FROM:	DATE OF DOCUMENT:	DATE RECEIVED	NO.:
Carolina Power & Light Company Raleigh, North Carolina 276 E. B. Bessac	12-8-71	12-13-71	

LTR.	MEMO:	EPORT:	OTHER:
X			

TO:	ORIG.:	CC:	OTHER:
Dr. Peter A. Morris	1		

ACTION NECESSARY ☐	CONCURRENCE ☐	DATE ANSWERED:
NO ACTION NECESSARY ☐	COMMENT ☐	BY:

CLASSIF:	POST OFFICE	FILE CODE:
U	REG. NO:	50-261 Misc. Appl.

DESCRIPTION: (Must Be Unclassified)
Ltr re their 9-29 & 11-5-71 ltrs... furnishing addl info re ECCS Per Rpt.. reanalyses of small break loss-of-coolant accidents & trans:

ENCLOSURES:
Figures 1 thru 6

(1 cy ea encl rec'd)

REMARKS:

REFERRED TO	DATE	RECEIVED BY	DATE
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Carolina Power & Light Company

Raleigh, North Carolina 27602

December 8, 1971

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
EMERGENCY CORE COOLING SYSTEM

Dear Dr. Morris:

This letter supplements the information submitted in Carolina Power & Light Company's letters of September 29, 1971 and November 5, 1971 concerning the Emergency Core Cooling System. Specifically, re-analyses of the small break loss-of-coolant accidents with the hot leg injection blocked are enclosed with this submittal.

The re-analyses are based on a single failure of one emergency diesel generator or power train resulting in two safety injection pumps delivering separately through three cold leg lines. The lowest resistance cold leg line is assumed to spill its flow to the containment through the break. The delivery curve for this case is presented in Figure 1. The pump discharge pressures indicated include the 5% reduction used in these analyses. The reactor coolant system volumes illustrate both quiet and froth levels for the range of break sizes and are presented in Figures 2 through 5, while the pressure transients are presented in Figure 6.

The peak clad temperature for the spectrum of breaks analyzed is less than 1300°F. In this evaluation, it was conservatively assumed that the axial power distribution was skewed to the top of the core. The heat transfer (LOCTA) analyses used a core froth volume as calculated by the Wilson correlation. While the core was uncovered, credit was taken for the steam generation in the covered portion of the core flowing past the higher uncovered elevation of the fuel rods.

There is no effect on the Steam Break Accident in that the original FSAR analysis considered only injection to the cold legs.

Additionally, the conclusions drawn in regard to the Steam Generator Tube Rupture Accident are not affected by the elimination of the automatic hot leg injection. In the event of such an accident, safety injection flow would be reduced when the accident had been identified and when a minimum on-scale

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water level had been established in the pressurizer. This procedure would minimize break flow to the secondary system. It should be noted that, in the case of a single tube rupture, one safety injection pump delivering to one safety injection line would provide sufficient water to return the pressurizer to an on-scale reading.

Yours very truly,

N. B. Bessac
Mgr. - Nuclear Generation

NBB/dds
Enclosures

cc: Messrs. C. D. Barham
G. P. Beatty
J. A. Jones

50-261
3011-2

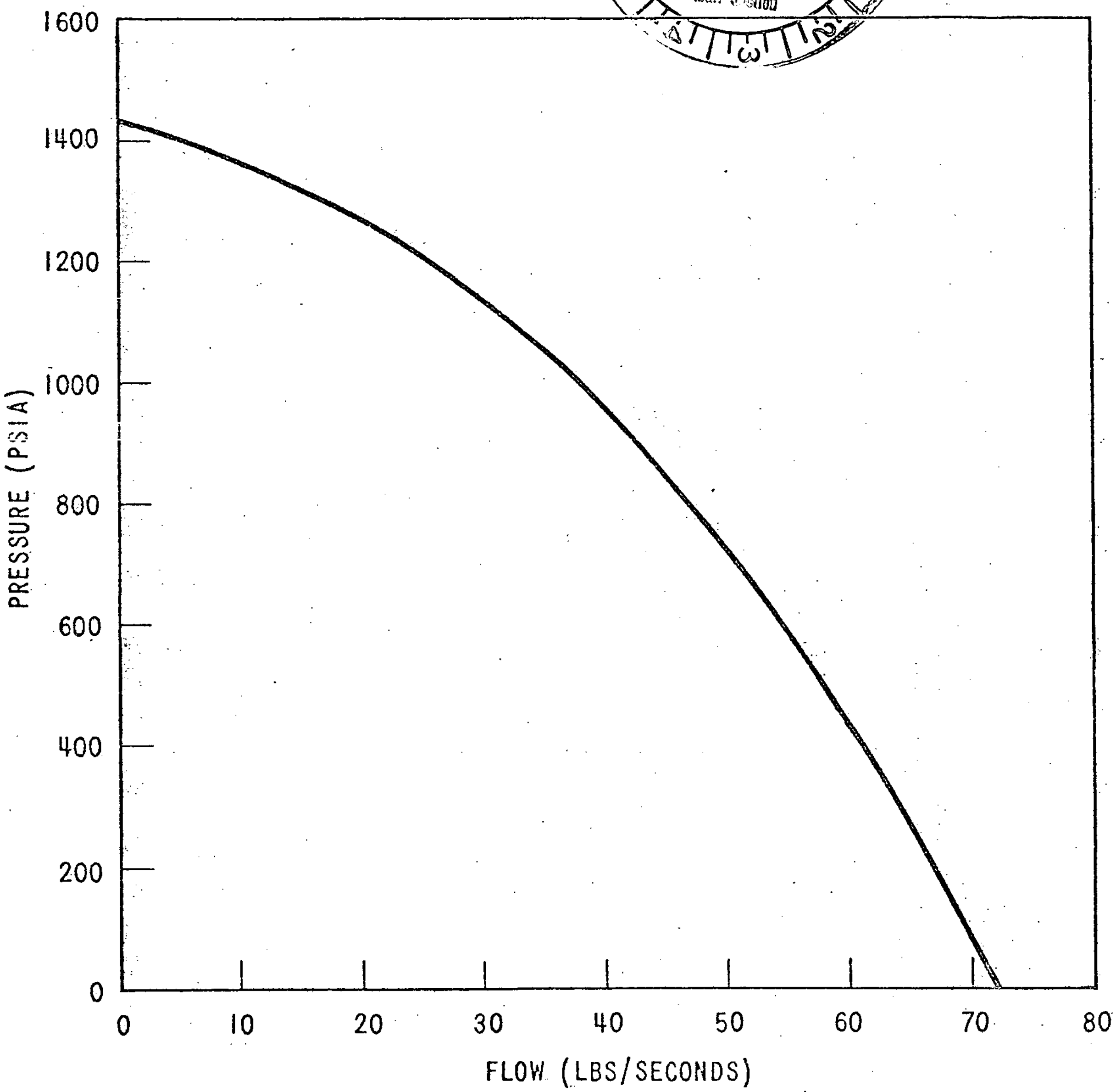

Figure 1.

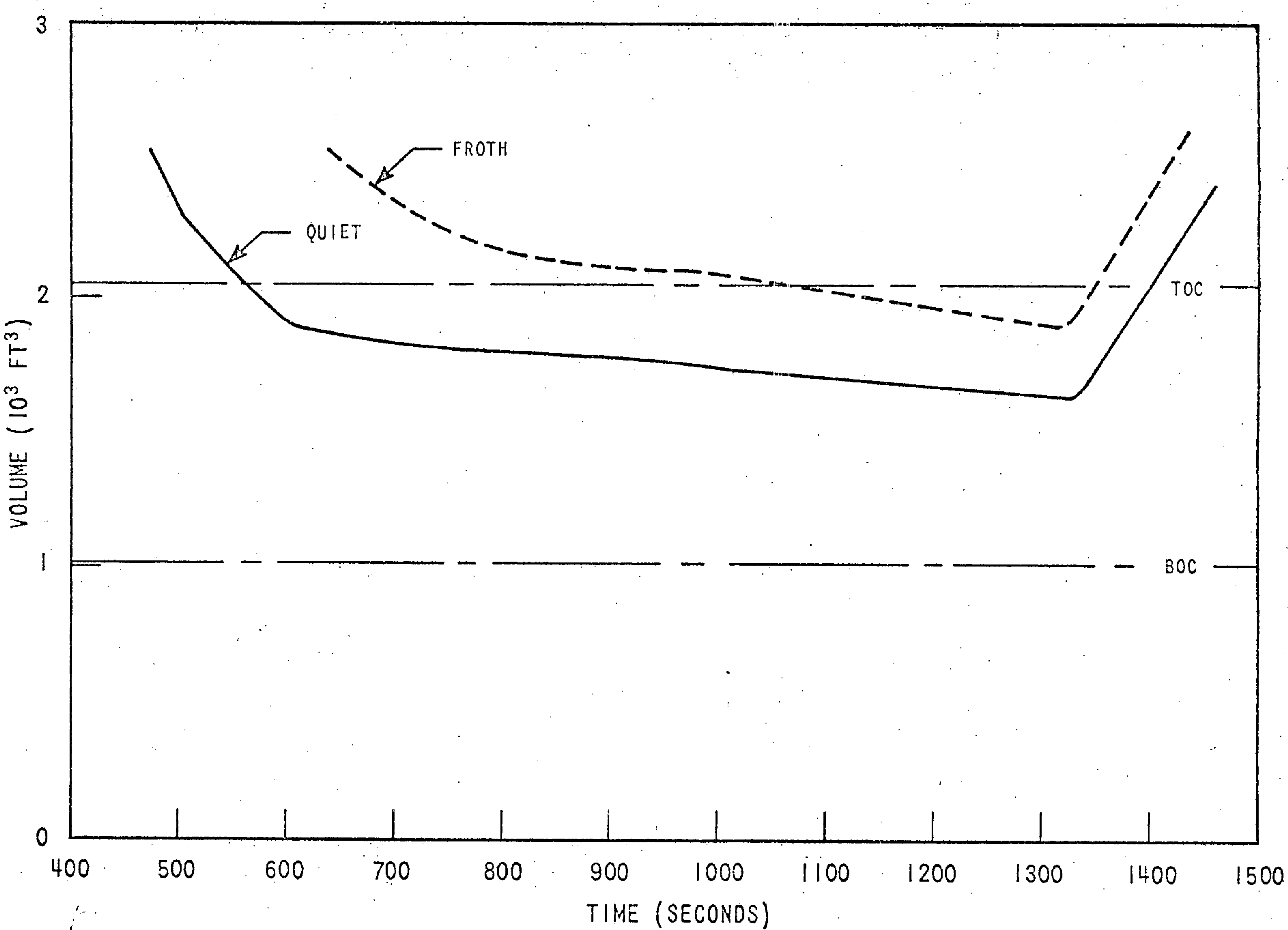

Figure 2. CPL Small Break LOCA Analyses Volume History
3.0 In. Dia. Break Minimum Safety Injection

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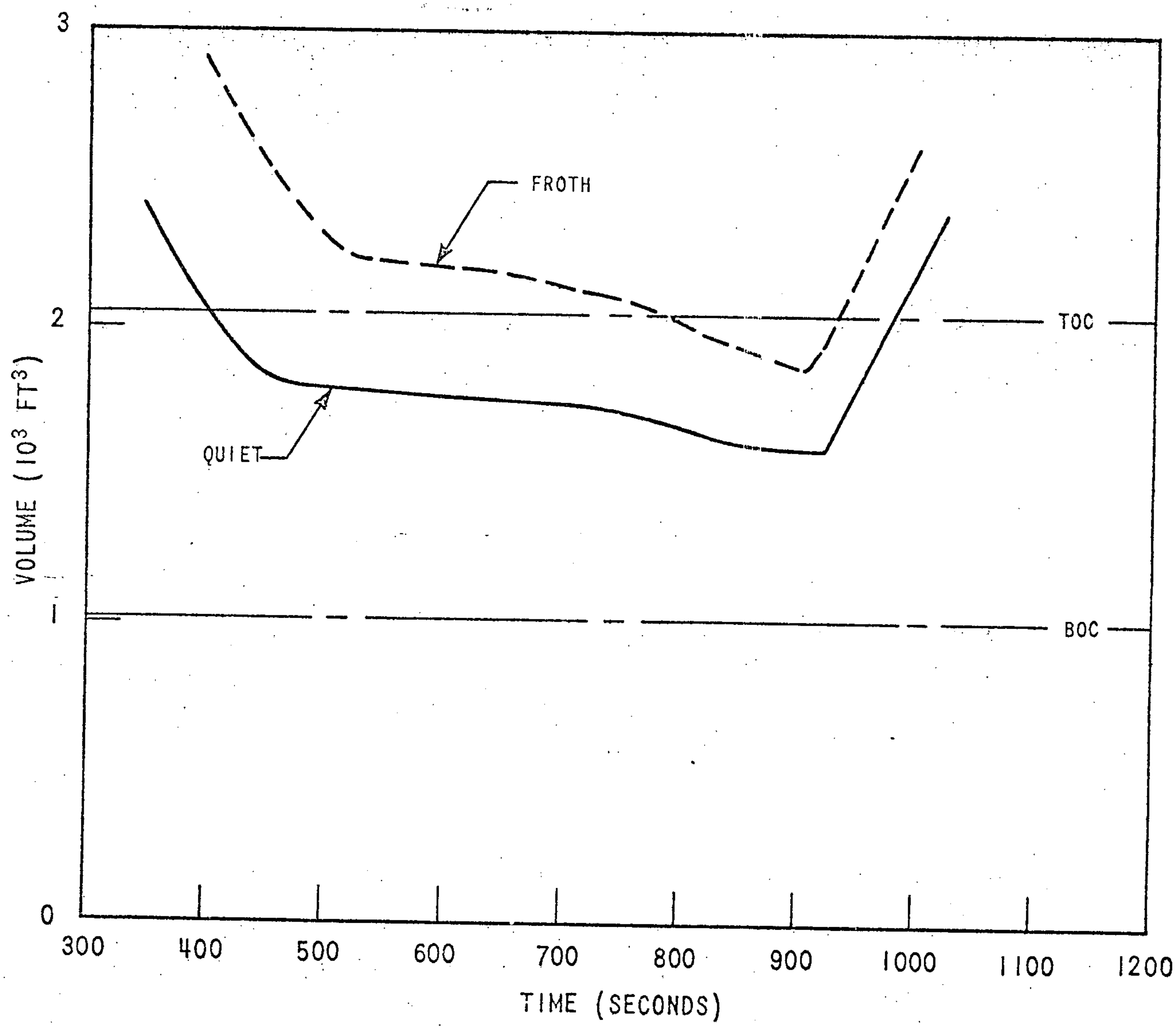

Figure 3. CPL Small Break LOCA Analyses Volume History
3.5 in. Dia. Break Minimum Safety Injection

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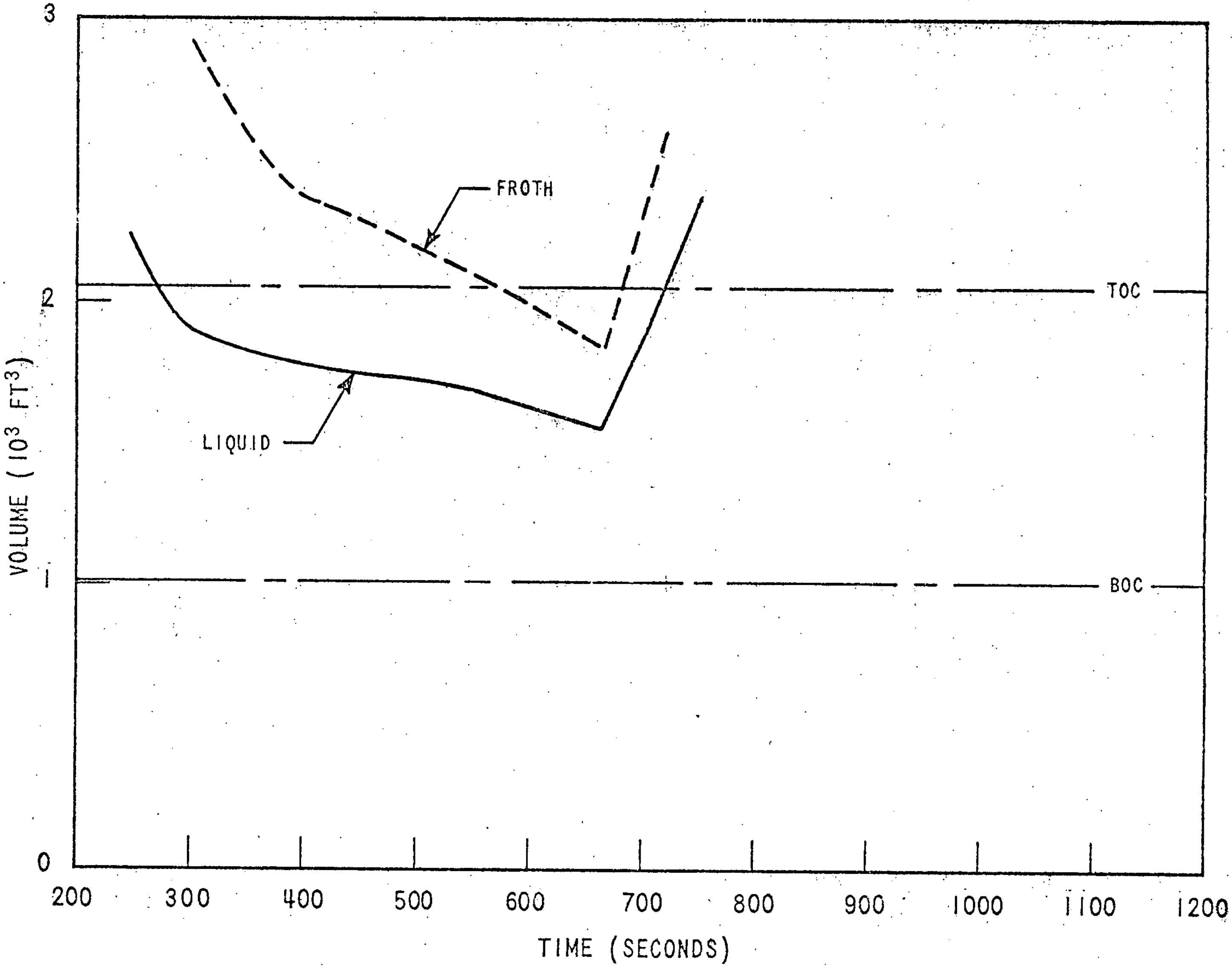

Figure 4. CPL Small Break LOCA Analyses Volume History
4.0 In. Dia. Break Minimum Safety Injection

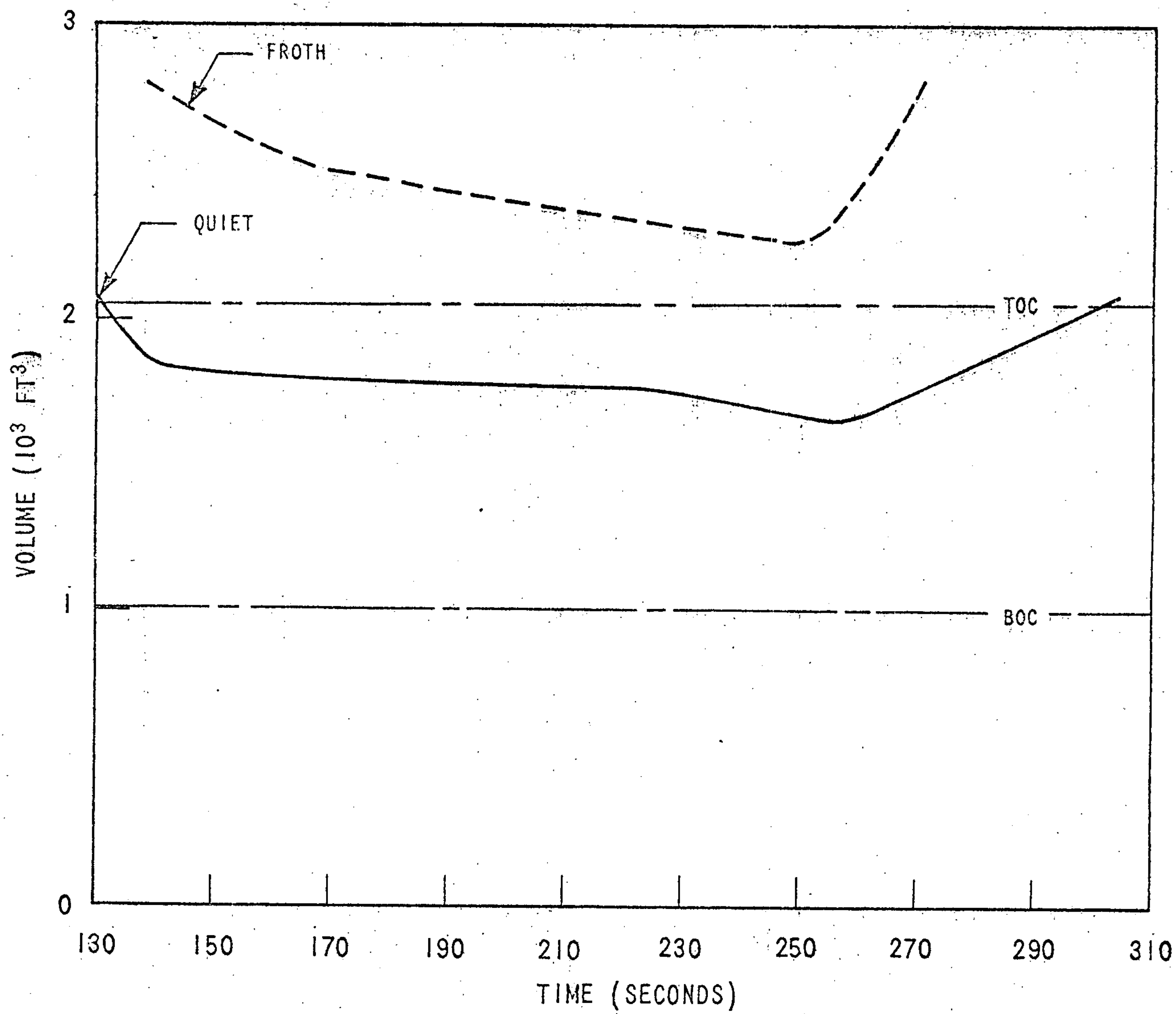

Figure 5. CPL Small Break LOCA Analyses Volume History
6.0 In. Dia. Break Min. Safety Injection

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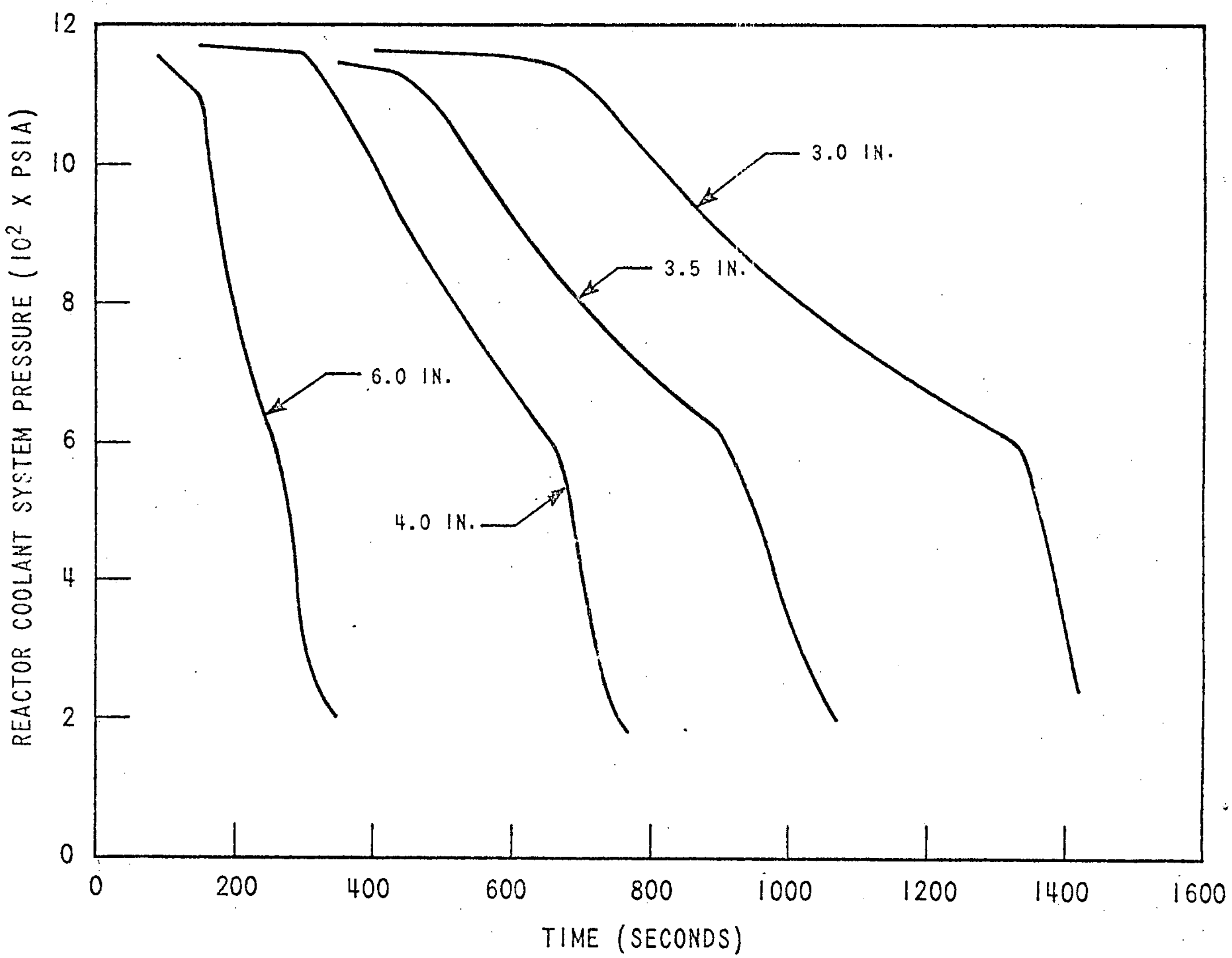

Figure 6. CPL Small Break LOCA Analyses Pressure History
Min. Safety Injection