Duquesne Light

Nuclear Division
P.O. Box 4
Shippingport, PA 15077-0004

Telephone (412) 456-6000

July 29, 1983

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Attn: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Request for Additional Information on N-1 Loop Operation

Gentlemen:

In accordance with your letter of May 27, 1983, we are providing the information requested on two-loop operation. We have evaluated the questions submitted by the Reactor Systems Branch (RSB) and the Procedures and Systems Review Branch (PSRB) in your letter. The information is provided in three parts:

Enclosure I - quantitative basis concerning a Steam Generator Tube Rupture (SGTR) event for N-1 loop operation which was requested by RSB.
Enclosure II - responses to the two questions posed by PSRB (Items 1 and 2)
Enclosure III- previously committed LOCA reanalysis using the NRC approved 1981 Westinghouse Evaluation Model for N-1 Loop operation.

We believe that the licensing process associated with N-1 loop considerations has departed from the technical issues unique to the N-1 loop condition towards resolution of generic multiplant issues related to N loop operation. Specifically, the issue of "qualification of the pressurizer power operated relief valves" and "operator response times" raised for the SGTR (Steam Generatur Tube Rupture) accidents during N-1 loop operations are not technical problems strictly associated with N-1 rather, these concerns are bounded by the N-Loop case for SGTR. We base this on the fact that the identical SGTR procedures utilized for N-1 loop operation would be characteristic of the actions taken by the operators for the N loop condition and are therefore similar in this regard. In addition, the parameters affecting the initial conditions and analyzed releases resulting from a SGTR for N-1 loop operation in all cases are comparable to, or more conservative than, the N loop case. One steam generator is adequate for decay heat removal and plant cooldown for either case. Provided as Table I is a comparison of parameters for both the N loop and N-1 loop case and any comments applicable to the conditions.

A001
1/1

The radiological consequences of a Steam Generator Tube Rupture event presented in the FSAR for N loop operation are based on a calculation of the leakage of primary coolant into the secondary side of the affected steam generator and subsequent discharge of radiological effluent via secondary side relief valves. The rate of reactor coolant leakage is dependent primarily on the capacity of the ECCS system and reactor coolant temperature after reactor trip which are not adversely affected by N-1 loop operation. Similarly, since initial power level and reactor coolant system fluid volume are lower, the discharge of steam required to remove decay heat and sensible fluid energy is also reduced.

The accumulated leakage in the affected steam generator is also dependent upon the time required for the operator to cool and depressurize the reactor coolant system to stop primary-to-secondary leakage. These actions can be completed coincident with a loss of offsite power using pressurizer and steam generator power operated relief valves. Neither the availability nor capacity of these components are reduced during N-1 loop operation. Hence, the timing of actions by the operator would not be significantly different for N-1 loop operation.

Evaluation of the net effect of N-1 operation on the radiological consequences of a design basis tube failure leads to the conclusion that the analysis for N loop operation is applicable.

The criteria of ANSI N660 "Proposed Standard for Time Response Design Criteria for Safety Related Operator Actions" state in the FOREWORD of the document that the criteria "are <u>not</u> intended to serve as a basis for <u>actual</u> operator action times, procedures, or training". Therefore, we have made no attempt to use this document as suggested to qualify the 30 minute operator response time but have utilized the SNUPPS simulator and three different licensed groups from our plant to make the determination that all requisite actions for the steam generator tube rupture can be completed within the 30 minute timeframe. Since the timing requirements for N660 were based on simulator analyses, we feel that this is an acceptable alternative. We do not believe that any document or standard can be used to quantify or qualify time response for operators during accident situations and that this can only be done through simulator training and best estimate plant response information recognizing that simulator response will vary somewhat from actual plant thermal hydraulic response dependent on the complexity of the simulator modeling of SGTR.

It was apparent from the three unannounced tests given on the simulator at SNUPPS that the primary requisite actions for SGTR, specifically, identification and isolation of the faulted generator and the subsequent cooldown and depressurization can be performed in the 30 minute timeframe.

<u>Maximum Time of 3 Tests</u>

180 seconds	identification of faulted steam generator
240 seconds	isolation of faulted steam generator
<u>780</u> seconds	cooldown and depressurization

1,200 seconds = 20 minutes

In conclusion, we believe that these tests were more conclusive with respect to operator response time than comparing existing procedures to ANSI N660 criteria since no document can adequately substitute for sound design, training, human factors considerations and live time simulation trials in evaluating operator response to accident conditions.

On this basis, we request that the subject of operator response time for SGTR and the qualification of the Pressurizer PORVs in mitigating the consequences of a SGTR for the N-1 loop condition be evaluated as a technical issue for N loop operation to expedite resolution of N-1, unless a specific parameter(s) unique to this configuration is identified as being less conservative than the N loop case. We have forwarded copies of the SGTR procedure to Mr. P. Tam for your use.

Notwithstanding this request and in concert with our desire to resolve the N-1 licensing ussues, we have evaluated the operation of the Pressurizer PORVs in the 3 loop SGTR Case for the conditions under which it is expected to operate (i.e., SGTR with Loss of Offsite Power) and included this information in Enclosure 1.

If you have any questions on this subject, please contact my office.

Very truly yours,

J. J. Carey

J. J. Carey
Vice President, Nuclear

Attachments

cc: Mr. W. M. Troskoski, Resident Inspector
U. S. Nuclear Regulatory Commission
Beaver Valley Power Station
Shippingport, PA 15077

U. S. Nuclear Regulatory Commission
c/o Document Management Branch
Washington, DC 20555

ENCLOSURE I

RSB Request for Additional Information

Question:

Your response to question 3 in reference 1 concerning a Steam Generator Tube Rupture (SGRT) during N-1 loop operation has been found to be inadequate. No quantitative basis was provided to substantiate your assertion that a SGTR while in the N-1 loop mode would be bounded by the FSAR calculation of a SGTR while in the N loop mode. Further, you have not provided suitable justification that the 30-minute (time to equalize RCS and faulted SG pressures) assumption in the FSAR can be met while in the N-1 loop mode. If suitable quantified bases cannot be provided, you should recalculate a SGTR in the N-1 loop mode, including an analysis of offsite dose consequences. Your analysis should specifically address the following:

- Assumption of loss of offsite power per GDC-17
- Justification for relying on non-safety related equipment for mitigation of the event (e.g., primary PORV, ADV's) should be provided.

The timing of actions taken by the operator should be justified on the basis of current or proposed procedures. The time response criteria of ANSI N660 (reference 2) should be taken into account.

Your response should contain calculated time variations of upper plenum pressure and temperature, saturation temperature, pressurizer level, level in the faulted steam generator, secondary relief and safety valve flows, and secondary temperature and pressure for each steam generator.

Please also provide a chronological listing of automatic actuations and operator actions, justified on the basis of current or proposed emergency procedures. If the 30-minute criterion assumed in the FSAR cannot be met, please provide justification for the current FSAR, N-loop operation SGTR analysis assumption.

Response

With respect to the issue of SGTR coincident with loss of offsite power, we cannot identify any parameter specific to N-1 that would represent a less conservative condition that the N loop case and therefore are evaluating it as an N Loop problem. Attachments 1 through 5 are our justifications for relying on non-safety related equipment for mitigation of a SGTR event coincident with loss of offsite power.

The containment environmental conditions under which the pressurizer PORVs are expected to operate during a SGTR coincident with a loss of offsite power for the 3 loop case envelope the expected N-1 condition. These results and assumptions are presented as Attachment 1.

Presented as Attachment 2 is our risk assessment of the probability of a steam generator tube rupture event coincident with loss of offsite power for N loop operation. Attachment 3 provides a comparison between the factors affecting SGTR for the N and N-1 loop. Attachment 4 details the conservatisms relative to the Unit 1 Technical Specifications and FSAR assumptions for SGTR event. Attachment 5 is a lcgic diagram which represents the various and diverse means of depressurizing the reactor coolant system during a SGTR. In consideration of the low probability of this event, analyses conservatism with respect to 10CFR100 releases, the expected containment atmosphere during the event, substantial capital already expended and the diverse depressurization means available, we feel that it does not warrant backfitting the pressurizer PORVs with fully qualified safety grade equipment.

We recognize that this accident was a design basis consideration for the SGTR, and, on a generic basis, the Final Safety Analysis Report does not provide substantial detail on which to justify the 30 minute operator response time. The Westinghouse Owner's Group Procedures Subcommittee will be addressing the 30 minute time response through the validation and verification efforts on the Emergency Response Guidelines to qualify this time frame. In addition, we will evaluate the final version of our plant specific emergency procedure for SGTR when our simulator is completed. This schedule will be consistent with the Control Room Design Review effort submitted in our Generic Letter 82-33 dated April 15, 1983. We have had three of our licensed groups timed on the SNUPPS simulator utilizing our current procedures to qualify the time to identify and isolate faulted steam generator and perform the subsequent depressurization/cooldown. These results were addressed in the cover letter and must consider the fact that our operators are not familiar with this control board (SNUPPS), as this is Duquesne Light's first use of this facility and that the modeling of the actual thermal hydraulic response will vary dependent on the simulator. Copies of the procedures for SGTR and safety injection have been forwarded to Mr. P. Tam for your use.

The September 1983 meeting of the Westinghouse Owner's Group (WOG) has an agenda item on the SGTR 30 minute response time issue that will be voted on to address this problem on behalf of the NTOL's (Seabrook, Shoreham, Harris, Catawba), who have open items in their safety evaluations in this regard. This issue was also identified as an open item (#16) in the Safety Evaluation of "Emergency Response Guidelines" in the June 1, 1983 D. G. Eisenhut letter to J. J. Sheppard of the WOG. The proposed WOG agenda items relative to SGTR include:

- justification of operator response time,
- consequences of delays (i.e., steam generator overfill),
- qualification of equipment used in mitigation of accidents, and
- limiting single failures.

We expect that this issue will be adequately funded, and we believe the results will be comparable to our preliminary analyses of the SGTR event and justify the limited conditions under which the pressurizer PORVs must operate. This effort will be responsive to the concerns expressed in the April 4, 1983 D. G. Eisenhut memorandum to the Commission entitled "Board Notification Regarding the Need for Rapid Primary System Depressurization Capability" in PWRs (BN-83-47) and the R. J. Mattson memorandum to D. G. Eisenhut dated March 27, 1983 entitled "Board Notification Regarding PORV's".

In the event that funding is not approved for this issue, we intend to stand on the existing design of the Pressurizer PORVs since our preliminary review indicates that these valves should function under the limited challenge imposed on them by the operating environment in containment during a SGTR.

Duquesne Light Company has spent $288,750 for testing the Masoneilan-type power operated relief valves through the EPRI Test Program to satisfy the NUREG-0737, Item II.D.1.A requirement. Results of the testing conducted at the Marshall Steam Station, where eleven different evaluation tests were performed with a total of 63 cycle operations, showed that in all cases the valves opened and closed on demand with no failures or damage to the valve and that the lowest recorded closing pressure was 2205 psig.

To satisfy the concerns of cold overpressure, we have upgraded and performed numerous modifications involving:

- NUREG-0578 Pressurizer Safety and Relief Valve Position Indication Acoustic Monitoring Modifications, Subcooled Margin Meter, the RCS Vent Modifications, and the Reactor Vessel Level Instrumentation Modification

- Installation of pressure switches in each pressurizer safety relief valve for monitoring pilot assembly leakage

- Pressurizer Spray Valves Replacement Modification (scheduled for the fourth refueling outage)

- Upgrading the Pressurizer Safety Relief Valves

- RCS Overpressure Protection Modification

Total expenditure of the above listed modifications, to date, is $7.3 million.

We have diligently followed and analyzed industry experience on these valves and are currently modifying the valves' air system based on our review of the Westinghouse Tech. Bulletin (NSD-TB-82-02) on the potential for vent port restriction due to orifices or elbows in the air line, which was dated April 15, 1982.

With due consideration for the substantial testing, modifications, man-rem exposure and capital investments associated with these valves to satisfy multi-plant issues, Duquesne Light Company does not intend to further enhance the qualifications and acceptability of these valves based on their capability to perform their intended function under the expected limited service condition for SGTR. We have not identified any failures of these valves in our review of events(1) that were service induced or caused by adverse environmental conditions.

The NRC staff review of NUREG-0651 concluded that during the SGTR cases, "no significant offsite doses or system performance inadequacies have occurred..., and only minor procedural and equipment deficiencies were noted". Moreover, no loss of offsite power occurred prior to or following a SGTR event.

References

1. NUREG 0651 Evaluation of Steam Generator Tube Rupture Events
   Tube Rupture Events
   NUREG 0886 Steam Generator Tube Experience
   NUREG 0909 NRC Report on the January 25, 1982 Steam Generator
   Tube Rupture at R.E. Genna Nuclear Power Plant
   Kemeany Report on the Accident at Three Mile Island
   NUREG CR-3226 Station Blackout Accident Analyses

ATTACHMENT 1

We have evaluated the conditions under which the pressurizer PORVs must operate for a steam generator tube rupture coincident with a loss of offsite power.

Our calculations indicate that the pressurizer discharge volume required for depressurization to the faulted steam generator pressure is relatively small in terms of total energy release and duration (<5 minutes) for the N loop condition.

We have performed a limited parametric review and iterative mass and energy calculations over incremental time periods and determined that the containment atmosphere is not significantly affected to the point that PORV operability is challenged during the period of PORV operation, as the energy is substantially absorbed by the pressurizer relief tank surge volume.

The assumptions and initial conditions utilized in performing the mass energy releases for SGTR are:

1. The pressurizer relief tank pressure, level and temperature are all at their alarm settings prior to the release (78% level, 22.7 psia, 125F)

2. PORV capacity 210,000 lbm/hr @ 2250 psia C_v = 46 critical flow coef. 0.9

3. Pressurizer liquid V_f = .02448 ft^3/lbm h_f = 642.3 BTU/lbm
   Pressurizer steam V_g = .2269 ft^3/lbm h_f = 1153.7 BTU/lbm

   based on a break flow stabilization pressure of 1750 psia

4. RCS Temperature reduced to 497F prior to depressurization in accordance with SGTR procedures.

5. Charging Pump Mass input (See figure 14.2-3 Updated FSAR attached)

6. 1400 ft^3 pressurizer
   1300 ft^3 pressurizer relief tank

7. Initial PRT Internal Energy = $93.225 \frac{BTU}{lbm}$ V_f = .0162 ft^3/lbm

8. Pressurizer level at time of PORV actuation = 50%

9. Curves for the pressurizer and relief tank are attached

Since the three PORVs fail closed and their respective isolation valves are being qualified under the EQ Program, we believe the valves are capable of performing their intended and limited safety function over the time frame of interest for a SGTR during the loss of offsite power scenario.

Our preliminary analyses indicate that the PRT would not rupture during a SGTR, however, if the PRT did rupture, the containment temperature expected for this short term energy release is approximately 230°F which would decay quickly due to cessation of the release and installed containment cooling systems. If the pressurizer relief tank surge volume cooling capacity, structural heat sinks, time delay associated with attaining equilibrium containment temperatures and time delay until any hypothetical pressurizer relief tank rupture occurs, are considered for the duration that the valve must function under SGTR with a loss of offsite power, it is highly improbable (i.e, $< 10^{-5}$) that at least one of the three installed PORVs would not function under these conditions.

<u>References</u>

1. BVPS Updated FSAR

2. Wark, K "Thermodynamics" McGraw/Hill New York 1927

3. Keenan, J. H. and Keyes, F.G. "Steam Tables" John Wiley and Sons, Inc. USA, 1969

4. BVPS OM Chapter 53, Procedure E-3

5. Westinghouse E-Spec 676270

6. Masoneilan Handbook for Control Valve Sizing Masoneilan Inc. 1977.

7. "Thermodynamics" Abbot and Van Ness, Schaums Outline, 1972

REV. 0 (1/82)

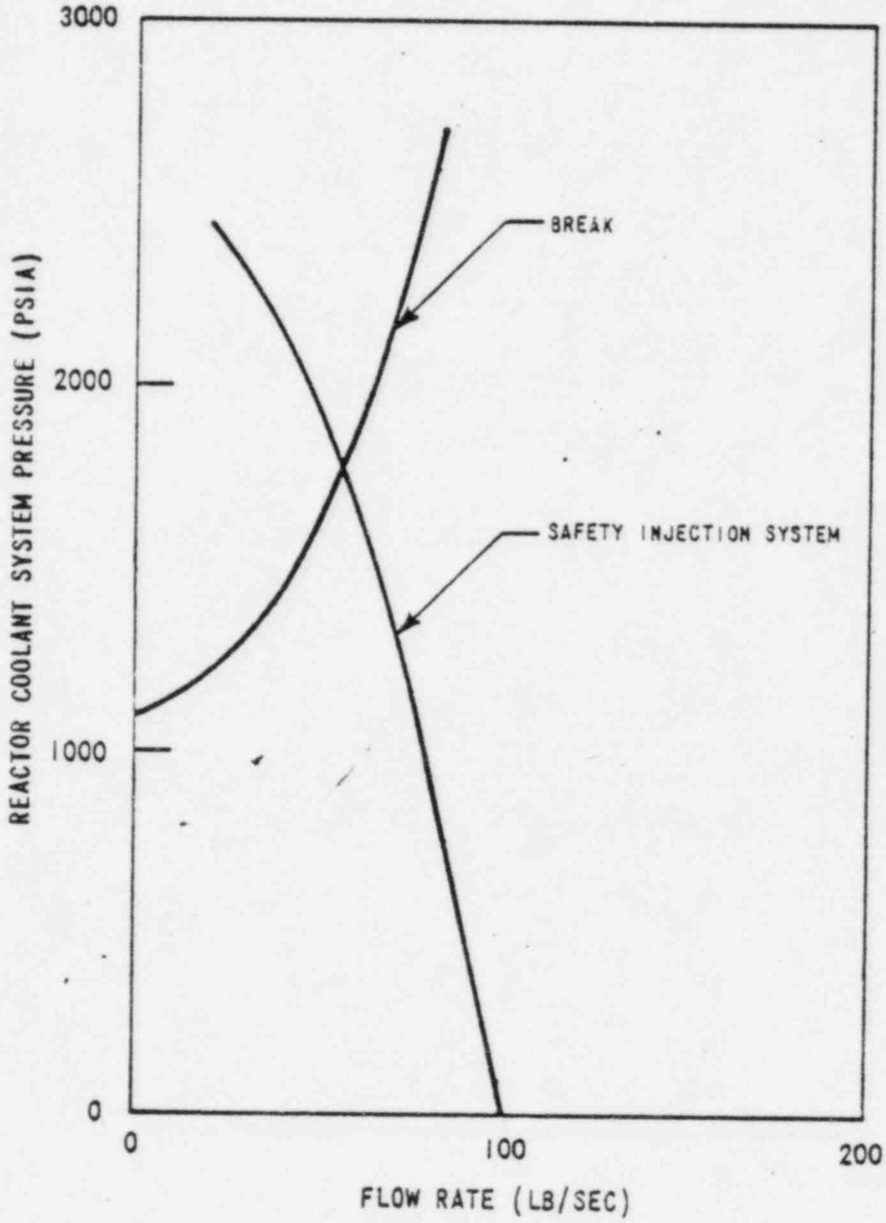


FIGURE 14.2-3
DLW - STEAM GENERATOR TUBE
RUPTURE EQUALIZING PRESSURE
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 0 (1/82)

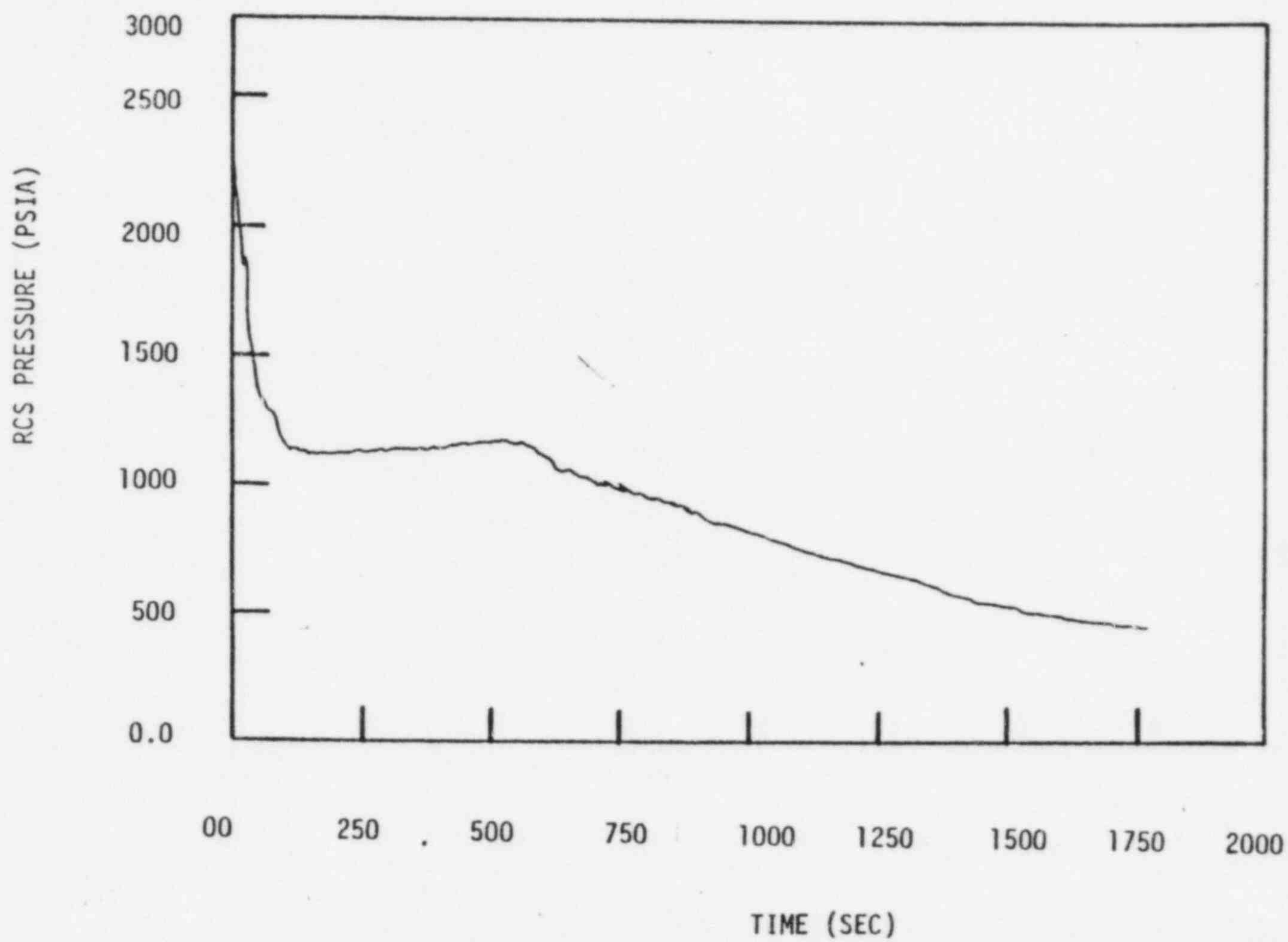
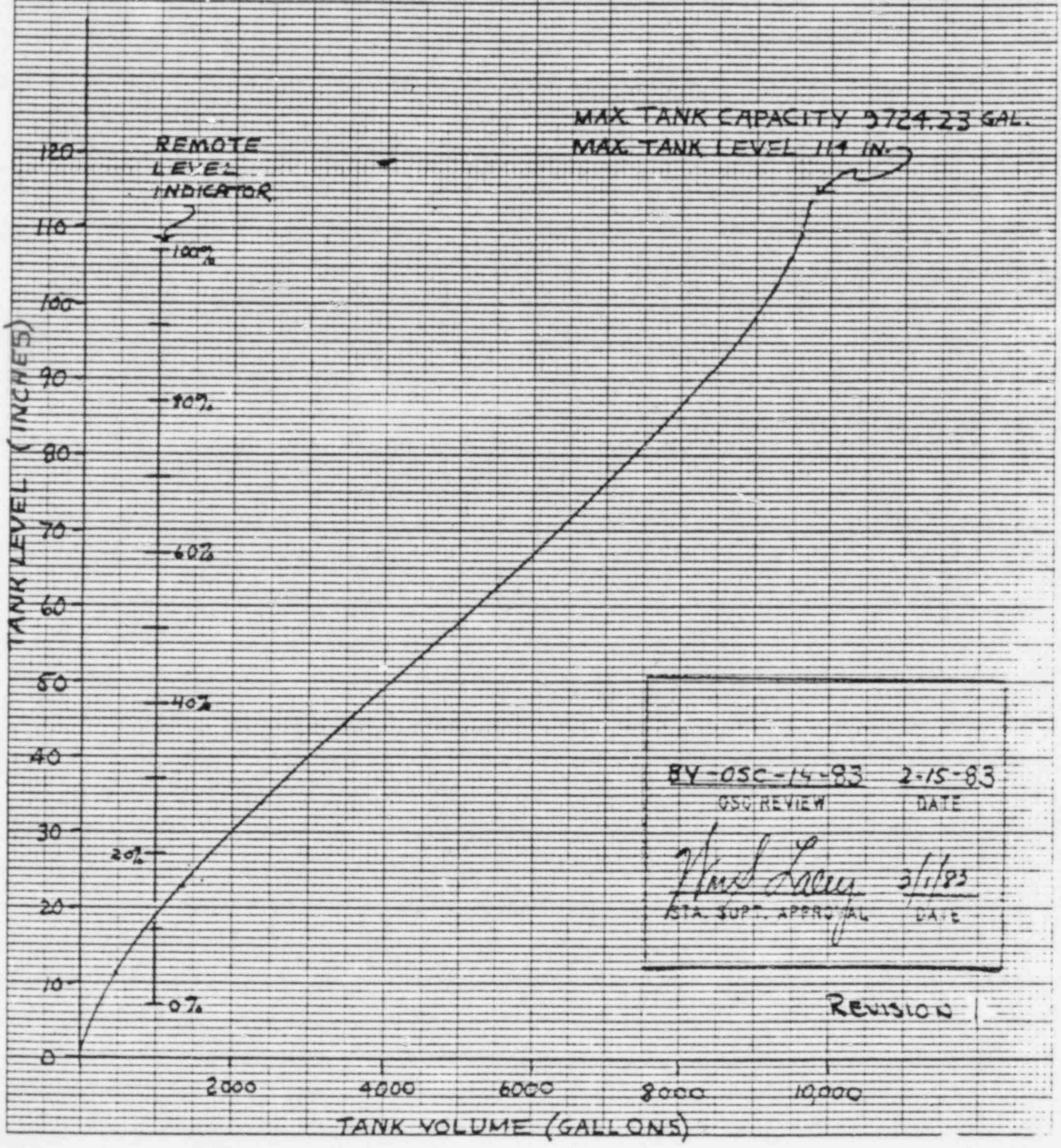
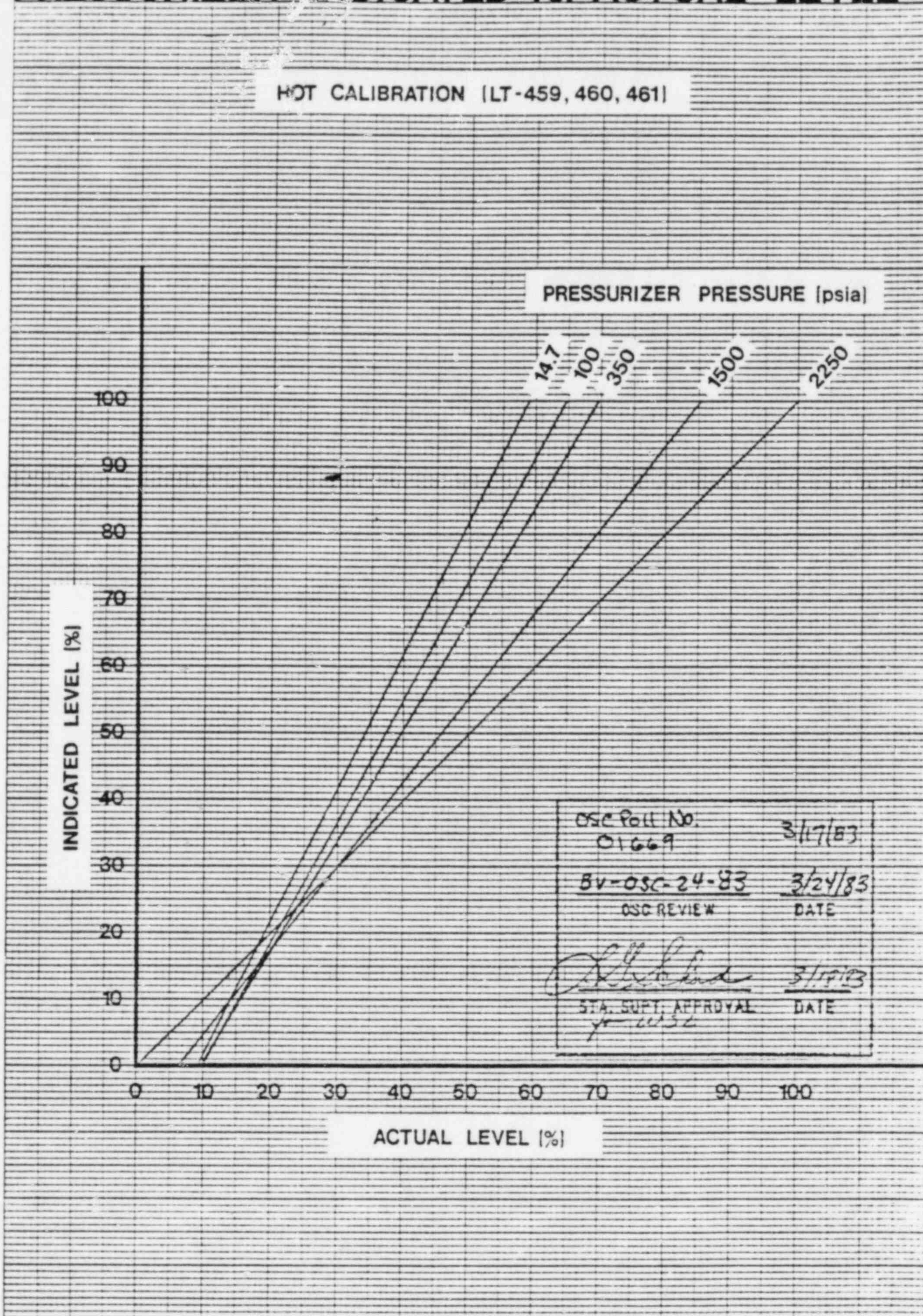


FIGURE 14.3-9A
RCS DEPRESSURIZATION TRANSIENT (3 INCH)
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

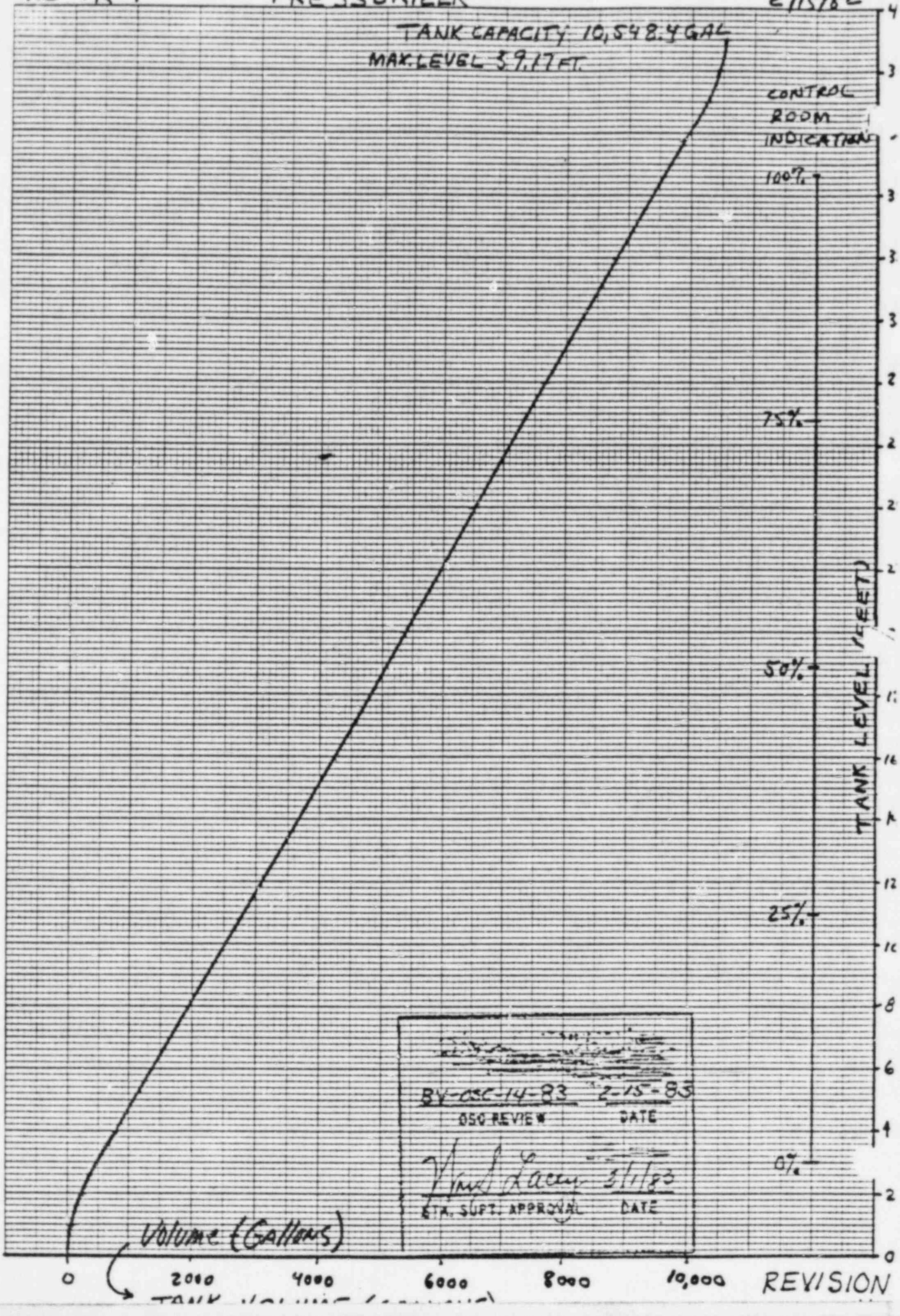
RC-TK-2 Pressurizer Relief Tank 12-27-82
CAB



PRESSURIZER INDICATED vs. ACTUAL LEVEL



RC-TK-1 PRESSURIZER 2/15/82



ATTACHMENT 2

Risk Assessment Evaluation for SGTR Coincident with Loss of Offsite Power

The following is a risk assessment addressing SGTR concerns using typical probability values.

1. An abbreviated method following NUREG/CR-2934 (SANDIA), Review and Evaluation of the Indian Point Probabilistic Safety Study (IPPSS), sequence to core melt for Tube Rupture (preliminary evaluation) is used for this case. This event addresses SGTR coincident with a stuck open secondary safety value. In the event of core meltdown, this may result in a direct radioactive material release to the atmosphere. The SGTR frequency used is from EPRI/NP-2330, dated January, 1982. No reduction in Tube Rupture probability for BV-1 was attempted considering all volatile treatment (AVT) for secondary chemistry. The four incidents reported in NUREGs-0651 and 0909 were for plants that initially used phosphate secondary chemistry. The staff indicated in NUREG-0651 that AVT has somewhat alleviated the concern over the recurrence of a Point Beach Unit-1 type tube rupture incident.

A) S/G TR	x	Failure of HPI System (ASP*)	x	Secondary Safeties demanded to open. (Assumes PORVs closed by Procedure)	x	At least one safety fails to close.

4.0 E-2 x 1.3 E-3 x 1.0 x .01 = 5.2 E-7

B) S/G TR	x	Secondary Safeties demanded to open.	x	At least one safety fails to close.	x	Failure of RHR Pumps (ASP*)

4.0 E-2 x 1.0 x .01 x 1.2 E-3 = 4.8 E-7

C) A + B = 1.0 E-6

*Value from Accident Sequence Precursor Study, NUREG-2497 (ORNL)

2. Loss of Offsite Power (LOP) and S/G Tube Rupture

   The Beaver Valley Unit-1 FSAR Accident Analysis documents this event. The three cases discussed below provides an assessment of the probability magnitude. Reasons why the occurrence of this event is considered highly unlikely are also presented.

   A) LOP and SGTR-coincident occurrence by unrelated causes.

      The coincident LOP and SGTR probability is considered negligible when the events are not causally related.

From WASH-1400, addressing LOP and LOCA, "Since the time of interest for this event is of the order of one minute, the likelihood of losing offsite power by a failure which is not causally related to the LOCA is negligible." It is assumed that the time span of interest for a SGTR would be comparable to that for a LOCA.

B) LOP during SGTR event by unrelated causes.

In this case, it is assumed that a SGTR event has occurred and at same time after the event there is a LOP due to an unrelated cause. In NUREG/CR-2497 (ORNL) the frequency of LOP was evaluated to be 4.1 E-2/yr. (4.8×10^{-6}/hr) in which was included the chance of rectifying the initiating event. A value of 1.8 E-3/demand (NUREG CR-2497) was used for failure of emergency power. Therefore, the probability for total loss of A/C power is calculated to be 7.4 E-5/yr. If it is assumed that LOP occurs within 30 minutes after tube rupture, the time to equalization of primary and secondary pressures, the probability can be estimated to be on the order of magnitude of E-9. For comparison, WASH-1400 assessed a point estimate of the failure rate of offsite power to be 2 E-5 failures per hour and the probability that both diesel generators Unit 1 will trip out to be 10 E-2. Although the methodology is somewhat different, it is noted that the WASH-1400 probability of total loss of electric power after a LOCA for 1 hour is 2.0 E-7.

C) LOP at S/G TR- Causally related

For this event consideration is given to the possibility of a SGTR resulting in a turbine generator trip and subsequent transient instability of the transmission grid due to the loss of generation. WASH-1400 assessed that the probability of losing offsite power due to LOCA induced power system transient is 10 E-3 and the failure of 2 diesel generator sets is E-2. These values are used below. Of concern here is the possible reduction of available depressurization capability. Upon LOP, based on Beaver Valley Unit-1 Operating Procedures, the pressurizer PORVs would be used to reduce primary system pressure thereby equalizing the pressures between the primary system and the secondary side of the affected steam generator. This action would serve to attenuate break flow into the secondary release path. The probability of PORV failure to operate upon demand is E-3 (upperbound, WASH-1400). The probability for this event can be calculated by:

LOP x PZR PORV Failure

E-3 x E-3 = E-6

assuming that the probability that the PZR PORV demand is one and emergency power is available. The total loss of AC power can be calculated by:

LOP x Failure of Emergency Power

E-3 x E-2 = E-5

These probabilities are considered conservatively high estimates for the following reasons:

1) It is somewhat doubtful that LOP would occur at all since none of the Tube Rupture Events reported in NUREGs 0651 and 0909 indicated a LOP.

2) From NUREG-0886 (2/82) "The probability of the design basis accident occurring during normal operation is small, and the probability that the accident would occur during the short period of time between the detection of a leak and that exceeding the Technical Specification leak rate limit and plant shutdown is even smaller."

3) No credit is taken for corrective action. Restoration of offsite power would increase the probability of full depressurization capability. Data from Appendix III of WASH-1400 approximates that a 64% restoration of offsite power within 30 minutes of an event such as a LOCA. On July 28, 1978, a total loss of offsite power occurred at BV-1 due to a main transformer fault and improper relay operation. Offsite power was restored in 17 minutes.

4) Only one of three PORVs is considered operable.

5) Beaver Valley Unit-1 has operated with AVT secondary chemistry.

ATTACHMENT 3

PARAMETER	COMPARISON OF N AND N-1 CASE (STOP VALVES CLOSED)	COMMENT
Pressurizer PORV Flow Rate	210,000 lbm/hr at 2350 psig saturated steam	No affect - 3 PORV's installed
RCS Pressure	2235 psig initial	No affect
Nuclear Power	100% plus calorimetric N loop 60% plus calorimetric N-1 loop	Makes N-1 SGTR more conservative
Decay Heat	Lower for N-1 loop due to lower initial power using ANS 5.1 Decay Heat	Makes N-1 SGTR more conservative
RCS Temperature	Lower T_{avg} for N-1	Makes N-1 SGTR more conservative
RCS Mass Inventory	Less for N-1 loop due to isolated loop inventory	Makes N-1 SGTR more conservative since less mass has to be cooled during post accident
Coolant Activity	Limits restricted by Technical Specification 3.4.8	Short term increased I 131 activity for N-1 permitted but compensated b, lower decay heat
Fission Product Inventory	Less for N-1 loop due to lower equilibrium levels of fission products	Makes N-1 SGTR more conservative
Steam Generator Tube Size	.875"	No affect
Steam Pressure	Higher for N-1 due to lower power	Makes initial conditions for N-1 SGTR more conversative due to lower tube ΔP, less break flow initial
Steam Generator Level	44% plus instrument errors	No affect as post trip steam pressures(would be the same, ie. same mass)
Break Flow	Slightly less for N-1 initially due to higher steam pressure during N-1, lower tube differential pressure	Makes N-1 SGTR more conservative
ECCS Flow Rates	Same for N and N-1 loop case	No affect
Cooldown	Faster for N-1 due to less mass in primary to be cooled and less decay heat	Makes N-1 SGTR more conversative
Site Meteorology	Same for N and N-1 loop case	No affect
Offsite Power Availability	Same for N and N-1 loop case	No affect
Secondary Heat Removal	Steam Dump, PORV's, and Safety Valves available for both cases	No affect, but due to less decay heat the offsite release would be less which would make N-1 more conversative

ATTACHMENT 4

Conservative FSAR Assumptions of SGTR Event

1. Analysis of tube thinning at Prairie Island established that even a tube with a 65% wall reduction did not rupture, while tube plugging is required at BVPS 1 for any tube with a wall reduction of 40%. (NUREG 0651, pg. C-6 and BVPS Technical Specification 3.4.5)

2. BVPS SGTR Analysis assumes 15X15 fuel while a 17X17 is actually used. This is conservative since the diffusion of radioactive isotopes in the fuel is temperature dependent and the 17X17 fuel operates at a lower temperature. Therefore, the release of fission products from the pellet to the clad gap is reduced. (FSAR Section 14.2.4)

3. Radiation monitors have been installed in the main steam relief system in conjunction with NUREG-0737, Item II.F.1 which would expedite operator identification of the faulted steam generator without benefit from the Air Ejector and Blowdown Radiation Monitor as described in the FSAR.

4. A qualified pressurizer and vessel head vent system has been installed to meet the requirements of NUREG-0737 Item II.B.1. This system would provide a limited depressurization rate due to a 7/32" orifice being installed in-line but would be available on a loss of all A.C. pending NRC review of the procedures previously submitted.

5. The radiological consequences of a SGTR event presented in the FSAR Section 14.2.4 produces a dose at the site boundary of 300 mRem whole body and 900 mRem thyroid which is substantially within the limits of 10CFR100, even if it is assumed the operator delays in taking action when warned by alarms and instruments. Therefore, as long as a single phase steam release is maintained, the accumulated release would be conservatively within 10 CFR 100 limits even if we assume the primary depressurization and isolation times already established during actual SGTR events and documented in NUREG 0909. Equalization times for actual events obtained from NUREG-0909 are summarized below:

   Point Beach . . . . 108 minutes

   Surry 2 . . . . 30 minutes

   Ginna . . . . 180 minutes

   Prairie Island . . . . 61 minutes

6. The activity release through a faulted steam generator, which is limited by the concentration in the reactor coolant assumed to result from 1% failed fuel, is highly conservative based on the fact the worst case coolant activity measured throughout

reactor operations at BVPS to date, was approximately 1 uCi/cc.

7. Conservative Meteorology data was also utilized in the activity release calculations. A X/Q value of 7.8×10^{-4} sec/m^3 was used in the FSAR, whereas the actual annual average X/Q value for BVPS is 7.1×10^{-5} sec/m^3. This would substantially reduce the offsite release rate below the projected FSAR assumed release.

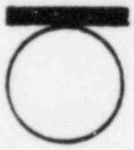
8. In the Model 51 steam generators utilized at Beaver Valley, approximately 2638 ft^3 of volume is available above the tap of the steam generator level span to the main steam isolation valve in the shortest run (438 ft^3) of steam pipe in the 1B Steam Generator.

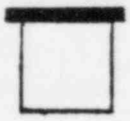
   The break flow rate through a double ended steam generator tube at a pressure of 2250 psia, which conservatively bounds the possible mass addition to the faulted steam generator is approximately 75 lbm/sec. assuming a nominal density of break flow of 50 lbm/ft^3, the volumetric flow rate is about 1.5 ft^3/sec which indicates that the operator has more than 20 minutes to fill a faulted steam generator to the MSIV <u>after</u> the indicated level has gone off-scale high.

   If a more realistic break flow rate were used at the equilibrium pressure where SI flow matches break flow and consideration given to the "shrink" in the steam generator level post trip, substantial time beyond 20 minutes could be realized. The Unit 1 FSAR analyses for SGTR was based on a mass transfer of 132,000 lbm to the secondary which is conservative with respect to the actual flows that would be anticipated for the expected duration.

ATTACHMENT 5

Legend/Notes:

 - "or" gate, requires any input signal to produce an output

 - "and" gate, requires all input signals to produce an output

 - ½ sources of offsite power required for component operation

1

For systems inside containment, air is supplied by 1 of 2 air compressors [IA-C-1A, 1B] which are powered from emergency power sources. Cooling to these units is normally supplied by a non-IE source (chilled water) but has a 1E backup source (river water). For systems outside containment, air is supplied by 1 of 3 non-IE power air compressors (SA-C-1A, 1B, 1C) which are operable if one offsite power source is available to the 480 volt busses. A diesel-driven backup compressor is also installed and verified operable on a weekly basis. A permanent modification to power 1 of 3 motor-driven air compressors and cooling supplies from a diesel generator power source is being designed for projected completion during the 5th refueling outage. The air system inside containment can be cross-connected with the outside air system consistent with containment integrity technical specifications.

References

Drawings RE21DN, RM155B, OM 36, OM 37, OM 38, OM 39
Manuals OM 6, 7, 29, 34, 36, 37, 38, 39 Sections 1-5
OM 34 Procedures J, L
DCP 295 File
IE/Circular 80-15 File
WOG Letter 82-155, 83-200
Westinghouse letter NS-PL-11697 6/30/83
NUREG 0737 Item II.B.1 File
EDS Report NUREG 0737
MED 00177 5/13/82
Schneider Summary Report EQ Status 6/24/82

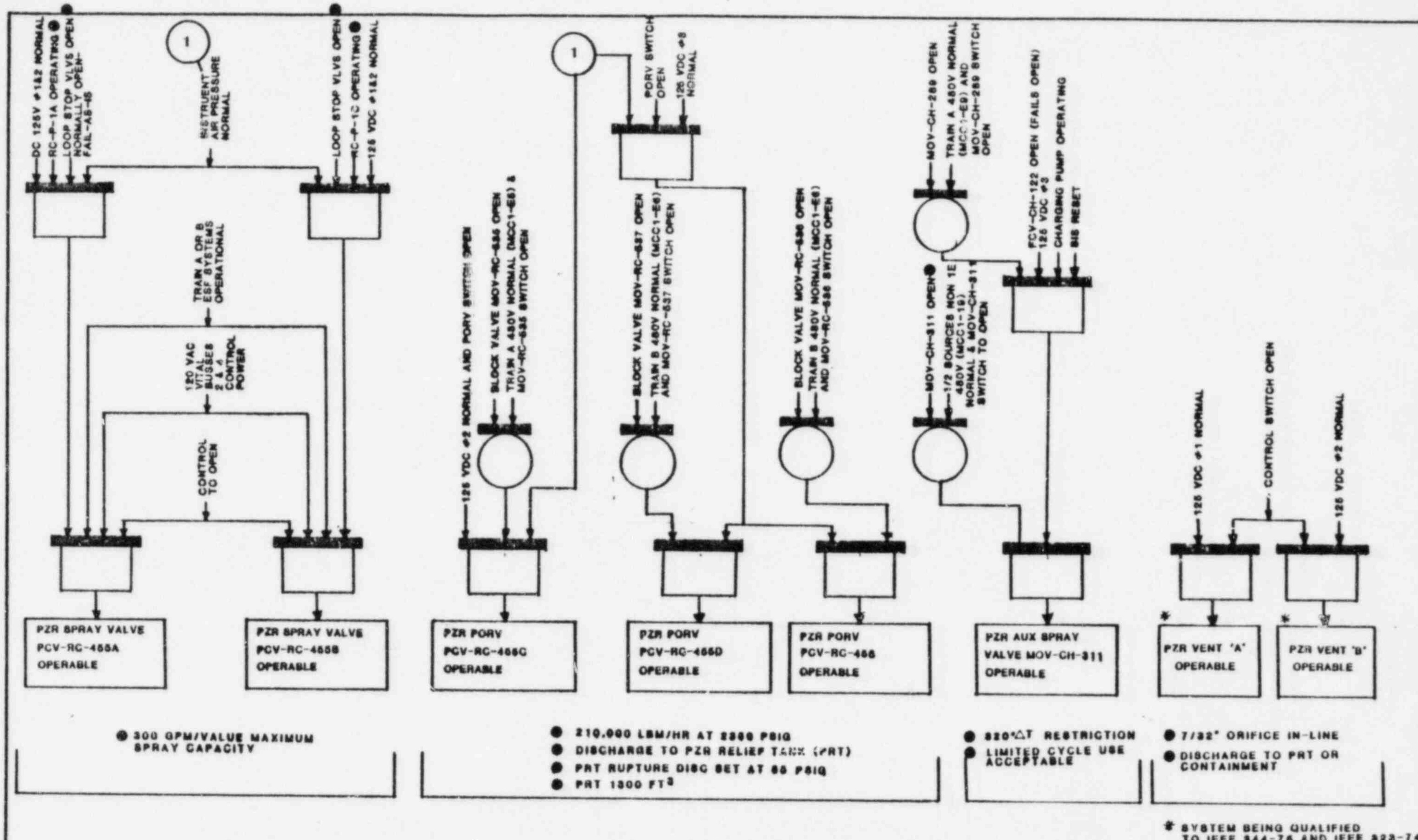


TABLE A - AVAILABLE DEPRESSURIZATION CAPABILITY

ATTACHMENT 5

ENCLOSURE II

PSRB Request for Additional Information

Item 1

The BVPS-1 Startup Report (Initial Startup Cycle) indicates that N-1 flow coastdown measurements were performed during Initial Startup. Confirm that the data is still valid (i.e., no changes to RCS piping, reactor internals, fuel design, S/G's or RCP's have occurred which would affect the original test results in a less conservative direction).

Response

An evaluation was conducted to review changes to the Reactor Coolant System and internals. The following is a summarized listing of minor changes made or scheduled since initial startup and the affects on flow coastdown:

Change	Affect
- Reactor Internals: Guide tube spit pin replacement modifications (Westinghouse design)	Insignificant
- Fuel Design: 1. Two optimized fuel assemblies 2. Pre-pressurization of fuel reduced from 500 to 450 psig	Insignificant
- Steam Generator Tubes: One tube was plugged during last cycle (Cycle 3)	Insignificant
- RCS Hot and Cold Leg Piping: No changes	No affect

Attached is a copy of the Test Results Report for the initial flow coastdown test (BVT 1.1-7.6.1) performed in 1976. The report identifies the case for two loop operation with a loop isolated. The test demonstrated measured core flows in excess of their corresponding FSAR curves (Reference Figure 3.8-6) for the N-1 loop case.

Based on our evaluation, it can be concluded there have been no significant changes that would affect the original test results of the N-1 flow coastdown measurements performed during the initial startup in a less conservative direction.

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station - Unit 1

Test Results Report

Date: 10/14/76

BVT 1.1-7.6.1 Title Reactor Coolant Flow Coastdown
Issue 5/2/76 Revision 7/2/75 JTG Approval Date 8/26/76
Test Date: Start 5/5/76 End 5/12/76 Partial Test No Complete Test Yes

Test Results: Satisfactory VII A,B,C,D,E,F. Unsatisfactory None
Retesting Recommended No Unreviewed Safety Question
Attachments: FIGS. 3.8-1 THRU 3.8-6, FIG. 2A Involved/Evaluated No / NA

Purpose-Scope:

To measure the rate at which reactor coolant flow changes following various reactor coolant loop loss of flow incidents, and to measure the protective system time delays associated with a loss of flow incident to determine the values assumed in the accident analysis are conservative.

Test Summary (Conclusion):

All values measured/calculated for this test satisfied the applicable acceptance criteria as specified in the procedure (shown on page 4 and Figs. 3.8-1 thru 6).

Individual loop flow data for this test was obtained from visicorder traces of one loop flow transmitter output voltage for each loop. These traces were reduced in accordance with the W startup procedure, DLW-SU-5.1.8, flow coastdown measurements. As backup data, RCS pressure, T_{ave} and flow in each loop were recorded (three indications per loop) before and after each test run.

All cases were analyzed for the first ten seconds of the flow coastdown transient and compared to the W revised FSAR curves and calculated minimum DNBR points for each case, which were also furnished. Appropriate allowances were made for the flow sensor time delay (0.608 seconds) which was extracted from the data (Fig. 2A).

Recommendation (if any):

None

Review		O.S.C. Approval	JTG Approval	
Test Engr.	Thomas S Psaras	J A Werling	S & W	J E Maidl 2/17/7
Test Supvr.	D. H. Williams	Chairman	W (Other)	E L Morris 2/17/
Sta. Supt.	J A Werling		D.L.Co.	J Carey 2/17/7

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station

Test Results Report Continuation Sheet

Test Summary (Conclusion): (Continued)

Initial attempts to compare the flow coastdown data obtained during this test with the curves furnished in the BVPS FSAR section 14 were not successful due to the difficulty in extracting exact points from the FSAR curves for data comparison. Upon request, W furnished coordinate values used to plot the curves shown in the FSAR. The initial plotting of the test data and the data furnished by W revealed numerous instances where the test data fell below the values for the FSAR curves by observable amounts. W personnel visited the site to review the raw test data reduction which resolved some of the deviations; however, in most cases, the test data indicated slightly quicker flow coastdown than the curves furnished in the FSAR for the initial few seconds of the transient. This problem was discussed with W personnel and it was agreed by all parties that the method used to correct for sensor time delay, although performed in accordance with W guidelines for data reduction, made comparison of the flow values determined during the onset of the transient to the FSAR values appear pessimistically low. This did not resolve the reason for the lower measured values during the later portions of the transients. After further discussion W indicated they would reanalyze the particular loss of flow transients with some changes to the input parameters of the code to more closely address the dynamic characteristics of the reactor coolant pumps installed at BVPS. This new analysis was performed and new curves were furnished by W for use in the analysis of the results of this test and for subsequent revision of the curves contained in the BVPS FSAR. W also furnished the coordinate values used to generate these curves to allow more precise comparison with the test data. In the following sections of this report, references to the FSAR curves and values are based on the reanalysis of these transients rather than the analysis performed initially for the FSAR. While the exact numbers quoted for the time interval from the initiation of the event until minimum DNBR is reached, as well as the specific values for the DNBR associated with the various transients have changed, the changes are relatively small in value and the original conclusions stated in the FSAR concerning these accidents are not affected by the results of the reanalysis.

The case of three reactor coolant loops operating with one reactor coolant pump coasting down (RC-P-1A) revealed an absence of the loop C input flow signal (From TP-436-1 via FT-RC-436) and failed to trace on the visicorder paper. However, as flow was verified in loop C by backup indication in the control room, this run was not repeated. Since flow in loop C was ≥ 100% in subsequent cases under similar conditions, the conservative assumption of 100% loop C flow was used to arrive at a core flow figure for this case. With the above assumption, the measured value for core flow was found acceptable in comparison to its FSAR curve (refer to Fig. 3.8-1). With the slowest full length control rod withdrawn to 228 steps (K-14 of CB-A), and all three reactor coolant pumps operating, RC-P-1A was tripped and the following monitored variable successfully met their acceptance criteria:

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station

Test Results Report Continuation Sheet

Test Summary (Conclusion): (Continued)

Low flow trip time delay (1.70 second), undervoltage trip time delay (1.17 second) and pump underfrequency trip time delay (0.53 second refer to data sheet on page 4).

The case of three reactor coolant loops operating with three reactor coolant pumps coasting down, proved that the measured minimum core flow is above the FSAR values for flow coastdown (Refer to Fig. 3.8-2). The slope of the measured inverse core flow was less than the FSAR curve slope for this case, as required (Refer to Fig. 2A). The test requirement that all three RCPs trip within 100 milliseconds of each other was proven in the above case (two RCPs trip within 100 milliseconds of each other was demonstrated in subsequent cases that followed).

The cases of two out of three loops operating with one or two reactor coolant pump(s) coasting down - one isolated loop, demonstrated measured core flows in excess of their corresponding FSAR curves (Refer to Figs. 3.8-6 and 3.8-4).

The cases of three reactor coolant loops operating with one or two reactor coolant pump(s) coasting down - no isolated loop, also demonstrated measured core flows above the values shown in the corresponding FSAR curves (Refer to Figs. 3.8-5 & 3.8-3).

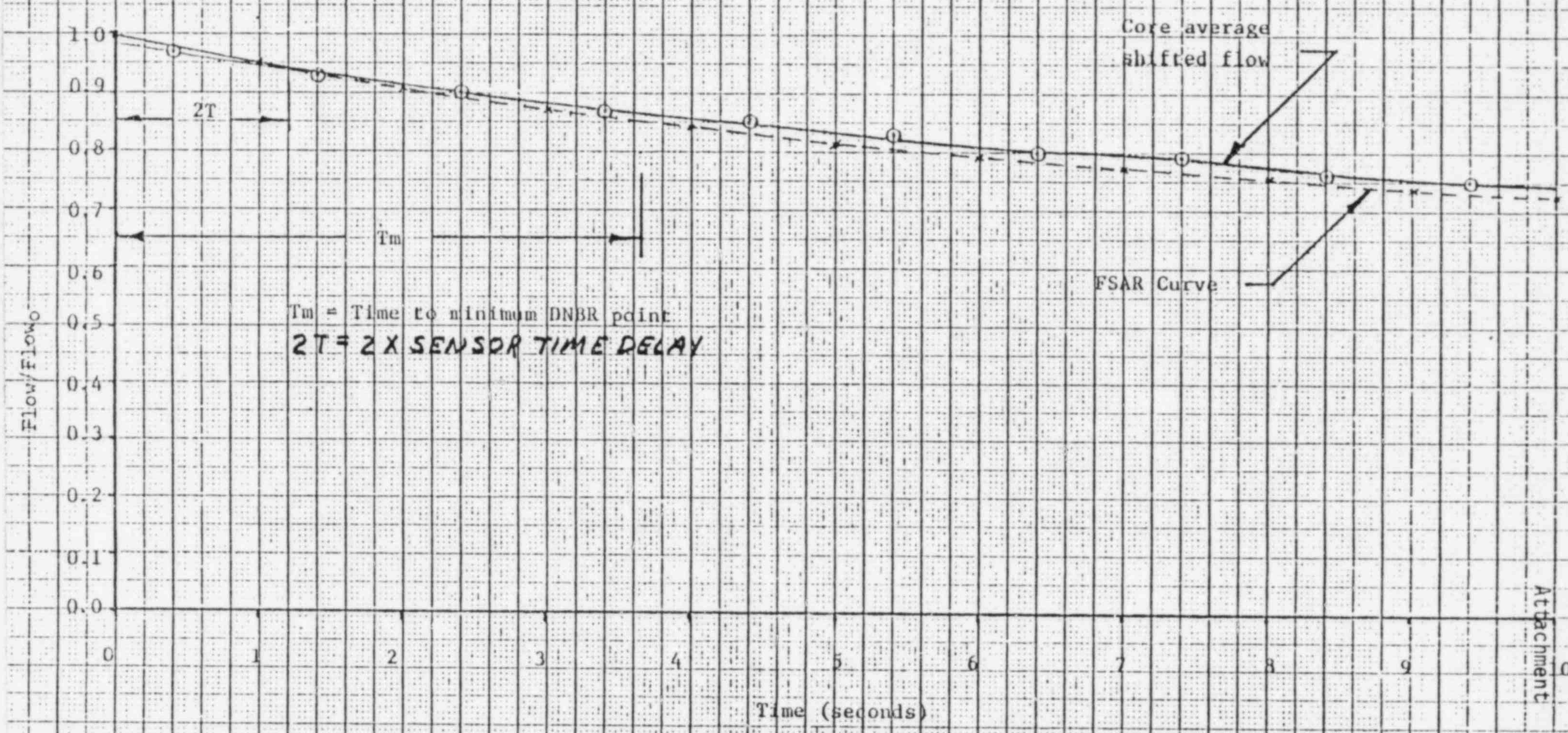
All cases examined above demonstrated that the measured core flows, through the flow coastdown transient (10 seconds), including the time identified as the point of minimum DNBR, were above the FSAR assumed core flows (Refer to Figs. 3.8-1 thru 3.8-6).

MWR #762226 was issued to trouble shoot TP-436-1 and FT-RC-436 to determine the cause of the signal failure. Since backup data for loop flow was available and traces taken later indicated sufficient flow in loop C, lack of this signal trace did not constitute a setback to core flow calculations. Also [FI-RC-414], RCS loop A flow indicator, failed to zero when all flow had ceased in the core, as indicated by remaining flow indicators in loops B & C (including second flow indicator in loop A). MWR #762227 was issued to request calibration of the flow indicator. MWR #762226 was closed on 6/8/76 while MWR #762227 is open.

REACTOR COOLANT SYSTEM FLOW COASTDOWN

TEST DATA

Low Flow Trip Time Delay (MEAS)	Low Flow Trip Time Delay (FSAR)	Under Volt Trip Time Delay Measured	Under Volt Trip Time Delay (FSAR)	Under Freq Trip Time Delay Measured	Under Freq Trip Time Delay (FSAR)	Inverse Core Flow Slope Measured	Inverse Core Flow Slope (FSAR)
(sec) 1.70	(sec) ≤ 2.42	(sec) 1.17	(sec) ≤ 1.20	(sec) 0.53	(sec) ≤ 0.60	$Flow_o/Flow.sec$ 0.0985	$Flow_o/Flow.sec$ ≤ 0.1043
All Above Obtained From Fig. 3.8-1 Case						From Fig. 3.8-2 Case	



Attachment

Figure 3.8-6
RCS flow
coastdown.
One out of twc
loop coastdowr
One loop isola

BYTLJ-7.4.1

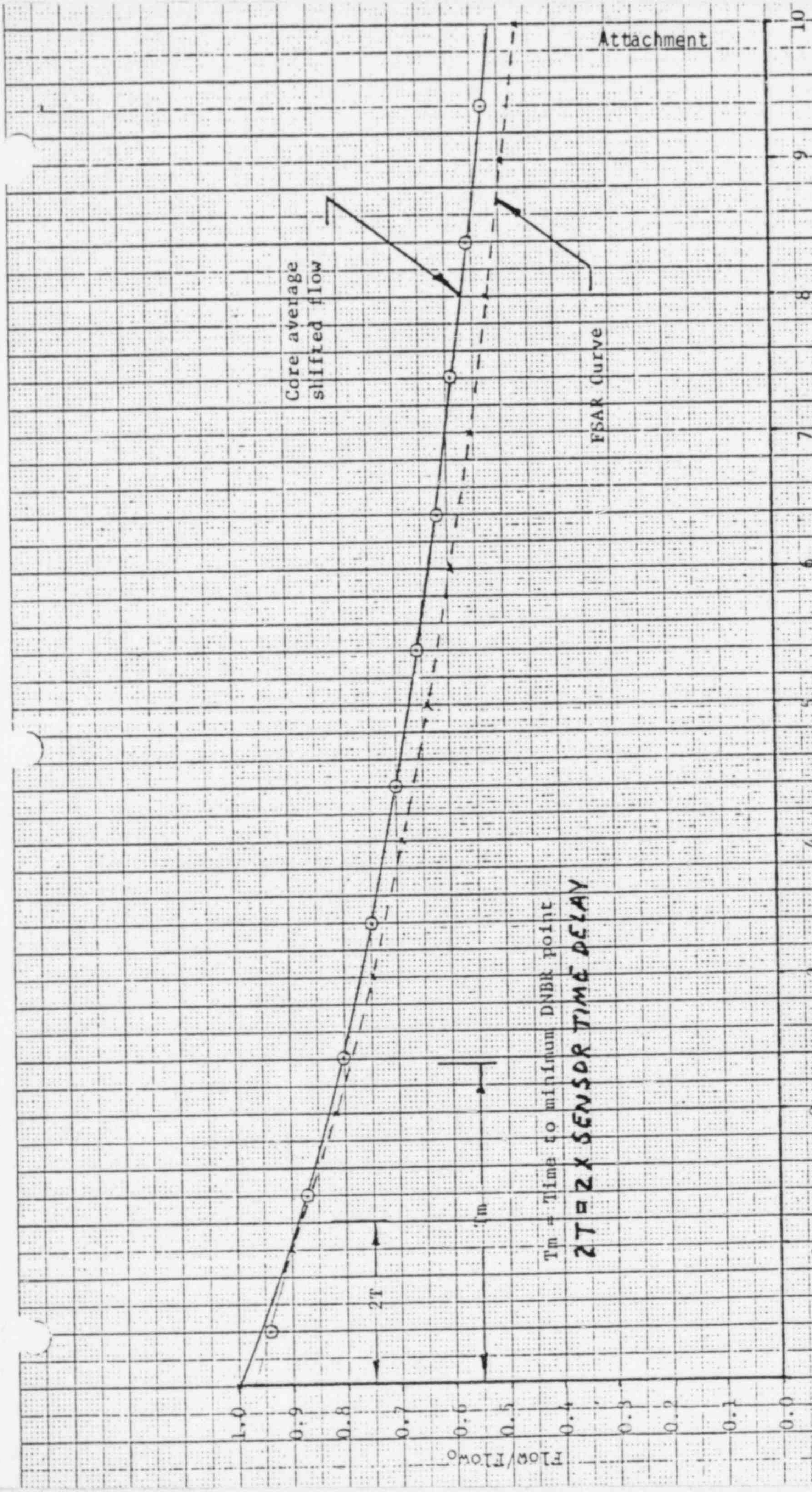


Figure 3.8-4
RCS Flow Coastdown. Two out of two loop coastdown. One loop isolated.

BVTLI-7.6.1

DATA SHEET N— 1 (Continued)

RCS Coastdown .lant Conditions

DATE 5/6/76

Plant * Parameter	2 Loops Operating 2 Loop Coastdown (No Isolated Loops)			2 Loops Operating 1 Loop Coastdown (1 Isolated Loop)			2 Loops Operating 2 Loop Coastdown (1 Isolated Loop)		
	Before	After	Initial	Before	After	Initial	Before	After	Initial
RCS Pressure [PI-RC402B]	2250 PSIG	2250 PSIG	JHS	2250 PSIG	2250 PSIG	JHS	2250 PSIG	2250 PSIG	JHS
Loop A Tavg [TI-RC412D]	540 F	538 F	JHS	530 F	530 F	JHS	546 F	546 F	JHS
Loop B Tavg [TI-RC422D]	542 F	541 F	JHS	544 F	545 F	JHS	347 F	546 F	JHS
Loop C Tavg [TI-RC432D]	541 F	540 F	JHS	543 F	542 F	JHS	546 F	545 F	JHS
Loop A Flow [FI-RC414]	25 %	0 %	JHS	0 %	10 %	JHS	10 %	10 %	JHS
Loop A Flow [FI-RC415]	20 %	0 %	JHS	0 %	0 %	JHS	0 %	0 %	JHS
Loop A Flow [FI-RC416]	20 %	0 %	JHS	0 %	0 %	JHS	0 %	0 %	JHS
Loop B Flow [FI-RC424]	106 %	0 %	JHS	105 %	110 %	JHS	105 %	0 %	JHS
Loop B Flow [FI-RC425]	106 %	0 %	JHS	105 %	110 %	JHS	105 %	0 %	JHS
Loop B Flow [FI-RC426]	106 %	0 %	JHS	105 %	110 %	JHS	105 %	0 %	JHS
Loop C Flow [FI-RC434]	105 %	0 %	JHS	104 %	0 %	JHS	104 %	0 %	JHS
Loop C Flow [FI-RC435]	105 %	0 %	JHS	104 %	0 %	JHS	103 %	0 %	JHS
Loop C Flow [FI-RC436]	105 %	0 %	JHS	105 %	0 %	JHS	104 %	0 %	JHS

APPROVED COPY

ISSUE 1, 5/2/75
REVISION 1, 7/21/75

Attachment

TVA T.T - 2/7/72

Attachment
Date 5/12/76
Recorded By [signature]

X. DATA SHEET NO. 2 (Continued) (See Figure 1 for sample calculations)

NOTE: % Flow = $\sqrt{\%\Delta P}$

LOOP AND CORE FLOW CALCULATIONS

2 Loops Operating, 1 Loop Coasting Down - 1 Isolated Loop ✓

PERCENT OF INITIAL FLOW

Time (t)	0 SEC.	2 SEC.	4 SEC.	6 SEC.	8 SEC.	10 SEC.
Loop 1A	0 %	0 %	0 %	0 %	0 %	0 %
Loop 1B	100 %	101 %	101 %	102 %	102 %	102 %
Loop 1C	100 %	85 %	73 %	63 %	55 %	47 %
Core	100 %	93 %	87 %	83 %	79 %	75 %

2 Loops Operating, 2 Loops Coasting Down - 1 Isolated Loop ✓

PERCENT OF INITIAL FLOW

Time (t)	0 SEC.	2 SEC.	4 SEC.	6 SEC.	8 SEC.	10 SEC.
Loop 1A	0 %	0 %	0 %	0 %	0 %	0 %
Loop 1B	100 %	87 %	75 %	65 %	59 %	53 %
Loop 1C	100 %	87 %	75 %	66 %	58 %	52 %
Core	100 %	87 %	75 %	66 %	59 %	53 %

APPROVED COPY

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops In Operation & Loop Stop Valves Open	LIMITS: 2 Loops In Operation & Isolated Loop Stop Valves Closed
Reactor Coolant System T_{avg}	$\leq$ 581°F	$\leq$ 570°F	$\leq$ 570°F
Pressurizer Pressure	$\geq$ 2220 psia*	$\geq$ 2220 psia*	$\geq$ 2220 psia*
Reactor Coolant System Total Flow Rate	$\geq$ 265,500 gpm	$\geq$ 189,000 gpm	$\geq$ 187,800 gpm

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

Item 2

It is not clear, from the BVPS-1 Startup Report for the Initial Start-up, that the "RTD Bypass Loop Flow Verification" test data and acceptance criteria are adequate to support N-1 operation in view of the fact that N-1 operation may reduce cold leg RTD bypass flow. Confirm that RTD response times will be acceptable for N-1 loop operation.

Response

Based on a review of the test data and Test Results Report for BVT 1.1-4.6.7, titled "RTD Bypass Loop Flow Verification", which is attached for your information (Attachment 1), the actual flow recorded in the RTD cold legs exceed the minimum required by at least 300% for each loop. This indicates that the flow would have to decrease to less than 1/3 the original value before it reached the minimum required value during N-1 loop operation. These minimum required values were calculated as part of the BVT 1.1.4.6.7 Test Results Report (Attachment 2), and were based on the RCS piping measurements and a one second transport time.

The Δp across the RCS piping is proportional to the flow rate. The total RCS flow rates are verified every 18 months per Tech. Spec. 4.2.5.2, Table 3.2-1, which gives minimum flow rate values for 3 loop, 2 loop w/stop valves open and 2 loop w/stop valves closed conditions. The limits for flow rates with stop valves closed and stop valves open for 2 loop operation are within 0.6% of each other.

In the most unlikely event of a low flow condition, a low flow alarm set at approximately 250 gpm would annunciate in the control room and would immediately alert operations personnel of the abnormal condition. The DNB related parameters for RCS Tavg, Pressurizer Pressure, and RCS flow rate are presently covered under Technical Specification 3.2.5, which requires the parameters are verified to be within limits every 12 hours for both N loop and N-1 loop conditions.

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station - Unit 1

Test Results Report

Date: 5/3/76

BVT 1.1-4.6.7 Title RTD Bypass Loop Flow Verification
Issue 1 Revision Rev.0 FR 3 JTG Approval Date 12/11/74
Test Date: Start 4/18/75 End 4/25/76 Partial Test Yes Complete Test No

Test Results: Satisfactory Yes Unsatisfactory No
Retesting Recommended No Unreviewed Safety Question
Attachments: None Involved/Evaluated No / NA

Purpose-Scope:

The purpose of this section of the test was to measure the actual flow rate in each RTD bypass loop to ensure the transport times are acceptable. The low flow alarm setpoint in the combined RTD bypass loop flow on each RCS loop was verified.

Test Summary (Conclusion):

		Flow Rate (GPM)		Transport Time (Sec)	
		Actual	Min. Req'd	Actual	Max. Allowed
1A Loop	Hot Leg	113	95	.84	1.0
	Cold Leg	175	49	.28	1.0
1B Loop	Hot Leg	113	100	.88	1.0
	Cold Leg	162	48	.30	1.0
1C Loop	Hot Leg	110	96	.87	1.0
	Cold Leg	165	49	.30	1.0

Recommendation (if any):

None

Review		O.S.C. Approval	JTG Approval	
Test Engr.	James Black		S & W	R E Adams 5/28/76
Test Supvr.	H. H. Williams	H P Williams	W (Other)	[illegible] 5/28/76
Sta. Supt.	H P Williams	Chairman	D.L.Co.	J Carey 5/28/76

Test Results Report Continuation Sheet

	Combined Hot and Cold Leg			
	Actual Total Flow (GPM)	Low Flow Alarm (GPM)	Low Flow Alarm (% of total flow)	
			Actual	Acceptance
1A Loop	288	263	91.3	90 ± 2
1B Loop	275	248	90.2	90 ± 2
1C Loop	275	246	89.5	90 ± 2

All hot and cold leg RTD bypass loops met acceptance for transport time. The low flow alarms for the combined hot and cold leg RTD bypass loops of all three RCS loops were reset to bring the setpoints within tolerance. The as left values are shown in the above tabulation.

Attachment 2
Page 1

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station - Unit 1

Test Results Report

Date: 1-27-76

BVT 1.1-4.6.7 Title RTD Bypass Loop Flow Verification
Issue 1 Revision 0 JTG Approval Date 12-11-75
Test Date: Start 4-18-75 End 1-26-76 Partial Test Yes Complete Test No

Test Results: Satisfactory Yes
Unsatisfactory No
Retesting Recommended No
Unreviewed Safety Question Involved/Evaluated No / NA
Attachments: Attachment 1
DLW-SU-5...9

Purpose-Scope:

This is a partial test, Section VII.A. The purpose of the test is to measure the length of the installed piping from reactor coolant loop connections to the last downstream RTD on the RTD manifold for both cold and hot leg RTD bypass loops. This measurement and pipe size is used to determine the flow rate required to obtain a transport time of 1.0 second.

Test Summary (Conclusion):

The length of of the various schedule piping was measured in the hot and cold bypass loops. The calculated flows required to obtain transport times of 1.0 second are listed in attachment 1.

No exact time has been specified as acceptance criteria for the bypass loop coolant transport time. The time has been increased from 0.5 seconds to 1.0 seconds. (Ref. DLW-SU-5.1.9, 6.0 - acceptance criteria). Due to the location of the piping tap off on the coolant loops, the bypass loop driving heads of the hot leg is considerably less than the cold leg thereby causing difficulty in obtaining available flow to achieve a 0.5 second transport time. The 1.0 second transport time reduces required flow by 50%.

Recommendations (if any):

None

Review

Test Engr. B. E. [illegible] for B. Cherry

Test Supvr. D. H. Williams

Sta. Supt. J. A. Werling

O.S.C. Approval

J. A. Werling
Chairman

JTG Approval

S & W [illegible] 2/12/76

W (Other) [illegible] 2/12/76

D.L.Co. [illegible] 2/12/76

X. DATA SHEET NO. 1

DATE: 1/27/76
INITIALS: JBB

A. Hot leg RTD bypass flow rate necessary to achieve a ~~0.5~~ 1.0 second | JBB

transport time.

Loop	RTD	Measured total 1" pipe length (ft) (3 paths) L (1)	Measured total 2" pipe length (ft) L (2)	Calculated total volume (ft)^3 VHL	Calcula require flow ra (gpm) FH
1A	TRB-411B, 412B, 411D	123½"	134"	.211	95
1B	TRB-421B, 422B, 421D	137"	139"	.222	100
1C	TRB-431B, 432B, 431D	121"	136½"	.213	96

| JBB

B. Cold leg RTD bypass flow rate necessary to achieve a ~~0.5~~ 1.0 second transp

time.

Loop	RTD	Measured total 1.5" pipe length (ft) L(1.5)	Measured total 2" pipe length (ft) L(2)	Calculated total vol. (ft^3) VCL	Calcul requir flow r gpm FC
1A	TRB-412C, 412D, 411C	48"	54"	.109	49
1B	TRB-422C, 422D, 421C	46½"	53"	.107	48
1C	TRB-432C, 432D, 431C	47"	55"	.110	49

APPROVED COPY

XI. APPENDIX 2

A. Calculated Transport Time

To calculate the flow rate necessary to achieve a ~~0.5~~ 1.0 second transport time, utilize the formula F= V x D/T where: JBB

F = Flow rate in hot or cold leg bypass loop

V = Total volume of pipe (hot or cold leg) from the Reactor Coolant System loop pipe connection to the last downstream RTD.

D = Volumetric conversion factor

T = Transport time

To determine the volume of the pipe, multiply the length of each pipe (1", 1 1/2", or 2") by the cross sectional area of each pipe. The cross-sectional areas for the pipes are as follows:

Area 1" schedule 160 pipe - 0.00362 square feet

Area 1 1/2" schedule 160 pipe - 0.00976 square feet

Area 2" schedule 160 pipe - 0.01556 square feet

Let L (1), L (1.5) and L (2) be the lengths of 1", 1.5" and 2" pipe respectively in feet.

For the cold leg bypass loop, there is only 1 1/2" and 2" piping.

V Cold Leg = VCL = L (1.5) x (.00976) + L (2) x (.01556)

D = 7.48 gallons/cubic foot

T = ~~0.00833~~ .01667 minutes, the required transport time

Then - $FCL = \dfrac{VCL \times 7.48}{\cancel{0.00833}\ .01667}$ gpm

JBB

The calculation of the required flow for the hot leg bypass is done in the same manner except that there is only 1" and 2" piping in the hot leg bypass loop.

XI. APPENDIX 2 (Continued)

B. Actual Transport Time

To determine the actual transport time, transpose the equation: $T = V \times D/F$ where F is now the actual flow rate as calculated above. ~~flow that is measured during the test.~~ This will give the actual transport time which can be compared to the required transport time.

JBB

Calculate the actual cold leg and hot leg RTD bypass loop flow rates by using the following equations:

$$F_h' = \frac{F_t}{(1 + F_c/F_h)}$$

$$F_c' = F_t - F_h'$$

where:

F_t - combined hot and cold leg RTD bypass loop flow rate.

F_c - measured cold leg RTD bypass loop flow rate.

F_h - measured hot leg RTD bypass loop flow rate.

F_c' - actual cold leg RTD bypass loop flow rate.

F_h' - actual hot leg RTD bypass loop flow rate.

Record these flows on Data Sheet # 2.

GES

APPROVED COPY

ENCLOSURE III

N-1 Loop LOCA Reanalysis

Attached is the results of the LOCA reanalysis performed for Beaver Valley Unit I using the 1981 Westinghouse Evaluation Model.

The Large Break N-1 LOCA ECCS analysis was performed at a power level of 1724 MWt. Other pertinent analysis assumptions include 17X17 standard fuel design. Also, the accumulator water volume remained 1025 cubic feet per accumulator. The analysis was performed with the NRC approved 1981 Westinghouse Evaluation Model as described in WCAP-9220-P-A Rev. 1.

The results of this analysis are presented in two attachments. Attachment A contains the analysis results, including tables and figures. Attachment B contains LOCA related technical specification values.

The CD = 0.4, 0.6 active loop break and CD = 0.4 inactive loop break sizes were analyzed in this study. The worst break size is the CD = 0.4 DECLG active loop break, and resulted in a peak clad temperature (PCT) of 1882°F at a total peaking factor (FQT) of 3.03. This analysis demonstrates conformance with the 10CFR50.46 requirements for Large Break ECCS LOCA Analyses.

MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS OF COOLANT ACCIDENT) WITH ONE COOLANT LOOP OUT OF SERVICE

An analysis specified by 10CFR50.46[1], "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors", for 2-loop operation of the Beaver Valley Station is presented in this section. The results of the loss of coolant accident analyses are shown in Table 15.4-2 and show compliance with the Acceptance Criteria. The analytical techniques used are in compliance with Appendix K of 10CFR50, and are described in Reference [2].

The boundary considered for loss of coolant accidents as related to connecting piping is defined in Section 3.6.

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 countermeasure will limit the consequences of the accident in two ways:

1. 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.

2. Injection of borated water provides heat transfer from the core and prevents excessive clad temperatures.

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, consistant with Appendix K of 10CFR50. Thereafter, the core heat tranfer is based on local conditions with transition boiling and forced convection to steam as the major heat tranfer mechanisms. During the refill period rod-to-rod radiation is the only heat transfer 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 injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appendix K of 10CFR50.

15.4.1.1 Thermal Analysis

15.4.1.1.1 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 (ECCS) are designed so that the reactor can be safely shutdown and the essential heat transfer geometry of the core preserved following the accident. Emergency safeguards systems present at the Beaver Valley station will be available during 2-loop operation.

The ECCS, even when operating during the injection mode with the most severe single active failure loss of a low-head SI pump, is designed to meet the Acceptance Criteria[1].

15.4.1.1.2 Method of Thermal Analysis

The description of the various aspects of the loss of coolant accident analysis is given in Reference [2]. This document describes the major phenomena modeled, the interfaces among 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] through [6]. The analyses presented were performed using the 1981 version of the Westinghouse Evaluation Model. This version includes the modifications to the models referenced above as specified by tne Nuclear Regulatory Commission (NRC) in Reference [7] and complies with Appendix K of 10CFR50. The 1981 Westinghouse Evaluation Model is documented in References [8], [11] and [12]. Containment data used to calculate ECCS backpressure is presented in Table 15.4-3.

The methods used to model ECCS performance for (N-1) loop operation are described in Reference [10]. Two distinct reactor coolant pipe cold leg break locations are possible during operation with a loop out of service; the break may occur either in an active loop or in the inactive loop. The SATAN nodalization scheme described in Reference [3] has been expanded in order to include the portion of the reactor coolant pipes in the isolated loop between the reactor vessel and the loop isolation valves in the blowdown calculation.

In the active loop break case scheme, element 52 is the inactive loop hot leg pipe length, element 54 is the cold leg pipe length, and element 53 is the inactive loop accumulator which feeds element 54, with the break location unchanged. For an inactive loop break as described in Reference [13], element 52 remains the hot leg, while elements 54 and 55 constitute the vessel side of the broken pipe. Element 56 represents the valved off pipe segment side of the break.

The WREFLOOD Code 19 element model is presented in Reference [5]. Figure 4.1 remains unchanged for active loop break analyses. To model the inactive loop break location a 14-element loop model was devised and reported in Reference [10].

The ECCS calculations were performed based on a core power peaking factor envelope calculated for 2-loop operation of Beaver Valley Unit 1. A design F_Q of 2.32 for N-loop operation results in a F_Q of 3.03 for N-1 loop operation. The normalized power versus core height [K(Z)] curve for N-1 loop operation is presented in Figure 15.4-17.

Figures 15.4-1 through 15.4-16 present the transients for the principal parameters for the break sizes analyzed. The following items are noted:

Figures 15.4-1a through 15.4-3c — The following quantities are presented at the clad burst location and at the hot spot (location of maximum clad temperature) both on the hottest fuel rod (hot rod):

(1) fluid quality
(2) mass velocity
(3) heat transfer coefficient.

The heat transfsr coefficient shown is calculated by the LOCTA-IV code.

Figures 15.4-4a through 15.4-6c	The system pressure shown is the calculated pressure in the core. The flow rate out the break is plotted as the sum of both ends for the guillotine break cases. This core pressure drop shown is from the lower plenum, near the core, to the upper plenum at the core outlet.
Figures 15.4-7a through 15.4-9c	These figures show the hot spot clad temperature transient and the clad temperature transient at the burst location. The fluid temperature shown is also for the hot spot and burst location. The core flow (top and bottom) is also shown.
Figures 15.4-10a through 15.4-10f	These figures show the core reflood transient.
Figures 15.4-11a through 15.4-12c	These figures show the Emergency Core Cooling System flow for all cases analyzed. As described earlier, the accumulator delivery during blowdown is discarded until the end of bypass is calculated. Accumulator flow, however, is established in refill reflood calculations. The accumulator flow assumed is the sum of that injected in the intact cold legs.
Figures 15.4-13a, b, c	The containment pressure transient is also provided.
Figures 15.4-14a, b, c	These figures show the core power transient.
Figure 15.4-15	This figure shows the break energy released to the containment during blowdown for the limiting case break.
Figure 15.4-16	This figure provides the containment wall condensing heat transfer coefficient for the limiting case break.

Figure 15.4-17 This figure provides the operating power peaking factor envelope. In addition to the above, Tables 15.4-4 and 15.4-5 present the reflood mass and energy release to the containment and the broken loop accumulator mass and energy flowrate to the contanment, respectively.

The clad temperature analysis is based on a total peaking factor of 3.03. The hot spot metal water reaction reached is 3.0 percent, which is well below the embrittlement limit of 17 percent, as required by 10CFR50.46. In addition, the total core metal-water reaction is less than 0.3 percent for all breaks as compared with the 1 percent criterion of 10CFR50.46.

Conclusions - Thermal Analysis

For breaks up to and including the doubled-ended severance of a reactor coolant pipe during 2-loop operation, 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 provides margin to the requirement of 2200°F, based on an F_q value of 3.03.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent 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 clad oxidation limits of 17 percent are not exceeded during or after quenching.

4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

15.4.1.1.3 Results

The sequence of events for each case analyzed is shown in Table 15.4-1.

Table 15.4-2 presents the peak clad temperatures and hot spot metal reaction for a range of break sizes and locations. This range of break sizes was determined to include the limiting case for peak clad temperature from sensitivity studies reported in References [9] and [10].

The SATAN-VI analysis of the loss of coolant accident is performed at 102 percent of the (N-1) License Power Rating. The peak linear power and core power used in the analyses are given in Table 15.4-2. Since there is margin between the value of the peak linear power density used in this analysis and the value expected in 2-loop operation, a lower peak clad temperature would be obtained by using the peak linear power density expected during operation.

For the results discussed below, the hot spot is defined to be the location of maximum peak clad temperature. This location is given in Table 15.4-2 for each break size analyzed.

15.4.7 References

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

2. Bordelon, F. M., Massie, H. W. and Zordon, T. A., "Westinghouse ECCS Evaluation Model - Summary," WCAP-8339, July 1974.

3. Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Spacetime Dependent Analysis of Loss of Coolant," WCAP-8302, June 1974 (Proprietary) and WCAP-8306, June 1974 (Non-Proprietary).

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

5. Kelly, R. D., et al., "Calculation Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)," WCAP-8170, June 1974 (Proprietary) and WCAP-8771, June 1974 (Non-Proprietary).

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

7. "Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10CFR50 Appendix K," Federal Register, November 1974.

8. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Supplementary Information," WCAP-8471, April 1975 (Proprietary) and WCAP-8472, April 1975 (Non-Proprietary).

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

10. Kemper, R. N., "Westinghouse Emergency Core Cooling System Evaluation Model for Analyzing (N-1) Loop Operation of Plants With Loop Isolation Valves," WCAP-8904, December 1976.

11. Eicheldinger, C., "Westinghouse ECCS Evaluation Model, 1981 Version," WCAP-9220-P-A (Proprietary), WCAP-9221-P-A (Non-Proprietary), Revision 1.

12. Rahe, E. P. (Westinghouse). Letter dated November 8, 1982 to James R. Miller (USNRC), letter number NS-EPRS-2679.

13. Eicheldinger, C. (Westinghouse). Letter dated September 7, 1977 to J. F. Stolz (USNRC), letter number NS-CE-1540.

TABLE 15.4-1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	Inactive Loop Break, C_D=0.4 DECLG (Sec)	Active Loop Break, C_D=0.6 DECLG (Sec)	Active Loop Break, C_D=0.4 DECLG (Sec)
START	0.0	0.0	0.0
Rx Trip Signal	0.48	0.46	0.46
S. I. Signal	4.41	2.11	2.66
Acc. Injection	23.60	9.58	12.40
End of Blowdown	41.07	23.25	26.33
Bottom of Core Recovery	56.55	36.57	39.11
Acc. Empty	61.96	46.43	51.03
Pump Injection	29.41	27.11	27.66
End of Bypass	41.07	23.25	26.27

TABLE 15.4-2

LARGE BREAK

	Inactive Loop Break, C_D=0.4 DECLG	Active Loop Break, C_D=0.6 DECLG	Active Loop Break, C_D=0.4 DECLG
Results			
Peak Clad Temp., °F	1882	1775	1882
Local Zr/H_2O Rxn, (max)%	2.7	2.3	3.0
Peak Clad Location, ft	9.0	7.25	9.0
Local Zr/H_2O Location, ft	9.0	7.25	7.0
Total Zr/H_2O Rxn, %	<0.3	<0.3	<0.3
Hot Rod Burst Time, sec	118.5	73.6	53.8
Hot Rod Burst Location, ft	6.5	6.25	6.0

Calculation

NSSS Power Mwt 102% of	1724
Peak Linear Power kw/ft 102% of	10.25
Peaking Factor (At License Rating)	3.03
Accumulator Water Volume, ft^3	1025

TABLE 15.4-3

CONTAINMENT DATA
(DRY CONTANMENT)

Free Volume	$1.89 \times 10^6 ft^3$
Initial Conditions	
Pressure	9.5 psia
Temperature	90°F
RWST Temperature	40°F
Service Water Temperature	35°F
Outside Temperature	35°F
Quench Spray System	
Number of Pumps Operating	2
Runout Flow Rate (each)	2200 gpm
Actuation Time	55 sec
Recirculation Spray System	
Number of Pumps Operating	4
Runout Flow Rate (each)	3300 gpm
Actuation Time	300 sec

STRUCTURAL HEAT SINKS[1]

Wall Number	Material	Thickness (ft)	Surface Area (sq ft)
1	Concrete	0.5	6,972
2	Concrete	1.0	77,446
3	Concrete	1.5	36,848
4	Concrete	2.0	17,010
5	Concrete	3.0	8,632
6	Carbon Steel	0.03125	18,270
	Concrete	4.5	

TABLE 15.4-3 (Continued)

CONTAINMENT DATA

STRUCTURAL HEAT SINKS[1]

Wall Number	Material	Thickness (ft)	Surface Area (sq ft)
7	Carbon Steel	0.03125	32,445
	Concrete	4.5	
8	Carbon Steel	0.04167	26,250
	Concrete	2.5	
9	Concrete	2.0	13,125
	Carbon Steel	0.03125	
	Concrete	10.0	
10	Stainless Steel	0.06875	3,270
11	Carbon Steel	0.02202	10,750
12	Carbon Steel	0.06242	748
13	Carbon Steel	0.1932	2,132
14	Carbon Steel	0.1833	5,479
15	Carbon Steel	0.0893	3,770
16	Carbon Steel	0.1041	10,938
17	Carbon Steel	1.020	600
18	Carbon Steel	0.0119	118,091
19[2]	Stainless Steel	0.0833	2,932.5

(1) All walls are painted with the exception of Walls 9 and 10. The thickness of paint is 5.0 mils for all painted walls with the exception of Wall 11, which has a paint thickness of 3.75 mils.

(2) Inactive Loop Pump Metal.

TABLE 15.4-4

MASS AND ENERGY RELEASE FOR LIMITING
BREAK REFLOOD TRANSIENT (ACTIVE
LOOP BREAK, C_D=0.4 DECLG)

Time (sec)	Break Mass Flow (lbm/sec)	Break Energy Flow (10^4 BTU/sec)
39.1	0.0	0.0
39.7	.0242	.00311
40.0	.0243	.00313
40.7	.0245	.00316
41.9	21.98	2.8353
51.6	204.50	8.6411
69.2	305.46	9.9087
91.5	317.78	9.7194
116.7	325.22	9.4086
173.9	337.12	8.7204
241.3	348.95	7.9802
326.4	354.77	7.2195

Accumulator nitrogen was released from the accumulators between 58.5 and 78.5 seconds at a flow rate of 181.6 lbm/sec.

TABLE 15.4-5

BROKEN LOOP ACCUMULATOR FLOW RATE TO
CONTAINMENT FOR THE LIMITING BREAK
(ACTIVE LOOP BREAK, C_D=0.4 DECLG)

Time (sec)	Mass Flow Rate (lbm/sec)
0.000	4121.948
1.010	3716.991
2.010	3417.334
3.010	3181.414
4.010	2989.343
5.010	2828.876
6.010	2691.456
7.010	2572.130
8.010	2466.572
9.010	2372.179
10.010	2287.153
11.010	2210.235
12.010	2140.266
13.010	2076.303
14.010	2017.505
15.010	1963.205
16.010	1912.901
17.010	1866.001
18.010	1822.347
19.010	1781.747
20.010	1744.203
21.010	1709.189
22.010	1676.418
23.010	1645.808
24.010	1617.065
25.010	1589.752
29.299	0.0

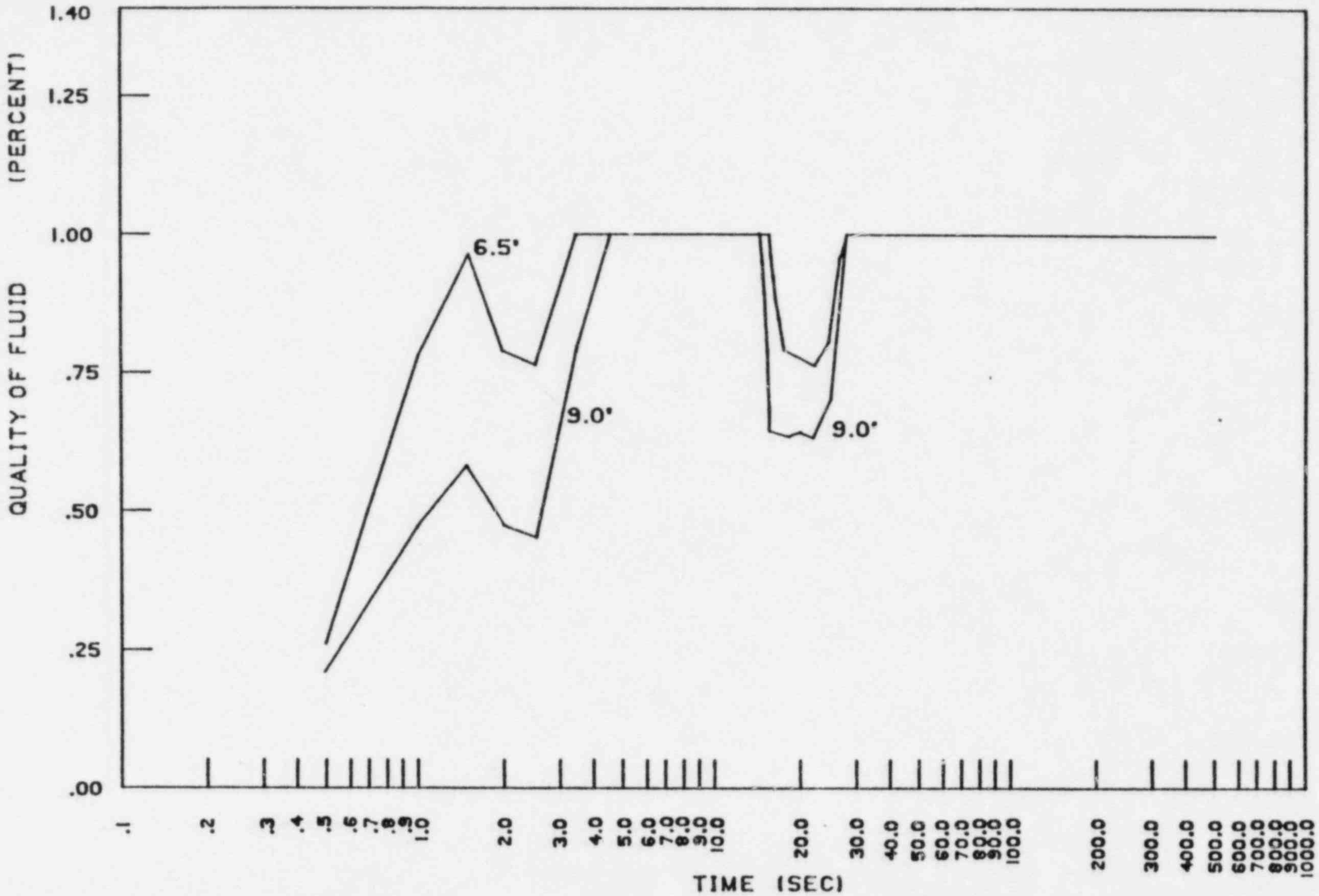


Figure 15.4-1A. Fluid Quality - DECLG (C_D = 0.4)
Inactive Loop Break

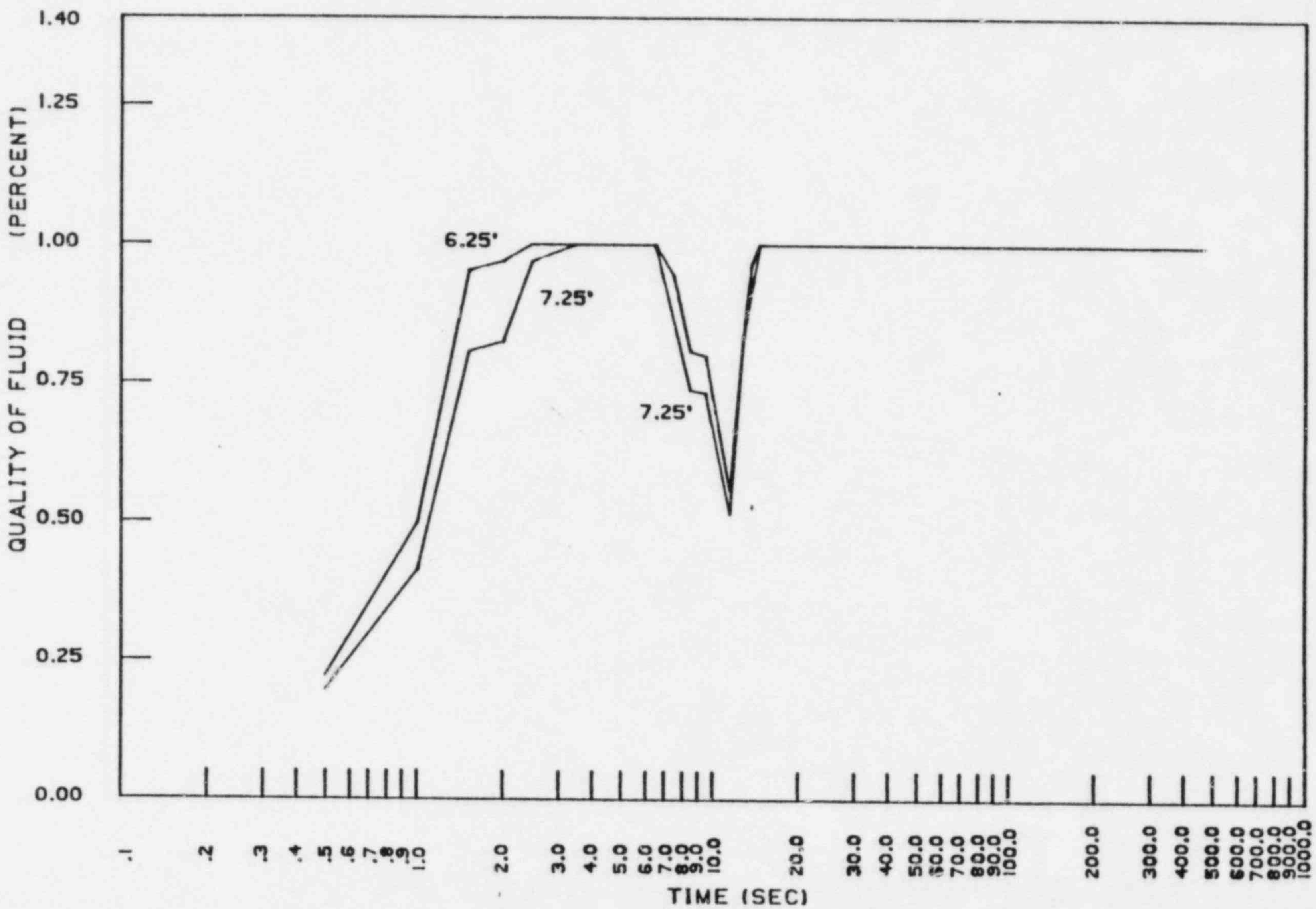


Figure 15.4-1B. Fluid Quality- DECLG (C_D=0.6)
Active Loop Break

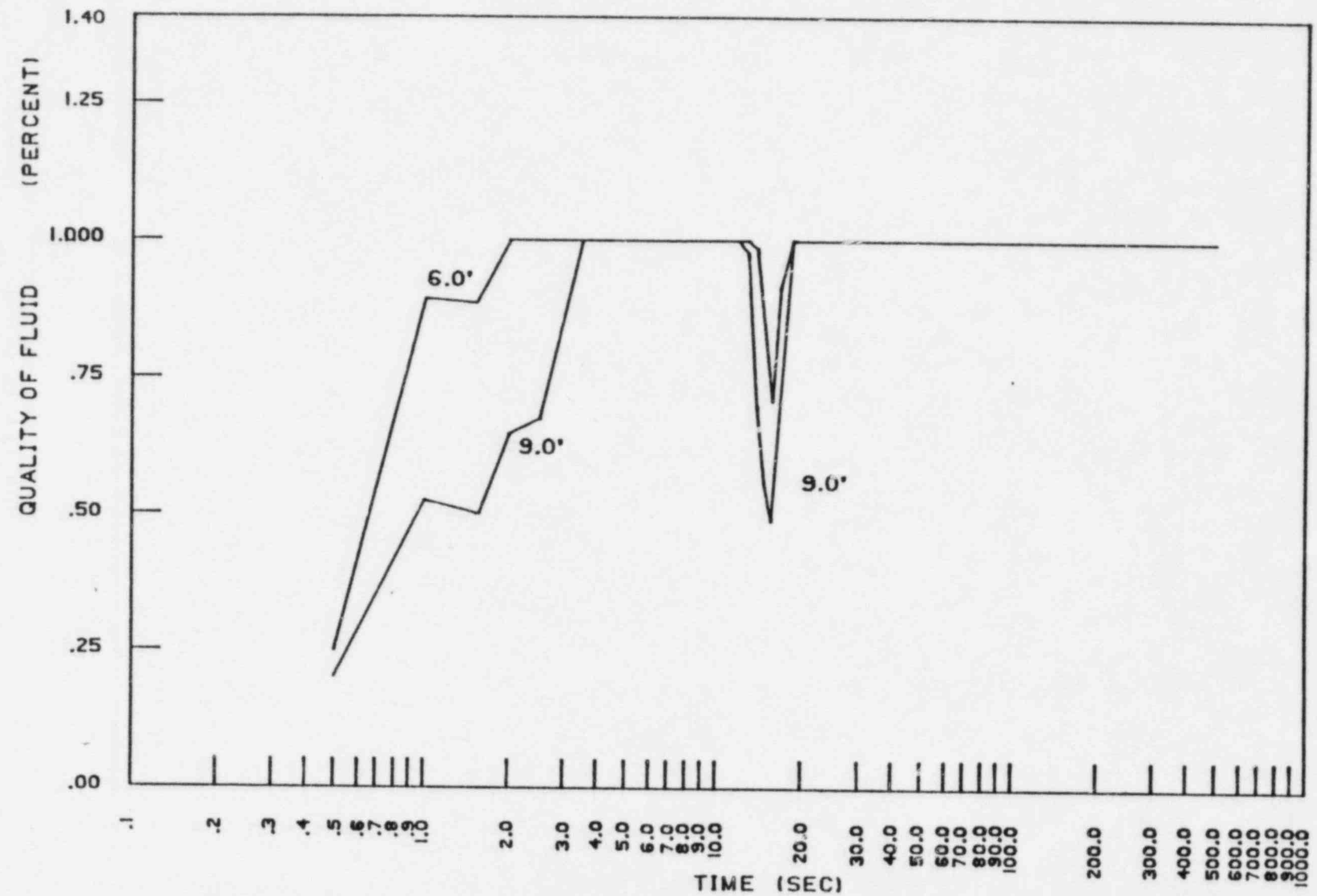


Figure 15.4-1C. Fluid Quality - DECLG (C_D =0.4)
Active Loop Break

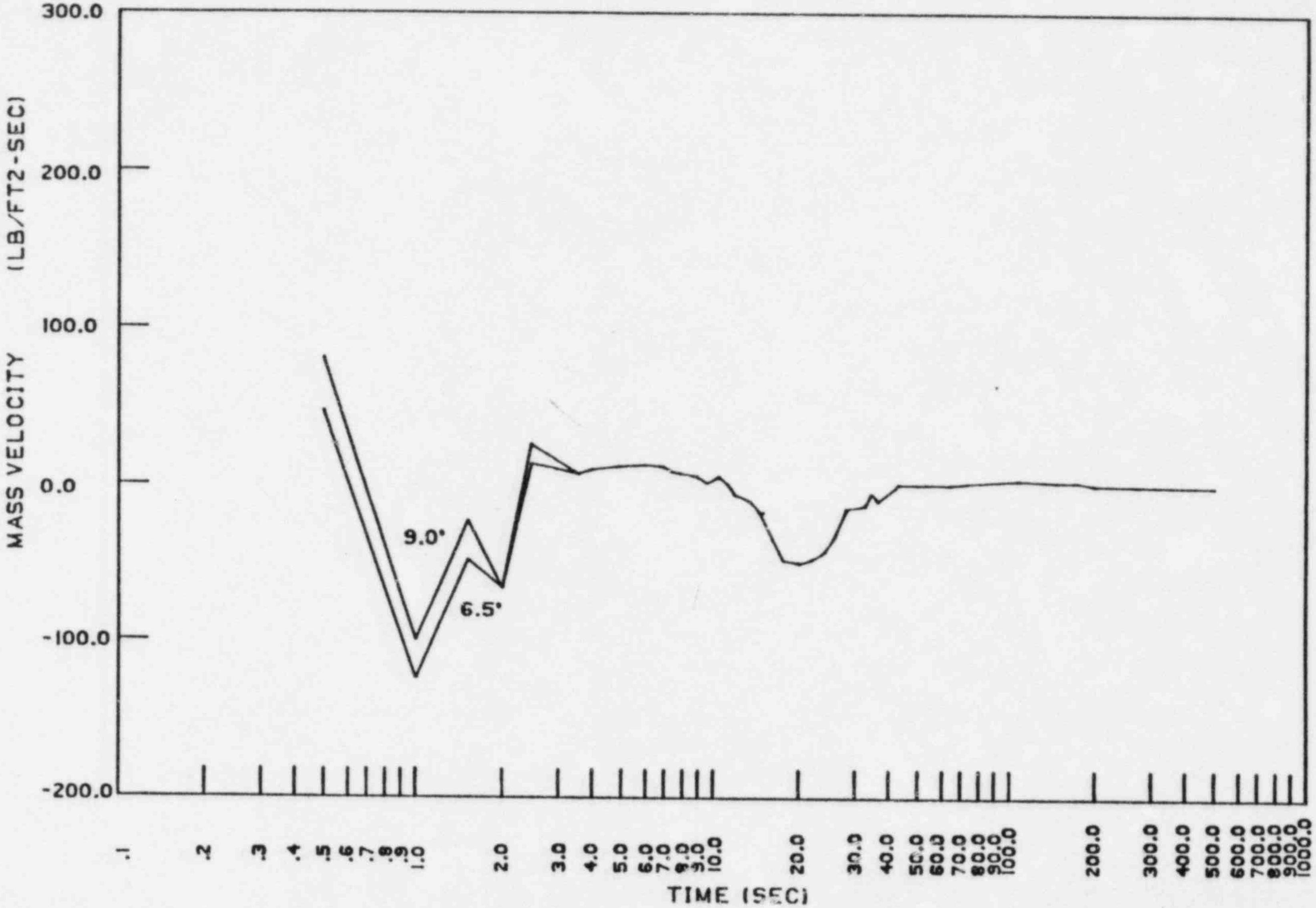


Figure 15.4-2A. Mass Velocity-DECLG (C_D=0.4) inactive Loop Break

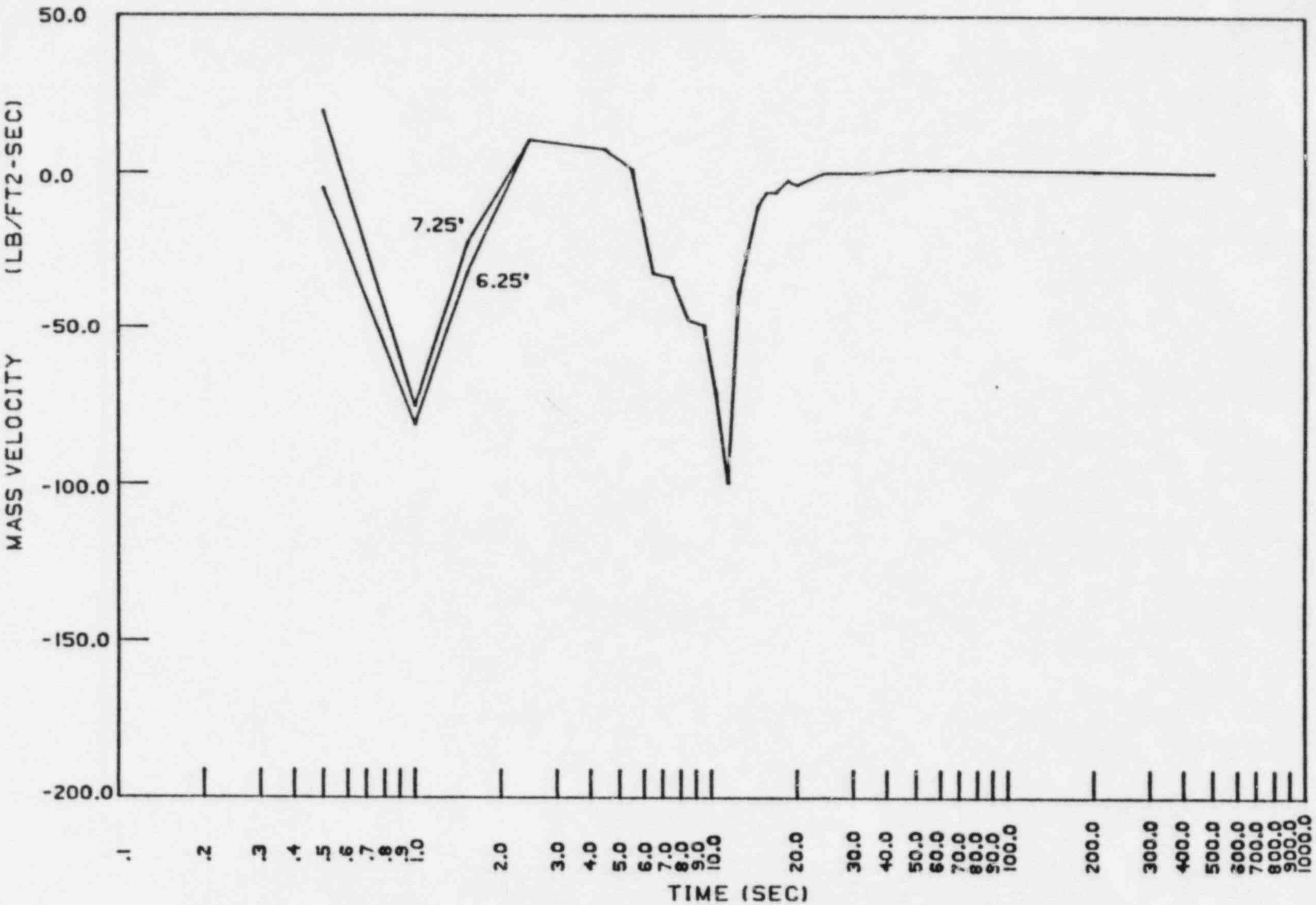


Figure 15.4-2B. Mass Velocity-DECLG (C_D=0.6)
Active Loop Break

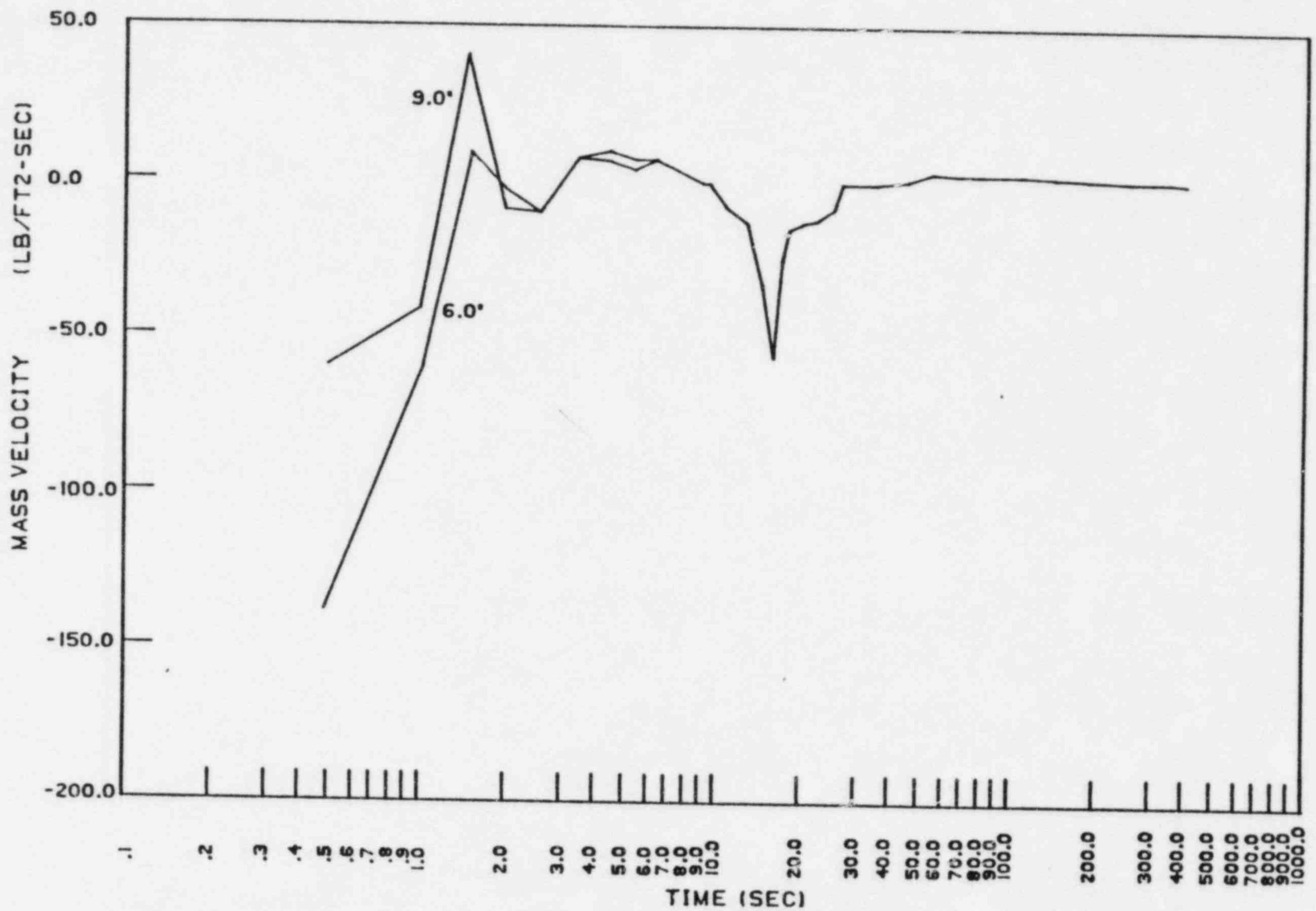


Figure 15.4-2C. Mass Velocity-DECLG (C_D=0.4)
Active Loop Break

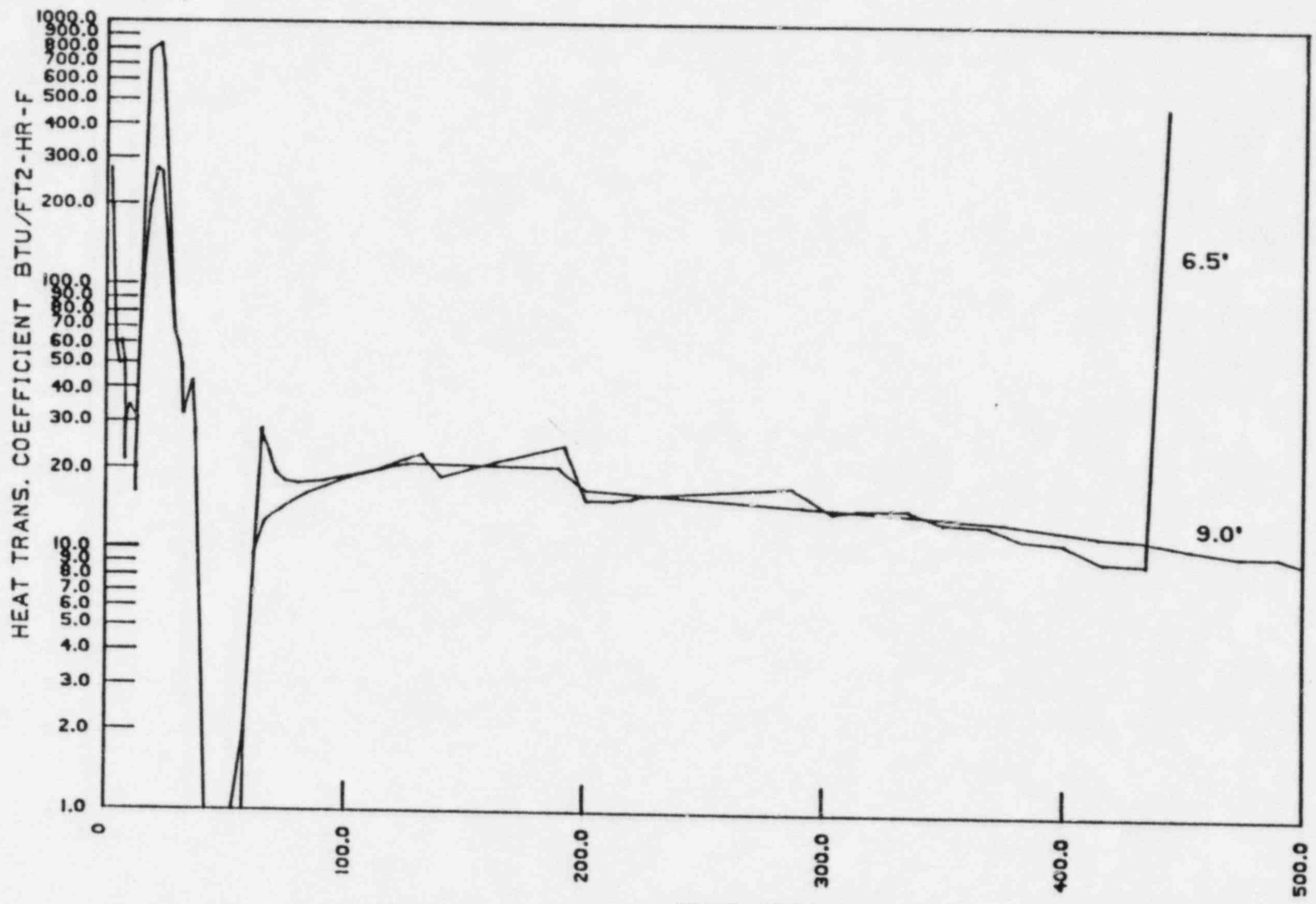


Figure 15.4-3A. Heat Transfer Coefficient - DECLG (C_D =0.4) Inactive Loop Break

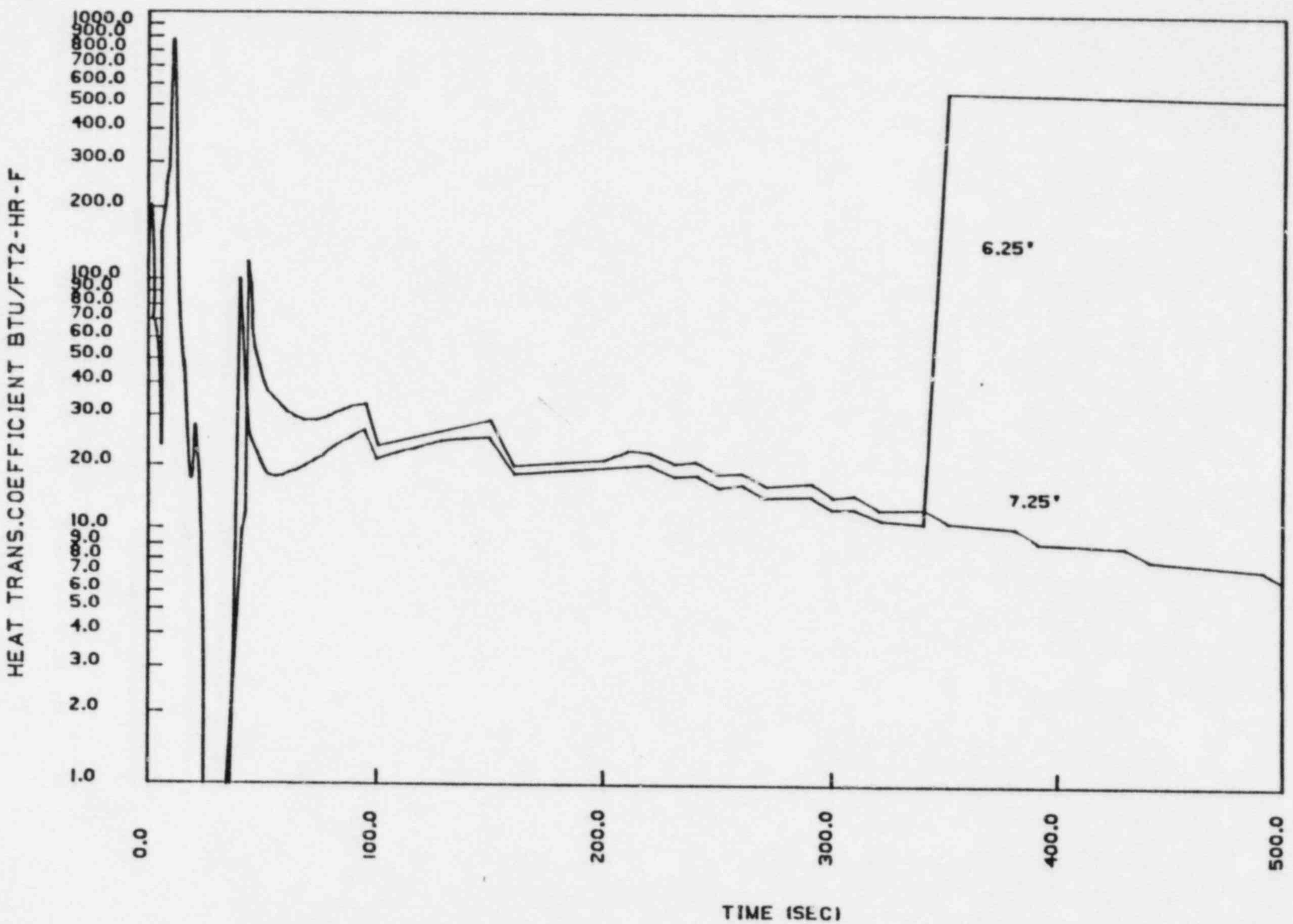


Figure 15.4-3B. Heat Transfer Coefficient - DECLG (C_D =0.6) Active Loop Break.

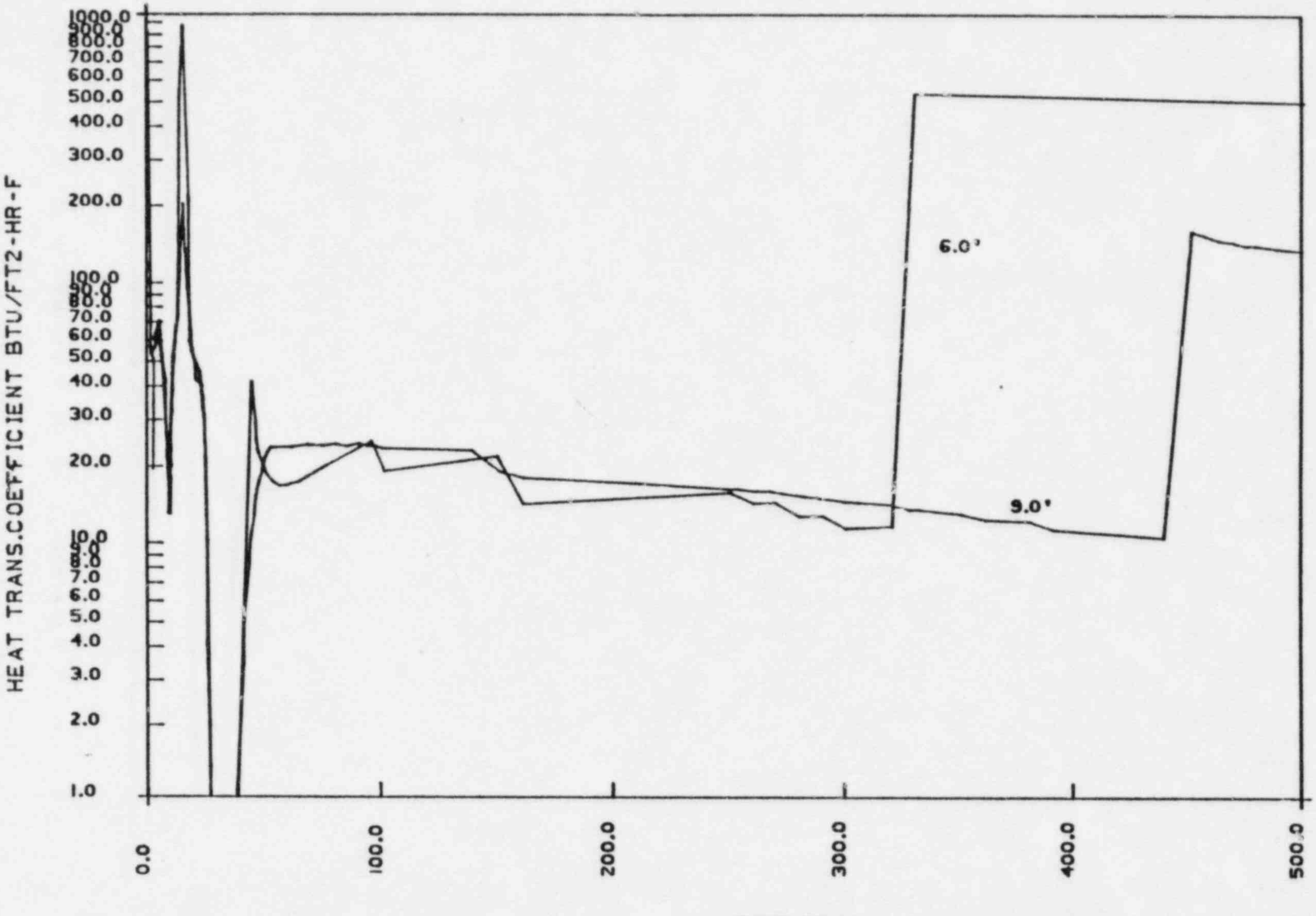


Figure 15.4-3C. Heat Transfer Coefficient - DECLG (C_D =0.4) Active Loop Break.

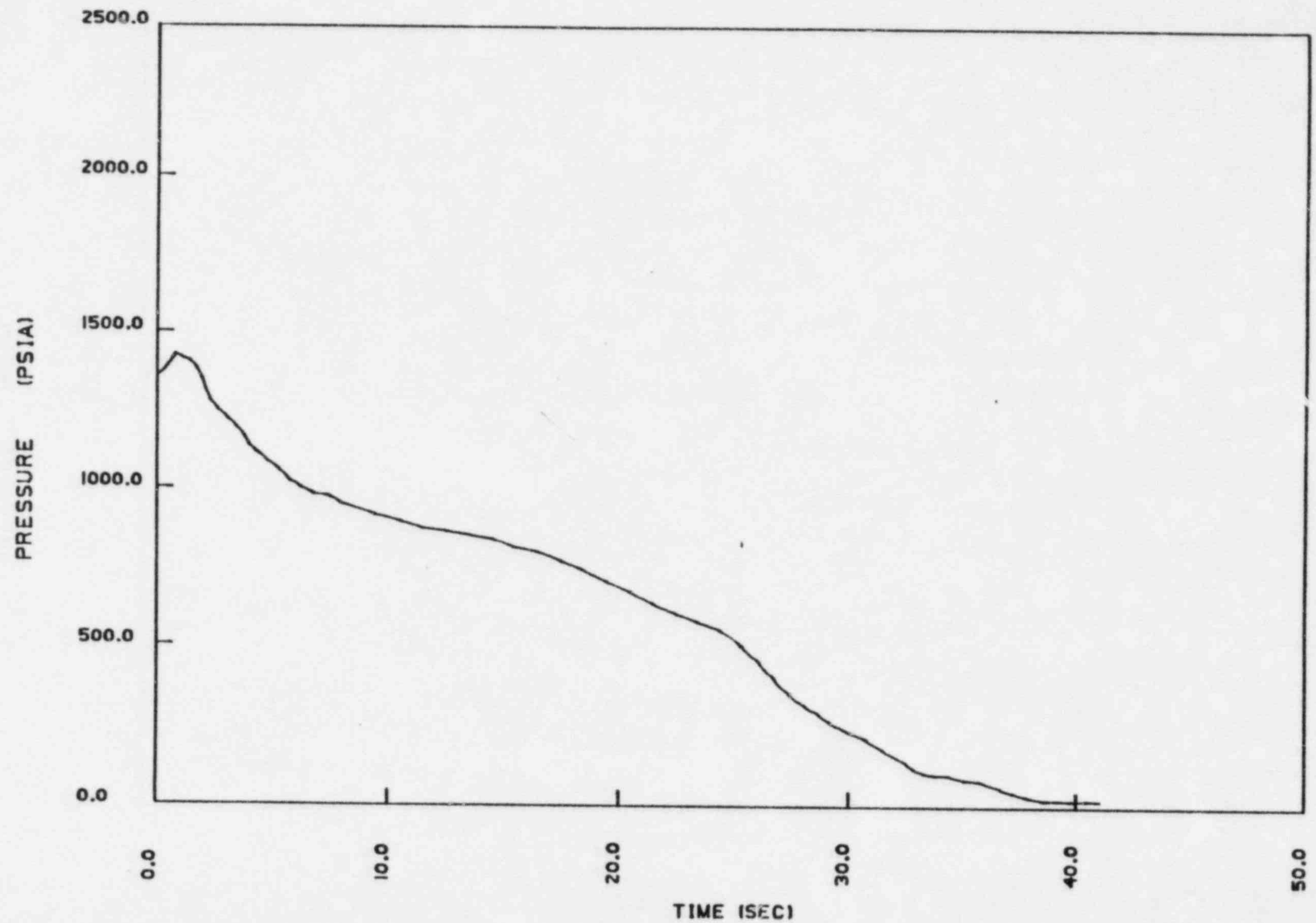


Figure 15.4-4A. Core Pressure - DECLG (C_D =0.4) Inactive Loop Break.

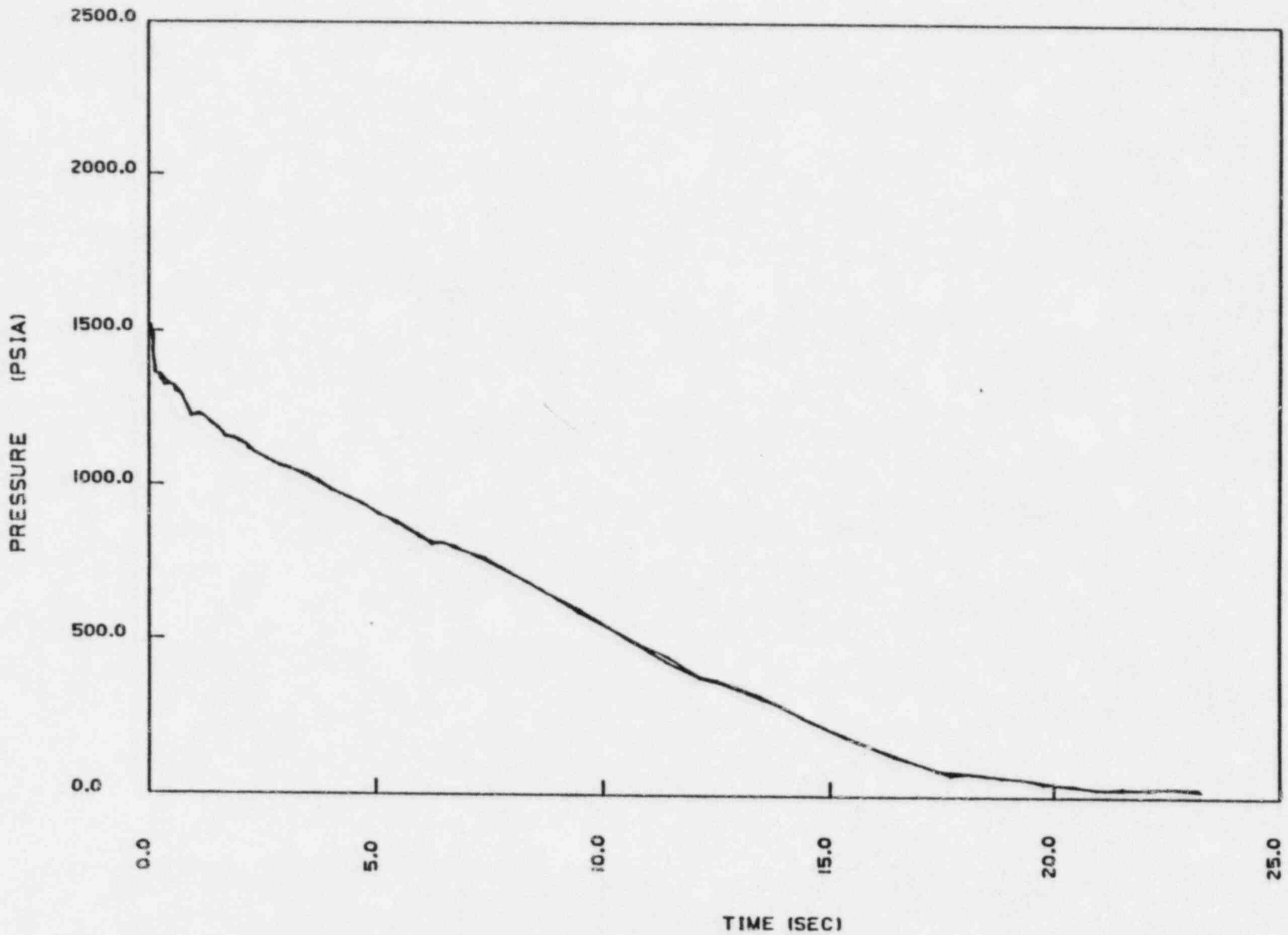


Figure 15.4-4B. Core Pressure - DECLG (C_D =0.6)
Active Loop Break.

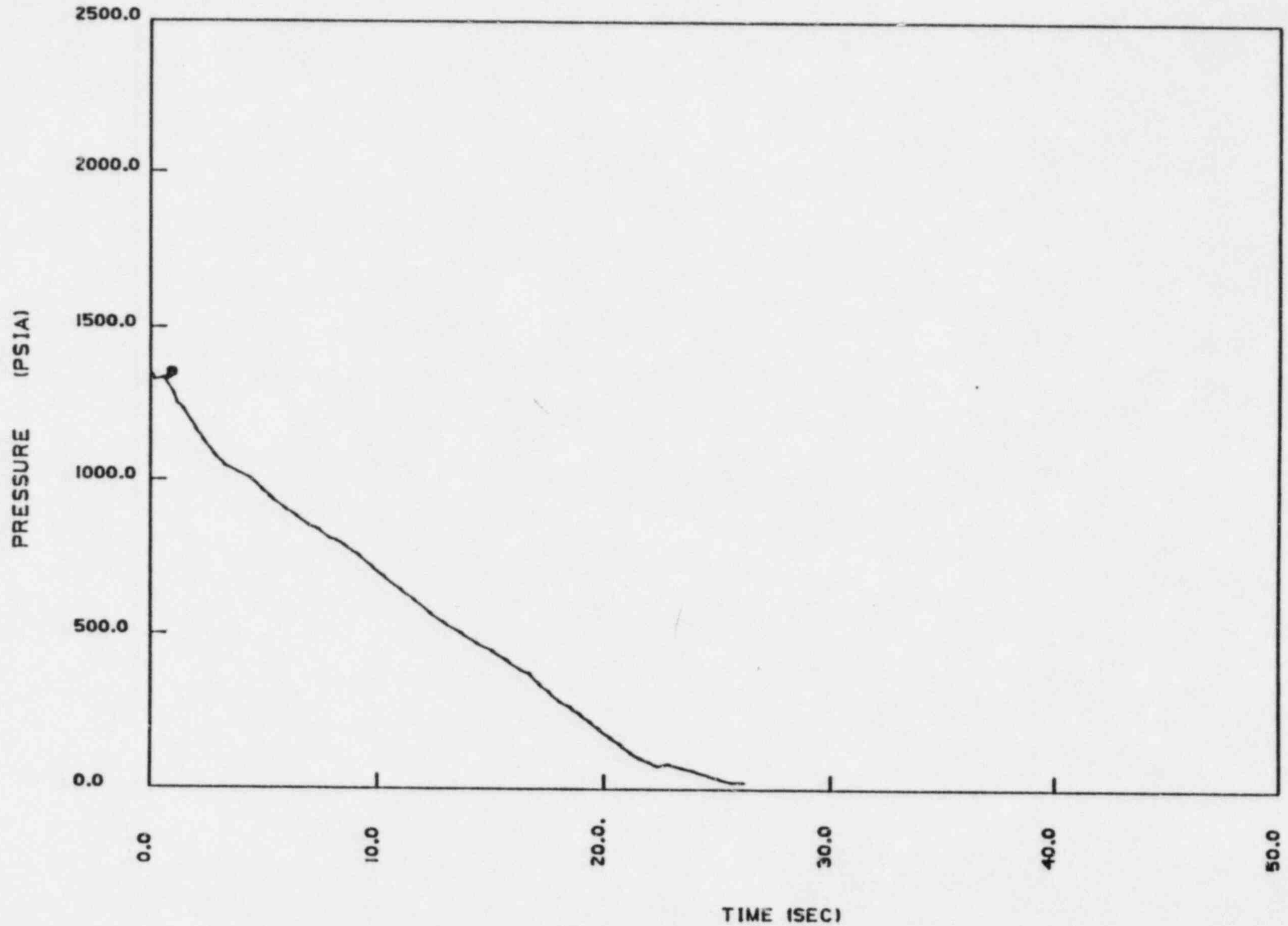


Figure 15.4-4C. Core Pressure - DECLG (C_D =0.4)
Active Loop Break.

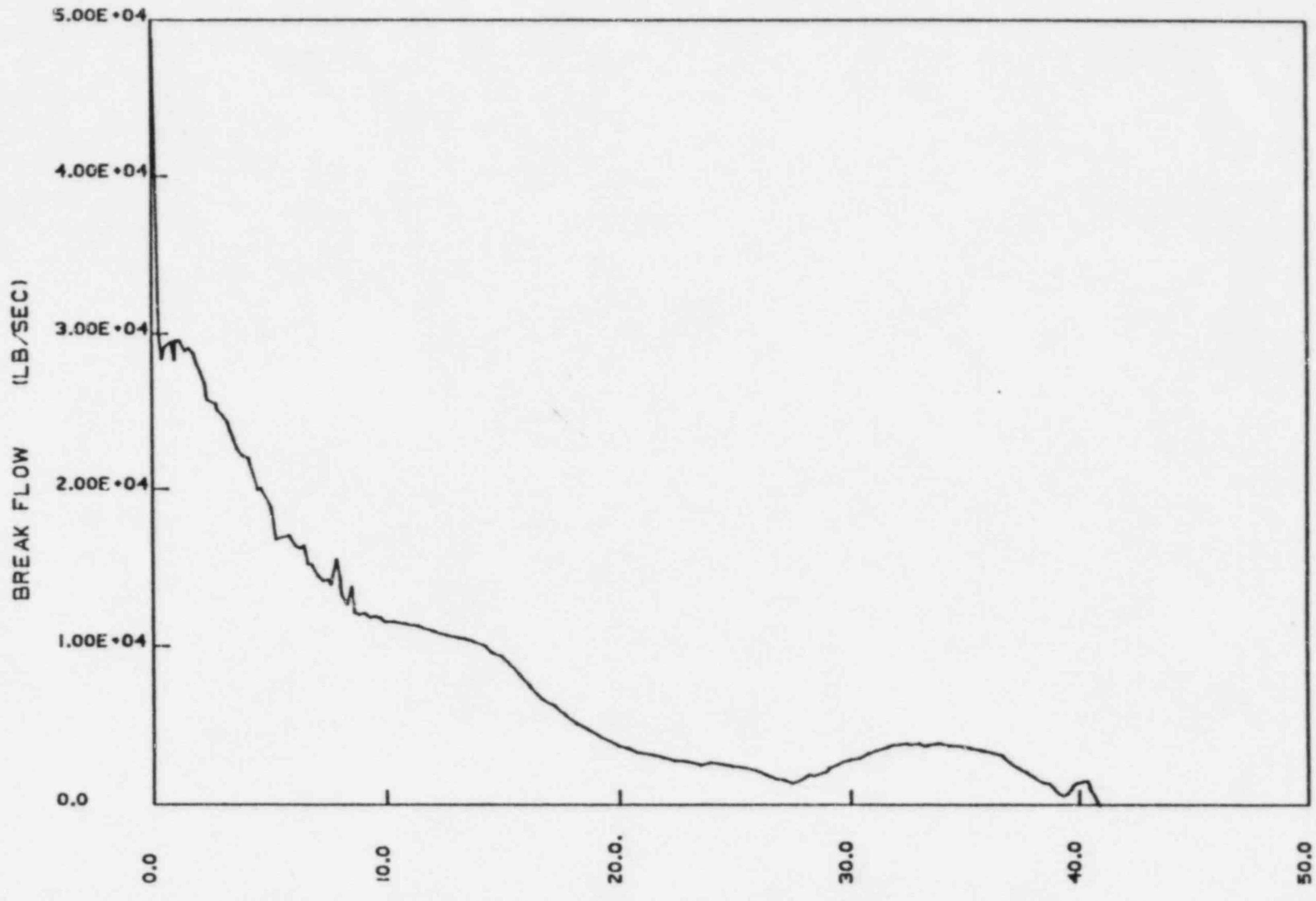


Figure 15.4-5A. Break Mass Flow, (C_D =0.4) DECLG Inactive Loop Break.

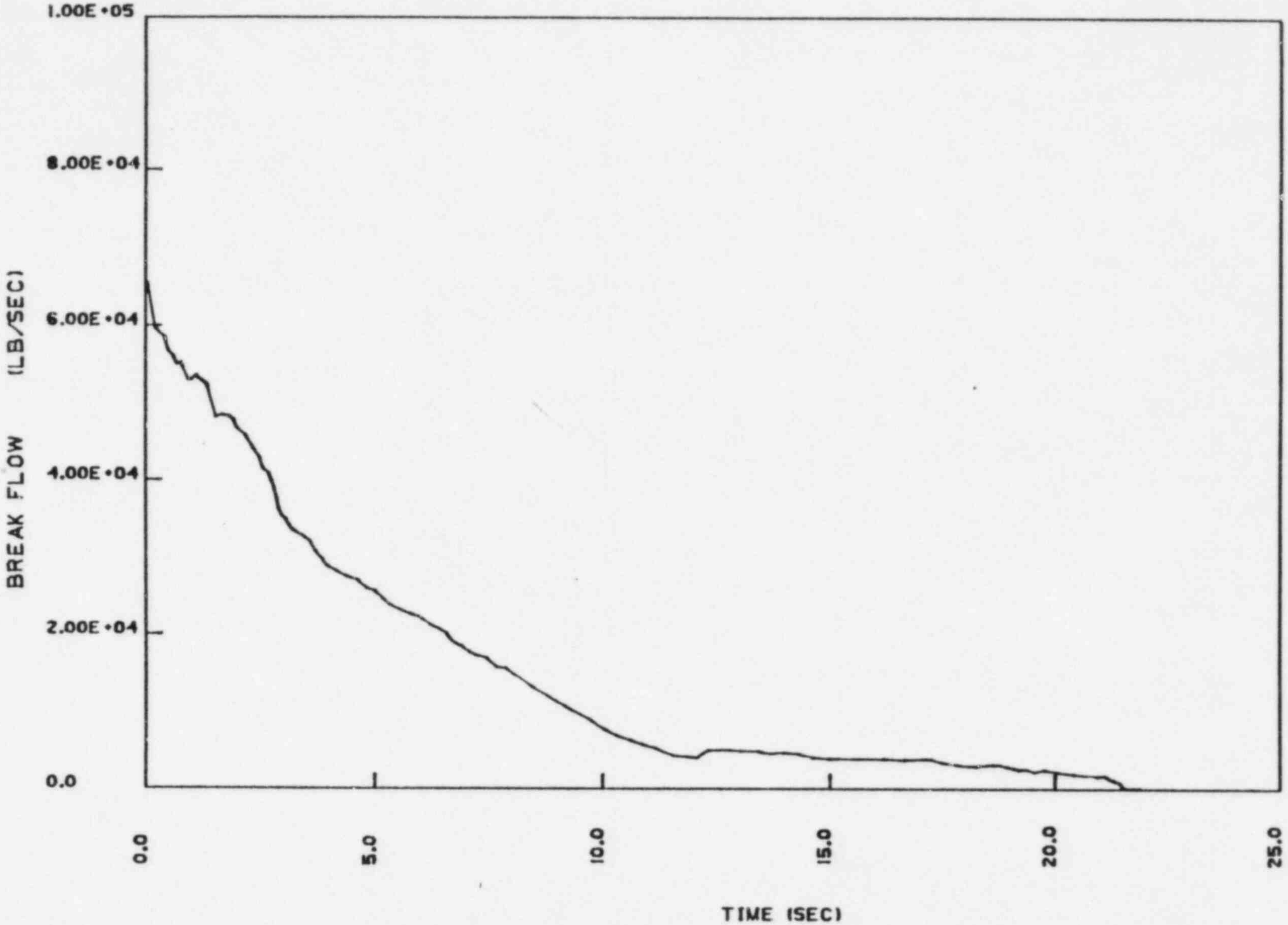


Figure 15.4-5B. Break Flow Rate - DECLG (C_D =0.6) Active Loop Break.

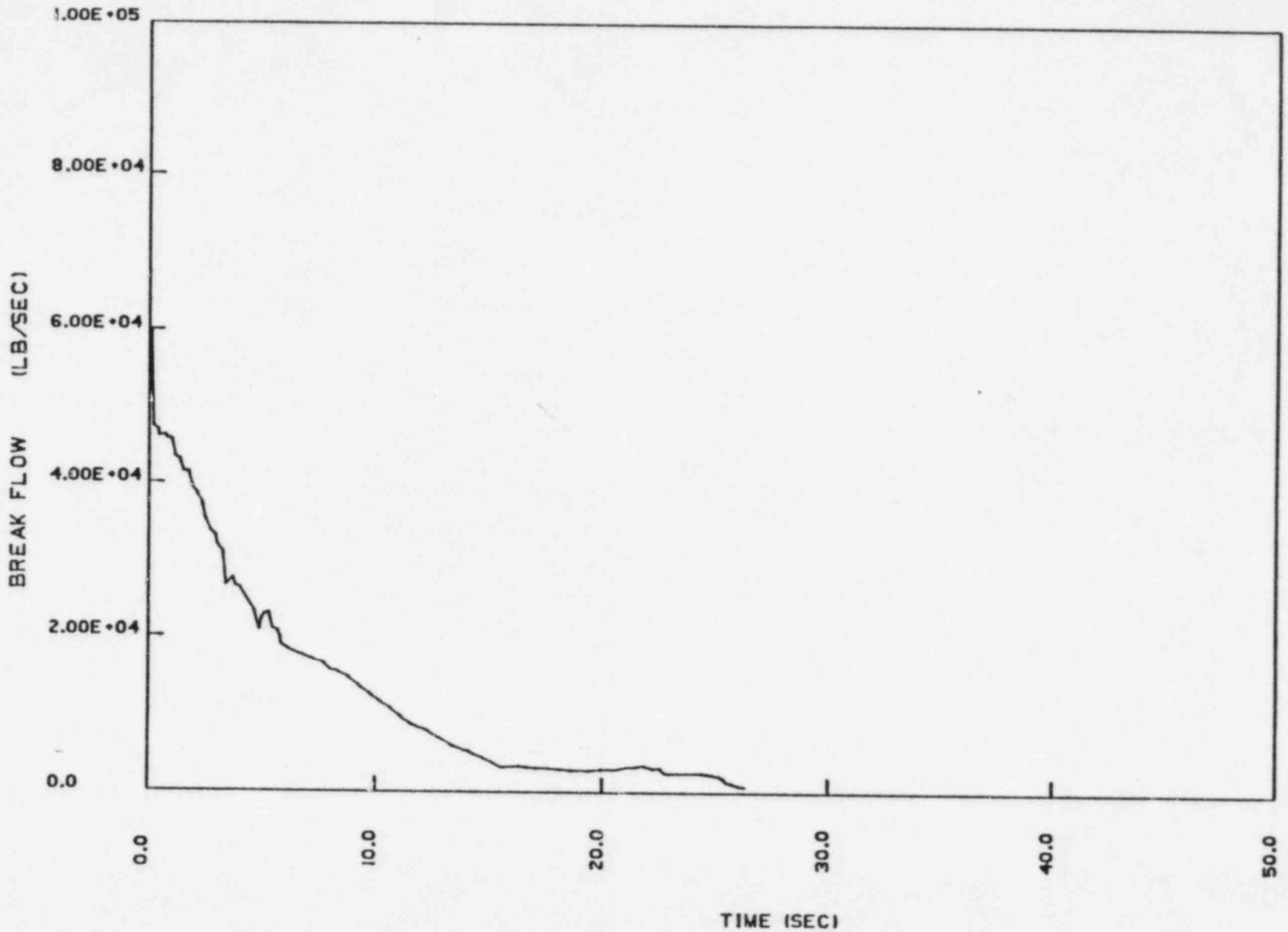


Figure 15.4-5C. Break Flow Rate - DECLG (C_D =0.4) Active Loop Break.

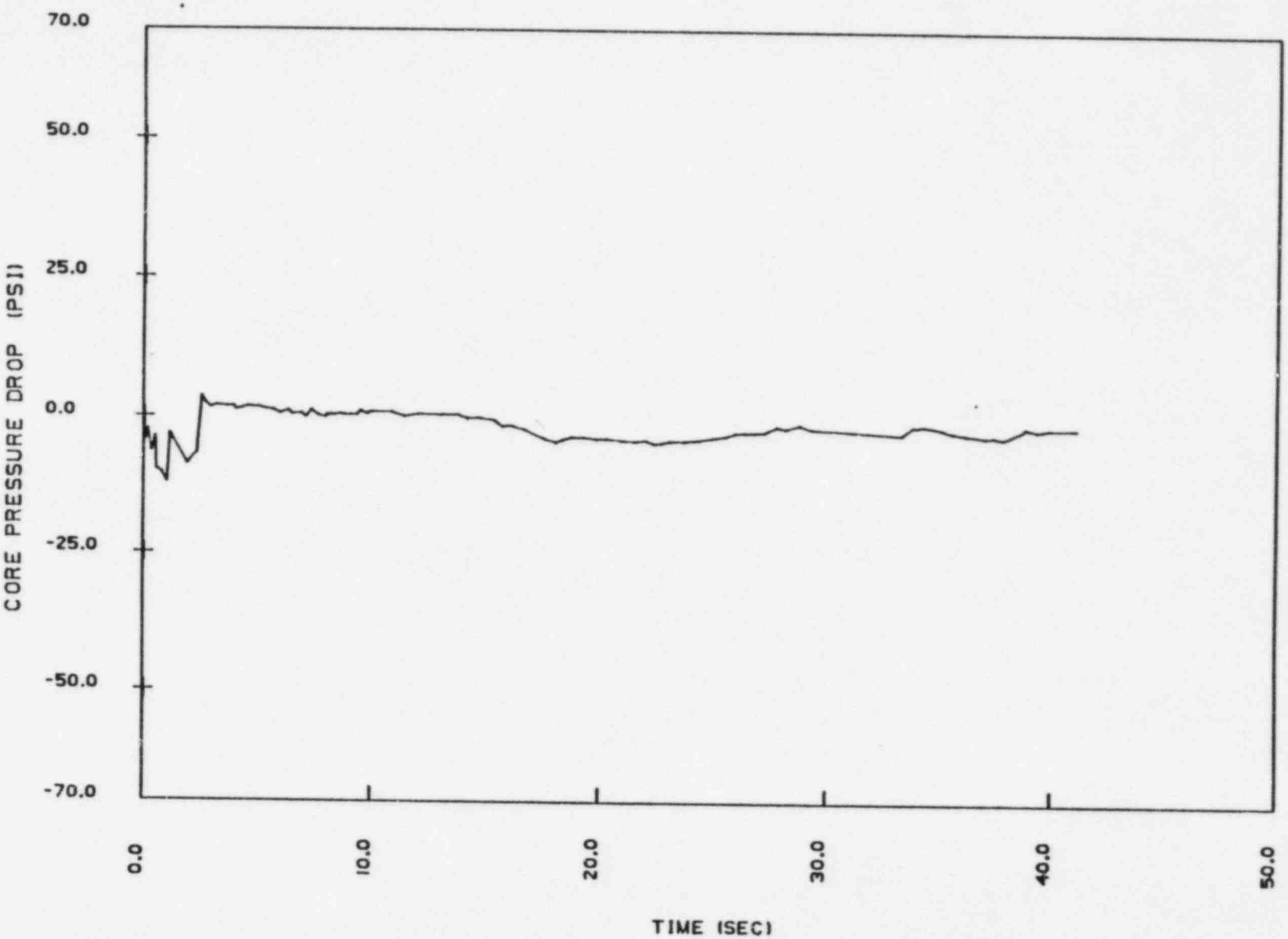


Figure 15.4-6A. Core Pressure Drop - DECLG (C_D =0.4) Inactive Loop Break.

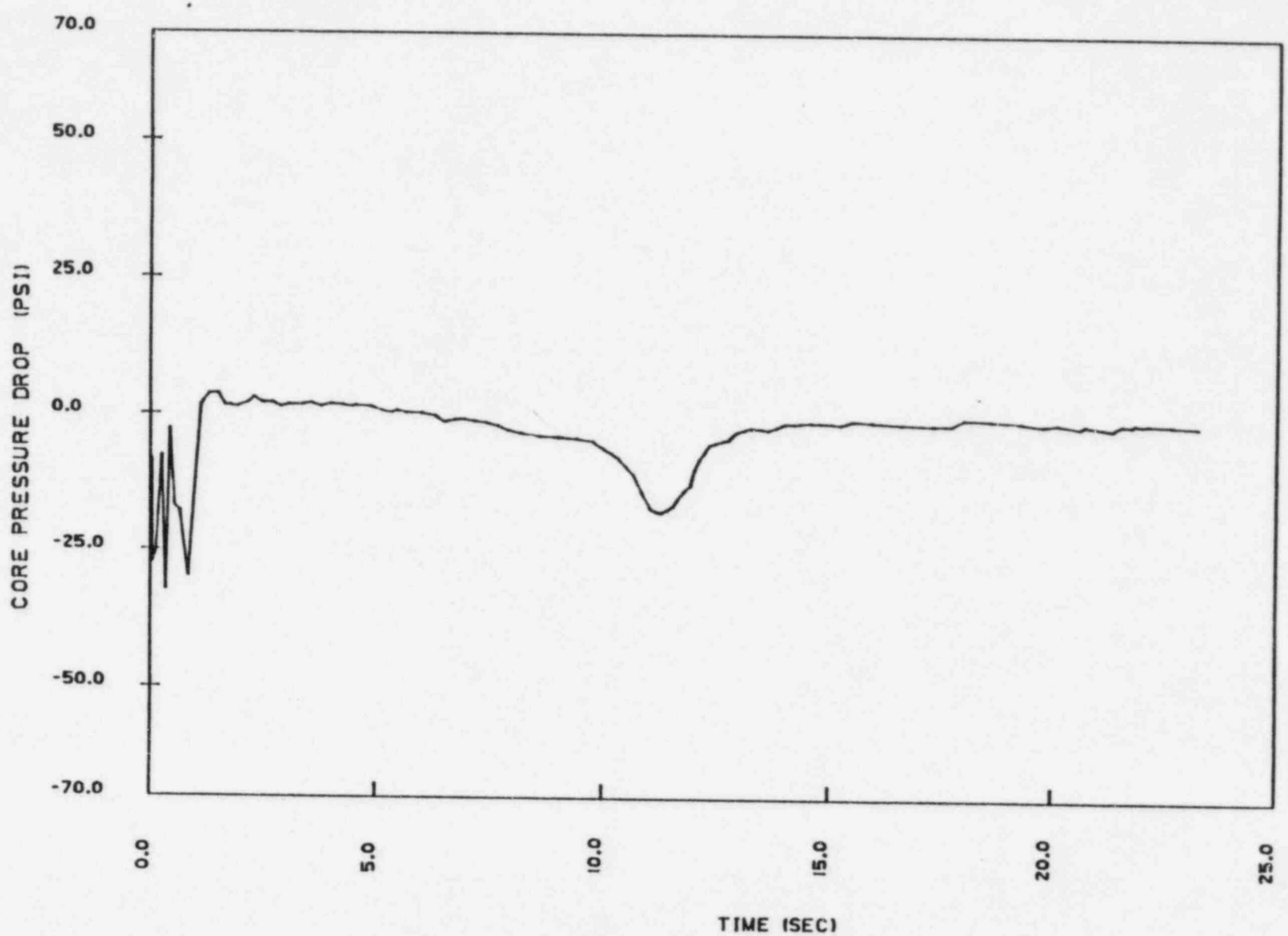


Figure 15.4-6B. Core Pressure Drop - DECLG (C_D =0.6) Active Loop Break.

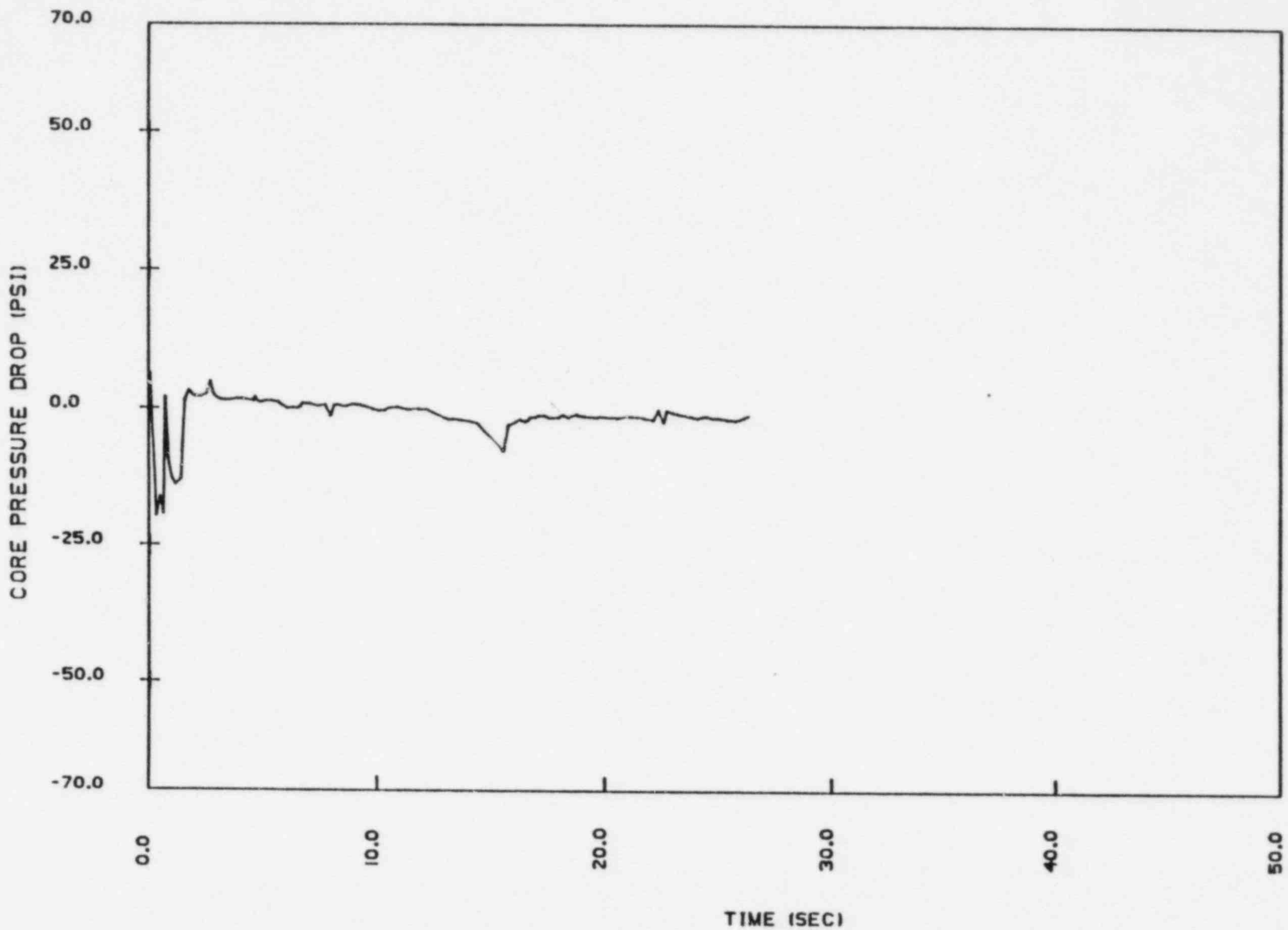


Figure 15.4-6C. Core Pressure Drop - DECLG (C_D=0.4)
Active Loop Break

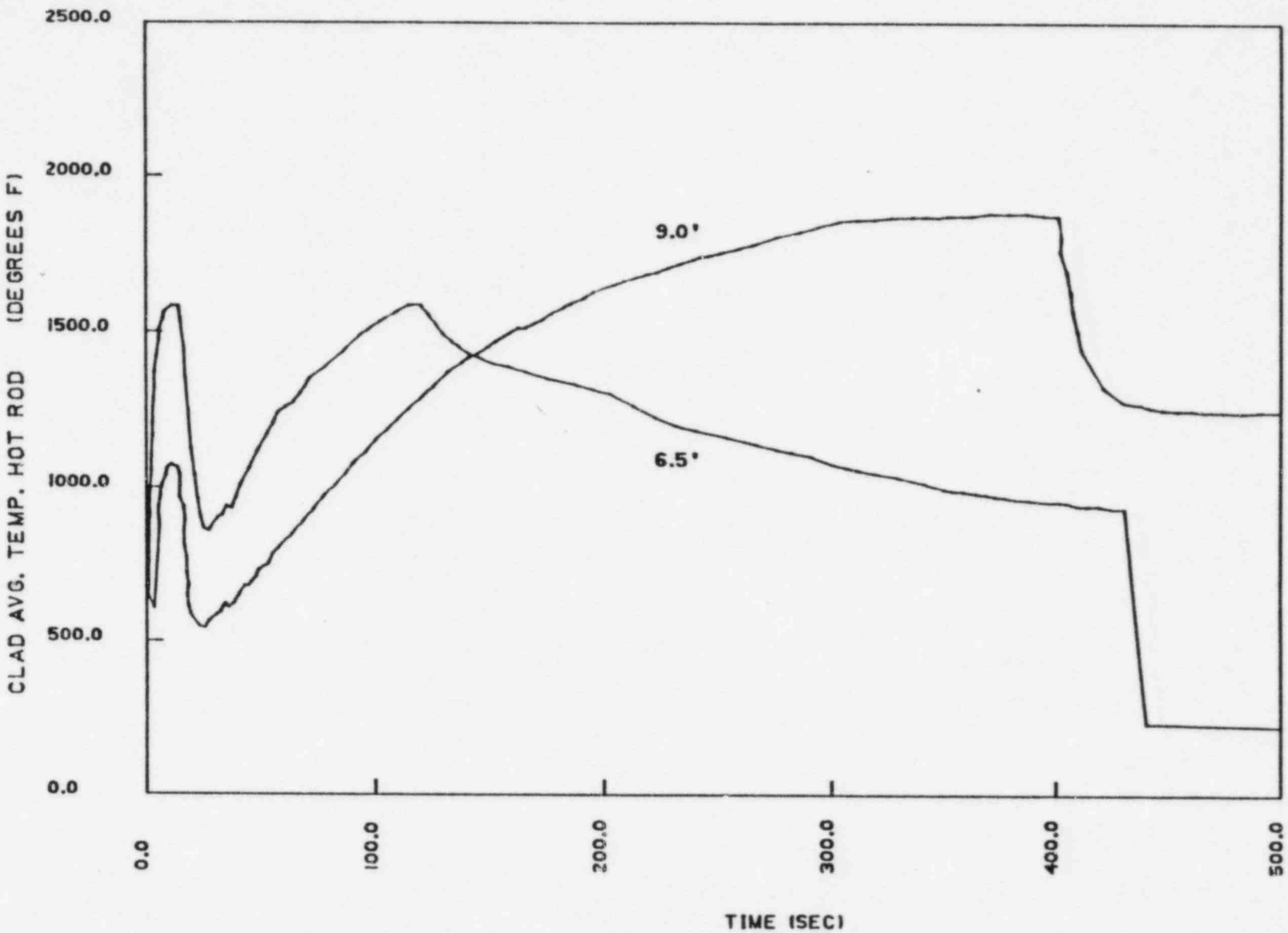


Figure 15.1-7A. Peak Clad Temperature - DECLG (C_D=0.4)
Inactive Loop Break

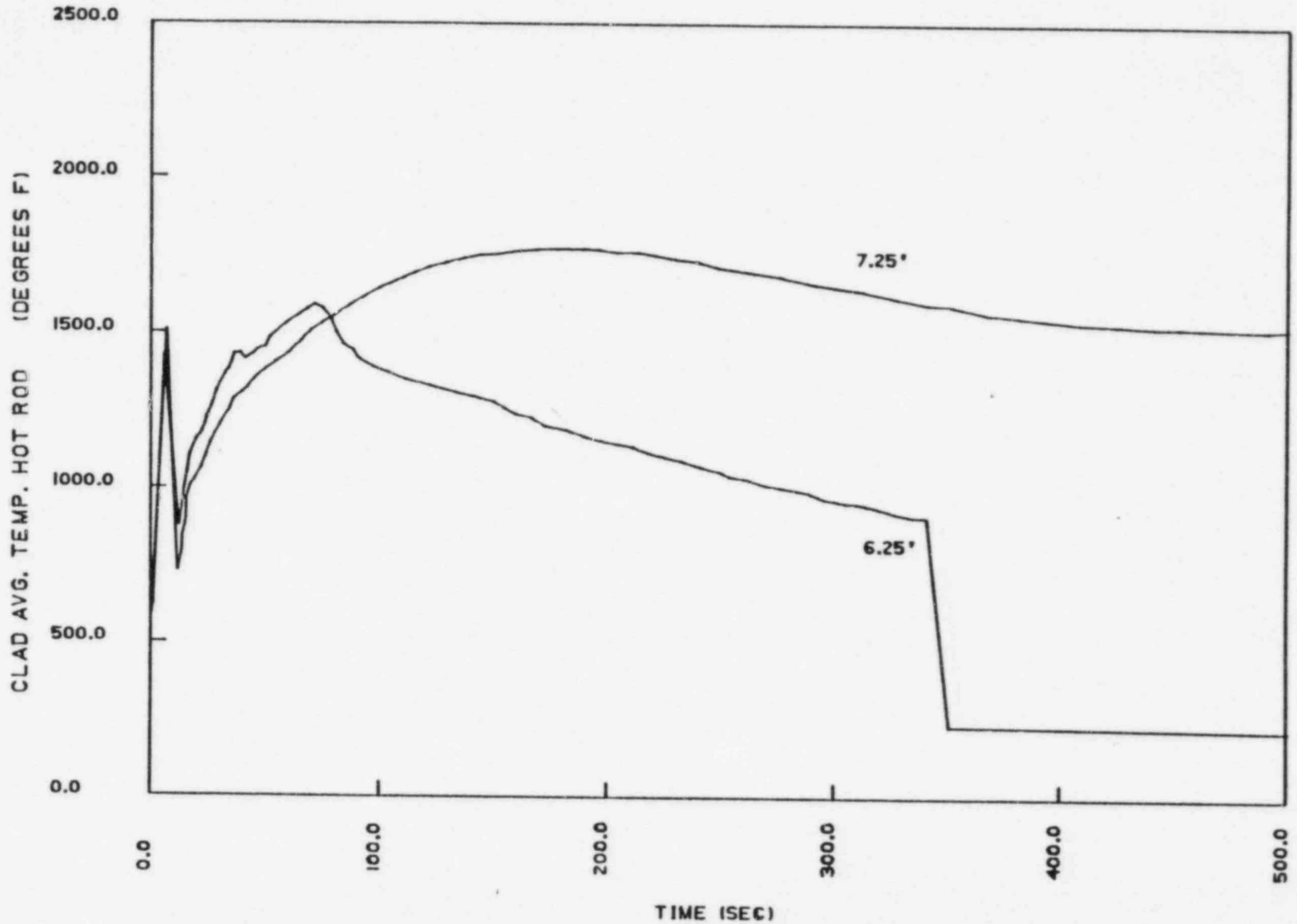


Figure 15.4-7B. Peak Clad Temperature - DECLG (C_D=0.6)
Active Loop Break

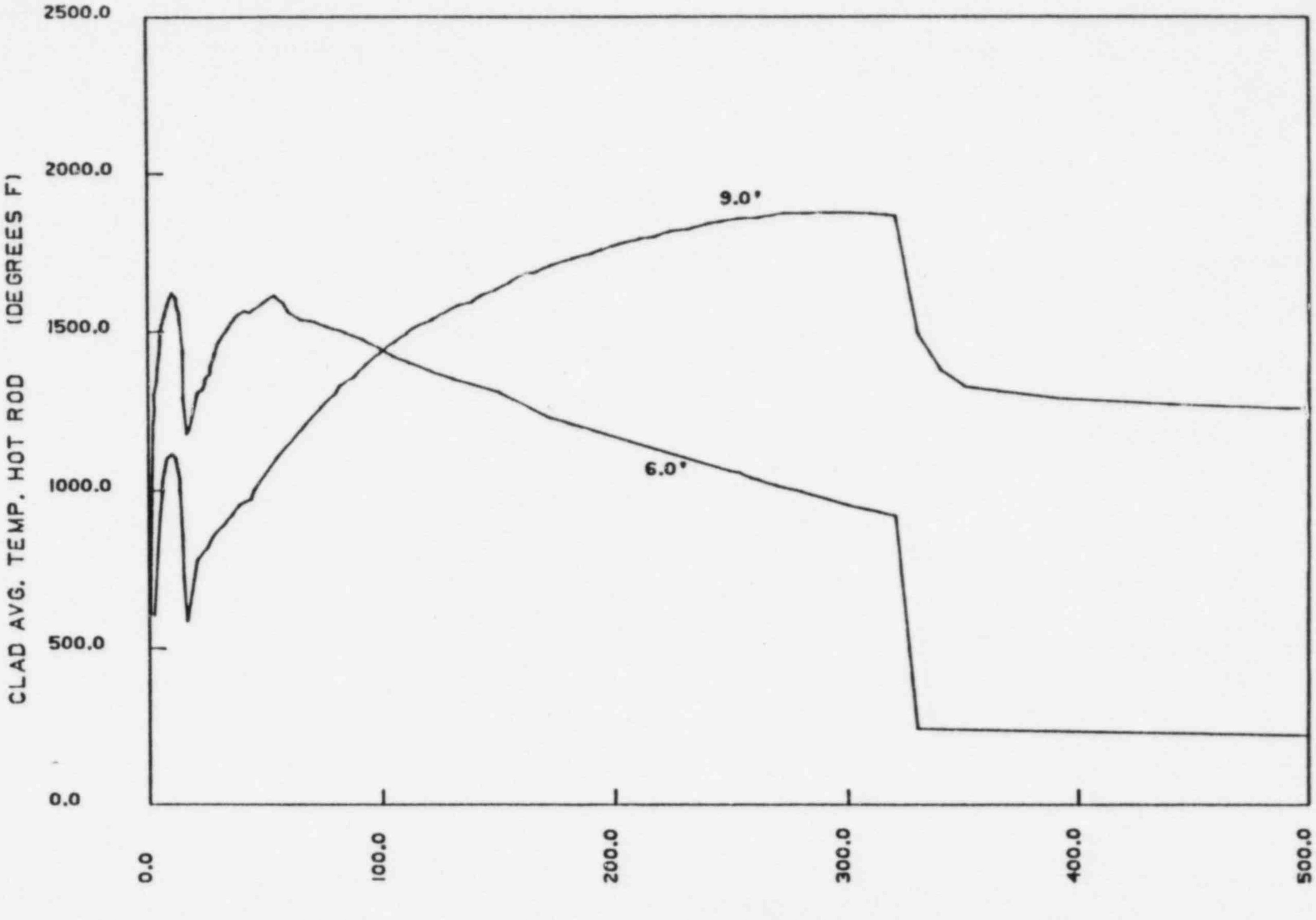


Figure 15.4-7C. Peak Clad Temperature – DECLG (C_D =0.4)
Active Loop Break

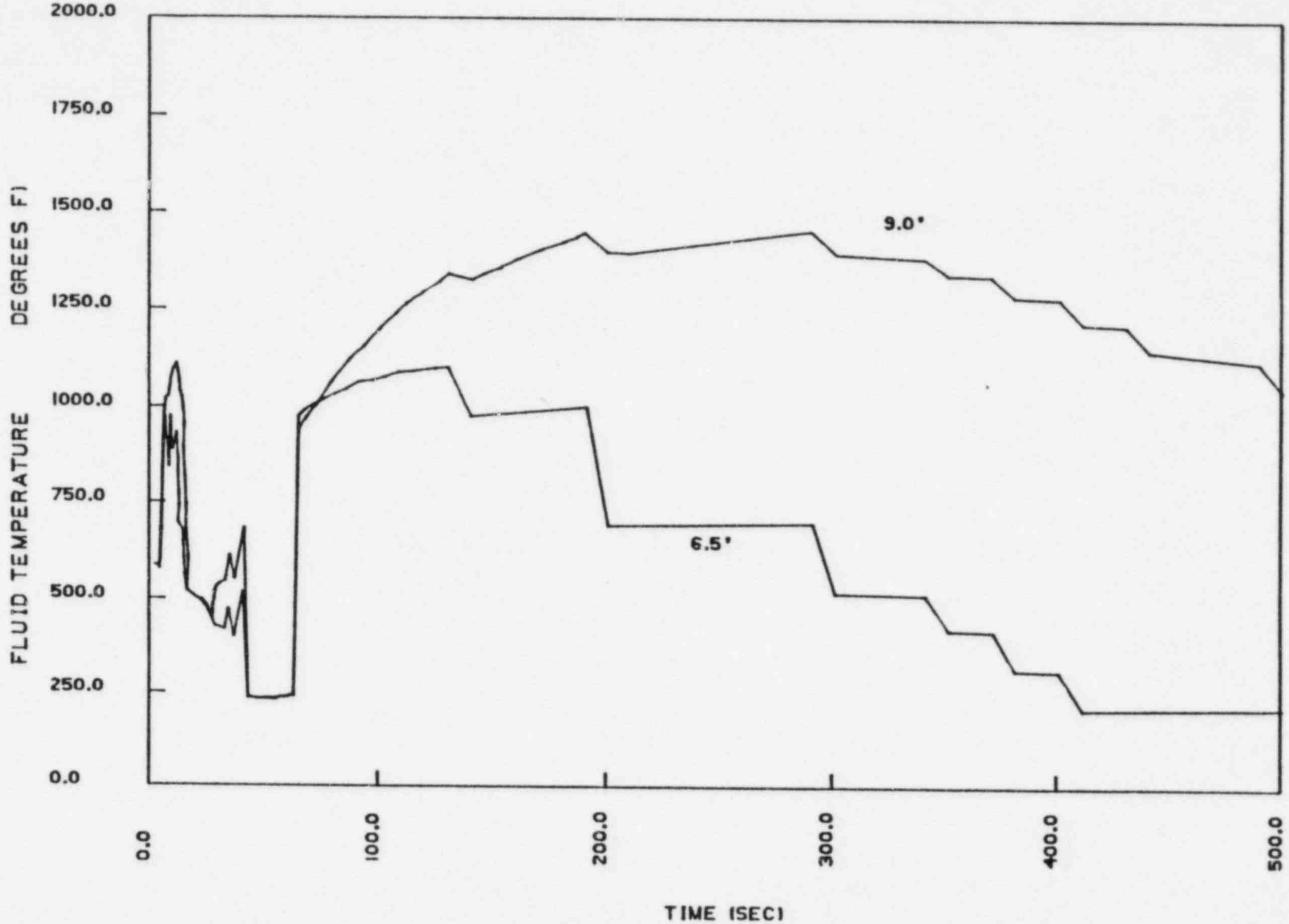


Figure 15.4-8A. Fluid Temperature - DECLG (C_D =0.4)
Inactive Loop Break

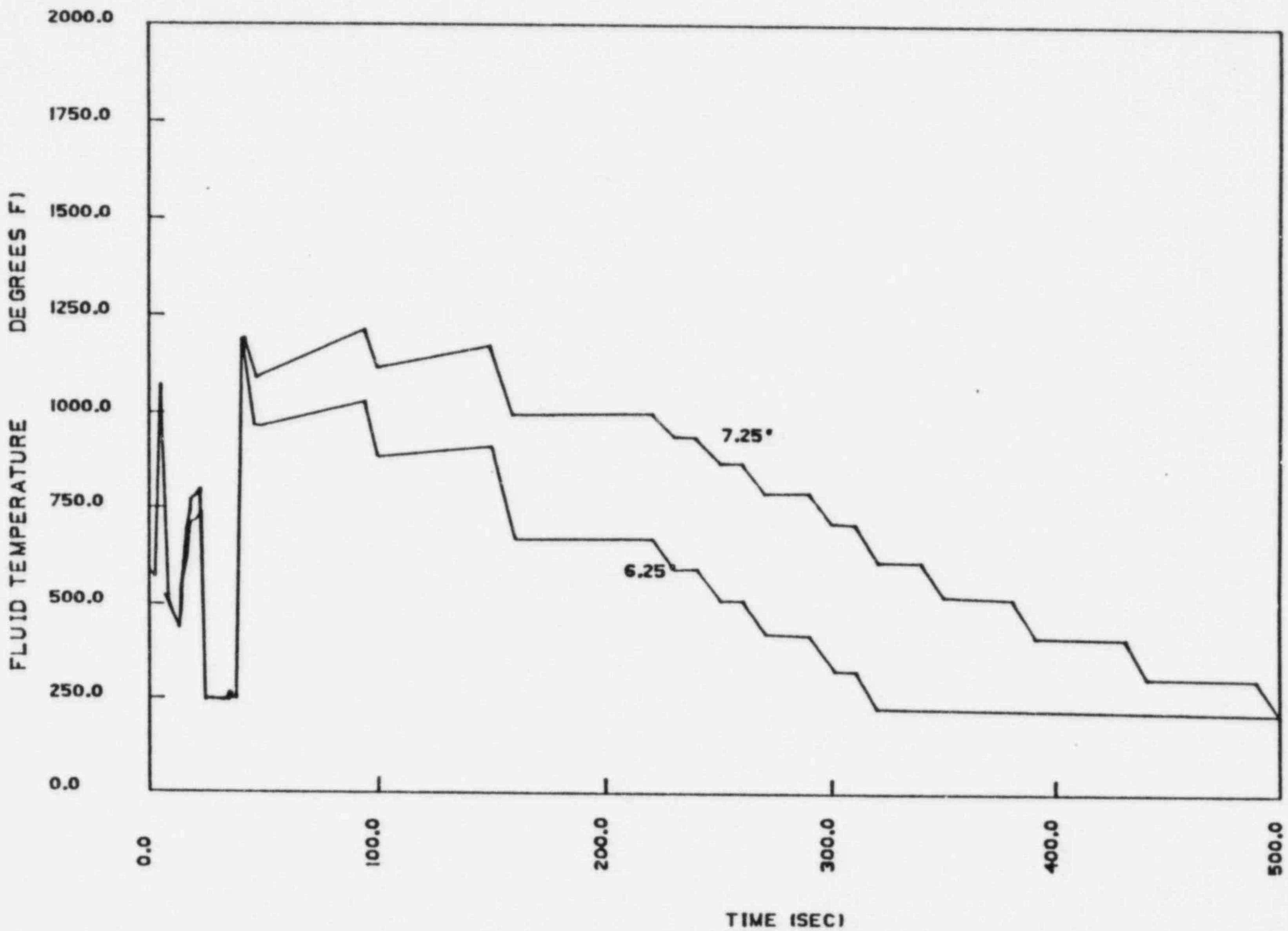


Figure 15.4-8B. Fluid Temperature - DECLG (C_D =0.6)
Active Loop Break

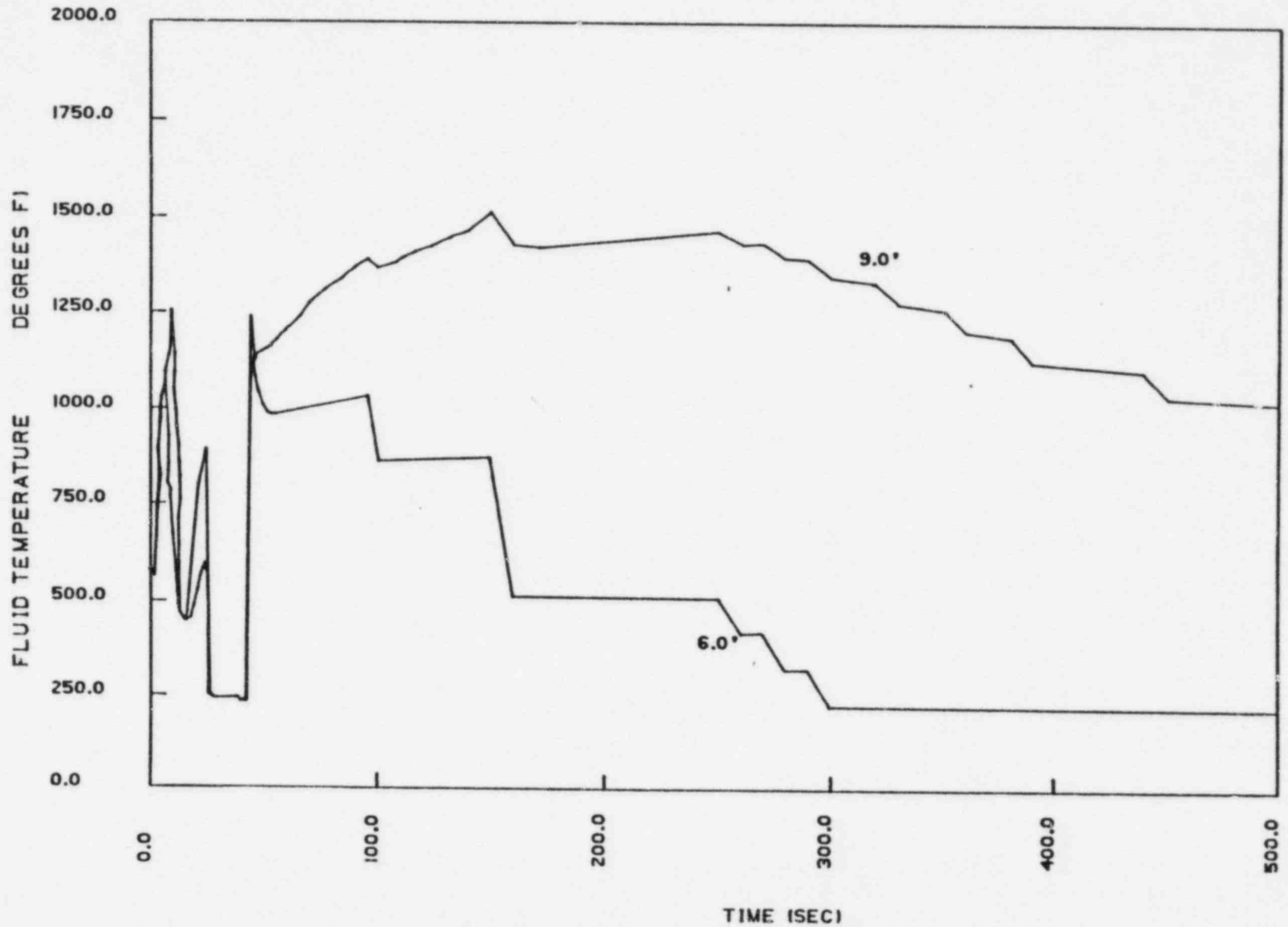


Figure 15.4-8C. Fluid Temperature - DECLG (C_D =0.4)
Active Loop Break

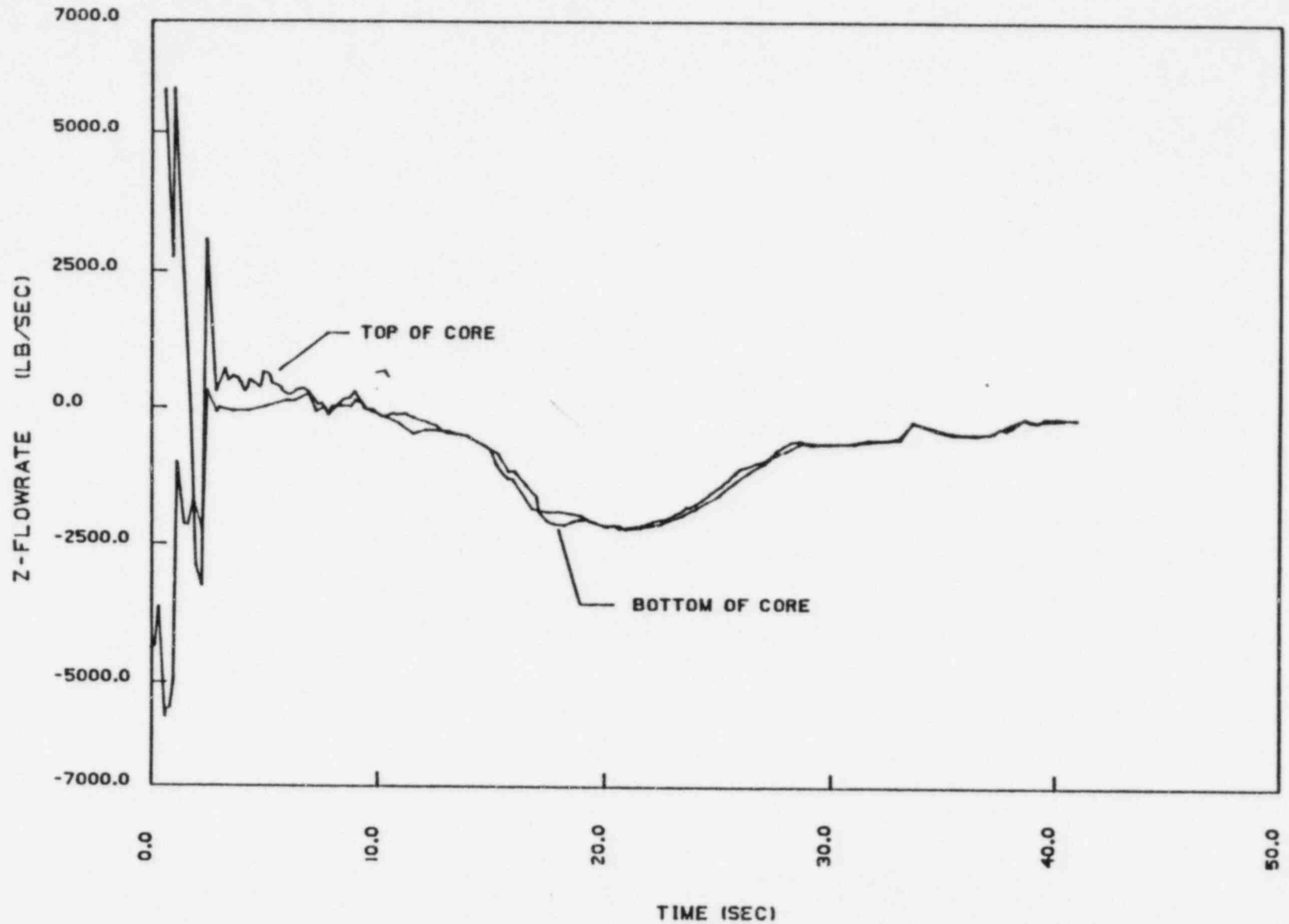


Figure 15.4-9A. Core Flow - Top and Bottom - DECLG (C_D=0.4)
Inactive Loop Break

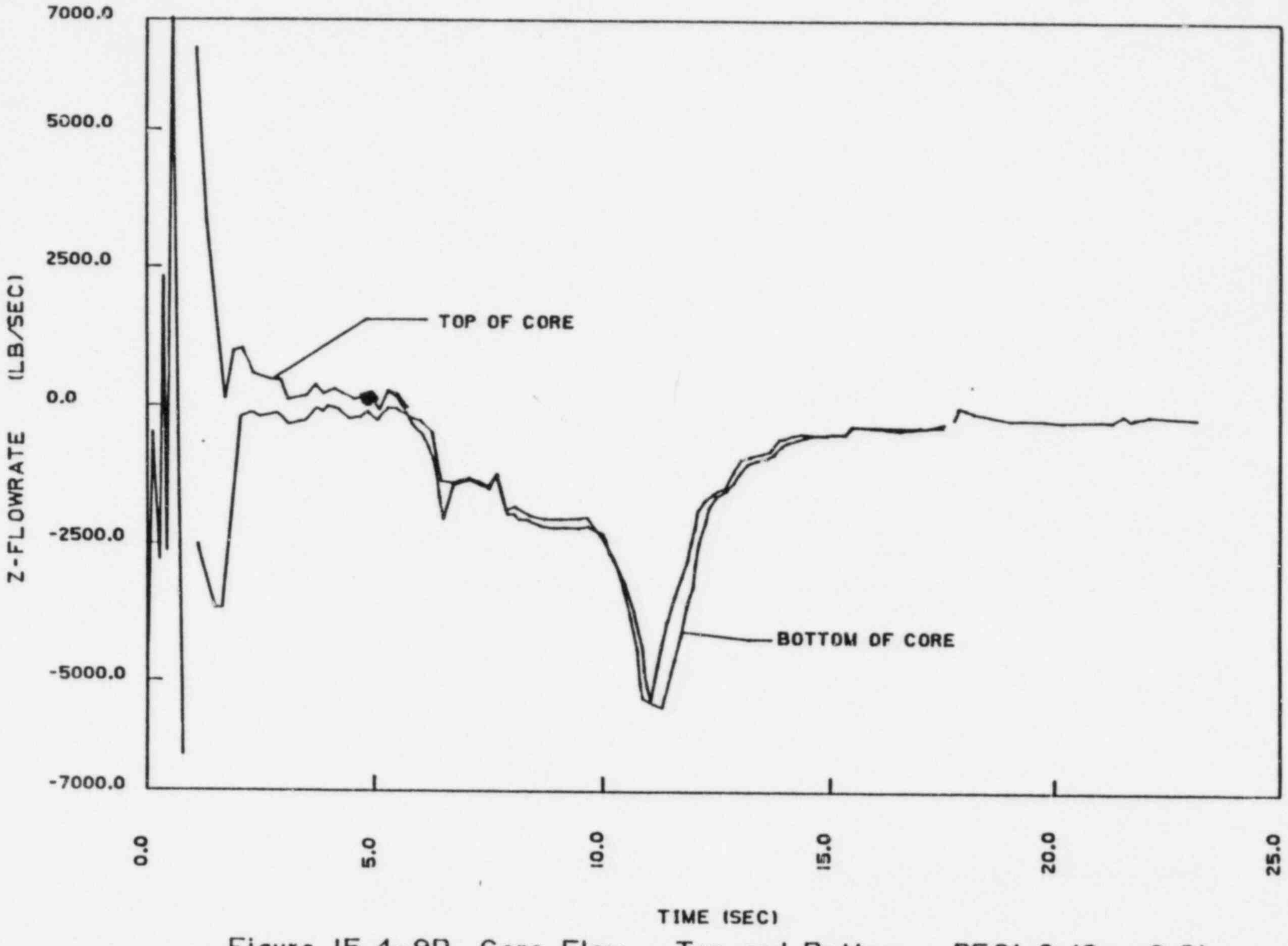


Figure 15.4-9B. Core Flow - Top and Bottom - DECLG (C_D =0.6)
Active Loop Break

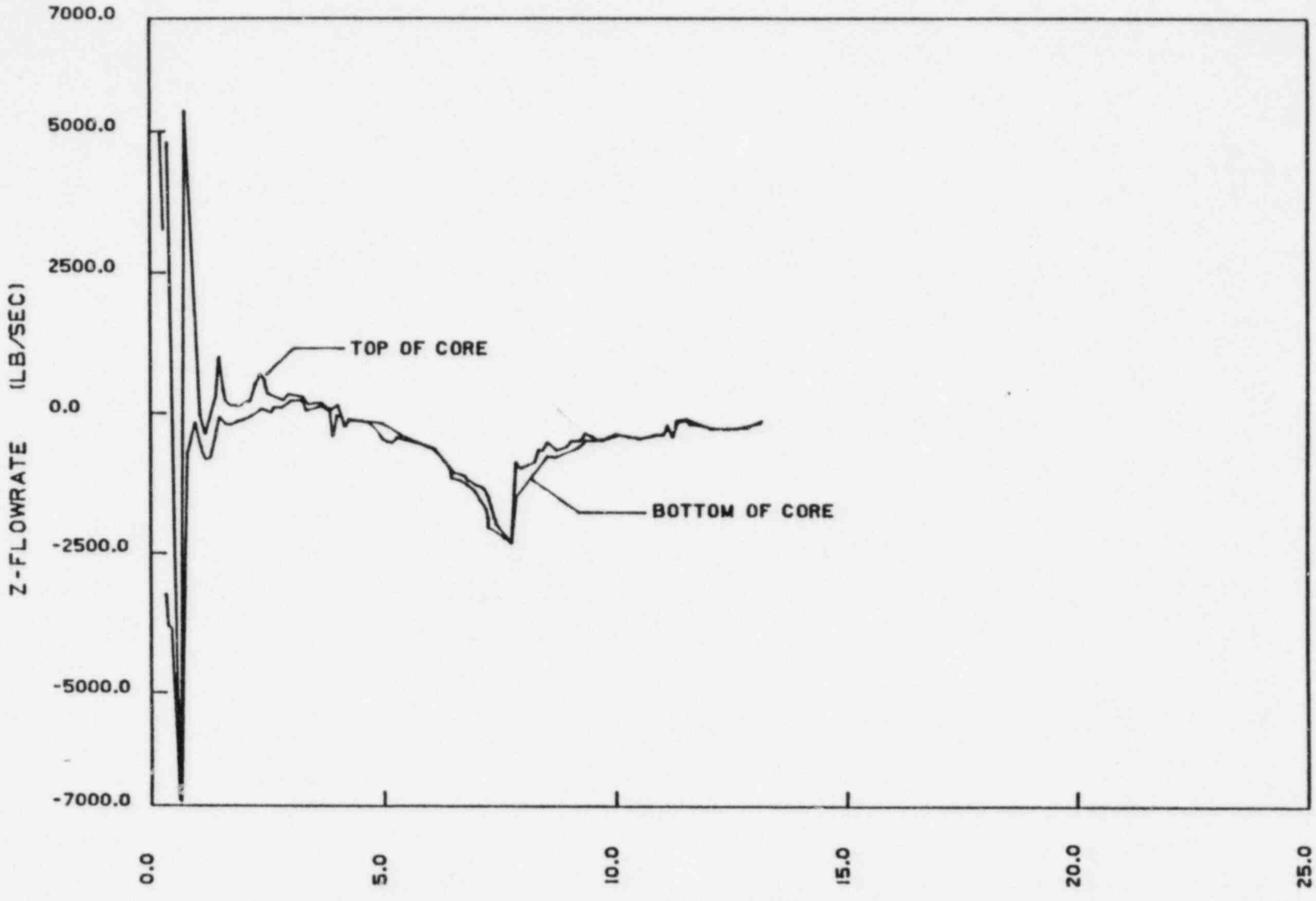


Figure 15.4-9C. Core Flow - Top and Bottom - DECLG (C_D =0.4)
Active Loop Break

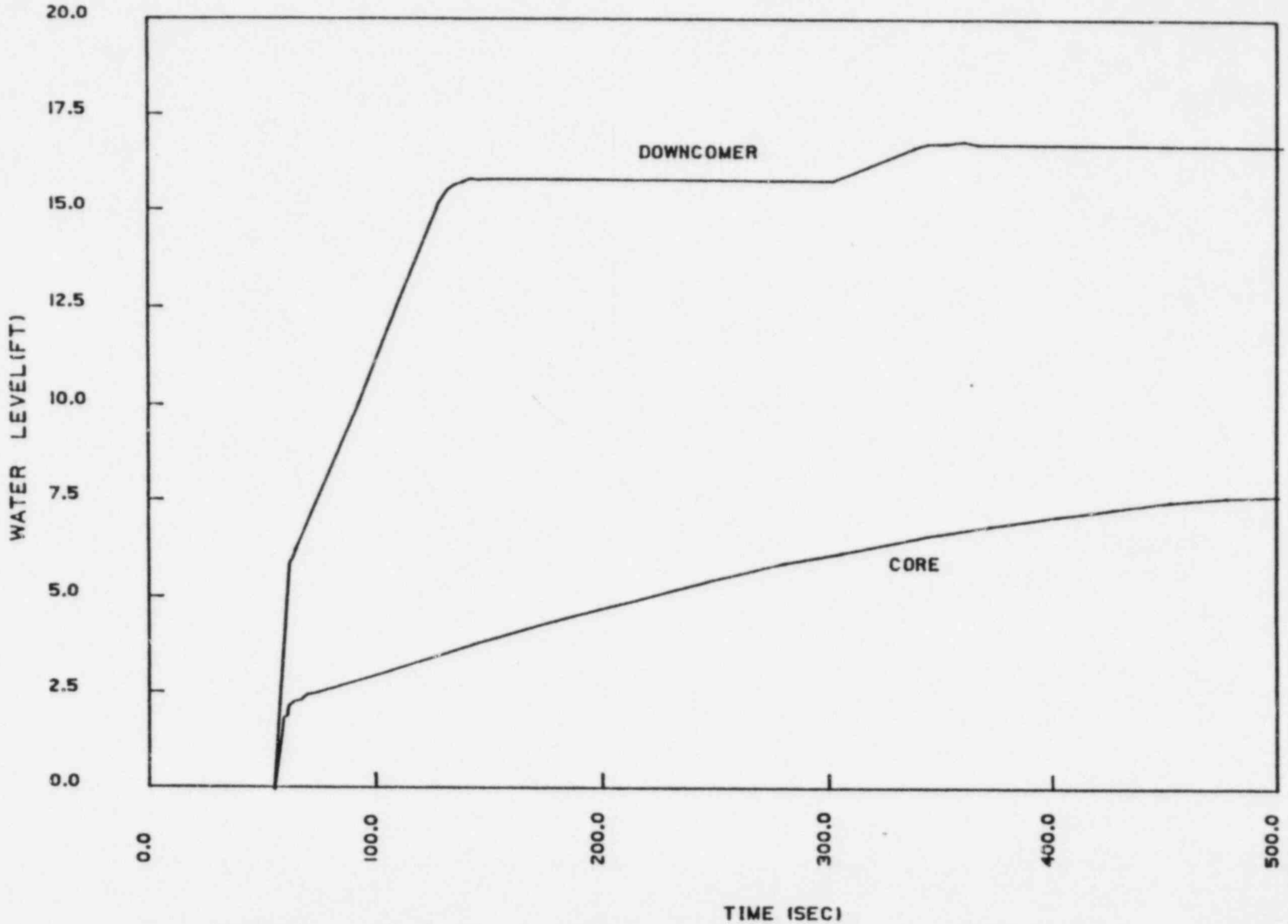


FIGURE 15.4-10A. Reflood Transient - DECLG (C_D =0.4) Inactive Loop Break Downcomer and Core Water Levels

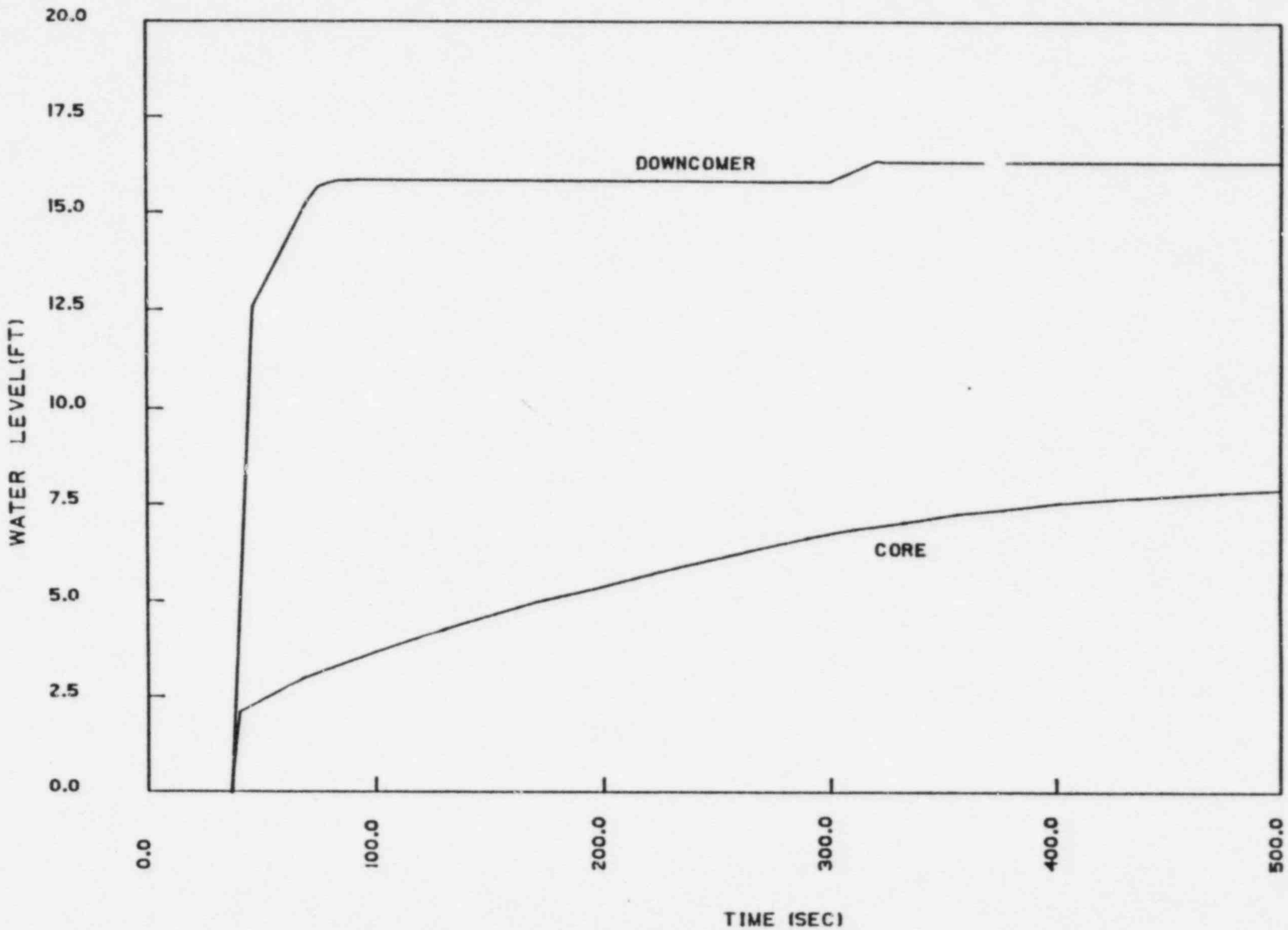


Figure 15.4-10B. Reflood Transient - DECLG (C_D =0.6) Active Loop Break
Downcomer and Core Water Levels

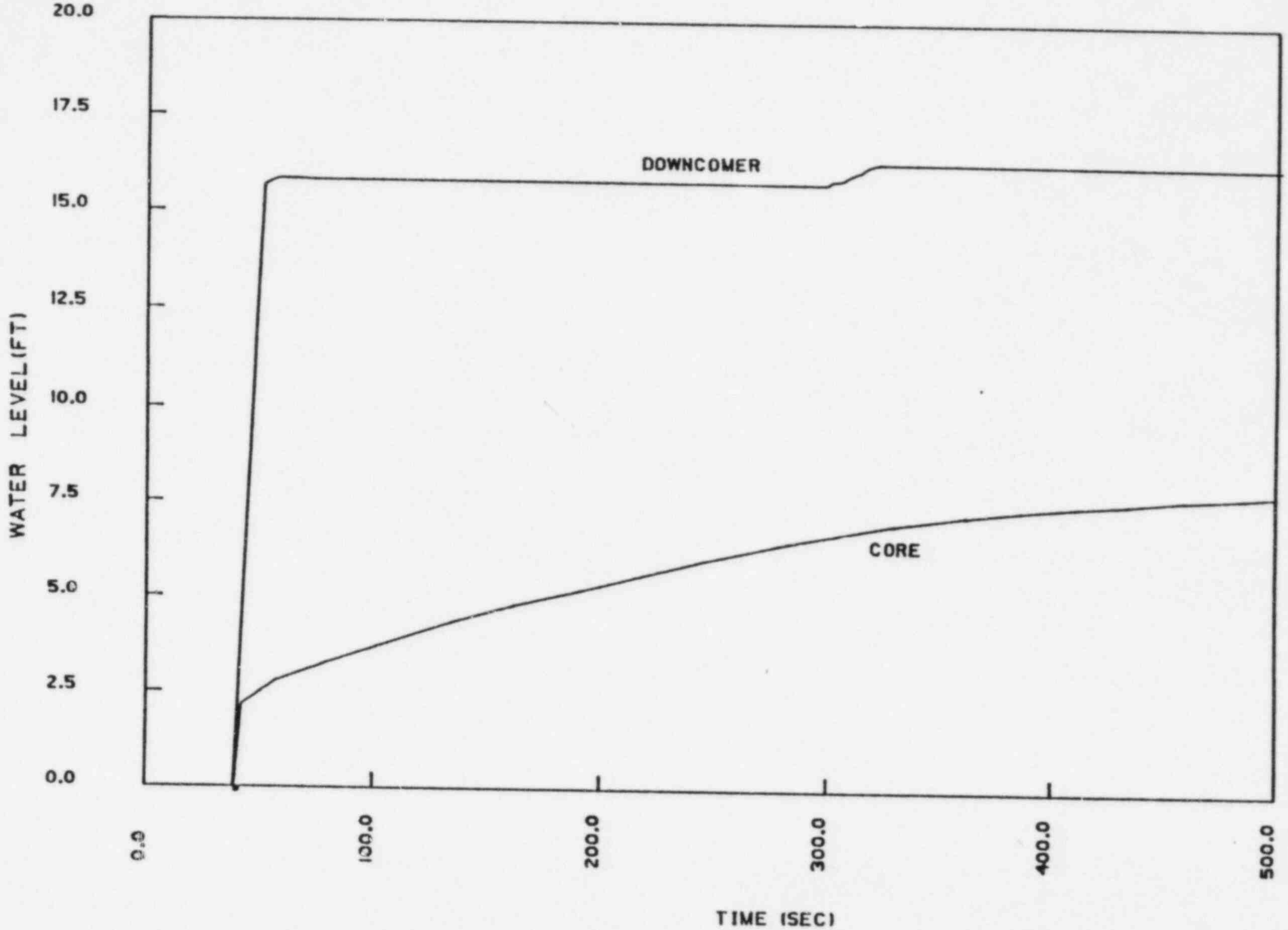


Figure 15.4-10C. Reflood Transient - DECLG(C_D =0.4) Active Loop Break Downcomer and Core Water Levels

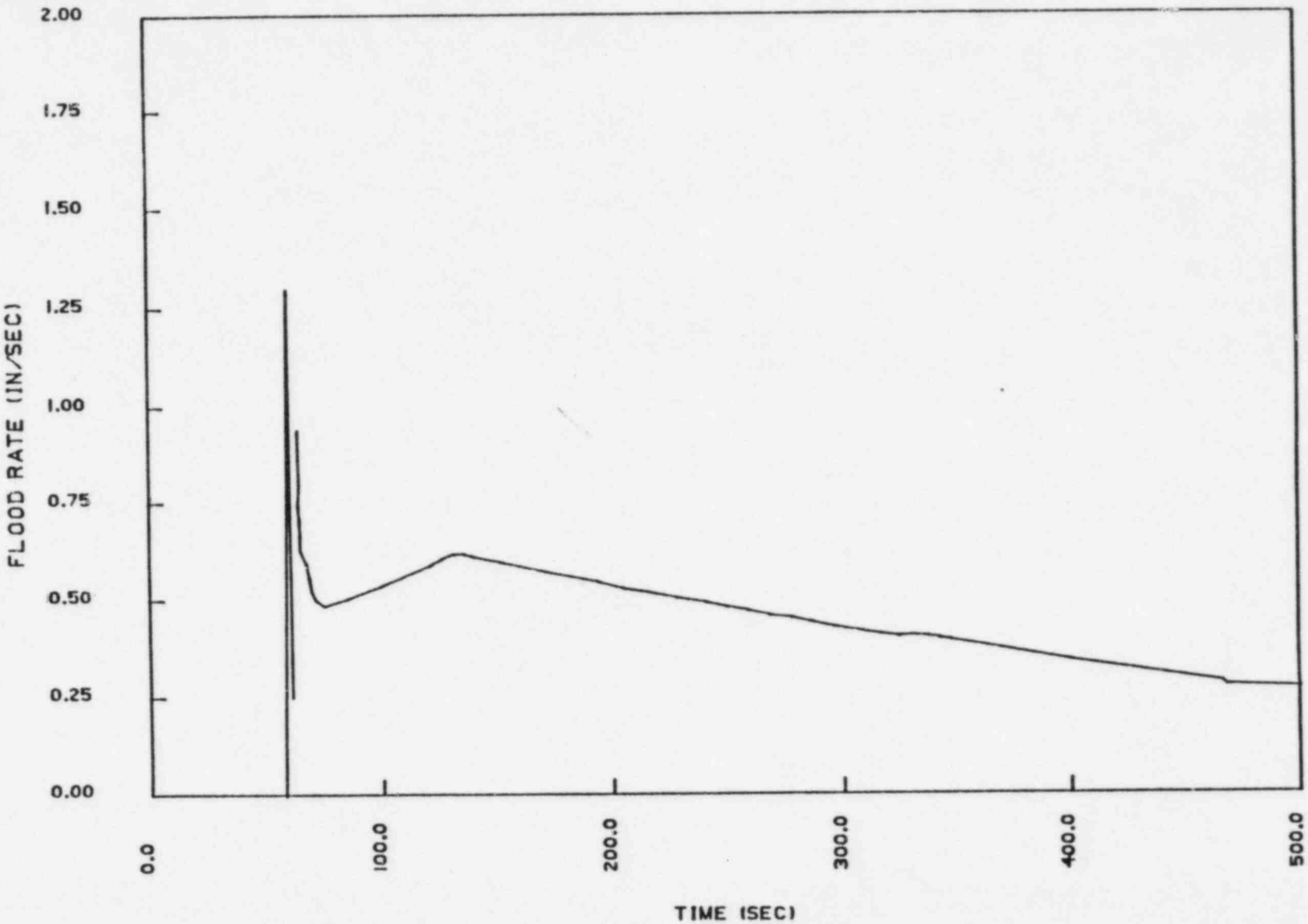


FIGURE 15.4-10D. Reflood Transient - DECLG (C_D=0.4) Inactive Loop Break Core Inlet Velocity

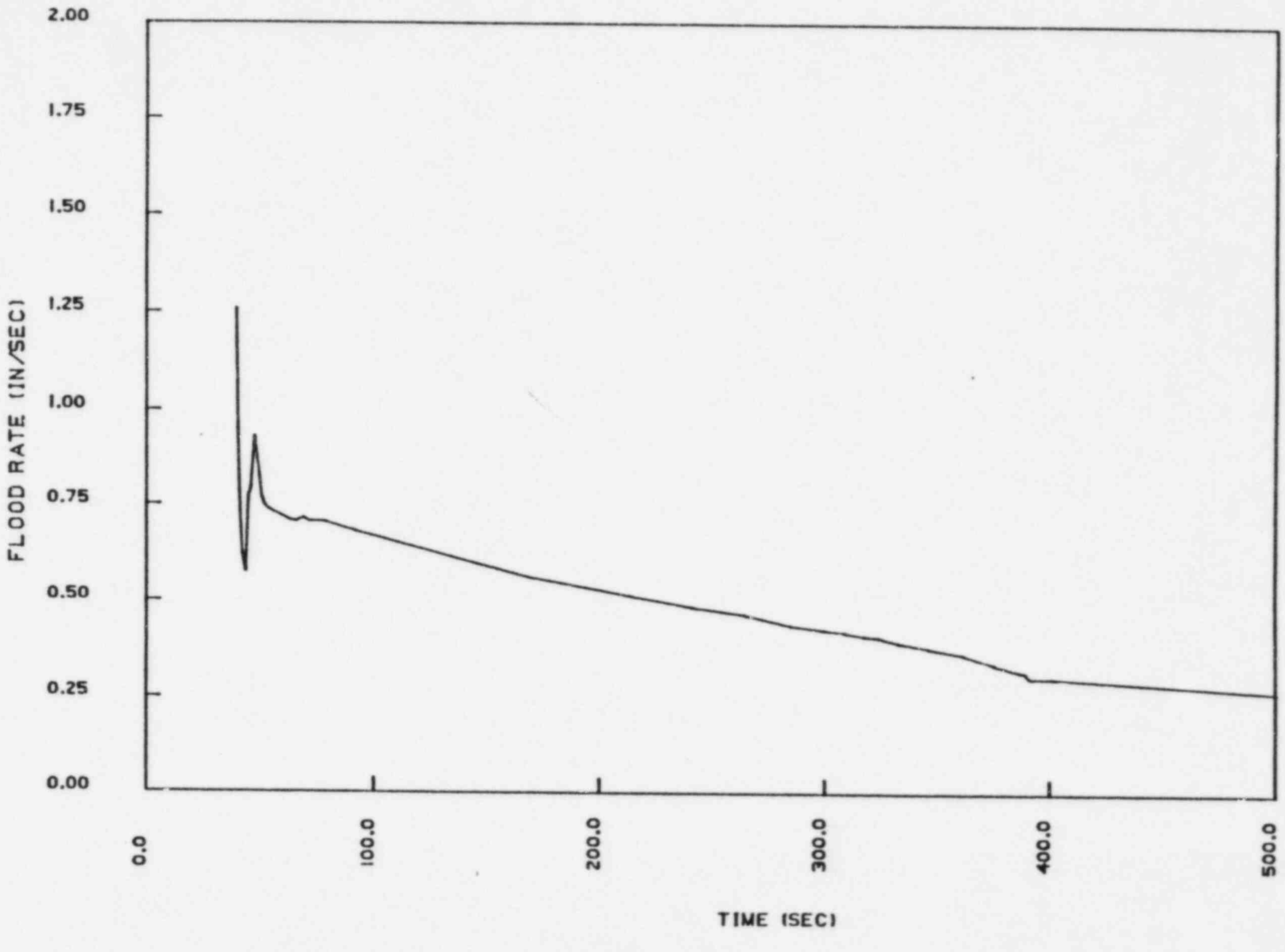


Figure 15.4-10E. Reflood Transient - DECLG (C_D =0.6) Active Loop Break
Core Inlet Velocity

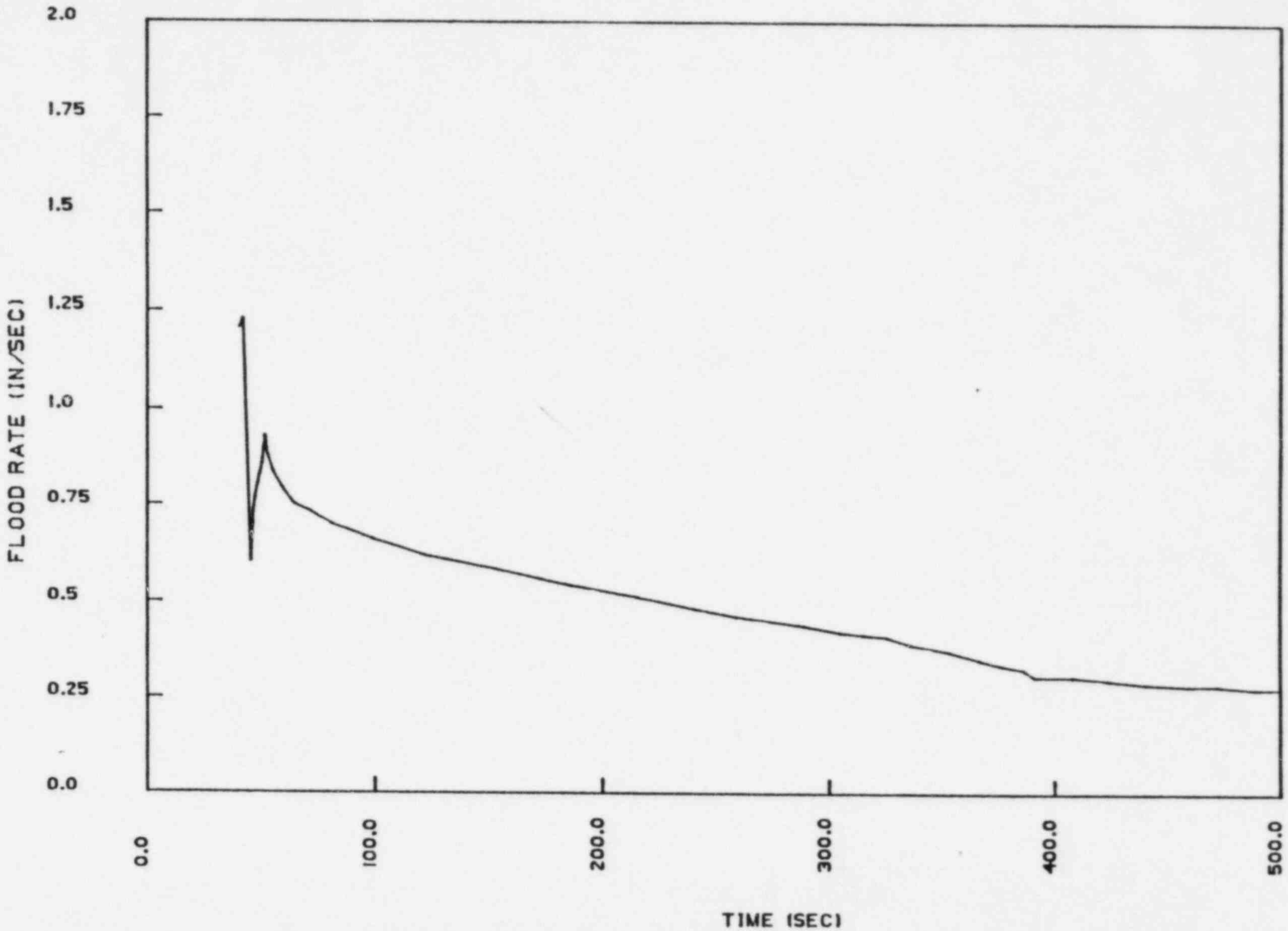


Figure 15.4-10F. Reflood Transient - DECLG(C_D =0.4) Active Loop Break

Core Inlet Velocity

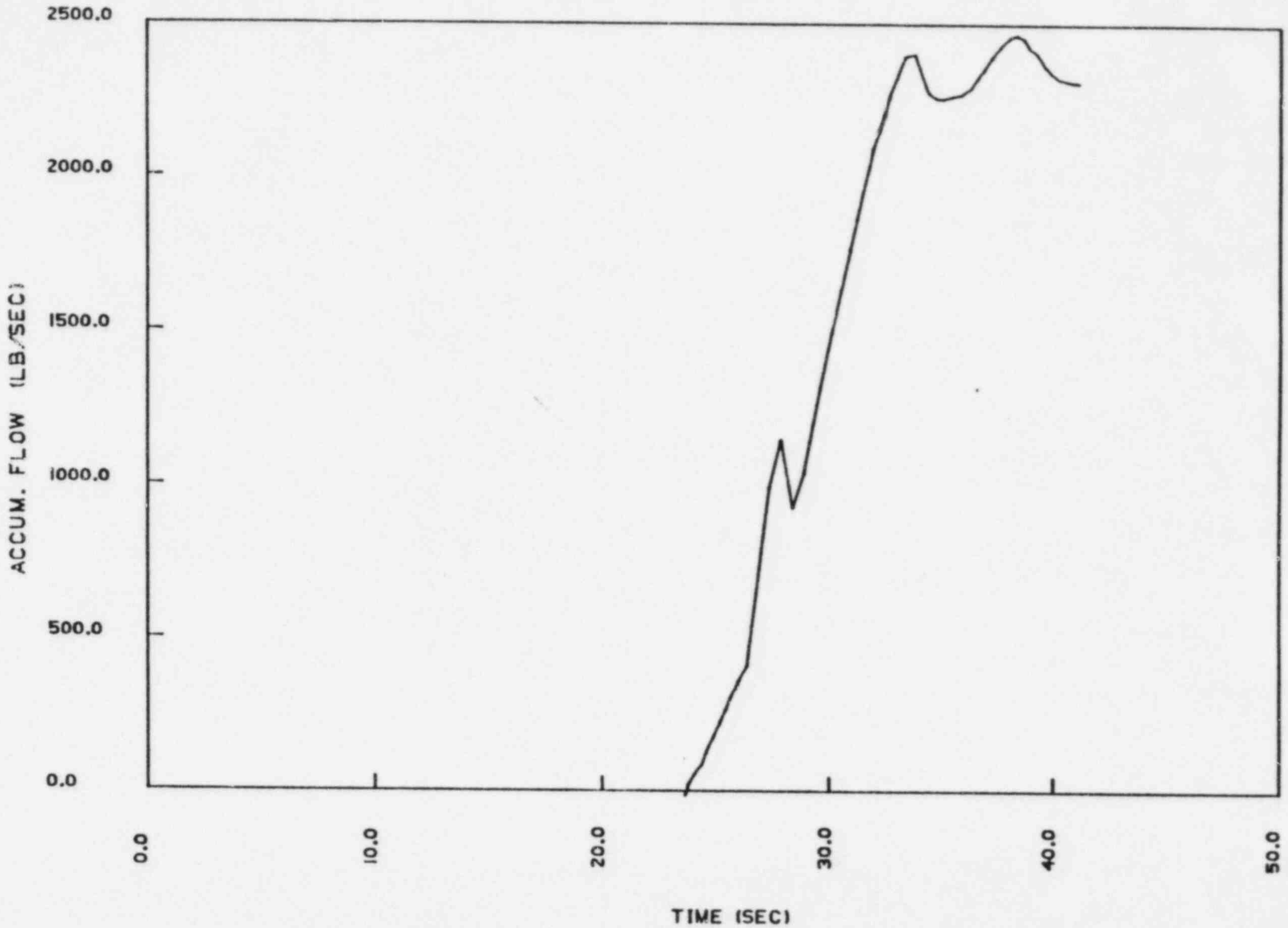


Figure 15.4-11A. Accumulator Flow (Blowdown) - DECLG (C_D =0.4)
Inactive Loop break

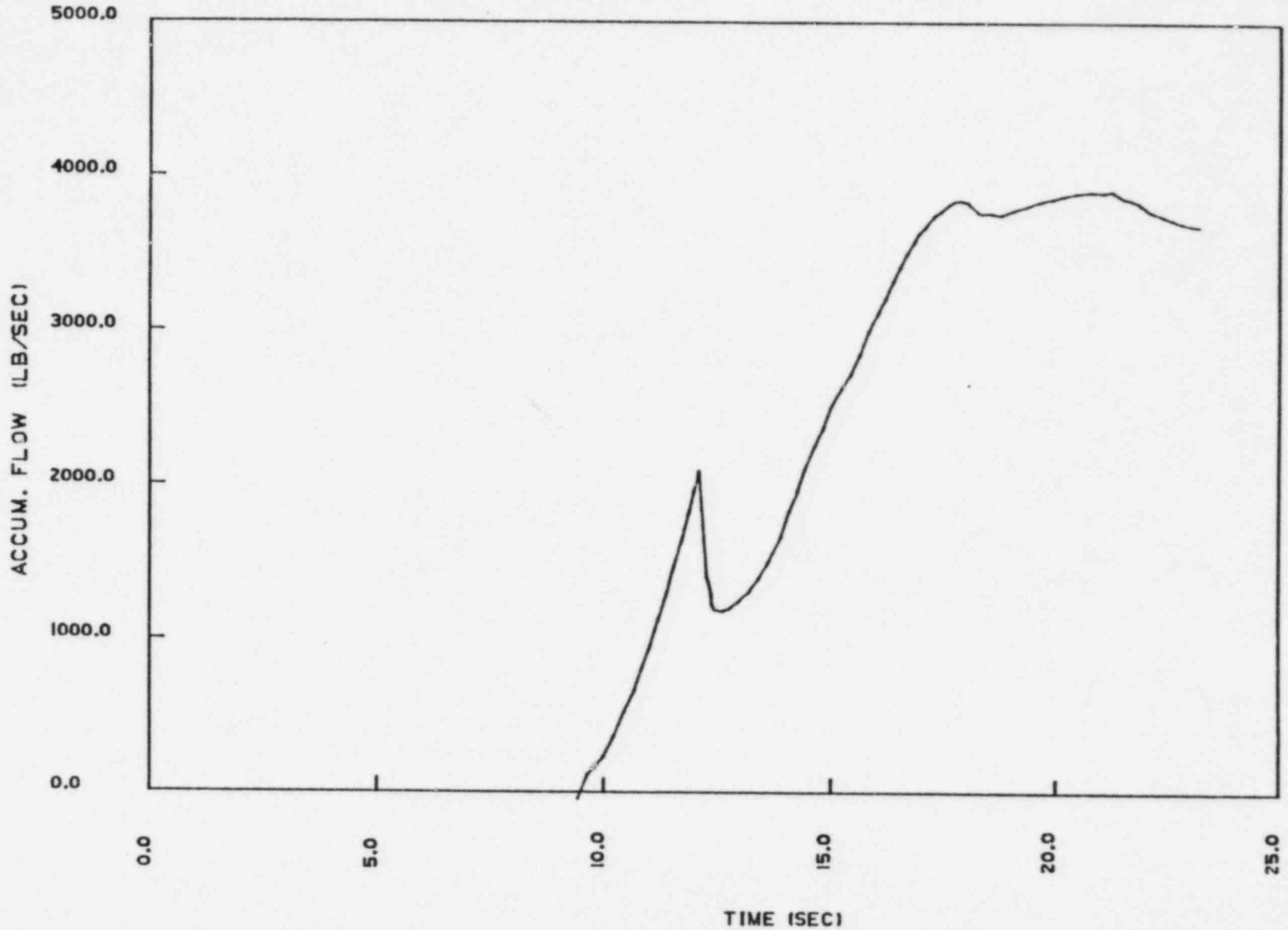


Figure 15.4-11B. Accumulator Flow (Blowdown) - DECLG (C_D =0.6)
Active Loop Break

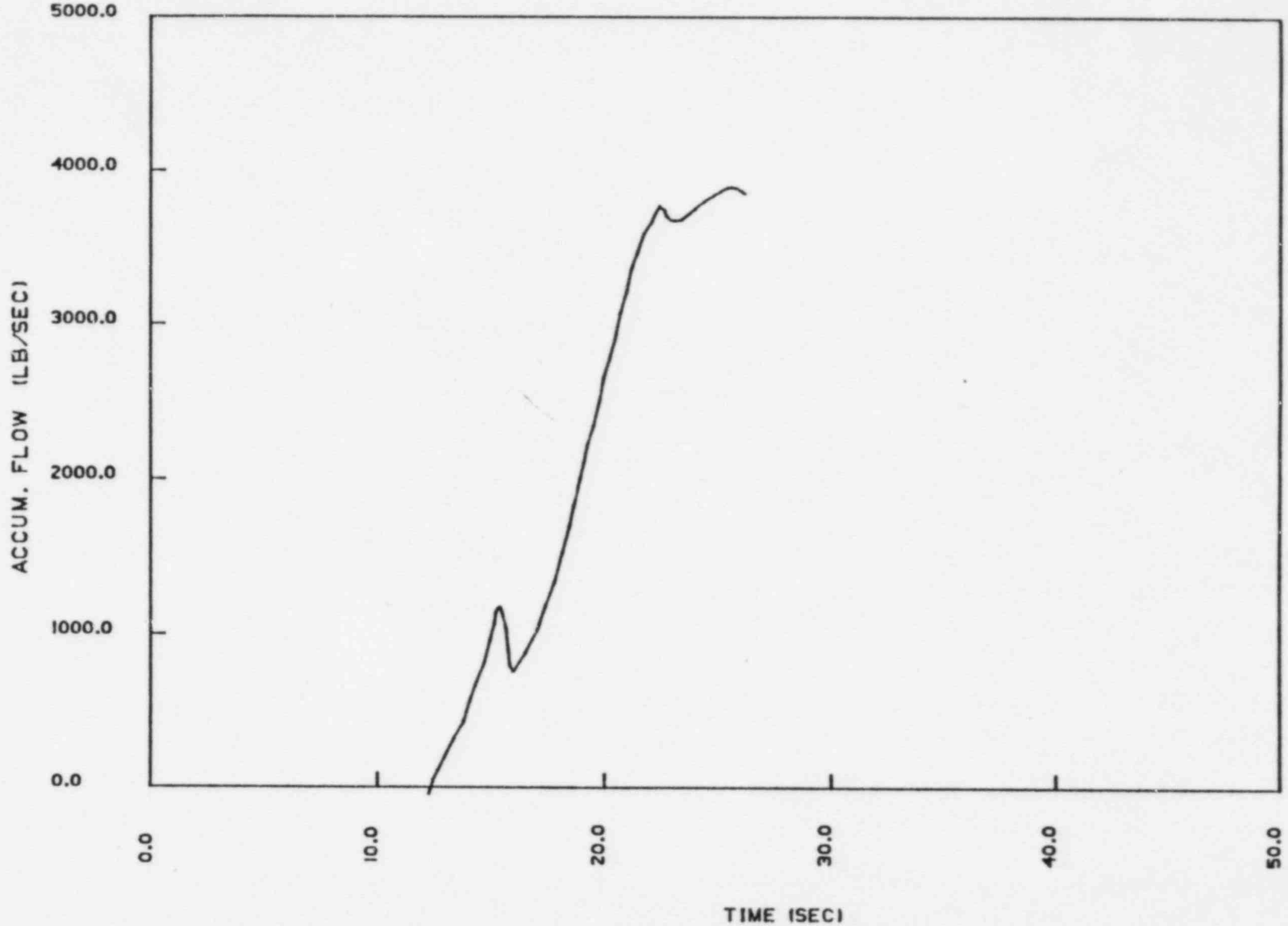


Figure 15.4-11C. Accumulator Flow (Blowdown) - DECLG (C_D =0.4)

Active Loop Break

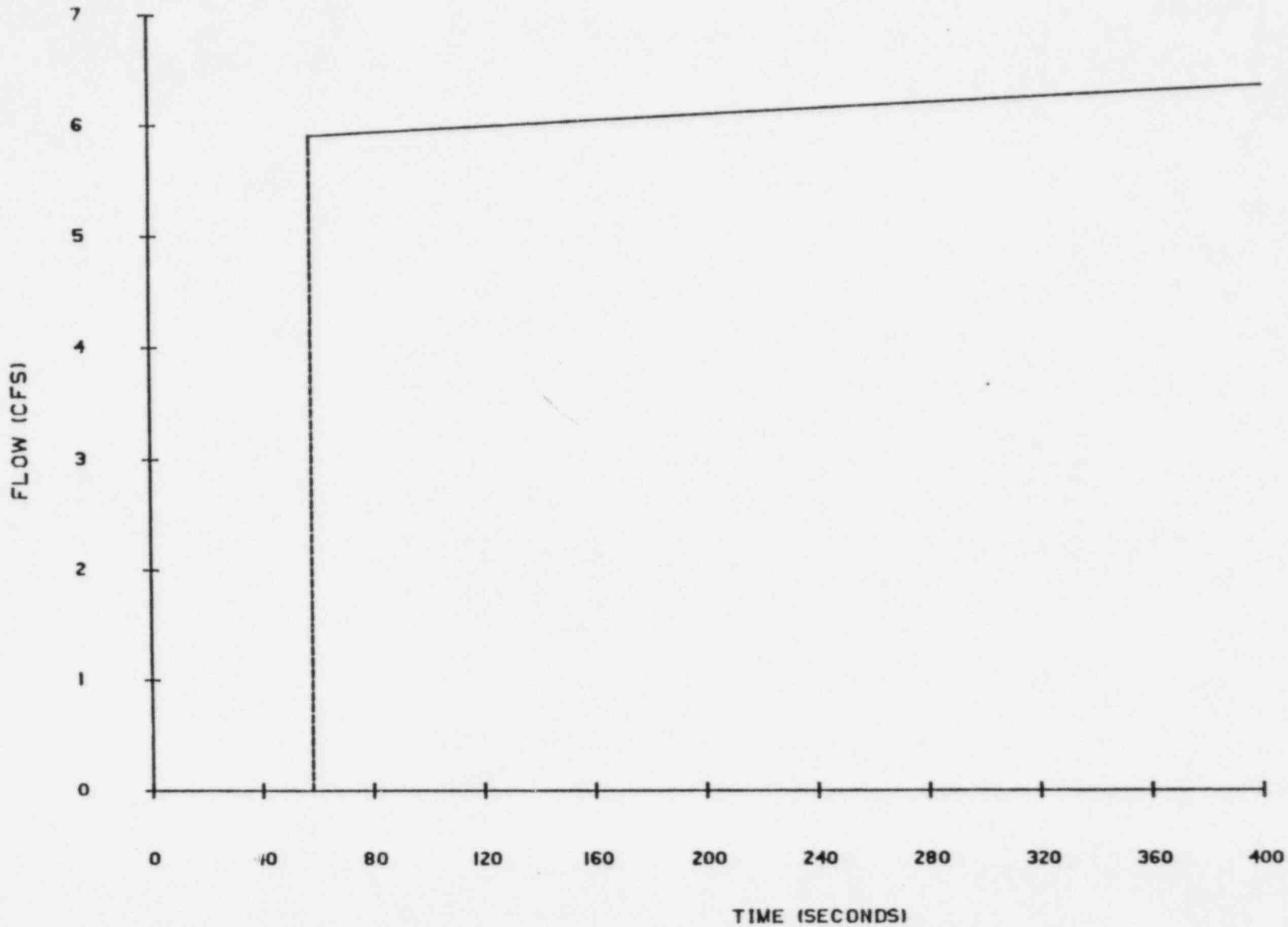


Figure 15.4-12A. Pumped Safety Injection Flow During Reflood: (C_D=0.4) DECLG, Inactive Loop Break

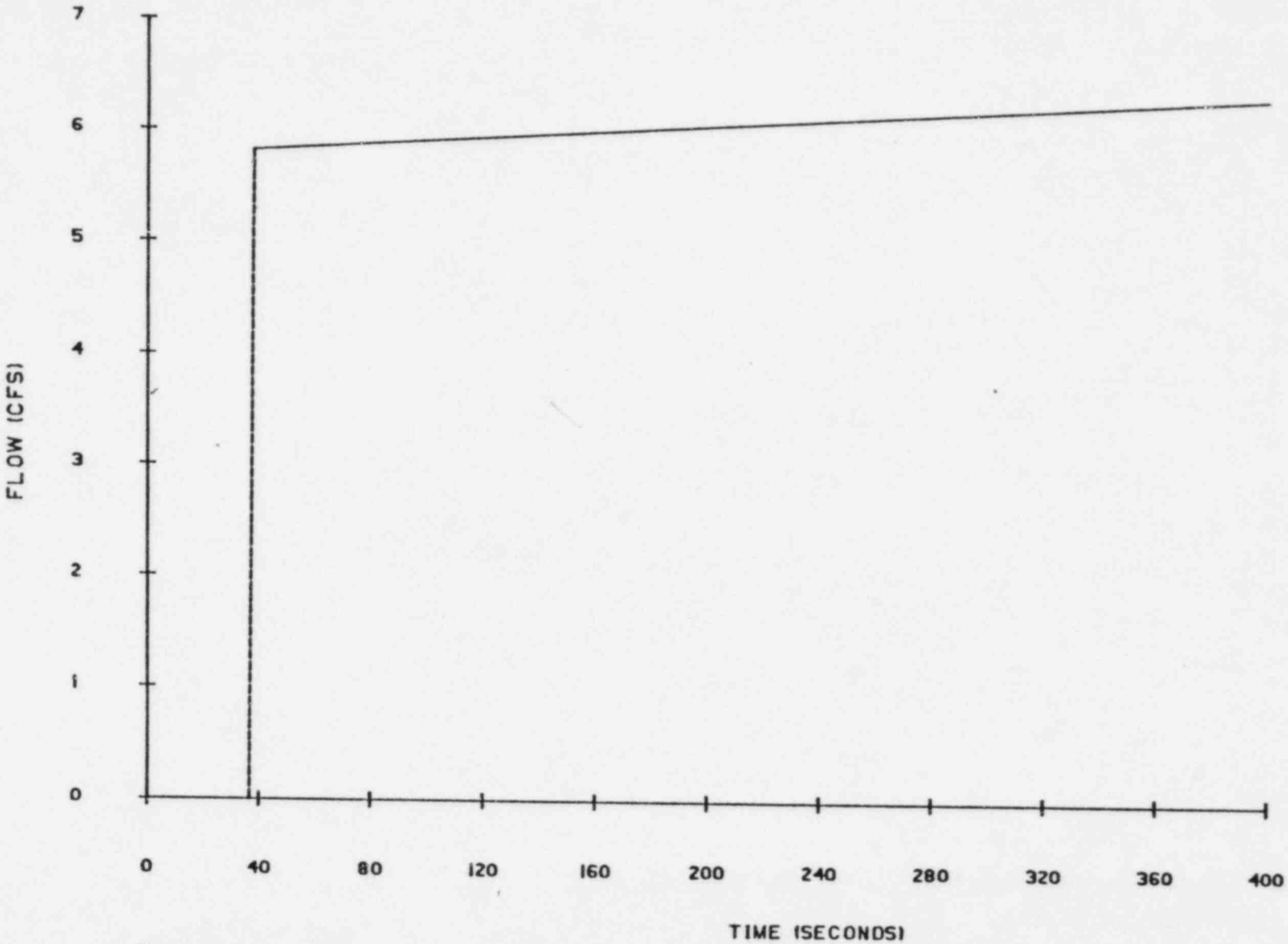


Figure 15.4-12B. Pumped Safety Injection Flow During Reflood: (C_D=0.6) DECLG, Active Loop Break

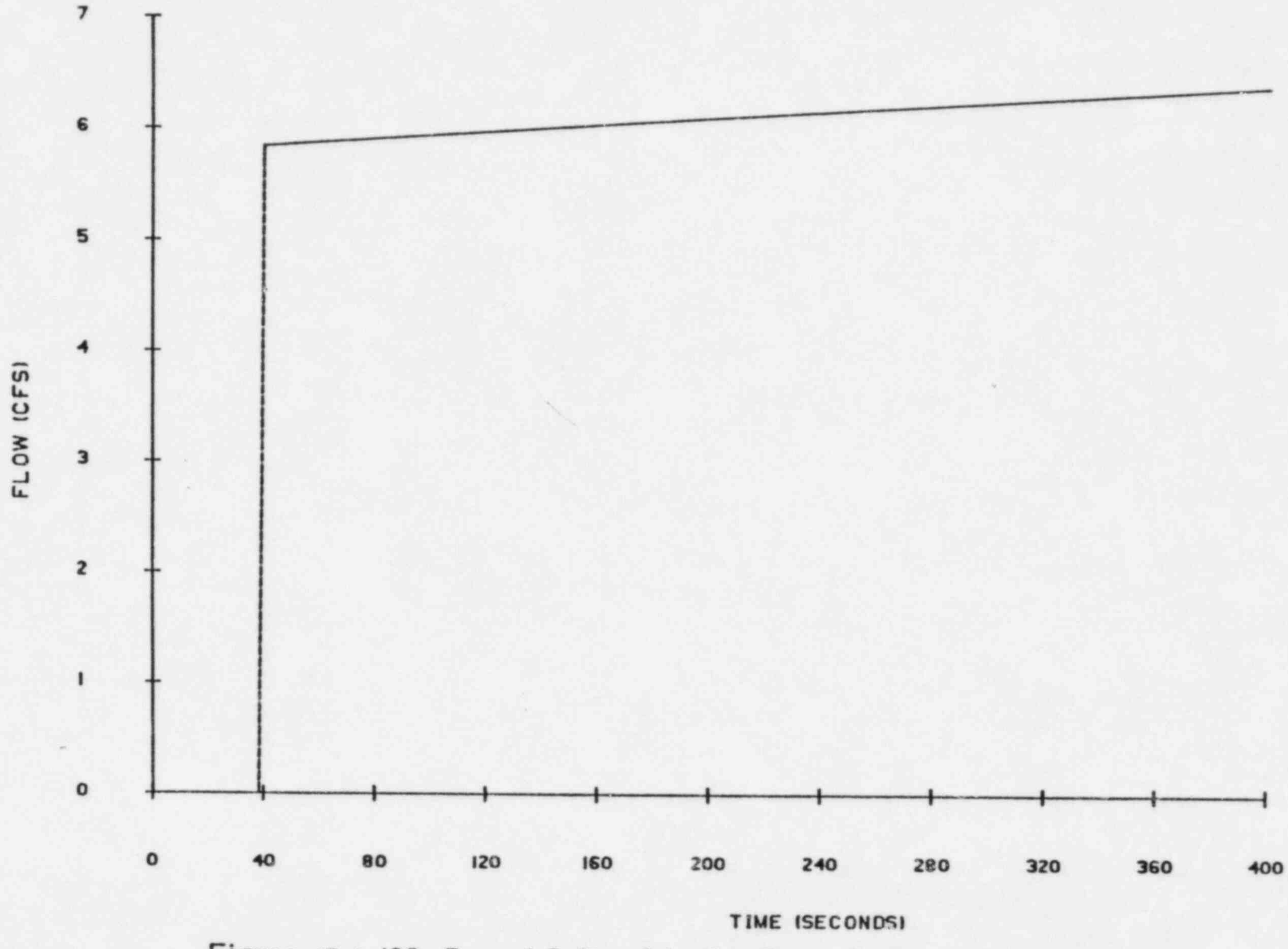


Figure 15.4-12C. Pumped Saftey Injection During Reflood: (C_D=0.4) DECLG, Active Loop Break

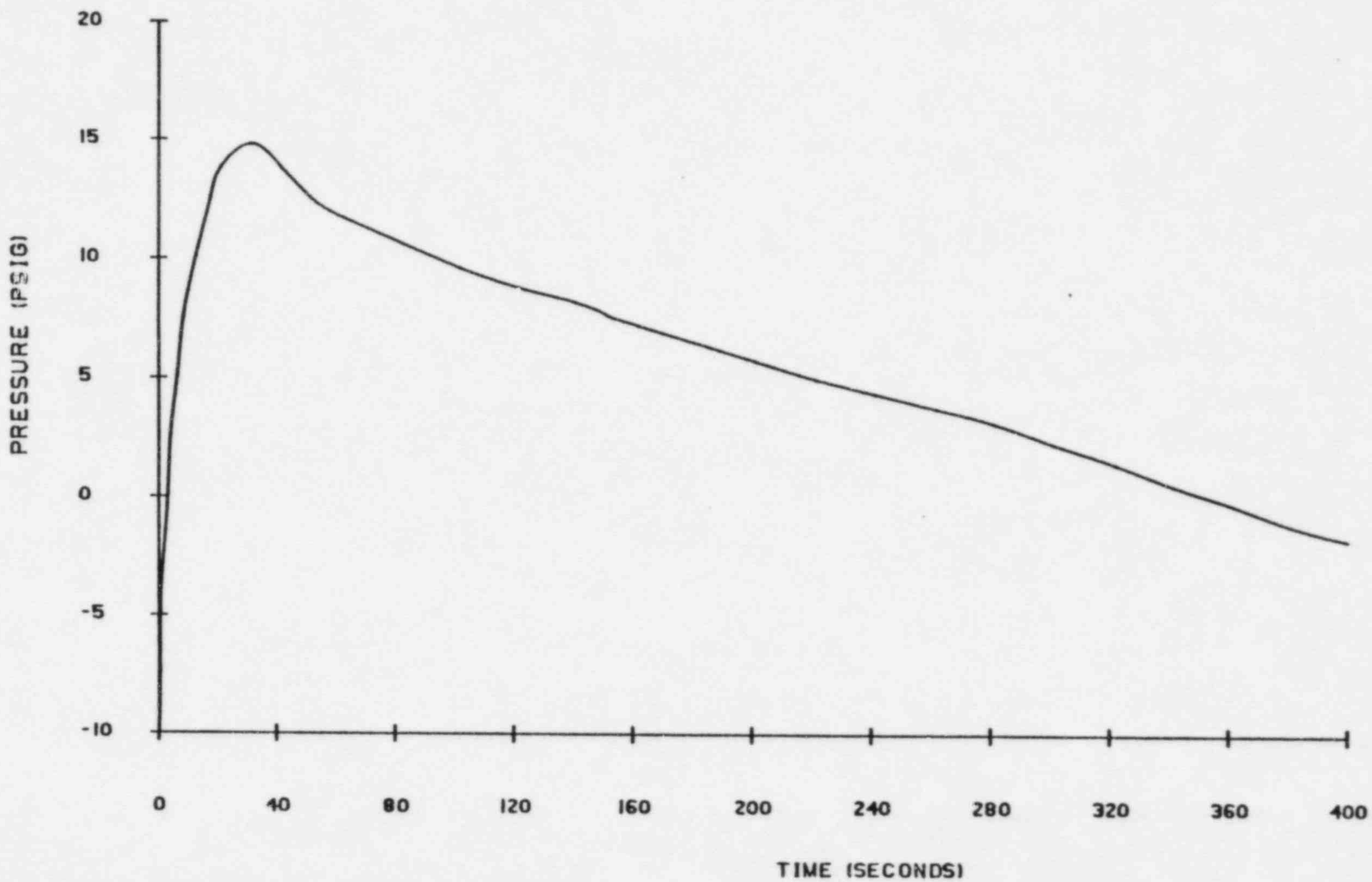


Figure 15.4-13A. Containment Pressure Transient, (C_D=0.4) Inactive Loop Break

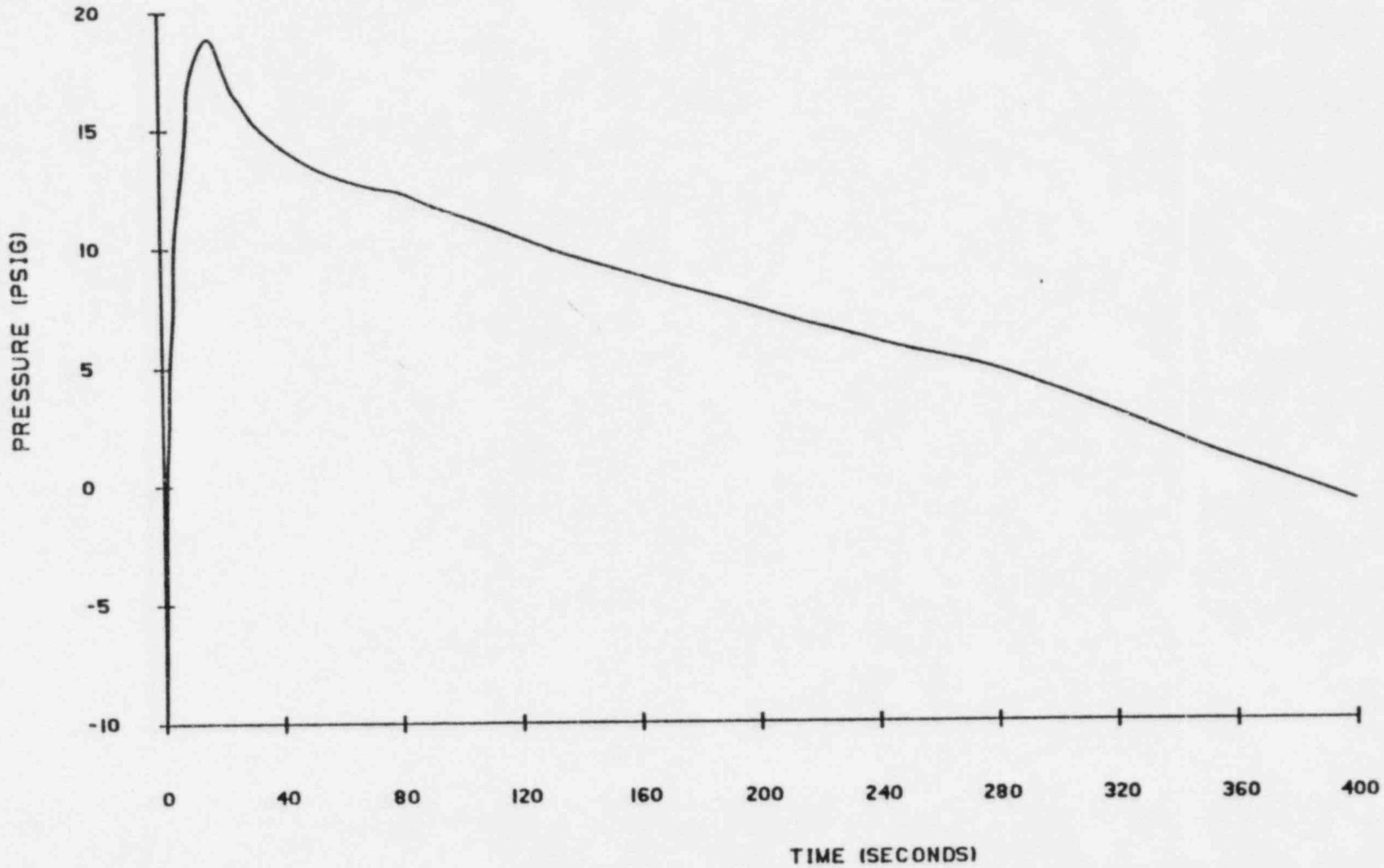


Figure 15.4-13B. Containment Pressure Transient, (C_D=0.6) DECLG Active Loop Break

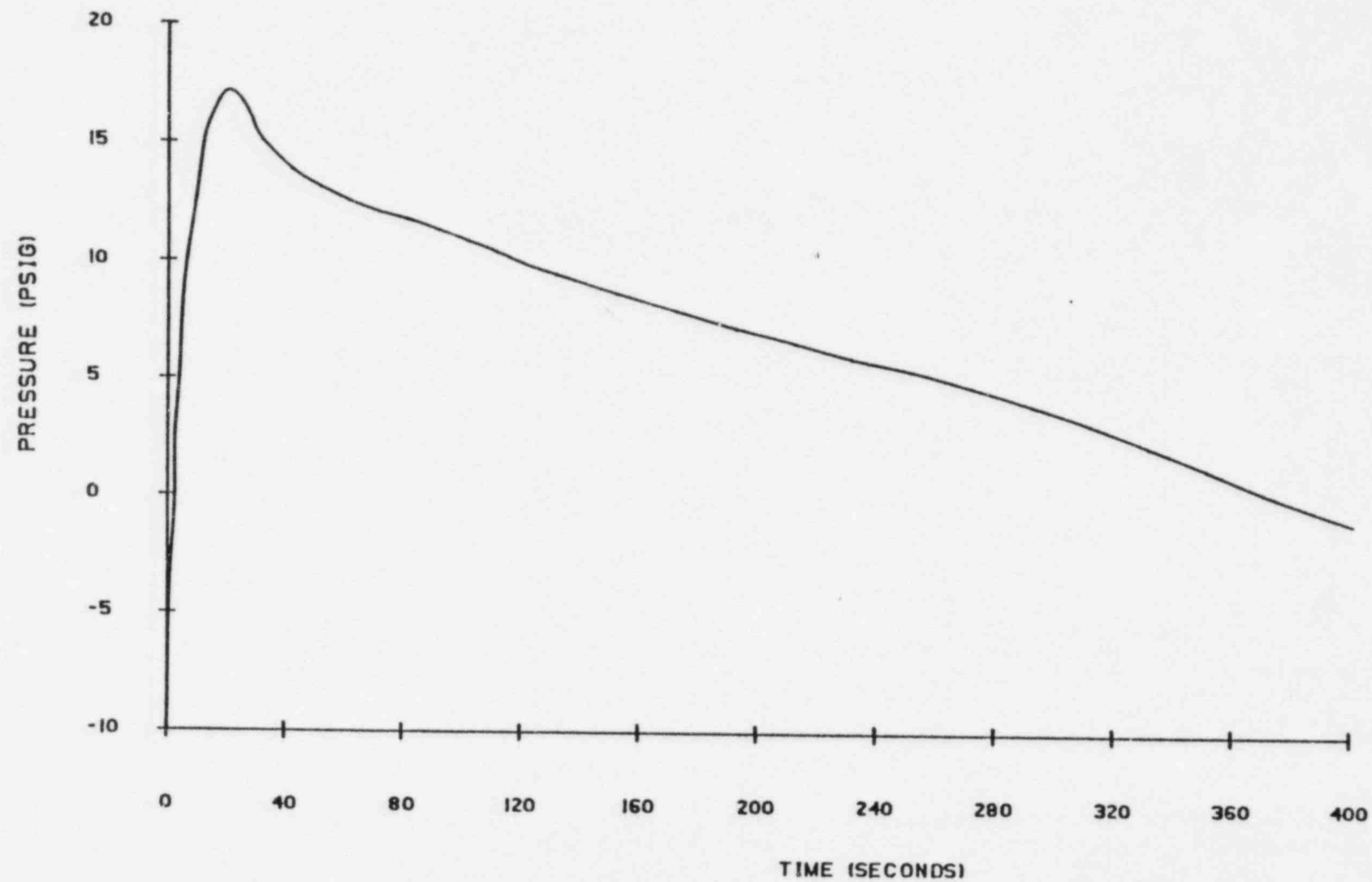


Figure 15.4-13C. Containment Pressure Transient, (C_D=0.4) DECLG Active Loop Break

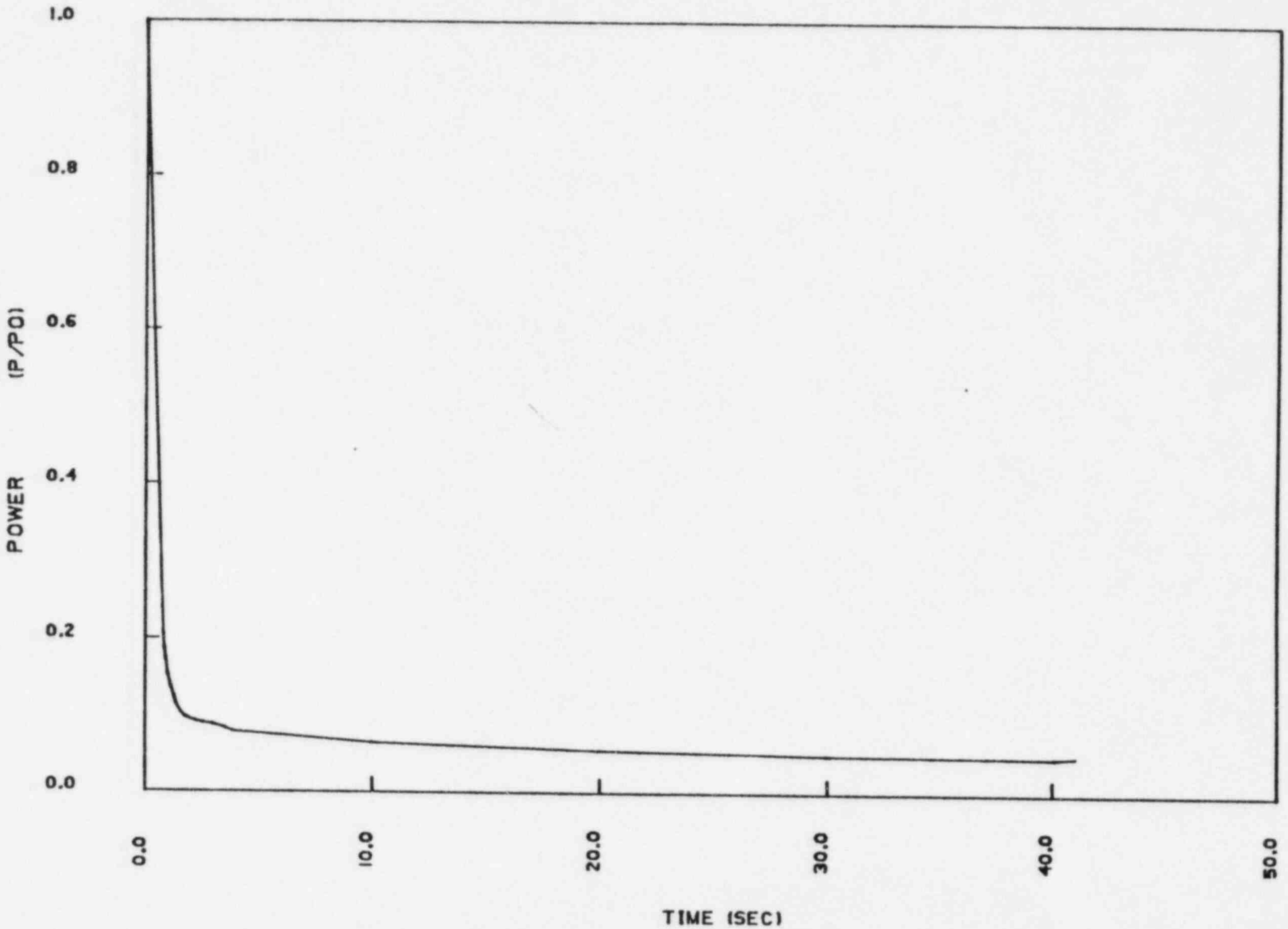


Figure 15.4-14A. Core Power Transient - DECLG (C_D=0.4)
Inactive Loop Break

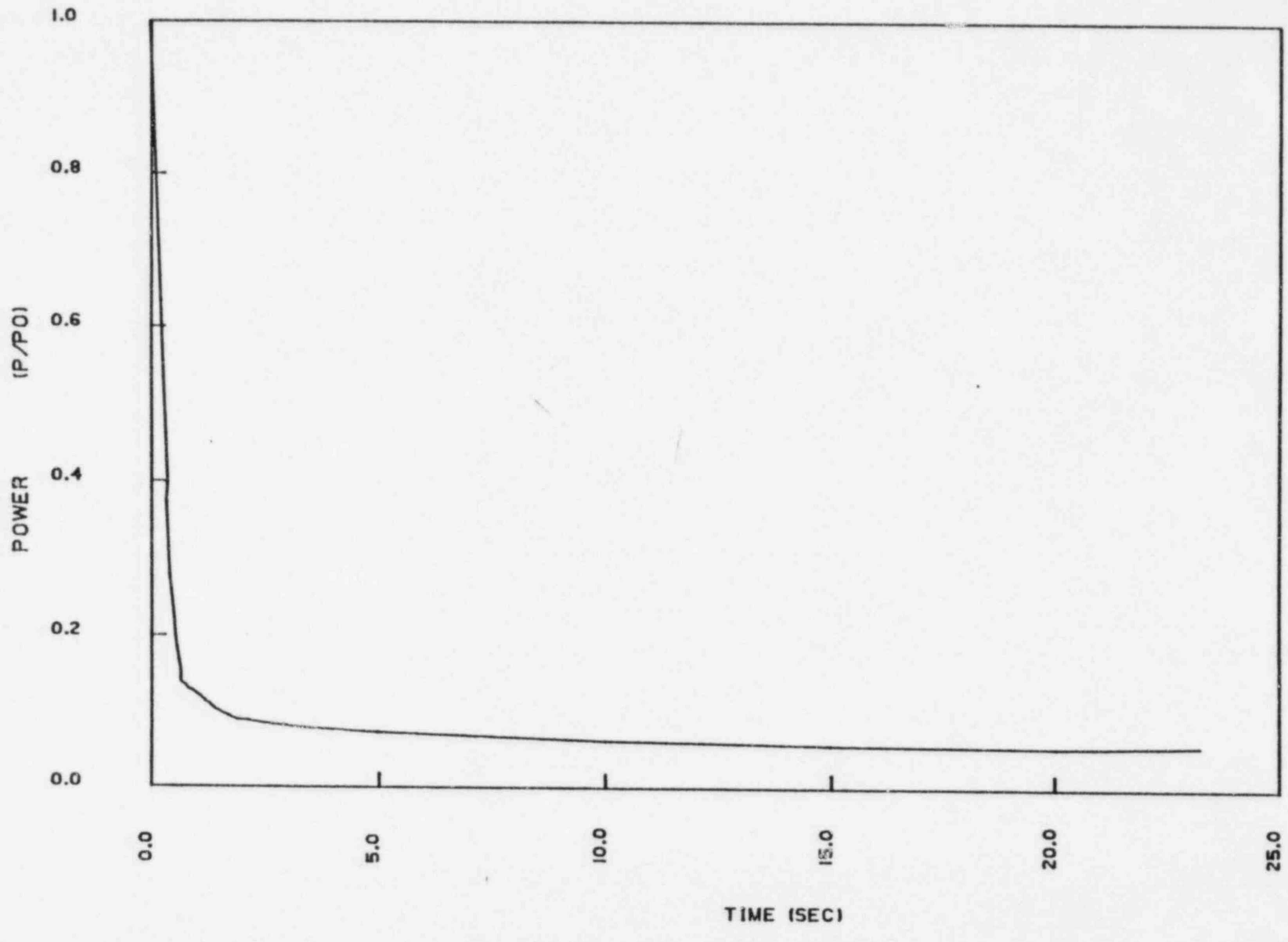


Figure 15.4-14B. Core Power Transient - DECLG (C_D =0.6)
Active Loop Break

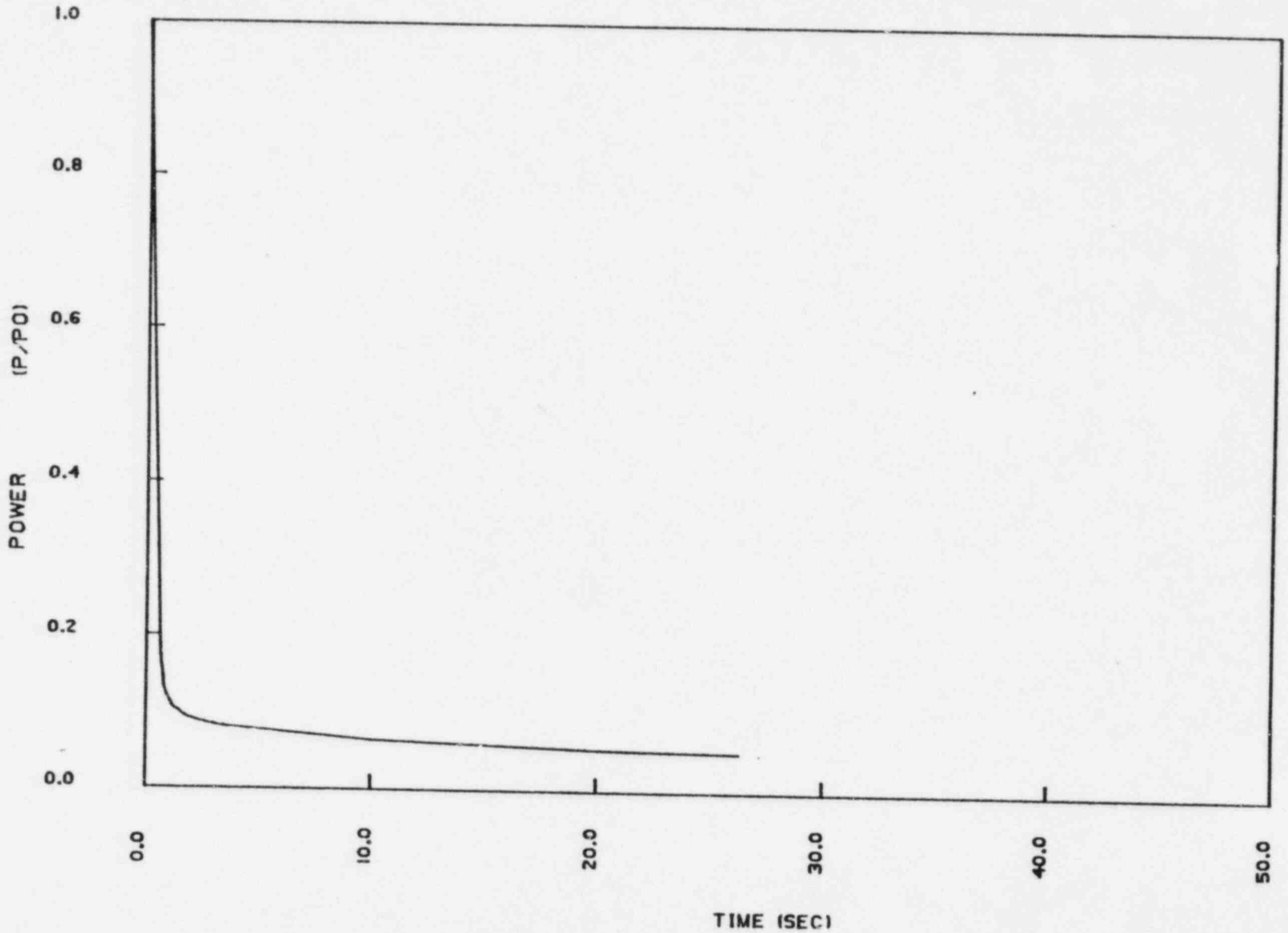


Figure 15.4-14C. Core Power Transient – DECLG (C_D=0.4)
Active Loop Break

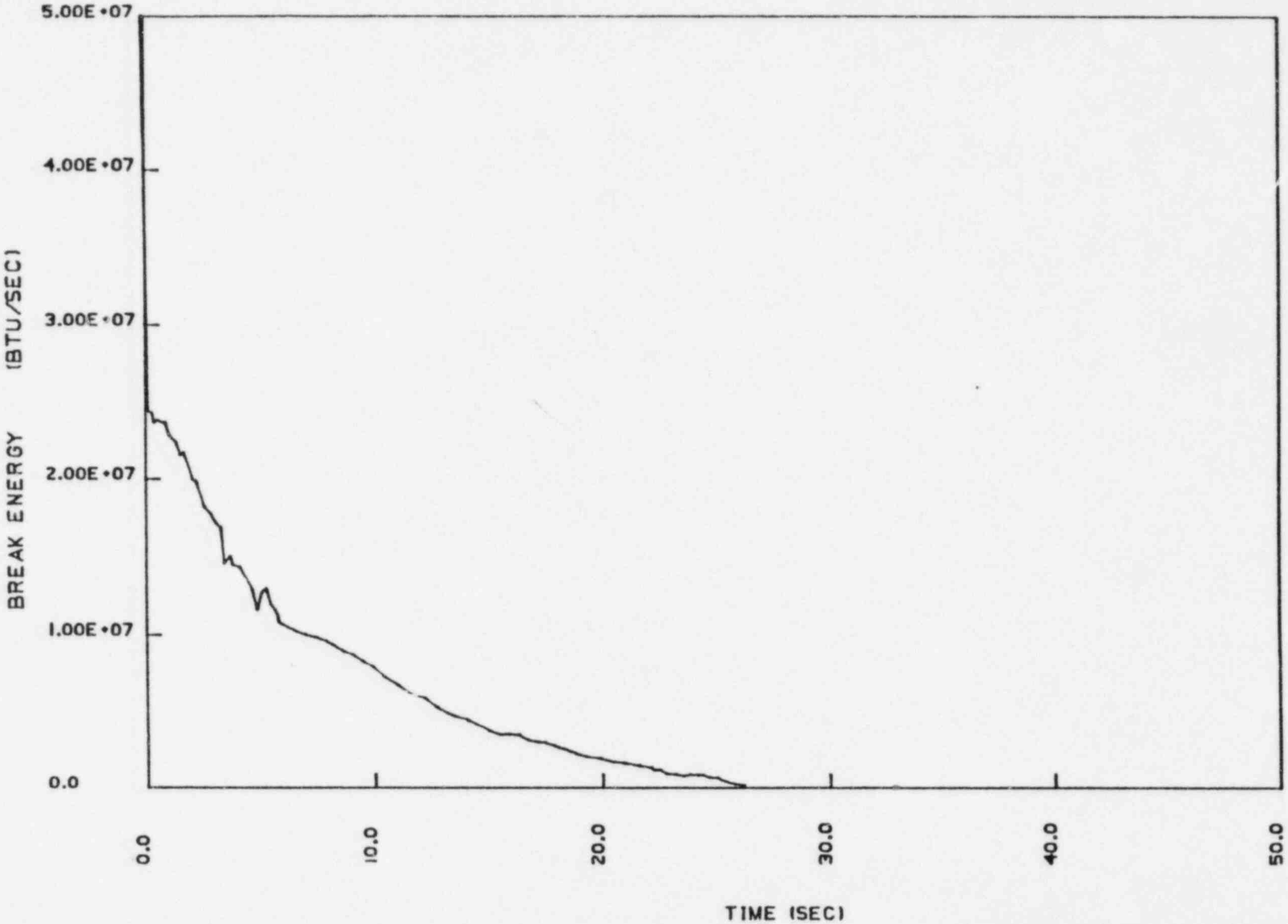


Figure 15.4-15. Break Energy Flow, (C_D =0.4) DECLG
Active Loop Break

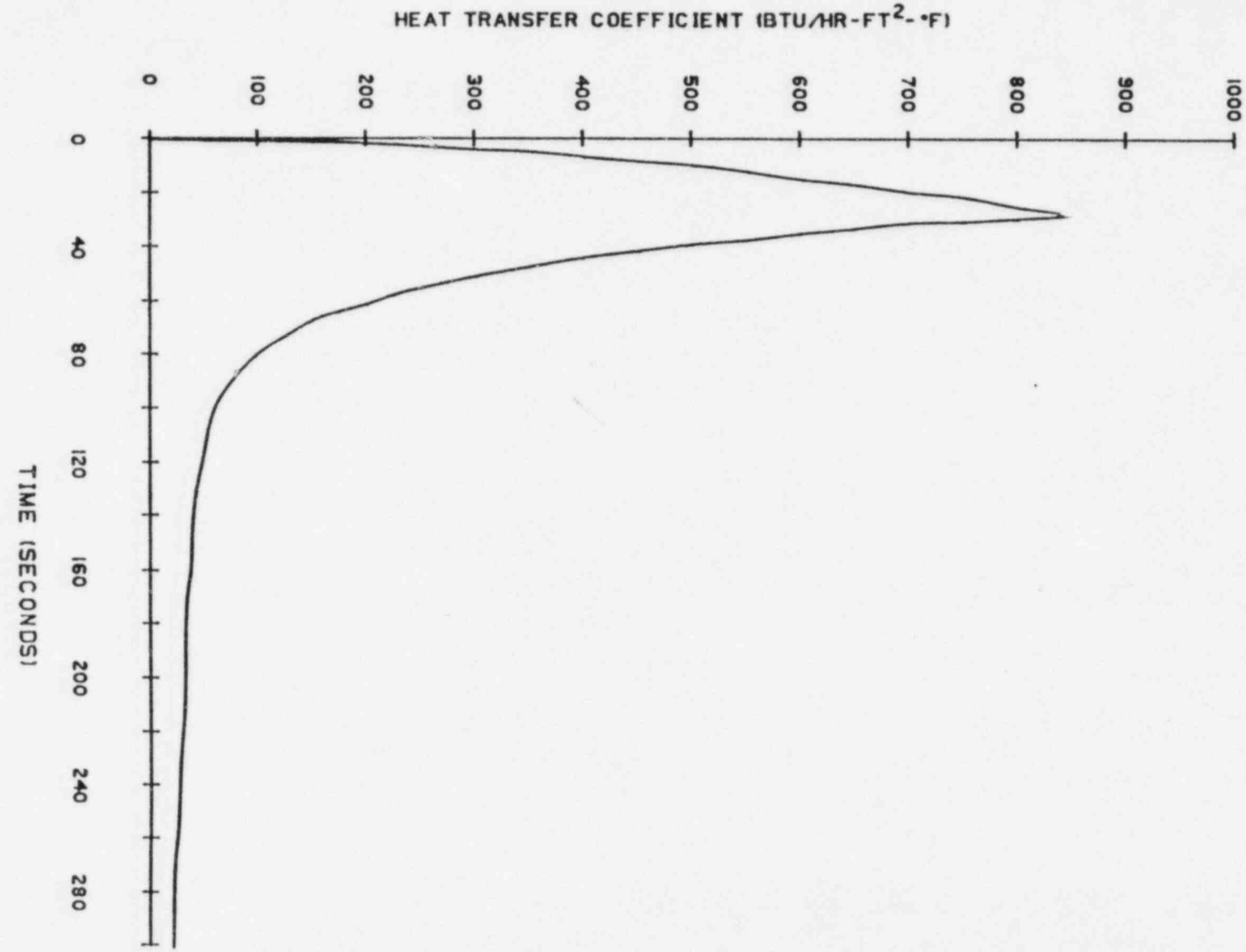


Figure 15.4-16. Containment Wall Condensing Heat Transfer Coefficient,
Active Loop Break, (C_D=0.4) DECLG

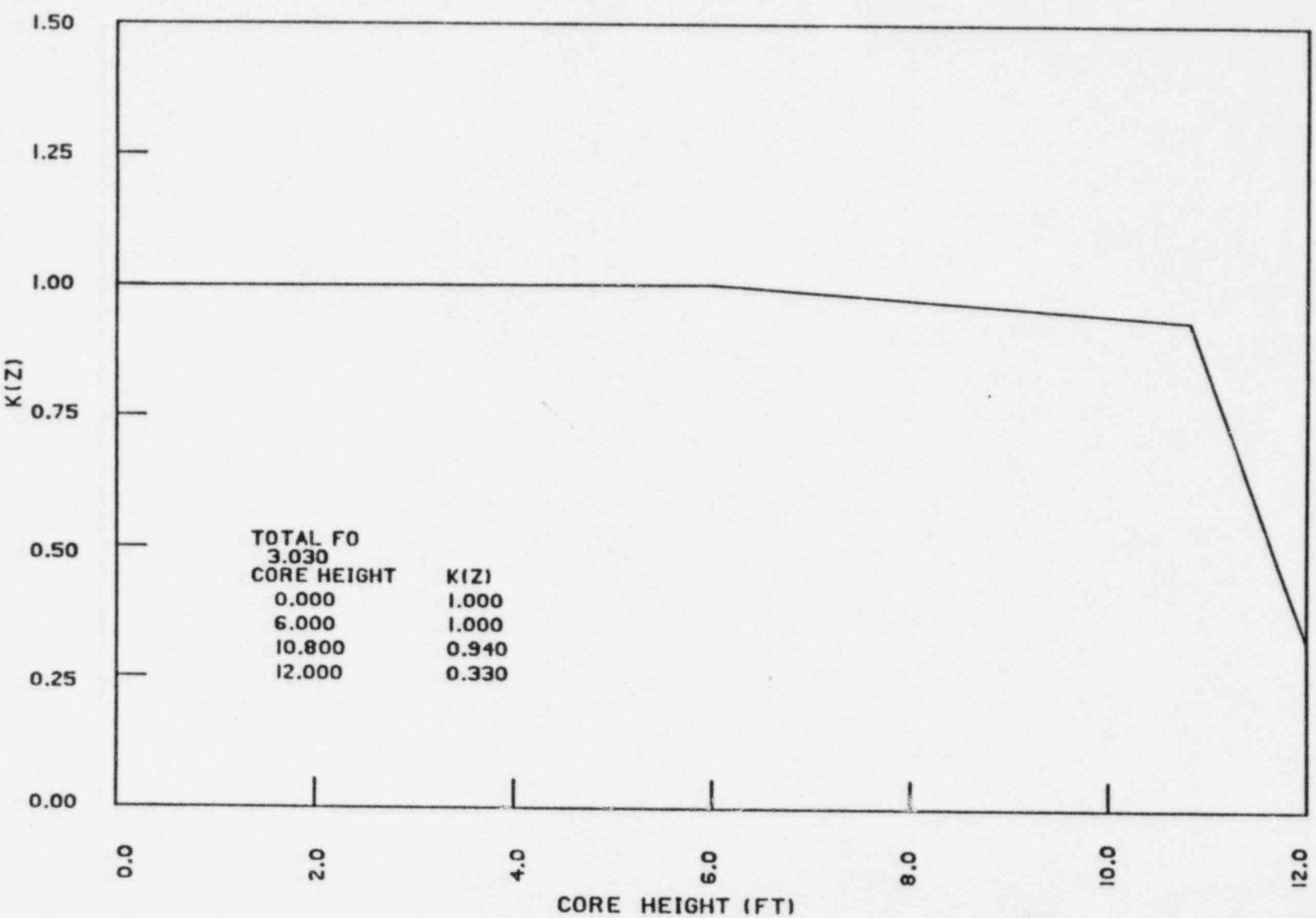


Figure 15.4-17. N-1 Loop K(Z) Envelope

ATTACHMENT B

ATTACHMENT B

LOCA RELATED TECH SPECS

N-1 LOOP OPERATION

Plant Name:	Beaver Valley Unit 1 (DLW)
Type/Date of LOCA Analysis:	CD = 0.4, 0.6 N-1 Active Loop Large Break CD = 0.4 N-1 Inactive Loop Large Break 1981 Model June 1983
Total Peaking Factor (FQT):	3.03
Cold Leg Accumulator Water Volume:	1025 cubic feet per accumulator (nominal)
Cold Leg Accumulator Gas Pressure:	600 psia (minimum)
K(Z) Curve:	See attached figure (15.4-17)