ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATION REVISIONS

FOR

REACTOR OPERATION WITH TWO

REACTOR COOLANT PUMPS

October 31, 1975

7912180968

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of five days is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The five days for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the five-day period is considered very remote.

A reactor coolant pump or low pressure injection pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One low pressure injection pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less. (1)

The low pressure injection system suction piping is designed for 300°F and 370 psig; thus the system with its redundant components can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident at hot shutdown. (5) The pressurizer code safety valve lift setpoint shall be set at 2500 psig ± 1% allowance for error and each valve shall be capable of relieving 300,000 lb/hr of saturated steam at a pressure no greater than 3% above the set pressure.

REFERENCES

(1) FSAR Tables 9-11 and 4-3 through 4-7.

(2) FSAR Sections 4.2.5.1 and 9.5.2.3.

(3) FSAR Section 4.2.5.4.

(4) FSAR Sections 4.3.10.4 and 4.2.4.

(5) FSAR Sections 4.3.7 and 14.1.2.2.3.

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ENCLOSURES:
Additional information relative to the Oconee Nuclear Station ECCS analysis, With Attach. 1-4......

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POWER BUILDING
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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

October 31, 1975

Mr. Benard C. Rusche
Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. A. Purple, Chief
Operating Reactors Branch #1

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Mr. Rusche:

With regard to your letters of October 14 and 15, 1975 concerning requests for additional information relative to the Oconee Nuclear Station ECCS analysis, Attachments 1 and 2 are provided.

As you are aware, Amendment 6 of the Oconee Unit 1 Facility Operating License, DPR-38, revised Technical Specifications to permit operation within the appropriate fuel and core design limits during the second fuel cycle. The proposed control rod withdrawal limit for four-pump operation (Figure 3.5.2-1A2) after 250 ± 5 full power days of operation was not included. The omission of this curve was due to the expected approval of Technical Specifications based on the ECCS Final Acceptance Criteria before the curve would be required. On July 9, 1975, an analysis of Oconee ECCS performance using an approved evaluation model was submitted. This submittal included revised control rod withdrawal curves for Oconee 1 Cycle 2 operation.

Currently, Oconee Unit 1, Cycle 2 measured boron concentration data indicate that it may be necessary to begin removal of the Group 7 control rods from the core as early as 235 EFPD, rather than the originally estimated 245-255 EFPD. The associated small discrepancy, approximately 20 ppmB, between the measured and the predicted boron concentration at this core lifetime, is well within the tolerance of calculational uncertainties. In order to assure the continued full power operation of Oconee 1, revised Final Acceptance Criteria control rod position limits and operational power imbalance envelopes have been prepared which permit withdrawal of Group 7 control rods 245 ± 10 EFPD (in lieu of 250 $\pm$ 5 EFPD proposed in July 9, 1975 submittal). Therefore, pursuant to 10CFR50, §50.90 and the provisions of the Commission's December 27, 1975 Order for Modification of License for Oconee Nuclear Station, Units 1, 2, and 3, it is requested that the control rod position limit and operational power imbalance envelope described in Attachment 3 be approved prior to attaining 235 EFPD on Oconee 1. Presently, with expected capacity factors, it is estimated that this will occur on November 28, 1975.

The limitation on power operation with one idle reactor coolant pump in each loop would be imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of operation. Pursuant to 10CFR50, §50.90, a revision to Oconee Technical Specification 3.1.1 is requested which will permit operation for a period of five days with one idle reactor coolant pump in each loop. This time period could allow repairs to the idle pumps and the return to an acceptable combination of operating reactor coolant pumps. The five days for this mode of operation is acceptable since this mode of operation is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the five day period is considered very remote. Proposed Technical Specification replacement pages are provided in Attachment 4.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

MST:vr

Attachments

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47 and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr.

William O. Parker, Jr., Vice President

ATTEST

John C. Goodman, Jr.

John C. Goodman, Jr.
Assistant Secretary

Subscribed and sworn to before me this 31st day of October, 1975.

Notary Public

My Commission Expires:

ATTACHMENT 1

RESPONSE TO MR. R. A. PURPLE'S LETTER

OF

OCTOBER 14, 1975

October 31, 1975

Question 1a

No discussion was offered as to the consequences of a break in an active cold leg of the fully active loop.

RESPONSE

During steady-state, three-pump operation, the pump in the active cold leg of the partially active loop supplies 44.6 percent of system flow, compared to the pump in an active cold leg of the fully active loop which supplies 34.7 percent of system flow. (Reverse flow in the inactive cold leg of the partially active loop amounts to 14 percent of system flow.) Therefore, placing the break at the discharge of the pump in the active cold leg of the partially active loop instead of at the discharge of the pump in an active cold leg of the fully active loop yields the most degraded positive flow through the core during the first half of the blowdown and results in higher cladding temperature. Thus, analyzing this break location is conservative in comparison with a break in an active cold leg of the fully active loop.

Question 1b

Technical Specifications will prohibit two pump operation unless an analysis is provided to support this mode of operation. Compare a break in the inactive cold leg to a break in the active cold leg.

RESPONSE

With two pumps operating, one in each loop, the maximum power level will be 51% FP, including 2 percent uncertainty, and the system flow rate is 50 percent of that for four pump operation at steady-state conditions. The idle pump in each loop is locked in position because flow is reversed in each of the inactive cold legs. (Approximately 18.8 percent of the reactor coolant flow from the downcomer plenum is directed back in each inactive cold leg.) If the flow reverses to the positive direction during the transient, an idle pump would act as a free spinning rotor with no power.

The core flow for a break in the inactive leg of a partially active loop with two pumps operating would be similar to that for a break in the active cold leg of the partially active loop with three pumps operating. During the LOCA transient, the positive driving force for both breaks is with two pumps and, therefore, the core flow would be approximately the same. The reflooding rate for the two pump case would be greater than for the three pump case because the core power is lower, 51 percent versus 77 percent, thus a lower cladding temperature rise would be expected than that predicted for the three-pump case.

A break at the pump discharge of either one of the active cold legs will cause a loss in positive flow during the first half of the transient compared to the above case. The transition from positive to negative flow should occur earlier. The negative flow would be substantially increased due to the decrease from two to one active pumps trying to force positive flow into the core region. The high negative flow rate through the core during the blowdown phase would provide good core cooling and remove a significant amount of the stored energy in the fuel. Thus the cladding temperature during this phase of the LOCA would be maintained relatively low. The reflooding phase should have the same improvement in clad temperature as described for the previous two pump case, i.e., a lower cladding temperature rise would be expected.

Therefore, the maximum cladding temperature for a LOCA during two pump operation should be approximately equal to or less than that calculated for a LOCA during three pump operation. Since the calculated peak cladding temperature for a LOCA that occurs during three pump operation gives a large margin (434F) relative to the 2200F limit, two pump operation will comply with the acceptance criteria for the ECCS set forth in 10CFR50 §50.46 and Appendix K. However, as no quantitative analysis has been performed, Technical Specifications for two pump operation will be revised to limit operation in this mode to periods not to exceed five days.

Considering the above information, the infrequency of operation with only two reactor coolant pumps, and the very low probability of a LOCA occurring in this limited time period, it is felt that this proposed restriction is appropriate.

Question 1c

Indicate and justify the worst-case pump status assumed at the time of the LOCA (tripped vs. powered).

RESPONSE

The partial loop analysis was performed assuming "pumps powered." Based on the results given in Section 5.5 of BAW-10103 for four pump operation, it was found that the "pumps-powered" case produced the highest peak cladding temperature, 2114F versus 2080F, for the "pumps-tripped" case. The difference of 34F indicates, however, that the LOCA analysis is relatively insensitive to assumptions regarding electrical power availability to the pump. Additionally, the ECCS performance analysis for three pump operation (submitted on August 1, 1975) considered two cases: a break in the active cold leg of a partially active loop and a break in the inactive cold leg of a partially active loop. These two cases can be regarded to correspond to the cases of "pumps tripped" and "pumps powered" since the core flow for a break in the active cold leg of a partially active loop and for a break in the inactive cold leg of a partially active loop is similar to the core flow with pumps tripped and powered, respectively, as can be seen by comparing Figures 2 and 4 of the partial loop analysis to Figure 5-7 of BAW-10103. Comparing results of the two partial loop cases, Figures 1 and 3 of the partial loop analysis, illustrates that the results are insensitive to a change in pump status. Additionally, since the maximum cladding temperature calculated for the partial loop analysis is 1766F, which is [illegible] less than for the same break at full power and flow conditions, a change in the pump status would not adversely affect the results.

Question 1d

Provide assurance that the PCT versus break size curve in BAW-10103 would not be significantly altered by either mode of partial loop operation.

RESPONSE

The partial loop analysis was performed assuming the worst case break (8.55 ft^2 DE, C_D = 1) reported in BAW-10103 at the maximum kw/ft limits shown in Figure 2-2. Historically, the above break has resulted in the highest cladding temperature for LOCA analysis. In general as the break size decreases, the duration of the blowdown increases which results in decreased maximum cladding temperature. Table 6-1 of BAW-10103 verifies this statement, i.e., the maximum cladding temperature decreased 195F when the discharge coefficient for a 8.55 ft^2 DE break was changed from 1.0 to 0.6.

As mentioned in the response to Question 1c, the core flows for the partial loop cases are similar to those shown in BAW-10103. Therefore, core flow for smaller breaks during partial loop operation would be similar to that shown in Section 6 of BAW-10103. With similar flow, the PCT versus break size curve should exhibit the same trend, i.e., decreasing PCT with break size. Since the PCT for the partial loop analysis is 313F less than that given for the worst break in BAW-10103, smaller breaks will exhibit larger margins of safety relative to the 2200F criterion.

Question 1e

Submit the LOCA parameters of interest identified in the "Minimum Requirements for ECCS Break Spectrum Submittals" dated April 25, 1975.

RESPONSE

The following are additional LOCA parameters of interest for the B&W Category 1 partial loop LOCA analysis.

Three Pumps, Break at Active Pump Inloop
With Idle Pump (CRAFT Run PP102 (Y1))

Figure 1 Reactor Vessel Pressure for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Figure 2 Core Water Level for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Figure 3 Downcomer Water Level for 8.55 ft^2 DE Break at Pump Discharge Partial Loop Operation, C_D = 1.0

Figure 4 Total Power for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Figure 5 Containment Pressure for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Three Pumps, Break at Idle Pump (CRAFT Run PP101 (1B))

Figure 6 Reactor Vessel Pressure for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Figure 7 Core Water Level for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Figure 8 Downcomer Water Level for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Figure 9 Total Power for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0

Computer Data for the Figures

Fig. No.		Version Date		
1	CRAFT 2, Version 5PP	4/17/75	PP102(Y1)	07/25/75
2	REFLOOD 2, No Loop Version 2	12/20/74	PR102(2I)	07/28/75
3	REFLOOD 2, No Loop Version 2	12/20/74	PR102(2I)	07/28/75
4	CRAFT 2, Version 5PP	4/17/75	PP102(Y1)	07/25/75
5	CONTEMPT, Version 15	11/15/74	PC100(FR)	07/11/75
6	CRAFT 2, Version 5PP	4/17/75	PP101(1B)	07/15/75
7	REFLOOD 2, No Loop Version 2	12/20/74	PR101(NJ)	07/15/75
8	REFLOOD 2, No Loop Version 2	12/20/74	PR101(NJ)	07/15/75
9	CRAFT 2, Version 5PP	4/17/75	PP101(1B)	07/15/75

Other Codes Used
(Figures Provided in Original Report)

Version Name	Version Date	Run Name	Run Date
THETA 1B, Version 6F	1/23/75	PT101(1J)	07/18/75
THETA 1B, Version 6F	1/23/75	PT1A2(HT)	07/30/75

Question 2

Provide information concerning measures to assure long term cooling capability.

RESPONSE

Boric Acid Concentration

In lieu of the three modes of operation proposed on April 16, 1975, to limit boron concentration buildup, Duke Power Company proposes to implement an alternate method of providing boron dilution during the long-term cooling phase following a postulated LOCA. This alternate method is a more viable method in that this mode is independent of the normal ECC mode, and only limited operator actions are required compared to those required for the three modes proposed earlier.

The alternate method of boron dilution consists of providing a gravity flow path for the reactor coolant from the hot leg nozzle to the reactor building sump through the decay heat drop line. The existing 1" ID decay heat vent line, with modification of valve LP-23 to allow operation from the control room, can provide gravity flow for the reactor coolant from the hot leg nozzle to the reactor building sump. However, this line is not single failure proof. To make this flow path single failure proof, an additional 1" ID drain line with two electric motor operated valves (LP-x and LP-y) will be installed on the decay heat drop line above Valve LP-1. These flow paths will result in drainage of highly concentrated water from the top of the core for all postulated loss-of-coolant accidents, allowing dilute water to enter the core and thus promote significant core circulation. The minimum driving head in these lines is 11.3 feet, and therefore, a gravity flow in excess of 40 gpm will exist in each of these lines. This would result in a core circulation in excess of 40 gpm; and based on the analysis submitted on April 16, 1975, it can be concluded that utilization of any one of these flow paths will limit boron concentration buildup to a C/C_o of less than 11. It should also be noted that utilization of these flow paths will not create any adverse effects on the normal ECC System.

The responses to the various questions applicable to this alternate method of boron dilution are as follows:

(a) The gravity flow paths for the proposed mode of boron dilution are identified in the attached PO drawings.

(b) The elevation of the 36" ID outlet nozzle is 809'6". The elevation of the highest point in the decay heat line above the points where the vent line and the drain line are tapped is 808'. The elevation of the decay heat vent line tapping is 797'. The elevation of the drain line to be installed will be 795'3".

(c) The electrical power supplies for LP1, 2, 23, X and Y have not been determined at present. They will be arranged such that LP-X and LP-Y have one source which is independent of the sources for valves LP-1, LP-2 or LP-23 such that a single electrical failure cannot affect both dilution paths. All valves will be powered from an Engineered Safeguards switchboard and all valve operators will be above the post-LOCA water level so that they cannot become submerged. The capacity of the emergency power source is more than adequate to carry these additional loads.

(d) The only operator action required for initiation of the boron dilution loop is to open the valves LP-1, LP-2, LP-23, LP-S and LP-Y from the control room. For large breaks, these valves may be opened within 24 hours following the LOCA; and for small breaks, the valves are to be opened only after the Reactor Coolant System is depressurized.

(e) Remote readouts of dilution flows are not required for this method of boron dilution as flow is assured due to the dependence upon the dilution flow.

(f) Since the minimum driving head in these drain lines is 11.3 feet, a gravity flow in excess of 40 gpm will exist in each of these lines ($Q = A\sqrt{2gh}$).

(g) Remote valve operability and flow through these lines can be verified at the time the necessary modifications are implemented.

(h) The design, engineering evaluation, and procurement effort for material and equipment needed for the necessary station modification are currently being initiated. The estimated time for the material and equipment acquisition is approximately two years and the station modification can be completed for each unit during the unit's first refueling following the material delivery. In the meantime, Mode 1, proposed on April 16, 1975, can be utilized as a temporary measure to provide added assurance of long-term cooling capability. The applicable emergency procedure will temporarily be revised by January 31, 1976, to incorporate the necessary operator action for utilization of this mode.

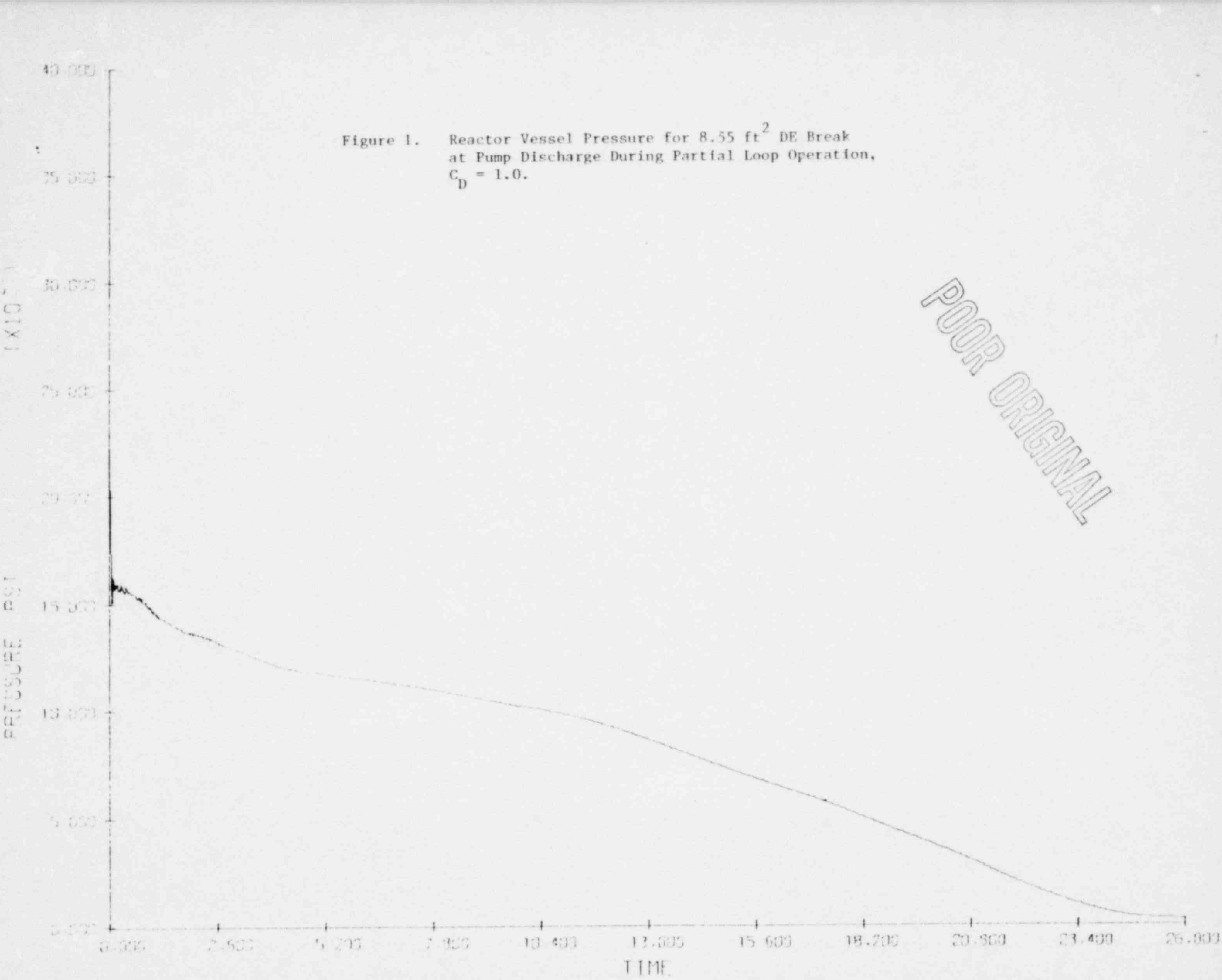


Figure 1. Reactor Vessel Pressure for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0.

Figure 2. Core Water Level for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0.

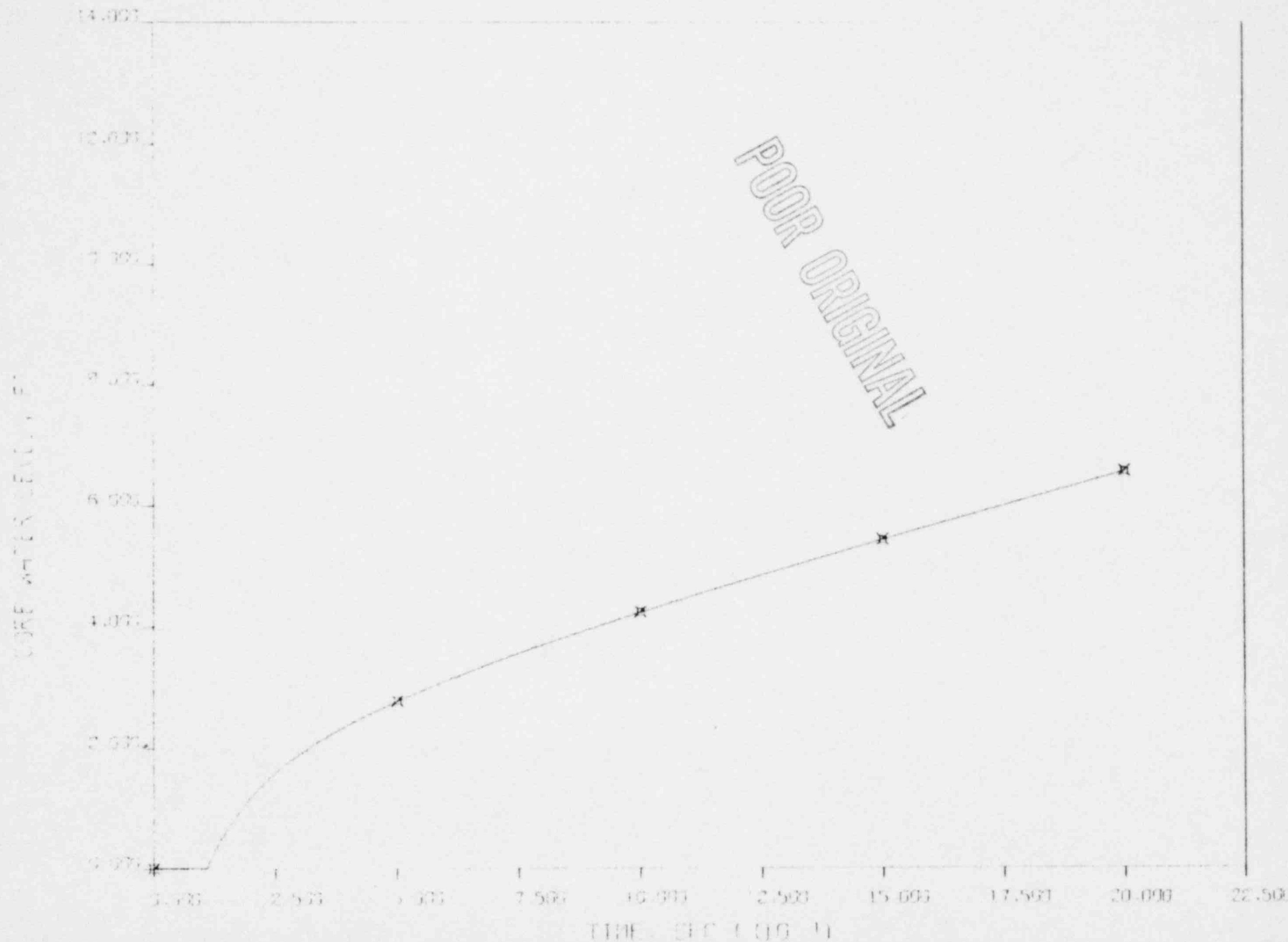


Figure 3. Downcomer Water Level for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, $C_D = 1.0$.

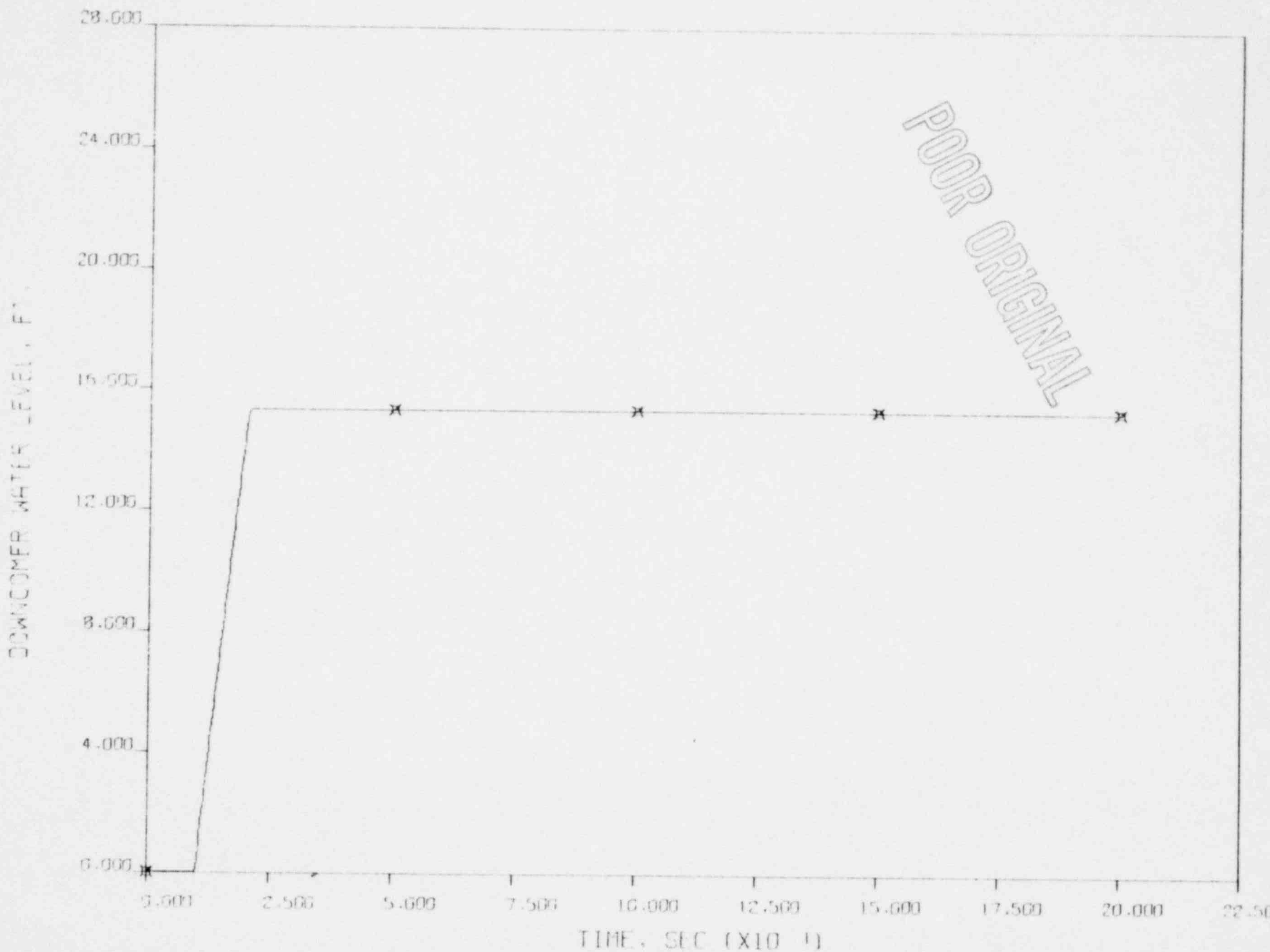


Figure 4. Total Power for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, $C_D = 1.0$.

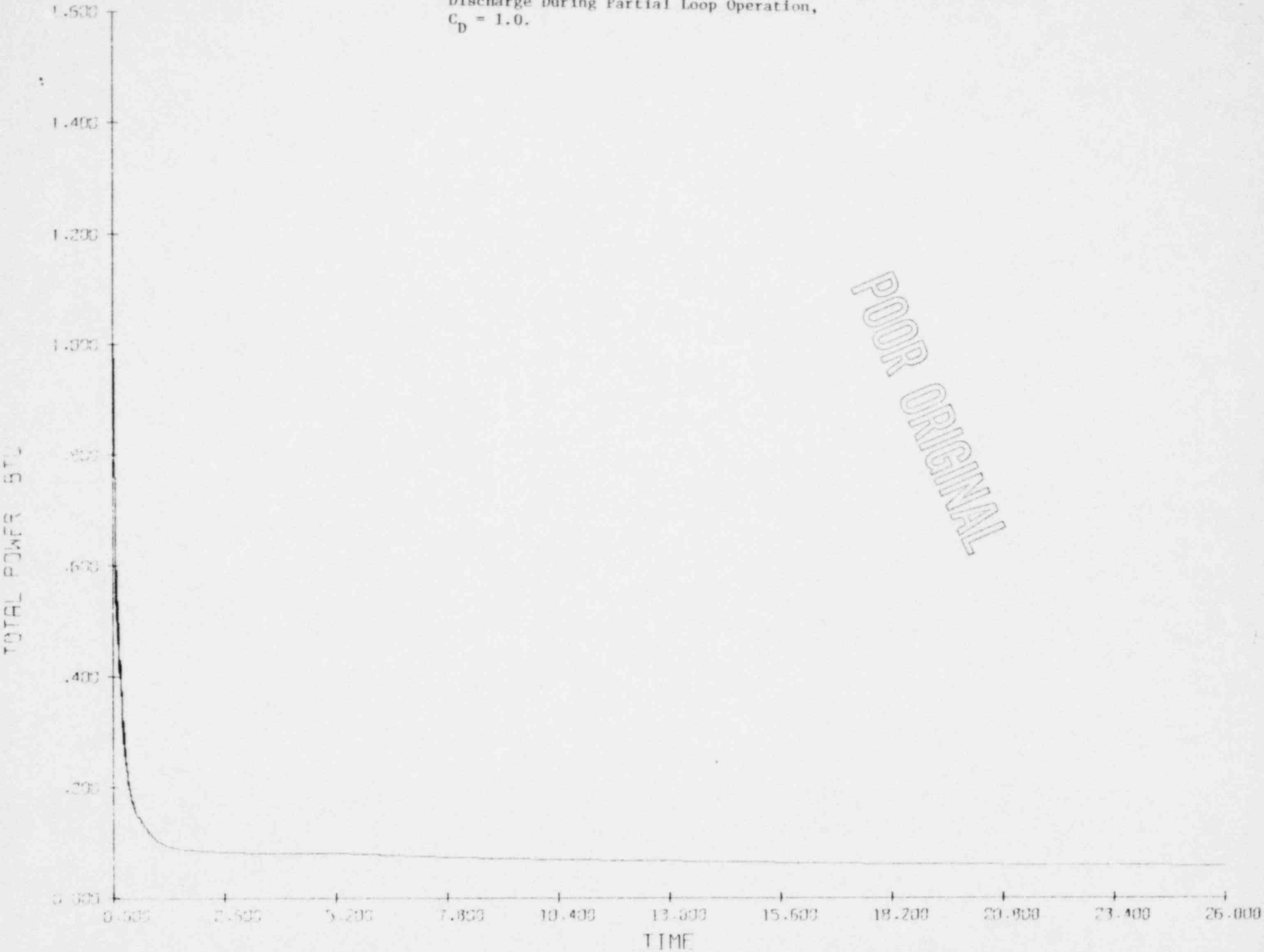


Figure 5. Containment Pressure for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation $C_D = 1.0$.

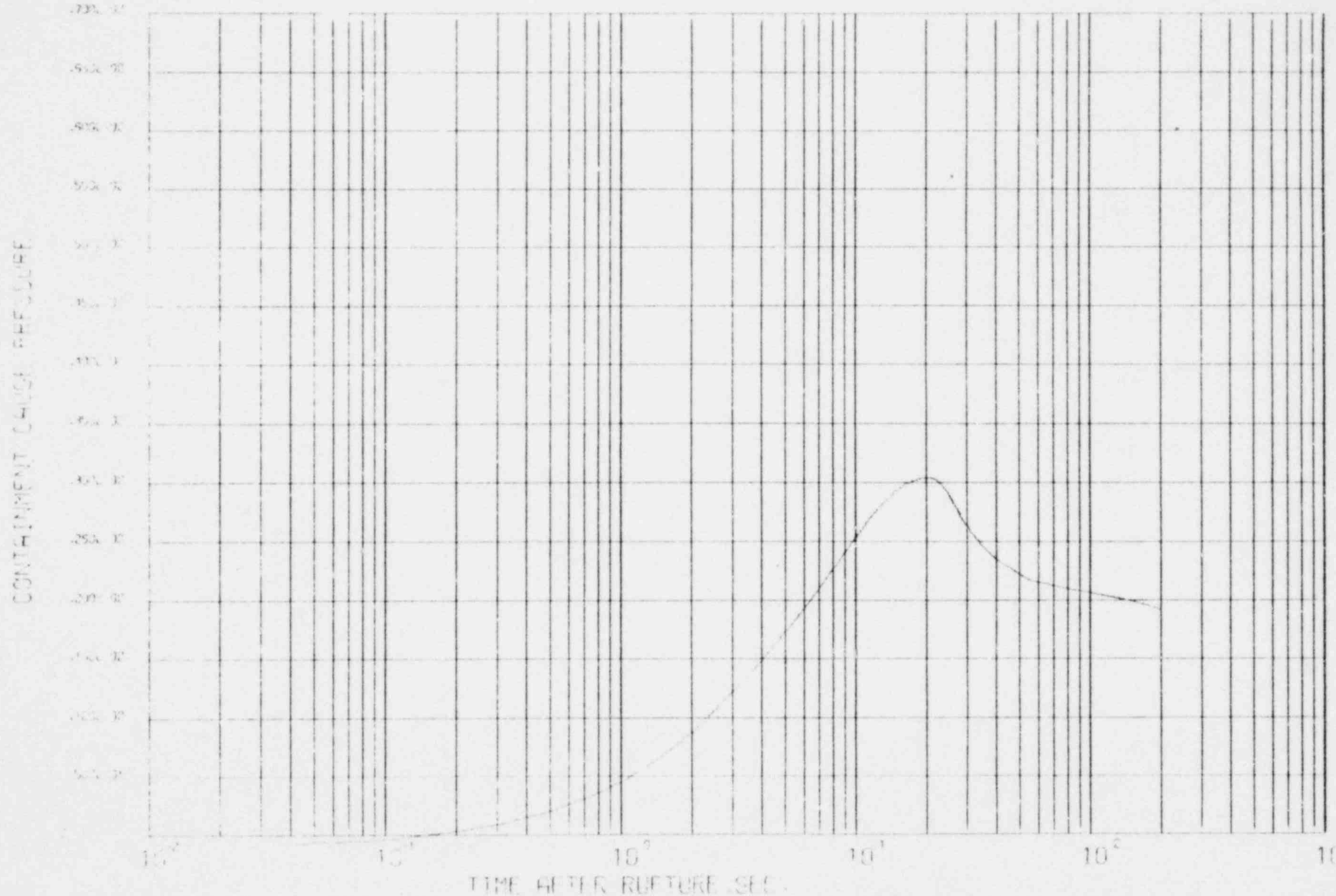


Figure 6. Reactor Vessel Pressure for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, C_D = 1.0.

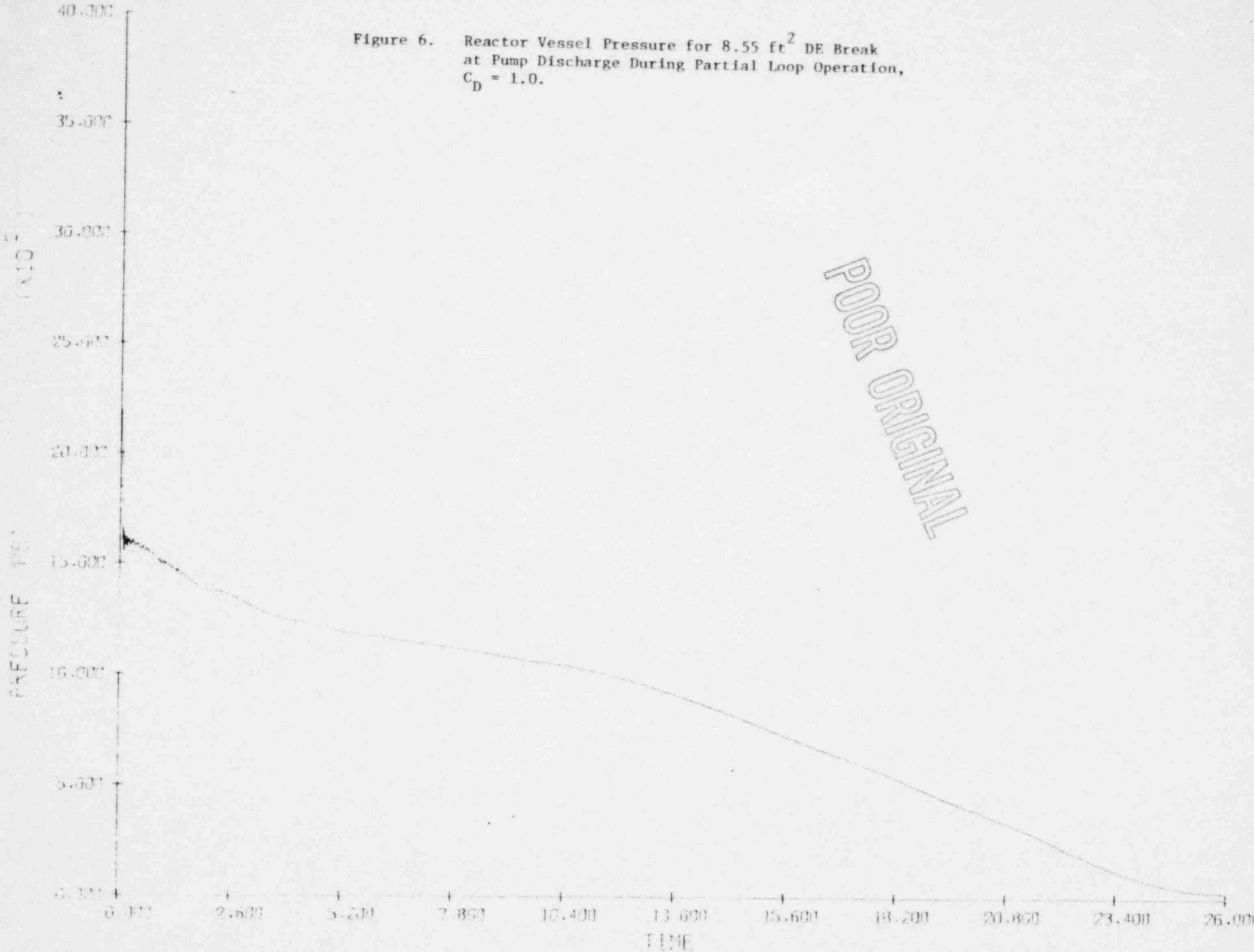


Figure 7. Core Water Level for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, $C_D = 1.0$.

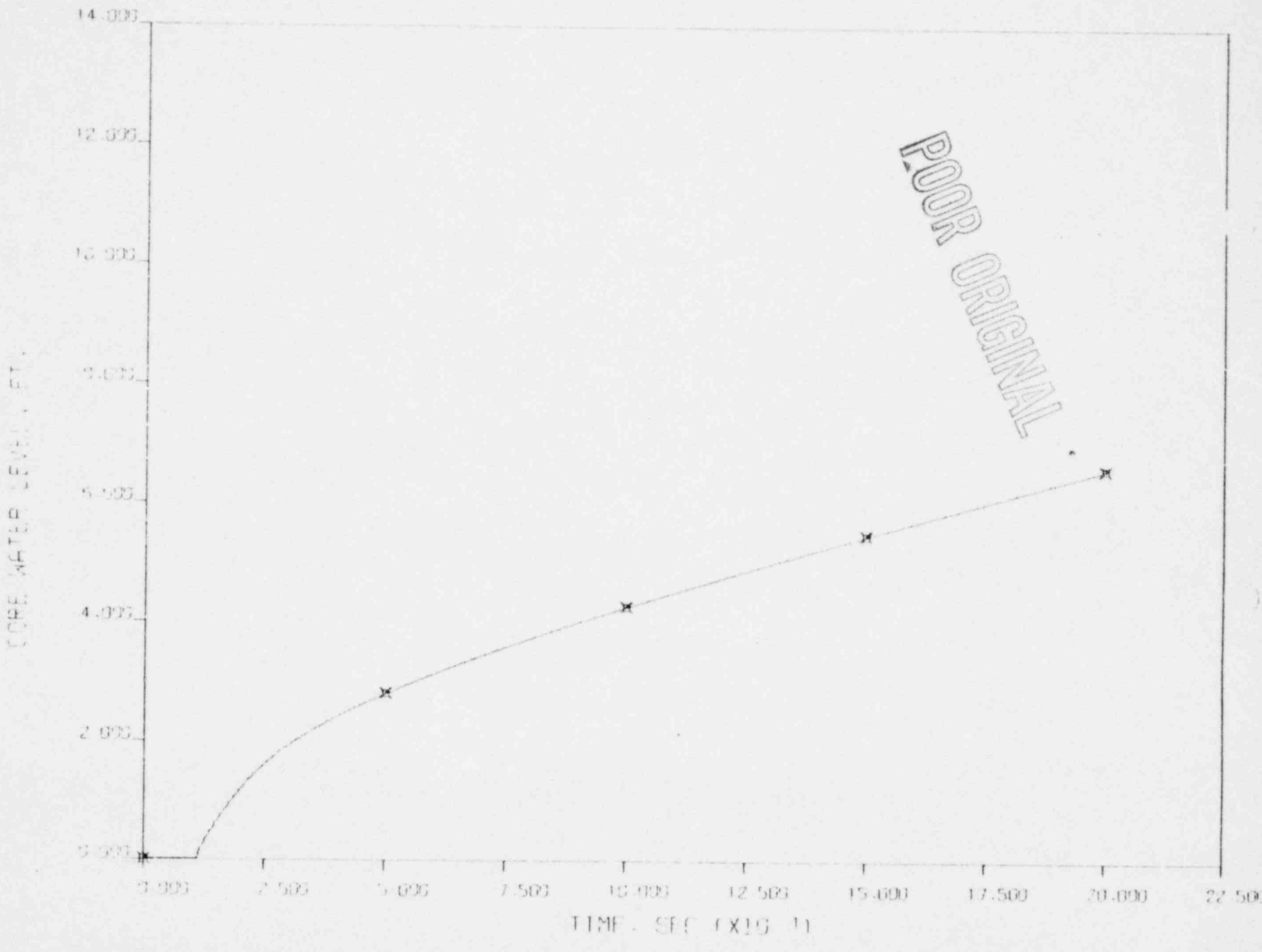


Figure 8. Downcomer Water Level for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, $C_D = 1.0$.

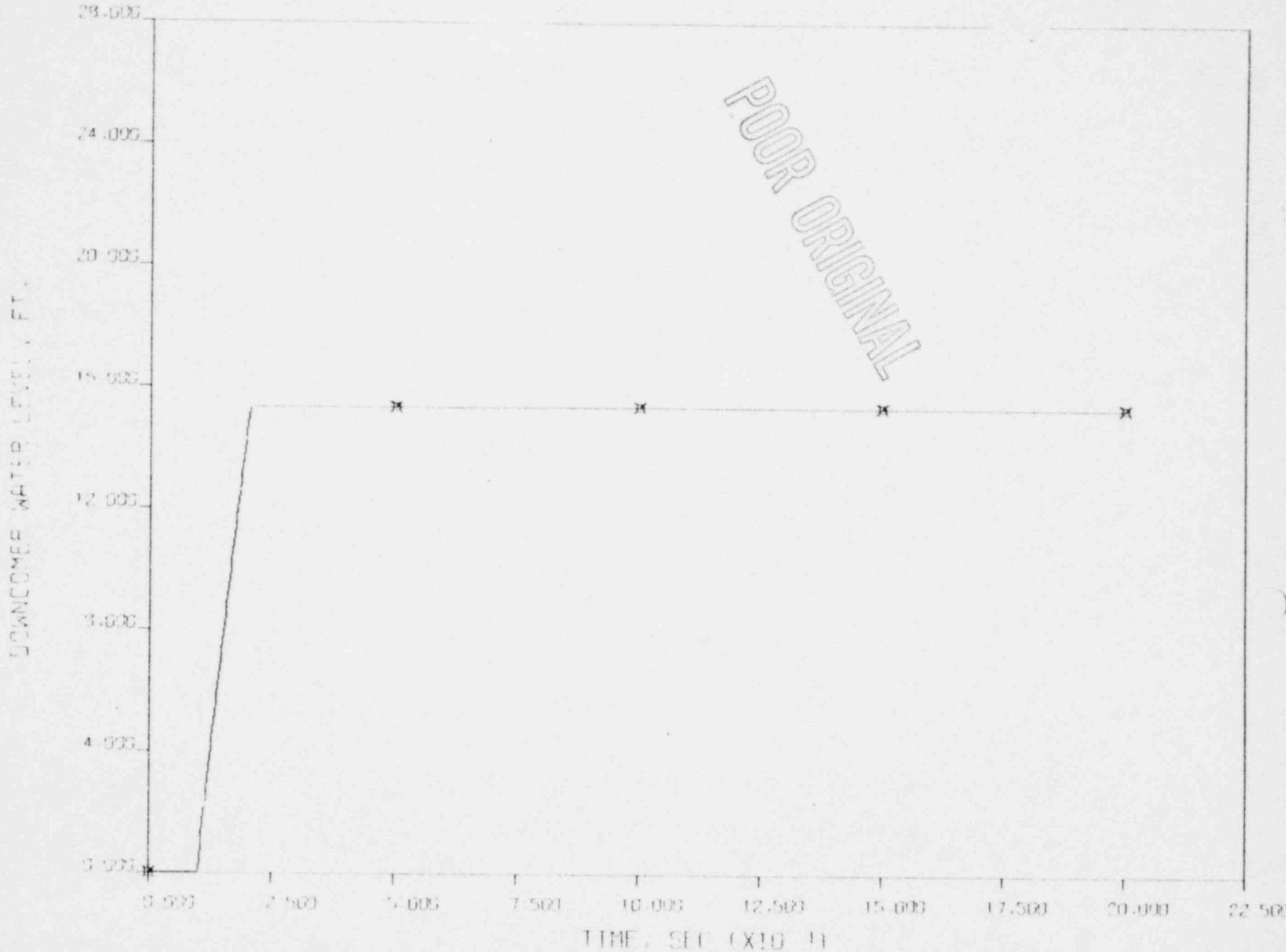
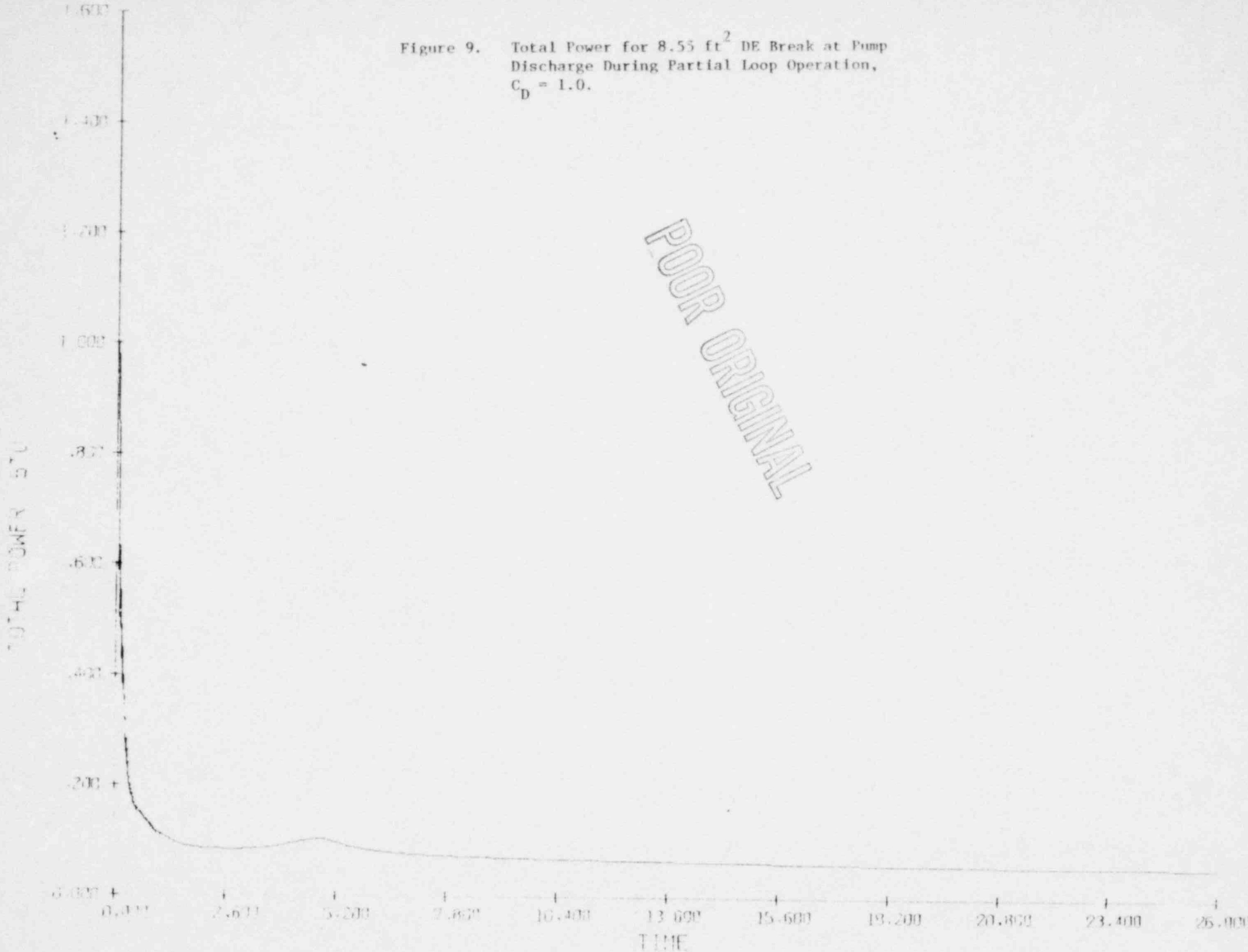


Figure 9. Total Power for 8.55 ft^2 DE Break at Pump Discharge During Partial Loop Operation, $C_D = 1.0$.



ATTACHMENT 2

RESPONSE TO MR. R. A. PURPLE'S LETTER

OF

OCTOBER 15, 1975

October 31, 1975

Question 1(a)

Confirm that your analyses considered a single failure or operator error that causes any manually controlled electrically-operated valve to move to a position that could adversely affect the ECCS (i.e., Service Water System Valves, Building Spray System Valves, Boron Dilution Valves, etc.).

RESPONSE

Analyses have been performed to evaluate the effects of a single failure or operator error that cause any manually-controlled, electrically-operated valve to move to a position that could adversely affect ECCS performance. Based on these analyses, it has been concluded that no station modifications or changes to the Technical Specifications are necessary to protect against such a single failure or operator error.

Questions 1(b) and 1(c)

Drawing PO-102A1 shows LPI valves LP-V4A and LP-V4B to be normally closed. To allow operation of the LPI-to-LPI crossover subsequent to a CFT line break and a single active component failure, these valves must be required by Station Technical Specifications to be open, power removed, and breakers locked open.

Your evaluation on page 2 of Attachment 3 for the DH cooler inlet and outlet valves does not appear to be correct. For a CFT line break and an inadvertent closure of a valve in the unaffected low pressure injection line, the LPI-to-LPI crossover would be rendered ineffective. Station Technical Specifications must require that power be disconnected and breakers locked open to LPI motor-operated valves downstream of the LPI-to-LPI crossover (valves normally open) and that a periodic test be performed to warn of abnormal leakage of the check valves in the LPI injection lines inside containment. These changes provide further assurance that abundant core cooling is available for a CFT line break and minimize the potential for a LOCA outside containment.

RESPONSE

The guillotine break of the core flood tank line between the reactor vessel nozzle and the first check valve has been previously analyzed in FSAR Supplement 14, submitted to the Commission on January 29, 1973. That analysis imposed the worst case single failure on the Low Pressure Injection System. The single failure imposed was that the active LPI pump was lined up to the core flood line where the break occurs and that the other LPI pump was inoperative by the criteria of an assumed single active failure. Although adequate core cooling was demonstrated in the short and long term, it was stated that increased long-term safety margin could be obtained by operator action. This action was stated as being easily taken within 15 minutes after the CFT line break and consisted of the operator opening the remotely operated cross connect valves at the LPI pump discharge and equalizing flow through the two LPI trains. These valves are provided with manual operators and could easily be manually opened within 15 minutes of a postulated CFT break.

With regard to single failure, the possibility of the LPI discharge header valves LP-17 (LP-V4A) and LP-18 (LP-V4B) or the decay heat cooler inlet or outlet valves failing in the closed position would have the same effect as the originally postulated single failure analysis. If this condition did occur, the valves could easily be manually opened to assure abundant long-term cooling. The reliance upon operator action to correct a single failure in one of the LPI discharge valves, decay heat cooler inlet or outlet valves, or the LPI-LPI crossover valves is consistent with station operating practices. Therefore, the requirement to assure abundant long-term cooling will, at most, require manual operation of one valve within 15 minutes of the postulated accident. This time of 15 minutes is conservative since these valves must be operated prior to switching suction of the LPI, HPI and Reactor Building spray pumps from the borated water storage tank to the Reactor Building sump. This is not expected to occur prior to 30 minutes from the postulated accident.

It is considered that the use of operator action within a reasonable time period is a satisfactory method of assuring abundant long-term cooling following a postulated core flood tank line break and the imposed single failure on the LPI system.

Question 1d

With regard to the failure open of a CFT vent valve, the position that this is a very low probability event is not sufficient justification of the Oconee design. Technical Specifications must require that powerto these valves be disconnected and the breakers locked open.

RESPONSE

The design of the core flood tank vent system incorporates an electrically-operated vent valve and a manually (locally) operated needle throttling valve. The needle valve, downstream of the electrically-operated valve, is provided to control the rate of core flood tank venting when the electrically-operated valve is opened. Oconee startup procedures require that the needle valve be positioned to a predetermined position and that this position be maintained during unit operation. This assures that if the electrically-operated valve were to fail open, the blowdown rate of the core flood tank would be limited. This is verified by a recent test on Oconee 2, during which pressure decay in the CFT from a maximum of 625 psi to the low pressure alarm, 580 psi, required 17 minutes when the electrically-operated valves were open. Therefore, in the event a CFT vent valve failed open the rate of pressure decrease would not adversely affect core flood tank performance and it is considered that a Technical Specification covering operation of the electrically-operated core flood tank vent valves is not necessary.

Question 1(e)

The following motor-operated valves do not appear to be addressed in Attachment 3:

HP-27	LP-17
HP-24	LP-18
HP-25	LP-21
	LP-22

Confirm that these valves could not move to a position that could adversely affect ECCS performance.

RESPONSE

Mr. R. A. Purple's letter of June 13, 1975 requested an evaluation of the effects of a single failure or operator error that causes any manually-controlled, electrically-operated ECCS valve to move to a position that could adversely affect ECCS performance. The analysis provided on July 9, 1975 addressed manually-controlled, electrically-operated ECCS valves as requested. The above listed valves are ES actuated valves which have been previously analyzed for the affects of single failure in Table 6-2 of the FSAR.

Question 2

With regard to the discussion on submerged equipment, the analysis is insufficient to allow an adequate evaluation. Specify the scope of the study in terms of systems considered in the analysis. Confirm that post-LOCA long-term cooling requirements were considered (i.e., systems needed to limit boric acid concentration in the reactor vessel). Provide the basis for the conclusion that certain Reactor Building isolation valves would be closed upon ES actuation before becoming submerged. For these valves, indicate the expected time of isolation after a worst-case break location and compare to the expected time at which the water level in the sump would first reach the valve motor. Specify the height above the containment floor of each of these valves.

RESPONSE

The analysis for submerged equipment utilized as-built arrangement plans to calculate a post-LOCA maximum water level of 8.3 feet, an elevation of 785' - 9-5/8". Included in this analysis are the effects of all potental water sources, including the borated water storage tank. Equipment arrangement in the lower portion of the Reactor Building was evaluated and it was determined that only those valves listed in Attachment 4 of our July 9, 1975 submittal could become submerged following a LOCA. None of these valves are utilized in meeting post-LOCA, long-term cooling requirements.

With regard to valves CS-5, HP-3 and HP-4, the following discussion is provided:

1. The Reactor Building is a cylindrical structure with a sloping floor from elevations 775'-0" to 777'-6" as shown in FSAR Figure 5-1. The valve elevations are 777'-7 5/8" for CS-5 and 780'-7" for HP-3 and HP-4.

2. The maximum Engineered Safeguards actuation time is seven seconds for a 0.4 ft^2 rupture (FSAR page 14-57a) and the valve closure times are 20 seconds for CS-5 and 28 seconds for HP-3 and HP-4.

3. The mass discharged to the Reactor Building as a function of time for a 5.0 ft^2 break is shown in FSAR Figure 14-63c. Since maximum valve closure

time is of the order of 35 seconds, it can be seen that the initial blowdown phase would be completed and the mass discharged is relatively independent of time and break size.

4. T calculated volume of the Reactor Building is approximately 58,500 ft/ft. Thus, the post-LOCA water level will be about 1.1 ft. in the period necessary for valves CS-5, HP-3, and HP-4 to close.

5. Valves HP-3 and HP-4 will have adequate time to close before they are submerged. In addition, each has a redundant isolation valve outside containment which would also assure containment integrity.

6. The quench tank isolation valve, CS-5, is a normally closed containment isolation valve, opened infrequently for limited periods to drain the quench tank if necessary or during maintenance periods. The valve is fully qualified for the post-LOCA environment; however, there is a possibility that the valve will become submerged prior to the time necessary for its closure. The redundant isolation valve outside containment will provide the backup required in the event CS-5 were open and failed to close.

Question 3(a)

Page 2: Describe the tests and provide the calculations upon which the CFT line resistance is based.

RESPONSE

For Oconee 1, the average L/D for the core flooding line, including valve losses and excluding entrance and exit losses, is 357. The friction factor (f) for turbulent flow for 14 inch schedule 140 pipe is 0.0130. This results in a line resistance (k = fl/D) of 4.641. For entrance and exit losses, the following were assumed:

k = 0.23	for a slightly rounded entrance
k = 1.0	for a sharp-edged exit
k = 0.597	for the flow restriction in the CFT line

The resulting core flooding tank line resistance is k = 6.468. A resistance of 6.5 was used in the analysis.

A test was performed at Oconee 1 which demonstrates the conservatism of the core flood tank line resistance assumed in the analysis. The test procedure consisted of filling the core flood tank with demineralized water to the operational "full" level, pressurizing the tank to 600 psig with nitrogen, and by means of the isolation valve, discharging the water into the reactor vessel and the fuel transfer canal. The tank pressure and water level in the tank were monitored throughout the discharge.

The results of the test verified the conservatism of the calculation of core flood tank line resistance.

Subsequent to the above test, flow restrictions were placed in each core flood line. A flow test was conducted on the restrictors at Alden Research Laboratories of Worcester Polytechnic Institute, Holden, Massachusetts for the purpose of establishing the maximum resistance coefficient. This was accomplished by flow calibration tests on a full-sized prototype insert. The results of the test showed that the maximum loss coefficient is 0.224 based on the throat area. Since CRAFT uses the core flooding line area (.7213 ft^2), the corrected loss coefficient for use in CRAFT is .597.

Question 3(b)

Page 9 indicated that the REFLOOD code version names are different on Figures 4 and 8. If the codes are not the same, describe the differences.

RESPONSE

The version name associated with Figure 8, run number RF143(SY) is incorrect. Both that run and run number RF141(IV) were run on the same version of REFLOOD, i.e., REFLOOD 2 Version 2-no loop-dated 12/20/74. This has been verified by checking the actual printout of both runs.

Question 3(c)

Figures 4 and 8: It is not obvious from these plots that flooding rates of less than 1 in/sec are not predicted. As indicated in the staff's "Minimum Requirements for ECCS Break Spectrum Submittals" dated April 25, 1974, resubmit these figures utilizing engineering graph paper to such a scale as to allow greater reading accuracy.

RESPONSE

Figures 4 and 8 have been redrawn on engineering graph paper and are attached.

Question 3(d)

Figure 2: Explain what is causing the distinct second reflood peak at about 60 seconds and relate to the same plot at the two-foot elevation in BAW-10103.

RESPONSE

The second peak results from three factors:

(1) The peak occurred after a switch in flooding rates.

(2) Flooding rate intervals in the Oconee 1 two-foot case are slightly different from the two-foot case in BAW-10103.

(3) For power peaks at the two-foot elevation, the FLECHT correlation is somewhat unstable.

Because of FLECHT instability, the slight shift in flooding rate intervals resulted in a temporary decrease in heat transfer coefficients for the Oconee 1 case during the time interval in question. This decreased cooling allowed a simultaneous increase in cladding temperature during that time. Later in the flooding rate interval, heat transfer coefficients increased. This sequence caused the cladding temperature to peak and then decrease.

Question 3(e)

Figure 5: Explain the drop in heat transfer coefficient at about 55 seconds. (Relative to BAW-10103)

RESPONSE

The drop in heat transfer coefficients at approximately 55 seconds is related to the factors described in the response to Question 3(d).

Question 3(f)

Explain why the hot spot shifted to the unruptured node at tht two-foot and four-foot elevation (relative to BAW-10103).

RESPONSE

The ruptured nodes cladding temperatures in the Oconee 1 study were 120F and 160F lower than that reported in BAW-10103 for the two and four-foot elevations, respectively. This means that the contribution of metal-water reaction, which increases at higher temperatures, is much less in Oconee 1. This tends to keep the ruptured node temperature below unruptured node temperatures for Oconee 1.

In addition, the Oconee 1 LOCA limits for these elevations yield peak cladding temperatures far below the 2200F limit. Had the linear heat rate been raised, the clad temperatures would have approached 2200F more closely. Then, it is probable that temperatures would have been higher in the ruptured nodes rather than in the unruptured nodes due to the effects of metal-water reaction on the inside and outside surfaces of the cladding.

Question 3(g)

Provide the value of volumetric average fuel temperature assumed in the Oconee 1 calculations (at 18 kw/ft with 580F sink temperature).

RESPONSE

For 18 kw/ft volumetric average fuel temperature is 3030F.

Question 3(h)

It is not apparent that you have sufficiently specified and justified all input parameters revised for the Oconee 1 analysis. For example, no explanation was given for the changes in initial pin pressure (page 7) relative to the generic calculation in BAW-10103. Confirm that all input changes have been identified and explained.

RESPONSE

The LOCA limits given in Figure 1 of the Oconee 1 ECCS Evaluation were analyzed at the worst pin pressure (time-in-life). Because the fuel parameters for the Oconee 1, Batch 4 fuel are different from that analyzed in BAW-10103, the initial pin pressures given in Page 7 of ATtachment 2 are different from the generic calculation. The effect of different fuel parameters on the worst time-in-life is illustrated in Figures 5-4 and 5-5 of BAW-10103. As shown in the time-in-life study of Section 5.4 of BAW-10103, the worst pin pressure will cause rupture during blowdown. In order to rupture during blowdown, the initial pin pressure was different for the Oconee 1 ECCS evaluation.

In addition to the pin pressure, other changes were as follows:

(a) Various pin dimensions (e.g., pellet diameter and plenum volumes), due to changes in fuel.

(b) System enthalpies and steam generator heat loads, due to a power of 2568 MWt versus 2772 MWt in BAW-10103.

(c) Core flood tank line resistance, due to the reasons given in the response to Question 3.a preceding.

The changes described here, and those listed in the Oconee 1 ECCS Evaluation Report, represent all changes made to the generic 177 lowered loop input in BAW-10103.

Question 4

It is noted that motor-operated valves LP-21, LP-22 and HP-24 from the BWST are shown normally closed. It appears that, assuming sufficient static head were available, the potential for a water hammer when ECC is injected into a dry line would be reduced considerably if these valves were normally left open. Please discuss.

RESPONSE

The BWST supply valve, LP-26, is open during unit operation to assure that a source of borated water is continuously available at valves LP-21, LP-22, and HP-24. The Low Pressure Injection and High Pressure Injection Systems are normally full of water from previous system uses. However, if it is assumed that the systems were dry downstream of LP-21, LP-22, and/or HP-24, the possibility of a water hammer is remote. This is primarily due to the fact that the lines between the ES valves and the pumps are relatively short and do not involve significant elevation differences and the fact that the pressure at the ES valves is limited to the static head of the BWST (approximately 24 psi). Therefore, when the ES valves are opened, the short lines, small elevation differences, the low pressure of the system, and the cushioning effect of entrapped air would prevent significant momentum buildup of the flowing water, limiting the possibility of significant water hammers in the system. Consequently, it is considered that these ES actuated valves do not need to be normally open.

Question 5

Discuss how it was intended that the LPI-to-LPI crossover would be actuated after a CFT line break (and a failure of the diesel on the unaffected low pressure injection line)... noting that one of the crossover valve motors would also be rendered inoperable.

RESPONSE

An analysis of the postulated CFT line break accident was presented in FSAR Supplement 14. As stated in Supplement 14, an increased safety margin can be obtained by operator action to initiate low pressure injection flow through the unbroken core flood line. Such operator action can eaily be taken within 15 minutes and consists of opening the remotely-operated cross connect valves at the discharge of the LPI pumps. In the event of a failure of one of the LPI crossover valves to operate remotely, the valves can be manually operated.

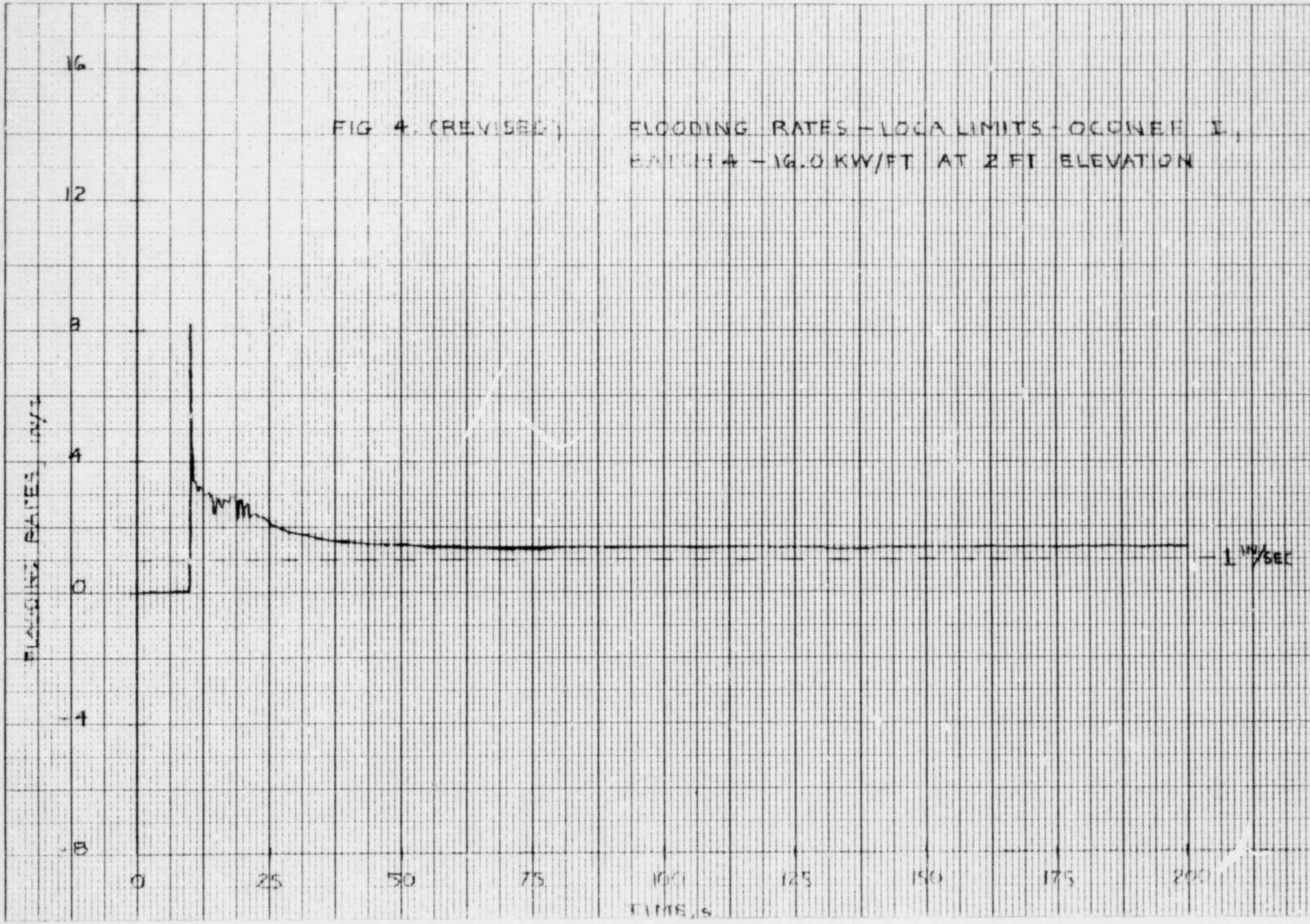


FIG 4. (REVISED) FLOODING RATES – LOCA LIMITS – OCONEE I,
[illegible] 4 – 16.0 KW/FT AT 2 FT ELEVATION

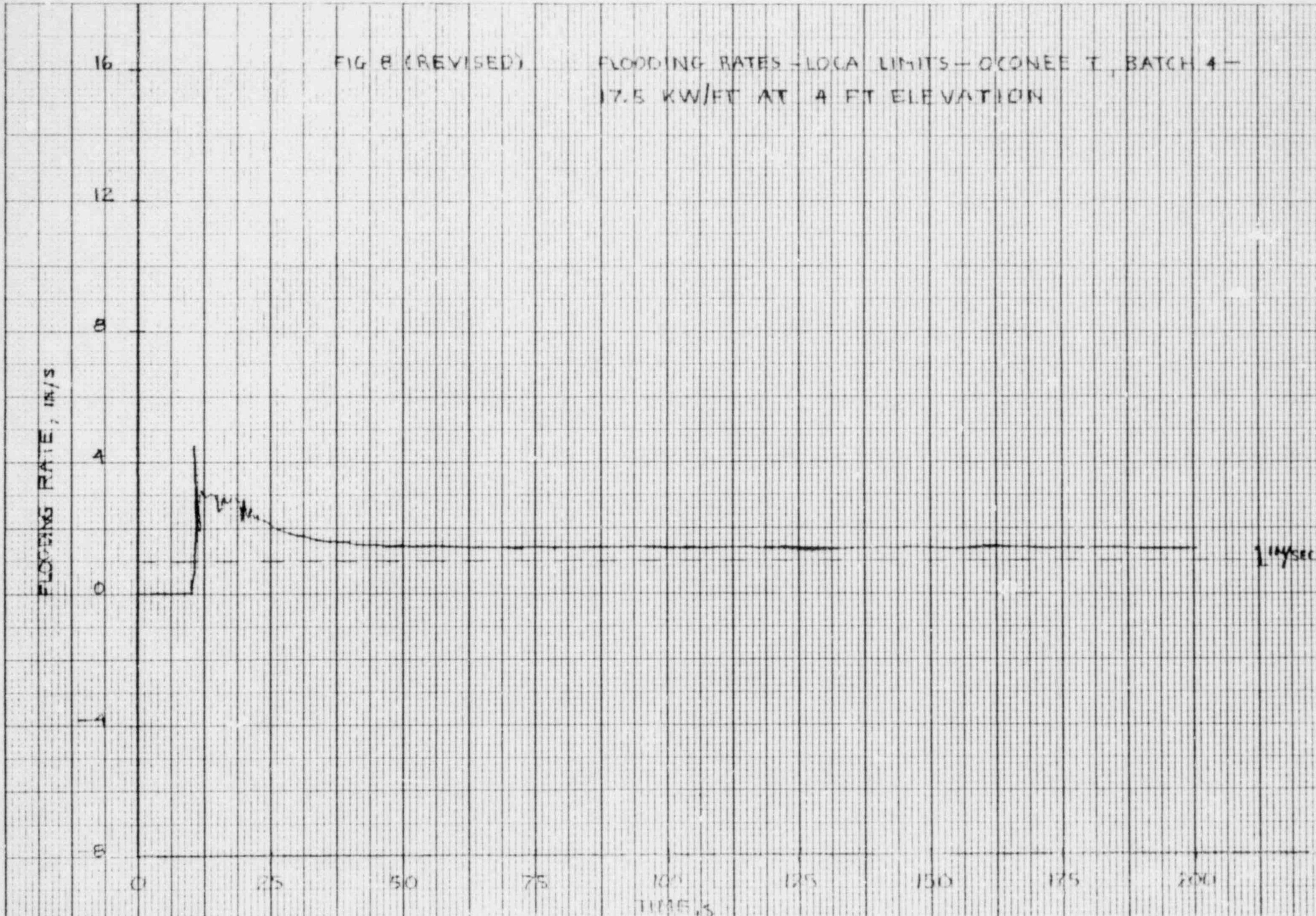


FIG 8 (REVISED) FLOODING RATES – LOCA LIMITS – OCONEE I, BATCH 4 – 17.5 KW/FT AT 4 FT ELEVATION

ATTACHMENT 3

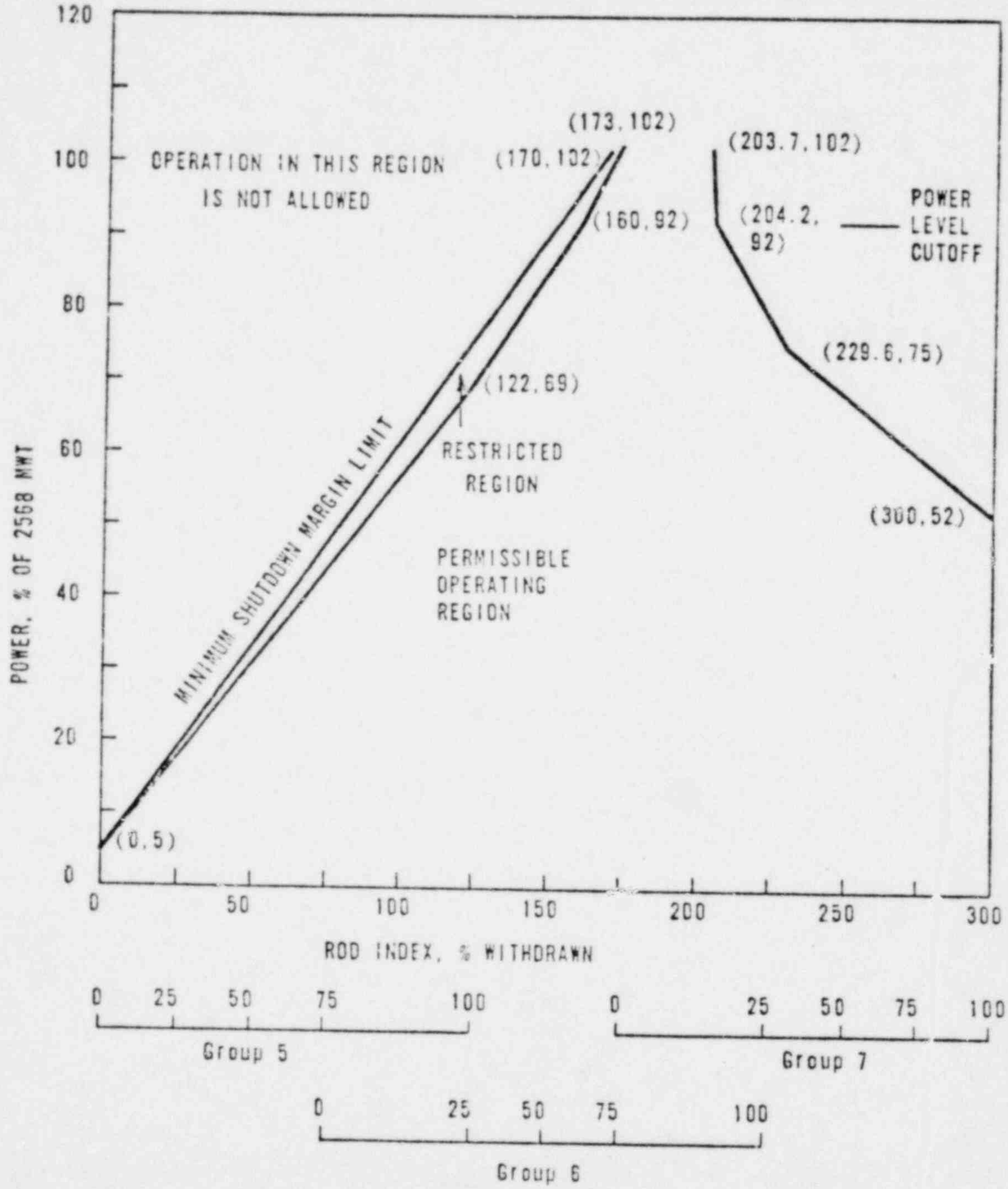
CONTROL ROD POSITION LIMITS

AND OPERATIONAL POWER IMBALANCE LIMITS

? OCONEE 1

October 31, 1975

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE TO THE PERIOD FROM 50±5 EFPD TO 245 ± 10 EFPD



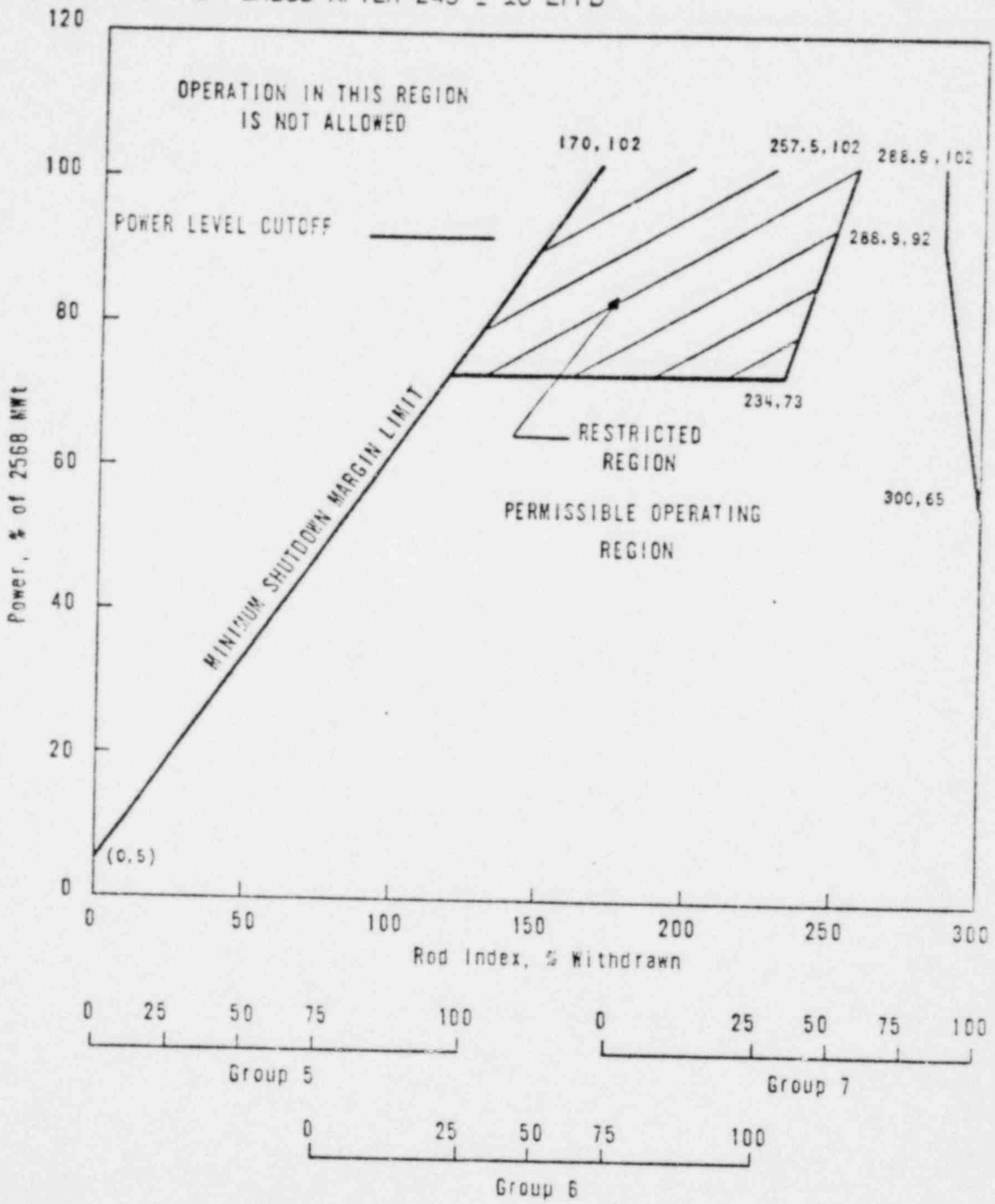
Rod index is the percentage sum of the withdrawal of Groups 5,6 and 7

UNIT 1
ROD POSITION LIMITS FOR
4 PUMP OPERATION

OCONEE NUCLEAR STATION

Figure 3.5.2-1A1

ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE
TO THE PERIOD AFTER 245 ± 10 EFPD

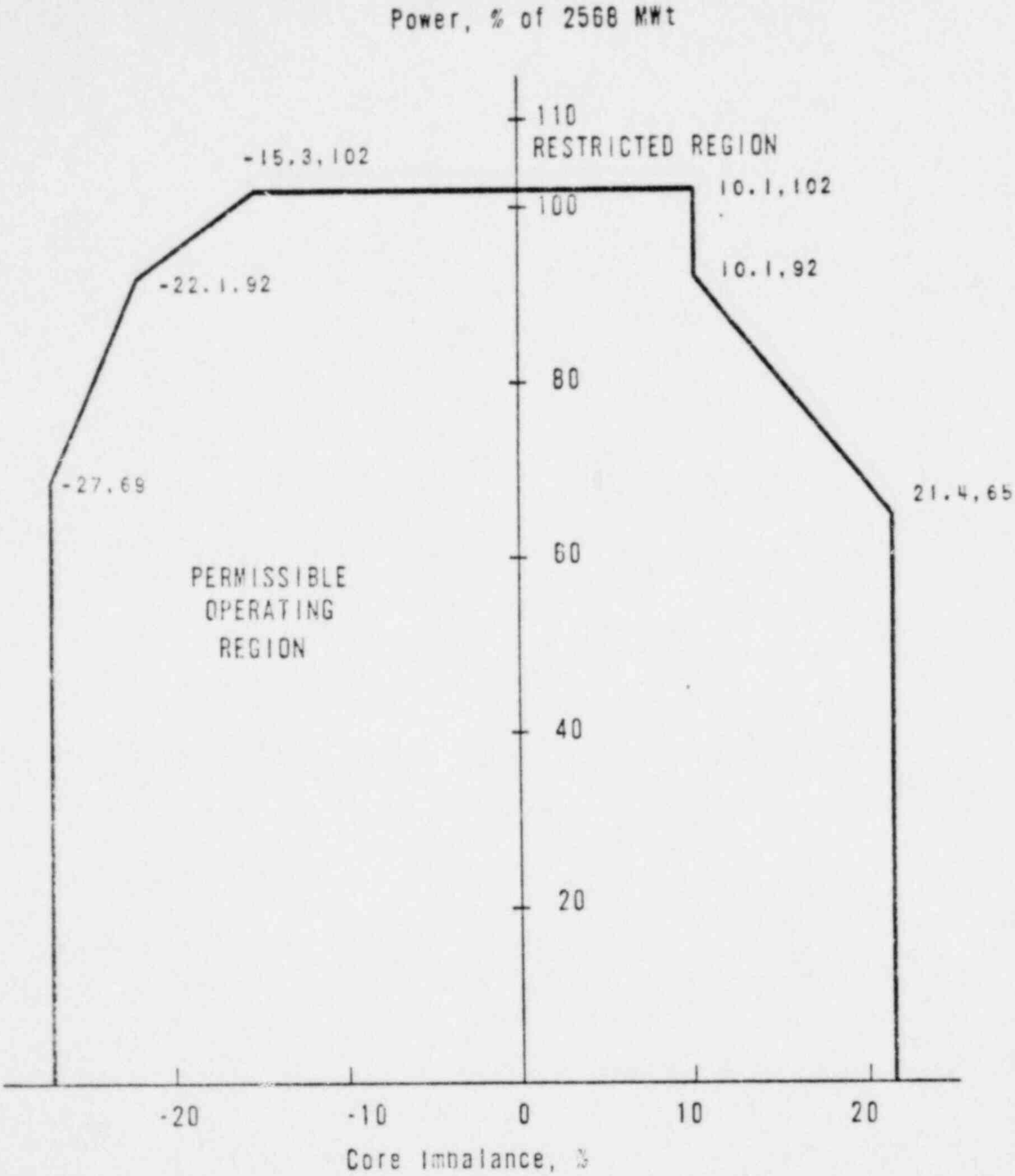


Rod index is the piecentage sum of the withdrawal of Groups 5,6 and 7

UNIT 1
ROD POSITION LIMITS FOR
4 PUMP OPERATION

OCONEE NUCLEAR STATION

Figure 3.5.2-1A2



UNIT 1
OPERATIONAL POWER IMBALANCE ENVELOPE

OCONEE NUCLEAR STATION

Figure 3.5.2-3A

ATTACHMENT 4

PROPOSED TECHNICAL SPECIFICATION REVISIONS

FOR

REACTOR OPERATION WITH TWO

REACTOR COOLANT PUMPS

October 31, 1975

3 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system components which must be met to ensure safe reactor operation.

Specification

3.1.1 Operational Components

a. Reactor Coolant Pumps

1. Whenever the reactor is critical, single pump operation shall be prohibited, single-loop operation shall be restricted to testing, and other pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

2. Except for test purposes and limited by Specification 2.3, power operation with one idle reactor coolant pump in each loop shall be restricted to five days. If the reactor is not returned to an acceptable RC pump operating combination at the end of the five day period, the reactor shall be in a hot shutdown condition within the next 12 hours.

3. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one low pressure injection pump is circulating reactor coolant.

b. Steam Generator

1. One steam generator shall be operable whenever the reactor coolant average temperature is above 250°F.

c. Pressurizer Safety Valves

1. All pressurizer code safety valves shall be operable whenever the reactor is critical.

2. At least one pressurizer code safety valve shall be operable whenever all reactor coolant system openings are closed, except for hydrostatic tests in accordance with the ASME Section III Boiler and Pressure Vessel Code.

Bases

The limitation on power operation with one idle RC pump in each loop has been imposed since the ECCS cooling performance has not been calculated in accordance with the Final Acceptance Criteria requirements specifically for this mode of reactor operation. A time period of five days is allowed for operation with one idle RC pump in each loop to effect repairs of the idle pump(s) and to return the reactor to an acceptable combination of operating RC pumps. The five days for this mode of operation is acceptable since this mode is expected to have considerable margin for the peak cladding temperature limit and since the likelihood of a LOCA within the five-day period is considered very remote.

A reactor coolant pump or low pressure injection pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One low pressure injection pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less. (1)

The low pressure injection system suction piping is designed for 300°F and 370 psig; thus the system with its redundant components can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident at hot shutdown. (5) The pressurizer code safety valve lift setpoint shall be set at 2500 psig ± 1% allowance for error and each valve shall be capable of relieving 300,000 lb/hr of saturated steam at a pressure no greater than 3% above the set pressure.

REFERENCES

(1) FSAR Tables 9-11 and 4-3 through 4-7.

(2) FSAR Sections 4.2.5.1 and 9.5.2.3.

(3) FSAR Section 4.2.5.4.

(4) FSAR Sections 4.3.10.4 and 4.2.4.

(5) FSAR Sections 4.3.7 and 14.1.2.2.3.