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Carolina Power & Light Company Raleigh, N.C. 27602 Mr. E.E. Utley	1-4-74	1-11-74	X			

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DESCRIPTION:
Ltr req change #1 to the tech specs...incorporated in this change are minimum performance characteristics for the safety injection pumps and the supporting safety analyses for the small break loss-of-coolant accident (LOCA)....trans the following...

PLANT NAME: H.B. Robinson #2

ENCLOSURES:
PROPOSED CHANGE TO TECH SPECS, consist of Rev & add'l pages, tables, & figs to the FSAR.

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Carolina Power & Light Company

January 4, 1974

File: NG-3514

Serial: NG-74-10

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RECEIVED JAN 11 1974 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. Donald J. Skovholt
Assistant Director for Operating Reactors
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

50-261

Dear Mr. Skovholt:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
SAFETY INJECTION PUMP PERFORMANCE

In response to your request of November 28, 1973, Carolina Power & Light Company submits Operating Plant Change No. 1 to the H. B. Robinson FSAR. Incorporated in this proposed change to the FSAR are minimum performance characteristics for the safety injection pumps and the supporting safety analyses for the small break loss-of-coolant accident (LOCA).

The revised minimum performance curve corresponds to the measured pump characteristics determined during the refueling outage and reported in our letters of May 25, 1973, and September 7, 1973. This performance, in combination with the measured Safety Injection System resistance, results in an integrated flow delivery that is conservative with respect to that used in the safety analyses reported to you in our letters of January 25, 1971, and December 8, 1971.

The section on small break LOCA in the FSAR has been revised to reflect the analysis provided to you in the December 8, 1971, letter. In addition, a sensitivity analysis has been incorporated which shows the effect of further pump performance degradation and fuel densification on the peak clad temperatures which occur during the small breaks of interest.

Page changes to incorporate the latest steamline break analyses have not been provided with this submittal. They will be provided as a part of the uprating amendment which will be submitted later this month. As stated in our letter of September 7, 1973, the analysis presented in WCAP-8114 uses a system delivery curve that is already 8% conservative with respect to measured delivery rates, and the analysis results allow an additional 7% degradation before a return to criticality or a DNB ratio less than

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1.30 would be experienced. In the analysis presented in WCAP-8243, for 2300 MWt operation, the same delivery rate was assumed. The delivery curve used in previous analyses of the steam break accident and the curve referred to above are incorporated in revised Figure 14.3.2-17 in the attachment and will be referenced in the uprating amendment.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:mvp
Attachments

cc: Messrs. N. B. Bessac
T. E. Bowman
B. J. Furr
B. Howell
D. V. Menscer
D. B. Waters

CAROLINA POWER & LIGHT COMPANY
H. B. ROBINSON STEAM ELECTRIC PLANT
UNIT 2

FINAL SAFETY ANALYSIS REPORT
OPERATING PLANT CHANGE NO. 1

FILING INSTRUCTIONS

Insert the Operating Plant Change No. 1 Transmittal Letter in the front of Volume 1.

Pages to be Removed		New Pages to be Inserted	
Page Number	Date	Page Number	Date
Section 6		Section 6	
6.2-18		6.2-18	January, 1974
		6.2-18a	January, 1974
Figure 6.2-7		Figure 6.2-7	January, 1974
Section 14		Section 14	
14-xi		14-xi	January, 1974
14-xii		14-xii	January, 1974
14.3.2-13		14.3.2-13	January, 1974
14.3.2-21		14.3.2-21	January, 1974
14.3.2-22		14.3.2-22	January, 1974
14.3.2-23		14.3.2-23	January, 1974
14.3.2-24		14.3.2-24	January, 1974
14.3.2-26		14.3.2-26	January, 1974
Figure 14.3.2-17		Figure 14.3.2-17	January, 1974
Figure 14.3.2-18		Figure 14.3.2-18	January, 1974
Figure 14.3.2-19			
Figure 14.3.2-20			
Figure 14.3.2-21			
Figure 14.3.2-22		Figure 14.3.2-22	January, 1974
Figure 14.3.2-23		Figure 14.3.2-23	January, 1974
Figure 14.3.2-24		Figure 14.3.2-24	January, 1974
Figure 14.3.2-25		Figure 14.3.2-25	January, 1974
Figure 14.3.2-26			
Figure 14.3.2-27			
Figure 14.3.2-28			

Design parameters are given in Table 6.2-5.

Refueling Water Storage Tank

In addition to its usual duty to supply borated water to the refueling canal for refueling operations, this tank provides borated water to the safety injection pumps, the residual heat removal pumps and the containment spray pumps for the loss-of-coolant accident. During plant operation it is aligned to the suction of the pumps. It is constructed of stainless steel.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling canal and a minimum of 300,000 gallons is available for delivery. This capacity provides an amount of borated water to assure:

a) A volume sufficient to refill the reactor vessel above the nozzles

b) The volume of borated refueling water needed to increase the concentration of initially spilled water to a point that assures no return to criticality with the reactor at cold shutdown and all control rods, except the most reactive RCC assembly, inserted into the core

c) A sufficient volume of water on the floor to permit the initiation of recirculation.

The water in the tank is borated to a concentration which assures reactor shutdown by at least 10% $\delta k/k$ when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration is approximately 1.4 weight percent boric acid. At $32^{\circ}F$ the solubility limit of boric acid is 2.2%. Therefore the concentration of boric acid in the refueling water storage tank is well below the solubility limit at $32^{\circ}F$.

Two level indications with low level alarms are provided.

A dynamic response analysis similar to that performed for the Containment Structure has been performed to determine the horizontal loads to be applied to this tank for the hypothetical earthquake. Vertical Seismic Loads equal to 0.133g have been applied simultaneously. Wave generation in the tank has been taken into account. A membrane stress analysis of the vertical cylindrical tank was performed considering the discontinuities at the base and top.

The allowable stress criteria are 95% yield for tension, 90% for compression and shear.

The design parameters are given in Table 6.2-6.

Safety Injection Pumps

The three high-head safety injection pumps for supplying borated water to the Reactor Coolant System are horizontal centrifugal pumps driven by electrical motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. The design parameters are presented in Table 6.2-7 and Figure 6.2-7 gives the performance characteristics of these pumps.

The performance characteristic in Figure 6.2-7 reflects the measured performance of the pumps as determined during the Cycle 1 - Cycle 2 refueling outage. The measured pump performance in combination with the measured system resistance results in a system delivery rate that is conservative with respect to those used in the steam break and the small break LOCA safety analyses, which are shown in Figure 14.3.2-17.

The two residual heat removal (low head) pumps of the Auxiliary Coolant System are used to inject borated water at low pressure to the Reactor Coolant System. They are also used to recirculate fluid from the containment sump and send it back to the reactor, the suction of the spray pumps or to suction of the high head safety injection pumps. These pumps are of the in-line centrifugal type, driven by electric motors. Parts of the pumps which contact

Operating Plant Change No. 1
January, 1974

SAFETY INJECTION PUMP MINIMUM PERFORMANCE CHARACTERISTICS

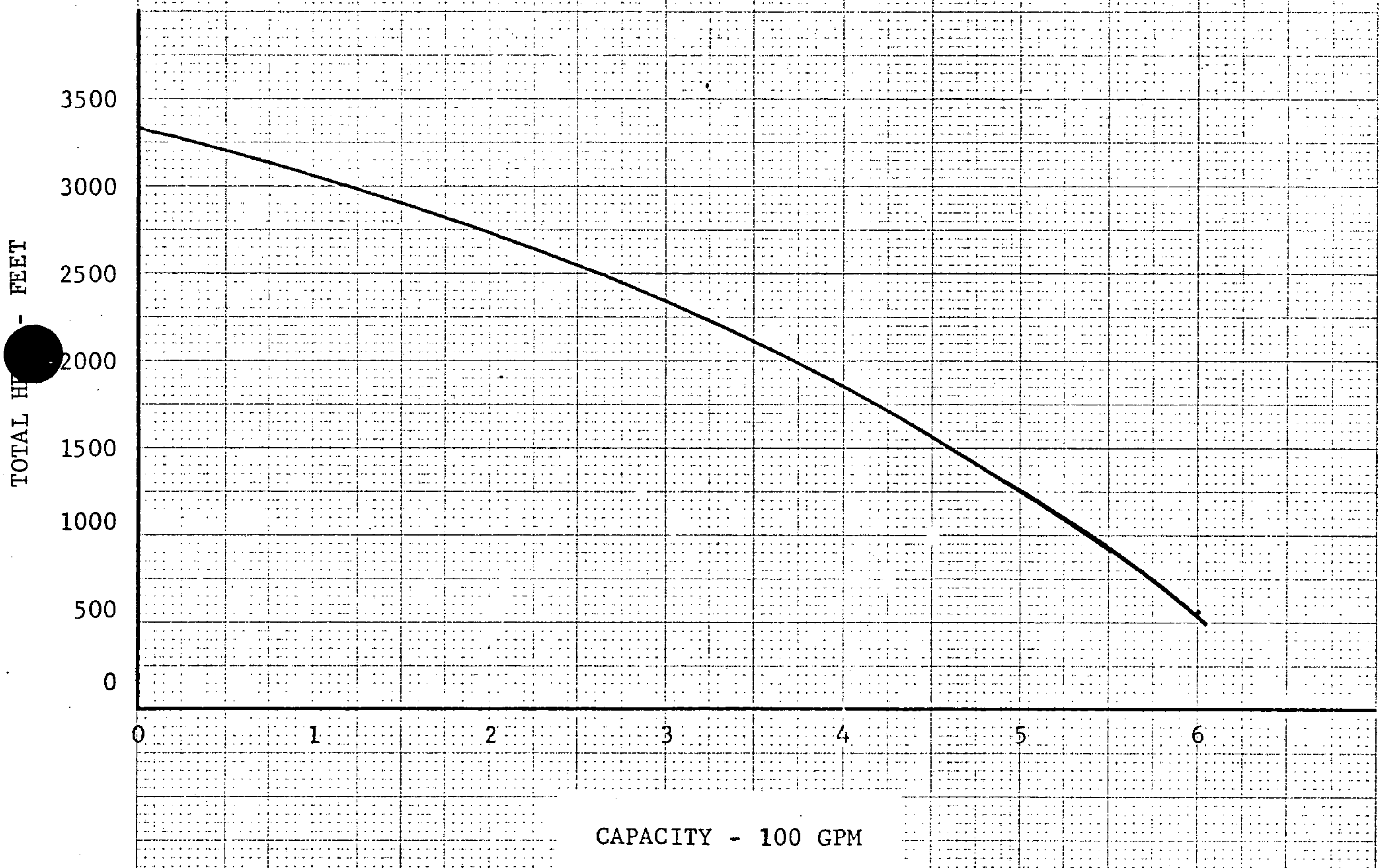


Figure 6.2-7

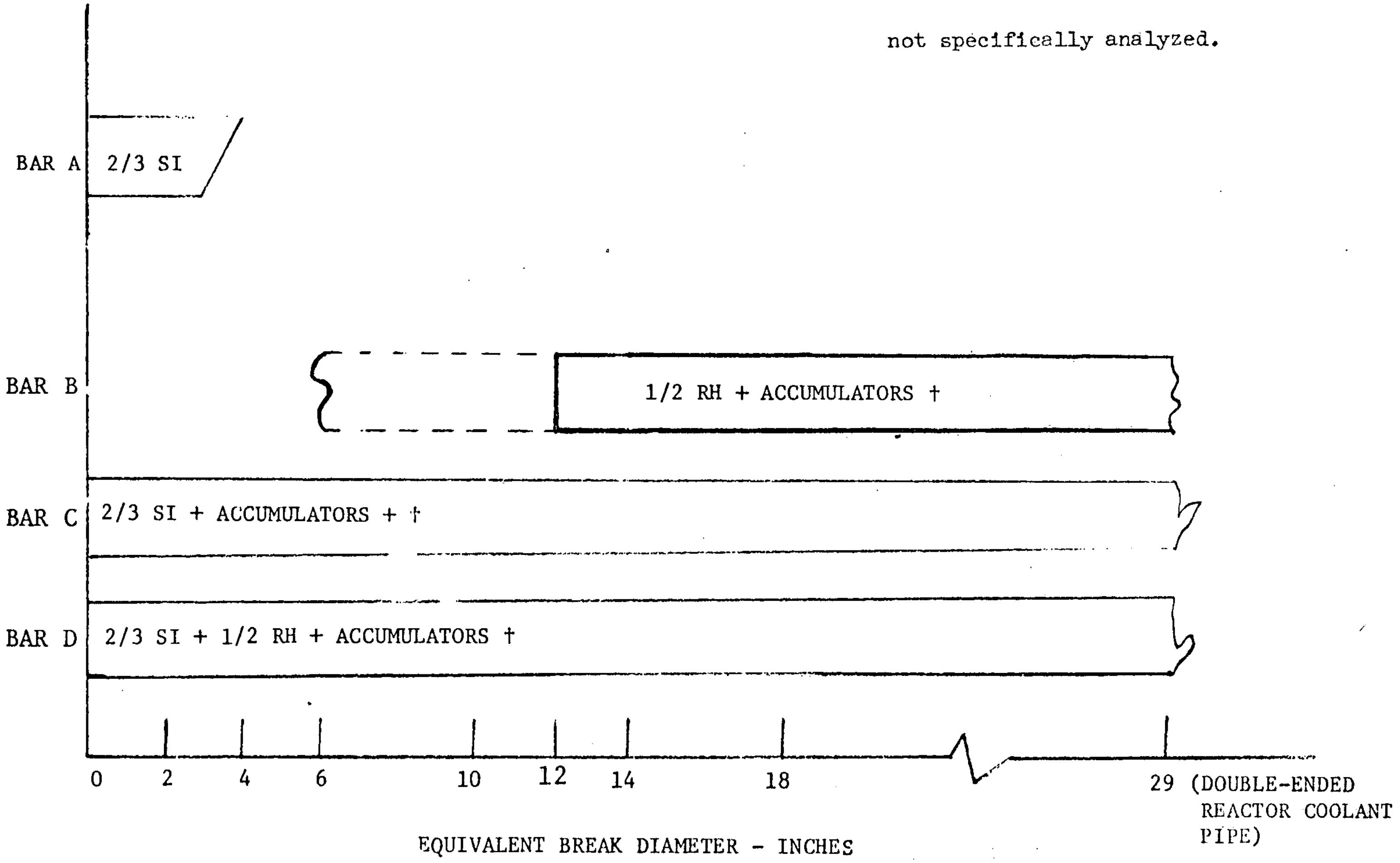
2/3 SI = TWO OF THREE SAFETY INJECTION PUMPS

1/2 RH = ONE OF TWO RESIDUAL HEAT REMOVAL PUMPS

CORE PROTECTION

Solid bar indicates capacity to meet core cooling criterion of no clad melting.

Dashed lines indicate expected performance not specifically analyzed.



RANGE OF PROTECTION BY SAFETY INJECTION SYSTEM
FIGURE 6.2-8

NOTE: FOR ALL CASES ONE OF TWO RECIRCULATION PUMPS REQUIRED FOR RECIRCULATION

† NO CREDIT IS TAKEN FOR THE ACCUMULATOR WHICH IS ATTACHED TO THE RUPTURED LEG IN THE CASE OF A COLD LEGBREAK

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

The code follows the pressure and mass in each volume as a function of time.

Conservatism in the Core Cooling Analysis

Some conservatisms which are inherent in the analytical models just presented are:

a) DNB is assumed to occur at 0.5 seconds for all breaks. This assumption is felt to be especially conservative for the smaller breaks where the flows remain high during the initial blowdown period.

b) When DNB occurs, it is assumed that the fuel rods can develop a condition of stable film boiling. No credit is taken for higher transition boiling coefficients that exist prior to the establishing of a stable film in the fuel rods. Conditions could exist by using a transition boiling model where a return to the nucleate boiling region would occur rather than entering stable film boiling.

c) The times the core becomes uncovered and recovered are calculated by the FLASH R code. Tests have verified that FLASH R underpredicts the amount of water remaining in the vessel during blowdown. A more realistic blowdown model would show that the core is uncovered for a shorter time period than that calculated in the above mentioned transients.

d) For the small breaks when long periods of blowdown exist the present analyses do not consider natural circulation in the core, which may result in significantly lower cladding temperatures.

Results

The capability of the Emergency Core Cooling System to meet the design criterion is analyzed for the following range of break sizes and location:

1. Large breaks, cold leg
   a) Double ended severance of the Reactor Coolant Pipe
   b) 6 ft^2
   c) 3 ft^2, and
   d) .5 ft^2

2. Small breaks, cold leg (SLAP)
   a) 6 inch
   b) 4 inch
   c) 3.5 inch
   d) 3 inch

For all of the above breaks the clad temperature transient is presented for the case where the contents of one accumulator tank was assumed spilled through the break. For hot leg breaks all of the accumulators empty into the reactor vessel. The above list of cold leg breaks result in more severe core temperature transients than the equivalent hot leg breaks. Thus the detailed analysis of hot leg breaks is not presented. Full flow from the safety injection pumps was assumed at 25 seconds.

Results - Large Area Ruptures

The power level used in the loss of coolant evaluations performed for the reactor includes a 2% increase above the maximum calculated core thermal rating of 2292 MWt to account for errors in the steam cycle calorimetric measurements.

Blowdown and Refill

Figures 14.3.2-1 to 14.3.2-4 are plots of the water volume in the reactor vessel for the large area ruptures. During blowdown, the volumes plotted represent an equivalent liquid volume which would result if the liquid and gas phases were completely separated. No credit is taken for an increased froth height due to voids created by boiling in the core. The volume of

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CASE	KNIGHT	BALLARD	LEE (L)	PLANS
GIAMBUSSO	PAWLICKI	SPANGLER	MAIGRET (L)	MCDONALD
BOYD	SHAO		SERVICE (L)	✓ DUBE w/Input
MOORE (L)(BWR)	STELLO	ENVIRO	SHEPPARD (E)	INFO
DEYOUNG(L)(PWR)	HOUSTON	MULLER	SMITH (L)	C. MILES
✓ SKOVHOLT (L)	NOVAK	DICKER	✓ TEETS (L)	B. KING
P. COLLINS	ROSS	KNIGHTON	WADE (E)	✓ A. Cabell
DENISE	IPPOLITO	YOUNGBLOOD	WILLIAMS (E)	
✓ REG OPR	TEDESCO	REGAN	WILSON (L)	
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Clad Perforation Model

Calculations are performed to determine the number of fuel rods that might fail during the thermal transient following a rupture in the primary cooling systems. In this analysis, fuel rods are considered to fail when the differences between the internal and external pressure exceeds the rod burst pressure.

The calculations are performed in the following manner:

A. The maximum clad temperature vs. time transients on the rods in the core are calculated assuming no change in the core geometry.

B. For each radial region of the core, a burst pressure vs. time curve is obtained by combining the temperature transient curve and the burst pressure vs. temperature curve.

C. The hot fuel volumes and the hot clad volumes obtained in the fuel
3 rod transient study are used to determine the hot void volume in the fuel rod as a function of time. The internal gas pressure distribution as a function of time is calculated considering the actual fuel rod power histories at the end of the equilibrium cycle when the maximum internal pressures are expected to exist.

D. All rods are assumed to fail if at any time during the transient the difference between internal gas pressure and external system pressure exceeds the burst pressure of the clad.

E. An evaluation is then performed to determine the rod with the lowest power rating (kw/ft) which fails. All rods above this power level then are considered as exhibiting rod bursting.

Results of the rod burst evaluation is presented in the table on page 14.3.2-19.

Results - Small Breaks

The analysis carried out and presented in the previous section demonstrated the adequacy of the accumulators to terminate core exposure and limit the temperature rise of the core for large area ruptures. For smaller breaks the discharge of fluid through the hole is less severe and for small enough breaks the high head safety injection pump is capable of maintaining flooding of the core hot spot for the entire blowdown. Where the hot spot remains covered no clad damage is expected.

Rupture of very small cross sections (up to about the equivalent of a 3/4" connecting pipe) will cause expulsion of coolant at a rate which can be accomodated by two of the three charging pumps well before the core is uncovered. Since instrument taps and sample connections are less than 3/4" diameter protection from rupture of this line is afforded by the charging pumps.

For smaller leaks, (up to about 1/2") these pumps would maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown. It should be noted that the safety injection pumps also provide protection for these small ruptures.

Should a larger break occur, resultant loss of pressure and pressurizer liquid level will cause reactor trip and initiation of safety injection supplementing the charging flow.

Using the SLAP code, break sizes of 3, 3.5, 4, and 6 inch equivalent diameters were re-analyzed.[21,22] This reanalysis is the latest of several performed prior to and during plant startup to justify continued operation based on measured safety injection pump and system performances.[16,17,18]

The analyses are based on hot leg injection being blocked,[19,20] with 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 14.3.2-17. The pump discharge pressures indicated include the 5% reduction used in these analyses.

The Reactor Coolant System pressure and volume for the range of break sizes are esented in Figure 14.3.2-18 and Figures 14.3.2-22 through 14.3.2-25, respectively. The volume figures illustrate both quiet and froth levels.

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.

Safety injection pump and system performance was again measured and reported in May, 1973[23]. The measured performance was the same as used in the small break analysis for reactor backpressure less than 900 PSIG. At higher backpressures, the measured performance was somewhat better than that used in the analysis. In order to access the effect of possible pump wear, the system performance was arbitrarily degraded by reducing both extremes on the pump performance curve (1) shutoff pressure and 2) runout flow by 5% each and constructing a curve of similar shape (to the original curve) through hose points. The resulting curve (Figure 14.3.2-17) shows a reduction of approximately 10% in "Flow to Reactor" at 1000 PSIG backpressure.

A series of calculations have been recently performed to determine the sensitivity of various pertinent parameters to typical three loop plant small break analysis results. One of the parameters studied was high head safety injection flow and indicates that a 10% reduction in flow, for H. B. Robinson, would result in an increase of approximately 300°F in peak clad temperature calculated during the small break loss of coolant accident.

To account for fuel densification in a small break loss of coolant accident analysis only the axial densification, which could result in a local power spike, must be considered. This local power spike is conservatively assumed to occur at the core elevation which has the highest calculated clad temperature during a LOCA. That elevation is generally near the top of the core. Additional calculations were made, again for a typical three loop plant, to determine the increase in peak clad temperature due to the local power spikes. The increase in peak clad temperature for the limiting break was approximately 4°F per ercent increase in local (hot spot) power.

The resulting peak clad temperature considering fuel clad collapse and reduced safety injection pump delivery is less than 1800°F.

The existence of a water filled loop seal was considered in the transient. That is, the plot of the water level in the core takes into account the depression of the core water level necessary to maintain a full downcomer and loop seal. This depicts a break for the worst break location, i.e., a cold leg break between the pump outlet and the reactor vessel inlet.

Therefore, from the results of analyses it is concluded that a break size of about 2 inches defines the upper limit of protection afforded by two high head safety injection pumps, considering minimum injection capability.

In the previous cases no credit was taken for operator action. Since time is available in a small break accident, it is expected that the operator will take control of the accident. By dumping steam through the steam generator relief valves the Reactor Coolant System can be depressurized. This depressurization of the Reactor Coolant System would result in less discharge through the break and greater addition from the Safety Injection System. The net result is a greater capability to maintain core flooding.

The action the operator would perform for this accident would be very similar to a normal cooldown. In a blackout situation the atmospheric dump valves are used, and when power is available the condenser dump would be used.

Conclusion

For breaks up to and including the double-ended severance of a reactor coolant pipe, the Safety Injection System with partial effectiveness will prevent clad melting and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved. The final core cooling systems design meets the core cooling criteria with substantial margin for all cases.

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22. CP&L Letter of March 20, 1972.

23. CP&L Letter of May 25, 1973.

24. CP&L Letter of September 7, 1973, et al.

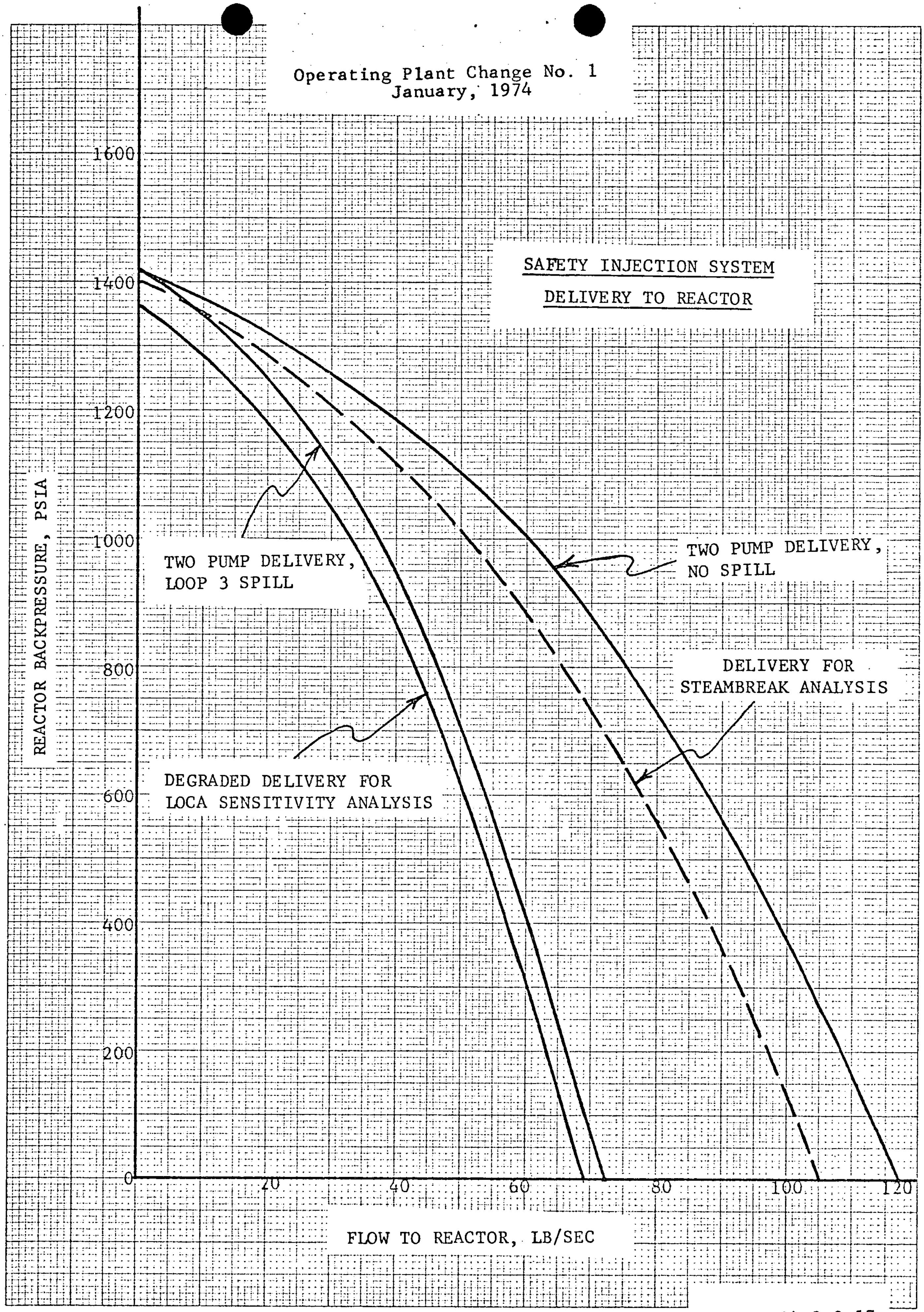
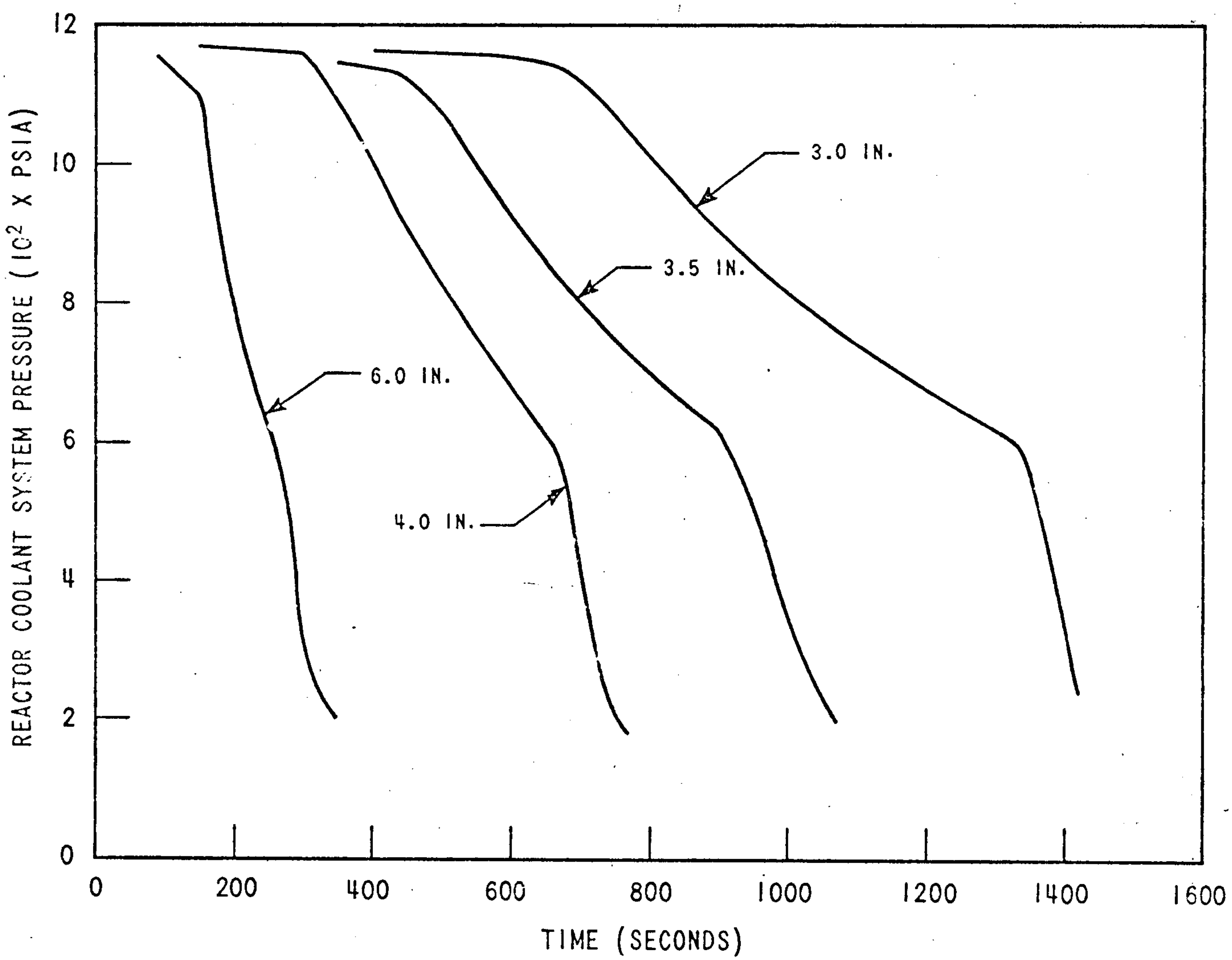


Figure 14.3.2-17



CPL Small Break LOCA Analyses Pressure History
Min. Safety Injection

Figure 14.3.2 - 18

Figures 14.3.2-19 thru 14.3.2-21
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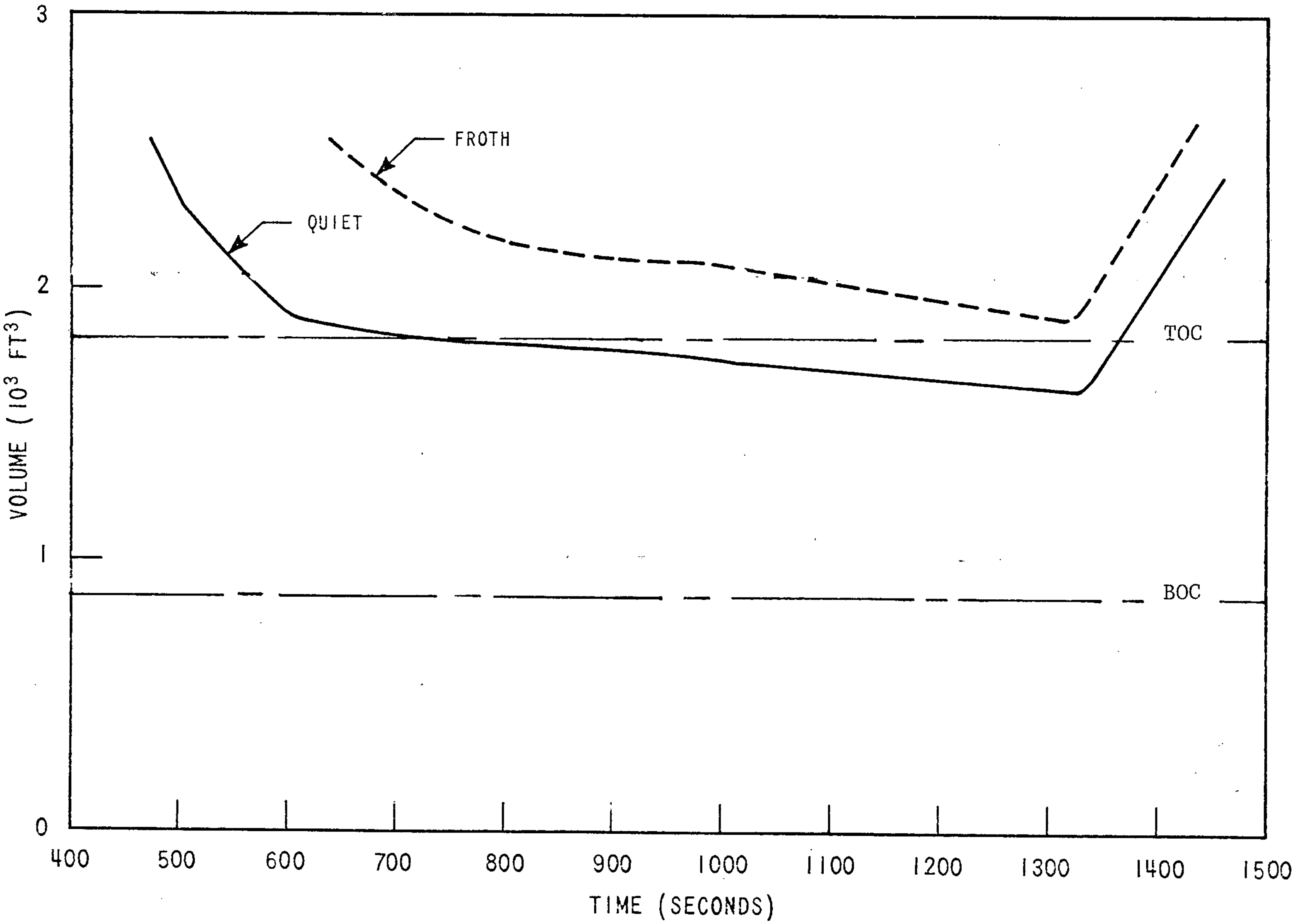
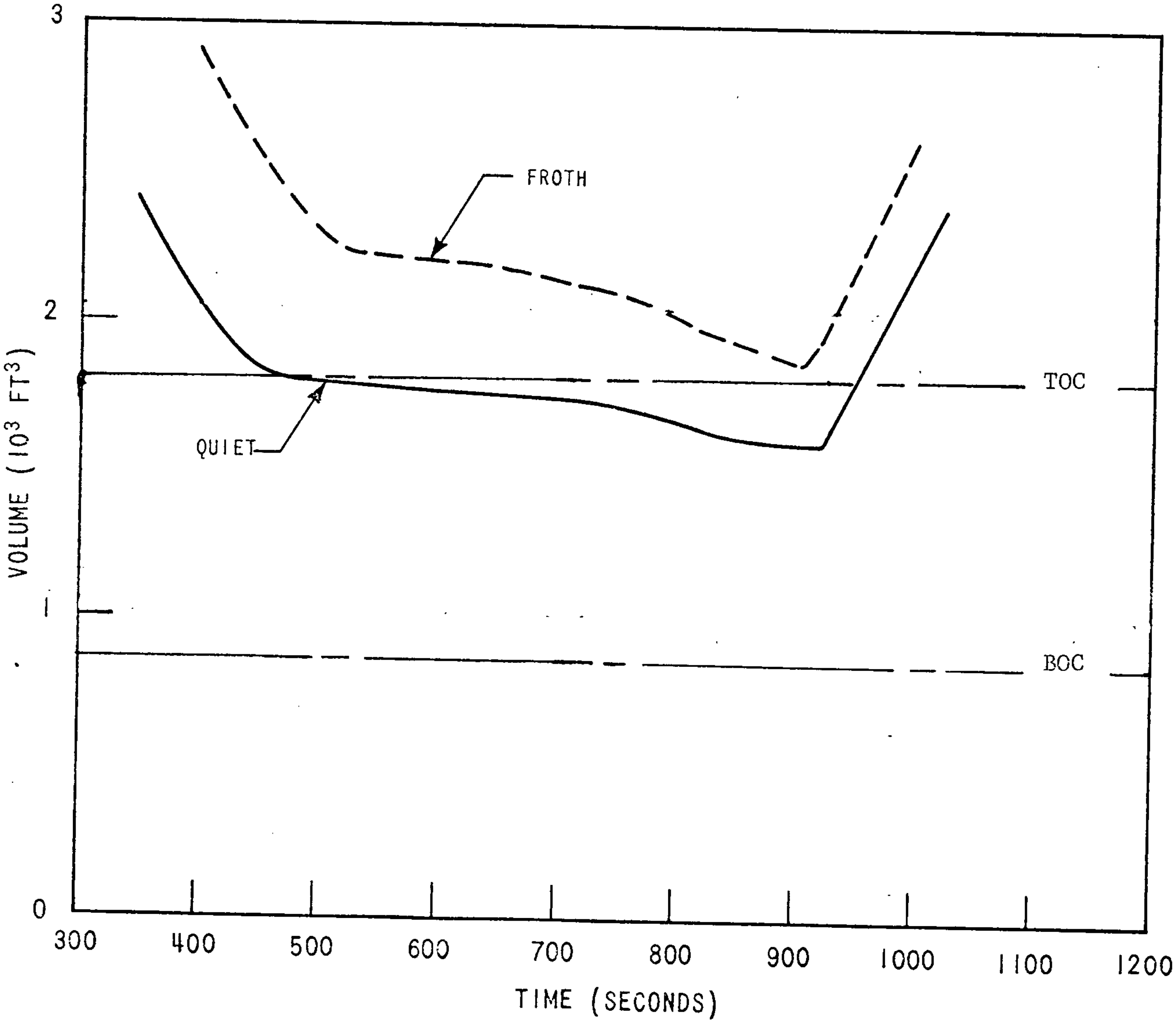
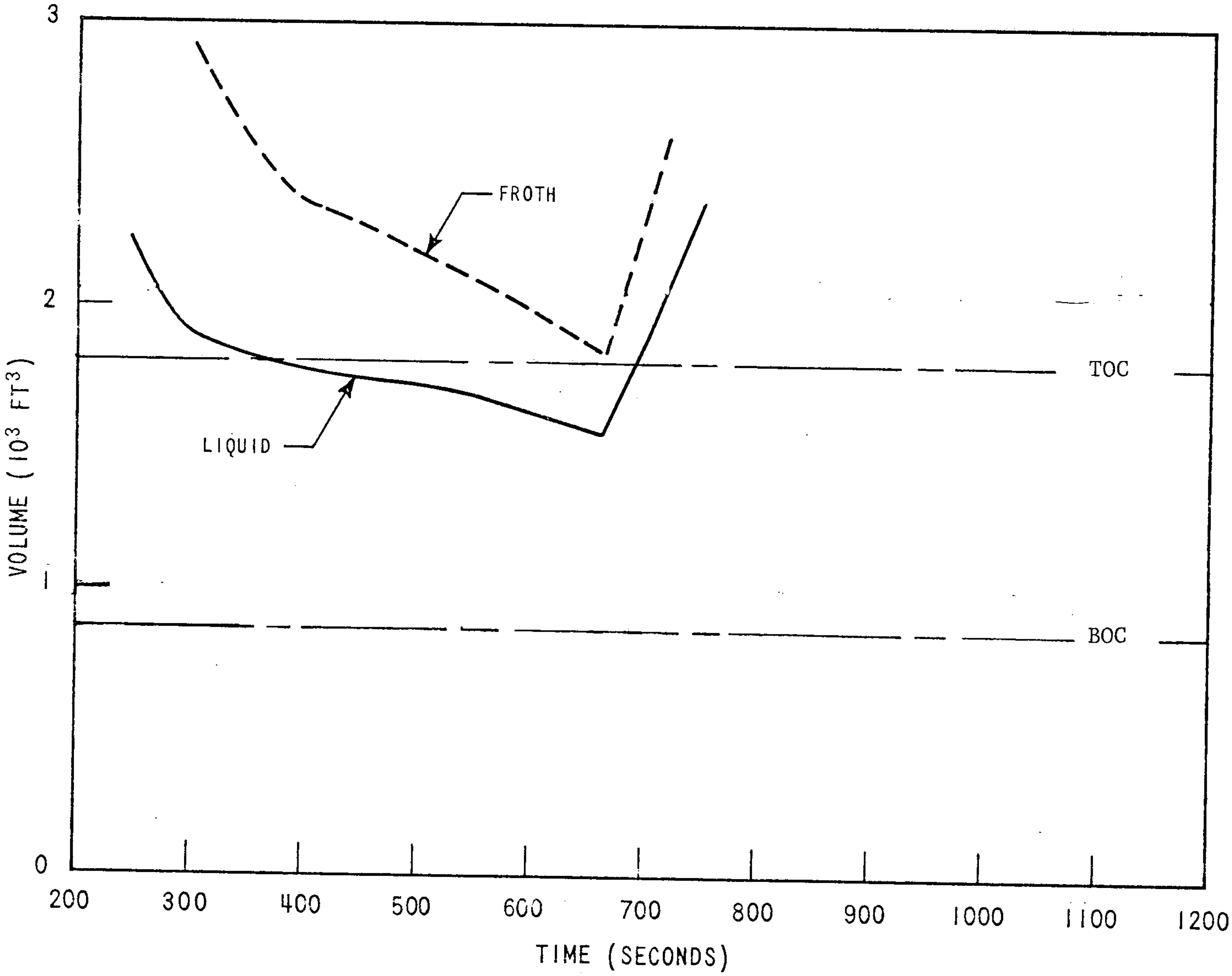


CPL Small Break LOCA Analyses Volume History
3.0 In. Dia. Break Minimum Safety Injection

Figure 14.3.2 - 22

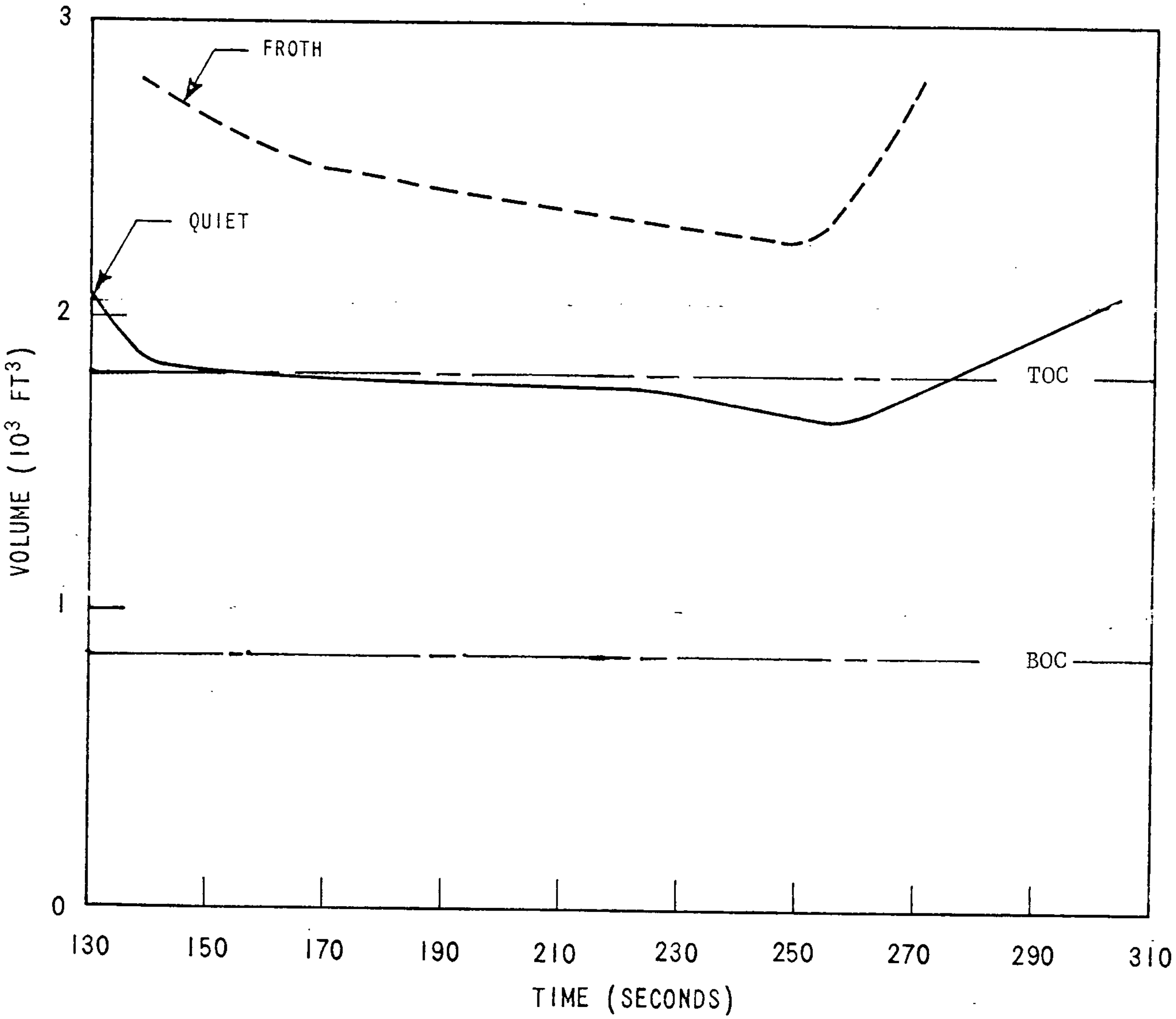


CPL Small Break LOCA Analyses Volume History
3.5 in. Dia. Break Minimum Safety Injection



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

Figure 14.3.2 - 24



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

Figure 14.3.2 - 25

Figures 14.3.2-26 thru 14.3.2-28
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