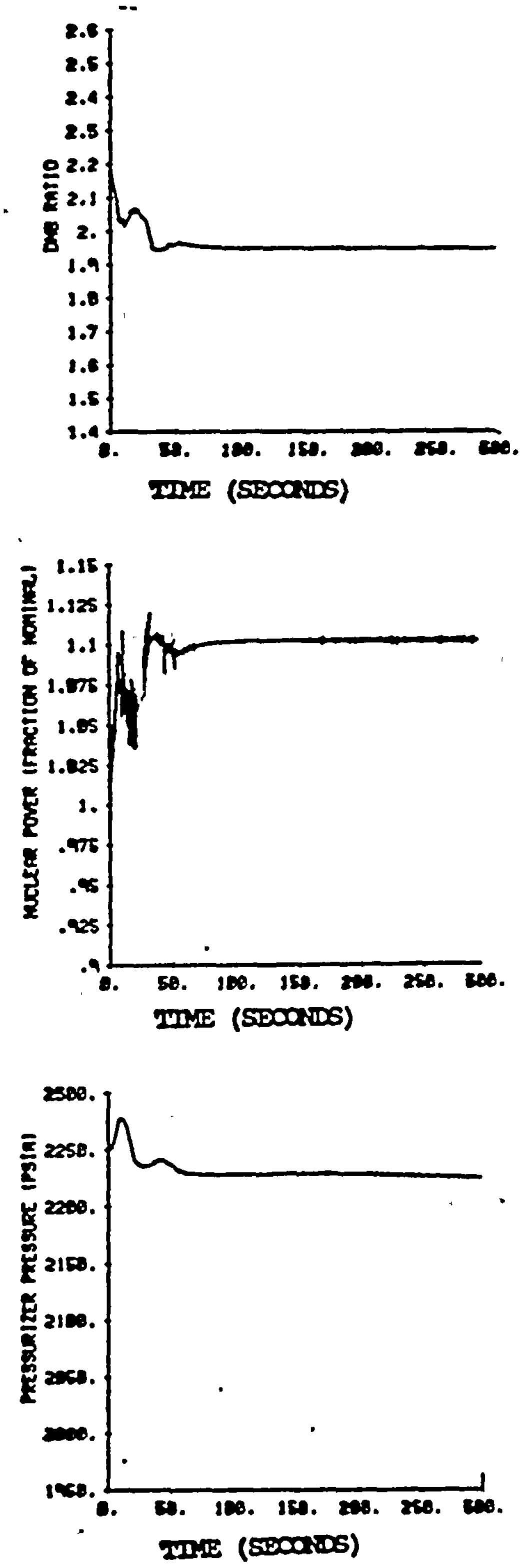
FIGURE 15.2.11-6
EXCESSIVE LOAD INCREASE WITH REACTOR CONTROL, BEGINNING OF LIFE, (MTC), MINIMUM FEEDBACK, DNBR, NUCLEAR POWER AND PRESSURIZER PRESSURE AS A FUNCTION OF TIME





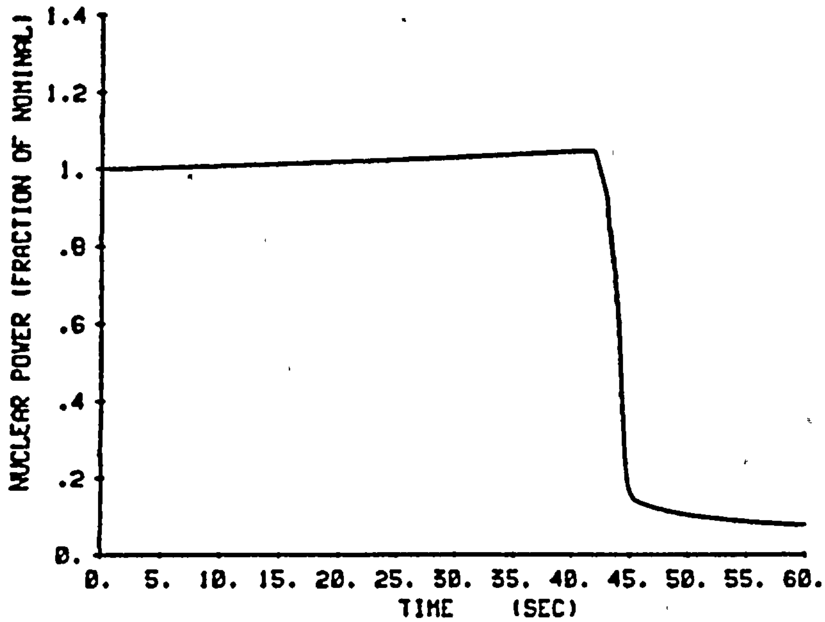
DIABLO CANYON UNITS 1 AND 2

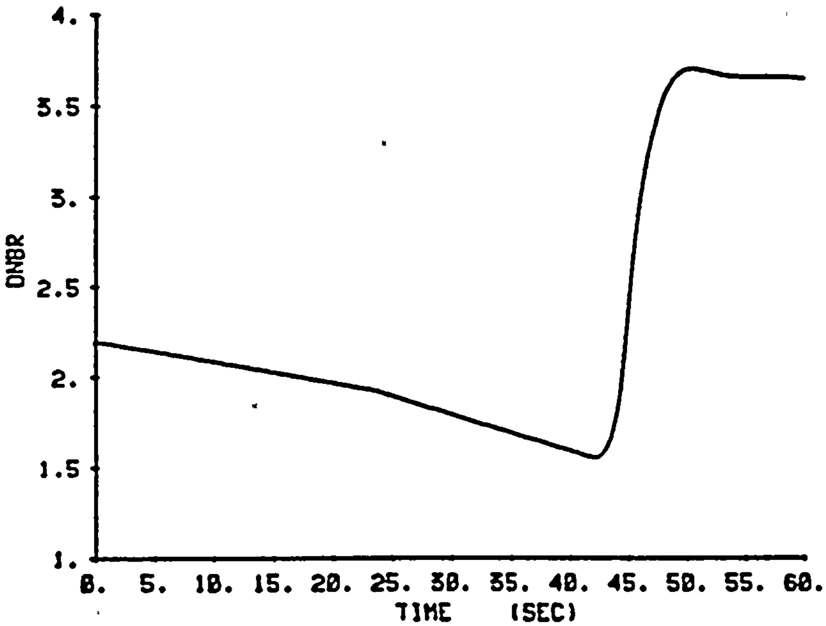
FIGURE 15.2.11-7
EXCESSIVE LOAD INCREASE WITH
REACTOR CONTROL. END OF LIFE.
(MTC). MAXIMUM FEEDBACK.
DELTA-T AND TAVG
AS A FUNCTION OF TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.11-8
EXCESSIVE LOAD INCREASE WITH
REACTOR CONTROL, END OF LIFE,
(MTC), MAXIMUM FEEDBACK, DNBR,
NUCLEAR POWER AND PRESSURIZER
PRESSURE AS A FUNCTION OF TIME

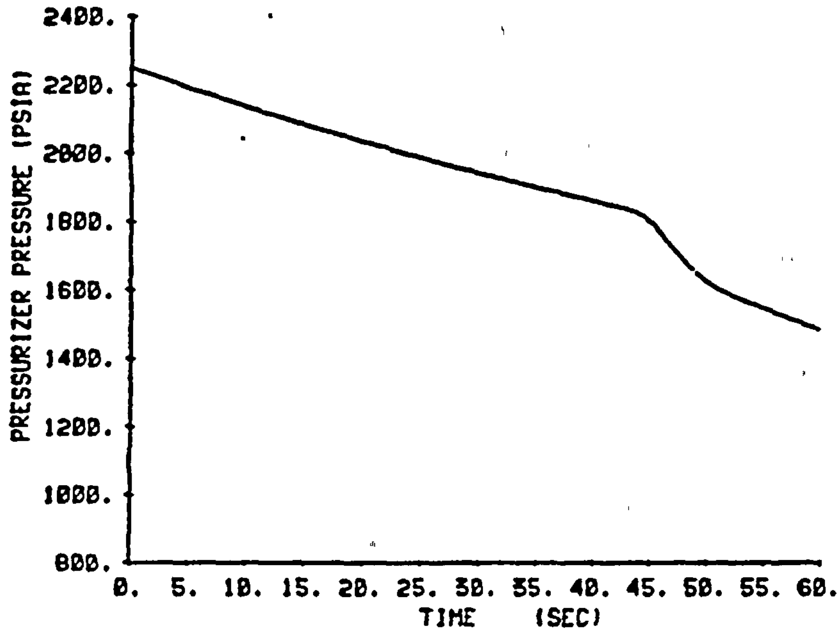


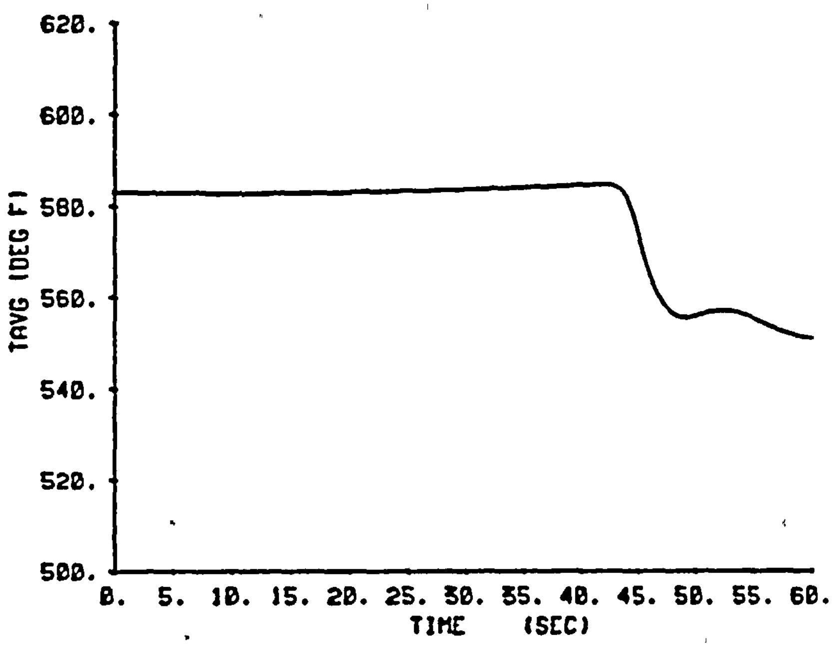


DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.12-1

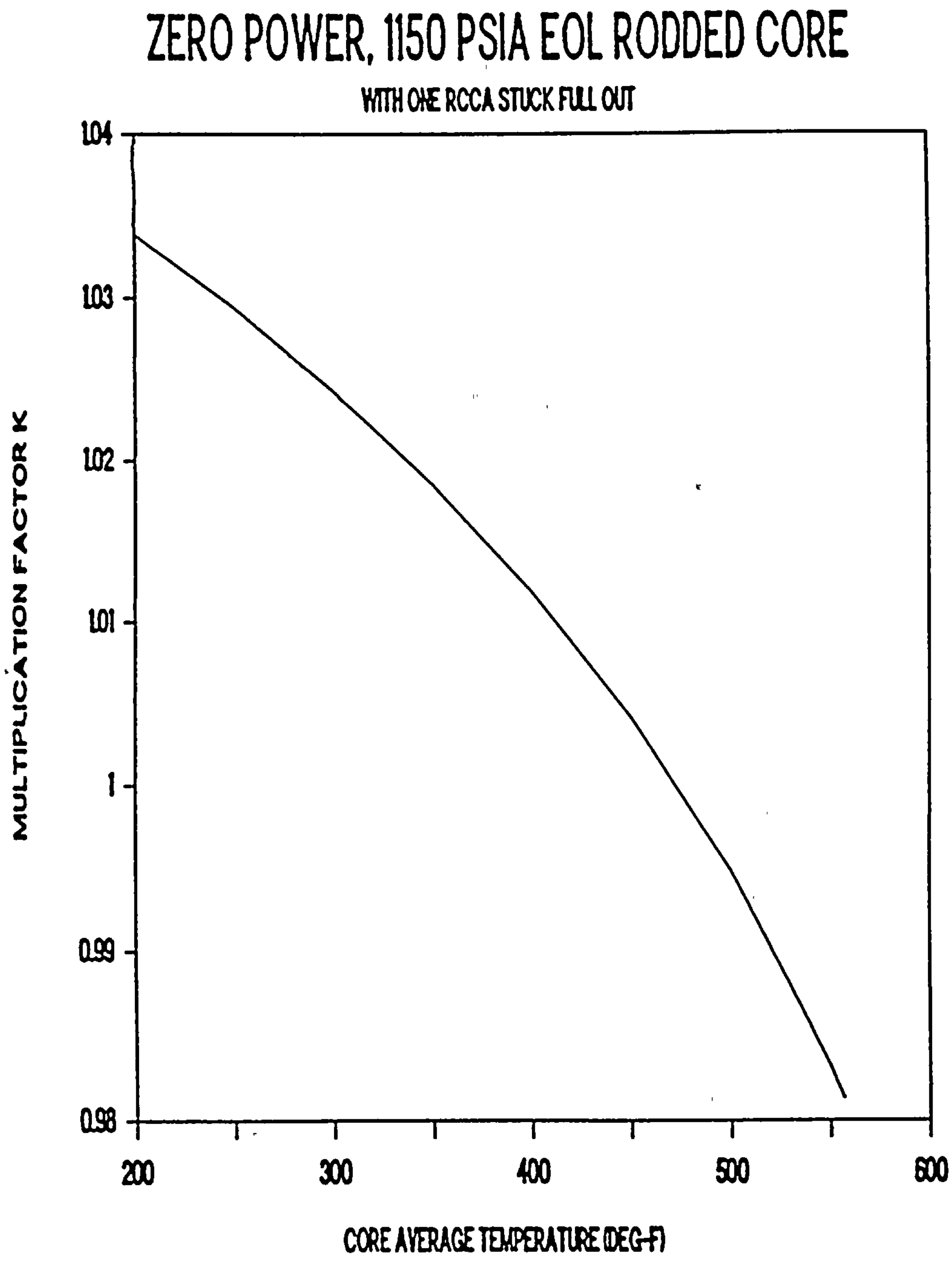
NUCLEAR POWER AND DNBR TRANSIENTS FOR ACCIDENTAL DEPRESSURIZATION OF REACTOR COOLANT SYSTEM





DIABLO CANYON UNITS 1 AND 2

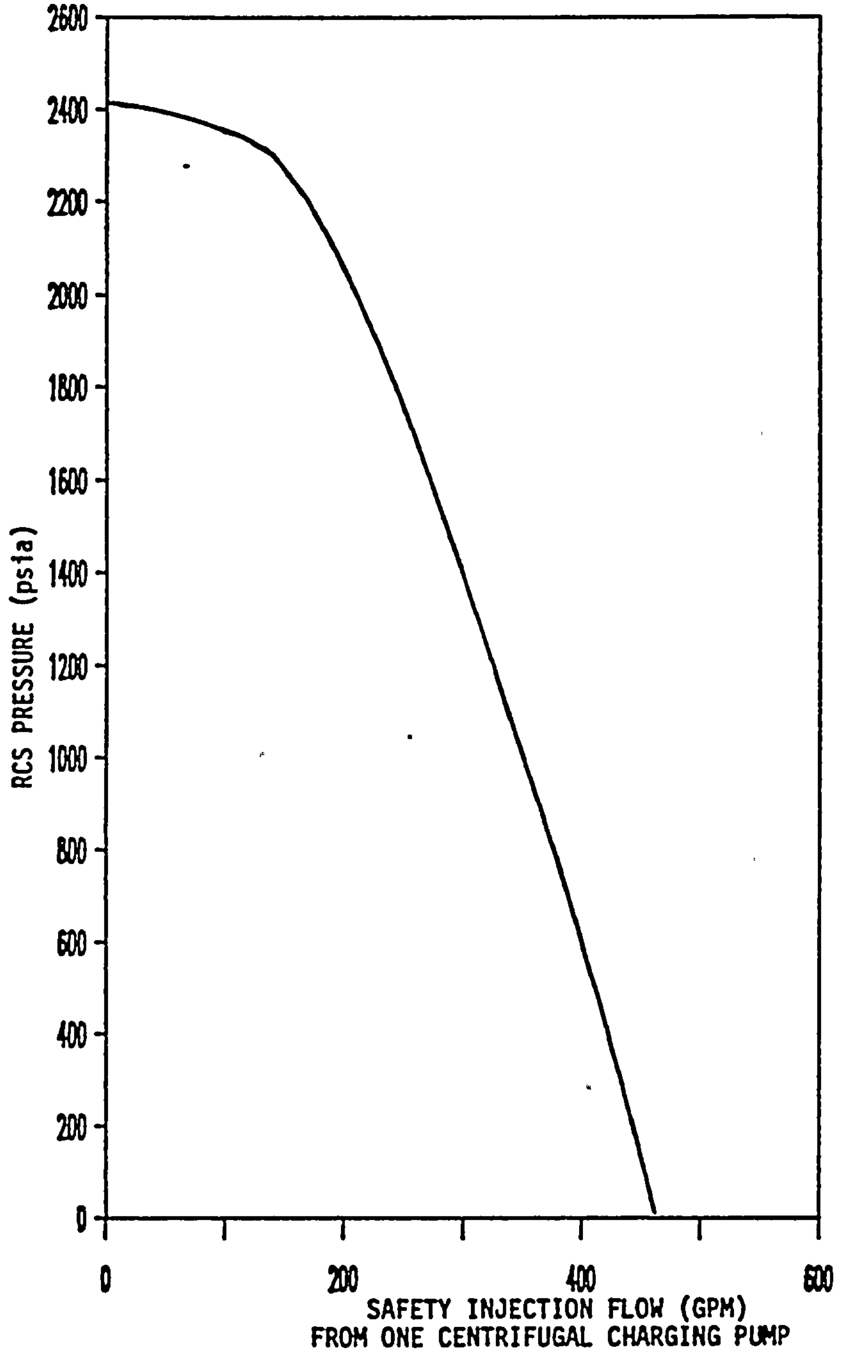
FIGURE 15.2.12-2
PRESSURIZER PRESSURE AND
VESSEL AVERAGE TEMPERATURE
TRANSIENTS FOR ACCIDENTAL
DEPRESSURIZATION OF REACTOR
COOLANT SYSTEM



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.13-1

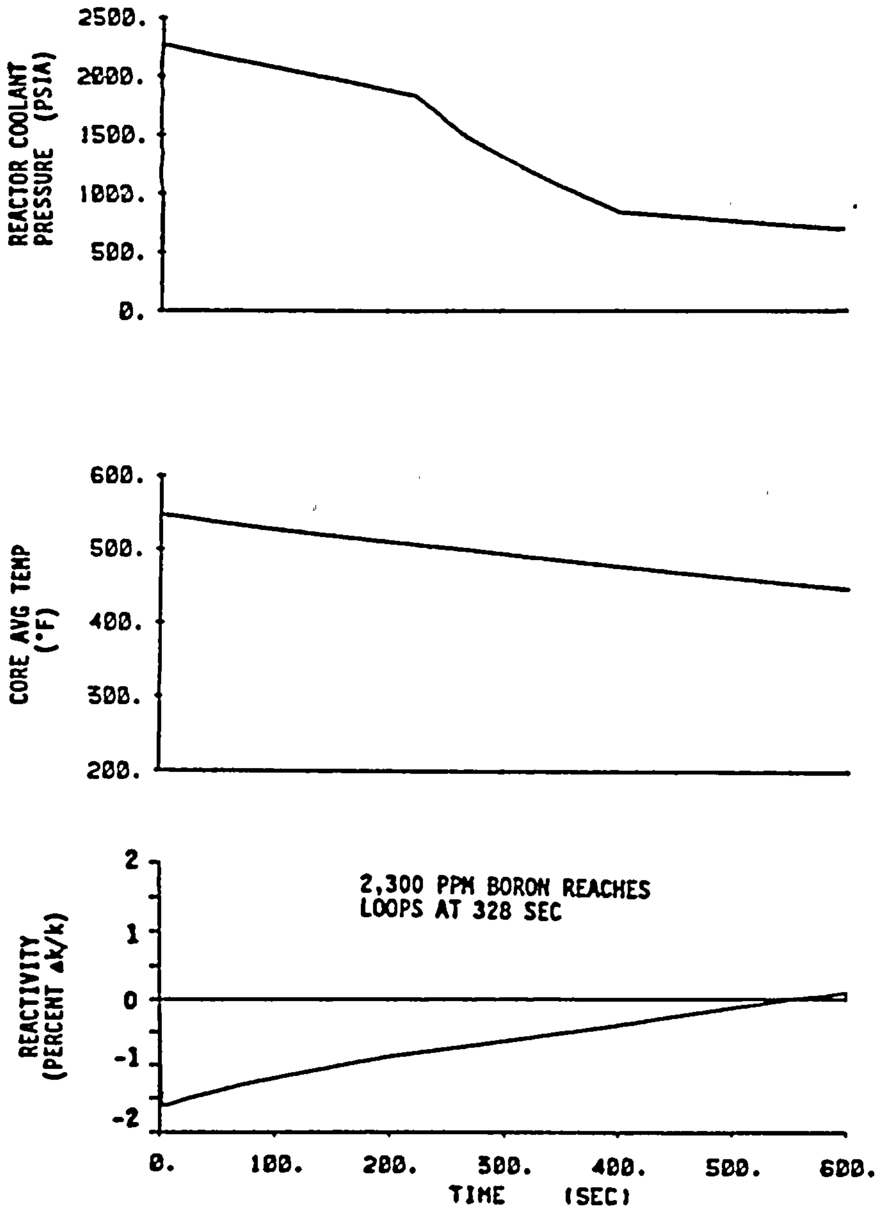
VARIATION OF KEFF WITH
CORE TEMPERATURE



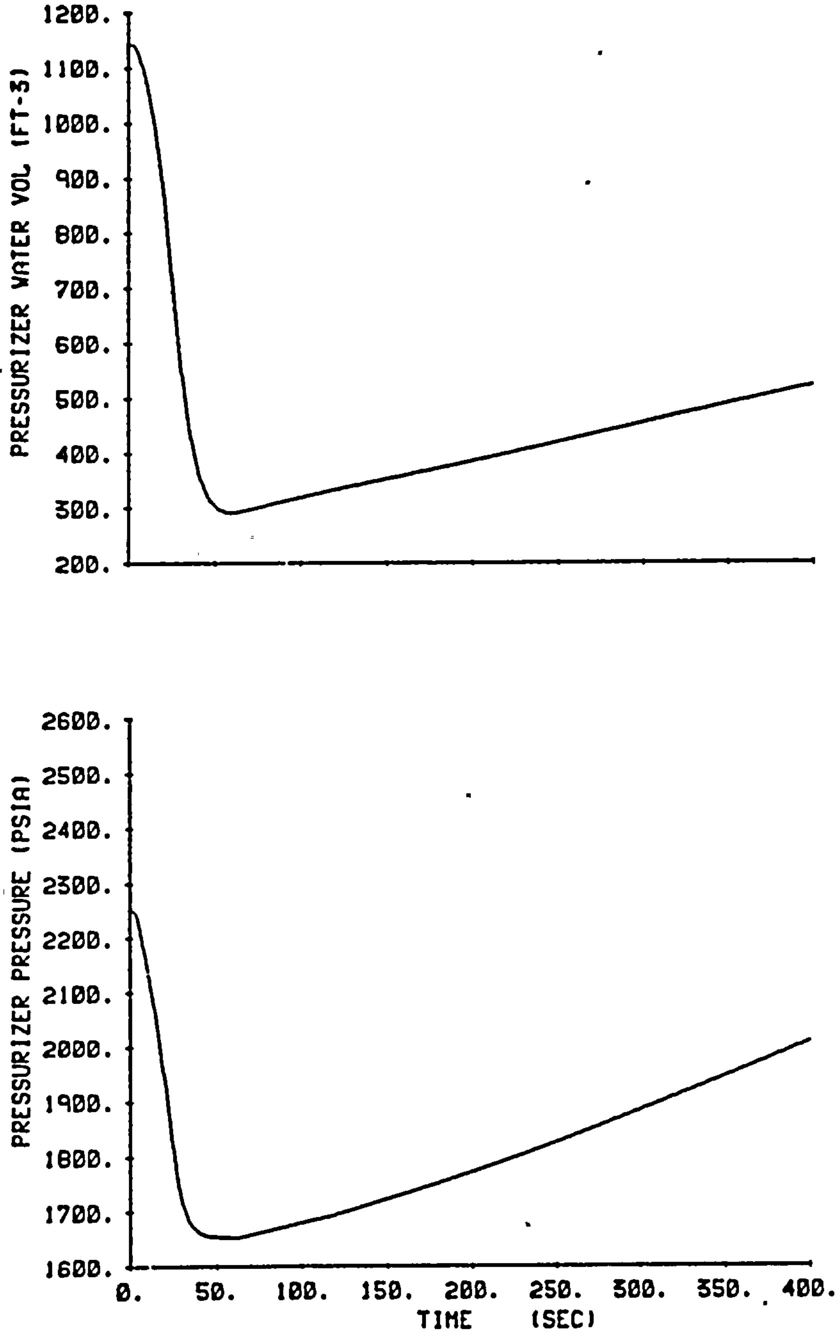
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.13-2

SAFETY INJECTION CURVE
MAIN STREAM DEPRESSURIZATION

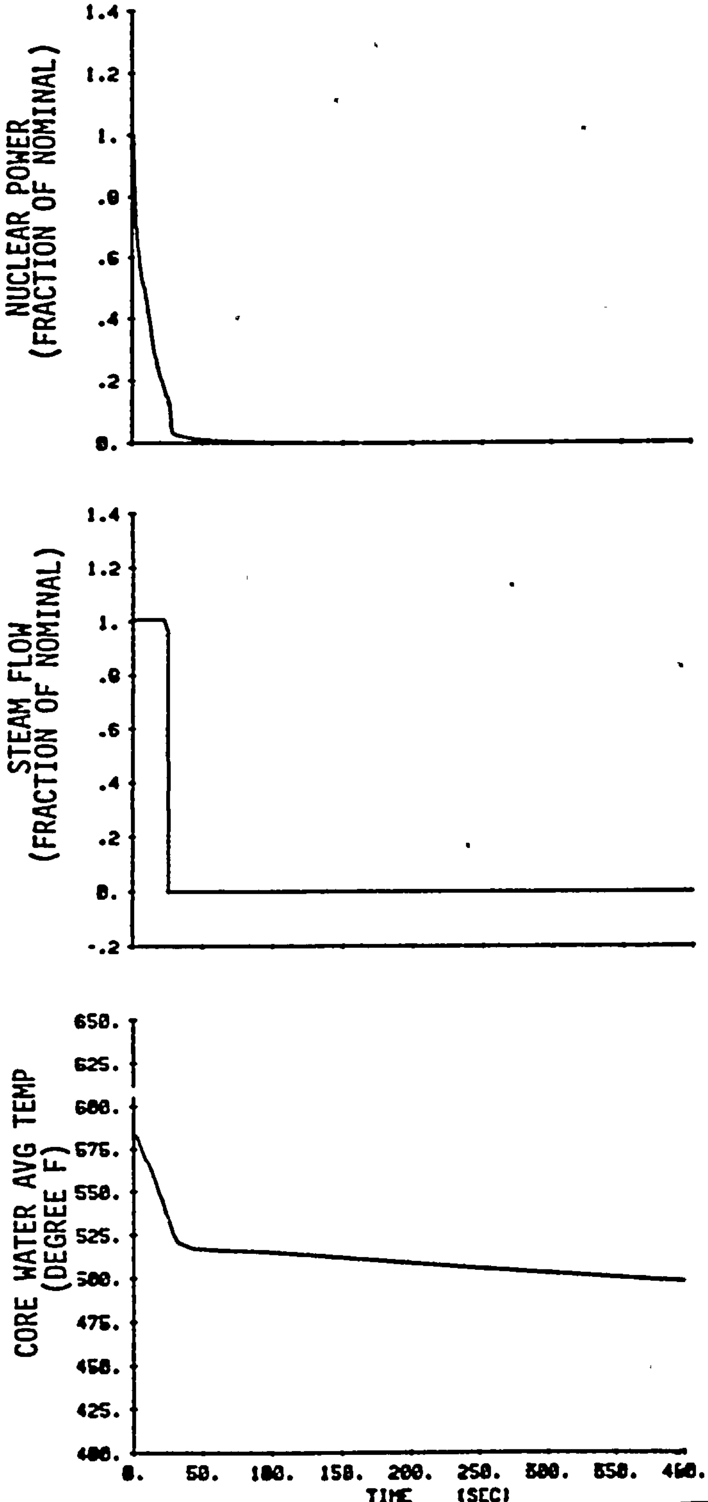


DIABLO CANYON UNITS 1 AND 2
FIGURE 15.2.13-3 TRANSIENT RESPONSE FOR A STEAM LINE BREAK EQUIVALENT TO 228 LBS./SEC AT 1015 PSIA WITH OUTSIDE POWER AVAILABLE



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.14-1
SPURIOUS ACTUATION OF SAFETY INJECTION SYSTEM AT POWER. PRESSURIZER WATER VOLUME AND PRESSURIZER PRESSURE VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.2.14-2
SPURIOUS ACTUATION OF SAFETY INJECTION SYSTEM AT POWER, NUCLEAR POWER, STEAM FLOW, AND CORE WATER AVERAGE TEMPERATURE VERSUS TIME

15.3 CONDITION III - INFREQUENT FAULTS

By definition, Condition III occurrences are faults that may occur very infrequently during the life of the plant. They will be accompanied with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system (RCS) or containment barriers. For the purposes of this report the following faults have been grouped into this category:

(1) Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, that actuates emergency core cooling

(2) Minor secondary system pipe breaks

(3) Inadvertent loading of a fuel assembly into an improper position

(4) Complete loss of forced reactor coolant flow

(5) Single rod cluster control assembly (RCCA) withdrawal at full power.

Each of these infrequent faults are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The time sequences of events during three Condition III faults of type (1) (small break loss-of-coolant accident) are shown in Table 15.3-1.

15.3.2 Minor Secondary System Pipe Breaks

15.3.2.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6-inch-diameter break or smaller.

15.3.2.2 Analysis of Effects and Consequences

Minor secondary system pipe breaks must be accommodated with the failure of only a small fraction of the fuel elements in the reactor. Since the results of analysis presented in Section 15.4.2 for a major secondary system pipe rupture also meet these criteria, separate analyses for minor secondary system pipe breaks is not required.

The analyses of the more probable accidental opening of a secondary system steam dump, relief, or safety valve is presented in Section 15.2.13. These analyses are illustrative of a pipe break equivalent in size to a single valve opening.

15.3.2.3 Conclusions

The analysis presented in Section 15.4.2 demonstrates that the consequences of a minor secondary system pipe break are acceptable since a departure from nucleate boiling ratio (DNBR) of less than the design basis values does not occur even for a more critical major secondary system pipe break.

15.3.4 Complete Loss of Forced Reactor Coolant Flow

15.3.4.1 Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor were not tripped promptly. The following provide necessary protection against a loss of coolant flow accident:

(1) Undervoltage or underfrequency on reactor coolant pump power supply buses

(2) Low reactor coolant loop flow

(3) Pump circuit breaker opening.

The reactor trip on reactor coolant pump bus undervoltage is provided to protect against conditions that can cause a loss of voltage to all reactor coolant pumps, i.e., station blackout. This function is blocked below approximately 10% power (Permissive 7).

The reactor trip on reactor coolant pump underfrequency is provided to open the reactor coolant pump breakers and trip the reactor for an underfrequency condition, resulting from frequency disturbances on the major power grid. The trip disengages the reactor coolant pumps from the power grid so that the pumps' flywheel kinetic energy is available for full coastdown.

The reactor trip on low primary coolant loop flow is provided to protect against loss-of-flow conditions that affect only one reactor coolant loop. It also serves as a backup to the undervoltage and underfrequency trips. This function is generated by two-out-of-three low-flow signals per reactor coolant loop. Above approximately 35% power (Permissive 8), low flow in any loop will actuate a reactor trip. Between approximately 10 and 35% power (Permissive 7 and Permissive 8), low-flow in any two loops will actuate a reactor trip. A reactor trip from opened pump breakers is provided as a backup to the low-flow signals. Above Permissive 7 a breaker open signal from any 2 of 4 pumps will actuate a reactor trip. Reactor trip on reactor coolant pump breakers open is blocked below Permissive 7.

Normal power for the reactor coolant pumps is supplied through buses from a transformer connected to the generator. Two pumps are on each bus. When a generator trip occurs, the buses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to supply coolant flow to the core. Following any turbine trip, where there are no electrical faults which require tripping the generator from the network, the generator remains connected to the network for approximately 30 seconds. The reactor coolant pumps remain connected to the generator thus ensuring full flow for 30 seconds after the reactor trip before any transfer is made.

15.3.4.2 Analysis of Effects and Consequences

This transient is analyzed by three digital computer codes. First the LOFTRAN (Ref 1) code is used to calculate the loop and core flow during the transient. The LOFTRAN code is also used to calculate the time of reactor trip based on the calculated flows and the nuclear power transient following reactor trip. The FACTRAN (Ref 2) code is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC (Ref 3) code is used to calculate the minimum DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The transients presented represent the minimum of the typical and thimble cells.

The following case has been analyzed:

All loops operating, all loops coasting down

The method of analysis and the assumptions made regarding initial operating conditions and reactivity coefficients are identical to those discussed in Section 15.2, except that following the loss of supply to all pumps at power, a reactor trip is actuated by either bus undervoltage or bus underfrequency.

15.3.4.3 Results

The calculated sequence of events is shown in Table 15.3-3. Figures 15.3.4-1 through 15.3.4-3 show the flow coastdown, the nuclear power coastdown, and the heat flux coastdown for the limiting complete loss of flow event. The reactor is assumed to trip on the undervoltage signal. A plot of DNBR versus time is given in Figure 15.3.4-4. This plot represents the limiting cell for the four-loop coastdown.

15.3.4.4 Conclusions

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the safety analysis limit values during the transient, and thus, no core safety limit is violated.

15.3.5 Single Rod Cluster Control Assembly Withdrawal at Full Power

15.3.5.1 Identification of Causes and Accident Description

No single electrical or mechanical failure in the rod control system could cause the accidental withdrawal of a single RCCA from the inserted bank at full power operation. The operator could deliberately withdraw a single RCCA in the control bank; this feature is necessary in order to retrieve an assembly should one be accidentally dropped. In the extremely unlikely event of simultaneous electrical failures that could result in single RCCA withdrawal, rod deviation and rod control urgent failure would both be displayed on the plant annunciator, and the rod position indicators would indicate the relative positions of the assemblies in the bank. The urgent failure alarm also inhibits automatic rod motion in the group in which it occurs. Withdrawal of a single RCCA by operator action, whether deliberate or by a combination of errors, would result in activation of the same alarm and the same visual indications.

Each bank of RCCAs in the system is divided into two groups of four mechanisms each (except Group 2 of Bank D which consists of five mechanisms). The rods comprising a group operate in parallel through multiplexing thyristors. The two groups in a bank move sequentially such that the first group is always within one step of the second group in the bank. A definite schedule of actuation and deactuation of the stationary gripper, movable gripper, and lift coils of a mechanism is required to withdraw the RCCA attached to the mechanism. Since the four stationary grippers, movable grippers, and lift coils associated with the four RCCAs of a rod group are driven in parallel, any single failure that would cause rod withdrawal would affect a minimum of one group, or four RCCAs. Mechanical failures are either in the direction of insertion or immobility.

In the unlikely event of multiple failures that result in continuous withdrawal of a single RCCA, it is not possible, in all cases, to provide assurance of automatic reactor trip so that core safety limits are not violated. Withdrawal of a single RCCA results in both positive reactivity insertion tending to increase core power, and an increase in local power density in the core area covered by the RCCA.

15.3.5.2 Analysis of Effects and Consequences

Power distributions within the core are calculated by the TURTLE code based on a macroscopic cross section generated by LEOPARD. The peaking factors calculated by TURTLE are then used by THINC to calculate the minimum DNB for the event. The plant was analyzed for the case of the worst rod withdrawn from Bank D inserted at the insertion limit, with the reactor initially at full power.

15.3.5.3 Results

Two cases have been considered as follows:

(1) If the reactor is in the automatic control mode, withdrawal of a single RCCA will result in the immobility of the other RCCAs in the controlling bank. The transient will then proceed in the same manner as Case 2 described below. For such cases as above, a trip will ultimately ensue, although not sufficiently fast in all cases to prevent a minimum DNBR in the core of less than the safety limit.

(2) If the reactor is in the manual control mode, continuous withdrawal of a single RCCA results in both an increase in core power and coolant temperature, and an increase in the local hot channel factor in the area of the failed RCCA. In terms of the overall system response, this case is similar to those presented in Section 15.2; however, the increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBR than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum core DNBR from falling below the safety limit value. Evaluation of this case at the power and coolant condition at which overtemperature ΔT trip would be expected to trip the plant shows that an upper limit for the number of rods with a DNBR less than the safety limit value is 5%.

15.3.5.4 Conclusions

For the case of one RCCA fully withdrawn, with the reactor in either the automatic or manual control mode and initially operating at full power with Bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNBR less than the safety analysis limit DNBR is 5% or less of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before DNB could occur. For Case 2 discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard.

15.3.6 References

1. T. W. T. Burnett, et al, LOFTRAN Code Description, WCAP-7907, June 1972.

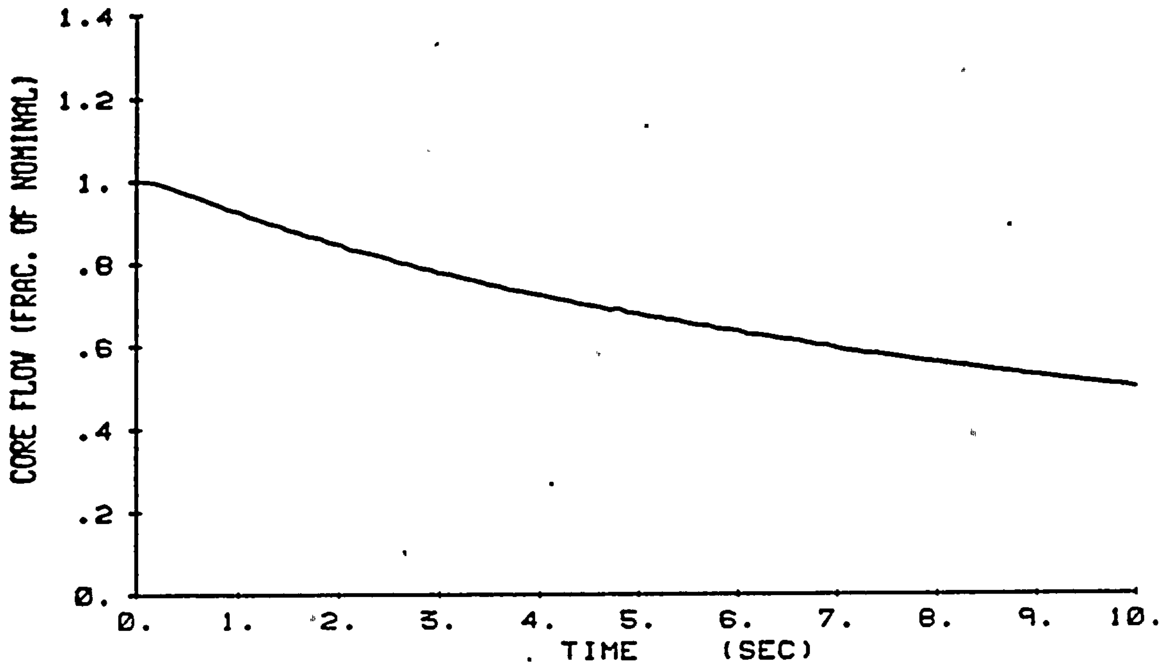
2. C. Hunin, FACTRAN, A Fortran-IV Code for Thermal Transients in UO_2 Fuel Rods, WCAP-7908, June 1972.

3. J. S. Shefcheck, Application of the THINC Program to PWR Design, WCAP-7359-L, August 1969 (proprietary), WCAP-7838, January 1972.

TABLE 15.3-3

TIME SEQUENCE OF EVENTS FOR CONDITION III EVENTS

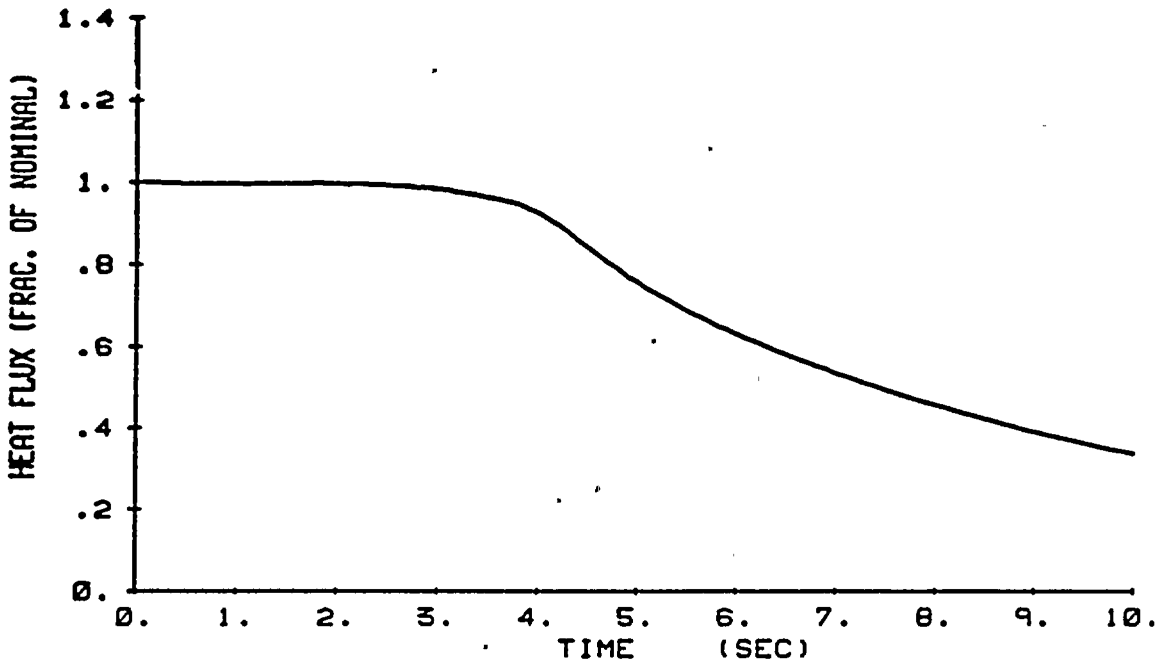
Accident	Event	Time, sec
Complete Loss of Forced Reactor Coolant Flow		
All loops operating, all pumps coasting down	Coastdown begins	0.0
	Rod motion begins	1.5
	Minimum DNBR begins	3.6



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.3.4-1

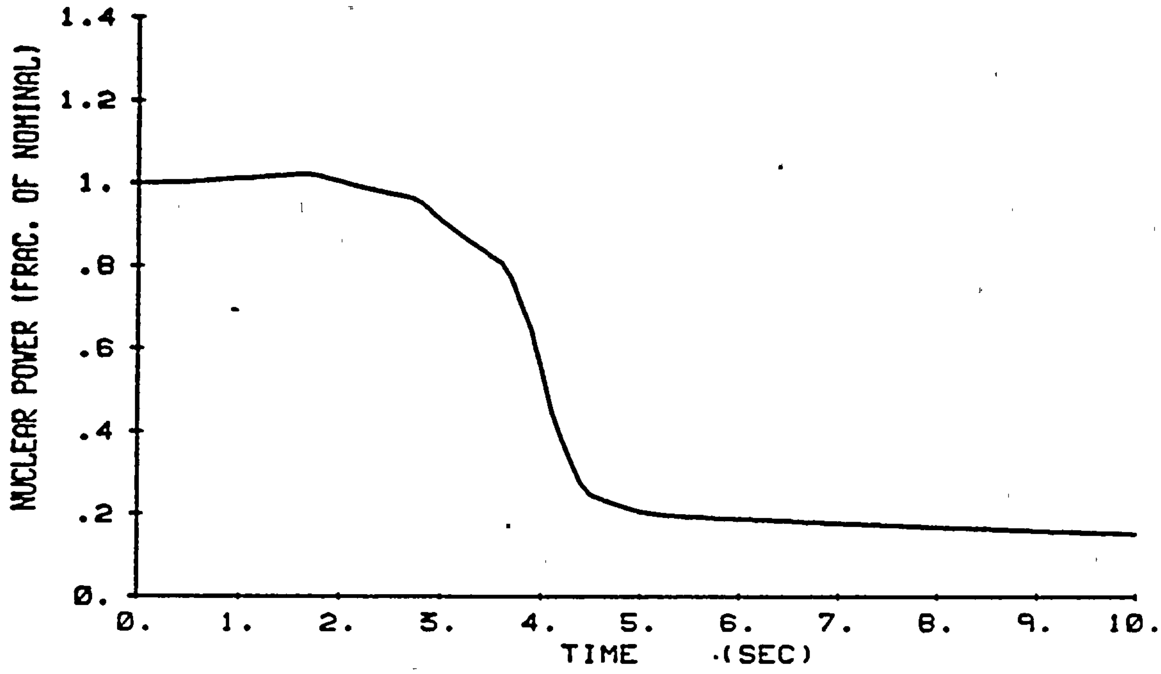
ALL LOOPS OPERATING.
ALL LOOPS COASTING DOWN.
FLOW COASTDOWN VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.3.4-2

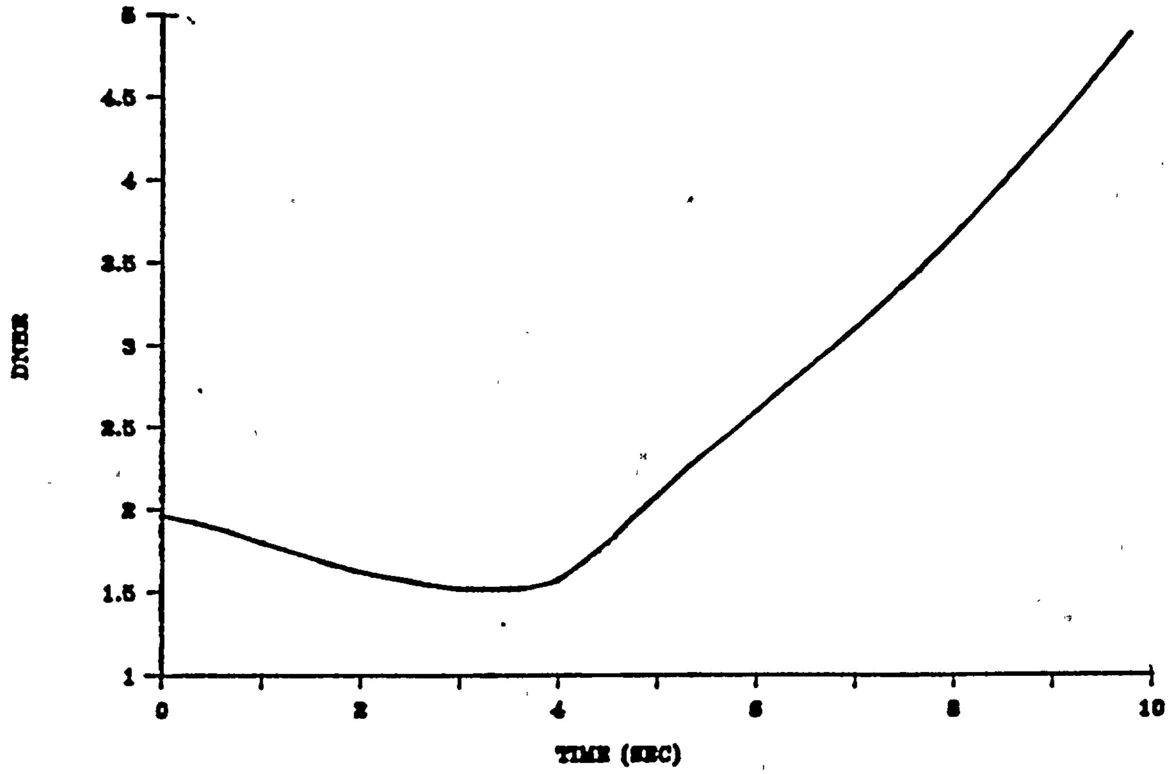
ALL LOOPS OPERATING,
ALL LOOPS COASTING DOWN,
HEAT FLUX VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.3.4-3

ALL LOOPS OPERATING,
ALL LOOPS COASTING DOWN,
NUCLEAR POWER VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.3.4-4

ALL LOOPS OPERATING,
ALL LOOPS COASTING DOWN,
DNBR VERSUS TIME

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults that are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences that must be designed against and represent limiting design cases. Condition IV faults shall not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system (ECCS) and the containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (RCS), i.e., loss-of-coolant accident (LOCA)

(2) Major secondary system pipe ruptures

(3) Steam generator tube rupture

(4) Single reactor coolant pump (RCP) locked rotor

(5) Fuel handling accident

(6) Rupture of a control rod mechanism housing (rod cluster control assembly [RCCA] ejection)

(7) Rupture of gas decay tank

(8) Rupture of a liquid holdup tank

(9) Rupture of a volume control tank.

Each of these nine limiting faults is analyzed in Section 15.4. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The analyses of thyroid and whole body doses, resulting from events leading to fission product release, are presented in Section 15.5. The fission product inventories that form a basis for these calculations are presented in Chapter 11 and Section 15.1. Also included is a discussion of system interdependency contributing to limiting fission product leakages from the containment following a Condition IV occurrence.

15.4.2 Major Secondary System Pipe Rupture

Two major secondary system pipe ruptures are analyzed in this section: rupture of a main steam line and rupture of a main feedwater pipe. The time sequence of events for each of these events is provided in Table 15.4-8.

15.4.2.1 Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam pipe would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high power peaking factors that exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the SIS.

The analysis of a main steam pipe rupture is performed to demonstrate that the following criteria are satisfied:

(1) Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safety features (ESF) there is no consequential damage to the primary system and the core remains in place and intact.

(2) Energy release to containment from the worst steam pipe break does not cause failure of the containment structure.

Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

The following functions provide the necessary protection against a steam pipe rupture:

(1) SIS actuation from any of the following:

(a) Two-out-of-four low pressurizer pressure signals

(b) High differential pressure signals between steam lines

(c) High steam line flow in two main steam lines (one-out-of-two per line) in coincidence with either low-low RCS average temperature or low steam line pressure in any two lines

(d) Two-out-of-three high containment pressure.

(2) The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

(3) Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater isolation valves that backup the control valves.

(4) Trip of the fast acting main steam line isolation valves on: (See Fig. 7.2-1 and Technical Specification[1] Table 3.3-5)

(a) High steam flow in two main steam lines in coincidence with either low-low T_{avg} or low steam line pressure in any two lines

(b) High-high containment pressure.

The fast-acting isolation valves are provided in each main steam line and will fully close within 10 seconds of a large steam line break. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would blow down even if one of the isolation valves fails to close. A description of steam line isolation is included in Chapter 10.

Steam flow is measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles that are of considerably smaller diameter than the main steam pipe are located inside the containment near the steam generators and also serve to limit the maximum steam flow for any break further downstream.

<u>15.4.2.1.2 Analysis of Effects and Consequences</u>

The analysis of the steam pipe rupture has been performed to determine:

(1) The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN code[2] has been used.

(2) The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC[3], has been used to determined if DNB occurs for the core conditions computed in (1) above.

The following conditions were assumed to exist at the time of a main steam line break accident.

(1) End of life (EOL) shutdown margin at no-load, equilibrium xenon conditions, and the most reactive assembly stuck in its fully withdrawn position: Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

(2) The negative moderator coefficient corresponding to the EOL rodded core with the most reactive rod in the fully withdrawn position: The variation of the coefficient with temperature and pressure has been included. The k_{eff} versus temperature at 1150 psia corresponding to the negative moderator temperature coefficient plus the Doppler temperature effect used is shown on Figure 15.2.13-1. The effect of power generation in the core on overall reactivity is shown on Figure 15.4.2-1.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underprediction of the reactivity feedback in the high-power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution was checked for the statepoints shown in Table 15.4-9. These core analyses considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the true reactivity for all statepoints in Table 15.4-9, verifying conservatism; i.e., underprediction of negative reactivity feedback from power generation.

(3) Minimum capability for injection of ~~high concentration~~ boric acid (2300 ppm) solution corresponding to the most restrictive single failure in the SIS. The characteristics of the injection unit used are shown on Figure 15.2.13-2. This corresponds to the flow delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration of boric acid that

from the safety injection lines downstream of the refueling water storage tank (RWST) isolation valves prior to the delivery of highly concentrated boric acid to the reactor coolant loops. This effect has been allowed for in the analysis. The modeling of the SIS in LOFTRAN is described in Reference 2.

For the cases where offsite power is assumed, the sequence of events in the SIS is the following: After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high-head injection pump starts. In 23 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept before the 2300 ppm boron reaches the core. This delay is inherently included in the modeling. In cases where offsite power is not available, an additional 15 second delay is assumed to be required to start the diesels and to load the necessary safety injection equipment onto them. That is, after a total of 38 seconds following an SIS signal, the SIS is assumed to be capable of delivering flow to the RCS.

(4) Four combinations of break sizes and initial plant conditions have been considered in determining the core power and RCS transients:

(a) Complete severance of a pipe outside the containment, downstream of the steam flow measuring nozzle, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available

(b) Complete severance of a pipe inside the containment at the outlet of the steam generator with the plant initially at no-load conditions with offsite power available

(c) Case (a) above with loss of offsite power simultaneous with the initiation of the safety injection signal. Loss of offsite power results in coolant pump coastdown.

(d) Case (b) above with the loss of offsite power simultaneous with the initiation of the safety injection signal.

(5) Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures are determined at EOL. The coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void in the region of the stuck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend on the core power, operating history, temperature, pressure, and flow, and thus are different for each case studied.

The values for the most limiting steamline break accident (Case B) are given in Table 15.4-9. A total of five time points are presented. This case is selected on the basis of hot channel factors, core power, and reactor coolant pressure. The other three cases are less severe relative to departure from nucleate boiling ratio (DNBR). The core parameters used for each of the four cases correspond to values determined from the respective transient analysis.

All the cases above assume initial hot shutdown conditions at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load, and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumes no-load condition at time zero.

However, since the initial steam generator water inventory is greatest at no-load, the magnitude and duration of the RCS cooldown are less for steam line breaks occurring at power.

(6) In computing the steam flow during a steam line break, the Moody Curve[4] for fL/D = 0 is used. The Moody Multiplier is 1 with a discharge at dry saturated steam conditions.

(7) Perfect moisture separation in the steam generator is assumed. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water carryover would reduce the magnitude of the temperature decrease in the core and the pressure increase in the containment.

15.4.2.1.3 Results

The results presented are a conservative indication of the events that would occur assuming a steam line rupture since it is postulated that all of the conditions described above occur simultaneously.

Figure 15.4.2-2 shows the RCS transient and core heat flux following a main steam pipe rupture (complete severance of a pipe) outside the containment, downstream of the flow measuring nozzle at initial no-load condition (Case A). The break assumed is the largest break that can occur anywhere outside the containment either upstream or downstream of the isolation valves. Offsite power is assumed available such that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs, the initiation of safety injection by high differential pressure between any steam line and the remaining steam lines or by high steam flow signals in coincidence with either low-low RCS temperature or low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast action isolation valves in the steam lines by the high steam flow signals in coincidence with either low-low RCS temperature or low steam line pressure. Even with the failure of one valve, release is limited to no more than 10 seconds assumed for isolation delay plus less than one second to reach the actuation setpoint. The steam line isolation valves are designed to be fully closed in less than 5 seconds after receipt of closure signal with no flow through them. With the high flow existing during a steam line rupture the valves will close considerably faster.

The steam flow on Figures 15.4.2-2 through 15.4.2-5 represents steam flow from the faulted steam generator and one intact steam generator.

As shown on Figures 15.4.2-2 and 15.4.2-5, the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck assembly) before boron solution at 2300 ppm enters the RCS from the SIS which is drawing from the RWST. The delay time consists of the time to receive and actuate the safety injection signal and the time to completely open valve trains in the safety injection lines. The safety injection pumps are then ready to deliver flow. At this stage a further delay time is incurred before 2300 ppm boron solution can be injected to the RCS due to the low concentration solution being swept from the safety injection lines. A peak core power well below the nominal full power value is attained.

The calculation assumes the boric acid is mixed with and diluted by the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends on the relative flow rates in the RCS and in the SIS. The variation of mass flowrate in the RCS due to water density changes is included in the calculation as is the variation of flowrate from the SIS and the accumulator due to changes in the RCS pressure. The SIS flow calculation includes the line losses in the system as well as the pump head curve. The accumulators provide an additional source of borated water after the RCS pressure decreases to below 610 psia. The core boron as a function of time is shown on Figure 15.4.2-6.

Figure 15.4.2-3 shows Case B, a steam line rupture at the exit of a steam generator at no-load. The sequence of events is similar to that described above for the rupture outside the containment except that criticality is attained earlier due to more rapid cooldown and a higher peak core average power is attained.

Figures 15.4.2-4 through 15.4.2-5 show the responses of the salient parameters for Cases C and D which correspond to the cases discussed above with additional safety injection delay due to loss of offsite power. The SIS delay time covers 38 seconds to start the diesels and load the necessary safety injection equipment onto them, for the appropriate valves to reach their final position, and for the high-head injection pump to reach full speed. In each case criticality is achieved later and the core power increase is slower than in the similar case with offsite power available. The ability of the emptying steam generator to extract heat from the RCS is reduced by the decreased flow in the RCS. For both these cases the peak core power remains well below the nominal full power value.

It should be noted that following a steam line break, only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power this heat is removed to the atmosphere via the steam line safety valves that have been sized to cover this condition.

15.4.2.1.4 Conclusion

A DNB analysis was performed for the four cases. It was found that the DNB design basis is met.

15.4.2.2 Major Rupture of a Main Feedwater Pipe

15.4.2.2.1 Identification of Causes and Accident Description

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater (AFW) to the affected steam generator. (A break upstream of the feedline check valve would affect the nuclear steam supply system [NSSS] only as a loss of feedwater. This case is covered by the evaluation in Section 15.2.8).

Depending on the size of the break and the plant operating conditions at the time of the break, the break could cause either an RCS cooldown (by excessive energy discharge through the break), or an RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.4.2.1, Rupture of a Main Steam Pipe. Therefore, only the RCS heatup effects are evaluated for a feedline rupture.

A feedline rupture reduces the ability to remove heat generated by the core from the RCS for the following reasons:

(1) Feedwater to the steam generators is reduced. Since feedwater is subcooled, its loss may cause reactor coolant temperatures to increase prior to reactor trip

(2) Liquid in the steam generator may be discharged through the break, and would then not be available for decay heat removal after trip

(3) The break may be large enough to prevent the addition of any main feedwater after trip.

The following provide the necessary protection against a main feedwater line rupture:

(1) A reactor trip on any of the following conditions:

(a) High pressurizer pressure

(b) Overtemperature ΔT

(c) Low-low steam generator water level in any steam generator

(d) Low steam generator level plus steam/feedwater flow mismatch in any steam generator

(e) Safety injection signals from any of the following:

1. Low steam line pressure coincident with high steam flow

2. High containment pressure

3. High steamline differential pressure

(Refer to Chapter 7 for a description of the actuation system.)

(2) An AFW system to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to Chapter 6 for a description of the AFW system.)

15.4.2.2.2 Analysis of Effects and Consequences

A detailed analysis using the LOFTRAN[2] code is performed in order to determine the plant transient following a feedline rupture. The code describes the plant thermal kinetics, RCS including natural circulation, pressurizer, steam generators, and feedwater system, and computes pertinent variables, including the pressurizer pressure, pressurizer water level, and reactor coolant average temperature.

Major assumptions are:

(1) The plant is initially operating at 102% of the ESF design rating

(2) Initial reactor coolant average temperature is 6.7°F above the nominal value, and the initial pressurizer pressure is 38 psi above its nominal value

(3) A conservatively high initial pressurizer level is assumed; initial steam generator water level is at the nominal value plus 5% in the faulted steam generator, and at the nominal value minus 5% in the intact steam generators

(4) No credit is taken for the pressurizer power-operated relief valves or pressurizer spray

(5) No credit is taken for the high pressurizer pressure reactor trip

(6) Main feed to all steam generators is assumed to stop at the time the break occurs. (All main feedwater spills out through the break.)

(7) Saturated liquid discharge only (no steam) is assumed from the affected steam generator through the feedline rupture initially. Saturated steam discharge is released after trip levels have been reached. This assumption minimizes energy removal from the NSSS during blowdown

(8) Reactor trip is assumed to be initiated when the low-low level trip setpoint in the ruptured steam generatator is reached. A low-low level setpoint of 0% narrow range span is assumed

(9) A double-ended break area of 1.360 ft^2 is assumed. This maximizes the blowdown discharge rate following the time of trip, which maximizes the resultant heatup of the reactor coolant

(10) No credit is taken for heat energy deposited in RCS metal during the RCS heatup

(11) No credit is taken for charging or letdown

(12) Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases

(13) Conservative core residual heat generation based on long-term operation at the initial power level preceding the trip is assumed. The 1979 ANS 5.1[5] decay heat standard plus uncertainty was used for calculation of residual decay heat levels

(14) The AFW is assumed to be initiated 10 minutes after the trip with the feed rate of 440 gpm. An additional 4 minutes is assumed before the feedlines are purged and the relatively cold (90°F) AFW enters two of the three unaffected steam generators.

15.4.2.2.3 Results

Results for two feedline break cases are presented. Results for a case in which offsite power is assumed to be available are presented in Section 15.4.2.2.3.1. Results for a case in which offsite power is assumed to be lost following reactor trip are presented in Section 15.4.2.2.3.2. The calculated sequence of events for both cases is listed in Table 15.4-8.

15.4.2.2.3.1 Feedline Rupture with Offsite Power Available

The system response following a feedwater line rupture, assuming offsite power is available, is presented in Figures 15.4.2-7 through 15.4.2-10. Results presented in Figures 15.4.2-8 and 15.4.2-10 show that pressures in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to the pressurizer safety valve setpoint. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the steam generators. Figure 15.4.2-8 shows that the water volume in the pressurizer increases in response to the heatup but remains below the initial water level. At approximately 5296 seconds, decay heat generation decreases to a level such that the total RCS heat generation (decay heat plus pump heat) is less than auxiliary feedwater heat removal capability, and RCS pressure and temperature begin to decrease.

The results show that the core remains covered at all times and that no boiling occurs in the reactor coolant loops.

15.4.2.2.3.2 Feedline Rupture with Offsite Power Unavailable

The system response following a feedwater line rupture without offsite power available is similar to the case with offsite power available. However, as a result of the loss of offsite power (assumed to occur at reactor trip), the reactor coolant pumps coast down. This results in a reduction in total RCS heat generation by the amount produced by pump operation.

The reduction in total RCS heat generation produces a milder transient than in the case where offsite power is available. Results presented in Figures 15.4.2-12 and 15.4.2-14 show that pressure in the RCS and main steam system remain below 110% of the respective design pressures. Pressurizer pressure decreases after reactor trip on low-low steam generator water level due to the reduction of heat input. Following this initial decrease, pressurizer pressure increases to a peak pressure of 2502 psia at 228 seconds. This increase in pressure is the result of coolant expansion caused by the reduction in heat transfer capability in the steam generators. Figure 15.4.2-12 shows that the water volume in the pressurizer increases in response to the heatup but remains at approximately the initial water level and does not fill the pressurizer. At approximately 2904 seconds, decay heat generation decreases to a level less than the auxiliary feedwater heat removal capability, and RCS temperature begins to decrease. The results show that the core remains covered at all times and that no boiling occurs in the reactor coolant loops.

15.4.2.2.4 Conclusion

Results of the analysis show that for the postulated feedline rupture, the assumed AFW system capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the reactor core.

15.4.4 Single Reactor Coolant Pump Locked Rotor

15.4.4.1 Identification of Causes and Accident Description

The accident postulated is an instantaneous seizure of an RCP rotor.

Following initiation of the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell-side of the steam generators is reduced, first because the reduced flow results in a decreased tube-side film coefficient and then because the reactor coolant in the tubes cools down while the shell-side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators causes an insurge into the pressurizer and a pressure increase throughout the RCS. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves in that sequence. The three power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect as well as the pressure-reducing effect of the spray is not included in the analysis.

15.4.4.2 Analysis of Effects and Consequences

Three digital computer codes are used to analyze this transient. The LOFTRAN[2] code is used to calculate the resulting loop and core coolant flow following the pump seizure. The LOFTRAN code is also used to calculate the time of reactor trip, based on the calculated flow, the nuclear power following reactor trip, and to determine the peak pressure. The thermal behavior of the fuel located at the core hot spot is investigated using the FACTRAN[6] code, using the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes the use of a film boiling heat transfer coefficient.

The following case is analyzed:

All loops operating, one locked rotor

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the RCPs is assumed to seize, the plant is assumed to be operating under steady state operating conditions with respect to the margin to DNB, i.e., nominal steady state power level, nominal steady state pressure, and nominal steady state coolant average temperature (+2.5°F for SG fouling).

When the peak pressure is evaluated, the initial pressure is conservatively estimated as 38 psi above nominal pressure (2250 psia) to allow for errors in the pressurizer pressure measurement and control channels. This is done to obtain the highest possible rise in the coolant pressure during the transient. To obtain the maximum pressure in the primary side, conservatively high loop pressure drops are added to the calculated pressurizer pressure. The pressure response is shown on Figure 15.4.4-1.

15.4.4.2.1 Evaluation of the Pressure Transient

After pump seizure and reactor trip, the neutron flux is rapidly reduced by control rod insertion effect. Rod motion is assumed to begin 1 second after the flow in the affected loop reaches 87% of nominal flow. No credit is taken for the pressure-reducing effect of the pressurizer relief valves, pressurizer spray, steam dump, or controlled feedwater flow after plant trip.

Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are assumed to initially open at 2500 psia and achieve rated flow at 2575 psia (3% accumulation).

15.4.4.2.2 Evaluation of the Effects of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core and, therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this hot spot condition represent the upper limit with respect to cladding temperature and zirconium-water reaction.

In the evaluation, the rod power at the hot spot is conservatively assumed to be greater than or equal to two and a half times the average rod power (i.e., $F_q \geq 2.5$) at the initial core power level.

15.4.4.2.3 Film Boiling Coefficient

The film boiling coefficient is calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties are evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficient at every time step based on the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density, and mass flowrate as a function of time are used as program input.

For this analysis, the initial values of the pressure and the bulk density are used throughout the transient since they are the most conservative with respect to cladding temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

15.4.4.2.4 Fuel Cladding Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and cladding (gap coefficient) has a pronounced influence on the thermal results. The larger the value of the gap coefficient, the more heat is transferred between pellet and cladding. Based on investigations on the effect of the gap coefficient upon the maximum cladding temperature during the transient, the gap coefficient was assumed to increase from a steady state value consistent with the initial fuel temperature to 10,000 BTU/hr-ft^2-°F in 0.5 seconds after the initiation of the transient. This assumption causes energy stored in the fuel to be released to the cladding at the initiation of the transient and maximizes the cladding temperature during the transient.

15.4.4.2.5 Zirconium-steam Reaction

The zirconium-steam reaction can become significant above 1800°F (cladding temperature). The Baker-Just parabolic rate equation shown below is used to define the rate of the zirconium-steam reaction.

$$\frac{d(w^2)}{dt} = 33.3 \times 10^6 \exp\left[-\frac{45,500}{1.986T}\right] \qquad (15.4\text{-}1)$$

where:

w = amount reacted, mg/cm^2
t = time, sec
T = temperature, °K
The reaction heat is 1510 cal/gm.

15.4.4.3 Results

Transient values of pressurizer pressure, flow coastdown, neutron flux, and hot channel heat flux are shown in the Figure 15.4.4-1 and Figures 15.4.4-3 through 15.4.4-5.

Maximum RCS pressure, maximum cladding temperature, and amount of zirconium-water reaction are contained in Table 15.4-10. Figure 15.4.4-2 shows the cladding temperature transient for the worst case.

15.4.4.4 Conclusions

(1) Since the peak RCS pressure reached during any of the transients is less than that which would cause stresses to exceed the faulted condition stress limits, the integrity of the primary coolant system is not endangered.

(2) Since the peak cladding surface temperature calculated for the hot spot during the worst transient remains considerably less than 2700°F and the amount of zirconium-water reaction is small, the core will remain in place and intact with no consequential loss of core cooling capability.

(3) The results of the transient analysis show that for four-loop operation, less than 10.0% of the fuel rods will have DNBRs below the safety analysis limit values.

15.4.6 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

15.4.6.1 Identification of Causes and Accident Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion and system depressurization together with an adverse core power distribution, possibly leading to localized fuel rod damage.

15.4.6.1.1 Design Precautions and Protection

Certain features of the DCPP are intended to preclude the possibility of an rod ejection accident, or to limit the consequences if the accident were to occur. These include a sound, conservative mechanical design of the rod housings, together with a thorough quality control (testing) program during assembly, and a nuclear design that lessens the potential ejection worth of RCCAs, and minimizes the number of assemblies inserted at high power levels.

15.4.6.1.2 Mechanical Design

The mechanical design is discussed in FSAR Section 4.2. Mechanical design and quality control procedures intended to preclude the possibility of an RCCA drive mechanism housing failure are listed below:

(1) Each full length control rod drive mechanism housing is completely assembled and shop tested at 4100 psi.

(2) The mechanism housings are individually hydrotested after they are attached to the head adapters in the reactor vessel head, and checked during the hydrotest of the completed reactor coolant system.

(3) Stress levels in the mechanism are not affected by anticipated system transients at power, or by the thermal movement of the coolant loops. Moments induced by the design-basis earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class I components.

(4) The latch mechanism housing and rod travel housing are each a single length of forged Type-304 stainless steel. This material exhibits excellent notch toughness at all temperatures that will be encountered.

A significant margin of strength in the elastic range together with the large energy absorption capability in the plastic range gives additional assurance that gross failure of the housing will not occur. The joints between the latch mechanism housing and head adapter, and between the latch mechanism housing and rod travel housing, are threaded joints reinforced by canopy-type rod welds. Administrative regulations require periodic inspections of these (and other) welds.

15.4.6.1.3 Nuclear Design

Even if a rupture of an RCCA drive mechanism housing is postulated, the operation of a plant utilizing chemical shim is such that the severity of an ejected RCCA is inherently limited. In general, the reactor is operated with the RCCAs inserted only far enough to permit load follow. Reactivity changes caused by core depletion and xenon transients are compensated by boron changes. Further, the location and grouping of control RCCA banks are selected during the nuclear design to lessen the severity of an RCCA ejection accident. Therefore, should a RCCA be ejected from its normal position during full-power operation, only a minor reactivity excursion, at worst, could be expected to occur.

However, it may be occasionally desirable to operate with larger than normal insertions. For this reason, a rod insertion limit is defined as a function of power level. Operation with the RCCAs above this limit guarantees adequate shutdown capability and acceptable power distribution. The position of all RCCAs is continuously indicated in the control room. An alarm will occur if a bank of RCCAs approaches its insertion limit or if one RCCA deviates from its bank. There are low and low-low level insertion monitors with visual and audio signals. Operating instructions require boration at low-level alarm and emergency boration at the low-low alarm.

15.4.6.1.4 Reactor Protection

The reactor protection in the event of a rod ejection accident has been described in Ref 7. The protection for this accident is provided by the power range high neutron flux trip (high and low setting) and high rate of neutron flux increase trip. These protection functions are described in detail in FSAR Section 7.2.

15.4.6.1.5 Effects on Adjacent Housings

Disregarding the remote possibility of the occurrence of an RCCA mechanism housing failure, investigations have shown that failure of a housing due to either longitudinal or circumferential cracking is not expected to cause damage to adjacent housings leading to increased severity of the initial accident.

15.4.6.1.6 Limiting Criteria

Due to the extremely low probability of an RCCA ejection accident, limited fuel damage is considered an acceptable consequence.

Comprehensive studies of the threshold of fuel failure and of the threshold of significant conversion of the fuel thermal energy to mechanical energy have been carried out as part of the SPERT project by the Idaho Nuclear Corporation (Ref 8). Extensive tests of zirconium-clad UO_2 fuel rods representative of those in PWR-type cores have demonstrated failure thresholds in the range of 240 to 257 cal/gm. However, other rods of a slightly different design have exhibited failures as low as 225 cal/gm. These results differ significantly from the TREAT (Ref 9) results, which indicated a failure threshold of 280 cal/gm. Limited results have indicated that this threshold decreases by about 10% with fuel burnup. The cladding failure mechanism appears to be melting for zero burnup rods and brittle fracture for irradiated rods. Also important is the conversion ratio of thermal to mechanical energy. This ratio becomes marginally detectable above 300 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods; catastrophic failure, (large fuel dispersal, large pressure rise) even for irradiated rods, did not occur below 300 cal/gm.

In view of the above experimental results, conservative criteria are applied to ensure that there is little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves. These criteria are:

(1) Average fuel pellet enthalpy at the hot spot below 225 cal/gm for unirradiated fuel and 200 cal/gm for irradiated fuel;

(2) Average cladding temperature at the hot spot below the temperature at which cladding embrittlement may be expected (2700°F);

(3) Peak reactor coolant pressure less than that which would cause stresses to exceed the faulted condition stress limits;

(4) Fuel melting will be limited to less than 10% of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits of Criterion (1) above.

15.4.6.2 Analysis of Effects and Consequences

The analysis of the RCCA ejection accident is performed in two stages: (a) an average core nuclear power transient calculation and (b) a hot spot heat transfer calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is pessimistically assumed to persist thoughout the transient.

A detailed discussion of the method on analysis can be found in Reference 10.

15.4.6.2.1 Average Core Analysis

The spatial kinetics computer code, TWINKLE (Ref 11), is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equations in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculating pointwise Doppler, and moderator feedback effects.

In this analysis, the code is used as one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and RCCA movement and the elimination of axial feedback weighting factors. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor. A further description of TWINKLE appears in Section 15.1.8.

15.4.6.2.2 Hot Spot Analysis

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using the detailed fuel and cladding transient heat transfer computer code, FACTRAN (Ref 6). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A parabolic radial power generation is used within the fuel rod.

FACTRAN uses the Dittus-Boelter (Ref 12) or Jens-Lottes (Ref 13) correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (Ref 14) to determine the film boiling coefficient after DNB. The DNB heat flux is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state temperature distribution to agree with that predicted by design fuel heat transfer codes.

For full power cases, the design initial hot channel factor (F_Q) is input to the code. The hot channel factor during the transient is assumed to increase from the steady state design value to the maximum transient value in 0.1 seconds, and remain at the maximum for the duration of the transient. This is conservative, since detailed spatial kinetics models show that the hot channel factor decreases shortly after the nuclear power peak due to power flattening caused by preferential feedback in the hot channel. Further description of FACTRAN appears in Section 15.1.8.

15.4.6.2.3 System Overpressure Analysis

Because safety limits for fuel damage specified earlier are not exceeded, there is little likelihood of fuel dispersal into the coolant. The pressure surge may therefore be calculated on the basis of conventional heat transfer from the fuel and prompt heat generation in the coolant.

The pressure surge is calculated by first performing the fuel heat transfer calculation to determine the average and hot spot heat flux versus time. Using this heat flux data, a THINC calculation is conducted to determine the volume surge. Finally, the volume surge is simulated in a plant transient computer code. This code calculates the pressure transient taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer spray and pressure relief valves. No credit is taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing (Ref 15).

15.4.6.2.4 Calculation of Basic Parameters

Input parameters for the analysis are conservatively selected on the basis of calculated values for this type of core. The more important parameters are discussed below. Table 15.4-11 presents the parameters used in this analysis.

15.4.6.2.5 Ejected Rod Worths and Hot Channel Factors

The values for ejected rod worths and hot channel factors are calculated using three dimensional calculations. Standard nuclear design codes are used in the analysis. No credit is taken for the flux-flattening effects of reactivity feedback. The calculation is performed for the maximum allowed bank insertion at a given power level as determined by the rod insertion limits. Adverse xenon distributions and part-length rod positions are considered in the calculations.

The total transient hot channel factor F_Q is then obtained by combining the axial and radial factors.

Appropriate margins are added to the results to allow for calculational uncertainties, including an allowance for nuclear power peaking due to densification.

15.4.6.2.6 Reactivity Feedback Weighting Factors

The largest temperature rises, and hence the largest reactivity feedbacks, occur in channels where the power is higher than average. Since the weight of regions is dependent on flux, these regions have high weights. This means that the reactivity feedback is larger than that indicated by a simple single channel analysis. Physics calculations were carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers that, when applied to single channel feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. In this analysis, since a one-dimensional (axial) spatial kinetics method is employed, axial weighting is not used. In addition, no weighting is applied to the moderator feedback. A conservative radial weighting factor is applied to the transient fuel temperature to obtain an effective fuel temperature as a function of time accounting for the missing spatial dimension. These weighting factors were shown to be conservative compared to three-dimensional analysis.

15.4.6.2.7 Moderator and Doppler Coefficient

The critical boron concentrations at the beginning-of-life (BOL) and end-of-life (EOL) are adjusted in the nuclear code in order to obtain moderator density coefficient curves which are conservative compared to actual design conditions for the plant. As discussed above, no weighting factor is applied to these results.

The Doppler reactivity defect is determined as a function of power level using the one-dimensional steady state computer code with a Doppler weighting factor of 1. The resulting curve is conservative compared to design predictions for this plant. The Doppler weighting factor should be larger than 1 (approximately 1.3), just to make the present calculation agree with design predictions before ejection. This weighting factor will increase under accident conditions, as discussed above.

15.4.6.2.8 Delayed Neutron Fraction

Calculations of the effective delayed neutron fraction (β_{eff}) typically yield values of 0.70% at BOL and 0.50% at EOL for the first cycle. The accident is sensitive to β if the ejected rod worth is nearly equal to or greater than β as in zero power transients. In order to allow for future fuel cycles, pessimistic estimates of β of 0.55% at beginning of cycle and 0.44% at end of cycle were used in the analysis.

15.4.6.2.9 Trip Reactivity Insertion

The trip reactivity insertion assumed is given in Table 15.4-11 and includes the effect of one stuck rod. These values are reduced by the ejected rod reactivity. The shutdown reactivity was simulated by dropping a rod of the required worth into the core. The start of rod motion occurred 0.5 seconds after the high neutron flux trip point was reached. This delay is assumed to consist of 0.2 seconds for the instrument channel to produce a signal, 0.15 seconds for the trip breaker to open, and 0.15 seconds for the coil to release the rods. The analyses presented are applicable for a rod insertion time of 2.7 seconds from coil release to entrance to the dashpot. The choice of such a conservative insertion rate means that there is over 1 second after the trip point is reached before significant shutdown reactivity is inserted into the core. This is a particularly important conservatism for hot full power accidents.

The rod insertion versus time is described in Section 15.1.4.

15.4.6.3 Results

The values of the parameters used in the analysis, as well as the results of the analysis, are presented in Table 15.4-11 and discussed below.

15.4.6.3.1 Beginning of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The worst ejected rod worth and hot channel factor were conservatively assumed to be 0.20% ΔK and 6.70, respectively. The peak hot spot cladding average temperature was 2434°F. The peak hot spot fuel center temperature exceeded the BOL melting temperature of 4900°F. However, melting was restricted to less than 10% of the pellet.

15.4.6.3.2 Beginning of Cycle, Zero Power

For this condition, control Bank D was assumed to be fully inserted and C was at its inserting limit. The worst ejected rod is located in control Bank D and was conservatively assumed to have a worth of 0.785% ΔK and a hot channel factor of 13. The peak hot spot cladding temperature reached 2660°F.

15.4.6.3.3 End of Cycle, Full Power

Control Bank D was assumed to be inserted to its insertion limit. The ejected rod worth and hot channel factors were conservatively assumed to be 0.21% ΔK and 6.50, respectively. This resulted in a PCT of 2218°F. The peak hot spot fuel center temperature exceeded the EOL melting temperature of 4800°F. However, melting was restricted to less than 10% of the pellets.

15.4.6.3.4 End of Cycle, Zero Power

The ejected rod worth and hot channel factor for this case were obtained assuming control Bank D to be fully inserted and Bank C at its insertion limit. The results were 0.85% ΔK and 21.5, respectively. The peak cladding and fuel center temperatures were 2632 and 3849°F, respectively.

A summary of the cases presented above is given in Table 15.4-11. The nuclear power and hot spot fuel cladding temperature transients for the BOL full power and EOL zero power cases are presented on Figures 15.4.6-1 through 15.4.6-4.

15.4.6.3.5 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered, less than 10% of the rods entered DNB based on a detailed three-dimensional THINC analysis. Although limited fuel melting at the hot spot was predicted for the full power cases, in practice melting is not expected since the analysis conservatively assumed that the hot spots before and after ejection were coincident.

15.4.6.3.6 Pressure Surge

A detailed calculation of the pressure surge for an ejection worth of one dollar at BOL, hot full power, indicates that the peak pressure does not exceed that which would cause stress to exceed the faulted condition stress limits. Since the severity of the present analysis does not exceed this worst case analysis, the accident for this plant will not result in an excessive pressure rise or further damage to the RCS.

15.4.6.3.7 Lattice Deformations

A large temperature gradient will exist in the region of the hot spot. Since the fuel rods are free to move in the vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradients across individual rods may produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion. In practice, no significant bowing is anticipated, since the structural rigidity of the core is more than sufficient to withstand the forces produced. Boiling in the hot spot region would produce a net flow away from that region. However, the heat from fuel is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling, sufficient to distort the lattice, is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio, and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It can be concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is conservatively ignored in the analyses.

<u>15.4.6.4 Conclusions</u>

Even on a pessimistic basis, the analyses indicate that the described fuel and cladding limits are not exceeded. It is concluded that there is no danger of sudden fuel dispersal into the coolant. Since the peak pressure does not exceed that which would cause stresses to exceed the faulted condition stress limits, it is concluded that there is no danger of further consequential damage to the reactor coolant system. The analyses have demonstrated that the upper limit in fission product release as a result of a number of fuel rods entering DNB amounts to 10% (Ref 15).

15.4.10 References

1. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended to the date of the most recent FSAR Update Revision.

2. T. W. T. Burnett, et. al., LOFTRAN Code Description, WCAP-7907, June 1972.

3. J. S. Shefcheck, Application of the THINC Program to PWR Design, WCAP-7359-L, August 1969 (Proriety), and WCAP-7838, January 1972.

4. F. S. Moody, "Transactions of the ASME," Journal of Heat Transfer, February 1965, Figure 3, page 134.

5. ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, 1979.

6. C. Hunin, FACTRAN, A Fortran N Code for Thermal Transients in UO_2 Fuel Rod, WCAP-7908, June 1972.

7. T. W. T. Burnett, Reactor Protection System Diversity in Westinghouse Pressurized Water Reactor, WCAP-7306, April 1969.

8. T. G. Taxelius, ed. "Annual Report - Spert Project, October 1968 September 1969", Idaho Nuclear Corporation IN-1370, June 1970.

9. R. C. Liimatainen and F. J. Testa, Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements, ANL-7225, January - June 1966, p. 177, November 1966.

10. D. H. Risher, Jr., An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1, December 1971.

11. D. H. Risher, Jr. and R. F. Barry, TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, WCAP-7979, November 1972.

12. F. W. Dittus and L. M. K. Boelter, University of California (Berkeley), Publs. Eng., 2,433, 1930.

13. W. H. Jens and P. A. Lottes, Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water, USAEC Report ANL-4627, 1951.

14. A. A. Bishop, et al., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

15. Risher, D. H., Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinectics Methods," WCAP-7588, Revision 1-A, January 1975.

TABLE 15.4-8

TIME SEQUENCE OF EVENTS FOR
MAJOR SECONDARY SYSTEM PIPE RUPTURES

Accident	Event	Time sec
Major Steam Line		
1. Case A	Steam line ruptures	0
	Pressurizer empty	14
	Criticality attained	51
	2300 ppm boron reaches loops	79
2. Case B	Steam line ruptures	0
	Pressurizer empty	16
	Criticality attained	35
	2300 ppm boron reaches loops	78
3. Case C	Steam line ruptures	0
	Pressurizer empty	15
	Criticality attained	69
	2300 ppm boron reaches loops	74
4. Case D	Steam line ruptures	0
	Pressurizer empty	18
	Criticality attained	48
	2300 ppm boron reaches loops	73

Accident	Event	Time, sec
Rupture of Main Feedwater Pipe (Offsite Power Available)	Feedline rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in affected steam generator	14
	Rod begins to drop	16
	Auxiliary Feedwater is started	614
	Feedwater lines are purged and auxiliary feedwater is delivered to two of three intact steam generators	864
	Total RCS heat generation (decay heat + pump heat) decreases to auxiliary feedwater heat removal capability	5296
	Peak pressurizer level after initial outsurge reached	5320

Accident	Event	Time, sec
Rupture of Main Feedwater Pipe (Offsite Power Unavailable)	Feedline rupture occurs	10
	Low-low steam generator level reactor trip setpoint reached in affected steam generator	14
	Rod begins to drop	16
	Reactor coolant pump coastdown	18
	Auxiliary feedwater is started	614
	Feedwater lines are purged and auxiliary feedwater is delivered to two of three intact steam generators	864
	Peak pressurizer level after initial outsurge reached	2544
	Total RCS heat generation decreases to auxiliary feedwater heat removal capability	2904

TABLE 15.4-9

CORE PARAMETERS USED IN STEAM BREAK DNB ANALYSIS

Parameter	Case B, Time Point				
	1	2	3	4	5
Reactor vessel inlet temperature to sector connected to affected steam generator, °F	324.6	324.6	324.6	324.7	325.8
Reactor vessel inlet temperature to remaining sector, °F	422.7	422.6	422.6	422.5	422.5
RCS pressure, psia	602.9	603.5	604.3	605.2	606.6
RCS flow, %	100	100	100	100	100
Heat flux, %	19.0	19.0	19.0	19.0	19.0
Time, sec	266.2	268.2	270.2	272.2	274.2

TABLE 15.4-10

SUMMARY OF RESULTS FOR LOCKED ROTOR TRANSIENT

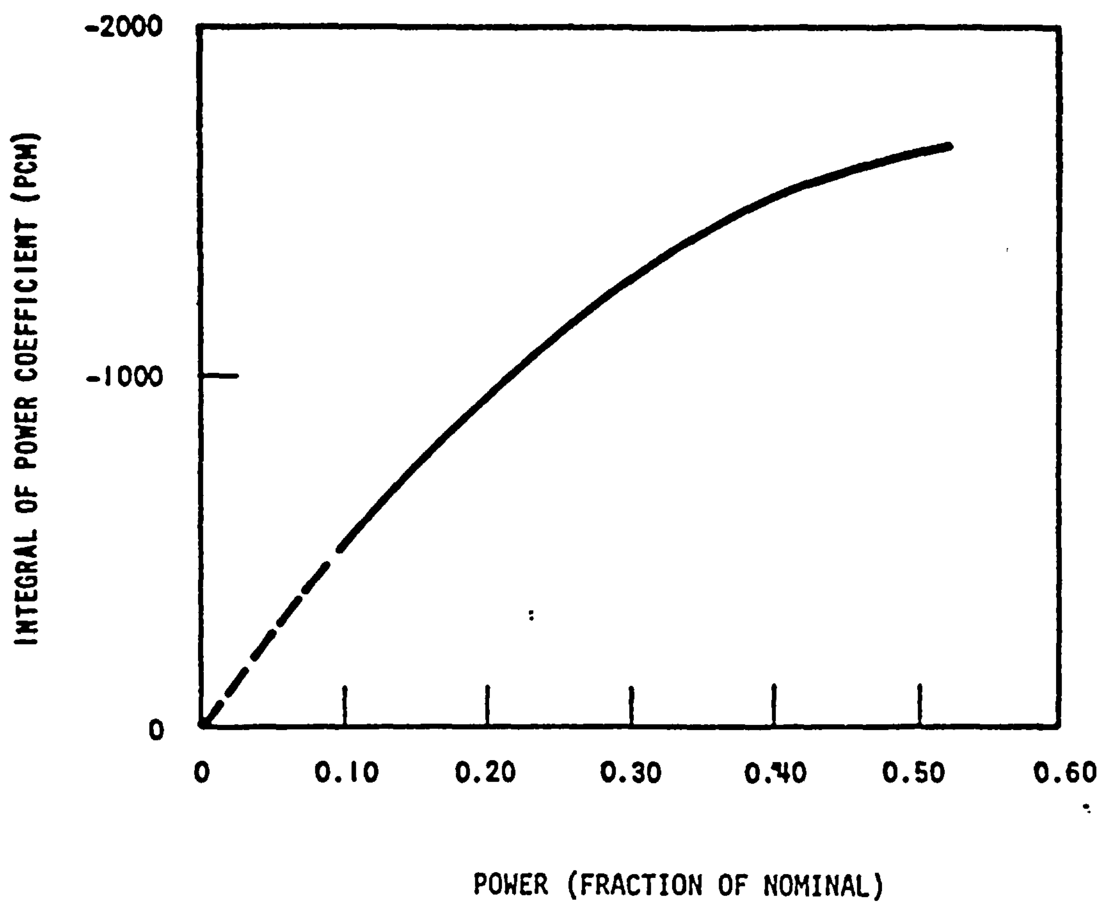
	4 Loops Operating Initially 1 Locked Rotor
Maximum RCS pressure, psia	2672
Maximum clad temperature, °F core hot spot	2040
Amount of Zr-H_2O at core hot spot, % by weight	0.7%

TABLE 15.4-11

PARAMETERS USED IN THE ANALYSIS OF THE ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENT

Time in Life	Beginning	Beginning	End	End
Power level, %	102	0.0	102	0.0
Ejected rod worth, %Δk	0.20	0.785	0.21	0.85
Delayed neutron fraction, %	0.55	0.55	0.44	0.44
Feedback reactivity weighting	1.30	2.071	1.30	3.55
Trip reactivity, %Δk	4	2	4	2
F_Q before rod ejection	2.50	-	2.50	-
F_Q after rod ejection	6.70	13	6.50	21.50
Number of operating pumps	4	2	4	2
Maximum fuel pellet average temperature, °F	4154	3509	3812	3408
Maximum fuel center temperature, °F	*	4025	*	3849
Maximum clad average temperature, °F	2434	2660	2218	2632
Maximum fuel stored energy, cal/gm	183	149	165	144

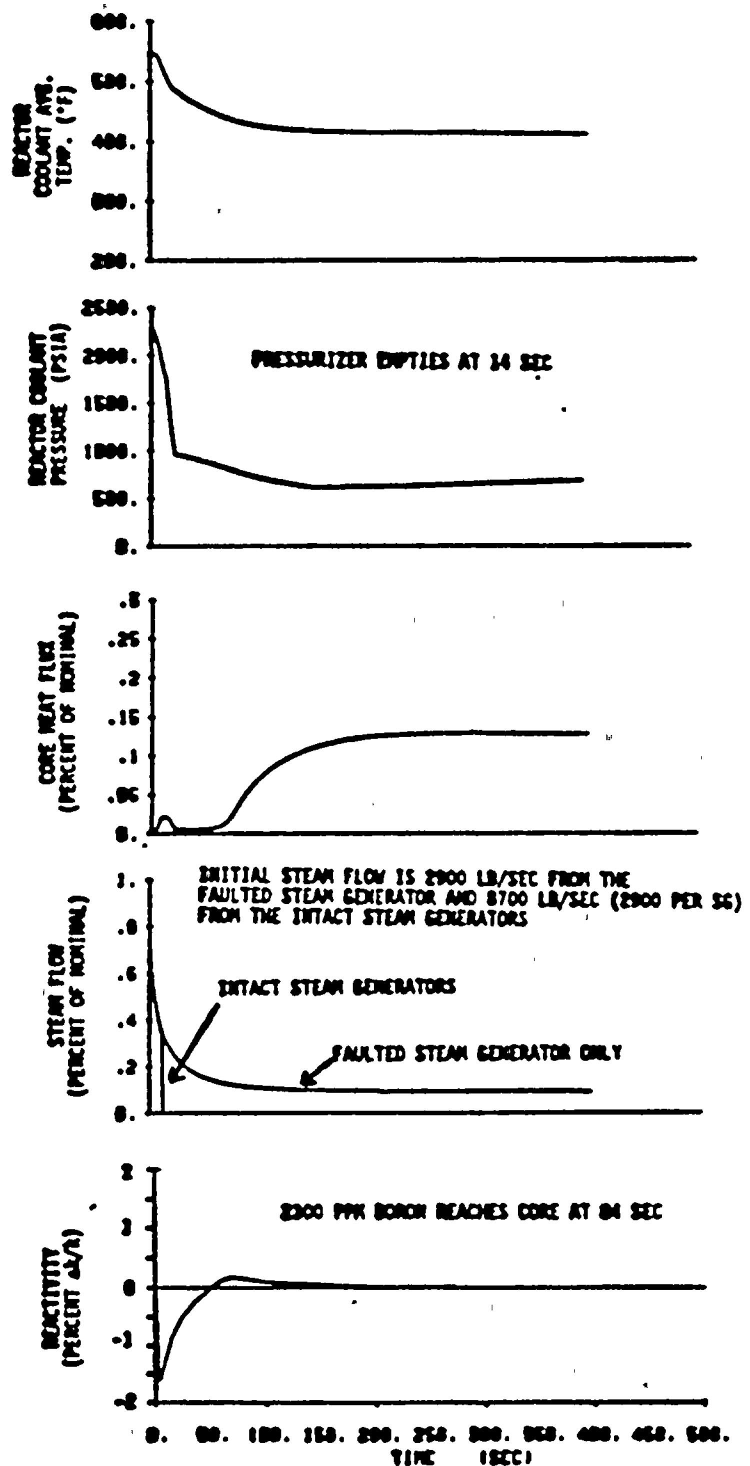
*Less than 10% fuel melt



DIABLO CANYON UNITS 1 AND 2

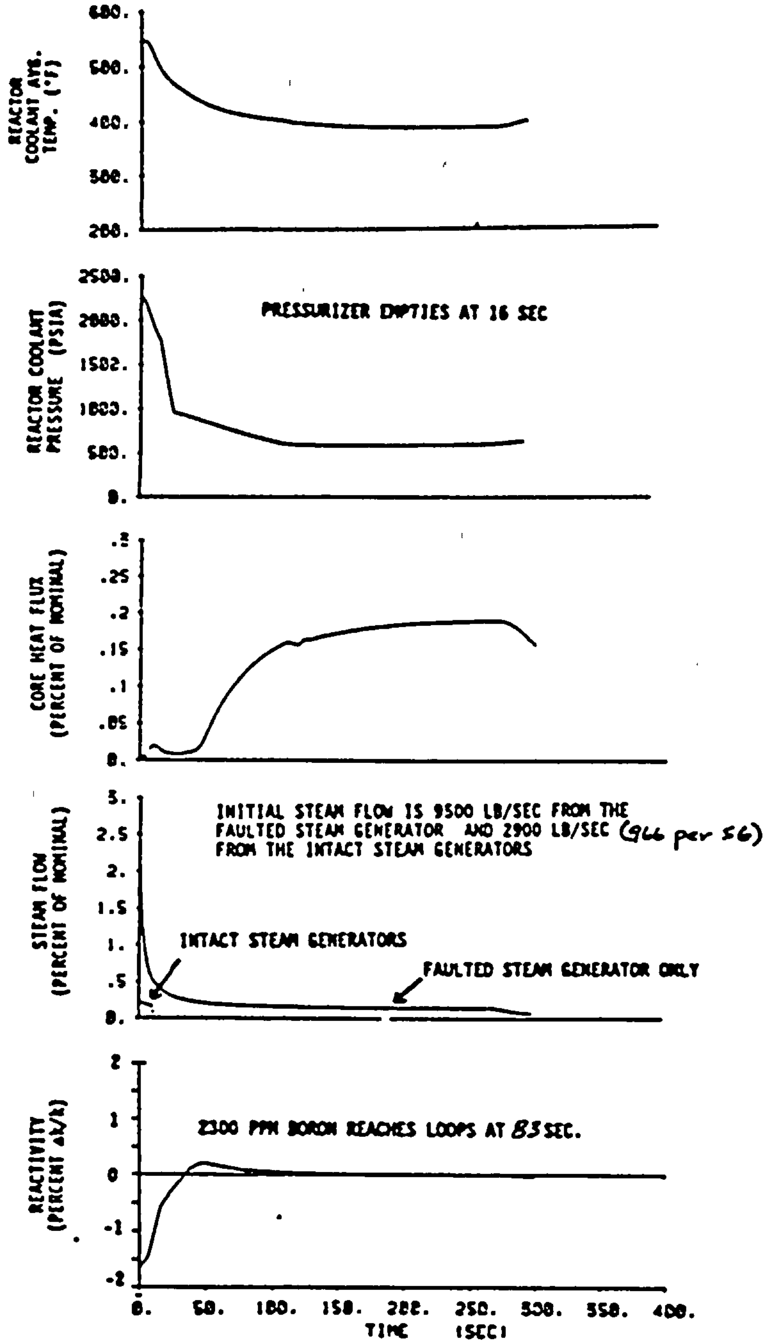
FIGURE 15.4.2-1

VARIATION OF REACTIVITY
WITH POWER AT CONSTANT CORE
AVERAGE TEMPERATURE



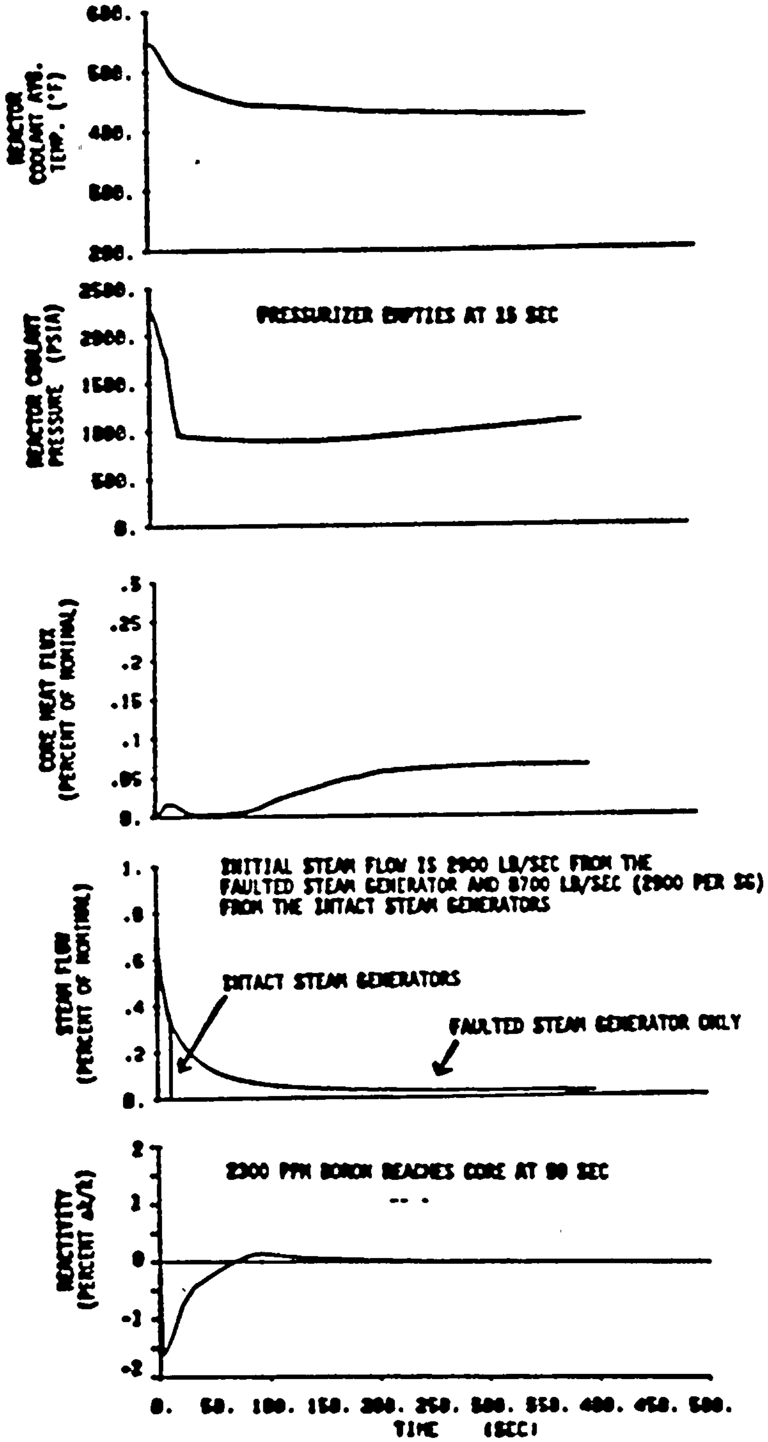
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.2-2
TRANSIENT RESPONSE TO STEAM LINE BREAK DOWNSTREAM OF FLOW MEASURING NOZZLE WITH SAFETY INJECTION AND OFFSITE POWER
(CASE A)



DIABLO CANYON UNITS 1 AND 2

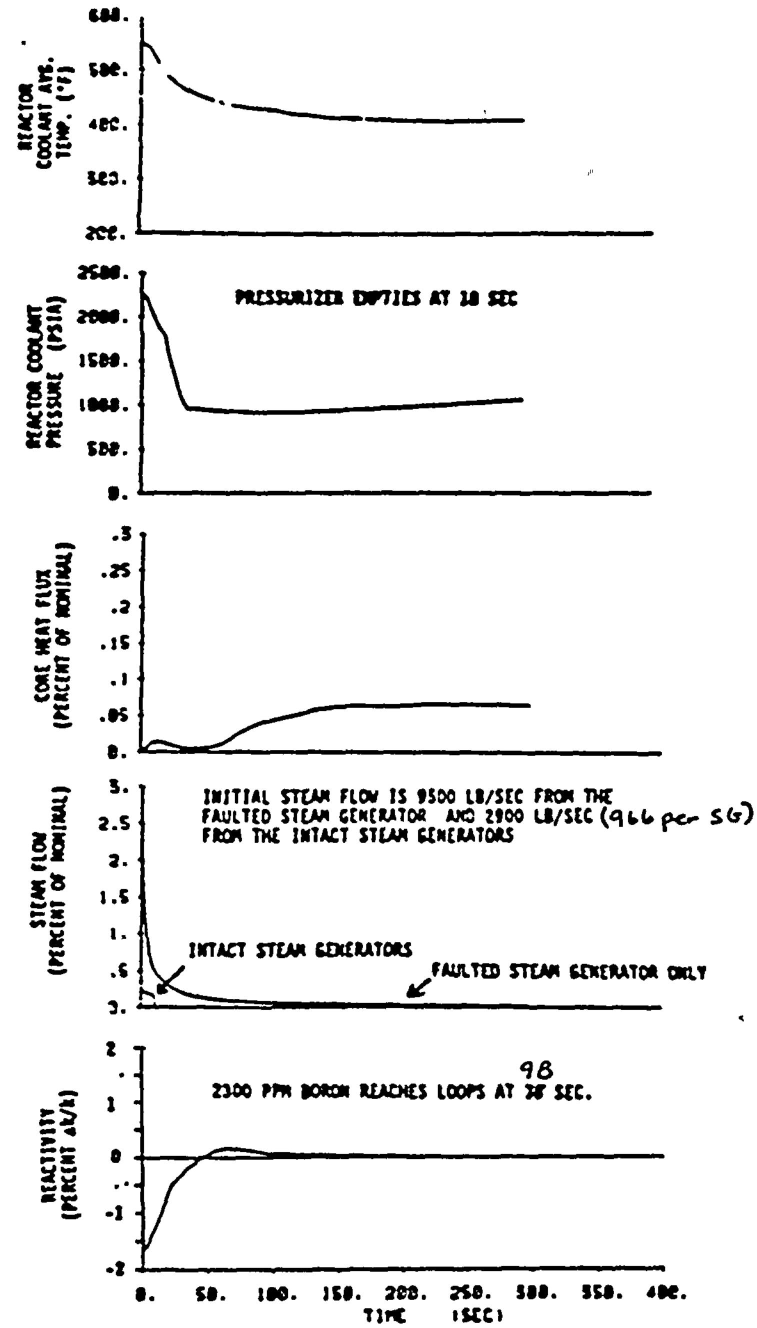
FIGURE 15.4.2-3
TRANSIENT RESPONSE TO STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITH SAFETY INJECTION AND OFFSITE POWER
(CASE B)



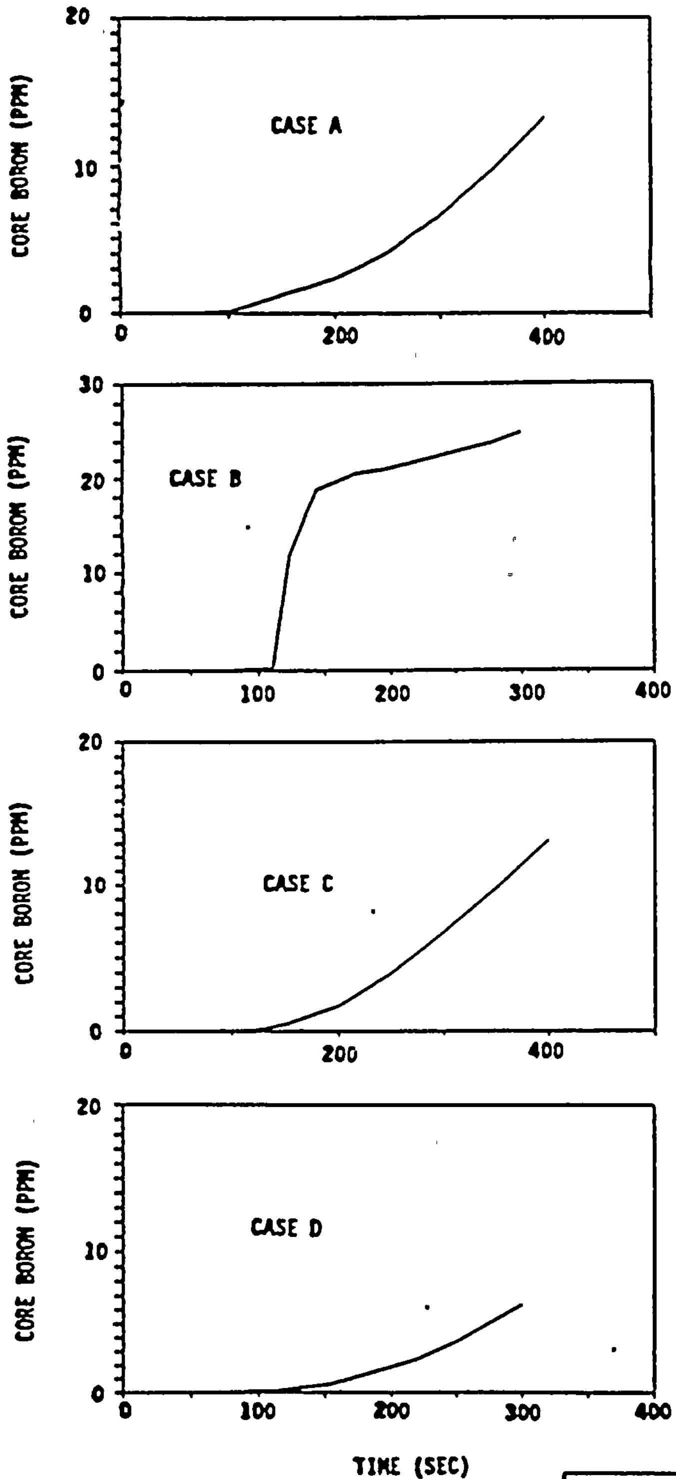
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.2-4

TRANSIENT RESPONSE TO STEAM LINE BREAK DOWNSTREAM OF FLOW MEASURING NOZZLE WITH SAFETY INJECTION AND WITHOUT OFFSITE POWER (CASE C)



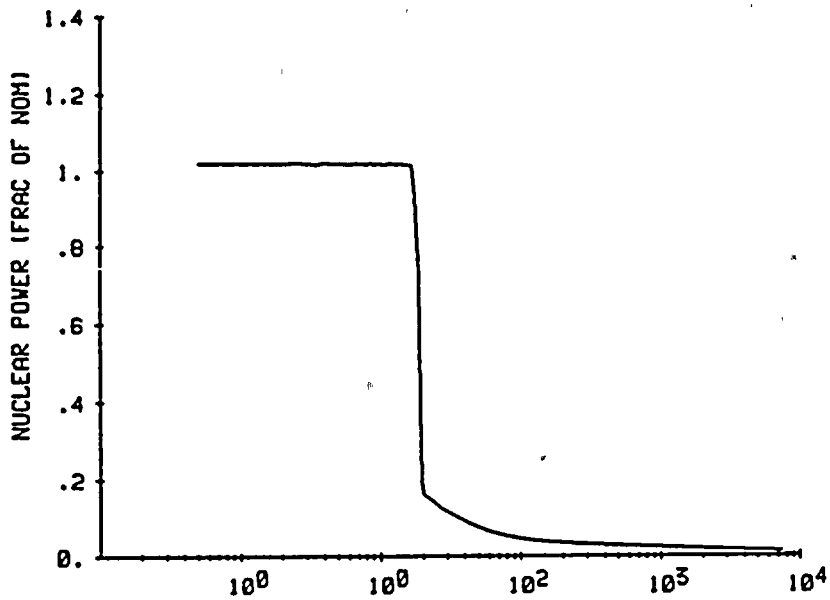
DIABLO CANYON UNITS 1 AND 2
FIGURE 15.4.2-5 TRANSIENT RESPONSE TO STEAM LINE BREAK AT EXIT OF STEAM GENERATOR WITH SAFETY INJECTION AND WITHOUT OFFSITE POWER (CASE D)

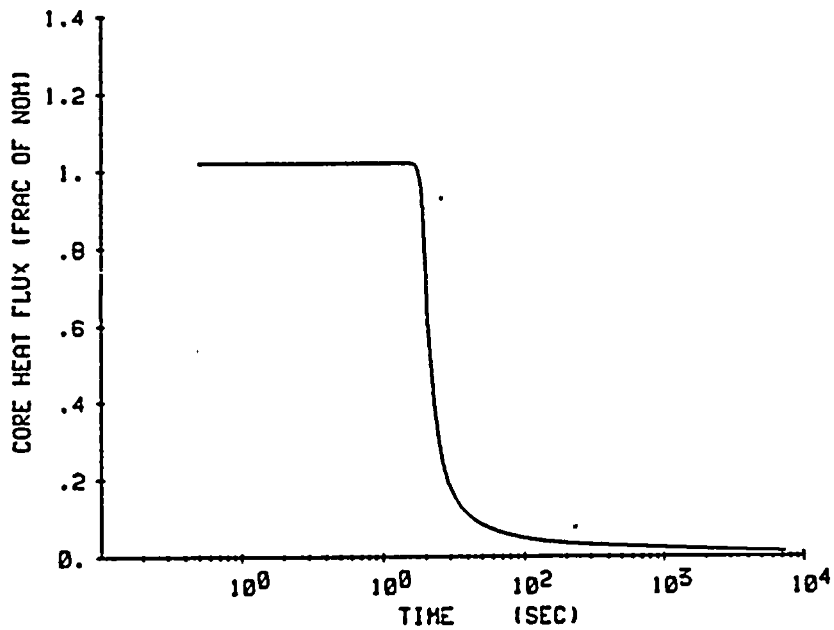


DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.2-6

CORE BORON VS. TIME

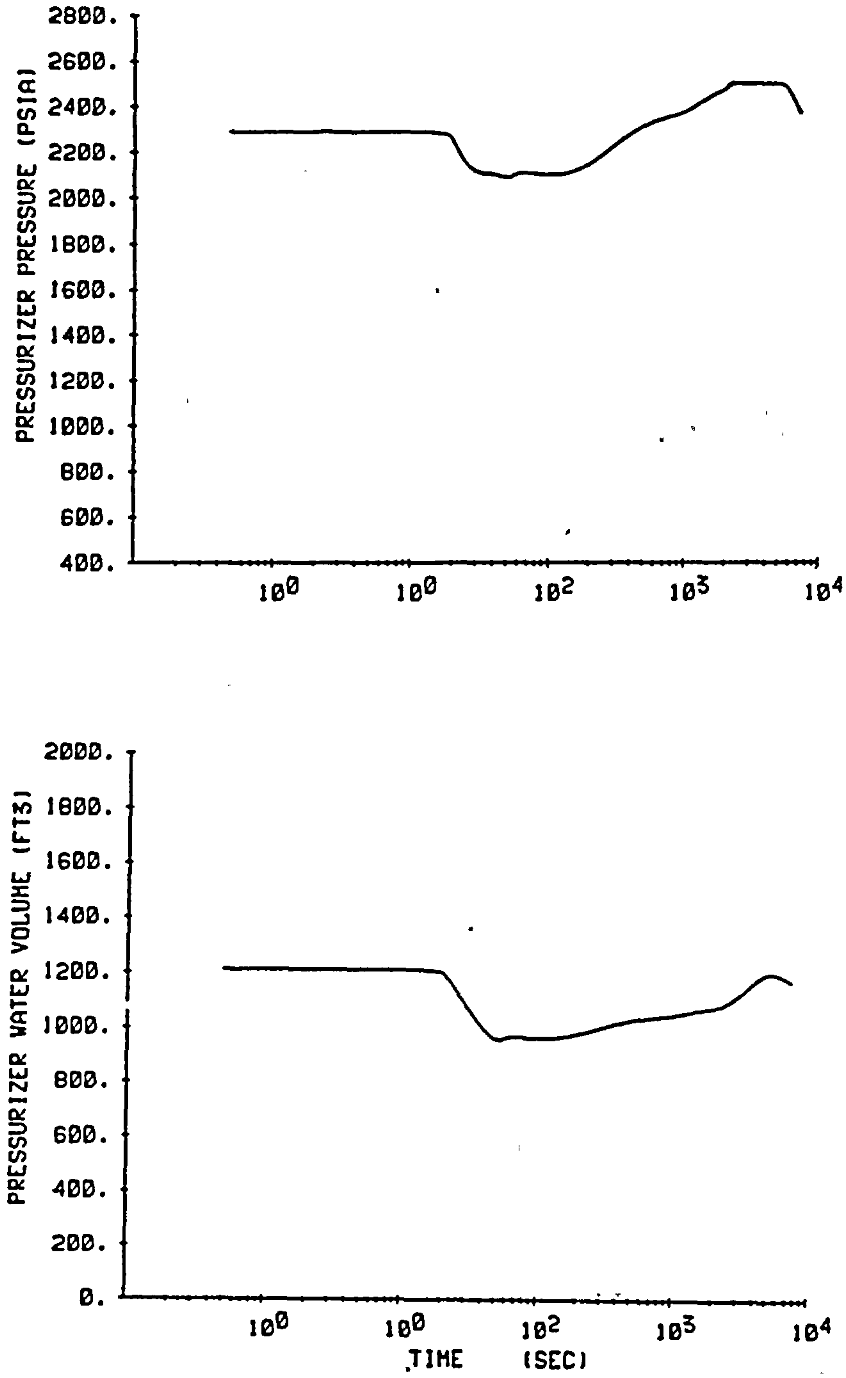




DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.2-7

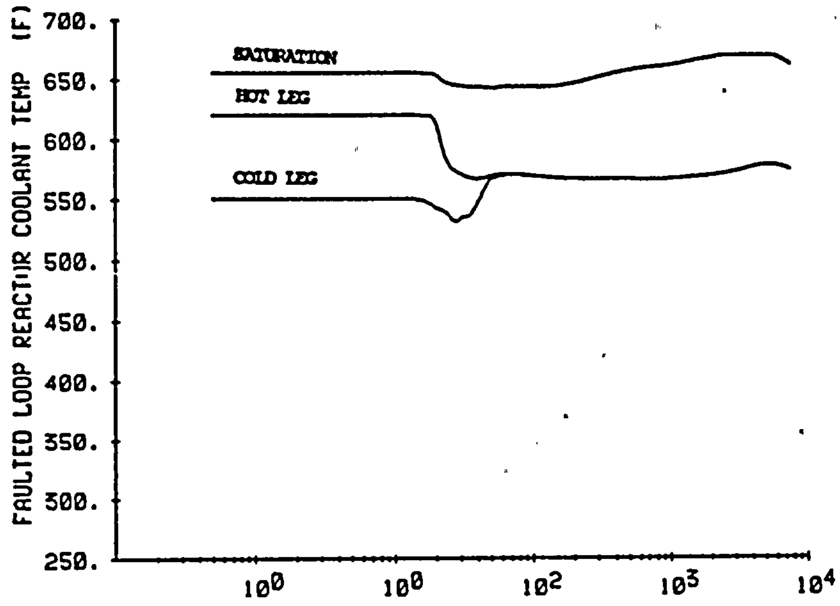
NUCLEAR POWER TRANSIENT AND CORE HEAT FLUX TRANSIENT FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE

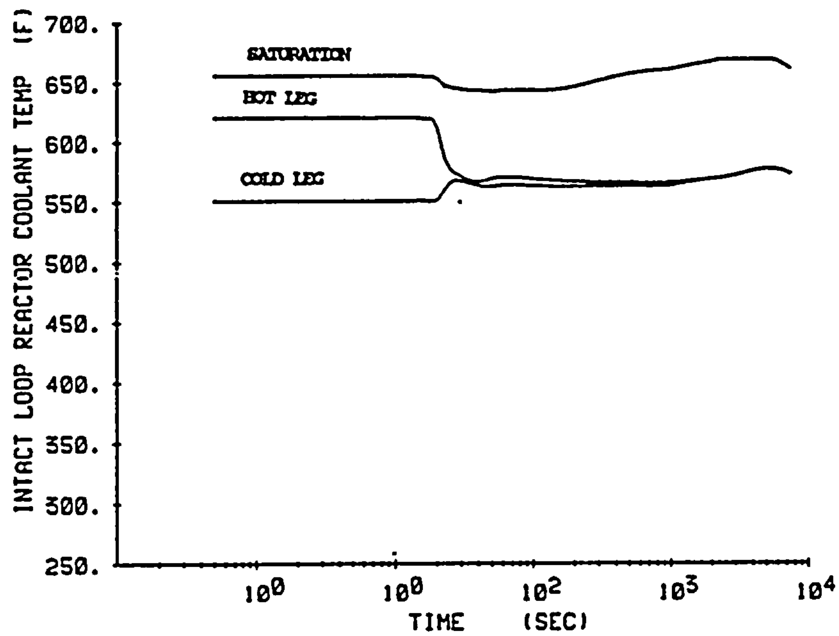


DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.2-8

PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE

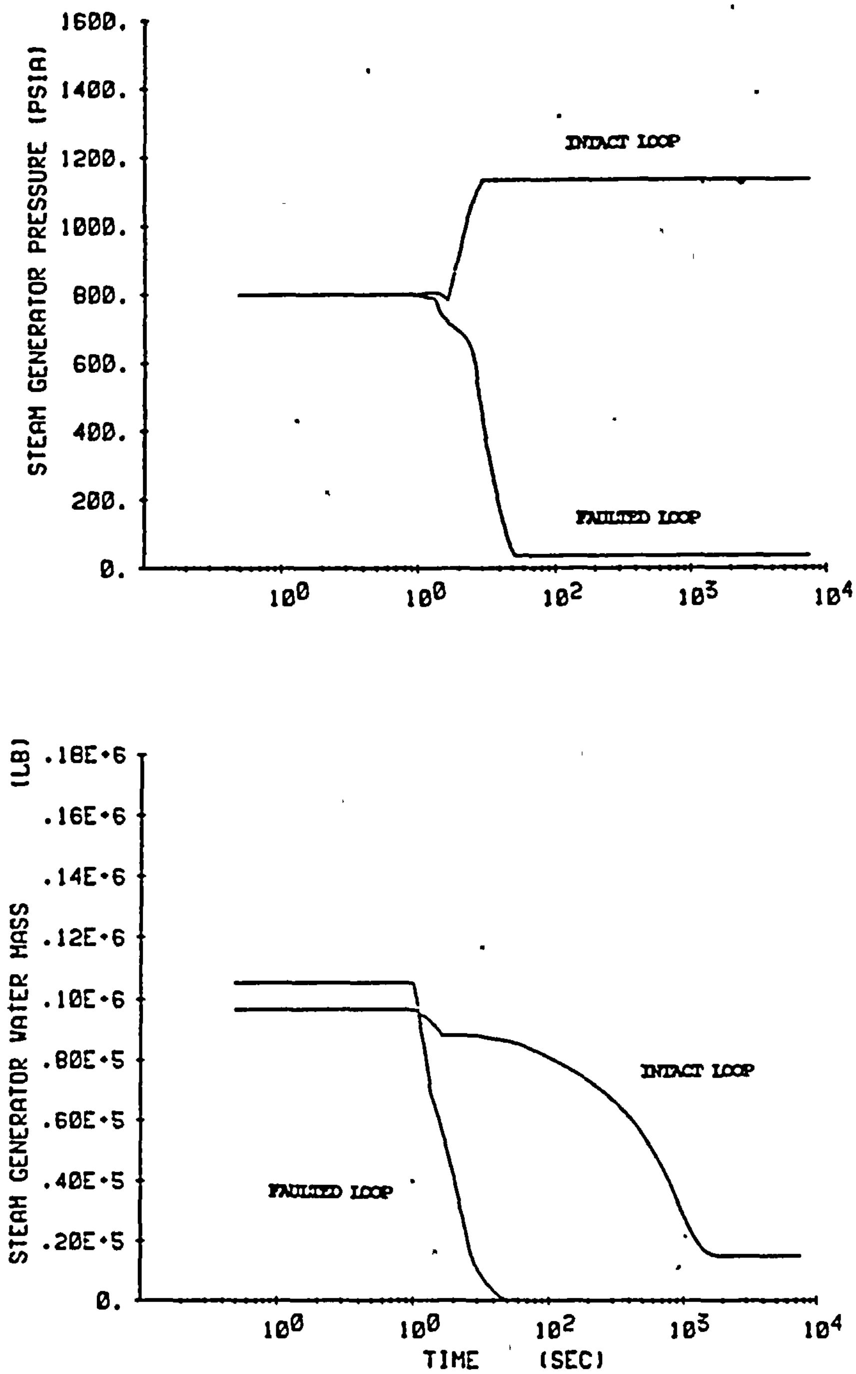




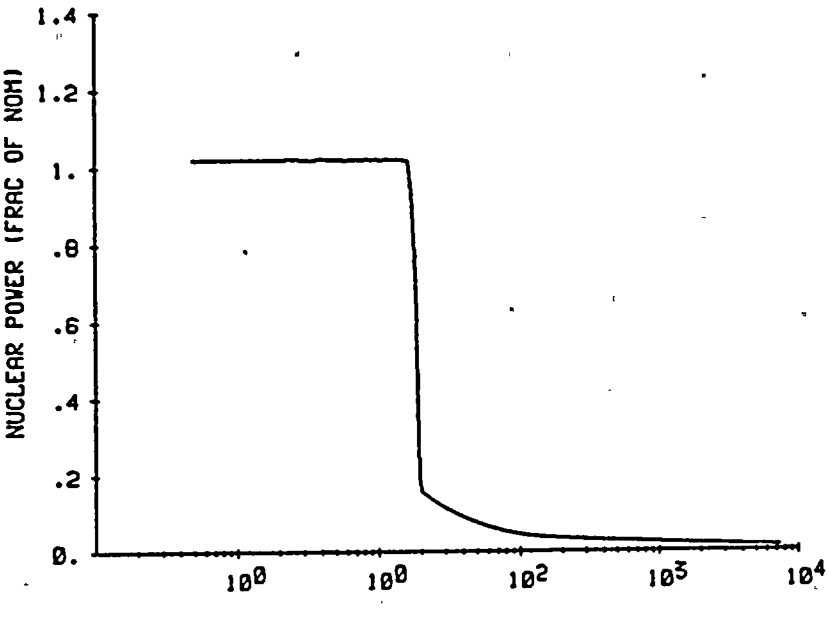
DIABLO CANYON UNITS 1 AND 2

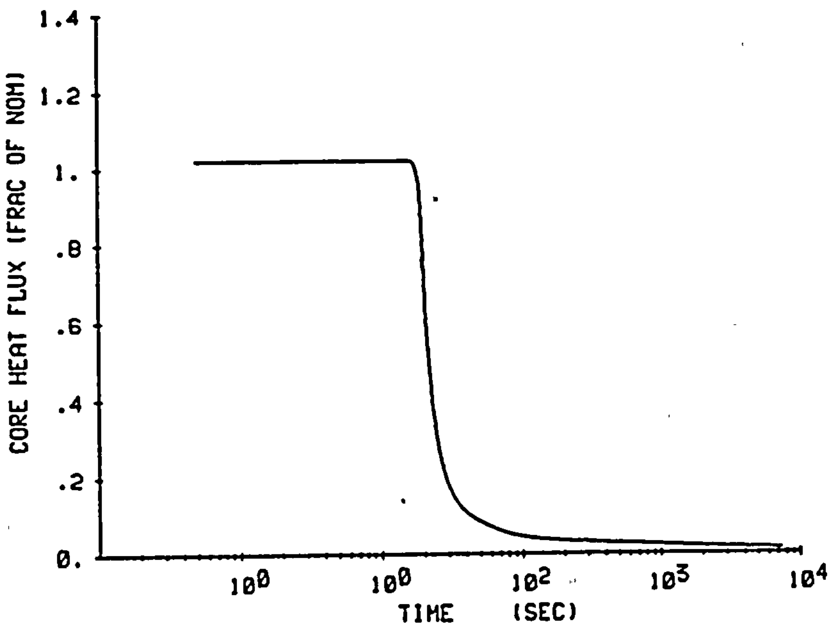
FIGURE 15.4.2-9

REACTOR COOLANT TEMPERATURE TRANSIENTS FOR THE FAULTED AND INTACT LOOPS FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE



DIABLO CANYON UNITS 1 AND 2
FIGURE 15.4.2-10
STEAM GENERATOR PRESSURE AND WATER MASS TRANSIENTS FOR MAIN FEEDLINE RUPTURE WITH OFFSITE POWER AVAILABLE

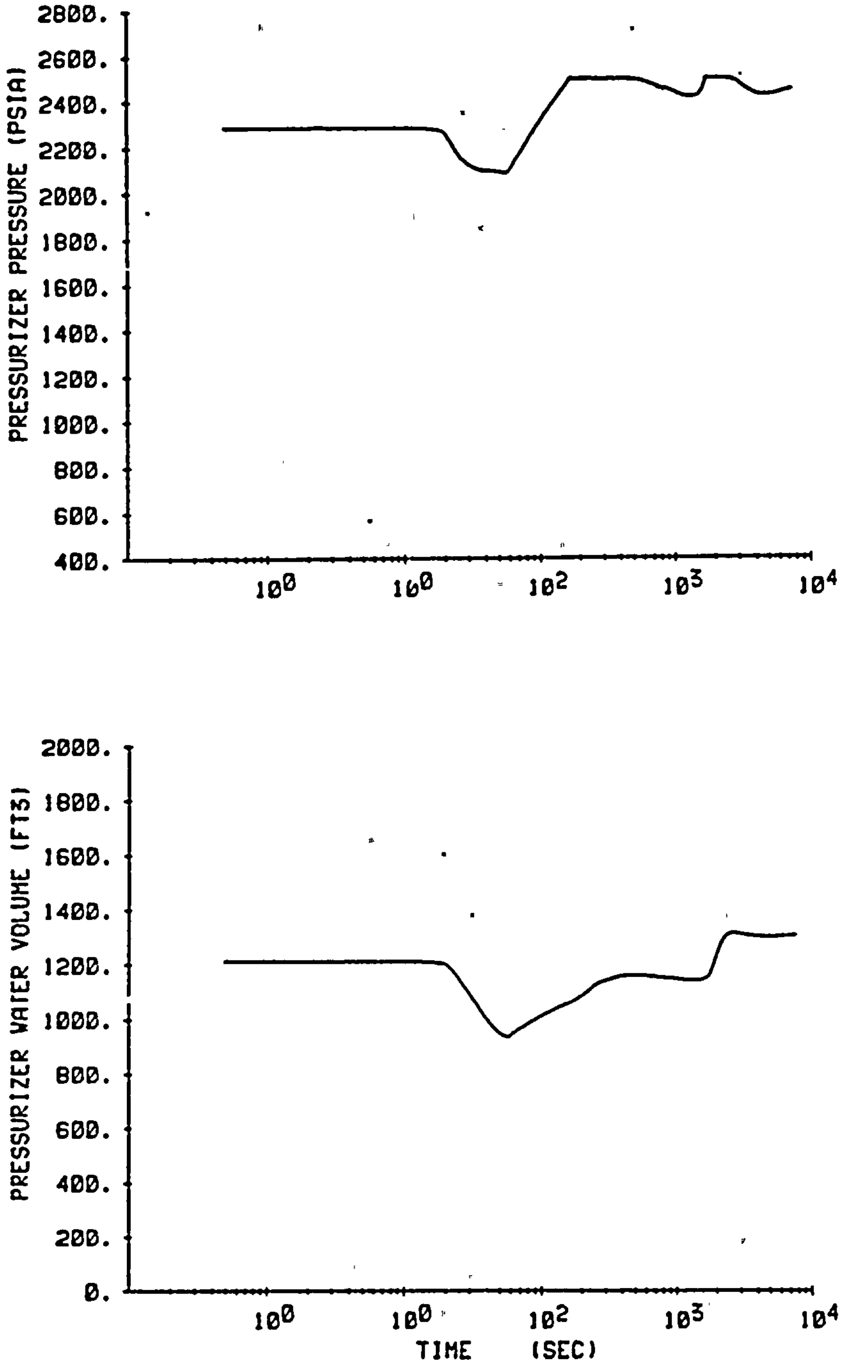




DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.2-11

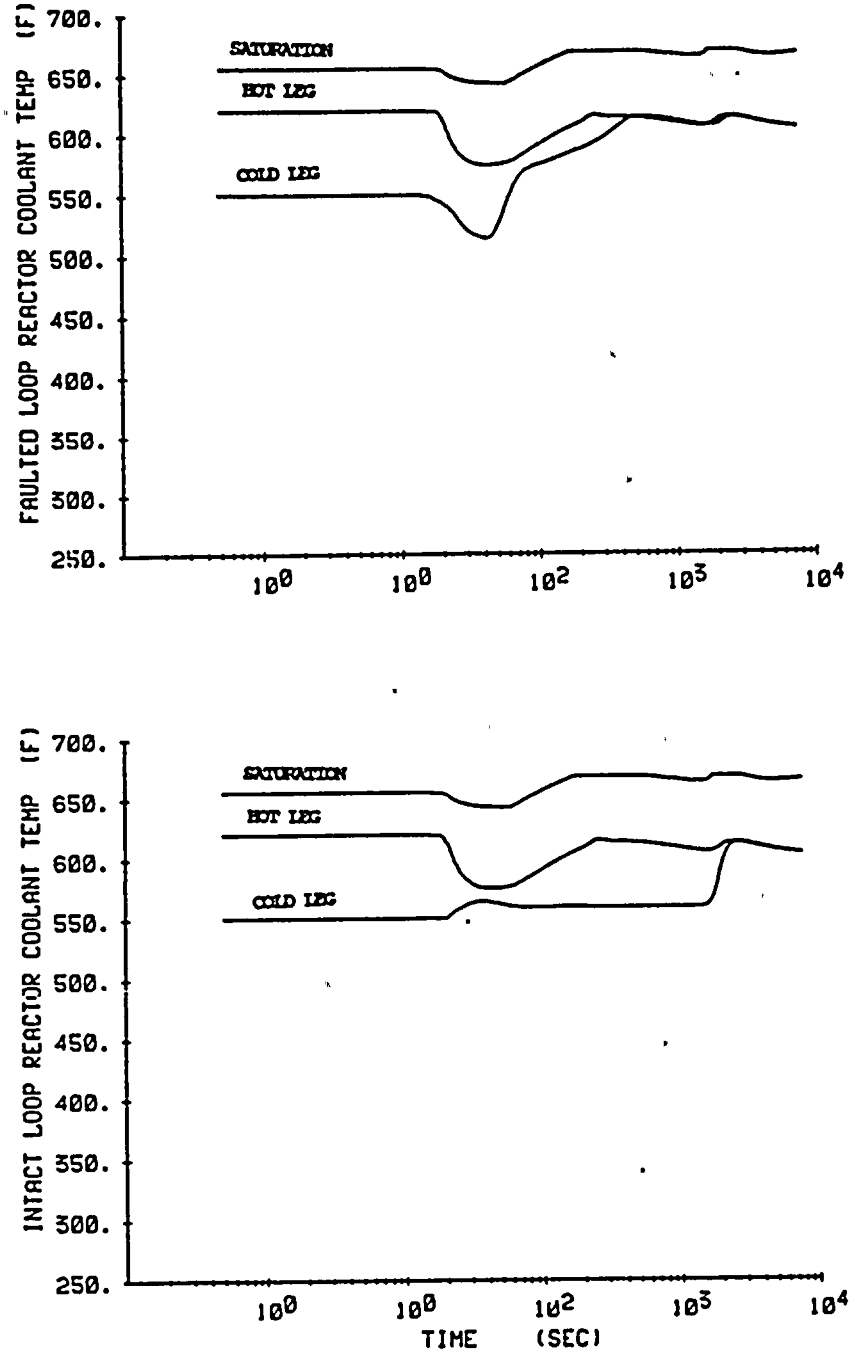
NUCLEAR POWER TRANSIENT AND CORE HEAT FLUX TRANSIENT FOR MAIN FEEDLINE RUPTURE WITHOUT OFFSITE POWER AVAILABLE



DIABLO CANYON UNITS 1 AND 2

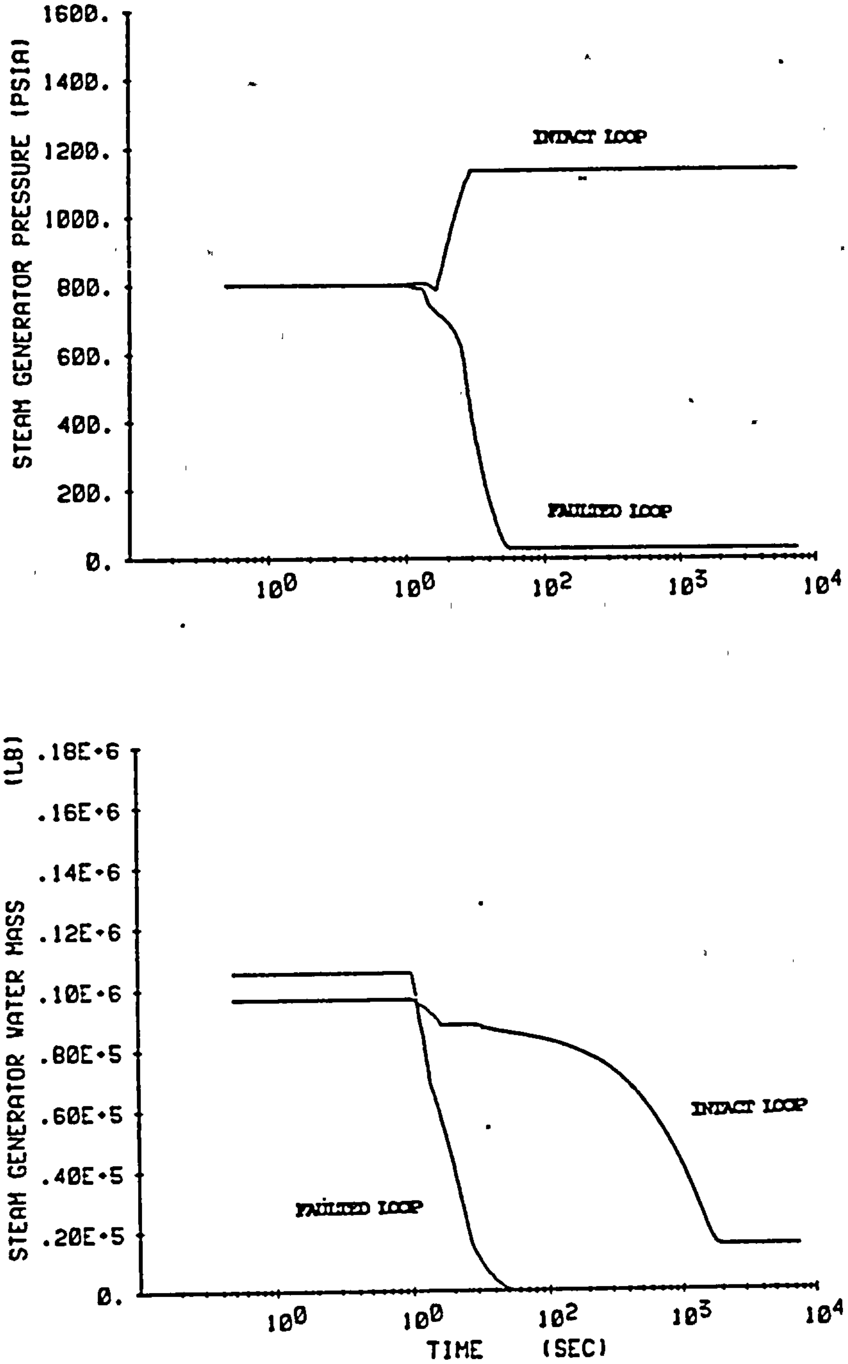
FIGURE 15.4.2-12

PRESSURIZER PRESSURE AND WATER VOLUME TRANSIENTS FOR MAIN FEEDLINE RUPTURE WITHOUT OFFSITE POWER AVAILABLE



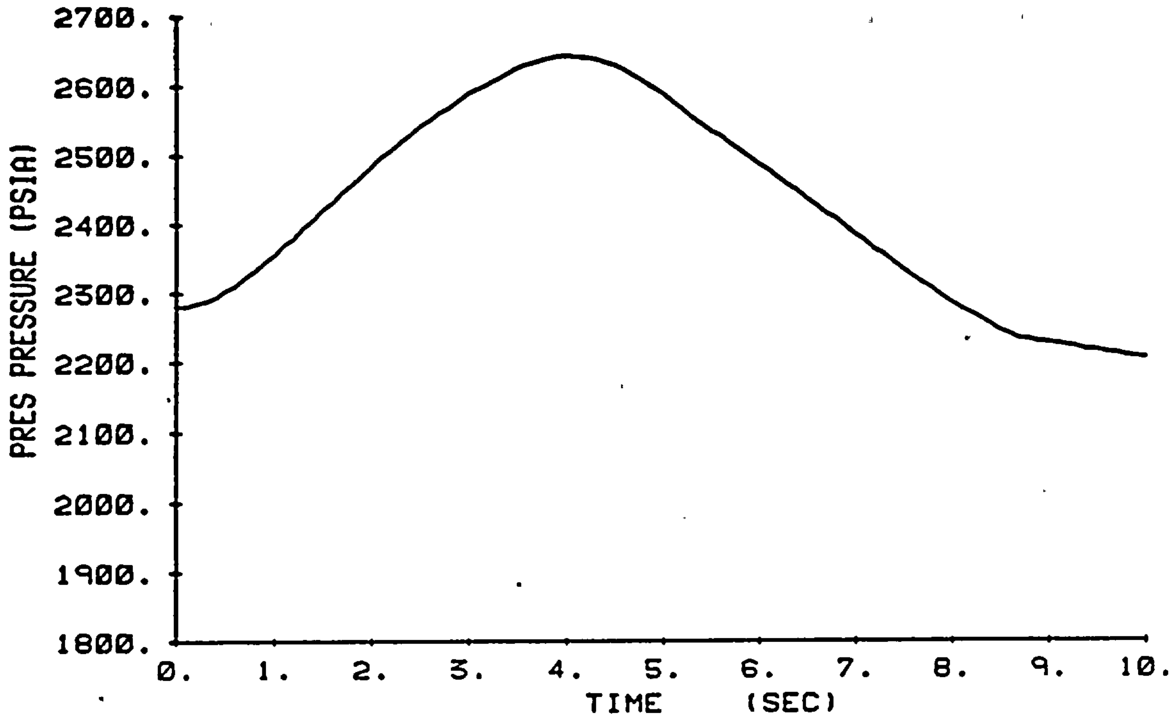
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.2-13
REACTOR COOLANT TEMPERATURE
TRANSIENTS FOR THE FAULTED
AND INTACT LOOPS FOR MAIN
FEEDLINE RUPTURE WITHOUT
OFFSITE POWER AVAILABLE



DIABLO CANYON UNITS 1 AND 2

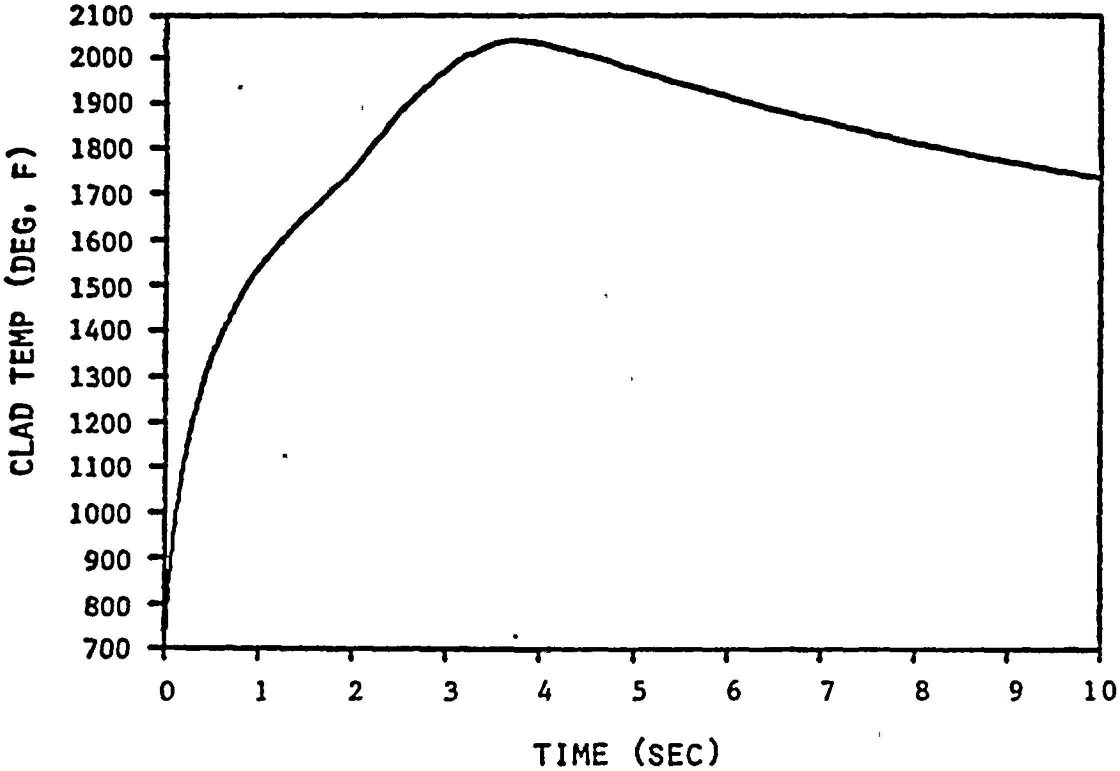
FIGURE 15.4.2-14
STEAM GENERATOR PRESSURE AND
STEAM GENERATOR WATER MASS
TRANSIENTS FOR MAIN
FEEDLINE RUPTURE WITHOUT
OFFSITE POWER AVAILABLE



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.4-1

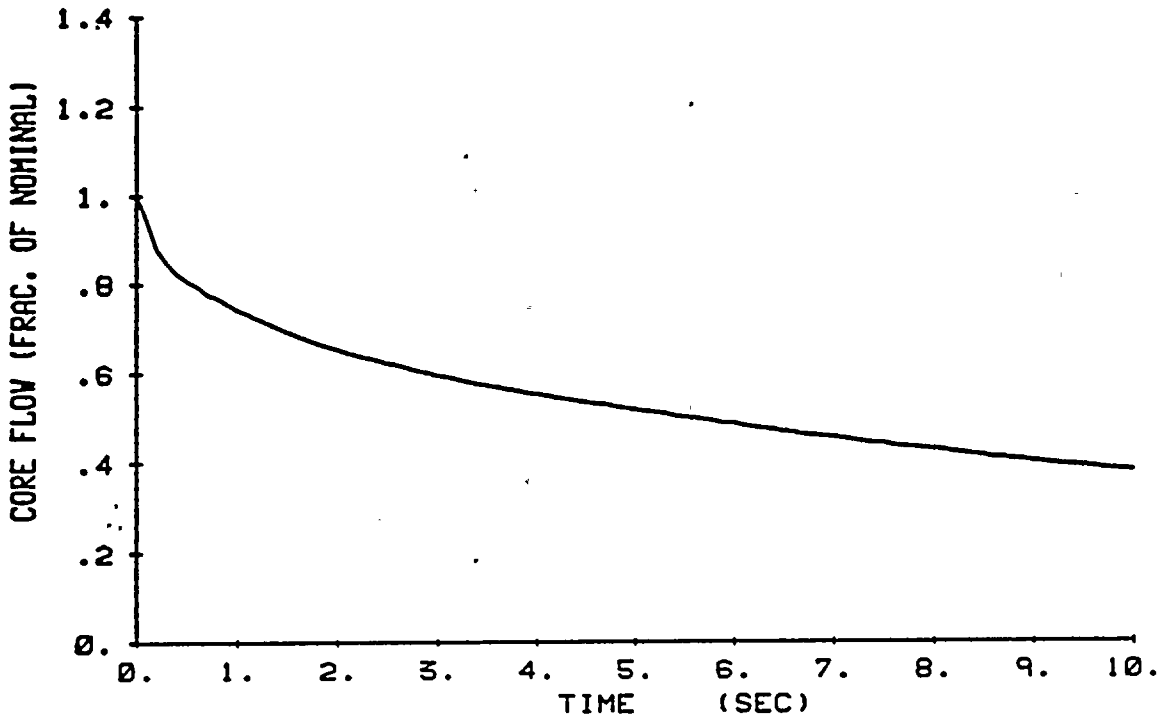
ALL LOOPS OPERATING,
ONE LOCKED ROTOR,
PRESSURE VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.4-2

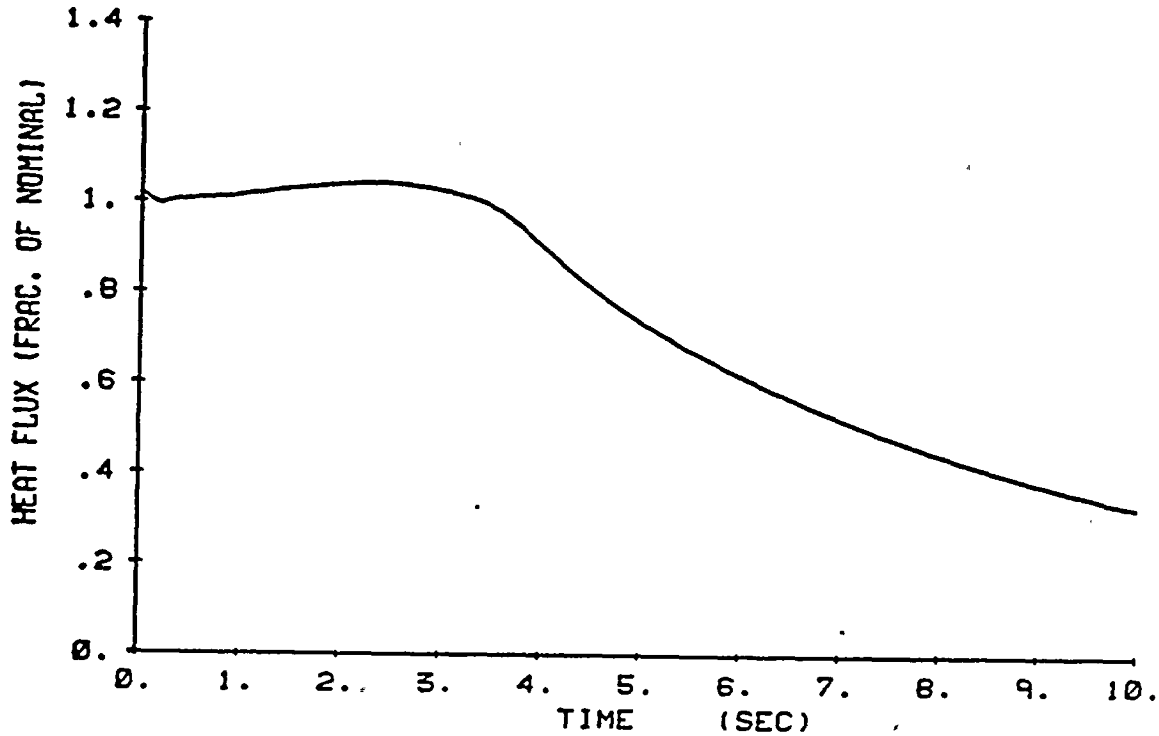
ALL LOOPS OPERATING,
ONE LOCKED ROTOR,
CLAD TEMPERATURE VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.4-3

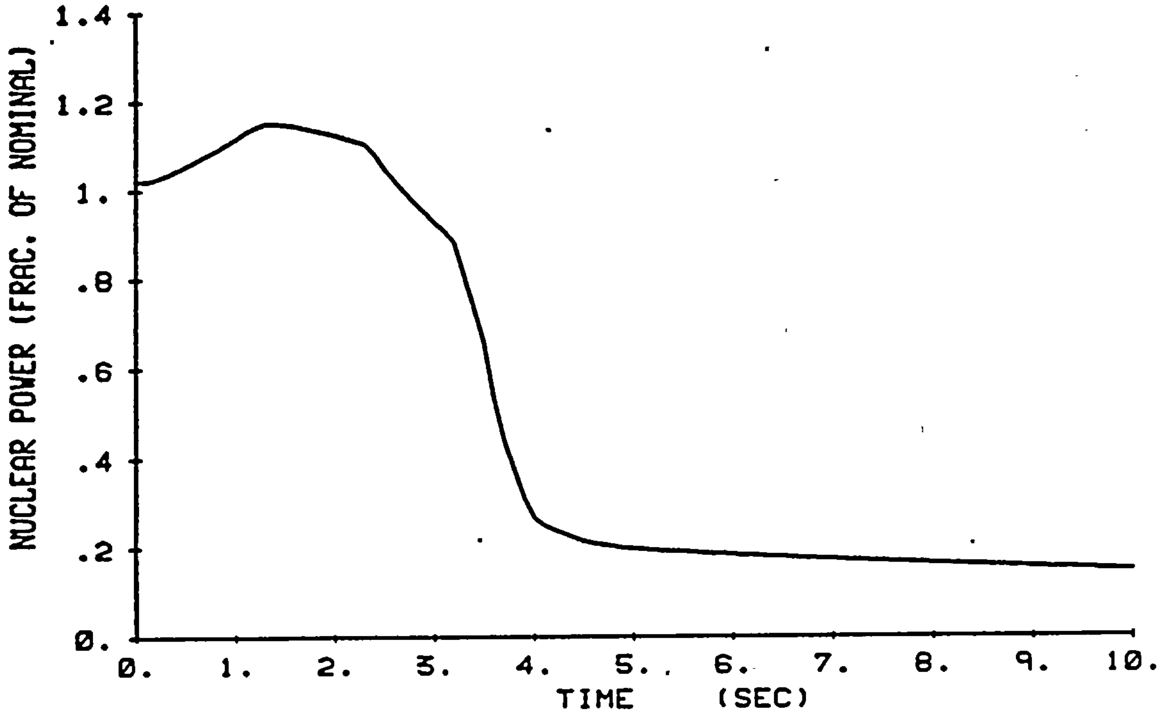
ALL LOOPS OPERATING.
ONE LOCKED ROTOR.
FLOW COASTDOWN VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.4-4

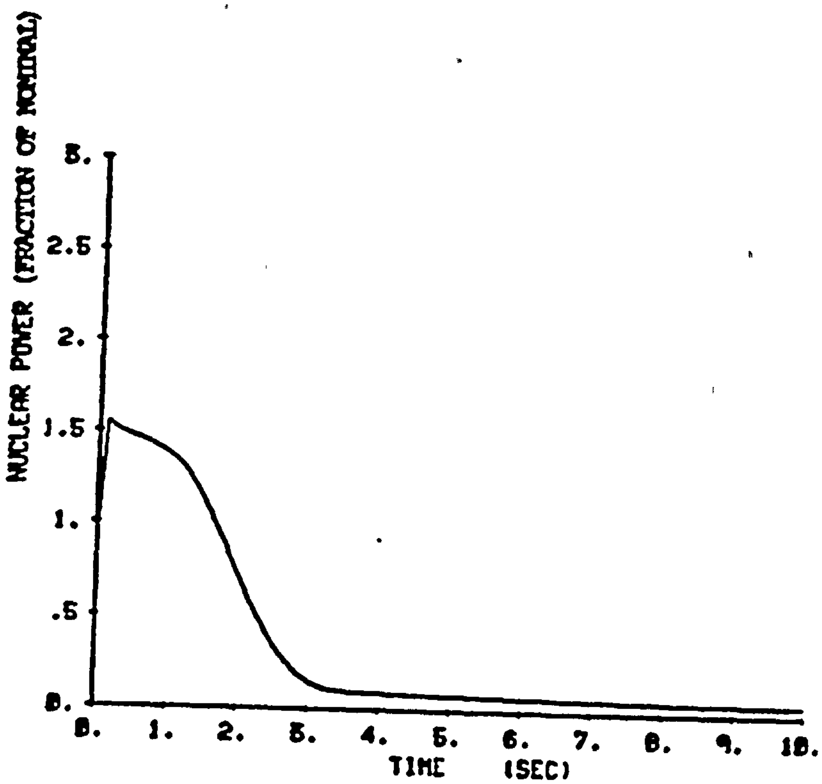
ALL LOOPS OPERATING,
ONE LOCKED ROTOR,
HEAT FLUX VERSUS TIME



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.4-5

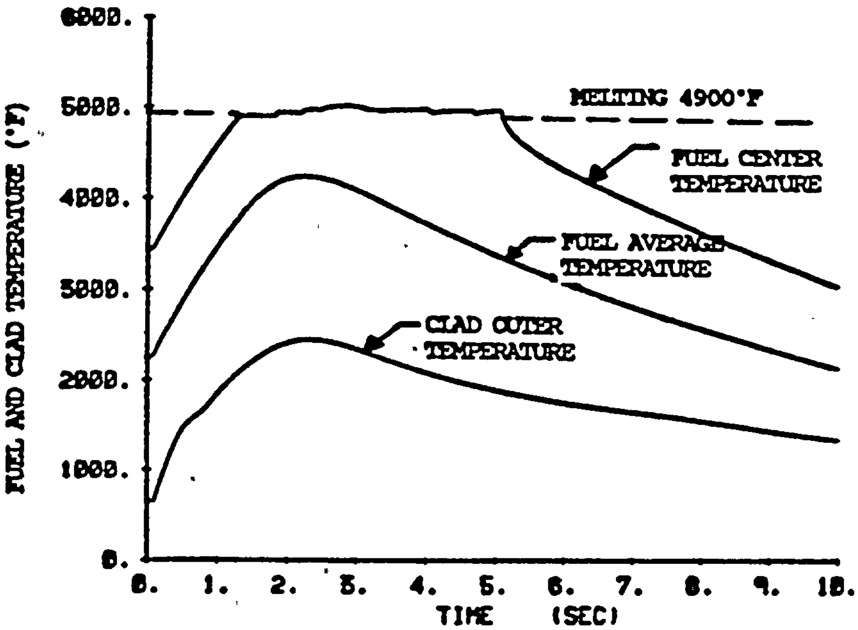
ALL LOOPS OPERATING.
ONE LOCKED ROTOR.
NUCLEAR POWER VERSUS TIME



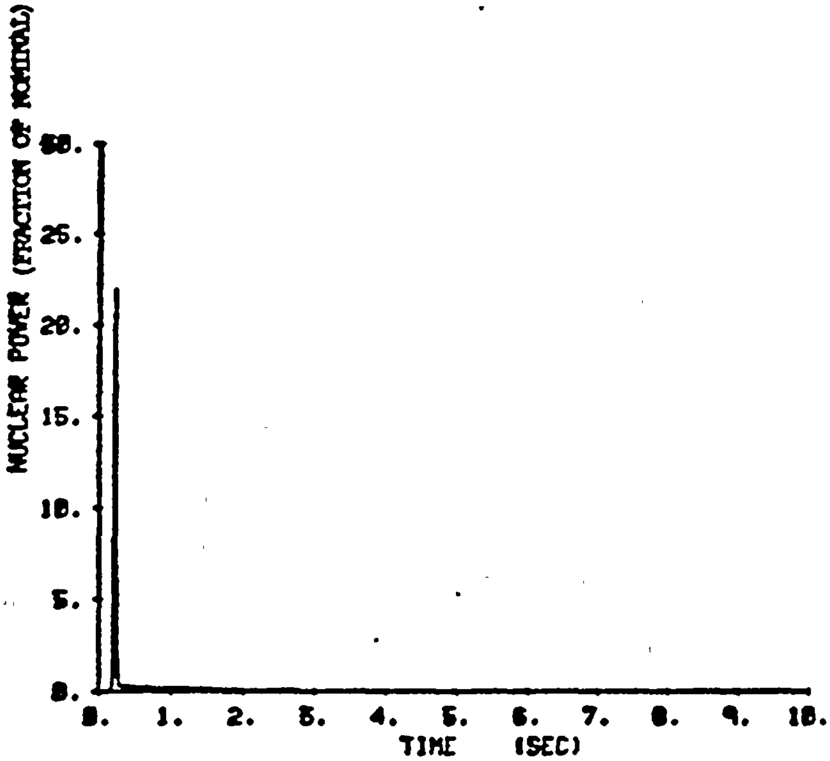
DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.6-1

NUCLEAR POWER TRANSIENT, BOL HFP, ROD EJECTION ACCIDENT



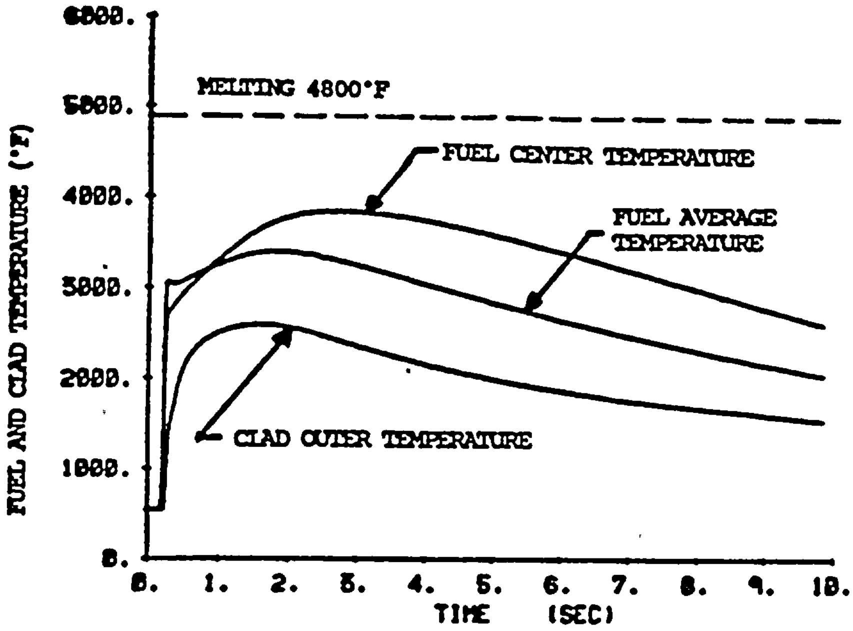
DIABLO CANYON UNITS 1 AND 2
FIGURE 15.4.6-2 HOT SPOT FUEL AND CLAD TEMPERATURES VERSUS TIME, BOL, HFP, ROD EJECTION ACCIDENT



DIABLO CANYON UNITS 1 AND 2

FIGURE 15.4.6-3

NUCLEAR POWER TRANSIENT, EOL HZP, ROD EJECTION ACCIDENT



DIABLO CANYON UNITS 1 AND 2
FIGURE 15.4.6-4 HOT SPOT FUEL AND CLAD TEMPERATURES VERSUS TIME, EOL, HZP, ROD EJECTION ACCIDENT

Appendix B

LOCA ANALYSIS

CONTENTS (Cont'd)

CONTENTS (Cont'd)

INSERT Y

INSERT Z

CHAPTER 15

ACCIDENT ANALYSIS

TABLES

FIGURES (Cont'd)

Chapter 15

FIGURES (Cont'd)

FIGURES (Cont'd)

FIGURES (Cont'd)

FIGURES (Cont'd)

15.3 CONDITION III - INFREQUENT FAULTS

By definition, Condition III occurrences are faults that may occur very infrequently during the life of the plant. They will be accompanied with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system (RCS) or containment barriers. For the purposes of this report the following faults have been grouped into this category:

(1) Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, that actuates emergency core cooling

(2) Minor secondary system pipe breaks

(3) Inadvertent loading of a fuel assembly into an improper position

(4) Complete loss of forced reactor coolant flow

(5) Single rod cluster control assembly (RCCA) withdrawal at full power.

Each of these infrequent faults are analyzed in this section. In general, each analysis includes an identification of causes and description of the accident, an analysis of effects and consequences, a presentation of results, and relevant conclusions.

The time sequences of events during three Condition III faults of type (1) (small break loss-of-coolant accident) are shown in Table 15.3-1.

15.3.1 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes that Actuate Emergency Core Cooling System

15.3.1.1 Identification of Causes and Accident Description

A loss-of-coolant accident (LOCA) is defined as a rupture of the RCS piping or of any line connected to the system. See Section 3.6 for a more detailed description of the LOCA boundary limits. Ruptures of small cross section will cause expulsion of the coolant at a rate that can be accommodated by the charging pumps that would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant that would be released to the containment contains fission products.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2250 psia. A makeup flow rate from one centrifugal charging pump is typically adequate to sustain pressurizer level at 2250 psia for a break through a 0.375-inch-diameter hole. This break results in a loss of approximately 17.5 lb/sec.

Should a larger break occur, depressurization of the RCS causes fluid to flow to the RCS from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low-pressure trip setpoint is reached. The safety injection system (SIS) is actuated when the appropriate pressurizer low-pressure setpoint is reached. Reactor trip and SIS actuation are also initiated by a high containment pressure signal. The consequences of the accident are limited in two ways:

(1) Reactor trip and borated water injection complement void formation in causing rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay

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

Before the break occurs, the plant is in an equilibrium condition; i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary system, system pressure increases and steam dump may occur. Makeup to the secondary side is automatically provided by the auxiliary feedwater (AFW) pumps. The safety injection signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates emergency feedwater flow by starting AFW pumps. The secondary flow aids in the reduction of RCS pressure. When the RCS depressurizes to below 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.

<u>15.3.1.2 Analysis of Effects and Consequences</u>

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 ~~25~~ 27 seconds after the generation of a safety injection signal.

For the analysis, the SIS delivery considers pumped injection flow that is depicted on 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% from the design head. The ~~25~~ 27 seconds 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 emergency core cooling system (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 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 on Figure 15.3-2. The extent to which the core is uncovered for the same break is presented on Figure 15.3-3. The maximum hot spot cladding temperature reached during the transient is ~~1280°F~~, including the 1358°F effects of fuel densification as described in Reference 3. The peak cladding temperature transient for the limiting break size is shown on Figure 15.3-4. The core steam flowrate for the 4-inch break is shown on 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 on Figure 15.3-6. Also, the hot spot fluid temperature for the worst break is shown on 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 on Figures 15.3-14a through 15.3-14f.

The core power (dimensionless) transient following the accident (relative to reactor scram time) is shown on Figure 15.3-8. The reactor shutdown time (~~4.4~~ 4.7 sec) is equal to the reactor trip signal processing time (2.0 sec) plus ~~2.4~~ 2.7 seconds for complete rod insertion. During this rod insertion period, the reactor is conservatively assumed to operate at rated power.

Several figures are also presented for the additional break sizes analyzed. Figures 15.3-9 and 15.3-10 present the RCS pressure transient for the 3-inch and 6-inch breaks, respectively, and Figures 15.3-11 and 15.3-12 present the core mixture height plots for both breaks. The peak cladding temperature transient for the ~~6~~ 3-inch break is shown on Figure 15.3-13. ~~No~~ The peak cladding temperature plot is shown on Figure 15.3-14 for the ~~3~~ 6-inch break~~, since no core uncovery occurs (see Figure 15.3-11) and no clad heatup is computed~~.

The small break analysis was performed with the Westinghouse ECCS Small Break Evaluation Model[12, 4] approved for this use by the Nuclear Regulatory Commission in May 1985.

Insert A

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.

15.3.2 Minor Secondary System Pipe Breaks

15.3.2.1 Identification of Causes and Accident Description

Included in this grouping are ruptures of secondary system lines which would result in steam release rates equivalent to a 6-inch-diameter break or smaller.

INSERT A

15.3.1.3.2 Small Break Core Coolable Geometry Evaluation for Grid Deformation

The small break LOCA analysis for the transition from 17X17 LOPAR to 17X17 VANTAGE-5 at the Diablo Canyon Power Plants was performed with the NOTRUMP Evaluation Model. This analysis resulted in a limiting break of 4-inch equivalent diameter with a peak clad temperature of 1358°F with an $F_Q(Z)$ of 2.50 and an Nuclear Enthalpy Rise Hot Channel Factor of 1.65. The effects of grid deformation were evaluated based on this analysis for a postulated small break LOCA.

The effects of the grid deformation on the small break LOCA analysis calculations were evaluated. The deformation is assumed to occur in the hot assembly which changes the hydraulic resistance of that assembly. This causes a decrease in vapor mass flow rate near the top of the core through the hot assembly (and around the hot rod) and an increase in the temperature of the vapor exiting the assembly. Consequently, the temperature of the fuel cladding increases, particularly at the top of the core where the peak clad temperature is calculated to occur.

A conservative evaluation was performed to determine the extent of the decrease in vapor mass flow rate through the hot assembly and the increase in peak clad temperature. The equivalent assembly-wide blockage (flow reduction) was determined as the blockage that, when applied to the assembly, resulted in the identical flow reduction seen in the subchannel having the theoretical maximum flow area reduction. The equivalent blockage used in the core coolable geometry analysis for the 17X17 VANTAGE-5 was based on the maximum of four (4) grids deforming.

INSERT A (continued)

Therefore, the number of deformed grids postulated to occur on an assembly is four (4) with the location of the grid deformation restricted to occur above the bottom two mixing vane grids and below the top two mixing vane grids. The flow rate reduction near the top of the core through the hot assembly (around the hot rod) based on the 39.3% maximum theoretical deformation (area reduction) of four grids resulted in an estimated increase to the small break peak clad temperature of 154°F at the top of the core.

Deformation of grids at Diablo Canyon Units 1 and 2 have been evaluated for effects on the limiting small break LOCA results. The peak clad temperature of the limiting small break has been conservatively estimated to increase by 154°F. This raises the peak clad temperature result of 1358°F to an estimated value of 1512°F. This result maintains considerable margin to the 2200°F peak cladding temperature limit of 10CFR50.46 and demonstrates coolable geometry during the transition from 17X17 LOPAR to 17X17 VANTAGE-5. Once a full core of 17X17 VANTAGE-5 has been achieved, the PCT penalty for the small break LOCA does not apply.

15.3.5.4 Conclusions

For the case of one RCCA fully withdrawn, with the reactor in either the automatic or manual control mode and initially operating at full power with Bank D at the insertion limit, an upper bound of the number of fuel rods experiencing DNBR $\leq$ 1.30 is 5% of the total fuel rods in the core.

For both cases discussed, the indicators and alarms mentioned would function to alert the operator to the malfunction before DNB could occur. For Case 2 discussed above, the insertion limit alarms (low and low-low alarms) would also serve in this regard.

15.3.6 References

1. Deleted in Revision 3

2. Deleted in Revision 3

3. J. M. Hellman, Fuel Densification Experimental Results and Model for Reactor Application, WCAP-8219, October 1973.

4. F. M. Bordelon, et al, LOCTA-IV Program: Loss-of-Coolant Transient Analysis, WCAP-8305 June 1974

5. S. Altomare and R. F. Barry, The TURTLE 24.0 Diffusion Depletion Code, WCAP-7758, September 1971.

6. R. F. Barry, LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094, WCAP 3269-26, September 1963.

7. Deleted in Revision 3

8. T. W. T. Burnett, et al, LOFTRAN Code Description, WCAP-7907, June 1972.

9. C. Hunin, FACTRAN, A Fortran-IV Code for Thermal Transients in UO_2 Fuel Rods, WCAP-7908, June 1972.

10. J. S. Shefcheck, Application of the THINC Program to PWR Design, WCAP-7359-L, August 1969 (proprietary), WCAP-7838, January 1972.

11. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended to the date of the most recent FSAR Update Revision.

12. P. E. Meyer, NOTRUMP, A Nodal Transient Small Break and General Network Code, WCAP-10079-P-A, August 1985.

13. H. Lee, S. D. Rupprecht, W. D. Tauche, W. R. Schwarz, Westinghouse Small Break ECCS Evauation Model Using the NOTRUMP Code, WCAP-10054-P-A, August 1985.

TABLE 15.3-1

TIME SEQUENCE OF EVENTS
FOR EACH SMALL BREAK LOCA ANALYSIS

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

FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

Table 15.3-1
Time Sequence Of Events
For Each SBLOCA Analysis

TABLE 15.3-2

SMALL COLD LEG BREAK
CLADDING PARAMETERS AND CALCULATION ASSUMPTIONS

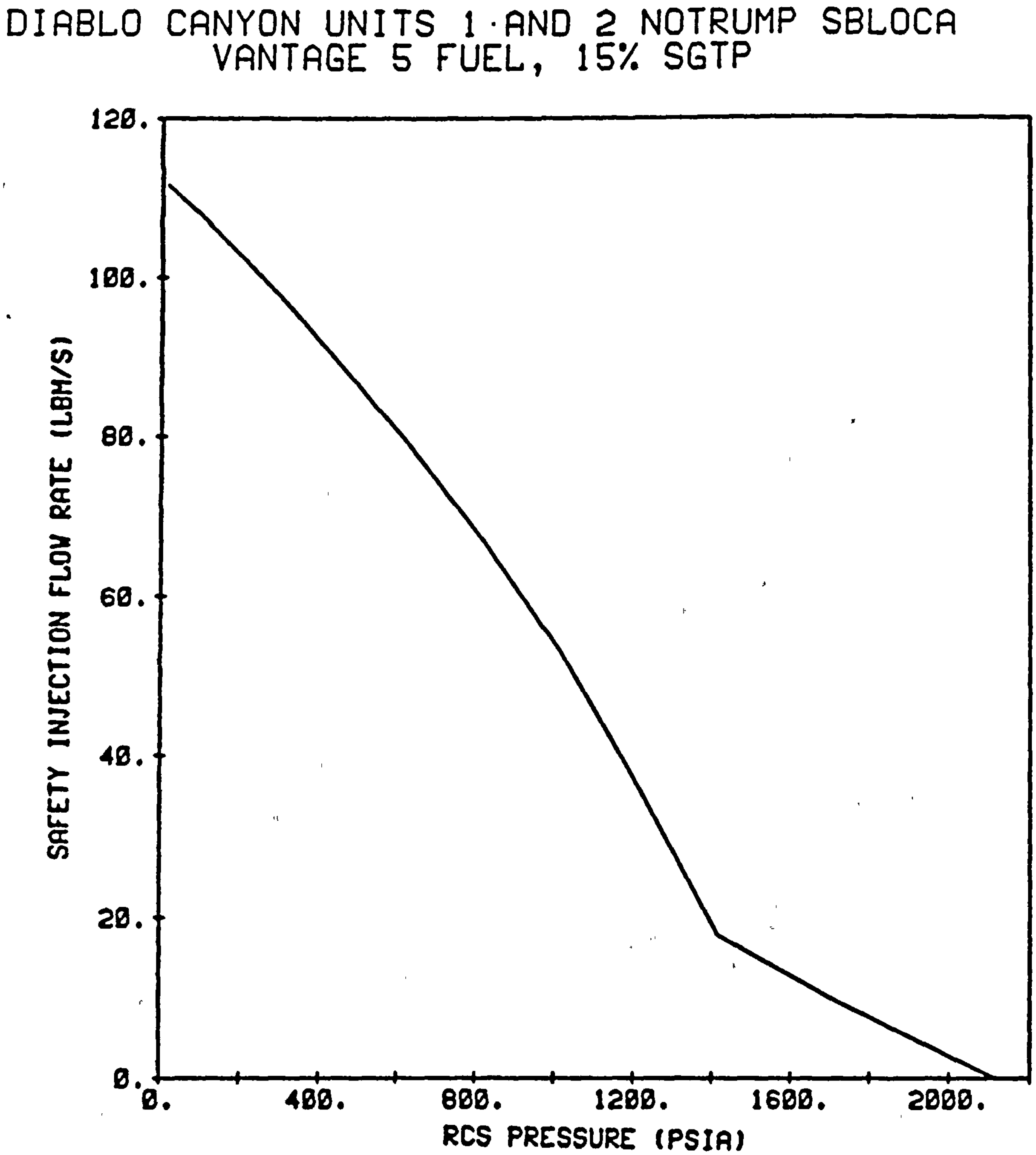
	Unit 2			Unit 1
	Equivalent Break Size			
Results	3 in.	4 in.	6 in.	4 in.
Peak clad temperature, °F	1023	1358	1099	1275
Peak clad location, ft.	12.0	12.0	12.0	12.0
Local Zr/H_2O Reaction (max), %	0.076	0.193	0.073	0.133
Local Zr/H_2O location, ft.	12.0	12.0	12.0	12.0
Total Zr/H_2O Reaction, %	<0.3	<0.3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

Calculation Assumptions

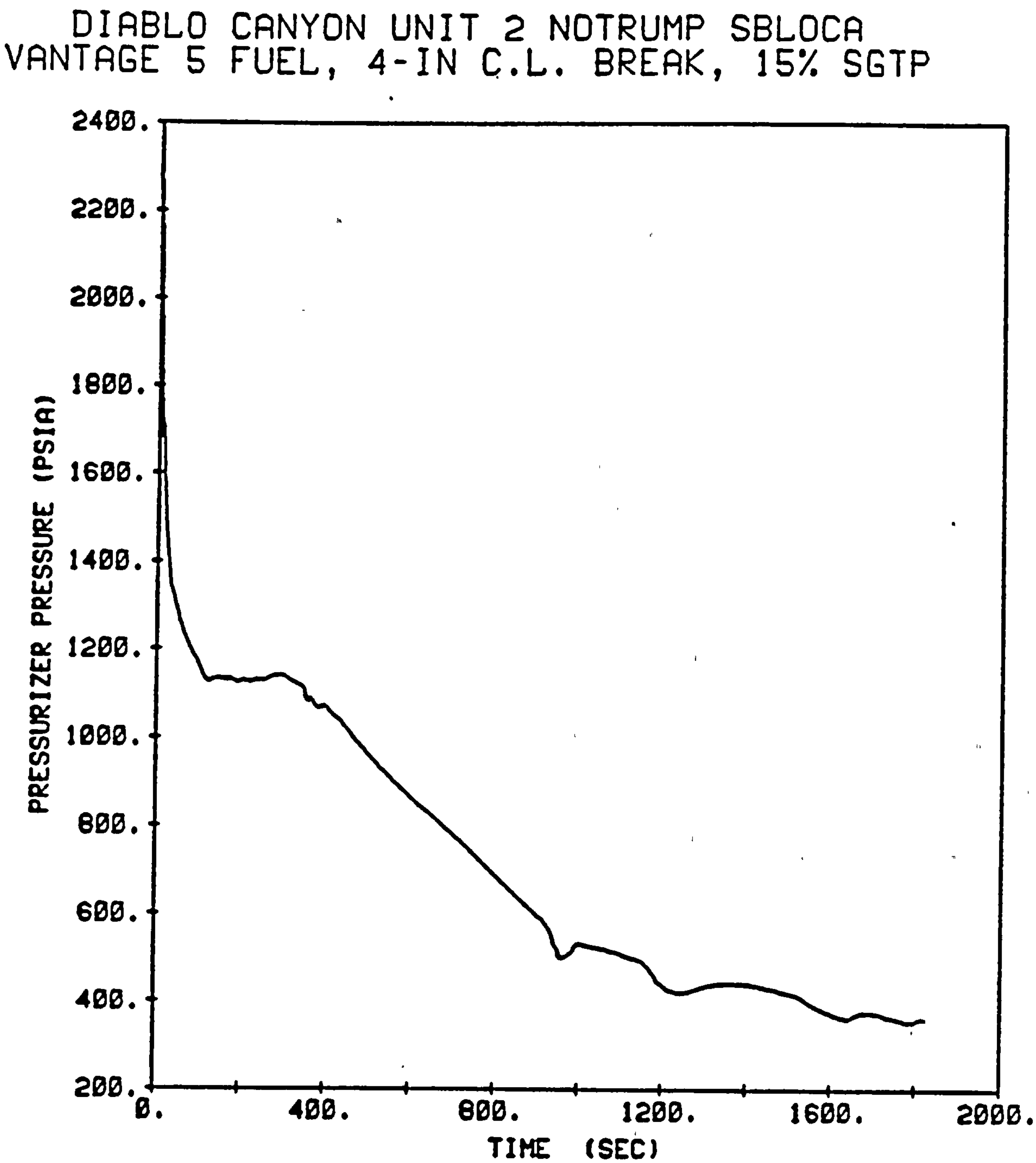
	Unit 2	Unit 1
NSSS power (MWt) is 102% of:	3411	3338
Peak linear power (KW/ft) is 102% of:	12.93	12.65
Peaking factor (at license rating) is:	2.50	2.50
Steam generator tube plugging is:	15%	15%

FSAR UPDATE
UNITS 1 AND 2
DIABLO CANYON SITE

Table 15.3-2
SBLOCA Cladding Parameters
And Calculation Assumptions

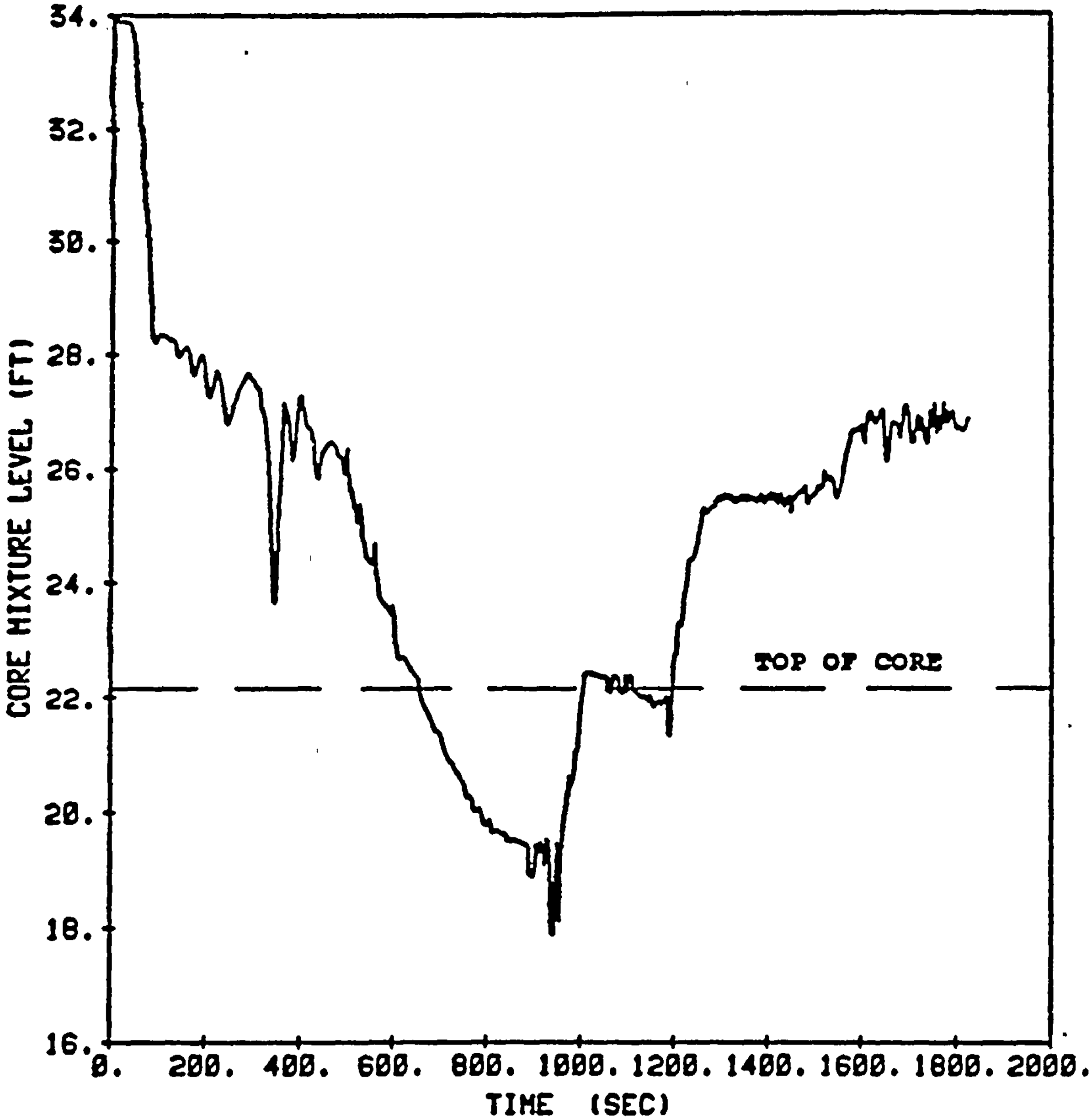


FSAR UPDATE
UNITS 1 AND 2 DIABLO CANYON SITE
Figure 15.3-1 Safety Injection Flow Rate For Small Break LOCA



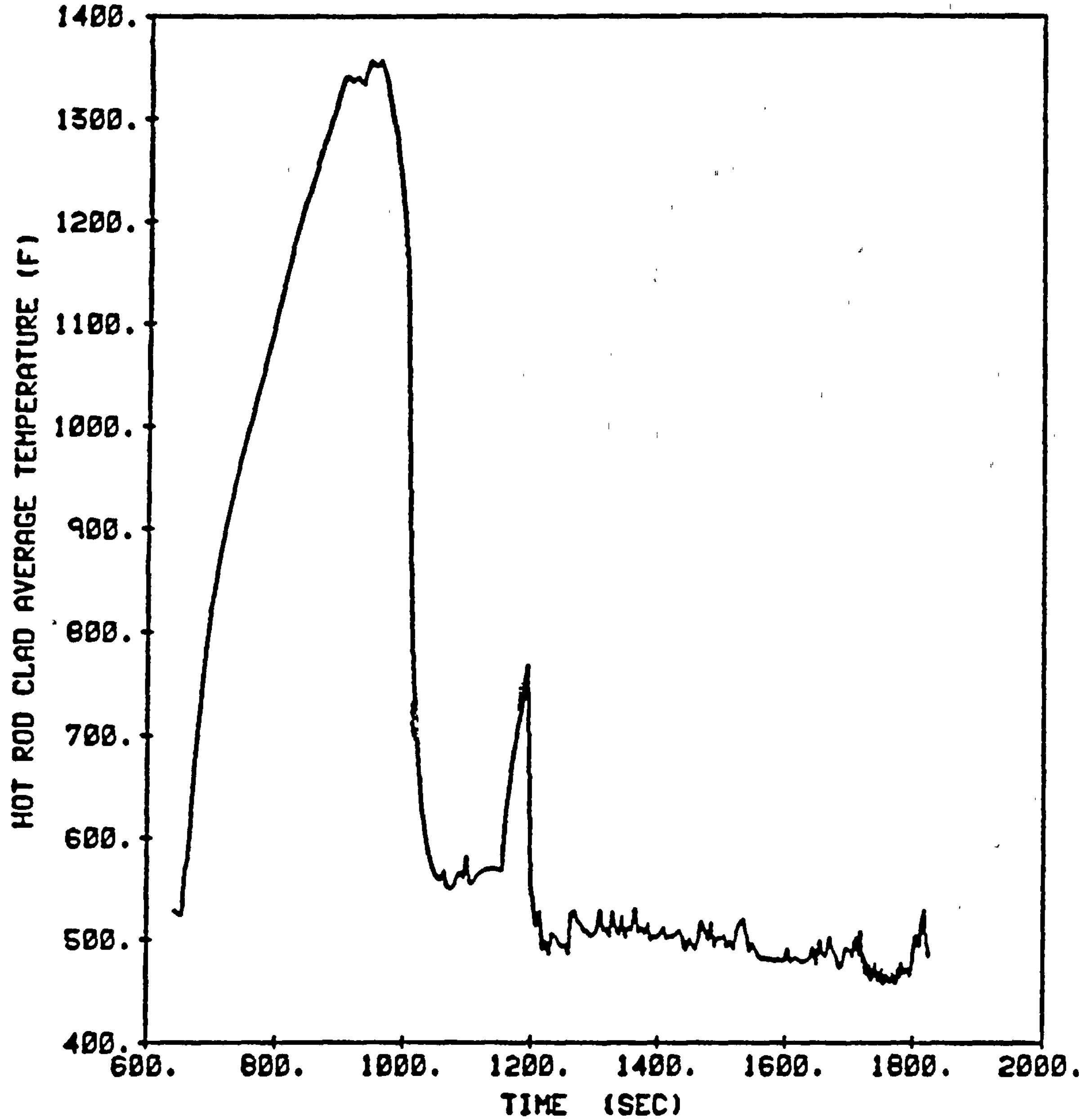
FSAR UPDATE
UNIT 2 **DIABLO CANYON SITE**
Figure 15.3-2 RCS Depressurization 4-inch Cold Leg Break

DIABLO CANYON UNIT 2 NOTRUMP SBLOCA
VANTAGE 5 FUEL, 4-IN C.L. BREAK, 15% SGTP



FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
Figure 15.3-3 Core Mixture Elevation 4-inch Cold Leg Break

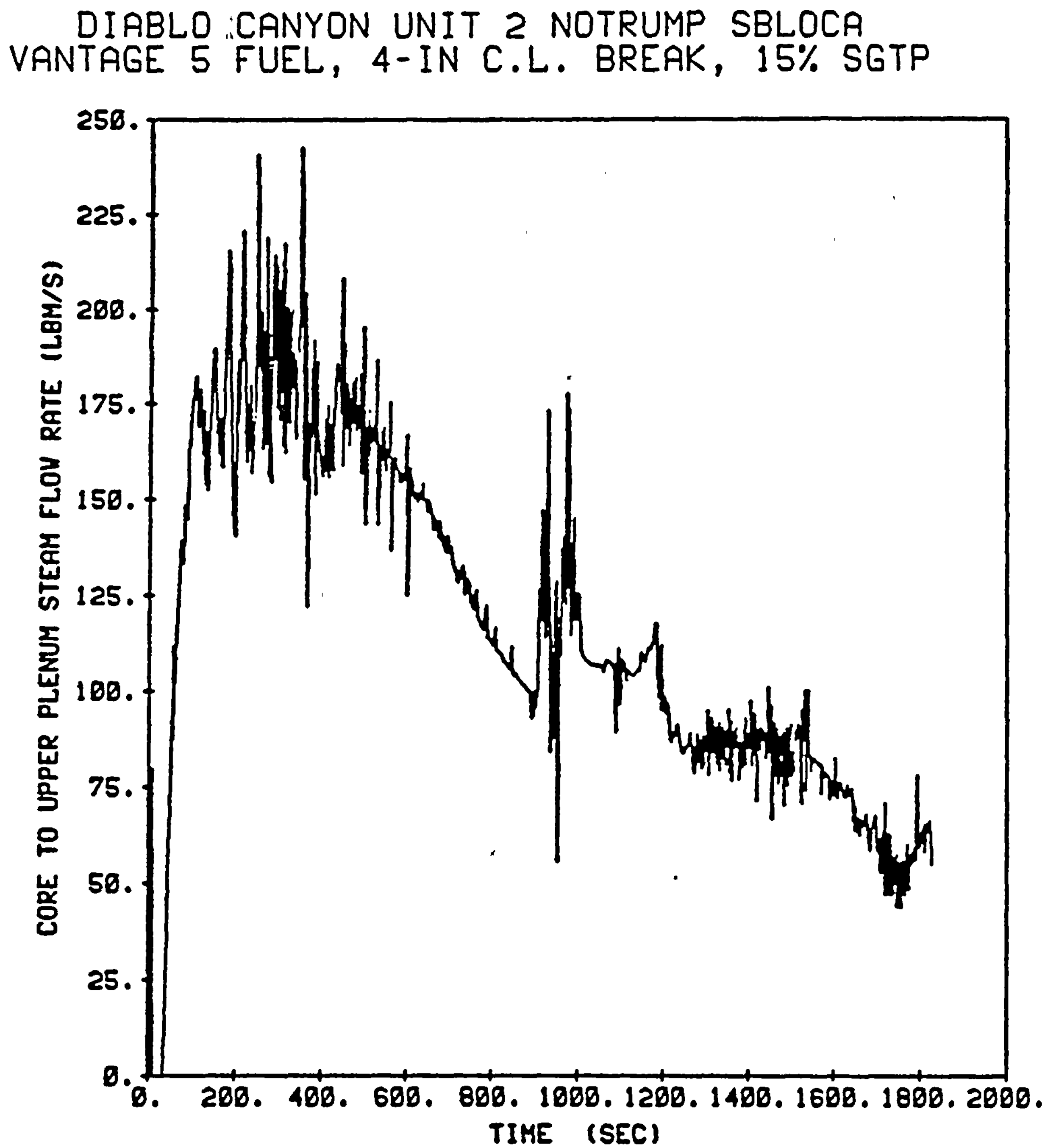
DIABLO CANYON UNIT 2 NOTRUMP SBLOCA
VANTAGE 5 FUEL, 4-IN C.L. BREAK, 15% SGTP



FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

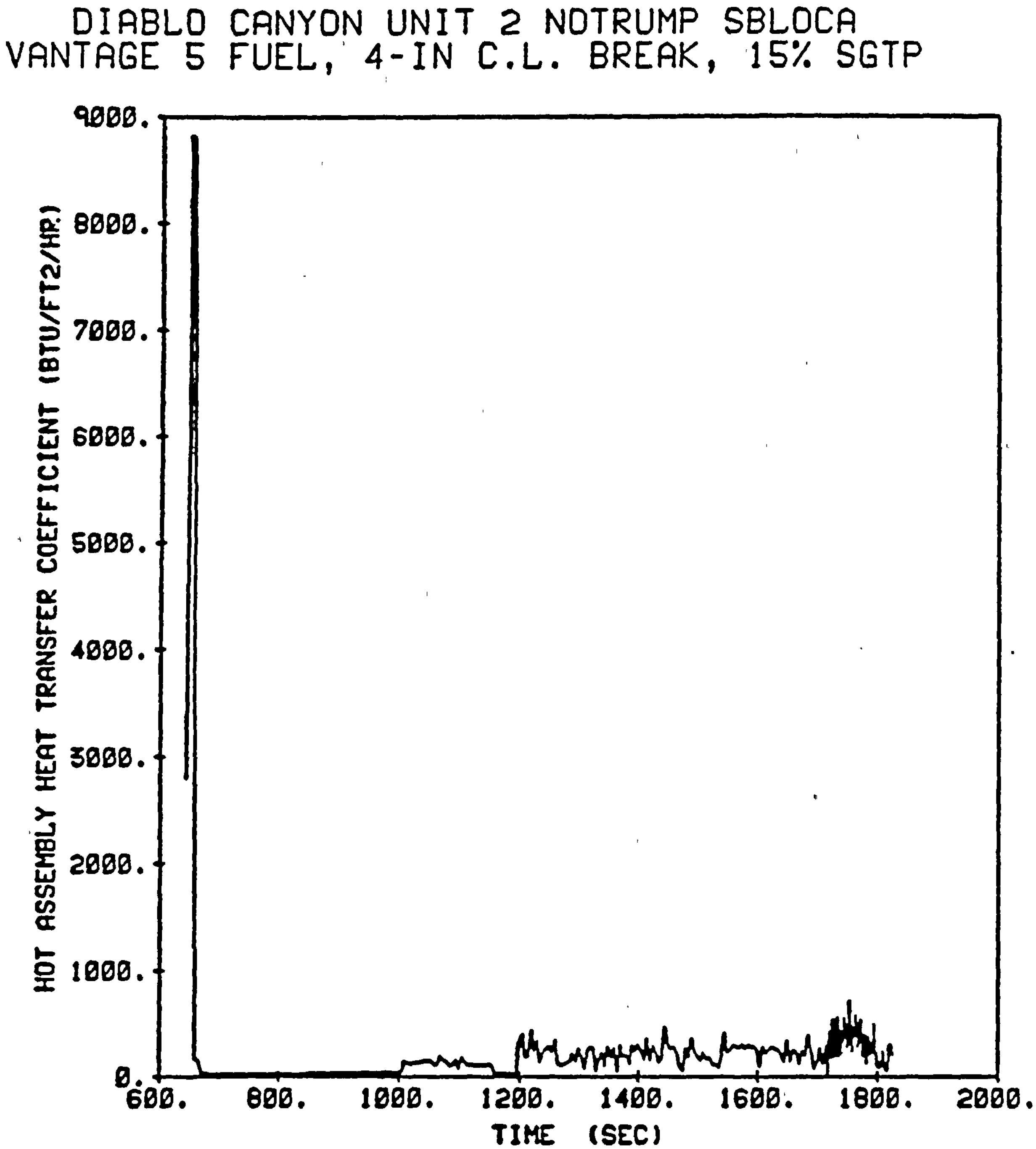
Figure 15.3-4
Clad Temperature Transient
4-inch Cold Leg Break



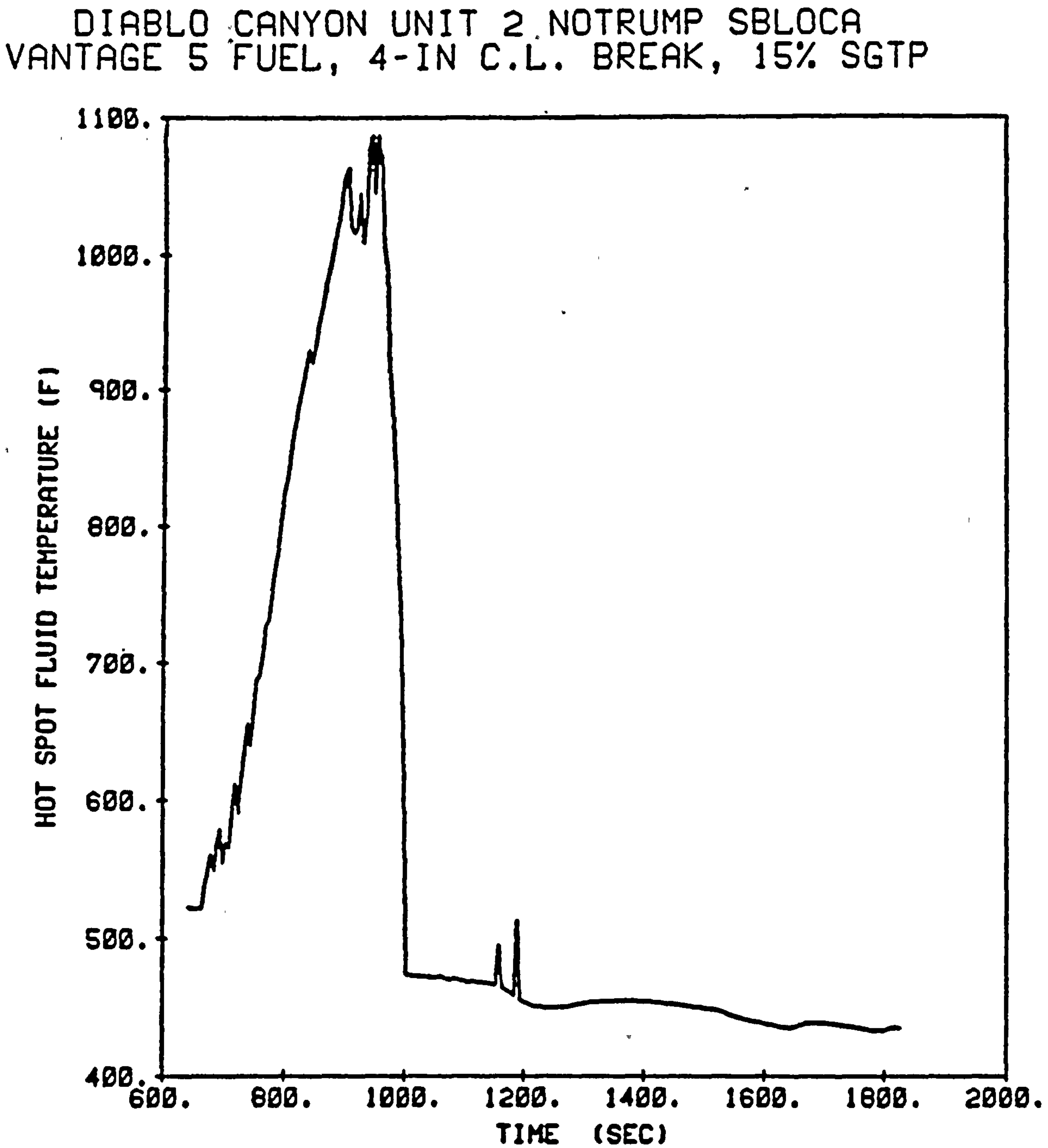
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

Figure 15.3-5
Steam Flow
4-inch Cold Leg Break



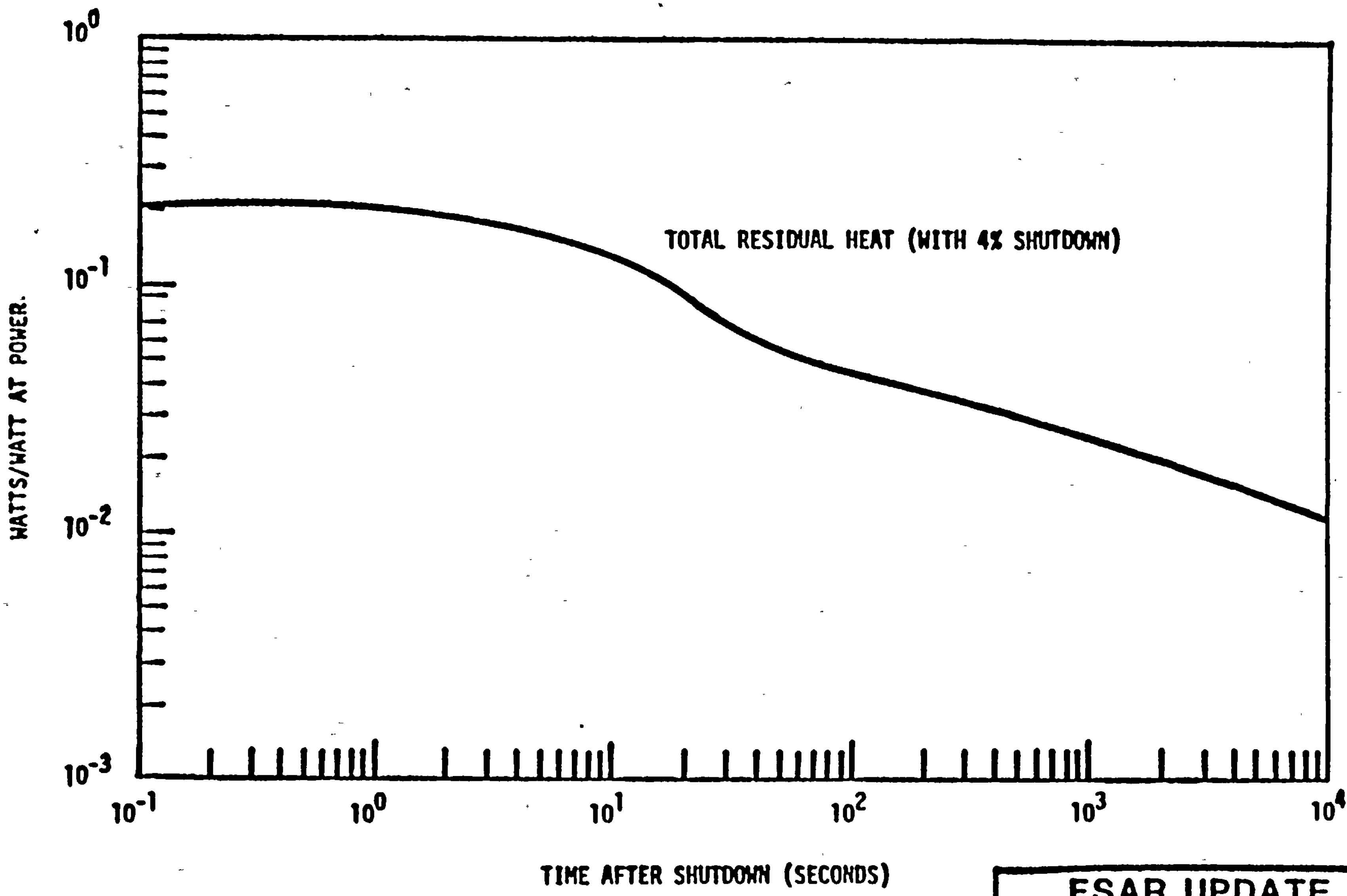
FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
Figure 15.3-6 Rod Film Coefficient 4-inch Cold Leg Break



FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

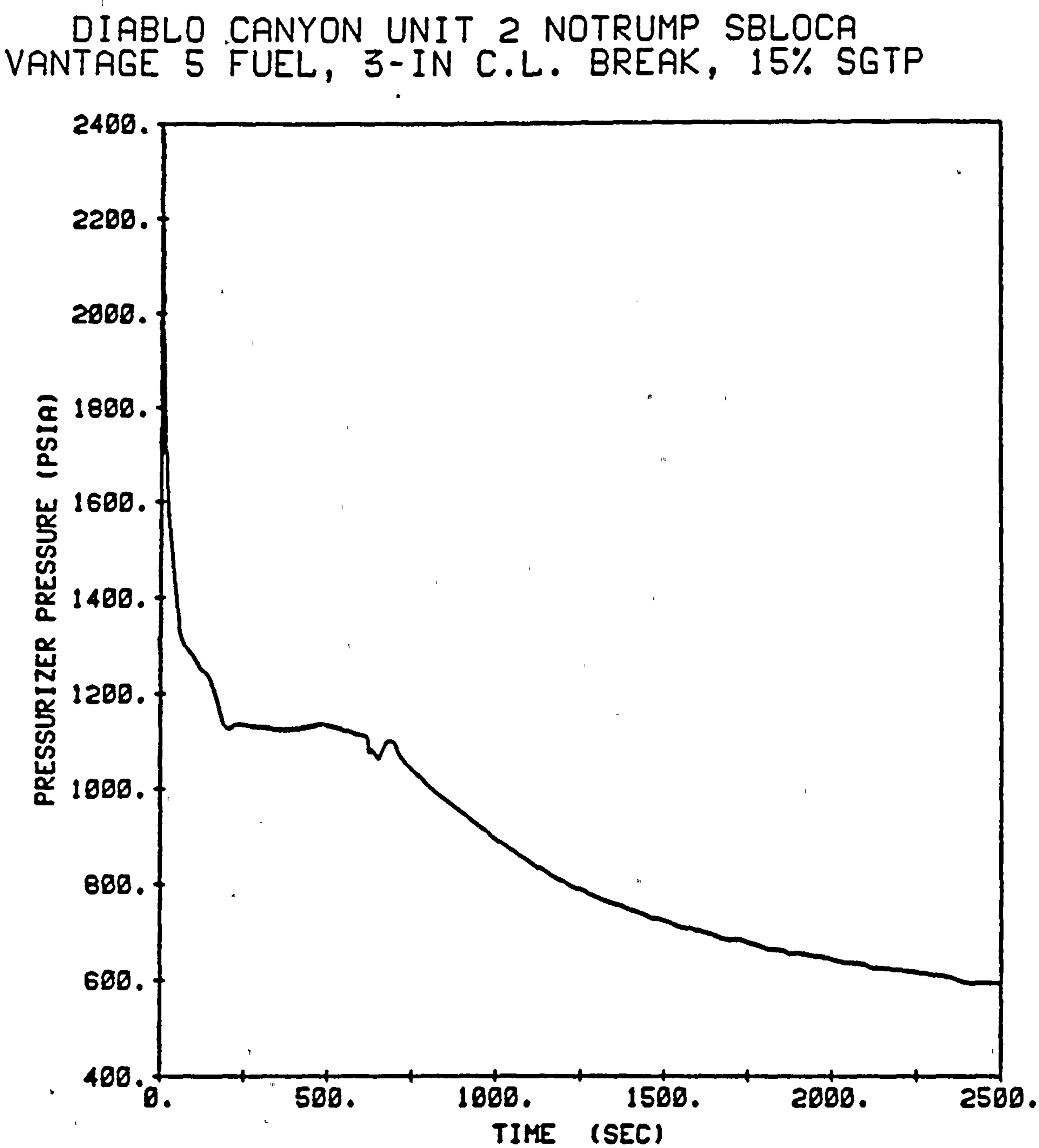
Figure 15.3-7
Hot Spot Fluid Temperature
4-inch Cold Leg Break



FSAR UPDATE

UNITS 1 AND 2
DIABLO CANYON SITE

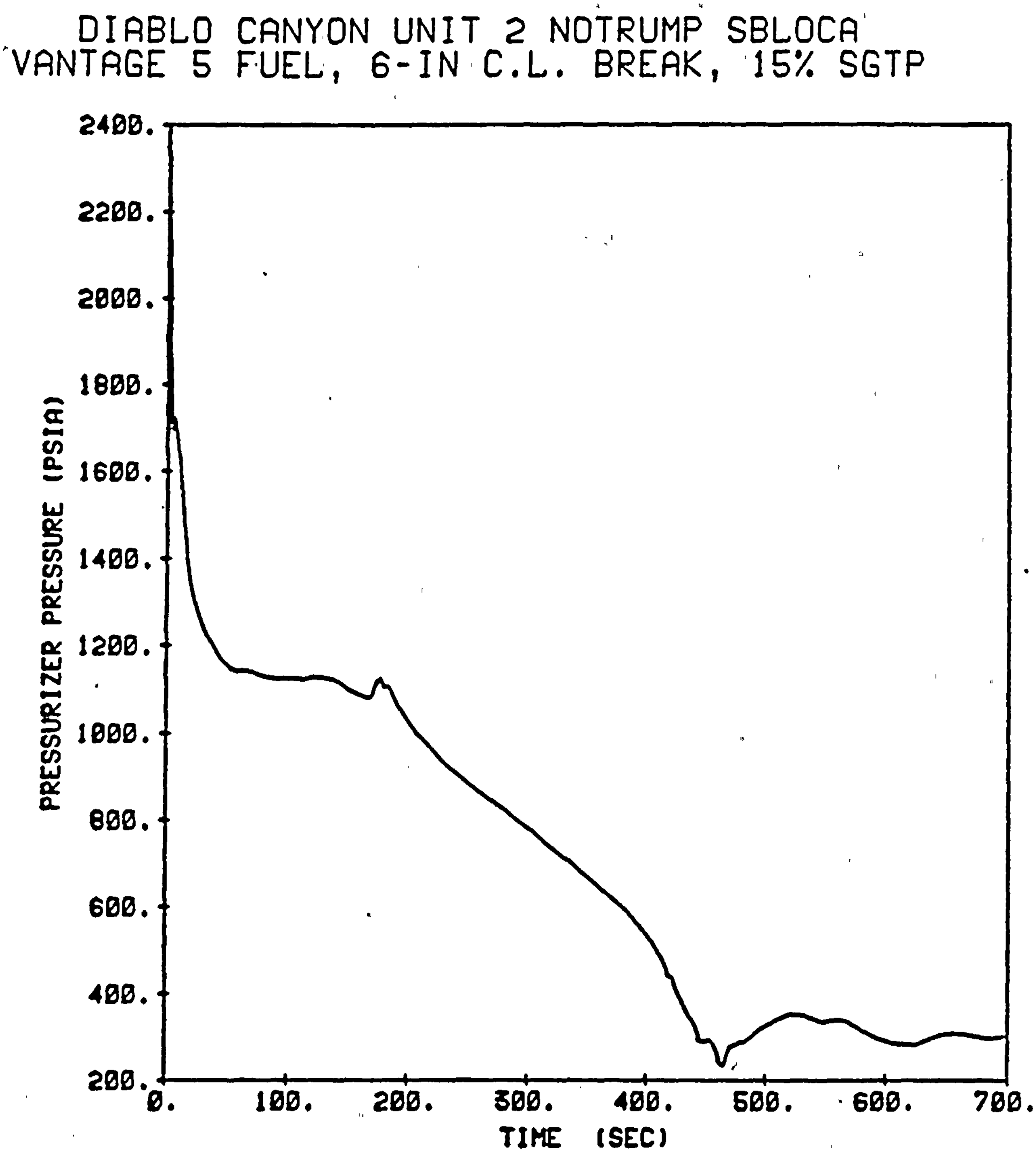
Figure 15.3-8
Core Power Distribution



FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

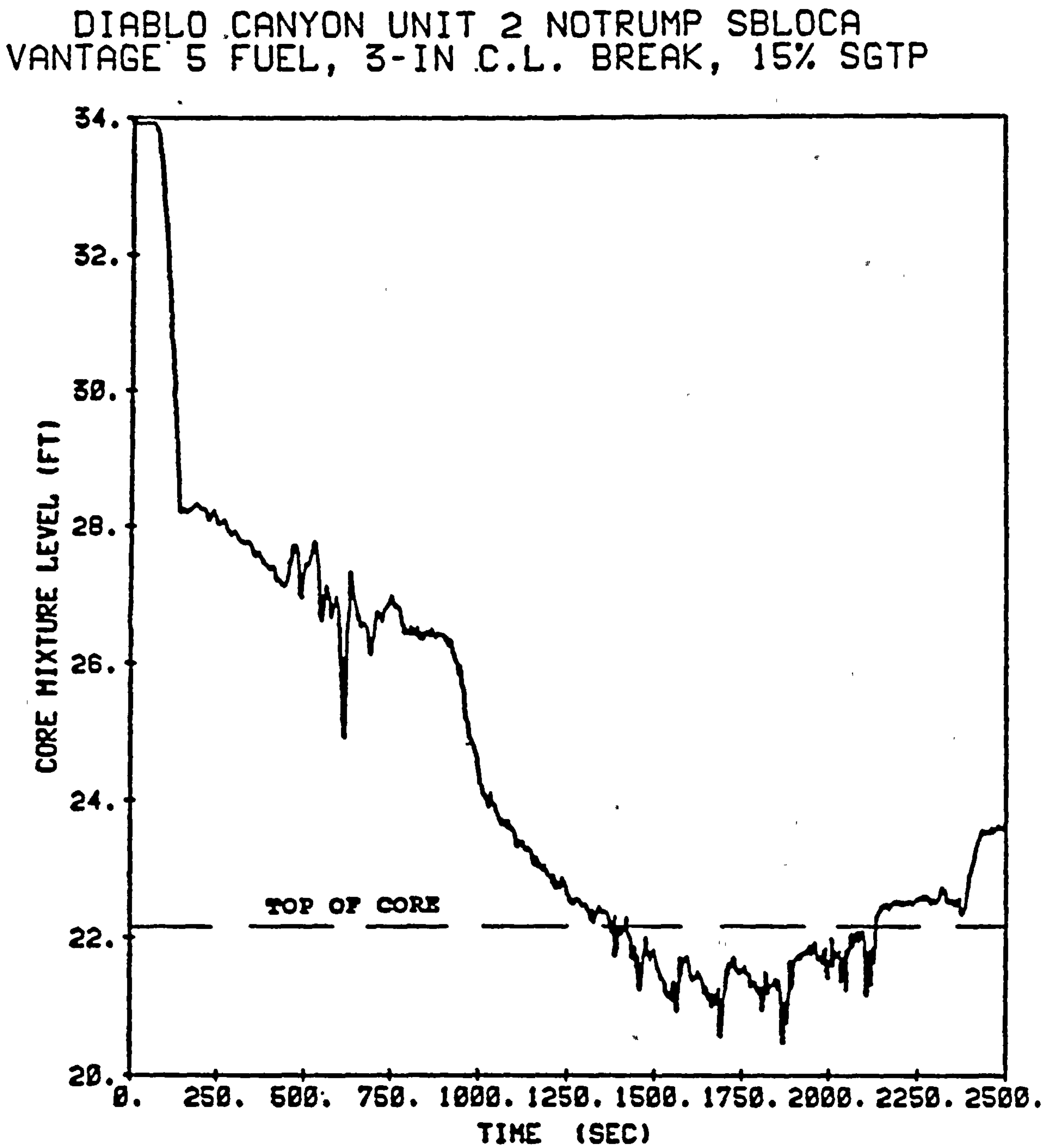
Figure 15.3-9
RCS Depressurization
3-inch Cold Leg Break



FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

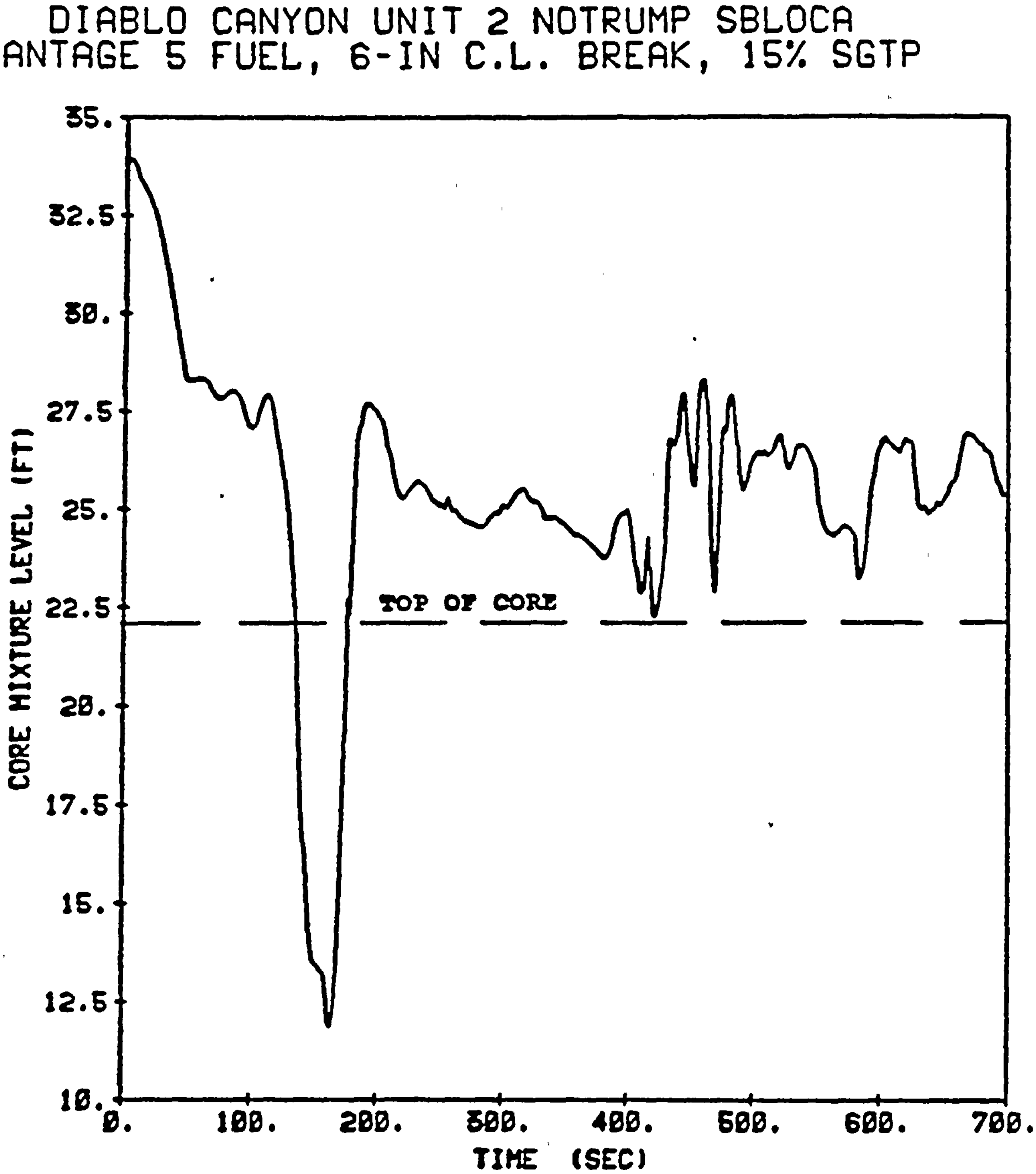
Figure 15.3-10
RCS Depressurization
6-inch Cold Leg Break



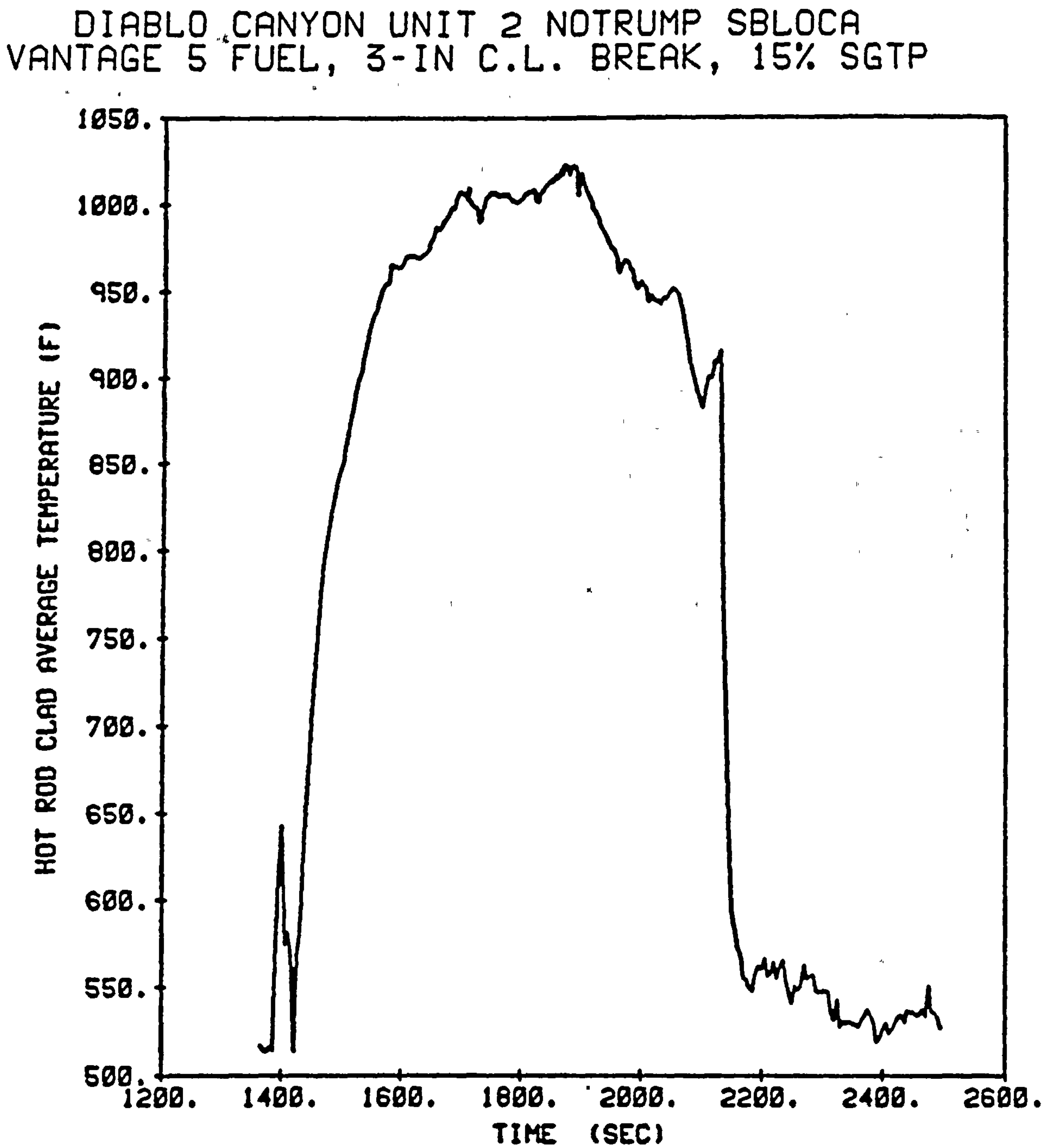
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

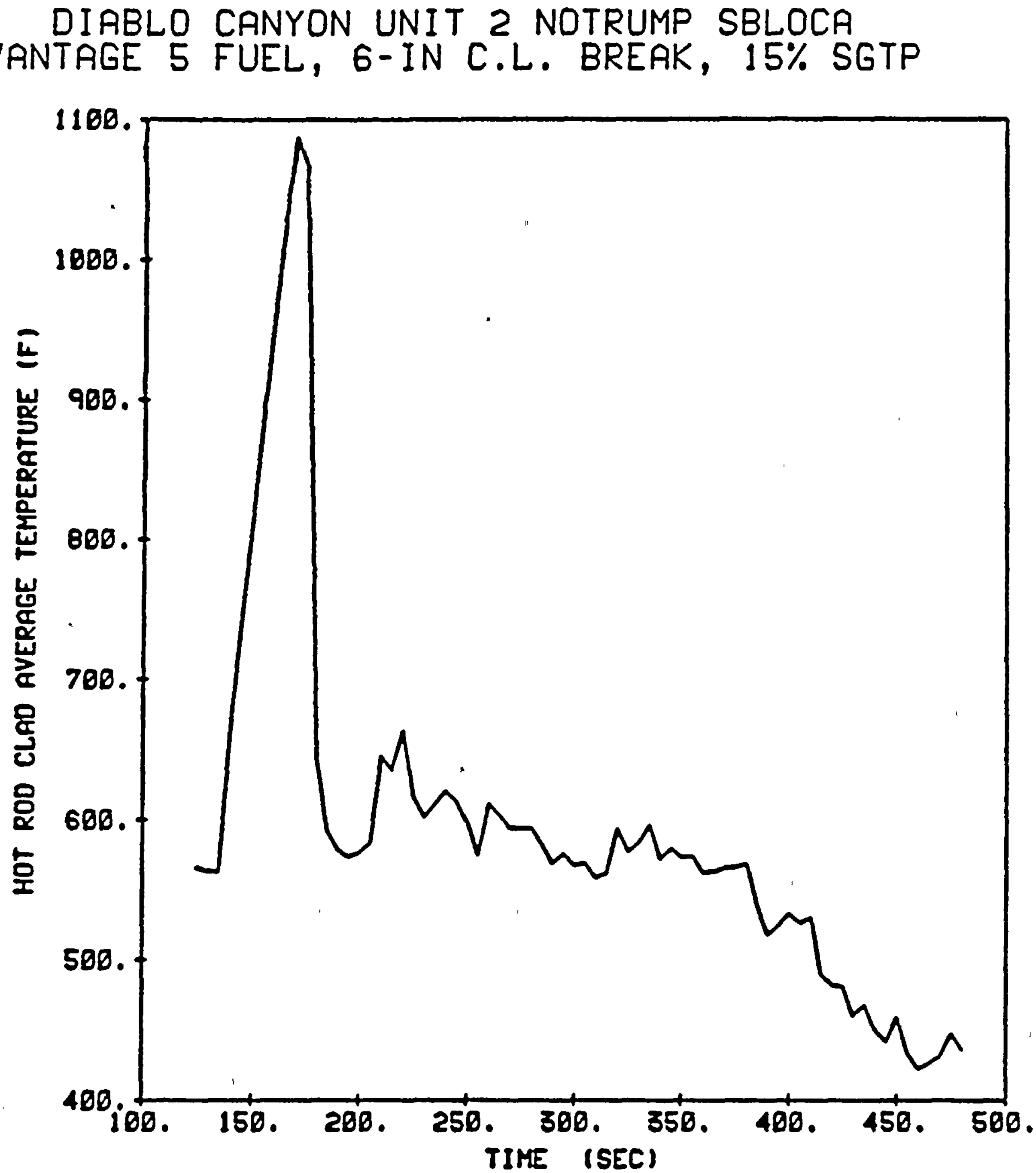
Figure 15.3-11
Core Mixture Elevation
3-inch Cold Leg Break



FSAR UPDATE
UNIT 2 **DIABLO CANYON SITE**
Figure 15.3-12 Core Mixture Elevation 6-inch Cold Leg Break



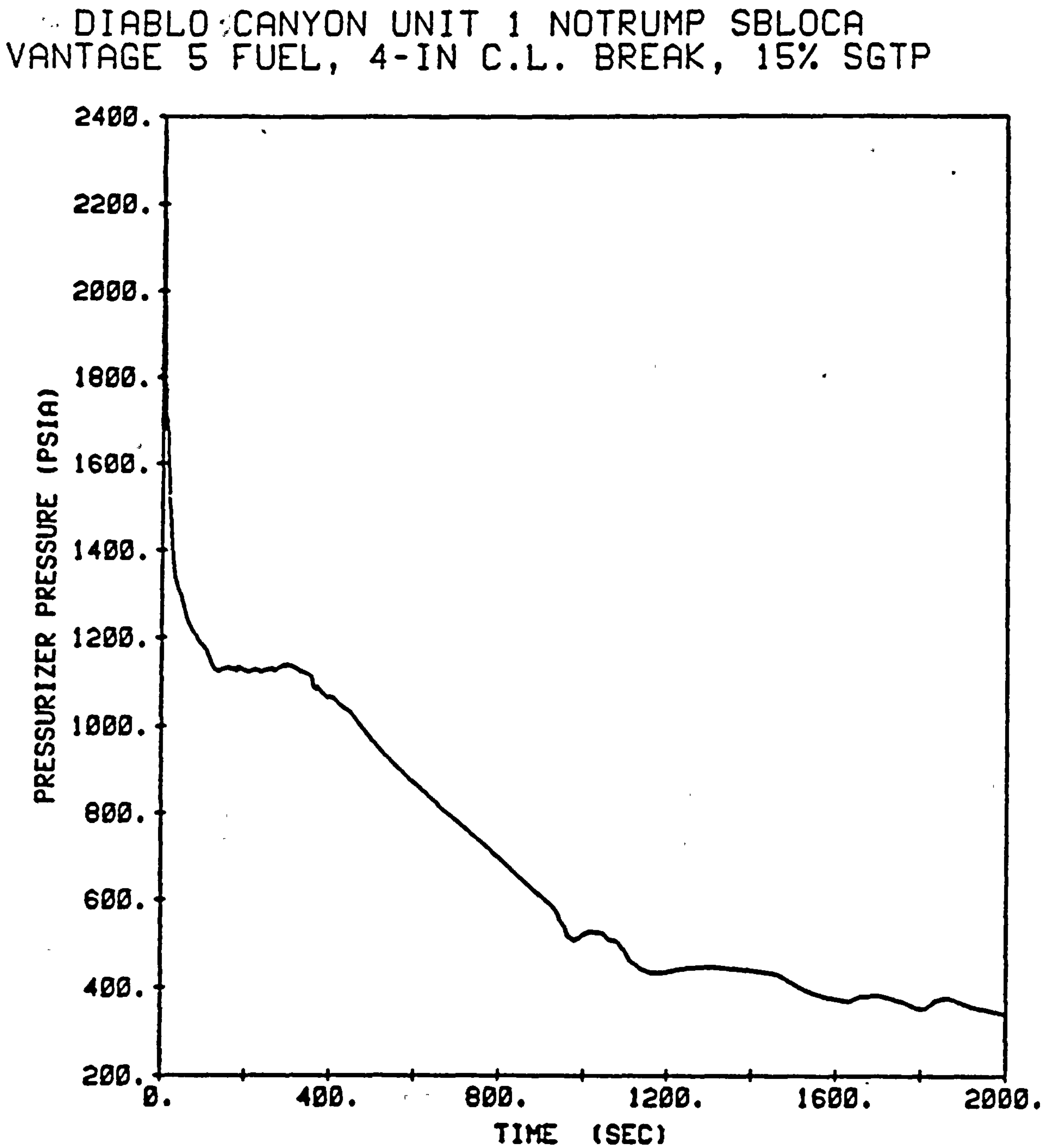
FSAR UPDATE
UNIT 2 DIABLO CANYON SITE
Figure 15.3-13 Clad Temperature Transient 3-inch Cold Leg Break



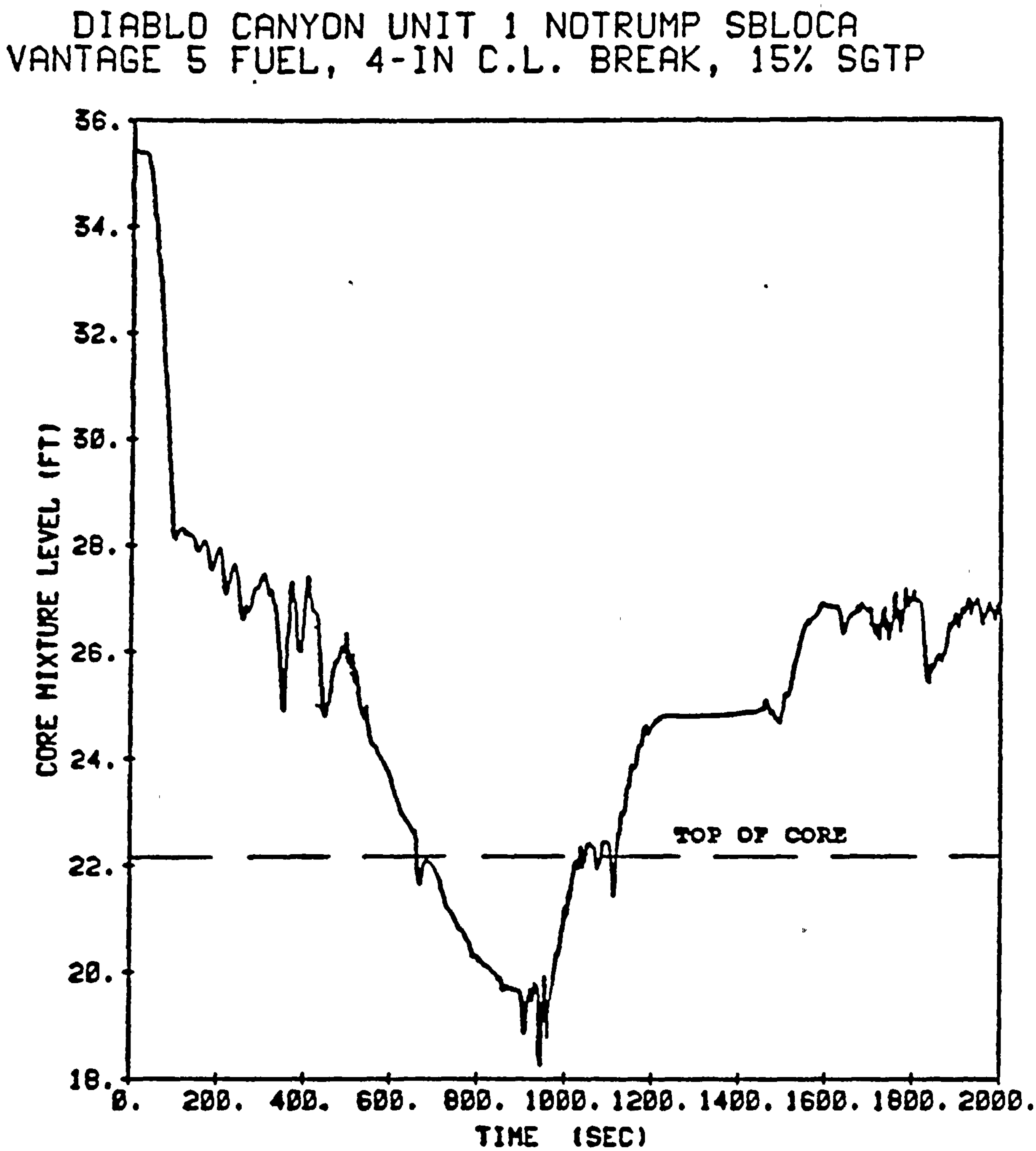
FSAR UPDATE

UNIT 2
DIABLO CANYON SITE

Figure 15.3-14
Clad Temperature Transient
6-inch Cold Leg Break



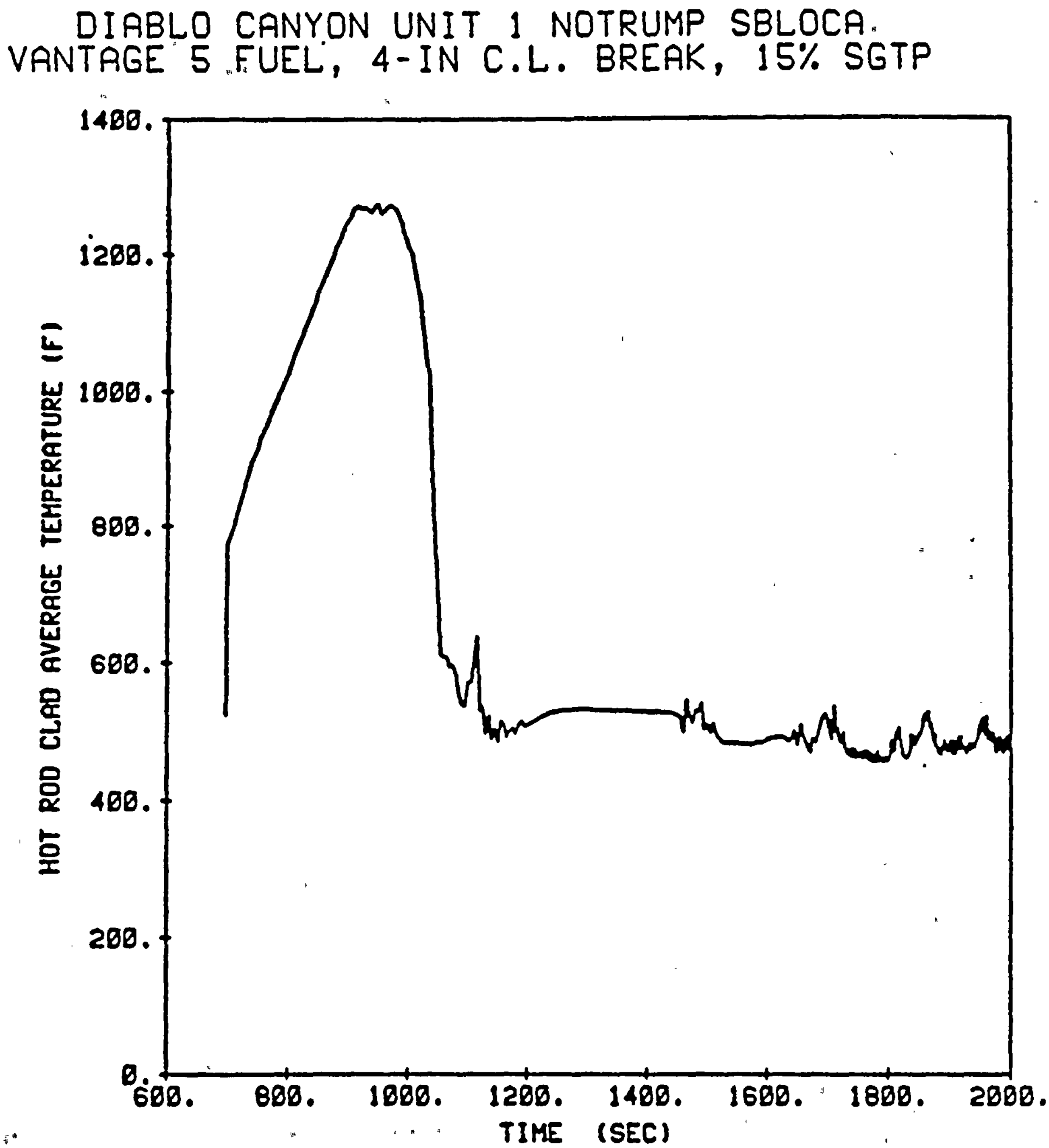
FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
Figure 15.3-14a RCS Depressurization 4-inch Cold Leg Break



FSAR UPDATE

UNIT 1
DIABLO CANYON SITE

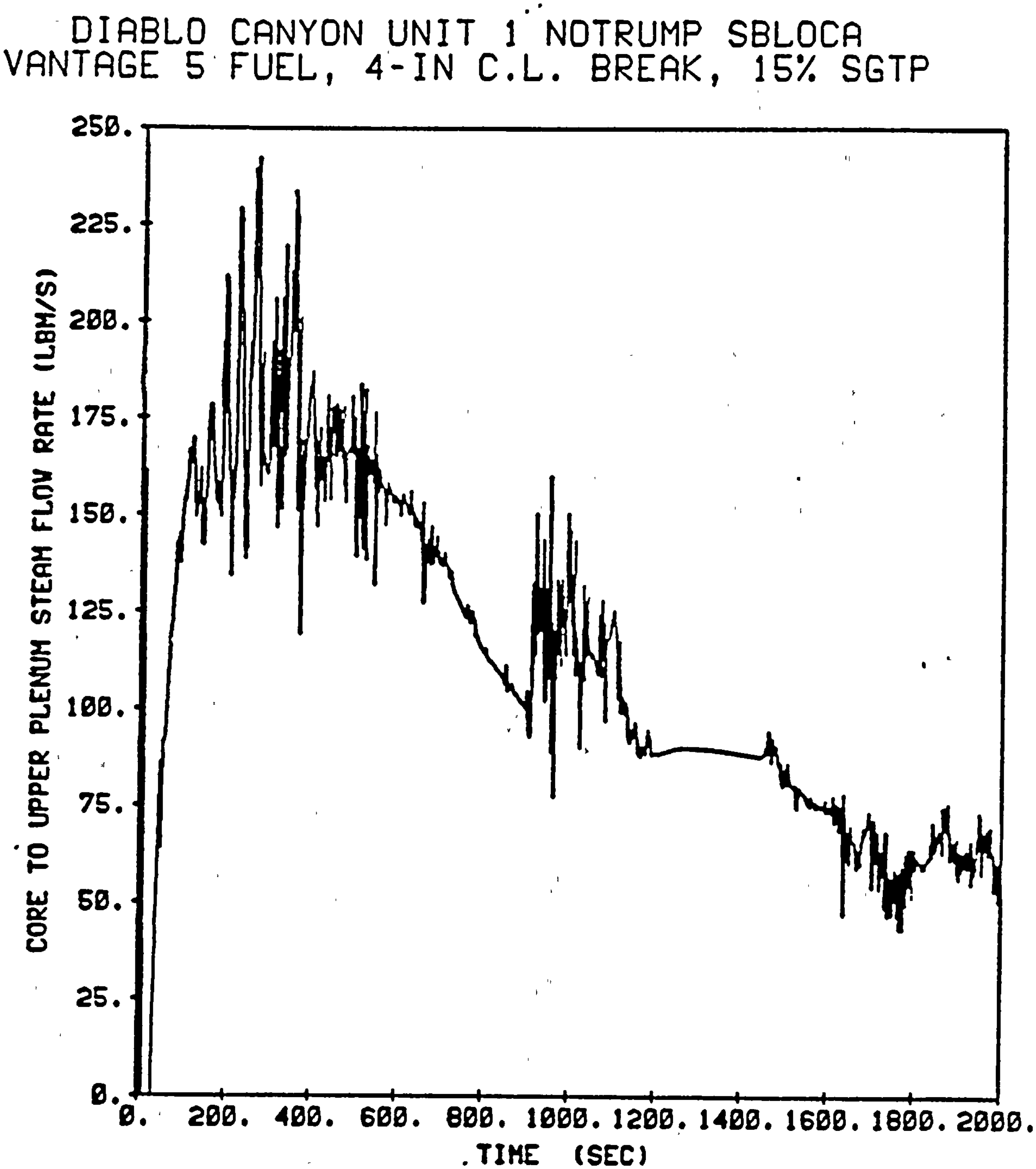
Figure 15.3-14b
Core Mixture Elevation
4-inch Cold Leg Break



FSAR UPDATE

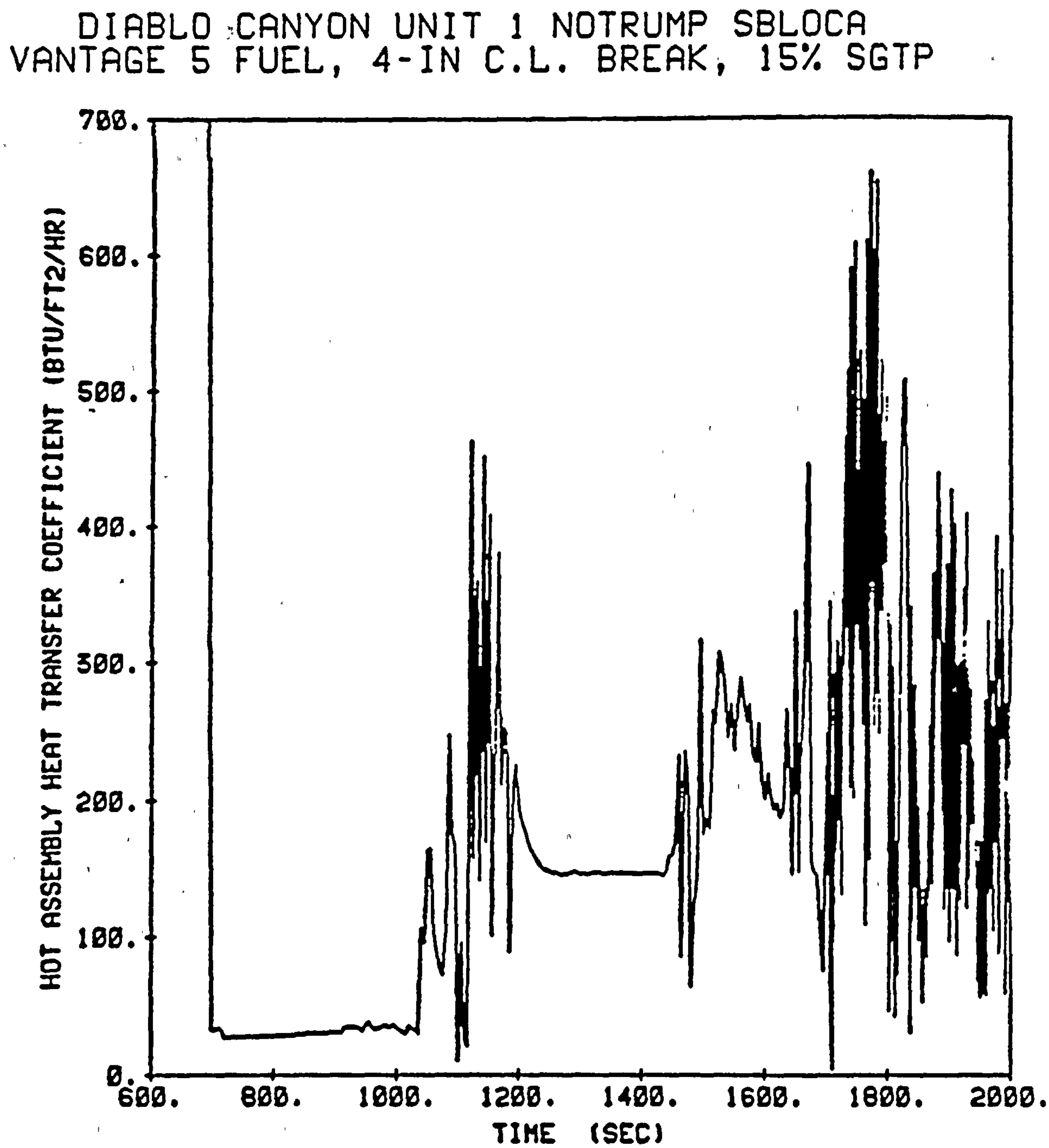
UNIT 1
DIABLO CANYON SITE

Figure 15.3-14c
Clad Temperature Transient
4-inch Cold Leg Break

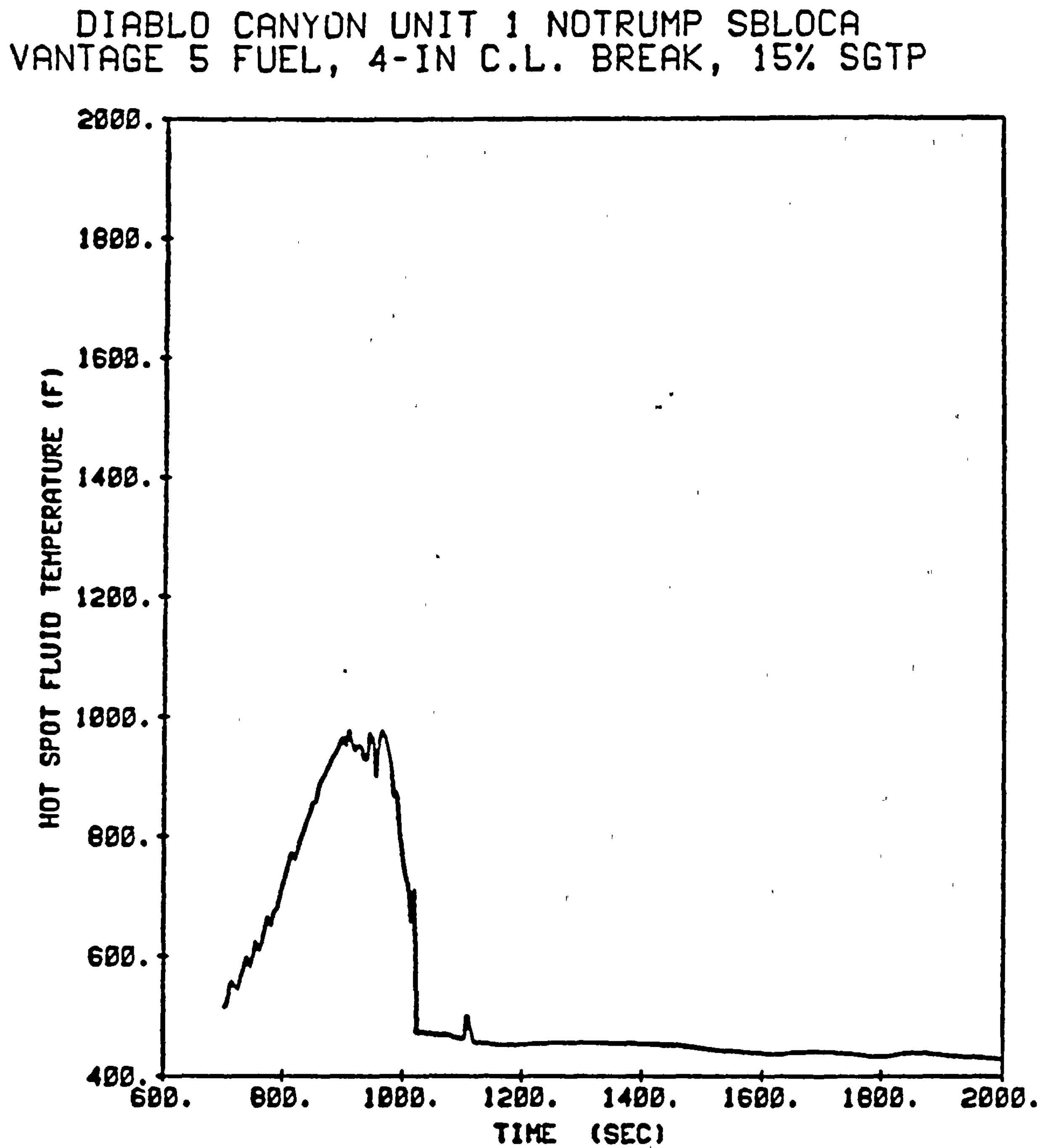


FSAR UPDATE
UNIT 1
DIABLO CANYON SITE

Figure 15.3-14d
Steam Flow
4-inch Cold Leg Break



FSAR UPDATE
UNIT 1 DIABLO CANYON SITE
Figure 15.3-14e Rod Film Coefficient 4-inch Cold Leg Break



FSAR UPDATE

UNIT 1
DIABLO CANYON SITE

Figure 15.3-14f
Hot Spot Fluid Temperature
4-inch Cold Leg Break

maximum overspeed condition caused by a large pressure head driving the pump in reverse. Such failure would require the failure of two check valves in the open position in conjunction with a rupture of the pipe on the suction side of the pump.

Despite the low probability of such a combination of failures, a shroud has been installed around the flexible coupling to eliminate all possibility of missiles being generated in the unlikely event of gross coupling failure.

INSERT A

~~6.3.3.2.14 Effect of Grid Deformation on ECCS Performance~~

~~The effects of grid distortion caused by a combined LOCA and seismic event have been evaluated for DCPP, even though no permanent distortion is predicted. Results are summarized in Section 3.7.3.15. The peak cladding temperature (PCT) in any assembly being postulated to deform is clearly lower than the PCT for the DBA, used to establish the technical specification limits. Increases in PCT due to confined channel geometry are more than compensated for by decreases in PCT due to lower power of the assemblies postulated to deform and enhance radiation heat transfer.~~

6.3.3.3 Alternate Analysis Methods

The method of break analysis and the spectrum of breaks analyzed are described in Section 6.3.1.1.

6.3.3.4 Fuel Rod Perforations

Results of the small pipe break and large pipe break analyses are presented in Sections 15.3 and 15.4, respectively.

6.3.3.5 Effects of ECCS Operation on the Core

When water in the RWST at its minimum boron concentration is mixed with the contents of the RCS, the resulting boron concentration ensures that the reactor will remain subcritical in the cold condition with all control rods, except the most reactive RCCA, inserted into the core.

The boron concentration of the accumulator and the RWST is below the solubility limit of boric acid at the respective temperatures.

Boron concentration (22,500 ppm boron, max) in the BIT is below the solubility limit of boric acid at the operating temperature (150°F) of the tank. There is also continuous recirculation between the BIT and the boric acid tanks. The heating and recirculation provided is adequate to ensure that precipitation will not occur in the BIT and associated piping.

6.3.3.6 Use of Dual Function Components

The ECCS contains components that have no other operating function as well as components that are shared with other systems. Components in each category are as follows:

6.3.3.2.14 Effect of Grid Deformation on ECCS Performance

(Insert A)

The effects of grid distortion caused by a combination of LOCA and Seismic loads have been evaluated for DCPP. The combined LOCA and Seismic structural analysis have shown that some peripheral VANTAGE 5 fuel assemblies will undergo loads capable of deforming the Zircaloy structural grids when 17X17 standard fuel is present with VANTAGE 5 assemblies. The details of the coolable geometry analysis appear in Section 15.4.1.1.3 and the results demonstrate that the core remains amenable to cooling with the deformed grid geometry.

15.4.1 Major Reactor Coolant System Pipe Ruptures (LOCA)

The analysis performed to comply with the requirements of 10 CFR 50.46[1] Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors, is presented in this section. The time sequence of events during a large double-ended cold leg guillotine (DECLG) break LOCA is shown in Tables 15.4-1 and 15.4-2. The results of the LOCA analysis are shown in Tables 15.4-3 and 15.4-4 and show compliance with the acceptance criteria. The analytical techniques used are in compliance with Appendix K of 10 CFR 50[1], and are described in Reference 34.

15.4.1.1 Thermal Analysis

15.4.1.1.1 Westinghouse Performance Criteria for ECCS

The reactor is designed to withstand thermal effects caused by a LOCA including the double-ended severance of the largest RCS pipe. The reactor core and internals together with the ECCS are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident.

The ECCS, even when operating during the injection mode with the most severe single active failure, is designed to meet the acceptance criteria of 10 CFR 50.46.

15.4.1.1.1.1 Sequence of Events and Systems Operations

Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety injection signal is generated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. However, no credit is taken during the LOCA blowdown for negative reactivity due to boron content of the injection water. In addition, the insertion of control rods to shut down the reactor is neglected in the large break analysis.

b. Injection of borated water provides the fluid medium for heat transfer from the core and prevents excessive clad temperatures.

15.4.1.1.1.2 Description of a Large Break LOCA Transient

The sequence of events following a large break LOCA is presented in Figure 15.4-1, Tables 15.4-1 and 15.4-2.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire RCS contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. Thereafter, the core heat transfer is based on local conditions with transition boiling, film boiling, and forced convection to steam as the major heat transfer mechanisms.

The heat transfer between the RCS and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and the main steam safety valves may actuate to limit the pressure. Makeup water to the secondary side is automatically provided by the auxiliary feedwater system. The safety injection signal actuates a feedwater isolation signal which isolates normal feedwater flow by closing the main feedwater isolation valves and also initiates emergency feedwater flow by starting the auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure.

When the RCS depressurizes to approximately 600 psia, the accumulators begin to inject borated water into the reactor coolant loops.

Since the loss of offsite power is assumed, the reactor coolant pumps are assumed to trip at the inception of the accident. The effects of pump coastdown are included in the blowdown analysis.

The blowdown phase of the transient ends after the RCS pressure (initially assumed at a nominal 2280 psia) falls to a value approaching that of the containment atmosphere. Prior to or at the end of the blowdown, the mechanisms that are responsible for the bypassing of emergency core cooling water injected into the RCS are calculated not to be effective. At this time (called end-of-bypass) refill of the reactor vessel lower plenum begins. Refill is complete when emergency core cooling water has filled the lower plenum of the reactor vessel which is bounded by the bottom of the fuel rods (called bottom of core recovery time).

The reflood phase of the transient is defined as the time period lasting from the end-of-refill until the reactor vessel has been filled with water to the extent that the core temperature rise has been terminated. From the later stage of blowdown and then the beginning-of-reflood, the safety injection accumulator tanks rapidly discharge borated cooling water into the RCS, contributing to the filling of the reactor vessel downcomer. The downcomer water elevation head provides the driving force required for the reflooding of the reactor core. The low head and high head safety injection pumps aid in the filling of the downcomer and subsequently supply water to maintain a full downcomer and complete the reflooding process. The safety injection pumped flow which is a function of pressure is given in Figures 15.4-~~45~~ through 15.4-52.
57

Continued operation of the ECCS pumps supplies water during long-term cooling. Core temperatures have been reduced to long-term steady state levels associated with dissipation of residual heat generation. After the water level of the refueling water storage tank reaches a minimum allowable value, coolant for long-term cooling of the core is obtained by switching to the cold leg recirculation phase of operation in which spilled borated water is drawn from the containment sump by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The containment spray system continues to operate to further reduce containment pressure. Approximately ~~24~~ hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel. 13.5

15.4.1.1.2 Method of Thermal Analysis

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Reference 1). The requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS system. Input parameters used for this analysis are presented in Tables 15.4-A and 15.4-B. Decay heat generated throughout the transient is also conservatively calculated as required by Appendix K of 10 CFR 50.

15.4.1.1.2.1 Large Break LOCA Evaluation Model

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase: (1) the thermal-hydraulic transient in the RCS, (2) the pressure and temperature transient within the containment, and (3) the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

INSERT B

~~The description of the various aspects of the LOCA analysis methodology is given in Reference 2. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the acceptance criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes, which are used in the LOCA analysis, are described in detail in References 3 through 6. Modifications to these codes are specified in References 33 through 35. The BART code is described in References 38 and 39. These codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD computer code is used to calculate this transient during the refill and reflood phases of the accident. The BART computer code is used to calculate the fluid and heat transfer conditions in the core during reflood. The COCO computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.~~

INSERT B

The description of the various aspects of the LOCA analysis methodology is given in Reference 2. The SATAN-VI (Ref 3), WREFLOOD (Refs 5,38), COCO (Ref 6), BASH (Ref 39), and LOCBART (Ref 4,38) codes which are used in the LOCA analysis are described in detail in the indicated references. These codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the LOCA. The SATAN-VI computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and the WREFLOOD/BASH computer codes are used to calculate this transient during the refill and reflood phases of the accident. The COCO computer code is used to calculate the containment pressure transient during all the three phases of the LOCA analysis. Similarly, the LOCBART computer code is used to compute the thermal transient of the hottest fuel rod during the three phases.

The large break analysis was performed with the approved 1981 EM + BASH version of the Evaluation Model (Reference 39). Specific features of these codes are presented in the following discussion.

SATAN-VI is used to calculate the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of blowdown these data are transferred to the WREFLOOD code and the COCO code for use in the determination of the containment pressure response during the reflood phase of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core pressure, and the core power decay transient are input to the LOCBART code.

INSERT B (continued)

WREFLOOD, using input from the SATAN-VI code, calculates the time to bottom of core recovery (BOC), RCS conditions at BOC and mass and energy release from the break during the reflood phase of the LOCA. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the Containment backpressure, the WREFLOOD and COCO codes are interactively linked. The BOC conditions calculated by WREFLOOD and the containment pressure transient calculated by COCO are used as input to the BASH code. Data from both the SATAN-VI code and the WREFLOOD code out to BOC are input to the LOCBART code which calculates core average conditions at BOC for use by the BASH code.

BASH provides a more realistic thermal-hydraulic simulation of the reactor core and RCS during the reflood phase of a large break LOCA. Instantaneous values of the accumulator conditions and safety injection flow at the time of completion of lower plenum refill are provided to BASH by WREFLOOD. Figure 15.4-2 illustrates how WREFLOOD has been replaced by BASH in calculating transient values of core inlet flow, enthalpy, and pressure for the detailed fuel rod model, LOCBART. A more detailed description of the BASH code is available in Reference 39. The BASH code provides a much more sophisticated treatment of steam/water flow phenomena in the reactor coolant system during core reflood. A more dynamic interaction between core thermal-hydraulics and system behavior is expected, and experiments have confirmed this expectation. The BART code has been coupled with a loop model to form the BASH code and BART provides the entrainment rate for a given flooding rate. The loop model determines the loop flows and pressure drops in response to the calculated core exit flow determined by BART. The updated inlet flow is used by BART to calculate a new entrainment rate to be fed back to the loop code. This process of transfering data between BART, the loop code and back to BART forms the calculational process for analyzing the reflood transient. This coupling of the BART code with a loop code produces a more dynamic flooding transient, which reflects the close coupling between core thermal-hydraulics and loop behavior.

INSERT B (continued)

The cladding heatup transient is calculated by LOCBART which is a combination of the LOCTA code with BART. A more detailed description of the LOCBART code can be found in Reference 39. During reflood, the LOCBART code provides a significant improvement in the prediction of fuel rod behavior. In LOCBART the empirical FLECHT correlation has been replaced by the BART code. BART employs rigorous mechanistic models to generate heat transfer coefficients appropriate to the actual flow and heat transfer regimes experienced by the fuel rods.

The analyses were performed using an upper head fluid temperature equal to the hot leg temperature. The effect of using hot leg temperature in the reactor vessel upper head region is described in Reference 13.

A reference three-break spectrum analysis was performed for Diablo Canyon Power Plant (DCPP) Unit 2, the unit with the higher power rating. The analysis presented here assumed a full core of Westinghouse VANTAGE 5 fuel. VANTAGE 5 fuel will be loaded into Unit 1 and Unit 2 during the cycles 4-5 reloads.

The containment back pressure is calculated using the methods and assumptions described in Reference 2, Appendix A. Both units have containments with similar internal steel and concrete structural heat sinks. Input parameters used for the DCPP analyses are presented in Table 15.4-5.

The containment initial conditions of 85°F and 14.7 psia are representatively low values anticipated during normal full power operation. The initial relative humidity was conservatively assumed to be 98.8%.

Modeling features necessary to account for the reactor barrel-baffle region and the reactor fuel assembly thimbles were included in this analysis as presented in References 38, 39,and 40. The impact of a no single failure assumption for the ECCS was examined by re-analyzing the most limiting break with maximum ECCS flows as required by Reference 39.

The large break analysis was performed with the approved December 1981 version of the Evaluation Model (Reference 34), with BART (Reference 37), which includes modifications delineated by E. P. Rahe in Reference 40.

SATAN-VI is used to calculate the RCS pressure, enthalpy, density, and the mass and energy flow rates in the RCS, as well as steam generator energy transfer between the primary and secondary systems as a function of time during the blowdown phase of the LOCA. SATAN-VI also calculates the accumulator water mass and internal pressure and the pipe break mass and energy flow rates that are assumed to be vented to the containment during blowdown. At the end of the blowdown and refill phases, these data are transferred to the WREFLOOD code. Also at the end-of-blowdown, the mass and energy release rates during blowdown are transferred to the COCO code for use in the determination of the containment pressure response during these phases of the LOCA. Additional SATAN-VI output data from the end-of-blowdown, including the core pressure, and the core power decay transient, are input to the LOCTA-IV code.

With input from the SATAN-VI Code, WREFLOOD uses a system thermal-hydraulic model to determine the core flooding rate (i.e., the rate at which coolant enters bottom of the core), the coolant pressure and temperature, and the quench front height during the refill and reflood phases of the LOCA. WREFLOOD also calculates the mass and energy flow addition to the containment through the break. Since the mass flow rate to the containment depends upon the core flooding rate and the local core pressure, which is a function of the containment backpressure, the WREFLOOD and COCO codes are interactively linked. With input and boundary conditions from WREFLOOD, the mechanistic core heat transfer model in BART calculates the hydraulic and heat transfer conditions in the core during reflood. LOCTA-IV is used throughout the analysis of the LOCA transient to calculate the fuel clad temperature and metal-water reaction of the hottest rod in the core. A schematic representation of the computer code interfaces for large break calculations is shown in Figures 15.4-A and 15.4-B.

15.4.1.1.3 Results

Table 15.4-3 presents the peak cladding temperatures, hot spot metal-water reaction, and other key results for a range of break sizes for DCPP Unit 2. The range of break sizes was determined to include the limiting case for peak cladding temperature (PCT) from sensitivity studies reported in References 11 and 12. Results obtained show the limiting break to be the DECLG, C_D = 0.4. Table 15.4-4 presents the same parameters for the limiting case (DECLG, C_D = 0.4) break for DCPP Unit 1. The low PCT value computed for Unit 1 makes it apparent that Unit 2 results are bounding. The peak linear power and core power used in the analyses are given in Tables 15.4-3 and 15.4-4. Since there is margin between the value of the peak linear power density used in this analysis and the value expected in operation, a lower PCT a lower PCT would be obtained by using the peak linear power density expected during operation.

For the results discussed below, the hot spot is defined to be the location of maximum PCT. This location is given in Tables 15.4-3 and 15.4-4 for each break size analyzed.

INSERT C

Figures 15.4-1 through 15.4-62 present the transients for the principal parameters for the break sizes analyzed, and where appropriate, the worst break maximum safeguards case. The following items are noted:

Figures	Subject
15.4-~~1~~ 3 through 15.4-~~12~~ 14	The following quantities are presented at the cladding burst location and at the hot spot (location of maximum cladding temperature), both on the hottest fuel rod (hot rod): (1) Fluid quality (2) Mass velocity (3) Heat transfer coefficient. The heat transfer coefficient shown is calculated by the LOCTA IV code.
15.4-~~13~~ 15 through 15.4-~~24~~ 26	The system pressure shown is the calculated pressure in the core. The flowrate out the break is plotted as the sum of both ends for the guillotine break cases. The core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.
15.4-~~25~~ 27 through 15.4-~~36~~ 38	These figures show the hot spot cladding temperature transient and the cladding temperature transient at the burst location. The fluid temperature shown is also for the hot spot and burst location. The core flow (top and bottom) is also shown.
15.4-~~37~~ 39 through 15.4-~~44~~ 46	These figures show the core reflood transient.
15.4-~~45~~ 47 through 15.4-~~52~~ 52	These figures show the ECCS flow for all cases analyzed. As described earlier, the accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in refill reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.

INSERT C

15.4.1.1.3.1 Large Break Core Coolable Geometry Evaluation for Grid Deformation

Analyses were performed to determine the effect on the large break LOCA Peak Cladding Temperature (PCT) as a result of deformed structural grids occurring on peripheral VANTAGE 5 fuel assemblies. Analyses performed to determine the loads on the fuel grids for combined LOCA and Seismic loads have determined that peripheral VANTAGE 5 fuel assemblies will experience some grid deformation when VANTAGE 5 fuel assemblies are mixed with the existing 17X17 LOPAR fuel. The maximum theoretical deformation of a VANTAGE 5 grid cell results in a flow area reduction of 39.3%. In order to model the deformation of the grids and the resulting flow channel confinement, various input parameters were modified in the SATAN-VI blowdown code and the LOCBART clad heat-up code. The THINC flow blockage model described in Ref(9) was used to determine the necessary input modifications in SATAN-VI and LOCBART. The result was an increase in PCT of 37°F (PCT = 2108°F) due to the flow area reduction of the deformed grids. The increase in PCT did not result in exceeding the 2200°F limit or the 17% maximum Zircaloy oxidation limit. Thus the VANTAGE 5 fuel retains a coolable geometry under combined LOCA and Seismic loads. The higher PCT applies only during the transition from the current 17X17 LOPAR fuel design to an all VANTAGE 5 core.

Figures	Subject
15.4-53 through 15.4-~~56~~ 54	These figures show the containment pressure transient.
15.4-~~57~~ 55 through 15.4-~~60~~ 58	These figures show the core power transient.
15.4-~~61~~ 59 Through 15.4-60	~~This~~ These figures show the break energy released to the containment during blowdown for the limiting cases.
15.4-~~62~~ 61 through 15.4-62	~~This~~ These figures provide the containment wall condensing heat transfer coefficient for the limiting cases.

In addition to the above, Tables 15.4-6 and 15.4-7 present the reflood mass and energy releases to the containment and the broken loop accumulator mass and energy flowrates to the containment, respectively, for the DCPP Unit 2 limiting break.

The cladding temperature analysis is based on a total peaking factor of ~~2.40~~ 2.50. The hot spot metal-water reaction reached is ~~7.22~~ 7.40%, which is well below the embrittlement limit of 17%, as required by 10 CFR 50.46. In addition, the total core metal-water reaction is less than 0.3% for all breaks as compared with the 1% criterion of 10 CFR 50.46.

The results of several sensitivity studies are reported in Reference 12. These results are for conditions that are not limiting in nature and hence are reported on a generic basis.

15.4.1.1.4 Conclusions - Thermal Analysis

For breaks up to and including the double-ended severance of a reactor coolant pipe, the ECCS will meet the acceptance criteria as presented in 10 CFR 50.46. That is:

(1) The calculated peak fuel element cladding temperature provides margin to the requirement of 2200°F.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.

(3) The cladding temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17% are not exceeded during or after quenching.

(4) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

15.4.2 Major Secondary System Pipe Rupture

Two major secondary system pipe ruptures are analyzed in this section: rupture of a main steam line and rupture of a main feedwater pipe. The time sequence of events for each of these events is provided in Table 15.4-8.

15.4.2.1 Rupture of a Main Steam Line

15.4.2.1.1 Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam pipe would result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam pipe rupture is a potential problem mainly because of the high power peaking factors that exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the SIS.

The analysis of a main steam pipe rupture is performed to demonstrate that the following criteria are satisfied:

(1) Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the engineered safety features (ESF) there is no consequential damage to the primary system and the core remains in place and intact.

(2) Energy release to containment from the worst steam pipe break does not cause failure of the containment structure.

Although DNB and possible cladding perforation following a steam pipe rupture are not necessarily unacceptable, the following analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

The following functions provide the necessary protection against a steam pipe rupture:

The liquid processed through the volume control tank contains dissolved fission and activation products, as well as undissolved radioactive noble gases. A spray nozzle located inside the tank on the inlet line strips part of the noble gases from the incoming liquid, and these gases are retained in the volume control tank vapor space. In addition, an overpressure of hydrogen cover gas is provided for the tank to control the hydrogen concentration in the reactor coolant.

The volume control tank is located in a vault which is a Design Class I structure, so that in the event of a rupture or spill all liquids are retained in the vault. The volume of the tank vault is sufficient to contain the full contents of the tank without spillage from the vault. Any gases released from the volume control tank are collected by the auxiliary building ventilation system and discharged via the auxiliary building vent.

15.4.9.2 Conclusions

The probability of a volume control tank rupture is small, but the probability of the release of all or part of the contents of a tank through operator error or valve failure should be considered somewhat greater. The release of the total contents of a volume control tank is taken as the postulated accident. Smaller leaks and spills from the volume control tank were found to have negligible environmental consequences and therefore are not included. The analysis of the radiological effects of this accident is contained in Section 15.5.

15.4.10 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 4, 1974.

2. F. M. Bordelon, et al, Westinghouse ECCS Evaluation Model - Summary, WCAP-8339, July 1974.

3. F. M. Bordelon, et al, SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant, WCAP-8302 (Proprietary), June 1974 and WCAP-8306, June 1974.

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

5. R. D. Kelly, et al, Calculational Model for Core Reflooding after a Loss-of Coolant Accident (WREFLOOD Code), WCAP-8170 (Proprietary), June 1974 and WCAP-8171, June 1974.

6. F. M. Bordelon, and E. T. Murphy, Containment Pressure Analysis Code (COCO), WCAP-8327 (Proprietary), June 1974 and WCAP-8326, June 1974.

7. "Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," Federal Register, November 1974.

8. F. M. Bordelon, et al, Westinghouse ECCS Evaluation Model - Supplementary Information, WCAP-8471-P-A (Proprietary) June 1975 and WCAP-8472-A, June 1975.

9. Westinghouse ECCS Evaluation Model - October 1975 Version, WCAP-8522, (Proprietary) November 1975, and WCAP-8523, November 1975.

10. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassallo of the Nuclear Regulatory Commission, Letter Number NS-CE-924, January 23, 1976.

11. W. J. Johnson, et al, Westinghouse ECCS - Four Loop Plant (17x17) Sensitivity Studies, WCAP-8565-P (Proprietary), July 1975, and WCAP-8566, July 1975.

12. R. Salvatori, Westinghouse ECCS - Plant Sensitivity Studies, WCAP-8340 (Proprietary), July 1974, and WCAP-8356, July 1974.

13. Letter from C. Eicheldinger of Westinghouse Electric Corporation to V. Stello of the Nuclear Regulatory Commission, Letter No. NS-CE-1163, August 13, 1976.

14. Westinghouse ECCS Evaluation Model - February 1978 Version, WCAP-9220-P-A, (Proprietary), February 1978, and WCAP-9221-A, February 1978.

15. J. M. Geets, MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System, WCAP-7909, June 1972.

16. F. S. Moody, "Transactions of the ASME," Journal of Heat Transfer, February 1965, Figure 3, page 134.

17. C. Hunin, FACTRAN, A Fortran IV Code for Thermal Transients in a UO_2 Fuel Rod, WCAP-7908, June 1972.

18. T. W. T. Burnett, Reactor Protection System Diversity in Westinghouse Pressurized Water Reactor, WCAP-7306, April 1969.

19. T. G. Taxelius, ed. "Annual Report - Spert Project, October 1968 September 1969", Idaho Nuclear Corporation IN-1370, June 1970.

20. R. C. Liimatainen and F. J. Testa, Studies in TREAT of Zircaloy-2-Clad, UO_2-Core Simulated Fuel Elements, ANL-7225, January - June 1966, p. 177, November 1966.

21. D. H. Risher, Jr., An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1, December 1971.

22. D. H. Risher, Jr. and R. F. Barry, TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, WCAP-7979, November 1972.

23. A. A. Bishop, et al., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

24. R. F. Barry, LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094, WCAP 3269-26, September 1963.

25. F. M. Bordelon, Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code), WCAP 7969, September 1972.

26. T. W. T. Burnett, et al., LOFTRAN Code Description, WCAP-7907-P-A (Proprietary), WCAP-7907-A (Nonproprietary), April 1984.

27. NUREG-0612, Control of Heavy Loads at Nuclear Power Plant, PGandE's Response, May 9, 1983.

28. F. W. Dittus and L. M. K. Boelter, University of California (Berkeley), Publs. Eng., 2,433, 1930.

29. W. H. Jens and P. A. Lottes, Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water, USAEC Report ANL-4627, 1951.

30. Technical Specifications, Diablo Canyon Power Plant Units 1 and 2, Appendix A to License Nos. DPR-80 and DPR-82, as amended to the date of the most recent FSAR Update Revision.

31. J. S. Shefcheck, Application of the THINC Program to PWR Design, WCAP-7359-L, August 1969 (Proprietary), and WCAP-7838, January 1972.

32. ANSI/ANS-5.1-1979, American National Standard for Decay Heat Power in Light Water Reactors, 1979.

33. F. M. Bordelon, et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471-P-A, April 1975 (Proprietary) and WCAP-8472-A, April 1975 (Nonproprietary).

34. "Westinghouse ECCS Evaluation Model, 1981 Version," WCAP-9220-P-A, Rev. 1 (Proprietary), WCAP-9221-A, Rev. 1 (Nonproprietary), February 1982.

35. Letter from C. Eicheldinger (Westinghouse) to D. B. Vassallo (NRC), Letter Number NS-CE-924, dated January 23, 1976.

36. Letter from E. P. Rahe, (Westinghouse) to Thomas, C. O., NRC, Letter Number NS-EPR-2673, October 27, 1982, Subject: "Westinghouse Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2 (Proprietary).

37. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), dated December 21, 1983.

38. ~~M. Young, et al., "BART A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," WCAP-9561-P, Addendum 3, June 1986.~~

39. ~~J. S. Chiou, et al., "Models for PWR Reflood Calculations Using the BART Code," WCAP-10062.~~

40. ~~Letter from E. P. Rahe (Westinghouse) to James R. Miller (NRC), NS-EPRS-2679, November 8, 1982.~~

INSERT D

INSERT D

38. Young. M. Y., "BART-A1: A Computer core for the Best Estimate Analysis of Reflood Transients", WCAP-9561-P-A, with Addenda 2-3 (Proprietary), March, 1984.

39. Kabadi, J. N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model using the BASH code", WCAP-10266-P-A Rev.2 (Proprietary), March, 1987.

40. Johnson, W. J.; Thompson, C. M., "Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version", WCAP-9168 (Proprietary) and WCAP-9169 (Non-Proprietary), September 1977.

TABLE 15.4-A

UNIT 2 INPUT PARAMETERS USED IN THE ECCS ANALYSES

Parameter	
Reactor core design thermal power (MWt)	3411 + 2%
Peak linear power (kW/ft)	~~13.067~~ 13.61 at 6.0 ft
Total peaking factor (F_Q) at peak	~~2.40~~ 2.50
Hot channel enthalpy rise factor ($F_{\Delta H}$)	~~1.62~~ 1.65
Power shape	Chopped cosine
Fuel assembly array	17 X 17 ~~standard~~ VANTAGE-5 275 ~~500~~psig backfill pressure
Nominal cold leg accumulator water volume (ft^3/accumulator)	850
Nominal cold leg accumulator tank volume (ft^3/accumulator)	1350
Minimum cold leg accumulator gas pressure (psia)	600
Steam generator initial pressure (psia) (accounts for ~~10%~~ 15% SGTP)	~~758.19~~ 788.9
Steam generator tube plugging level (%)	~~10~~ 15
Initial flow in each loop (lb/sec)	~~9336.18~~ 8991.61
Vessel inlet temperature (°F) (accounts for ~~10%~~ 15% SGTP)	~~542.65~~ 548.49
Vessel outlet temperature (°F) (accounts for ~~10%~~ 15% SGTP)	~~609.15~~ 614.91
Reactor coolant pressure (psia)	2280

TABLE 15.4-B

UNIT 1 INPUT PARAMETERS USED IN THE ECCS ANALYSES

Parameter	
Licensed core power, 102% of:	3338 MWt ~~+ 2%~~
Peak linear power	~~12.79~~ 13.32 kW/ft
Total core peaking factor (F_Q)	~~2.40~~ 2.50
Hot channel enthalpy rise factor ($F_{\Delta H}$)	~~1.62~~ 1.65
Power shape	Chopped cosine
Fuel assembly array	17 X 17 ~~600~~ 275 psig ~~chamfered pellets~~ VANTAGE-5
Nominal cold leg accumulator water volume	850 ft^3/ accumulator
Nominal cold leg accumulator tank volume	1350 ft^3/ accumulator
Minimum cold leg accumulator gas pressure	600 psia
Steam generator initial pressure (accounts for ~~10%~~ 15% tube plugging)	~~760.06 psia~~ 788.9 psia
Steam generator tube plugging level	~~10%~~ 15%
Initial flow in each loop	~~9257.225~~ 8987.75 lbm/sec
Vessel inlet tempeature (accounts for ~~10%~~ 15% SGTP)	~~542.2°F~~ 540.49
Vessel outlet temperature (accounts for ~~10%~~ 15% SGTP)	~~608.0°F~~ 614.95
Reactor coolant pressure	2280 (psia)

TABLE 15.4-1

LARGE BREAK
TIME SEQUENCE OF EVENTS
DCPP UNIT 2

	DECL[(A)] C_D=0.8 (seconds)	DECL C_D=0.6 (seconds)	DECL C_D=0.4 (seconds)
Start	0.0	0.0	0.0
Rx Trip Signal	0.694	0.703	0.718
S.I.S.	0.53	0.61	0.75
Accumulator Injection starts	12.3	14.8	20.2
End of Blowdown	25.67	30.28	37.11
Bottom of Core Recovery	37.77	42.60	50.48
Accumulator Empty	48.33	51.74	58.18
Pump Injection	25.53	25.61	25.75
End of Bypass	25.67	30.28	37.11

(A) DECL stands for a double-ended cold leg break. C_D is the discharge coefficient.

TABLE 15.4-1 (Cont'd)

LARGE BREAK
TIME SEQUENCE OF EVENTS
DCPP UNIT 2

	DECL(A) C_D=0.4 MAXIMUM SAFETY INJECTION (seconds)
Start	0.0
Rx Trip Signal	0.718
S.I.S.	0.75
Accumulator Injection	20.2
End of Blowdown	36.84
Bottom of Core Recovery	49.41
Accumulator Empty	58.93
Pump Injection	25.75
End of Bypass	36.84

(A) DECL stands for a double-ended cold leg break. C_D is the discharge coefficient.

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 15.4-2

LARGE BREAK
TIME SEQUENCE OF EVENTS
DCPP UNIT 1

	DECL[A] C_D=0.4 (seconds)
Start	0.0
Rx Trip Signal	0.72
S.I.S.	0.75
Accumulator Injection	20.4
End of Blowdown	37.67
Bottom of Core Recovery	51.54
Accumulator Empty	58.59
Pump Injection	25.75
End of Bypass	37.67

(A) DECL stands for a double-ended cold leg break. C_D is the discharge coefficient.

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 15.4-3

LARGE BREAK ANALYSIS
RESULTS - DCPP UNIT 2

	DECL C_D=0.8	DECL C_D=0.6	DECL C_D=0.4
PEAK CLAD TEMP, (°F)	1747.2	1971.2	2071.1
PEAK CLAD LOCATION, (Ft)	6.0	6.25	6.25
LOCAL Zr/H_2O Rxn, Max (%)	2.10	5.92	7.40
LOCAL Zr/H_2O LOCATION,(Ft)	6.25	6.25	6.00
TOTAL Zr/H_2O Rxn, (%)	<0.3	<0.3	<0.3
HOT ROD BURST TIME, (Sec)	48.96	42.2	47.2
HOT ROD BURST LOCATION, (Ft)	6.25	6.25	6.0

CALCULATION

NSSS POWER MWt, 102% of	3411
PEAK LINEAR POWER, kW/Ft 102% of	13.61
PEAKING FACTOR (at license rating)	2.50[1]
ACCUMULATOR WATER VOLUME	850 Ft^3/Tank
STEAM GENERATOR TUBE PLUGGING	15% Per Steam Generator

----

[1]The Technical Specification "Heat Flux Hot Channel Factor - $F_Q(Z)$", (Spedification 3/4.2.2) is 2.45. A peaking factor limit of 2.45 is more restrictive than the 2.50 value assumed for the large break LOCA analyses.

TABLE 15.4-3 (Cont'd)

LARGE BREAK ANALYSIS
RESULTS - DCPP UNIT 2

	DECL C_D=0.4 MAXIMUM SAFETY INJECTION	DECL C_D=0.4 GRID DEFORMATION[2] TRANSITION CORE
PEAK CLAD TEMP, (°F)	1877.6	2108.0
PEAK CLAD LOCATION, (Ft)	6.25	8.0
LOCAL Zr/H_2O Rxn, Max (%)	3.01	7.37
LOCAL Zr/H_2O LOCATION, (Ft)	6.00	8.0
TOTAL Zr/H_2O Rxn, (%)	<0.3	<0.3
HOT ROD BURST TIME, (Sec)	48.22	44.0
HOT ROD BURST LOCATION, (Ft)	6.00	6.0
CALCULATION		
NSSS POWER MWt, 102% of	3411	
PEAK LINEAR POWER, kW/Ft 102% of	13.61	
PEAKING FACTOR (at license rating)	2.50[3]	
ACCUMULATOR WATER VOLUME	850 Ft^3/Tank	
STEAM GENERATOR TUBE PLUGGING	15% per Steam Generator	

----

[2]The grid deformation results apply only during the transition from 17X17 LOPAR to VANTAGE-5 fuel. Once a full core of VANTAGE-5 fuel is achieved then the maximum calculated peak cladding temperature will be 2071°F.

[3]The Technical specification for "Heat Flux Hot Channel Factor - $F_Q(Z)$", (Spedification 3/4.2.2) is 2.45. The peaking factor limit of 2.45 is more restrictive than the 2.50 value assumed for the large break LOCA analyses.

TABLE 15.4-4

LARGE BREAK ANALYSIS
RESULTS - DCPP UNIT 1

	DECL C_D=0.4
PEAK CLAD TEMP, (°F)	2042.2
PEAK CLAD LOCATION, (Ft)	6.25
LOCAL Zr/H_2O Rxn, Max (%)	7.40
LOCAL Zr/H_2O LOCATION, (Ft)	6.25
TOTAL Zr/H_2O Rxn, (%)	<0.3
HOT ROD BURST TIME, (Sec)	55.12
HOT ROD BURST LOCATION, (Ft)	6.25

CALCULATION

NSSS POWER MWt, 102% of	3338
PEAK LINEAR POWER, kW/Ft 102% of	13.32
PEAKING FACTOR (at license rating)	2.50[4]
ACCUMULATOR WATER VOLUME	850 Ft^3/Tank
STEAM GENERATOR TUBE PLUGGING	15% per Steam Generator

---

[4] The Techincal Specification "Heat Flux Hot Channel Factor - $F_Q(Z)$", (Specification 3/4.2.2) is 2.45. A peaking factor limit of 2.45 is more restrictive than the 2.50 value assumed for the large break LOCA analyses.

TABLE 15.4-5

CONTAINMENT INPUT PARAMETERS USED FOR LOCA ECCS BACK PRESSURE ANALYSIS

NET FREE VOLUME, CU FT		2,630,000
INITIAL CONDITIONS		
Pressure,	Psia	14.7
Temperature,	°F	85
RWST temperature,	°F	39
Service water temperature,	°F	45
Outside temperature,	°F	39
Spray System		
Number of pumps operating		2
Runout flowrate, gpm		3250
Actuation time, sec		18
Safeguards Fan Coolers		
Number of fan coolers operating		5
Fastest postaccident initiation of the fan coolers, sec		20
Structural Heat Sinks		
Thickness, in.		Area, Ft^2
42.0, Concrete		65,749
12.0, Concrete		24,054
24.0, Concrete		14,313
12.0, Concrete		48,183
108.0, Concrete		20,492
30.0, Concrete		33,867
1.68, Steel		8,525
1.92, Steel		4,015
6.99, Steel		1,771

DCCP UNITS 1 & 2 FSAR UPDATE

TABLE 15.4-5 (Cont'd)

CONTAINMENT INPUT PARAMETERS USED

FOR LOCA ECCS BACK PRESSURE ANALYSIS

Structural Heat Sinks (Cont'd)

Thickness, in.	Area, Ft^2
0.5656, Steel	43,396
0.088, Steel	24,090
0.22, Steel	10,597
0.088, Steel	8,470
0.102, Steel	23,438
0.071, Steel	20,266
0.708, Steel	26,050
0.127, Steel	33,000
0.773, Steel	11,004
0.375, Steel	99,616
1.569, Steel	1,530
1.098, Steel	21,022
0.745, Steel	6,755
0.960, Steel	792
0.144, Stainless Steel	9,737
0.654, Stainless Steel	943
0.642, Steel	1,373
3.0, Steel	575
0.75, Steel	17,542

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 15.4-6

REFLOOD MASS AND ENERGY RELEASES

DCPP UNIT 2 DECLG, C_D=0.4

Time, Sec	M(total), Lbm/Sec	Mh(total), BTU/Sec
50.482	0.00	0.00
51.382	4.872	6.312×10^3
61.268	75.36	9.319×10^4
77.793	89.25	1.0985×10^5
97.443	102.14	1.2528×10^5
117.743	117.00	1.4121×10^5
137.743	324.33	2.0156×10^5
158.043	355.74	2.0426×10^5
246.893	426.17	2.0569×10^5

DCPP UNITS 1 & 2 FSAR UPDATE

TABLE 15.4-7

BROKEN LOOP INJECTION SPILL DURING BLOWDOWN

DCPP UNIT 2 DECLG, C_D=0.4

Time, SEC	M(total), Lbm/Sec	Mh(total) BTU/SEC
0.0	3168.75	$1.7316X10^5$
1.01	2949.24	$1.6117X10^5$
2.01	2772.53	$1.5151X10^5$
3.01	2625.10	$1.43454X10^5$
4.01	2499.24	$1.36606X10^5$
5.01	2391.42	$1.30683X10^5$
6.01	2295.99	$1.25469X10^5$
7.01	2211.06	$1.20827X10^5$
8.01	2134.78	$1.16659X10^5$
9.01	2065.56	$1.12876X10^5$
10.01	2002.17	$1.09412X10^5$
12.01	1889.67	$1.03265X10^5$
14.01	1792.82	$9.79719X10^4$
16.01	1708.55	$9.55922X10^4$
18.01	1643.57	$8.93245X10^4$
20.01	1569.04	$8.57431X10^4$
22.01	1510.49	$8.25440X10^4$
24.01	1458.98	$7.96797X10^4$
26.01	1466.77	$7.62693X10^4$
27.01	1446.28	$7.50914X10^4$

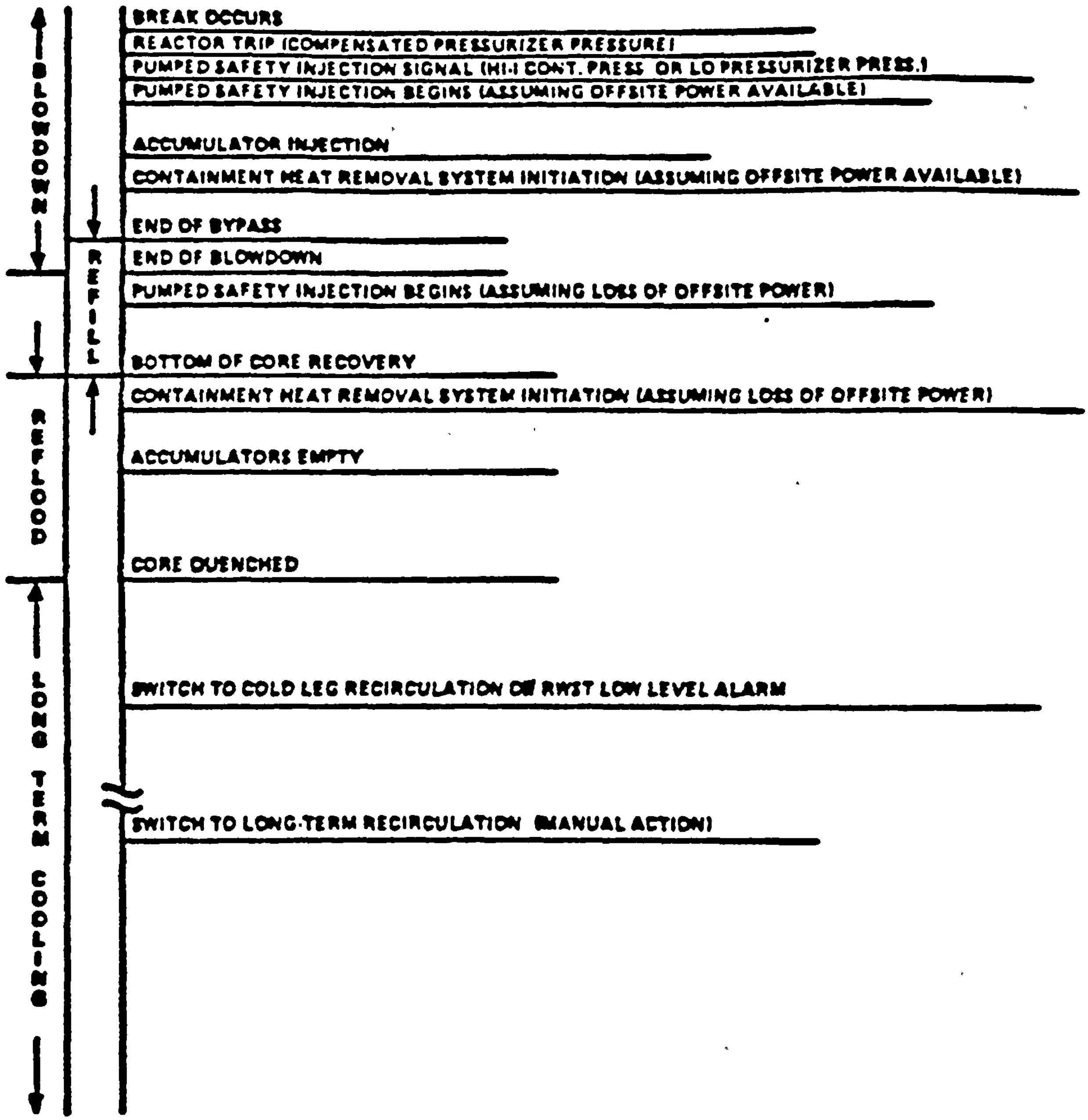


Figure 15.4-1 SEQUENCE OF EVENTS FOR LARGE BREAK LOCA ANALYSIS
DIABLO CANYON POWER PLANTS UNITS NO.1 & 2

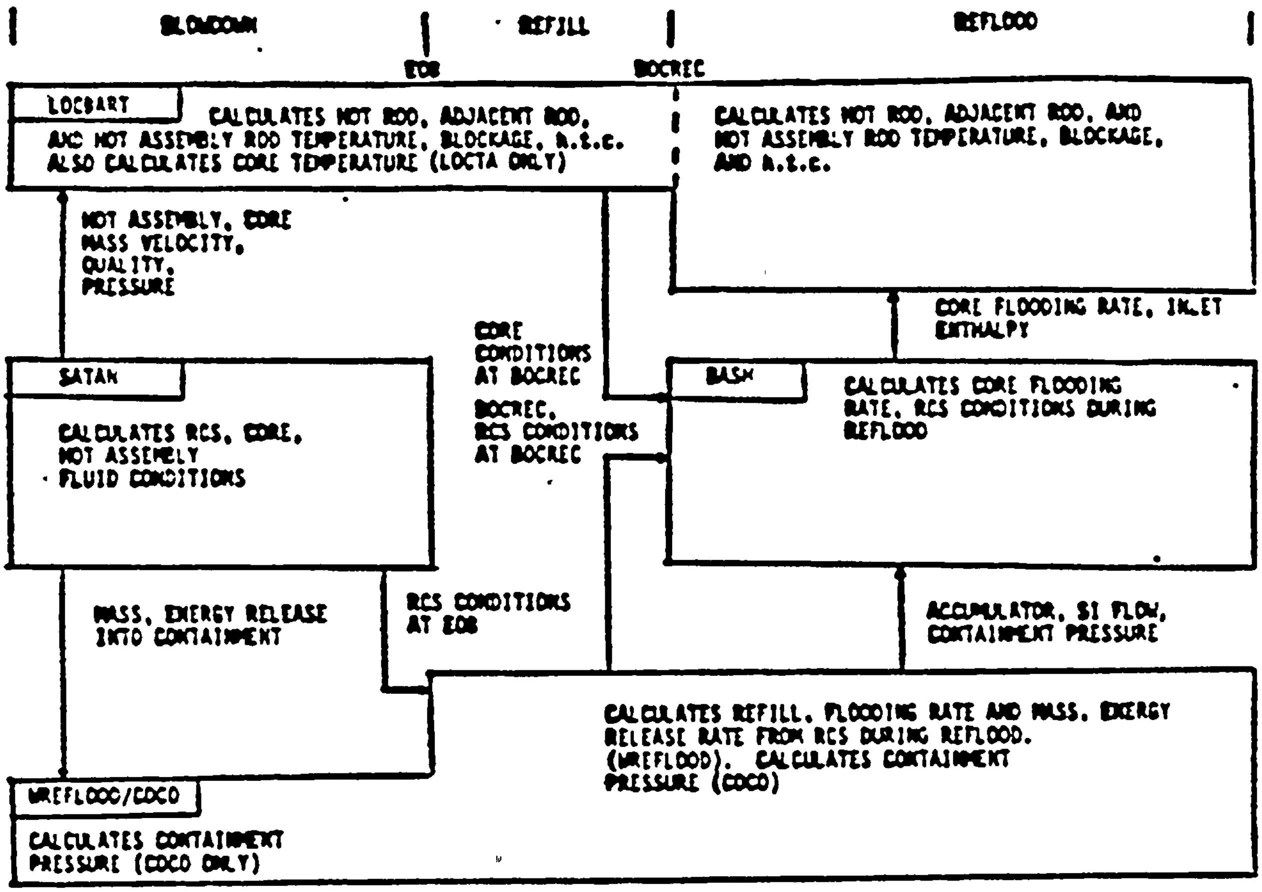


Figure 15.4-2 COMPUTER CODE INTERFACE DESCRIPTION
1981 EM + BASH LARGE BREAK LOCA MODELS
DIABLO CANYON POWER PLANTS UNITS NO.1 & 2

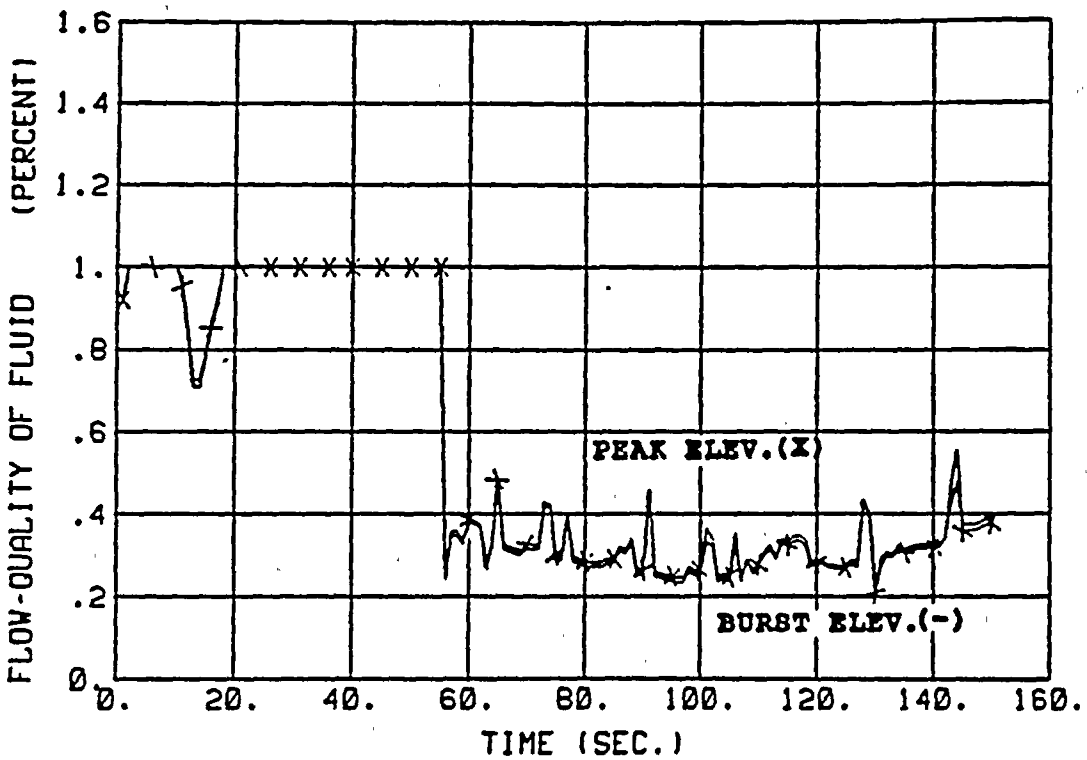


Figure 15.4-3 Fluid Quality - DECLG (C_D=0.8), Unit 2

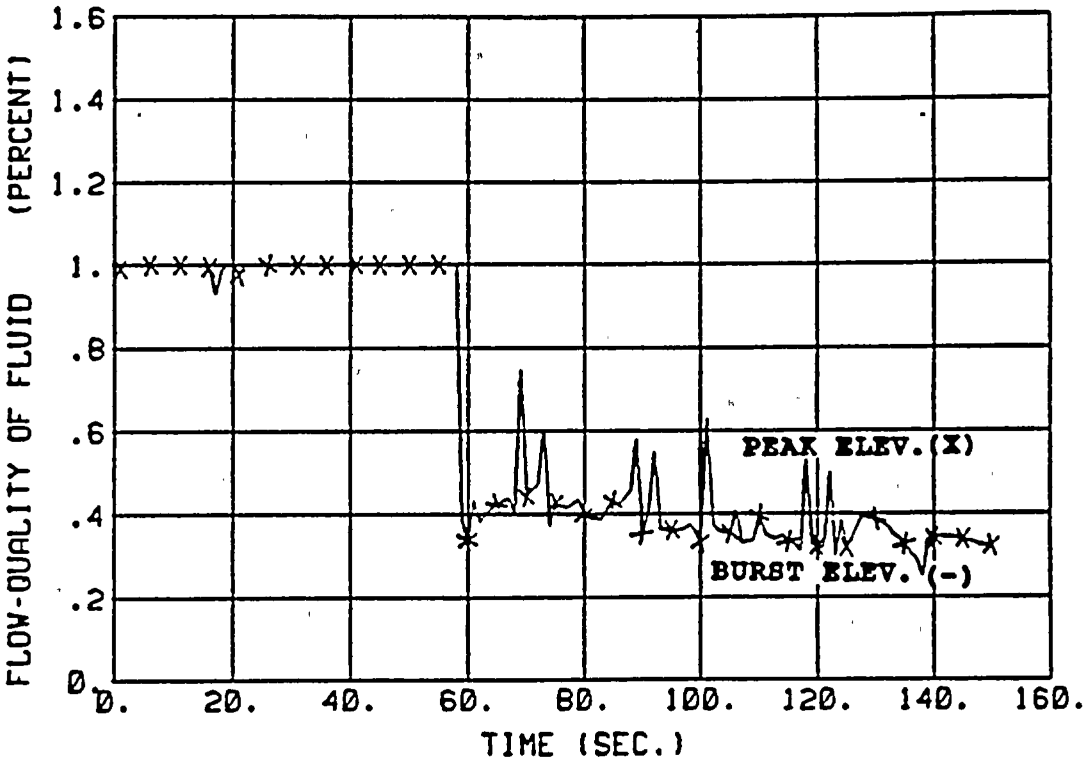


Figure 15.4-4 Fluid Quality - DECLG (C_D=0.6), Unit 2

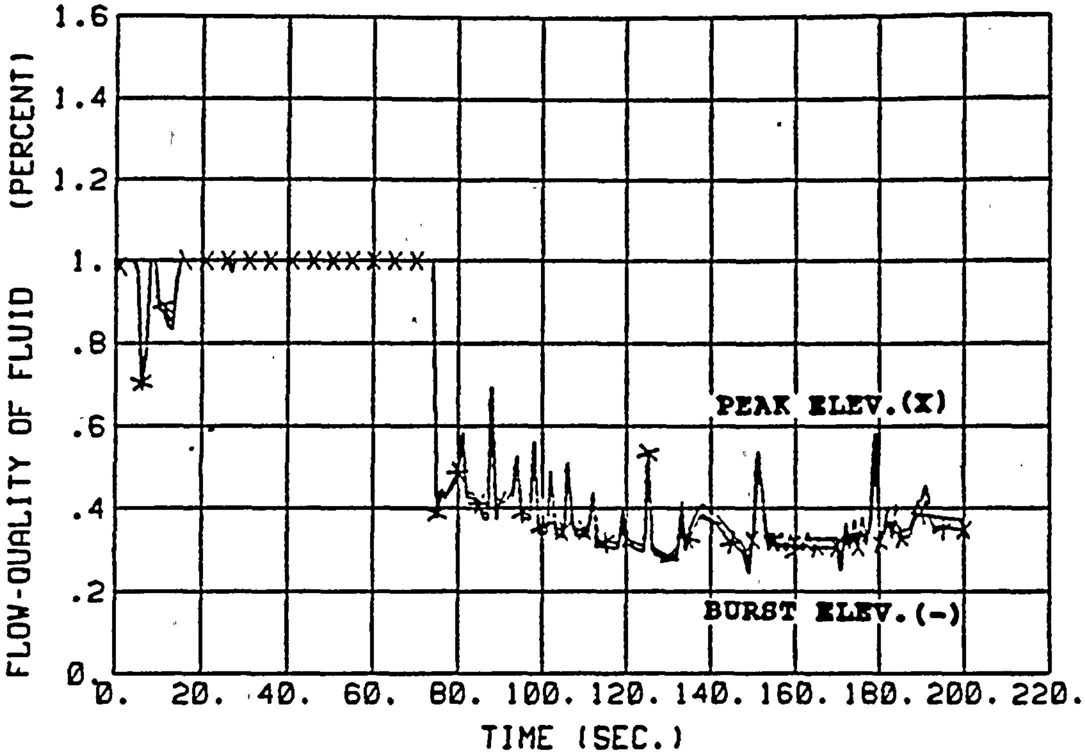


Figure 15.4-5 Fluid Quality - DECLG (C_D=0.4), Unit 2

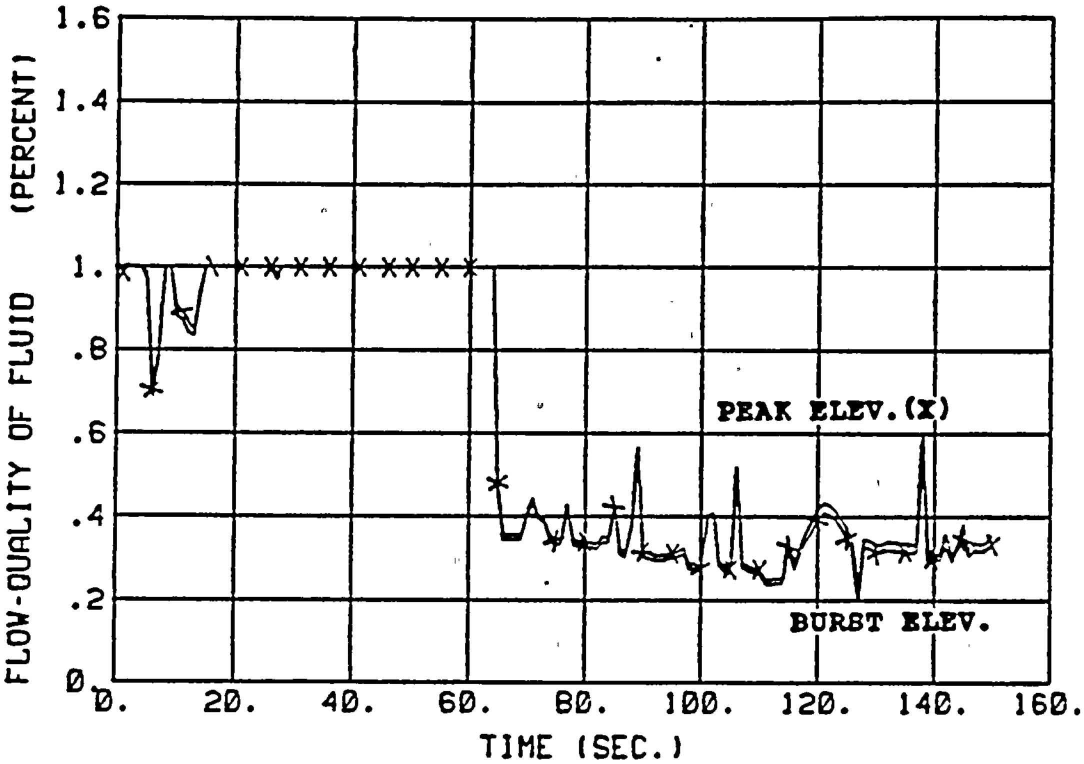


Figure 15.4-5A Fluid Quality - DECLG (C_D=0.4), Unit 2
Maximum Safety Injection

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

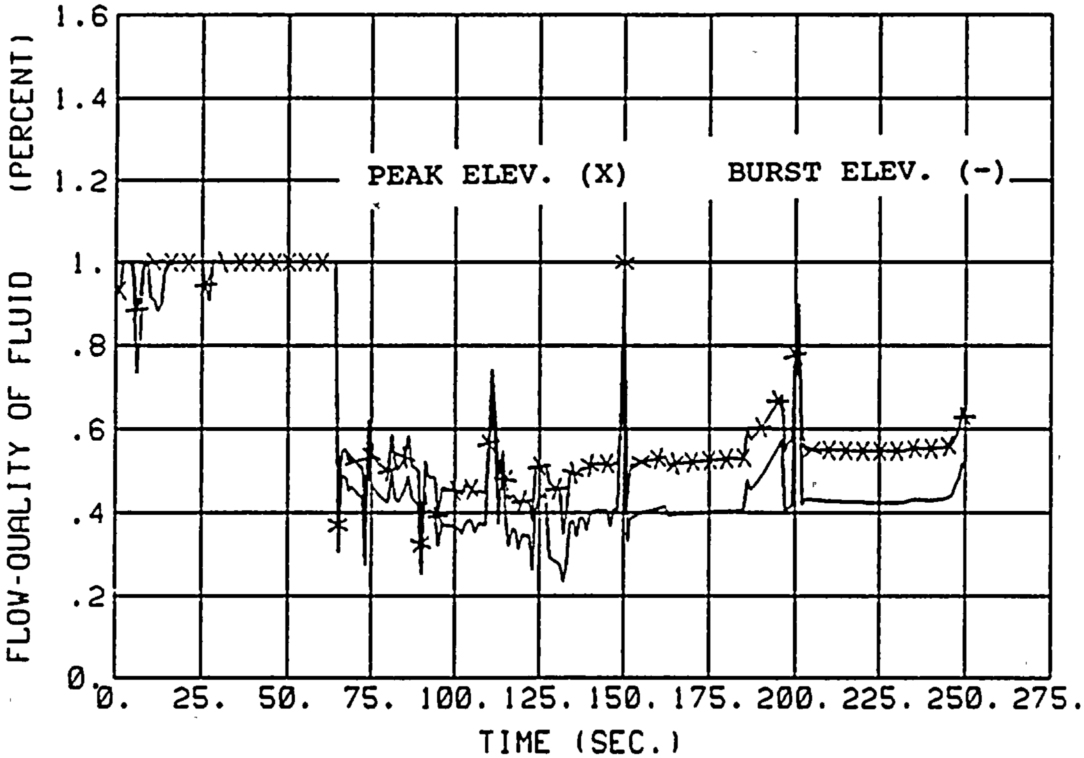


Figure 15.4-5B Fluid Quality - DECLG (C_D=0.4), Unit 2 Grid Deformation

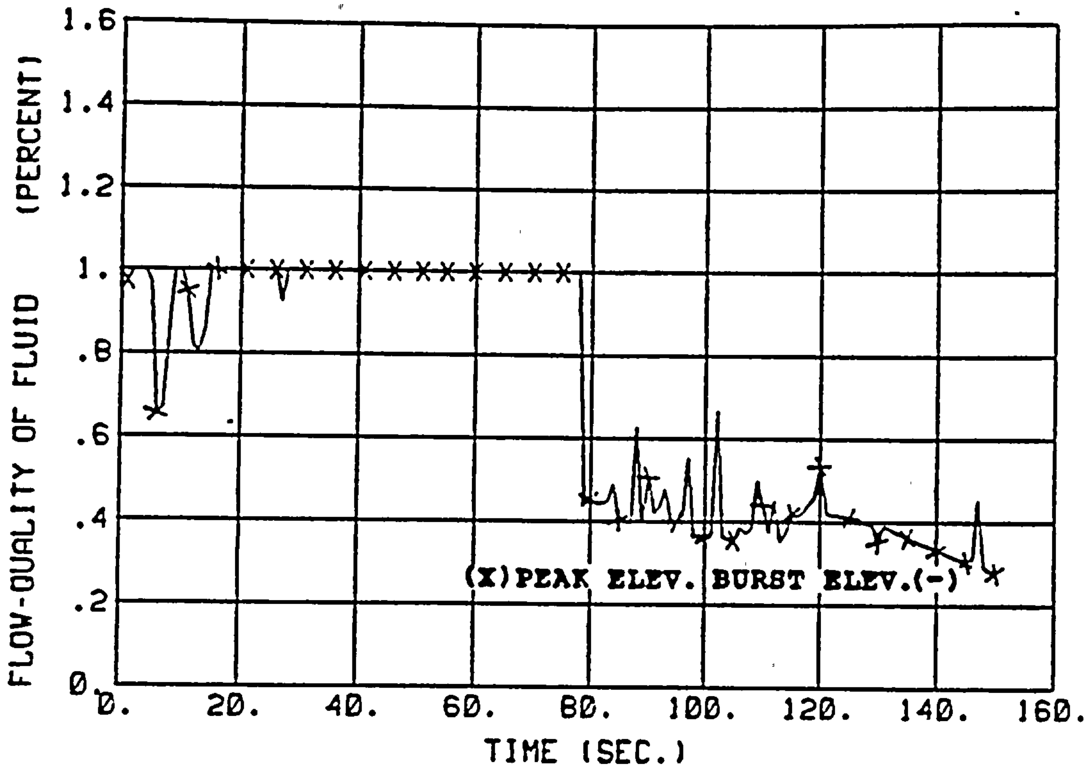


Figure 15.4-6 Fluid Quality - DECLG (C_D=0.4), Unit 1

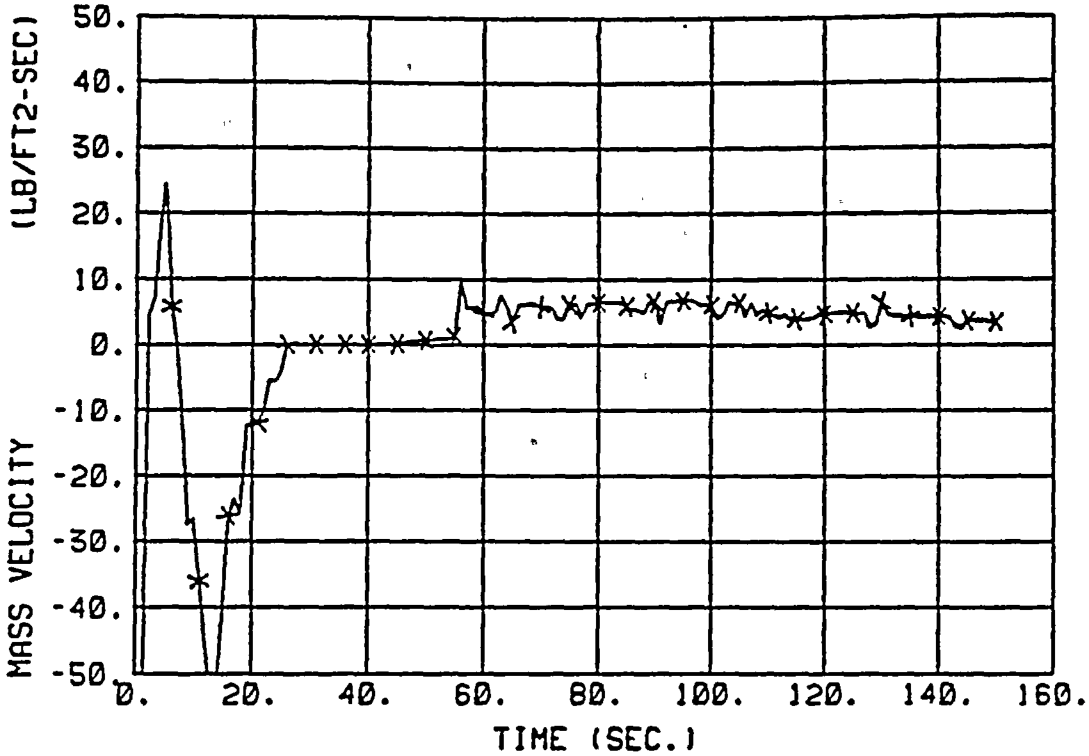


Figure 15.4-7 Mass Velocity - DECLG (C_D=0.8), Unit 2

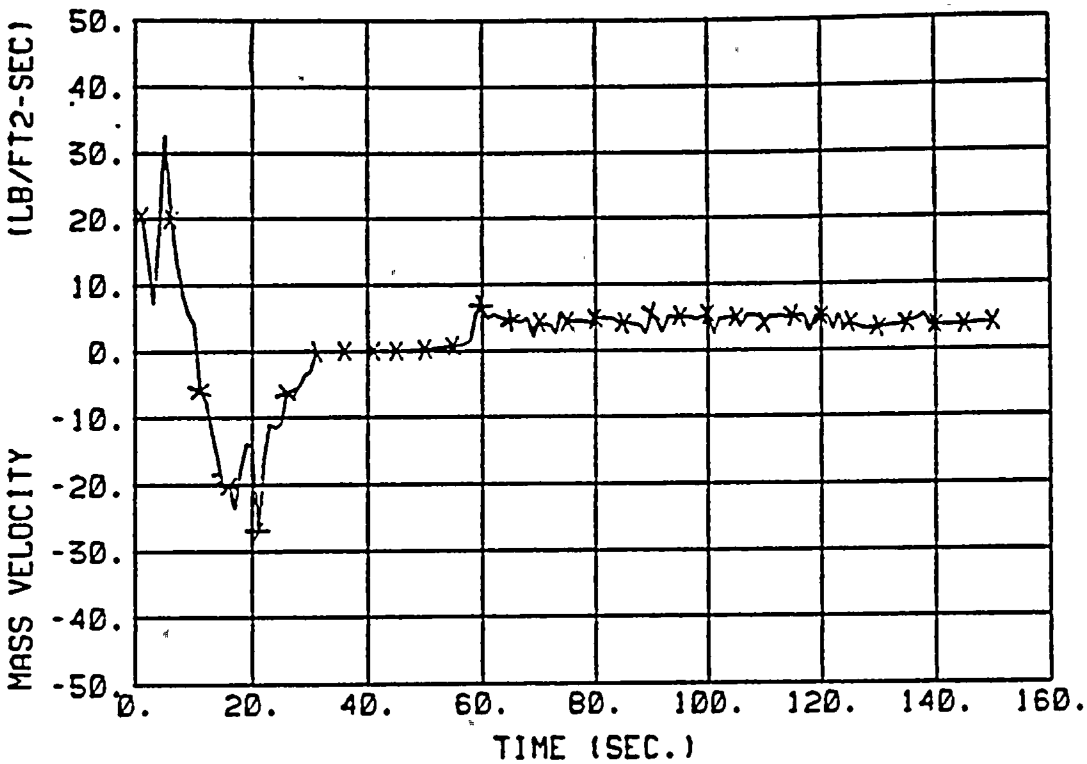


Figure 15.4-8 Mass Velocity - DECLG (C_D=0.6), Unit 2

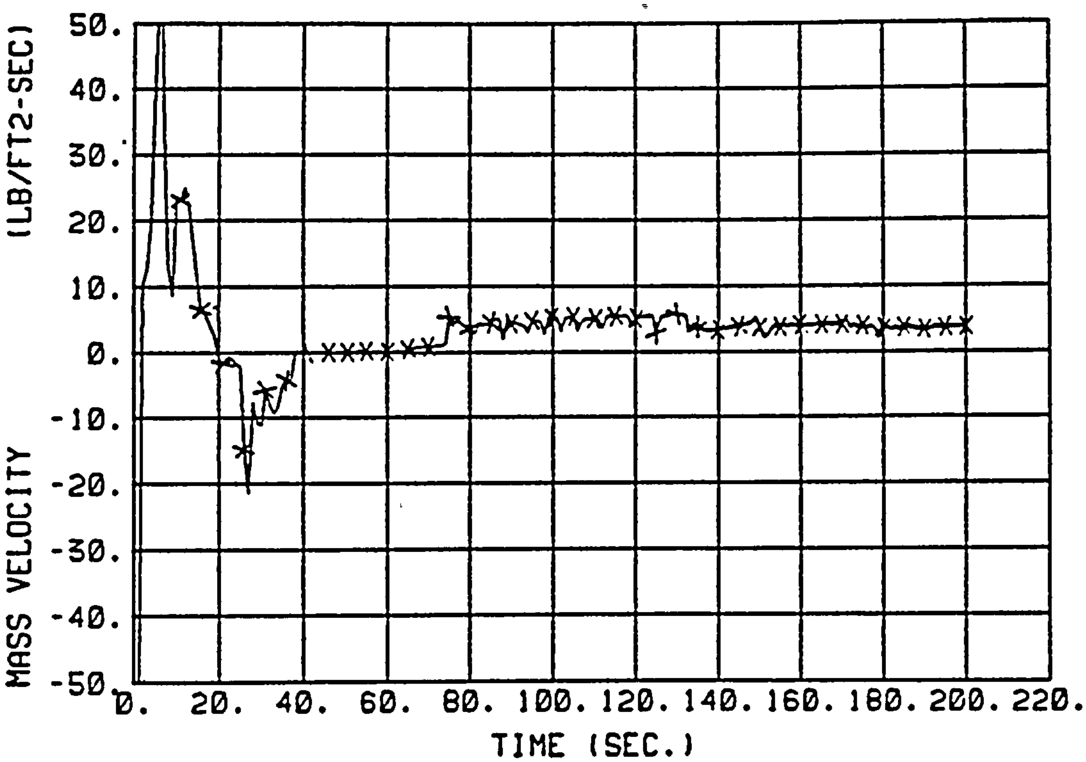


Figure 15.4-9 Mass Velocity - DECLG (C_D=0.4), Unit 2

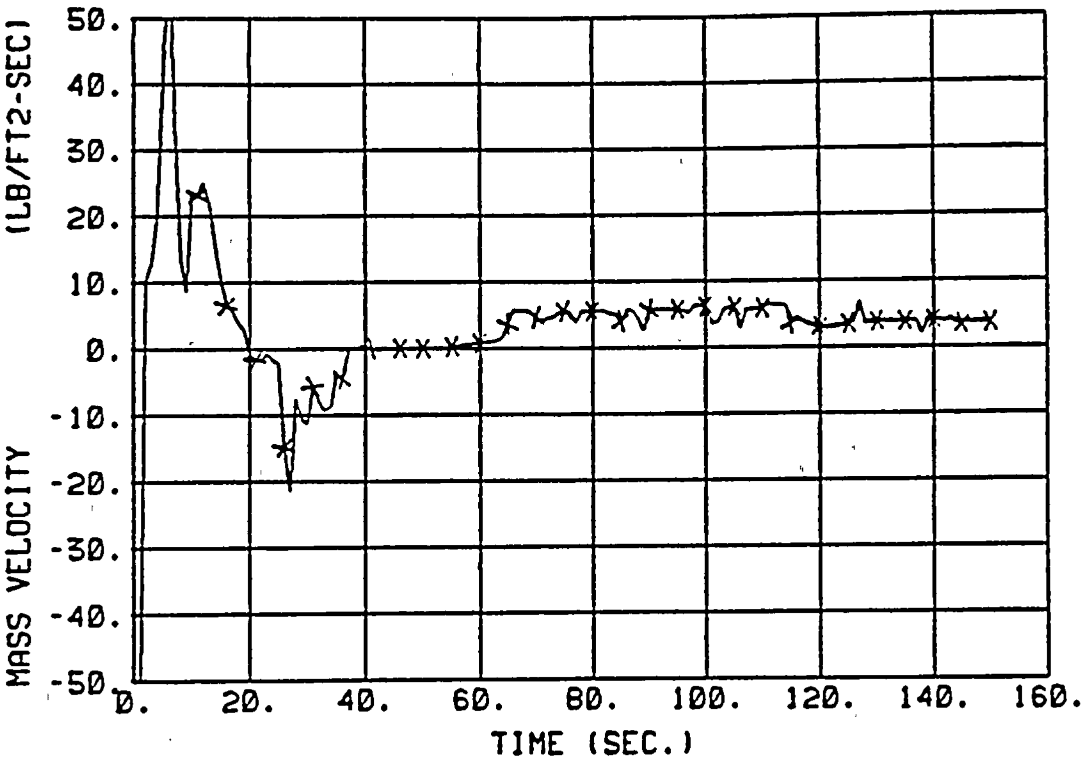


Figure 15.4-9A Mass Velocity - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

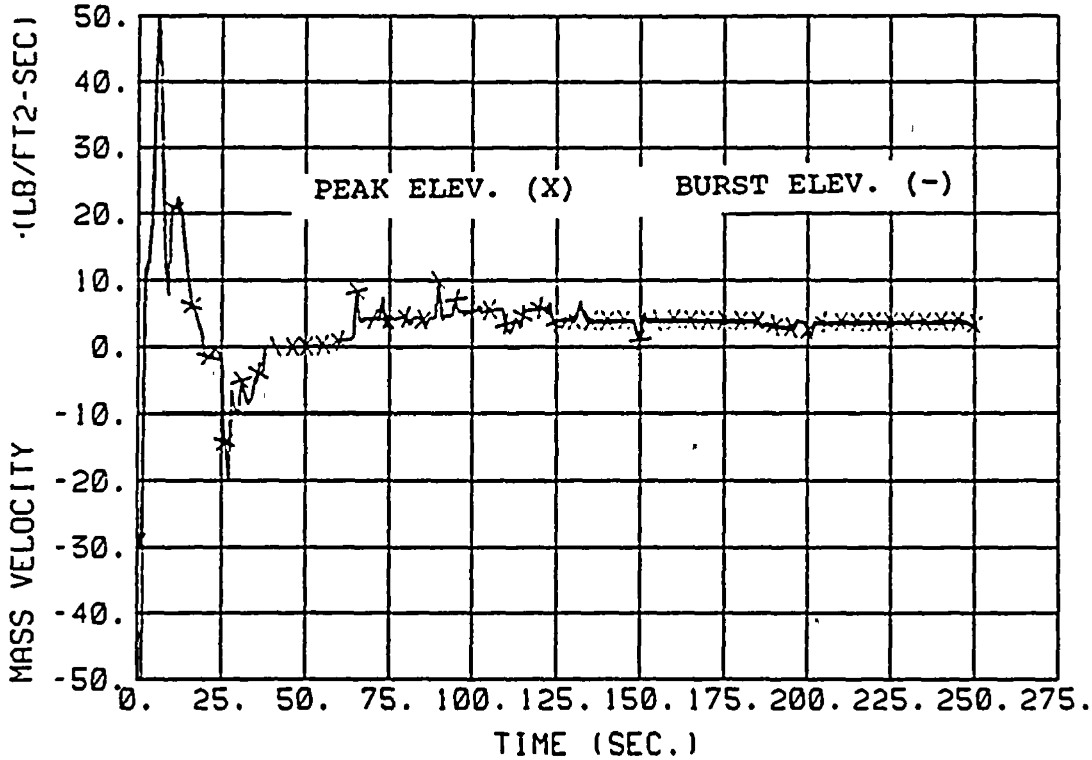


Figure 15.4-9B Mass Velocity - DECLG (C_D=0.4), Unit 2 Grid Deformation

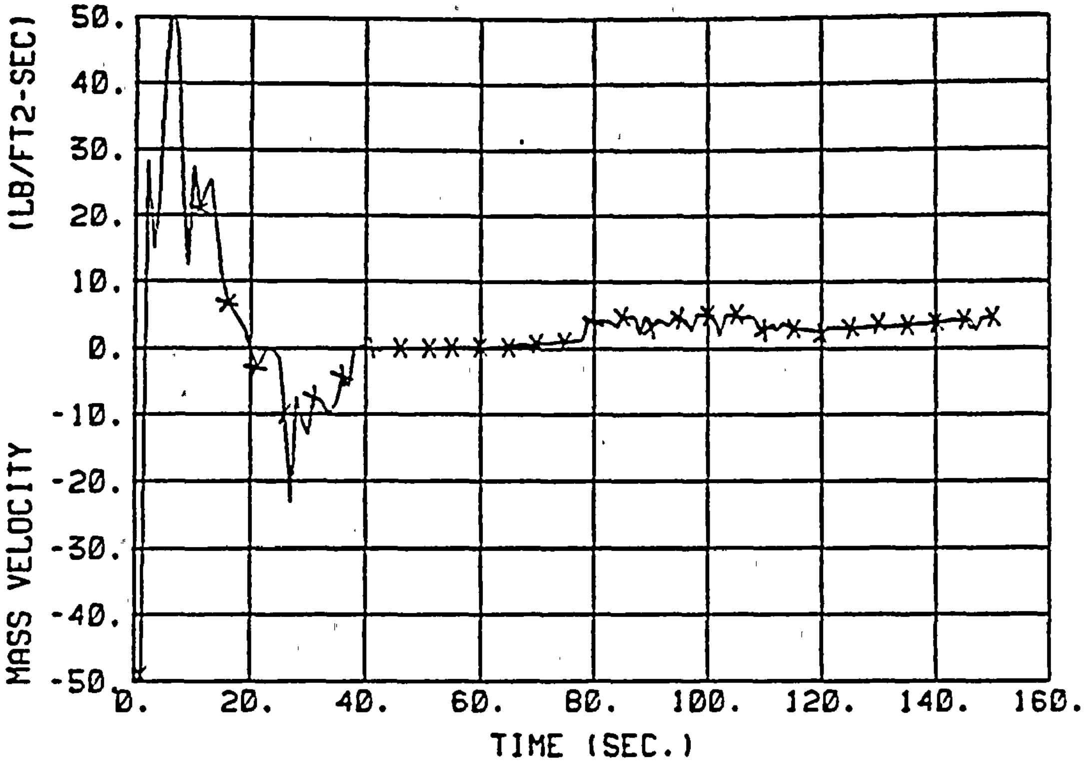


Figure 15.4-10 Mass Velocity - DECLG (C_D=0.4), Unit 1

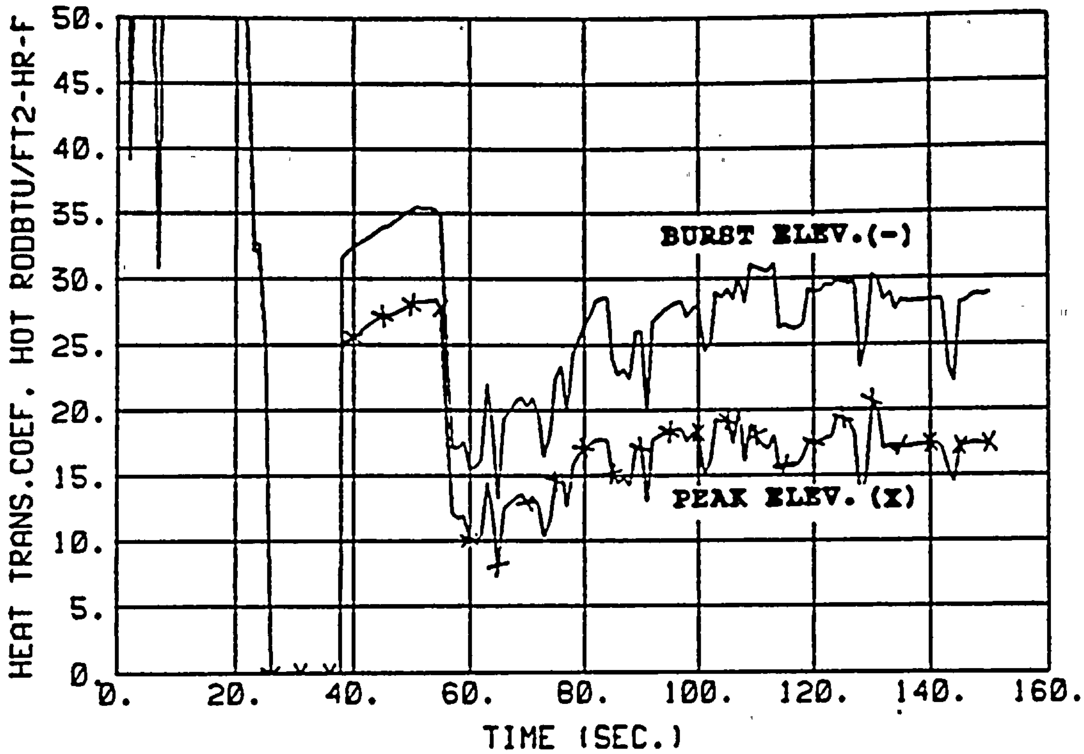


Figure 15.4-11 Heat Transfer Coefficient - DECLG (C_D=0.8), Unit 2

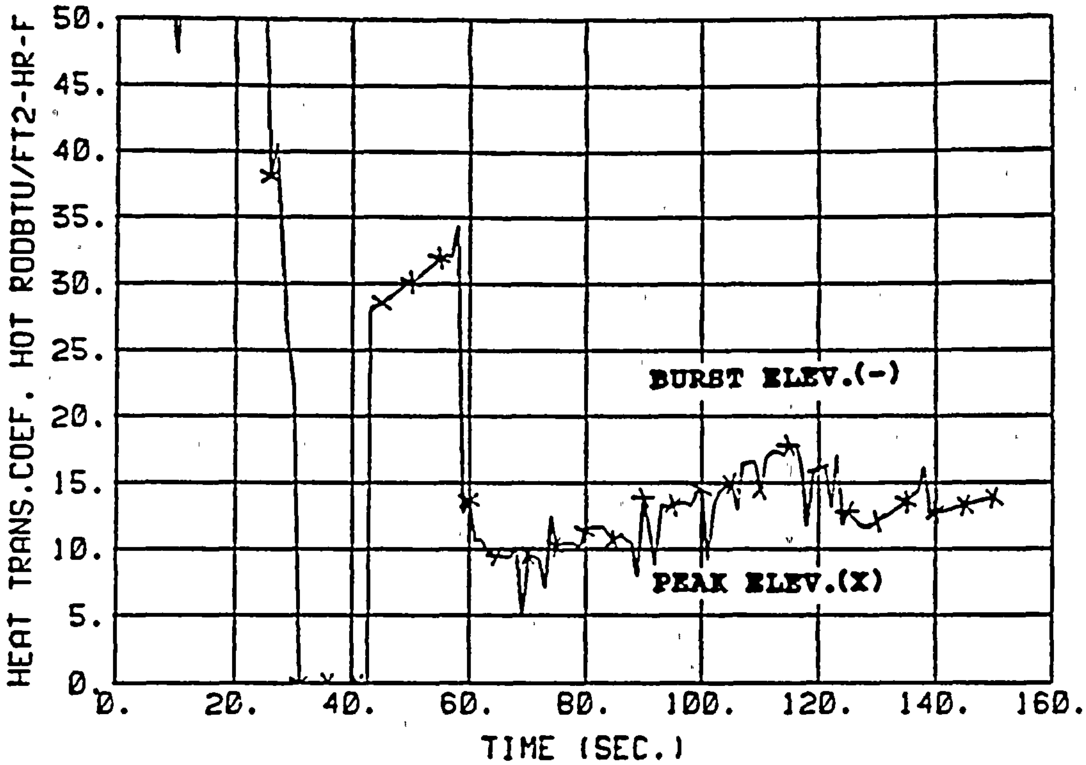


Figure 15.4-12 Heat Transfer Coefficient - DECLG (C$_D$=0.6), Unit 2

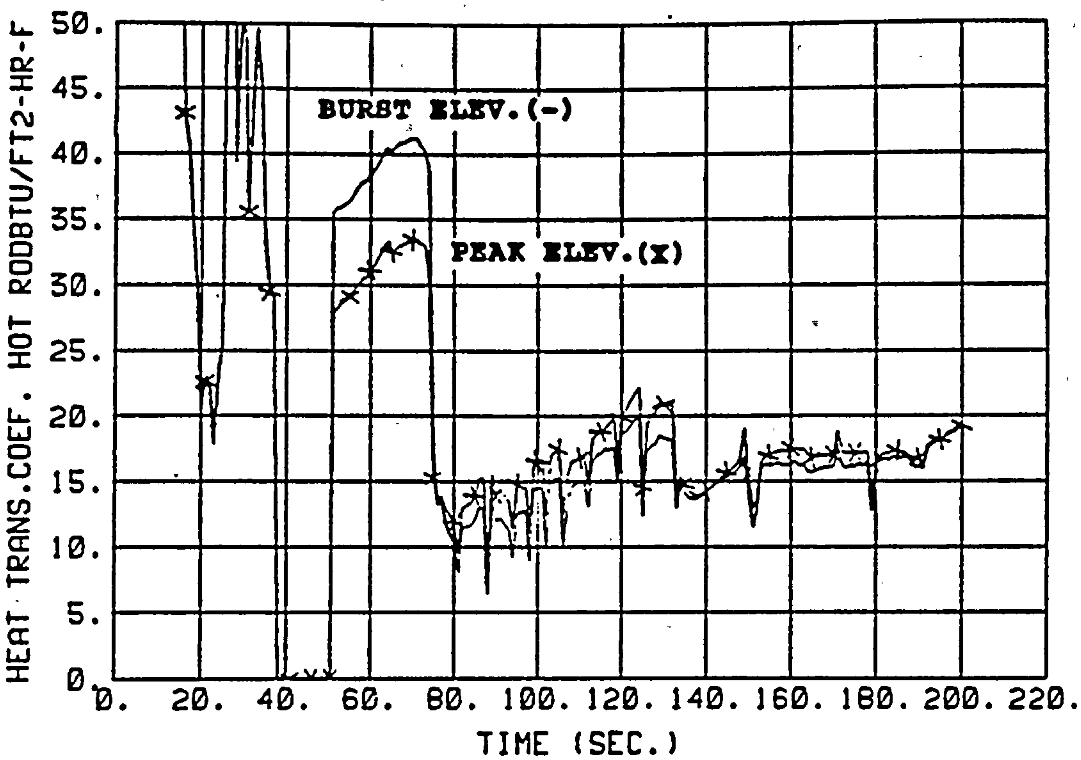


Figure 15.4-13 Heat Transfer Coefficient - DECLG (C_D=0.4), Unit 2

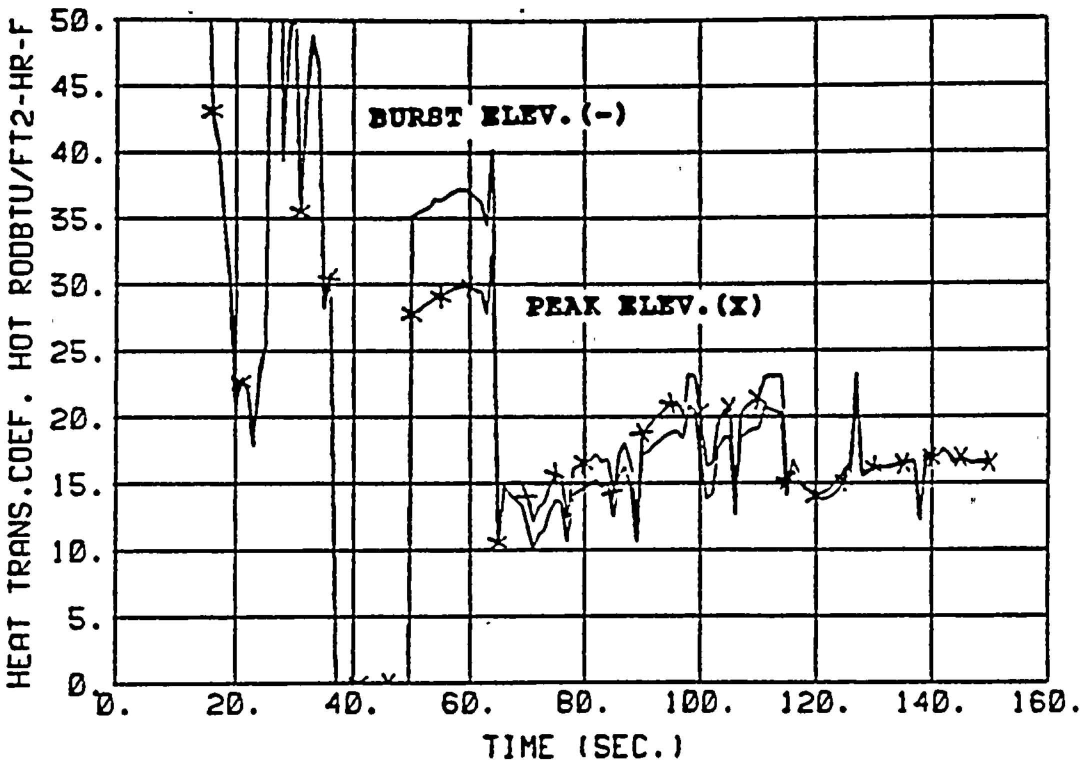


Figure 15.4-13A Heat Transfer Coefficient - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

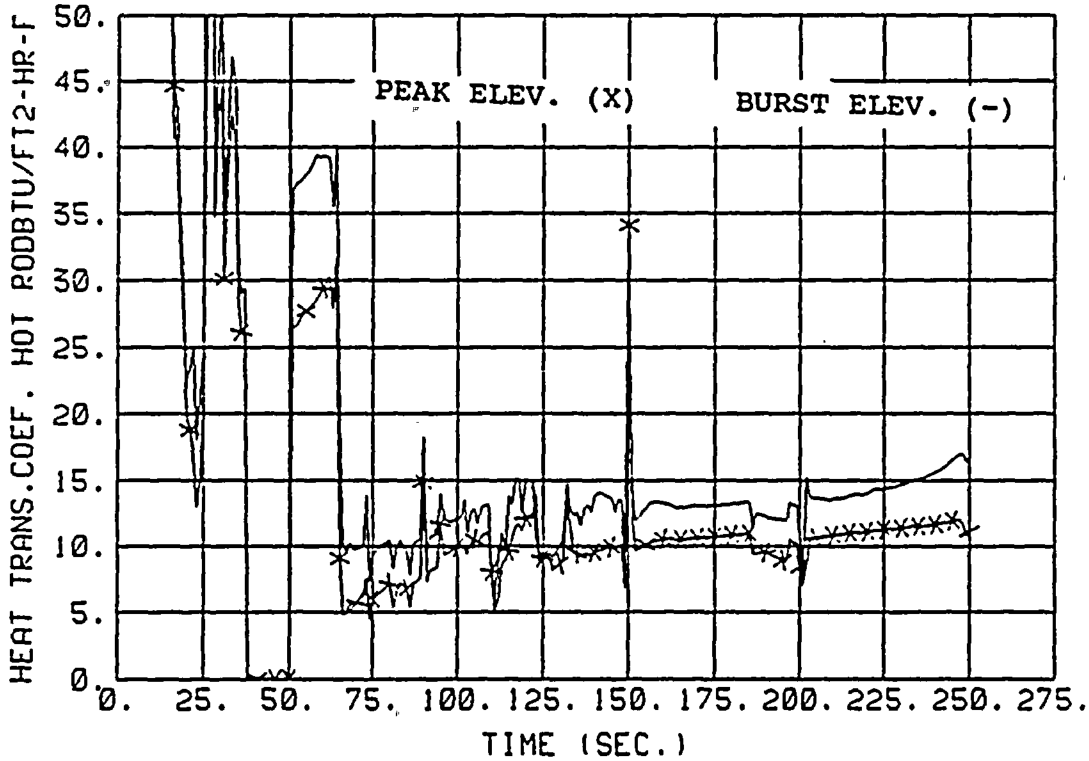


Figure 15.4-13B Heat Transfer Coefficient - DECLG (C_D=0.4), Unit 2 Grid Deformation

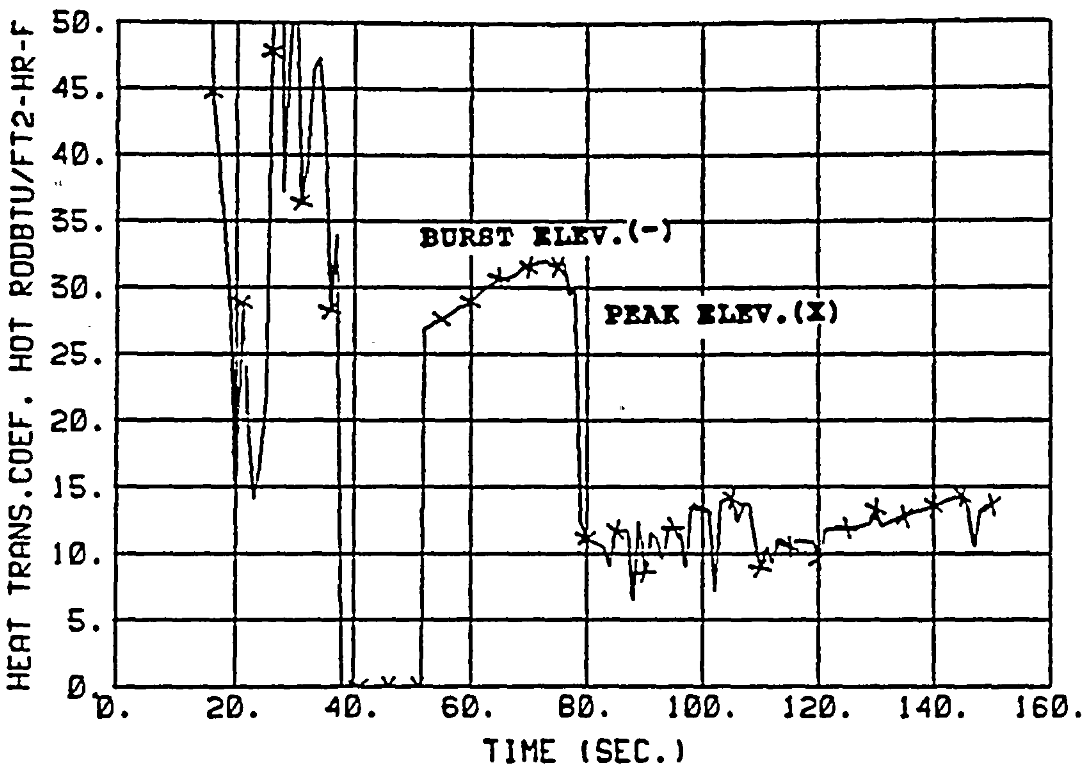


Figure 15.4-14 Heat Transfer Coefficient - DECLG (C_D=0.4), Unit 1

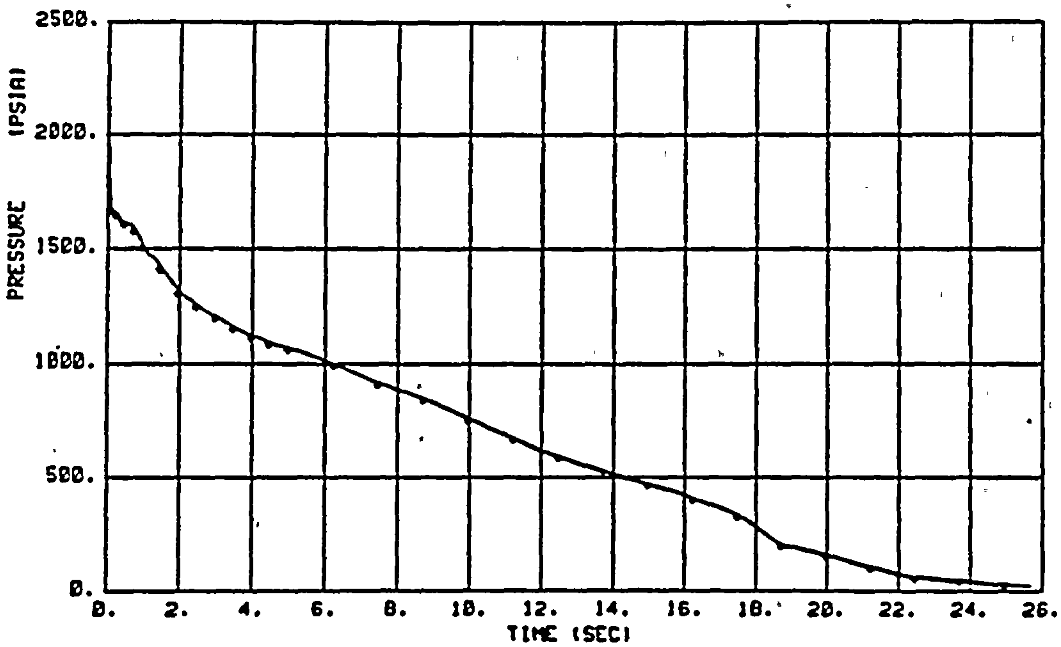


Figure 15.4-15 Core Pressure - DECLG (C_D=0.8), Unit 2

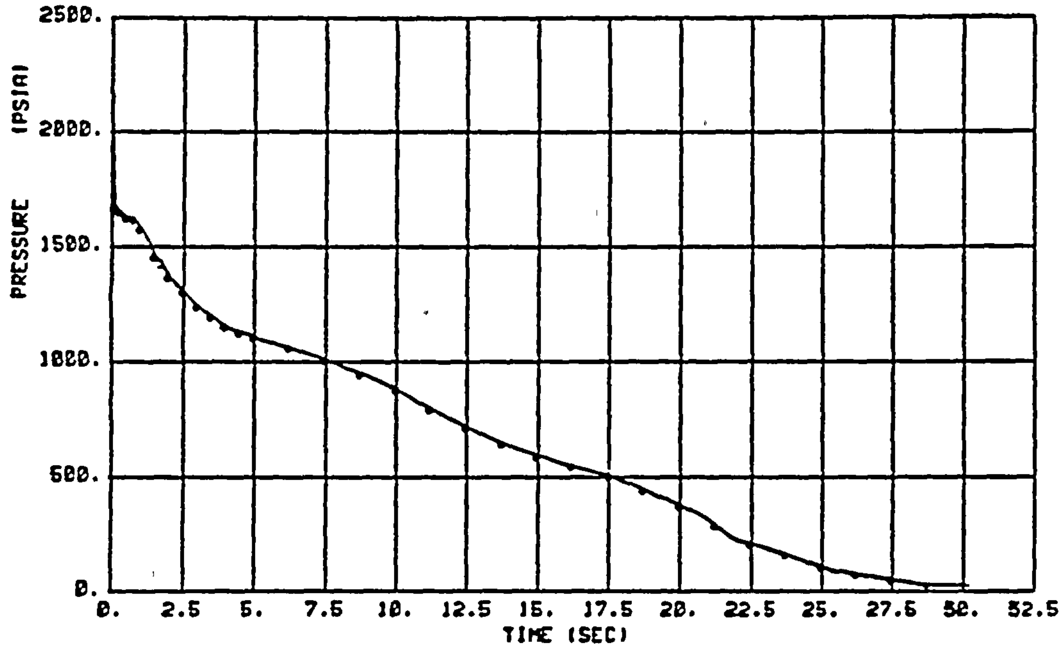


Figure 15.4-16 Core Pressure - DECLG (C_D=0.6), Unit 2

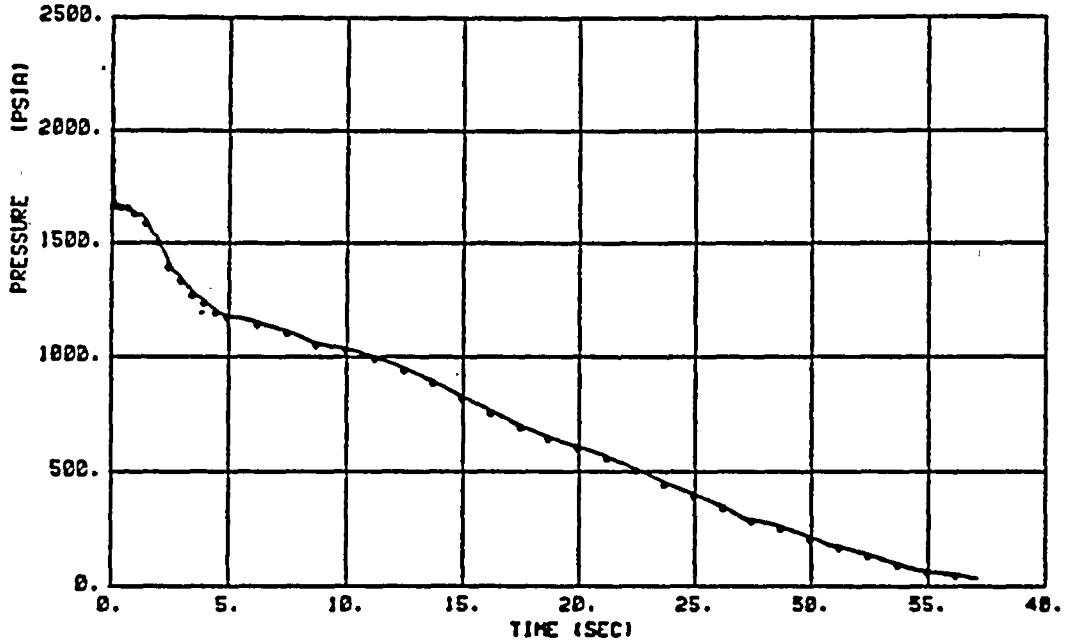


Figure 15.4-17 Core Pressure - DECLG (C_D=0.4), Unit 2

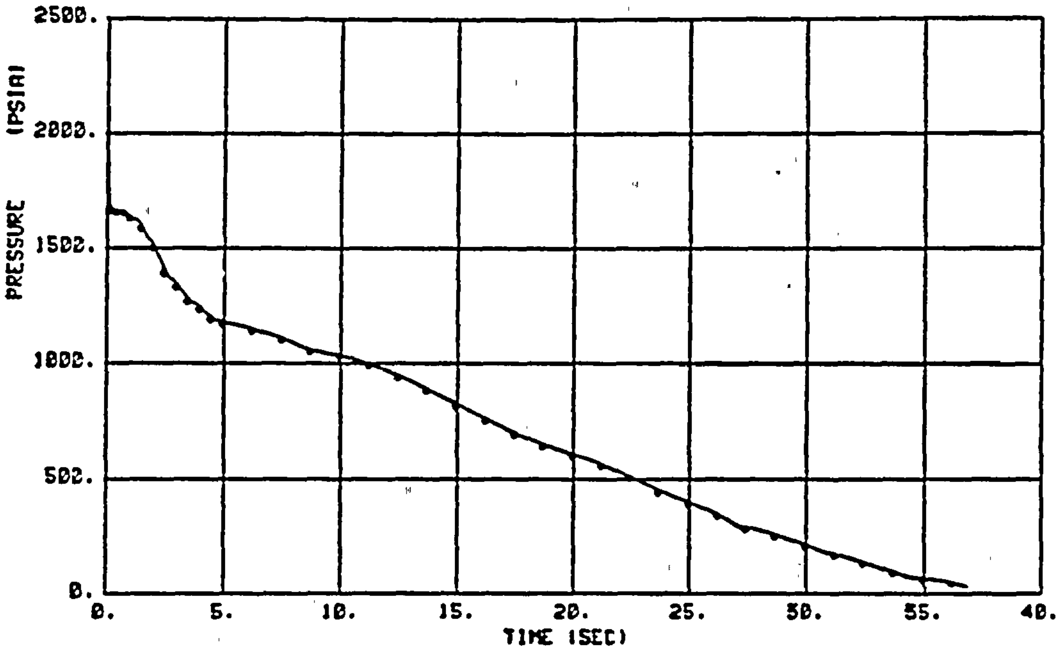


Figure 15.4-17A Core Pressure - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

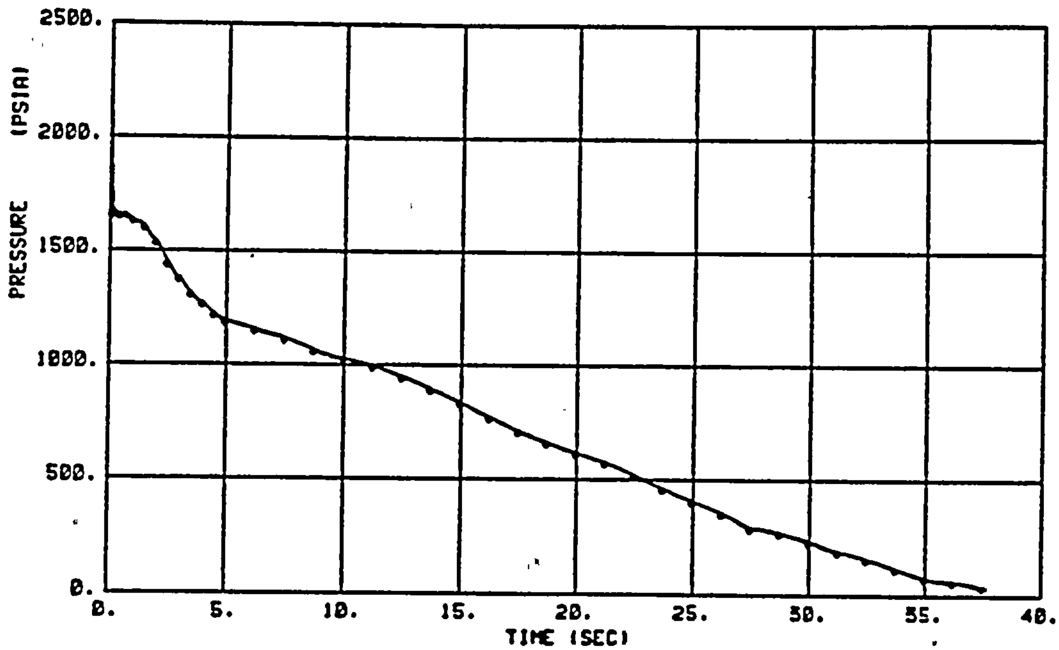


Figure 15.4-18 Core Pressure - DECLG (C_D=0.4), Unit 1

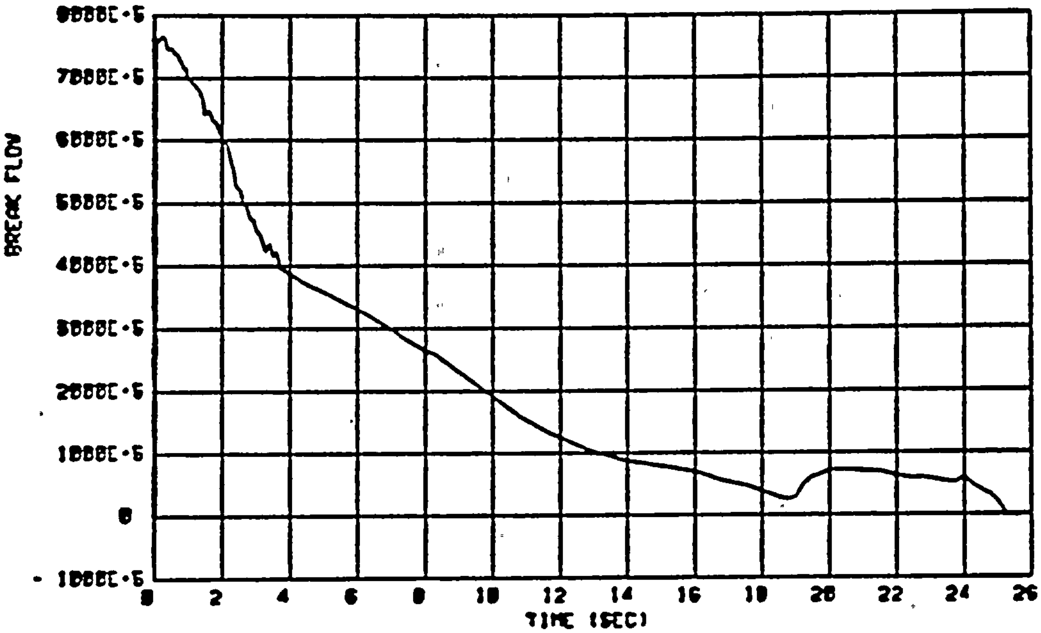


Figure 15.4-19 Break Flowrate - DECLG (C_D=0.8), Unit 2

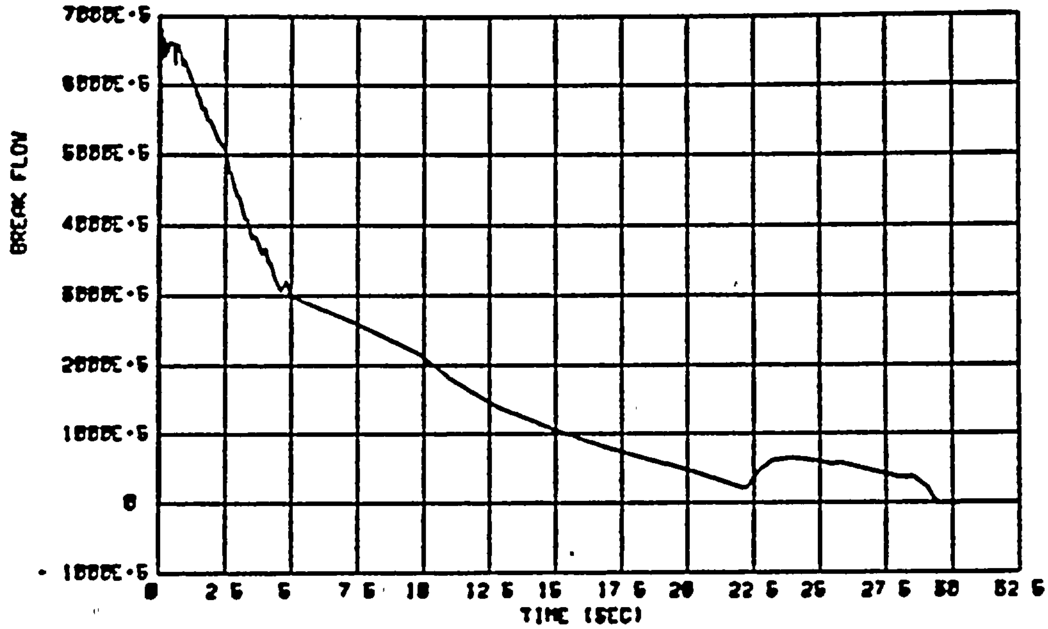


Figure 15.4-20 Break Flowrate - DECLG (C_D=0.6), Unit 2

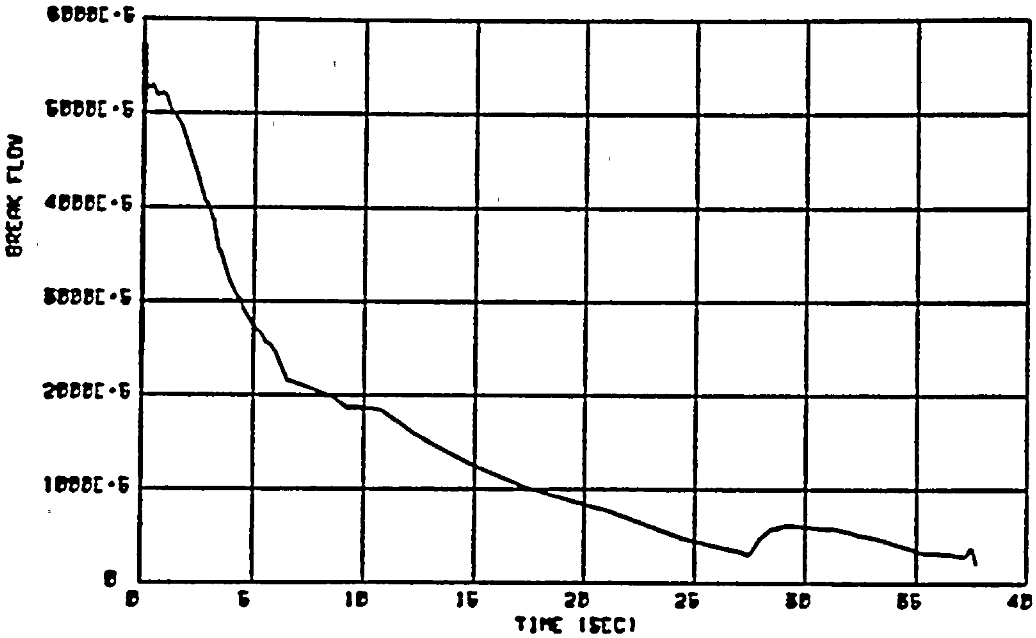


Figure 15.4-21 Break Flowrate - DECLG (C_D=0.4), Unit 2

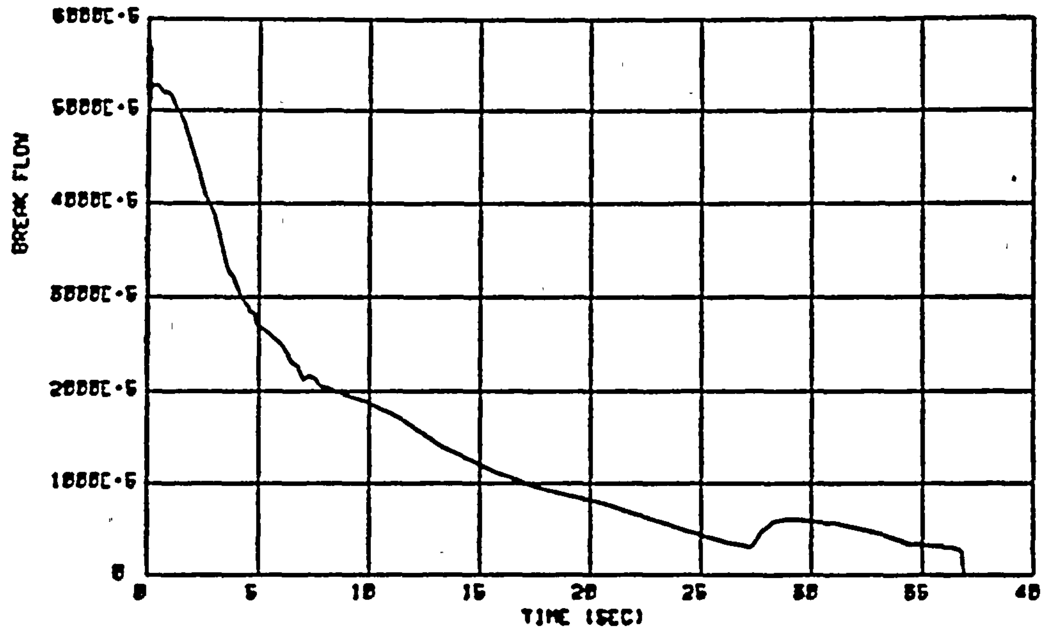


Figure 15.4-21A Break Flowrate - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

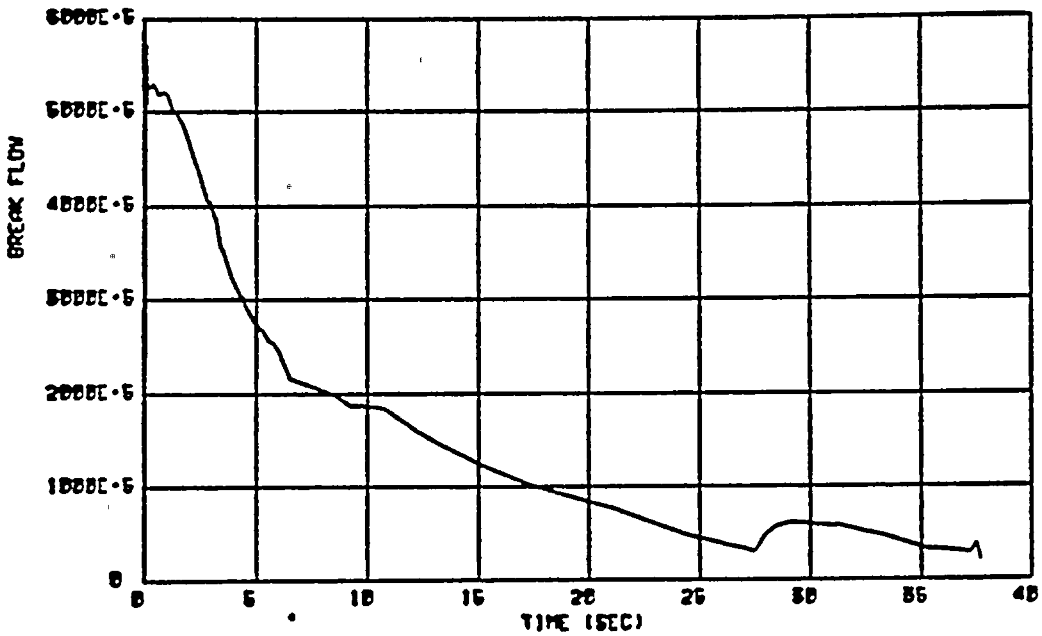


Figure 15.4-22 Break Flowrate - DECLG (C_D=0.4), Unit 1

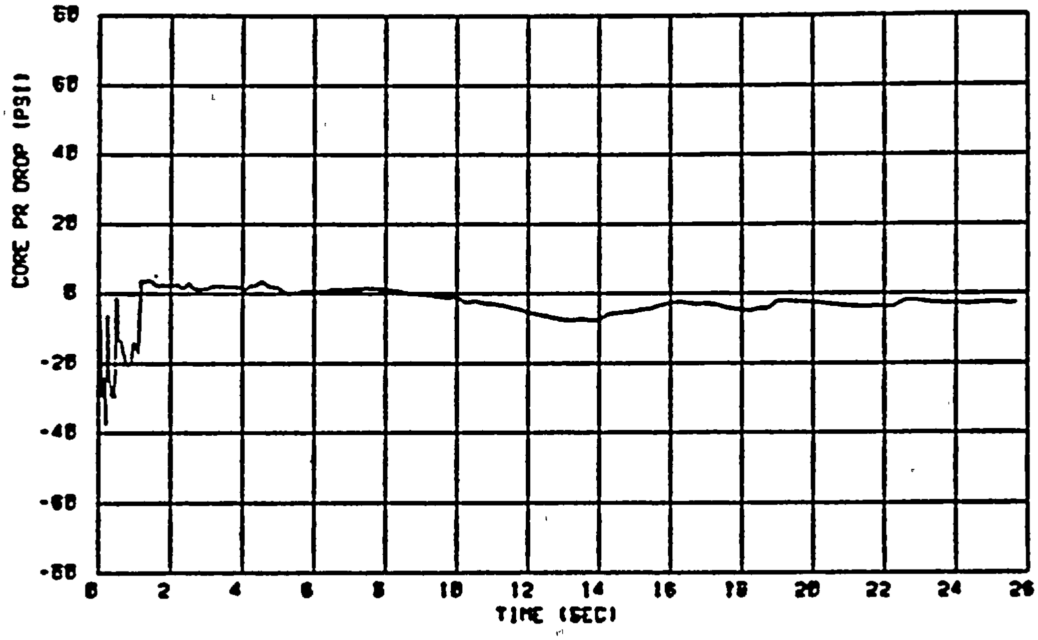


Figure 15.4-23 Core Pressure Drop - DECLG (C_D=0.8), Unit 2

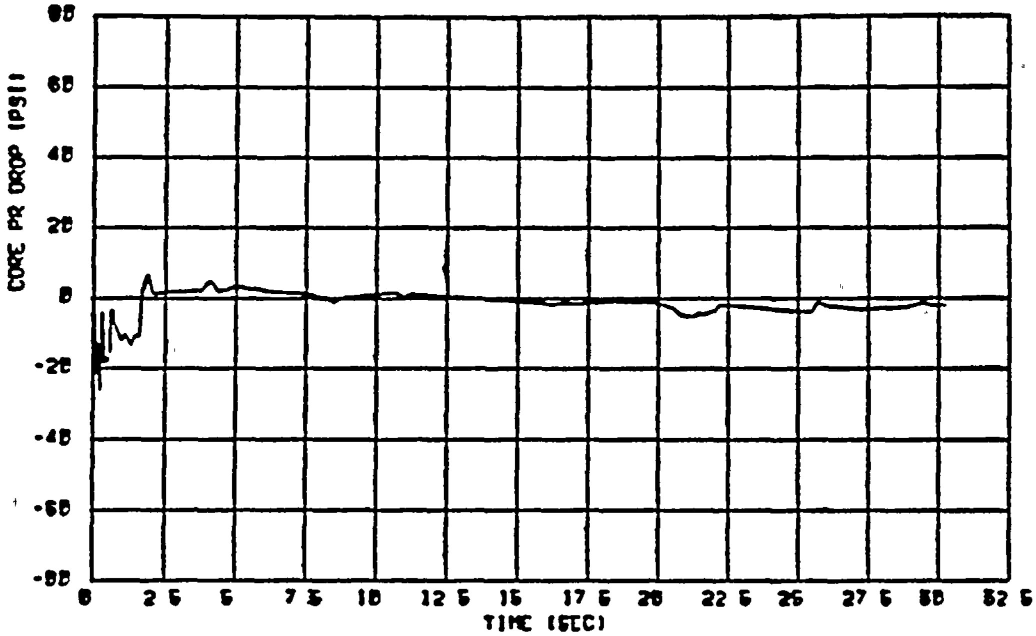


Figure 15.4-24 Core Pressure Drop - DECLG (C_D=0.6), Unit 2

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

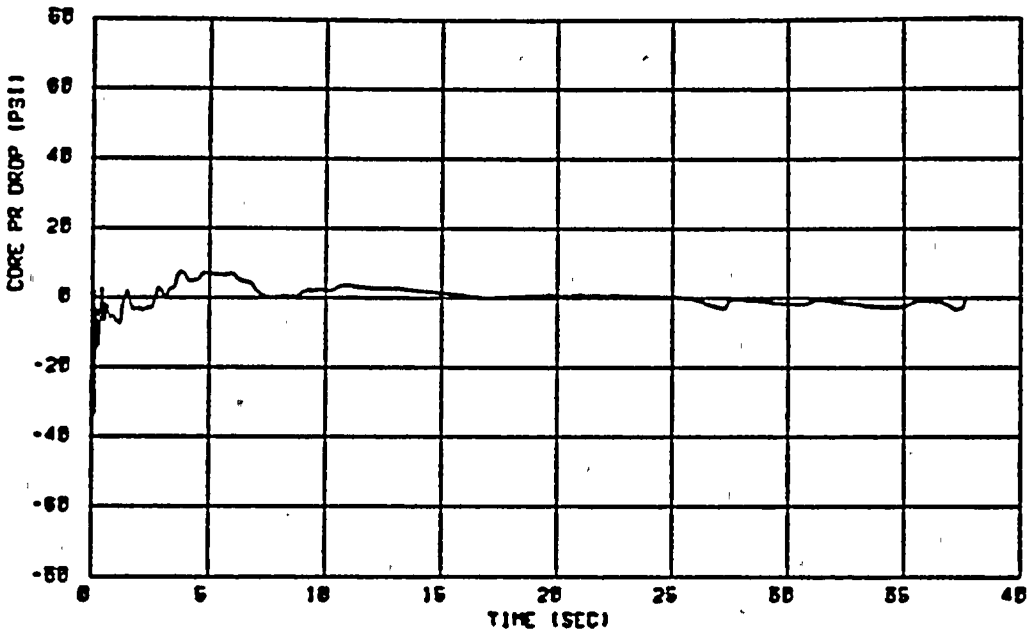


Figure 15.4-25 Core Pressure Drop - DECLG (C_D=0.4), Unit 2

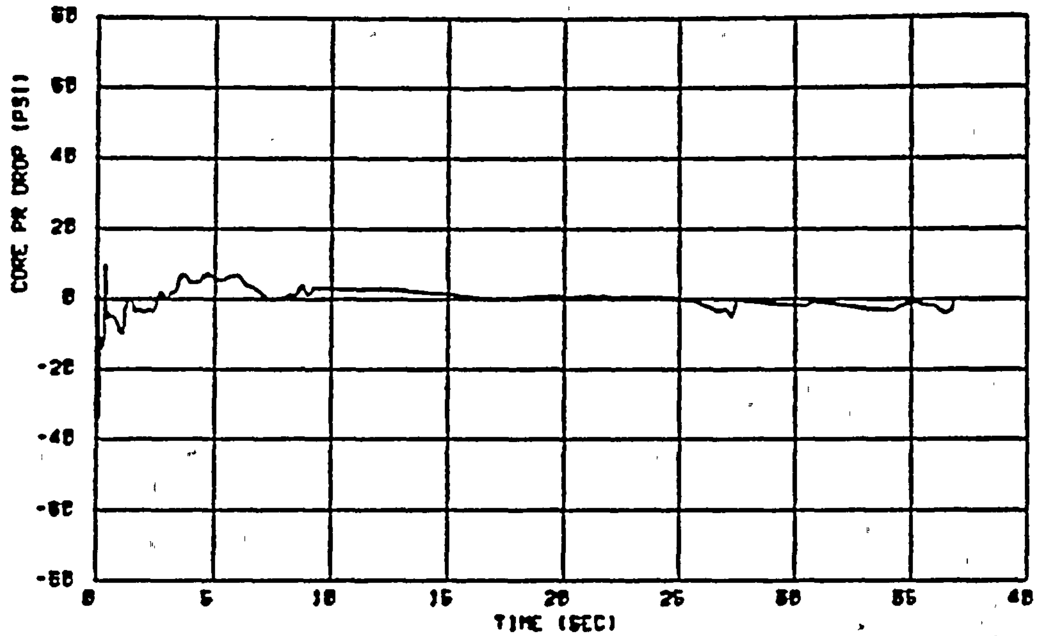


Figure 15.4-25A Core Pressure Drop - DECLG (C_D=0.4), Unit 2
Maximum Safety Injection

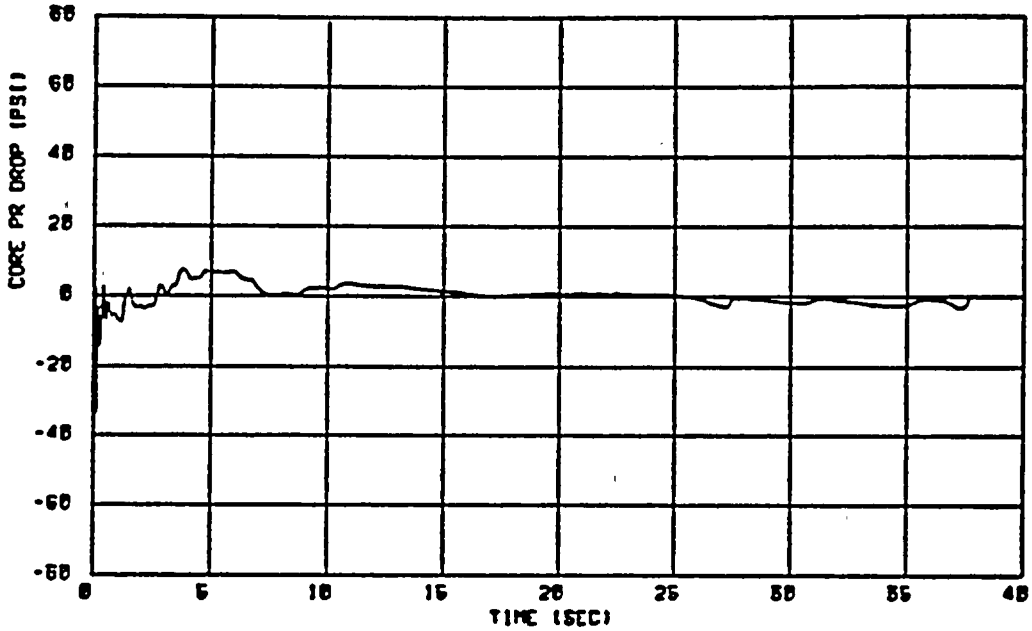


Figure 15.4-26 Core Pressure Drop - DECLG (C_D=0.4), Unit 1

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

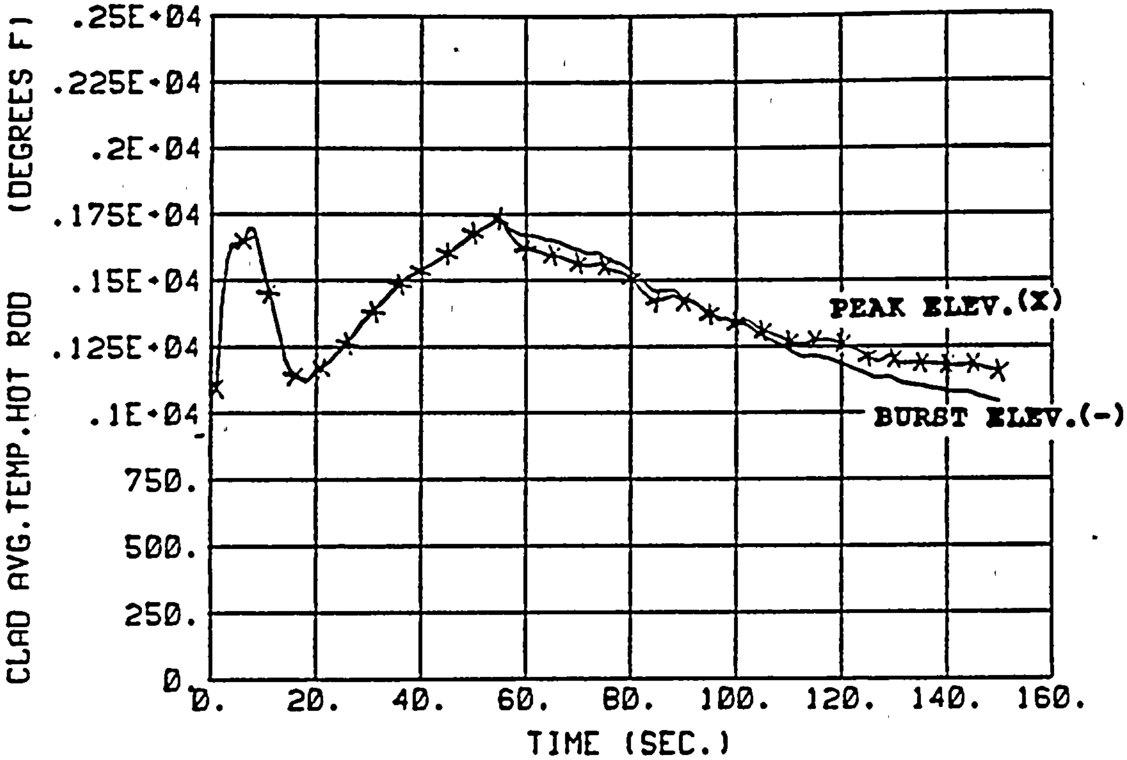


Figure 15.4-27 Peak Clad Temperature - DECLG (C_D=0.8), Unit 2

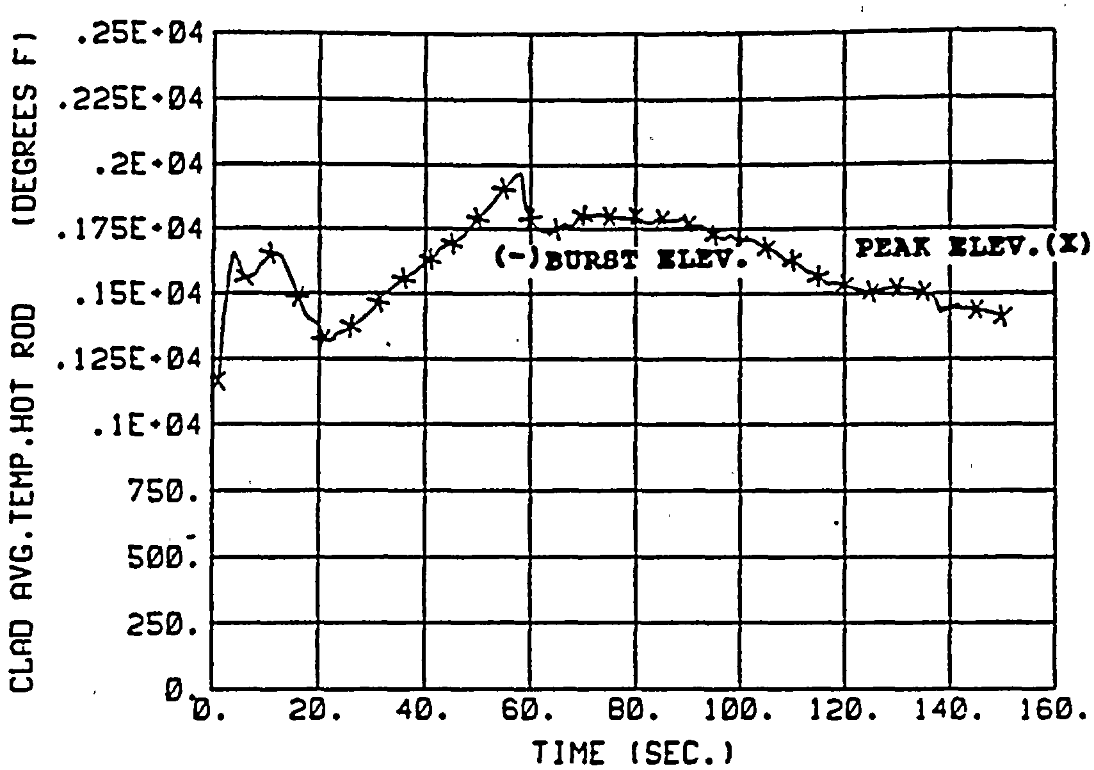


Figure 15.4-28 Peak Clad Temperature - DECLG (C_D=0.6), Unit 2

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

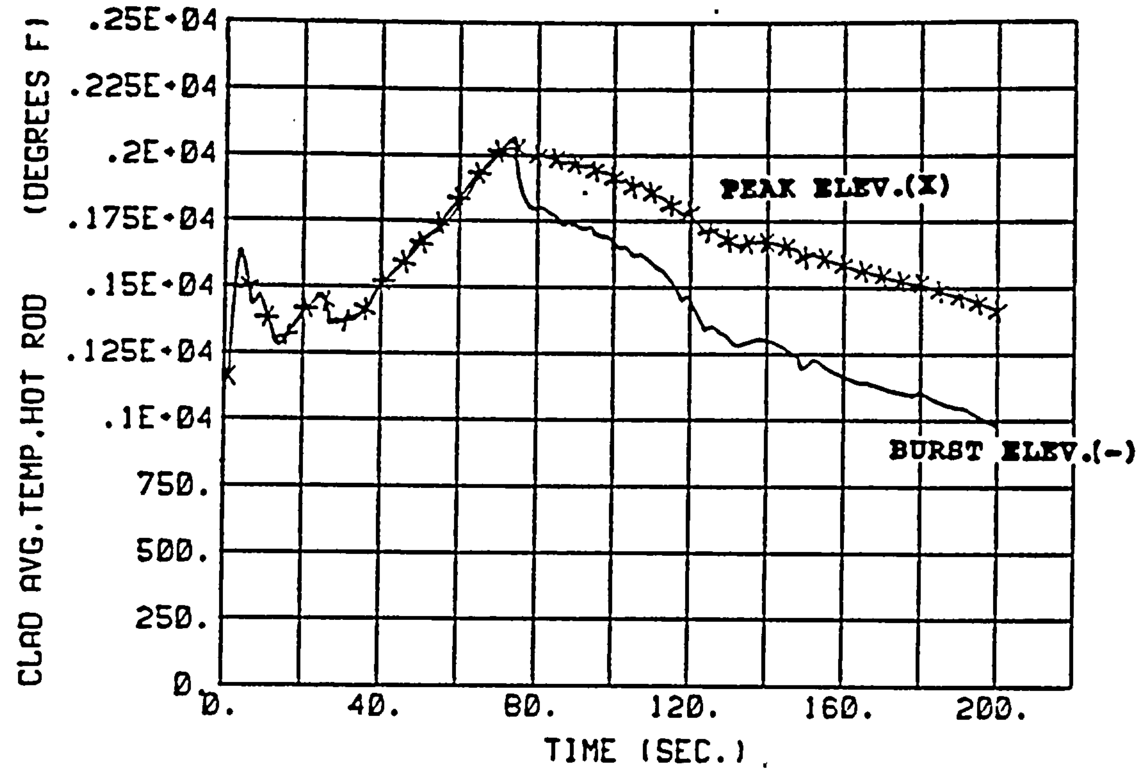


Figure 15.4-29 Peak Clad Temperature - DECLG (C_D=0.4), Unit 2

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

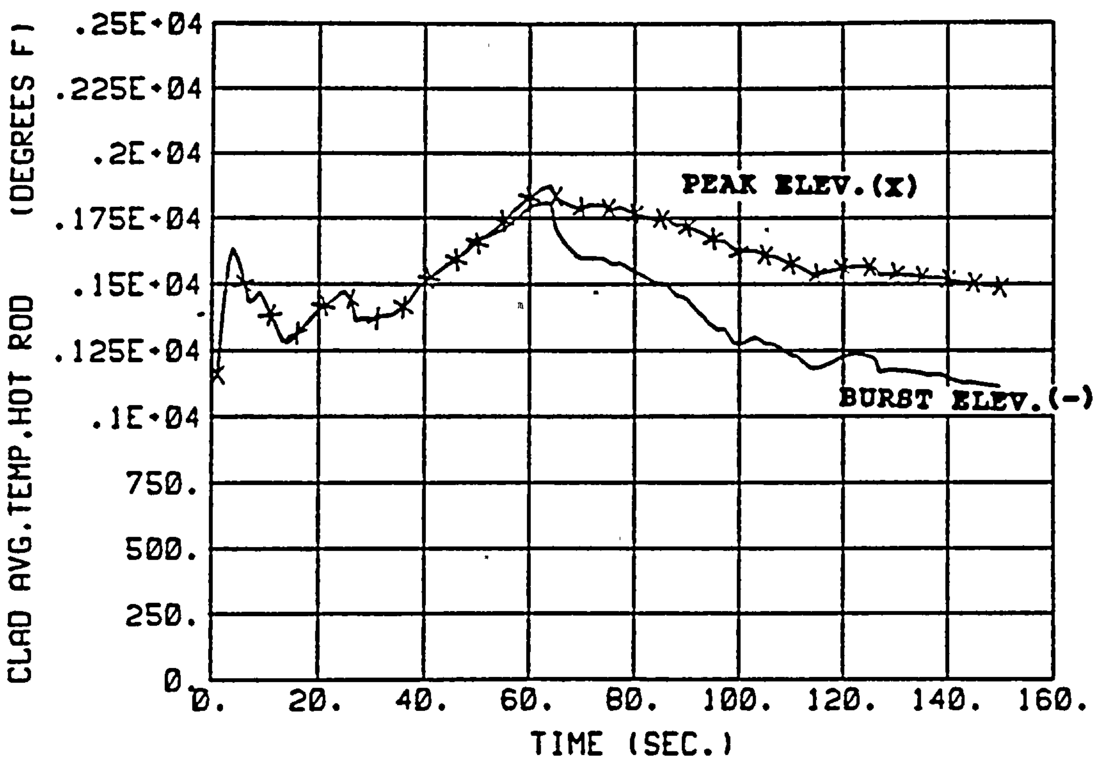


Figure 15.4-29A Peak Clad Temperature - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

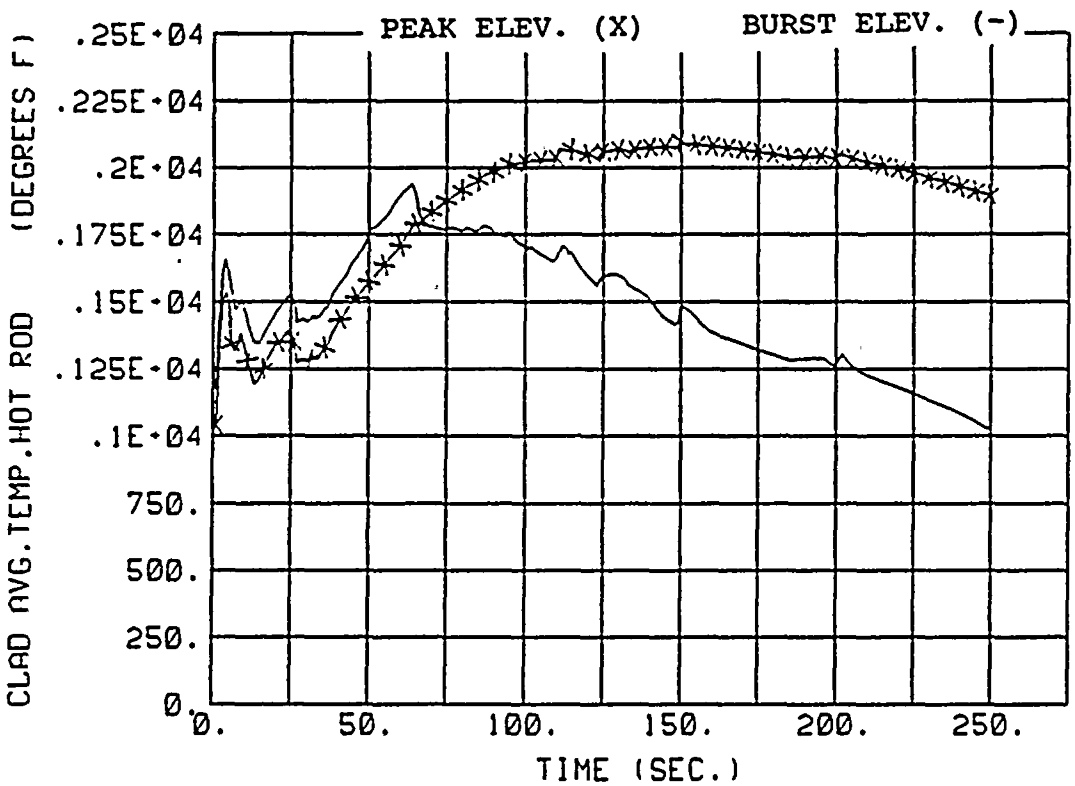


Figure 15.4-29B Peak Clad Temperature - DECLG (C_D=0.4), Unit 2 Grid Deformation

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

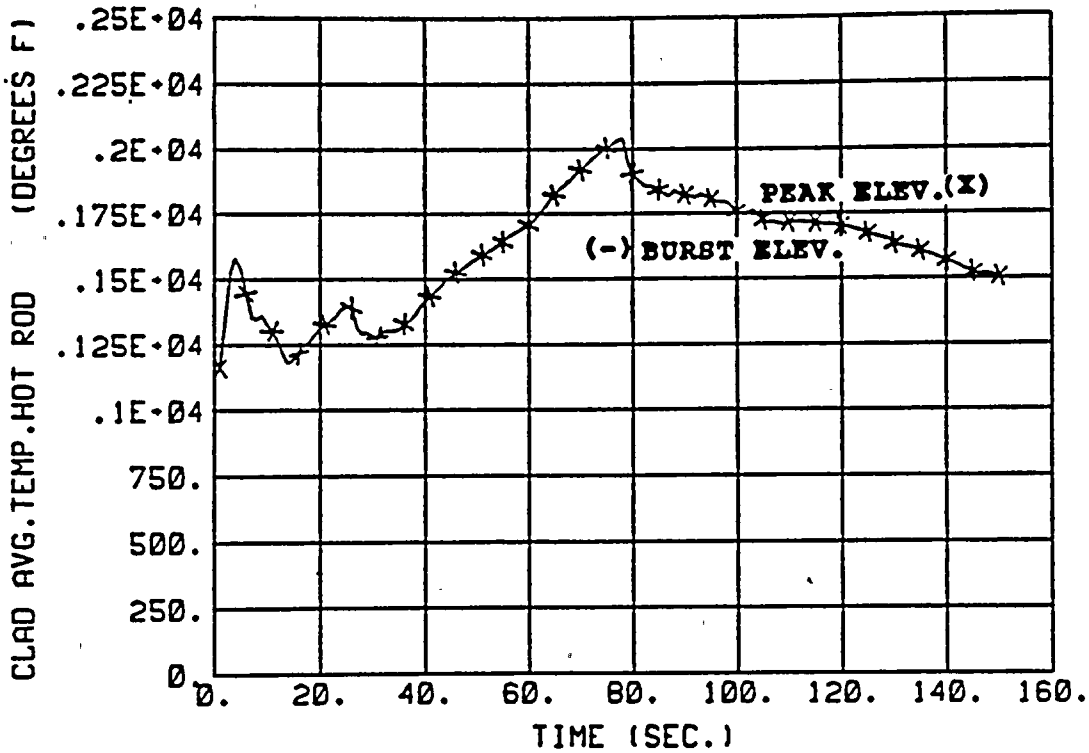


Figure 15.4-30 Peak Clad Temperature - DECLG (C_D=0.4), Unit 1

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

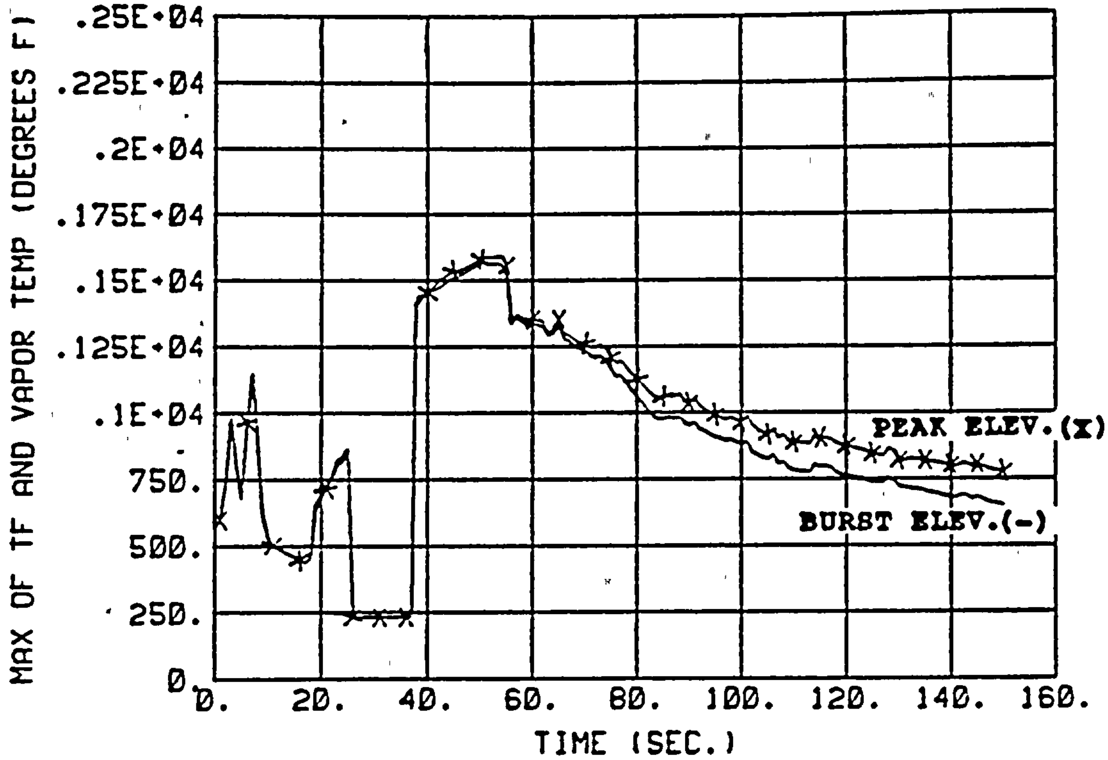


Figure 15.4-31 Fluid Temperature - DECLG (C_D=0.8), Unit 2

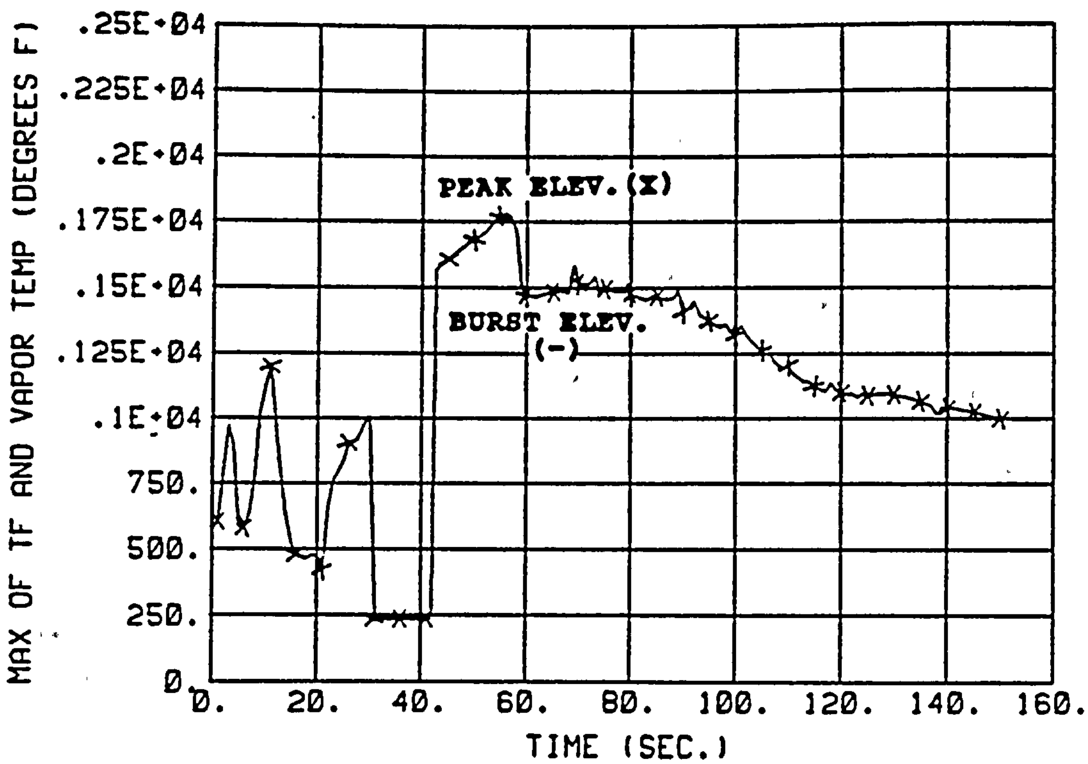


Figure 15.4-32 Fluid Temperature - DECLG (C_D=0.6), Unit 2

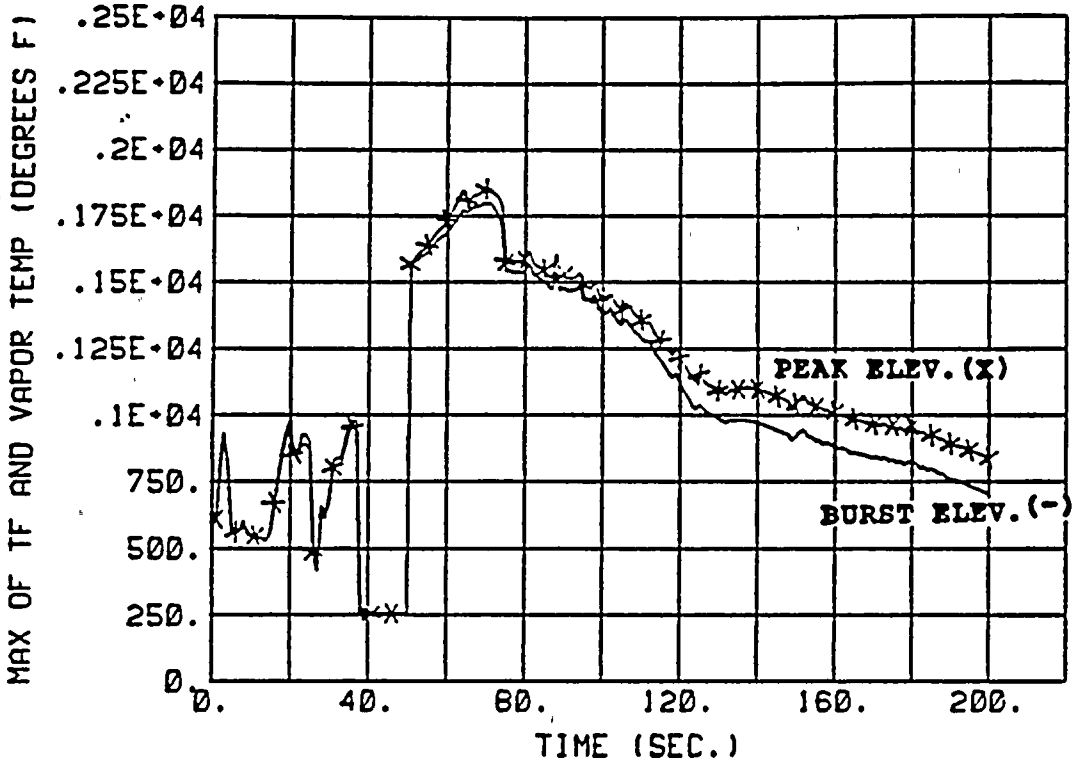


Figure 15.4-33 Fluid Temperature - DECLG (C_D=0.4), Unit 2

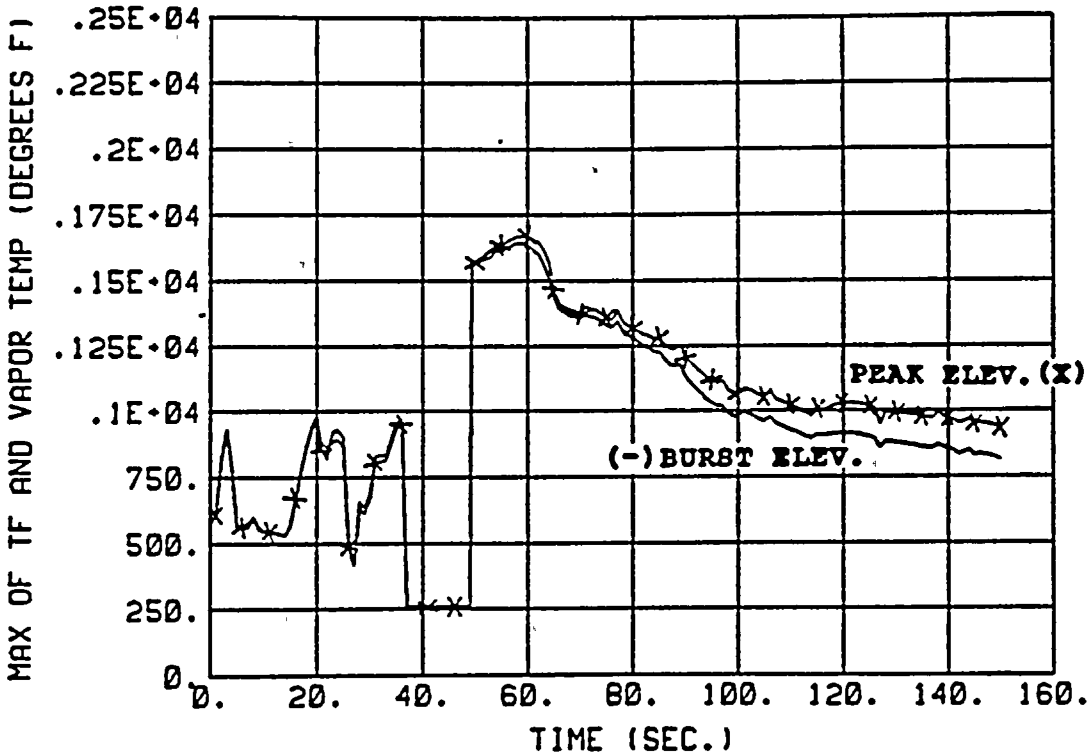


Figure 15.4-33A Fluid Temperature - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

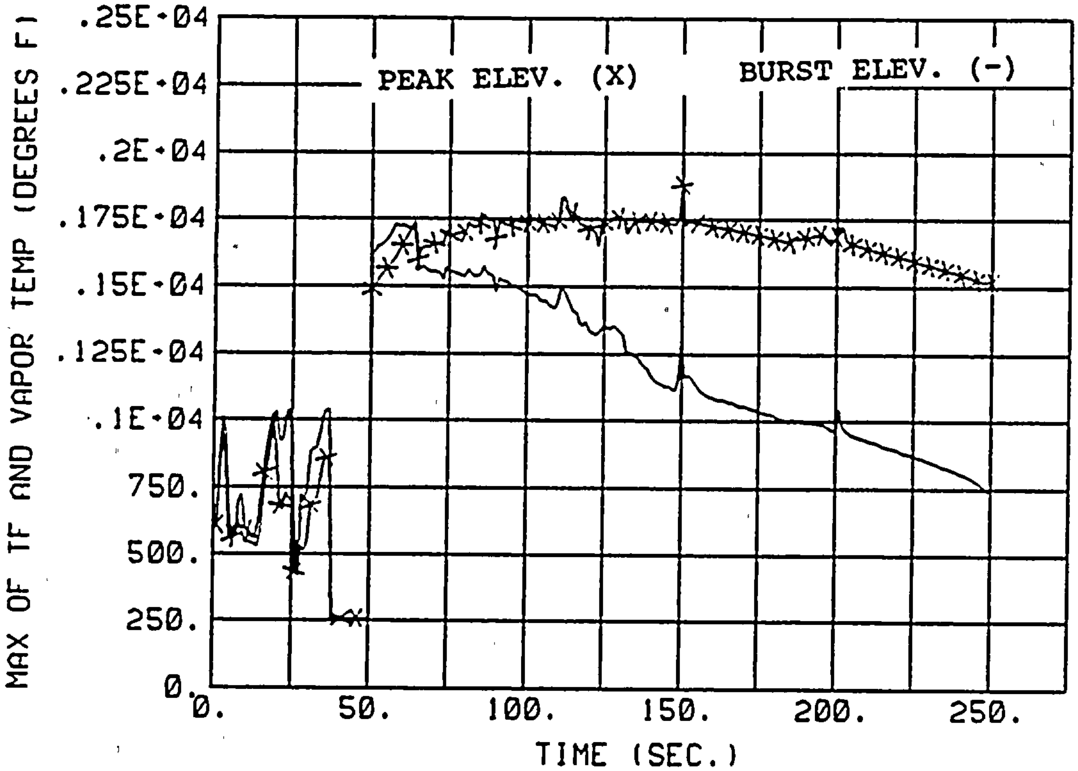


Figure 15.4-33B Fluid Temperature - DECLG (C_D=0.4), Unit 2 Grid Deformation

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

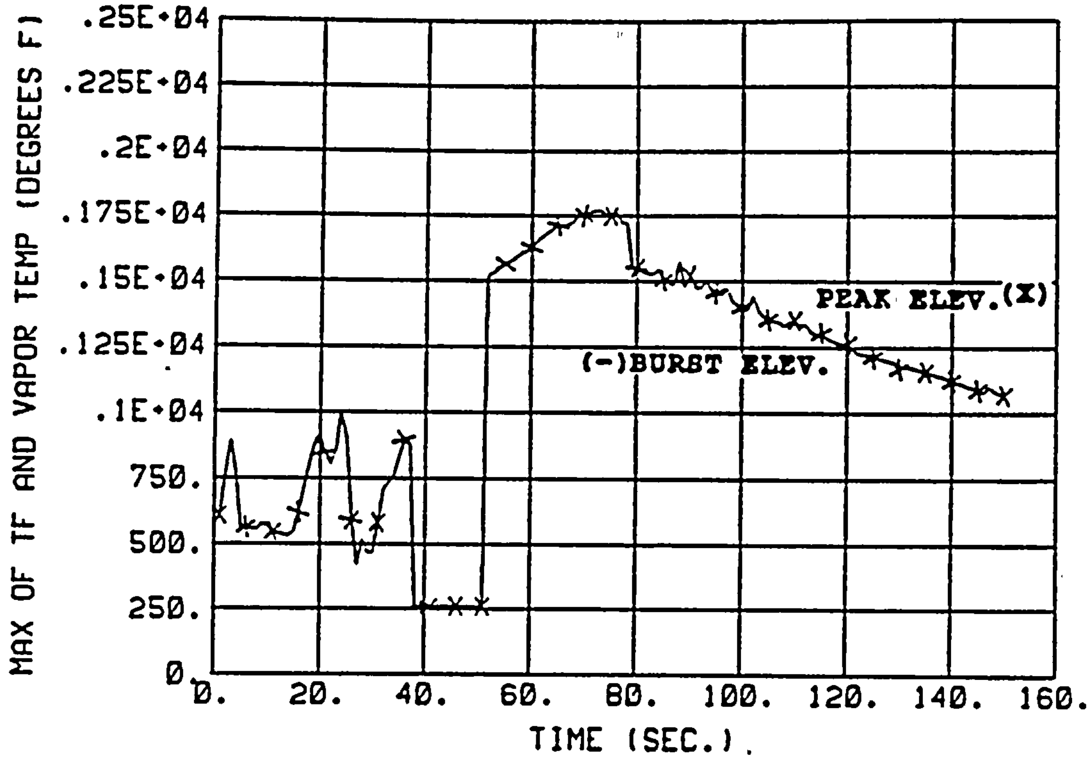


Figure 15.4-34 Fluid Temperature - DECLG (C_D=0.4), Unit 1

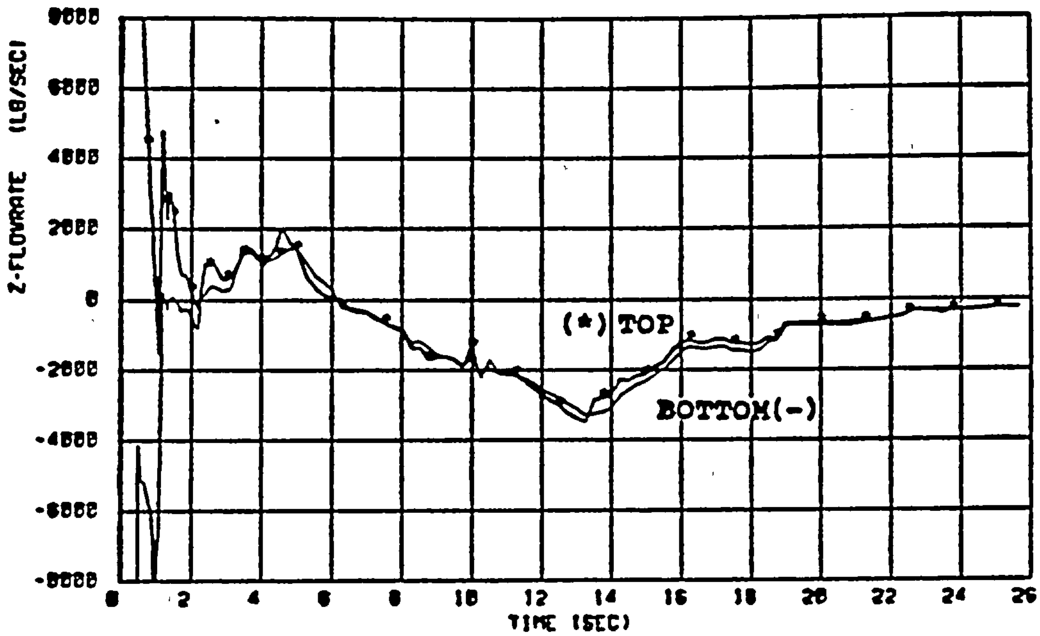


Figure 15.4-35 Core Flow - Top and Botton - DECLG (C_D=0.8), Unit 2

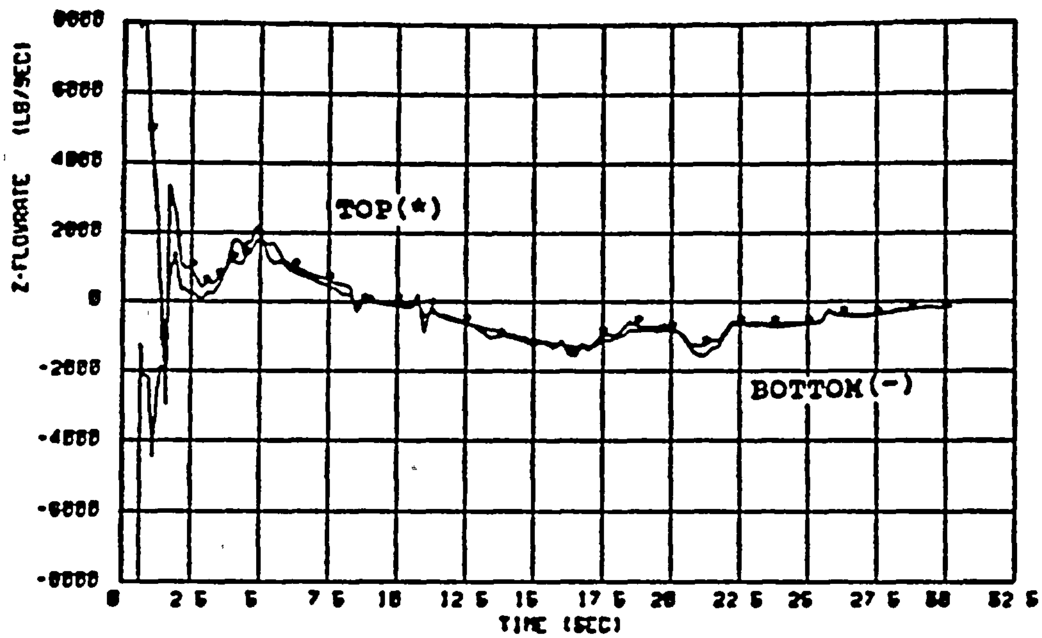


Figure 15.4-36 Core Flow - Top and Bottom - DECLG (C_D=0.6), Unit 2

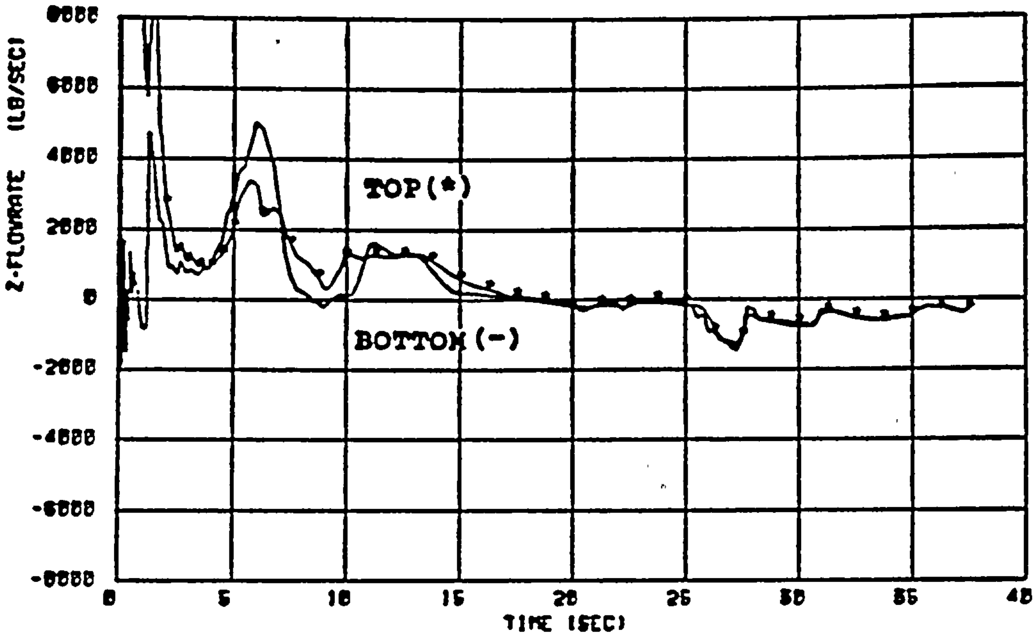


Figure 15.4-37 Core Flow - Top and Bottom - DECLG (C_D=0.4), Unit 2

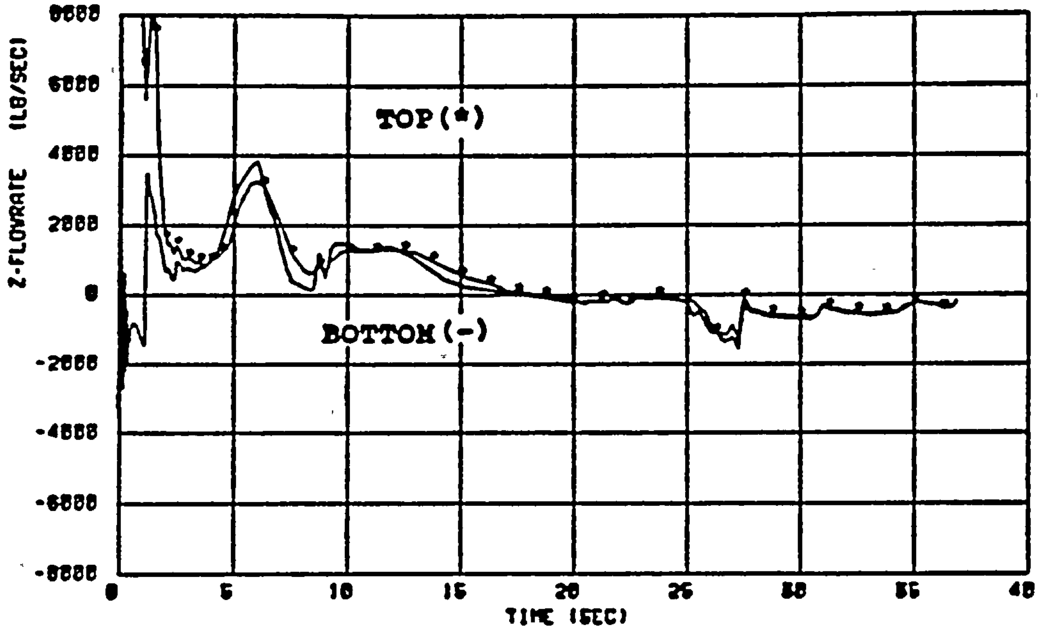


Figure 15.4-37A Core Flow - Top and Bottom - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

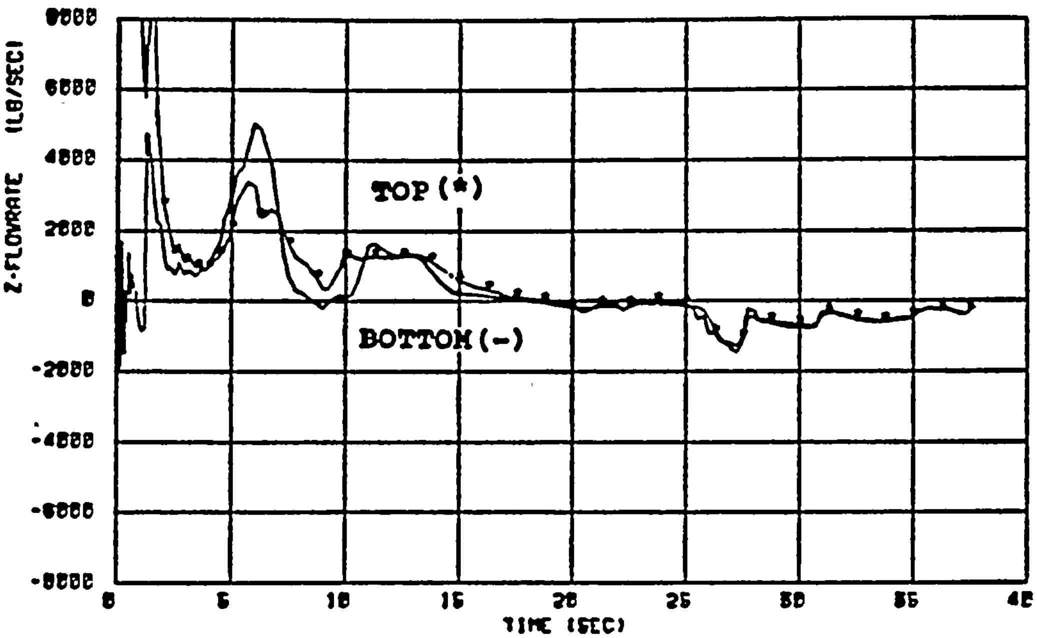


Figure 15.4-38 Core Flow - Top and Bottom - DECLG (C_D=0.4), Unit 1

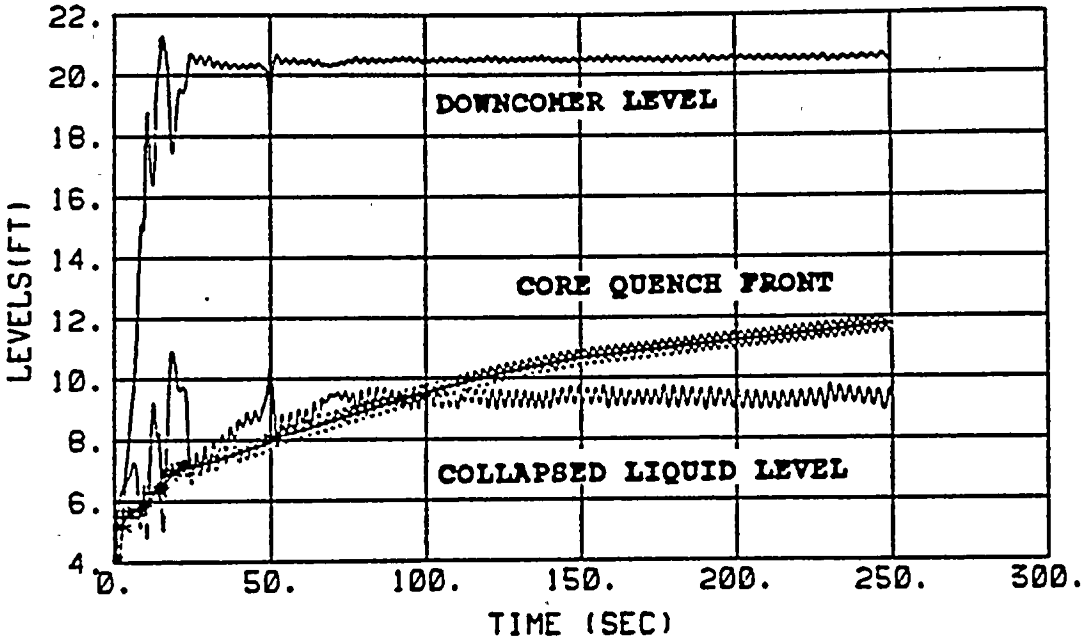


Figure 15.4-39 Reflood Transient - DECLG (C_D=0.8), Unit 2 Downcomer and Core Water Levels

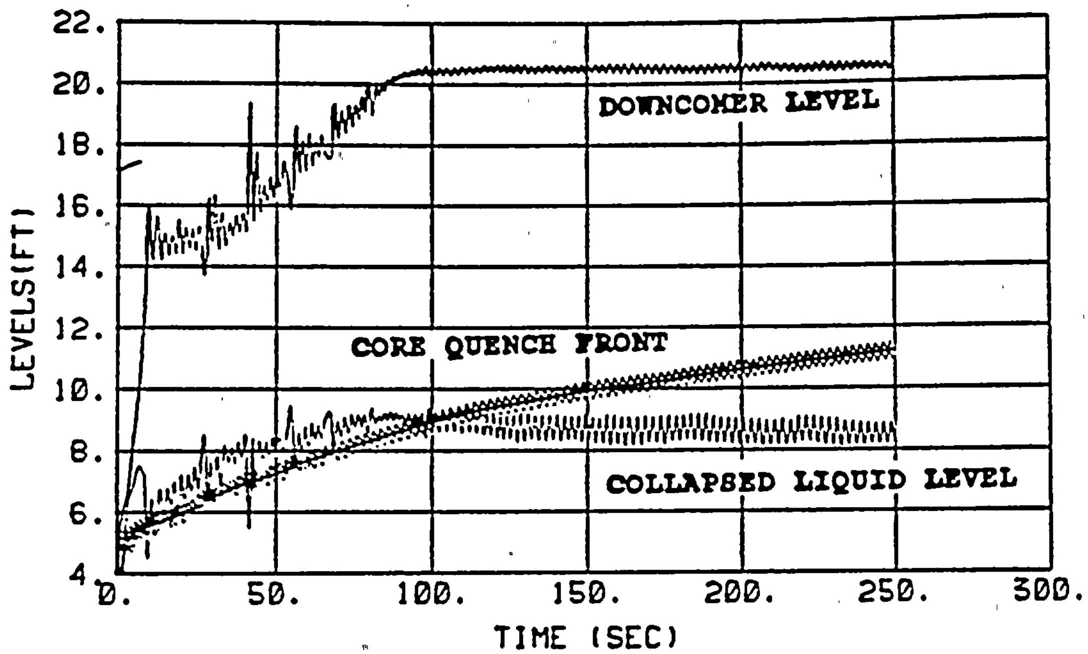


Figure 15.4-40 Reflood Transient - DECLG (C_D=0.6), Unit 2 Downcomer and Core Water Levels

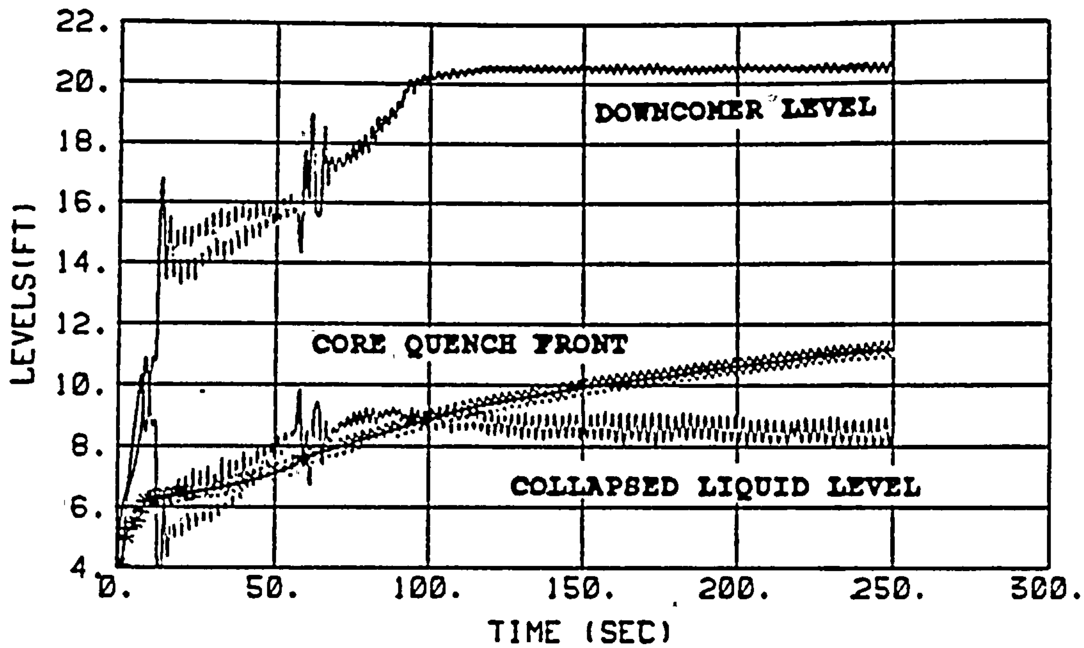


Figure 15.4-41 Reflood Transient - DECLG (C_D=0.4), Unit 2 Downcomer and Core Water Levels

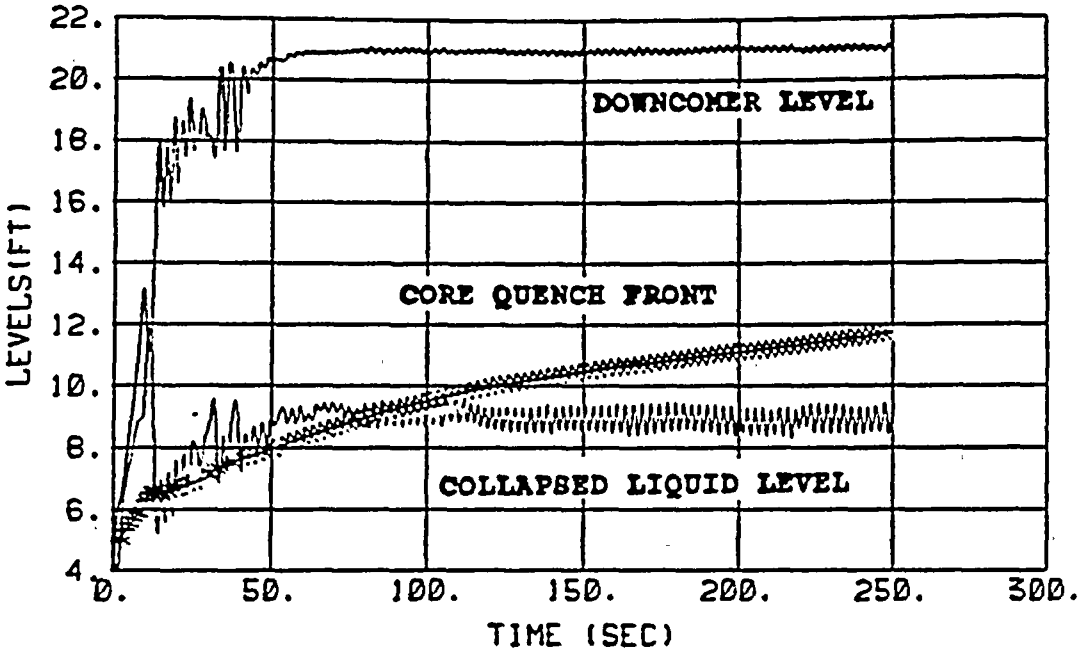


Figure 15.4-41A Reflood Transient - DECLG (C_D=0.4), Unit 2
Downcomer and Core Water Levels
Maximum Safety Injection

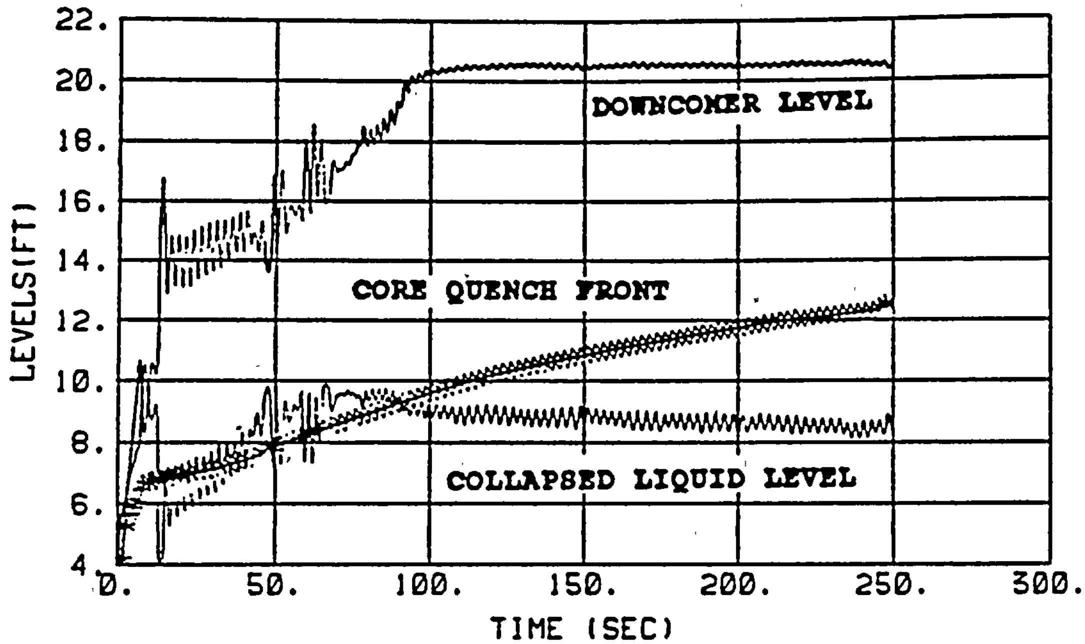


Figure 15.4-42 Reflood Transient - DECLG (C_D=0.4), Unit 1 Downcomer and Core Water Levels

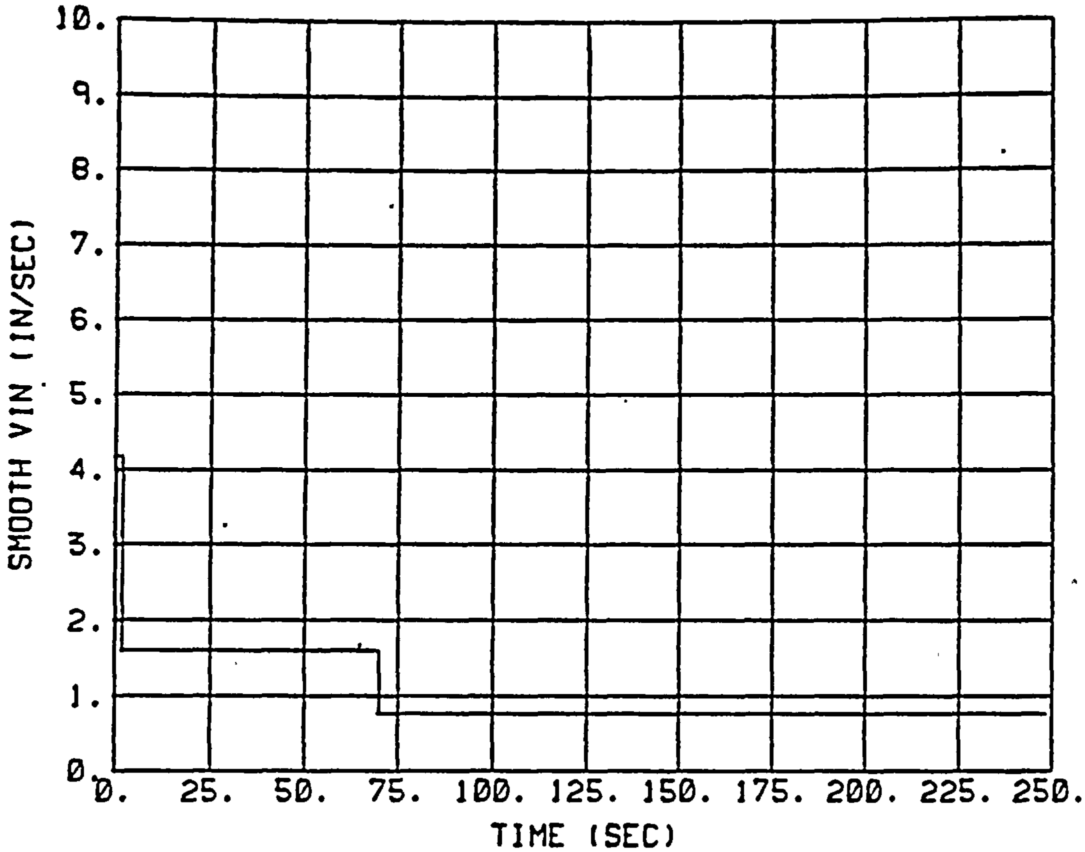


Figure 15.4-43 Reflood Transient - DECLG (C_D=0.8), Unit 2
Core Inlet Velocity

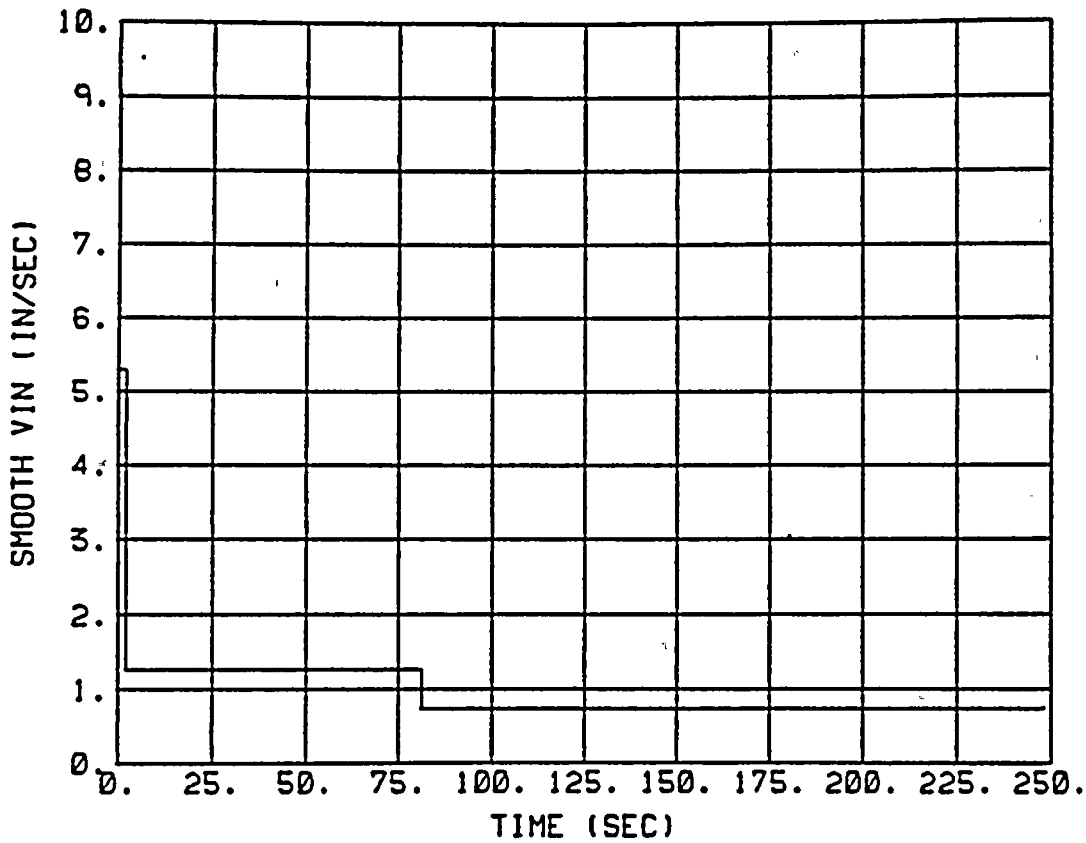


Figure 15.4-44 Reflood Transient - DECLG (C_D=0.6), Unit 2 Core Inlet Velocity

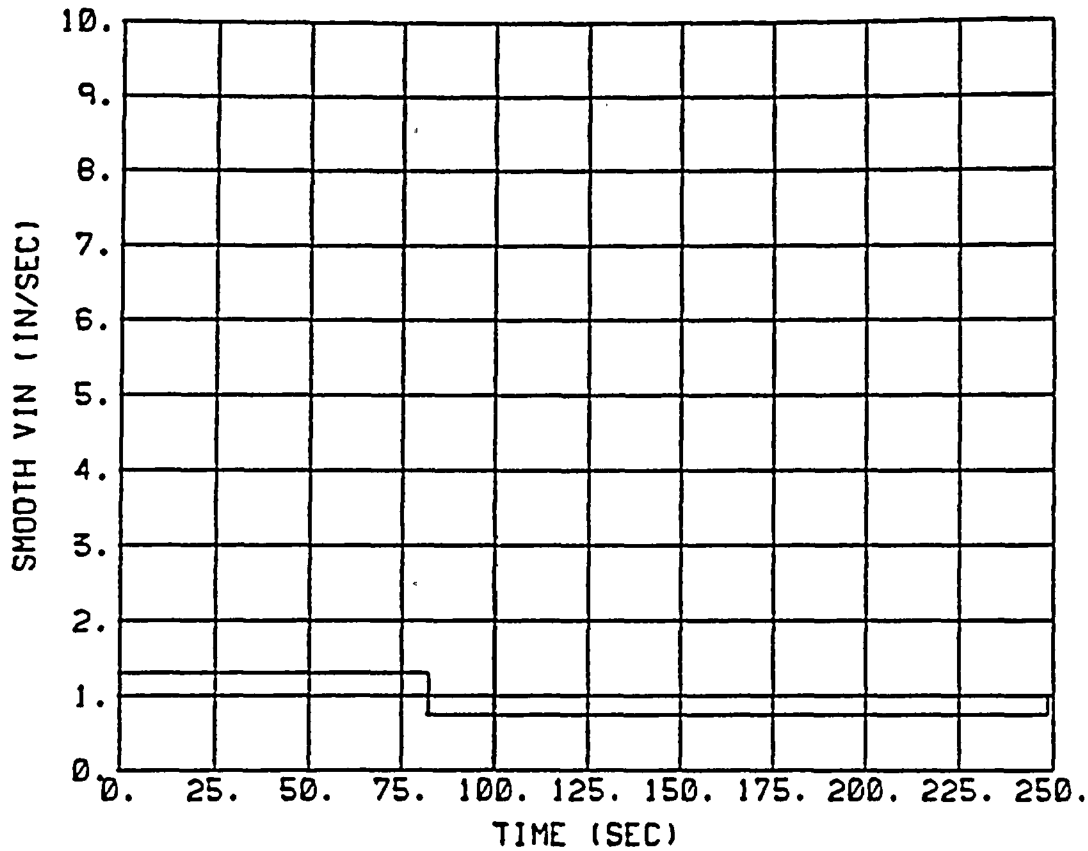


Figure 15.4-45 Reflood Transient - DECLG (C_D=0.4), Unit 2 Core Inlet Velocity

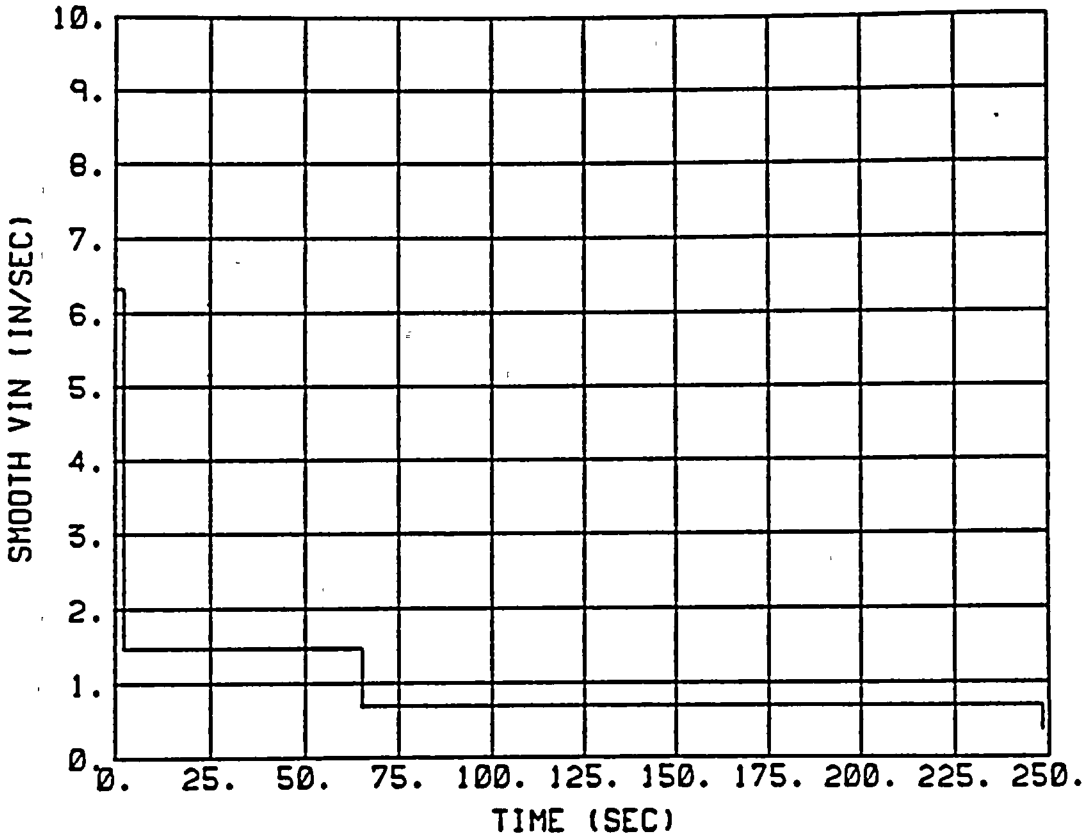


Figure 15.4-45A Reflood Transient - DECLG (C_D=0.4), Unit 2
Core Inlet Velocity
Maximum Safety Injection

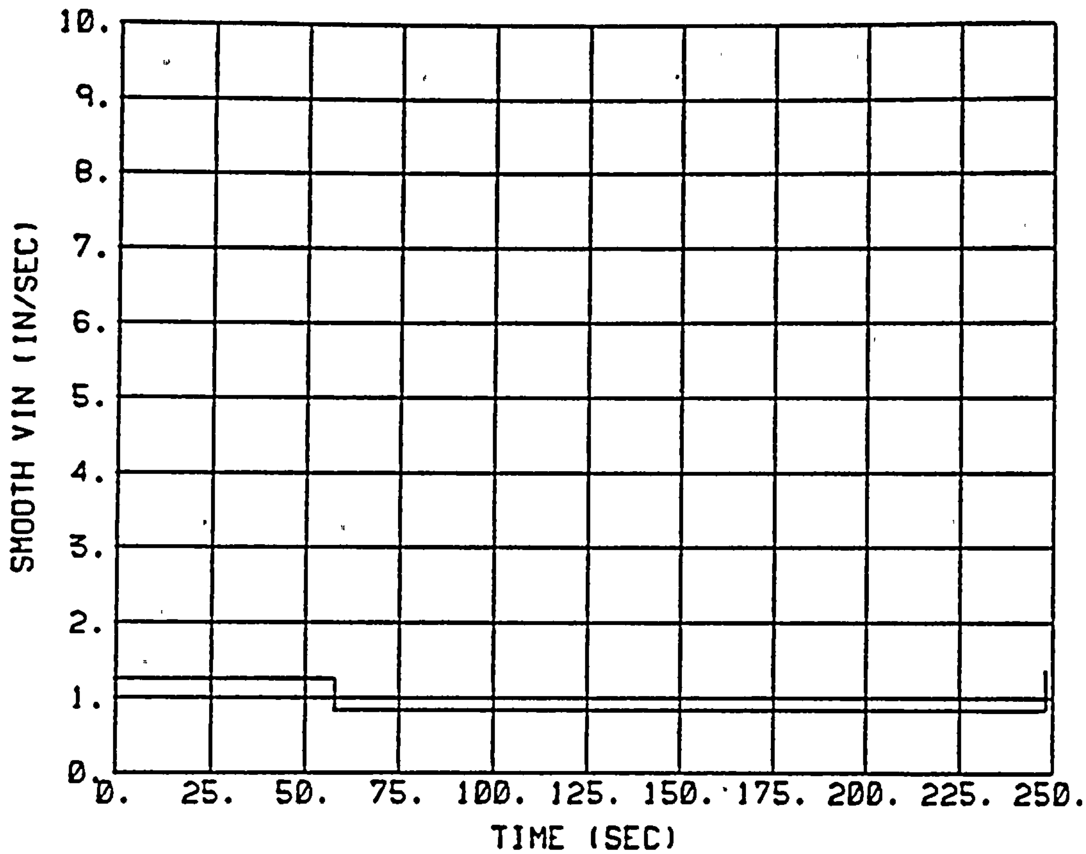


Figure 15.4-46 Reflood Transient - DECLG (C_D=0.4), Unit 1
Core Inlet Velocity

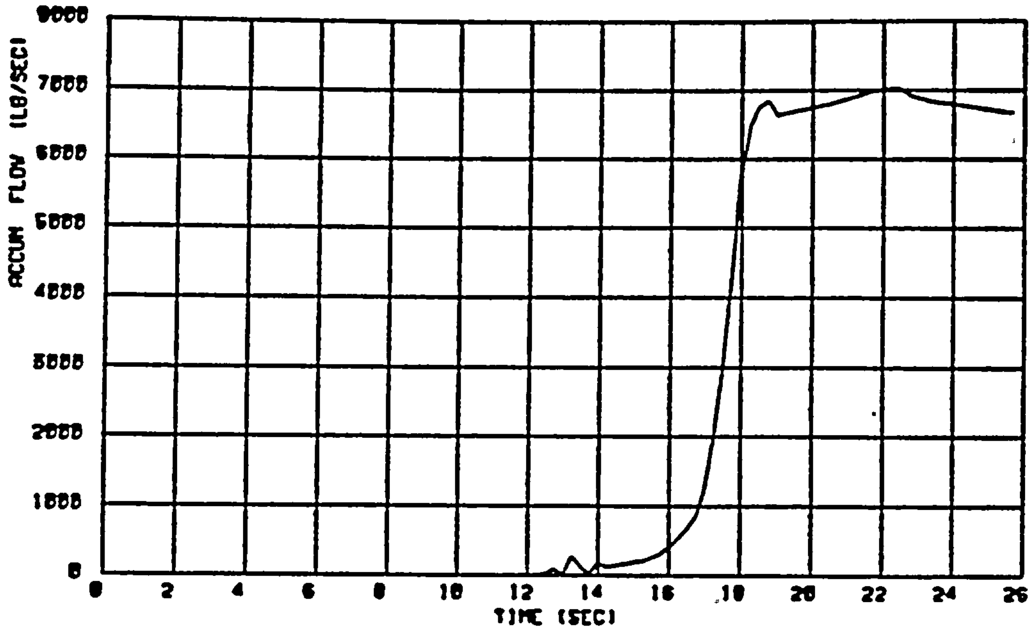


Figure 15.4-47 Accumulator Flow (Blowdown) - DECLG (C_D=0.8), Unit 2

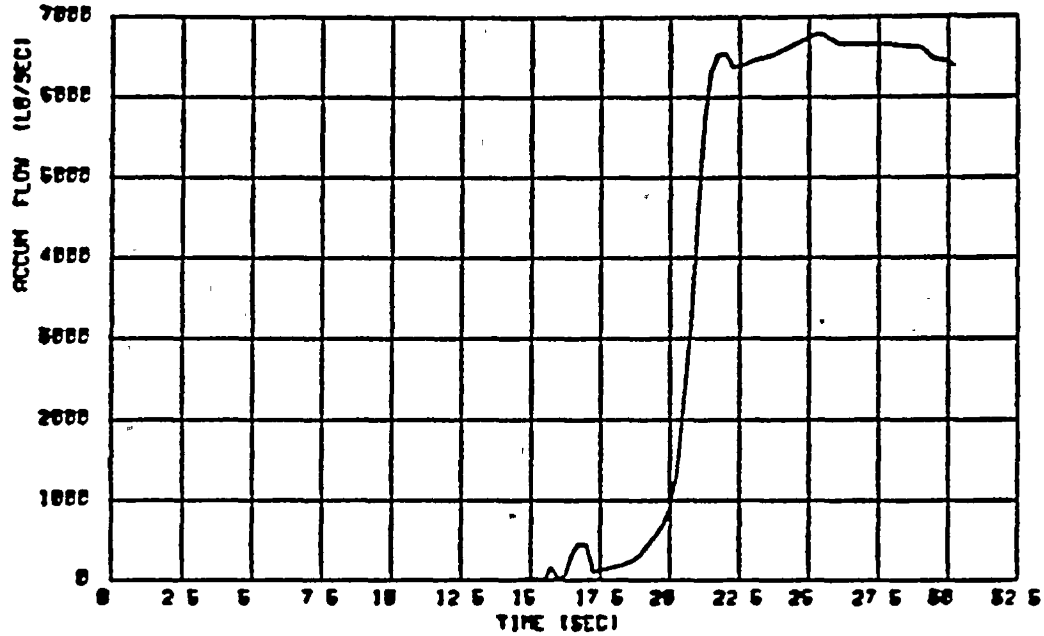


Figure 15.4-48 Accumulator Flow (Blowdown) - DECLG (C_D=0.6), Unit 2

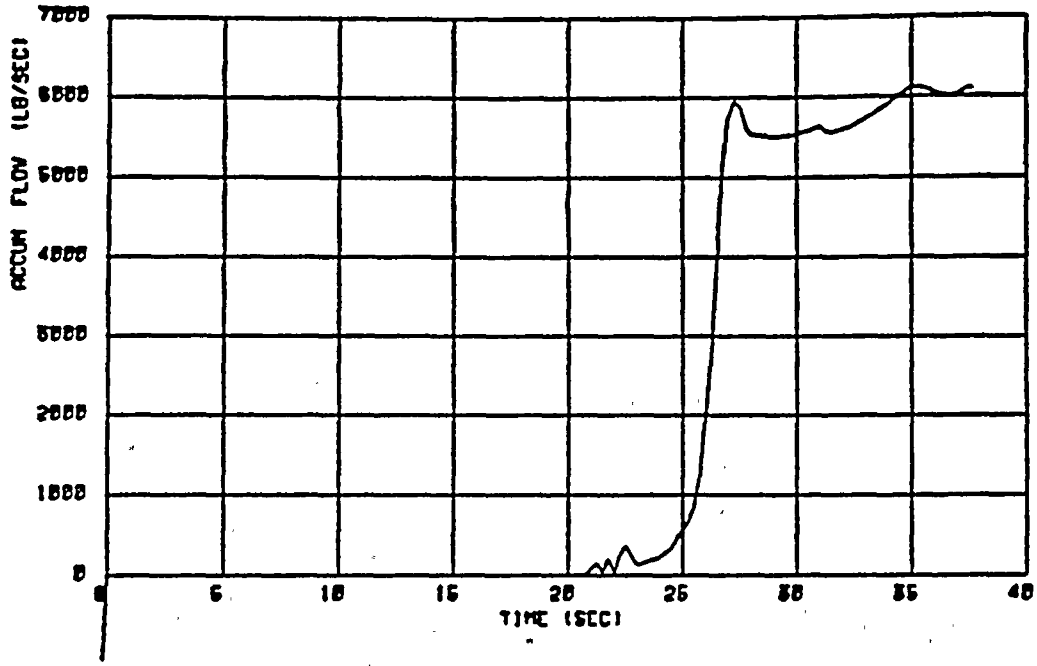


Figure 15.4-49 Accumulator Flow (Blowdown) - DECLG (C_D=0.4), Unit 2

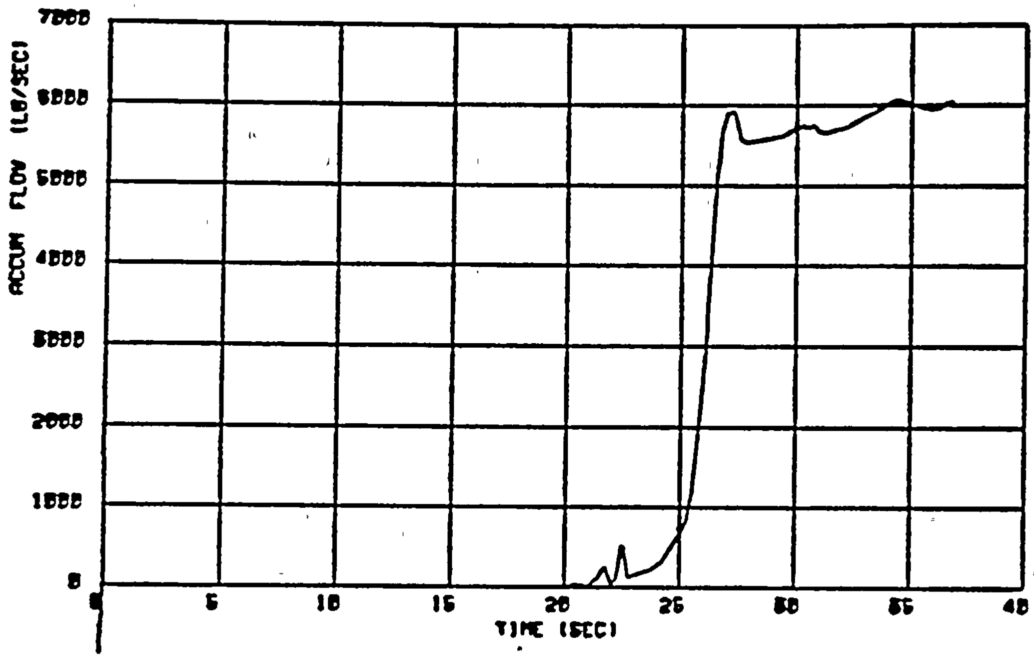


Figure 15.4-49A Accumulator Flow (Blowdown) - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

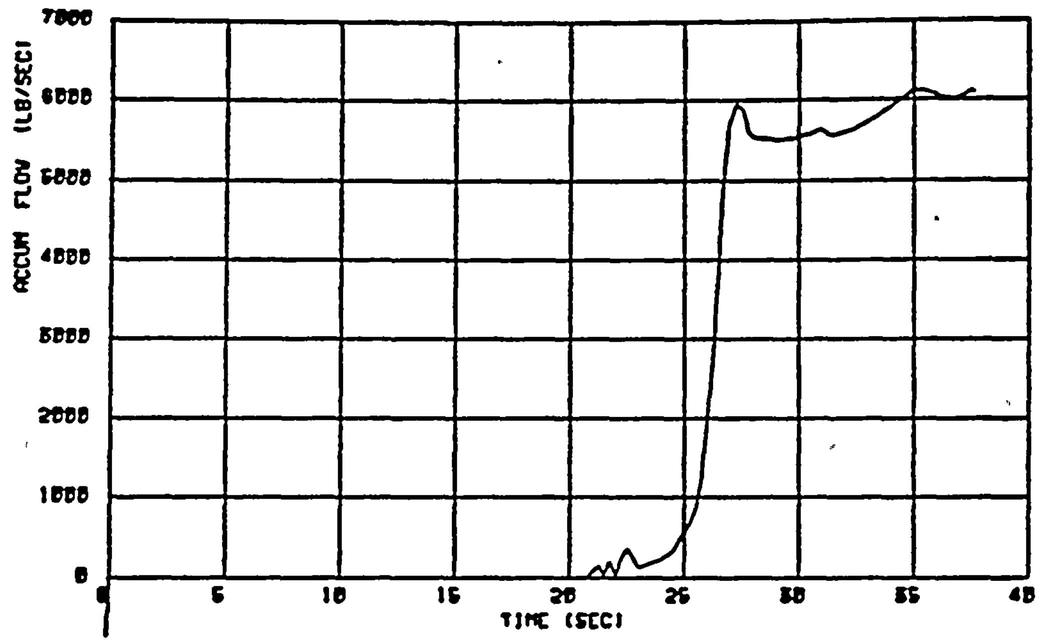


Figure 15.4-50 Accumulator Flow (Blowdown) - DECLG (C_D=0.4), Unit 1

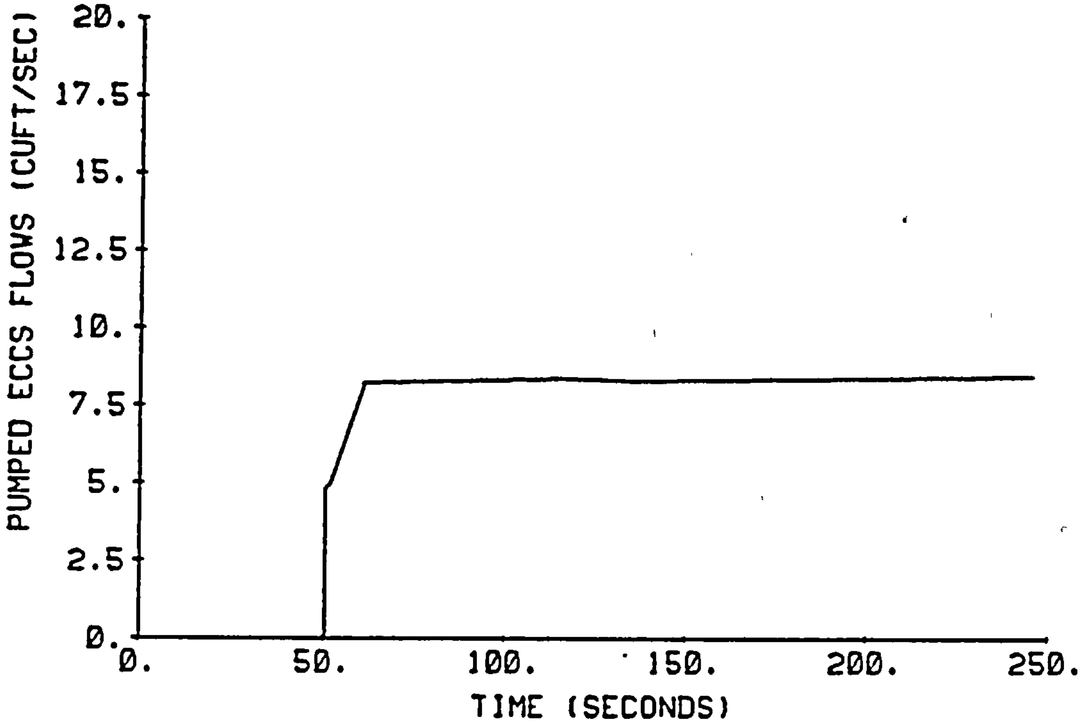


Figure 15.4-51 Pumped ECCS Flow (Reflood) - DECLG (C_D=0.4), Unit 2

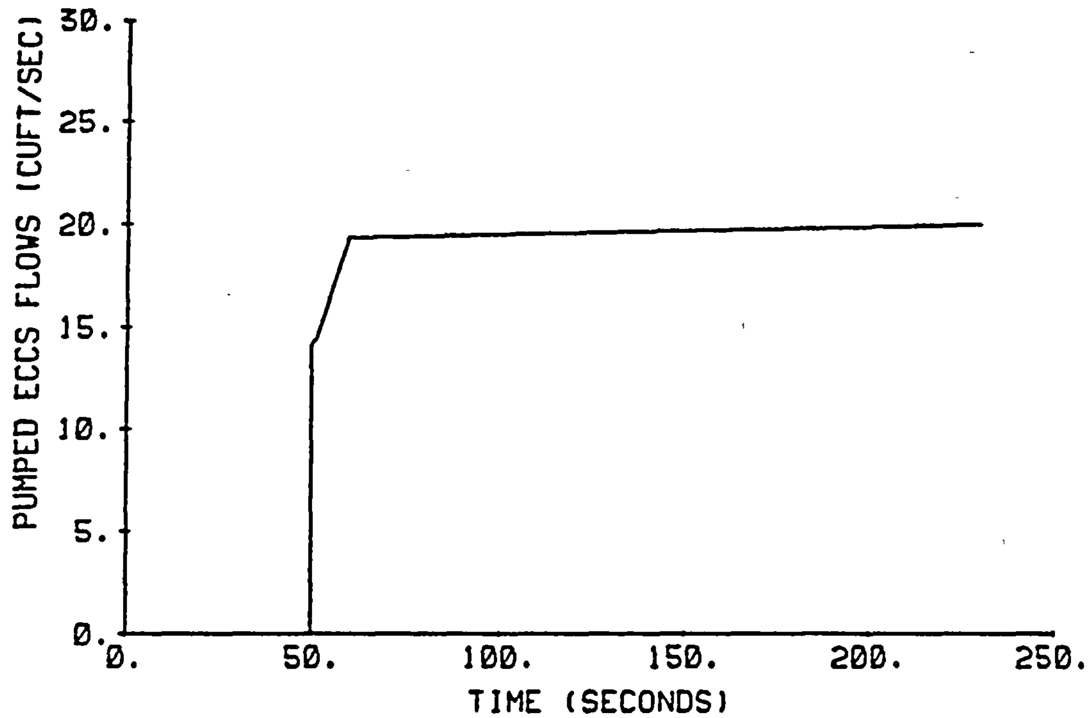


Figure 15.4-51A Pumped ECCS Flow (Reflood) - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

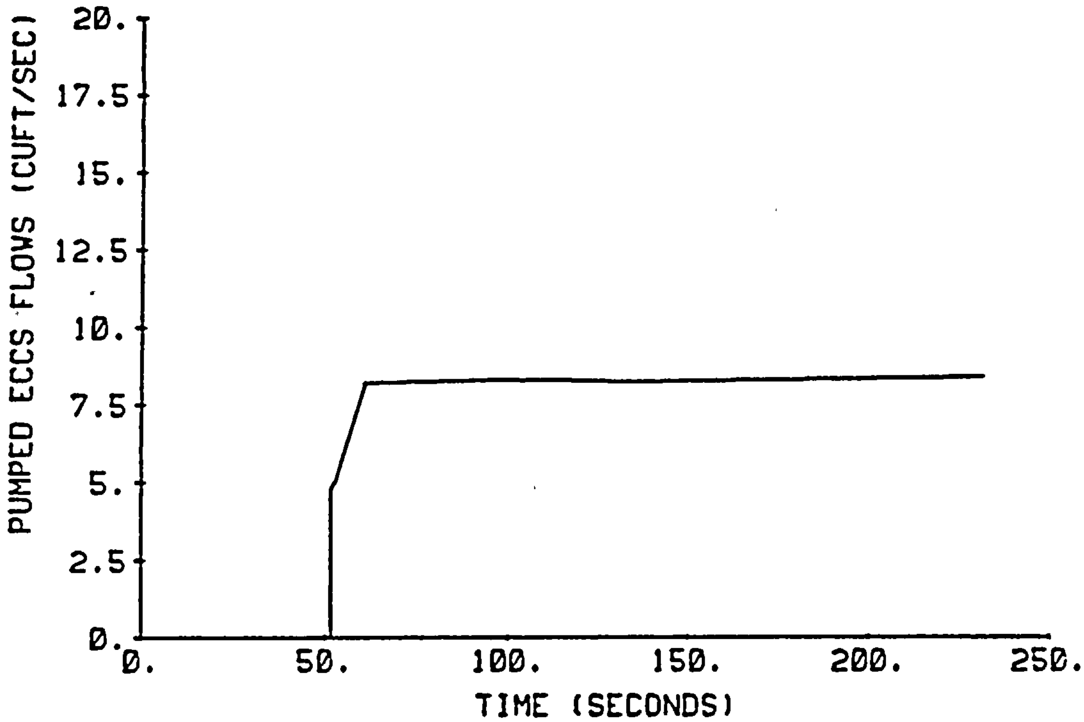


Figure 15.4-52 Pumped ECCS Flow (Reflood) - DECLG (C_D=0.4), Unit 1

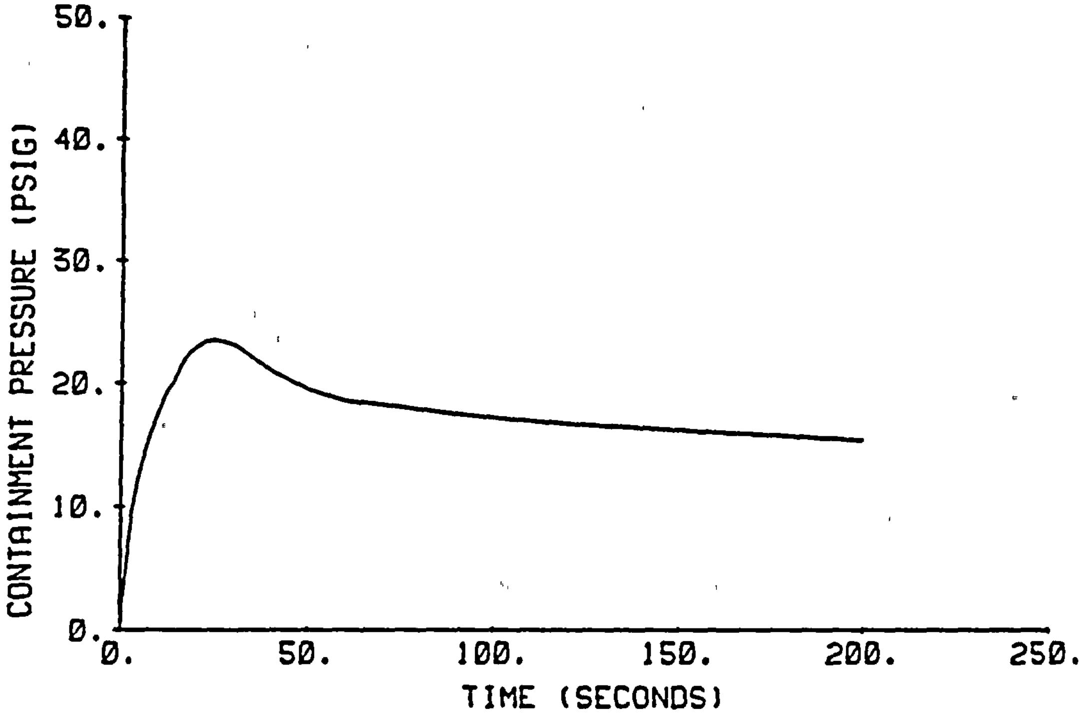


Figure 15.4-53 Containment Pressure - DECLG (C_D=0.4), Unit 2

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

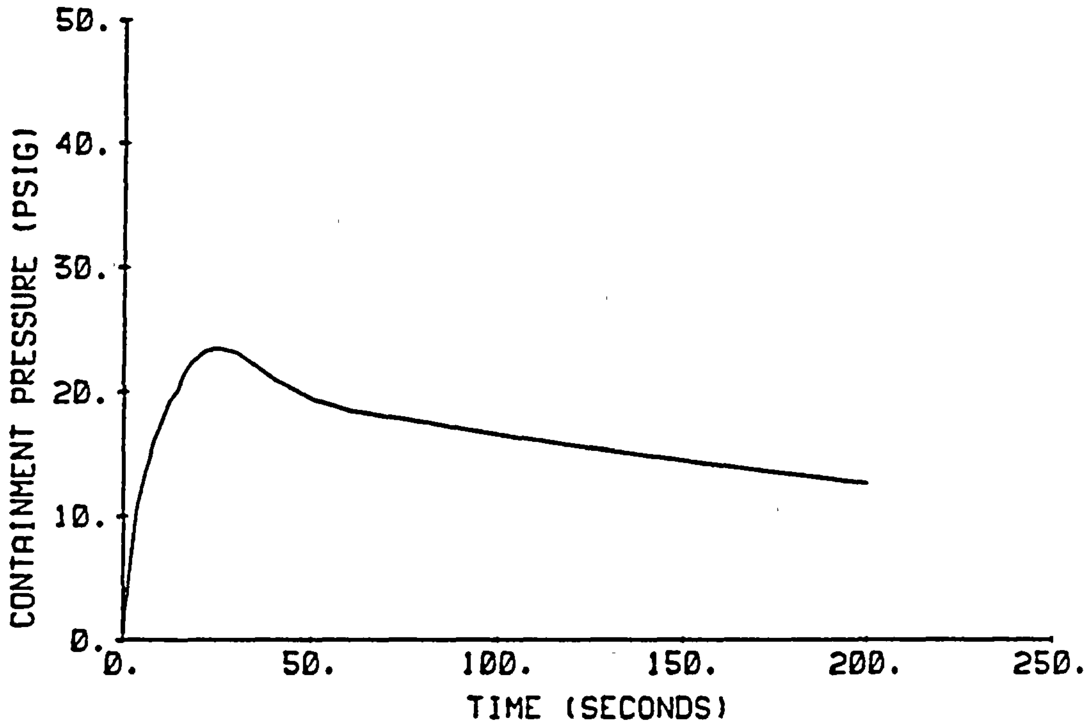


Figure 15.4-53A Containment Pressure - DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

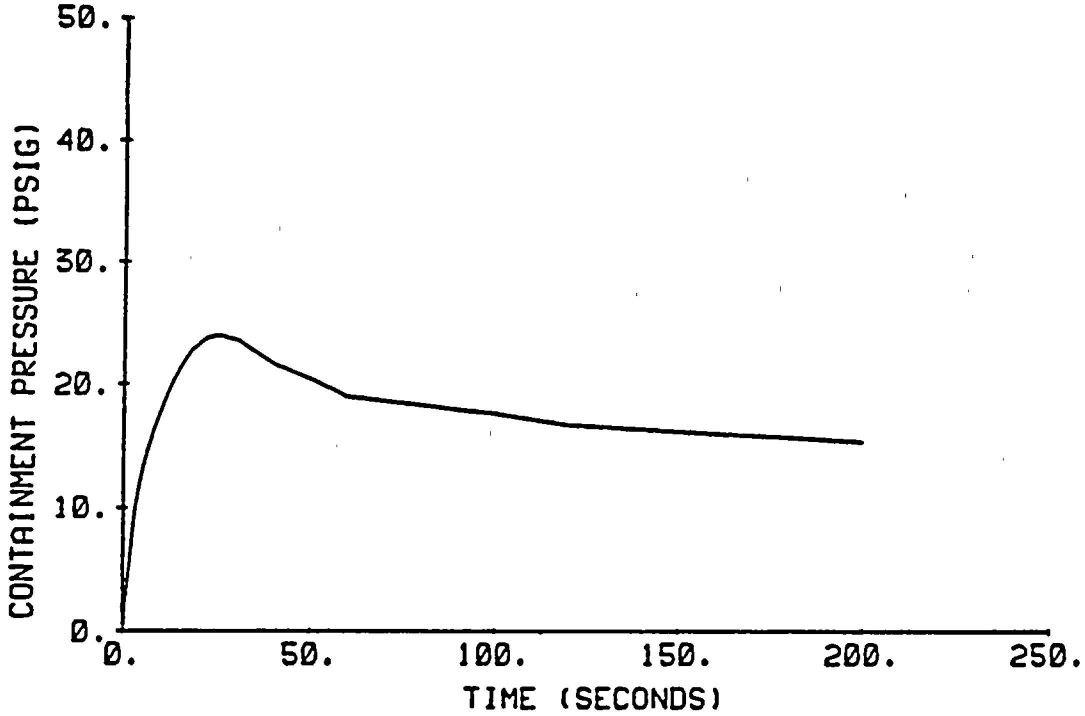


Figure 15.4-54 Containment Pressure - DECLG (C_D=0.4), Unit 1

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

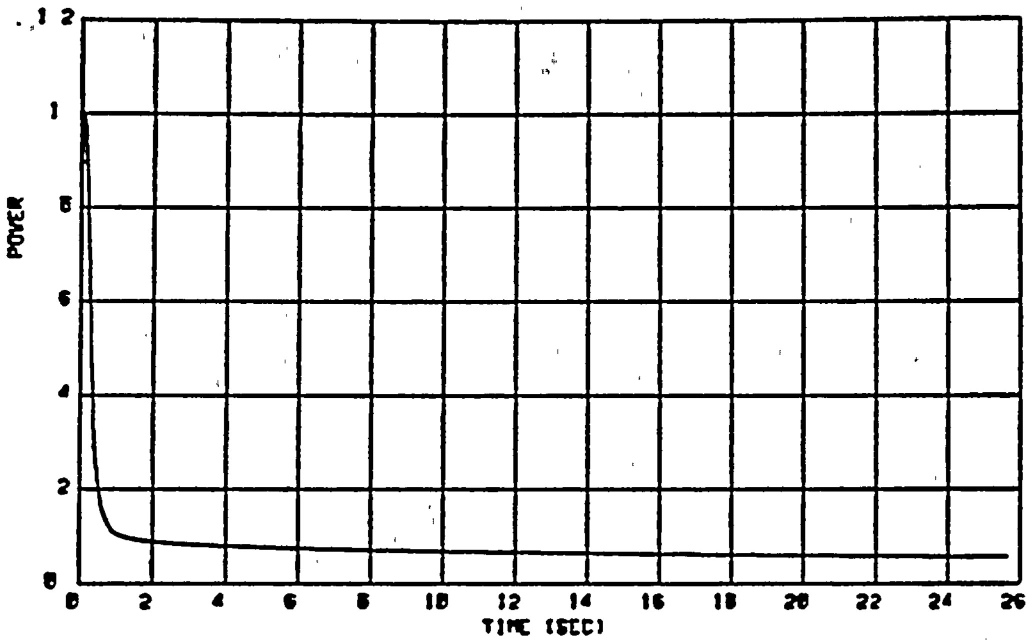


Figure 15.4-55 Core Power Transient - DECLG (C_D=0.8), Unit 2

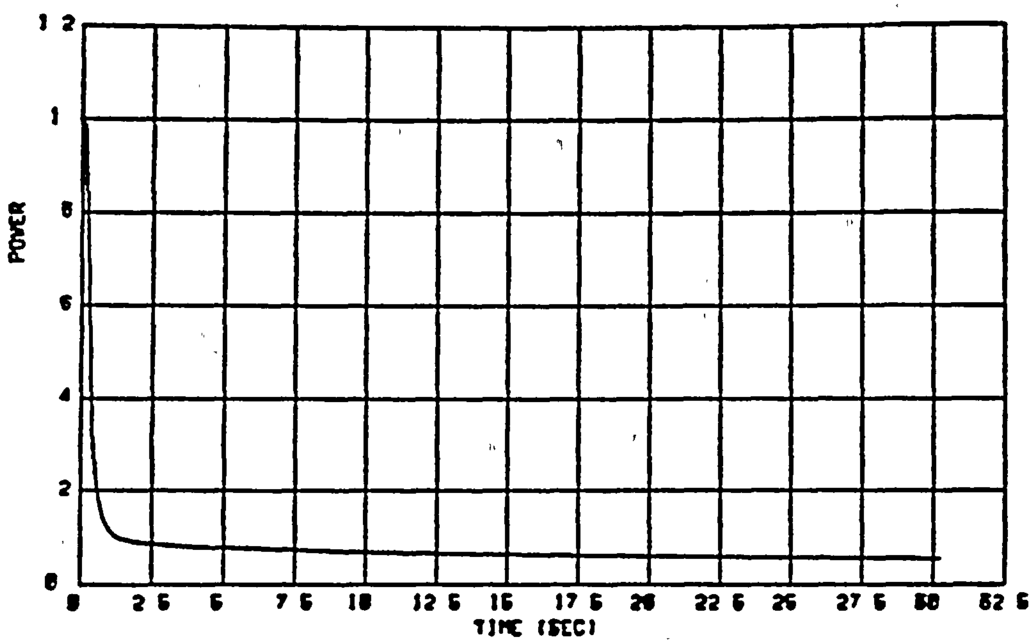


Figure 15.4-56 Core Power Transient - DECLG (C_D=0.6), Unit 2

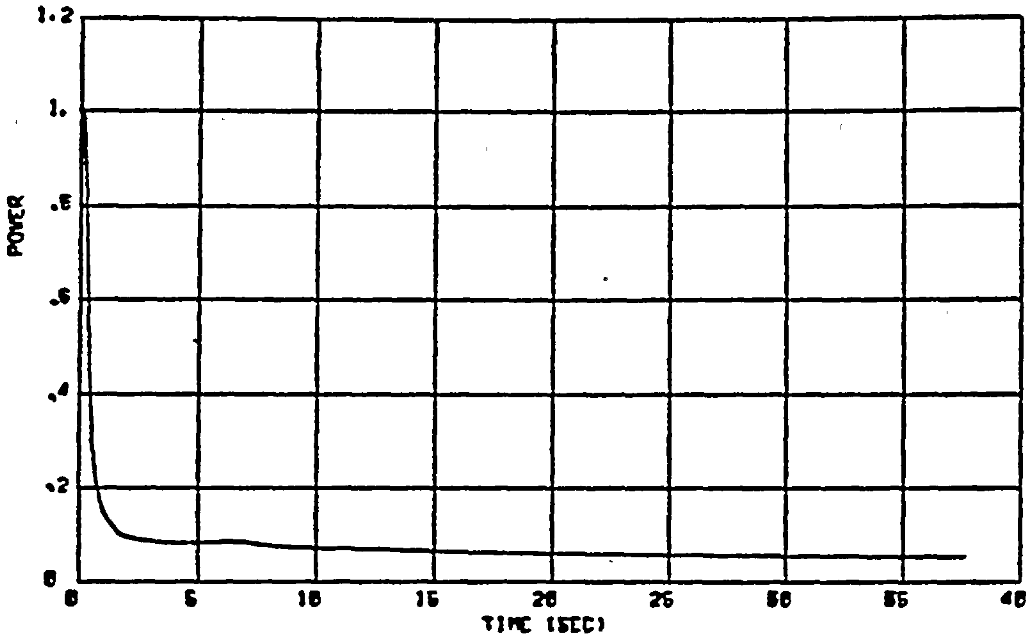


Figure 15.4-57 Core Power Transient - DECLG (C_D=0.4), Unit 2

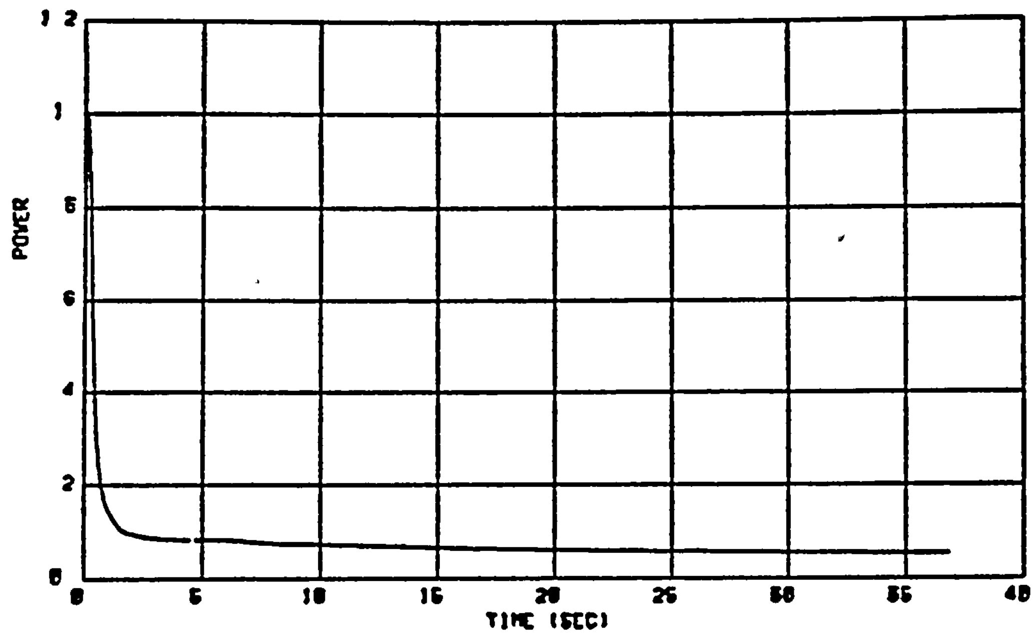


Figure 15.4-57A Core Power Transient - DECLG (C_D=0.4), Unit 2
Maximum Safety Injection

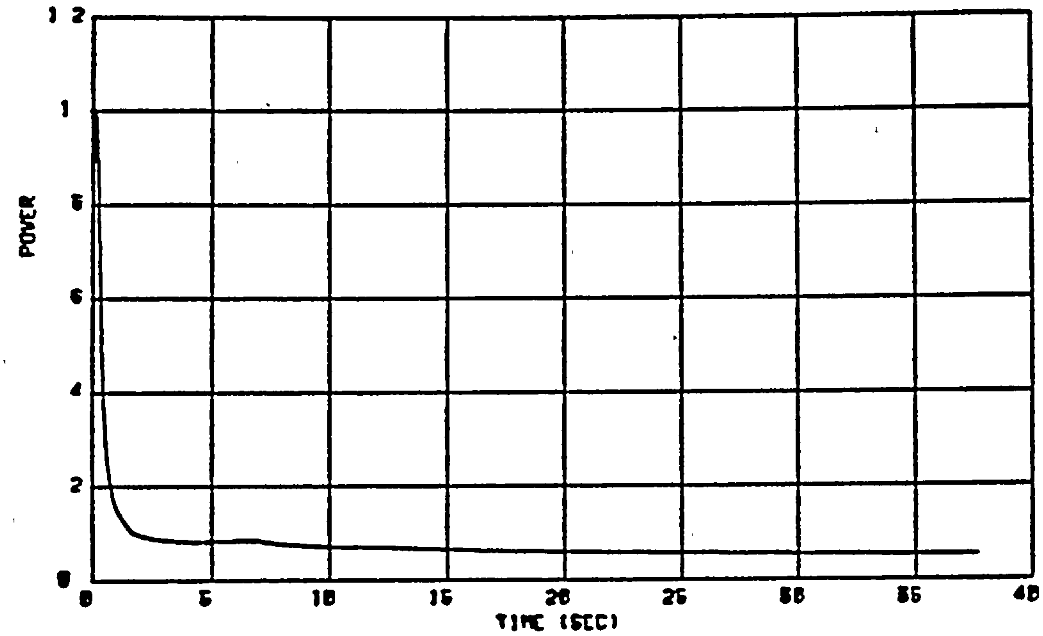


Figure 15.4-58 Core Power Transient - DECLG (C_D=0.4), Unit 1

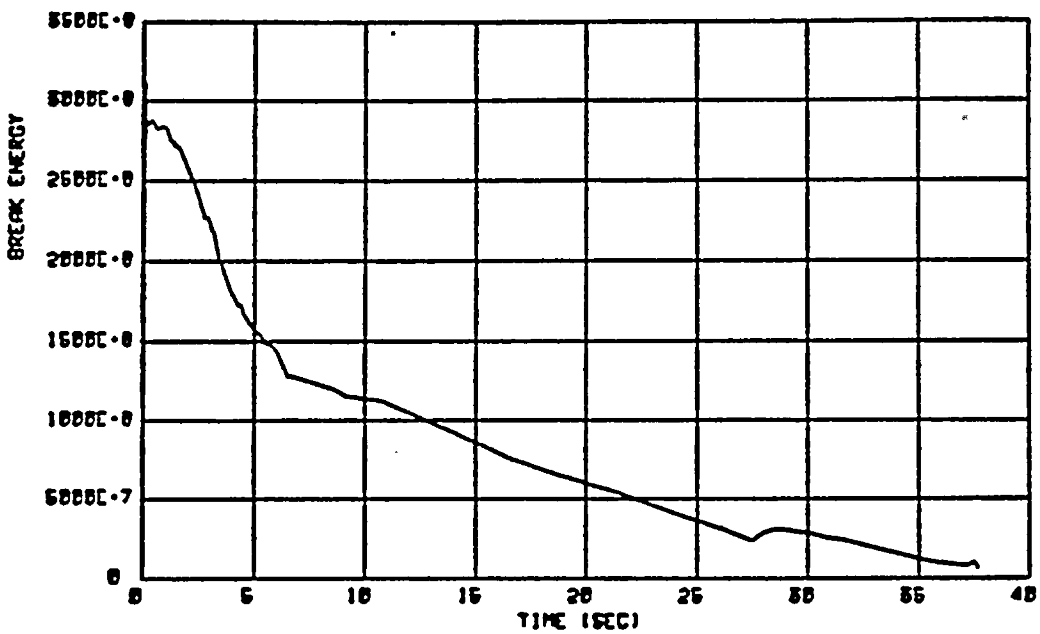


Figure 15.4-59 Break Energy Released (Blowdown)-DECLG (C_D=0.4), Unit 2

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

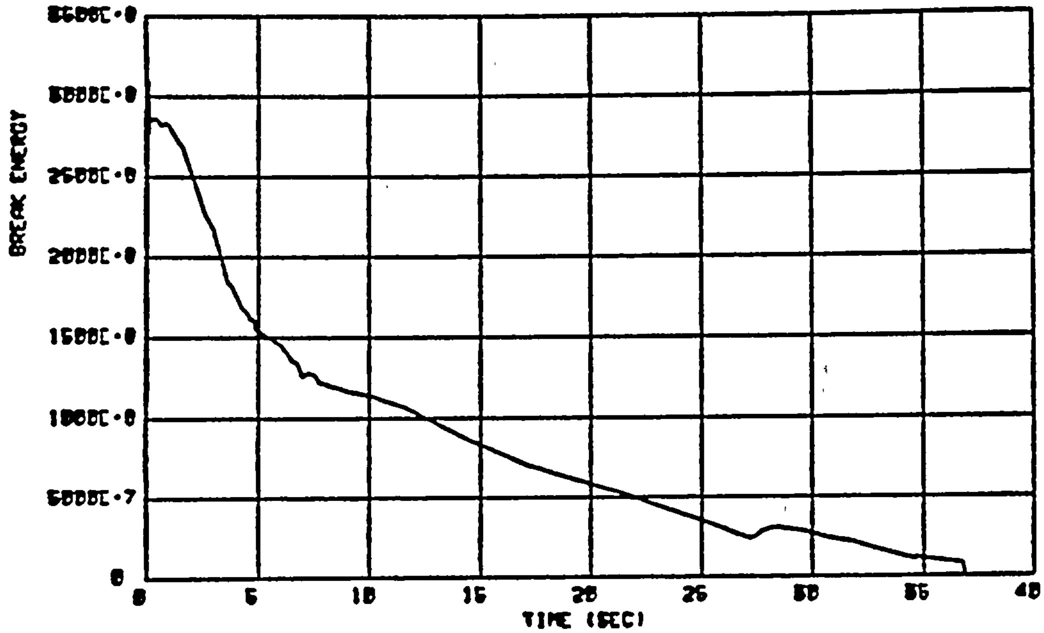


Figure 15.4-59A Break Energy Released (Blowdown)-DECLG (C_D=0.4), Unit 2 Maximum Safety Injection

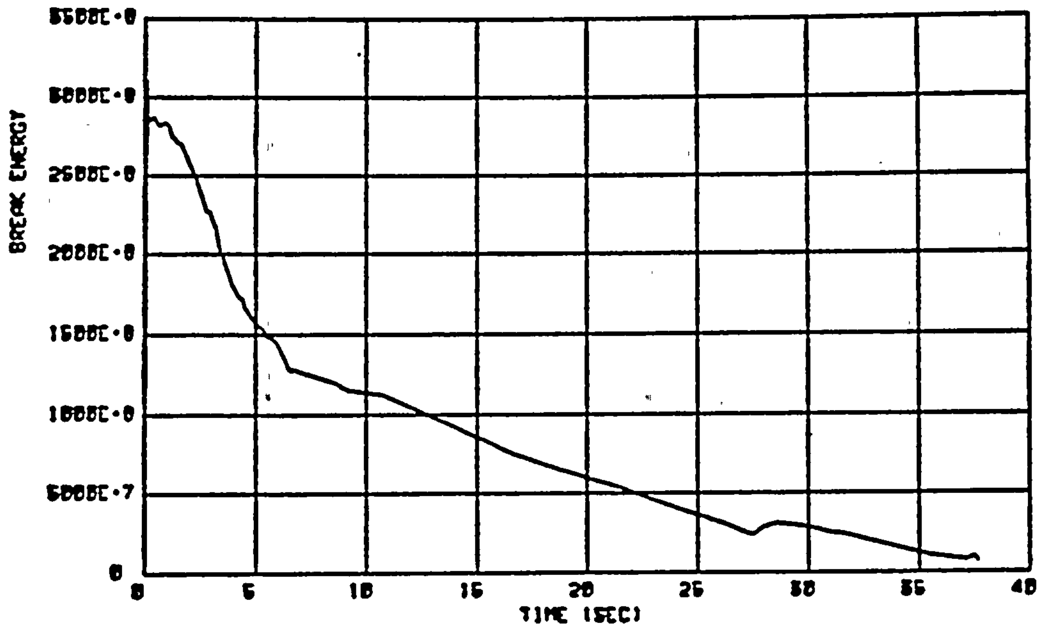


Figure 15.4-60 Break Energy Released (Blowdown)-DECLG (C_D=0.4), Unit 1

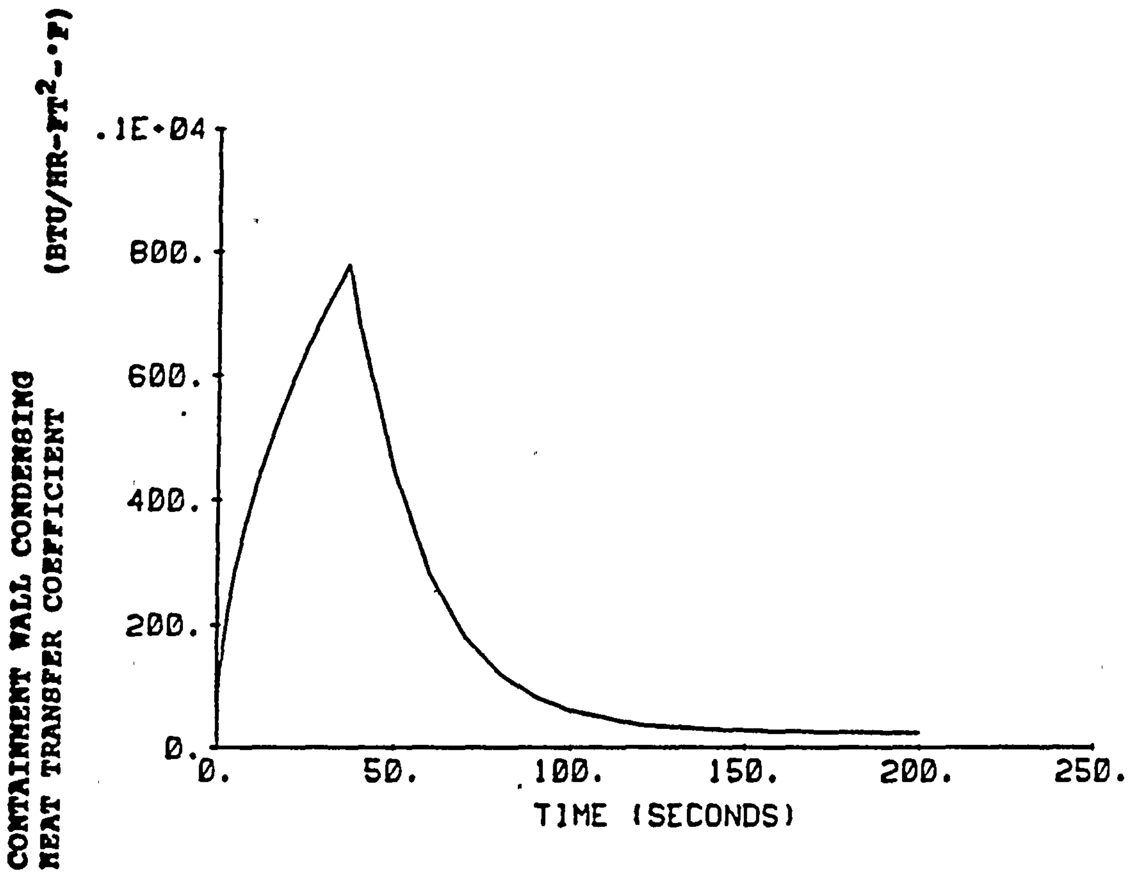


Figure 15.4-61 Containment Wall Condensing - DECLG (C_D=0.4), Unit 2 Heat Transfer Coefficient

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

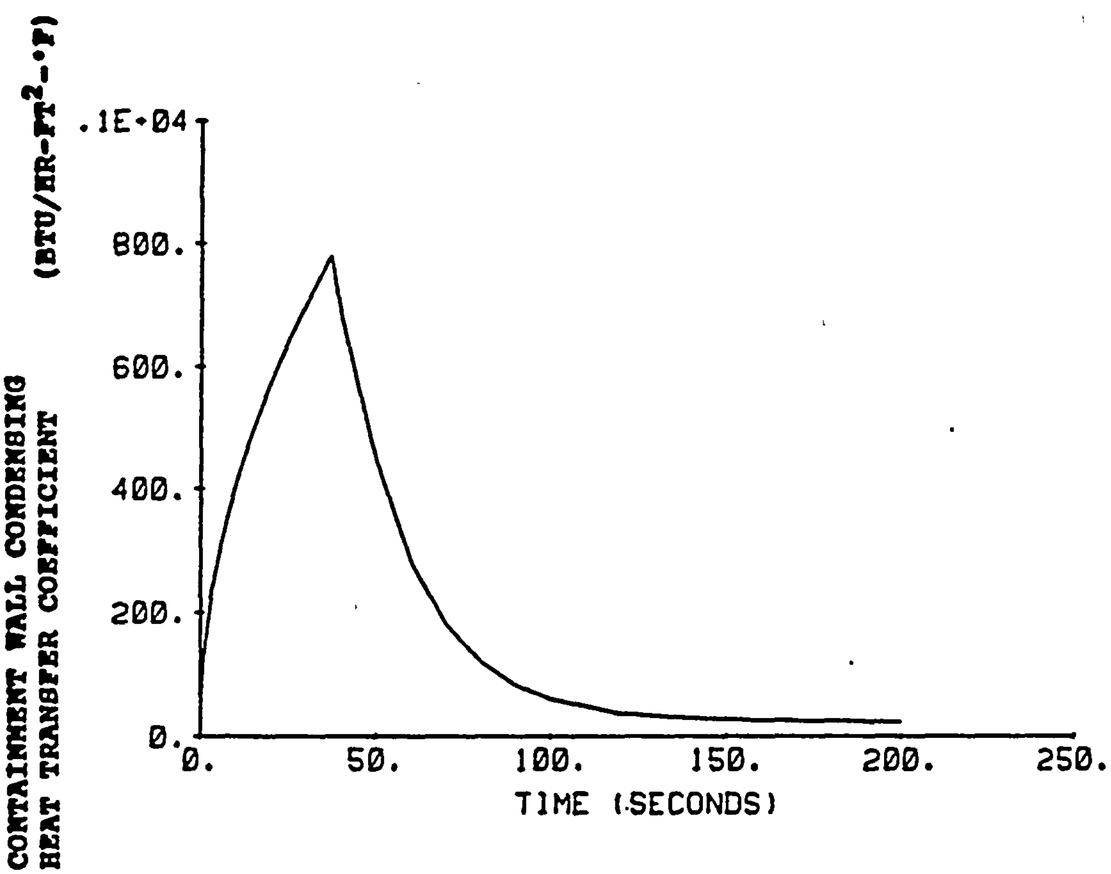


Figure 15.4-61A Containment Wall Condensing Heat Transfer Coefficient - DECLG (C_D=0.4), Unit 2 - Maximum Safety Injection

DIABLO CANYON POWER PLANTS (DCPP)

UNITS 1 AND 2 FSAR UPDATE

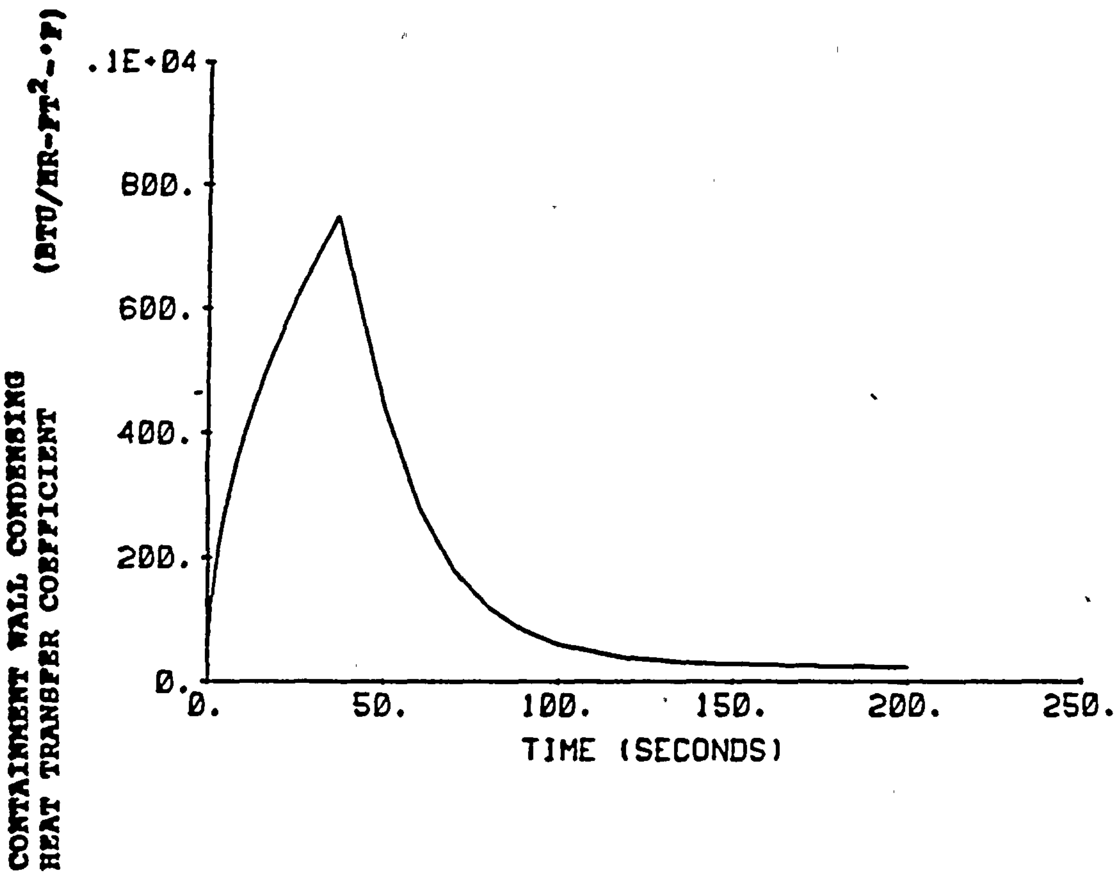


Figure 15.4-62 Containment Wall Condensing - DECLG (C_D=0.4), Unit 1 Heat Transfer Coefficient