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Decay Heat Removal Problem Associated With
Recovery From a Very Small Break LOCA
for CE System 80 PWR

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Nuclear Systems Analysis Section
May, 1977

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Roisman 79-98 7907160249

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2. Physical Arrangement and Characteristics

The physical arrangement of the reactor coolant system for a typical System 80 plant is shown in Figure 1. The plant has two steam generators each with a single hot leg piping connection (42 inch) and two cold leg piping connections (30 inch) with a reactor coolant pump installed in each cold leg. The pressurizer is attached to one of the hot leg pipes by an 8 foot long surge pipe (12 inch). The reactor vessel has two corresponding hot leg piping nozzles and four cold leg nozzles.

The plant elevations corresponding to various points in the reactor coolant system are given in Table 1. Other reactor coolant system characteristics of interest are given in Table 2.

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3. Predicted Post-LOCA Modes of Decay Heat Removal

The conditions predicted to be associated with each quasi-steady-state post-LOCA mode of decay heat removal are detailed in Appendix A. These conditions are for a very small break (≤ 0.1 ft^2) for which the steam generators must remove a significant portion of the decay heat; otherwise, reactor coolant system repressurization occurs since the break is too small to accommodate the transport of all decay heat to the environs. Modes 1-6 are based on loss of off-site power, no operator actions, minimum core cooling (one train) response using cold leg injection, closure of main steam isolation valves, and steam generators at saturation conditions corresponding to the lowest safety valve set point.

Modes 1-7 given in Appendix A define

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the conditions which are thought to exist in the reactor coolant system to provide decay heat removal during the relatively slow changing post-LOCA period. These conditions are summarized in Table 3.

The achievement of Modes 1-3 appears straightforward although there might be some concern if the transition to core boiling is not accompanied by a relatively rapid drainage of the steam generator tubes. If an adequate condensing surface is not exposed quickly, excessive steam bubbling might occur within the tubes.

Operation in Mode 4 appears reasonable to achieve although the reactor operator will be unaware as to what is happening to reactor vessel level.

Mode 5 is thought to be the most

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uncertain operation which will take place. The onset of Mode 5 cannot be predicted or controlled since there is no level information available to the operator. The flooding of the hot leg pipes and steam generator inlets will disrupt normal steam flow from the core to the steam generators and will lead to an intermittent voiding of the hot leg piping as the steam is forced through. It appears reasonable that adequate decay heat transport might be achieved during this transition period although there should be some concern about the mechanical effects of slug flow on steam generator tubes.

In order to re-establish natural circulation decay heat removal which is Mode 6, it is necessary that Mode 5 terminate by filling the steam generator tubes with liquid and

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establishing the density gradients required for natural circulation. It is not clear how this can be accomplished while maintaining an adequate heat transport rate from the core region through the hot leg piping and inlet side tubing. There could be a problem with a persisting slug flow from the reactor vessel and significant accumulations of non-condensable gases in the steam generator U-tubes. If enough non-condensable gases have accumulated, natural circulation cannot be re-established (unless the gases can be removed by some means).

If Mode 6 cannot be established, it will be necessary to wait until the decay heat level is sufficiently low to provide prolonged core cooling by a rapid filling (subcooling) of the reactor vessel and establishment of shutdown

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Cooling System (SCS) operation before core boiling is re-initiated. This variation of Mode 7 might be difficult to achieve unless the reactor vessel water level is sufficiently low to permit a large cold water addition to the core region. The proposed maneuver may be difficult to achieve without a reactor vessel level indication to guide the control of water addition during the waiting and reflood periods.

4. Available Quantative Results

Information concerning a Combustion Engineering analysis for the small break LOCA for a System 80 plant is included as Appendix B. This analysis is for cold leg breaks which are 0.1, 0.05, and 0.005 square feet in size (about 4, 3, and 1 inch diameter: respectively). The analysis assumes loss of off-site power, the worst single

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failure of emergency equipment, and no operator action. The steam generators are assumed to remain at the saturation pressure corresponding to the lowest setpoint of the safety valves (1285 psia).

The results shown in Figure 1 of Appendix B indicate that the reactor vessel level will drop into the hot leg piping for each assumed break size within 15 to 30 minutes following the break, thus leading to the predicted occurrence of the first six modes of operation outlined in Appendix A. The results show that the core will remain covered at all times including Mode 4 operation.

Some correlation can be recognized between the curves in Appendix B and the modes of operation predicted in Appendix A.

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5. Area of Concern

The transition from core boiling to natural circulation which is predicted in Appendix A (Mod 5) to occur during reactor vessel level recovery following a very small break will require that the steam generator U-tubes become refilled. It is not clear how this can be accomplished, particularly if a significant amount of non-condensable gas has accumulated in the tubes. If the refilling is not accomplished, plant recovery from a very small break LOCA might be unacceptably delayed pending additional fuel cooldown.

Adding to this concern is the uncertainty associated with unknown vessel level, the adequacy of emergency operating instructions and training for this event, and the consequences of the unstable slug flow conditions which are predicted to develop as

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the reactor level recovers.

This operational difficulty appears to be generic for pressurized water reactors although it may be aggravated somewhat in the CE System 80 because of the lower head and capacity of the high pressure injection pumps. In PWR systems designed by other vendors, higher head pumps with greater flow capacity at a given pressure assure that for a greater range of very small breaks the reactor vessel level will not drop below the hot leg nozzles and thereby disrupt natural circulation.

Additional Considerations

This qualitative consideration of recovery from [v]ery small break LOCA did not include the [prob]lem of assuring a continuous long-term [sourc]e of clean auxiliary feedwater for the ste-

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generator. This difficulty might materialize if the recovery should be delayed pending additional fuel cooldown.

It should also be recognized that long-term recovery could involve recirculation of water from the primary containment sump if makeup water from the refueling water tank is no longer available or water accumulations inside of containment become limiting. The likelyhood of requiring recirculation will increase if a delay for additional cooldown is experienced. This recirculation mode of operation with the reactor at high pressure is not an established design requirement and has not been evaluated for feasibility.

Additional complications appear likely if it is required to use the high pressure pumps for reactor makeup from the sump while

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operating the Shutdown Cooling System (SCS) since it involves some common piping including a common return pipe. This mode of operation has also not been a design requirement. In addition, the high pressure pumps do not have minimum flow protection while taking suction from the sump. If the flow rate to the reactor is very low, pump damage could be a consideration.

The effect on primary containment of very small breaks was not considered. A low-low pressurizer pressure will appear early in the event and initiate containment isolation. A subsequent high-high containment pressure will initiate containment spray. The operator may not be able (or desire) to intercept the automatic spray. A prolonged isolation due to delay for additional cooldown will increase

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the likelyhood of containment spray.

The effect of the safety injection tank starting to dump at about 610 psia was not evaluated. Unless the nitrogen pressure is reduced or the water level is lowered sufficiently, a dump will occur automatically. The tanks can be isolated manually below 415 psia but they are programmed to reopen if the pressure exceeds 500 psia.

A loss of offsite power was assumed in preparing this report. If the very small break should not be accompanied by an offsite power loss, the effect of continued operation of the reactor coolant pumps would need to be evaluated. The loss of system overpressure at a high average reactor coolant temperature might lead to unacceptable pump or seal damage if the operator does not manually

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trip the pumps within a few minutes. Physical damage might be induced by excessive vibration or a severe flow instability. Even if offsite power is lost, the operator must take steps within about one-half hour to re-establish seal cooling by loading the CVCS pumps on the diesel generators. If not done, an additional loss of coolant will develop through the pump seals. The consequence of this scenario was not developed.

The very small break postulated for this report was assumed to be located at the top of a cold leg pipe. Other break locations were not considered in detail. However, the break location is not thought to be a major influence on the existence of the predicted operating modes given in Appendix A, i.e., there exists one or more break sizes and locations which will result in the occurrence of each predicted mode.

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Table 1

Reactor Coolant System Elevations

Component		Elevation (ft)*
Pressurizer:	Vessel High Point (inside)	570
	Vessel Low Point (inside)	529
	Hot Leg Connection	512
	Full Power Water Level	554
	After Scram Water Level	544
	Top of Heaters	539
	Horizontal Surge Pipe (center)	519
Steam Generator:	Highest Tube (center)	551
	Lowest Tube (center)	543
	Full Power Water Level	556
	Tubesheet (face)	517
Reactor Vessel:	Vessel High Point (inside)	530
	Hot Leg Nozzle (high point)	512
	Hot Leg Nozzle (low point)	508.5
	Active Fuel (high point)	505
	Active Fuel (low point)	492.5

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* Approximate values.

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Table 2

Reactor Coolant System Characteristics

Full Power Temperatures (°F):	Core Entry	565
	Core Midplane	593
	Core Exit	621
	Pressurizer Liquid	653
	SG Steam	553
	SG Feedwater	450
Full Power Pressures (psia):	RC Pump Discharge	2319
	Core Midplane	2288
	Core Exit	2260
	SG Steam	1070
Safety Valve Settings (psia):	Pressurizer	2500
	Steam Generator	1285
Volumes (ft^3):	Pressurizer (steam space)	900
	Pressurizer (above heaters)	1500
	Pressurizer (total)	1800
	Reactor Vessel (above hot legs)	2700*
Actuation Pressures (psia):	Safety Injection Pumps	1600
	Safety Injection Tanks	610
	Reactor Scram	1750

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* Approximate value.

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Table 3

Post-LOCA Modes of Decay Heat Removal[a]

Mode	Condition	Press. Level	RV Level	System Press.	SG Tube Level
1	Natural Circulation[b]	R-Decr.	Full	R-Decr.	Full
2	Natural Circulation[c]	S-Incr.	S-Decr.[d]	Stable	Full
3	Transition	Drain	S-Decr.[e]	Stable	Drain
4	Core Boiling	Empty	S-Decr.[f]	Stable	Empty
5	Transition	Empty	S-Incr.[e]	S-Decr.	Refill
6	Natural Circulation[c]	Refill	S-Incr.	S.-Decr.	Full
7	Forced Circulation[g]	Full	Stable	S-Decr.	Full

a. Nomenclature: Press. Level is pressurizer level.
RV Level is reactor vessel level.
System Press. is system pressure.
SG Tube Level is steam generator tube level.
R-Decr. is rapidly decreasing.
S-Decr. is slowly decreasing.
S-Incr. is slowly increasing.
Stable is no significant changes.

b. With pressurizer pressure controlling.

c. With reactor vessel pressure controlling.

d. Reactor vessel steam bubble has formed.

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e. Reactor vessel level has reached top of hot leg nozzles.

f. Reactor vessel level will reach a minimum and start to increase.

g. Using Shutdown Cooling System.

(PROPOSED) YELLOW CREEK NUCLEAR PLANT
REACTOR COOLANT SYSTEM

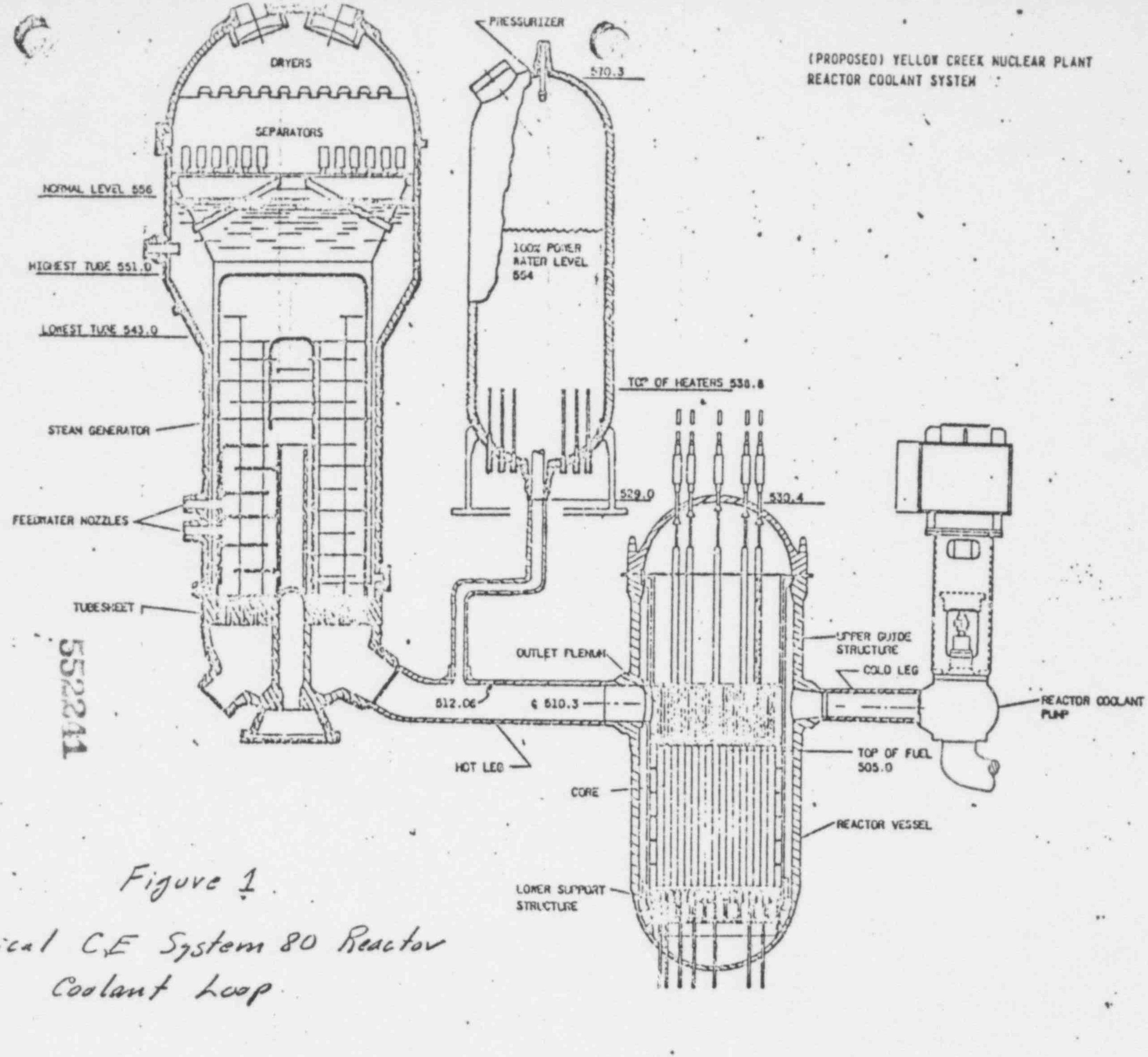

Figure 1

Typical C.E System 80 Reactor
Coolant Loop

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Appendix A

Conditions Associated With Post-LOCA Modes of Decay Heat Removal For Very Small Break LOCA *, 1

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* Numbered notes are included at end of appendix.

Mode 1[2]

Decay Heat Removal by Natural Circulation[3]
With Pressurizer Pressure Controlling

a. Pressurizer steam bubble pressure is above saturation pressure corresponding to reactor vessel top plenum effective liquid temperature and is superheated to pressurizer effective metal temperature.

b. Pressurizer level and pressure are decreasing rapidly (steam bubble expanding) due to fluid loss through the break in excess of makeup capability.

c. Pressurizer liquid is partially flashing to steam (thereby cooling) in an attempt to maintain constant pressure as the steam bubble expands.

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d. Pressurizer steam bubble is superheating to pressurizer effective metal temperature (thereby cooling walls).

e. Without an effective heat input other than from the pressurizer metal to compensate for the heat removed by flashing the pressurizer liquid and heat loss to the environs, the pressurizer pressure is decreasing rapidly as the steam bubble expands.

f. Pressurizer steam bubble may expand until superheated steam is voiding and condensing in the circulating liquid at pressurizer connection to the hot leg piping.

Mode 2 [2]

Decay Heat Removal by Natural Circulation [3]

With Reactor Vessel Pressure Controlling

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a. Steam bubble forms in reactor vessel plenum when pressurizer pressure decreases to below the saturation pressure corresponding to top plenum liquid temperature.

b. Reactor vessel level starts decreasing (top plenum steam bubble expanding) due to fluid loss through the break in excess of makeup capability.

c. Reactor vessel top plenum liquid is partially flashing to steam (thereby cooling) in an attempt to maintain constant pressure as the steam bubble expands.

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d. Decay heat is providing an effective heat input to compensate for the heat removed by flashing the top plenum liquid, thereby assuring a liquid temperature and corresponding steam bubble pressure which follows closely

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the slowly changing effects of decay heat reduction, heat loss to the environs, and heat transport through the break.

e. Pressurizer surge line is filled with semi-quiescent liquid which is subcooling to below the reactor vessel top plenum effective liquid temperature due to heat loss through the surge line in excess of heat gain by transport from the hot leg pipe.

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f. Pressurizer steam bubble pressure is in equilibrium with slowly decreasing reactor vessel steam bubble pressure and is most likely contracting (pressurizer level slowly increasing) as steam from the pressurizer bubble is condensed on the pressurizer wall at a rate which is commensurate

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with heat loss to the environs.

Mode 3[2]

Decay Heat Removal During Transition From Natural Circulation[3] to Core Boiling[4] Mode

a. Reactor vessel level reaches top of hot leg nozzles and starts to extend into hot leg pipes and steam generator inlets due to fluid loss through the break in excess of makeup capability.

b. Reactor vessel steam bubble enters pressurizer connection and causes a heatup and gravity drain of any water in the pressurizer.

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c. Reactor vessel steam bubble enters steam generator inlets and causes a gravity drain of water in the U-tubes back to the hot and cold leg piping.

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d. Natural circulation through the reactor core and steam generators is disrupted.

e. Water in the reactor core starts to boil agressively due to loss of natural circulation core cooling.

Mode 4[2]

Decay Heat Removal by Core Boiling[4]

a. Reactor vessel level continues to decrease due to fluid loss through the break in excess of makeup capability.

b. Reactor vessel steam bubble remains at saturation pressure corresponding to reactor core exit temperature and is fed directly by steam formed in the core at a rate which is commensurate with the decay heat input, steam generator condensation

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rate, and the volumetric expansion, and which follows closely the slowly changing effects of decay heat reduction, heat loss to the environs, and heat transport through the break.

c. Liquid from condensation inside steam generator tubes is returned to reactor vessel by gravity counterflow along the bottom of each partially filled hot leg pipe and by gravity flow to the pump loop seals and the cold leg pipes.

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d. Reactor core exit temperature and reactor vessel steam bubble pressure are decreasing in response to a decay heat reduction without a commensurate reduction in heat loss to the environs and heat transport through the break, and eventually in response to

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a reduction in steam generator pressure (and temperature) due to heat loss to the environs exceeding decay heat transferred after the safety valves are closed. (If operator action is considered, the steam generator pressure may be reduced by opening atmospheric dump valves and thereby reduce the steam generator and reactor coolant system temperatures).

e. Fluid loss through the break is decreasing and makeup rate is increasing due to reduction in reactor vessel pressure.

f. Reactor vessel level starts to increase as makeup capability exceeds fluid loss through the break.

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Mode 5[2]

Decay Heat Removal During Transition From Core Boiling[4] to Natural Circulation[3] Mode

a. Reactor vessel level continues to increase to top of hot leg nozzles as makeup capability exceeds fluid loss through break.

b. Steam generator inlets are flooded or severely choked by slug flow and entrainment thereby disrupting steam flow to the tubes.

c. Reactor core exit temperature and corresponding reactor vessel steam bubble pressure start to increase in response to disruption of steam flow to the tubes.

d. Reactor vessel level will decrease in response to the steam pressure increase, thereby releasing intermittent steam bubbles

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to the steam generator tubes.

e. Reactor vessel steam bubble pressure will be communicated somewhat equally to the water filled hot and cold leg pipes due to a liquid coupling through the gap between the core barrell and the hot leg nozzles and a steam coupling through the bypass holes around the vessel flange area.

f. Reactor vessel level decrease (pressure increase) will be accompanied by a compensatory water level rise (perhaps equalized) and steam pressure increase inside the steam generator tubes.

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g. If sufficient water is available in the reactor vessel top plenum and if steam in the tubes can be condensed before the next cycle of steam bubbles reaches the steam generator, the tubes might become completely filled with

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liquid and the density gradients established which are required for natural circulation.

h. If sufficient quantities of non-condensable gases (principally dissolved hydrogen and oxygen released during depressurization) have accumulated in the steam generator U-tubes due to prolonged sweeping of the core region and subsequent concentration during steam condensation at the top of the tubes, it will <u>not</u> be possible to refill the tubes and thereby establish natural circulation.

i. (insert page 33A)

<u>Mode 6</u>[2]

<u>Decay Heat Removal After Re-establishing Natural Circulation[3] With Reactor Vessel Pressure Controlling</u>

a. Natural circulation is <u>assumed</u> to be re-established with a reactor vessel level

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i. If natural circulation is not re-established, system pressure increases until it reaches a pressurizer safety valve set point and expells water (with probable intermittent slug or two-phase flow instability) until a level is established in the hot leg pipes; thereby re-establishing decay heat removal by condensing in the steam generators with a subsequent reduction in pressure as the system cools again. (Note that makeup flow cuts off before the safety valve set point is reached and that no remotely operable relief valve is available to reduce system pressure. Pressure reduction and re-establishment of makeup flow must be achieved by system cooldown using core boiling and the steam generators.)

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above the hot leg nozzles and with the steam generator tubes full of liquid.

b. Reactor vessel level continues to increase (top plenum steam bubble contracts) as makeup capability exceeds fluid loss through the break.

c. The rate of level increase will be commensurate with the reactor vessel steam bubble pressure which responds to changes in the decay heat input, heat loss to the environs, and heat transport to the steam generator and through the break.

d. Pressurizer surge line is filled with liquid and pressurizer contains a trapped steam bubble which is in pressure equilibrium with the reactor vessel steam bubble.

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e. Pressurizer steam bubble is contracting as steam is condensed on pressurizer walls at a rate which is commensurate with heat loss to environs.

f. Pressurizer will eventually fill with liquid unless limited by non-condensable gases as the pressurizer steam bubble condenses.

Mode 7[5]

Decay Heat Removal by Forced Circulation[6] Using Shutdown Cooling System

a. Natural circulation has been previously established with reactor vessel level above hot leg nozzles and the pressurizer completely filled with liquid.

b. Reactor vessel level has been stabilized by adjusting makeup capability to equal

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fluid loss through the break. (In the long term, operator action may be used to re-establish a pressurizer steam bubble and establish a stable pressurizer level.)

c. Shutdown Cooling System (SCS) is in operation to provide forced circulation decay heat removal from the reactor core.

d. Reactor vessel top plenum is partially filled with steam bubble at saturation pressure corresponding to top plenum liquid temperature (unless pressurizer pressure control has been established).

e. Reactor vessel pressure is below SCS design operating pressure and decreasing at a rate commensurate with decay heat input, heat loss to environs, heat transport through break, and heat removal by the SCS.

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Notes

1. A very small break LOCA (≤ 0.1 ft^2 for CE) is one for which the steam generators must remove a significant portion of the decay heat; otherwise, reactor coolant system repressurization occurs since the break is too small to facilitate the transport of all decay heat to the environs.

2. Assuming a very small break, reactor scram, loss of off-site power, no operator actions, minimum core cooling (one train) response using cold leg injection, closure of main steam isolation valves, and steam generators at saturation conditions corresponding to safety valve set points (as long as possible). The Safety Injection Tanks are assumed not to dump.

3. Decay heat removal is by natural circulation through the reactor core and steam generator tubes with the core exit temperature (also assumed to be reactor vessel top plenum effective liquid temperature) established by the decay heat rate and natural convection heat transfer and transport coefficients.

4. Decay heat removal is by boiling heat transfer in the core, saturated steam heat transport from the reactor vessel to steam generators, and condensing heat transfer inside the steam generator tubes.

5. Assuming a very small break, loss of off-site power, operator actions permissible where required, closure of main steam isolation and steam dump valves, and minimum core cooling (one train) response

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using SCS for heat removal and safety injection, and charging pumps for makeup.

6. Decay heat removal is by forced circulation through the reactor core and SCS heat exchangers using SCS pumps in the shutdown cooling mode of operation with the core exit temperature (also assumed to be the reactor vessel top plenum effective liquid temperature) established by the decay heat rate and forced convection heat transfer and transport coefficients.

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Appendix B

Small Break LOCA Analysis
by Combustion Engineering

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel 203/688-1911
Telex: 9-9297

L.D.

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POWER
SYSTEMS

~~N8M2-50-D302~~

September 18, 1975
TD-CE-161
9/24

Tennessee Valley Authority
204 Union Building
Knoxville, Tennessee 37902

Attention: Mr. D. R. Patterson
Chief Mechanical Engineer

Gentlemen:

Subject: Nuclear Steam Supply Systems
TVA X-24 and X-25 Installation
TVA Order No. 60-84840
CE Contracts 14074 and 14174

SMALL BREAK LOCA

Ref.(A): D.V. Graf to D.R. Patterson, TD-CE-117, 6/11/75, subject: Meeting Minutes of 5/27 & 5/28 in Windsor

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During the Emergency Feedwater Meeting of 5/27 and 5/28, 1975, C-E agreed to provide more information concerning our analysis of the subject accident. This commitment was documented in Item 23 of Reference (A). The attached report describes the results of our analysis and we expect that it should satisfy your concerns. If you desire more information on this subject, please contact Mr. F. Z. Stiteler in Knoxville.

SEP 19 1975 --DRP:AH

CC: WEIngle, 401 UB-K
RHHodges, 100 DB-K
SKesler, 407 FB-K
LMMontgomery, 401 AB-K
RMPierce, 515 LB-K (2)
HCRussell, 423 MIB-K
X21/22 Task Fo, 433 MIB-K
RLL/FZS;BB Task Fc, 507 NKT-K

Very truly yours,

R. L. Lumpkin, Jr. for
Project Manager

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Att.

cc: W. B. Wade, TVA, w/o att
J. Ginder, CE, w/o att
H. Brown, CE, w/o att

DRP:AH SEP 19 1975

MASTER FILE

N8-02-CA-D302

SMALL BREAK LOCA

Introduction

In a meeting held in Windsor on 5/27-5/28/75, Mr. Sabin of TVA expressed a concern for a class of small break LOCA's whose depressurization rates are relatively slow compared to those small breaks presented in CESSAR[1], Section 6.3.3.3. He worried about the possibility of inadequate HPSI flow for such breaks and also asked about the LOCA requirements for Emergency Feedwater and operator actions.

This transmittal constitutes a response to action item number 23 of Reference 2.

Summary

The results herein demonstrate that for any cold leg break less than or equal to 0.1 ft^2 in size, adequate liquid inventory is maintained in the reactor vessel to keep the active fuel completely covered at all times.

The results shown in CESSAR demonstrate that for those small breaks in which some degree of core uncovery is predicted, the ECCS criteria[3] are met and peak clad temperatures are less than 1200°F.

It has been assumed that for all breaks no operator actions are performed. At some later time following the LOCA, operator procedures for long term cooling will be implemented. This memo, however, does not address the long term cooling phase of the accident. Long term cooling performance and operator procedures will be discussed in a future submittal.

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Since NRC requires the assumption of loss-of-offsite power, main feedwater flow is not available. For breaks larger than 0.1 ft^2, those presented in CESSAR, Emergency Feedwater (EFW) was not represented and indeed was found to have little influence on the performance of such relatively large break sizes. However, for breaks equal to or smaller than 0.1 ft^2, EFW was found absolutely necessary. For these smaller breaks, the steam generators must remove a significant part of the decay energy generated in the core. If they do not, then primary side re-pressurization occurs since the break is too small to emit all of the steam boiled off in the core.

Discussion

Calculations of breaks 0.1 ft^2 and smaller have been performed with the CELEVEL code. This code was specifically written for application to such small breaks.

The predicted two-phase level transients for several breaks 0.1 ft^2 and smaller are shown in Figure 1. The corresponding pressure transients are given in Figure 2. These results demonstrate that although the depressurization rates for such small sized break sizes are gradual, adequate HPSI pump inflow is provided to keep the core covered at all times.

The Safety Injection Tanks (SIT) become activated at approximately 900 seconds for the 0.1 ft^2 case. Neither the .05 ft^2 nor the .005 ft^2 cases resulted in SIT activation before the level transient is turned around. For both breaks, the level transient bottoms out at an elevation above the top of the active fuel and recovers from there via HPSI inflow alone.

For all of these breaks, steam generator heat transfer plays an important role. Since EFW is supplied to each steam generator, an adequate secondary liquid inventory is maintained to support heat removal by condensation of the steam in the U-tubes on the primary side. The secondary

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inventory is assumed to be saturated at a temperature corresponding to the lower setpoint of the safety valves (1270 psia). The condensation film coefficient used was approximately 200 BTU/hr-ft^2-^{o}F. This is a conservative value which properly accounts for the influence of non-condensibles.

To clearly indicate the influence of EFW, Figure 3 has been included. Results are shown for the reactor vessel pressure and level transients for cases with and without emergency feedwater. As shown, if EFW is not available a re-pressurization occurs which in turn reduces HPSI inflow and the level transient is significantly aggravated. Neither case shown in Figure 3 should be construed to specifically represent the performance of System 80. EFW is available on all System 80 plants. The case without EFW is shown only for demonstrative purposes. For both cases shown, the HPSI capacities assumed were much less than those of System 80. Thus, the level transient are more severe.

Major Assumptions

As required by NRC[3], the following conservative assumptions were applied in analyzing all small breaks ($\leq$ 1.3 ft^2):

1. Loss of offsite power assumed upon a scram signal.
2. The worst single failure of emergency equipment was assumed (failure of one emergency diesel generator to start).
3. An uncertainty margin of +20% was added to the calculated decay heat generated in the reactor.
4. Conservative setpoints and delays were assumed for scram, SIAS and equipment startup.

Assumption (1) results in a description of the accident in which the reactor coolant pumps begin coastdown, the main steam isolation valves shut, and the main feedwater pumps stop upon receipt of a scram signal. Main steam and feedwater flows are ramped to zero in 0.5 seconds.

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Assumption (2) allows credit for only one train of ECCS pumps and one train of Emergency Feedwater (EFW). Thus, only one HPSI pump and one EFW pump are available to satisfy LOCA requirements. It was additionally assumed that 25% of the HPSI inflow spills out through the break.

The flow capacity requirements for the HPSI and EFW pumps are proportional to values of decay heat. Assumption (3), therefore, results in a proportionate increase in these requirements. The HPSI pump capacity assumed is shown in Figure 4. The EFW capacity was assumed to be 875 gpm (minimum).

Assumption (4) resulted in the following setpoint and delay assumptions:

a) Reactor scram occurs upon a low pressurizer pressure (LPP) indication of 1728 psia (minimum). Upon receipt of the LPP signal there is a 0.9 second delay before the control rods begin motion followed by a 3.5 second interval to fully insert the rods.

b) SIAS occurs upon a pressurizer pressure indication of 1550 psia (minimum). The SIAS automatically starts the emergency diesel generators which provide power to the HPSI pumps and the EFW pumps. The total time from receipt of an SIAS to the time the HPSI pumps provide flow is 30 seconds (maximum). The total time delay from SIAS until EFW reaches the steam generators is 45 seconds (maximum).

References

1. Combustion Engineering, Inc., Standard Safety Analysis Report (CESSAR), Docket No. STN-50-470, Amendment No. 31, Section 6.3.3.3.
2. D. V. Graf to D. R. Patterson, TD-CE-117, subject: "Meeting Minutes of 5/27 and 5/28 in Windsor."
3. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3, Friday, January 4, 1974.

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FIGURE 1

SYSTEM 80

TWO PHASE MIXTURE LEVEL

IN THE INNER VESSEL

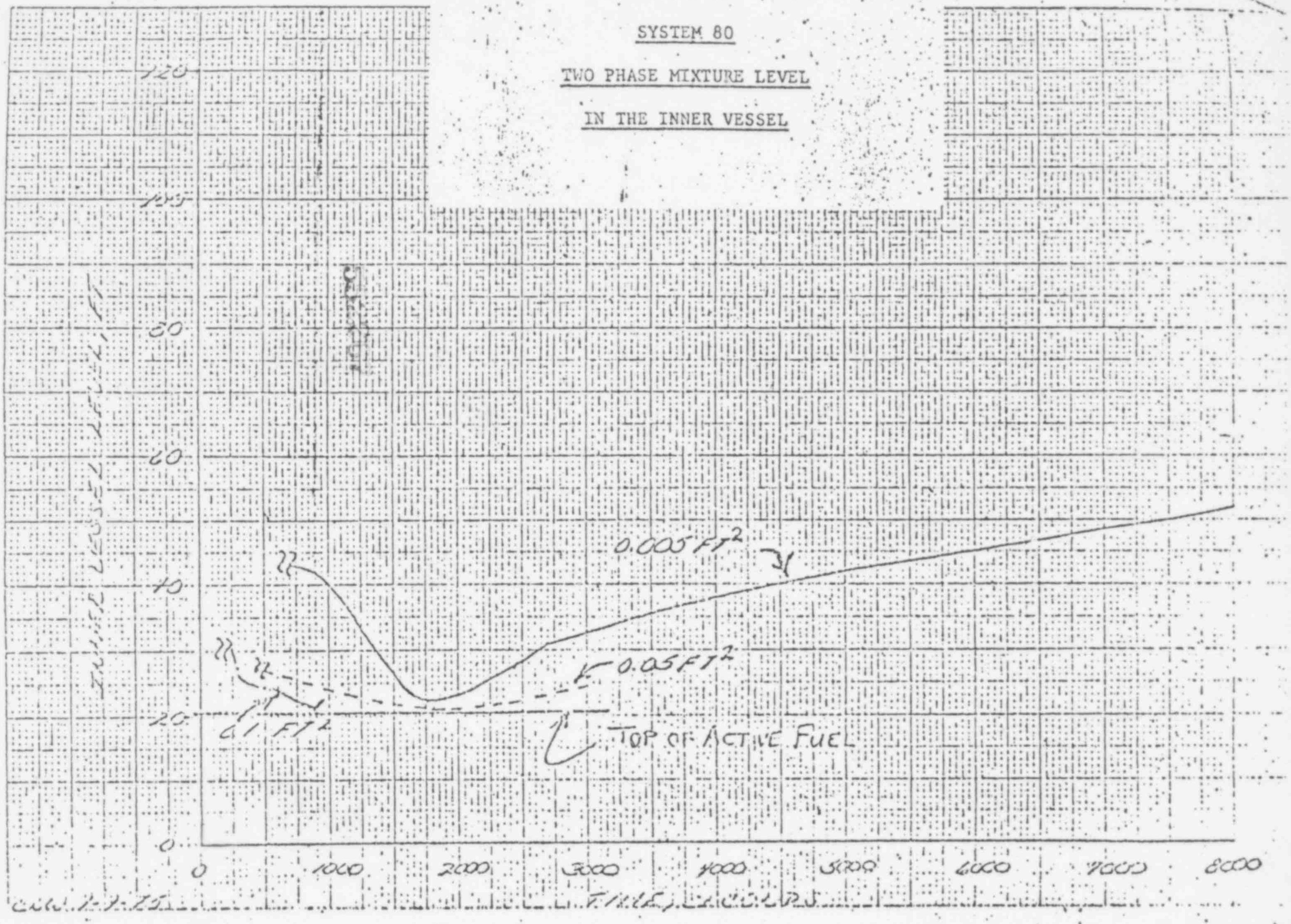


FIGURE 2

SYSTEM 80

PRIMARY SYSTEM PRESSURE

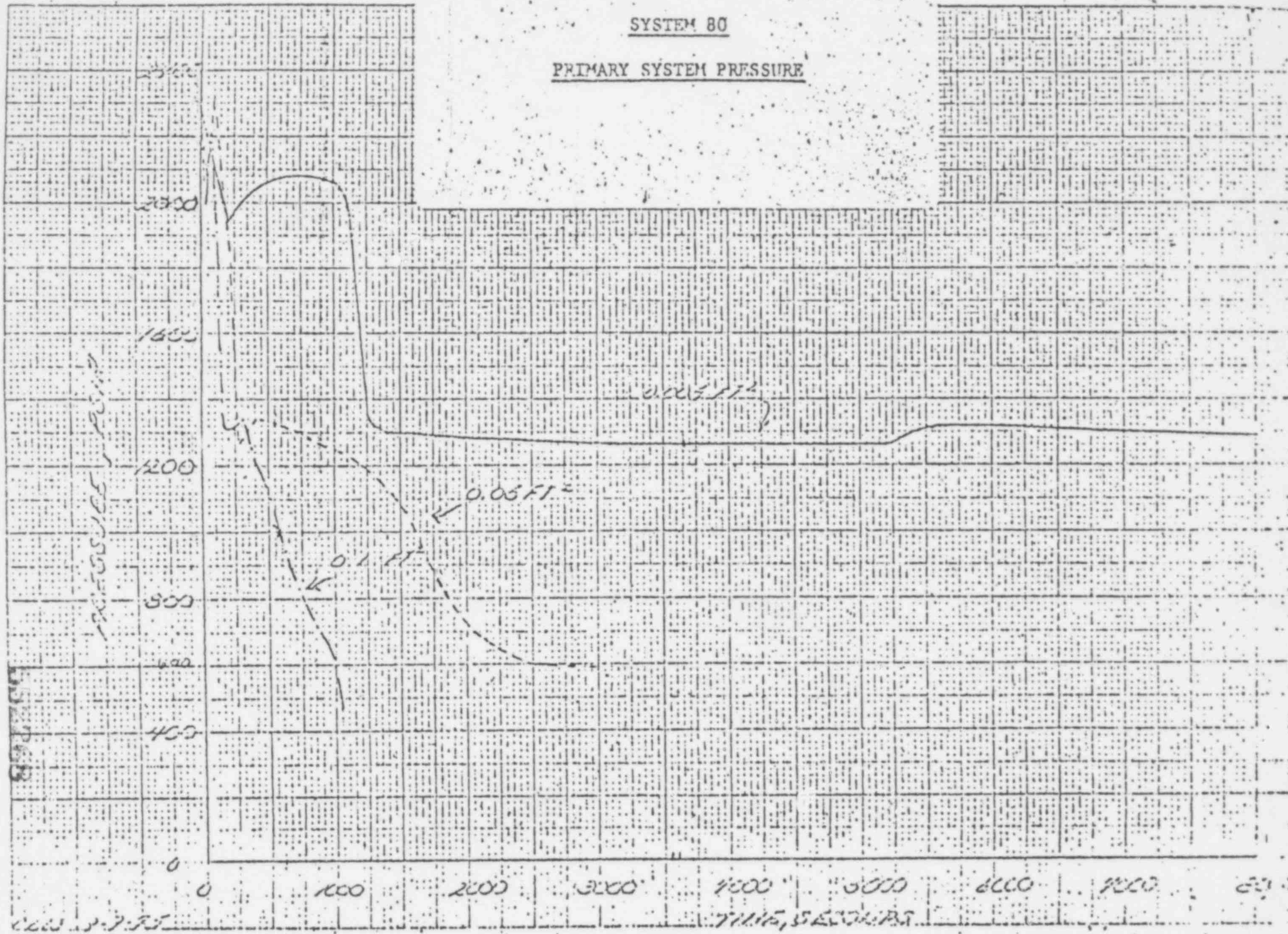


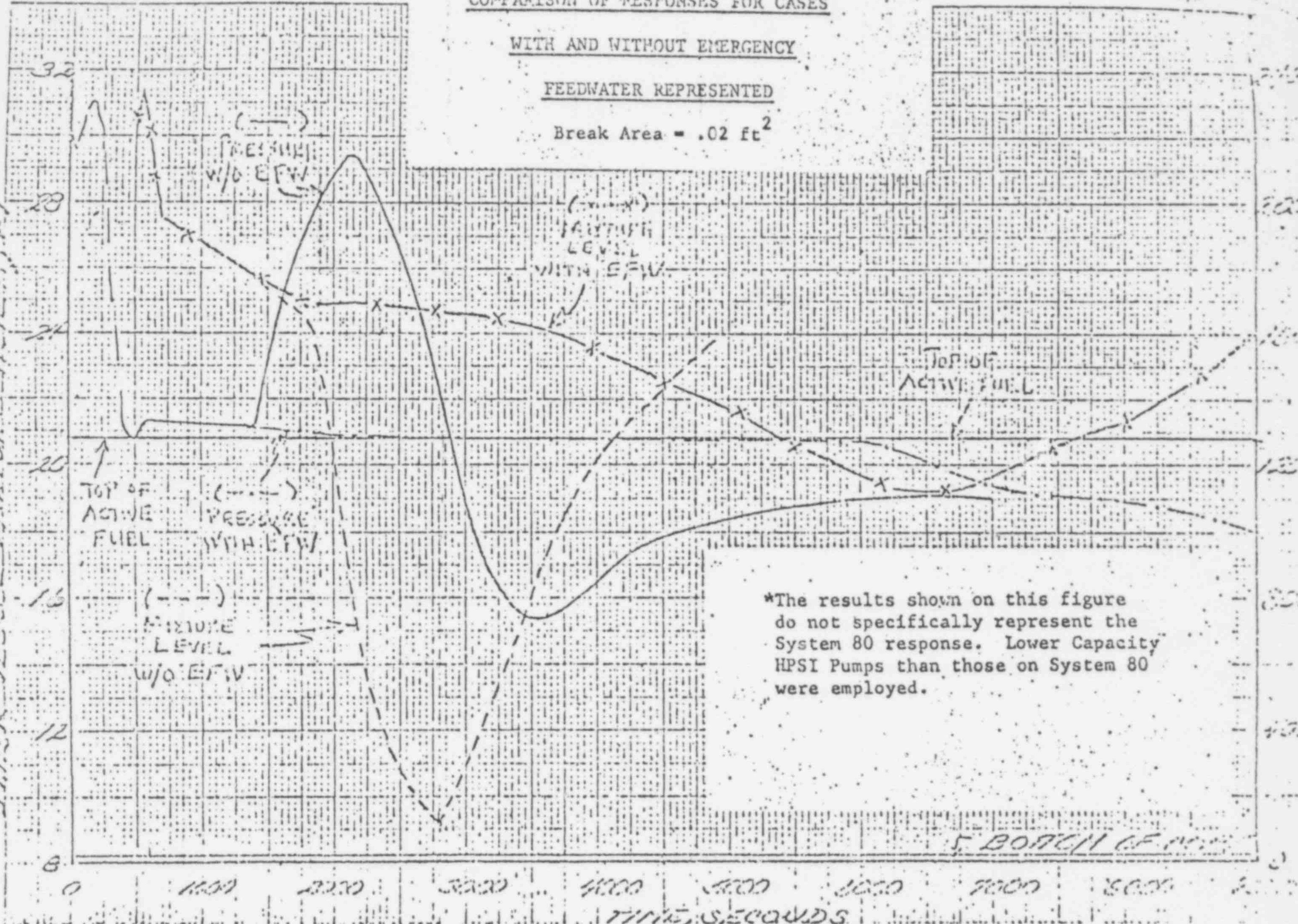
*FIGURE 3

COMPARISON OF RESPONSES FOR CASES

WITH AND WITHOUT EMERGENCY

FEEDWATER REPRESENTED

Break Area = .02 ft^2



*The results shown on this figure do not specifically represent the System 80 response. Lower Capacity HPSI Pumps than those on System 80 were employed.

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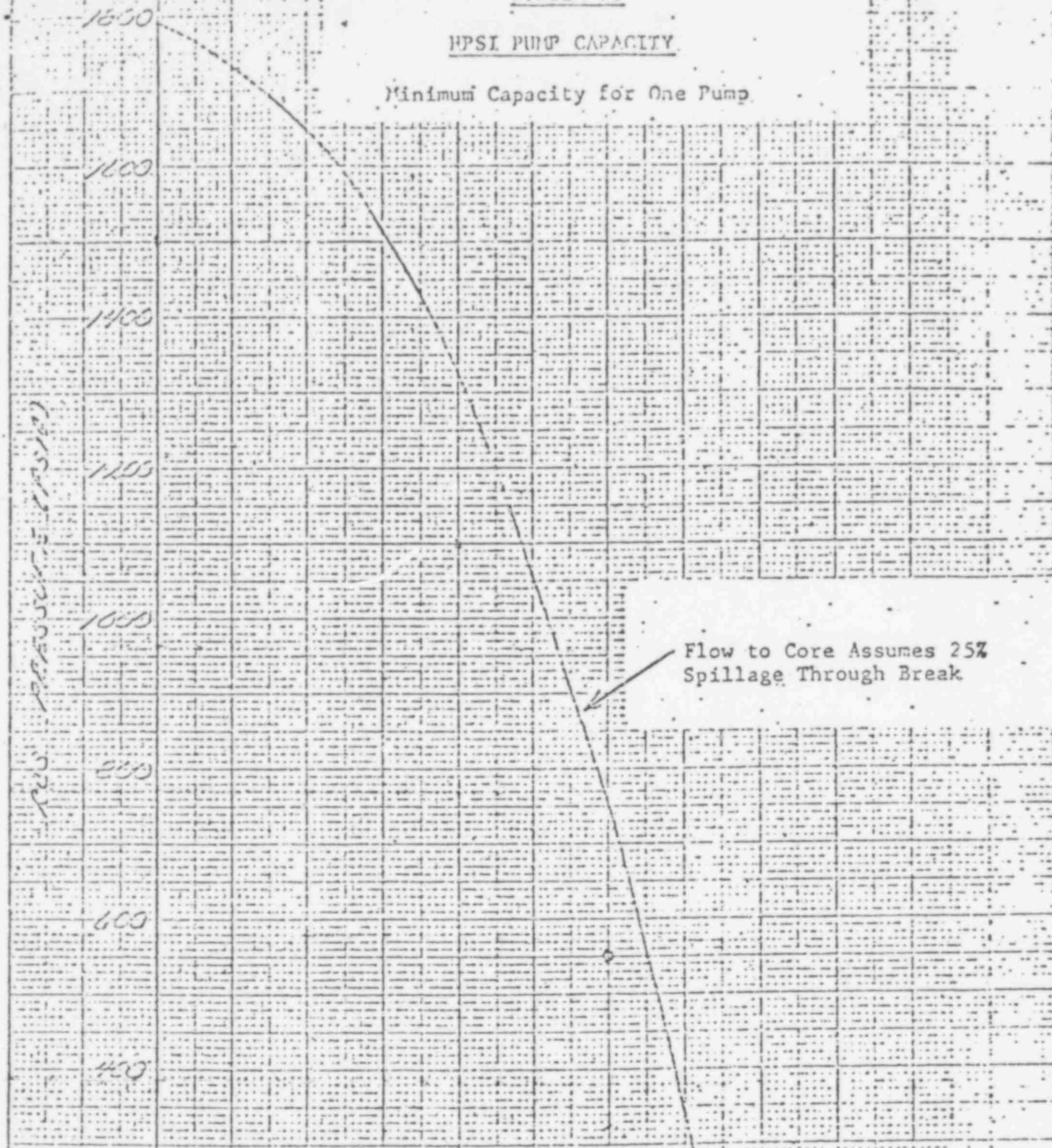
SYSTEM 80

HPSI PUMP CAPACITY

Minimum Capacity for One Pump



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