SNUPPS

Standardized Nuclear Unit Power Plant System

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Nicholas A. Petrick
Executive Director

April 17, 1984

SLNRC 84-0069 FILE: 0278
SUBJ: Revision in Diesel Generator Start Time

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Docket Nos. STN 50-482 and STN 50-483

Reference: SLNRC 82-01, dated 1/7/82, SNUPPS ECCS Analysis

Dear Mr. Denton:

A revision to the emergency diesel generator start time by two seconds (from 10 seconds to 12 seconds) is defined herein. This change in start time is based on providing a more practical method in implementing Technical Specification surveillance requirements in the operation of the Callaway Plant, Unit No. 1 and the Wolf Creek Generating Station, Unit No. 1.

The effects of delaying the start of the emergency diesel generators by two seconds have been evaluated by examining the consequences on the accident analyses presented in Chapter 15 of the FSAR and the containment pressurization calculations presented in Chapter 6 of the FSAR. The evaluation considered those accidents presented in Chapter 15 that require safety injection. The only accidents potentially affected by this change are the loss of coolant accident (LOCA) and the main steam-line break (MSLB). This evaluation has shown the following results:

- The LOCA analysis for the maximum safety injection flow case of the double-ended cold leg guillotine break (C_D = 0.6) showed an increase in peak clad temperature from 2106°F to 2174°F. It should be noted that the peak clad temperature reported in the FSAR (2105°F- Revision 9) was based on Westinghouse's generic penalty for maximum safety injection flow. Westinghouse has evaluated the minimum safety injection flow case with a two second delay in the start of the emergency diesel generator and found this case to be not limiting (2114°F versus 2174°F). The increase in peak clad temperature (2106°F to 2174°F) of

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68°F results from the combination of the changes from considering low head and accumulator injection interaction, the LOCTA modification in the axial conduction model, and the two second delay in the emergency diesel generator start time. It is estimated that of this 68°F increase in peak clad temperature, the two second delay in the emergency diesel generator start time results in an increase in peak clad temperature of approximately 10°F. This LOCA analysis was performed using the 1978 model. Use of the 1981 ECCS evaluation model and consideration of the reduction in initial pellet temperature modelling change would yield a net margin of approximately 200°F to the 50.46 criteria for the SNUPPS plants.

- The limiting steamline break analysis is the case with offsite power available, where diesel start is not involved. A review of the steamline break analysis with loss of offsite power, shows that the two second delay in the emergency diesel generator start would have an insignificant effect, if any, on the results in terms of reactivity and the DNBR evaluation. The offsite power available case is still limiting for the steamline break.

- The effects of the delayed emergency diesel generator start time on the effectiveness of the containment sprays and the containment air coolers were examined with respect to the containment pressurization calculations for the LOCA and MSLB. For both of these cases, sufficient margin was allowed in the previous calculations for full functioning of these systems, that the 2 second increase in emergency diesel generator start time had no effect on the pressurization analysis.

Attached to this letter are the page changes in the FSAR that will be formally implemented in Revision 15. It should be noted that there are numerous references to the emergency diesel generator start time in the FSAR and the FSAR changes presented herein address only those areas that could have been affected by the modification.

Your expeditious review of the above modification is requested. If there are any questions, please do not hesitate to call us.

Very truly yours,

J. O. Cermak for

Nicholas A. Petrick

JOC/nldllal6&17
Attachment

cc:	D. F. Schnell	UE	J. Neisler/B. Little	USNRC/CAL
	G. L. Koester	KGE	W. Schum/A. Smith	USNRC/WC
	D. T. McPhee	KCPL	B. L. Forney	USNRC/RIII

failure of these systems is assumed to be consistent with the mass and energy release data assumptions for each break analyzed. The total heat removed by each of the containment heat removal systems up to the time of the calculated peak containment pressure is listed in Table 6.2.1-8. The design bases of the containment heat removal systems are discussed in Section 6.2.2.

The functional performance of the containment and the ECCS also rely upon the operation of the containment isolation system, as described in Section 6.2.4. Required isolation operations are assumed for purposes of the containment design evaluation in Section 6.2.1.1.3.

d. Parameters Affecting Capability for Post-Accident Pressure Reduction

The principal parameters which affect post-accident pressure reduction are 1) the heat absorbed by the heat sinks inside the containment, 2) the heat removed by the containment air coolers, and 3) the heat transferred to the containment sump by the containment spray system.

A conservative amount of heat sink material has been calculated, and its heat absorption capability has been considered in the containment design evaluation in Section 6.2.1.1.3. The parameters describing the heat sinks credited with heat absorption are provided in Table 6.2.1-4.

The pressure reduction capability of the containment air coolers and the containment spray system consider the parameters provided in Table 6.2.1-3. The assumed start time of these active heat removal systems considers a diesel start time of 10 seconds, load sequencing times, and the maximum startup time of the systems.

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e. Parameters Affecting Heat Removal from the Containment

Heat is transferred from the containment to the outside environment via the fan coolers and residual heat removal heat exchangers through the component cooling water and essential service water systems and released to the ultimate heat sink. A small amount of heat is also transferred through the containment wall and dome to the outside atmosphere.

The component cooling water system is described in Section 9.2.2, the essential service water system is described in Section 9.2.1, and the ultimate heat sink is described in Section 9.2.5.

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In support of case c, large break LOCA (DECLG C_D = 0.6, Maximum SI with 12 second diesel generator start) of Table 15.6-10, an evaluation of the assumptions used in the LOCA and MSLB containment pressurization calculations with respect to the full functioning times of the containment spray system and the containment air coolers was performed. The evaluation shows that the containment pressurization calculations for both LOCA and MSLB provided sufficient margin so that a 12-second diesel generator start time does not change the assumed full functioning times of the containment sprays and the containment air coolers. Therefore, additional LOCA and MSLB containment pressurization calculations are not required for case c of Table 15.6-10, since this case is bounded by the previously performed containment calculations.

The small-break analysis was performed with the October 1975 version of the Westinghouse ECCS Evaluation Model (Ref. 7, 12, 13, and 14).

15.6.5.3.2 Input Parameters and Initial Conditions

Table 15.6-9 lists important input parameters and initial conditions used in the analysis.

The analysis presented in this section was performed with a reactor vessel upper head temperature equal to the RCS cold leg temperature. The effect of using the cold leg temperature in the reactor vessel upper head is described in Reference 23. In addition, the analysis in this section utilized the upflow barrel-baffle methodology described in Reference 19.

The bases used to select the numerical values that are input parameters to the analysis have been conservatively determined from sensitivity studies (Ref. 20, 21, 22). In addition, the requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core-peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated.

15.6.5.3.3 Results

Large-Break Results

Based on the results of the LOCA sensitivity studies (Ref. 20, 21, and 22), the limiting large break was found to be the double-ended cold leg guillotine (DECLG) break. Therefore, only the DECLG break is considered in the large-break ECCS performance analysis. Calculations were performed for a range of Moody break discharge coefficients. The results of these calculations are summarized in Tables 15.6-10 and 15.6-11. INSERT A

¶ The mass and energy release data for the break resulting in the highest calculated peak clad temperature are presented in Section 6.2.1.5.

Figures 15.6-7 through 15.6-30 present the parameters of principal interest from the large-break ECCS analyses. For all cases analyzed, transients of the following parameters are presented:

Insert A For Page 15.6-22

The worst break in the spectrum of break sizes analyzed was a discharge coefficient (C_d) of 0.6. This worst break was analyzed with a 12 second diesel generator start time, to study the effect of a 2 second delay in the start of the diesel generator. Two cases are presented in Table 15.6-11 for the C_d = 0.6 DECLG break: the minimum safety injection case and the maximum safety injection case (Reference 27 methodology). The maximum safety injection case proved to be the most limiting.

a. Hot spot clad temperature

b. Coolant pressure in the reactor core

c. Water level in the core and downcomer during reflood

d. Core reflooding rate

e. Thermal power during blowdown

The containment pressure transient resulting from a LOCA is presented in Section 6.2.1.5.

For the limiting break analyzed, the following additional transient parameters are presented:

a. Core flow during blowdown (inlet and outlet)

b. Core heat transfer coefficients

c. Hot spot fluid temperature

d. Mass released to containment during blowdown

e. Energy released to containment during blowdown

f. Fluid quality in the hot assembly during blowdown

g. Mass velocity during blowdown

h. Accumulator water flow rate during blowdown

i. Pumped safety injection water flow rate during reflood

The maximum clad temperature calculated for a large break is 2174°F ~~2080 F*~~, which is less than the Acceptance Criteria limit of 2200 F of 10 CFR 50.46. The maximum local metal-water reaction is 6.18 percent, which is well below the embrittlement limit of 17 percent, as required by 10 CFR 50.46. The total core metal-water reaction is less than 0.3 percent for all breaks, compared with the 1-percent criterion of 10 CFR 50.46, and the clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. As a result, the core temperature will continue to drop, and the ability to remove decay heat generated in the fuel for an extended period of time will be provided.

~~*An error was discovered in the worst single failure assumption for the large break ECCS analysis (see Ref. 26). Evaluation of the impact of this error (see Ref. 27) indicated that the maximum cladding temperature calculated for the large break could be 2106 F, which is less than the Acceptance Criteria limit of 2200 F of 10 CFR 50.46. The NRC reviewed this issue and agreed that new analysis results were not necessary.~~

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~~5/82~~

18. Letter NS-TMA-2014, dated December 11, 1978, Anderson, T. M. (Westinghouse) to Tedesco, R. L. (NRC).

19. Johnson, W. J. and Thompson, C. M., "Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version," WCAP-9168 (Proprietary) and WCAP-9169 (Non-Proprietary), September 1977.

20. "Westinghouse ECCS Evaluation Model Sensitivity Studies," WCAP-8341 (Proprietary) and WCAP-8342 (Non-Proprietary), July 1974.

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

22. Johnson, W. J., Massie, H. W., and Thompson, C. M., "Westinghouse ECCS-Four Loop Plant (17x17) Sensitivity Studies," WCAP-8565-P-A (Proprietary) and WCAP-8566-A (Non-Proprietary), July 1975.

23. Letter NS-TMA-2030, dated February 12, 1979, Anderson, T. M. (Westinghouse) to Denton, H. R. (NRC).

24. DiNunno, J. J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, Division of Licensing and Regulation, AEC, Washington, D.C., 1962.

25. USNRC NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," by Postma, A. K. and Tam, P. S., dated January 1978.

26. Maximum Safety Injection Worst Single Failure, NS-EPR-2538, December 22, 1981, letter from E. P. Rahe of Westinghouse Electric Corporation to R. L. Tedesco, Assistant Director of Licensing, and T. P. Speis, Assistant Director for Reactor Safety of the U.S. NRC.

~~27. SNUPPS ECCS Analysis, SLNRC-82-01, File 0278, January 7, 1982. Letter from N. A. Petrick of SNUPPS to H. R. Denton, Director, Office of Nuclear Reactor Regulation, U.S. NRC.~~

27. Letter NS-EPR-2538, dated December 22, 1981, Rahe, E.P. (Westinghouse) to Tedesco, R. L. (NRC).

TABLE 15.6-10

TIME SEQUENCE OF EVENTS FOR
LOSS-OF-COOLANT ACCIDENTS

Accident	Event	Time (sec)
Large break LOCA		
a. DECLG C_D=0.8	Start	0.0
	Reactor trip signal	0.82
	Safety injection signal	1.0
	Accumulator injection begins	13.9
	End-of-bypass	24.2
	Pump injection begins	26.0
	End-of-blowdown	27.0
	Bottom of core recovery	37.4
	Accumulator empty	47.9
b. DECLG C_D=0.6 (Minimum SI With 12 Second Diesel Generator Start)	Start	0.0
	Reactor trip signal	0.83
	Safety injection signal	1.15
	Accumulator injection begins	15.9
	Pump injection begins	28.6
	End-of-bypass	26.5
	End-of-blowdown	29.1
	Bottom of core recovery	40.9
	Accumulator empty	50.2
d. DECLG C_D=0.4	Start	0.0
	Reactor trip signal	0.82
	Safety injection signal	1.42
	Accumulator injection begins	20.7
	Pump injection begins	26.4
	End-of-bypass	34.6
	End-of-blowdown	37.7
	Bottom of core recovery	48.9
	Accumulator empty	56.9
Small break LOCA		
a. 3 inch	Start	0.0
	Reactor trip signal	29.7
	Top of core uncovered	623
	Accumulator injection begins	N/A
	Peak clad temperature occurs	1351
	Top of core covered	2300
c. DECLG C_D=0.6 (Maximum SI with 12 Second Diesel generator Start)	Start	0.0
	Reactor trip signal	0.83
	Safety injection signal	1.15
	Accumulator injection begins	15.9
	Pump injection begins	28.6
	End-of-bypass	26.5
	End-of-blowdown	29.1
	Bottom of core recovery	40.5
	Accumulator empty	51.3

TABLE 15.6-11

LARGE BREAK LOCAL RESULTS *
FUEL CLADDING DATA

Results	DECLG C_D=0.8	DECLG C_D=0.6		DECLG C_D=0.4
	Case a	Case b	Case c	Case d
Peak clad temperature, F	2053.6	2114.5	2174.2	1624
Peak clad location, ft	7.5	7.5	7.5	7.5
Local Zr/H_2O reaction, max. (%)	4.38	5.38	6.18	0.85
Local Zr/H_2O location, ft	7.5	7.5	7.5	7.5
Total Zr/H_2O reaction, %	<0.3	<0.3	<0.3	<0.3
Hot rod burst time, sec	29.2	27.8	27.8	N/A
Hot rod burst location, ft	6.0	6.0	6.0	N/A

* Refer to Section 15.6.5.3.3 and Table 15.6-10 for a definition of Cases a through d.

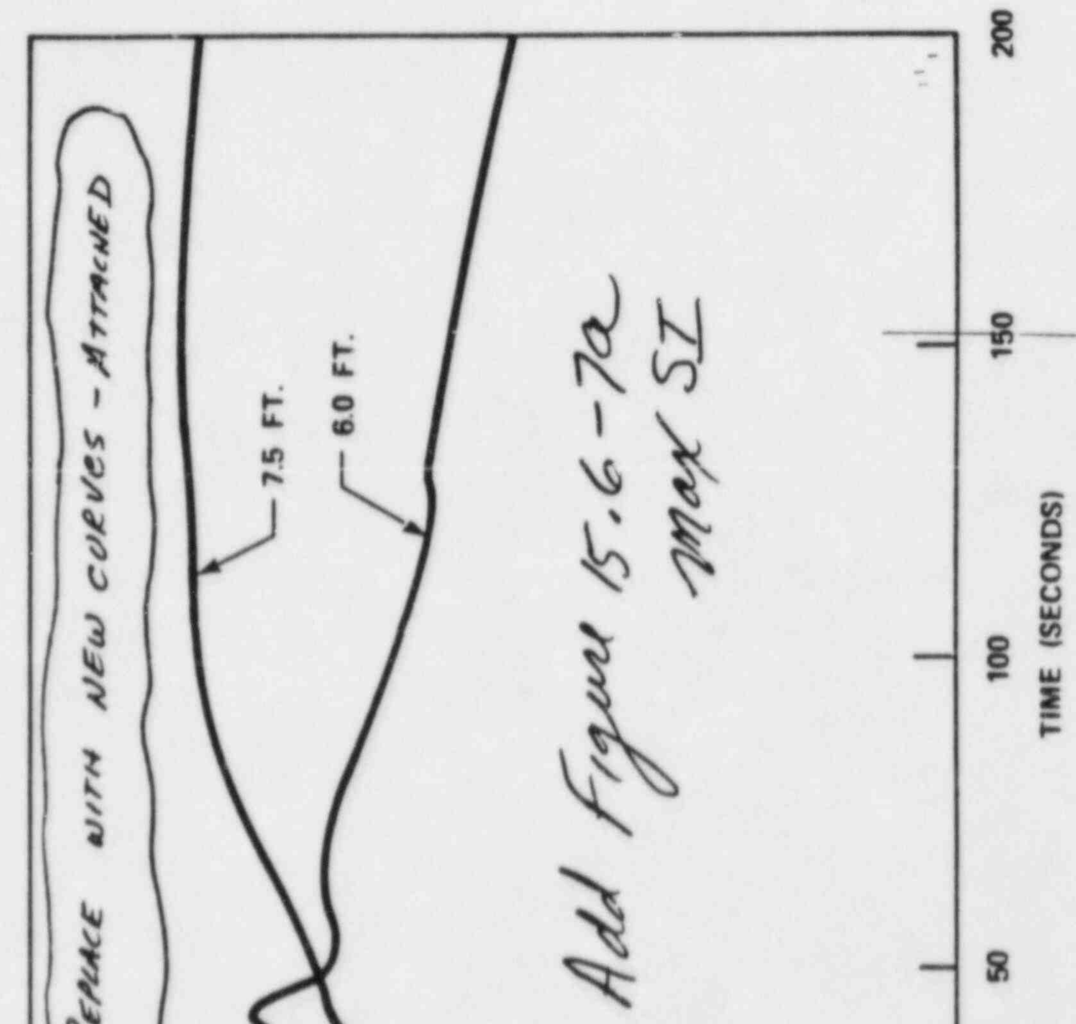

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FIGURE 15.6-7

PEAK CLAD TEMPERATURE – DECLG (C_D = 0.6)

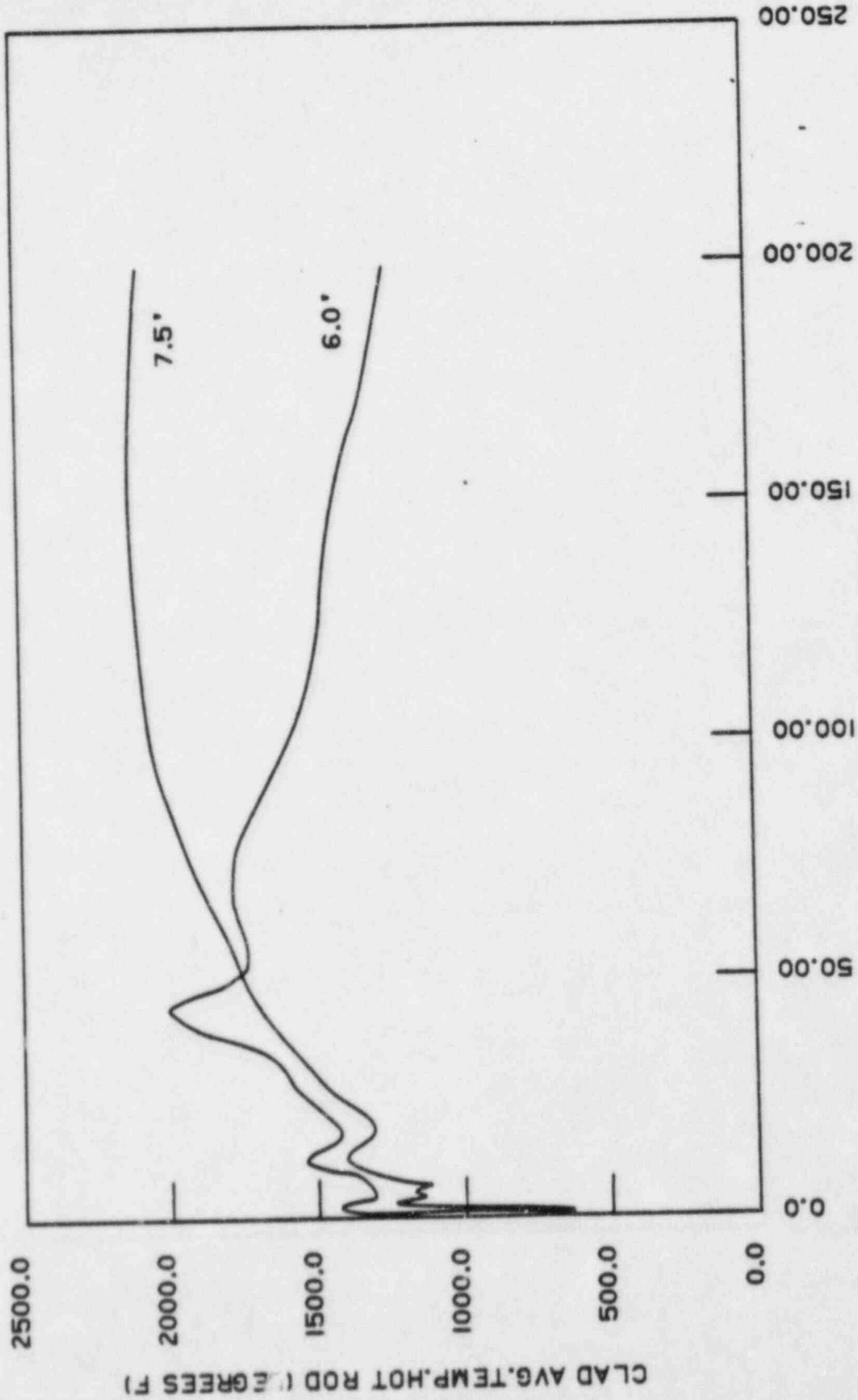

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FIGURE 15.6-7
PEAK CLAD TEMPERATURE
DECLG (C_D=0.6 - MIN S.
12 SEC D/G START)

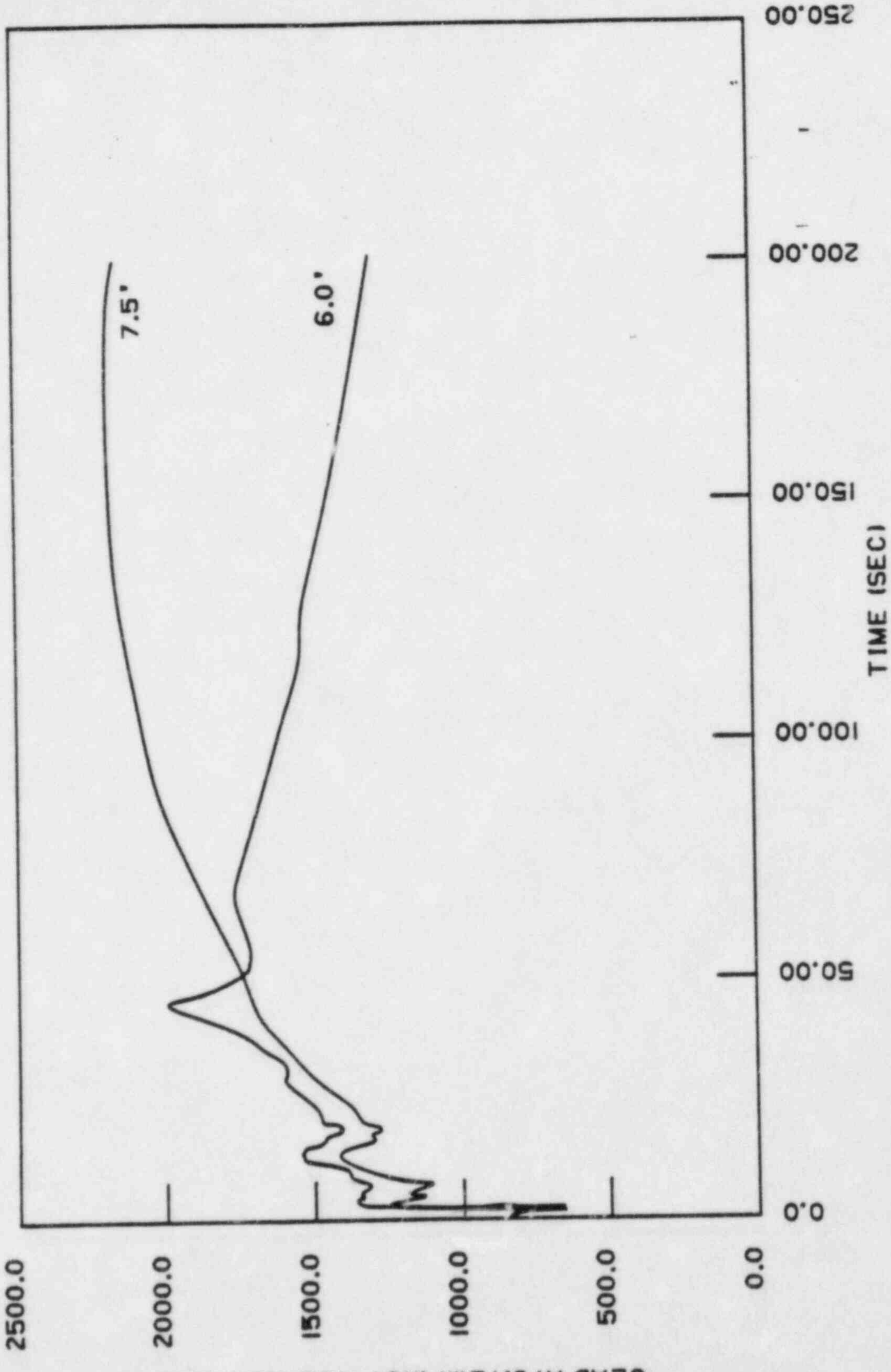

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FIGURE 15.6-[illegible]7a
PEAK CLAD TEMPERATURE
DECLG (C_D=0.6-MAX SI-
12SEC D/G START)

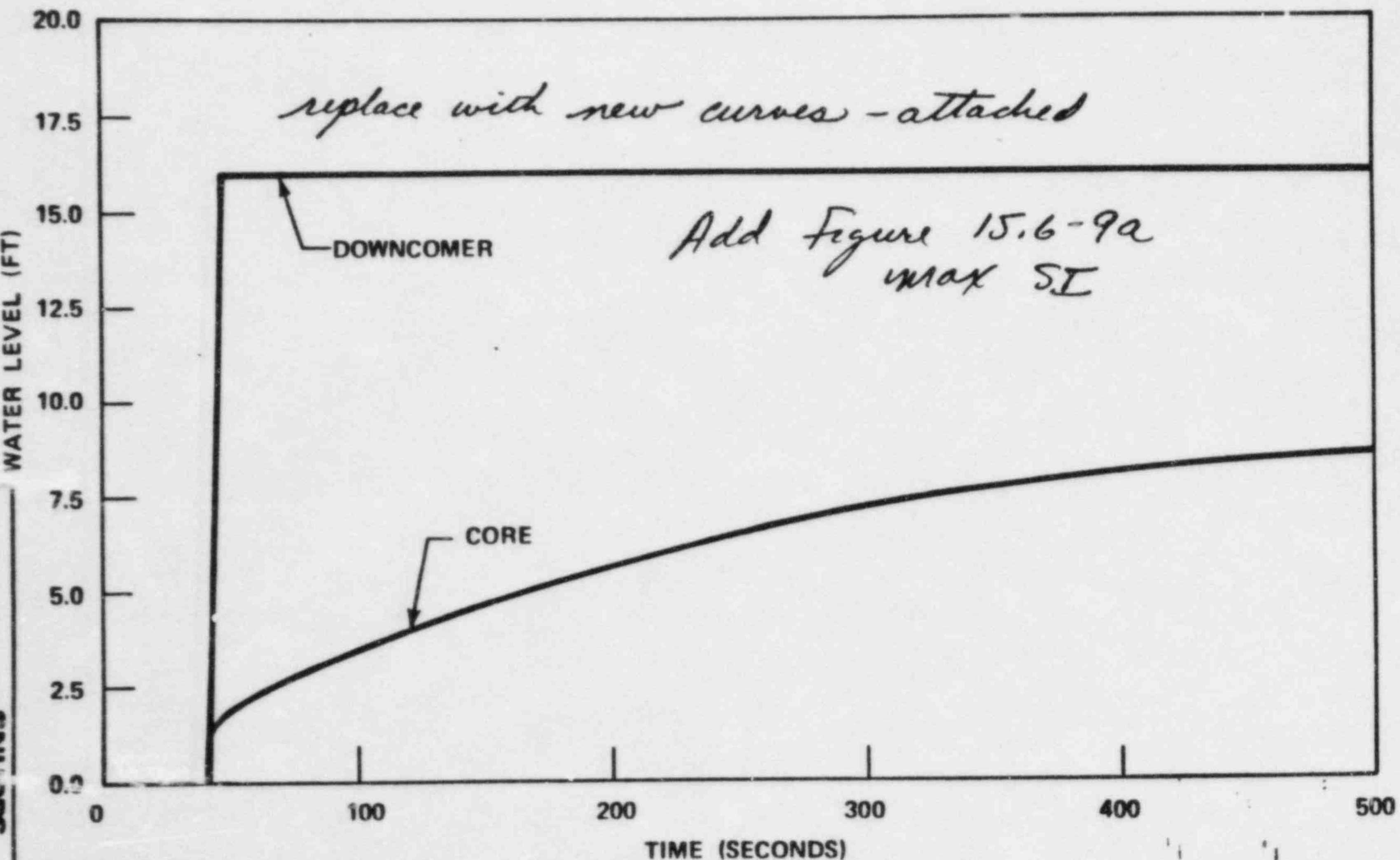

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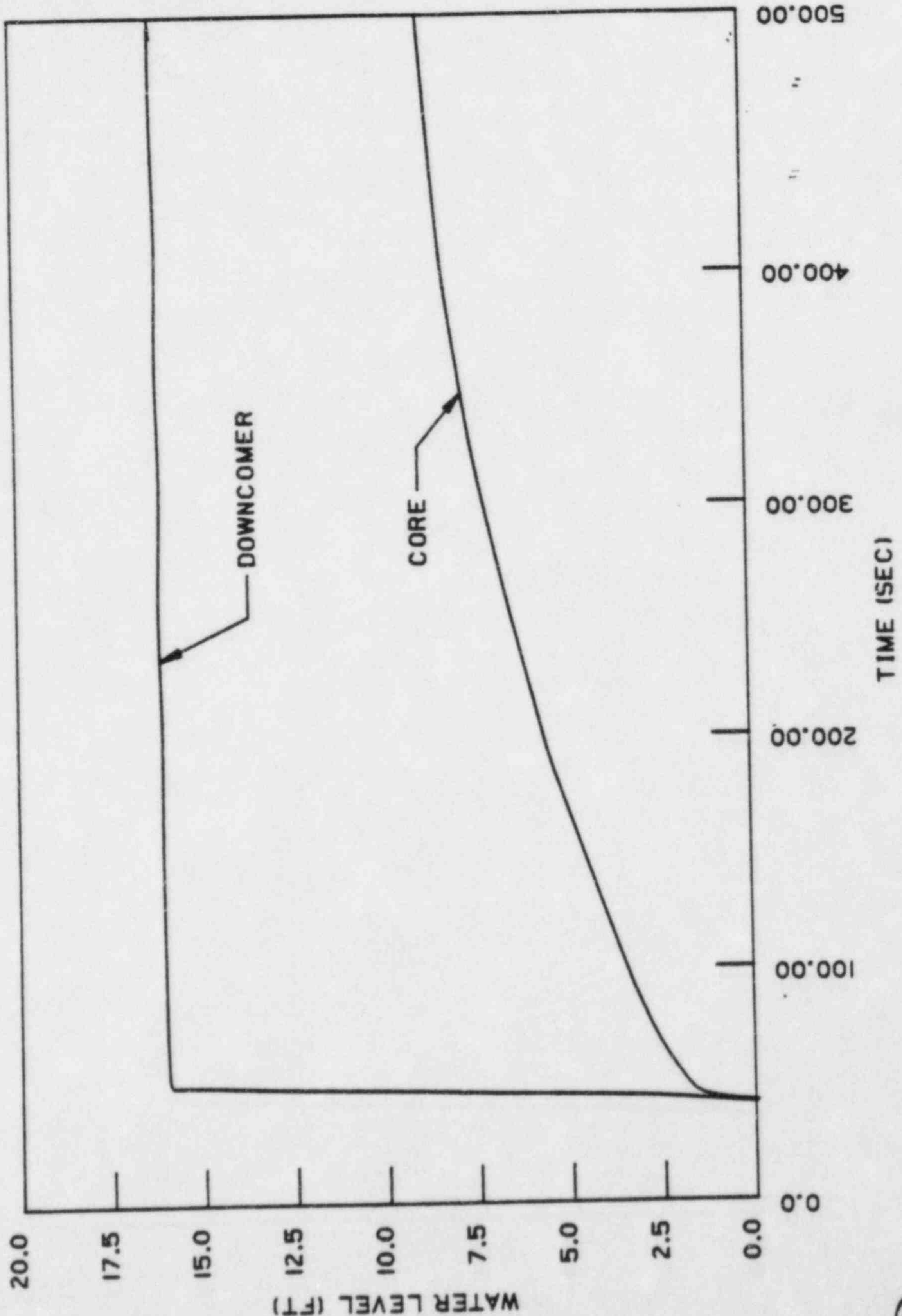
FIGURE 15.6-9

DECLG (C_D = 0.6) DOWNCOMER AND CORE WATER LEVELS DURING REFLOOD



MinSI-12 SEC D/G St

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FIGURE 15.6-9

DECLG (C_D=0.6) DOWNCOMER AND CORE WATER LEVELS DURING REFLOOD

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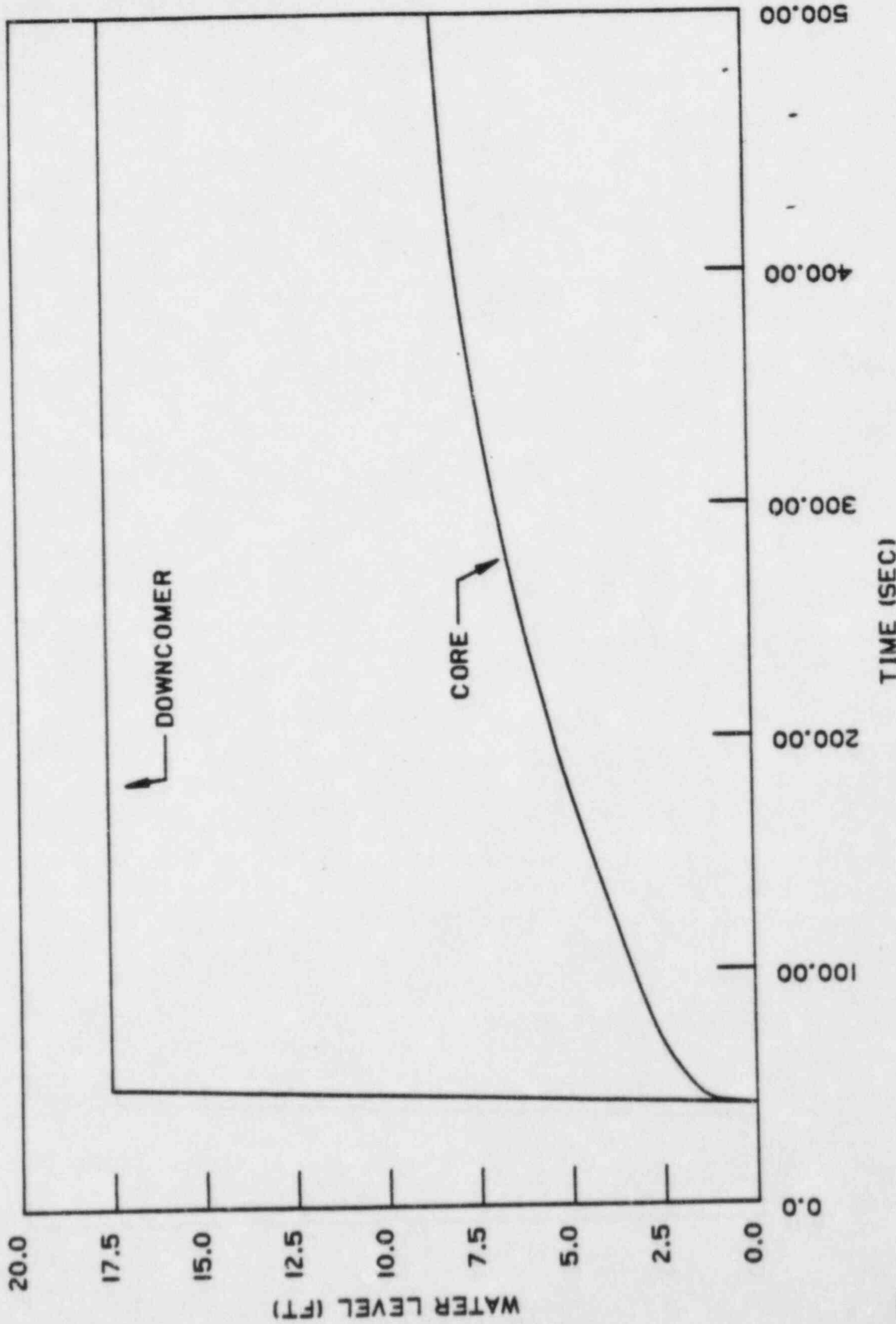

12SEC D/G STAR

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FIGURE 15.6-9a
DECLG (C $_D$=0.6 MAX SI)
DOWNCOMER AND CORE WATE
LEVELS DURING REFLOOD

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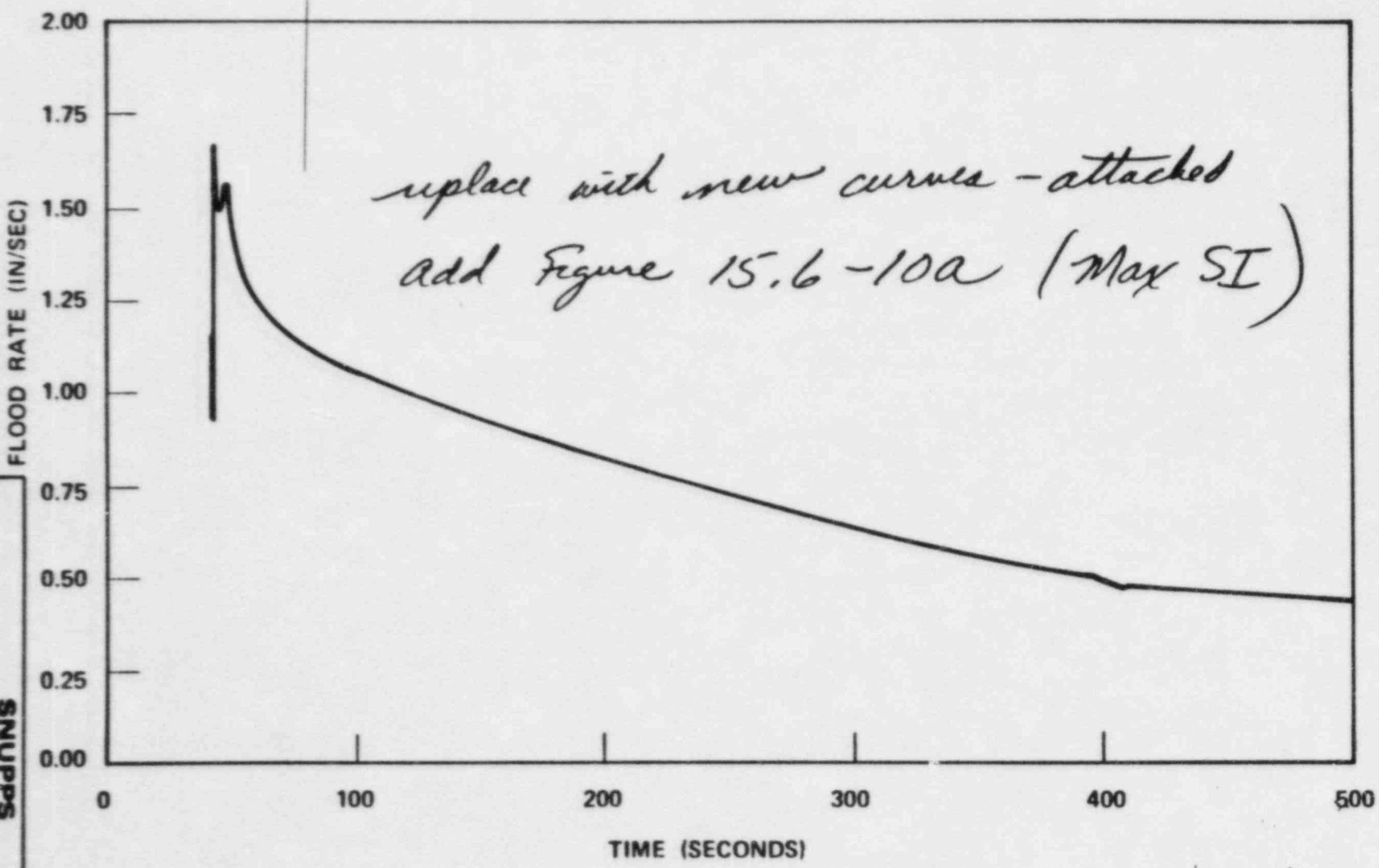

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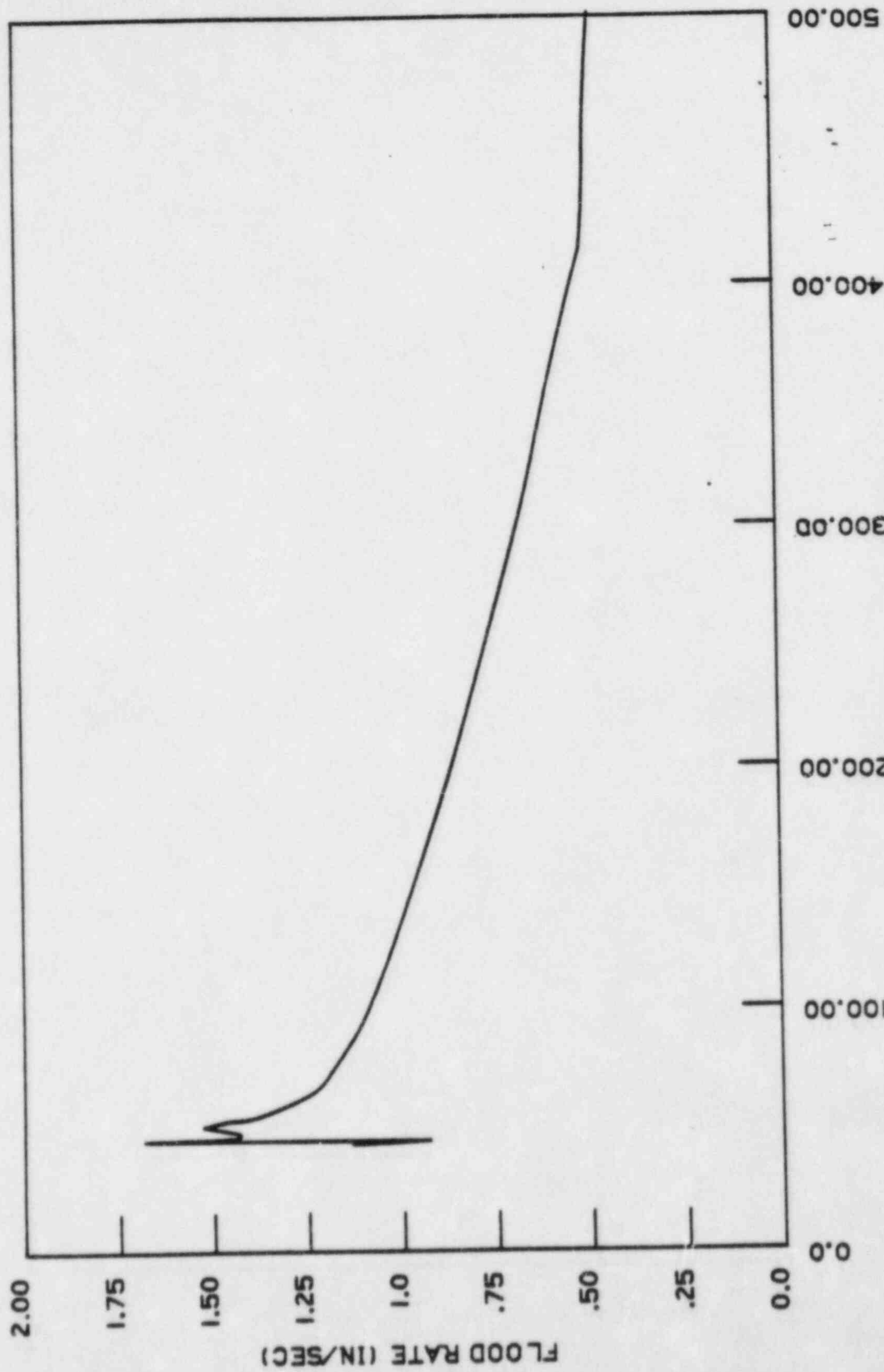
FIGURE 15.6-10

DECLG (C_D = 0.6) CORE INLET VELOCITY DURING REFLOOD



Min SI-
12 SEC D/G
START

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FIGURE 15.6-10

DECLG (C_D=0.6)

CORE INLET VELOCITY

DURING REFLOOD

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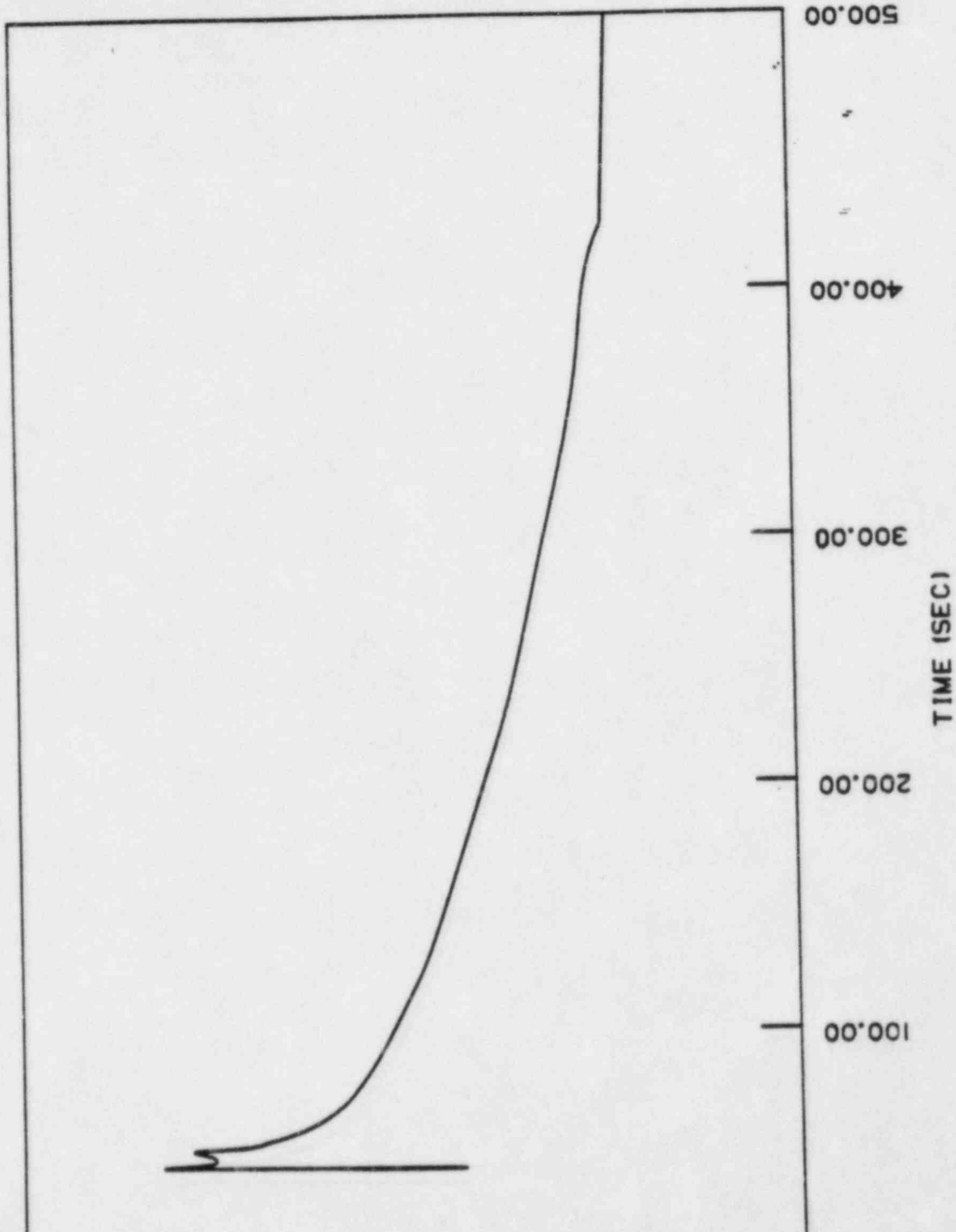

12 SEC D/G START

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FIGURE 15.6-10a
DECLG (C_D=0.6 MAX SI)
CORE INLET VELOCITY
DURING REFLOOD

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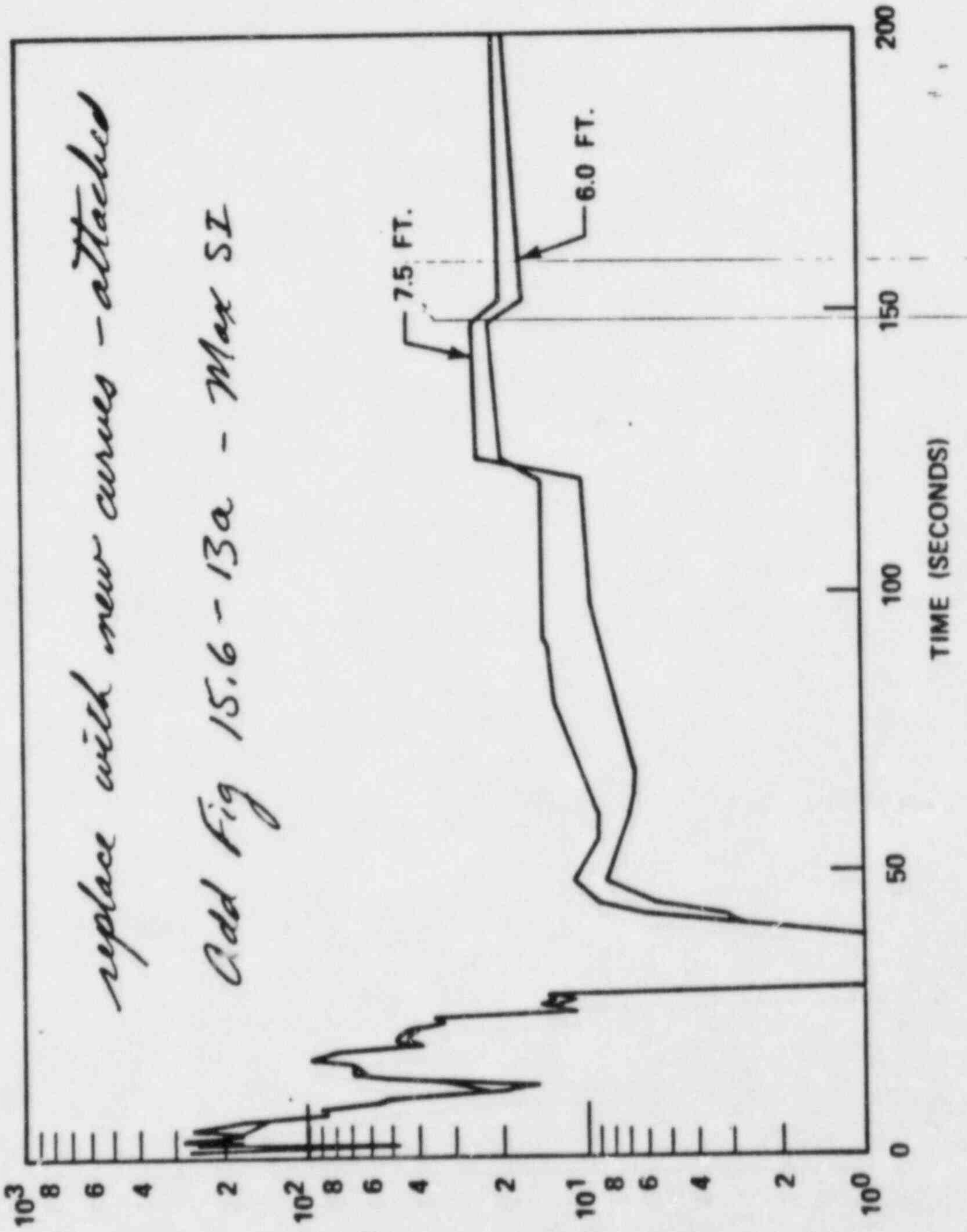

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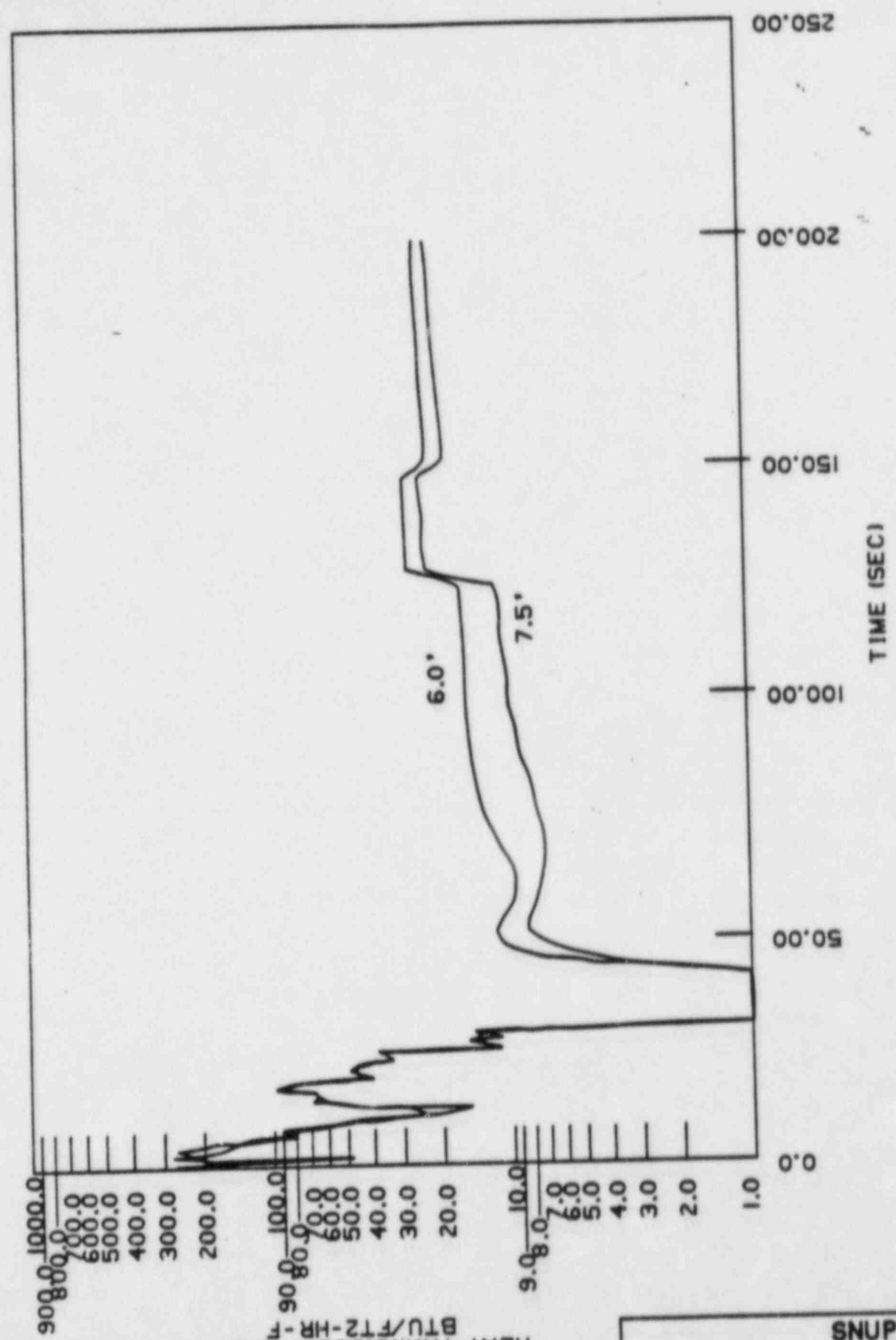
FIGURE 15.6-13

CORE HEAT TRANSFER COEFFICIENT – DECLG (C_D = 0.6)



MIN SI-
12 SEC D/G
START

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FIGURE 15.6-13
CORE HEAT TRANSFER
COEFFICIENT
DECLG (C_D=0.6)

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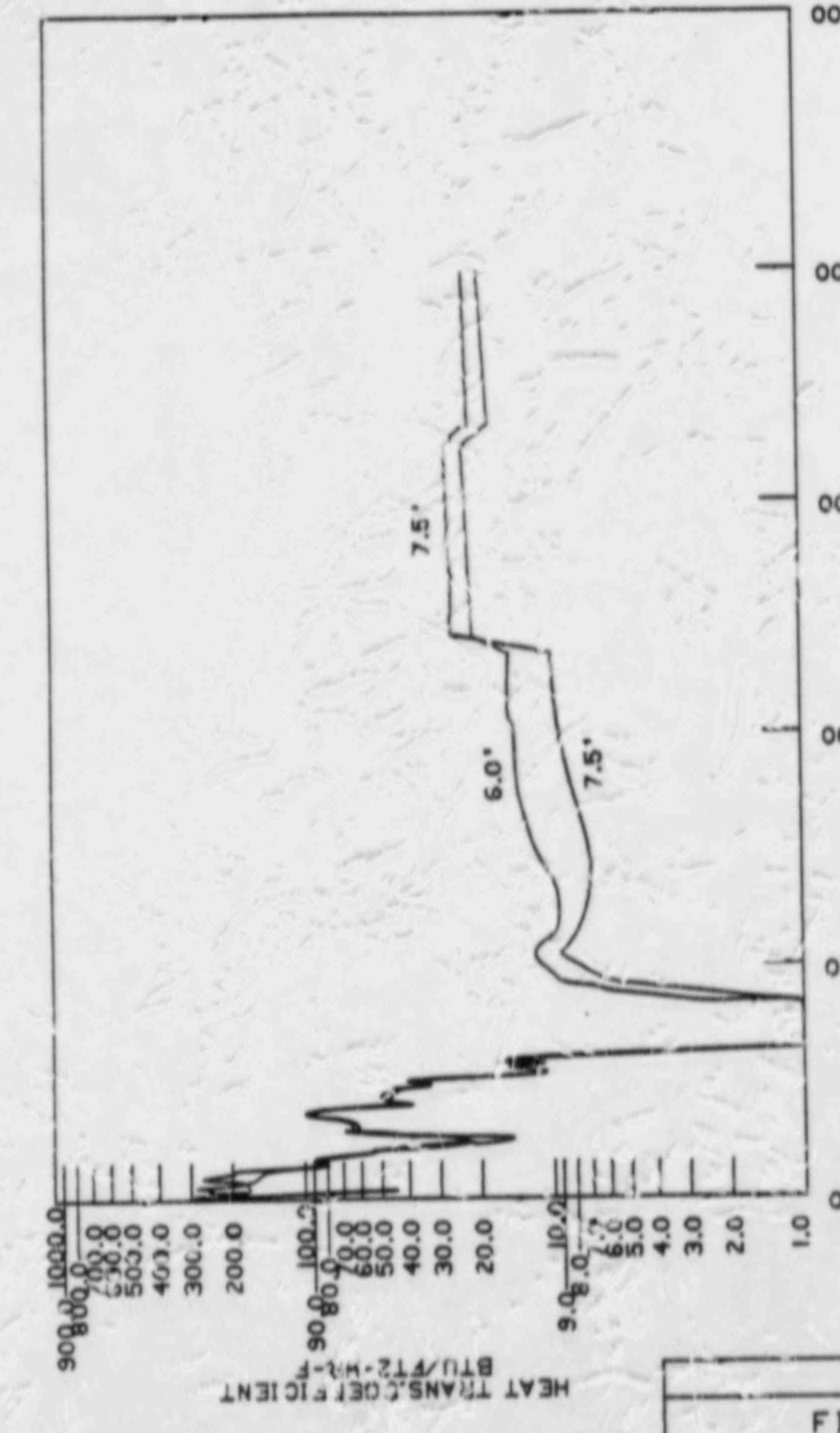

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12 Sec D/G Sta

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FIGURE 15.6-13a
CORE HEAT TRANSFER
COEFFICIENT
DECLG (C_D=0.6 MAX S

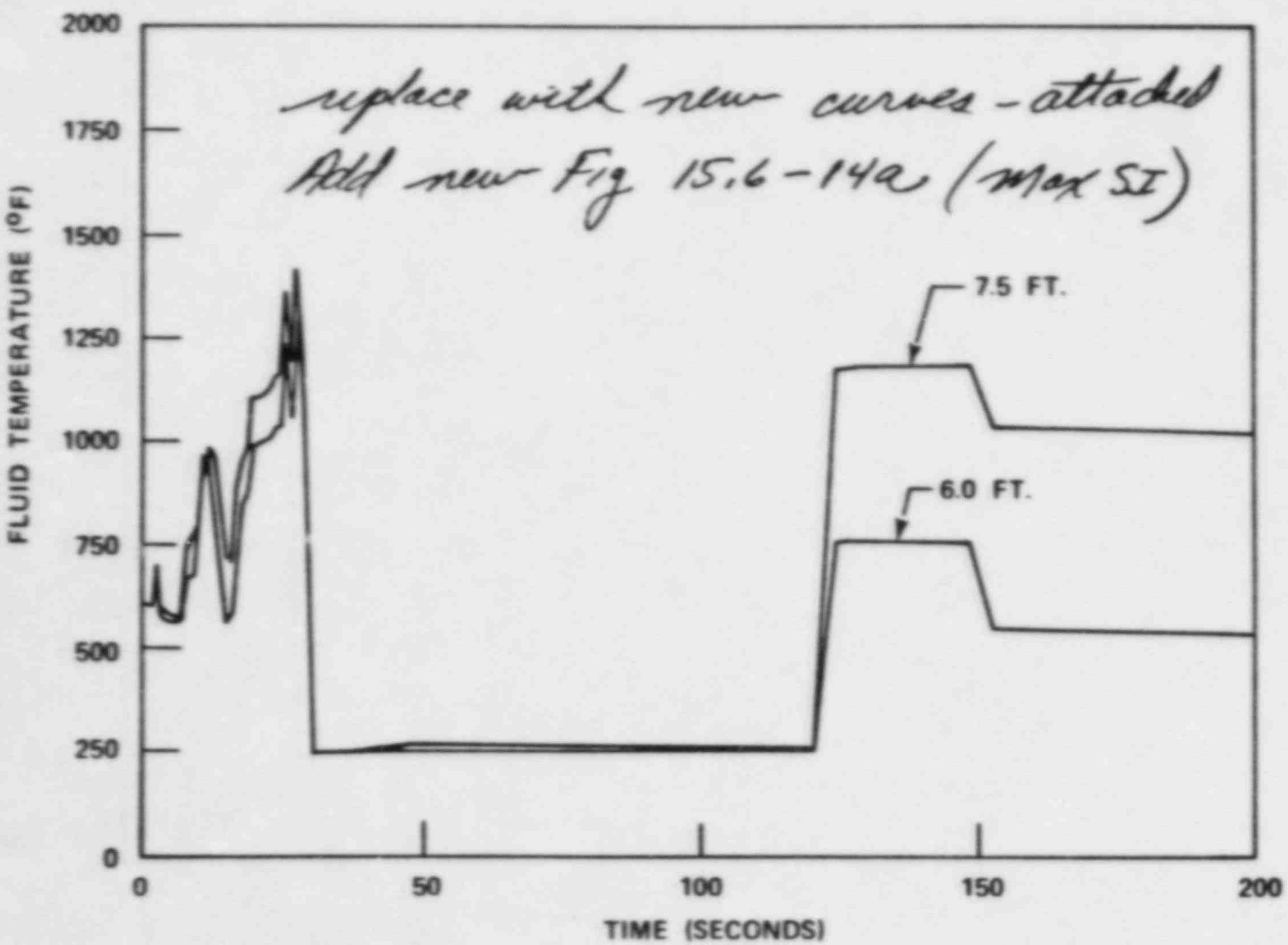

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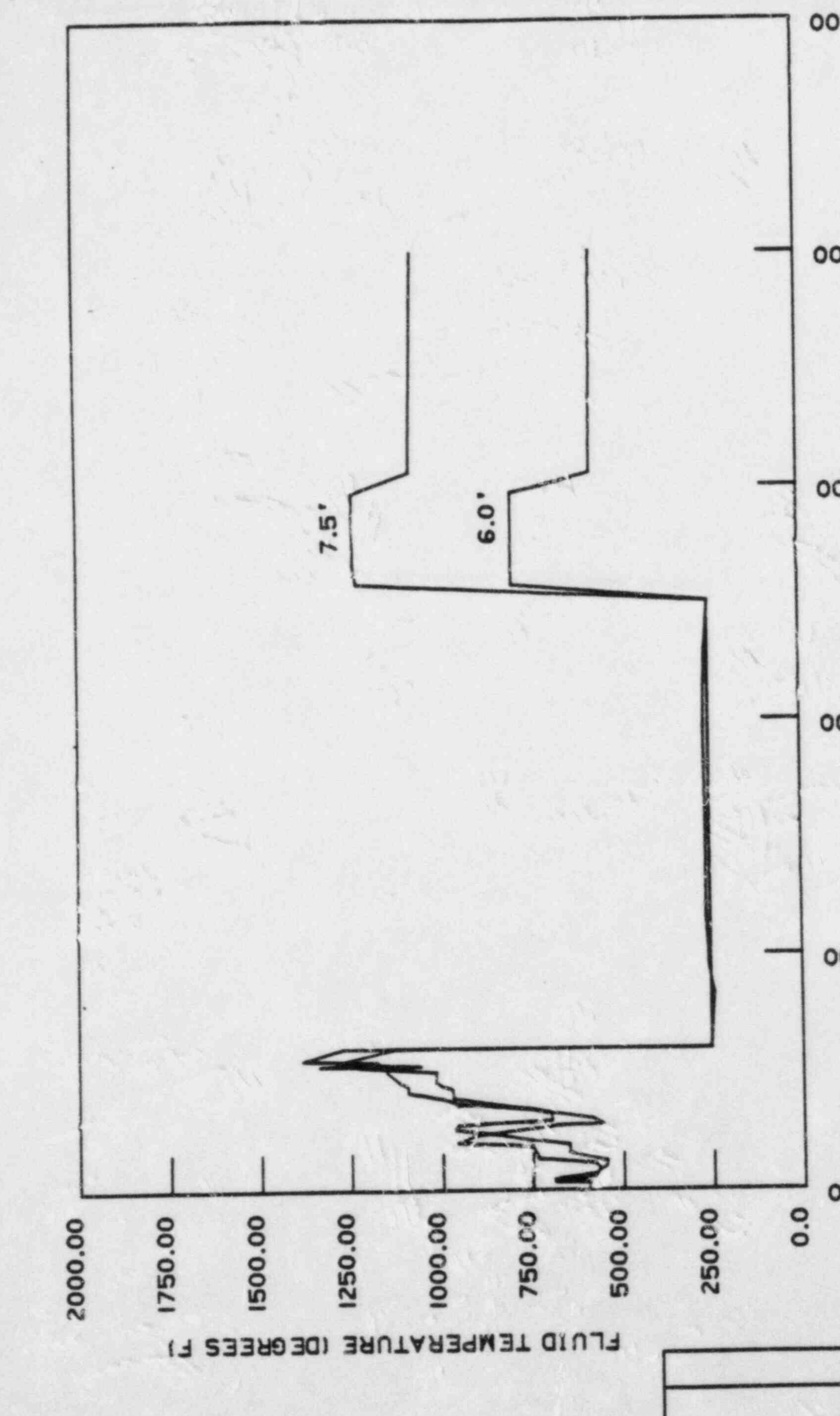
FIGURE 15.6-14

FLUID TEMPERATURE – DECLG (C_D = 0.6)



Min SI-12
Sec D/G
Start

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FIGURE 15.6-14
FLUID TEMPERATURE
DECLG (C_D=0.6)

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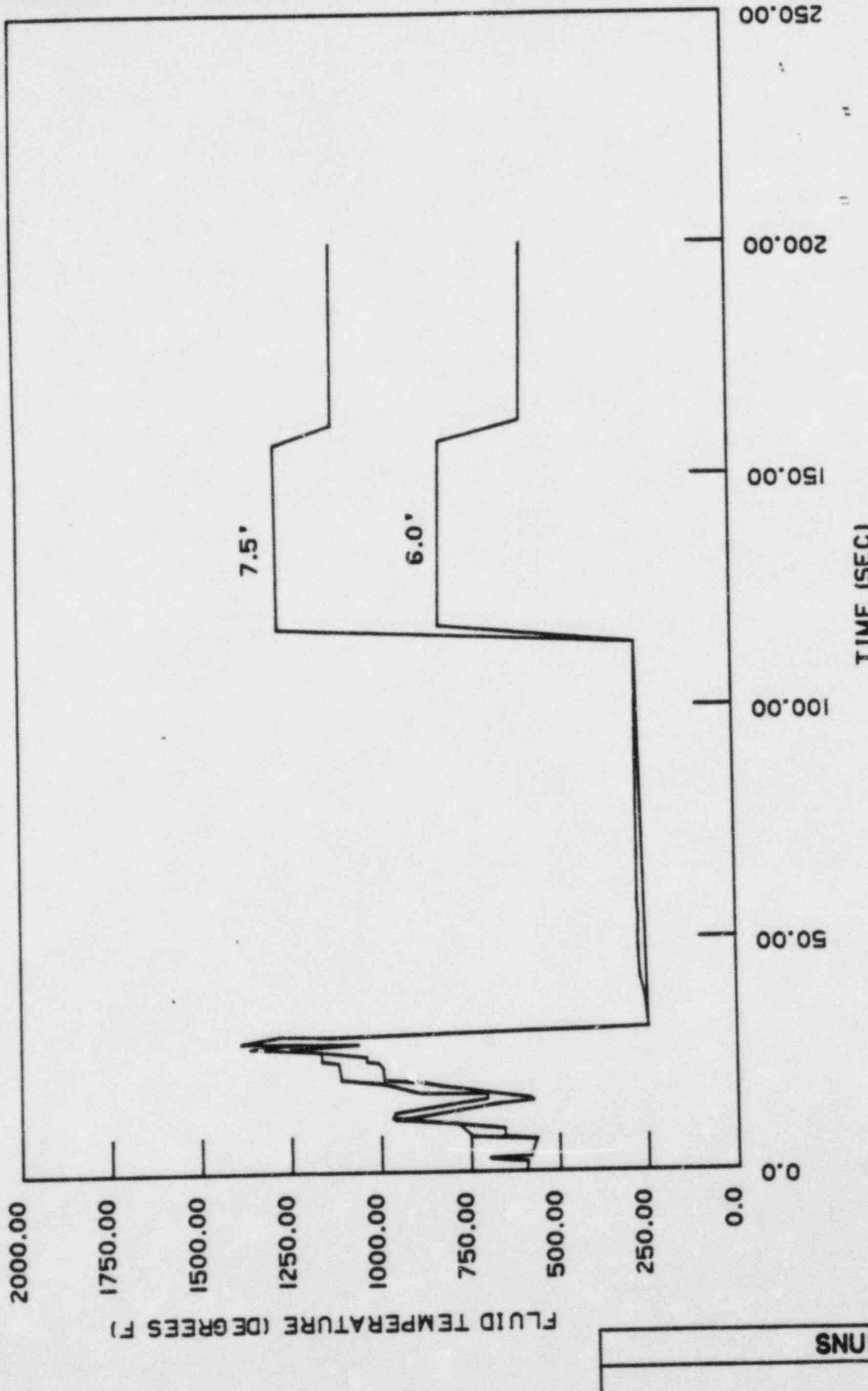

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FIGURE 15.6-14a
FLUID TEMPERATURE,
DECLG (C_D=0.6 MAX SI)

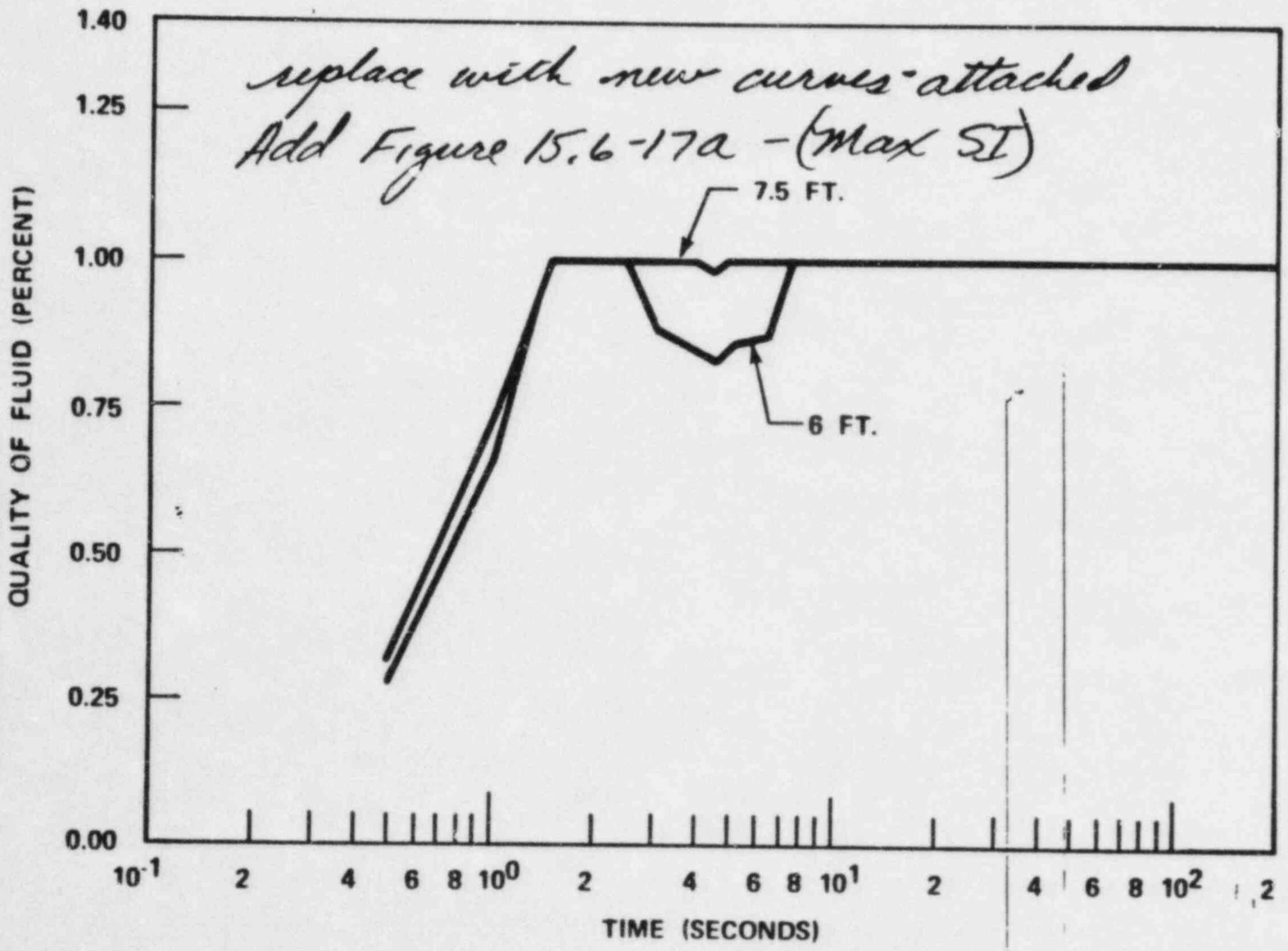

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FIGURE 15.6-17

FLUID QUALITY – DECLG (C_D = 0.6)

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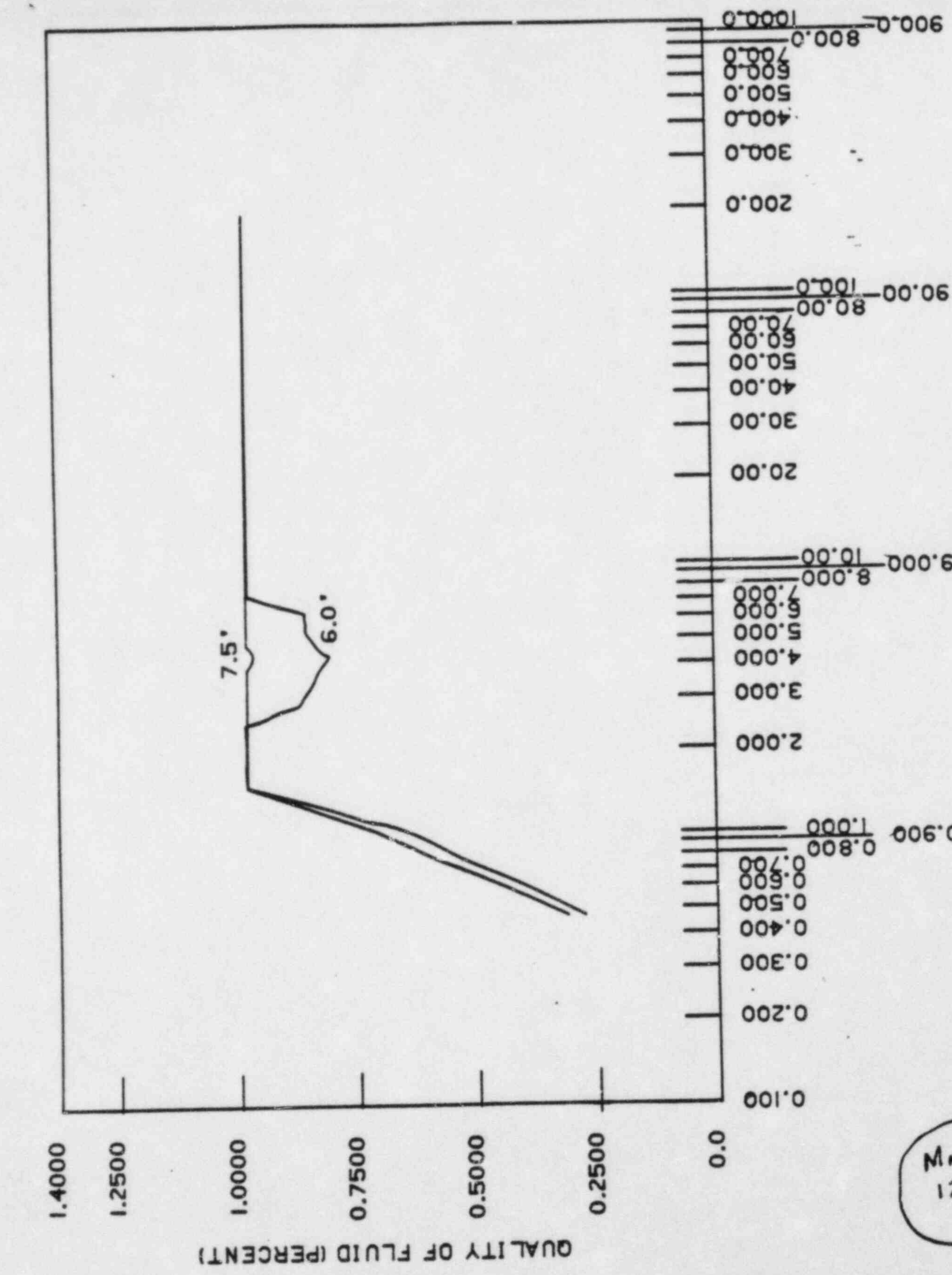

MIN SI-
12 SEC D/G
START

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FIGURE 15.6-17
FLUID QUALITY
DECLG (C_D=0.6)

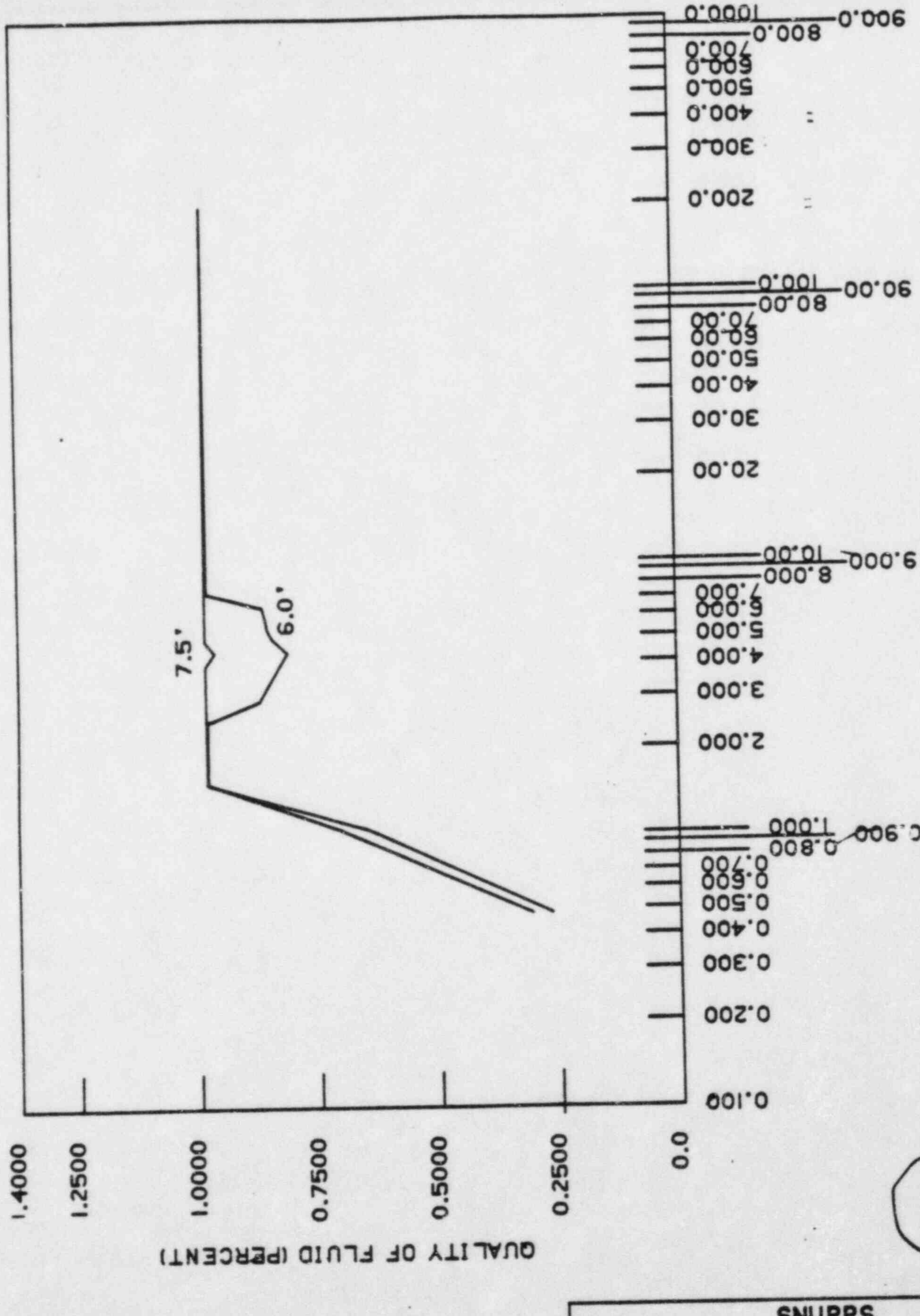

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FIGURE 15.6-[redacted]17a
FLU[redacted]D QUALITY
DECLG (C_D=0.6 MAX SI)

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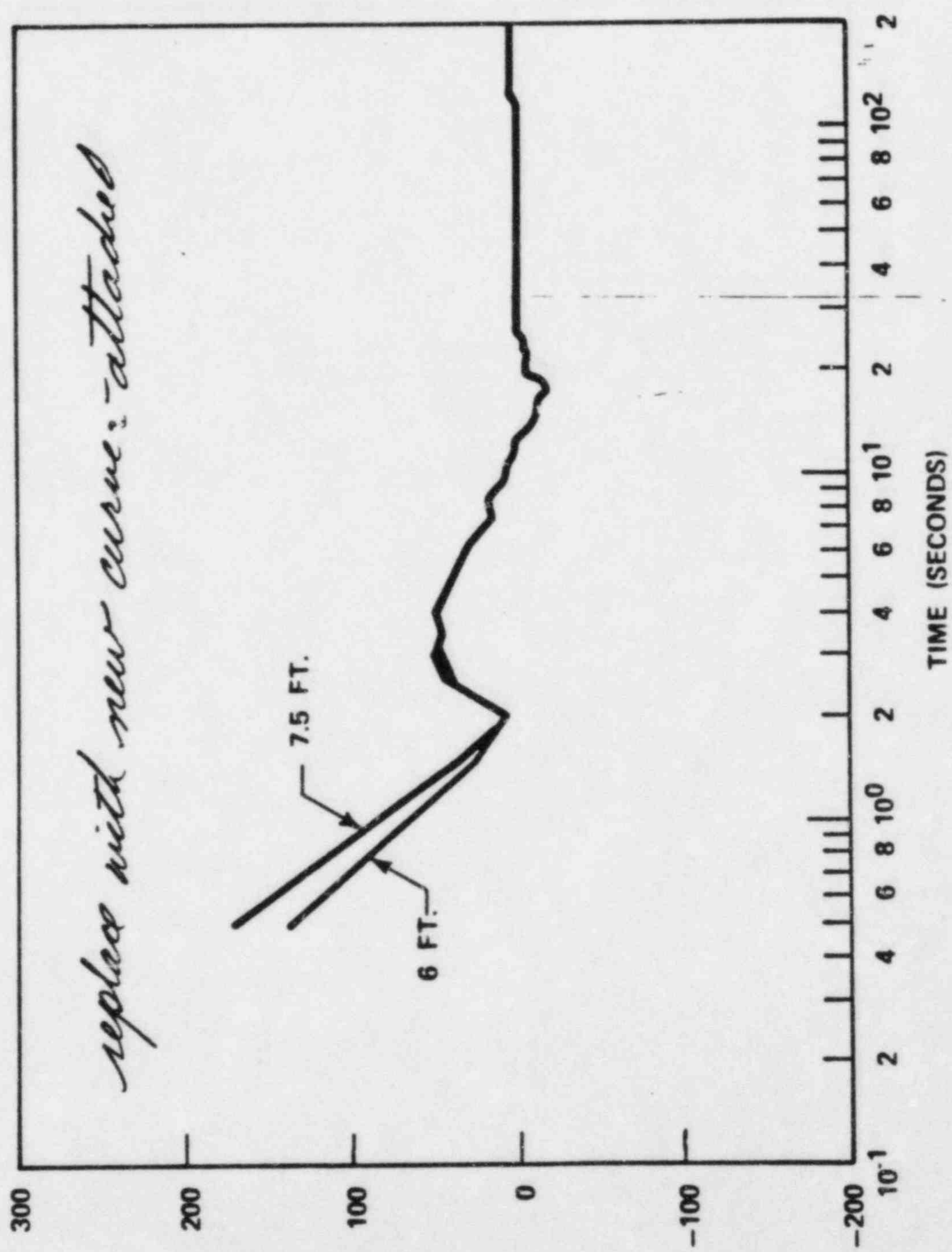

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FIGURE 15.6-19

MASS VELOCITY – DECLG (C_D = 0.6)

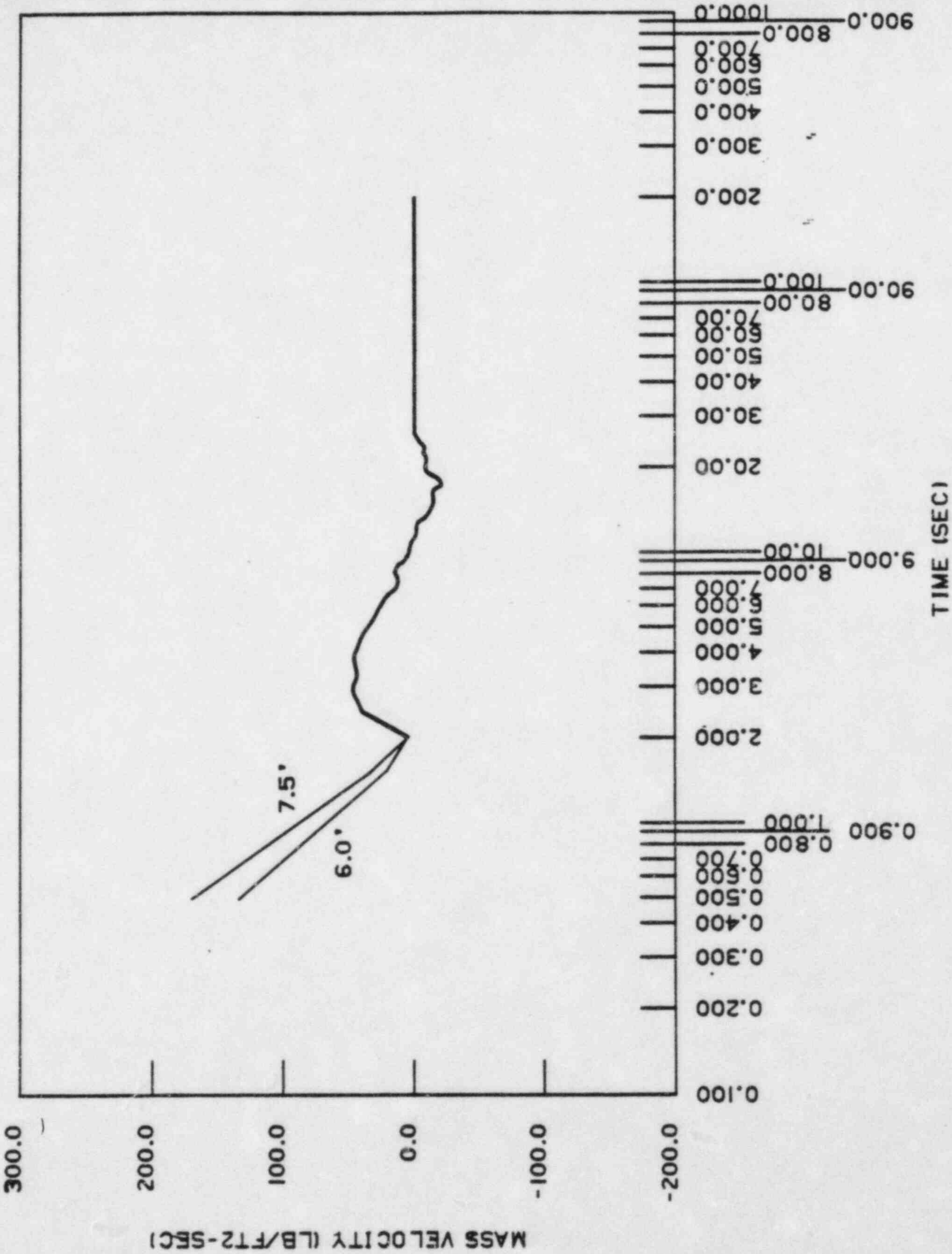

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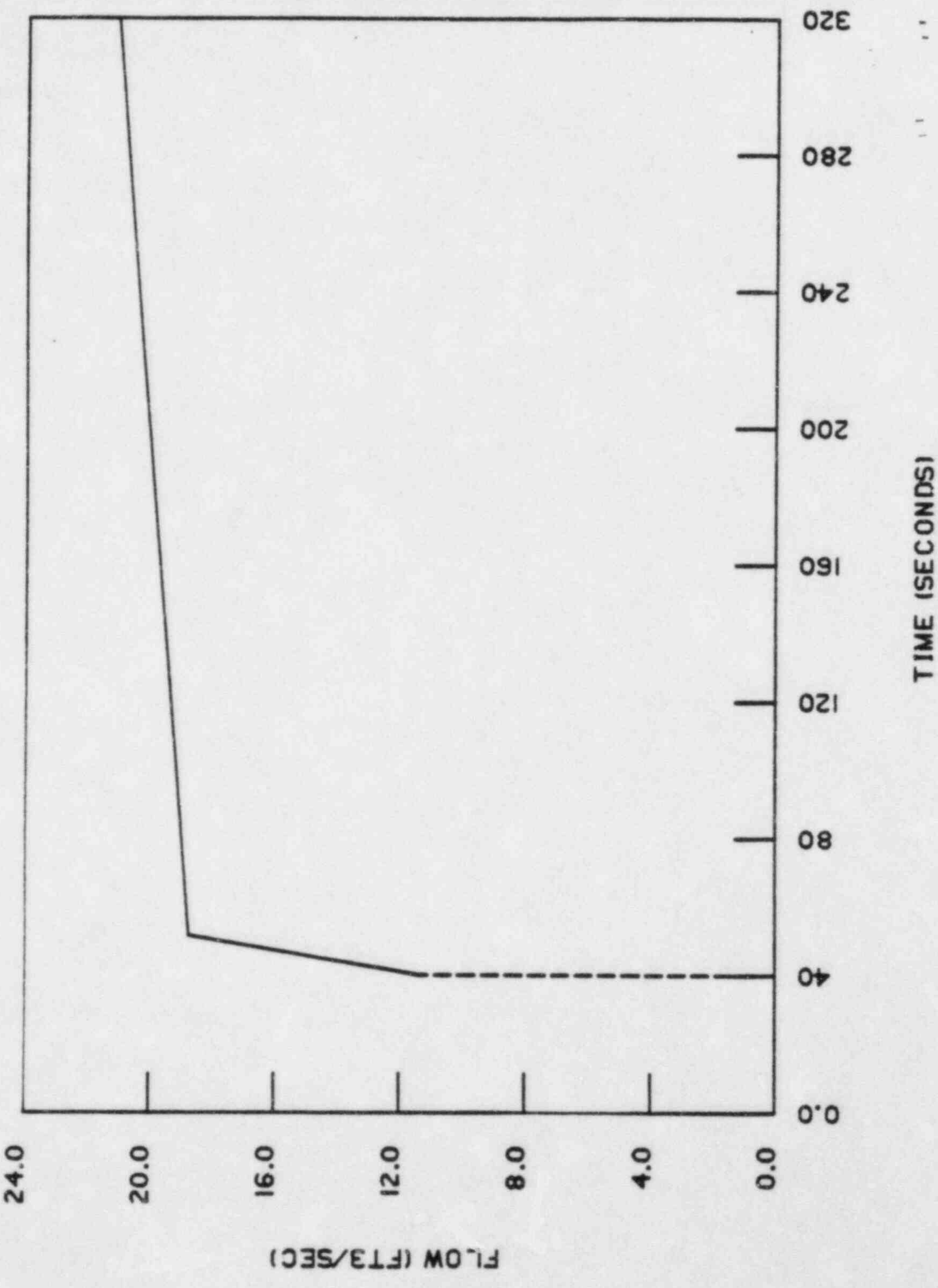
FIGURE 15.6-19
MASS VELOCITY
DECLG (C_D=0.6)



12 SEC D/G START

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FIGURE 15.6-20a
PUMPED ECCS FLOW
DURING REFLOOD
DECLG (C_D=0.6 MAX SI)