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Group Vice President

March 17, 1992

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NO. 50-446
ADVANCE FSAR SUBMITTAL - UNIT 2 UPDATE FOR OPTIMIZED FUEL ASSEMBLY AND UNIT 2 ACCIDENT ANALYSES METHODOLOGIES

Gentlemen:

Amendment 84 to the CPSES FSAR was transmitted in TU Electric letter TXX-92082, dated February 28, 1992. As stated in TXX-92082, a portion of the Unit 2 update was not available in time for incorporation into Amendment 84. The remaining portions of the Unit 2 update are attached. The material in Amendment 84 and this letter should provide the balance of information required for the review of Unit 2.

For Large and Small Break LOCA, separate Figures which show the Unit 2 plant parameters have been added. The current FSAR Figures for Large and Small Break LOCA are being relabeled from "Units 1 and 2" to "Unit 1," but have not been provided as part of the attachment. Only the Figures which are relevant to the Unit 2 submittal are attached.

As the NRC Staff is aware, a license amendment has been requested concerning the operation of the Flux Doubling Actuation System during a Boron Dilution Event in Operating Modes 3, 4, and 5. Several Unit 2 parameters and results for this event remain labeled as "[TBD]" in the attachment pending resolution of the license amendment request.

To facilitate NRC Staff review of the Unit 2 update, the attachment is organized as follows:

1. A marked-up copy of the revised FSAR pages (changes are indicated in the margin by the word "draft").

2. A description/justification of each change.

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The material attached to this letter will be incorporated in FSAR Amendment 85, which is currently scheduled for May, 1992. If you have any questions regarding this submittal, please contact David Bize at (214) 812-8879.

Sincerely,

William J. Cahill, Jr.

By: JSMarshall
J. S. Marshall
Generic Licensing Manager

Attachment
DNB/dnb

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (2)
Mr. M. B. Fields (NRR)

A recent measurement in the second cycle of a 121 assembly, 12 foot, core is compared with a simplified one dimensional core average axial calculation in Figure 4.3-25. This calculation does not give explicit representation to the fuel grids.

The accumulated data on power distributions in actual operation is basically of three types:

1. Much of the data is obtained in steady state operation at constant power in the normal operating configuration;

2. Data with unusual values of axial offset are obtained as part of the excore detector calibration exercise which is performed every 92 EFPD; 84

3. Special tests have been performed in load follow and other transient xenon conditions which have yielded useful information on power distributions.

These data are presented in detail in Reference [8]. Figure 4.3-26 contains a summary of measured values of F_Q as a function of axial offset for five plants from that report.

4.3.2.2.8 Testing and Operations Support DRAFT

A very extensive series of physics tests is performed on the first cores. These tests and the criteria for satisfactory results are described in Chapter 14. Since not all limiting situations can be created at BOL, the main purpose of the tests is to provide a check on the calculational methods used in the predictions for the conditions of the test. Limited tests are performed at the beginning of each reload cycle to verify that the core is loaded as designed and can be safely operated. The methodology is described in References 35 and 36 are employed to predict core characteristics required for physics testing and reactor operations. DRAFT

28. Moore, J. S., "Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods," WCAP-7806, December 1971.

29. Nodvik, R. J., "Saxton Core II Fuel Performance Evaluation," WCAP-3385-56, Part II, "Evaluation of Mass Spectrometric and Radio-chemical Analyses of Irradiated Saxton Plutonium Fuel," July 1970.

30. Leamer, R. D., et al., "PUO-2-UO2 Fueled Critical Experiments," WCAP-3726-1, July 1967.

31. Vassallo, D. B. "Interium Safety Evaluation Report on Westinghouse Fuel Rod Bowing," USNRC, April 1976.

32. Camden T. M., et. al., "PALADON - Westinghouse Nodal Computer Code," WCAP-9485-PA, December 1978. 84

33. Ankey, R. D., "PALDON - Westinghouse Nodal Computer Code," WCAP-9485-PA, Supplement 1, September 1981. 84

34. Skaritka, "Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021, Rev. 1, October 1982. 84

35. Edwards, D. J., "Control Rod Worth Analysis RXE-90-005, December, 1990. DRAFT

36. Edwards, D. J., et. al., "Steady State Reactor Physics Methodology," RXE-89-003-A, July, 1989. DRAFT

TABLE 4.3-2B | 84

(Sheet 1 of 3)

NUCLEAR DESIGN PARAMETERS

(First Cycle - Unit 2)

Core Average Linear Power, kW/ft, including densification effects	5.45	
Total Heat Flux Hot Channel Factor, F_Q	2.32	
Nuclear Enthalphy Rise Hot Channel Factor, $F^N_{\Delta H}$	1.55	
Reactivity Coefficients+	Design Limits	Best Estimate
Doppler-only Power, Lower Curve Coefficients, pcm/% Power++	-19.4 to -12.6	-14 to -10
(See Figure 15.0-2), Upper Curve	-10.2 to -6.0	-12.0 to -8.0
Doppler Temperature Coefficient, pcm/°F++	-2.9 to -0.91	-1.9 to -1.3
Moderator Temperature Coefficient, pcm/°F++	+5 to -40	0 to -36.4
Rodded Moderator Density, pcm/gm/cc++	$\leq 0.43 \times 10^5$	$\leq 0.28 \times 10^5$
Boron Coefficient, pcm/ppm++	-16 to -7	-16 to -10
Boron Coefficient for Boron Dilution, pcm/ppm++		
Modes 1 and 2	-13.3	-12.7
Mode 5	-14.0	-13.7

DRAFT

\+ Uncertainties are given in Section 4.3.3.3

++ Note: 1 pcm = (percent mille) = 10^{-5} $\Delta\rho$ where $\Delta\rho$ is calculated from two statepoint values of keff by $\ln(K_2/K_1)$.

TABLE 4.3-2B | 84

(Sheet 2)

NUCLEAR DESIGN PARAMETERS

(First Cycle - Unit 2)

Delayed Neutron Fraction and Lifetime		
β_{eff} BOL, (EOL)	0.0075, (0.0044)	
ℓ^*, BOL, (EOL) μsec	20.7 (21.3)	DRAFT
Control Rods		
Rod Requirements	See Table 4.3-3	
Maximum Bank Worth, pcm	< 2000	
Maximum Ejected Rod Worth	See Section 8.2.2	
Bank Worth+++, pcm++	BOL, HZP, Xe free	EOL HZP, Equilibrium XE
Bank D	770	690
Bank C	1150	1170
Bank B	1180	1190
Bank A	360	420
Radial Factor (BOL to EOL)		
Unrodded	1.42 to 1.32	
D Bank	1.56 to 1.52	
D + C	1.58 to 1.54	

+++These are typical values for Ag-In-Cd or Hf and will vary somewhat (about 7% higher) with hybrid B_4C absorber design.

Draft Version

For transients which may be DNB limited the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits illustrated in Figure 15.0-1. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications.

DRAFT

The axial power shapes used in the DNB calculations are discussed in Section 4.4.

The radial and axial power distributions described above are input to the THINC Code as described in Section 4.4.

For transients which may be overpower limited the total peaking factor (F_q) is of importance. All transients that may be overpow limited are assumed to begin with plant conditions including power distributions which are consistent with reactor operation as defined in the Technical Specifications.

For overpower transients which are slow with respect to the fuel rod thermal time constant, for example the Chemical and Volume Control System malfunction that results in a decrease in the boron concentration in the reactor coolant incident which lasts many minutes, and the excessive increase in secondary steam flow incident which may reach equilibrium without causing a reactor trip, the fuel rod thermal evaluations are performed as discussed in Section 4.4. For overpower transients which are fast with respect to the fuel rod thermal time constant, for example the uncontrolled rod cluster control assembly bank withdrawal from subcritical or low power startup and rod cluster control assembly ejection incidents which result in a large power rise over a few seconds, a detailed fuel heat transfer calculation must be performed. Although the fuel rod thermal time constant is a function of system conditions, fuel burnup and rod power, a typical value at beginning-of-life for high power rods is approximately 5 seconds.

TABLE 15.0-2

(Sheet 3)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (Δ k/gm/cc)	Doppler	Initial NSSS Thermal Power Output Assumed[b] (MWt)	ITDP (Unit 2 Only)	
Complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, THINC	+5 (Unit 2)	0.0 (Unit 1)	upper[a]	3425	Yes	84
Reactor coolant pump shaft seizure (locked rotor)	LOFTRAN, FACTRAN	+5	-	upper[a]	3425	No	84
15.4 Reactivity and Power Distribution Anomalies							
Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, THINC	Refer to Section 15.4.1.2		Defect 1% (Unit 1) Defect 0.82% (Unit 2)	0		84
Uncontrolled rod cluster control assembly bank withdrawal at power	LOFTRAN	+5	+.43	lower and upper[a]	3425 2082 403	Yes	DRAFT
Rod cluster control assembly misalignment	TURTLE, LOFTRAN,	-	Section 15.4.3	Section 15.4.3	3425	Yes	84
Chemical and Volume Control System malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	NA	NA	NA	No	84

TABLE 15.6-2

(Sheet 5)

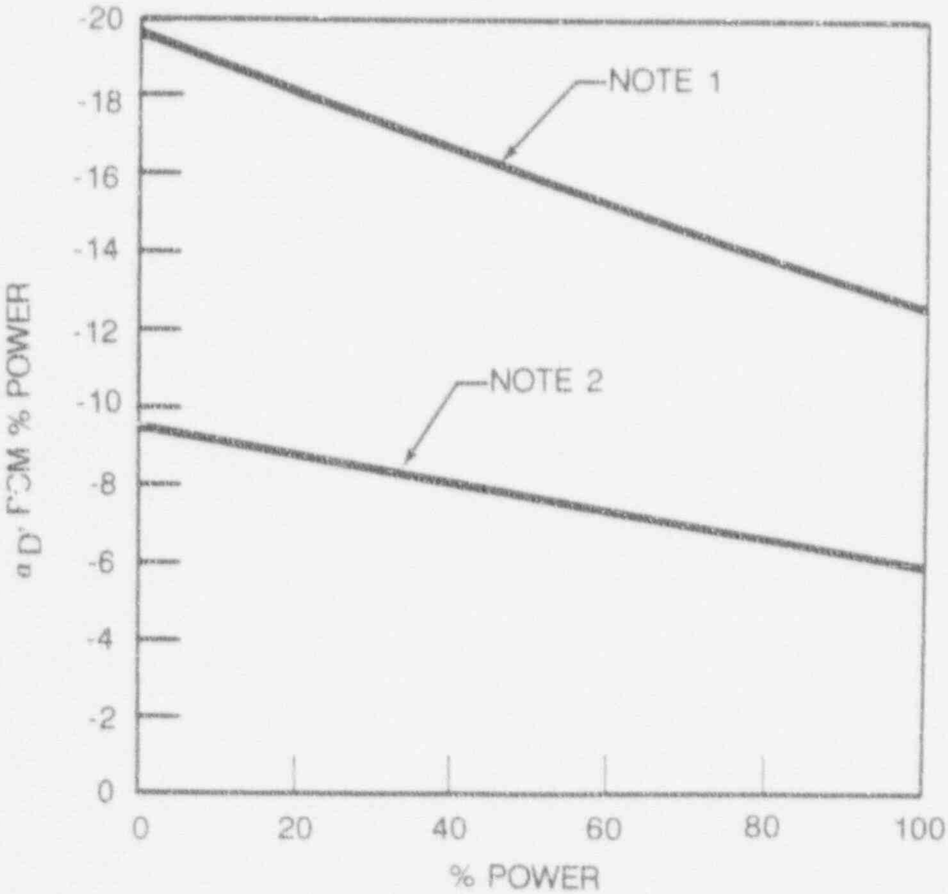
SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Faults	Computer Codes Utilized	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Moderator Density (Δk/gm/cc)	Doppler	Initial NSSS Thermal Power Output Assumed[b] (MWt)	ITDP (Unit 2 Only)	
							84
Loss of coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	LARGE BREAK (UNITS 1 & 2) SATAN-VI, WREFLOOD, COCO, LOCTA-IV	See Section 15.6.5, references	-	See Section 15.6.5, references	3411[c]	No	DRAFT 76
	SMALL BREAK (UNIT 1) WFLASH, LOCTA IV						DRAFT
	SMALL BREAK (UNIT 2) NOTRUMP, LOCTA-IV						DRAFT

[c]This power output does not include the thermal power generated by the reactor coolant pumps (see Table 15.6-5). 76

NOTE 1 - "UPPER CURVE" MOST NEGATIVE DOPPLER ONLY POWER DEFECT = -1.6 $\Delta\rho$ (0 TO 100% POWER)

NOTE 2 - "LOWER CURVE" LEAST NEGATIVE DOPPLER ONLY POWER DEFECT = -0.78 $\Delta\rho$ (0 TO 100% POWER)



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Doppler Power Coefficient Used in the Accident Analysis

FIGURE 15.0-2B

d.	Initial steam release from affected steam generator (lb) (0 to 30 min)	164,000 (Unit 1) 176,000 (Unit 2)	DRAFT DRAFT 66
e.	Long term steam release from affected steam generator (lb) (0 to 8 hr)	0 (Units 1 and 2)	DRAFT 66 66
f.	Steam release from three unaffected steam generators		66 66
	(lb) (0 to 2 hr)	438,000 (Unit 1) 413,000 (Unit 2)	DRAFT DRAFT
	(2 to 8 hr)	860,000 (Unit 1) 873,000 (Unit 2)	DRAFT DRAFT

3. Dispersion data

a.	EAB and LPZ distances	1544m and 4 miles	66
b.	x/Q	@ EAB 2.6×10^{-4}sec/m^3 (0 - 2 hr) @ LPZ (0 - 8 hr) 2.3×10^{-5}sec/m^3	66 66 66 66

4. Dose data

a.	Method of dose calculations	See Appendix 15B	66
b.	Dose conversion assumptions	See Appendix 15B	66

The following conditions were assumed for an inadvertent boron dilution while in these modes: 52

1) The boron concentration required to meet a SDM of 1.6% $\Delta K/K$ (Unit 1) and [TBD] $\Delta K/K$ (Unit 2) is very conservatively estimated to be 1465 (Unit 1) and [TBD] (Unit 2) ppm. This corresponds to a critical C_B of 1350 ppm (Unit 1) and [TBD] ppm (Unit 2), assuming a very conservative, constant boron worth of 13.9 pcm/ppm (Unit 1) and [TBD] pcm/ppm (Unit 2). 84

2) Dilution flow rate is limited by design to a maximum of 167 gpm. 74

3) A minimum RCS water volume of 4169 (Unit 1) and [TBD] (Unit 2) ft^3. This is a conservative estimate of the active volume of the RCS while on one train of RHR, and is a very conservative estimate of the active RCS volume with one reactor coolant pump operating. 84 14

Q212.78
Q212.136

Dilution During Startup 14

Startup is a transitory mode of operation. In this mode the plant is being taken from one long term mode of operation, Hot Standby, to another, Power. The plant is maintained in the Startup mode only for the purpose of startup testing at the beginning of each cycle. During this mode of operation the plant is in manual control, i.e., T_{avg}/ rod control is in manual. All normal actions required to change power level, either up or down, require operator initiation. The Technical Specifications require a SDM of 1.6% $\Delta K/K$ (Unit 1) and 1.3% $\Delta K/K$ (Unit 2) and four reactor coolant pumps operating. Other conditions assumed are: 14 DRAFT 14

Q212.78
Q212.136

1) Dilution flow rate is limited by the design of the CVCS and RMWS. The makeup flow rate is limited to a maximum of 167 gpm for startup. 74

2) A minimum RCS water volume of 9000 (Unit 1) and 9100 (Unit 2) DRAFT
ft^3. This is a very conservative estimate of the active RCS
volume, minus the pressurizer volume. 14

3) Initial C_B for criticality is assumed to be 1600 ppm (Unit 1) DRAFT
and 1500 ppm (Unit 2) with a very conservative, constant boron
worth of 12.5 pcm/ppm (Unit 1) and 13.3 pcm/ppm (Unit 2).

<u>Dilution During Full Power Operation</u> 14

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Q212.136

The plant may be operated at power two ways, automatic T_{avg}/rod control or under manual (operator) rod control. The Technical Specifications require an available shutdown margin of 1.6% $\Delta K/K$ (Unit 1) and 1.3% $\Delta K/K$ (Unit 2) and four reactor coolant pumps operating. With the plant at power and the RCS at pressure, the dilution rate is limited by the capacity of the centrifugal charging pumps (analysis is performed assuming two charging pumps are in operation even though normal operation is with one pump). Conditions assumed for this mode are: 15 84 14

1) Dilution flow rate is limited by the design of the CVCS and RMWS. The makeup flow rate is limited to a maximum of 167 gpm. When the pressurizer level control is in manual, the maximum dilution flow rate is 167 gpm and when in automatic pressurizer level control, the dilution is limited to the maximum letdown flow rate (approximately 125 gpm). 74

Q212.78
Q212.136

2) A minimum RCS water volume of 9000 (Unit 1) and 9100 (Unit 2) DRAFT
ft^3. This is very conservative estimate of the active RCS
volume, minus the pressurizer volume. 14

3) Initial C_B for criticality is assumed to be 1600 ppm (Unit 1) DRAFT
and 1500 ppm (Unit 2) with a very conservative, constant boron
worth of 12.5 pcm/ppm (Unit 1) and 13.3 pcm/ppm (Unit 2).

the Intermediate Range. Too fast a power escalation (due to an 15
unknown dilution) would result in reaching P-6 unexpectedly, leaving 14
insufficient time to manually block the Source Range reactor trip.
Failure to perform this manual action results in a reactor trip and
immediate shutdown of the reactor.

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Q212.137

However, in the event of an unplanned approach to criticality or 14
dilution during power escalation while in the Startup mode, the plant
status is such that minimal impact will result. The plant will
slowly escalate in power to a reactor trip on the Power Range Neutron
Flux - High, low setpoint (nominally 25% RTP). After reactor trip DRAFT
there is at least 21.5 (Unit 1) and 17.9 (Unit 2) minutes for operator
action prior to return to criticality. The required operator action
is the opening of valves 1,2-LCV-112D and E to initiate boration and 14
the closing of valves 1,2-LCV-112B and C to terminate dilution.

Q212.78
Q212.136

Dilution During Full Power Operation 14

With the reactor under manual rod control and no operator action taken 14
to terminate the transient, the power and temperature rise will cause
the reactor to reach the Overtemperature N-16 trip setpoint resulting
in a reactor trip. After reactor trip there is at least 21.5 (Unit DRAFT
1) and 17.9 (Unit 2) minutes for operator action prior to return to
criticality. The required operator action is the opening of valves
1,2-LCV-112D and E and the closing of valves 1,2-LCV-112B and C. The 14
boron dilution transient in this case is essentially equivalent to an
uncontrolled rod withdrawal at power. A reactor trip occurs when 74
either the Hi Neutron Flux or the Overtemperature N-16 setpoint is
reached. The maximum reactivity insertion rate for a boron dilution DRAFT
transient is conservatively estimated to be 1.2 pcm/sec (Unit 1) and
1.2 pcm/sec (Unit 2) and is within the range of insertion rates
analyzed for uncontrolled rod withdrawal at power.

It should be noted that prior to reaching the Overtemperature N-16 reactor trip the operator will have received an alarm on Overtemperature N-16 and an Overtemperature N-16 turbine runback. With the reactor in automatic rod control the pressurizer level controller will limit the dilution flow rate to the maximum letdown rate, approximately 125 gpm. If a dilution rate in excess of the letdown rate is present, the pressurizer level controller will throttle charging flow down to match the letdown rate. 14

Q212.78
Q212.136

Thus with the reactor in automatic rod control, a boron dilution will result in a power and temperature increase such that the rod controller will attempt to compensate by slow insertion of the control rods. This action by the controller will result in at least three alarms to the operator: 14

1) rod insertion limit - low level alarm, 14

2) rod insertion limit - low-low level alarm if insertion continued after (1) above, and 14

3) axial flux difference alarm (ΔI outside of the target band). 14

Given the many alarms, indications, and the inherent slow process of dilution at power, the operator has sufficient time for action. For example, the operator has at least 28 (Unit 1) and 23 (Unit 2) minutes from the rod insertion limit low-low alarm until 1.6% $\Delta K/K$ (Unit 1) and 1.3% $\Delta K/K$ (Unit 2) is inserted at beginning-of-life. The time would be significantly longer at end-of-life, due to the low initial boron concentration, when shutdown margin is a concern. 14 DRAFT 14

Q212.66

The above results demonstrate that in all modes of operation an inadvertent boron dilution is precluded, or responded to by automatic functions, or sufficient time is available for operator action to terminate the transient. Following termination of the dilution flow and initiation of boration, the reactor is in a stable condition with the operator regaining the required shutdown margin. 14

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds) Unit 1	Time (seconds) Unit 2	
				84
2. Dilution during hot shutdown and hot standby	Dilution begins	0		74 74
	Flux doubling occurs and source of dilution is automatically isolated	416		84 74 74
	Minimum margin to loss of shutdown occurs	683		74 84
3. Dilution during full power operation				
a. Automatic reactor control	Operator receives low-low rod insertion alarm due to dilution	0	0	DRAFT 74 74
	Shutdown margin lost	1728	1434	DRAFT
b. Manual control	Dilution begins	0	0	DRAFT

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE REACTIVITY AND POWER DISTRIBUTION ANOMALIES

Accident	Event	Time (seconds) Unit 1	Unit 2	
				84
	Reactor trip setpoint reached for Overtemperature N-16	60	68	DRAFT
	Rods begin to fall into core	62	68.5	DRAFT 5
	Shutdown margin is lost (if dilution continues after trip)	1350	1142	DRAFT
Rod cluster control assembly ejection accident				
1. Beginning-of-life, full power	Initiation of rod ejection	0.0	0.0	84
	Power range high neutron flux setpoint reached	0.05	0.05	84
	Peak nuclear power occurs	0.13	0.14	84

80 break transient is much less severe than an equivalent sized break in
the cold leg. Thus, the transient due to the inadvertent opening of
a pressurizer safety valve would be much less severe than the small
break cold leg transients discussed in Section 15.6.5. Initially the
event results in a rapidly decreasing RCS pressure until this
pressure reaches a value corresponding to the hot leg saturation
pressure. At this time, the pressure decrease is slowed
considerably. The pressure continues to decrease throughout the
84 transient. The effect of the pressure decrease would be to decrease
or increase the neutron flux depending on the moderator density
feedback, but the Reactor Control System (if in the automatic mode)
functions to maintain the power and average coolant temperature
essentially constant until the reactor trip occurs. Initially, the
pressurizer level increases due to expansion caused by the
depressurization, and then decreases following the reactor trip.

The reactor may be tripped by the following Reactor Protection System signals:

5 1. Overtemperature N-16.
5 2. Pressurizer low pressure.

An inadvertent opening of a pressurizer safety valve is classified as an American Nuclear Society (ANS) Condition II event, a fault of moderate frequency. See Section 15.0.1 for a discussion of Condition II events.

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

<u>Method of Analysis</u>

The accidental depressurization transient is analyzed by employing the detailed digital computer code LOFTRAN [1]. 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.

Plant characteristics and initial conditions are discussed in Section 15.0.3. For Unit 2, this transient is analyzed with the Improved 84
Thermal Design Procedure as described in WCAP-8567[27]. In order to give conservative results in calculating the departure from nucleate boiling ratio (DNBR) during the transient, the following assumptions are made:

1. Initial conditions of maximum core power, maximum reactor coolant 84
   average temperature and minimum reactor coolant pressure, plus uncertainties are assumed, resulting in the minimum initial margin to DNB (see Section 15.0.3). For Unit 2, uncertainties in initial conditions are included in the limit DNBR as described in WCAP-8567[27].

2. The most positive moderator temperature coefficient of reactivity 84
   is assumed in order to provide a conservatively high amount of positive reactivity feedback due to changes in the moderator density. The spatial effect of void 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 84
   to flatten the core power distribution.

3. The least negative Doppler coefficient of reactivity is assumed 84
   such that the resultant amount of negative feedback is conservatively low in order to maximize any power increase due to moderator reactivity feedback.

Plant systems and equipment which are necessary to mitigate the effects of RCS depressurization caused by an inadvertent safety valve opening are discussed in Section 15.0.8 and listed in Table 15.0-6.

70 6. Single Active Failure

78 The failure of the main feedwater regulating valve, in its as-is position, is the limiting single active failure with respect to the filling of the ruptured steam generator. For this single failure scenario, 100% main feedwater flow to the ruptured steam generator is conservatively assumed to continue until the closure of the main feedwater isolation valve, and the elapsed time from reactor trip to automatic feedwater isolation is conservatively maximized. For additional conservatism, no credit is taken for the early identification and isolation of the ruptured steam generator by the reactor operators.

Results

Q212.66

81 Evaluation of the steam generator tube rupture event indicates that DNBR is not reached and, thus, no clad damage would be expected in this transient. This is consistent with the fact that when the reactor is at power, the reactor coolant pumps are operating and, for this event, only a small fraction of the total primary system fluid inventory has leaked to the secondary side. Thus, it is very unlikely that DNB would occur as a result of the reduced RCS flow. The RCS depressurization that results due to flow out of the tube rupture presents another possibility for obtaining a low DNBR. However, the depressurization that occurs in a steam generator tube rupture is much less than considered in the depressurization transient analyzed in Section 15.6.1 for the Inadvertent Opening or a
DRAFT Pressurizer Safety or Relief Valve. In the analysis of that event, it was determined that the DNBR remains above the limit value throughout the transient and, thus, no clad damage is expected. From
81 this, it is concluded that no clad damage is expected in the steam generator tube rupture event.

78 Representative transient responses for the design-basis steam generator tube rupture event are shown in Figures 15.6-3 and 15.6-3A.

76 described. Equalization of the primary and secondary system pressures terminates the leakage flow into the ruptured steam generator in sufficient time to prevent filling the ruptured steam
78 generator. The volume available prior to filling the ruptured steam generator is greater than for the scenario described in Section 15.6.3.2.

76 The time dependent mass releases used to assess the radiological consequences of the postulated steam generator tube rupture are calculated from the RETRAN02 thermal-hydraulic analysis described above. Time-dependent values of the leakage rate into the ruptured steam generator and the flashing fraction were also used to assess the radiological consequences for the 0-2 hour time period following the event. Following the closure of the atmospheric relief valve block valve, the additional radiological dose is due to the leakage from the primary system into the intact steam generators and the initial concentration of radioactivity contained in the intact steam generators.

70 Two separate iodine spikes are considered:

70 <u>Case I</u> A reactor transient has occurred prior to the tube rupture and raised the primary coolant iodine concentration to
72 60 uCi/gm Dose Equivalent Iodine-131 (DEQ I-131). The resulting preaccident isotopic iodine concentrations are shown in Table 15.6-3.

70 <u>Case II</u> The reactor trip or primary system depressurization associated with the postulated accident creates an iodine spike in the primary system. The spike is assumed to increase the iodine appearance rate (inleakage from the defective fuel rods to the primary coolant) to 500 times the equilibrium appearance rate. The concurrent iodine spike appearance rates are presented in Table 15.6-4.

The assumptions below are used to determine the initial primary and secondary activities and to calculate the activity released and the

offsite doses for the postulated steam generator tube rupture accident. 70

1. The initial primary coolant iodine activity (i.e., prior to any iodine spike considerations) is assumed to be at 1.0 70
uCi/gm DEQ I-131. 70

2. The primary coolant activity has been leaking into the secondary side at one gpm for a period of time long enough to establish equilibrium activity concentrations is the steam generators. 70

3. All noble gas activity transported from the primary system to the secondary system and all noble gas and iodine activity initially in the steam region of the steam generators is immediately released to the environment. The initial iodine activity in the water region of the ruptured steam generator increases over time due to the unflashed portion of the leakage. 76

4. Due to the pressure differential between the primary and secondary sides, a fraction of the primary coolant that leaked to the defective steam generator flashes to steam. This flashed fraction does not mix with the steam generator water and, therefore, is not subjected to any iodine removal process in the steam generator. However, the flashed fraction experiences iodine removal in the condenser when that path is available. 76

5. Radioactive decay of parent iodines to noble gas products is considered during the iodine spiking processes and as unflashed iodine accumulates in the steam generators. the radioactive decay of the parent iodines is conservatively assumed to not decrease the activity of the parent iodines. 76

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 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.

DRAFT The blowdown phase of the transient ends when the RCS pressure (initially assumed at 2250 psia for Unit 1 and 2280 psia for Unit 2) 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 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.

Q212.40

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 engineered safety features sumps by the low head safety injection (residual heat removal) pumps and returned to the RCS cold legs. The Containment 6
Spray System continues to operate to further reduce Containment pressure. Within approximately 16 hours after initiation of the 78
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.

Description of Small Break LOCA Transient

As contrasted with the large break, the blowdown phase of the small break occurs over a longer time period. Thus, for the small break LOCA there are only three characteristic stages, i.e., gradual blowdown in which the decrease in water level is checked, core recovery, and long term recirculation.

15.6.5.3 Core and System Performance

15.6.5.3.1 Mathematical Model

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10CFR50 [3].

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, including the thermal-hydraulic transient in the RCS, the pressure and temperature transient within the Containment, and the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of inter-related computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in Reference [5]. 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 [6]
DRAFT through [9]. Modifications to the model are described in References [10], [11], [12], [29] and [30]. These codes are used to assess the
6 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 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.

DRAFT The analysis presented here was performed for Unit 1 with the February 1978 version of the evaluation model which includes the modifications delineated in References [15], [16], [17] and [17a]. The analysis was performed for Unit 2 with the approved 1981 version of the evaluation model which includes modifications delineated in References [15], [16a], [17] and [17a].

6 The analysis in this section was performed with the upper head fluid temperature equal to the reactor coolant system cold leg fluid temperature.

6 The upper head fluid temperature has been made equal to the cold leg temperature by increasing the upper head cooling flow [20].

Small Break LOCA Evaluation Model

DRAFT The WFLASH program used in the analysis of the small break LOCA for Unit 1 is an extension of the FLASH-4 code [13] developed at the Westinghouse Bettis Atomic Power Laboratory. The NOTRUMP computer program was used in the analysis of the small break LOCA for Unit 2, which incorporates a number of advanced features. Among these new features are the utilization of nonequilibrium thermal calculation in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flow limitations, mixture level tracking logic in multiple-stack fluid nodes and regime-dependent heat transfer
DRAFT correlations. The NOTRUMP [31, 32] small break LOCA 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." The WFLASH and NOTRUMP programs permit a detailed spatial representation of the RCS.

The PCS 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. Detailed descriptions of WFLASH and DRAFT
NOTRUMP are given in References [14] and [31, 32, 33], respectively.

The use of WFLASH and NOTRUMP in the analysis involves, among other DRAFT
things, the representation of the reactor core as a heated control
volume with the associated bubble rise model to permit a transient
mixture height calculation. The multi-node 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.

Clad thermal analyses are performed with the LOCTA-IV code [9] which DRAFT
uses the RCS pressure, fuel rod power history, steam flow past the
uncovered part of the core and mixture height history from the WFLASH
and NOTRUMP hydraulic calculations as input.

Q212.80

The small break analysis for Unit 1 was performed with the October, DRAFT
1975 version of the Westinghouse ECCS Evaluation Model (refer to
References [9], [14], [14a] and [14b]) and for Unit 2 with the
May, 1985 NOTRUMP ECCS Evaluation model (refer to References [33]). 6

Schematic representations of the computer code interfaces are given in DRAFT
Figures 15.6-5 and 15.6-6 for Large Break LOCA and Small Break LOCA,
respectively.

15.6.5.3.2 Input Parameters and Initial Conditions

Table 15.6-5 lists important input parameters and initial conditions used in the analysis.

The analysis presented in this section was performed with a reactor 6
vessel upper head temperature equal to the RCS cold leg temperature.
The effect of using the cold leg temperature in the reactor vessel
upper head is described in Reference [20]. In addition, the large 36
break analysis in this section utilized the upflow barrel-baffle
methodology described in Reference [25].

Q212.134

Q212.65

DRAFT The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from sensitivity studies (refer to References [18], [19] and [20] for Unit 1 and References [18], [19], [20] and [20a] for Unit 2). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant

DRAFT overall conservatism in the analysis. For the Unit 1 large and small break LOCA base cases, the worst single failure assumed in the analyses is one RHR pump and one safety injection train, respectively. For Unit 2, the worst large break LOCA (C_D=0.6) is analyzed in accordance with the methodology in Reference [28] and found to be most limiting when no single failure (maximum safeguards) is used. The

72 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. Decay heat generated through the transient is also conservatively calculated.

15.6.5.3.3 Results

78 The results of the base case LOCA analyses and the peak cladding temperature (PCT) penalties associated with subsequent safety evaluations are described in this section.

<u>Large Break Results</u>

DRAFT Based on the results of the LOCA sensitivity studies (References [18], [19] and [20] for Unit 1 and References [18], [19], [20] and [20a] for Unit 2), the limiting large break was found to be the double ended cold leg guillotine (DECLG). Therefore, only the DECLG break is considered in the large break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 15.6-1 and 15.6-6.

The mass and energy release data for the break resulting in the highest calculated peak clad temperature are presented in Section 6.2.1.5.

For Unit 1, figures 15.6-7 through 15.6-23 and 15.6-47A present the transients for the principle parameters. The following items are noted: DRAFT 6

Figures 15.6-7A - 15.6-9D:	Quality, mass velocity, and clad heat transfer coefficient for the hotspot and burst locations	6
Figures 15.6-10A - 15.6-12D:	Core pressure, break flow, and core pressure drop. The break flow is the sum of the flowrates from both ends of the guillotine break. The core pressure drop is taken as the pressure just before the core inlet minus the pressure just beyond the core outlet	6 72
Figures 15.6-13A - 15.6-15D:	Clad temperature, fluid temperature, and core flow. The clad and fluid temperatures are for the hot spot and burst locations	6
Figures 15.6-16A - 15.6-17D:	Downcomer and core water level during reflood, and flooding rate	6
Figures 15.6-18A - 15.6-19D:	Emergency core cooling system flowrates, for both accumulator and pumped safety injection	6
Figures 15.6-20A - 15.6-21D:	Containment pressure, and core power transients	6

6	Figures 15.6-22 - 15.6-23:	Break energy release during blowdown, and the containment wall condensing heat transfer coefficient for the worst break
DRAFT	Figures 15.6-47A	Large Break pumped safety injection flow rate
6	Section 6.2.1.5:	Presents a discussion of the containment pressure transient resulting from a LOCA.

DRAFT For Unit 2, figures 15.6-49 through 15.6-75 present the parameters of principal interest from the large break ECCS analyses: For all cases analyzed, transients of the following parameters are presented:

DRAFT a. Hot spot clad temperature. (Figures 15.6-49, 15.6-49A, 15.6-64, 15.6-70)

DRAFT b. Coolant pressure in the reactor core. (Figures 15.6-50, 15.6-65, 15.6-71)

DRAFT c. Water level in the core and downcomer during reflood. (Figures 15.6-51, 15.6-51A, 15.6-66, 15.6-72)

DRAFT d. Core reflooding rate. (Figures 15.6-52, 15.6-52A, 15.6-67, 15.6-73)

DRAFT e. Thermal power during blowdown. (Figures 15.6-53, 15.6-68, 15.6-74)

DRAFT f. Containment Pressure (Figures 15.6-54, 15.6-54A, 15.6-69, 15.6-75)

The Containment pressure transient resulting from a LOCA is presented in Section 6.2.1.5.

For the limiting break analyzed for Unit 2, the following additional transient parameters are presented: DRAFT

a. Core flow during blowdown (inlet and outlet). (Figure 15.6-55) DRAFT

b. Core heat transfer coefficients. (Figure 15.6-56) DRAFT

c. Hot spot fluid temperature. (Figure 15.6-57) DRAFT

d. Mass released to Containment during blowdown. (Figure 15.6-58) DRAFT

e. Energy released to Containment during blowdown. (Figure 15.6-59) DRAFT

f. Fluid quality in the hot assembly during blowdown. (Figure 15.6-60) DRAFT

g. Mass velocity during blowdown. (Figure 15.6-62) DRAFT

h. Accumulator water flow rate during blowdown. (Figure 15.6-61) DRAFT

i. Pumped safety injection water flow rate during reflood. **(Figure 15.6-63)** **DRAFT**

j. Large Break pumped safety injection flow rate for the worst single failure of one RHR pump (Figure 15.6-47A) DRAFT

The maximum clad temperature calculated for the large break base case is 2011°F for Unit 1 and 1808°F for Unit 2 which are less than the 10CFR50.46 Acceptance Criteria limit of 2200°F. The maximum local metal-water reaction is 3.92 percent for Unit 1 and 2.04 percent for Unit 2, which are well below the embrittlement limit of 17 percent as required by 10CFR50.46. The total core metal-water reaction is less than 0.3 percent for both units for all breaks, as compared with the 1 percent criterion of 10CFR50.46, and the clad temperature transient is terminated at a time when the core geometry DRAFT

is still amenable to cooling. As a result, the core temperature will continue to drop and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

DRAFT

A number of safety evaluations for the large break LOCA analysis for both units have been performed subsequent to the calculations reported above. For Unit 1, these safety evaluations were for reduced accumulator water volume, thimble tube modeling, reduced RHR flow due to miniflow, steam generator feedwater flow split modification, 1% uniform steam generator tube plugging and a 2.1% correction in steam generator flow area, steam generator tube seismic-LOCA assumption and resulted in a total peak cladding temperature (PCT) penalty of 55.0°F for Unit 1. The final limiting PCT for the Unit 1 large break, including the penalties from these safety evaluations, is 2065.7°F.

DRAFT

For Unit 2, these safety evaluations were due to revision of thimble tube modeling, fuel rod modeling, steam generator tube seismic-LOCA assumption, modification in feedwater flow split, a reduction in the minimum SI flow and steam generator tube plugging (1%). Due to the reduction in the minimum SI flow, the minimum SI flow case becomes more limiting compared to the base case. This is because the reduction in the SI flow affects the PCT (1804°F) associated with the minimum SI flow case only, whereas the maximum SI case (1808°F) remains unaffected. The above noted changes result in a peak clad temperature (PCT) penalty of 111°F for Unit 2. The limiting peak clad temperature (PCT) for the Unit 2 large break LOCA, including penalties, is 1915°F.

DRAFT

The PCTs for both Units are below the 10CFR50.46 Acceptance Criterion for peak clad temperature of 2200°F.

<u>Small Break Results</u>

As noted previously, the calculated peak clad temperature resulting from a small break LOCA is less than that calculated for a large break. Based on the results of the LOCA sensitivity studies (Reference [19]) the limiting small break was found to be less than a 10 inch diameter rupture of the RCS cold leg. A small break specturm analysis showed that the limiting small breaks were 4 inch and 3 inch for Unit 1 and Unit 2, respectively. The results of the analyses are summarized in Tables 15.6-1 and 15.6-7. 6 DRAFT

Figures 15.6-34 through 15.6-46, 15.6-47B, and 15.6-48 present the principle parameters of interest for the Unit 1 small break ECCS analyses. For all cases analyzed the following transient parameters are presented: DRAFT

1. RCS pressure

2. Core mixture height

3. Hot spot clad temperature

4. Core power after reactor trip

For the limiting break analyzed for Unit 1, the following additional transient parameters are presented: DRAFT

1. Core steam flow rate

2. Core heat transfer coefficient

3. Hot spot fluid temperature

The maximum calculated peak clad temperature for the Unit 1 small break base case is $1788^{o}F$. These results are well below all Acceptance Criteria limits of 10CFR50.46 and in all cases are not limiting when compared to the results presented for Unit 1 large breaks. DRAFT

78

78 A number of safety evaluations for the small break LOCA analysis have
DRAFT been performed subsequent to the calculations reported above. For Unit 1, these safety evaluations were for reduced safety injection flow rates, an increase in the auxiliary feedwater line purge volume, a lower low pressurizer pressure safety injection signal setpoint, increased auxiliary feedwater flow and an increase in the signal processing delay time, correction of $Zr-H_2O$ reaction error and resulted in a total PCT penalty of 247°F. The final limiting PCT for the small break, including penalties from these safety evaluations, is 2035°F. This is less than the final limiting large
78 break PCT and remains well below the 2200°F Acceptance Criterion value in 10CFR50.46.

DRAFT The results of the analyses for Unit 2 are summarized in Tables 15.6-1 and 15.6-7. The principle parameters of interest for Unit 2 small break ECCS analyses are presented in Figures 15.6-47B and 15.6-76 through 15.6-89. The following is a list of the above parameters:

DRAFT 1. RCS pressure

DRAFT 2. Core mixture height

DRAFT 3. Hot spot clad temperature

DRAFT 4. Core power after reactor trip

DRAFT For the limiting break analyzed for Unit 2, the following additional transient parameters are presented:

DRAFT 1. Core steam flow rate

DRAFT 2. Core heat transfer coefficient

DRAFT 3. Hot spot fluid temperature

The maximum calculated peak cladding temperature for the Unit 2 small break case is $1434^{\circ}F$. These results are well below the Acceptance Criteria limits of 10CFR50.46 and in all cases are not limiting when compared to the results presented for Unit 2 large breaks. DRAFT

15.6.5.4 Environmental Consequences

66

To demonstrate in a conservative manner that the operation of a nuclear power station does not present any undue radiological hazard to the general public, a hypothetical accident involving a gross release of fission products is evaluated. No mechanism for such a release has been postulated because it would require a number of simultaneous failures to occur in the engineered safety features. The 66
core fission product inventory is assumed to be released into the containment as described in TID-14844 [21]. Numerical values for the total core fission product inventory of the isotopes considered in calculating the radiation doses are listed in Table 15.6-8.

The radiological evaluation of this accident is divided into two parts: internal (thyroid) dose from inhalation of iodines in the leakage plume, and external (whole body) exposure as a result of immersion in the leakage plume.

The radiological consequences due to the release of core fission 66
products during a postulated loss-of-coolant accident are evaluated in the following sections:

1. Radiological consequences of containment leakage 66

   The integrated thyroid doses and the integrated whole body doses 66
   are calculated using methods and assumptions in conformance with Regulatory Guide 1.4. The assumptions used in the analysis are listed below.

a. Twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full-power operation of the core is immediately available for leakage from the reactor containment. Of this 25 percent, 91 percent is in the form of elemental iodine, 5 percent is in the form of particulate iodine, and 4 percent is in the form of organic iodides.

b. All (i.e., 100 percent) of the equilibrium radioactive noble gas inventory developed from maximum full-power operation of the core is immediately available for leakage from the reactor containment.

c. The effects of radiological decay during holdup in the containment are taken into account.

66 d. The containment volume is divided into separate regions by concrete floors at different elevations (see Section 6.5.2). A radial gap between the concrete floors and the inner wall of the Containment Building permits a limited amount of convective
72 mixing between these regions. The region not covered by containment spray is treated as a separate unsprayed volume which is assumed to mix with the volumes in the sprayed areas at a mixing rate of two turnovers per hour.

The Containment Spray System is actuated by a high containment pressure signal. For a discussion of the sequence of events of
66 spray system operation, see Section 6.5.2. A sodium hydroxide spray is used to reduce the amount of fission product iodine available for release during the LOCA. The containment spray solution is assumed to interact with the elemental iodine and particulate iodine. The mathematical model which calculates the iodine spray removal coefficient is presented in Section
69 6.5.2. For each region the calculated elemental iodine removal coefficients are above 10 hr^{-1}. The removal coefficient for elemental iodine used in the offsite dose calculation is limited to a maximum value of 10 hr^{-1} [22]. A conservative value of 1.07 hr^{-1} is used for the particulate iodine removal

coefficient, although higher removal coefficients have been 66
calculated. The elemental iodine removal effectiveness may be expected to diminish after the concentration in the containment
atmosphere has been reduced by several orders of magnitude. The 66
elemental iodine removal effectiveness of the spray system is conservatively assumed to cease after a decontamination of 100 in the containment atmosphere has been achieved.

e. The iodine and noble gases available for release to the 66
environment are assumed to leak from the Containment at a maximum leak rate of 0.10 percent of the containment volume per day for the first 24 hours, and at 0.05 percent of the containment volume per day for the duration of the accident.

f. The duration of the accident is considered to be 30 days.

g. A ground-level release is assumed. Atmospheric dilution 66
factors are discussed in Section 2.3 [23] and listed in Table 15.6-9.

h. No credit is taken for depletion of fission products in the 72
plume due to ground deposition or radioactive decay in transit.

i. For the first 8 hours after the accident, the breathing rate of persons offsite is assumed to be 3.47×10^{-4} cubic meters per second (m^3/sec). Eight to 24 hours following the accident, the breathing rate is assumed to be 1.75×10^{-4} m^3/sec.
From 24 hours through 30 days after the accident, the rate is 66
assumed to be 2.32×10^{-4} m^3/sec.

66

j. The mathematical model and dose conversion factors presented in 66
Appendix 15B are used for evaluating the radiological consequences of the LOCA

k. Other assumptions are detailed in Table 15.6-9. 66

66 Using the assumptions presented above and the mathematical models presented in Appendix 15B, the doses at the EAB were conservatively calculated to be 117.5 rem to the thyroid and 1.48 rem to the whole body; the doses at the LPZ were conservatively calculated to be 22.2 rem to the thyroid and 0.29 rem to the whole body.

66 2. Radiological consequences of engineered safety features equipment
Q022.19g leakage outside containment.

66 Following a postulated LOCA, a potential source of fission product release is the leakage of water from engineered safety features (ESF) equipment located outside the containment. Such leakage could occur during the recirculation phase through components such as pump flanges, valves, and heat exchangers. The fission products could then be released from the water into the atmosphere resulting in offsite radiological consequences that contribute to the total dose from the LOCA.

66

66 An analysis of the offsite effects attributable to ESF equipment leakage is performed based on the following conservative assumptions:

66 a. 50 percent of the halogens originally present in the core are intimately mixed with the coolant water and are assumed to be available for release through ESF equipment outside containment (see Table 15.6-10).

66 b. All of the noble gases produced from the decay of halogens which remain in the leakage water are released to rooms housing the leaking components.

78 c. The leakage from all ESF components is conservatively assumed to start 10 minutes after the LOCA and continue for the duration of the accident at a rate of 2 gallons per minute.

d. An iodine partition factor of 0.1 is assumed. This factor is 66
taken as the fraction of iodine in the leakage that becomes
airborne.

66

e. Gaseous radioactivity released to rooms housing the leaking 66
components is considered to be immediately swept away by the
ventilation system and released to the atmosphere. (See
Section 9.4.5 and Figures 1.2-16, 1.2-35 and 9.4-9). 53

Q312.12

f. An iodine adsorber efficiency of 95 percent is applied since the 66
ventilation exhaust passes through HEPA filters and iodine
adsorbers prior to release to the atmosphere. The iodine
adsorbers are designed to the requirements of NRC Regulatory 59
Guide 1.52 (See Appendix 1A(B)) as discussed in Section 9.4.3.

g. No credit is taken for an elevated release; all meteorological 66
parameters are considered to be identical to those previously
defined in this section.

Q312.12

Based upon the foregoing model, the thyroid and whole body dose 66
contributions due to ESF equipment leakage are conservatively
calculated to be 26.9 rem and 0.742 rem, respectively, for the EAB.
The LPZ doses are conservatively calculated to be 14.1 rem to the
thyroid and 0.817 rem to the whole body.

3. Total dose due to a postulated LOCA 66

The total dose attributed to a postulated LOCA is the combined 66
doses due to containment leakage and ESF equipment leakage. The
combined EAB doses are 145 rem to the thyroid and 2.2 rem to the
whole body. The combined LPZ doses are 36.3 rem to the thyroid 72
and 1.1 rem to the whole body. As expected, the doses are below
the values set forth in 10CFR100.

Q312.12

72 The dose to personnel engaging in mineral extraction operations within the exclusion area, in the event of a postulated LOCA, would be less than the dose values of 300 rem to the thyroid and 25 rem to the whole body set forth in 10CFR100.

66

4. Dose to the control room occupants

66 In the event of a Design Basis Accident (DBA), the safety injection actuation signal or a high radiation signal from the control room air intake monitors will initiate emergency recirculation and pressurization of the Control Room air conditioning system.
46 Later, the emergency ventilation air makeup system can be brought into operation as described in Section 9.4.1.

66 The control room doses were analyzed for various design basis accidents. It was determined that the LOCA doses represent the limiting case. Therefore the methodology and the doses calculated for the LOCA are reported here.

The following assumptions are applied in the calculations of the dose to the control room occupants following the LOCA:

66 a. The basic assumptions presented in Items 1 and 2, above, are applied, except a constant breathing rate of 3.47×10^{-4} m^3/sec is assumed throughout the accident.

b. The control room pressurization (air intake) and recirculation iodine adsorbers are assigned a 99 percent decontamination efficiency for both elemental and organic iodines in accordance with Table 2 of Regulatory Guide 1.52 (See Appendix 1A(B)). 52
The pressurization adsorbers are arranged in series with the recirculation adsorbers (see Figure 9.4-1) during the emergency pressurization mode, thus providing an equivalent decontaminating efficiency in excess of 99 percent for both elemental and organic iodines from the pressurization makeup air. 59

c. The control room air-conditioning system runs either in the emergency recirculation mode or the emergency ventilation mode during a LOCA. 66

d. During the emergency recirculation mode of operation, a constant air intake flow rate of 800 ft^3/min is assumed. This makes up for losses caused by leaks and maintains the control room atmosphere at a positive pressure of 0.125 inch water gauge relative to adjacent areas. 72

Since both recirculation trains are actuated by the safety injection signal, the outside air intake flow rate during dual train operation is conservatively estimated to be 1600 cfm. If both trains are assumed to operate for one hour, the calculated thyroid dose would decrease, due to the additional iodine filtration. The calculated whole body gamma and beta skin doses would increase slightly due to the additional intake of outside air. In both cases, the calculated doses remain below the limits specified in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19. The reported control room doses are the higher values of the two cases. 66 78

66 e. During the emergency ventilation mode of operation, 3800 ft^3/min of outdoor air is used to introduce fresh air into the control room.

f. The emergency ventilation mode of operation is under administrative control so that the dose to the control room occupants is minimized, and the need for air change is satisfied.

The operating mode sequence used in this analysis is as follows:

	Time Period	Operating Mode	Time in Mode	Air Intake Flow Rate	Filtered Recirculation Flow Rate
66	0 to 96	emergency	96 hours	800	7200
66	hours(1)	recirculation		ft^3/min	ft^3/min
66					
66	96 to 117	emergency	21 hours	3800	4200
66	hours	ventilation		ft^3/min	ft^3/min
66	117 to 720	emergency	603 hours	800	7200
66	hours	recirculation		ft^3/min	ft^3/min

66 (1) Since both recirculation trains are actuated by the SI signal, one train must be turned off manually by the operators within one (1) hour.

g. The distance from the Containment to the control room air intake is 94 feet, and the air intake is located 56 feet above
78 ground. The distance from the primary plant vent stack (i.e. the ESF leakage release point) to the closest air intake is 138 ft.

h. Atmospheric dilution factors are determined from the following equation based on Reference [24]:

$$X/Q = [U(\pi \sigma_y \sigma_z + A/(K+2))]^{-1}$$ 66

where:

U = wind speed at an elevation of 10 meters (m/sec)

σ_y, σ_z = standard deviation of the gas concentration in 45
the horizontal and vertical crosswind directions, respectively, 46
both being evaluated at a distance of 94
feet for the containment leakage source and at a distance of 138 78
feet for the ESF leakage source

$$K = \frac{3}{(S/D)^{1.4}}$$

S = distance between containment surface or primary plant vent and closest control room intake 78

D = diameter of Containment for the containment leakage source and a combination of portions of the Containment and Auxiliary Building for the ESF leakage source 78

A = projected area of Containment Building (3265 m^2) for the containment leakage source and 2088 m^2 for the ESF leakage source 78

Table 15.6-12 summarizes the X/Q values calculated utilizing this expression.

78 i. The total unfiltered infiltration rate in the control room is 12 cfm, including 10 cfm due to ingress/egress and 2 cfm leakage from the ductwork passing through the control room pressure boundary. Leakage through the closed dampers due to the pressure differential is also included. The damper leakage air will be filtered by the recirculation filtration units.

66 j. Habitability of the control room is based on the following occupancy factors:

Time Period	Occupancy Factor
0 to 24 hours	1.0
1 to 4 days	0.6
4 to 30 days	0.4

66 k. The air volume in the control room used to determine exposures to operators is 423,032 ft^3.

66 l. The models for the major contributors to the control room dose are provided in Appendix 15B.

78 Using the above assumptions and procedures, the thyroid dose is conservatively calculated to be 27.4 rem in the control room for the duration of the accident. The thyroid dose can be further
46 reduced by the use of the fullface respirators which are available at all times in the control room. Air packs are provided in the control room and emergency control center for the use of operators leaving the control room either to go offsite, or to some control
66 point in the station, or to the control room from offsite. The use of air packs reduces the thyroid dose considerably during such
78 movements. The total whole body gamma dose is conservatively calculated to be 3.76 rem. This calculated dose includes whole
66 body dose contributions from containment sources (both direct and scattered

radiation), the external passing cloud, control room atmosphere, 66
activity buildup on filters, and streaming through doors and penetrations. These calculated doses are less than the limiting values specified in 10CFR50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19.

The skin dose received in the control room during the accident 78
period is conservatively calculated to be 44.0 rem. This
calculated beta skin dose is less than the 75 rem limit allowed if 66
special protective clothing and eye protection are used. Therefore, special protective clothing and eye protection are provided for use, if required, to reduce the beta skin doses to the operators to within acceptable limits in accordance with 10CFR50, Appendix A, GDC 19.

5. Environmental consequences of containment purging to control 66
containment hydrogen concentration after a LOCA

Purging of the containment atmosphere provides a backup method for 66
controlling potential hydrogen accumulation in the Containment following a postulated LOCA. The use of the hydrogen purge system (see Section 6.2.5.2.2) is precluded by redundant electric hydrogen recombiners located in the Containment Building (see Section 6.2.5.2.1). The electric hydrogen recombiners are the primary means of controlling post- LOCA hydrogen buildup. Thus, an analysis of the radiological consequences of containment purging is not provided.

Q022.8
6. Environmental consequences of releases through the containment 2
pressure relief line in the event of a LOCA

Q022.8
An analysis of the radioactive effluents escaping the Containment 72
to the environment after a LOCA, via the line through the controlled access area exhaust system, was performed using the following assumptions:

Q022.20
2 a. The maximum containment air/steam mass release to the environment assuming critical flow was calculated as 5,427 lb_m.

b. Only reactor coolant activity is assumed to be released and the largest release occurs for a 3 inch break. Hence, only the 3 inch break is analyzed.

66 c. A preaccident iodine spike was considered in determining the primary reactor coolant activity. The corresponding reactor coolant iodine concentrations are listed in Table 15.6-3. The noble gas activity concentrations are presented in Table 15.1-4.

Q312.21
5 d. The containment pressure relief line isolation valve closure time including instrumentation delays will not exceed 5 seconds. The radioactive fission products are assumed to be released from the Containment through the pressure relief line for a period of 38.1 seconds. This includes 33.1 seconds to initiate the low pressurizer pressure trip setpoints (see Table 15.6-1) corresponding to the 3 inch line break conditions.

e. No credit was taken for radioactive decay.

f. No credit was taken for an elevated release.

Based on the foregoing assumptions, the doses to the thyroid and whole body were conservatively calculated to be 2.03 rem and
72 1.61×10^{-3} rem, respectively, at the exclusion area boundary. The doses from this accident are well within the values set forth in 10CFR100.

15.6.6 A NUMBER OF BWR TRANSIENTS

This section is not applicable to the CPSES.

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TABLE 15.6-1
(Sheet 1 of 6)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A
DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec)		
		Unit 1	Unit 2	84
Inadvertent opening of a	Safety valve			
pressurizer safety valve	opens fully	0.0	0.0	84
	Overtemperature N-16			5
	reactor trip setpoint			5
	reached	13.3	32.1	84
	Rods begin to drop	14.3	34.1	84
	Minimum DNBR occurs	14.8	34.6	84
				5
Steam generator tube				70
rupture	Start SGTR, Loop 4	5.0	5.0	DRAFT
	Reactor trip, turbine			70
	trip, loss of offsite			70
	power, AFW initiation	405	405	DRAFT
				78
	Close MSIV, Loop 4	785	785	DRAFT
	Isolate AFW, Loop 4	905	905	DRAFT

TABLE 15.6-1
(Sheet 2)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec) Unit 1	Unit 2	
		Unit 1	Unit 2	DRAFT
	Begin RCS cooldown	1205	1205	DRAFT
	End RCS cooldown	1632	1632	DRAFT
	Begin RCS depressurization	1752	1752	DRAFT
	End RCS depressurization	1907	1907	DRAFT
	Terminate ECCS flow	1967	1967	DRAFT
Large break LOCA				5
1. DECLG C_D = 1.0	Start	0.0	-	
	Reactor trip signal	0.793	-	6
	Safety injection signal	1.05	-	6
	Accumulator injection			6
	begins	12.8	-	6
	End-of-bypass	23.49	-	6
	End-of-blowdown	24.73	-	6
	Pump injection begins	26.05	-	6
	Bottom of core recovery	36.83	-	6
	Accumulator empty	47.02	-	6
2. DECLG C_D = 0.4	Start	0.0	0.0	DRAFT
	Reactor trip signal	0.841	0.53	DRAFT
	Safety injection signal	1.65	1.62	DRAFT
	Accumulator injection			6
	begins	20.4	19.6	DRAFT

TABLE 15.6-1

(Sheet 3)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec) Unit 1	Unit 2	
				DRAFT
	Pump injection begins	26.65	26.62	DRAFT
	End-of-bypass	33.65	35.73	DRAFT
	End-of-blowdown	37.66	35.73	DRAFT
	Bottom of core recovery	47.68	48.25	DRAFT
	Accumulator empty	56.91	54.08	DRAFT
3. DECLG C_D = 0.6	Start	0.0	0.0	DRAFT
	Reactor trip signal	0.815	0.52	DRAFT
	Safety injection signal	1.32	1.3	DRAFT
	Accumulator injection			6
	begins	15.6	15.2	DRAFT
	End-of-bypass	25.52	28.69	DRAFT
	End-of-blowdown	28.65	28.69	DRAFT
	Pump injection begins	26.32	26.3	DRAFT
	Bottom of core recovery	39.07	40.7	DRAFT
	Accumulator empty	49.88	47.8	DRAFT
DECLG C_D = 0.6	Start	-	0.0	DRAFT
(Maximum SI)	Reactor trip signal	-	0.63	DRAFT
	Safety injection signal	-	1.3	DRAFT
	Accumulator injection			DRAFT
	begins	-	15.0	DRAFT
	End-of-bypass	-	28.69	DRAFT
	End-of-blowdown	-	28.69	DRAFT
	Pump injection begins	-	26.3	DRAFT

TABLE 15.6-1

(Sheet 4)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec) Unit 1	Time (sec) Unit 2	
				DRAFT
	Bottom of core recovery	-	40.4	DRAFT
	Accumulator empty	-	48.3	DRAFT
4. DECLG C_D = 0.8	Start	0.0	0.0	DRAFT
	Reactor trip signal	0.802	0.51	DRAFT
	Safety injection signal	1.15	1.12	DRAFT
	Accumulator injection			6
	begins	13.5	12.6	DRAFT
	End-of-bypass	23.72	24.96	DRAFT
	End-of-blowdown	25.43	24.96	
	Pump injection begins	26.15	26.1	DRAFT
	Bottom of core recovery	37.03	36.5	DRAFT
	Accumulator empty	47.65	44.7	DRAFT
Small break LOCA				6
				6
1. 2 inch	Start	N/A	0.0	DRAFT
	Reactor trip signal	N/A	62.9	DRAFT
	Safety injection signal	N/A	73.9	DRAFT
	Top of core uncovered	N/A	2381.2	DRAFT
	Accumulator injection			DRAFT
	begins	N/A	N/A	DRAFT
	Peak clad temperature			DRAFT
	occurs	N/A	4062.6	DRAFT
	Top of core covered	N/A	5512.5	DRAFT

TABLE 15.6-1

(Sheet 5)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec) Unit 1	Unit 2	
		Unit 1	Unit 2	DRAFT
2. 3 inch	Start	0.0	0.0	DRAFT
	Reactor trip signal	33.1	21.6	DRAFT
	Top of core uncovered	622.7	990.5	DRAFT
	Accumulator injection begins	2594.2	1999.8	DRAFT
	Peak clad temperature occurs	1363.9	1841.8	DRAFT
	Top of core covered	2308.6	3263.9	DRAFT
3. 4 inch	Start	0.0	0.0	DRAFT
	Reactor trip signal	20.8	12.7	DRAFT
	Top of core uncovered	325	623.5	DRAFT
	Accumulator injection begins	795	887.6	DRAFT
	Peak clad temperature occurs	836.2	948.0	DRAFT
	Top of core covered	848	1342.2	DRAFT

TABLE 15.6-1
(Sheet 6)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (sec) Unit 1	Unit 2	
				DRAFT
4. 6 inch	Start	0.0	N/A	DRAFT
	Reactor trip signal	13.1	N/A	DRAFT
	Top of core uncovered	133	N/A	DRAFT
	Accumulator injection begins	335	N/A	DRAFT
	Peak clad temperature occurs	212.6	N/A	DRAFT
	Top of core covered	367.7	N/A	DRAFT

TABLE 15.6-5

INPUT PARAMETERS USED IN THE LOCA ANALYSIS

76

	UNIT 1 LARGE BREAK	UNIT 1 SMALL BREAK	UNIT 2 LARGE BREAK	UNIT 2 SMALL BREAK
Licensed core power (MW_t)	3411[a]	3411[a]	3411[a]	[illegible][a]
Reactor Coolant Pump Heat (MW)	0	0[b]	0[b]	0[c]
Peak linear power, includes 102% factor (kW/ft)	12.88	11.66	12.63	12.88
Total peaking factor, F^T_Q	2.32	2.32	2.32	2.32
Power Shape	Chopped cosine	See Figure 15.6-48	Chopped Cosine	See Figure 15.6-69
Fuel assembly array	17 x 17	17 x 17	Optimized 17 x 17	Optimized 17 x 17
Accumulator water volume, nominal (ft^3/accumulator)	850	950	850	850
Accumulator tank volume, nominal (ft^3/accumulator)	1350	1350	1350	1350
Accumulator gas pressure, minimum (psia)	600	600	600	600
Safety injection pumped flow	See Figure 15.6-47A[d]	See Figure 15.6-47B[d]	See Figure 15.6-47A[d]	See Figure 15.6-47B[d]
Containment parameters	See Section 6.2	NA	See Section 6.2	NA
Initial loop flow (lb/sec)	9902	8743	9881	9868
Vessel inlet temperature (°F)	556.5	564.8	558.3	564.1
Vessel outlet temperature (°F)	617.2	627.6	616.7	623.3
Reactor coolant pressure (psia)	2250	2280	2280	2280
Steam pressure (psia)	965	1000	994.7	1000.0
Steam generator tube plugging level (%)	0[d]	0[d]	0[d]	.5

[a] Two percent is added to this power to account for calorimetric error.

[b] Pump heat is assumed to be generated in the core.

[c] Evaluated at 1% SGTP.

[d] Evaluated pump flow.

76
78
DRAFT
DRAFT

TABLE 15.6-6

(Sheet 2)

LARGE BREAK - ANALYSIS INPUT AND RESULTS

Results, Unit 1	DECLG, C_D = 0.8 Minimum SI	DECLG, C_D = 0.6 Minimum SI	DECLG, C_D = 0.5 Maximum SI	DECLG, C_D = 0.4 Minimum SI	
Peak clad temperature (^{o}F)	1775	1804	1808	1769	DRAFT
Location (feet)	7.5	7.5	7.5	7.5	DRAFT
Maximum local clad/water reaction (%)	1.97	1.92	2.04	1.85	DRAFT DRAFT
Location (feet)	7.25	7.5	7.5	7.25	DRAFT
Total core clad/water reaction (%)	<0.3	<0.3	<0.3	<0.3	DRAFT DRAFT
Hot rod burst time (seconds)	75.8	72.4	74.8	95.8	DRAFT
Location (feet)	7.25	6.75	7.0	7.25	DRAFT

DRAFT

TABLE 15.6-6

(Sheet 1 of 2)

LARGE BREAK - ANALYSIS INPUT AND RESULTS

Quantities in the Calculations:		Q212.134
Licensed core power rating	102% of 3411 MWt	6
Total core peaking factor	2.32	6
Peak linear power	102% of 12.63 KW/ft for Unit 1,	6
	102% of 12.62 KW/ft for Unit 2	DRAFT
Accumulator water volume	850 cubic feet per tank	DRAFT
Accumulator pressure	600 psia	6
Number of Safety Injection Pumps Operating	3	6
Steam Generator Tube Plugging Level	0 percent	6
Fuel Parameters - Cycle 1, Region 1		78
		76

Results, Unit 1	DECLG, C_D = 1.0	DECLG, C_D = 0.8	DECLG, C_D = 0.6	DECLG, C_D = 0.4	6
Peak clad temperature (°F)	1969.2	1945.2	2010.7	1577.8	6
Location (feet)	7.5	7.5	7.5	7.5	6
Maximum local clad/water reaction (%)	3.4	3.08	3.92	0.72	6
Location (feet)	7.5	7.5	7.5	7.5	6
					6
Total core clad/water reaction (%)	0.3	0.2	0.3	0.3	6
					6
Hot rod burst time (seconds)	26.2	29.0	27.5	NA	6
Location (feet)	6.0	6.0	6.0	NA	6
					78
					76

DRAFT

TABLE 15.6-7

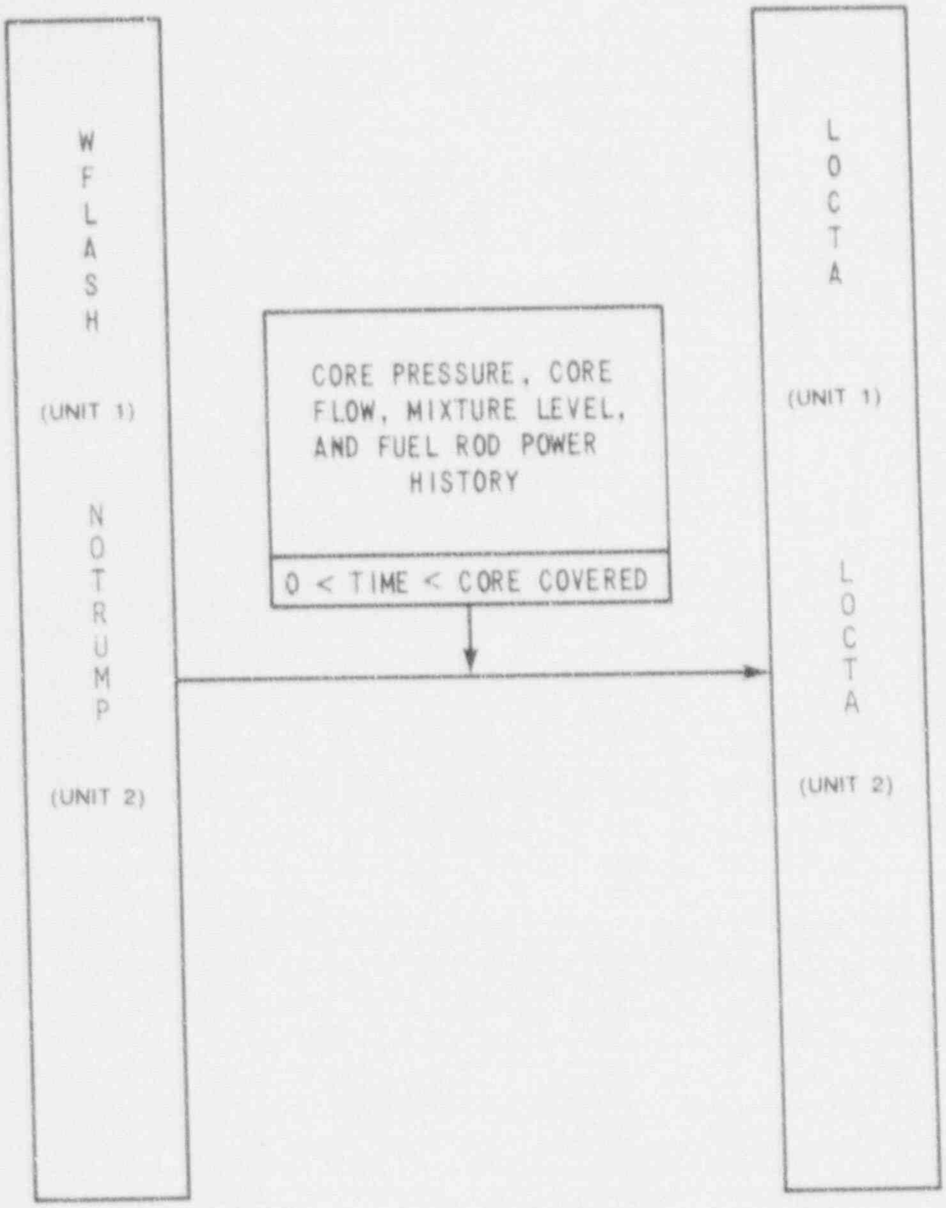
SMALL BREAK LOCA RESULTS

FUEL CLADDING DATA

	Pipe Break Size			
	3 Inch	4 Inch	6 Inch	
Unit 1 Results				DRAFT
Peak clad temperature (°F)	1499.2	1787.5	1343.5	6
Peak clad temperature location (ft)	11.25	11.50	11.0	6
Local Zr/H_2O reaction, maximum (%)	0.683	3.21	0.382	6
Local Zr/H_2O location (ft)	11.25	11.25	11.0	6
Total Zr/H_2O reaction (%)	<0.3	<0.3	<0.3	6
Hot rod burst time (sec)	N/A	758.1	N/A	6
Hot rod burst location (ft)	N/A	11.25	N/A	

	Pipe Break Size			DRAFT
	2 Inch	3 Inch	4 Inch	DRAFT
Unit 2 Results				DRAFT
Peak clad temperature (°F)	1005.3	1433.8	1290.9	DRAFT
Peak clad temperature location (ft)	11.5	11.75	11.5	DRAFT
Local Zr/H_2O reaction, maximum (%)	0.05	0.60	0.11	DRAFT
Local Zr/H_2O location (ft)	11.5	11.75	11.5	DRAFT
Total Zr/H_2O reaction (%)	<1.0	<1.0	<1.0	DRAFT
Hot rod burst time (sec)	N/A	N/A	N/A	DRAFT
Hot rod burst location (ft)	N/A	N/A	N/A	DRAFT

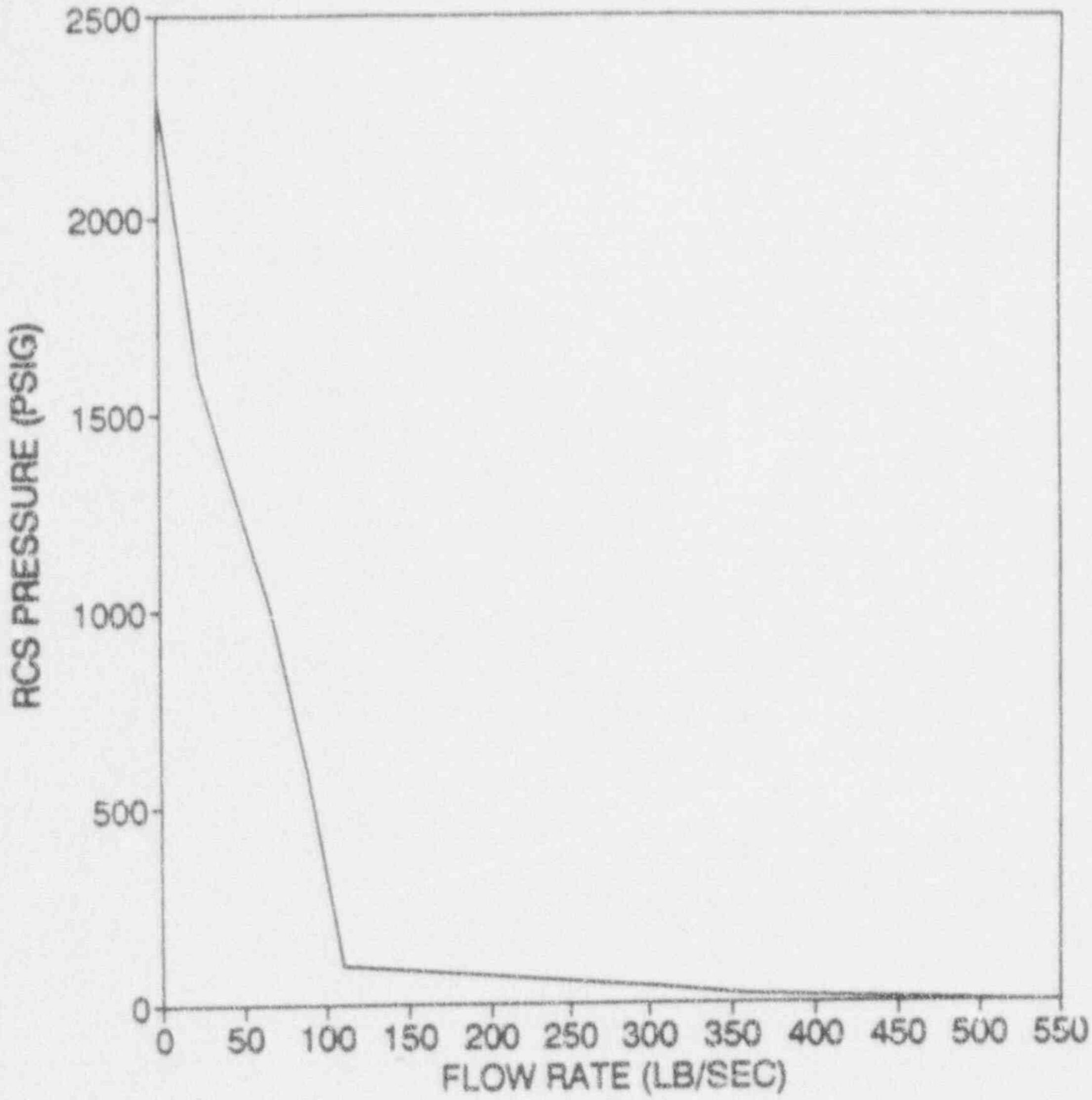
11,946-3



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Code Interface Description for Small Break Model

FIGURE 15.6-6

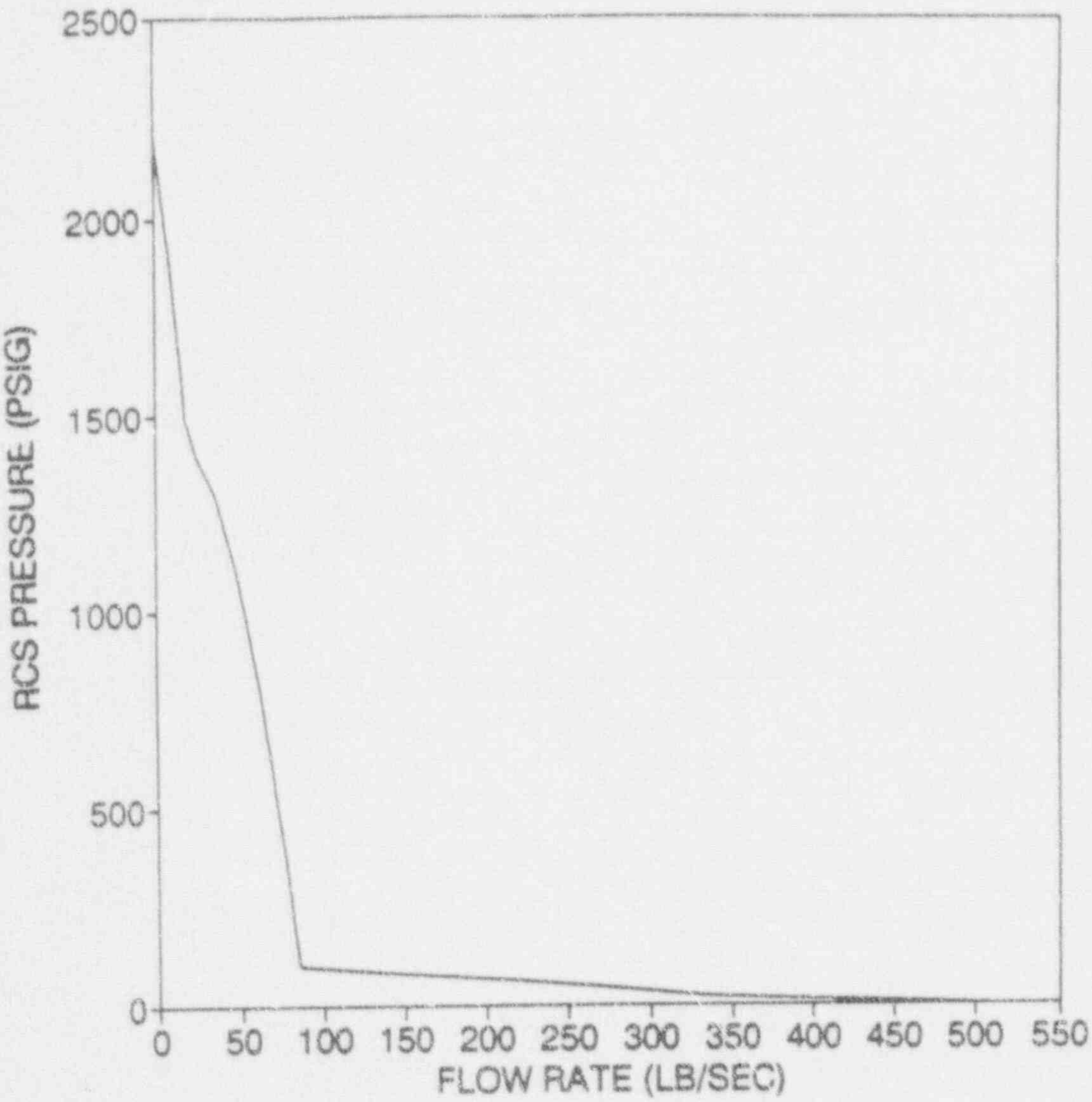


COMANCHE PEAK S.E.S
FINAL SAFETY ANALYSIS REPORT
UNITS 1 and 2

Large Break Safety Injection Flow Rate

Figure 15.6-47A

007 30824



COMANCHE PEAK S.E.S
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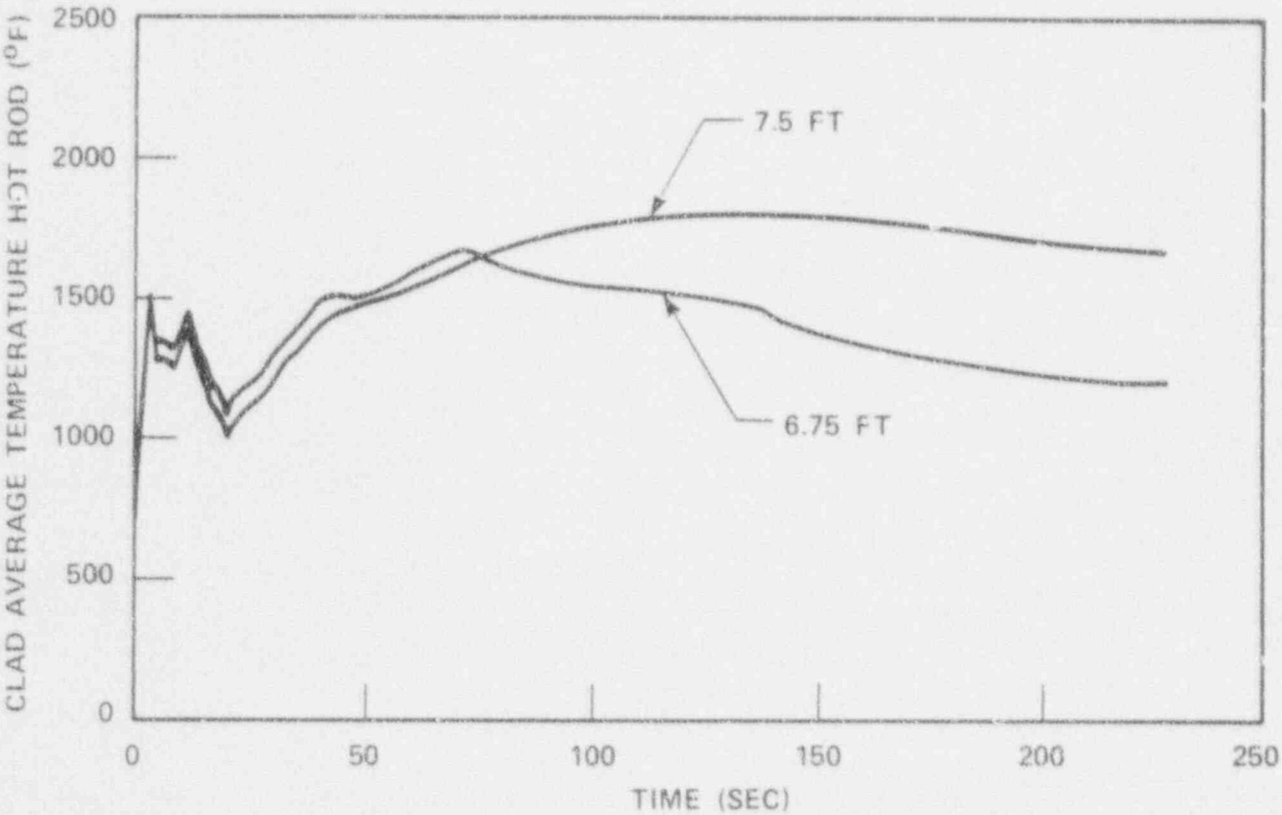
UNITS 1 and 2

Small Break Safety Injection Flow Rate

Figure 15.6-47B

006 30824



COMANCHE PEAK S.E.S.
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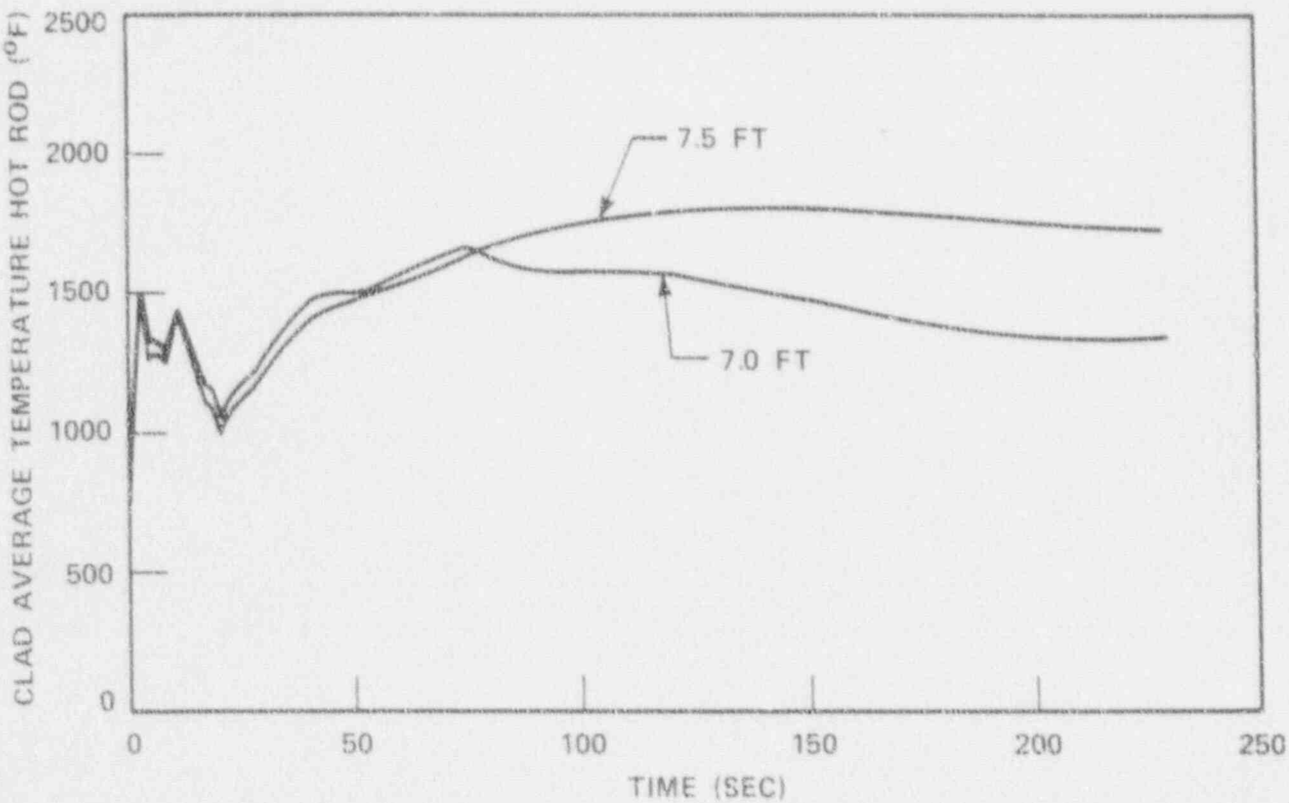
UNIT 2

Peak Clad Temperature
DECLG (CD = 0.6) Min SI

FIGURE 15.6-49

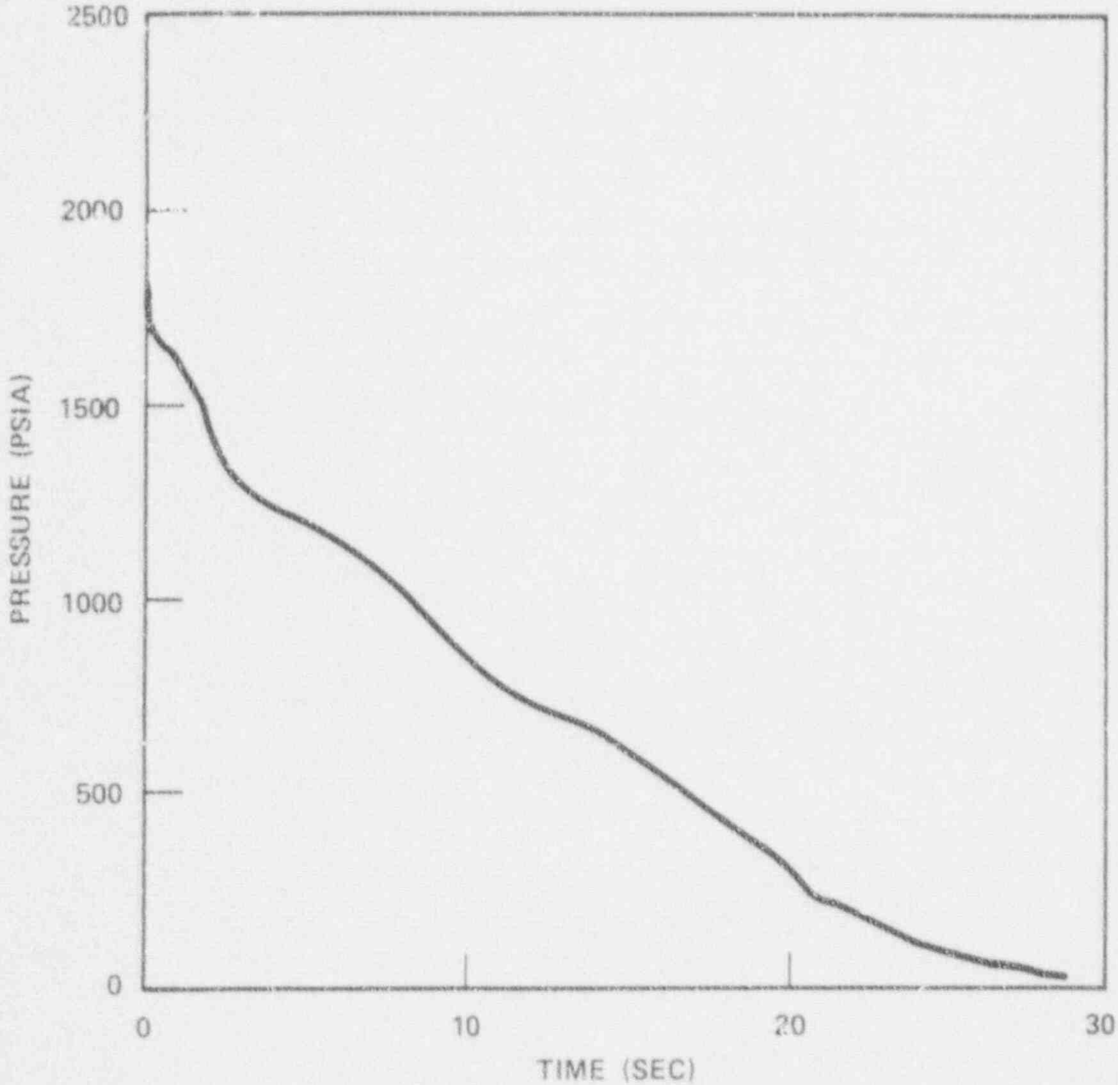
10457-6



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Peak Clad Temperature
DECLG (CD = 0.6) Max SI

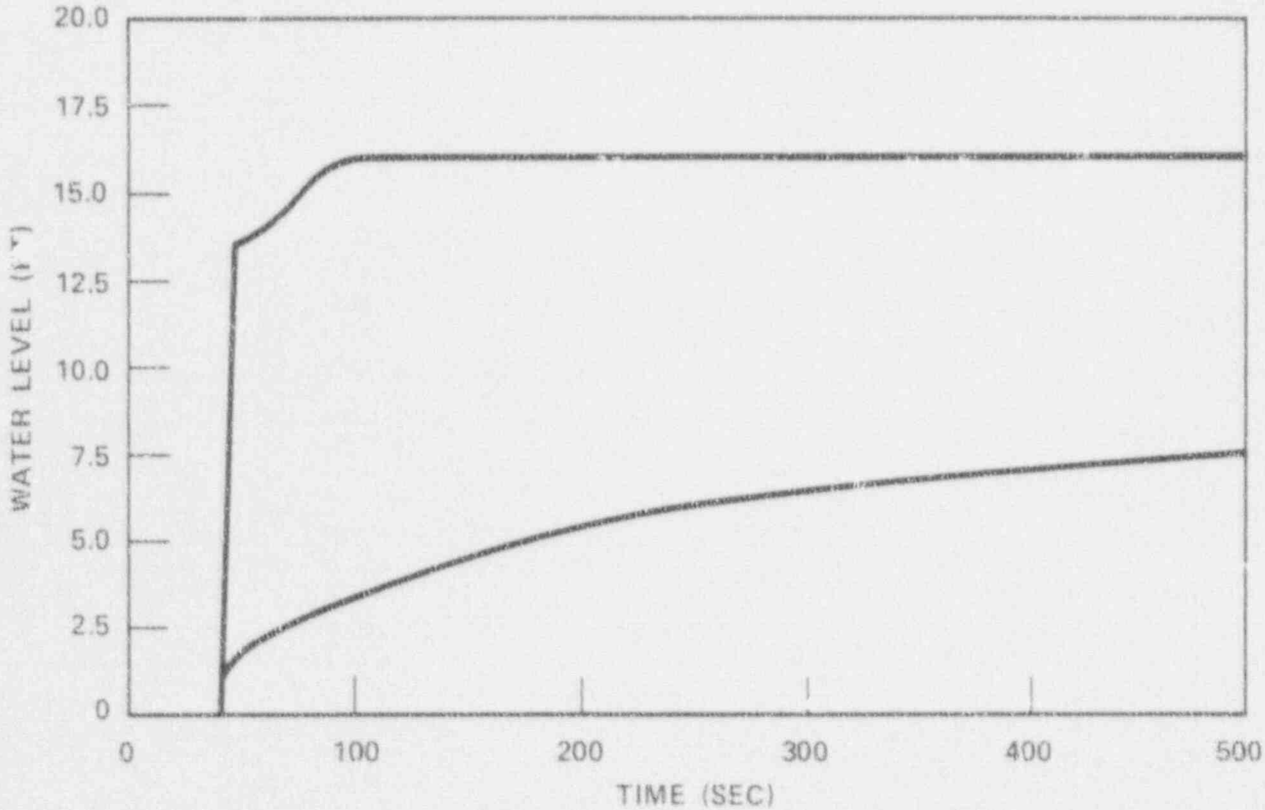
FIGURE 15.6-49A



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Pressure
DECLG (CD = 0.6)

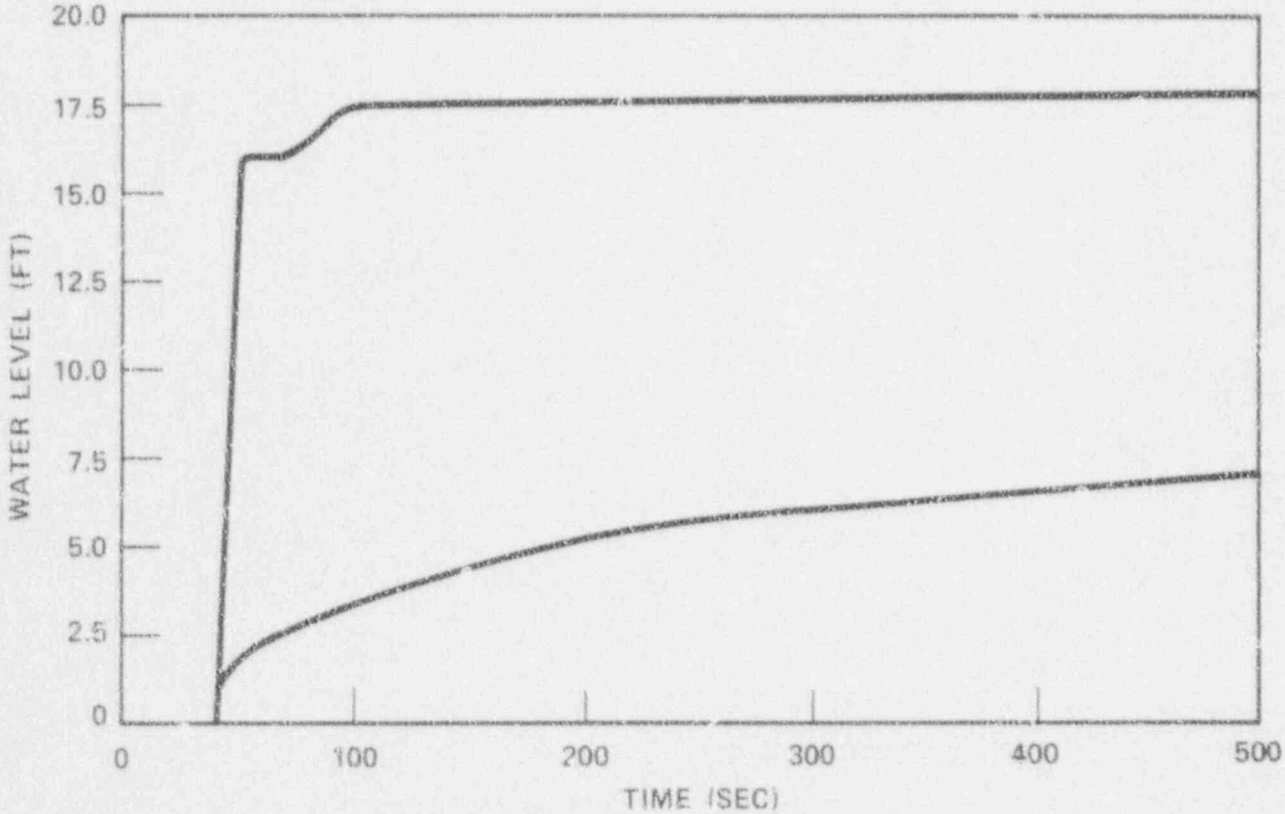
FIGURE 15.6-50



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient – Core & Downcomer Water Levels DECLD (CD = 0.6) Min SI

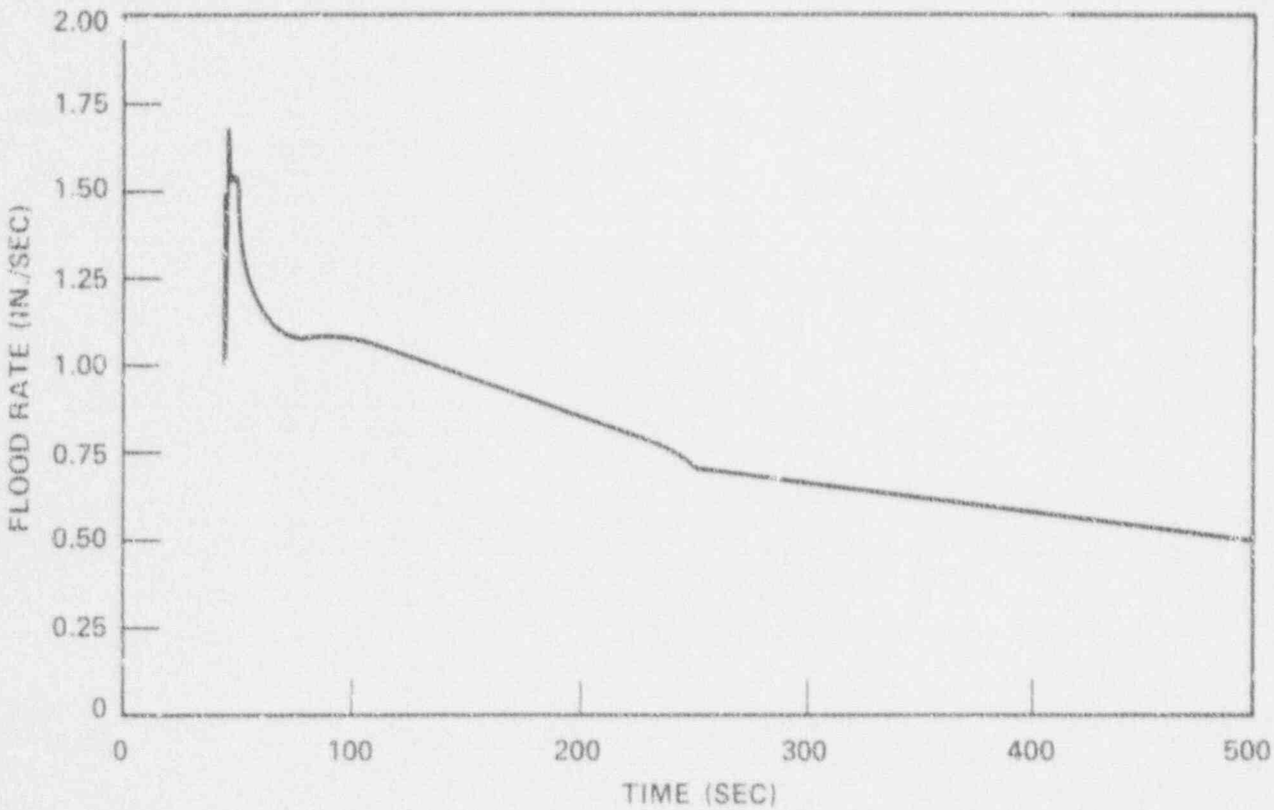
FIGURE 15.6-51



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient – Core & Downcomer Water Levels DECLD (CD = 0.6) Max SI

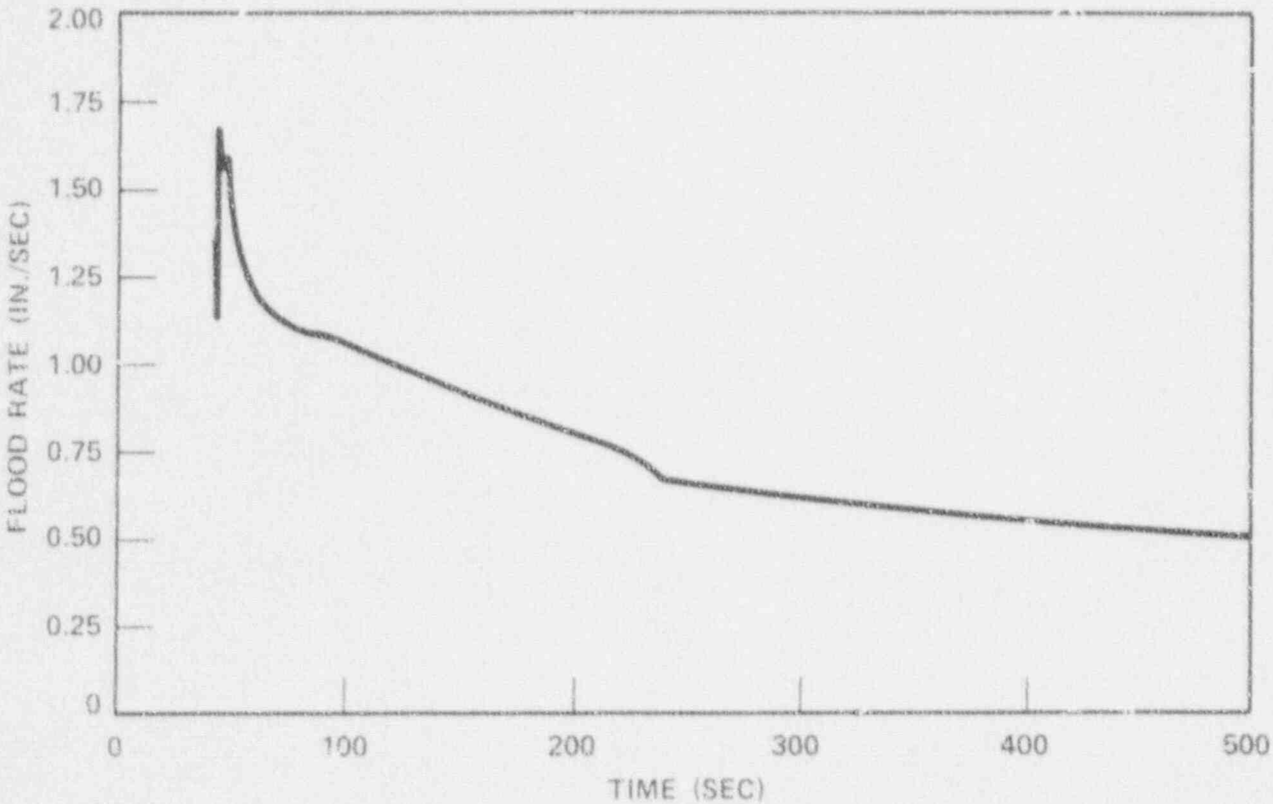
FIGURE 15.6-51A



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient Core Inlet Velocity
DECLG (CD = 0.6) Min SI

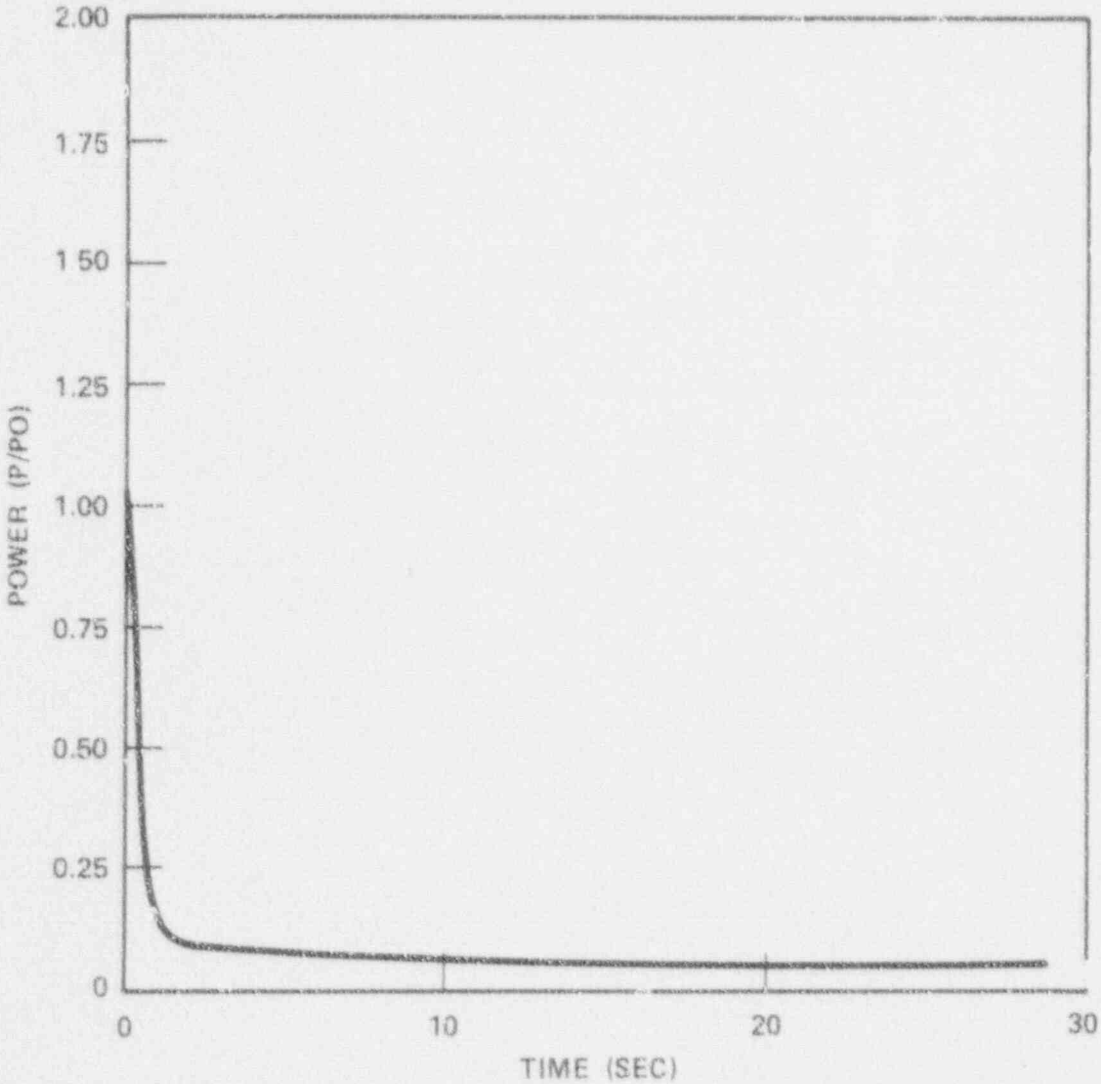
FIGURE 15.6-52



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient Core Inlet Velocity
DECLG (CD = 0.6) Max SI

FIGURE 15.6-52A

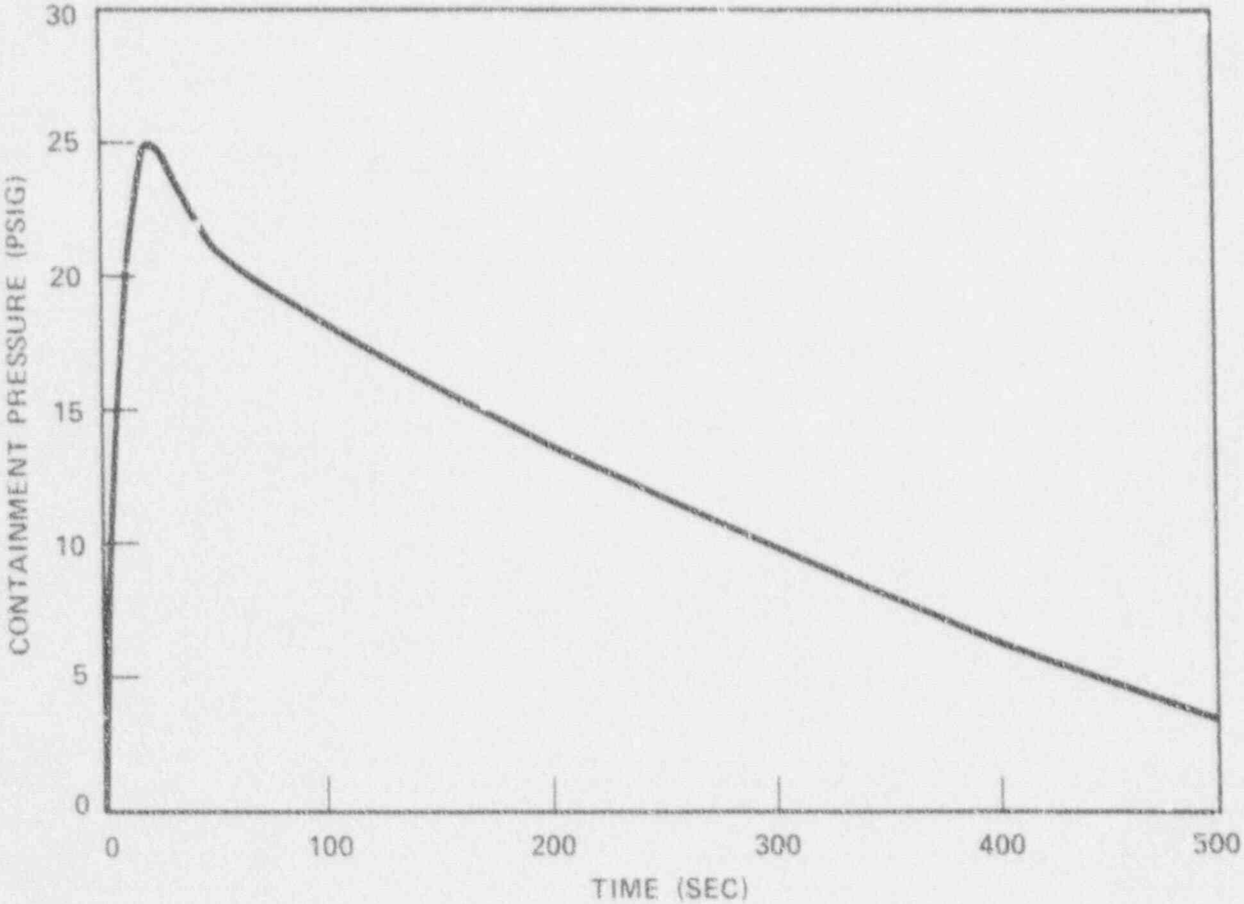


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Power Transient
DECLG (CD = 0.6)

FIGURE 15.6-53

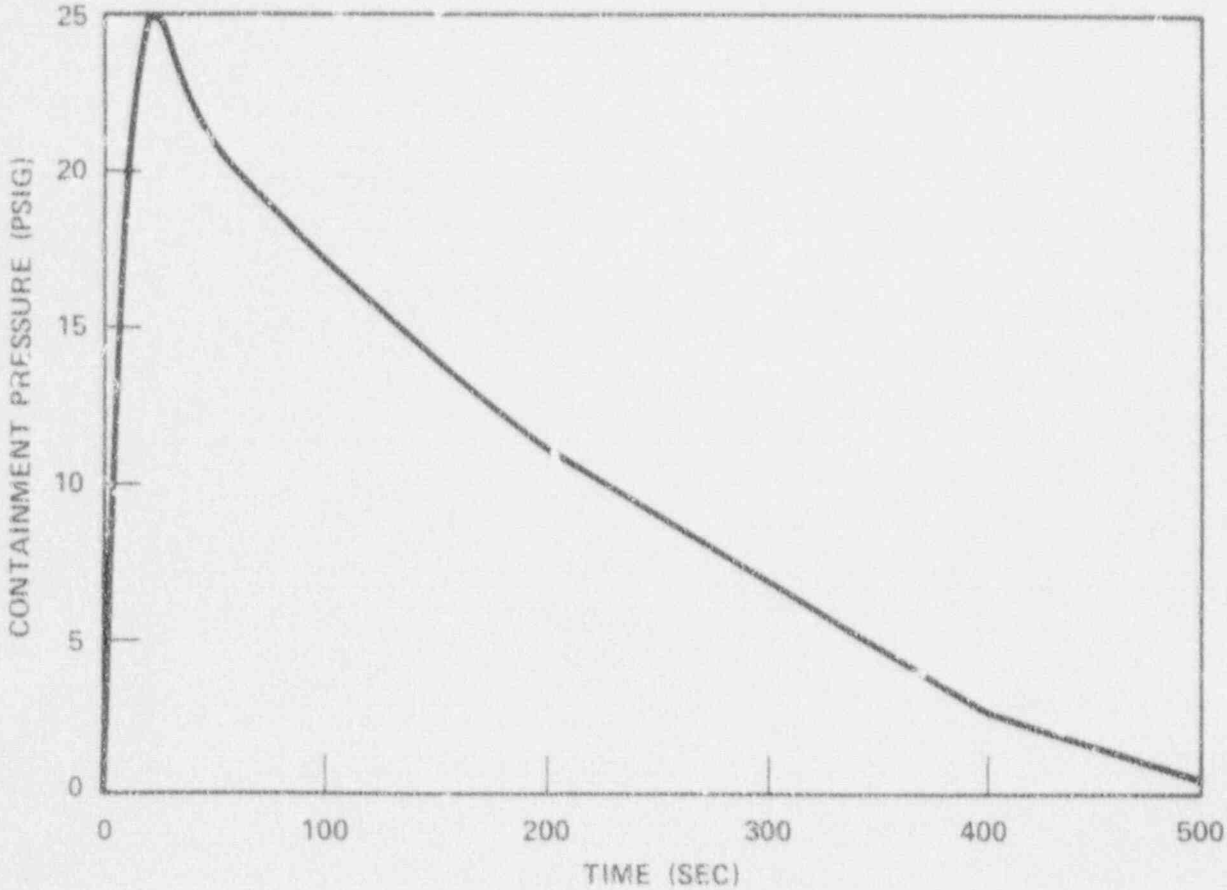
10457-12



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Containment Pressure
DECLG (CD = 0.6) Min SI

FIGURE 15.6-54

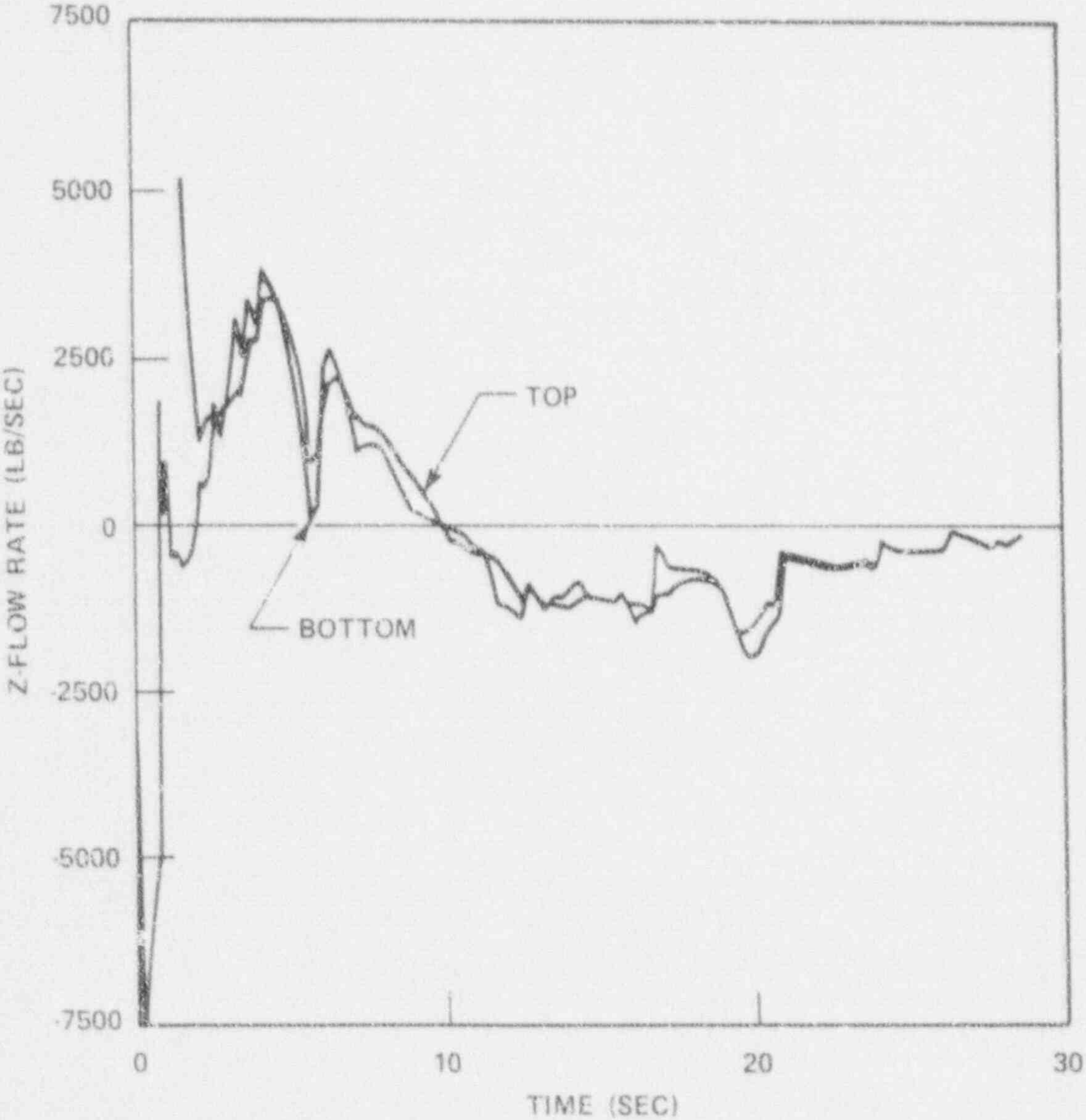


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Containment Pressure
DECLG (CD = 0.6) Max SI

FIGURE 15.6-54A

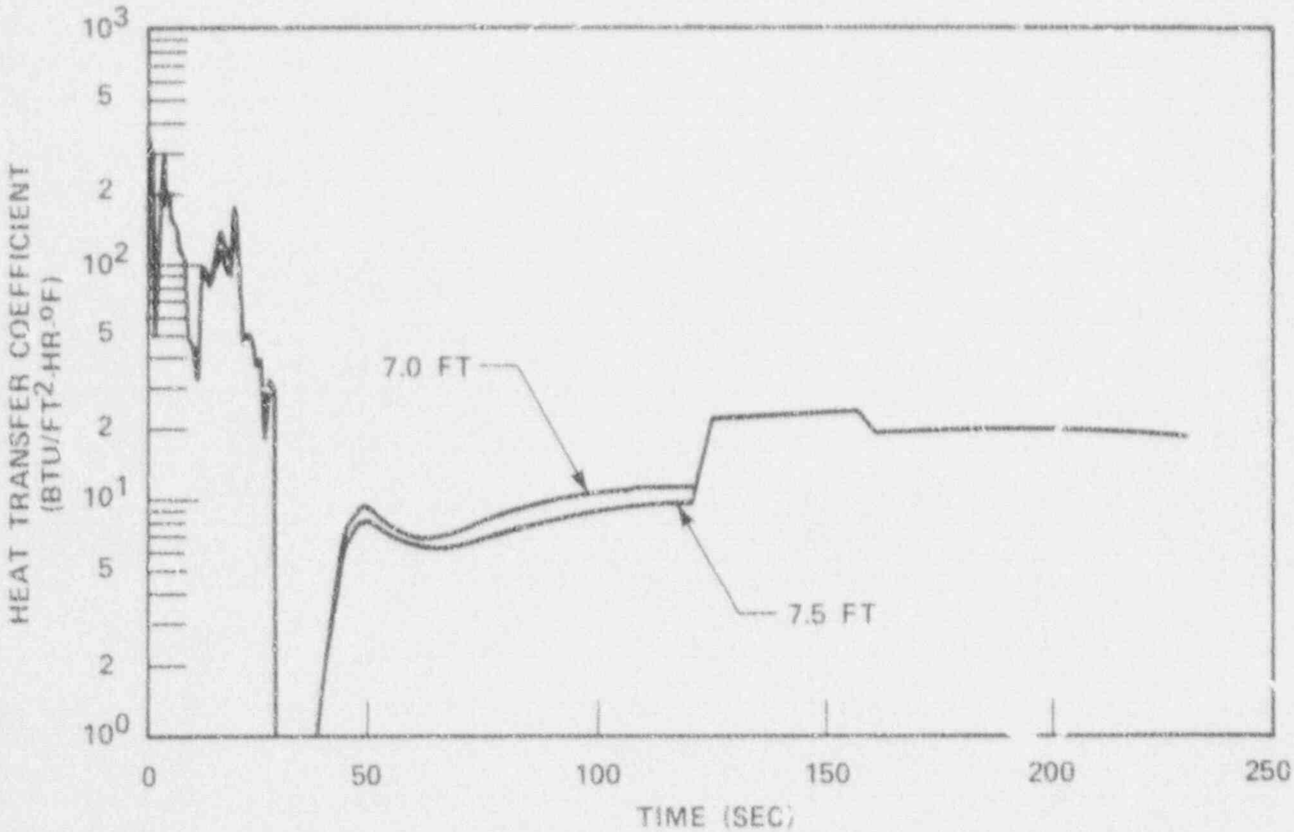
10457-14



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Flow (Top and Bottom)
DECLG (CD = 0.6) Max SI

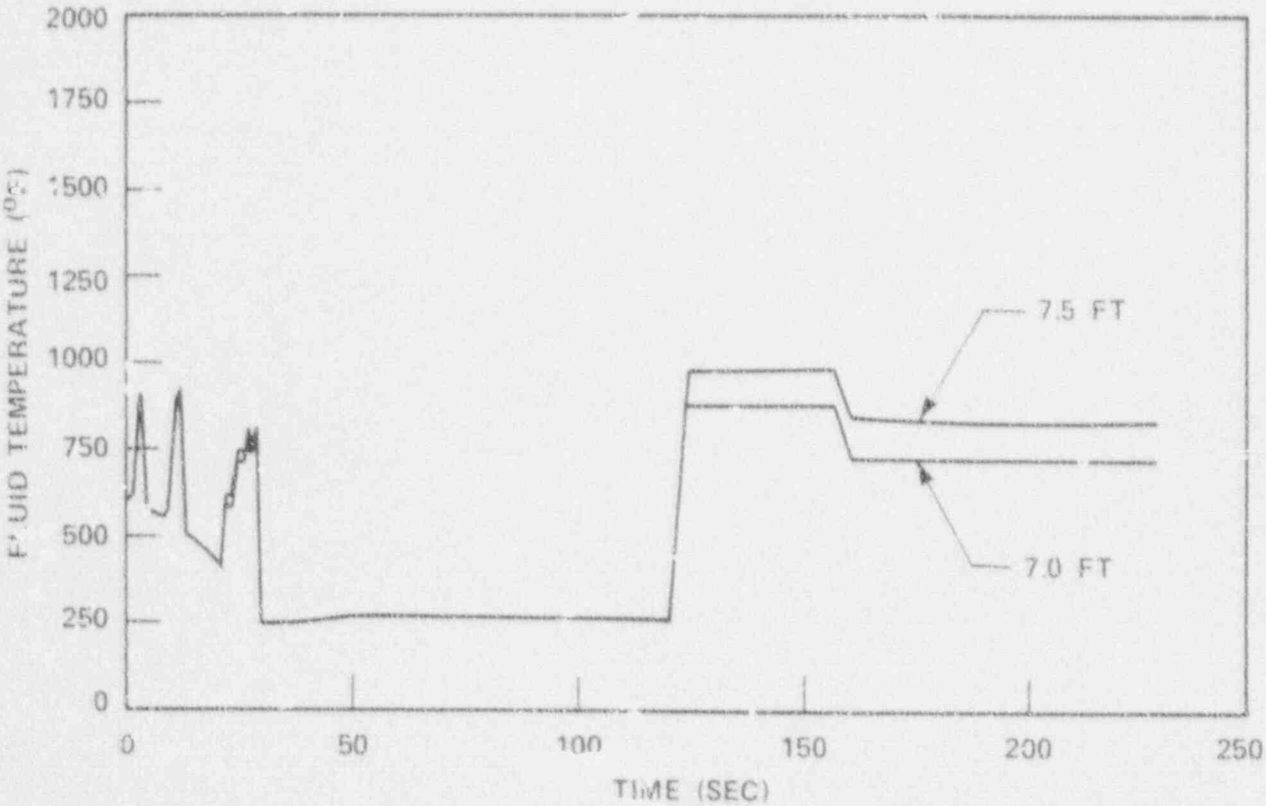
FIGURE 15.6-55



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Heat Transfer Coefficient
DECLG (CD = 0.6) Max SI

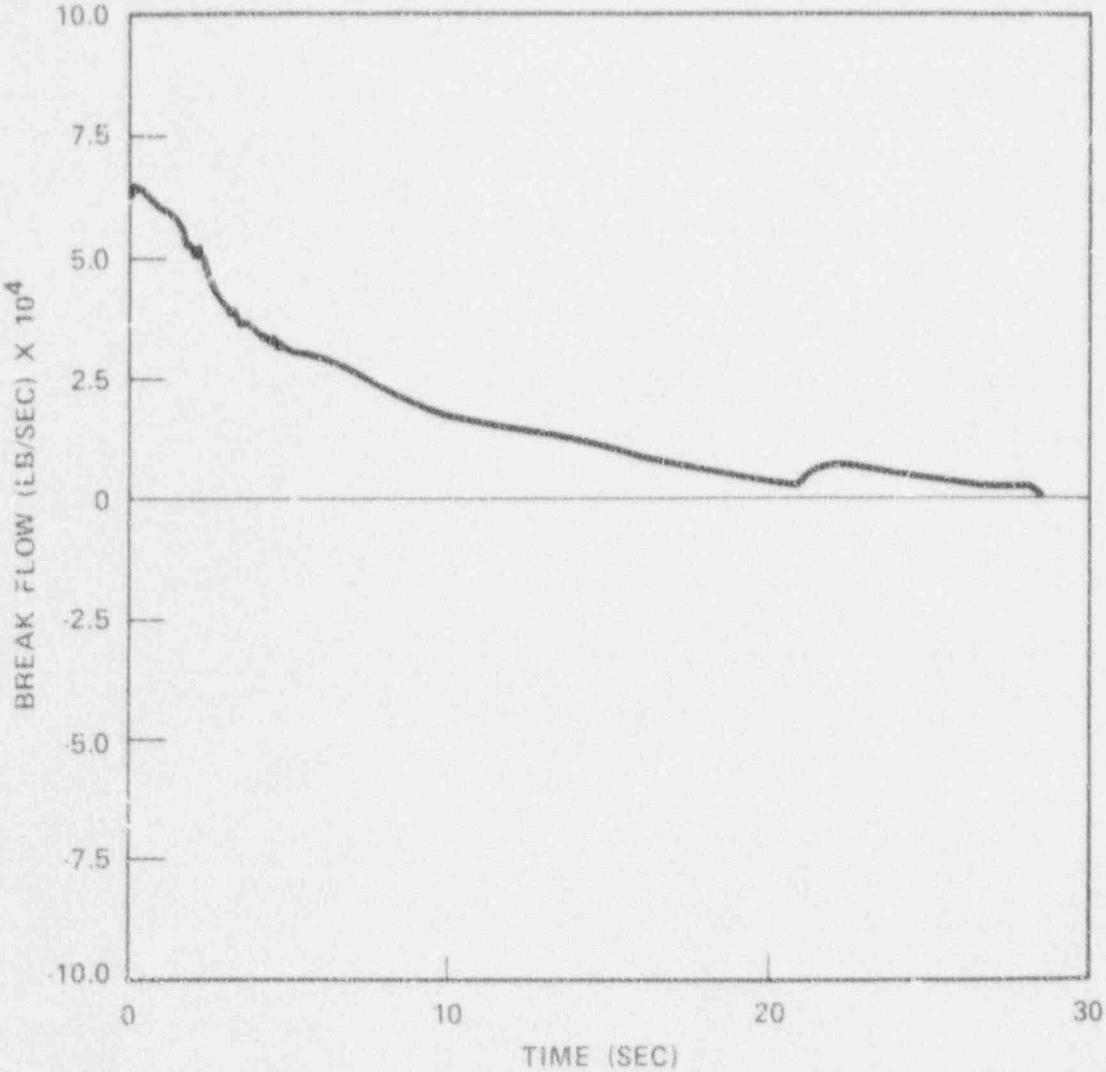
FIGURE 15.6-56



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Fluid Temperature
DECLG (CD = 0.6) Max SI

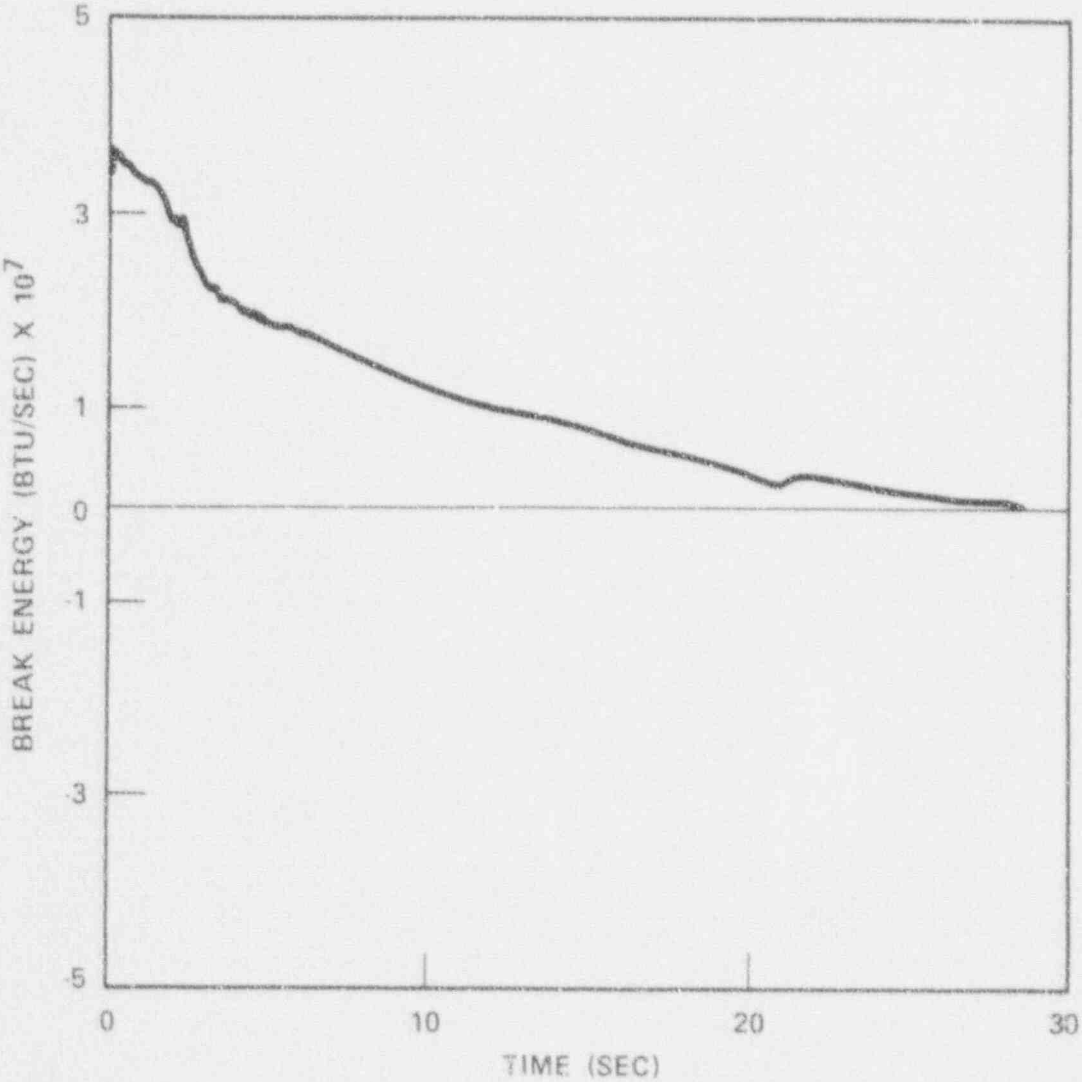
FIGURE 15.6-57



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Break Flow Rate
DECLG (CD = 0.6) Max SI

FIGURE 15.6-58

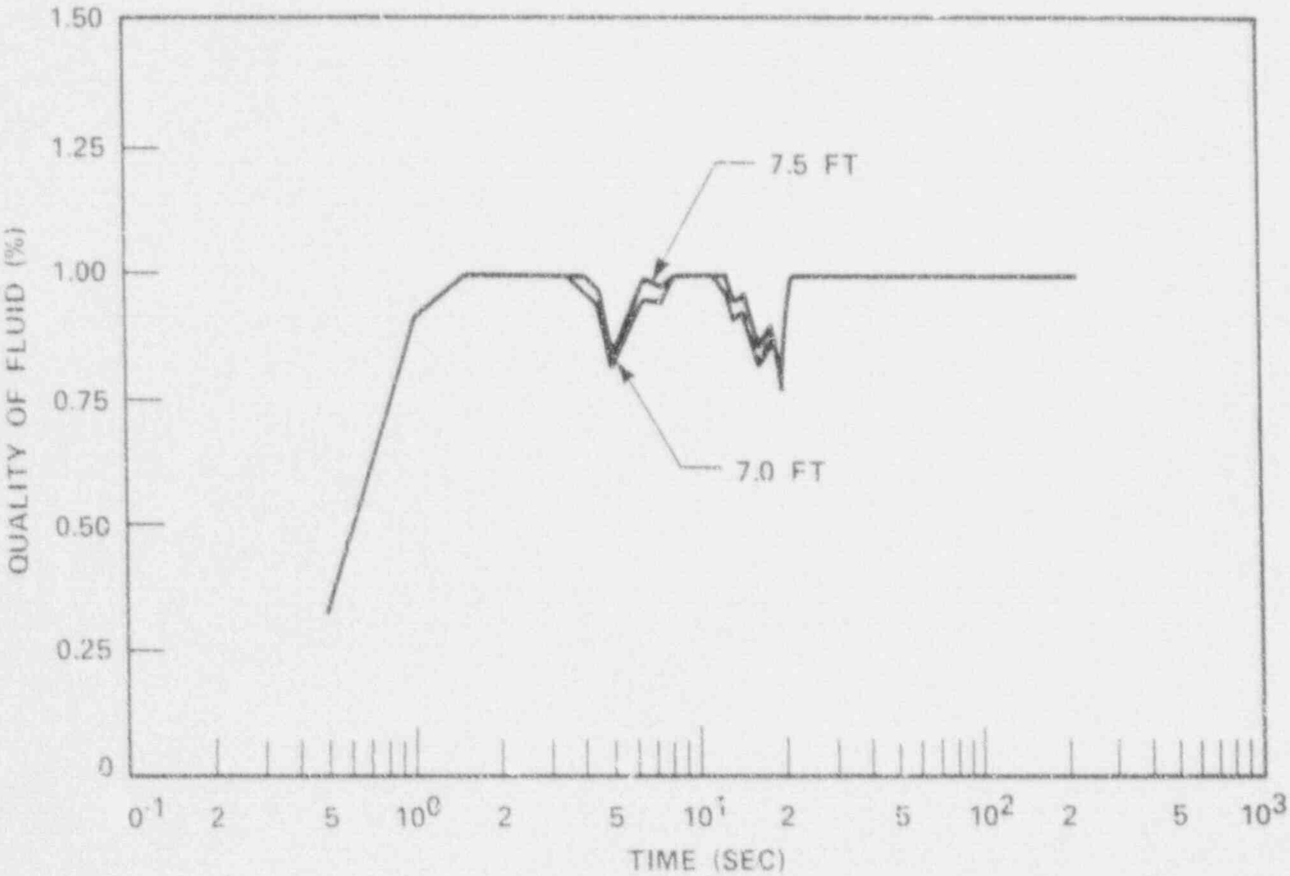


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Break Energy Released to Containment
(CD = 0.6) Max SI

FIGURE 15.6-59

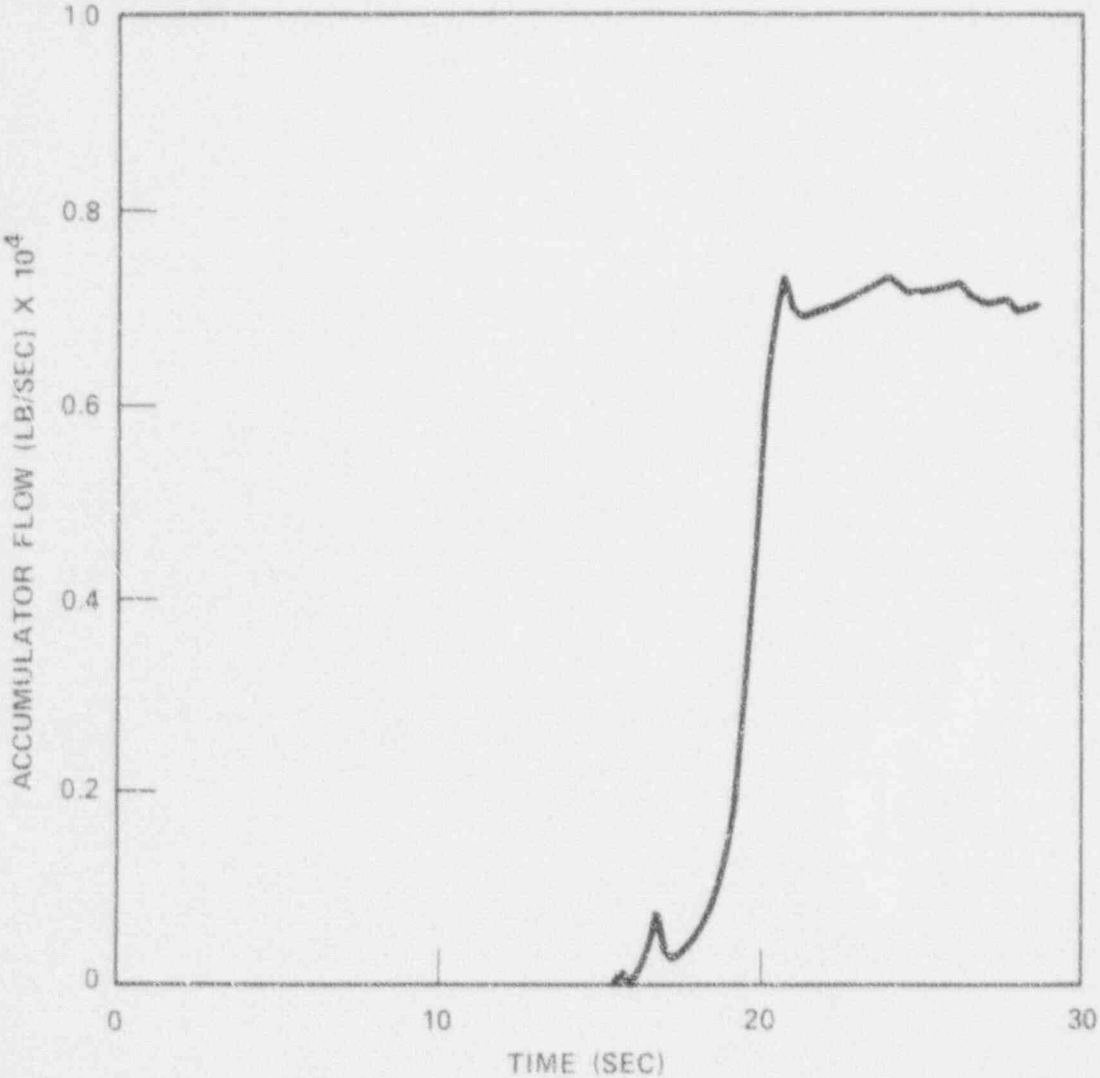
10457-19



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Fluid Quality
DECLG (CD = 0.6) Max SI

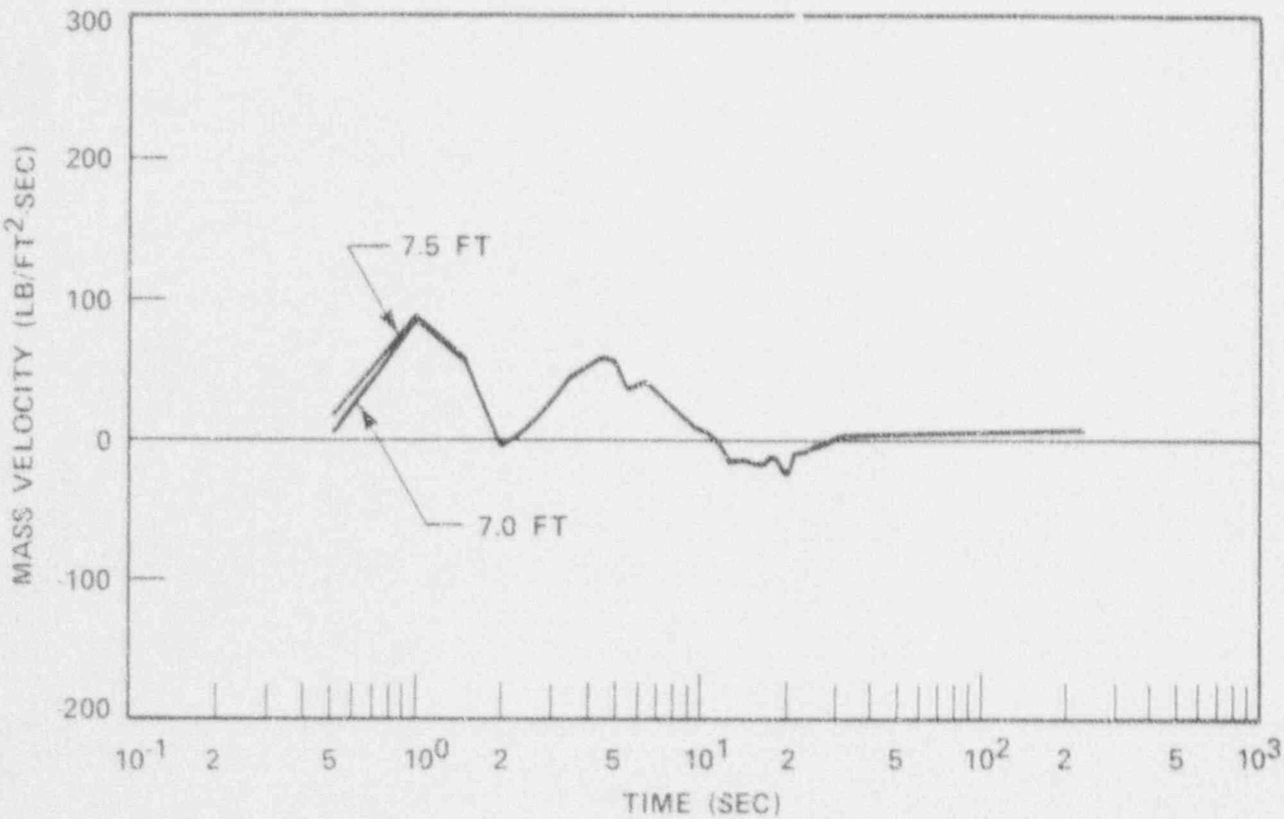
FIGURE 15.6-60



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Accumulator Flow (Blowdown)
DECLG (CD = 0.6) Max SI

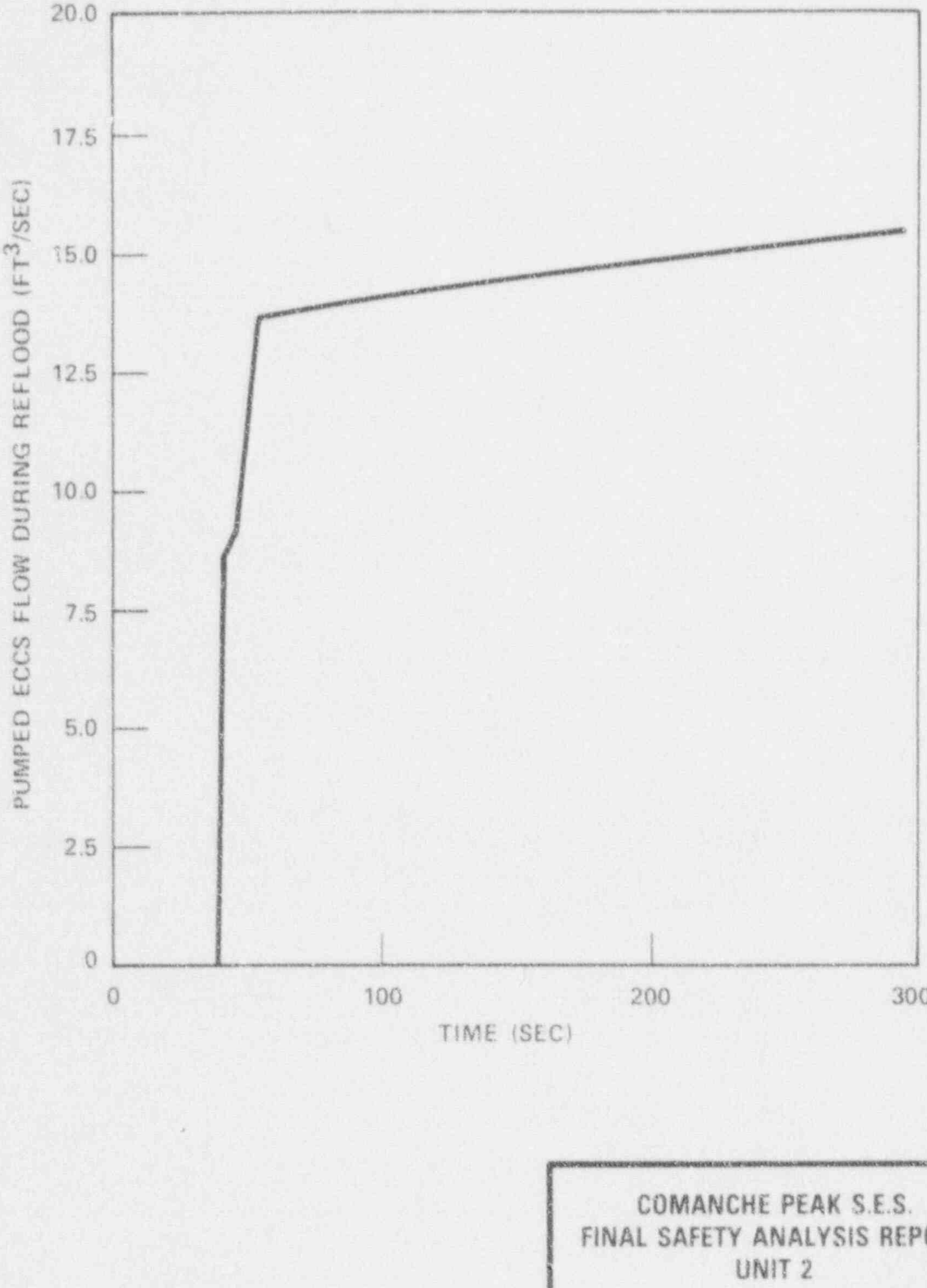
FIGURE 15.6-61



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Mass Velocity
DECLG (CD = 0.6) Max SI

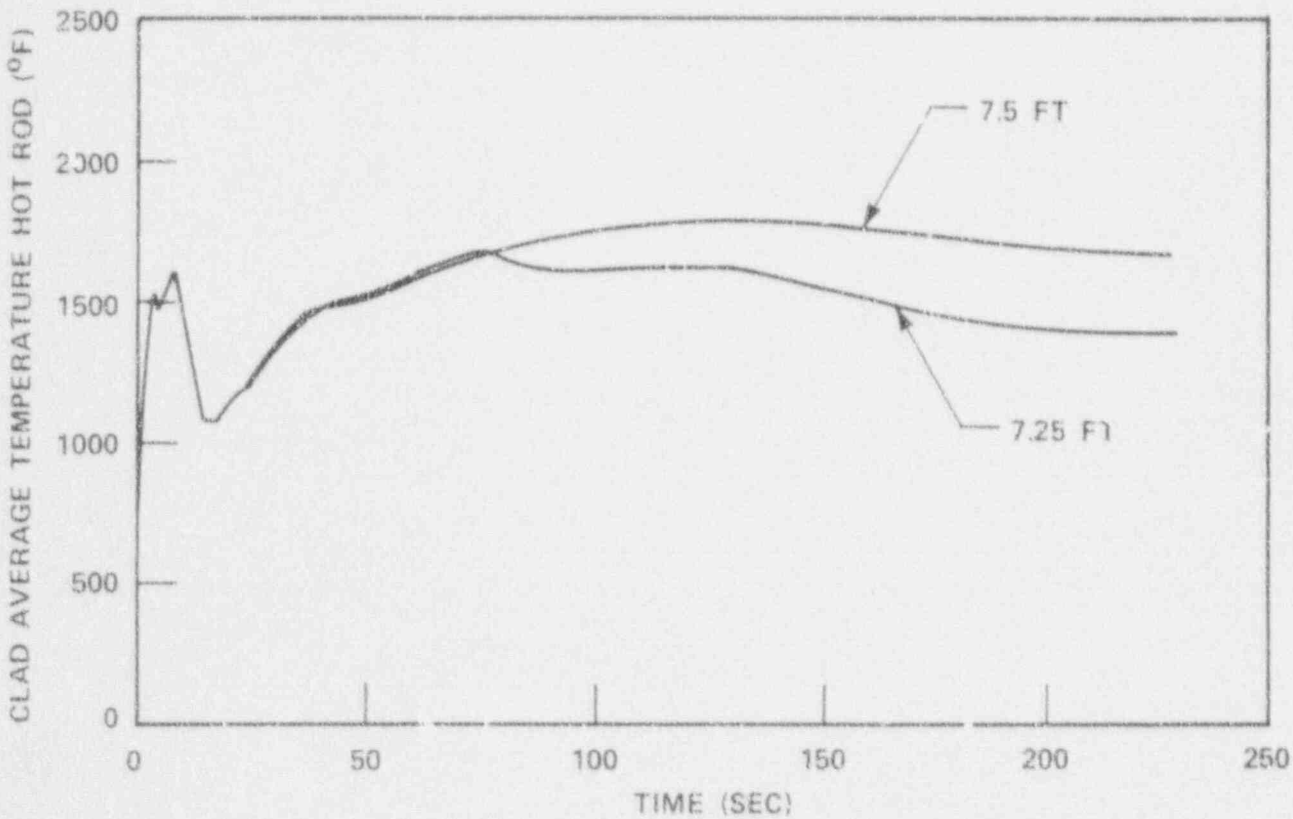
FIGURE 15.6-62



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Pumped ECCS Flow (Reflood)
DECLG (CD = 0.6) Max SI

FIGURE 15.6-63

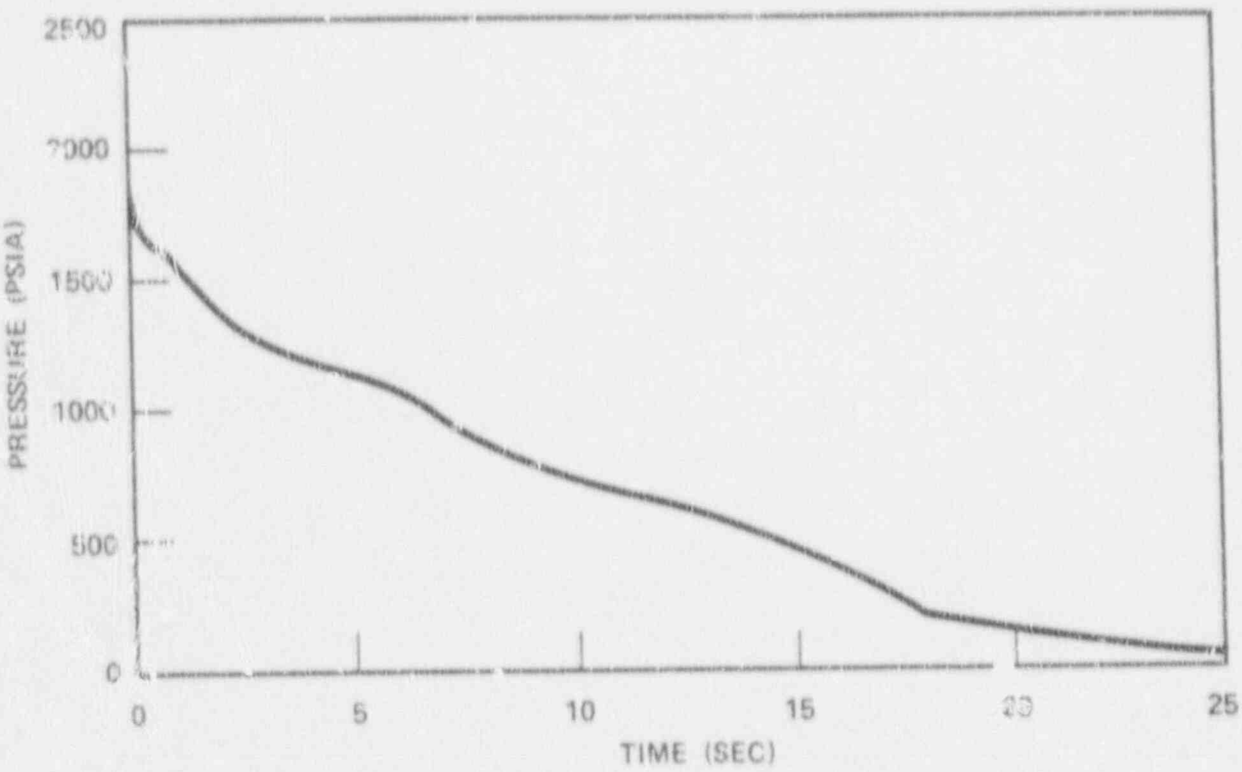


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Peak Clad Temperature
DECLG (CD = 0.8)

FIGURE 15.6-64

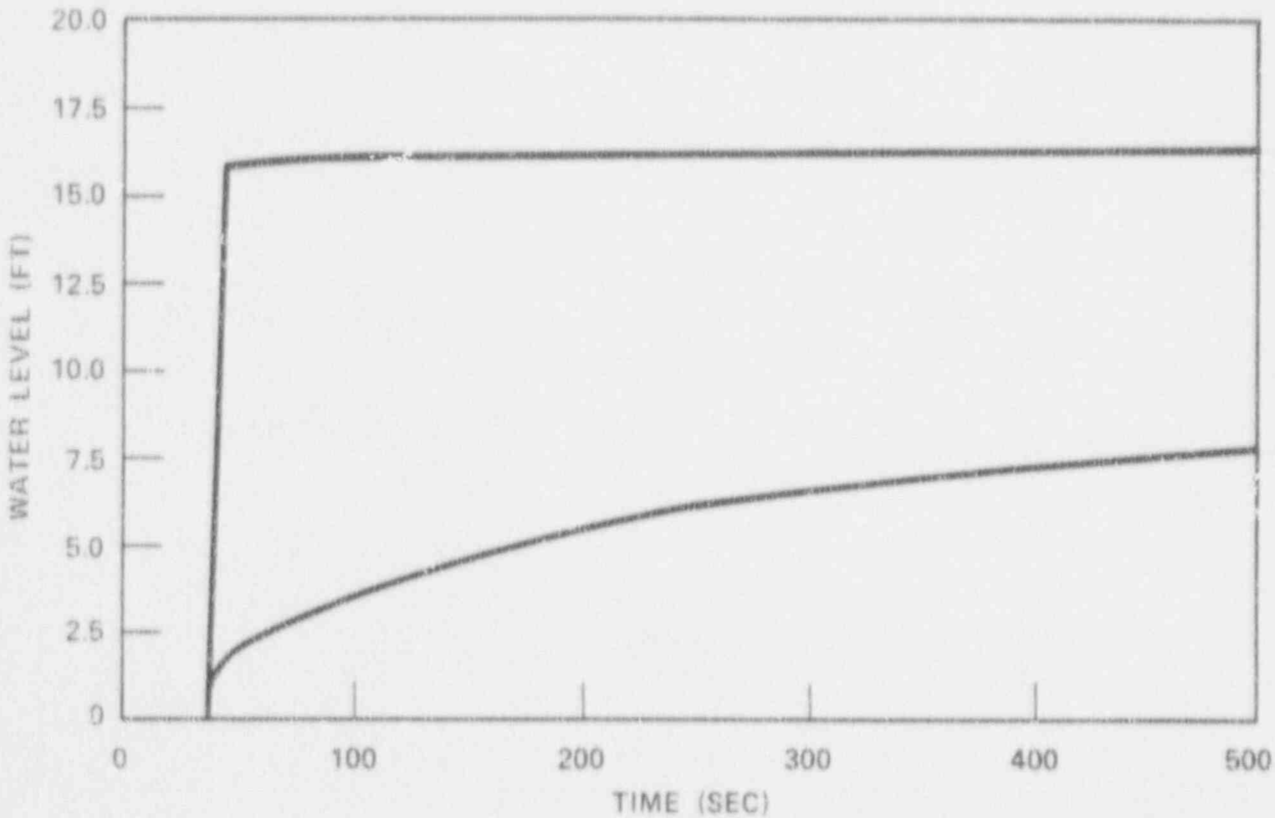
10457-24



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Pressure DECLG (CD = 0.8)

FIGURE 15.6-65

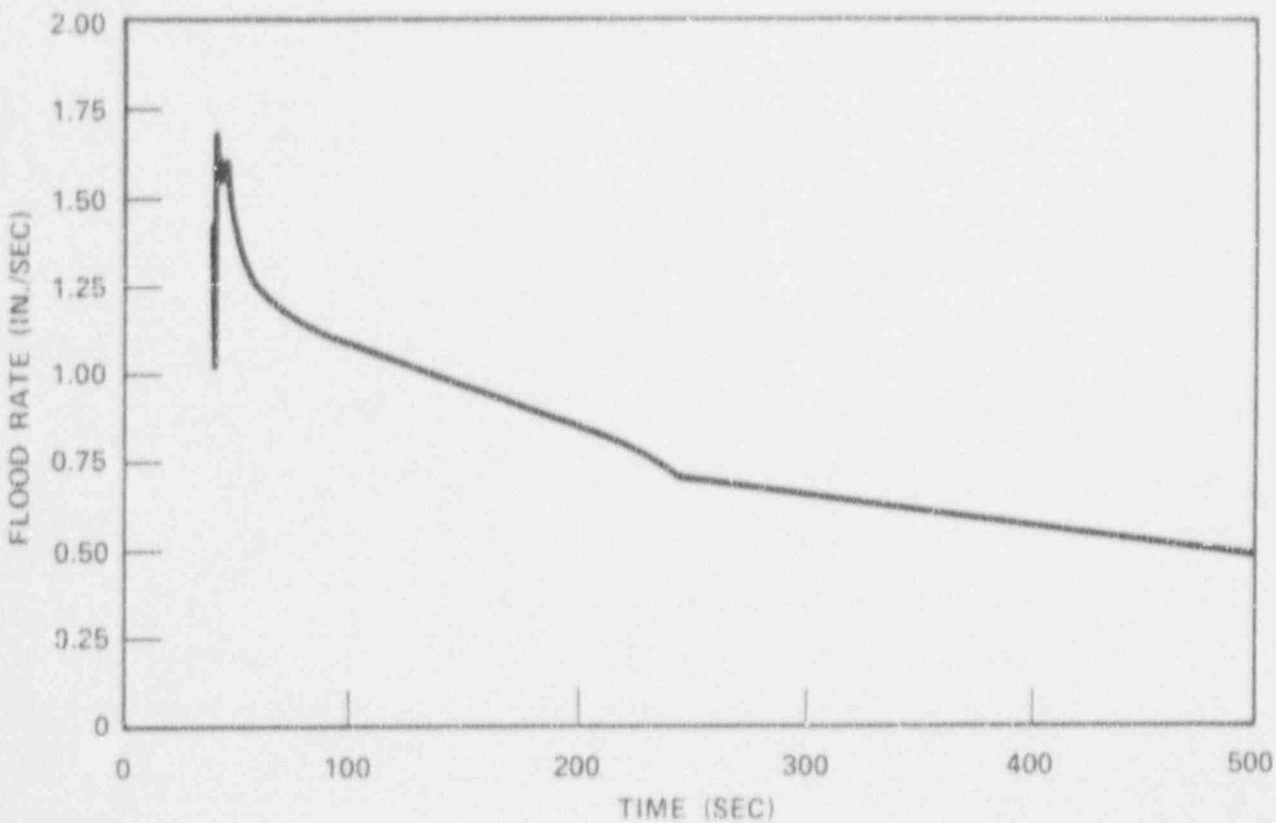


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient – Core & Downcomer Water Levels DECLG (CD = 0.8)

FIGURE 15.6-66

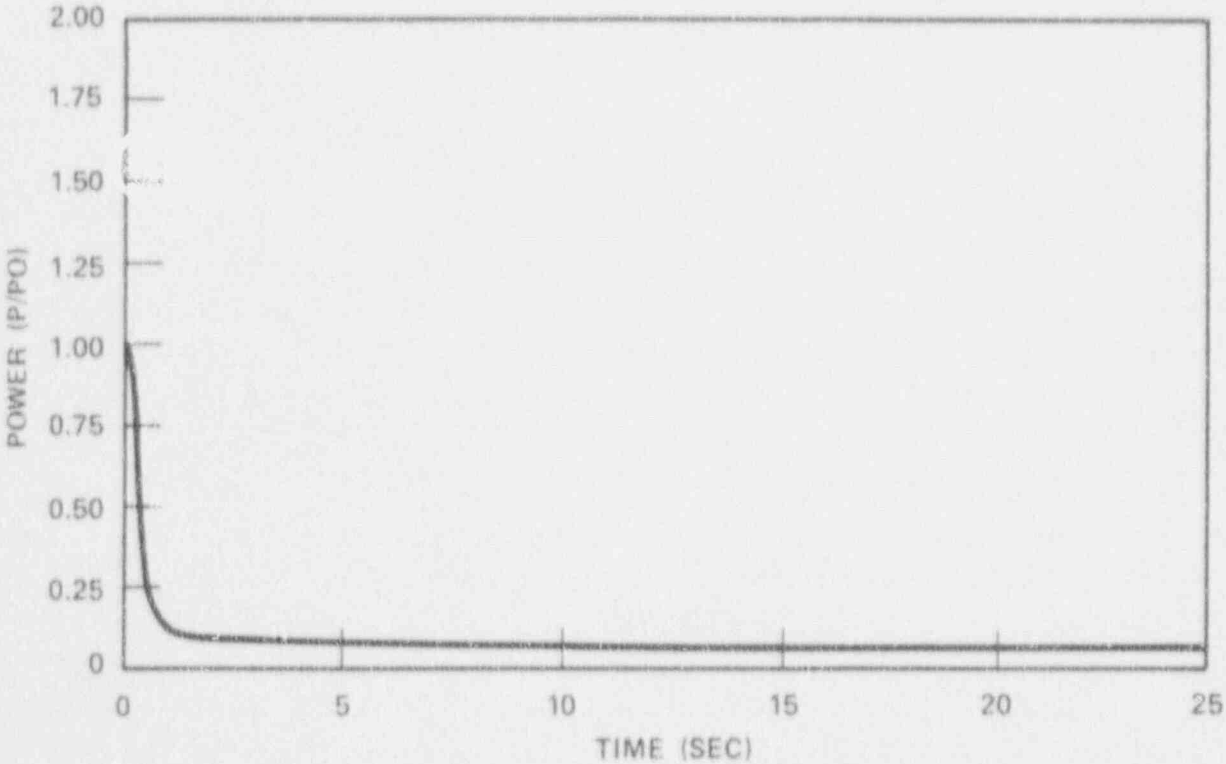
10467-26



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient Core Inlet Velocity
DECLG (CD = 0.8)

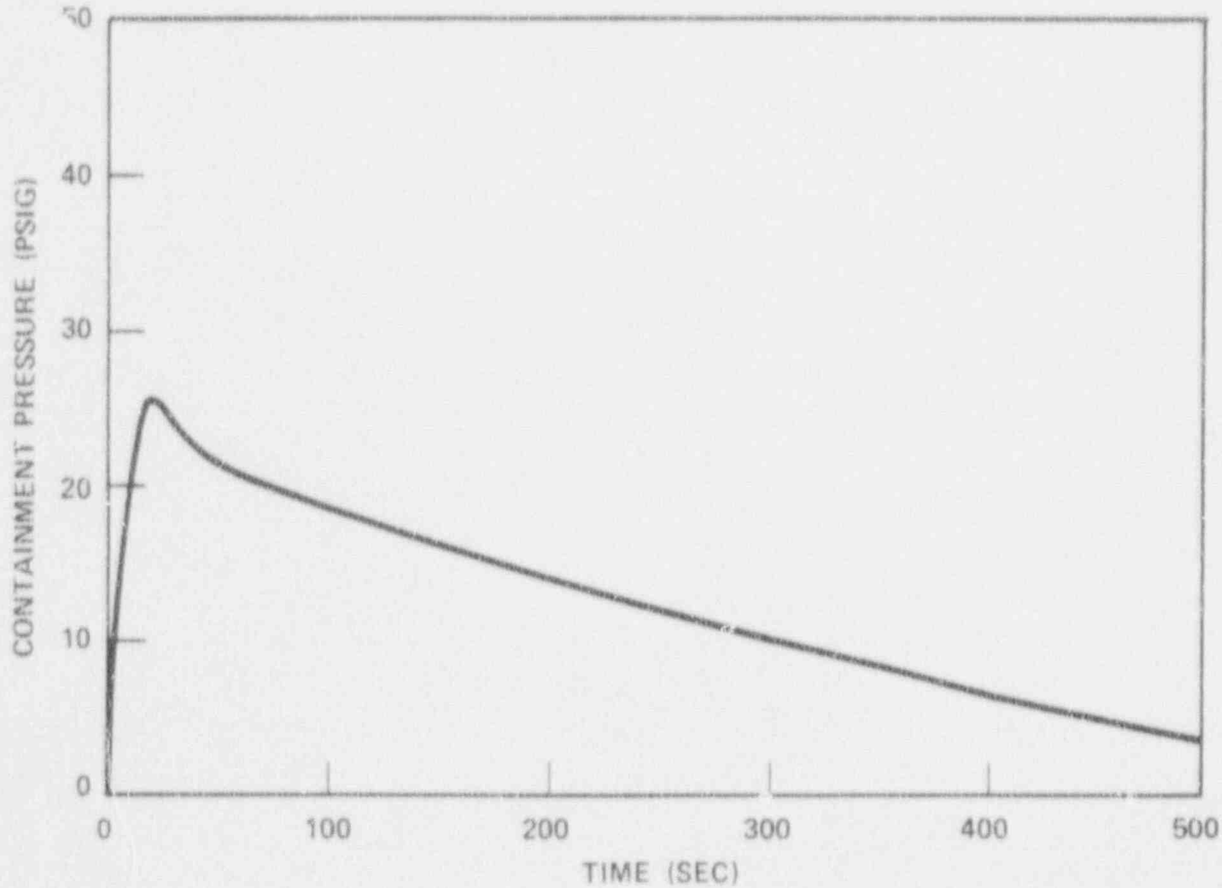
FIGURE 15.6-67



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Power Transient
DECLG (CD = 0.8)

FIGURE 15.6-68

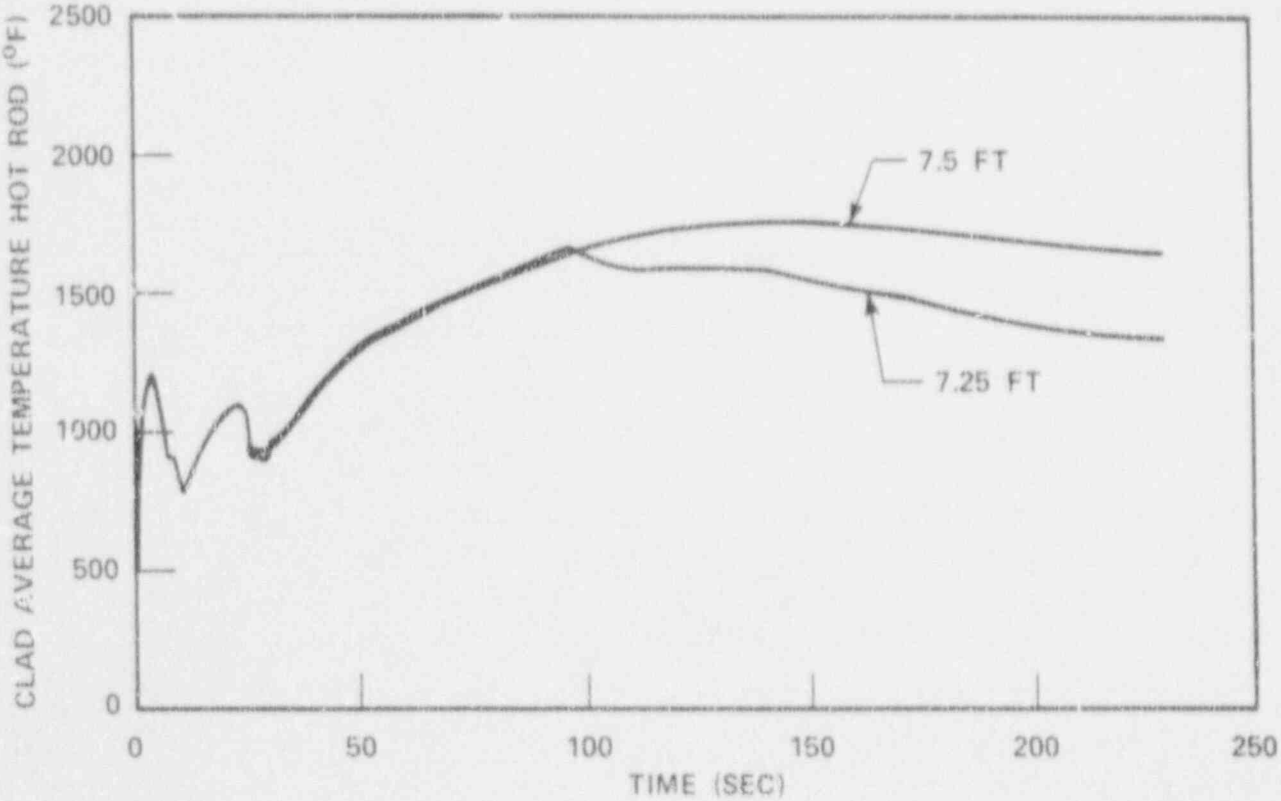


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Containment Pressure
DECLG (CD = 0.8)

FIGURE 15.6-69

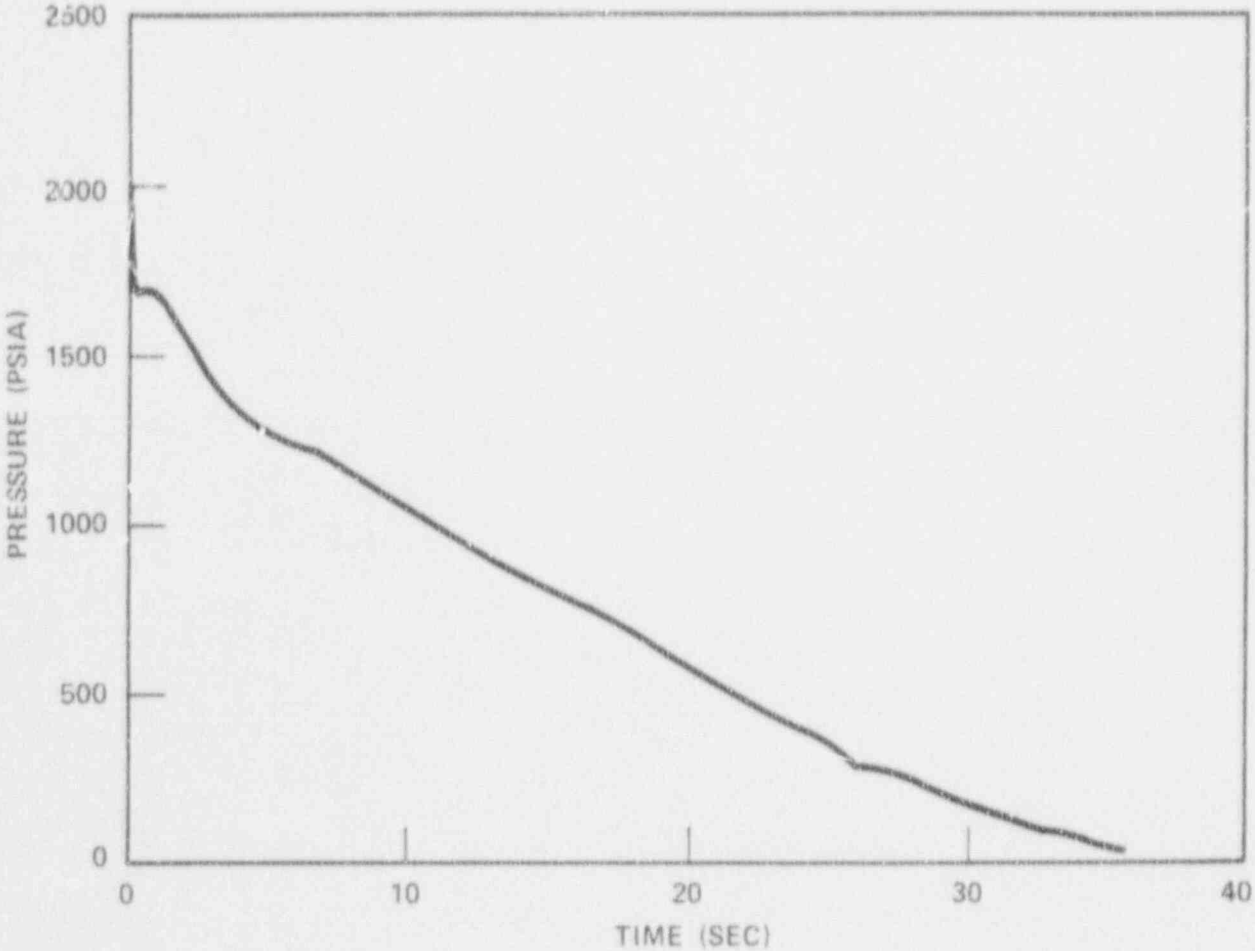
10457-29



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Peak Clad Temperature
DECLG (CD = 0.4)

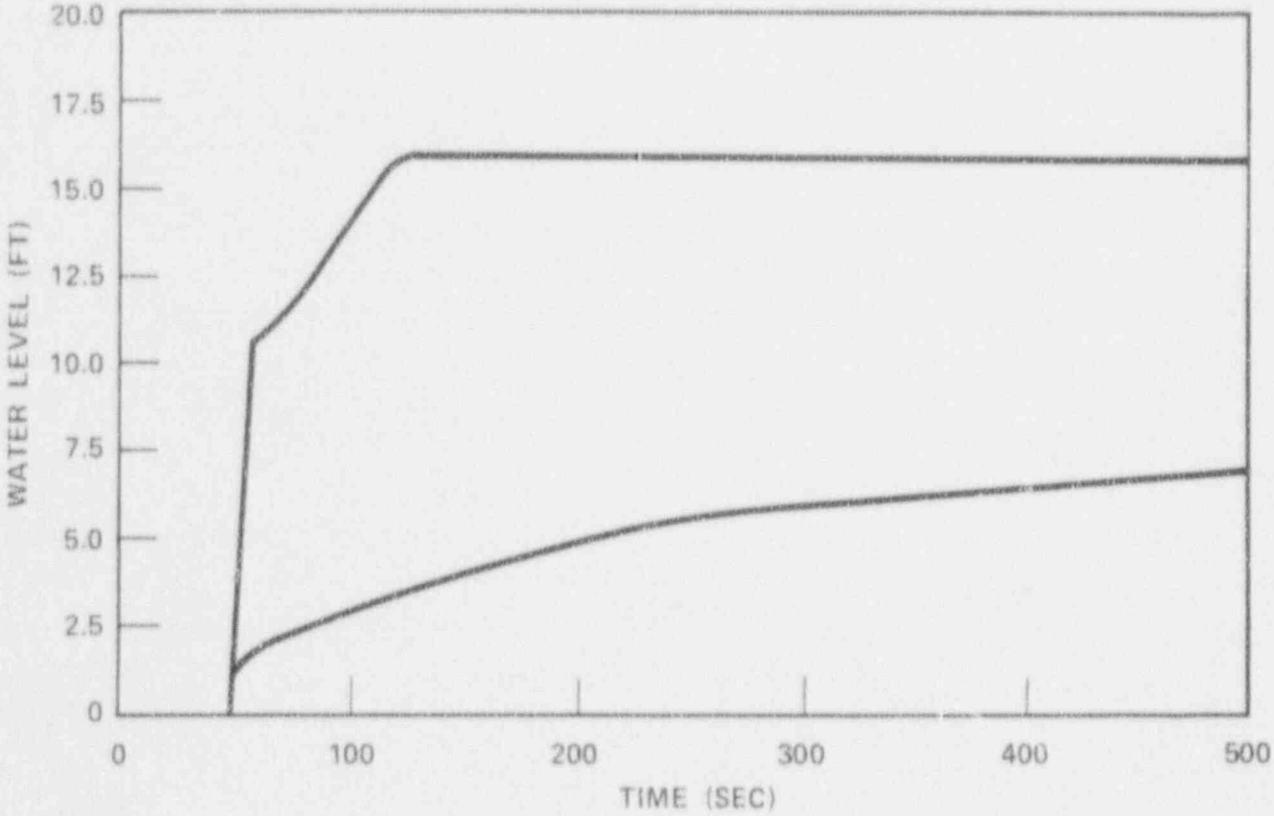
FIGURE 15.6-70



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Pressure
DECLG (CD = 0.4)

FIGURE 15.6-71

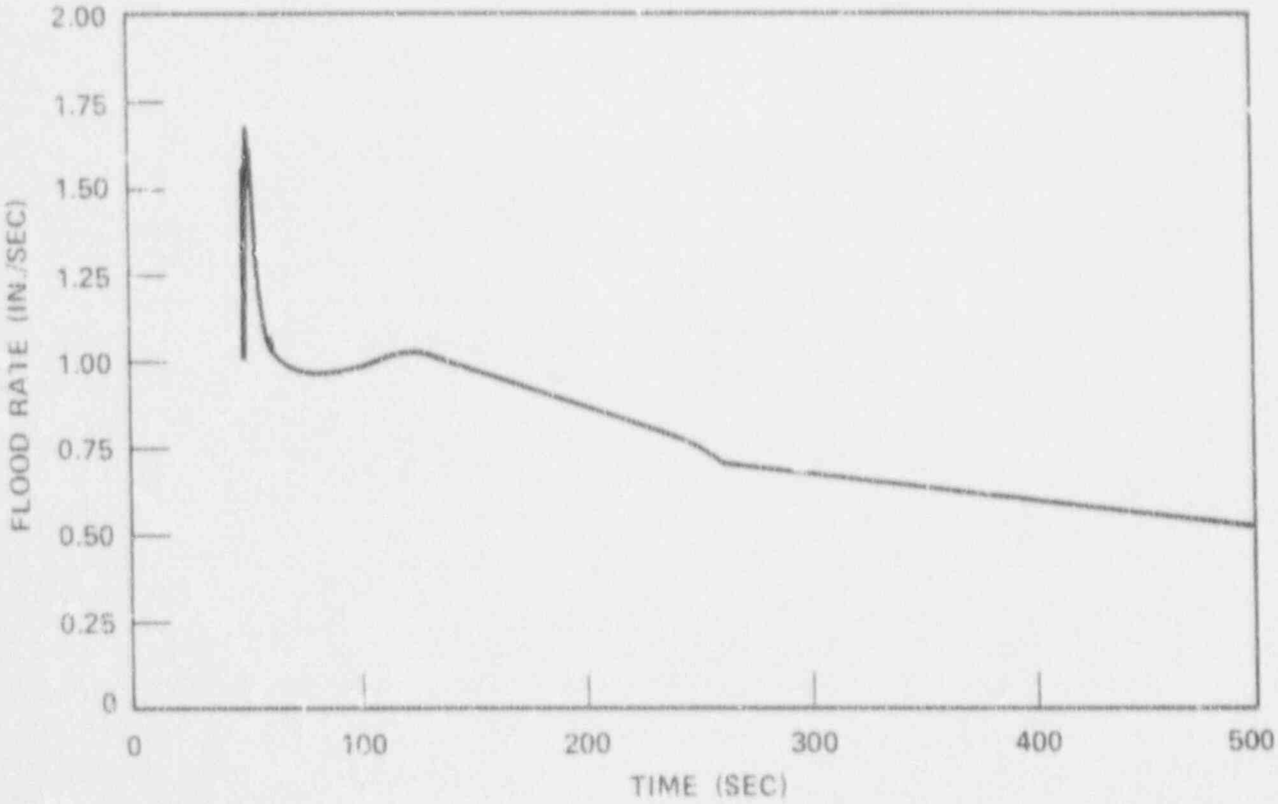


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient – Core & Downcomer Water Levels DECLG (CD = 0.4)

FIGURE 15.6-72

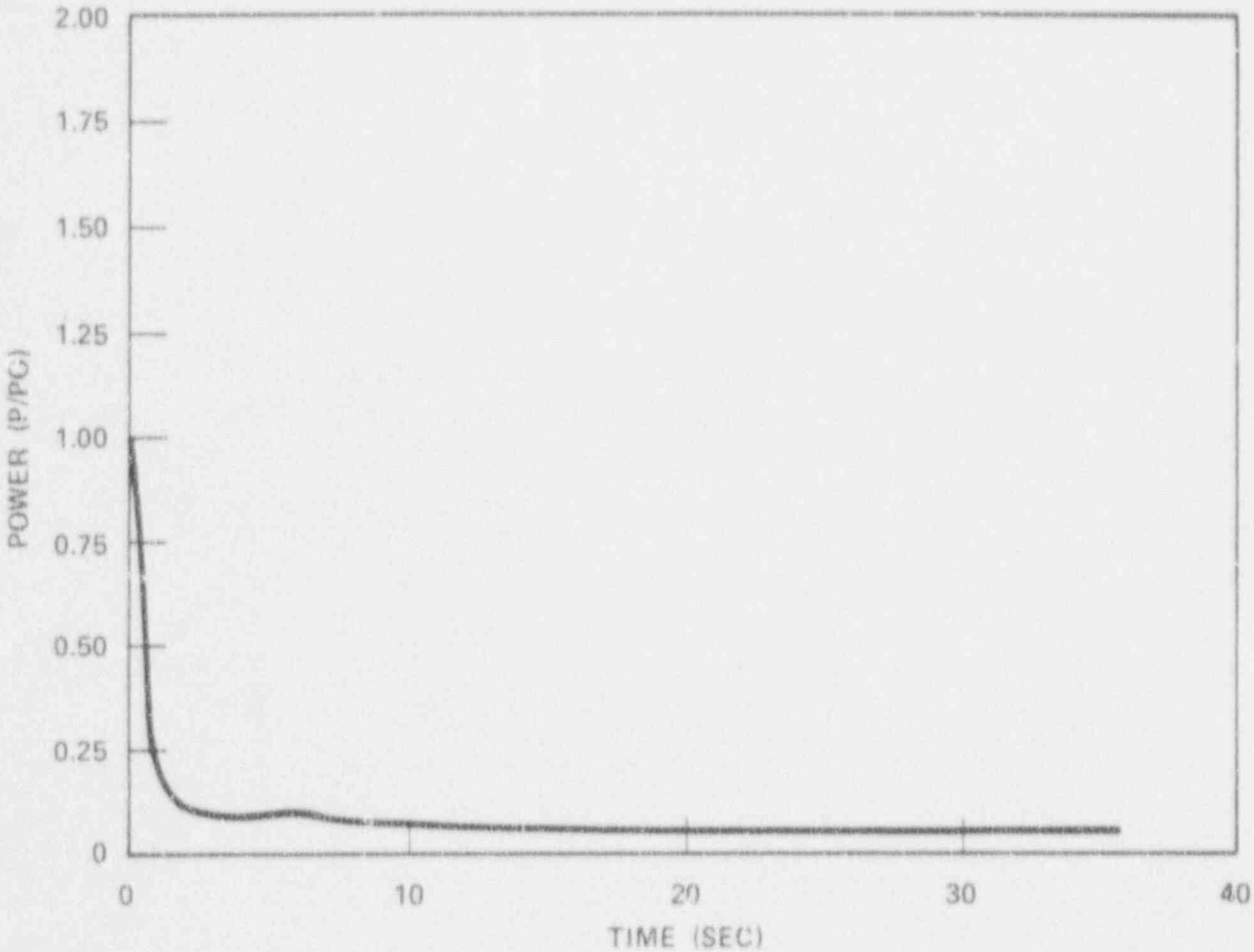
10457-32



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Reflood Transient Core Inlet Velocity
DECLG (CD = 0.4)

FIGURE 15.6-73

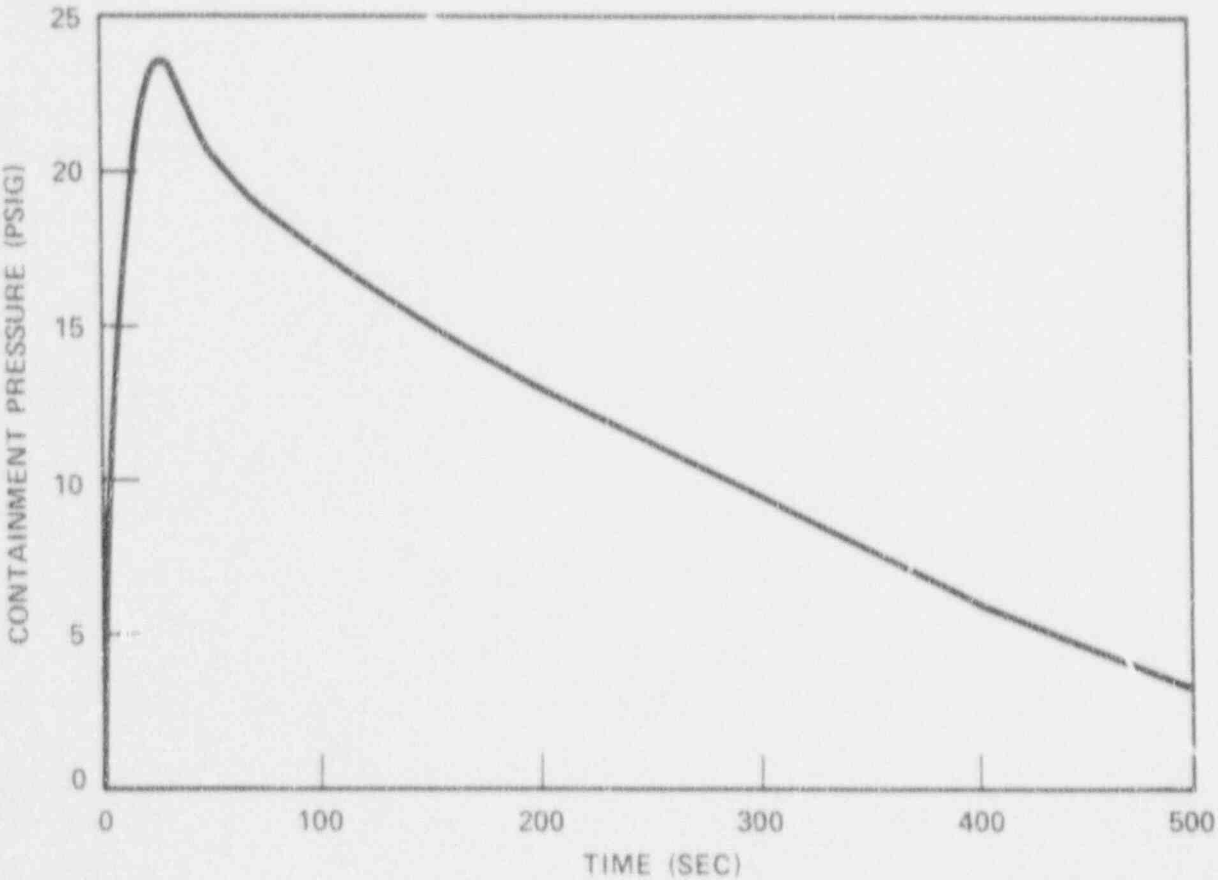


COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Power Transient
DECLG (CD = 0.4)

FIGURE 15.6-74

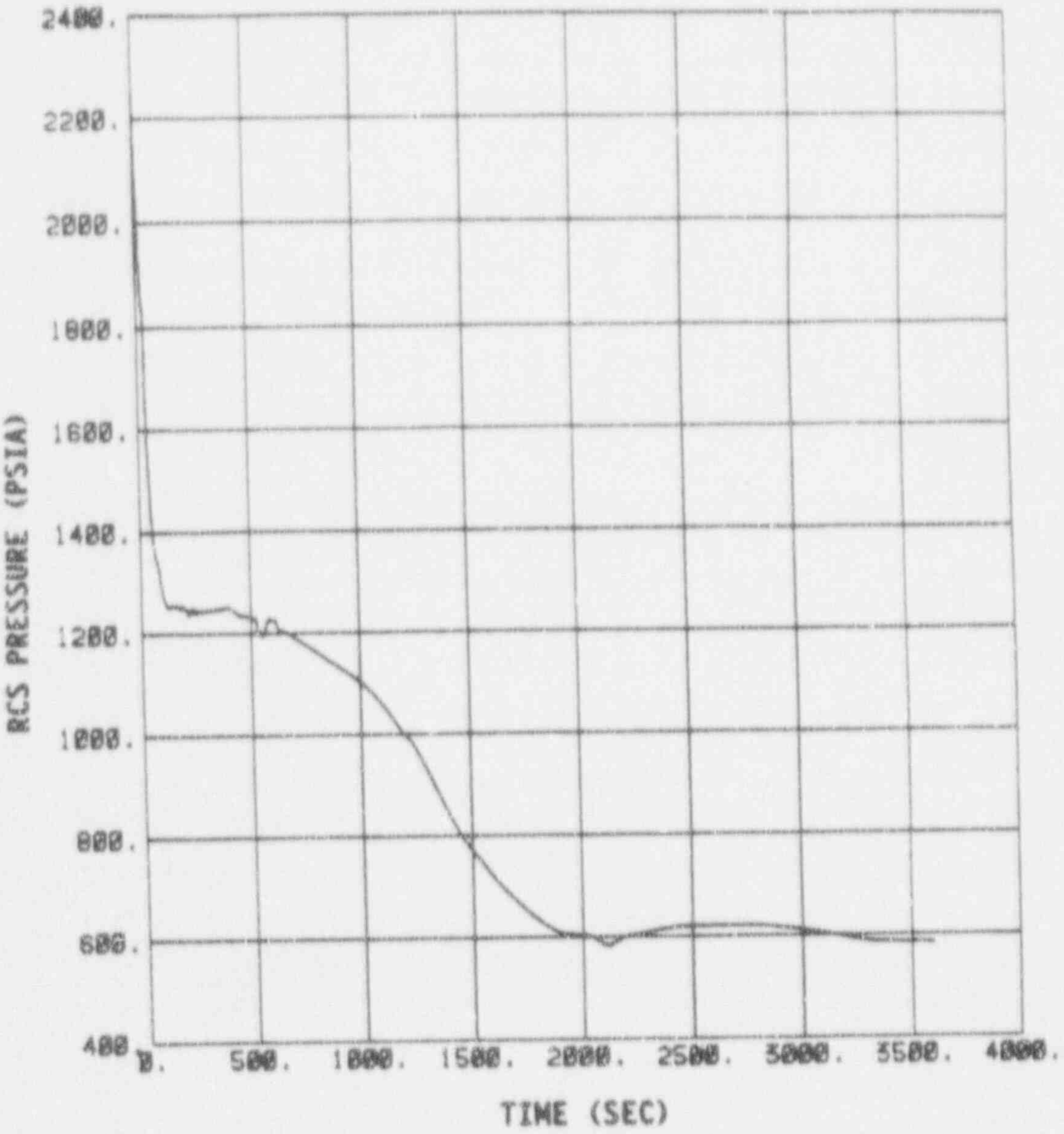
1045734



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Containment Pressure
DECLG (CD = 0.4)

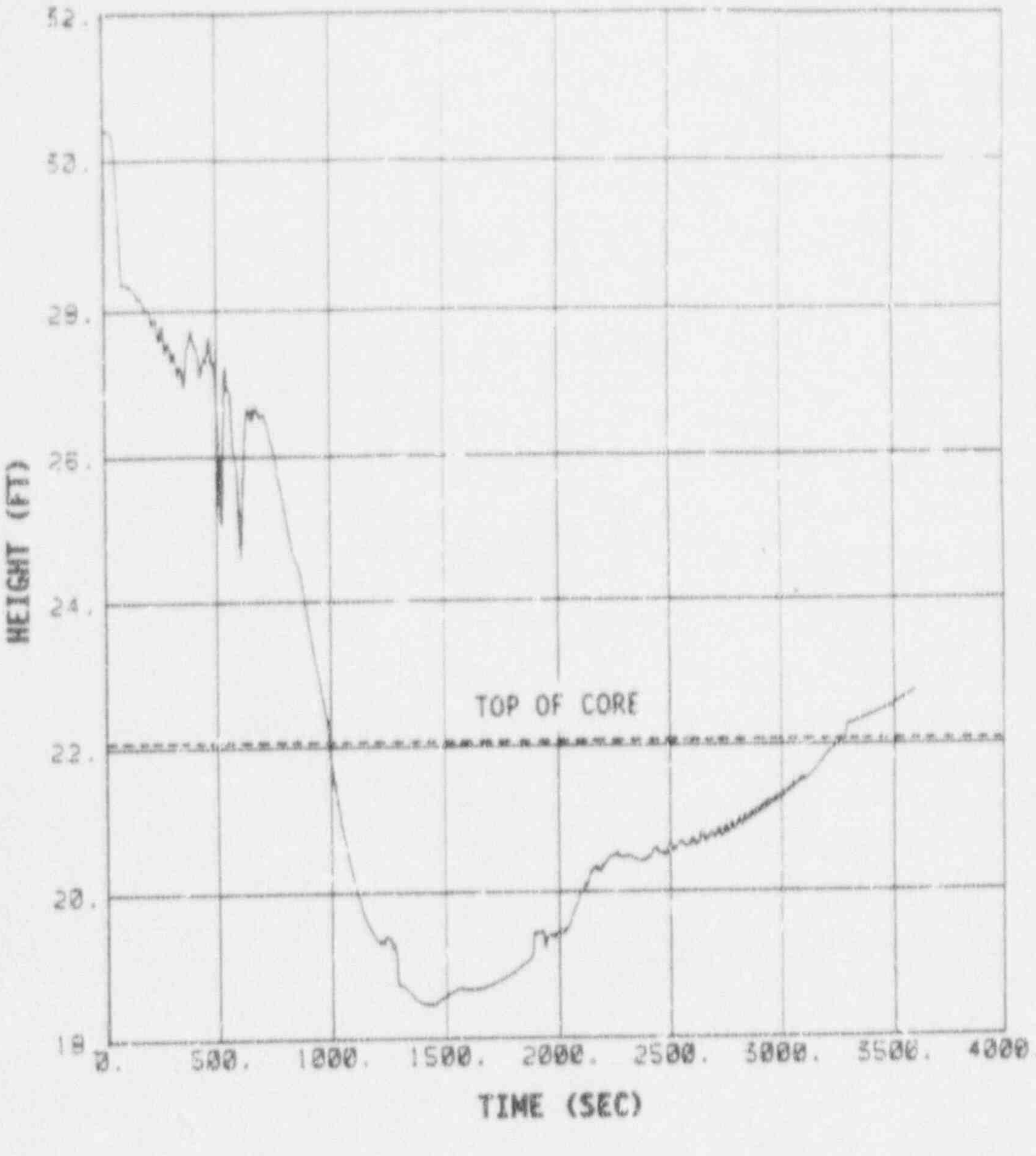
FIGURE 15.6-75



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

RCS Depressurization Transient
(3 Inch Break)

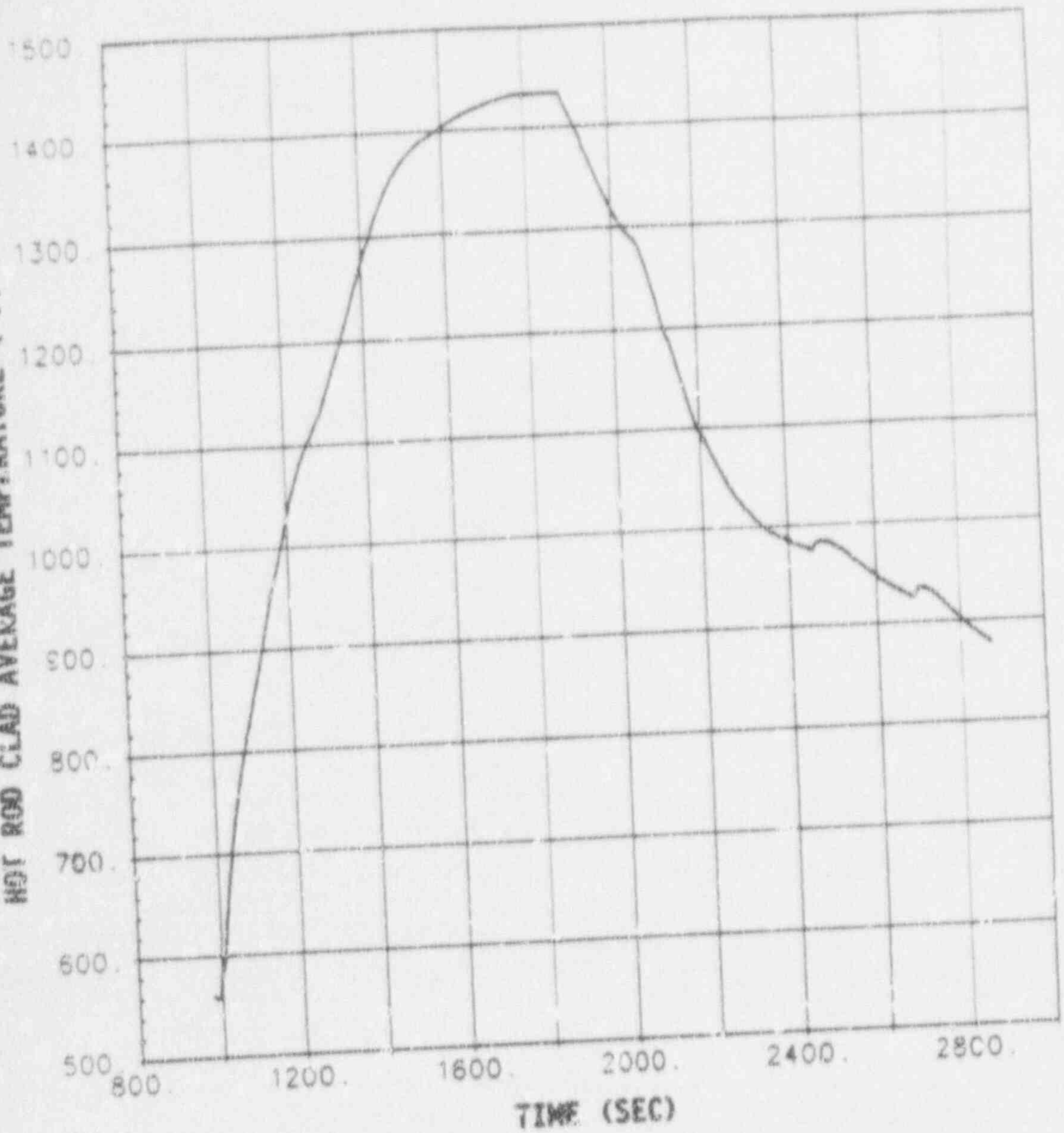
FIGURE 15.6-76



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Mixture Height
(3 Inch Break)

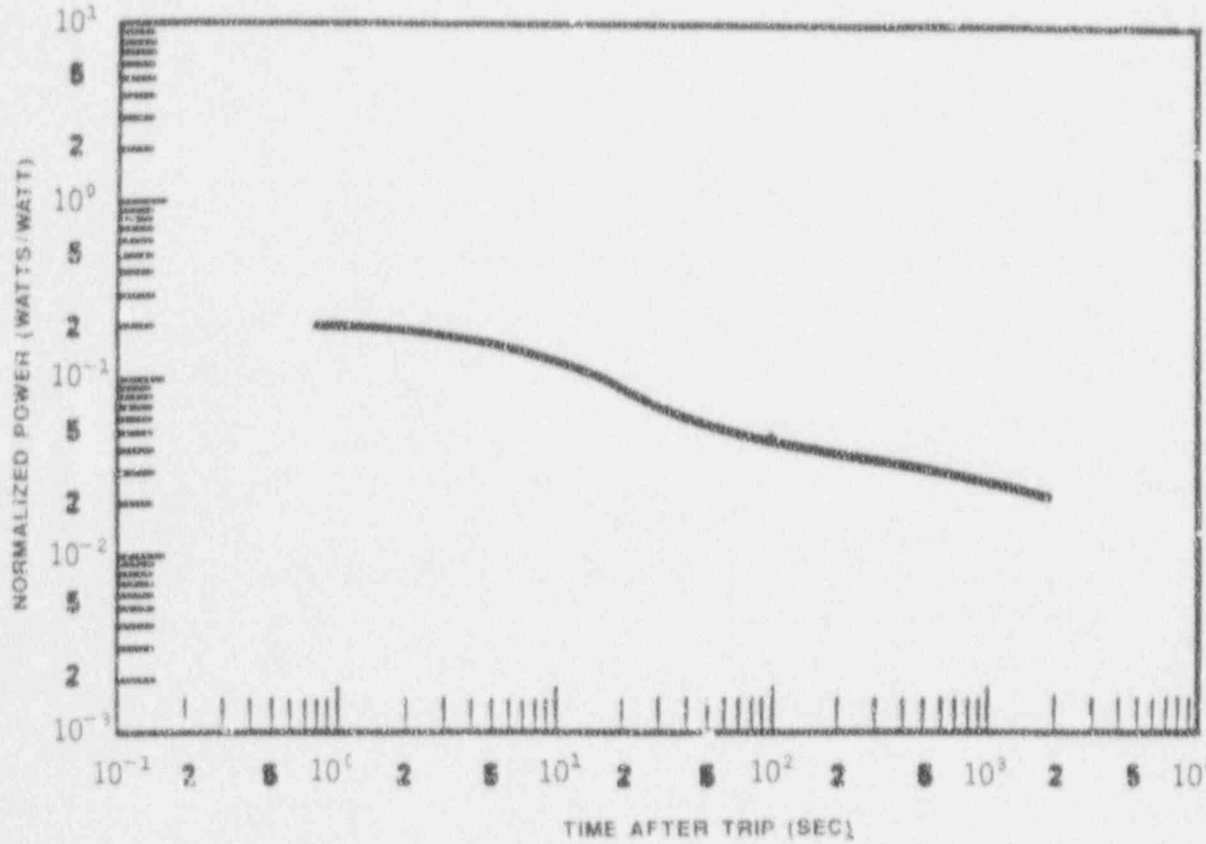
FIGURE 15.6-77



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Clad Temperature Transient
(3 Inch Break)

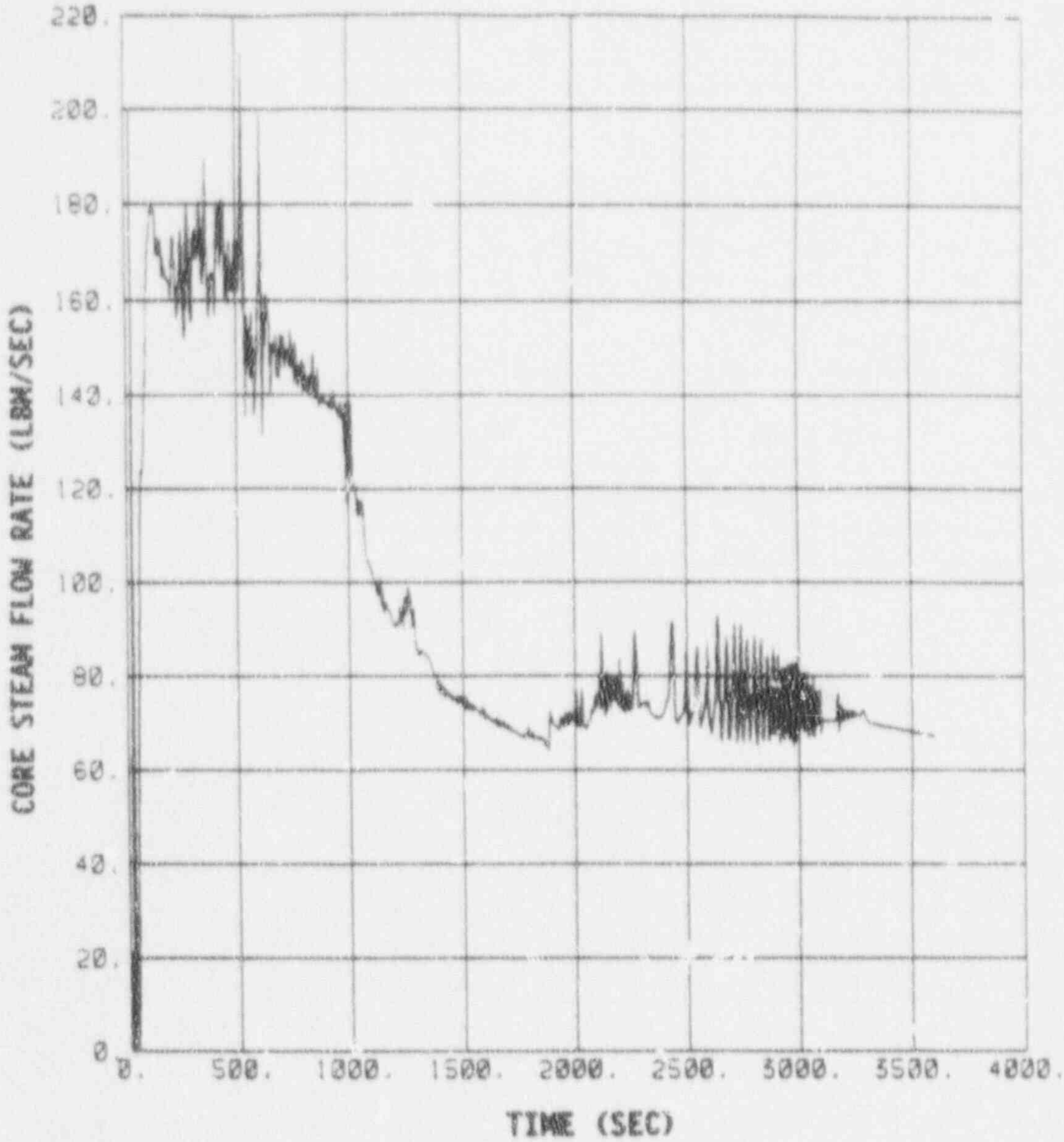
FIGURE 15.6-78



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Power After Reactor Trip

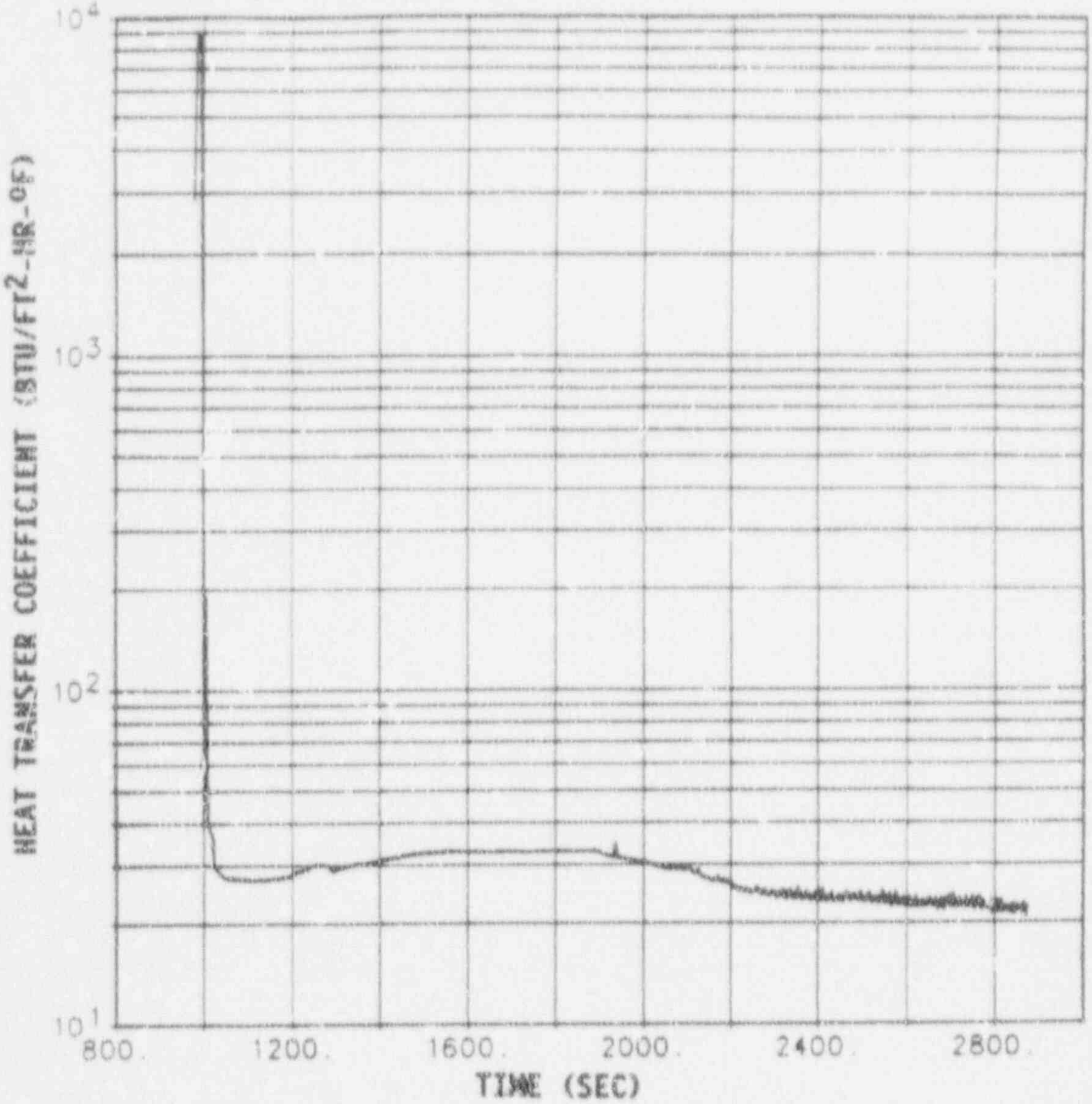
FIGURE 15.6-79



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Steam Flow
(3 Inch Break)

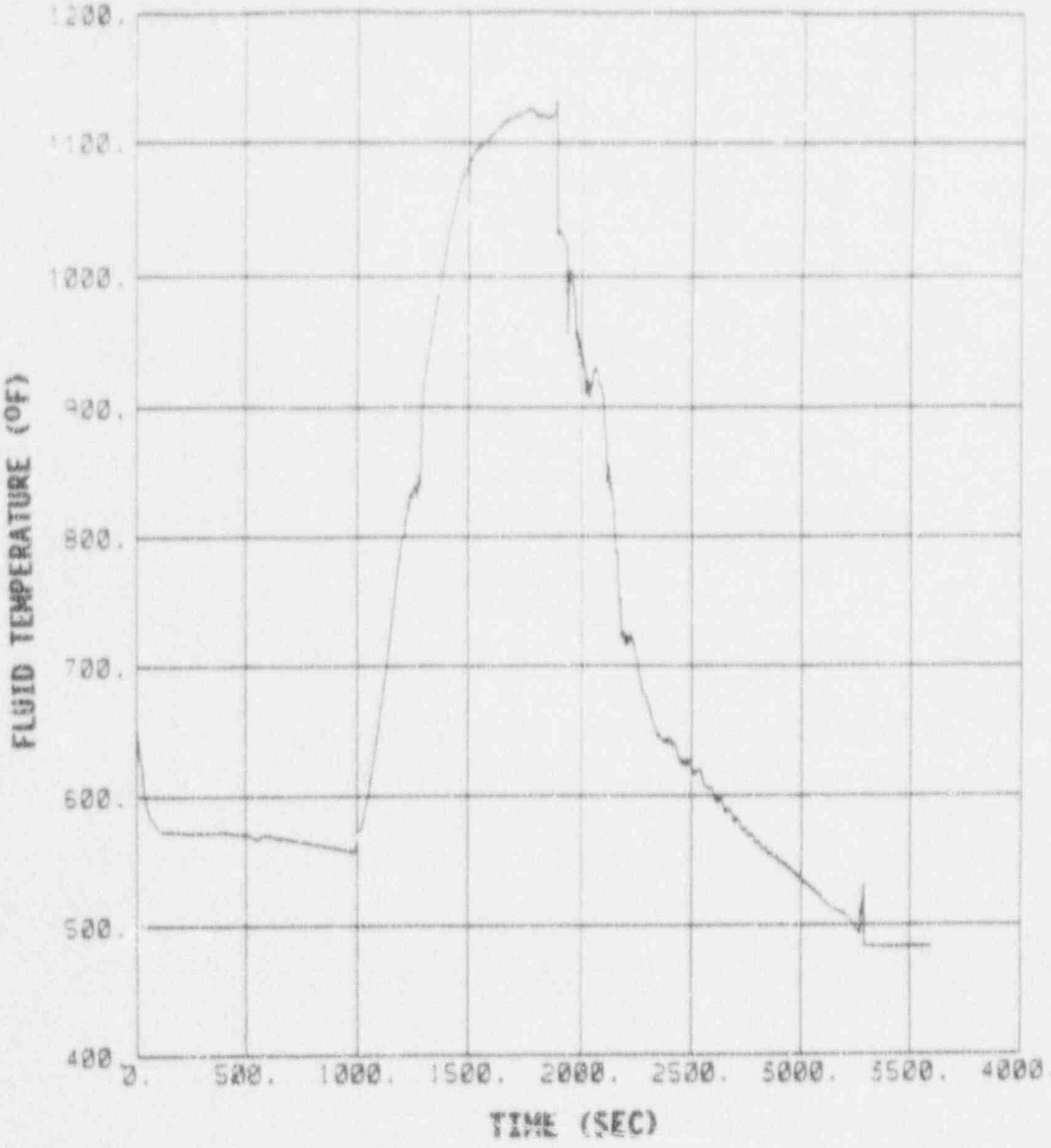
FIGURE 15.6-80



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Rod Film Heat Transfer Coefficient
(3 Inch Break)

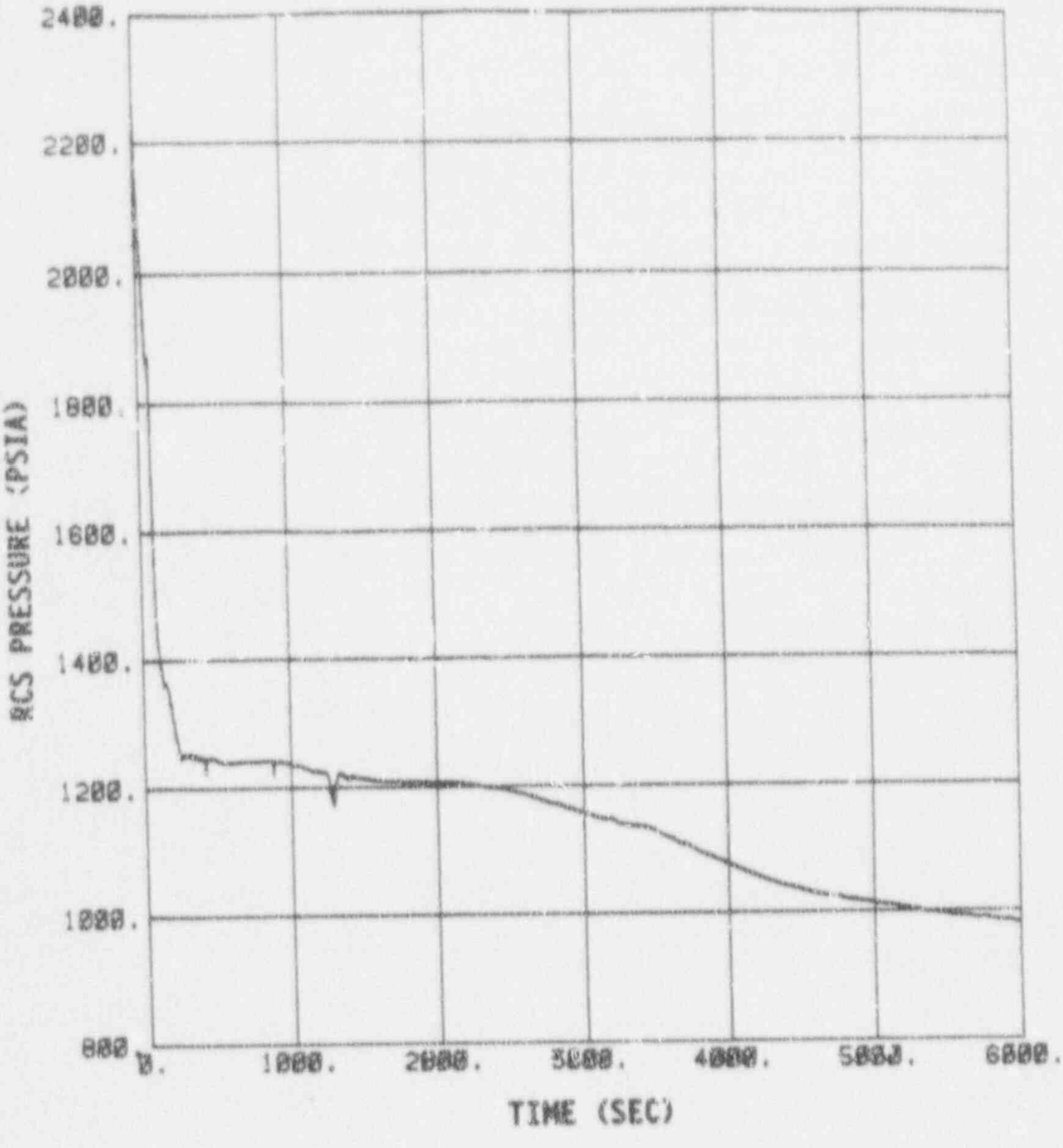
FIGURE 15.6-81



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Hot Spot Fluid Temperature
(3 Inch Break)

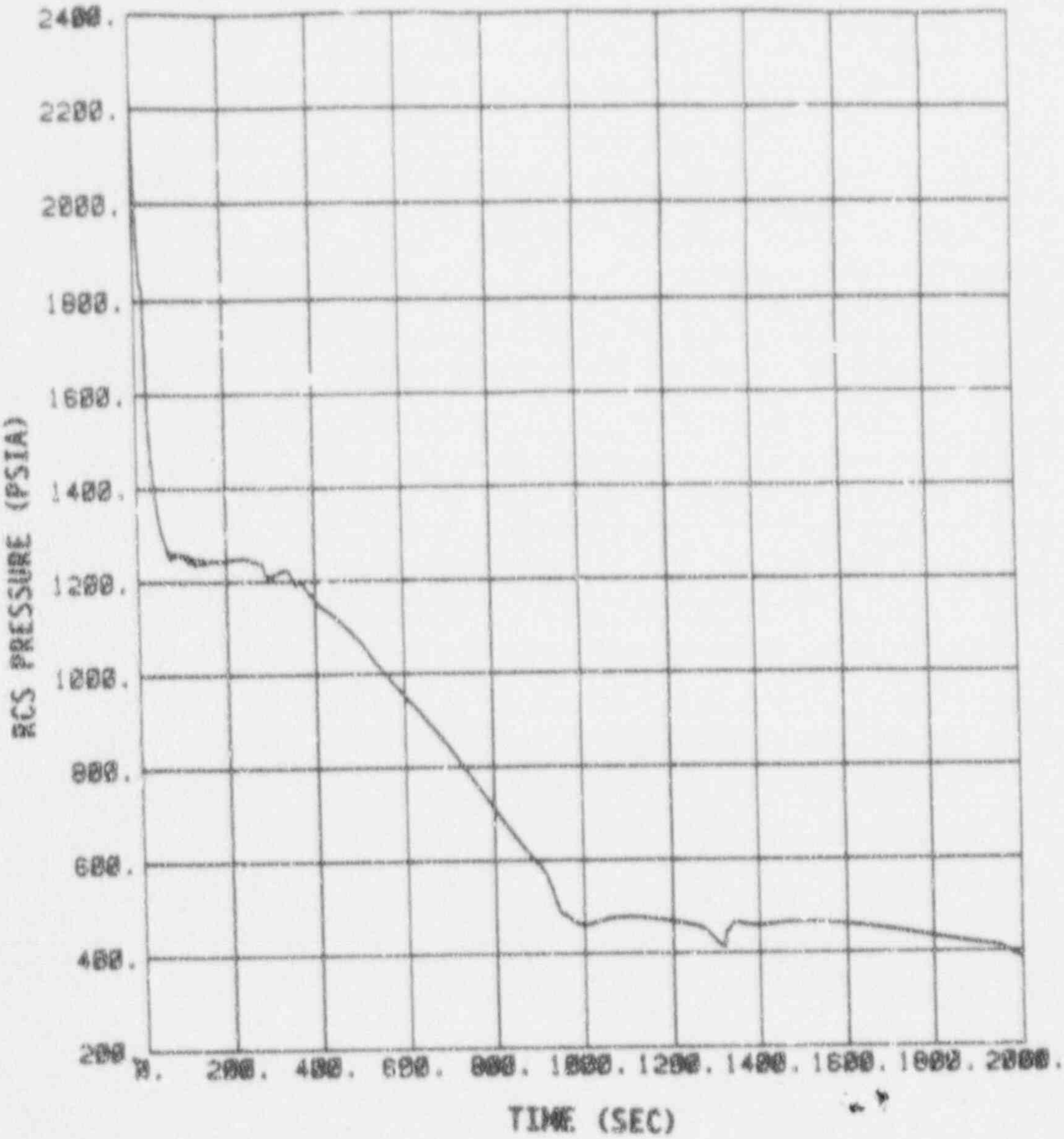
FIGURE 15.6-82



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

RCS Depressurization Transient
(2 Inch Break)

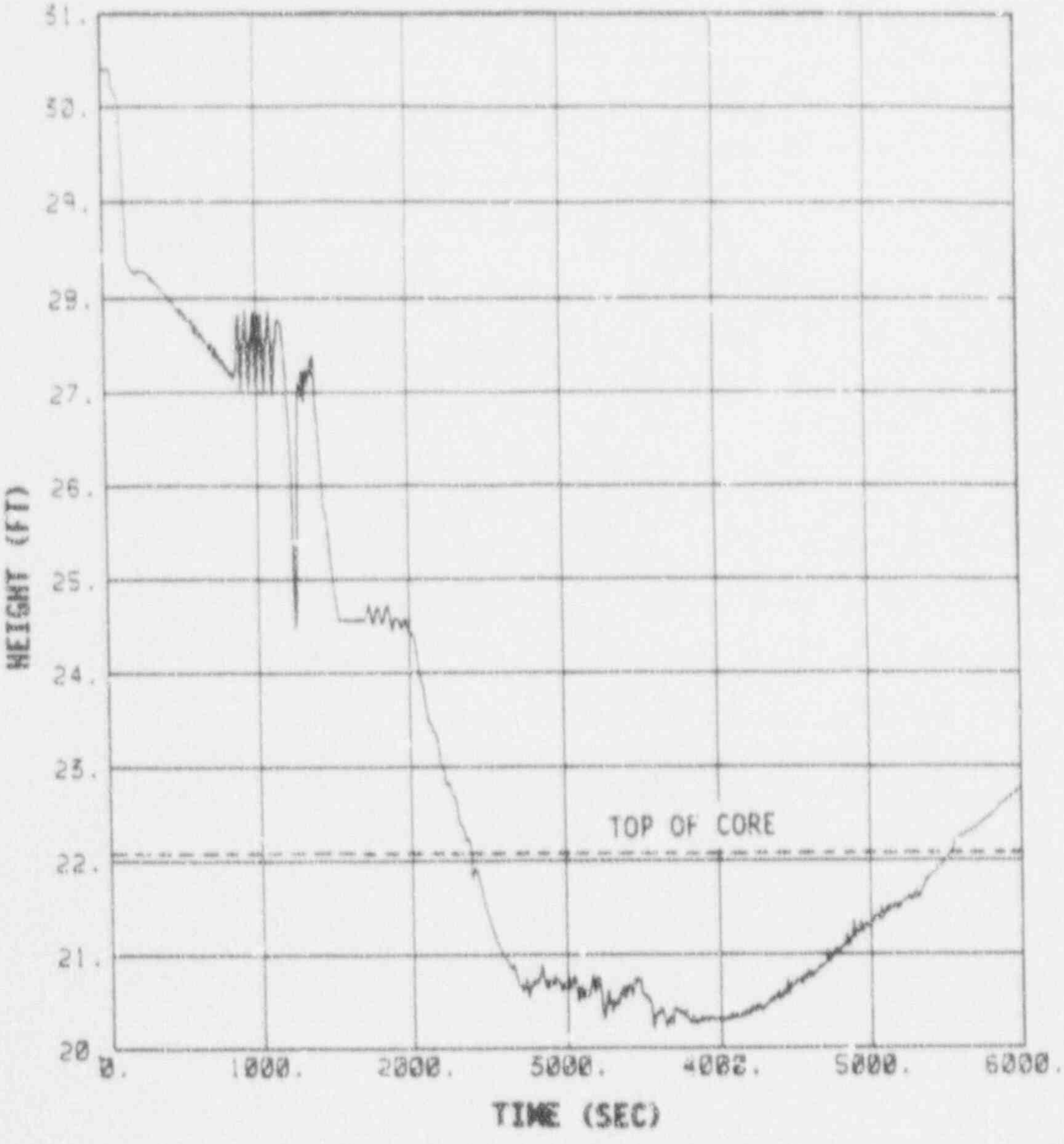
FIGURE 15.6-83



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

RCS Depressurization Transient
(4 inch Break)

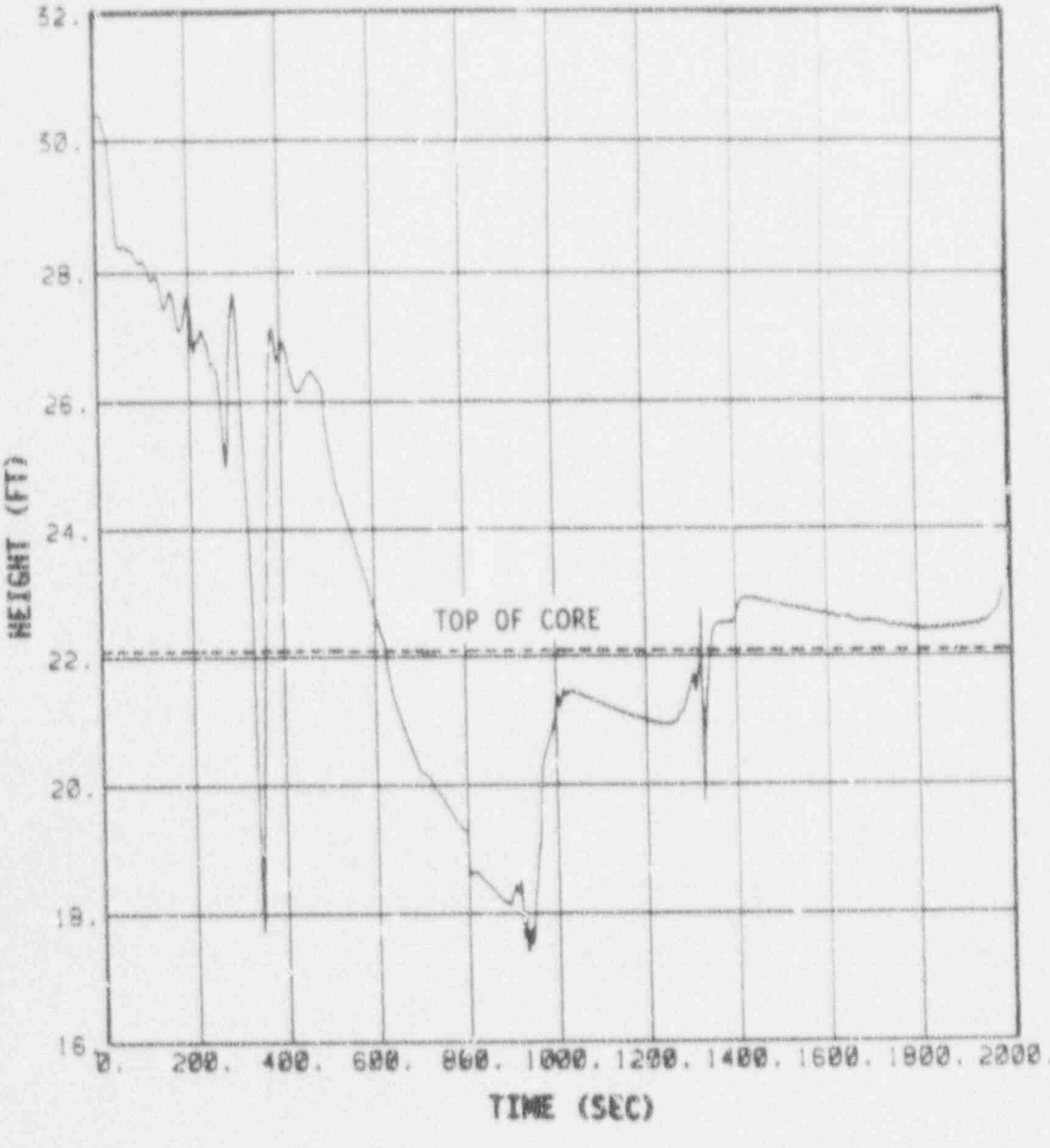
FIGURE 15.6-84



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Mixture Height
(2 Inch Break)

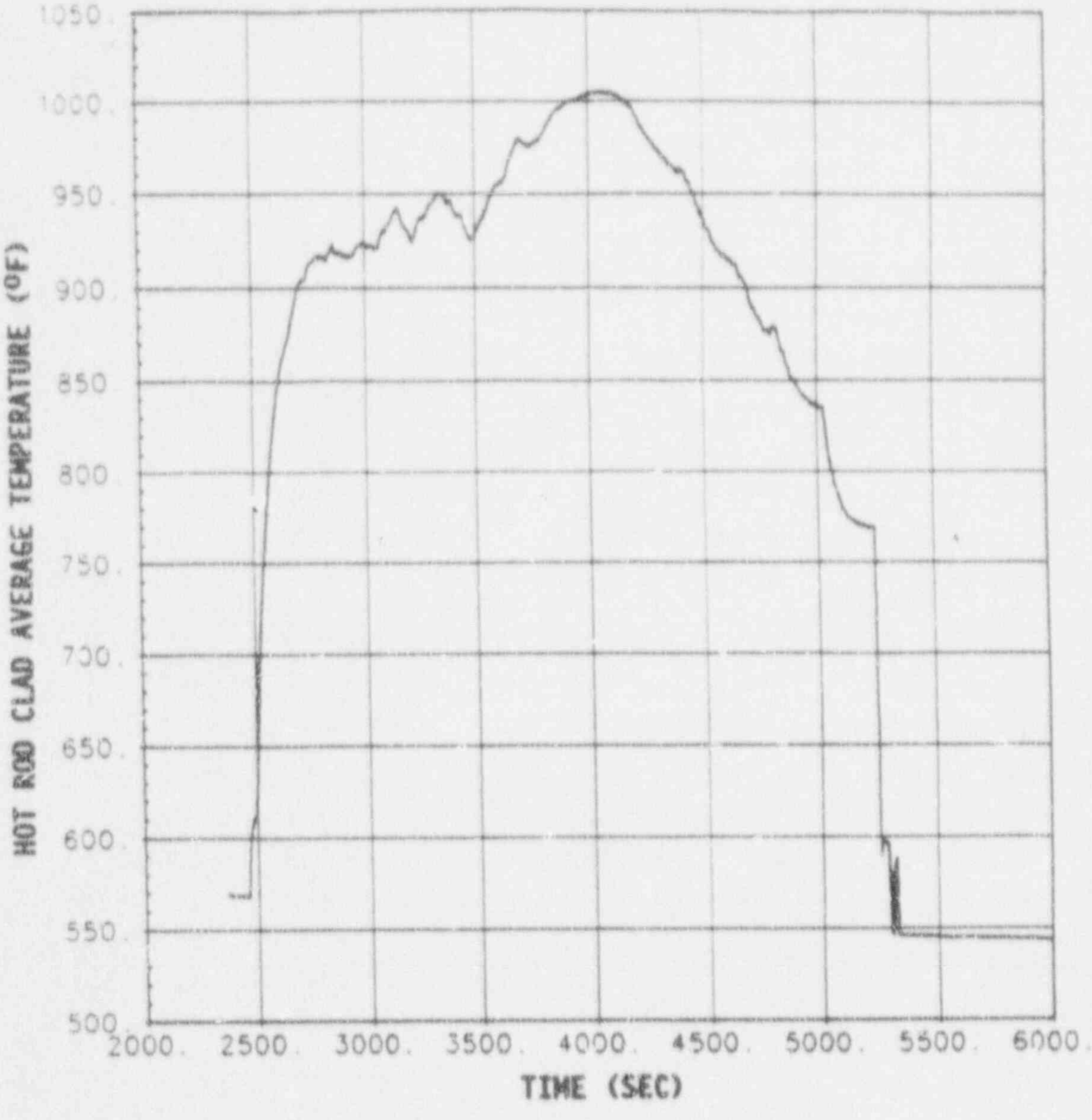
FIGURE 15.6-85



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Core Mixture Height
(4 Inch Break)

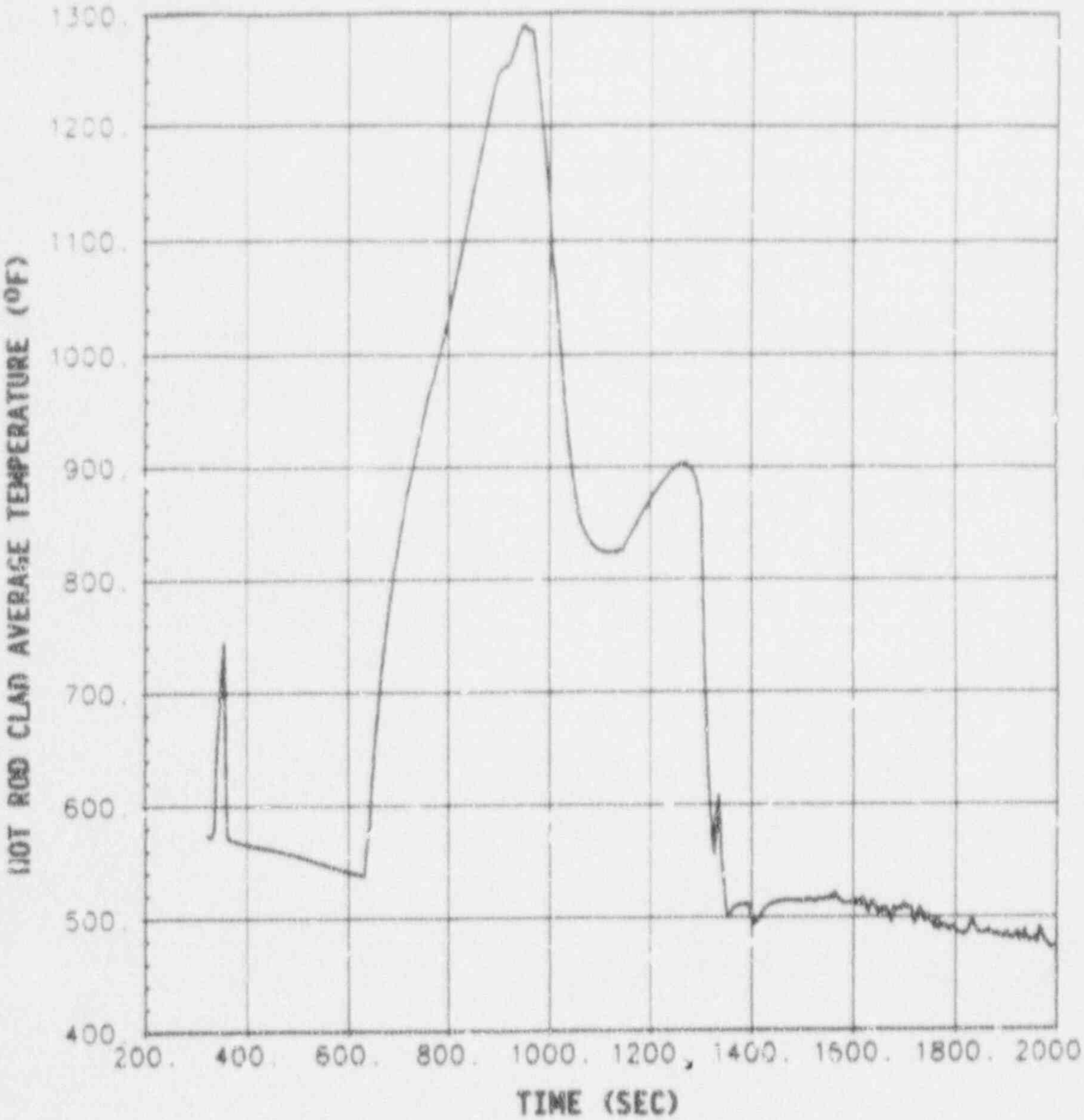
FIGURE 15.6-86



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Clad Temperature Transient
(2 Inch Break)

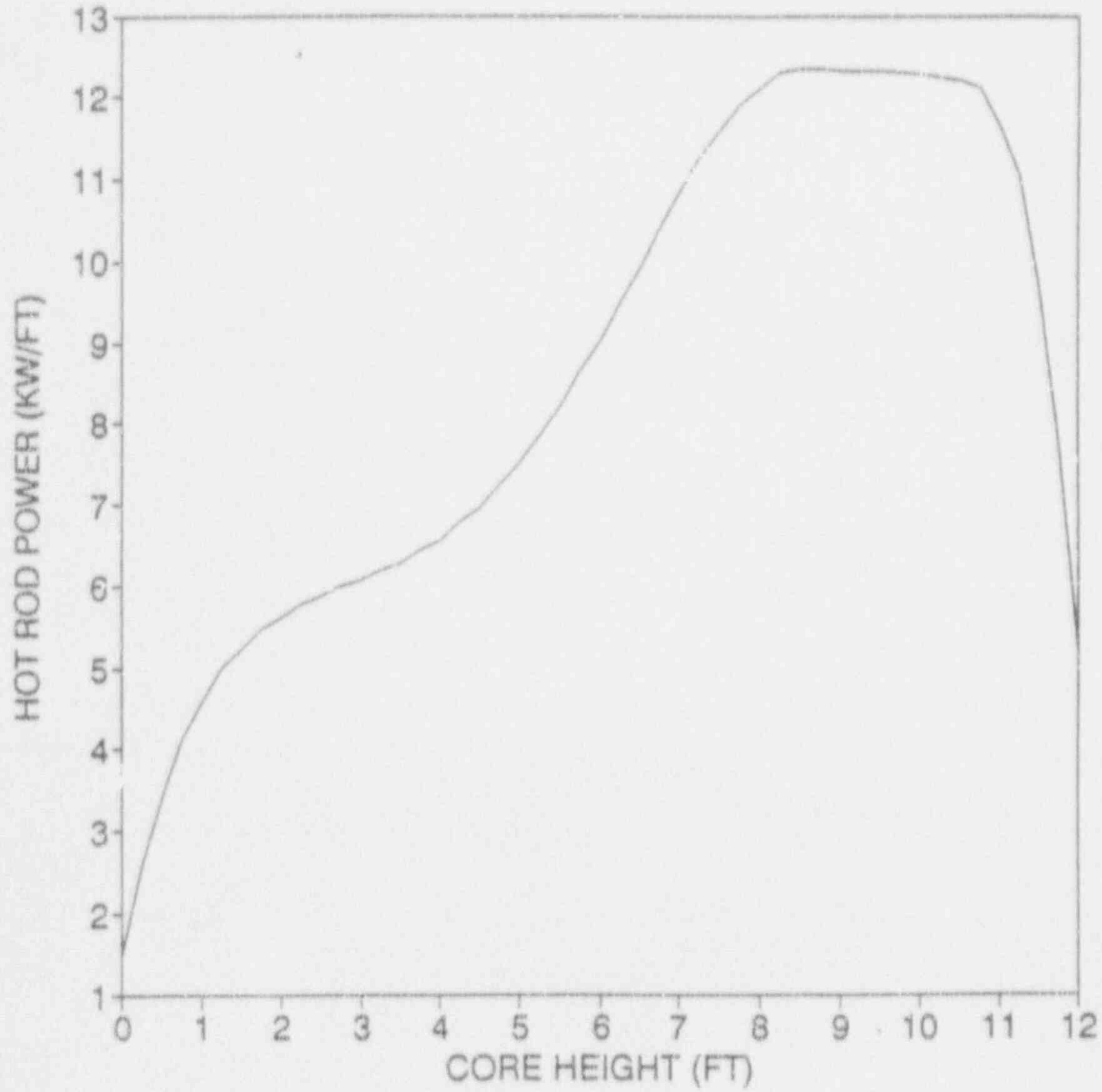
FIGURE 15.6-87



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Clad Temperature Transient
(4 Inch Break)

FIGURE 15.6-88



COMANCHE PEAK S.E.S.
FINAL SAFETY ANALYSIS REPORT
UNIT 2

Small Break
Power Distribution

FIGURE 15.6-89

FSAR Page (as amended)	Group	Description
4.3-33, 74	2	Expand discussion to recognize that operations support is required in addition to testing and to identify that NRC approved TU Electric methodology is employed for both testing and operation support. Revision: NRC has approved TU Electric Control Rod Analysis methodology for generic application. The TU Electric Steady State Physics methodology has been approved for use on Unit 1. The FSAR has been revised to reflect the intent to apply the methodology to Unit 1 reload cycles and Unit 2 initial startup and reload cycles. FSAR Change Request Number: 91-141.99 SER/SSER Impact: Yes Safety evaluation written by NRC on TU Electric Steady State Physics topical only identified applicability to Unit 1. Applicability is being expanded to include use on Unit 2.
Table 4.3-2, B	2	See Sheet No(s):1,2 Correct Nuclear Design Parameters Correction: The boron coefficient for boron dilution was not reanalyzed for a change in the Unit 2 shutdown margin until after Amendment 84 was incorporated into the FSAR. This number is being corrected for the reanalysis. In addition, a typographical error was corrected for neutron lifetime (20.7 microseconds vice 21.7). FSAR Change Request Number: 91-141.35 SER/SSER Impact: No
15.0-12	3	Correct Statement Concerning Axial Power Shapes Correction: In Amendment 84, changes were made to reflect the Unit 2 differences, but a statement concerning the axial power shapes used in the DNB calculations was not changed to reflect Unit 2. The Unit 2 analyses do not use a 1.55 chopped cosine shape. The axial power shapes for each unit are discussed in section 4.4. FSAR Change Request Number: 91-141.05 SER/SSER Impact: No
Table 15.0-2	2	See Sheet No(s):3 Correct Reactivity Coefficients Used in the Rod Withdrawal at Power Accident Correction: In Amendment 84, the reactivity coefficients were modified to reflect the Unit 2 analyses and the reanalyses associated with the change to a positive moderator co-

FSAR Page (as amended)	Group	Description
		efficient in Unit 1. The value for the moderator density coefficient was inadvertently left out of the Amendment. The correct value of +.43 delta k/gm/cc has been added to the Table for the Rod Withdrawal at Power Accident. FSAR Change Request Number: 91-141.05 SER/SSER Impact: No
Table 15.0-2	2	See Sheet No(s):5 Add Unit 2 Computer Codes and Initial Conditions for the Large and Small Break Loss of Coolant Accidents Addition: Table 15.0-2 summarizes the Computer Codes and the range of initial conditions assumed in the analyses. The Table has been revised to include the Unit 2 codes and initial conditions. FSAR Change Request Number: 91-141.05 SER/SSER Impact: No
Figure 15.0-2, B	2	Correct Doppler Power Coefficients Used in the Accident Analysis Correction: In Amendment 84, Figure 15.0-2B was inserted for the doppler coefficients used in the Unit analyses. The Figure inadvertently inverted the upper and lower doppler curves. A new Figure 15.0-2B has been provided. FSAR Change Request Number: 91-141.05 SER/SSER Impact: No
Table 15.1-3	2	See Sheet No(s):3 Parameters for Postulated Main Steam Line Break Accident Revision: The Unit 2 parameters have been added to Table 15.1-3. Unit 2 has a different parameters because the steam generators are different from Unit 1. The parameters have also changed for Unit 1 because of a change in the relief valve setpoint. FSAR Change Request Number: 91-141.99 SER/SSER Impact: No
15.4-33, 34	2	See Page No(s):37, 38 Add Unit 2 Parameters and Results for Boron Dilution Event During Startup and Power Operation Addition: The Boron Dilution Event has bee reanalyzed for Unit 2. The Unit 2 input parameters (e.g., RCS volume, boron boron worth, etc.) and results (e.g., time for operator

FSAR Page (as amended)	Group	Description
		action, etc.) have been added to the discussion. FSAR Change Request Number: 91-141.52 Related SER Section: 15.2.3.1 SER/SSER Impact: No
Table 15.4-1	2	See Sheet No(s):4, 5 Add Unit 2 Sequence of Events for Boron Dilution Event During Startup and Power Operation Addition: The Boron Dilution Event has been reanalyzed for Unit 2. A new column has been added to Table 15.4-1 to add the times for the Boron Dilution Event during startup and power operation. FSAR Change Request Number: 91-141.52 Related SER Section: 15.2.3.1 SER/SSER Impact: No
15.6-14	2	Remove Specific Referance to the DNBR of 1.30 Addition: Each occurrence of "DNBR is less than 1.30" (departure from nucleate boiling ratio) has been replaced with a generic term, "less than the limit value" because the DNBR is different for each unit and may change in the future if the fuel type changes. or if a different analytical method is used in the analysis. This change does not have a material effect and has been made to preclude repetitive changes in the future. The DNBR limit is listed in Technical Specification 3/4.2 for each unit. FSAR Change Request Number: 91-141.99 SER/SSER Impact: No
15.6-22, 24	2	See Page No(s):26 thru 35 Add Unit 2 Assumptions, Initial Conditions and Results for the Large and Small Break Loss of Coolant Accidents Addition: The Large and Small Break Loss of Coolant Accidents were analyzed differently for Unit 2. The Large Break LOCA was analyzed using the approved 1981 ECCS evaluation model. The Small Break LOCA was analyzed using the May, 1985 NOTRUMP ECCS evaluation model. The assumptions, initial conditions and results, which are different for Unit 2, have been added. FSAR Change Request Number: 91-141.01 SER/SSER Impact: No
15.6-32	1	Changes to Peak Clad Temperature (PCT) Penalties and Final Limiting PCT for Unit 1 Large Break LOCA

FSAR Page (as amended)	Group	Description
		Revision: An error in the ECCS calculation for Unit 1 resulted in a PCT penalty of 7.2 degrees Fahrenheit (F) which increased the total PCT penalty to 55 degrees F and the final limiting PCT to 2065.7 degrees F. The 7.2 degree F penalty was for steam generator tube collapse due to concurrent seismic and LOCA loads. TU Electric notified the NRC that the total PCT penalty exceeded 50 degrees F, in accordance with 10CFR50.46, via letter TXX-91230 dated July 31, 1991, and provided a schedule for reanalysis. FSAR Change Request Number: 91-141.99 Related SSER Section: SSER23 15.3.8 SER/SSER Impact: Yes The Large Break PCT is stated as 2058.5 degrees F.
15.6-32	1	Changes to Peak Clad Temperature (PCT) Penalties and Final Limiting PCT for Unit 2 Large Break LOCA Revision: Errors in the ECCS calculation for Unit 2 result in a total PCT penalty which exceeds 50 degrees Fahrenheit (F). TU Electric notified the NRC that the total PCT penalty exceeded 50 degrees F, in accordance with 10CFR50.46, via letter TXX-91270 dated July 31, 1991, and provided a schedule for reanalysis. FSAR Change Request Number: 91-141.99 SER/SSER Impact: No
15.6-34	1	Changes to Peak Clad Temperature (PCT) Penalties and Final Limiting PCT for Unit 1 Small Break LOCA Revision: A correction to the ECCS calculation has been made to account for zirconium-water reaction, and safety injection and auxiliary feedwater flow adjustment. The correction increases the total PCT penalty to 247 degrees Fahrenheit (F) and the final limiting small break PCT to 20[illegible] degrees F. FSAR Change Request Number: 91-141.99 Related SSER Section: SSER23 15.3.8 SER/SSER Impact: Yes The Small Break PCT is stated as 1895.5 degrees F.
15.6-49, 50	2	See Page No(s):51 Add Unit 2 References to the Reference List Addition: The source materials for the Unit 2 analyses have been added to the reference list. FSAR Change Request Number: 91-141.02 SER/SSER Impact: No

FSAR Page (as amended)	Group	Description
Table 15.6-1	2	See Sheet No(s):01 thru 06 Add Unit 2 Times to Sequences of Events for the Loss of Coolant Accidents and Steam Generator Tube Rupture Addition: The Loss of Coolant Accidents have been analyzed for Unit 2. A new column has been added to Table 15.6-1 for the Unit 2 sequence of events. The times for the Steam Generator Tube Rupture event are identical to Unit 1 because the same analysis is used for both units. FSAR Change Request Number: 91-141.04 SER/SSER Impact: No
Table 15.6-1	3	See Sheet No(s):5 Table 15.6-1, Time Sequence of Events for Decrease in Reactor Coolant Inventory Correction: Incorrect duplicate entries have been removed for the 3 inch break. Duplicate entries for Accumulator Injection, Peak Clad Temperature, and Top of Core Covered, stated times which were the times for the 6 inch break. FSAR Change Request Number: 91-141.99 SER/SSER Impact: No
Table 15.6-5	2	Add Unit 2 Parameters and Results of Loss of Coolant Accidents to Table 15.6-5 Addition: Table 15.6-5 summarizes the input parameters for the LOCA analysis. The Large and Small Break LOCAs have been analyzed differently for Unit 2. The input parameters for Unit 2 have been added to the Table. FSAR Change Request Number: 91-141.45 SER/SSER Impact: No
Table 15.6-5	3	Table 15.6-5, Input Parameters Used in the LOCA Analysis Correction: The LOCA analyses are run at 3651 megawatts for the thermal hydraulic analyses and at 3411 megawatts to determine the cladding heatup. FSAR Change Request Number: 91-141.99 SER/SSER Impact: No
Table 15.6-6	2	See Sheet No(s):01 and 02 Add Unit 2 Parameters and Results for Large Break LOCA to Table 15.6-6 Addition: Table 15.6-6 summarizes the input parameters and re-

FSAR Page (as amended)	Group	Description
		sults for the Large Break LOCA analysis. The Large Break LOCA was analyzed differently for Unit 2. The Unit 2 input parameters and results have been added to the Table. FSAR Change Request Number: 91-141.45 SER/SSER Impact: No
Table 15.6-7	2	Add Unit 2 Small Break LOCA Results to Table 15.6-7 Addition: Table 15.6-7 summarizes the results for the Small Break LOCA analysis. The Small Break LOCA was analyzed for Unit 2 with a different methodology. The Unit 2 results have been added to Table 15.6-7. FSAR Change Request Number: 91-141.45 SER/SSER Impact: No
Figure 15.6-6	2	Add Unit 2 Figure for Small Break LOCA Computer Code Addition: The Unit 2 Small Break LOCA analysis uses the NOTRUMP code. Figure 15.6-6 has been changed to reflect this difference from Unit 1. FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-47, (A&B)	2	Update Existing Safety Injection Flow Rate Figures Revision: The Safety Injection Flow Rate Figures for the Large and Small Break LOCAs have been updated to account for plant changes to the Residual Heat Removal system. FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-49	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-49, A	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-50	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No

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Figure 15.6-51	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-51, A	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-52	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-52, A	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-53	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-54	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-54, A	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-55	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-56	2	Add Unit 2 Figures for Large Break LOCA Addition:

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		FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-57	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-58	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-59	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-60	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-61	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-62	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-63	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-64	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No

FSAR Page (as amended)	Group	Description
Figure 15.6-65	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-66	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-67	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-68	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-69	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-70	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-71	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-72	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-73	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No

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Figure 15.6-74	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-75	2	Add Unit 2 Figures for Large Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-76	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-77	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-78	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-79	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-80	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-81	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-82	2	Add Unit 2 Figures for Small Break LOCA Addition:

FSAR Page (as amended)	Group	Description
		FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-83	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-84	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-85	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-86	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-87	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-88	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No
Figure 15.6-89	2	Add Unit 2 Figures for Small Break LOCA Addition: FSAR Change Request Number: 91-141.47 SER/SSER Impact: No