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ATTACHMENT A

Technical Specification
Page Revisions

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

April, 1980

2 SAFETY LIMTTS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, and coolant temperaure during four-loop and three-loop operation, and reactor coolant flow during four-loop operation.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figures 2.1-1 and 2.1-2 for four and three-loop operation respectively (additional limitations on three-loop operation are described in section 3.1.A). The safety limit is exceeded if the point defined by the combination of Reactor Coolant System average temperature and power level is at any time above the appropriate pressure line.

The Region 1 fuel residence time shall be limited to 21,000 effective full power hours (EFPH) under design operation conditions. The licensee may propose to operate individual assemblies from Region 1 in excess of 21,000 EFPH by providing an analysis which includes the effect of clad flattening or a change in operation conditions. Any such analysis, if proposed, shall be approved by the Regulatory Staff prior to operation in excess of 21.000 EFPH.

The following DNB related parameters pertain to four loop steady state operation at power levels greater than 98% of rated full power (in excess of 2703 MWt):

a. Reactor Coolant System $T_{avg} \leq 573.5^{o}F$
b. Pressurizer Pressure $\geq$ 2220 psia
c. Reactor Coolant System Total Flow Rate $\geq$ 340,800 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Rod withdrawal block and load runback occurs if reactor trip setpoints are approached within a fixed limit.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than 1.30.[4]

The ranges on reactor coolant system temperature, pressure and loop coolant flow [5] during steady-state, four-loop, power operation are specified to assure that the values assumed in the accident analyses are not exceeded during normal plant operation.

Compliance with the specified ranges on reactor coolant system temperature and pressurizer pressure is demonstrated by verifying that the parameters are within their applicable ranges at least once each 12 hours.

Compliance with the specified range on Reactor Coolant System total flow rate is demonstrated by verifying the parameter is within it's range after each refueling cycle.

References

[1] FSAR Section 3.2.2

[2] FSAR Section 3.2.1

[3] FSAR Technical Specification 3.10

[4] FSAR Section 14.1.1

[5] "Analysis and Evaluation of Non-LOCA Transients for Operation with 95% Reactor Coolant System Thermal Design Flow and with 25% Uniform Steam Generator Tube Plugging," dated April, 1980.

References

(1) FSAR Section 9

(2) FSAR Section 6.2

(3) FSAR Section 6.2

(4) FSAR Section 6.3

(5) FSAR Section 14.3.5

(6) FSAR Section 1.2

(7) FSAR Section 8.2

(8) FSAR Section 9.6.1

(9) FSAR Section 14.3

(10) Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and Appendix K of 10CFR50", dated December 1978, and "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10CFR50.46 and 10 CFR Part 50, Appendix K," dated April, 1980.

(11) Letter from William J. Cahill, Jr. of Consolidated Edison Company of New York, to Robert W. Reid of the Nuclear Regulatory Commission, dated July 13, 1976. Indian Point Unit No. 2 Small Break LOCA Analysis.

(12) Indian Point Unit No. 3 FSAR Sections 6.2 and 6.3 and the Safety Evaluation accompanying "Application for Amendment to Operating License" sworn to by Mr. William J. Cahill, Jr. on March 28, 1977.

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Ojbectives:

To ensure:

1. Core subcriticality after reactor trip,

2. Acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation and transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and

3. Limit potential reactivity insertions caused by hypothetical control rod ejection.

Specifications:

3.10.1 Shutdown Reactivity

The shutdown margin shall be at least as great as shown in Figure 3.10-1.

3.10.2 Power Distribution Limits

3.10.2.1 At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

(a) $F^N_{\Delta H} \leq 1.55 \left[1 + 0.2\ (1-P)\right]$

(b) For $\leq$ 6% steam generator tube plugging:

$F_Q(Z) \leq (2.31/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.62) \times K(Z)$ for $P \leq .5$

(c) For $>$ 6% but $\leq$ 12% steam generator tube plugging:

$F_Q(Z) \leq (2.25/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.50) \times K(Z)$ for $P \leq .5$

where P is the fraction of full power at which the core is operating; K(Z) is the fraction given in Figure 3.10-2a (for $\leq$ 6% tube plugging) or Figure 3.10-2b (for $>$ 6% but $\leq$ 12% tube plugging); and Z is the core height location of F_Q.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

The upper bound envelope of the total peaking factor (F_Q) of specification 3.10.2.1 times the normalized peaking factor axial dependence of Figures 3.10-2a and b has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10. The results of the loss of coolant accident analyses based on the specified F_Q times the normalized envelope of Figures 3.10-2a and b indicate a peak clad temperature of less than 2200°F for the double-ended cold leg guillotine break with C_D=0.6, the worst case break.(1) (2)

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the specified limit of $F^N_{\Delta H}$ there is a 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control assure that the total peaking factor upper bound envelope of specified F_Q times Figures 3.10-2a and b is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though flux difference is then within the limits specified by the procedure.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the control rod bank more than 190 steps withdrawn (i.e. normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ±5 percent ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure 3.10-5 shows a typical construction of the target flux difference band at BOL and Figure 3.10-6 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux

accident for an isolated fully inserted rod will be worse if the residence time of the rod is long enough to cause significant non-uniform fuel depletion. The 4 week period is short compared with the time interval required to achieve a significant non-uniform fuel depletion.

The required drop time to dashpot entry is consistent with safety analysis.

REFERENCE

1. Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in Accordance with the Acceptance Criteria of 10 CFR 50.46 and Appendix K of 10 CFR 50." See also Consolidated Edison Company's letter to NRC dated January 5, 1979 which submitted the results of this reanalysis based on the Westinghouse ECCS Evaluation Model approved by NRC letter to Westinghouse dated August 29, 1978.

2. Indian Point Unit No. 2, "Analysis of the Emergency Core Cooling System in accordance with the acceptance criteria of 10CFR50.46 and 10 CFR Part 50, Appendix K," dated April, 1980.

Figure 3.10-2 a

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

(For S.G. tube plugging levels up to 6%)

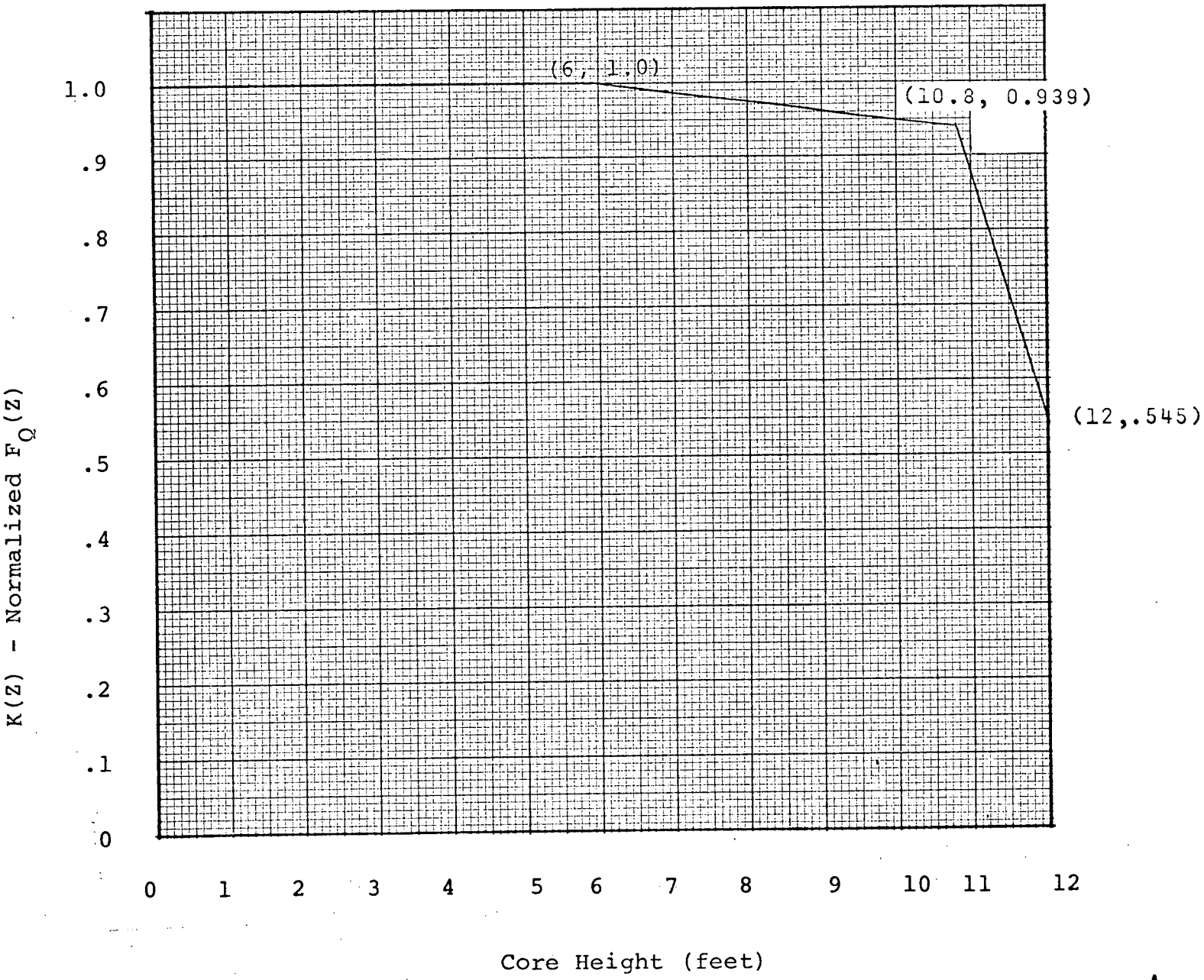

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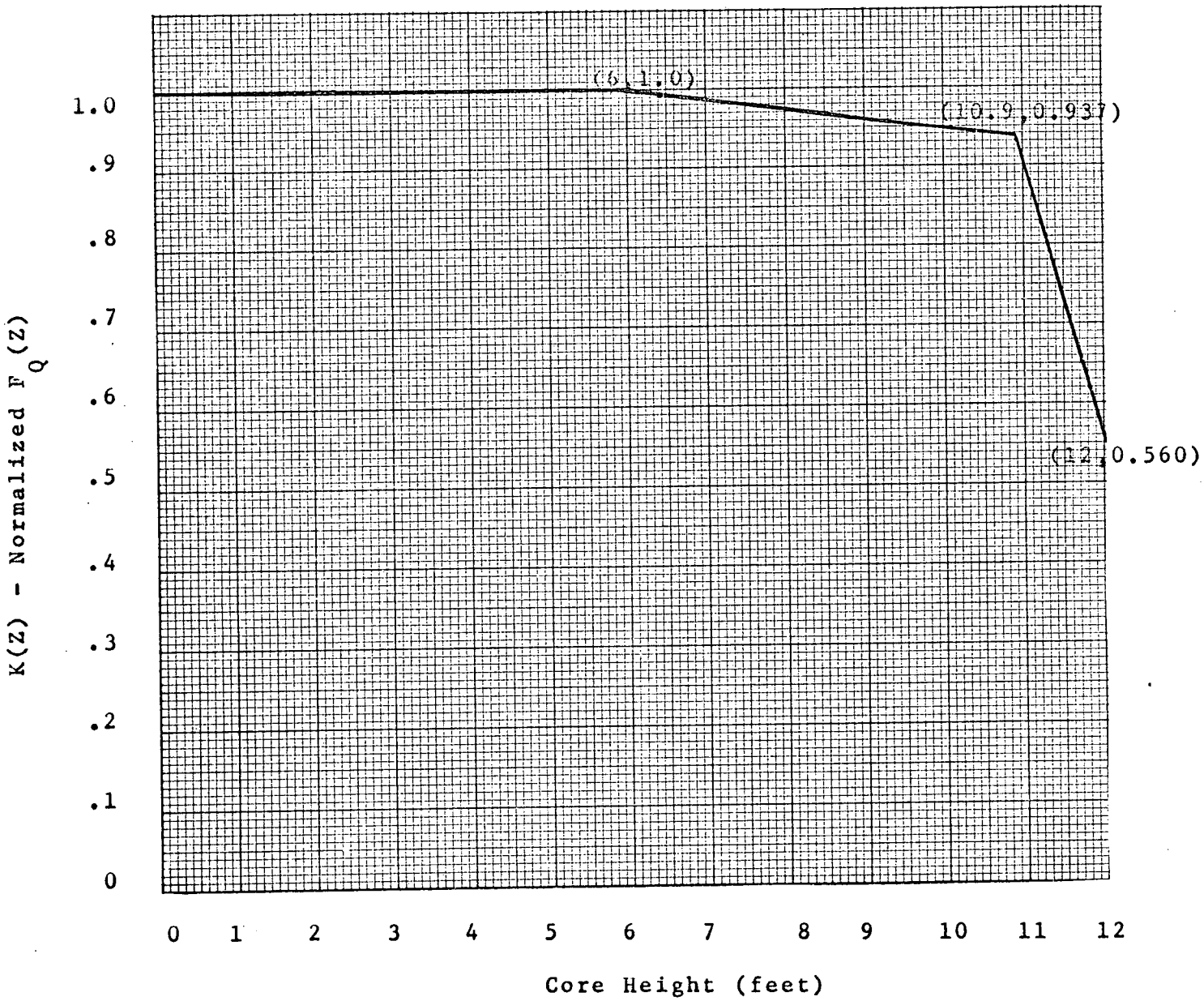
Figure 3.10-2b

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE
(For S.G. tube plugging levels > 6% but ≤ 12%)



Amendment No.

ATTACHMENT B

Safety Evaluation

Consolidated Edison Company of New York, Inc.

Indian Point Unit No. 2

Docket No. 50-247

April, 1980

Safety Evaluation

The proposed changes, contained in Attachment A to this Application, would modify certain operational parameters in the Indian Point Unit No. 2 Technical Specifications based on the most recent ECCS-LOCA and DNB-NonLOCA transient evaluations.

As presented in the attached Enclosure 1, an ECCS reanalysis has been performed for the limiting case (DECLG, C_D=0.6) using a 12% uniform steam generator tube plugging limit (versus 6% used in the previous analysis). This reanalysis demonstrates that with a maximum total nuclear peaking factor (F_Q) of 2.25 and steam generator tube plugging up to 12%, ECCS performance will meet the acceptance criteria of 10CFR50.46 and 10CFR Part 50, Appendix K. Accordingly, changes are proposed to specification 3.10 to permit an F_Q of $\leq$2.31 up to 6% steam generator tube plugging (as presently allowed) and to conservatively require an F_Q of $\leq$2.25 for steam generator tube plugging greater than 6% but $\leq$12%. No other plant parameters are required to be changed.

In addition, the ECCS reanalysis of Enclosure 1 has been evaluated with regard to the recent issue raised by the NRC concerning those portions of NSSS vendors' ECCS evaluation models dealing with fuel cladding swelling, the incidence of rupture and fuel assembly blockage (Ref: draft NUREG-0630). This issue was addressed in Mr. Darrell G. Eisenhut's November 9, 1979 and November 27, 1979 letters to all licensees and was responded to for Indian Point Unit No. 2 by Con Edison's January 8, 1980 and January 24, 1980 letters to NRC. The same reevaluation addressed in those letters for the present plant operating mode (i.e., $F_Q \leq$2.31 and Plugging $\leq$ 6%) was performed for the proposed operating mode (i.e., $F_Q \leq$2.25 and Plugging $\leq$12%). As demonstrated in Enclosure 1A, it has been determined that all the acceptance criteria are still met at the proposed peaking factor of 2.25 using the NRC fuel models presented in draft NUREG-0630. Therefore, the reanalysis presented in Enclosure 1 is valid.

At the present time, Indian Point Unit No. 2 has a reactor coolant system flow measurement uncertainty of $\pm$4.2% at full power. When this flow uncertainty is applied in the negative direction to the last actual flow measurement taken at Indian Point Unit No. 2, the margin above the minimum required thermal design flow is found to be less than 1/2%. Accordingly, to gain additional margin to the minimum required reactor coolant flow, reanalyses and/or reevaluations, as presented in the attached Enclosure 2, were performed assuming 95% of the thermal design flow. As demonstrated in Enclosure 2, all safety limits are still met at the reduced flow condition.

The analyses addressed in Enclosure 2 have assumed a steam generator plugging level of 25%. However, ECCS/LOCA reanalyses have not been performed to justify operation at steam generator plugging levels up to 25% at this time. All that is justified is that the 95% flow non-LOCA transient reanalyses/reevaluations described in Enclosure 2 are valid for steam generator plugging levels up to 25%. Therefore, changes are proposed to specification 2.1 to permit a reduction in the minimum required reactor coolant system flow to $\geq$ 340, 800 gpm commensurate with the results of the analyses presented in Enclosure 2.

The proposed changes have been reviewed by both the Station Nuclear Safety Committee and the Consolidated Edison Nuclear Facilities Safety Committee. Both Committees concur that the proposed changes do not represent a significant hazards consideration and will not cause any change in the types or an increase in the amounts of effluents or any change in the authorized power level of the facility.

Enclosure 1

ANALYSIS OF THE EMERGENCY CORE COOLING SYSTEM IN ACCORDANCE WITH THE ACCEPTANCE CRITERIA OF 10CFR50.46 AND 10 CFR PART 50, APPENDIX K

CONSOLIDATED EDISON COMPANY
OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FACILITY OPERATING LICENSE
NO. DPR-26

April, 1980

ANALYSIS OF THE EMERGENCY CORE COOLING SYSTEM IN ACCORDANCE WITH THE ACCEPTANCE CRITERIA OF 10CFR50.46 AND 10 CFR PART 50, APPENDIX K

The analysis specified by the Nuclear Regulatory Commission in the "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors per 10CFR50.46 and 10 CFR Part 50, Appendix K (Reference 1), is presented herein for Indian Point Unit No. 2 for the limiting case (DECLG, C_D=0.6) at a 12% uniform steam generator tube plugging limit (versus 6% for the previous analysis).

The analytical techniques used are in compliance with 10 CFR Part 50, Appendix K, and are described in Reference 2. The individual computer codes which comprise the Westinghouse Emergency Core Cooling System (ECCS) evaluation model are described in detail in References 3,4,5 and 6 along with the code modifications specified in References 7,8 and 9. The analysis presented here was performed with the February 1978 version of the evaluation model which includes modifications delineated in References 10, 11, 12 and 13.

The boundary considered for loss of coolant accidents as related to connecting pipings is defined in Section 4.1.3 of the IP3 FSAR.

Should a major break occur, depressurization of the Reactor Coolant System results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. A Safety Injection

System signal is actuated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat.

b. Injection of borated water provides heat transfer from the core and prevents excessive clad temperature.

At the beginning of the blowdown phase, the entire Reactor Coolant System contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with 10CFR Part 50, Appendix K. Thereafter, the core heat transfer is based on local conditions with transition boiling and forced convection to steam as the major mechanisms. During the refill period, rod-to-rod radiation is the only mechanism.

When the Reactor Coolant System pressure falls below 600 psia, the accumulators begin to inject borated water. The conservative assumption is made that accumulator water injection bypasses the core and goes out through the break until the termination of bypass. This conservatism is consistent with 10CFR Part 50, Appendix K.

Thermal Analysis

Westinghouse Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss-of-coolant accident including the double ended severance of the largest Reactor Coolant System pipe. The reactor core and internals together with the Emergency Core Cooling System are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident. The Emergency Core Cooling System even when operating with the most severe single active failure is designed to meet the Acceptance Criteria.

Method of Thermal Analysis

The description of the various aspects of the LOCA analysis is given in Reference 2 . This document describes the major phenomena modeled, the interface anong the computer codes and features of the codes which maintain compliance with the Acceptance Criteria. The individual codes are described in detail in References 3,4,5 and 6.

The analysis presented in this submittal is performed for the DECLG break with C_D = 0.6 and with a reactor vessel fluid inlet temperature of 521.4 F (vessel average temperature of 549°F) and consistent secondary-side initial conditions. The analysis includes the effects of 12% uniform steam generator tube plugging. Reference 14 demonstrated the 0.6 DECLG to be limiting and the present analysis differs from Reference 14 in the level of steam generator tube plugging assumed (12% versus 6%) and in the maximum total peaking factor (2.25 vs. 2.31).

The analysis considers the reactor vessel upper head fluid temperature to be equal to the Reactor Coolant System hot leg fluid temperature. The effect of using the hot leg temperature in the reactor vessel upper head region is described in Reference 15. Reference 16 presents a break spectrum sensitivity study using the increased upper head fluid temperature. Analyses for small breaks in the reactor coolant piping are presented in Reference 17.

Results

Table IP2-1 presents the peak clad temperature and hot spot metal water reaction. The time sequence of events during the large break is shown in Table IP2-2.

The SATAN-IV blowdown and reflood analysis of the loss-of-coolant accident, and the peak linear power used in the rod heat up calculations (LOCTA-IV code) is based on 102 percent of the licensed core power rating. Since there is margin between the value of the peak linear power density used in this analysis and the value expected in operation, a lower peak clad temperature would be expected by using the peak linear power density expected during operation.

For the results discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table IP2-1.

Figures IP2-1 through IP2-17 present the transient behavior of the principal parameters, as follows:

Figures IP2-1 through IP2-3	These figures show the fluid quality, the mass velocity and the heat transfer coefficient (as calculated by the LOCTA-IV code) at the hot spot (location of maximum clad temperature) and burst location, on the hottest fuel rod (hot rod).
Figures IP2-4 through IP2-6	These figures show the core pressure, the flow rate out of the break (the sum of both ends for the guillotine break) and the core pressure drop (from the lower plenum near the core, to the upper plenum at the core outlet).
Figures IP2-7 through IP2-9	These figures show the clad temperature transient at the hot spot and the burst location, the fluid temperature (also for the hot spot and burst location), and the core flow (top and bottom).
Figures IP2-10 through IP2-11	These figures show the core reflood transient parameters (water level and flooding rate).
Figures IP2-12 through IP2-13	These figures show the Emergency Core Cooling System flow. The accumulator flow assumed is the sum of that injected in the intact cold legs.

Figure IP2-14	This figure shows the containment pressure transient.
Figure IP2-15	This figure shows the core power transient.
Figure IP2-16	This figure shows the break energy released to the containment.
Figure IP2-17	This figure shows the containment wall condensing heat transfer coefficient.

In addition to the above, Tables IP2-4 and IP2-5 present the reload mass and energy releases to the containment, and the broken loop accumulator mass and energy release to the containment respectively.

The clad temperature analysis is based on a total peaking factor of 2.25. The hot spot metal-water reaction reached is well below the embrittlement limit of 17% as required by 10CFR50.46. In addition, the total core metal-water reaction is less than 0.3% as compared to the 1% criterion of 10CFR50.46.

The results of several sensitivity studies are reported in Reference 18. These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

Conclusions-Thermal Analysis

For breaks up to and including the double ended severance of a reactor coolant pipe, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46. That is:

1. The calculated peak fuel element clad temperature does not exceed 2200°F based on a total core peaking factor of 2.25.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of Zircaloy in the reactor.
3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17% are not exceeded during or after quenching.
4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the longlived radioactivity remaining in the core.

References

1. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors", 10CFR50.46 and Appendix K of 10CFR50.46. Federal Register, Volume 39, Number 3, January 4, 1974.

2. Bordelon, F. M., et al., "The Westinghouse ECCS Evaluation Model - Summary", WCAP-8339; July 1974.

3. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant", WCAP-8306 (Non-Proprietary), June 1974.

4. Bordelon, F. M., et al, "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8301 (Proprietary), WCAP-8305 (Non-Proprietary), June 1974.

5. Kelly, R.D., et al., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)", WCAP-8171 (Non-Properietary), June 1974.

6. Bordelon, F. M., and Murphy, E. T., "Containment Pressure Analysis Code (COCO)", WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), June 1974.

7. Bordelon, F. M., et al., "The Westinghouse ECCS Evaluation Model - Supplementary Information",WCAP-8471 (Proprietary), WCAP-8472 (Non-Proprietary), January 1975.

8. "Westinghouse ECCS Evaluation Model, October 1975 Version", WCAP-8622 (Proprietary), and WCAP-8623 (Non-Proprietary), November 1975.

9. Letter from C. Eicheldinger of Westinghouse Electric Corporation to D. B. Vassalo of the Nuclear Regulatory Commission, Letter Number NS-CE-924, dated January 23, 1976.

10. Kelly, R. D., Thompson, C. M., et al., "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing Large LOCA's During Operation With One Loop Out of Service for Plants Without Loop Isolation Valves," WCAP-9166, February, 1978.

11. Eicheldinger C., "Westinghouse ECCS Evaluation Model, February 1978 Version", WCAP-9220-P-A (Proprietary), WCAP-9221-A (Non-Proprietary), February, 1978.

12. Letter from T. M. Anderson of Westinghouse Electric Corporation to John Stolz of the Nuclear Regulatory Commission, Letter Number NS-TMA-1981, Novmeber 1, 1978.

References (Cont'd)

13. Letter from T. M. Anderson of Westinghouse Electric Corporation to Tedesco of the Nuclear Regulatory Commission, Letter Number NS-TMA-2014, dated December 11, 1978.

14. Letter from W. J. Cahill, Jr. of Consolidated Company of New York, Inc. to Harold R. Denton of the Nuclear Regulatory Commission, dated January 5, 1979. Indian Point Unit No. 2 Analysis of Emergency Core Cooling System in accordance with the acceptance criteria of 10CFR50.46 and Appendix K of 10 CFR50.

15. Letter from C. Eichelidinger of Westinghouse Electric Corporation to V. Stello of the Nuclear Regulatory Commission, Letter Number NS-CE-1163, dated August 13, 1976.

16. Beck, H. S., et al., "Westinghouse ECCS - Four Loop Plant (15x15) Sensitivity Studies with Upper Head Fluid Temperature at T Hot", WCAP-8855-A(Non-Proprietary), May 1977.

17. Letter from William J. Cahill, Jr. of Consolidated Edison Co. of New York, Inc. to Robert W. Reid of the Nuclear Regulatory Commission, dated July 13, 1976. Indian Point Unit No. 2 Small Break LOCA Analysis.

18. Salvatori, R., "Westinghouse ECCS - Plant Sensitivity Studies", WCAP-8340 (Proprietary), WCAP-8356 (Non- Proprietary), July 1974.

TABLE IP2-1

LARGE BREAK - RESULTS AND ANALYSIS INPUT

	DECLG (C_D=0.6)
Results	
Peak Clad Temp. °F	2182
Peak Clad Location Ft.	7.5
Local Zr/H_2O Rxn(max)%	4.75
Local Zr/H_2O Location Ft.	7.5
Total Zr/H_2O Rxn %	<0.3
Hot Burst Time sec.	32.8
Hot Rod Burst Location Ft.	6.0
Calculation	
NSSS Power Mwt 102% of	2758
Peak Linear Power kw/ft 102% of	13.01
Peaking Factor (At License Rating)	2.25
Accumulator Water Volume (ft^3 per tank)	716
Accumulator Pressure (psia)	600
Number of Safety Injection Pumps operating	2
Steam generator tube Plugging Level (%)	12 (uniform)

TABLE IP2-2

LARGE BREAK - TIME SEQUENCE OF EVENTS

OCCURRENCE TIME (SECONDS)

	DECLG (C_D=0.6)
Accident Initiation	0.0
Reactor Trip Signal	0.54
Safety Injection Signal	1.18
Start Accumulator Injection	15.9
Start Pumped ECC Injection	26.18
End of ECC Bypass	29.2
End of Blowdown	31.86
Bottom of Core Recovery	45.1
Accumulator Empty	49.0

TABLE IP2-3

LARGE BREAK CONTAINMENT DATA

NET FREE VOLUME	2.61×10^6 ft^3
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90 °F
RWST Temperature	40 °F
Service Water Temperature	35 °F
Outside Temperature	-20 °F
SPRAY SYSTEM	
Number of Pumps Operating	2
Runout Flow Rate	3000 gpm
Actuation Time	20 secs
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers Operating	5
Fastest Post Accident Initiation of Fan Coolers	30 secs
STRUCTURAL HEAT SINKS	
Thickness (In)	Area (Ft^2)
1. 0.007 Paint, 0.375 steel, 54.0 concrete	45,684
2. 0.007 Paint, 0.5 steel, 42.0 concrete	28,613
3. 12.0 concrete	15,000
4. 0.375 stainless steel, 12.0 concrete	10,000
5. 12.0 concrete	61,000
6. 0.5 steel	68,792
7. 0.007 Paint, 0.375 steel	81,704

TABLE IP2-3 (Continued)

LARGE BREAK CONTAINMENT DATA

Thickness (In)		Area (Ft^2)
8.	0.25 steel	27,948
9.	0.007 Paint, 0.1875 steel	69,800
10.	0.125 steel	3,000
11.	0.138 steel	22,000
12.	0.0625 steel	10,000
13.	0.019 stainless steel, 1.25 insulation, 0.75 steel, 54.0 concrete	785
14.	0.019 stainless steel, 1.25 insulation, 0.5 steel, 54.0 concrete	6849
15.	0.025 stainless steel, 1.5 insulation 0.5 steel, 54.0 concrete	3816
16.	0.025 stainless steel, 1.5 insulation, 0.375 steel, 54.0 concrete	4362

TABLE IP2-4

REFLOOD MASS AND ENERGY RELEASE TO THE CONTAINMENT

0.6 DECLG BREAK

Time (sec)	Mass (lbm/sec)	Energy (BTU/sec)
45.2	0.0	0.0
45.9	5.1	6589.2
50.3	26.6	34370.
62.3	29.5	38134.
80.0	55.2	69084.
98.6	82.6	101874.
116.8	102.4	125627.
169.2	364.0	215487.

TABLE IP2-5

BROKEN LOOP ACCUMULATOR MASS AND ENERGY

RELEASE TO THE CONTAINMENT

0.6 DECLG BREAK

Time (sec)	Mass (lbm/sec)	Energy (BTU/sec)
0.0	4288.	255679.
2.0	3417.	206976.
4.0	2997.	178708.
6.0	2673.	159373.
8.0	2431.	144934.
10.0	2239.	133482.
12.0	2080.	124035.
14.0	1946.	116012.
16.0	1830.	109084.
18.0	0.	0.

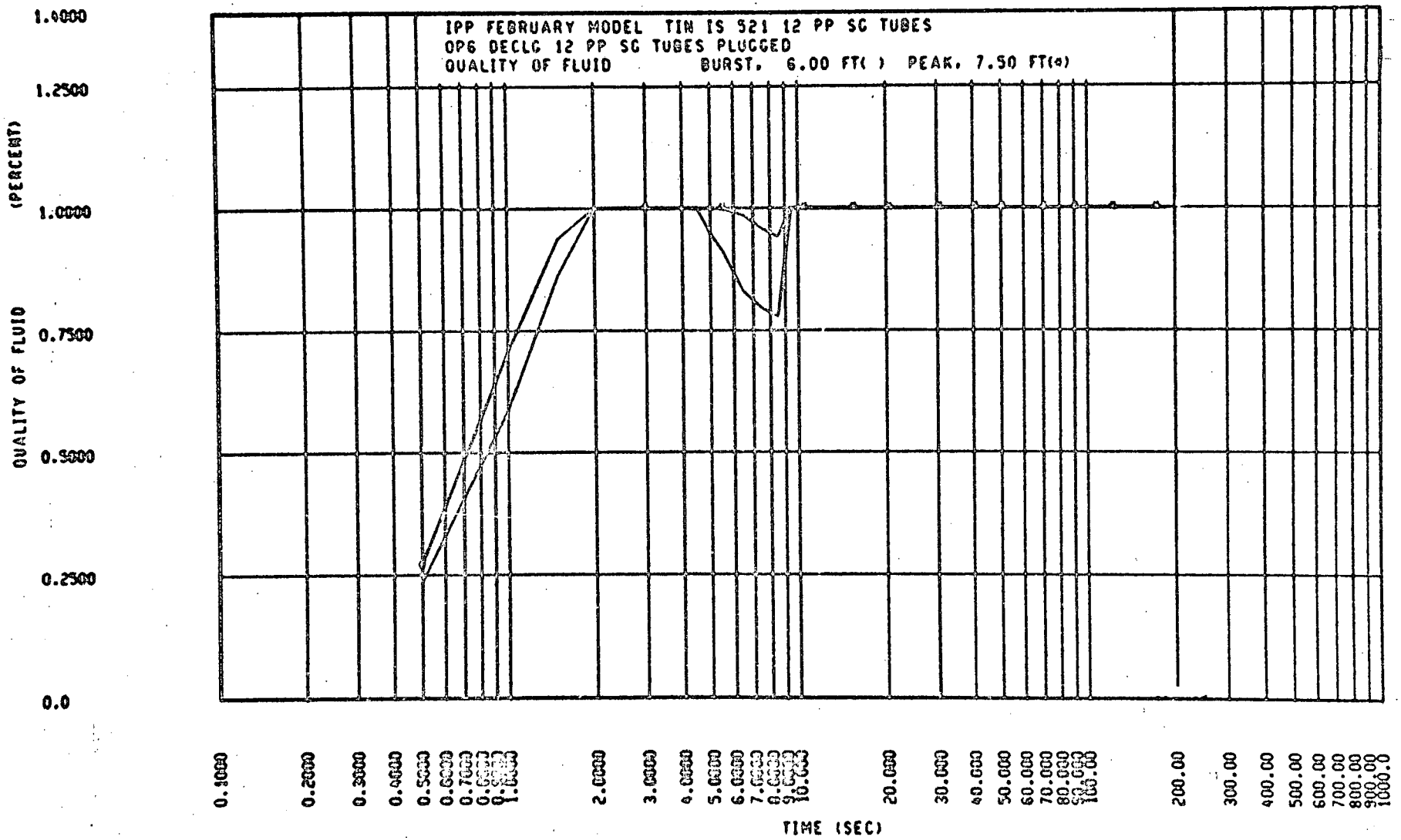


Figure IP2-1: Fluid Quality

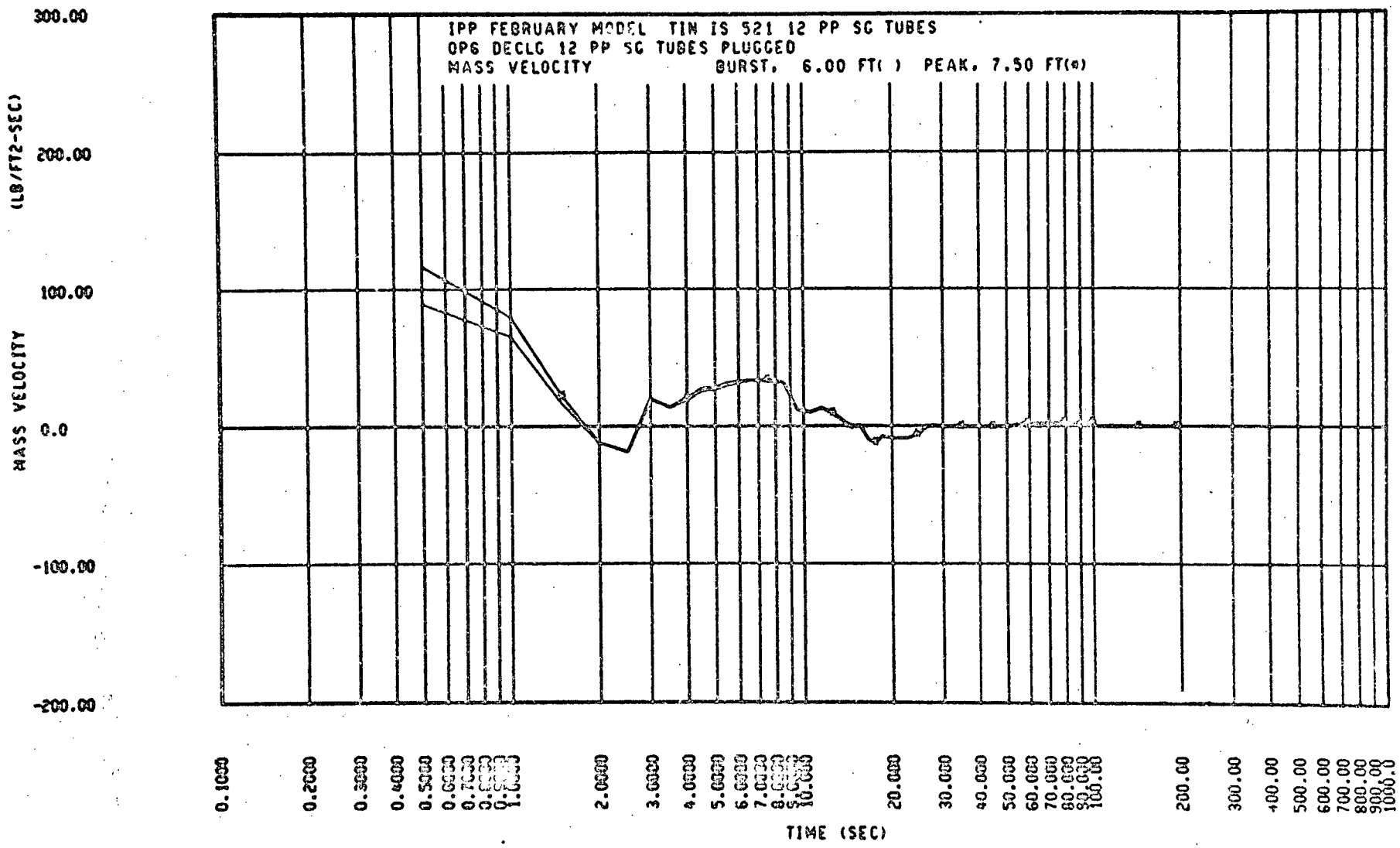


Figure IP2-2: Mass Velocity

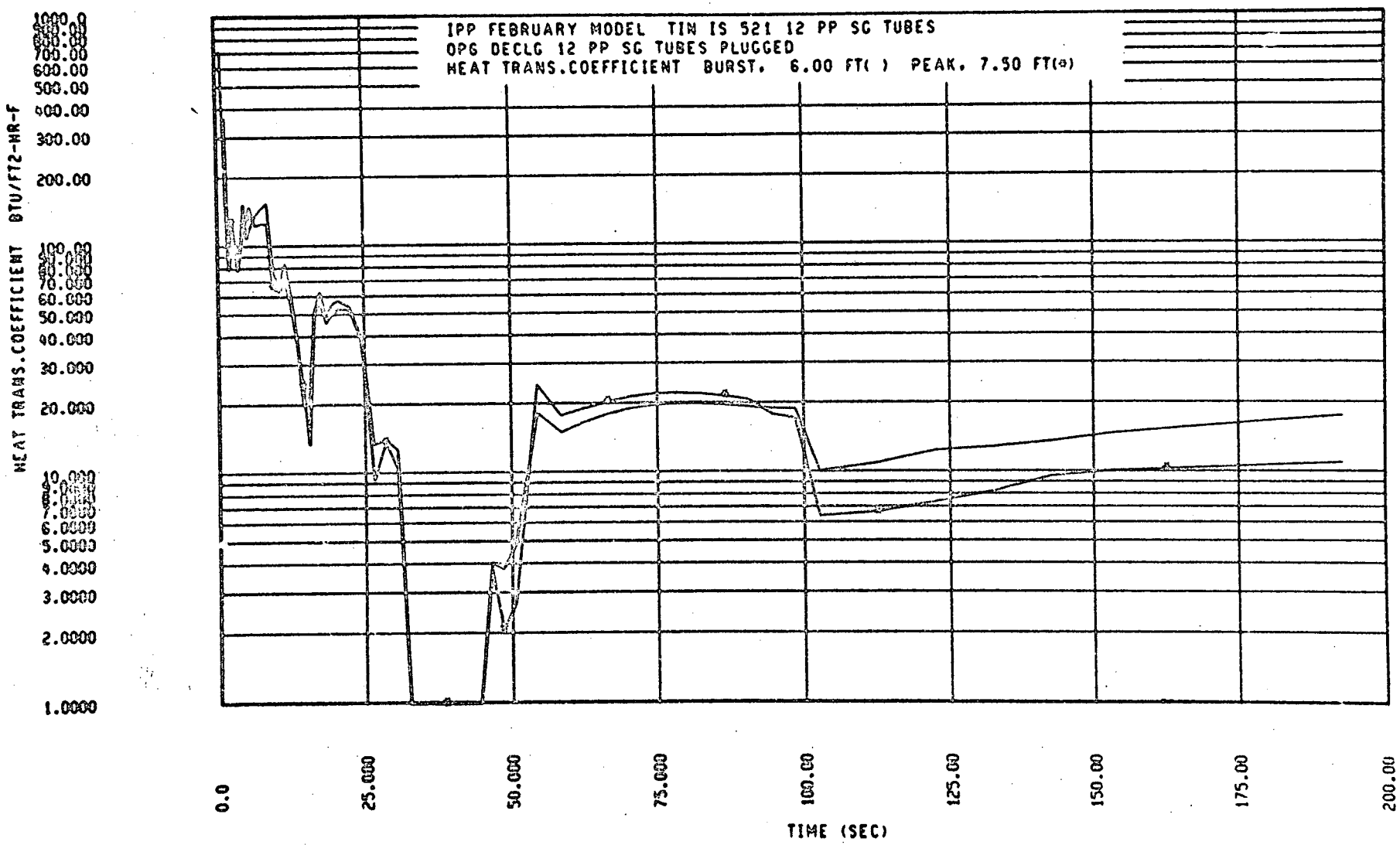


Figure IP2-3: Heat Transfer Coeffficient

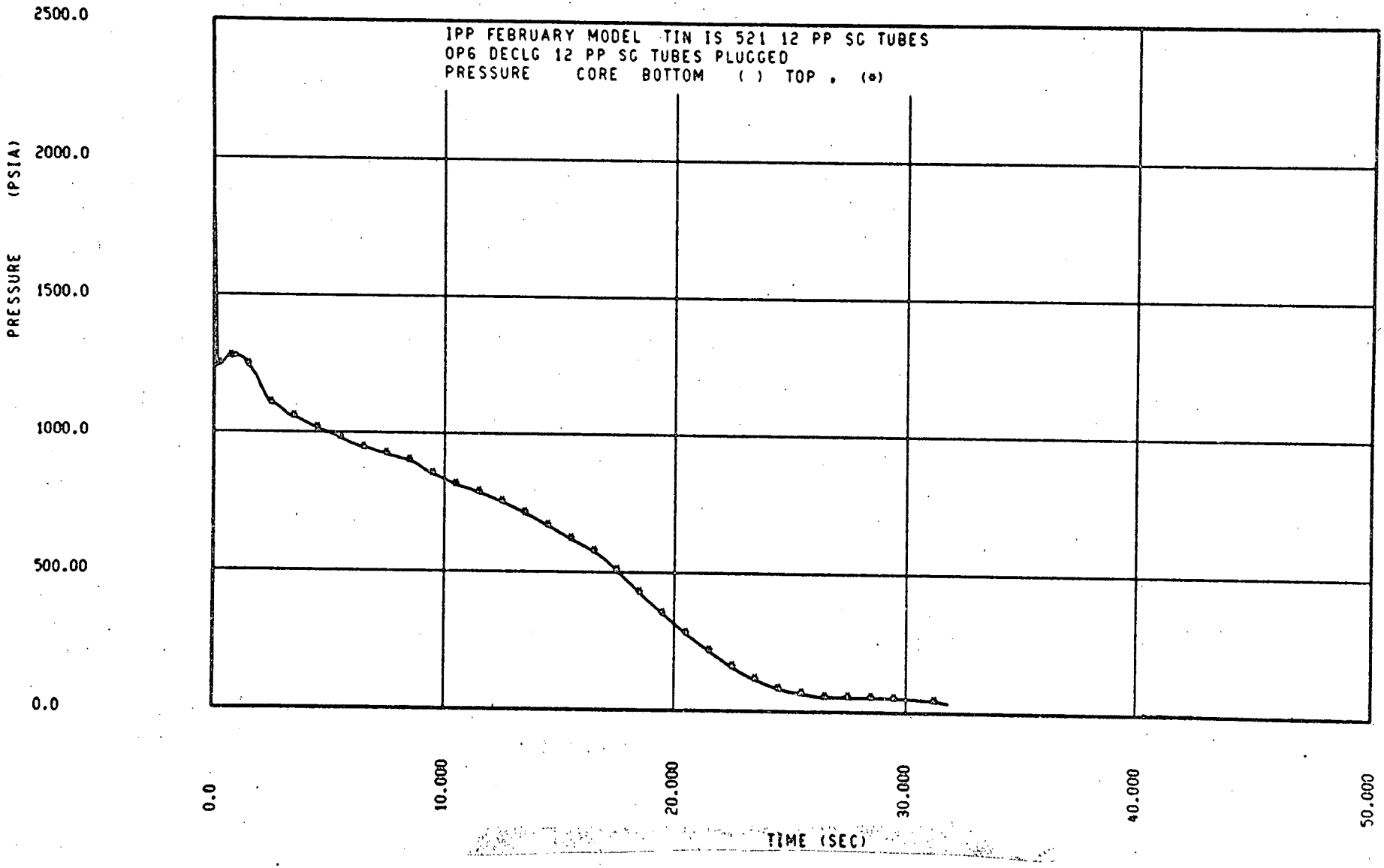


Figure IP2-4: Core Pressure

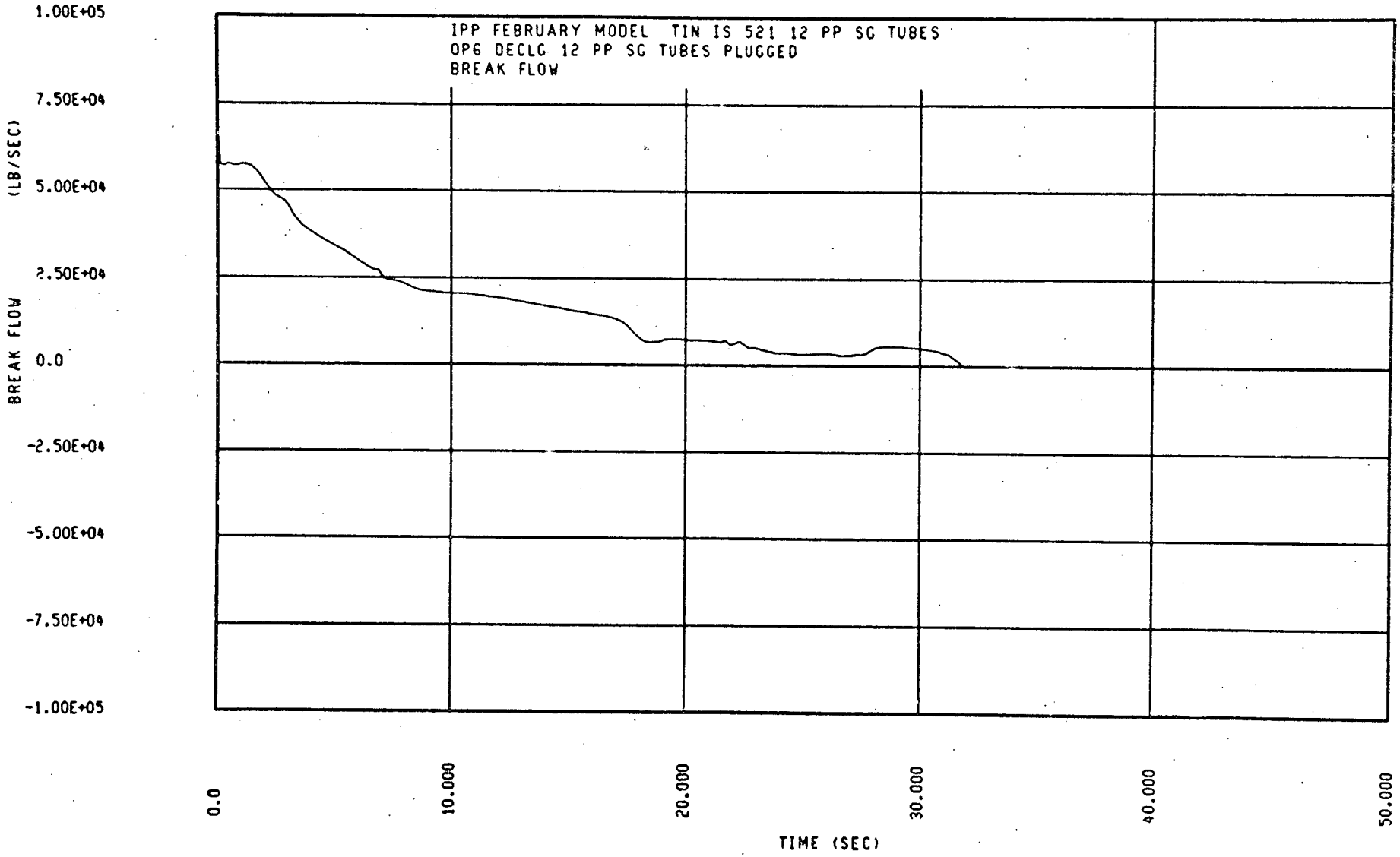


Figure IP2-5: Break Flow Rate

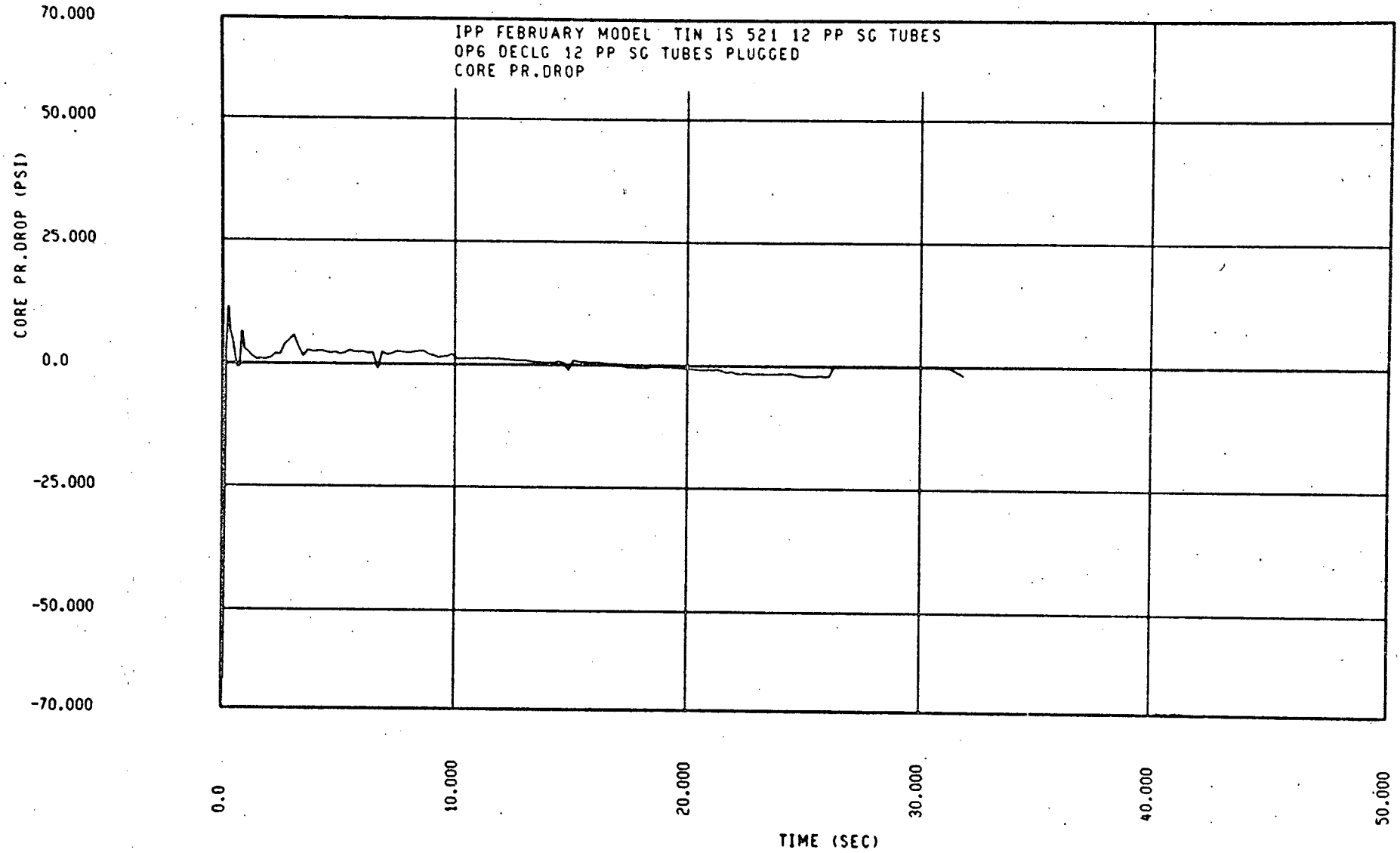


Figure IP2-6: Core Pressure Drop

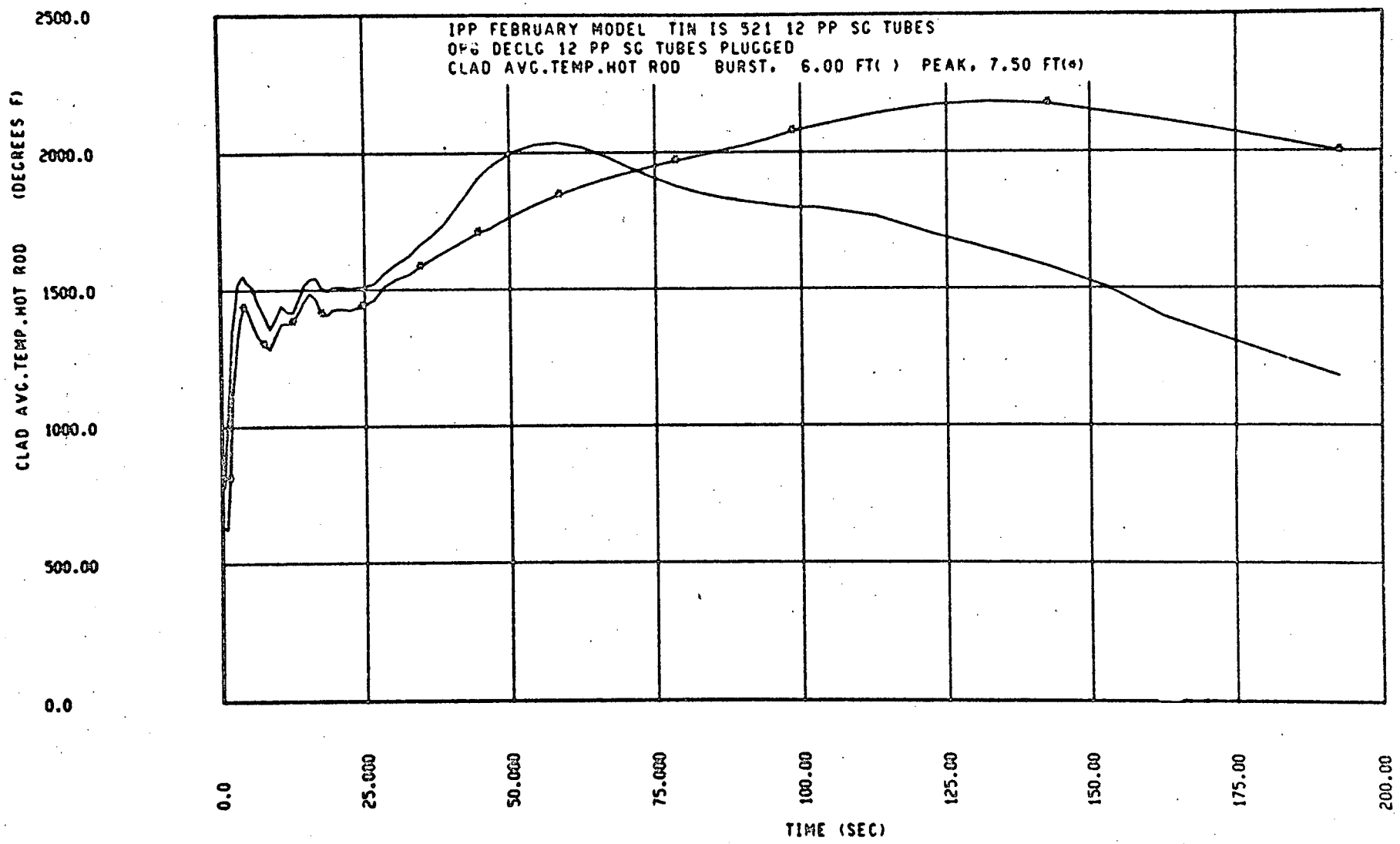


Figure IP2-7: Peak Clad Temperature

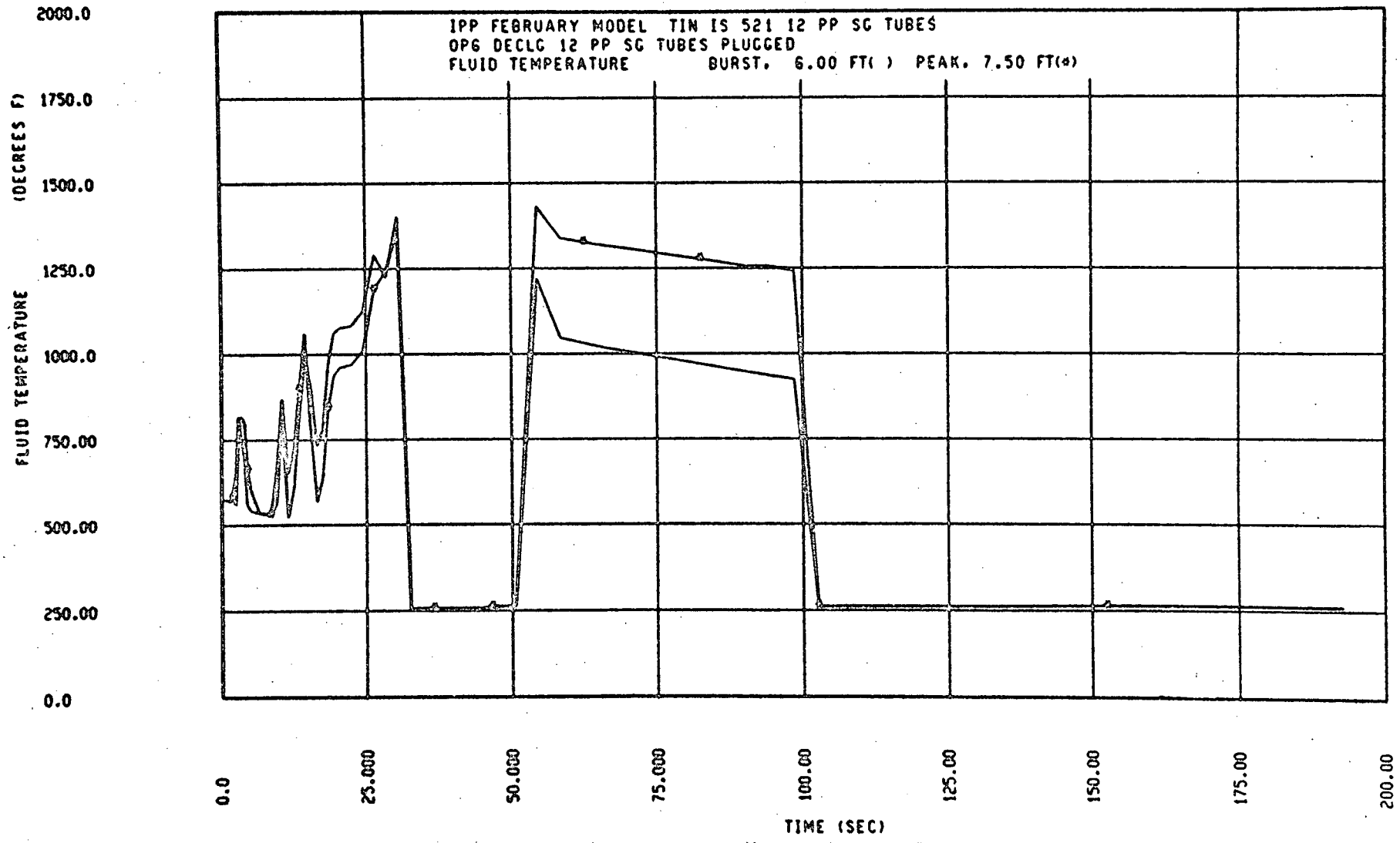


Figure IP2-8: Fluid Temperature

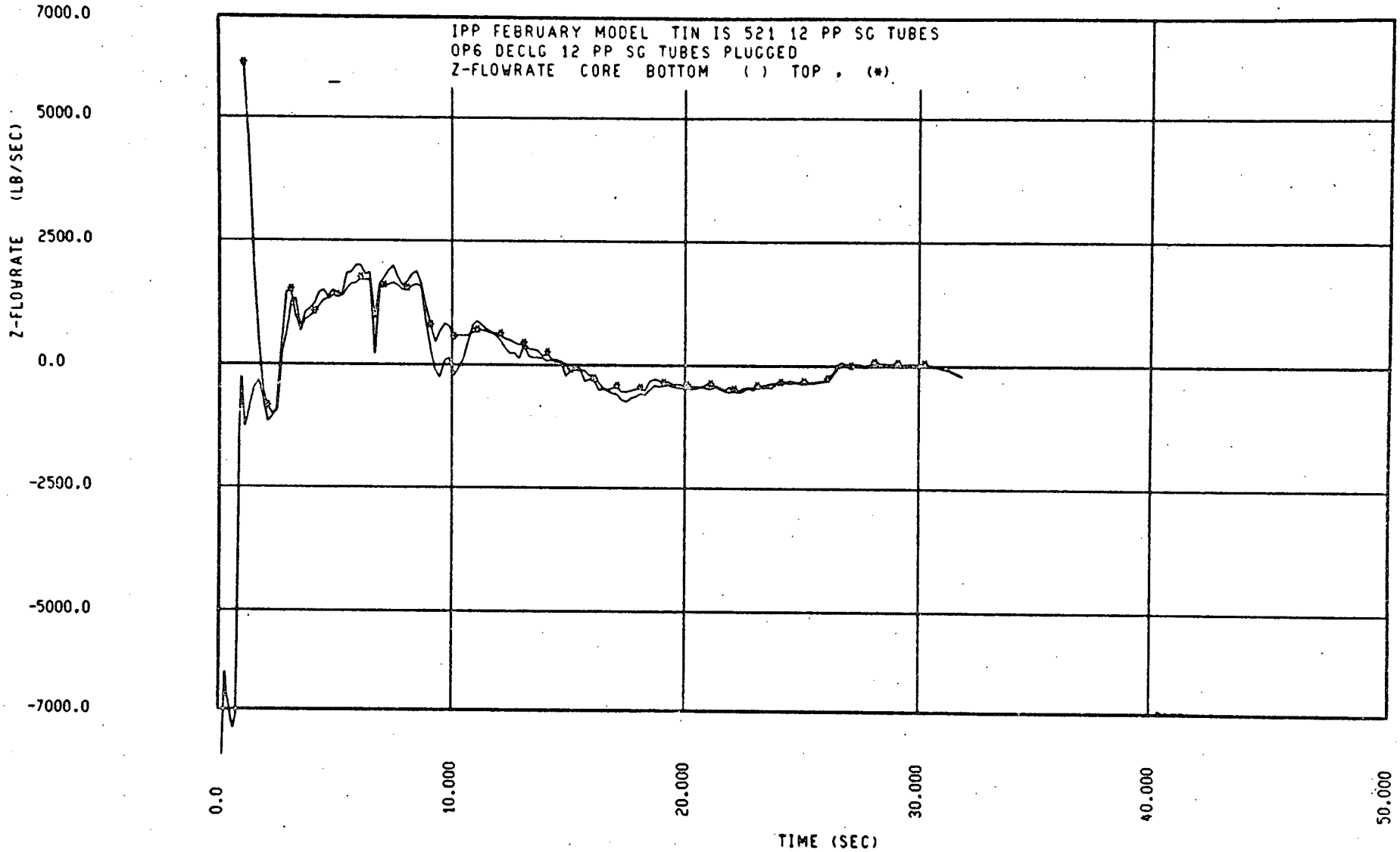


Figure IP2-9: Core Flow-Top and Bottom

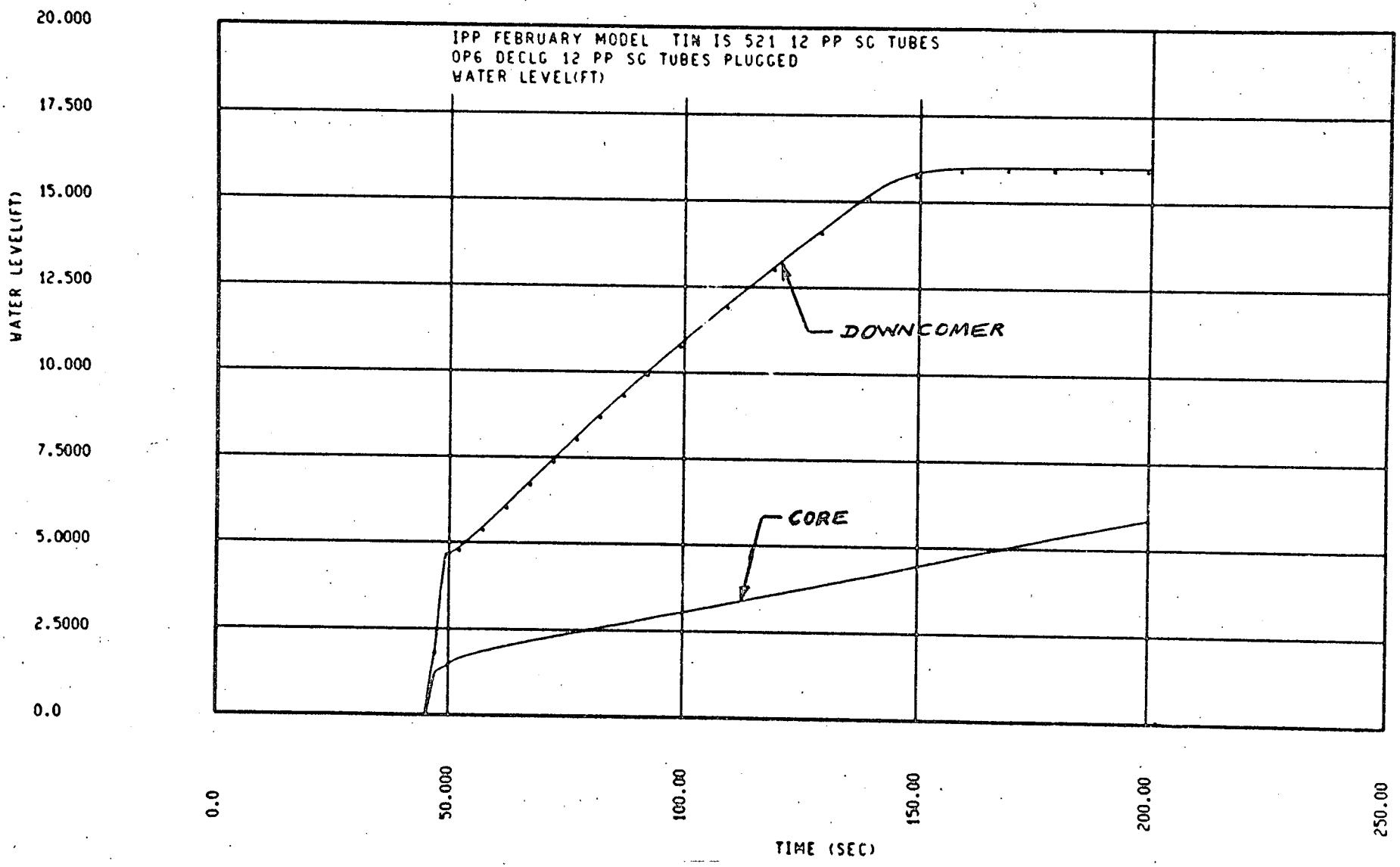


Figure IP2-10: Reflood Transient-Core & Downcomer Water Level

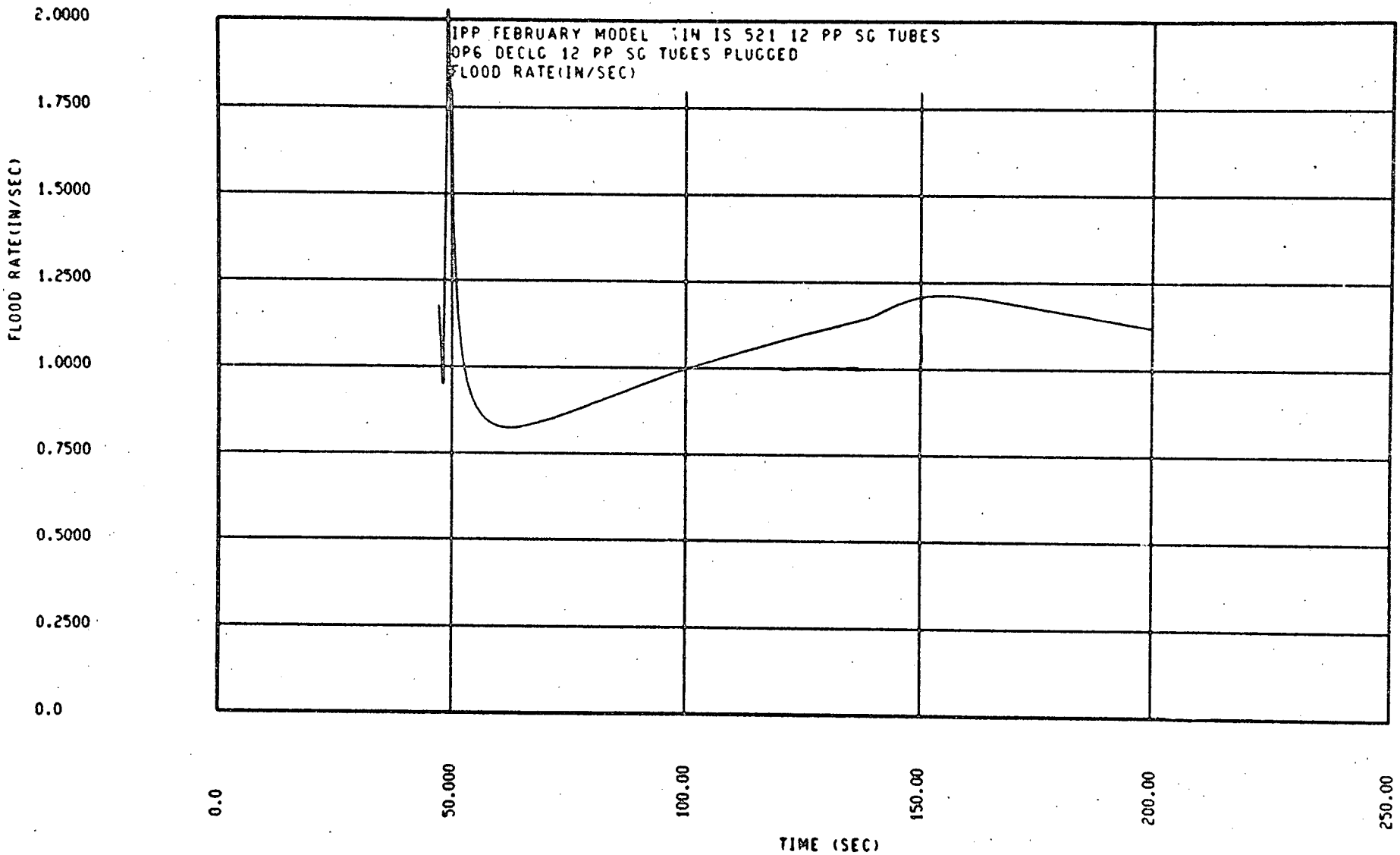


Figure IP2-11: Reflood Transient-
Core Inlet Velocity

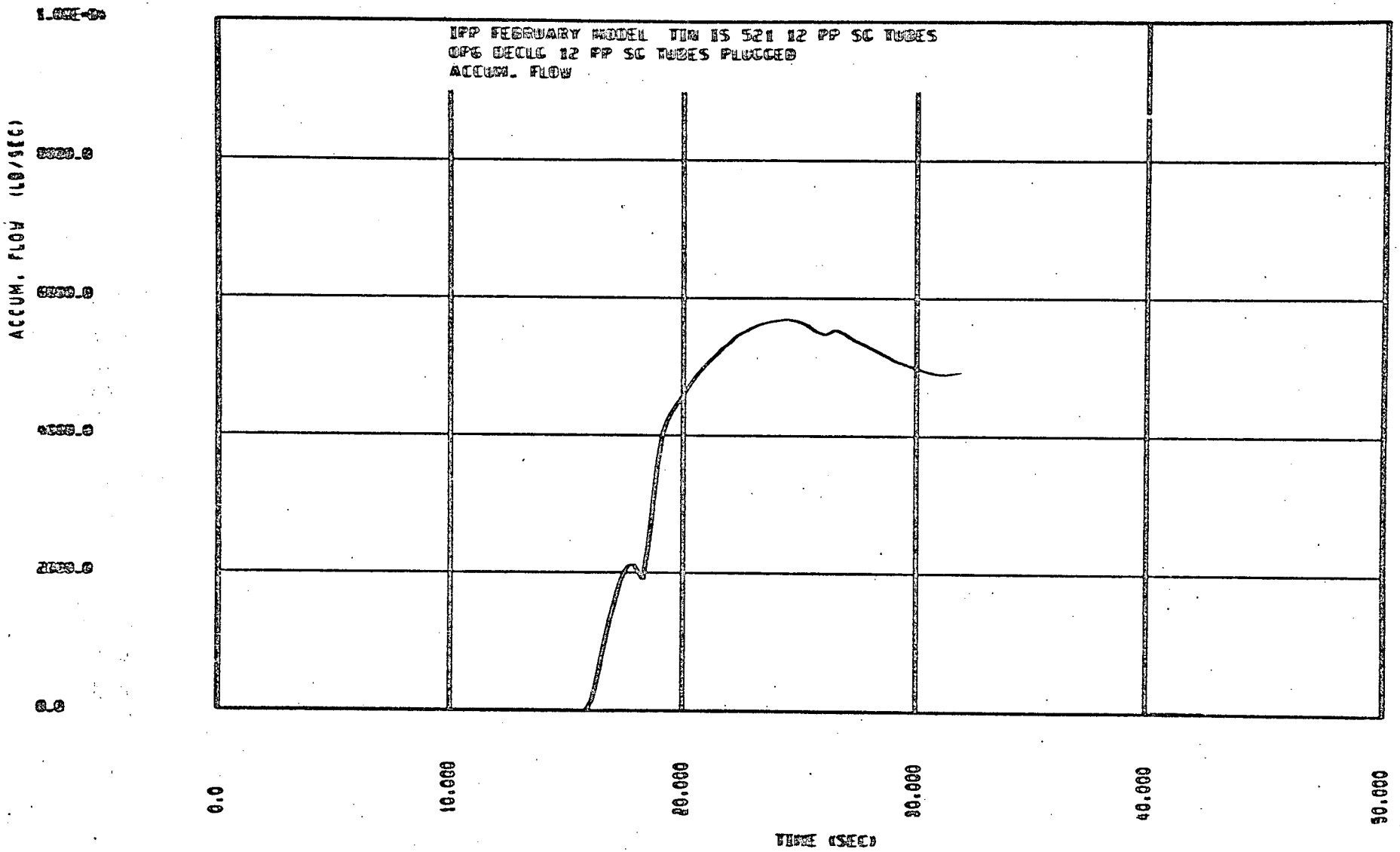
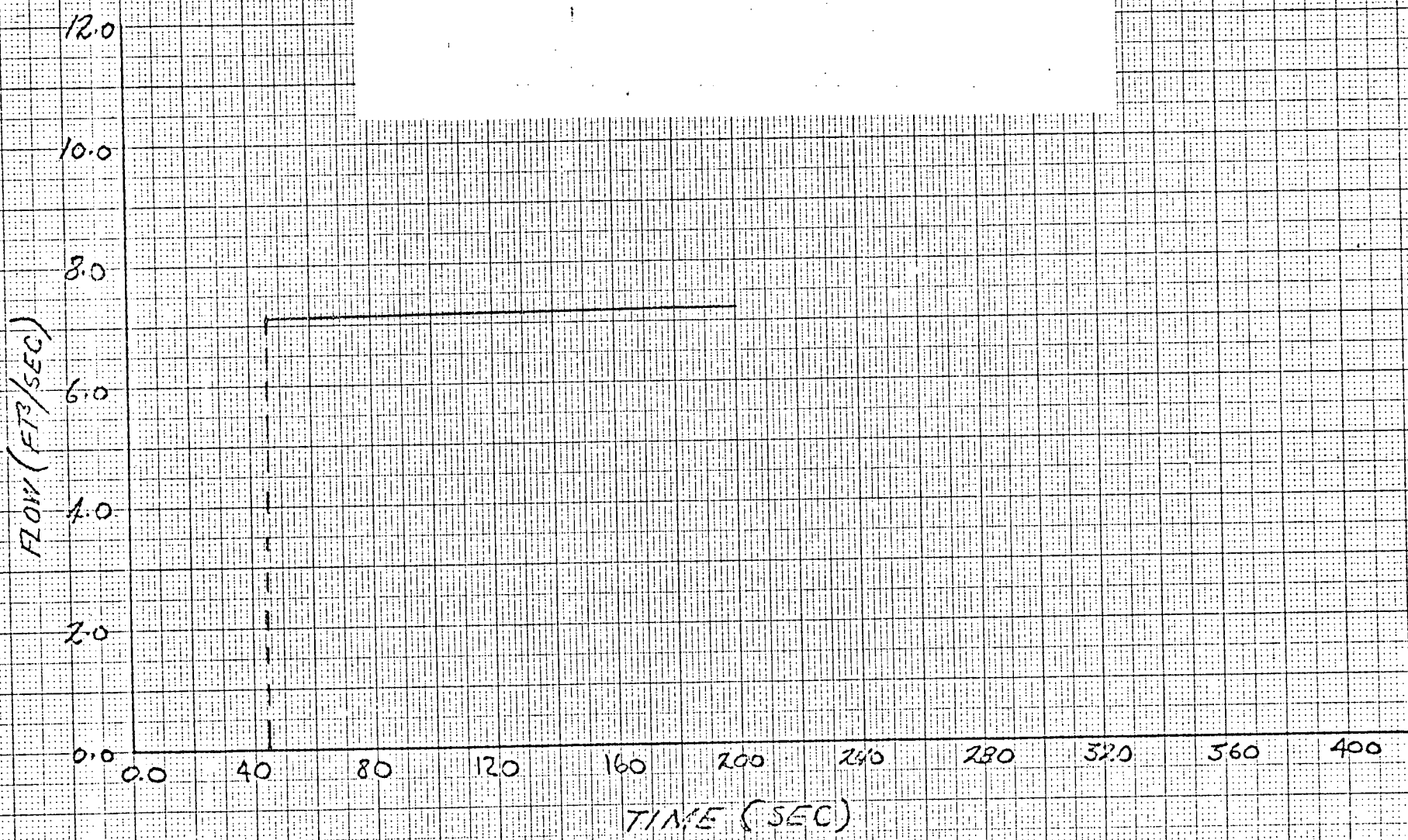


Figure IP2-12: Accumulate Flow
(Blowdown)

Figure IP2-13: Pumped ECCS Flow
(Reflood)



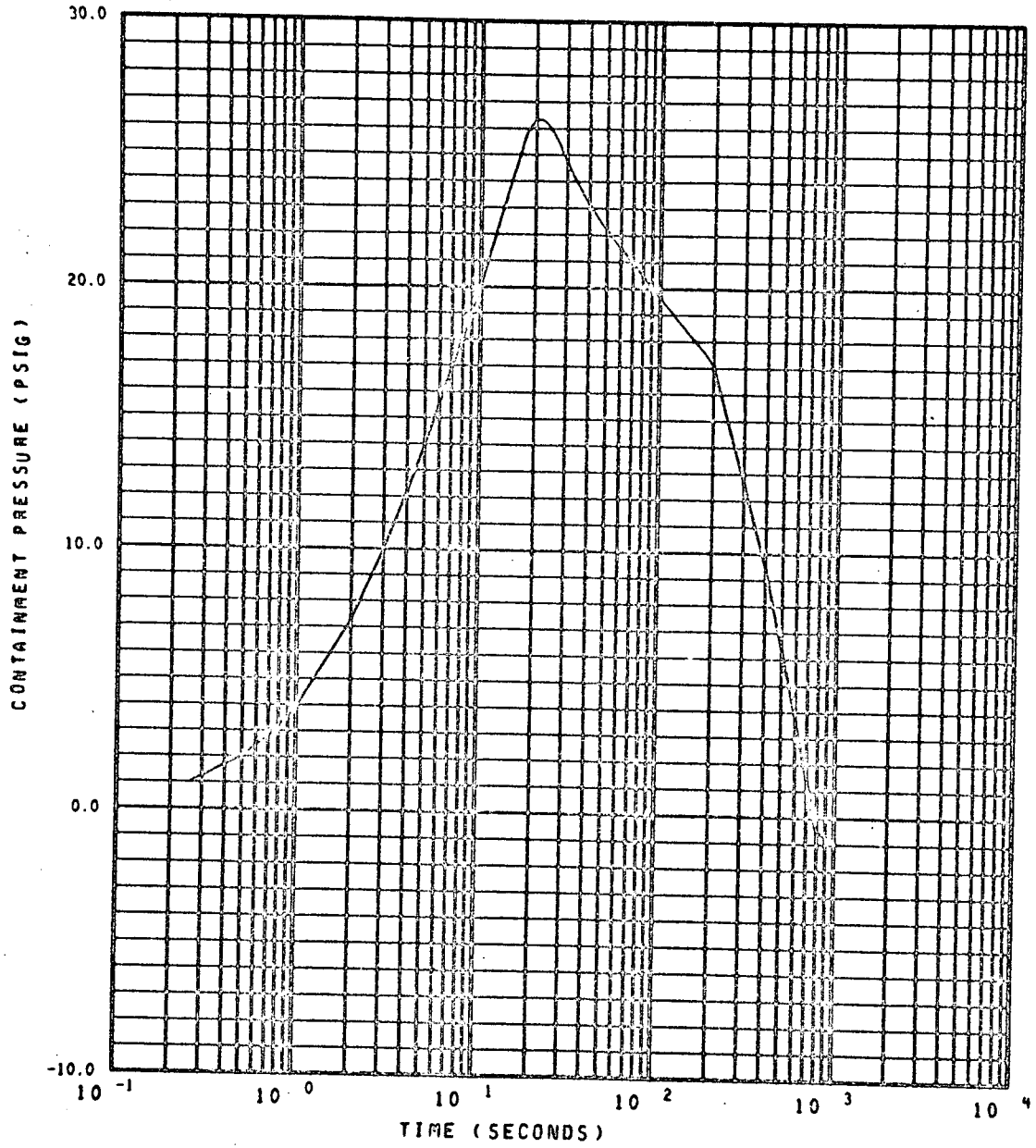


Figure IP2-14: Containment Pressure Transient

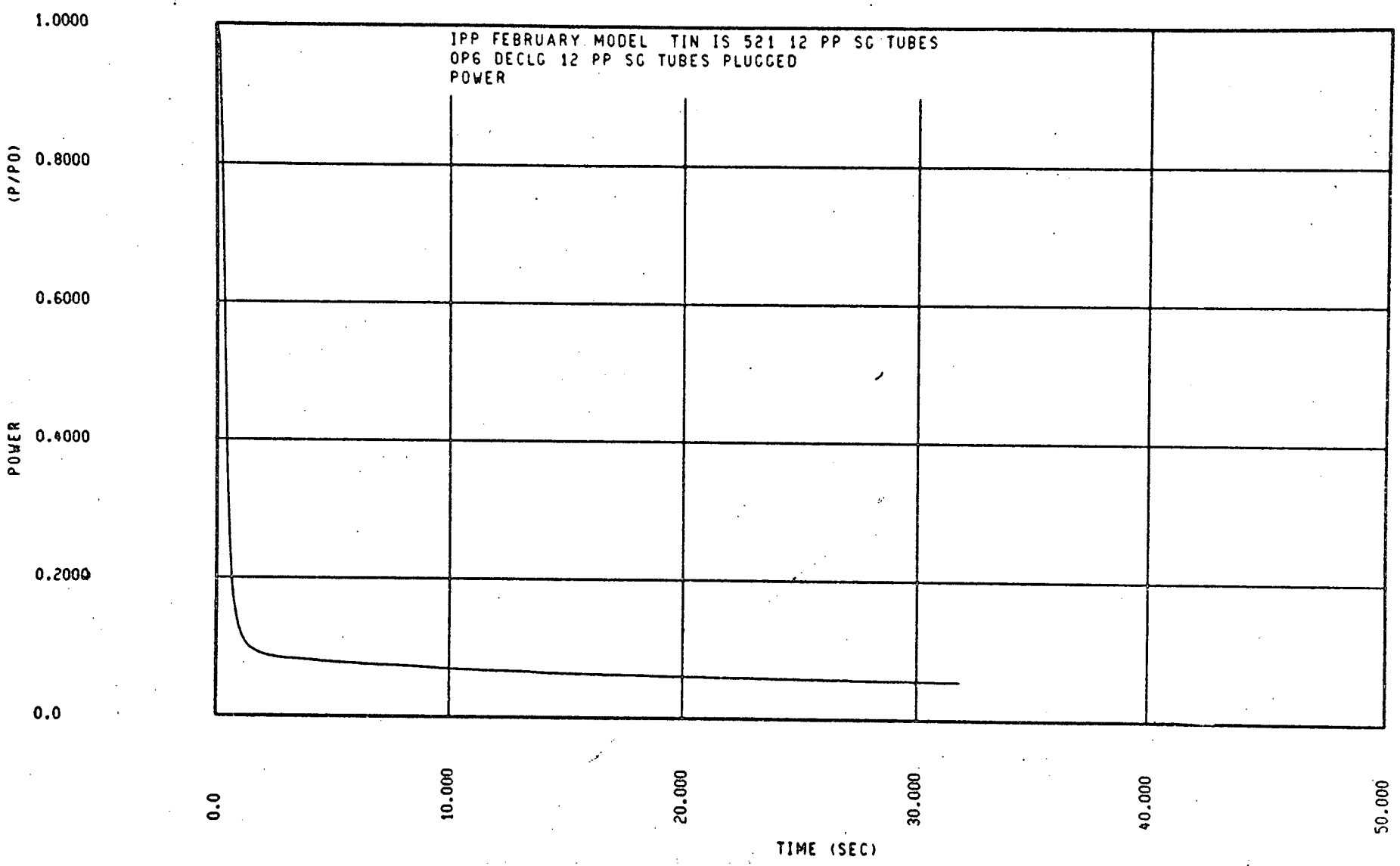


Figure IP2-15: Normalized Core Power

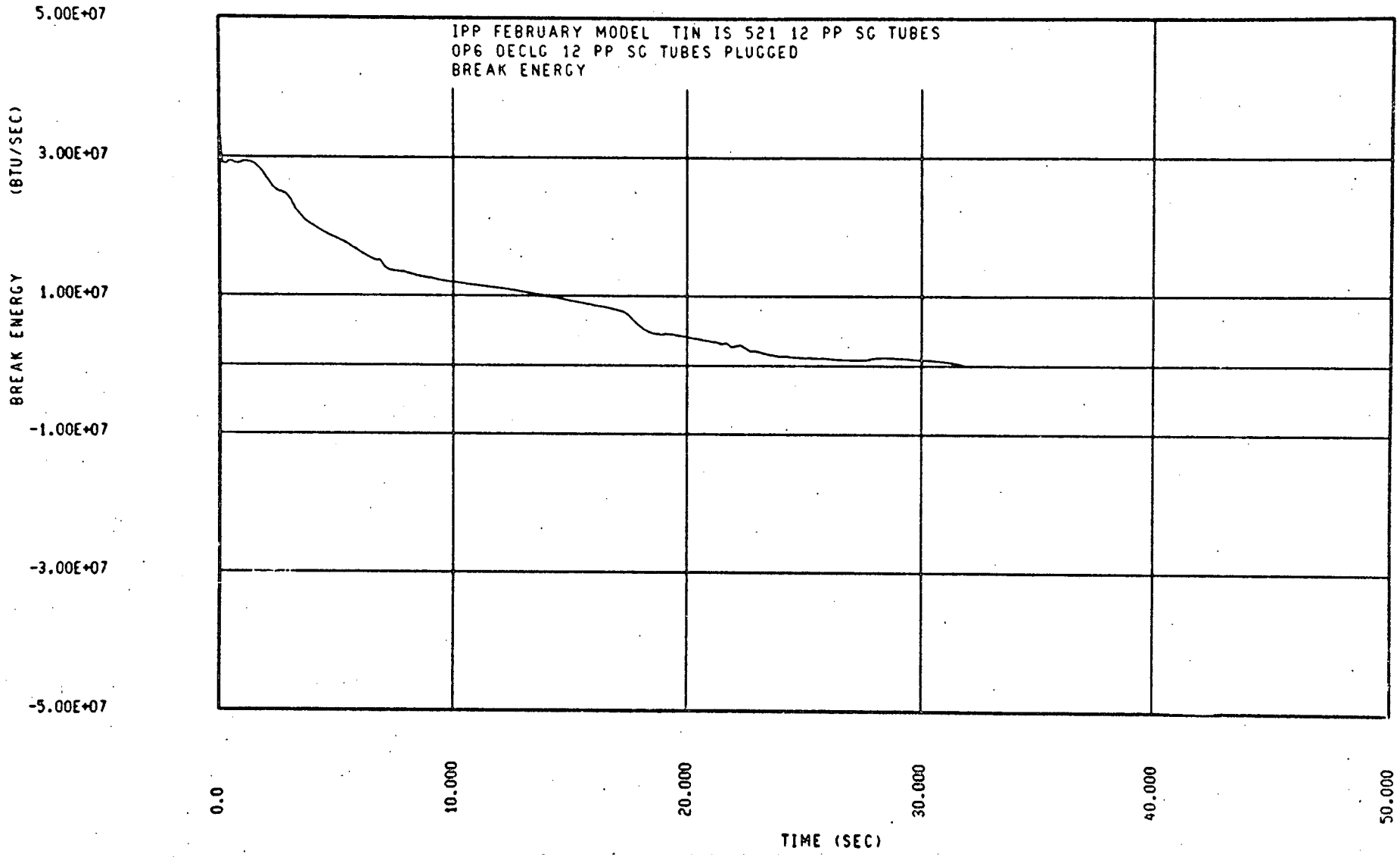
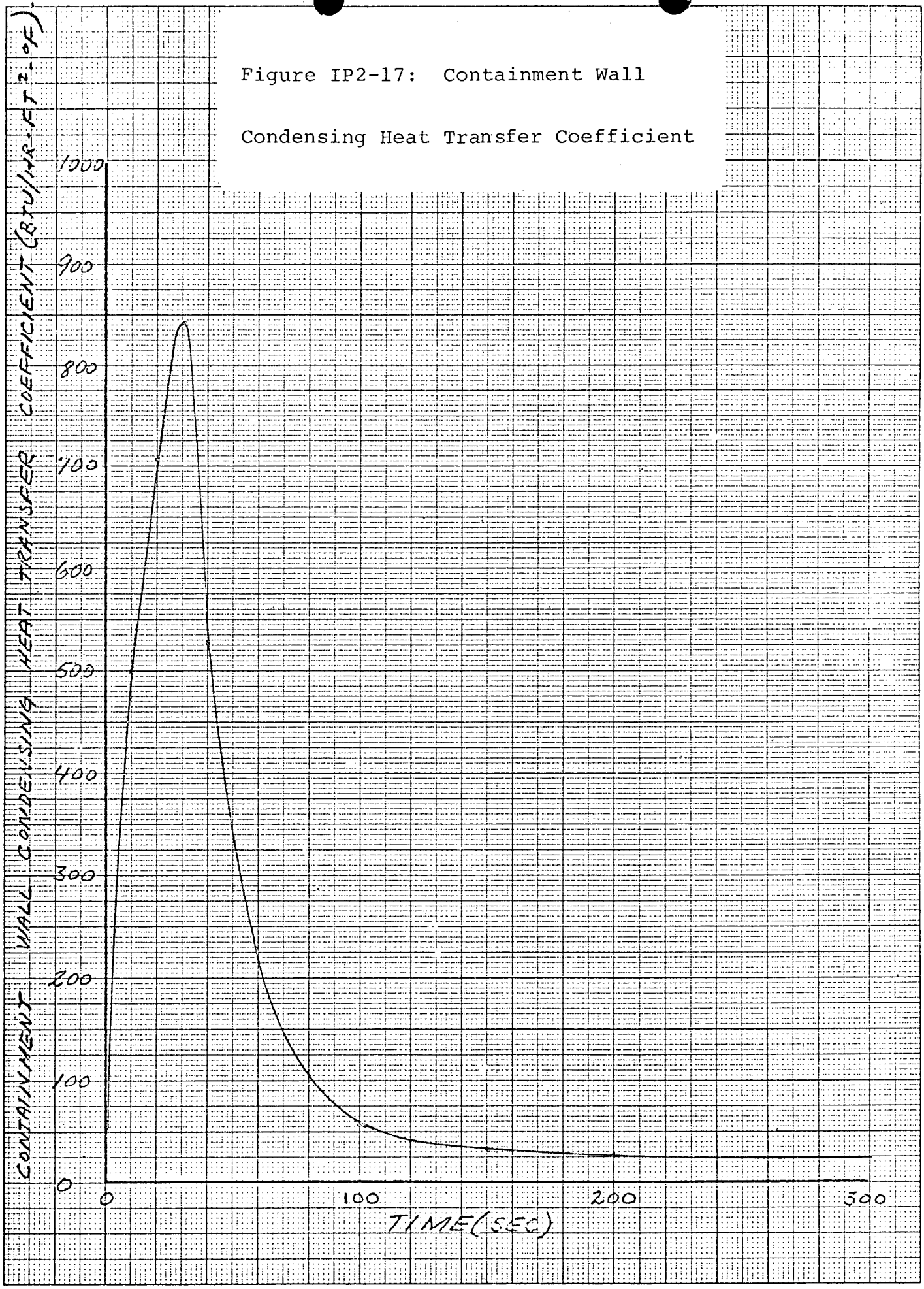


Figure IP2-16: Break Energy Released to Containment

Figure IP2-17: Containment Wall

Condensing Heat Transfer Coefficient



Enclosure 1A

EVALUATION OF THE POTENTIAL IMPACT OF USING FUEL MODELS PRESENTED IN DRAFT NUREG-0630 ON THE ECCS REANALYSIS FOR 12% STEAM GENERATOR TUBE PLUGGING

CONSOLIDATED EDISON COMPANY
OF NEW YORK, INC.
INDIAN POINT UNIT NO. 2
DOCKET NO. 50-247
FACILITY OPERATING LICENSE
NO. DPR-26

April, 1980

A. Evaluation of the potential impact of using fuel rod models presented in draft NUREG-0630 on the Loss of Coolant Accident (LOCA) analysis for Indian Point Unit 2.

This evaluation is based on the limiting break LOCA analysis identified as follows:

BREAK TYPE - DOUBLE ENDED COLD LEG GUILLOTINE

BREAK DISCHARGE COEFFICIENT 0.6

WESTINGHOUSE ECCS EVALUATION MODEL VERSION MODIFIED* FEBRUARY, 1978

* The fuel rod burst model was modified to factor in heatup rate dependence as documented in WCAP-8970-P-A "Westinghouse Emergency Core Cooling System Small Break, October 1975 Model." Fuel rod burst curves used in this analysis represented clad heatup rates of 10 Degrees F/Second for the Hot Rod and 10 Degrees F/Second for the Average Hot Assembly Rod.

CORE PEAKING FACTOR 2.25 - used the February, 1978 model

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR THE BURST REGION OF THE CLAD - 2036.°F = PCT_B

ELEVATION - 6.0 Feet

HOT ROD MAXIMUM TEMPERATURE CALCULATED FOR A NON-RUPTURED REGION OF THE CLAD - 2182.°F = PCT_N

ELEVATION - 7.5 Feet

CLAD STRAIN DURING BLOWDOWN AT THIS ELEVATION 0.0 Percent
MAXIMUM CLAD STRAIN AT THIS ELEVATION - 7.27 Percent

Maximum temperature for this node* occurs when the core reflood rate is greater than 1.0 inch per second and reflood heat transfer is based on the flecht calculation.

AVERAGE HOT ASSEMBLY ROD BURST ELEVATION - 7.5 Feet

HOT ASSEMBLY BLOCKAGE CALCULATED - 21.4 Percent

1. <u>BURST NODE</u>

The maximum potential impact on the ruptured clad node is expressed in letter NS-TMA-2174 in terms of the change in the peaking factor limit (FQ) required to maintain a peak clad temperature (PCT) of 2200°F and in terms of a change in PCT at a constant FQ. Since the clad-water reaction rate

*non-burst node.

increases significantly at temperatures above 2200.°F, individual effects (such as ΔPCT due to changes in several fuel rod models) indicated here may not accurately apply over large ranges, but a simultaneous change in FQ which causes the PCT to remain in the neighborhood of 2200.°F justifies use of this evaluation procedure.

From NS-TMA-2174:

For the Burst Node of the clad:

- 0.01 ΔFQ → ∿ 150°F BURST NODE ΔPCT

- Use of the NRC burst model and the revised Westinghouse burst model could require an F_Q reduction of 0.027
- The maximum estimated impact of using the NRC strain model is a required FQ reduction of 0.03.

Therefore, the maximum penalty for the Hot Rod burst node is:

$$\Delta PCT_1 = (.027 + .03)\ (150°F/.01) = 855°F$$

Margin to the 2200°F limit is:

$$\Delta PCT_2 = 2200.°F - PCT_B = 164°F$$

The FQ reduction is required to maintain the 2200°F clad temperature limit is.

$$\Delta FQ_B = (\Delta PCT_1 - \Delta PCT_2)\ \left(\frac{.01\ \Delta FQ}{150°F}\right)$$

$$= (855 - 164)\ \left(\frac{.01}{150}\right)$$

$$= 0.046 \text{ (but not less than zero).}$$

2. NON-BURST NODE

The maximum temperature calculated for a non-burst section of clad typically occurs at an elevation above the core mid-plane during the core reflood phase of the LOCA transient. The potential impact on that maximum clad temperature of using the NRC fuel rod models can be estimated by examining two aspects of the analyses. The first aspect is the change in pellet-clad gap conductance resulting from a difference in clad strain at the non-burst maximum clad temperature node elevation. Note that clad strain all along the fuel rod stops after clad burst occurs and use of a different clad burst model can change the time at which burst is calculated. Three sets of LOCA analysis results were studied to establish an acceptable sensitivity to apply

generically in this evaluation. The possible PCT increase resulting from a change in strain (in the Hot Rod) is +20.°F per percent decrease in strain at the maximum clad temperature locations. Since the clad strain calculated during the reactor coolant system blowdown phase of the accident is not changed by the use of NRC fuel rod models, the maximum decrease in clad strain that must be considered here is the difference between the "maximum clad strain" and the "clad strain during blowdown" indicated above.

Therefore:

$$\Delta PCT_3 = \left(\frac{20°F}{.01 \text{ strain}}\right) (\text{MAX STRAIN} - \text{BLOWDOWN STRAIN})$$

$$= \left(\frac{20}{.01}\right) (.0727 - 0 \quad)$$

$$= 145 \ °F$$

The second aspect of the analysis that can increase PCT is the flow blockage calculated. Since the greatest value of blockage indicated by the NRC blockage model is 75 percent, the maximum PCT increase can be estimated by assuming that the current level of blockage in the analysis (indicated above) is raised to 75 percent and then applying an appropriate sensitivity formula shown in NS-TMA-2174.

Therefore,

$$\Delta PCT_4 = 1.25°F \ (50 - \text{PERCENT CURRENT BLOCKAGE}) + 2.36°F \ (75-50)$$

$$= 1.25 \ (50 - 21.4) + 2.36 \ (75-50)$$

$$= 95 \ °F$$

If PCT_N occurs when the core reflood rate is greater than 1.0 inch per second $\Delta PCT_4 = 0$. The total potential PCT increase for the non-burst node is then

$$\Delta PCT_5 = \Delta PCT_3 + \Delta PCT_4 = 145 + 0 = 145°F$$

Margin to the 2200°F limit is

$$\Delta PCT_6 = 2200°F - PCT_N = 18 \ °F$$

The FQ reduction required to maintain this 2200°F clad temperature limit is (from NS-TMA-2174)

$$\Delta FQ_N = (\Delta PCT_5 - \Delta PCT_6) \left(\frac{.01 \Delta FQ}{10°F \Delta PCT}\right)$$

$$\Delta FQ_N = 0.127 \text{(but not less than zero)}.$$

The peaking factor reduction required to maintain the 2200°F clad temperature limit is therefore the greater of ΔFQ_B and ΔFQ_N, or; $\Delta FQ_{PENALTY} \doteq 0.127$

B. The effect on LOCA analysis results of using improved analytical and modeling techniques (which are currently approved for use in the Upper Head Injection plant LOCA analyses) in the reactor coolant system blowdown calculation (SATAN computer code) has been quantified via an analysis which has recently been submitted to the NRC for review. Recognizing that review of that analysis is not yet complete and that the benefits associated with those model improvements can change for other plant designs, the NRC has established a credit that is acceptable for this interim period to help offset penalties resulting from application of the NRC fuel rod models. That credit for two, three and four loop plants is an increase in the LOCA peaking factor limit of 0.12, 0.15 and 0.20 respectively.

C. The peaking factor limit adjustment required to justify plant operation for this interim period is determined as the appropriate ΔFQ credit identified in section (B) above, minus the $\Delta FQ_{PENALTY}$ calculated in section (A) above (but not greater than zero).

$$\text{FQ ADJUSTMENT} = 0.20 - 0.127 \rightarrow 0$$

Enclosure 2

ANALYSIS AND EVALUATION OF NON-LOCA TRANSIENTS FOR OPERATION WITH 95% REACTOR COOLANT SYSTEM THERMAL DESIGN FLOW AND WITH 25% UNIFORM STEAM GENERATOR TUBE PLUGGING

Consolidated Edison Company
Of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
Facility Operating License
No. DPR-26

April, 1980

ACCIDENT ANALYSIS

The impact of a reduction in thermal design flow for Indian Point Unit 2 on the non-LOCA accident analyses presented in Chapter 14 of the FSAR has been assessed. In general, all of the transients are sensitive to the steady state primary flow. The approach used was to identify the impact of a reduction in Thermal Design Flow on each accident. The evaluation was consistent with the following assumptions:

Thermal Design Flow, gpm	85,200
S.G. Tube Plugging, %	25
Maximum Allowed Power, MWt	2758
T_{avg} at 100% Power, °F	569.5*

In addition, a study was made of each currently applicable accident analysis to identify margins to safety limits which could be used to offset flow reduction penalties.

In general, reanalysis and evaluation techniques were based on the assumptions and methods employed in the FSAR; exceptions are noted in the discussion of each incident. Based on the above work, reanalysis of the limiting transients is performed to verify safety conclusions. Cycle 4 parameters were used unless otherwise stated.

Control Rod Withdrawal From a Subcritical Condition

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 14.1.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion causes a heatup of the moderator. However, since the power rise is rapid and is followed by an immediate reactor trip, the moderator temperature rise is small. Thus nuclear power response is primarily a function of the Doppler temperature coefficient. The increase in temperature would result in more Doppler feedback thus reducing the nuclear power excursion as presented in the FSAR which would partially compensate for the flow reduction. Therefore,

* All analyses/evaluations are applicable to both T_{avg} = 569.5°F or T_{avg} = 549°F.

the transient is only moderately sensitive to reactor coolant flow.

The most recent analysis shows that for a 80 x 10^{-5} Δk/sec reactivity insertion rate the peak heat flux achieved is 85.3% of nominal with a resultant peak fuel average temperature of 816°F. A 5% reduction flow would degrade heat transfer from the fuel by a maximum of 5%. Thus peak fuel and clad temperatures would also increase by a maximum of 5%, yielding maximum fuel and clad temperatures of 856°F and 624°F. These values are still significantly below fuel melt (5080°F) and zirconium-H_2O reaction (1800°F) limits. Thus temperature increases will not result in a violation of safety limits.

Uncontrolled Control Rod Assembly Withdrawal at Power

An uncontrolled control rod assembly withdrawal at power produces a mismatch in steamflow and core power, resulting in an increase in average temperature. The following reactor trips protect the core from falling below a DNBR of 1.30:

(1) Nuclear Overpower
(2) High Pressure and Level
(3) Overpower and Overtemperature ΔT

Although the current core limits were verified to be conservative*, the setpoints corresponding to the core limits had to be regenerated for the reduced flow. It was found that the current setpoints were conservative and need not be changed. To verify this the range of insertion rates for which the overtemperature trip is activated for full power conditions were rerun and the DNBR did not fall below 1.30. Therefore the conclusions reached in the FSAR are valid.

Control Rod Assembly Drop

The drop of a Control Rod Assembly results in a step decrease in reactivity which produces a similar reduction in core power, thus reducing the coolant average temperature. The highly negative moderator temperature coefficient (-35 pcm/°F) assumed in the analysis results in a power increase (overshoot) above the turbine runback value causing a temporary imbalance between core power and secondary

*Accommodated by the generic margin present in the current Cycle 4 analysis, with a rod bow penalty based on the recently approved partial bow model. (Letter from John F. Stolz to T. M. Anderson, April 5, 1979, "WCAP-8691")

power extraction capability. The effect of a 5% reduction in initial RCS flow would be a smaller reduction in coolant average temperature. Thus the power overshoot would be less than the value shown in Section 14.1.4 of the FSAR. Based on the FSAR transient, statepoints were evaluated consistent with a 5% reduction in flow. The results of this DNB evaluation showed that the DNBR limit of 1.30 can be accommodated with margin in the current cycle.

Boron Dilution

For a boron dilution incident during refueling or startup, while the reactor is subcritical, Section 14.2.5 of the FSAR shows that the operator has sufficient time to identify the problem and terminate the dilution before the reactor becomes critical. Tube plugging has no affect on the analysis at refueling condition since only the reactor vessel and RHR system volumes are considered. For a dilution during startup, the effective volume of primary coolant in the steam generator tubes has been reduced by $\sim$25% (660 ft^3). Thus the volume of the reactor coolant is reduced from 10,390 ft^3 to 9730 ft^3. The minimum time required for the reactor to return critical is reduced to approximately 2 hours. Thus adequate time is available for the operator to recognize and terminate the dilution flow from startup conditions.

For dilution at power the same RCS volume reduction must be considered. With the reactor in automatic control, the time to lose shutdown margin would be reduced from 14 to 13.2 minutes. The FSAR analysis, however, assumed a total shutdown margin of 1 percent $\Delta k/k$. Current cycles are design to 1.95 percent $\Delta k/k$ shutdown margin. Thus using the same FSAR methods, however, reducing RCS volume and increasing shutdown margin the operator action time will increase. Therefore the operation has adequate time to stop dilution before the reactor returns critical.

With the reactor in manual control, the reduction in RCS volume increases the effective reactivity insertion rate. But it is still less than the conservative 1.2×10^{-5} $\Delta k/sec$ used in FSAR. However, as in the automatic boron dilution case, the shutdown margin has been increased to 1.95% $\Delta k/k$ which more than compensates for the volume reduction. The operator action time will increase. Thus FSAR safety margins are maintained.

Startup of an Inactive Reactor Coolant Loop

An inadvertent startup of an idle reactor coolant pump results in the injection of cold water into the core. The nuclear power (and corresponding heat flux) transient presented is conservative; the lower loop flow would also result in a slightly lower reactivity insertion rate. Therefore, although flow has been reduced to 95% of the FSAR case, the conservative value of heat flux, coupled with the decrease in T_{inlet}, would still maintain adequate margins to DNB.

Excessive Heat Removal Due to Feedwater System Malfunction

The addition of excessive feedwater and inadvertent opening of the feedwater bypass valve are excessive heat removal incidents which result in a power increase due to moderator feedback. Section 14.1.10 of the FSAR presents two cases. The first case assumes a 0 moderator coefficient, this case is used to demonstrate inherent transient attenuation capability during a feedwater reduction. A reduction in thermal design flow will have a negligible effect on stability since the reactivity insertion is identical to the FSAR case. DNB is not a consideration for this case since DNBR's do not fall below the steady state value. This is due to the relatively large reduction in T_{avg}. The reduction in flow, however, will result in the initial steady state DNBR being reduced from 1.90 to 1.8 (this value corresponds to the DNBR which was calculated at time zero in the Loss of Flow THINC analysis). Thus there is still adequate margin to safety limits.

The second case assumes a large negative moderator coefficient. The reduction in thermal design flow will result in a slower cooldown, hence a lower reactivity insertion rate than shown in the FSAR. The integral reactivity insertion due moderator temperature reduction will be less than the FSAR case, thus producing a lower peak nuclear power. Therefore, the reduction in DNBR from the steady state value ($\sim$1.8 at reduced flow) would be no greater than that shown in the FSAR. The FSAR shows a reduction of $\sim$.4. Thus the 5% flow reduction will result in a minimum DNBR of $\sim$1.40. Therefore considerable margin is available to safety limits. Protection is assured for both cases via the overtemperature Delta-T protection setpoints.

Excessive Load Increase

An excessive load increase event, in which the steam load exceeds the core power, results in a decrease in reactor coolant system temperature, which is very similar to the feedwater malfunction cases. Two cases are presented in FSAR Section 14.1.11;

- BOL in manual control
- BOL in automatic control

Both cases show considerable margin to safety limits. Reactor trip is not encountered with minimum DNBR's of 1.66 being calculated. The 5% reduction in thermal design flow will have the same effect on the nuclear power transient as in the Excessive Heat Removal Accident. For there was a maximum DNBR reduction of ∿12%. Thus the overall impact is due to the reduction in the initial steady state DNB margin. A 5% reduction in flow would result in a similar reduction in calculated DNBR's. Thus the flow reduction will yield a minimum DNBR > 1.30. Thus there is still margin to safety limits. The FSAR analysis showed that there was sufficient margin such that no reactor trip was necessary to assure DNB protection. The over-temperature Delta-T setpoints will protect against DNBR < 1.3. Thus if this accident was re-analyzed, the protection setpoints could result in a reactor trip therefore preventing violation of core limits. The adequacy of this protection was verified in the Rod Withdrawal at Power Reanalysis.

Loss of Flow [Complete analysis in Appendix I]

A 5% reduction in thermal design flow could have an adverse effect on maintaining DNBR > 1.3. The adequacy of the statepoint for cycle 4 was verified for the reduced flow rate and the statepoint at the reduced flowrate was found to yield a DNBR > 1.3. Therefore the conclusions reached in the FSAR are valid.*

Locked Rotor

The FSAR (Section 14.1.6) shows that the most severe locked rotor incident is an instantaneous seizure of a reactor coolant pump rotor at 100% power with three loops operating. Following the incident, reactor coolant system temperature rises

* Loss of flow analysis included partial rod bow.

until shortly after reactor trip. A reduction in RCS flow will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the transient. The flow reduction in the affected loop is so rapid that the time of reactor trip (low flow setpoint is reached) is unchanged.

Therefore the nuclear power and heat flux response will be unchanged. However, the reduction in flow and coolant mass inventory will result in slightly higher system pressures and clad temperatures. Clad temperature calculations were not performed since minimum DNB did not fall below 1.3. The peak system pressure will increase ∿ 100 psig above the previous value, however, the maximum calculated value was 2530 psia. This is significantly below the pressure at which vessel stress limits are exceeded (∿ 600 psia exists to this limit), thus considerable margin exists to absorb any pressure increase. The 25% reduction in steam generator tubes would result in approximately a 5% reduction in primary mass which decreases the heat capacity of the RCS by the same amount. This would not result in higher peak temperatures or pressures since the peak values are reached in considerably less than on loop transport time constant.

Thus operation at reduced flow will not cause safety limits to be exceeded for a locked rotor accident.

Loss of External Electrical Load

The results of a loss of load is a core power level which momentarily exceeds the secondary system power extraction causing an increase in core water temperature. The most recent analysis shows a peak pressurizer pressure of 2550 psia following reactor trip and a minimum DNBR of 1.67. A reduction in loop flow and RCS mass inventory will result in a more rapid pressure rise than is currently shown. The effect will be minor, however, since the reactor is tripped on high pressurizer pressure. Thus the time to trip will be decreased which will result in a lower total energy input to the coolant. Therefore, although the initial margin to DNB will be reduced, the minimum transient DNBR will be only marginally affected. In addition, the overtemperature Delta-T setpoints will assure adequate margin to DNB. The 5% reduction in flow will lead to an conservative increase is system pressure to 2600. The pressurizer will not fill, and the system will not violate any pressure boundaries. Therefore no safety limits are reached.

Loss of Normal Feedwater/Station Blackout *

This transient is analyzed to determine that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These criteria are assured by applying the more stringent requirement that the pressurizer must not be filled with water. The effect of reducing initial core flow would be a larger and more rapid heatup of the primary system. The resulting coolant density change would increase the volume of water in the pressurizer. The analysis in FSAR Sections 14.1.9 and 14.1.12 show considerable margin is available. The assumption that the initial power is 102% of 3216.5 has approximately 20% conservatism included. So the ∿20% conservatism in power more than compensates for the 5% reduction in flow. Therefore the FSAR conclusions are still valid.

Rupture of a Control Rod Drive Mechanism Housing, Control Rod Ejection

The rod ejection transient is analyzed at full power and hot standby for both beginning and end of life conditions (Section 14.2.6 of the FSAR). A reduction in core flow will result in a reduction in heat transfer to the coolant which will increase clad and fuel peak temperatures. All cases have significant margin to fuel failure limits. The effect of reducing flow by 5% is to primarily increase the peak clad temperatures by ∿50°F. The current analysis shows that for all cases a value of 400°F can be accommodated before peak clad limits are reached (2700°F). The fuel temperatures will also increase, however, the increase much less than the clad increase due to the rapid nature of the transient. In addition to the output there is a significant degree of conservatism in the inputs. Also for the full power cases, the initial hot spot fuel temperatures were calculated assuming an F_Q of 2.55. Due to LOCA considerations the F_Q limit will be ∿2.0. This results in more than a 175°F reduction in initial temperature which translates into a > 75°F reduction in peak transient fuel temperatures. Thus the above considerations will clearly compensate for the reduction in thermal design flow.

* Station blackout is covered by the Loss of Flow for the initial few seconds, and by the loss of normal feedwater there after.

Steamline Break

The steamline break transient is analyzed for hot zero power, end of life conditions (Section 14.2.5 of the FSAR) for the following cases:

- Hypothetical Break (steam pipe rupture)
  - Inside Containment with and without power
  - Outside Containment with and without power
- Credible Break (Dump valve opening)

A steamline break results in a rapid depressurization of the steam generators which causes a large reactivity insertion to the core via primary cooldown. The acceptance criteria for this accident is that no DNB must occur following a return to power. This limit, however, is highly conservative since steamline break is classified as a Condition III event. A reduction in core flow will result in a reduction in heat transfer from the fuel to the coolant. Thus the return to power as shown in the current analysis would be conservative with respect to the lower initial flow conditions. In addition, the time of Safety Injection actuation would be unaffected by flow conditions for the Hypothetical Breaks. This coupled with the slower return to power would result in a significant reduction in peak average power for the cases with and without power. Thus a 5% reduction in flow will not result in a violation of safety limits.

CONCLUSION

To assess the effect on accident analysis of operation of IP 2 with significant levels of steam generator tube plugging, a safety analysis was performed. An evaluation or reanalysis was performed to identify the effect of a flow reduction on the FSAR transients and to quantify margins available to offset penalties. Based on above, a 5% reduction in thermal design flow and a steam generator tube plugging level of 25% will not result in violation of safety limits for the transients evaluated.

APPENDIX I

Loss of Reactor Coolant Flow

As demonstrated in the FSAR, Section 14.1.6, the most severe loss of flow transient is caused by the simultaneous loss of electrical power to all four reactor coolant pumps. This transient was reanalyzed to determine the effect of steam generator tube plugging on the minimum DNBR reached during the incident. Tube plugging may result in a decrease in margin to safety limits due to the following effects:

- Higher loop resistances result in a more rapid flow coastdown.
- Lower initial flows result in less margin to the 1.30 DNBR limit.

Thus reanalysis is required.

Method of Analysis

Analysis methods and assumptions used in the reevaluation were consistent with those employed in the FSAR. These assumptions included:

1. Initial operating conditions most adverse with respect to the margin to DNBR, i.e., maximum steady state power level (102% of nominal), minimum pressure (2220 psia), and maximum temperature (573.5).

2. High value (absolute) of Doppler coefficient and zero moderator temperature coefficient;

3. Time from loss of power to all pumps to the initiation of control rod assembly motion (reactor trip) of 1.2 seconds; and

4. 4% ΔK trip reactivity from full power.

The flow coastdown was calculated by the PHOENIX code, and the resultant system transient was simulated using the LOFTRAN code. The calculation was performed for a zero constant moderator coefficient value.

Results

The minimum value of DNBR presented for this incident is >1.30. Figures 1 through 4 show the flow coastdown, nuclear power, heat flux, and minimum DNB ratio vs. time.

Conclusions

Steam generator tube plugging appreciably affects the results of the complete loss of flow transient, however, the minimum DNBR remains above 1.30 for this incident. This case was analyzed since it is the most limiting one presented in the FSAR. Loss of a single pump with all loops in service or with a single loop out of service were less limiting.

FIGURE 1

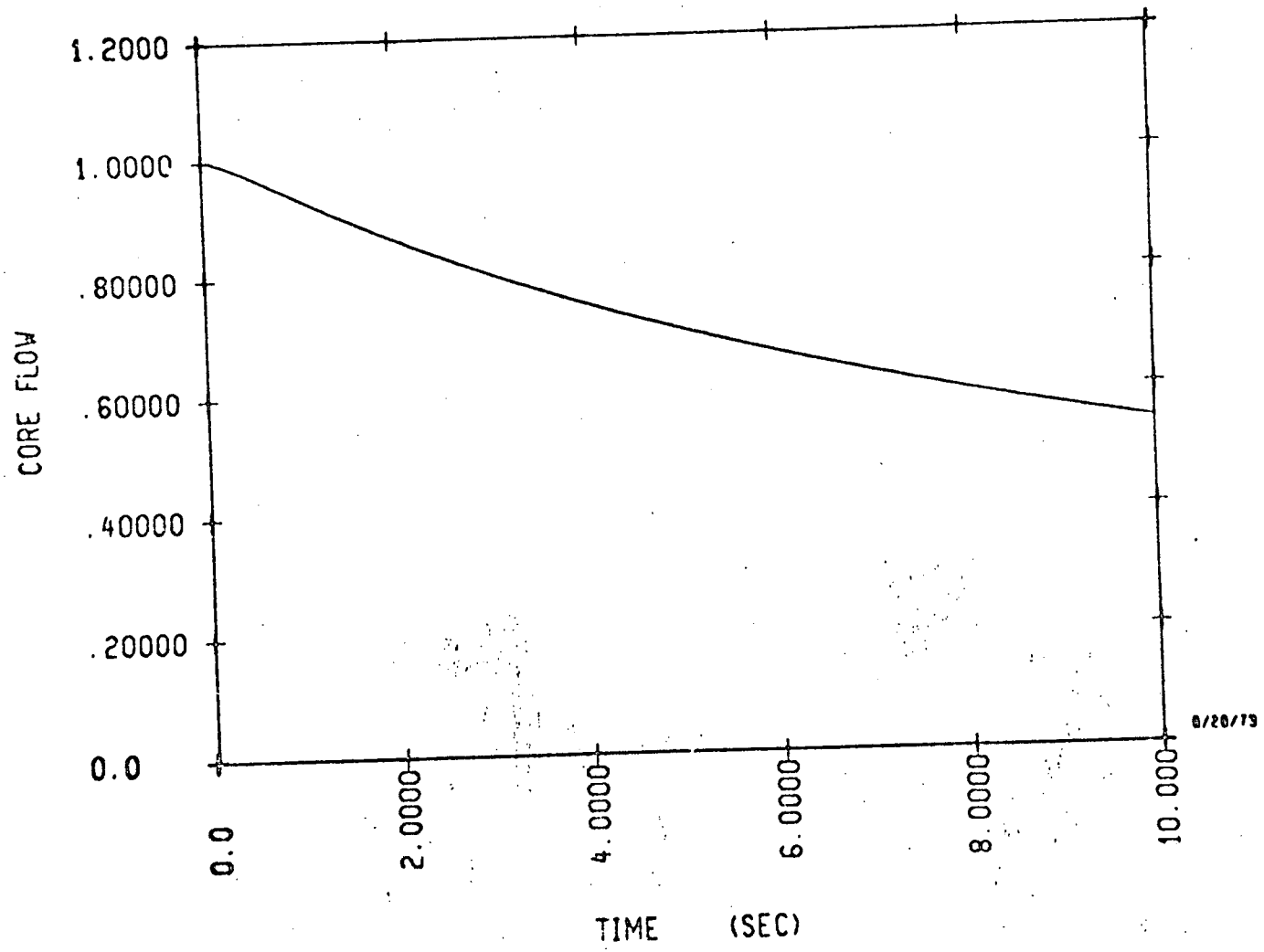

1.2000
1.0000
.80000
.60000
.40000
.20000
0.0
CORE FLOW
0.0
2.0000
4.0000
6.0000
8.0000
10.000
TIME (SEC)
8/20/73

FIGURE 2

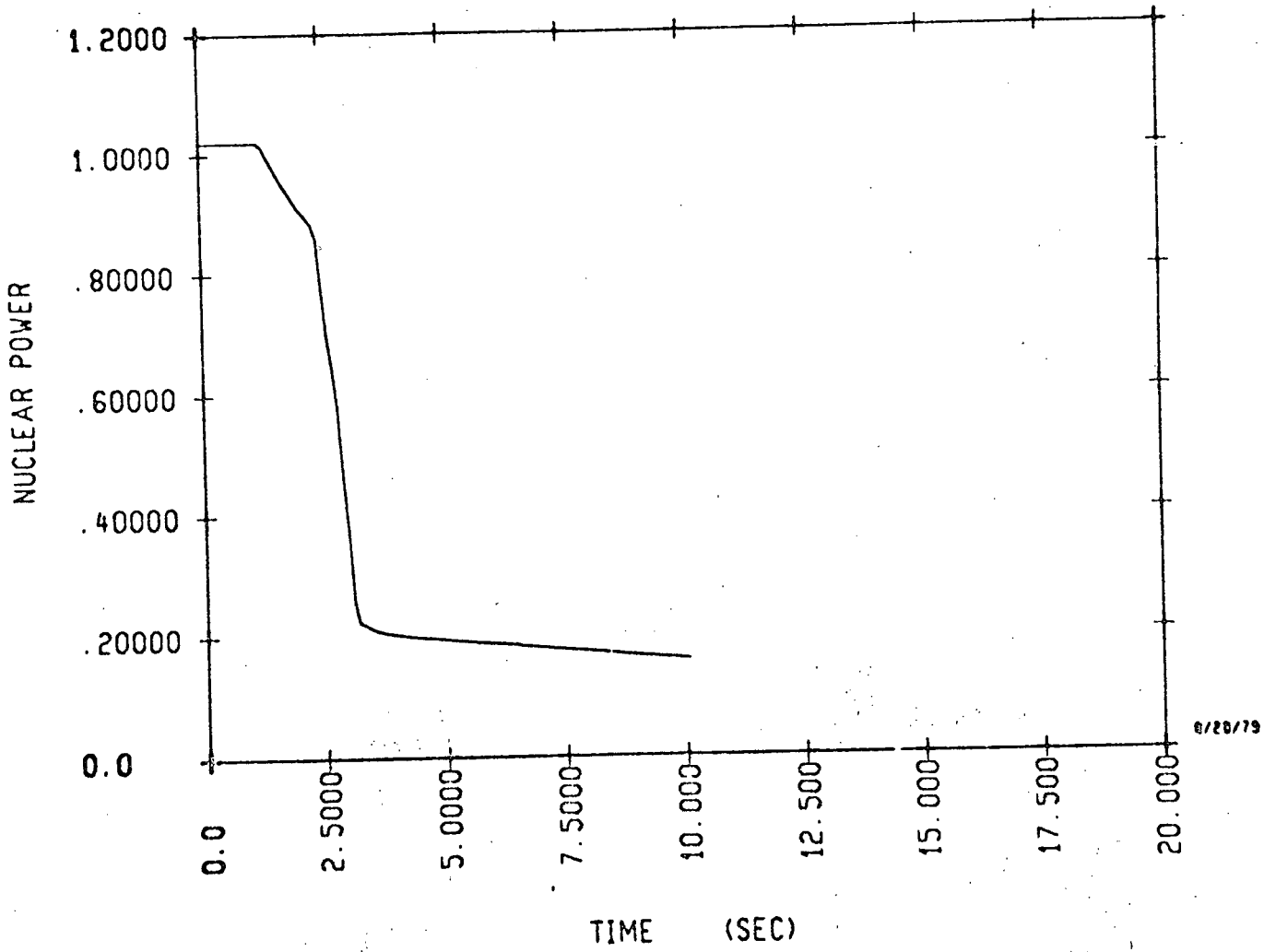

1.2000
1.0000
.80000
.60000
.40000
.20000
0.0
NUCLEAR POWER
0.0
2.5000
5.0000
7.5000
10.000
12.500
15.000
17.500
20.000
TIME (SEC)
8/28/79

FIGURE 3

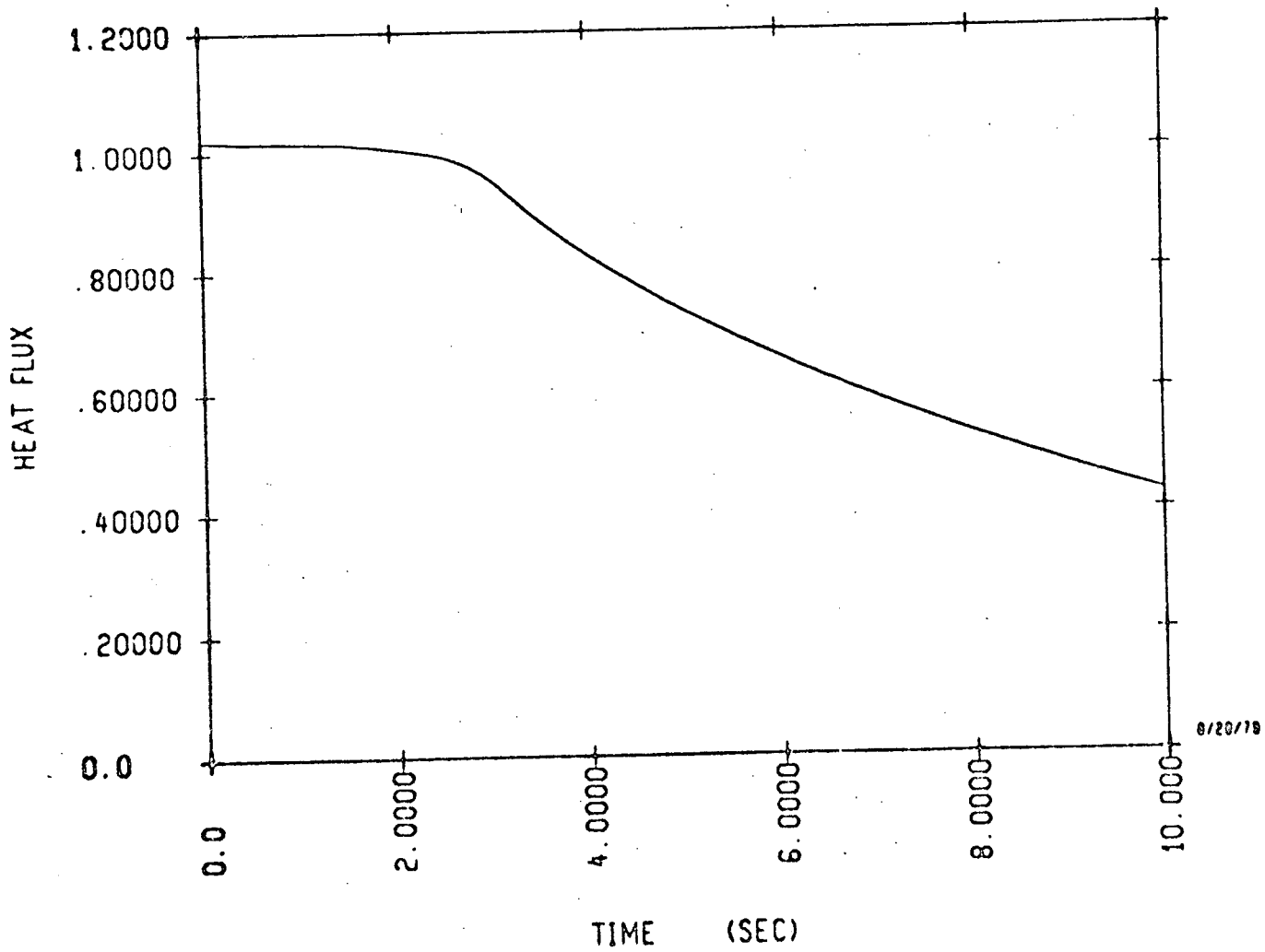
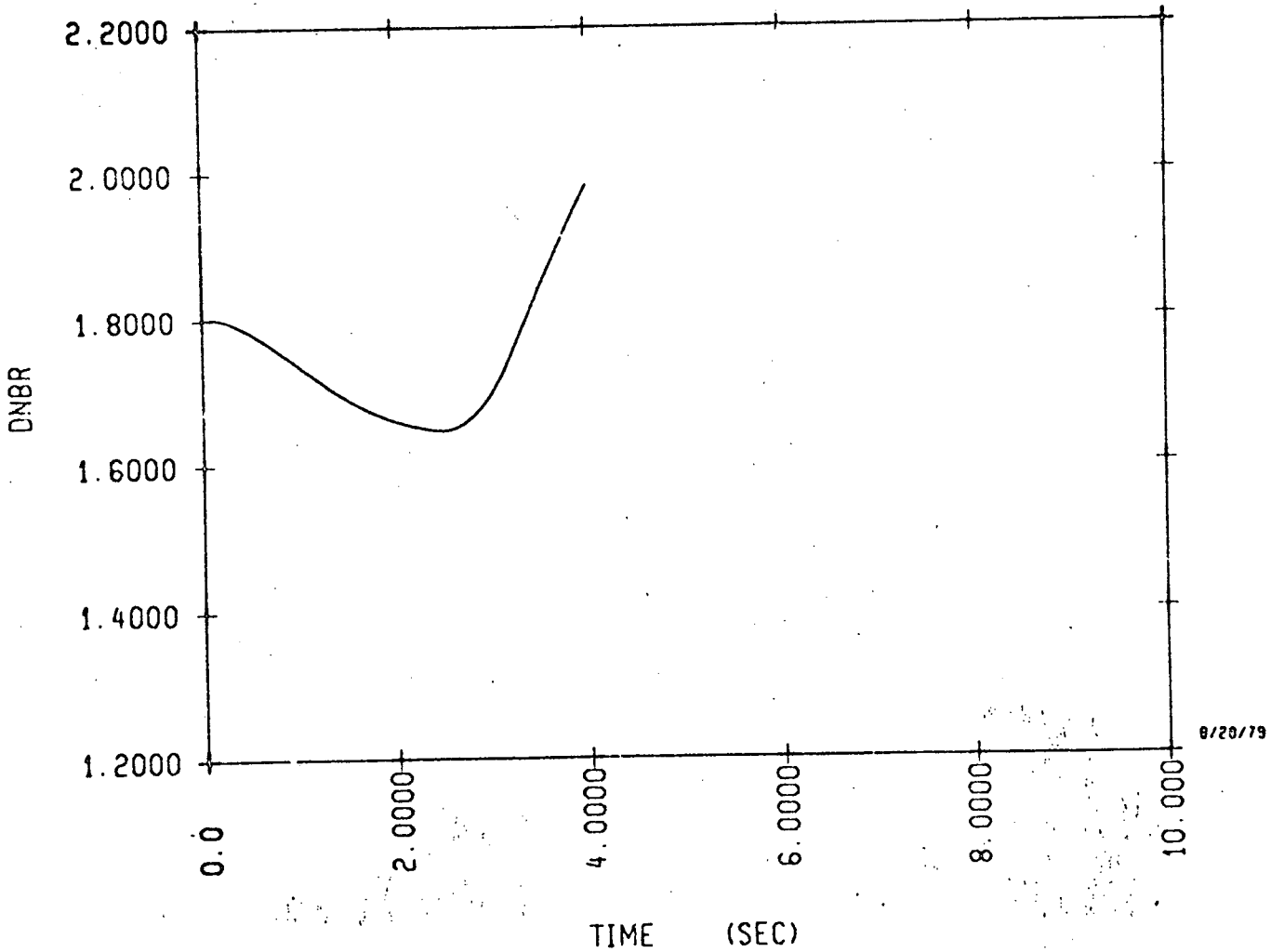

1.2000
1.0000
.80000
.60000
.40000
.20000
0.0
HEAT FLUX
0.0
2.0000
4.0000
6.0000
8.0000
10.000
TIME (SEC)
8/20/79

FIGURE 4


2.2000
2.0000
1.8000
1.6000
1.4000
1.2000
DNBR
0.0
2.0000
4.0000
6.0000
8.0000
10.000
TIME (SEC)
8/20/79