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Mr. Purple	1 signed			SENT NRC PDR XXX SENT LOCAL PDR XXX

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DESCRIPTION: Ltr re our ltr of 6-5-75, and 7-3-75 trans the following:

ACKNOWLEDGED

DO NOT REMOVE

PLANT NAME: Kewaunee

ENCLOSURES: Calculation and submittal of the ECCS performance per the 3-15-75 version of the Westinghouse model. Analysis conforms to direction provided by re letters.....

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FOR ACTION/INFORMATION

wtm 7-11-75

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WISCONSIN PUBLIC SERVICE CORPORATION

P.O. Box 1200, Green Bay, Wisconsin 54305

July 8, 1975

RECEIVED 7-11 U.S. NUCLEAR REGULATORY COMMISSION Mail Section WM

RECEIVED JUL 11 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

Mr. Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Purple:

Subject: Docket 50-305
Operating License DPR-43
ECCS Evaluation Model Analysis

References: Letter from Mr. R. A. Purple to Mr. E. W. James, dated June 5, 1975. Letter from Mr. K. R. Goller to Mr. E. W. James, dated July 3, 1975

Please find enclosed a copy of Westinghouse's calculation and submittal to Wisconsin Public Service of the ECCS performance per the March 15, 1975 version of the Westinghouse model. This analysis is provided to conform to the direction provided by the referenced letters.

In our letter of June 16, 1975 we stated that the required July 9, 1975 submittal date was not practical. We did, however, direct Westinghouse to proceed with the calculation immediately upon receipt of your June 5, 1975 letter, prior to our review and evaluation of the validity of the direction provided by that letter. The analysis results were received on July 7 following overtime analysis activities on Westinghouse's part. The technical information required to evaluate the Commission's request for reanalysis on a technical basis was received on July 2, 1975. Due to the very limited amount of time available for our review of these documents, it was impossible for our staff to review the technical justification and/or adequacy of the request by the Commission and the enclosed Westinghouse provided information.

The enclosed Westinghouse analysis supposedly responds to items I through IV of the reanalysis direction letter of June 5, 1975.

In regards to Item V, we wish to reference our submittal of August 31, 1972, Amendment No. 21 to the Kewaunee Nuclear Power Plant Application for construction permit and operating license which addressed this concern.

In regards to Item VI, our evaluation indicated that six motor operated valves will be flooded post LOCA. Four of the flooded valves

7376

are associated with Residual Heat Removal System and the other two valves are associated with the steam generator blowdown system. Due to the electrical design of the motor power circuit and control circuits flooding of these valves would not result in valve operator movement. It is noted that operation of these valves Post LOCA would not be dependable and operation Post LOCA is not necessary.

Our review of the technical information in regards to the December 25, 1974 version of the Westinghouse evaluation model, the March 15, 1975 version of the Westinghouse evaluation model and attached enclosure is in progress. When our review of this technical information is complete, we would prefer to discuss this entire matter with your staff to assure that we are in agreement as to the orderly conclusion of this activity.

Yours very truly,

E. W. James
Senior Vice-President
Power Supply & Engineering

EWJ/ml

Attach.

15.4 CONDITION IV - LIMITING FAULTS

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR Part 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the Emergency Core Cooling System and the containment. For the purposes of this report the following faults have been classified in this category:

1. Major rupture of pipes containing reactor coolant up to and including double ended rupture of the largest pipe in the Reactor Coolant System (loss of coolant accident).

2. Major secondary system pipe ruptures.

3. Steam generator tube rupture

4. Single reactor coolant pump locked rotor.

5. Fuel handling accident.

6. Rupture of a control rod mechanism housing (rod cluster control assembly ejection).

The analysis of thyroid and whole body doses, resulting from events leading to fission product release, appears [ ]. The fission product inventories which form a basis

for these calculations are presented in Chapter [ ] and Section [ ]. [ ] also includes the discussion of systems interdependency contributing to limiting fission product leakages from the containment following a Condition IV occurrence.

The time sequence of events during each Condition IV fault is shown in Table [1].

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE RUPTURES (LOSS OF COOLANT ACCIDENT)

The analysis specified by 10CFR50.46 [1] "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors", is presented in this section. The results of the loss of coolant accident analysis are shown in Table [2] and show compliance with the Acceptance Criteria. The analytical techniques used are in compliance with Appendix K of 10CFR50, and are described in the topical report, "Westinghouse ECCS Evaluation Model Summary" [2]. The results for the small break lossofcoolant accident are presented in Section 15.3.1 and are in conformance with 10CFR50.46 and Appendix K of 10CFR50.

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

Should a major break occur, depressurization of the Reactor Coolant System results in a pressure decrease in the pressurizer. Reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. A Safety Injection System signal is actuated when the appropriate setpoint is reached. These countermeasures will limit the consequences of the accident in two ways:

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 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, consistent with Appendix K of 10CFR50 [1]. Thereafter the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

When the Reactor Coolant System pressure falls below [700] 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.

Thermal Analysis

Westinghouse Performance Criteria for Emergency Core Cooling System

The reactor is designed to withstand thermal effects caused by a loss of coolant accident including the double ended severance of the largest Reactor Coolant System pipe. The reactor core and internals together with the Emergency Core Cooling System are designed so that the reactor can be safely shutdown 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].

Method of Thermal Analysis

The description of the various aspects of the LOCA analysis is given in WCAP8339 [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 code [6] to determine the ECCS backpressure are presented in Table [3]. The analysis presented here was performed with the MARCH 15, 1975 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 MARCH 15, 1975 WESTINGHOUSE ECCS EVALUATION MODEL IS DOCUMENTED IN REFERENCE [9].

Results

Table [2] presents the peak clad temperatures and hot spot metal reaction for the 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 SATAN VI analysis of the loss of coolant accident is performed at 102 percent of Engineered Safeguards Design Rating.* INSERT* BELOW The peak linear power, and core power used in the analyses are given in Table [2]. The equivalent core parameter at the license application power level are also shown in Table [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] for each break size analyzed.

* THE LOCTA IV ANALYSIS IS PERFORMED AT 102 PERCENT OF NSSS LICENSED POWER.

Figures [1] through [17] present the transients for the principal parameters for the break sizes analyzed. The following items are noted:

Figure [1a-1c] The system pressure shown is the calculated pressure in the core.

Figure [2a-2c] This figure shows the fraction of core power.

Figure [3a-4c] These figures show the core reflood transients.

Figure [5a-5c] This figure shows the hot spot clad temperature transient and the clad temperature transient at the burst location.

Figure [6a-6c] This figure shows the containment pressure transient.

Figure [7 - 8] The core flow (top and bottom) and the pressure drop across the core is shown for the worst break.

Figure [9-10] These figures show the Emergency Core Cooling System flow for the worst break. As described earlier the accumulator delivery during blowdown is discarded until the end of bypass is calculated.

Accumulator flow, however, is established in refill flood calculations. The accumulator flow assumed is the sum of that injected in the intack cold legs.

Figure [11-14] 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) for the worst break:

1. mass velocity
2. fluid quality
3. fluid temperature
4. heat transfer coefficient

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

Figure [15-16] The mass and energy released to the containment is provided for the worst break.

Figure [17] The containment wall condensing heat transfer coefficient is provided for the worst break.

The clad temperature analysis is based on a total peaking factor of [2.15]. The hot spot metal water reaction reached is [8.09%], 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.

The results of several sensitivity studies are reported in reference [7]. These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

Conclusions - Thermal Analysis

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

1. The calculated peak fuel element clad temperature does not exceed the 2200°F limit with a total Fq of 2.15.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircalloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17% are not exceeded during or after quenching.

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

The time sequence of events for all breaks analyzed is shown in Table [1].

References for 15.4.1

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.D., "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.

LARGE BREAK

LIST OF FIGURES

(a, b, c refer to one break (each) for the entire list of figures)

1a, 1b, 1c — core pressure
2a, 2b, 2c — core power
3a, 3b, 3c — core flooding rate
4a, 4b, 4c — core and downcomer water level
5a, 5b, 5c — peak clad temperature
6a, 6b, 6c — containment pressure
7 core flow - top and bottom
8 core pressure drop
9 accumulator flow during blowdown
10 pumped ECCS flow
11 mass velocity
12 fluid quality
13 fluid temperature
14 film heat transfer coefficient
15 break mass release
16 break energy release
17 containment wall heat transfer coefficient

wps

TABLE 1

LARGE BREAK

TIME SEQUENCE OF EVENTS

	DECL ($C_D = 1.0$) (Sec)	0.6 DECL ($C_D = 0.6$) (Sec)	0.4 DECL ($C_D = 0.4$) (Sec)
START	0.0	0.0	0.0
Reactor Trip Signal	0.58	0.59	0.6
Safety Injection Signal	0.43	0.51	0.6
Accumulator Injection	5.9	7.8	10.4
End of Blowdown	14.92	16.9	20.4
Bottom of Core Recovery	29.89	31.61	34.2
Accumulator Empty	37.55	39.45	42.7
Pump Injection	25.43	25.51	25.6
End of Bypass Time	14.92	16.85	20.35

TABLE 2

LARGE BREAK

	DECL ($C_D = 1.0$)	0.6 DECL ($C_D = 0.6$)	0.4 DECL ($C_D = 0.4$)
Results			
Peak Clad Temp. °F	2101.	2100.	2200.
Peak Clad Location Ft.	9.0	9.0	9.0
Local Zr/H_2O Rxn(max)%	6.14	6.08	8.09
Local Zr/H_2O Location Ft.	9.0	9.0	9.0
Total Zr/H_2O Rxn %	<0.3	<0.3	<0.3
Hot Rod Burst Time sec	53.6	52.2	24.4
Hot Rod Burst Location Ft.	6.0	6.0	5.75

Calculation	
ESDRPower Mwt 102% of	1722
Peak Linear Power kw/ft 102% of	13.53
Peaking Factor (At License Rating)	2.15
Accumulator Water Volume (FT3)	1250

Fuel region + cycle analyzed	Cycle	Region
UNIT 1	1	3
UNIT 2 (If applicable)	N/A	N/A

LARGE BREAK

TABLE 3

CONTAINMENT DATA

(DRY CONTAINMENT)

NET FREE VOLUME	1.37×10^6 ft^3
INITIAL CONDITIONS	
Pressure	14.7 psia
Temperature	90 °F
RWST Temperature	70 °F
Service Water Temperature	32 °F
Outside Temperature	-20 °F
SPRAY SYSTEM	
Number of Pumps Operating	2
Runout Flow Rate	1600 gpm/each
Actuation Time	15 secs
SAFEGUARDS FAN COOLERS	
Number of Fan Coolers Operating	4
Fastests Post Accident Initiation of Fan Coolers	15 secs

LARGE BREAK

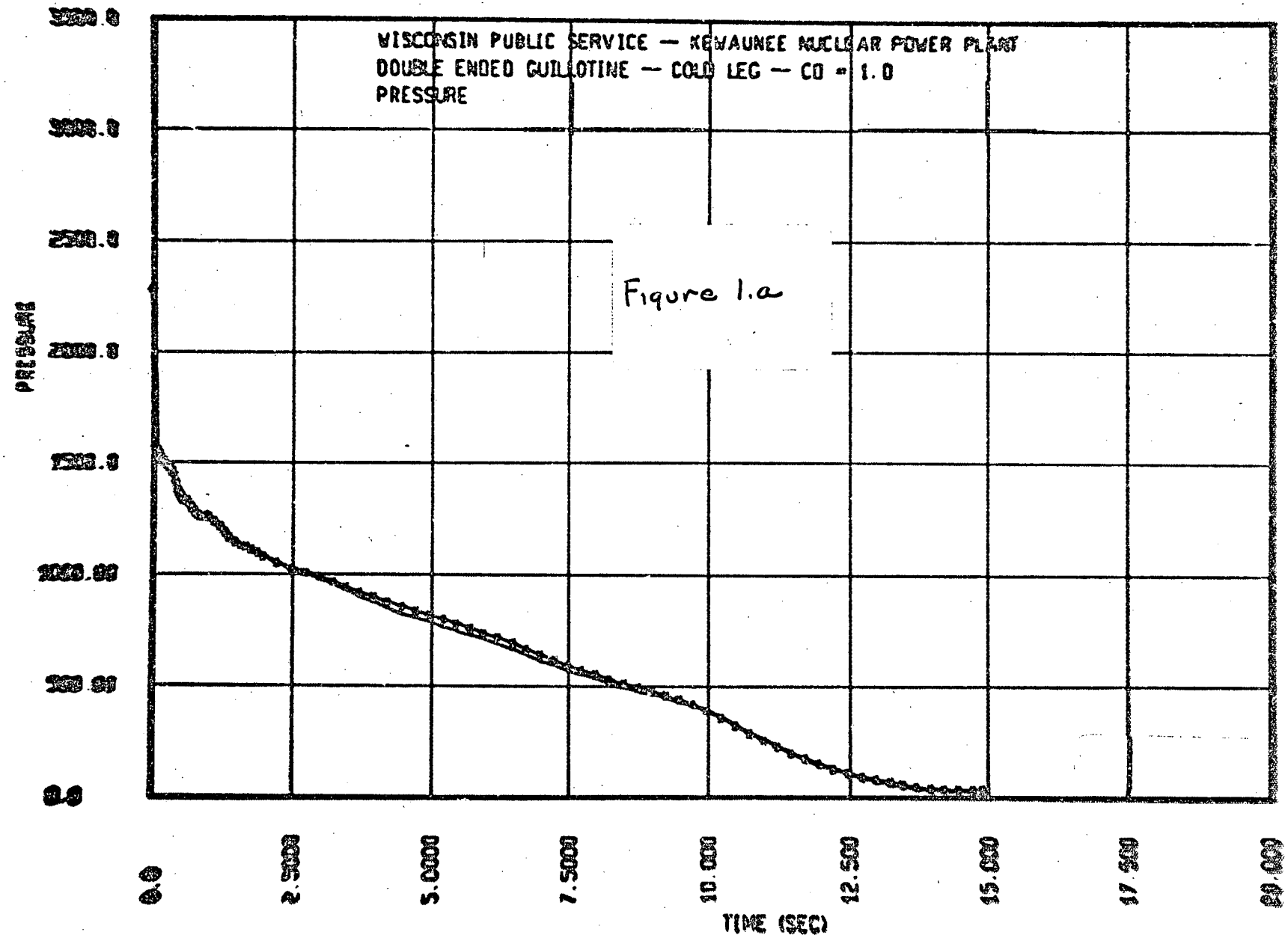
TABLE 3 (Continued)

CONTAINMENT DATA

(DRY CONTAINMENT)

STRUCTURAL HEAT SINKS

Thickness (In)	Area (Ft^2)
1.5 steel	41300
.75 steel	32000
.25 steel 12 concrete	7860
.375 steel	6800
.25 steel	32000
.5 steel	44000
.145 steel	1695
.09 steel	12400
.1 steel	6000
.1875 steel	35125
1.44 steel	2800
12 concrete	40800
6 concrete	25070
3 concrete	7570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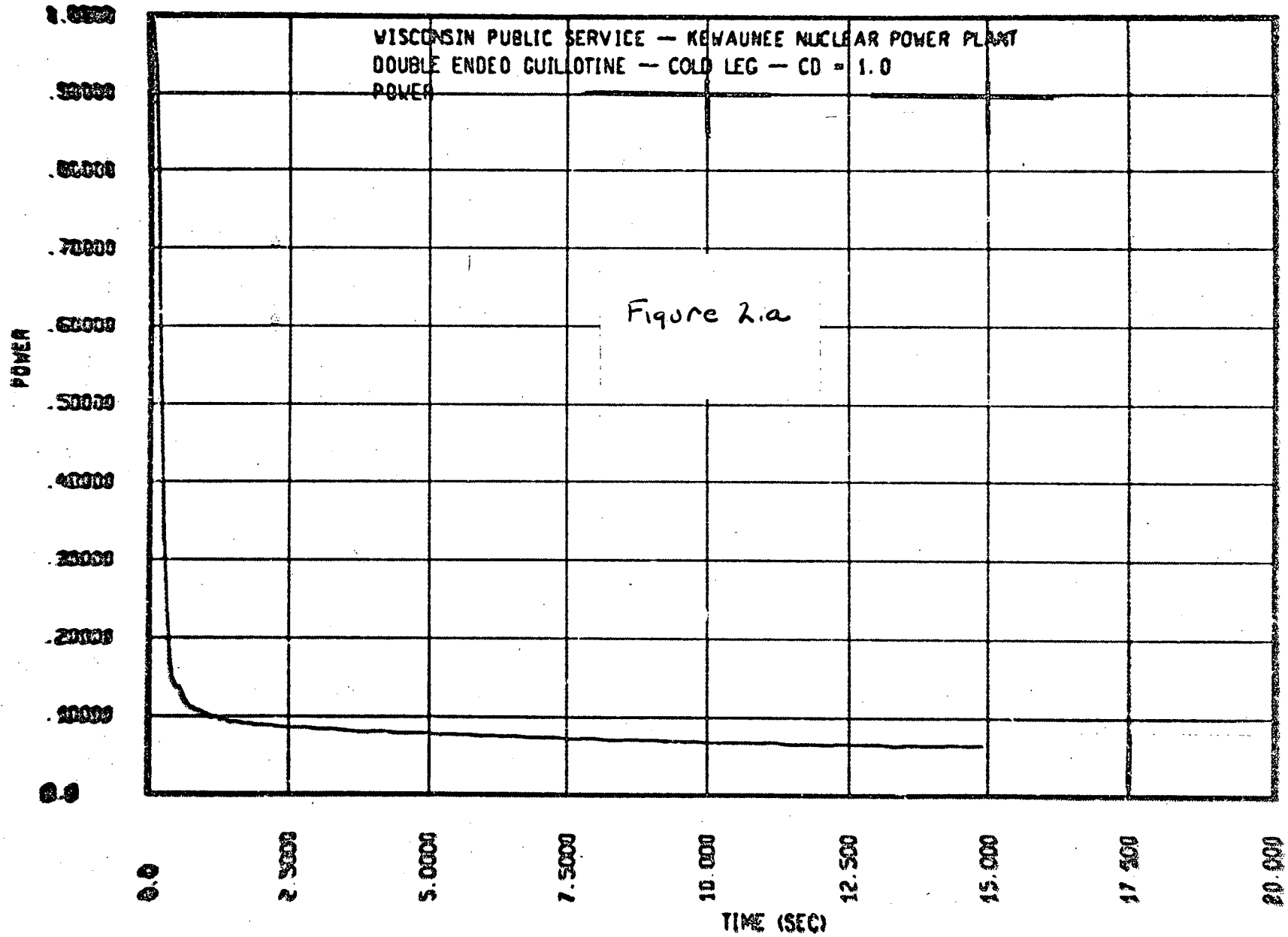
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 1.0
PRESSURE
PRESSURE
TIME (SEC)
0.0
2.5000
5.0000
7.5000
10.000
12.500
15.000
17.500
20.000

Figure 1.a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DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 1.0
POWER
POWER
TIME (SEC)
0.0
2.5000
5.0000
7.5000
10.000
12.500
15.000
17.500
20.000
.50000
.40000
.30000
.20000
.10000
0.0

Figure 2.a

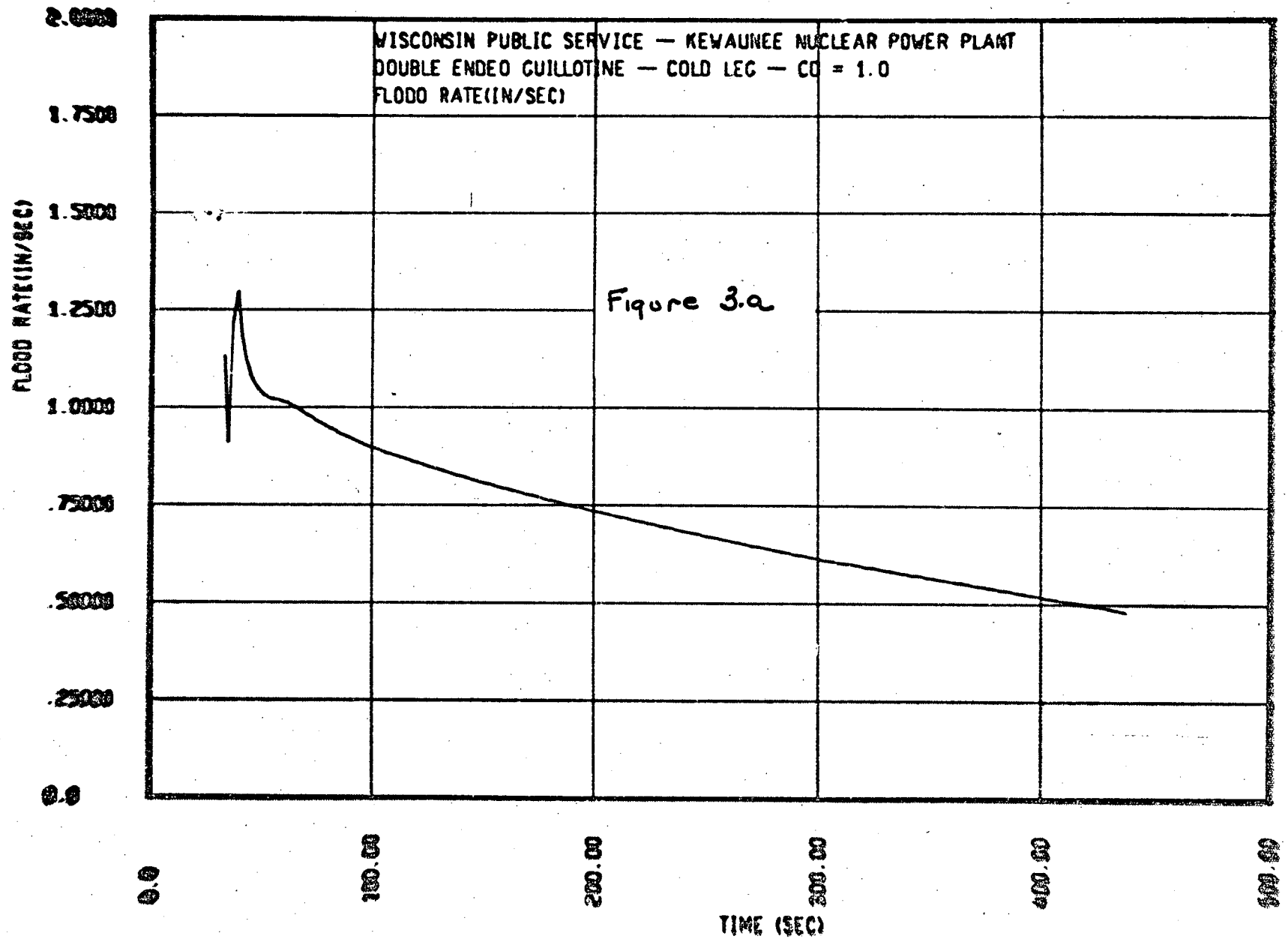
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 1.0
FLOOD RATE(IN/SEC)
FLOOD RATE(IN/SEC)
2.0000
1.7500
1.5000
1.2500
1.0000
.75000
.50000
.25000
0.0
0.0
100.00
200.00
300.00
400.00
500.00
TIME (SEC)

Figure 3.a

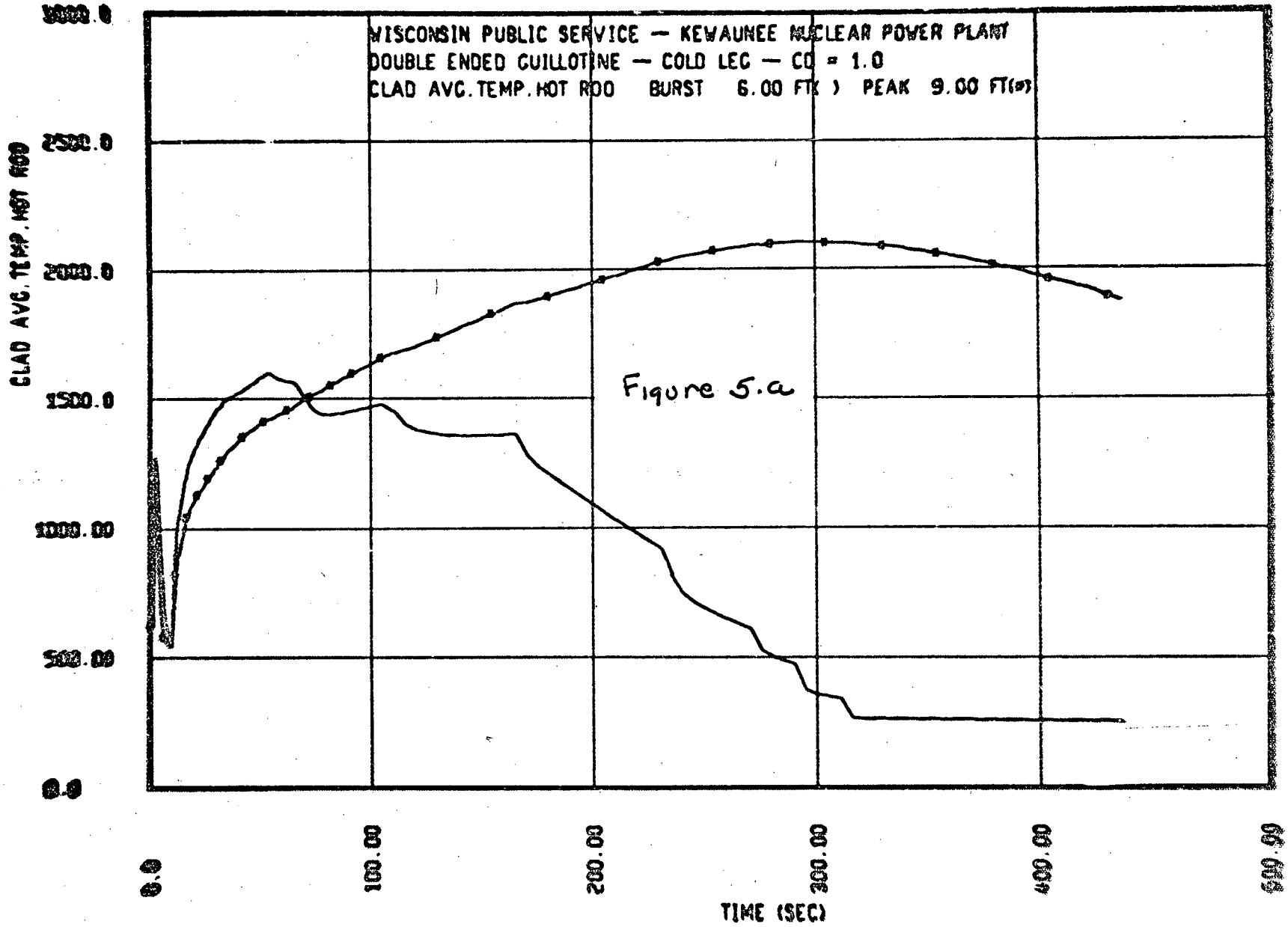
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 1.0
CLAD AVG.TEMP.HOT ROD BURST 6.00 FT( ) PEAK 9.00 FT(*)
CLAD AVG.TEMP.HOT ROD
3000.0
2500.0
2000.0
1500.0
1000.00
500.00
0.0
0.0
100.00
200.00
300.00
400.00
500.00
TIME (SEC)

Figure 5.a

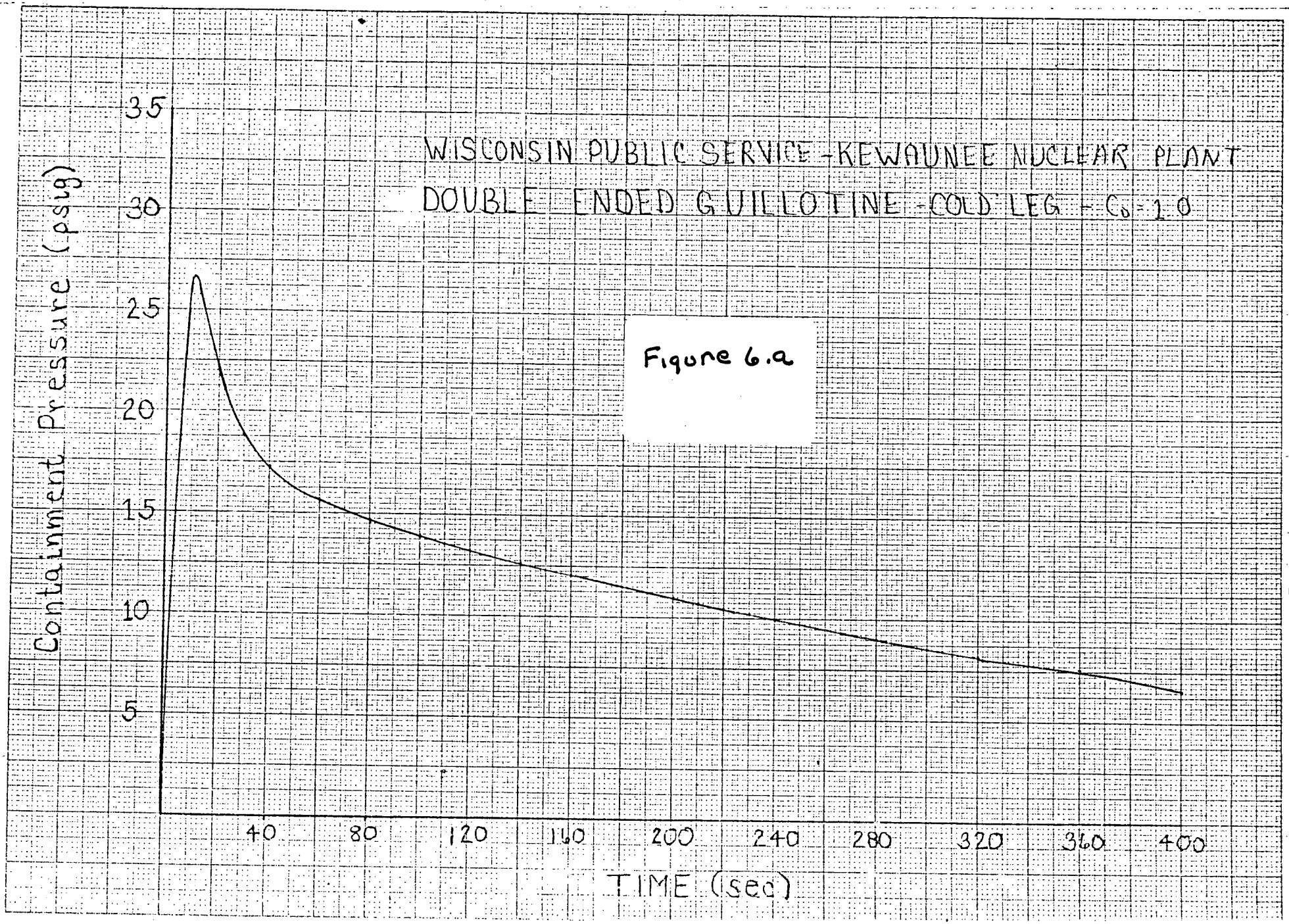
WISCONSIN PUBLIC SERVICE - KEWAUNEE NUCLEAR PLANT
DOUBLE ENDED GUILLOTINE - COLD LEG - C_D = 1.0
Containment Pressure (psig)
35
30
25
20
15
10
5
40
80
120
160
200
240
280
320
360
400
TIME (sec)

Figure 6.a

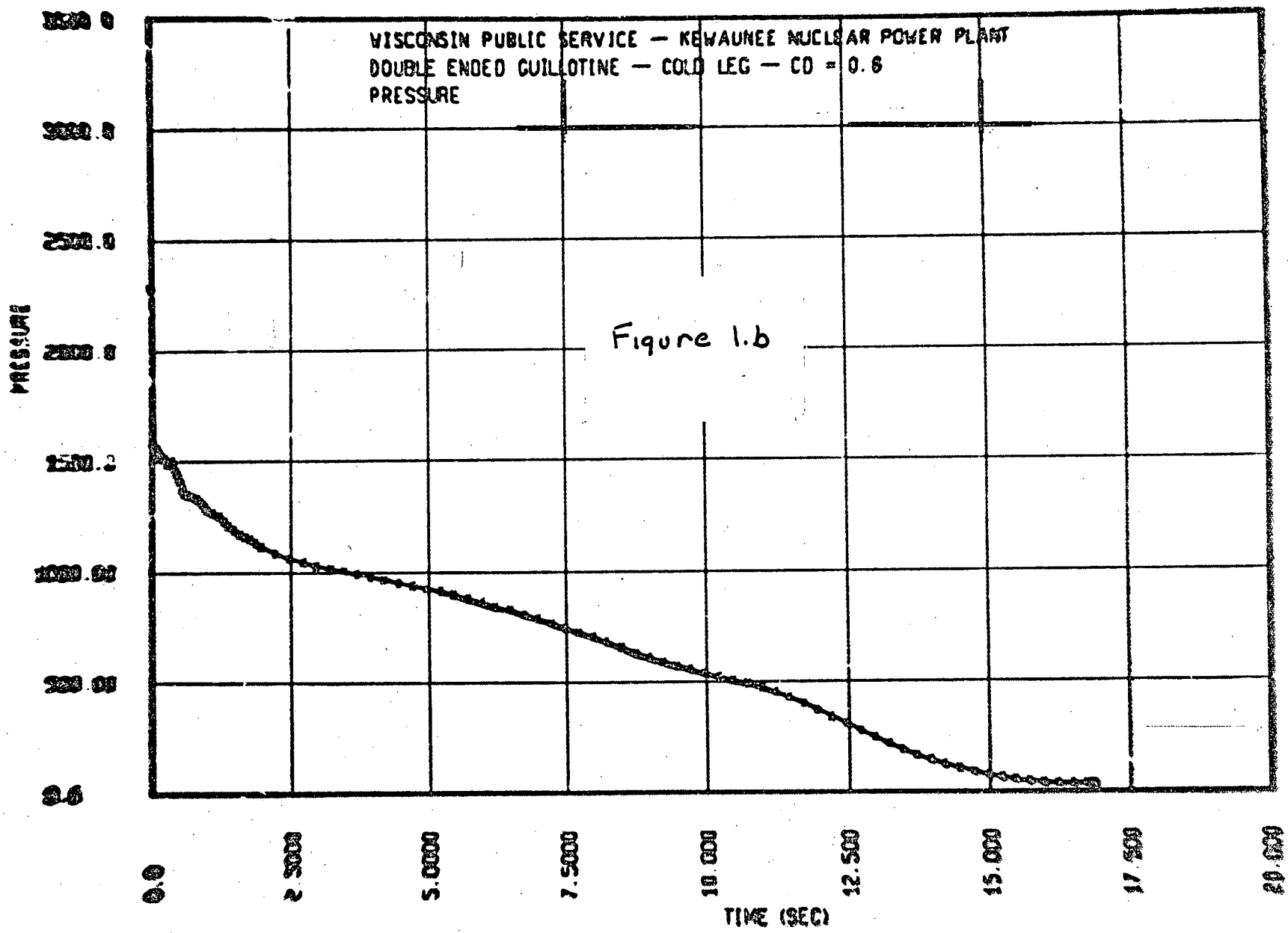
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.6
PRESSURE
PRESSURE
TIME (SEC)
0.0
2.5000
5.0000
7.5000
10.000
12.500
15.000
17.500
20.000

Figure 1.b

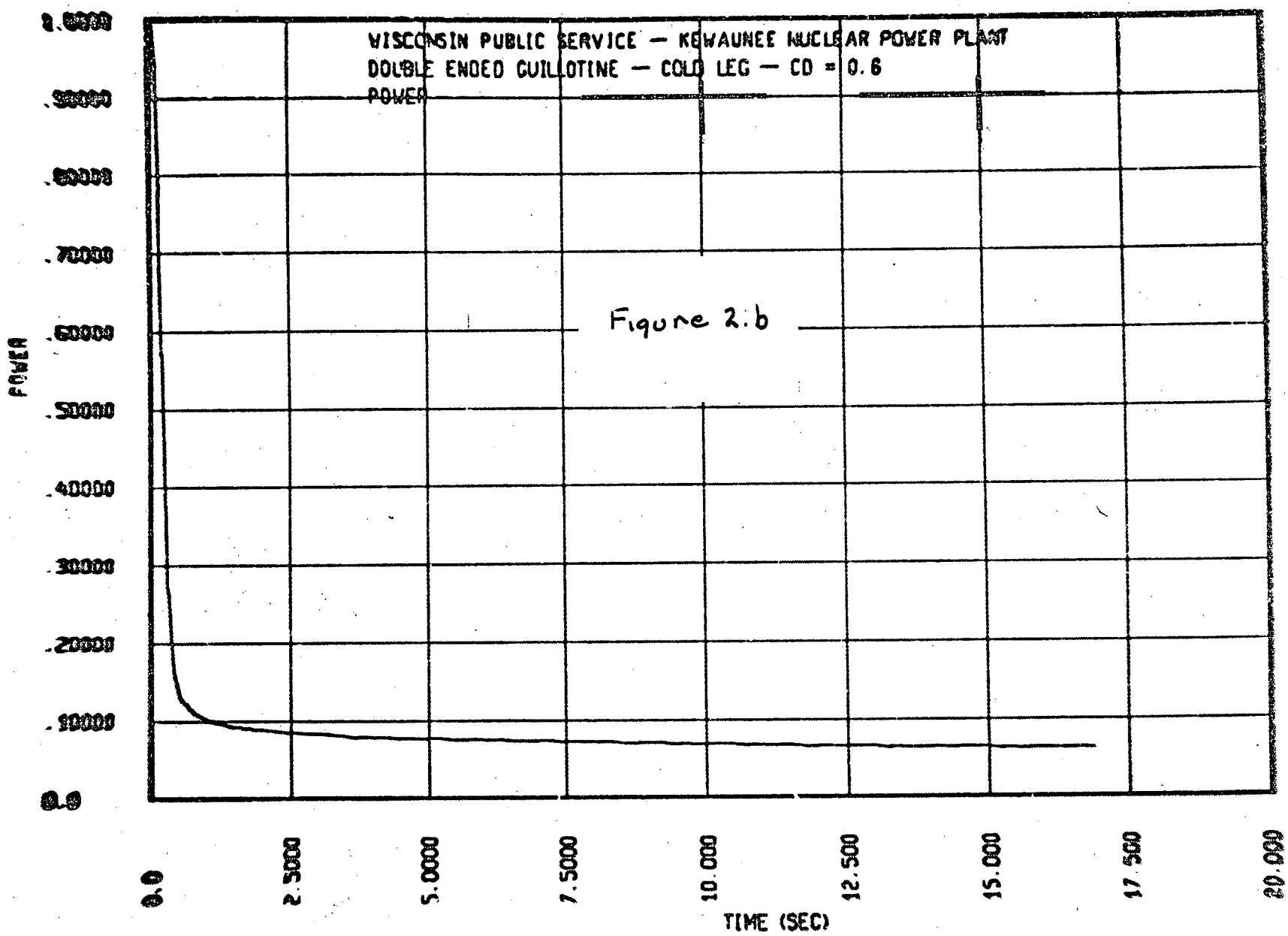
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.6
POWER
POWER
1.0000
.90000
.80000
.70000
.60000
.50000
.40000
.30000
.20000
.10000
0.0
0.0
2.5000
5.0000
7.5000
10.000
12.500
15.000
17.500
20.000
TIME (SEC)

Figure 2.b

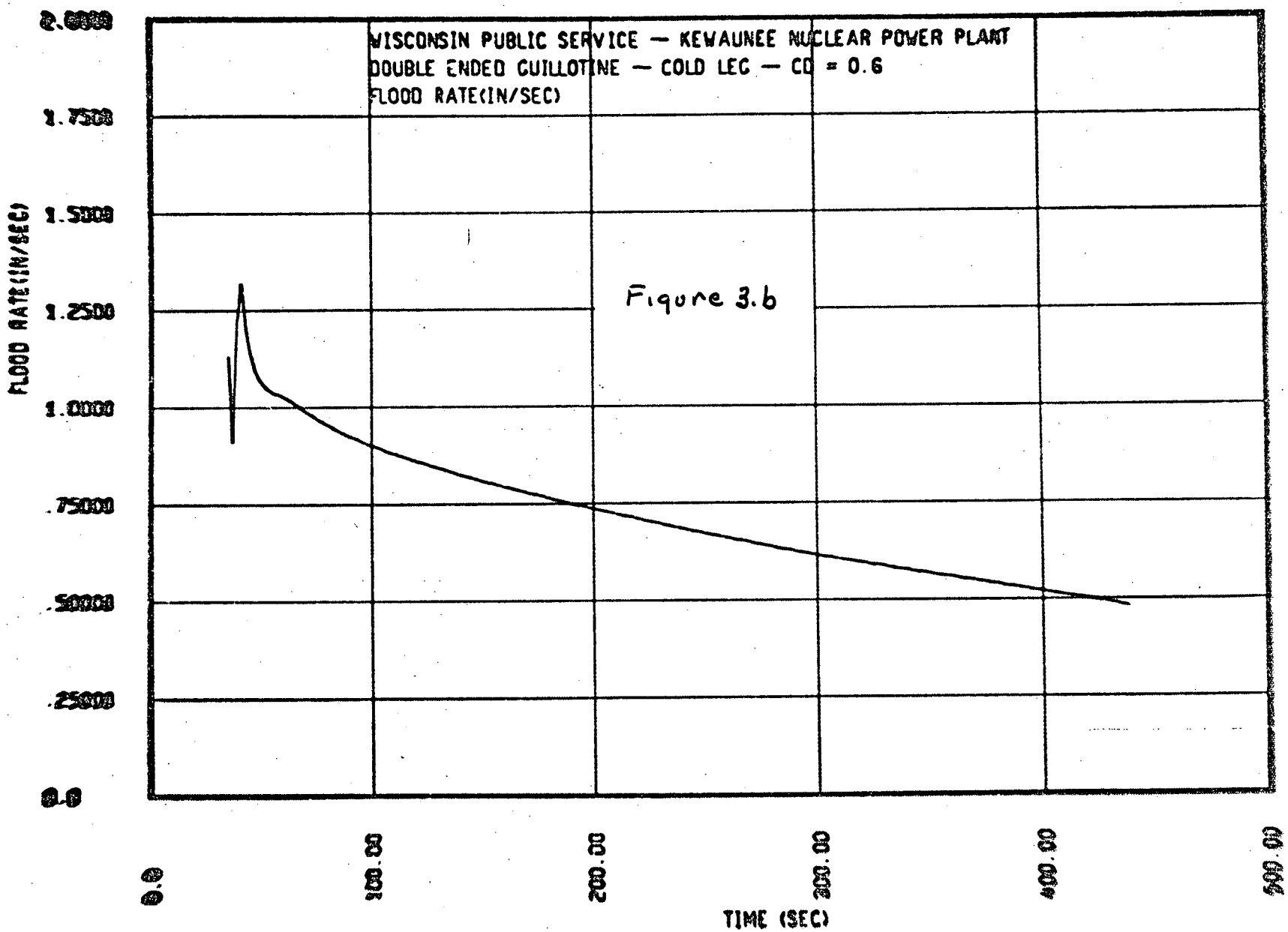
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.6
FLOOD RATE(IN/SEC)
FLOOD RATE(IN/SEC)
2.0000
1.7500
1.5000
1.2500
1.0000
.75000
.50000
.25000
0.0
0.0
100.00
200.00
300.00
400.00
500.00
TIME (SEC)

Figure 3.b

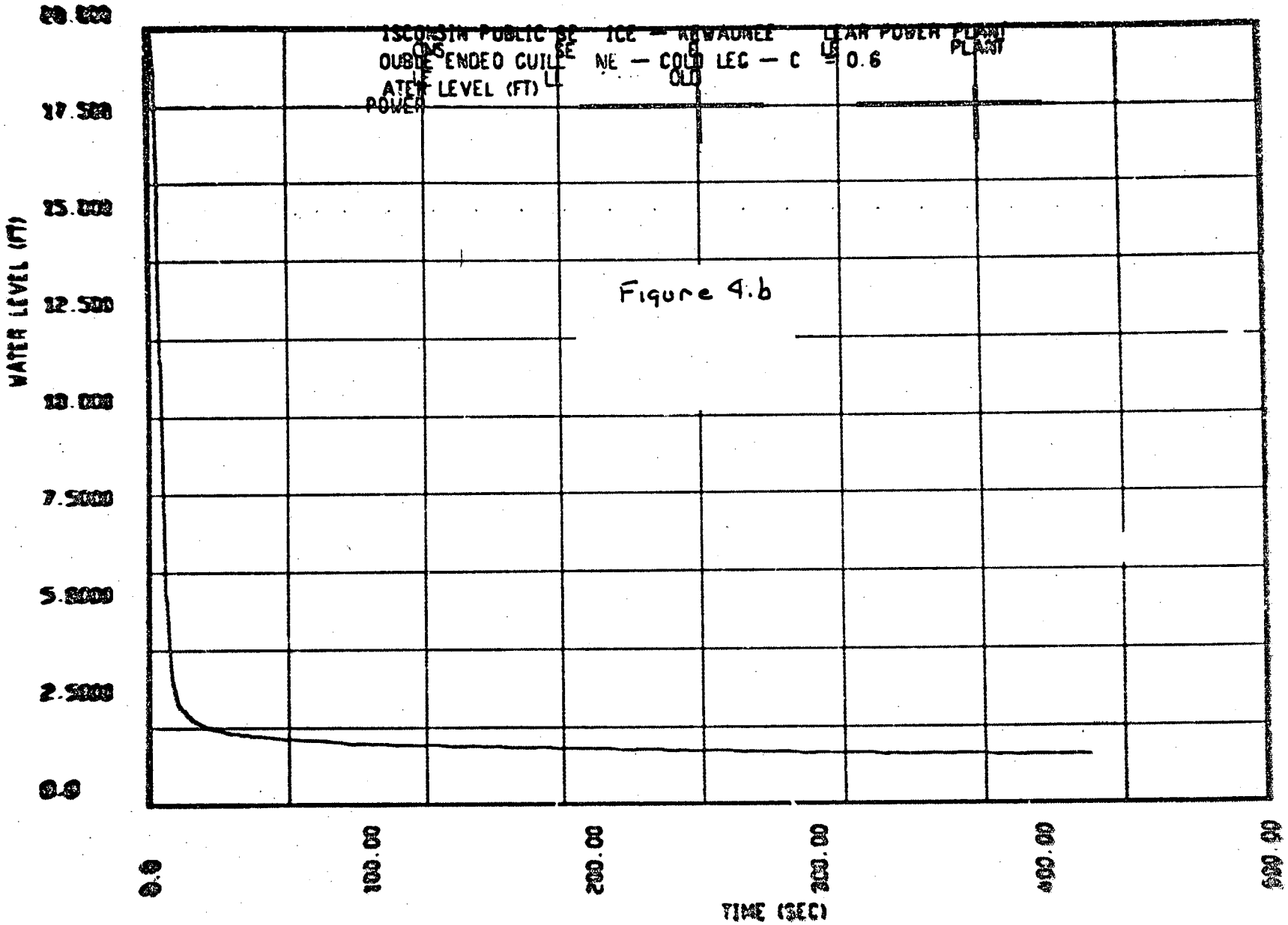
WISCONSIN PUBLIC SERVICE CORP. — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.6
WATER LEVEL (FT)
TIME (SEC)
0.0
2.5000
5.0000
7.5000
10.000
12.500
15.000
17.500
20.000
100.00
200.00
300.00
400.00
500.00

Figure 4.b

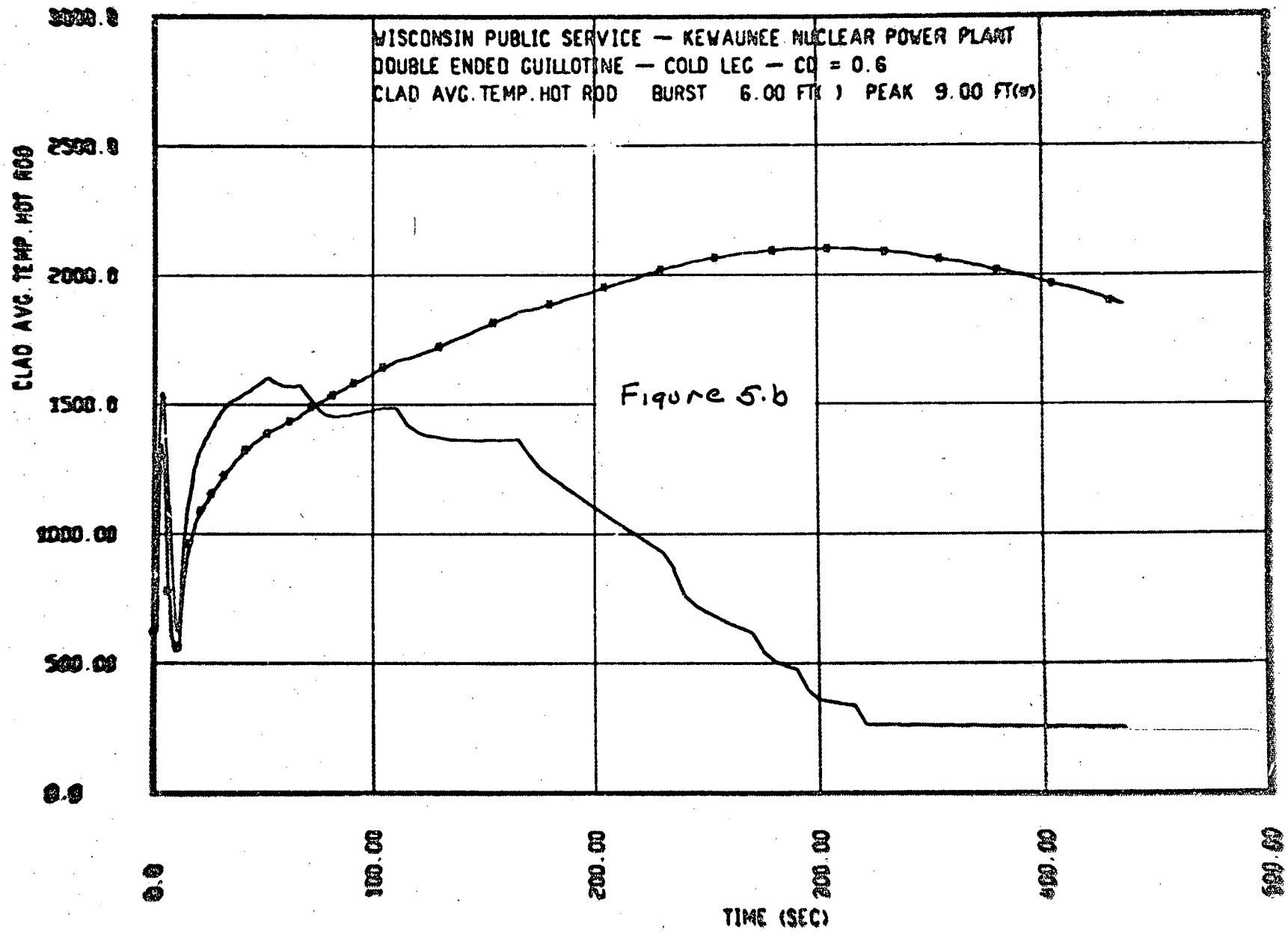
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.6
CLAD AVG. TEMP. HOT ROD BURST 6.00 FT( ) PEAK 9.00 FT(m)
CLAD AVG. TEMP. HOT ROD
3000.0
2500.0
2000.0
1500.0
1000.00
500.00
0.0
0.0
100.00
200.00
300.00
400.00
500.00
TIME (SEC)

Figure 5.b

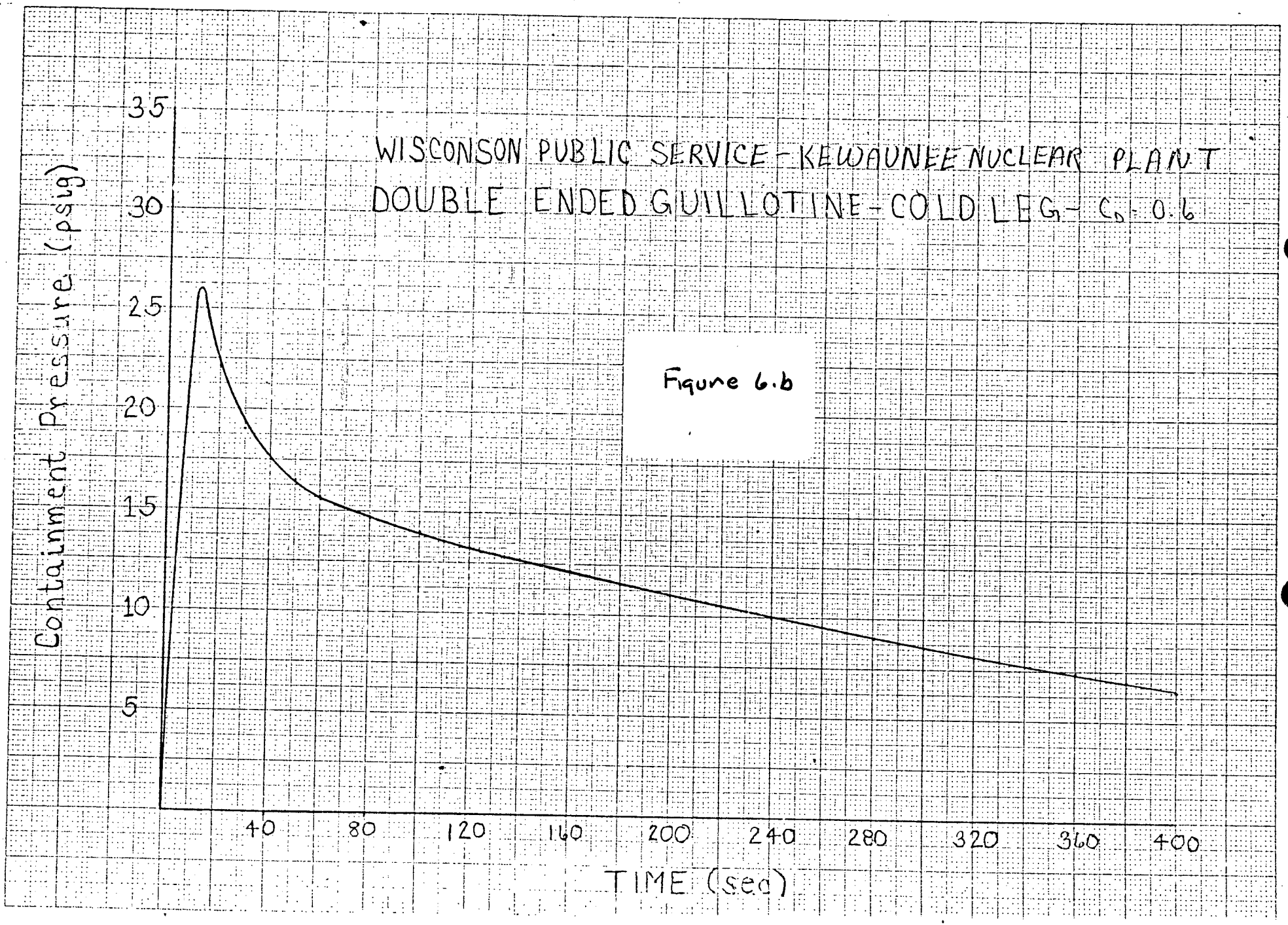
WISCONSON PUBLIC SERVICE - KEWAUNEE NUCLEAR PLANT
DOUBLE ENDED GUILLOTINE - COLD LEG - C_D - 0.6
Containment Pressure (psig)
35
30
25
20
15
10
5
40
80
120
160
200
240
280
320
360
400
TIME (sec)

Figure 6.b

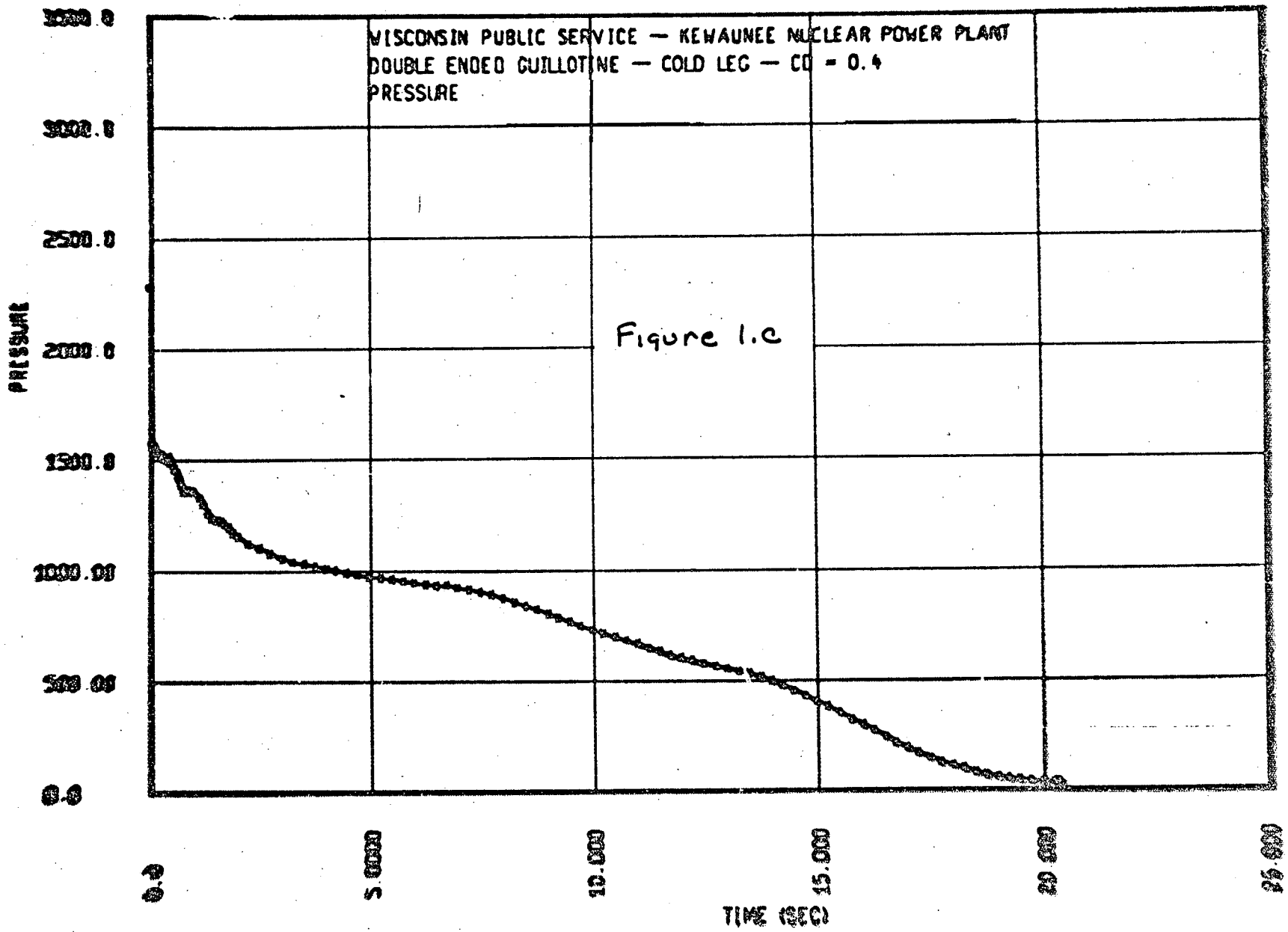
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.4
PRESSURE
PRESSURE
3500.0
3000.0
2500.0
2000.0
1500.0
1000.00
500.00
0.0
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 1.c

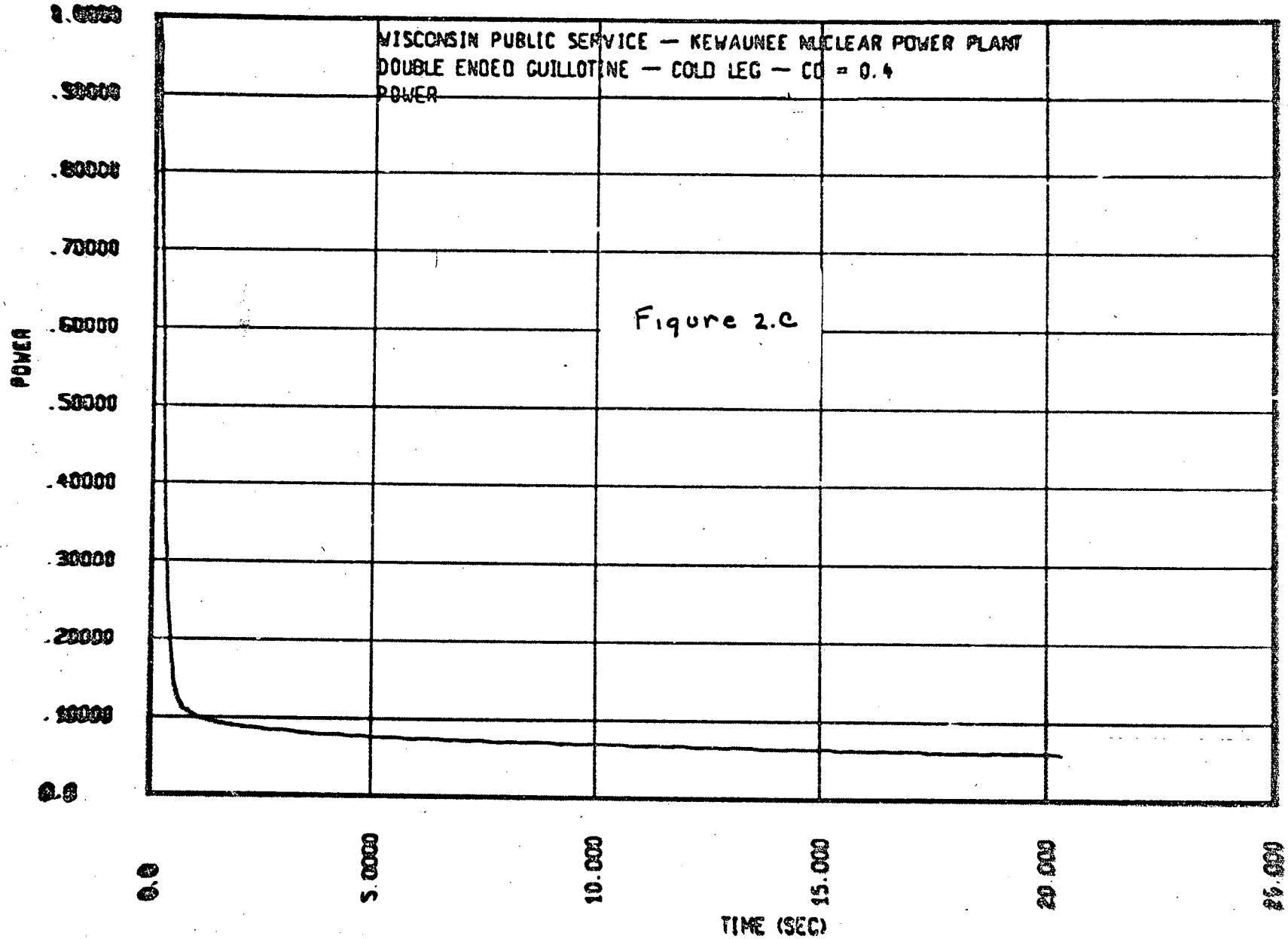
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.4
POWER
POWER
1.00000
.90000
.80000
.70000
.60000
.50000
.40000
.30000
.20000
.10000
0.0
0.0
5.0000
10.000
15.000
20.000
25.000
TIME (SEC)

Figure 2.c

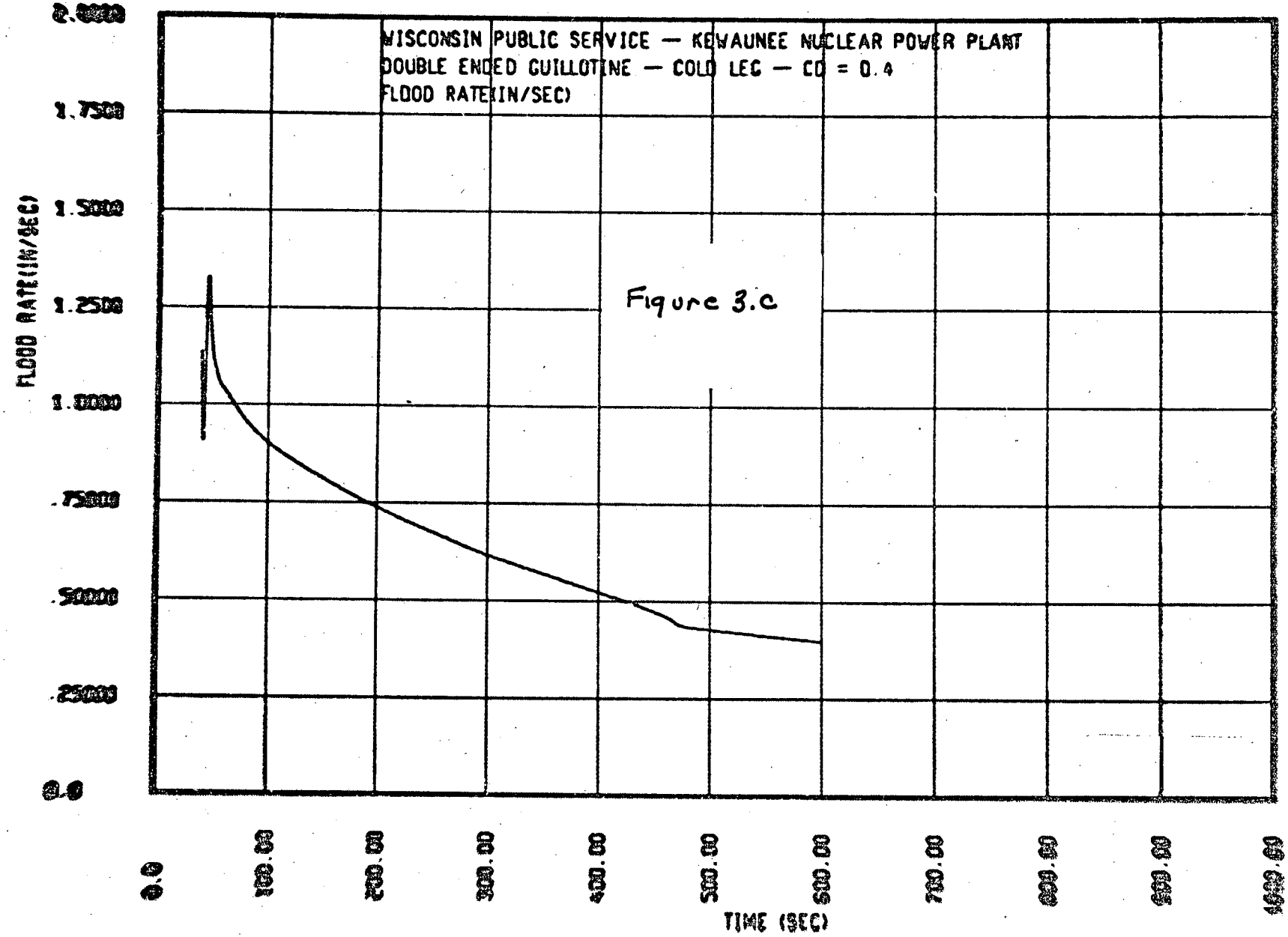
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.4
FLOOD RATE(IN/SEC)
FLOOD RATE(IN/SEC)
2.0000
1.7500
1.5000
1.2500
1.0000
.75000
.50000
.25000
0.0
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.00
TIME (SEC)

Figure 3.c

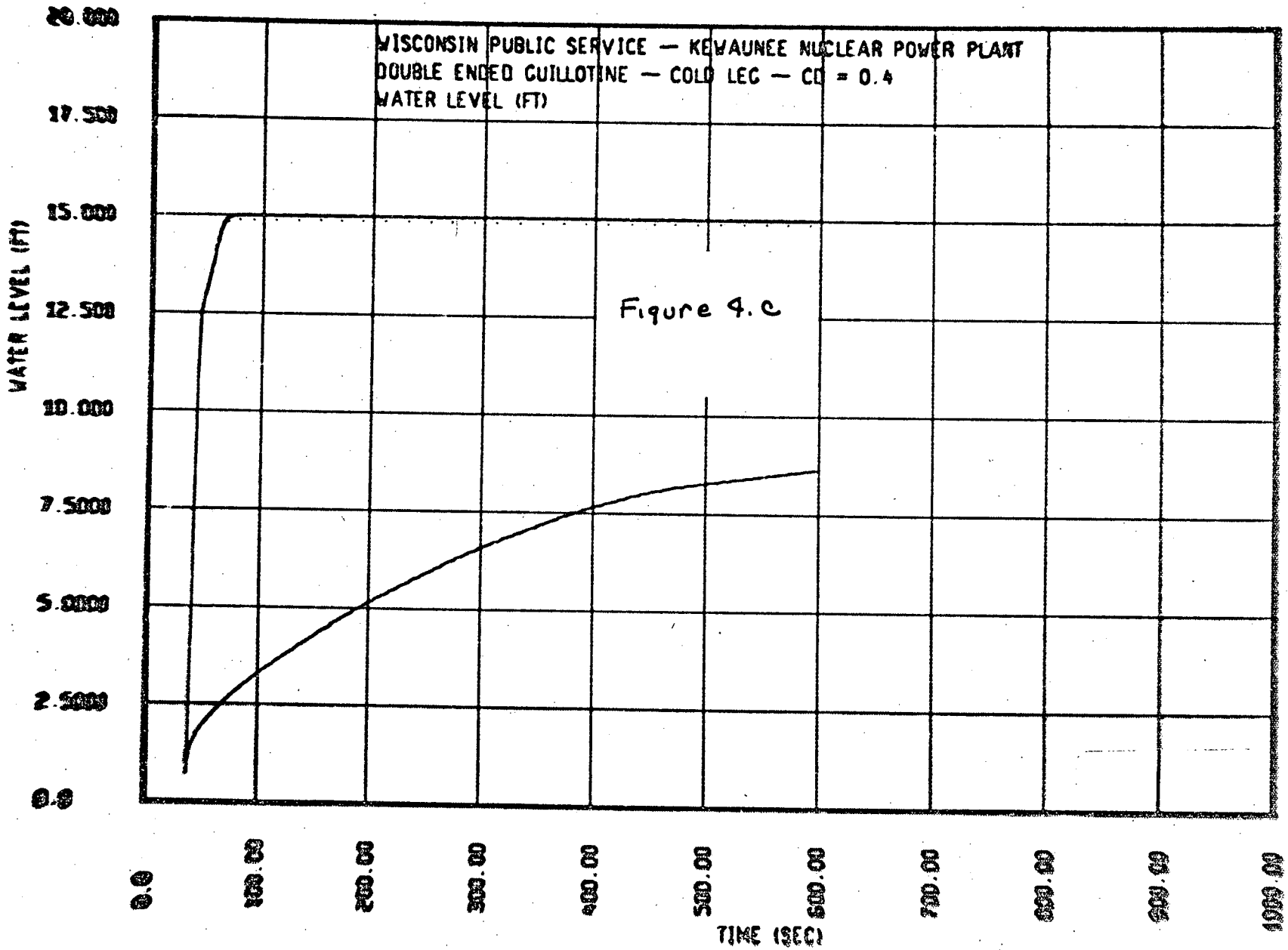
WISCONSIN PUBLIC SERVICE — KEWAUNEE NUCLEAR POWER PLANT
DOUBLE ENDED GUILLOTINE — COLD LEG — CD = 0.4
WATER LEVEL (FT)
WATER LEVEL (FT)
20.000
17.500
15.000
12.500
10.000
7.5000
5.0000
2.5000
0.0
0.0
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.00
1000.00
TIME (SEC)

Figure 4.c