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| TO: Mr Lear | ORIG one signed | CC | OTHER | SENT NRC PDR XX<br>SENT LOCAL PDR XX | | |
| CLASS | UNCLASS XXXXXX | PROP INFO | INPUT | NO CYS REC'D 1 | DOCKET NO: 50-261 | |

DESCRIPTION:

Ltr trans the following:

ACKNOWLEDGED
DO NOT REMOVE

PLANT NAME: Robinson #2

ENCLOSURES:

Addl info for ECCS analysis....in response to our 6-18, 6-27, and their 3-14 submittal.....

(40 cys encl rec'd)

## FOR ACTION/INFORMATION 8-6-75 ehf

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CP&L
Carolina Power & Light Company

July 24, 1975





50-261

FILE: NG-3514 (R) SERIAL: NG-75-962



Mr. George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing
Office of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555



Dear Mr. Lear:

H. B. ROBINSON UNIT NO. 2
LICENSE NO. DPR-23
ADDITIONAL INFORMATION FOR ECCS ANALYSIS

In response to your letters of June 18, 1975 and June 27, 1975, the following additional information is filed in order to complete your evaluation of our ECCS submittal of March 14, 1975 for the H. B. Robinson Unit No. 2 Plant. In response to the requests, Carolina Power & Light Company submits information concerning submerged valve motors within containment, single failure or operator error causing any manually controlled, electrically operated valve to move to a position that could adversely affect the ECCS, additional information concerning the prevention of boron precipitation following a LOCA, a reanalysis of the large break LOCA ($C_D$ = 1.0) with the approved March 15, 1975 Westinghouse evaluation model, justification of the fuel region used in the ECCS analysis, and additional information on LOCA requested by the NRC Staff.

In regard to your concern on submerged valves, a thorough investigation of piping layout drawings supplemented by visual inspection of the Robinson containment reveals that no electric motor-operated valves would be submerged following a postulated loss of coolant accident. Thus, no modifications or other qualifications of ECCS performance during short-term or long-term cooling are required.

In regard to single failure or operator error causing adverse movement of an electrically operated, manually controlled valve, CP&L has studied the ECCS operation and plant procedures and has determined that the following valves will be de-energized at their respective motor control centers: 862 A&B, 863 A&B, 864 A&B, 865 AB&C, and 878 A&B in the open positions; and 866 A&B in the closed position. De-energization of these valves will be maintained during normal plant operations except for periodic tests of the valves, or associated valves and parts of the ECCS in communication with the valves, when they will be re-energized to provide proper



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valve lineups for the test condition. Following a LOCA event, positions of some of these valves must be changed in order to provide for long-term cooling and injection capability to prevent boron precipitation. The valves will be re-energized appropriately to provide the proper valve lineups. The air supply to Valves 605 and 758 will also be shut off to assure the valves remaining closed during the post-accident condition.

Operators on shift will be assigned responsibility for the motor control centers such that, following a LOCA, they can re-energize the above valves as required and enable repositioning of the valves for long-term cooling. This is an interim measure until a modification can be performed to the motor control circuitry to enable control from the control room and still satisfy the single failure/operator error criteria.

With respect to prevention of boron precipitation in the vessel following a LOCA, certain modifications to our submittal of April 18 are in order following review of later information transmitted to you on plants similar in design to the Robinson Plant. First, to assure that motor-operated Valves 866 A&B, 750 and 751 can be operated in the post-LOCA environment and assure pumped flow into the RCS hot leg, these valves will be required to be opened within two hours following a LOCA. Injection into the hot legs will be prohibited by blocking the flow paths with closure of valves outside the containment. These valves will later be opened at approximately 18 hours after the initiation of the accident to assure delivery to the RCS hot leg and, thus, avoid high concentrations of boric acid in the reactor vessel which would approach solubility limits. Valves 744 A&B have operators which were converted to Class "H" operators and are specifically designed to operate after long-term exposure to the lost-LOCA environment.

To assure long-term cooling capability incorporating considerations for passive failures, Emergency Instruction EI-1 will be modified to incorporate the above changes in system lineups. The modifications to the procedure are attached for your information.

Also attached are revisions to the ECCS analyses submitted on October 2, 1974 and March 14, 1975 which incorporate the use of the March 15, 1975 Westinghouse evaluation model to calculate the LOCA transient for the double-ended cold leg guillotine break with a Moody discharge coefficient of 1.0. The peak clad temperature of 1795°F is well below the limit of 2200°F and is only 21°F greater than the value obtained with the December 27, 1974 model. Included with this attachment is a justification of the use of Fuel Region 5 as the limiting region in the analyses.

Additional attachments respond to informal staff questions and supply information pertinent to computation of reactor vessel boric acid concentrations, the effects of reactor coolant pumps running on the resultant

peak clad temperature following a LOCA, and the amount of time available for switchover from the injection phase to the recirculation phase following a design basis LOCA.

We trust this information is suitable for your use.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:bn

Attachments

CC: Messrs. N. B. Bessac
P. W. Howe
J. A. Jones
R. E. Jones
W. B. Kincaid
J. B. McGirt
D. B. Waters

4. Verify with the monitor lights that the valves associated with safety injection are in the position after "S" initiation. (Light indicates when a valve is in safeguard position.) Valves SI-865A, B & C will not indicate since the breakers for these valves are locked open.

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PAGE 11 & 11a

5.

(b) The hot leg injection valves to the reactor coolant loops SI-866A and SI-866B do not receive an "S" signal. The breakers for the valves are locked in the open position. These valves will only be operated and used as stated in Appendix A. 4.b page _______ and Appendix A, 5 page _____.

(d) Delete because the breakers for these valves are locked in the open position with the valves open.

(e) Change "e" to "d" because "d" has been deleted.

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APPENDIX A

DETAILED RECOVERY PROCEDURE - LOSS OF REACTOR COOLANT

A. Discussion

The objectives of this procedure are as follows:

1. Provide emergency core cooling to prevent damage to the fuel cladding and release of excessive radioactivity.

2. Provide long-term control and cooldown of the reactor by recirculation of spilled reactor coolant, injected water, and containment spray system drainage.

3. Adjust the chemistry of the spray solution to increase the effectiveness of iodine removal from the containment atmosphere.

B. Symptoms (Unique to Loss of Coolant)

1. Rising containment pressure.

2. High containment radiation alarm.

3. Rising water level in the containment sump.

C. Immediate Actions

Automatic Actions

Refer to the automatic actions following the generation of the safety injection signal and the high-high containment pressure signal.

Manual Actions

1. Injection phase

(a) Manually reset safety injection as soon as the startup sequencing of safeguards equipment is completed.

(b) Verify valve position SI-863 A&B, SI-860 A&B, SI-861 A&B, SI-841 A&B and SI-855 closed.

SI-856 A&B, RHR-744 A&B, SI-870 A&B, SI-867 A&B and SI-869 open.

(c) If two residual heat removal pumps are operating, stop one for pump protection if the reactor coolant system pressure is still above 130 psig 15 minutes after the accident.

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(d) Check the status of the control room ventilation to ensure that control room intake duct isolation has occurred and that emergency recirculation has been initiated.

(e) Stop the diesel generators provided outside power has not been interrupted.

2. <u>Small Break</u>

In case of a small break, characterized by a slowly decreasing RCS pressure, additional actions can be taken to help RCS depressurization.

(a) Sample each steam generator secondary side.

(1) Manually reset containment isolation signal.

(2) Open the blowdown sample isolation valves.

(3) Observe the sample line radiation monitor RM-19.

(4) If the blowdown sample isolation valves automatically close due to high radiation level, determine which steam generator is leaking, isolate it and dump steam from the other steam generators.

(b) If activity level is acceptable, transfer steam dump control from $T_{avg}$ control to pressure control.

(c) If the condenser is not available, use atmospheric steam relief.

(d) Slowly increase the rate of steam dump and begin cooldown of the reactor coolant system after RCS has been borated to the cold shutdown condition.

(e) The reactor coolant pumps can continue to run if the following conditions are met:

(1) The reactor coolant pressure is within the limits shown by the pressure temperature curve, Figure GP-3 of operating instruction GP-1.

(2) Component cooling water and seal injection flow have not been lost.

(3) The No. 1 seal leak-off rate is normal.

(f) Refer to GP-1D.

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(d) Check the status of the control room ventilation to ensure that control room intake duct isolation has occurred and that emergency recirculation has been initiated.

(e) Stop the diesel generators provided outside power has not been interrupted.

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(2) Component cooling water and seal injection flow have not been lost.

(3) The No. 1 seal leak-off rate is normal.

(f) Refer to GP-1D.

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## 3. Large Break

(a) Secure the turbine in accordance with normal procedure.

(b) For a coincident station blackout the following additional actions are required:

(1) Verify automatic start of steam-driven auxiliary feedwater pump.

(2) Ensure that both battery chargers are operating.

(c) When the refueling water storage tank low level alarm is actuated, stop one safety injection pump if three are operating and stop one containment spray pump if two are operating.

(d) Dispatch people to MCC5 and 6 and to the RHR pit and establish communications.

Note: These people are to operate breakers for SI-864 A&B, 862 A&B, 866 A&B and to close valves RHR-752 A&B. These breakers have been locked in the open position to prevent a single failure or operator error causing adverse movement of an electrically-operated manually-controlled valve.

(e) When the refueling water storage tank low-low level alarm is actuated, terminate injection flow as follows:

(1) Stop the operating safety injection containment spray and residual heat removal pumps.

(2) Close valves SI-856 A&B.

Note: The time between stopping the injection phase and starting the RHR, HHSI and spray pumps on recirculation phase should be $<10$ minutes to avoid uncovering the core and loss of cooling capability.

## 4. Initial Recirculation Phase

(a) Have the breakers for the following valves unlocked and closed: SI-864 A&B, SI-862 A&B, SI-866 A&B.

(b) Close SI-864 A&B, SI-862 A&B and SI-869.

(c) Open SI-866 A&B after SI-869 is shut.

(d) Visually verify the position of SI-869.

(e) Have the breakers for the following valves opened and locked open: SI-864 A&B, SI-862 A&B, SI-869 and SI-866 A&B.

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3. Large Break

(a) Secure the turbine in accordance with normal procedure.

(b) For a coincident station blackout the following additional actions are required:

(1) Verify automatic start of steam-driven auxiliary feedwater pump.

(2) Ensure that both battery chargers are operating.

(c) When the refueling water storage tank low level alarm is actuated, stop one safety injection pump if three are operating and stop one containment spray pump if two are operating.

(d) Dispatch people to MCC5 and 6 and to the RHR pit and establish communications.

Note: These people are to operate breakers for SI-864 A&B, 862 A&B, 866 A&B and to close valves RHR-752 A&B. These breakers have been locked in the open position to prevent a single failure or operator error causing adverse movement of an electrically-operated manually-controlled valve.

(e) When the refueling water storage tank low-low level alarm is actuated, terminate injection flow as follows:

(1) Stop the operating safety injection containment spray and residual heat removal pumps.

(2) Close valves SI-856 A&B.

Note: The time between stopping the injection phase and starting the RHR, HHSI and spray pumps on recirculation phase should be $<10$ minutes to avoid uncovering the core and loss of cooling capability.

4. Initial Recirculation Phase

(a) Have the breakers for the following valves unlocked and closed: SI-864 A&B, SI-862 A&B, SI-866 A&B.

(b) Close SI-864 A&B, SI-862 A&B and SI-869.

(c) Open SI-866 A&B after SI-869 is shut.

(d) Visually verify the position of SI-869.

(e) Have the breakers for the following valves opened and locked open: SI-864 A&B, SI-862 A&B, SI-869 and SI-866 A&B.

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(f) Close the RHR pump suction valves RHR-752 A&B. (Manual valves located in RHR pump pit.)

(g) Close RHR-743 (RHR pump miniflow line isolation valve, located in overhead in pipe alley).

(h) Open RHR-750 and RHR-751 (to establish a backup flow path to hot leg).

Note: At a time not greater than two hours after the accident perform the following valve operations.

(a) Shut SI-869 and verify its position by a local visual check.

(b) Unlock the circuit breakers for SI-866 A&B to restore power to the motor operators and open SI-866 A&B.

(c) Manually close RHR-743.

(d) Open RHR-750 and RHR 751.

Do not perform steps (c) and (d) unless the plant is in the recirculation phase.

Note: Valves RHR-752 A&B and RHR-743 should be closed prior to restarting RHR pumps, if possible, because of possible high radiation levels.

(i) Verify that the cooling water low flow alarms for the residual heat removal, containment spray and safety injection pumps are not actuated.

(j) Open CCW-749 A&B and start the second component cooling water pump if it is not running.

(k) Open SI-863 A&B.

(l) Close RHR-744 A&B.

(m) Open SI-860 A&B and SI-861 A&B.

(n) Start A or B residual heat removal pump and two safety injection pumps to establish recirculation flow to the cold legs.

(o) Check the recirculation flow on FI-940 and FI-943.

(p) Start A or B containment spray pump and control flow at approximately 1160 gpm as read on FI-958 A by throttling the pump discharge valves by opening the breakers when the desired flow is reached.

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(q) Throttle SI-845 C until a reading of 12 gpm is obtained on FI-949.

(r) When the level in the spray additive tank reaches 0%, close SI-845 C or SI-845 A and B.

(s) Manually reset containment isolation by depressing the reset button.

(t) Weekly sample recirculation loop fluid to determine solution boron concentration and pH; make adjustments as necessary.

5. Eighteen Hours after the Accident

(a) Close SI-867 A&B and SI-870 A&B.

(b) Open SI-869.

(c) Start the second residual heat removal pump if operable; if not, go to step #6.

(d) Open RHR-744 A&B, then throttle these valves to give approximately 2250 gpm flow as read on FI-605. This may be accomplished by pulling the breakers when the desired flow is obtained.

Note: The above is done to reduce the concentration of boron in the reactor vessel which resulted from a leak in the cold legs. It has been estimated that the boron concentration would be approximately 23% eighteen hours after the accident.

(e) Continue containment spray.

6. Loss of the Second Residual Heat Removal Pump

In the event that the second residual heat removal pump is not operable, inject water into the hot legs thru valve SI-869 for twelve hours and then inject into the cold legs by opening SI-867 A&B and SI-870 A&B, then close SI-869. Alternate between the hot and cold legs every twelve hours until the second residual heat removal pump is operable, then return to step #5.

7. Loss of High Head Hot Leg Recirculation

In the event that a passive failure occurs and the high head hot leg injection is isolated by closing SI-869, proceed to initiate hot leg injection via the low head injection system. The RHR system valves have previously been aligned as follows: RHR-750 and 751 open, RHR-743 closed, and RHR-752 A&B closed. Open valve RHR-743 (residual heat removal minimum recirc. valve) and the flow of water would be thru valves RHR-750 and RHR-751 to loop #2 hot leg. During this operation, injection to the cold legs would be maintained using the high head recirculation path. Valves RHR-744 A&B would be closed during this mode of hot leg injection. Continue using containment spray.

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8. Loss of High Head Recirculation

In the event that there is not any safety injection pumps operable, use the residual heat removal pumps for recirculation thru valves RHR-744 A&B to the cold legs. If it is eighteen hours after the accident and hot leg injection is needed, alternate every twelve hours using step #7.

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H. B. ROBINSON STEAM ELECTRIC PLANT
UNIT NO. 2
EMERGENCY INSTRUCTION #1
APPENDIX "A" CHECKOFF SHEET

1. Injection Phase

(a) Safety injection reset after startup of safeguards equipment.

(b) Verify valve position.

| | | |
|---|---|---|
| | SI-863 A closed | ______ |
| | SI-863 B closed | ______ |
| | SI-860 A closed | ______ |
| | SI-860 B closed | ______ |
| | SI-861 A closed | ______ |
| | SI-861 B closed | ______ |
| | SI-841 A closed | ______ |
| | SI-841 B closed | ______ |
| | SI-855 closed | ______ |
| | SI-856 A open | ______ |
| | SI-856 B open | ______ |
| | RHR-744 A open | ______ |
| | RHR-744 B open | ______ |
| | SI-870 A open | ______ |
| | SI-870 B open | ______ |
| | SI-867 A open | ______ |
| | SI-867 B open | ______ |
| | SI-869 open | ______ |

(c) Stop one RHR pump if two are operating. ______

(d) Control room intake duct isolation has occurred. ______

(e) Stop diesel generators if outside power is available. ______

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2. Small Break

(a) Sample each steam generator secondary side and log results in control operators log. ______

(1) Reset containment isolation signal. ______

(2) Open S.G. blowdown valves. ______

(3) R-19 is not alarming. ______

(4) If R-19 is alarming, determine which S.G. is leaking and isolate it and dump steam from the other S.G.

S.G. Leaking # ______

(b) S.G. activity level acceptable. Steam dump control transferred from $T_{avg}$ control to pressure control. ______

(c) Is the condenser available for steam dump?

If not, use atmospheric steam relief. ______

(d) RCS has been borated to cold shutdown condition. ______

(e) (1) RCS pressure is within limits shown by pressure-temperature curve Figure GP-3 of GP-1. ______

(2) Component cooling and seal injection to RCP's have not been lost. ______

(3) No. 1 seal leak-off rate is normal. A___ B___ C___ ______

(f) Proceed to GP-1D for cooldown. ______

3. Large Break

(a) Turbine secured and on turning gear. ______

(b) For a coincident station blackout.

(1) Steam-driven auxiliary feedwater pump running. ______

(2) A&B battery chargers operating. ______

(c) Refueling water storage tank low level alarm actuated. ______

Stop one of three safety injection pumps. ______

Stop one of two containment spray pumps. ______

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(d) Dispatch people to: MCC #5 ______

MCC #6 ______

RHR Pit ______

Communications established. ______

(e) Refueling water storage tank low-low level alarm actuated. ______

Stop operating safety injection pumps. ______

Stop operating residual heat removal pumps. ______

Stop operating containment spray pumps. ______

Close SI-856 A. ______

Close SI-856 B. ______

4. Initial Recirculation Phase

(a) Breakers for the following valves unlocked and closed:

SI-864 A ______

SI-864 B ______

SI-862 A ______

SI-862 B ______

SI-866 A ______

SI-866 B ______

(b) Close valves SI-864 A ______

SI-864 B ______

SI-862 A ______

SI-862 B ______

SI-869 ______

(c) After SI-869 is closed, open SI-866 A ______

open SI-866 B ______

(d) Visually verify position of SI-869. ______

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| | | |
|---|---|---|
| (e) | Breakers for the following valves are open and locked open: | |
| | SI-864 A | |
| | SI-864 B | |
| | SI-862 A | |
| | SI-862 B | |
| | SI-866 A | |
| | SI-866 B | |
| | SI-869 | |
| (f) | RHR-752 A closed | |
| | RHR-752 B closed | |
| (g) | RHR-743 closed | |
| (h) | RHR-750 open | |
| | RHR-751 open | |
| (i) | Cooling water low flow alarms for the following pumps are not actuated: | |
| | Residual Heat Removal | |
| | Safety Injection | |
| | Containment Spray | |
| (j) | CCW-749 A open | |
| | CCW-749 B open | |
| | Two component cooling water pumps running | |
| (k) | Open SI-863 A | |
| | Open SI-863 B | |
| (l) | Close RHR-744 A | |
| | Close RHR-744 B | |
| (m) | Open SI-860 A | |
| | Open SI-860 B | |
| | Open SI-861 A | |
| | Open SI-861 B | |

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(n) Start A or B residual heat removal pump ______

A ______

B ______

Start two safety injection pumps ____ ______

(o) FI-940 ______ gpm

FI-943 ______ gpm

(p) Start A or B containment spray pump ______

(approx. 1160 gpm) FI-958 A. ______ gpm

(q) Throttle SI-845 until 12 gpm is obtained on FI-949. ______ gpm

(r) When 0% is reached in spray additive tank closed SI-845 C ______

SI-845 B ______

SI-845 A ______

(s) Reset containment isolation. ______

5. Eighteen Hours after the Accident

(a) Close SI-867 A ______

Close SI-867 B ______

Close SI-870 A ______

Close SI-870 B ______

(b) Open SI-869 ______

(c) Start second residual heat removal pump. ______

(d) Throttle valves RHR-744A and RHR-744B to give approx. 2250 gpm on FI-605. ______ gpm

6. Loss of Second Residual Heat Removal Pump

(a) Second residual heat removal pump inoperable # ______.

(b) Hot leg injection started by opening SI-869 ______ hrs.

(c) Cold leg injection started by opening SI-867 A&B and SI-870 A&B and Closing SI-869 ____ hrs.

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Note: Alternate between hot leg injection and cold leg injection every twelve hours.

(d) Second residual heat removal pump is operable - Proceed to #5.

7. Loss of High Head Hot Leg Recirculation

(a) High head hot leg injection isolated - SI-869 closed ________

(b) RHR-750 open ________

RHR-751 open ________

RHR-743 closed ________

RHR-752 A closed ________

RHR-752 B closed ________

RHR-744 A closed ________

RHR-744 B closed ________

(c) Open RHR-743 ________

8. Loss of High Head Hot Leg and Cold Leg Recirculation

(a) All safety injection pumps are inoperable. ________

(b) For cold leg injection open RHR-744 A&B. ________

RHR-743 closed ________

(c) For hot leg injection position or verify the following valves:

RHR-750 open ________

RHR-751 open ________

RHR-744 A closed ________

RHR-744 B closed ________

RHR-752 A closed ________

RHR-752 B closed ________

RHR-743 open ________

Alternate between hot leg and cold leg injection every twelve hours.

The region 5 cycle 3 fuel was used as the limiting fuel region in the CP&L H. B. Robinson FAC ECCS analysis because:

1. It is the minimum burnup region.
2. It has a higher gap pressure.
3. It has a high enrichment.
4. It has high fuel temperatures and gap pressures as a function of kw/ft.

The combination of the above items makes region 5 fuel the limiting region.

## 2.2 Thermal Analysis

### 2.2.1 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 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 during the injection mode with the most severe single active failure, is designed to meet the Acceptance Criteria(1).

### 2.2.2 Method of Thermal Analysis

The description of the various aspects of the LOCA analysis is given in WCAP-8339(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 separate reports (3-6).

The containment parameters used in the containment analysis(6) to determine the ECCS backpressure are presented in Table 2.3. The analysis presented here was performed with the December 25, 1974, version of the Westinghouse evaluation model. This version includes the modifications to the models referenced above as specified by the staff in Reference 8 and complies with Appendix K, 10CFR50.46. The analysis for the $C_D$ = 1.0 break was performed with the 3-15-75 version of the Westinghouse model, which is documented in Reference 9.

## 2.3 Results

Table 2.2 presents the peak clad temperatures and hot spot metal reaction for a range of break sizes. This range of break sizes was determined to include the limiting case for peak clad temperature from sensitivity studies reported in Reference 7.

The analysis of the loss of coolant accident is performed at 102 percent of Engineered Safeguards Design Rating. The peak linear power, and core power used in the analyses are given in Table 2.2. The equivalent core parameter at the license application power level are also shown in Table 2.2. 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 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 2.2 for each break size analyzed.

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 long-lived radioactivity remaining in the core.

The time sequence of events for all breaks analyzed is shown in Table 2.1.

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

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

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

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

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

7. Buterbaugh, T. L., Johnson, W. J. and Kopelic, S. C., "Westinghouse ECCS - Plant Sensitivity Studies", WCAP-8356, July 1974.

8. Federal Register, "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", November 1974.

9. Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model Supplementary Information", WCAP-8472, April 1975.

TABLE 2.1

LARGE BREAK

TIME SEQUENCE OF EVENTS

| Event | Time After Break Initiation (Seconds) | | |
|---|---|---|---|
| | DECLG ($C_D$ = 1.0) | DECLG ($C_D$ = 0.6) | DECLG ($C_D$ = 0.4) |
| Start | 0.0 | 0.0 | 0.0 |
| Reactor Trip Signal | 0.626 | 0.640 | 0.660 |
| Safety Injection Signal | .54 | .67 | .82 |
| Accumulator Injection | 10.1 | 12.8 | 17.0 |
| End of Blowdown | 22.1 | 24.0 | 28.6 |
| Bottom of Core Recovery | 42.36 | 43.1 | 47.89 |
| Accumulator Empty | 49.372 | 52.3 | 56.8 |
| Pump Injection | 25.54 | 25.67 | 25.82 |

TABLE 2.2

LARGE BREAK

ANALYSIS RESULTS

| Analysis Results | DECLG $C_D = 1.0$ | DECLG $C_D = 0.6$ | DECLG $C_D = 0.4$ |
|---|---|---|---|
| Peak Clad Temperature, $^{o}F$ | 1795 | 1867 | 2200 |
| Peak Clad Temperature Location, Ft. | 7.25 | 6.75 | 6.25 |
| Local $Zr/H_2O$ Reaction (max), % | 1.757 | 2.0 | 9.6 |
| Local $Zr/H_2O$ Location, Ft. | 7.25 | 6.75 | 5.75 |
| Total $Zr/H_2O$ Reaction, % | < 0.3 | < 0.3 | < 0.3 |
| Hot Rod Burst Time, sec | 41.8 | 35.6 | 26.5 |
| Hot Rod Burst Location, Ft. | 5.75 | 5.0 | 5.75 |

| Calculation | |
|---|---|
| Core Power, Mwt 102% of | 2300 |
| Peak Linear Power, kw/ft 102% of | 13.45 |
| Peaking Factor (At License Rating) | 2.30 |
| Accumulator Water Volume, $Ft^3$* | 825 |

| Fuel region + cycle analyzed | Cycle | Region |
|---|---|---|
| | 3 | 5 |

*Setpoints on Accumulator Water Volume High and Low Limits are modified to allow this minimum volume to be maintained.

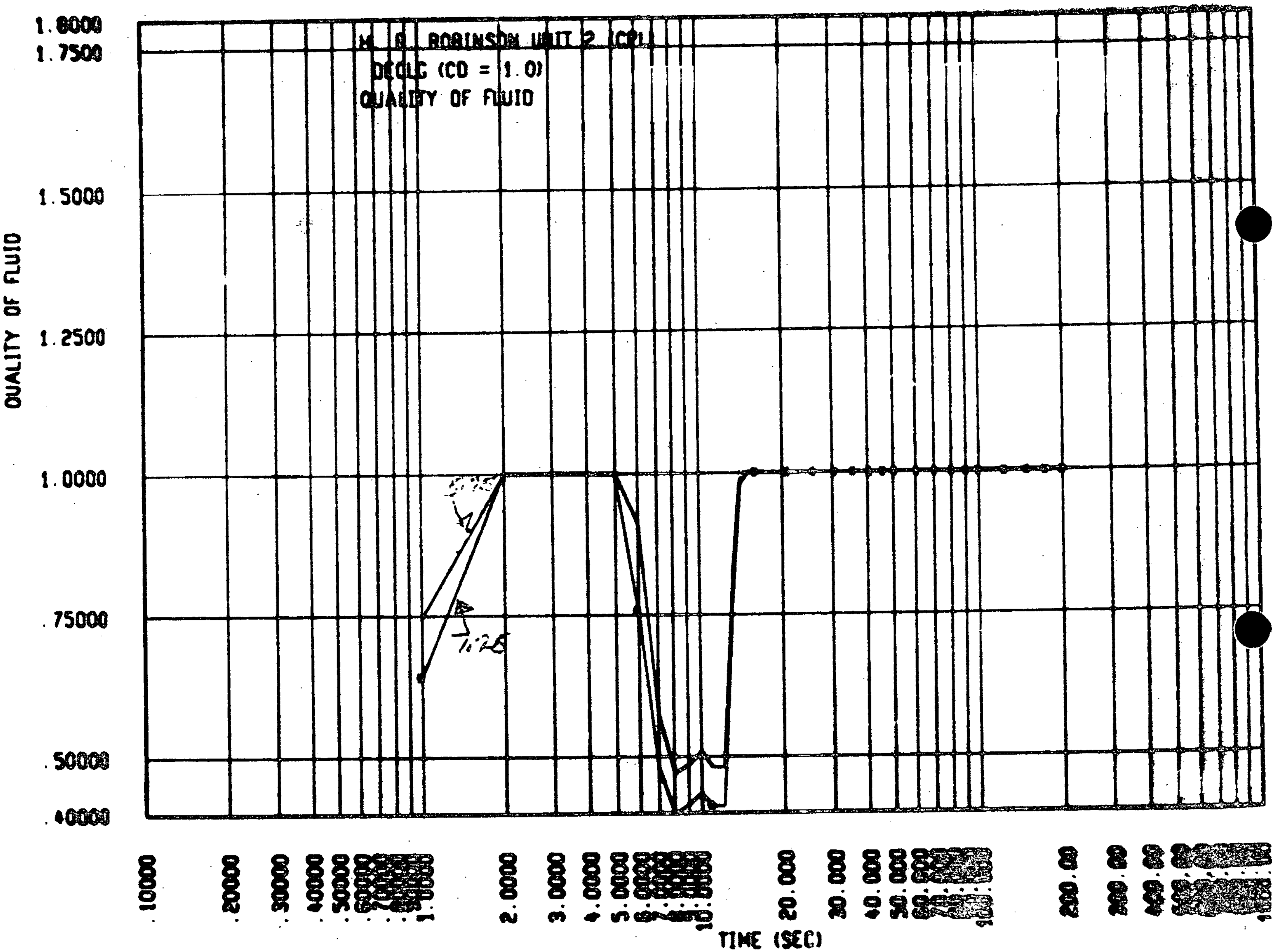


Figure 26a - Fluid Quality (DECLG (CD = 1.0)

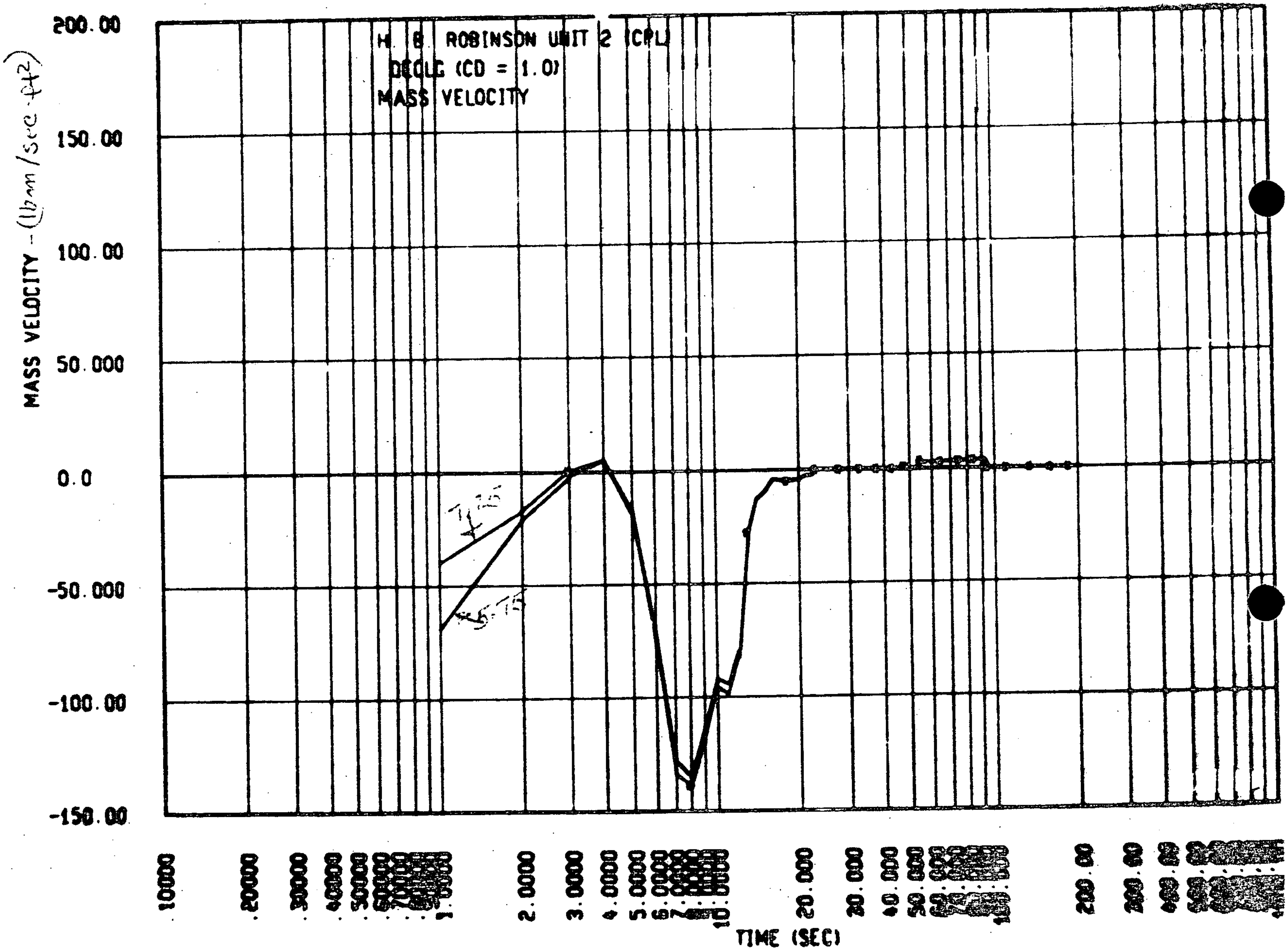


Figure 2-2a - Mass Velocity - DECLG (CD=1.0)

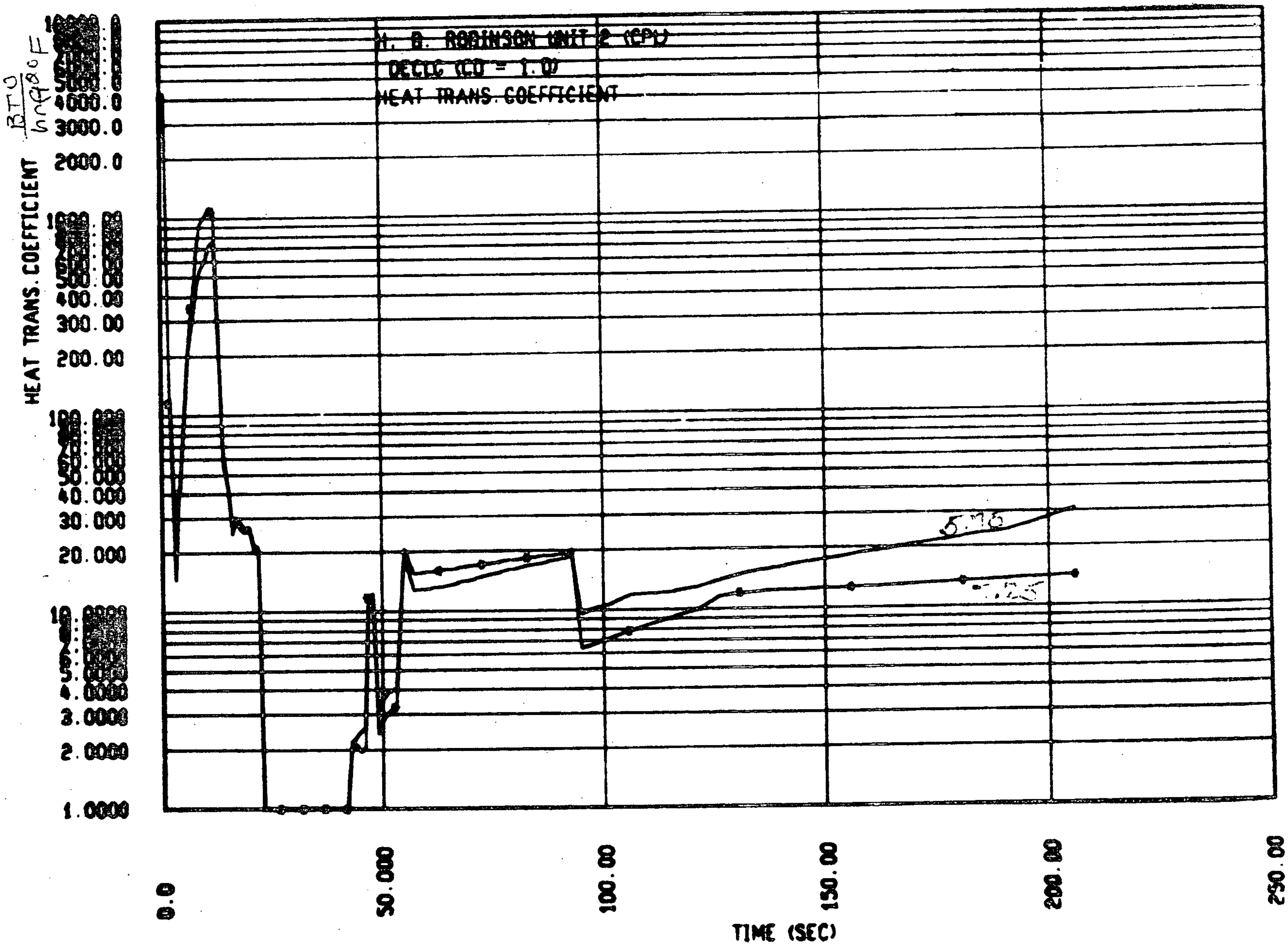


Figure 2.3a - Film Heat Transfer Coefficient DECLG (CD=1.0)

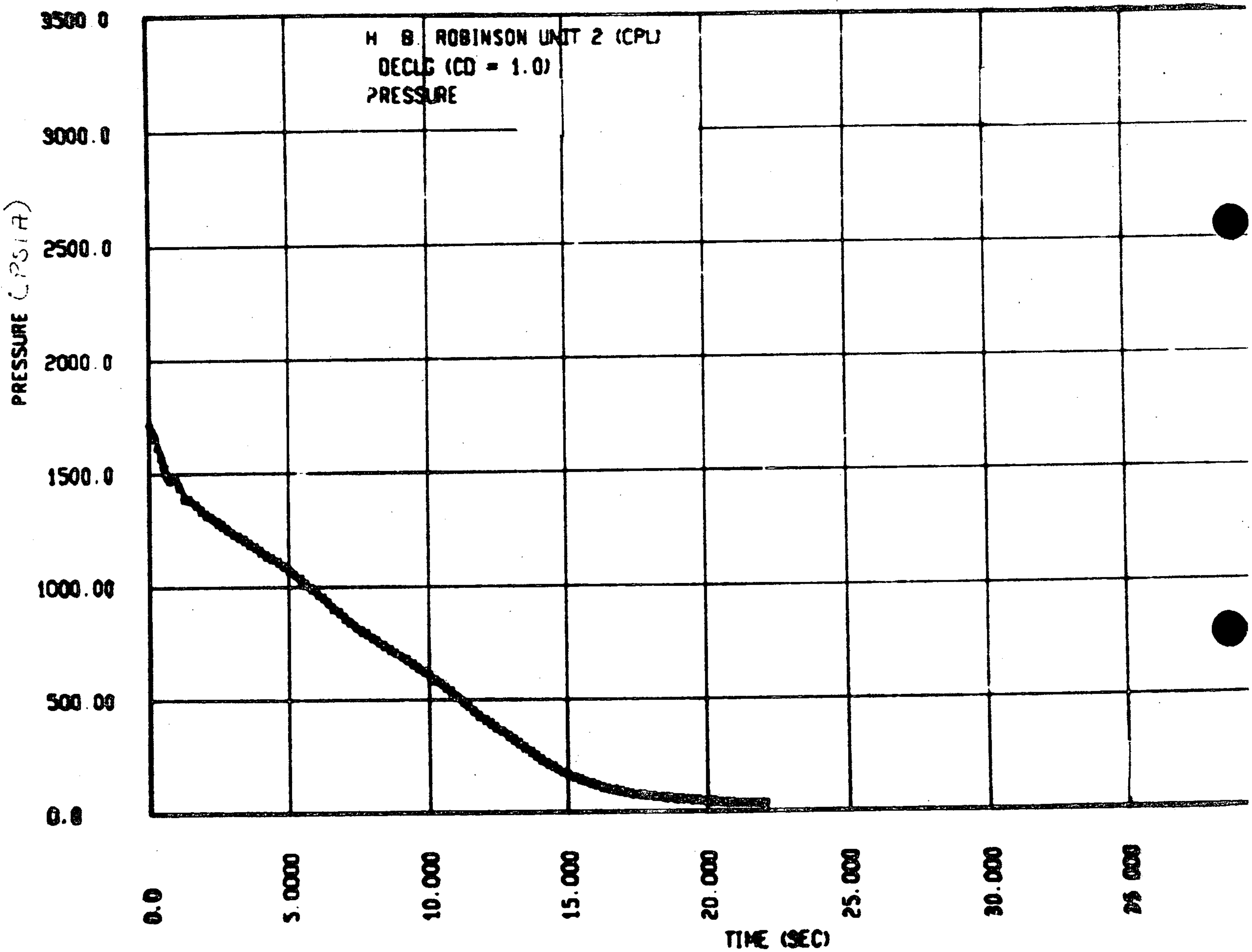


Figure 3.4a - Core Pressure DECLG (CD=1.0)

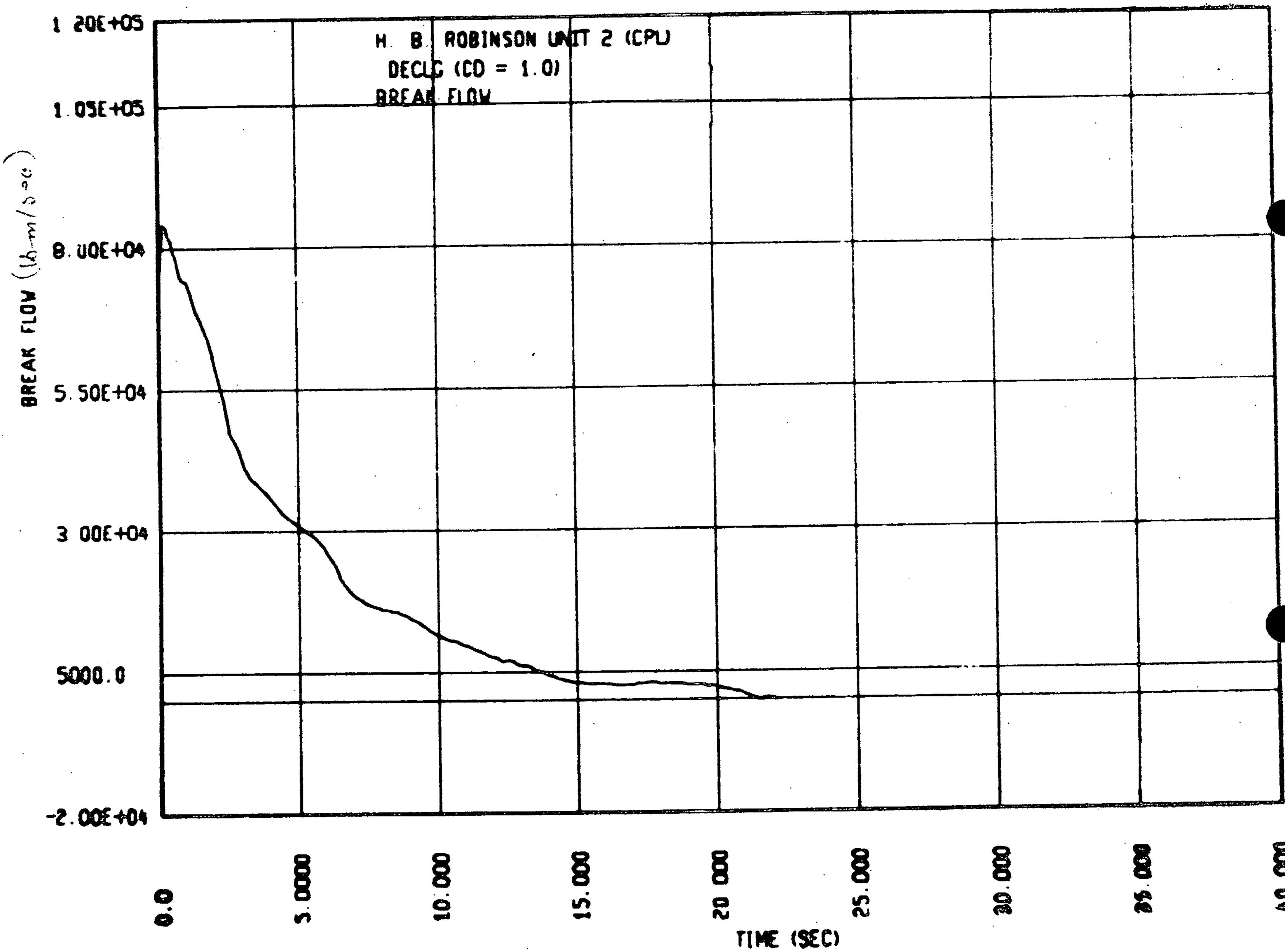


Figure 2.5a Break flowrate - DECLG ($C_D$=1.0)

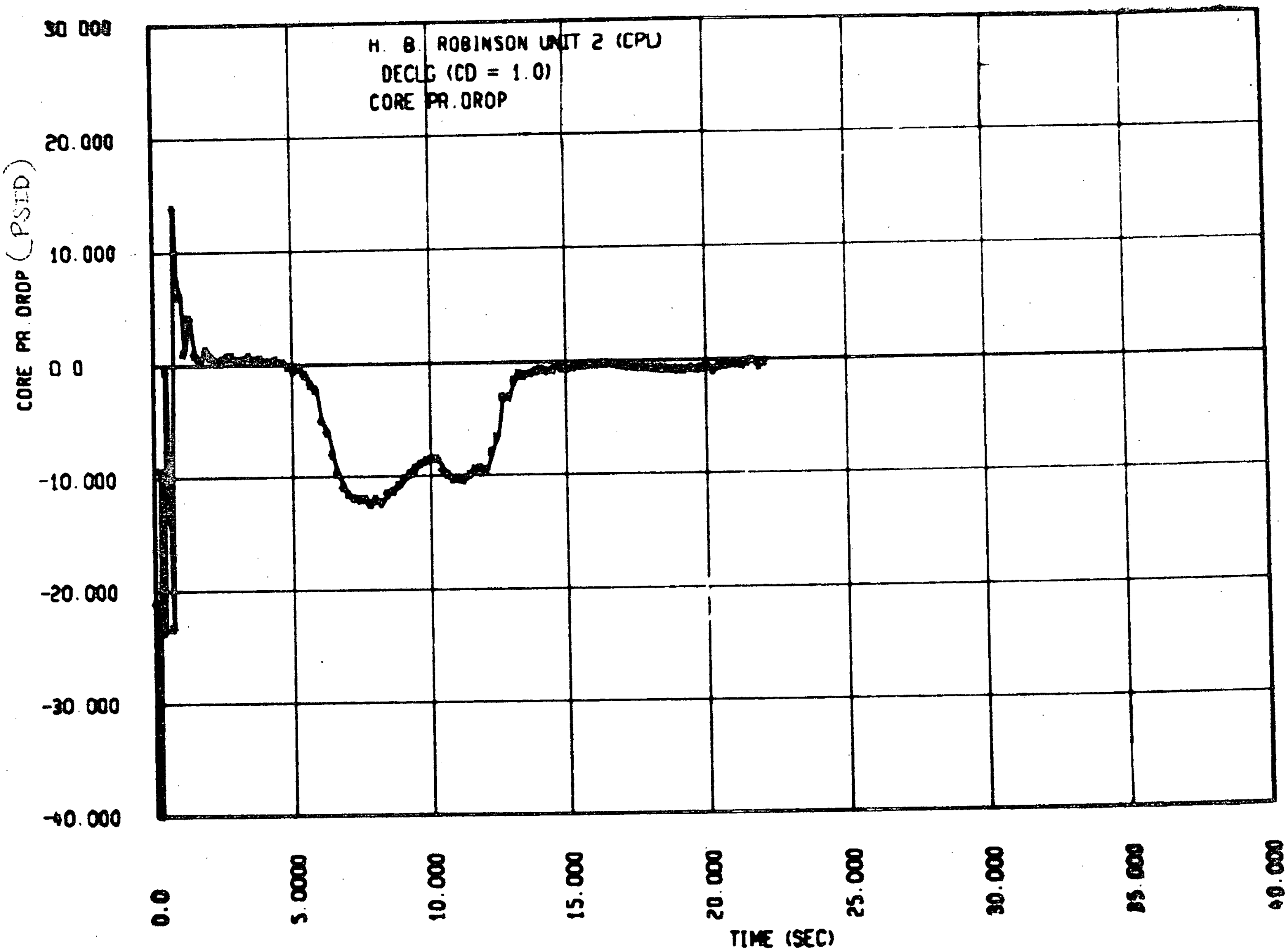


Figure 3.6a - Core Pressure Drop - DECLG (CD=1.

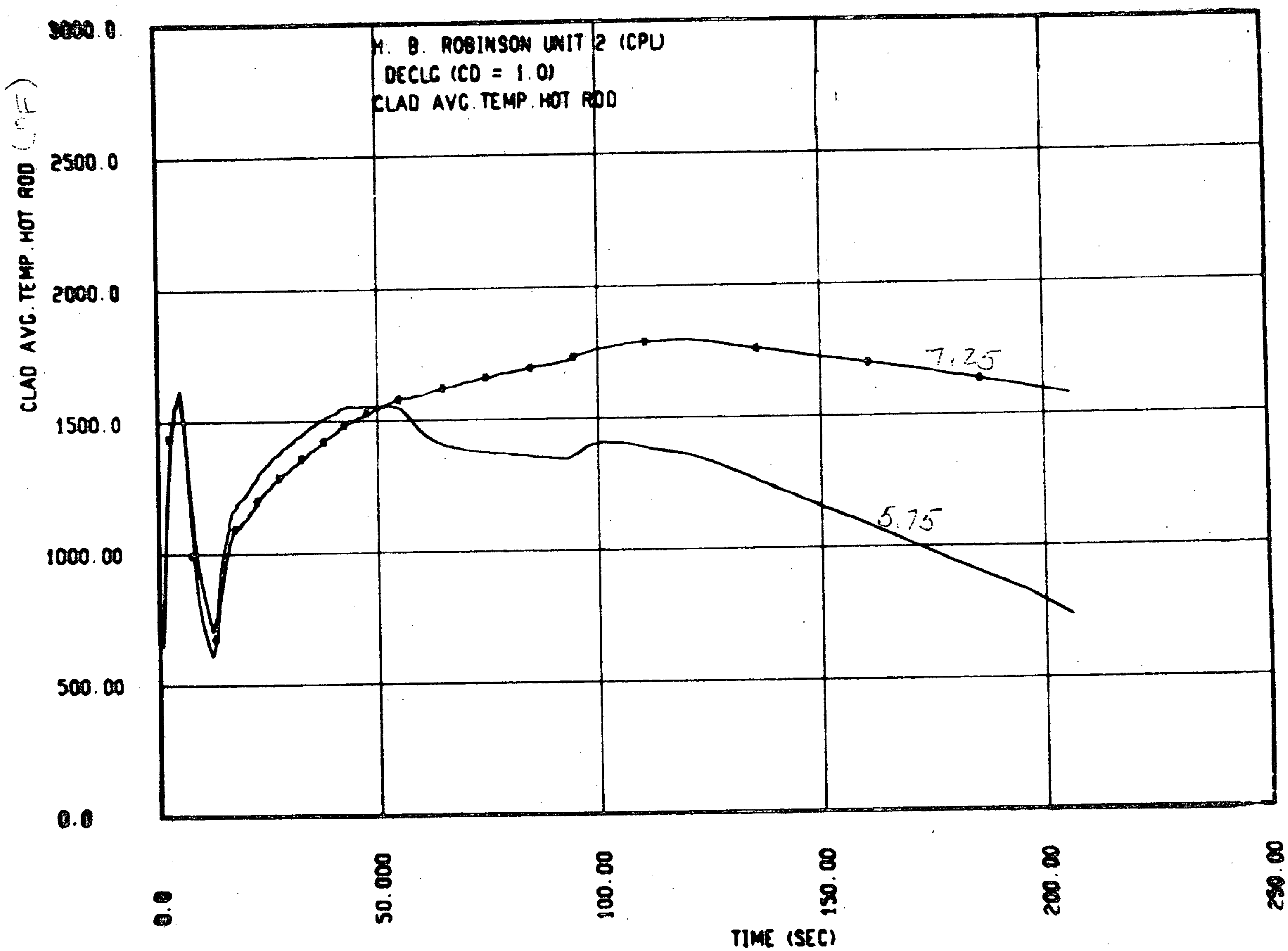


Figure 2-7a Peak Clad Temperature - DECLG (Cd=1.0)

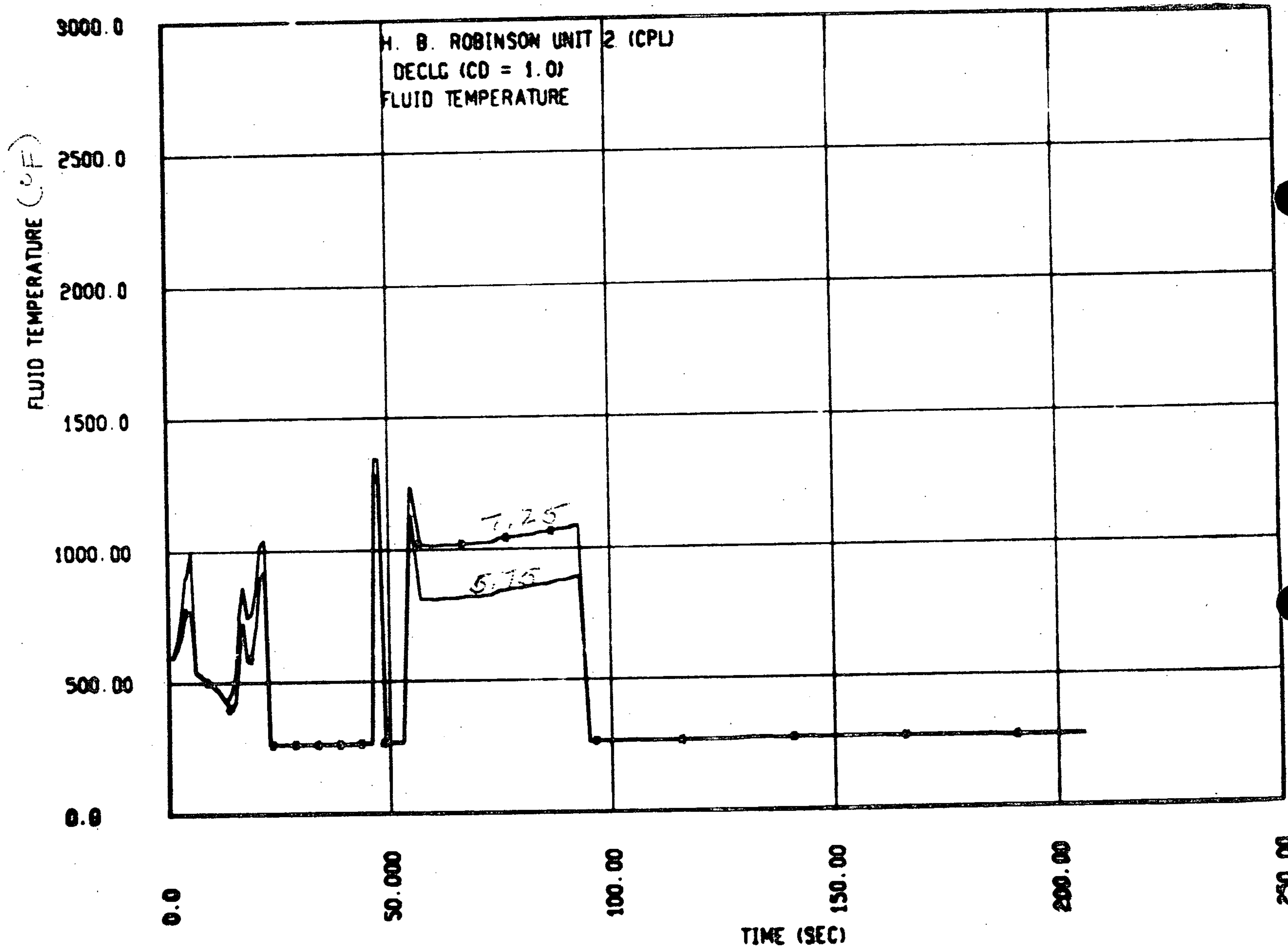


Figure 2.8a - Fluid temperature - DECLG (CD=1.0)

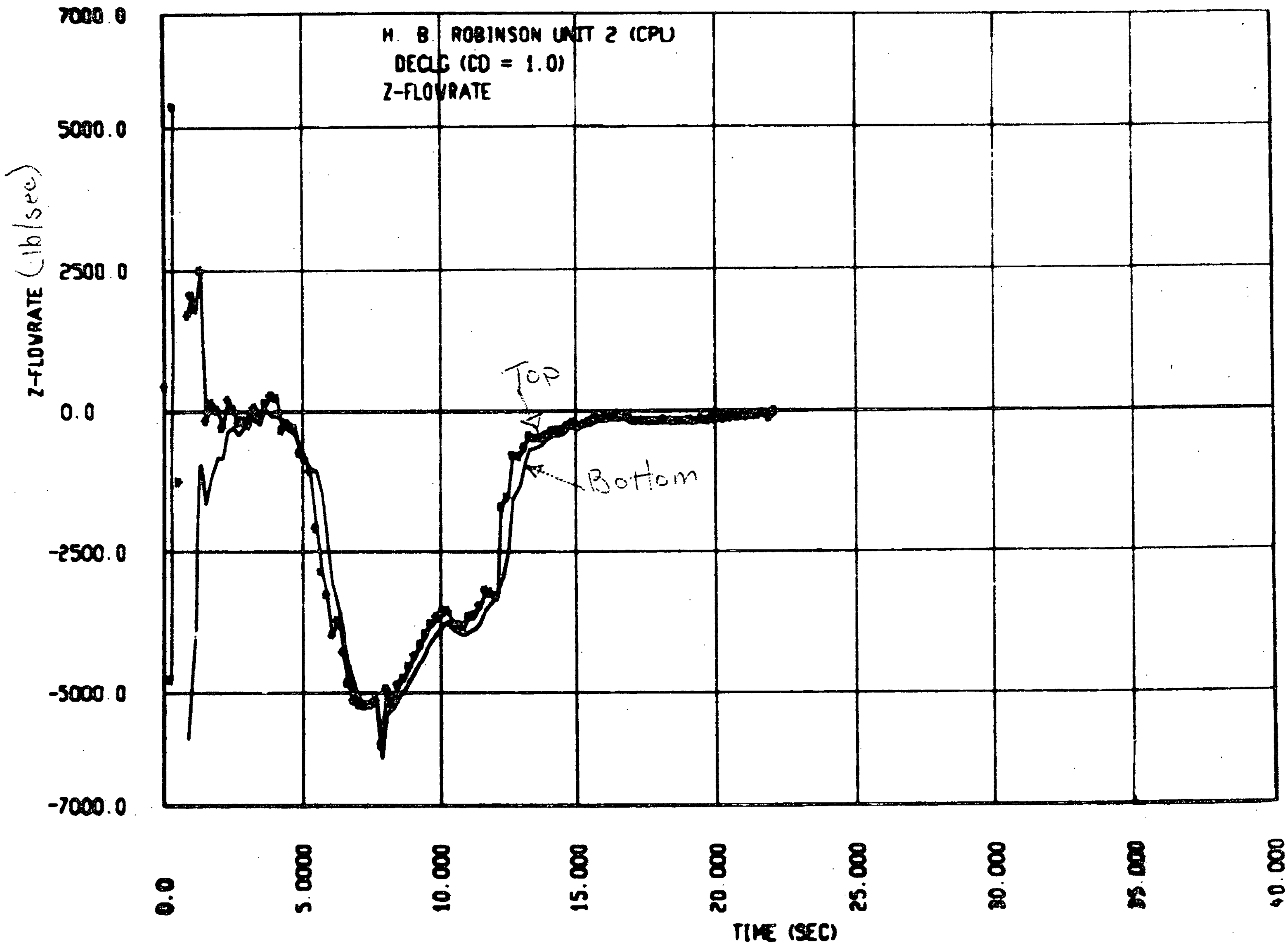


Figure 2.7c Core flow - Top and Bottom - DECLG ($C_D$=1.0)

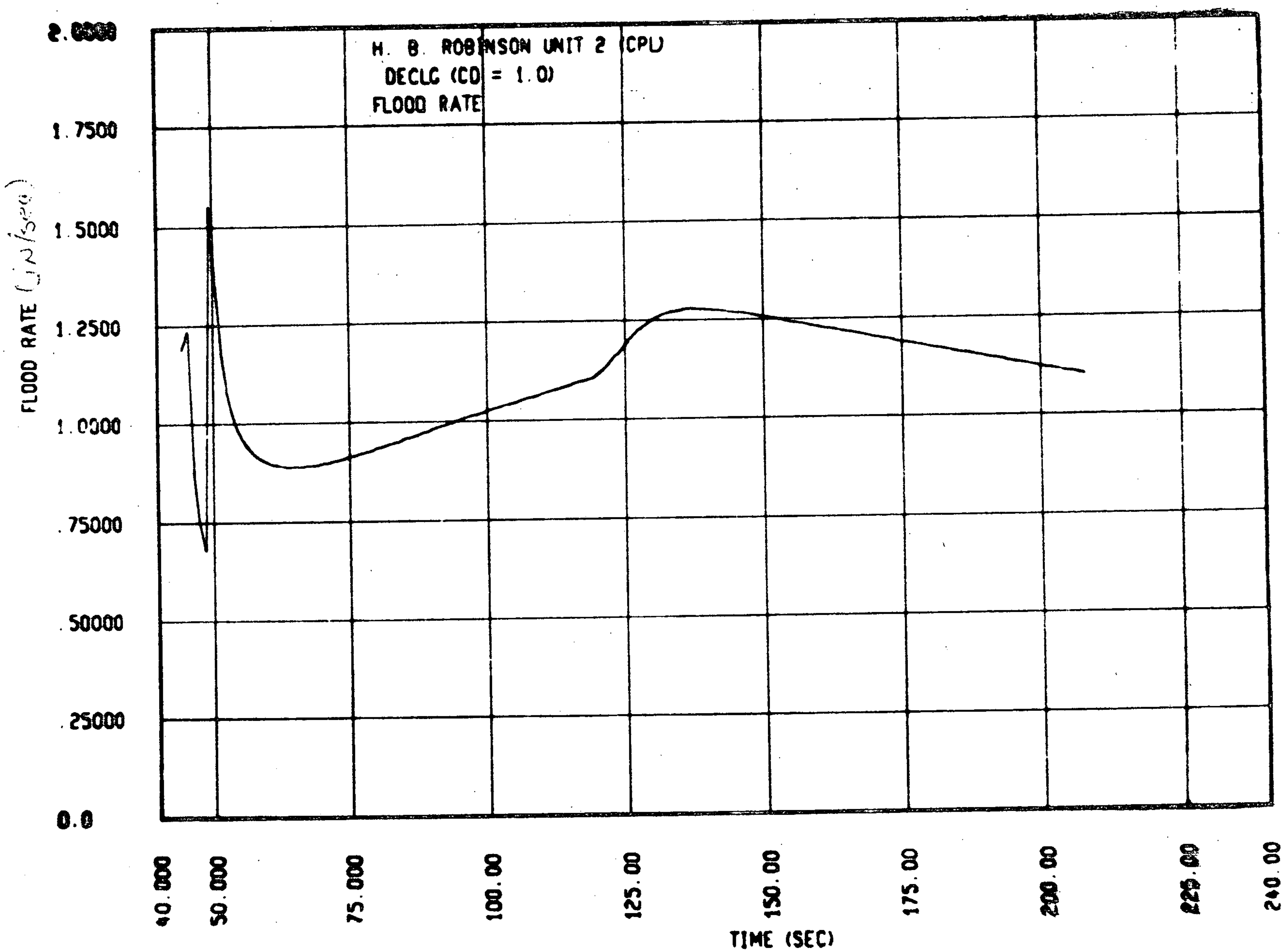


Figure 3.10a-1 Reflood Transient - Core Flooding Rate DEPNG (CD=1.0)

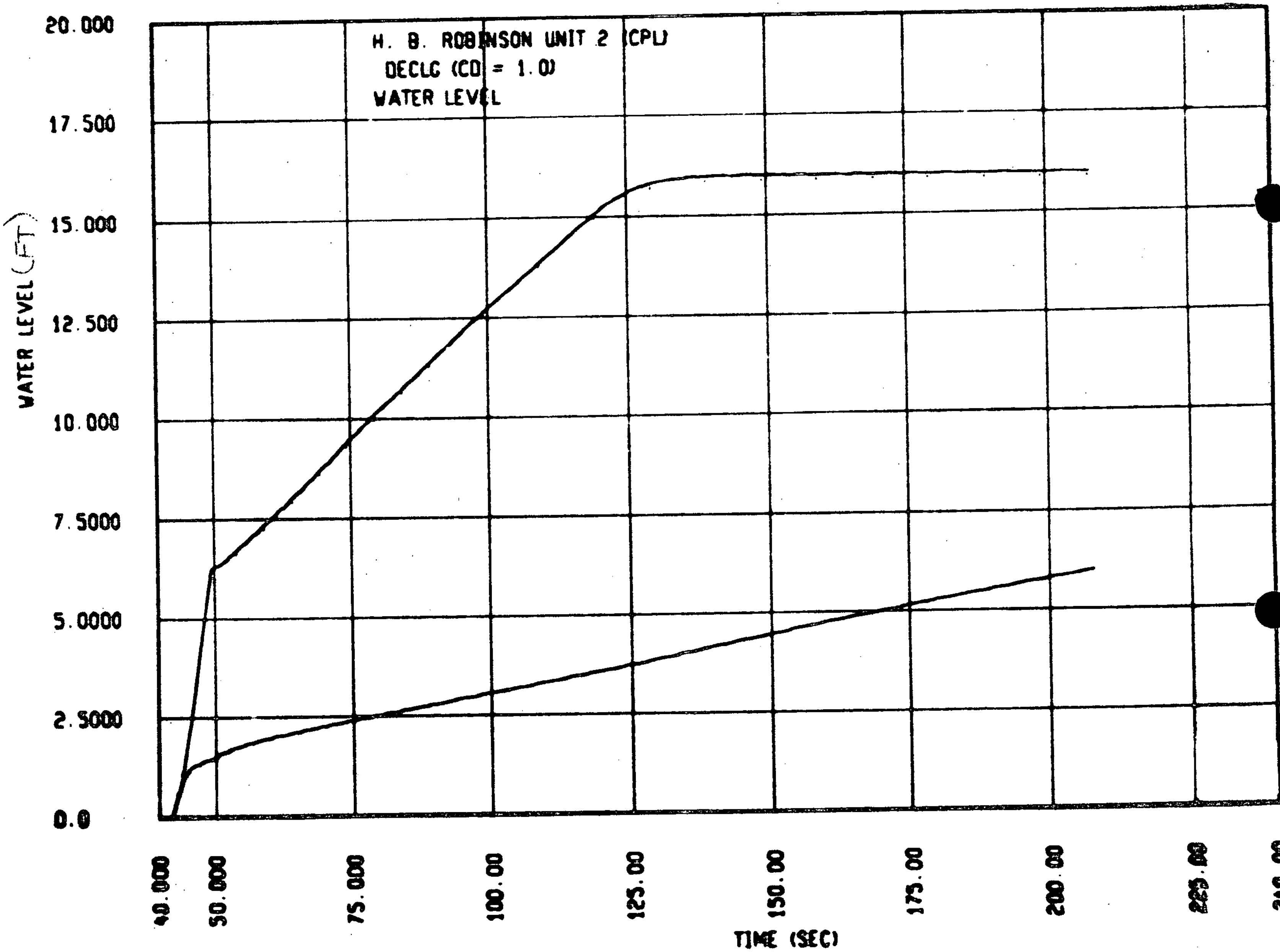


Figure 2-10.a-2 Reflood Transient - Core and Downcomer Water Height DECLG (CD=1.0)

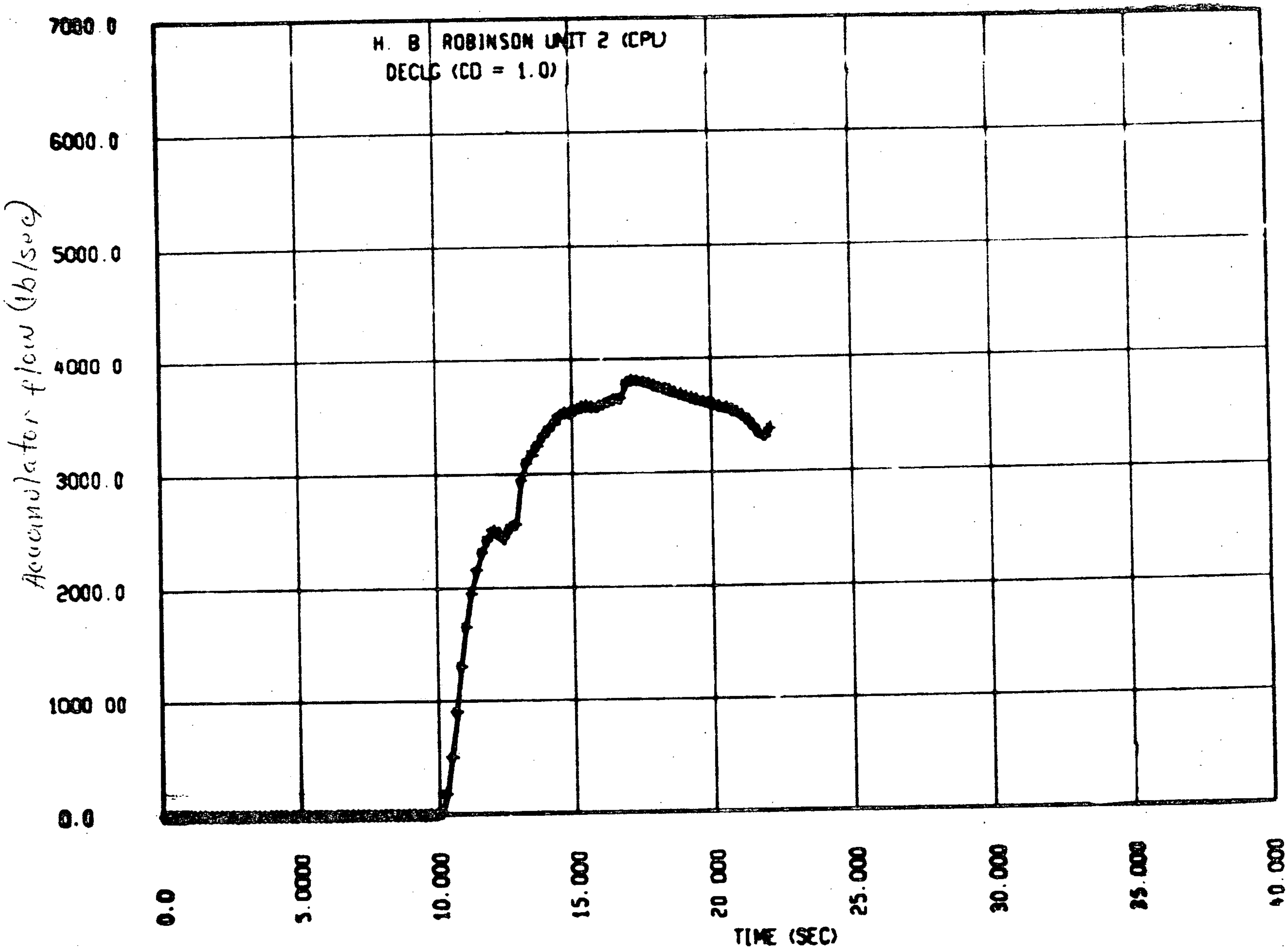


Figure 11.a - Accumulator Flow during Blowdown - DECLG ($C_D$ = 1.0)

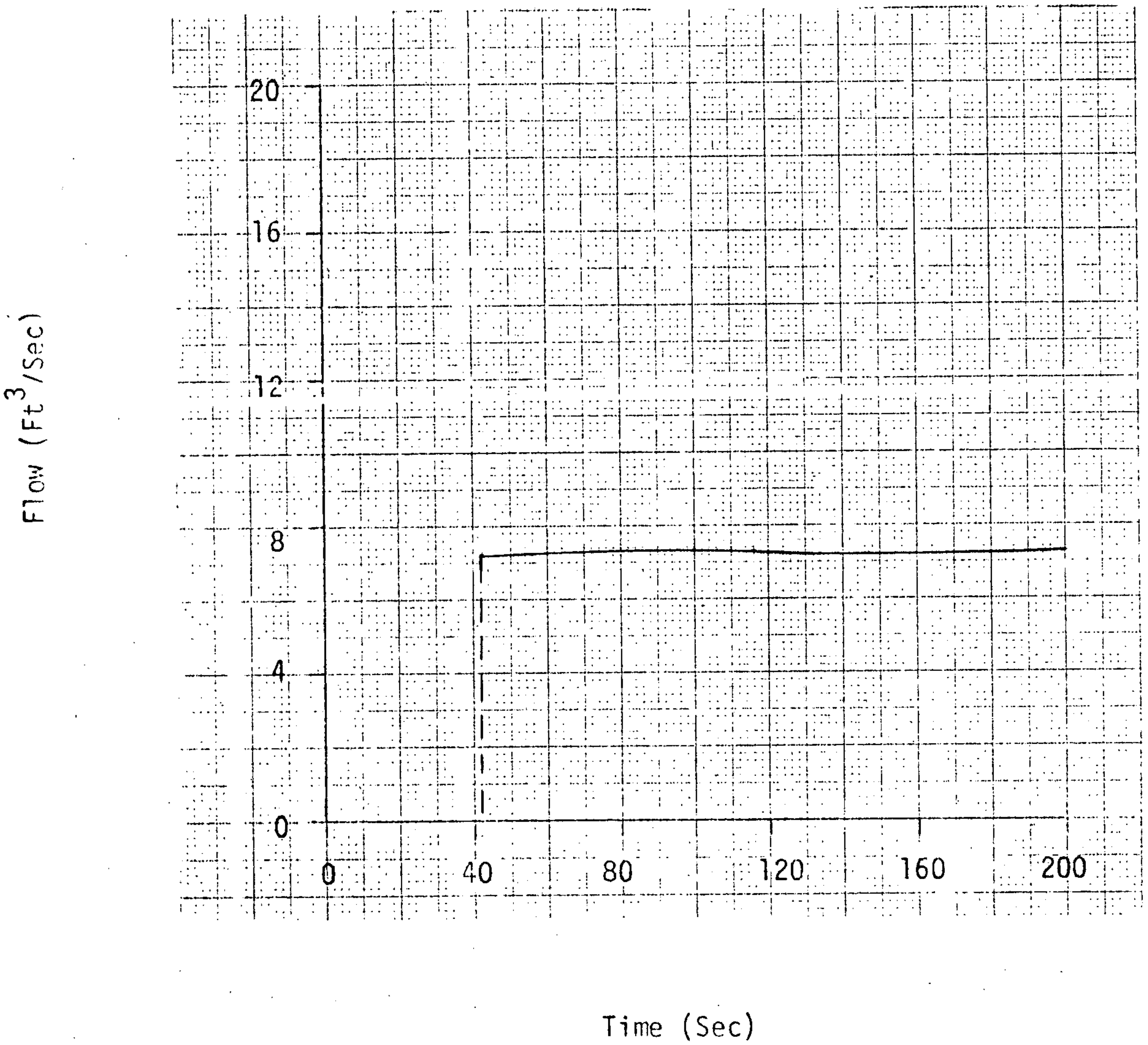


Figure 2-12a Pumped ECCS Flow (Reflood) - DECLG ($C_D$=1.0)

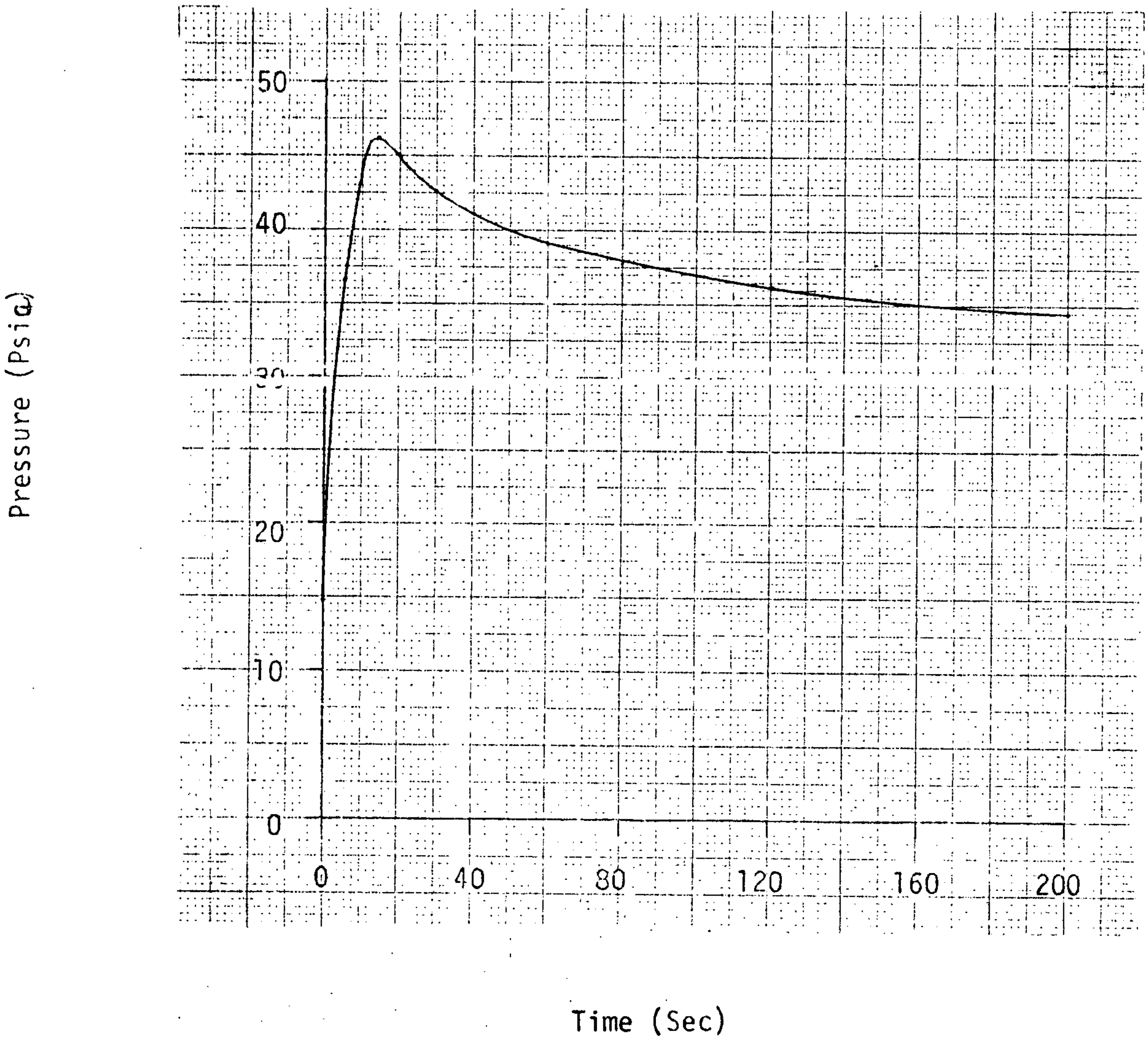


Figure 2-13a Containment Pressure - DECLG ($C_D = 1.0$)

## INFORMATION FOR COMPUTATION OF VESSEL BORIC ACID CONCENTRATION - H. B. ROBINSON UNIT NO. 2

| Turkey Point | H. B. Robinson | |
|---|---|---|
| 2200 MWt | 2300 MWt | Core Power |
| $3.0 \times 10^6$ | $3.3 \times 10^6$ | Sump Inventory (lbm) |
| 2030 | 2045 | Initial ppm |
| 854 | 854 | Effective vessel val. ($ft^3$) |
| 93 | 98 | SI Enthalpy |
| 87.2 | 82 | Sensible Heat |
| 131.5 | | SI flow rate req'd |

Q - For a $C_D$ of 0.4, the generic 3-loop analysis showed a positive core flow between 10-13 seconds compared to Figure 2-9c which shows a negative core flow during this period. . .

Re-run this worst-case with reactor coolant pumps <u>running</u> to confirm that the expected decrease in core flow after 10 seconds (with pumps running) would not cause an increase in PCT.

A - Figure 1 shows plots of the core flow during blowdown for the Double Ended Cold Leg Break with $C_D$ = 0.4 for two conditions:*

1) The "design" assumption that the RCS pumps trip at the time of the pipe break.

2) Assuming that the RCS pumps run at 100% speed during blowdown.

All other input assumptions were identical for these cases and the results are that the design case (pumps trip) had a peak clad temperature 26 F higher than the pumps run case.

This confirms the results and conclusions reached in the generic 3 loop sensitivity study presented in WCAP 8356.

*Applies for H. B. Robinson Unit No. 2

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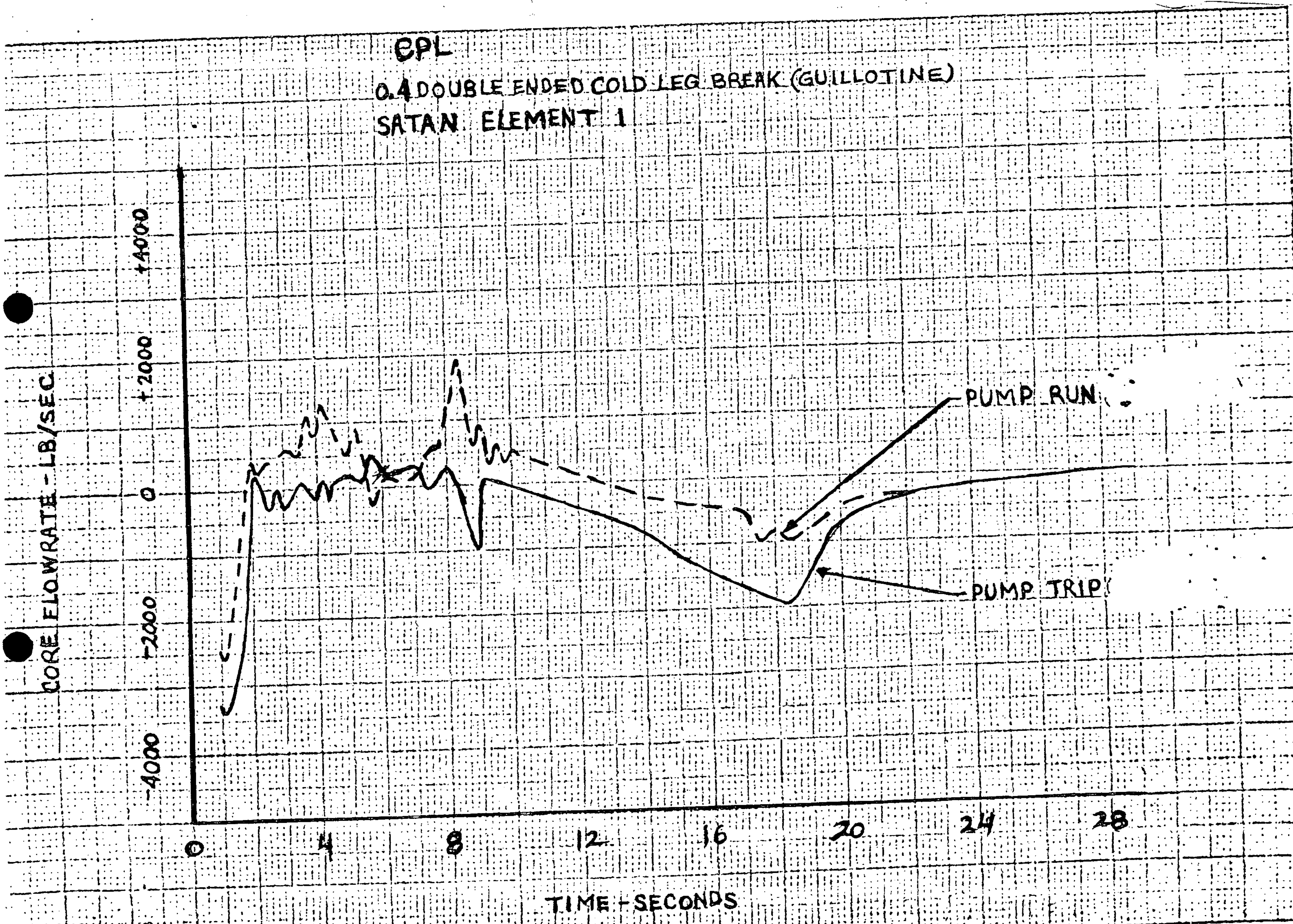


CPL
0.4 DOUBLE ENDED COLD LEG BREAK (GUILLOTINE)
SATAN ELEMENT 1
CORE FLOWRATE - LB/SEC
+4000
+2000
0
-2000
-4000
PUMP RUN
PUMP TRIP
0
4
8
12
16
20
24
28
TIME - SECONDS

Time for Switchover from Injection Phase to Recirculation Phase Following a Loss of Coolant Accident

Concern has been expressed over the time available for operator action to complete valve alignments during the switchover from the injection phase following a Loss of Coolant Accident (LOCA) where water is injected into the cold legs of the primary coolant system from the supply available in the Refueling Water Storage Tank to the recirculation phase where water is drawn from the containment sump by the residual heat removal pumps and injected into the cold legs. In this switchover period, all pumps in the Safety Injection System such as the containment spray, high head and low head pumps are off and no water is being supplied to the core. Due to residual decay heat present in the core, boil off of the vessel and loop contents would occur and after a certain period of time, the water level in the vessel would reach the top of the core. If the water level were allowed to decrease below the top of the core, fuel clad heatup would reoccur with a possibility of the allowable peak clad temperatures being exceeded.

The time available for switchover has been calculated by the reactor vendor, Westinghouse, and is attached to this discussion. As may be seen, the minimum time is 15 minutes, assuming both safeguards' trains are operating. Under this assumption, initiation of switchover occurs about 26 minutes after the design basis LOCA when the low-low level alarm for the RWST is received and the pumps are secured. With one train of safeguards in operation, the alarm is received 52 minutes after the LOCA and the resulting time available for switchover is about 18 minutes.

During the switchover period after the operating pumps have been secured, valves in the Refueling Water Storage Tank line which have been open during the injection phase must be closed to isolate the RWST and the low head pump suction. In addition, the Safety Injection Systems must be aligned to provide a flow of water from the containment sump to the low head pump suction, and from downstream of the RHR heat exchangers to the suction of the high head and containment spray pumps. The detailed procedure to be followed during this time period and the valve position changes to be performed are covered in Appendix A of Emergency Instruction EI-1 of which a preliminary version has been submitted. A study of the time required to perform all of the required actions of the procedure from the time the operating pumps are secured prior to switchover to the time they are restarted following switchover has been performed by the Robinson Plant operating staff. The time required is estimated to be of the order of seven minutes, accounting for operator reaction time, valve closure and opening times, pump starting times, etc. A note is incorporated in the procedure which states that switchover must be completed within 10 minutes and stating the concern of possible core uncovery if that time is exceeded. This conservatism of allowing only 10 minutes instead of the full 15 minutes as calculated by Westinghouse and the conservatism contained in the Westinghouse calculation itself is assurance that the switchover can be performed in a deliberate manner with no concern of the possibility of reuncovering the core.

Attachment

DWB/rt

The minimum time available to the operator for switchover from the injection to the recirculation mode has been calculated. The time to core recovery, assuring a complete interruption of ECCS, is 15 minutes for two trains operating and 18 minutes for one train operating.

Calculation of time to uncover top of core if ECCS flow is interrupted.

Initial conditions

Power = 2297 - ESDR power

t = 26.3 min - time when lo-lo level signal is reached with two trains operating.

Pcont = 14.7 psia

decay heat is ANS infinite +20% = .022 at 26.3 min

RWST - 300,000 gal

Lo-lo level signal = 7% or 21,000 gal

$$t = \frac{300{,}000 - 21{,}000 \text{ gal}}{5{,}300 \times 2 \frac{\text{gal}}{\text{min}}} \approx 26.3 \text{ min}$$

1 Lo-head pump max flow = 3,000 gpm

1 Hi-head pump max flow = 600 gpm

1 Spray pump runout flow = <u>1,700 gpm</u>

5,300 gpm

Core area = 41.9 $ft^2$

Core height = 12 ft

Upper plenum area = 82.7 $ft^2$

Upper plenum ht = 11 ft

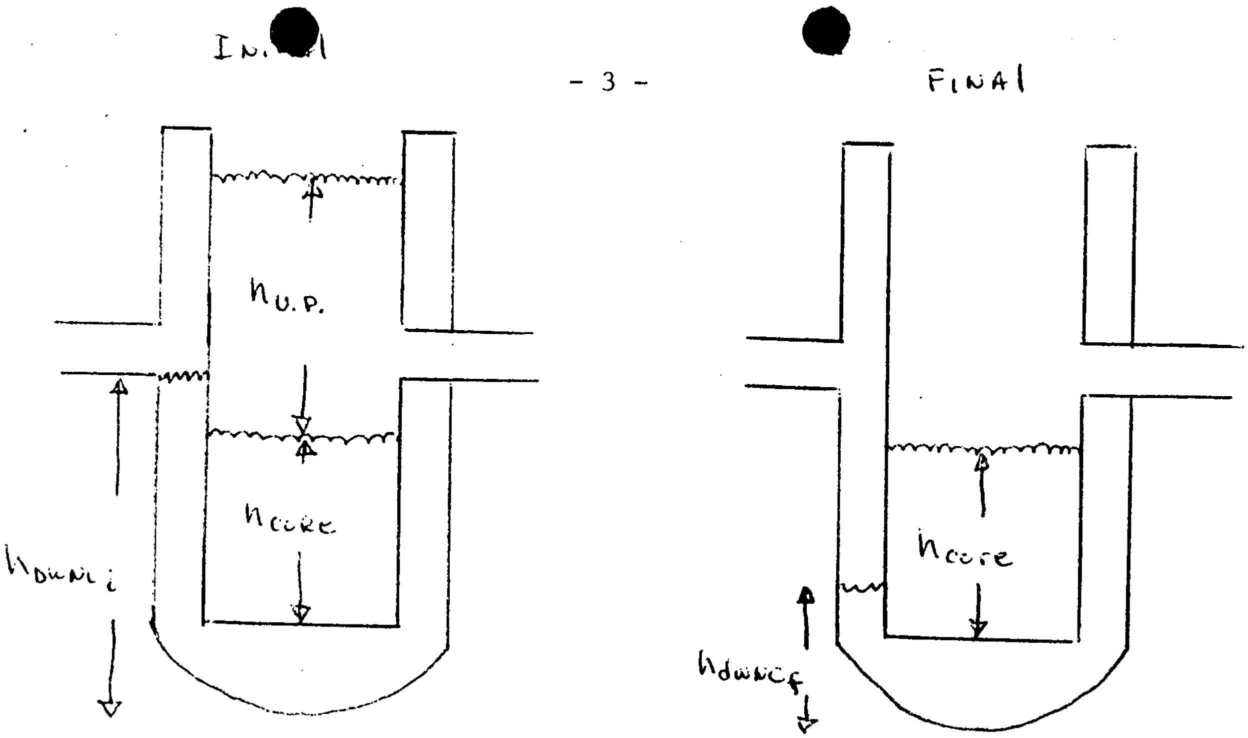
Downcomer area = 36.14 $ft^2$

Downcomer ht = 16.0 ft

Fluid properties at P = 14.7

$v_f$ = .016719 $v_g$ = 26.799

$h_f$ = 180.17 $h_{fg}$ = 970.3 $h_g$ = 1150.5




$$Q \text{ decay} = (2297 \times .022) \times \frac{3.413 \times 10^6}{3600} = 47{,}909 \text{ BTU/sec}$$

$$\text{Boil off} = \frac{47{,}909 \text{ BTU/sec}}{970.3 \text{ BTU/lbm}} = 49.37 \text{ lbm/sec}$$

Superficial steam velocity - core $\frac{(49.37 \times 26.799)}{41.9} = 31.57$ ft/sec

Average core velocity = ½ (31.57) = 15.79 ft/sec

Superficial steam velocity - upper plenum $\frac{(49.37 \times 26.795)}{82.7} = 15.99$

Calculation of void fractions in the core and upper plenum using Yeh correlation

$$= .925 \left(\frac{\rho_g}{\rho_f}\right)^{.289} \left(\frac{V_g}{Vbcr}\right)^{a}$$

where

$$a = .47 \text{ if } \frac{V_g}{Vbcr} \geq 1$$

$$a = .67 \text{ if } \frac{V_g}{Vbcr} < 1$$

$$Vbcr = 2/3 \sqrt{g \; Rbcr}$$

$$Rbcr = \left(\frac{1.53}{2/3}\right)^{2} \frac{G}{P_f}$$

$$G = 12 \; (500 - .707 \; tsat) \; 10^{-6} = .004201 \; \frac{lbf}{ft}$$

$Rbcr = .044$ ft

$Vbcr = .795$

$\frac{V_g}{Vbcr}$ 1 for core and upper plenum, therefore, $a = .47$

$\alpha$ core = .646 $\quad \alpha$ upper plenum = .6498

Calculate mixture height in upper plenum

$\rho_f$ hdwnc$_i$ = $\rho_f$ h core + $\rho_f$ hup

= (1 - $\alpha$ core) $\rho_f$ h core t (1 - up) $\rho_f$ hup

$$hup = \frac{hdwnc_i - (1 - \alpha core)\ h\ core}{(1 - \alpha up)}$$

$$hup = \frac{16 - (1 - .646)\ 12}{(1 - .6498)} = 33.55 \text{ ft}$$

Since 33.55 >11 ft - height of upper plenum hup will be set to 11 ft.

Ths is a conservative assumption.

Initial liquid mass in core upper plenum and downcomer.

It is seen from calculating effects of metal heat and energy addition that downcomer water may be assumed bubble free.

$$M_{ff} = \frac{1}{.016719} \left[36.14 \times 16 + (1 - .646)(41.9 \times 12) + (1 - .650)(82.7 \times 11)\right]$$

$M_{ff}$ = 64,286 lbm

Height of liquid in downcomer when core is about to uncover hdwnc$_f$

$\rho_f$ hdwnc$_f$ = $\rho_f$ h core (1 - core)

= 12 (1 - .646) = 4.24 ft

Final liquid mass

$$M_{ff} = \frac{1}{.016719} \left[36.14 \times (4.24) + (1 - .646)\ (41.9 \times 12)\right]$$

$$M_{ff} = 19{,}828 \text{ lbm}$$

Mass evaporated = 64,286 - 19,828 = 44,458 lbm

$$\text{Time} = \frac{44{,}458 \text{ lbm} \times 970.3 \frac{\text{BTU}}{\text{lbm}}}{47{,}909 \text{ BTU/sec}} = 15.0 \text{ min}$$

For one train of ECCS operating

Total flow = 5,300 gpm

$$t = \frac{279{,}000 \text{ gal}}{5{,}300 \text{ gpm}} = 52.6 \text{ min}$$

Decay heat fraction = .019 at 52.6 min

Q decay = 41,375.9 BTU/sec

Boil off = 42.64 lbm/sec

Superficial steam velocity = 27.27 ft/sec

Average vel in core = 13.63 ft/sec

Superficial steam velocity = 13.81 ft/sec

Void fractions

$\alpha$ core = .602

$\alpha$ up = .606

Mixture height in up

$hup = 28.48$ ft

set to 11 ft

Initial mass in core, up and dwnc

$M_{ff} = 67,993$ lbm

Height of liquid in downcomer when core is about to uncover

$hdwnc_f = 4.776$ ft

Mass liquid final

$M_{ff} = 22,293$

Mass evaporated = 45,700 lbm

Time to boil off to top of core = 17.86 minutes

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