Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 15 1979

Docket Nos.: 50-269, 270, 287, 289, 302, 312, 313, 346

FACILITIES: Oconee Nuclear Station, Units No. 1, 2 & 3
Three Mile Island Nuclear Station, Unit No. 1
Crystal River Unit No. 3, Nuclear Generating Station
Rancho Seco Nuclear Generating Station
Arkansas Nuclear One, Unit No. 1
Davis-Besse Nuclear Power Station, Unit No. 1

LICENSEES: Duke Power Company
Metropolitan Edison Company
Florida Power Corporation
Sacramento Municipal Utility District
Arkansas Power & Light Company
Toledo Edison Company

SUBJECT: SUMMARY OF MEETING HELD ON JULY 18, 1979 TO DISCUSS BABCOCK & WILCOX (B&W) SMALL BREAK LOSS-OF-COOLANT ACCIDENT (LOCA) ANALYSES (REACTOR COOLANT PUMPS OPERATING VERSUS TRIPPED)

On July 18, 1979, members of the NRC staff met with B&W and representatives of the B&W Owners' Group (TMI-2 Follow-up Subcommittee) in Bethesda, Maryland to discuss preliminary calculations performed by B&W which indicated that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the reactor coolant pumps (RCPs) can increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core. A list of attendees is provided as Enclosure 1 to this summary.

BACKGROUND

Subsequent to the accident at Three Mile Island, Unit No. 2 (TMI-2) on March 28, 1979, the Office of Inspection and Enforcement (IE) issued several Bulletins requiring certain actions to be taken by all holders of operating licenses for each operating reactor. Item 4.c of IE Bulletin 79-05A required the licensees for B&W designed pressurized water reactors (PWRs) to revise their operating procedures to specify that, in the event of high pressure injection (HPI) initiation with RCPs running, at least one RCP per loop would remain operating. Similar requirements, applicable to reactors designed by other PWR vendors, were contained in other IE Bulletins.

Prior to the accident at TMI-2, Westinghouse and its licensees generally adopted the position that the operator should promptly trip all operating RCPs in the LOCA situation. This Westinghouse position, led to a series of meetings between the NRC staff and Westinghouse, as well as with the other PWR vendors, to discuss this issue. In addition, more detailed analyses concerning this matter were requested by the staff. This meeting was arranged to discuss B&W's preliminary calculations concerning the effect of continued operation of the RCPs during a small break LOCA.

7909100644

K.B.
Memo

DISCUSSION

As discussed above, the current emergency procedures being used by all B&W operating plants direct the operators to keep at least one RCP per loop operating in the event of a LOCA, provided offsite power is available. B&W previously presented analyses which demonstrated that the reactor core would be adequately cooled during a small break LOCA provided either the RCPs remained operating during the entire course of the event or were tripped at the beginning of the accident. At the request of the NRC staff, B&W performed additional analyses, for various break sizes, assuming that the RCPs were operating at the beginning of the accident and were then inadvertently tripped a short time into the accident. These calculations were performed using a six node CRAFT-2 model.

B&W's calculations showed that for a range of breaks between 0.025 ft^2 and 0.2 ft^2, if RCPs remained operating until the reactor coolant system (RCS) contained a high void fraction and were then tripped, the core would be uncovered for an extended period of time (approximately 600 seconds assuming RCP trip at 90% void fraction and two HPI trains operating). B&W explained that with RCPs operating, more liquid would be lost through the break than could be injected with the HPI system, thus leading to high void fractions in the RCS. As long as RCPs remained operating, pumping a two-phase mixture of water and steam, cooling of the core would be insured. However, once the RCPs were tripped (either manually or through loss of offsite power) the liquid and steam would separate. If the void fraction in the RCS was approximately 63% or greater at the time the RCPs were tripped, the liquid remaining in the reactor vessel, following the liquid/steam separation, would not be sufficient to cover the core. A period of time would be needed for the HPI system to pump enough water into the vessel to recover the core. B&W's calculations assumed an adiabatic heat-up of the core of 5^oF/Sec. during the period the core was uncovered. Under certain conditions, these calculations indicated that 10 CFR Part 50 Appendix K limits would be exceeded.

B&W stated that the critical break size was bounded by these analyses. Below 0.025 ft^2 the void fraction would not exceed the 63% value and thus, if RCPs were tripped, the core would remain covered. Above 0.2 ft^2, the system would depressurize rapidly enough to insure low pressure injection (LPI) initiation for continued core cooling.

The NRC staff expressed several concerns with the model used for the B&W calculations:

(1) the six node CRAFT-2 model was found to produce void fractions several percent lower than the 23 node model normally used for small break analyses,

(2) no liquid carryout was assumed to occur from the core during the reflooding period,

(3) a homogeneous void distribution model was used in CRAFT-2 during the time the RCPs were operating. This assumption maximizes the liquid lost through the break and may be overly conservative, and

(4) the adiabatic heat-up model used when the core is uncovered may also be overly conservative.

B&W stated that on all of the B&W plants no RCS penetrations exist in that range of break sizes and therefore, felt the probability of a break within the bounds of the analyses was highly unlikely. The staff felt that enough uncertainty existed within the analyses that the bounds of the break size could be greater than that presented in the analyses. In addition, the staff asked B&W if having two pressurizer safety valves stick open would put the plant into a small break LOCA situation that was within the critical break size. B&W stated they would look into this matter.

B&W stated that it would continue to do additional analyses in this area to confirm its preliminary calculations; in addition, if prompt tripping of the RCPs during a LOCA situation would be required, it was investigating the possibility of designing a hard-wired RCP trip which would trip the pumps on initiation of HPI (low RCS pressure-indicative of a LOCA) in conjunction with low RCP current (indicative of a high void fraction in the RCS).

The NRC staff stated that it would review the calculations presented and determine the advisability of tripping the RCPs upon indication of a LOCA. The slides used in the presentation are included as Enclosures 2 through 15 of this summary.

A similar analysis was presented by B&W for a main steam line break. The results of that analysis showed that the time RCPs were tripped had little effect on the severity of the accident.

Two additional areas were discussed at the end of the meeting:

(1) <u>Outstanding Information Requirements</u>

The list of outstanding information requirements, tabulated by B&W in a letter from J. Taylor (B&W) to R. Mattson (NRC) dated June 13, 1979, was compared to a list compiled by the NRC staff. The staff stated that it needed information in three areas which were not covered in the B&W letter. These include:

(A) NRC concerns regarding the presence of non-condensible gas within the RCS following a small break LOCA,

(B) responses to questions concerning the CRAFT-2 code small break analyses raised by the ACRS (ECCS Subcommittee) June 19, 1979, and

(C) the ability of the RCPs to operate in a highly voided reactor coolant system.

The staff agreed that a letter formalizing these commitments would be sent in the near future.

(2) Analyses Required in the Next 3 to 6 Months

A subgroup of members of the Bulletins & Orders and the Lessons Learned Task Forces has determined that, in addition to the evaluation of small break LOCAs, two additional tasks should be undertaken:

(A) an assessment of the symptoms of inadequate core cooling, and

(B) an evaluation of accidents and transients beyond current design analyses.

In both cases, the emphasis should be on information needed for operator training, and information needed for the preparation of improved emergency procedures. The staff suggested that these programs be completed by December 1, 1979. The B&W Owner's Group stated they would have difficulty meeting this schedule. They plan to meet with the staff in mid-August to discuss progress on the two additional analyses.

CONCLUSIONS

Following this meeting, the staff held discussions with the other PWR vendors to pass on the information presented by B&W. Analyses performed by all the vendors (B&W, Westinghouse and Combustion Engineering) supported the tripping of RCPs immediately under LOCA conditions. On July 26, 1979, IE Bulletin Nos. 79-05C & 79-06C was issued requiring certain actions to be taken by all holders of operating licenses for PWR facilities. This Bulletin required, as immediate action for all PWR licensees, to revise their operating procedures such that upon reactor trip and initiation of HPI caused by low reactor coolant system pressure the operator will immediately trip all operating RCPs. A copy of this Bulletin is attached as Enclosure 16 to this summary.

R A Capra

R. A. Capra, Project Manager
Bulletins & Orders Task Force

Enclosures:
As stated

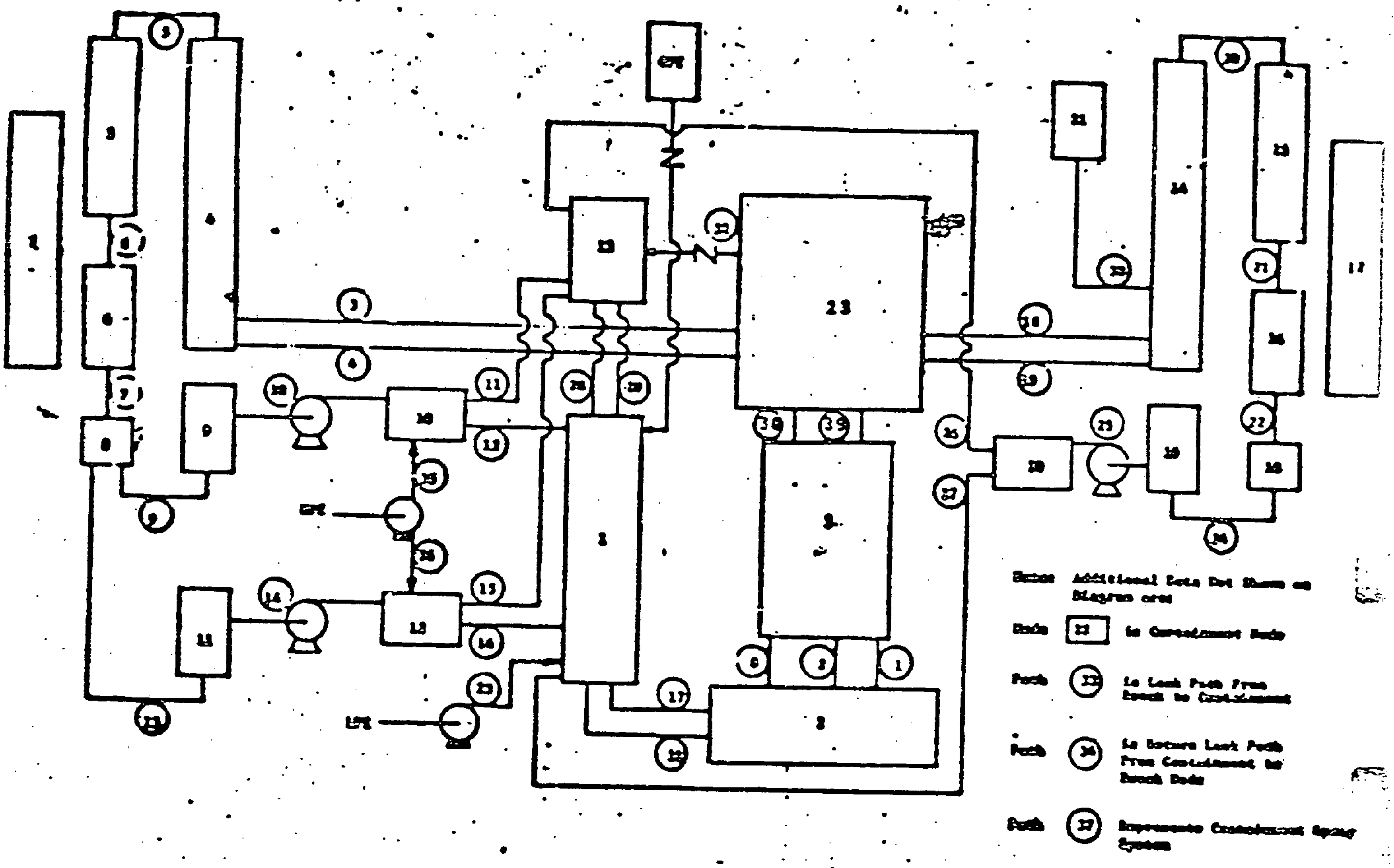
Distribution:
See attached pages

ENCLOSURE 1

LIST OF ATTENDEES

Duke Power Company	D. C. Holt (Systems Engineer, Licensing) K. Conody (Manager, Projects & Licensing) R. Gill (Oconee Head - Licensing) P. M. Abraham (Nuclear Engineer)
General Public Utilities Service Corporation	L. Lanese (Control & Safety Analysis Engineer)
Florida Power Corporation	P. Y. Baynard (Assistant Manager - Nuclear Support Services)
Sacramento Municipal Utility District	S. Anderson (Nuclear Engineer)
Arkansas Power & Light Company	D. H. Williams (Production Engineer) D. G. Mardis (Production Engineer - Licensing)
Toledo Edison Company	F. Miller (Nuclear Systems Engineer)
Consumers Power	T. J. Sullivan (Executive Engineer)
Babcock & Wilcox	H. Bailey (Licensing Engineer) R. E. Ham (Product Line Manager, Engineering Services) D. Hallman (Manager, Plant Performance Services) E. R. Kane (Manager, Operating Plant Licensing) C. E. Parks (Adv. Eng.) M. A. Haghi (Nuclear Engineer) G. O. Geissler (Manager, Generic Licensing Unit) E. A. Womack (Manager, Plant Design)
NRC	B. Sheron (Bulletins & Orders Task Force - Analysis) R. F. Audette " " " " " P. S. Tam (ACRS) J. G. Stampels (ACRS) E. J. Brown (Structures & Components Standards Branch) S. L. Wookey (Operating Reactors Branch No. 4) D. Garner (Project Manager, Rancho Seco) M. B. Fairtile (Project Manager, Oconee 1, 2, 3) G. Kelley (Bulletins & Orders Task Force - Systems) P. E. Norian (Bulletins & Orders Task Force - Analysis) W. L. Jensen " " " " " Z. R. Rosztoczy (Bulletins & Orders Task Force Group Leader Analysis) T. H. Novak (Deputy Director, Bulletins & Orders Task Force R. A. Capra (B&W Project Manager, Bulletins & Orders Task Force)

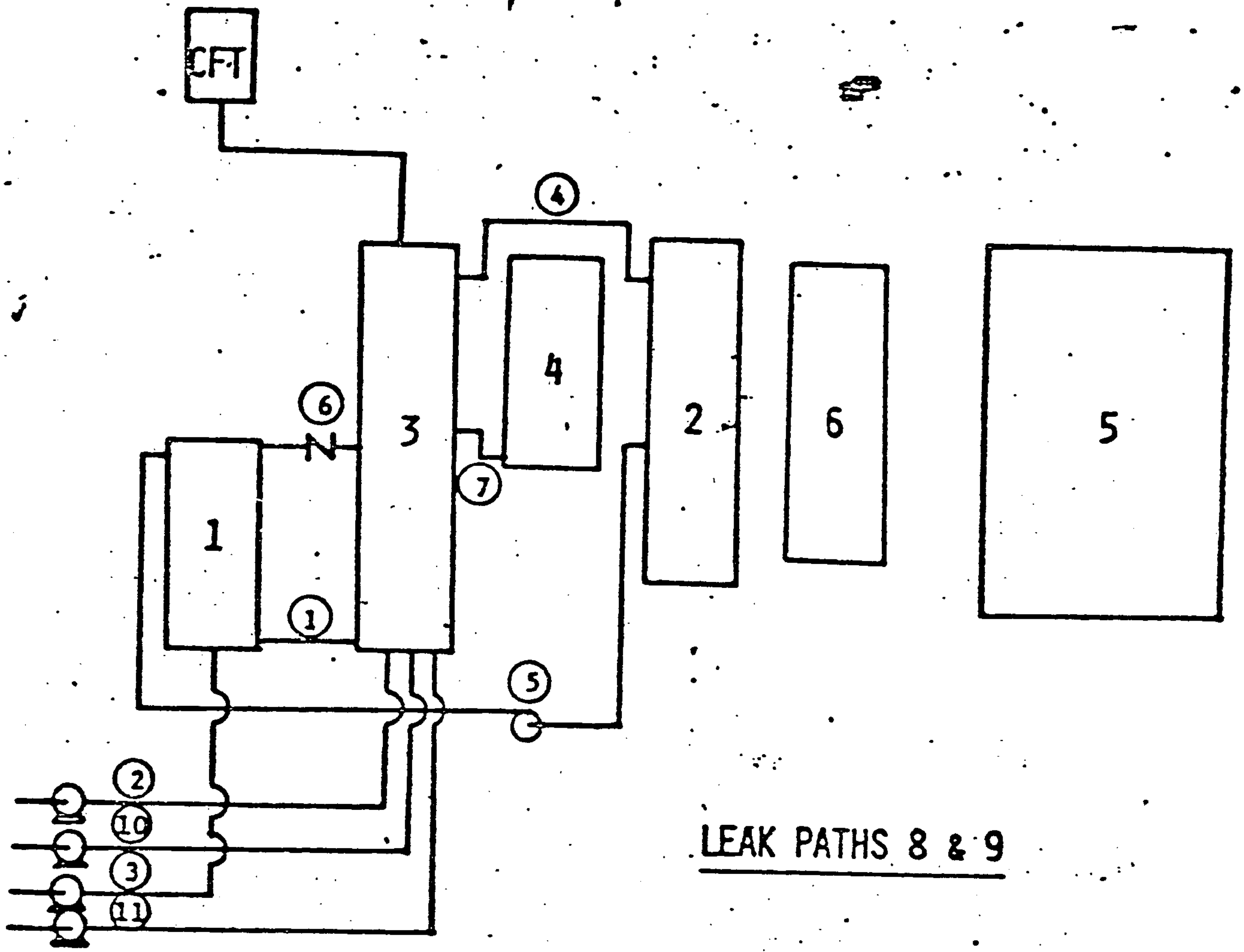
Figure 1. CRAFT2 Noding Diagram for Small Breaks



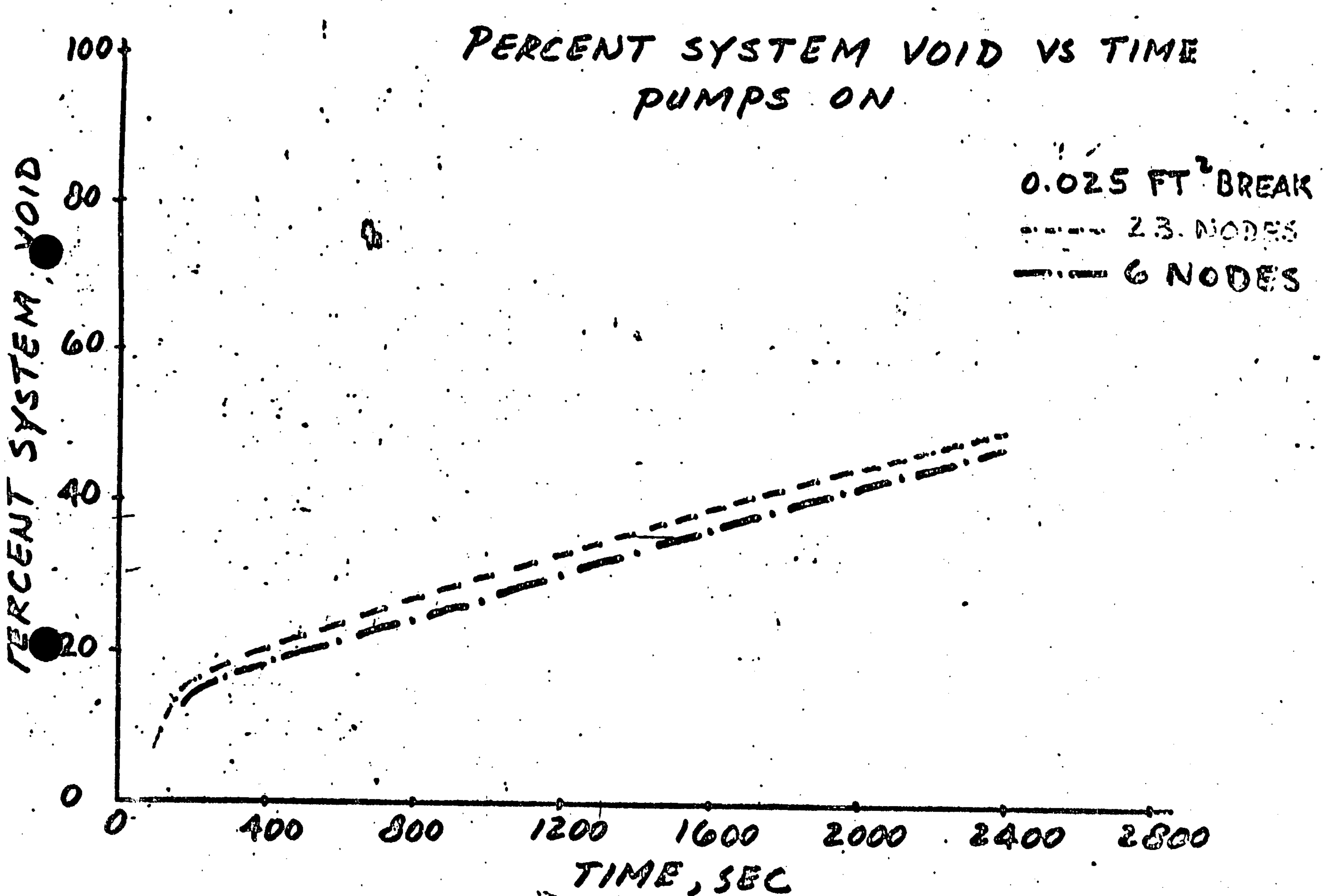
Node No.	Identification
1	Downcomer
2	Lower Plenum
3	Core
4,14	Hot Leg Piping
5,15	SG & Upper Head
6,16	Steam Generator Tubes
7,17	Secondary, SG
8,18	SG Lower Head
9,11,19	Cold Leg Piping
10,12,20	Cold Leg Piping
13	Upper Downcomer
21	Pressurizer
22	Containment
23	Upper Plenum

Path No.	Identification
1,2	Core
3,4,18,19	Hot Leg Piping
5,20	Hot Leg, Upper
6,21	SG Tubes
7,22	SG Lower Head
8	Core Bypass
9,13,24	Cold Leg Piping
10,14,25	Pumps
11,12,15,16,26,27	Cold Leg Piping
17,31	Downcomer
23	LPI
28,29	Upper Downcomer
30	Pressurizer
32	Vent Valve
33,34	Leak & Return Path
35,36	HPI
37	Containment Sprays

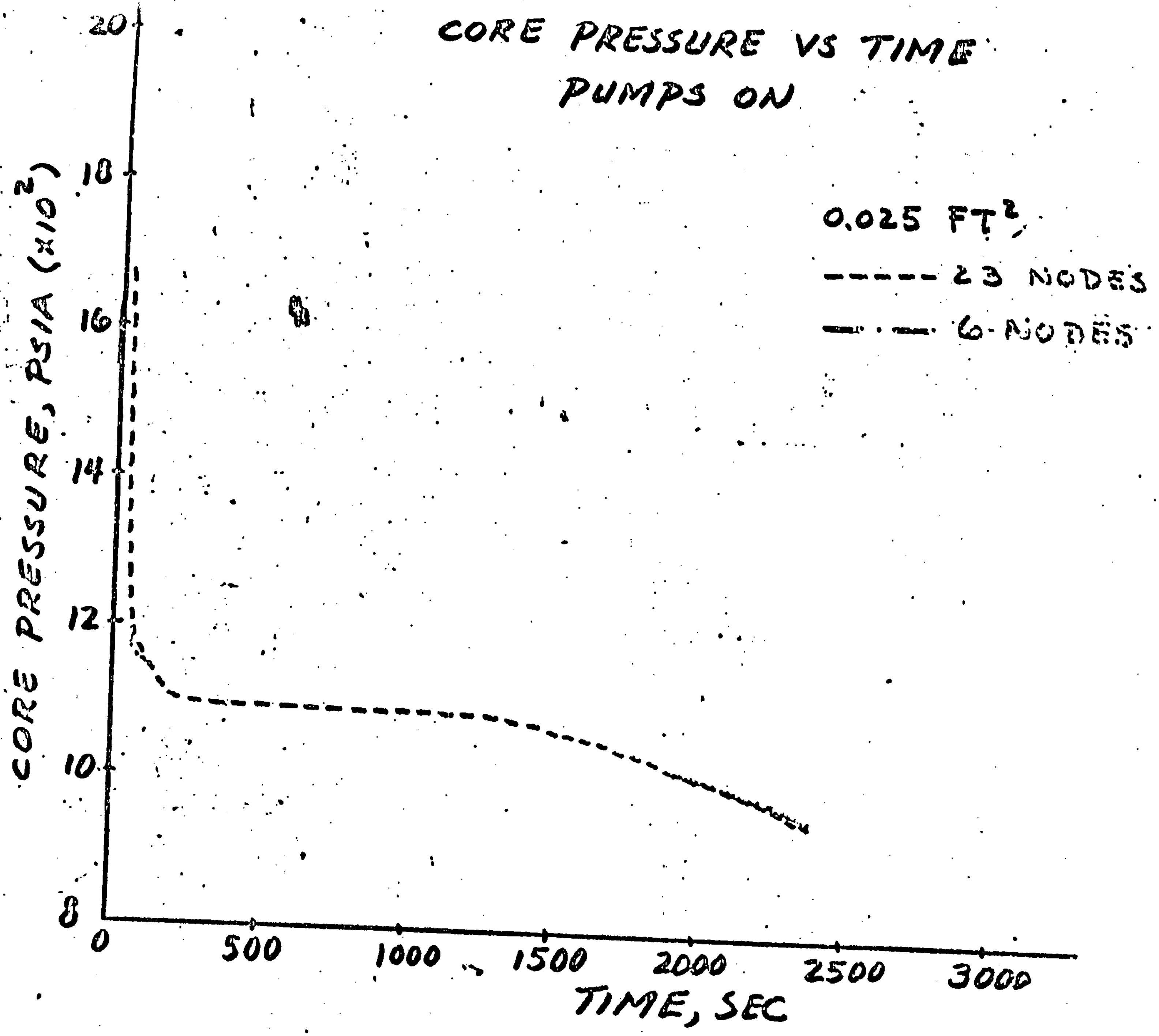
CRAFT2 NODING DIAGRAM FOR SMALL BREAKS
(6 NODE MODEL)



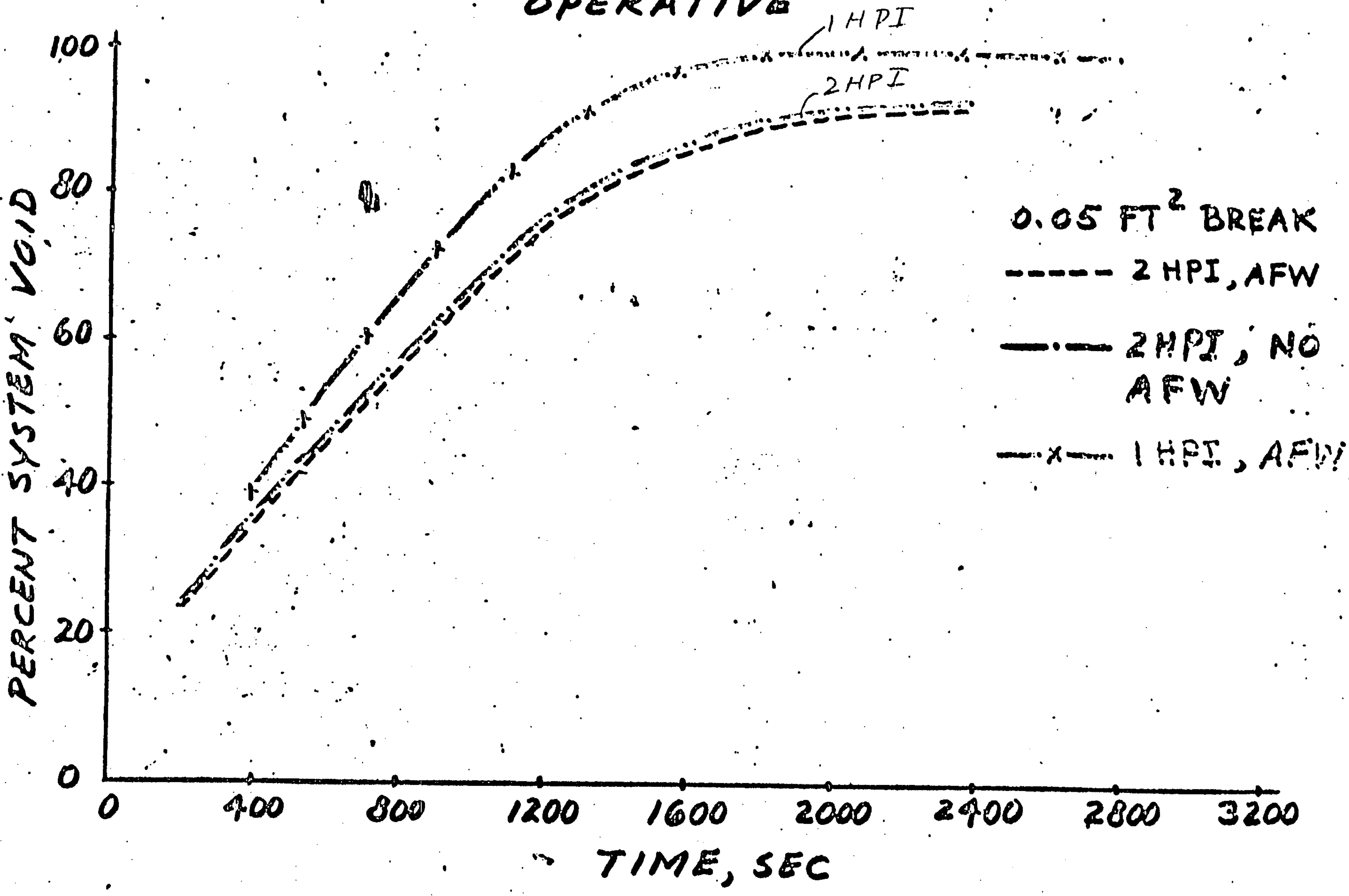
Enclosure 4



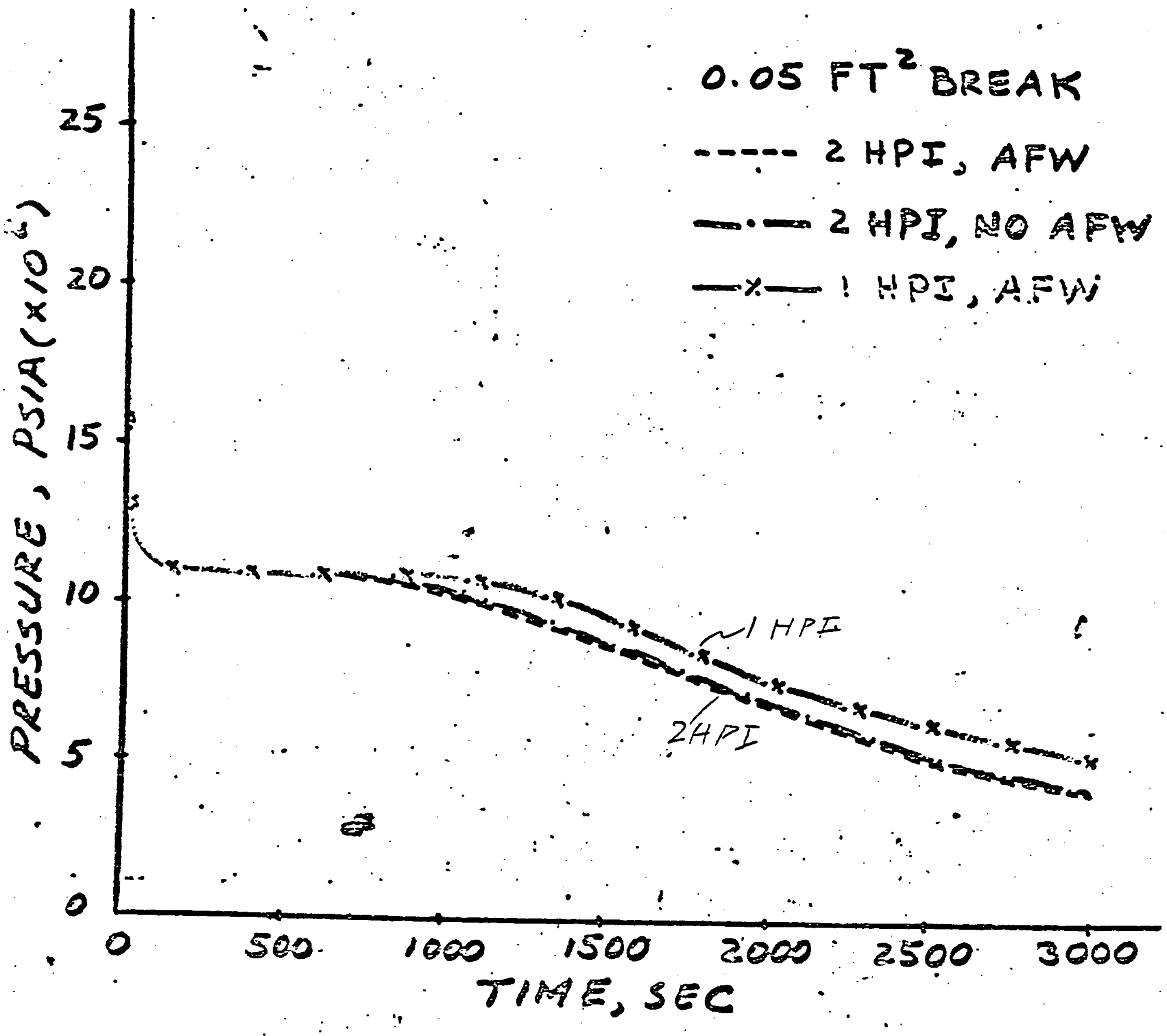
PERCENT SYSTEM VOID VS TIME
PUMPS ON
0.025 FT^2 BREAK
23 NODES
6 NODES
PERCENT SYSTEM VOID
100
80
60
40
20
0
0
400
800
1200
1600
2000
2400
2800
TIME, SEC

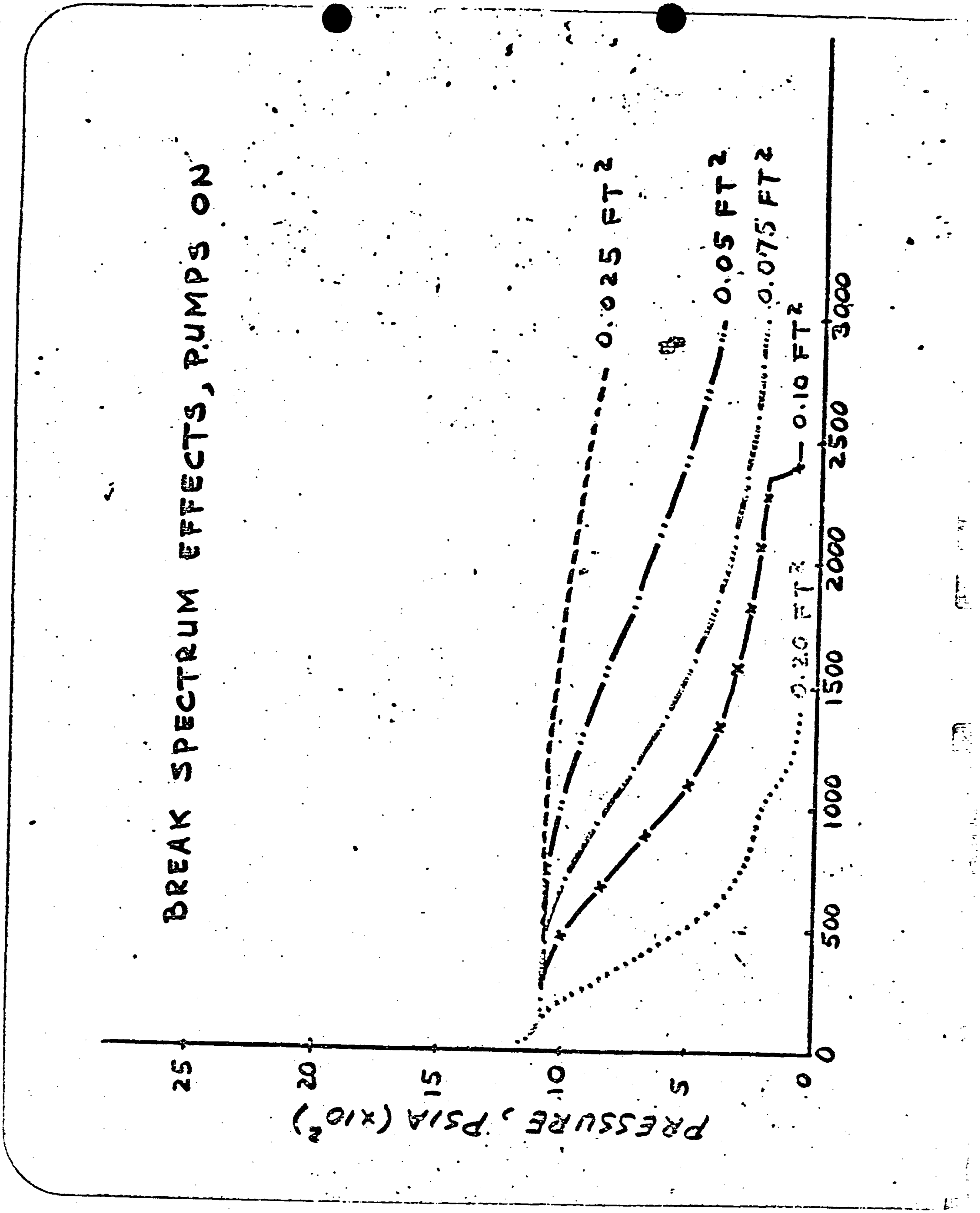

CORE PRESSURE VS TIME
PUMPS ON
0.025 FT²
23 NODES
6 NODES
CORE PRESSURE, PSIA (x10²)
20
18
16
14
12
10
8
0
500
1000
1500
2000
2500
3000
TIME, SEC

EFFECT OF EQUIPMENT WITH PUMPS OPERATIVE



EFFECT OF EQUIPMENT WITH PUMPS OPERATIVE





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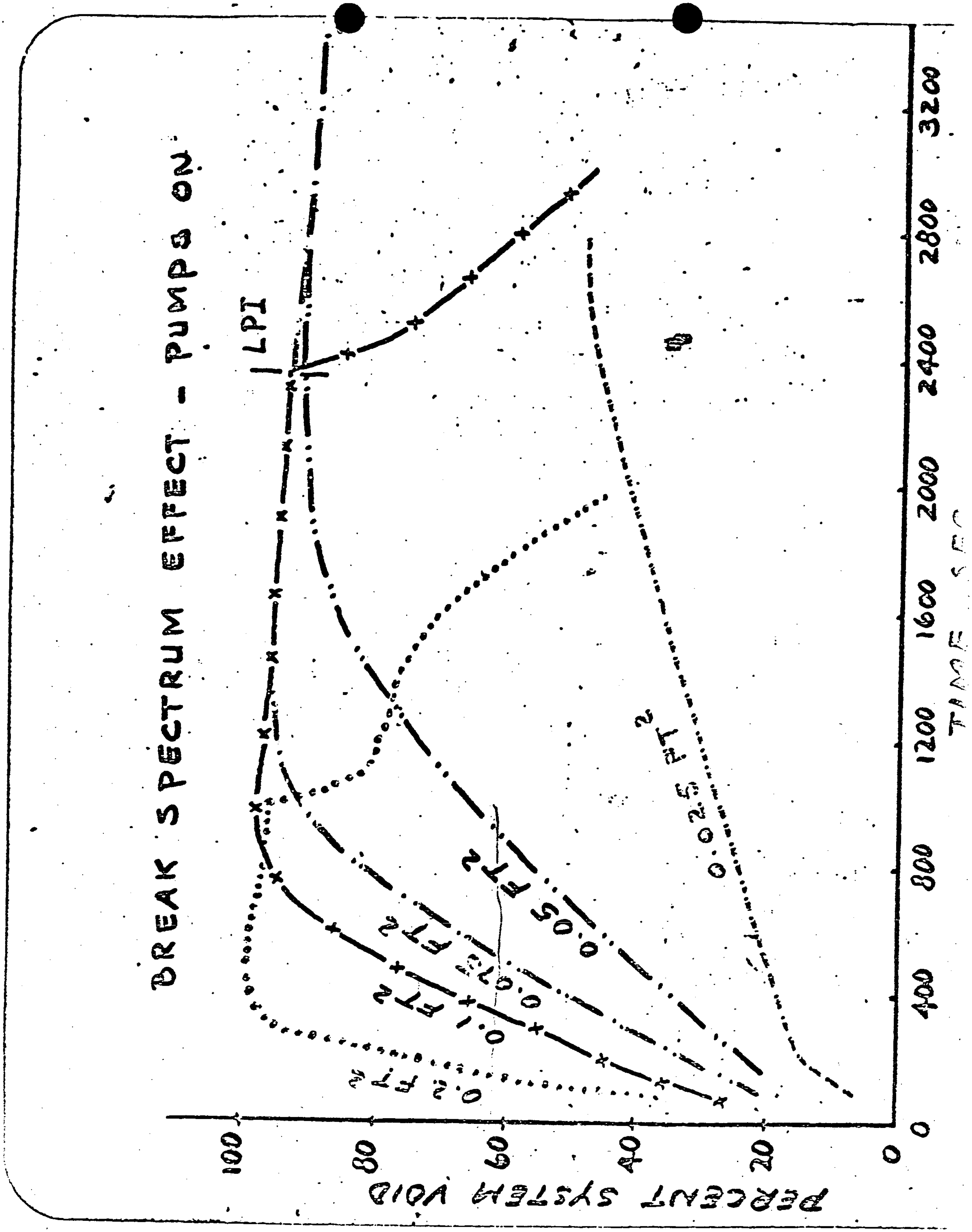
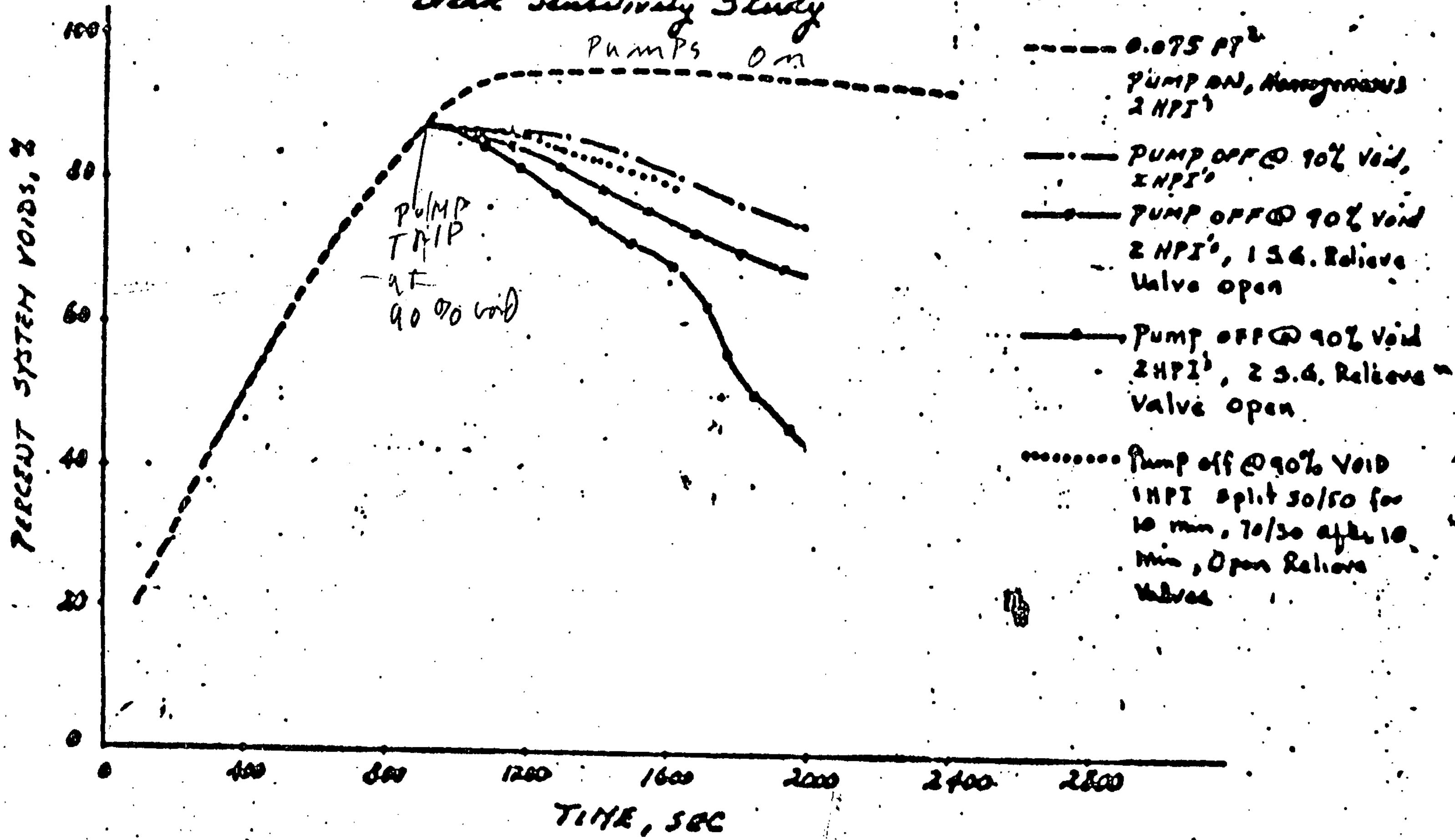

BREAK SPECTRUM EFFECT - PUMPS ON
LPI
0.2 FT2
0.1 FT2
0.075 FT2
0.05 FT2
0.025 FT2
PERCENT SYSTEM VOID
0
20
40
60
80
100
0
400
800
1200
1600
2000
2400
2800
3200

FIGURE 12: Average System Void Fraction for 0.075 FT^2 Break Sensitivity Study



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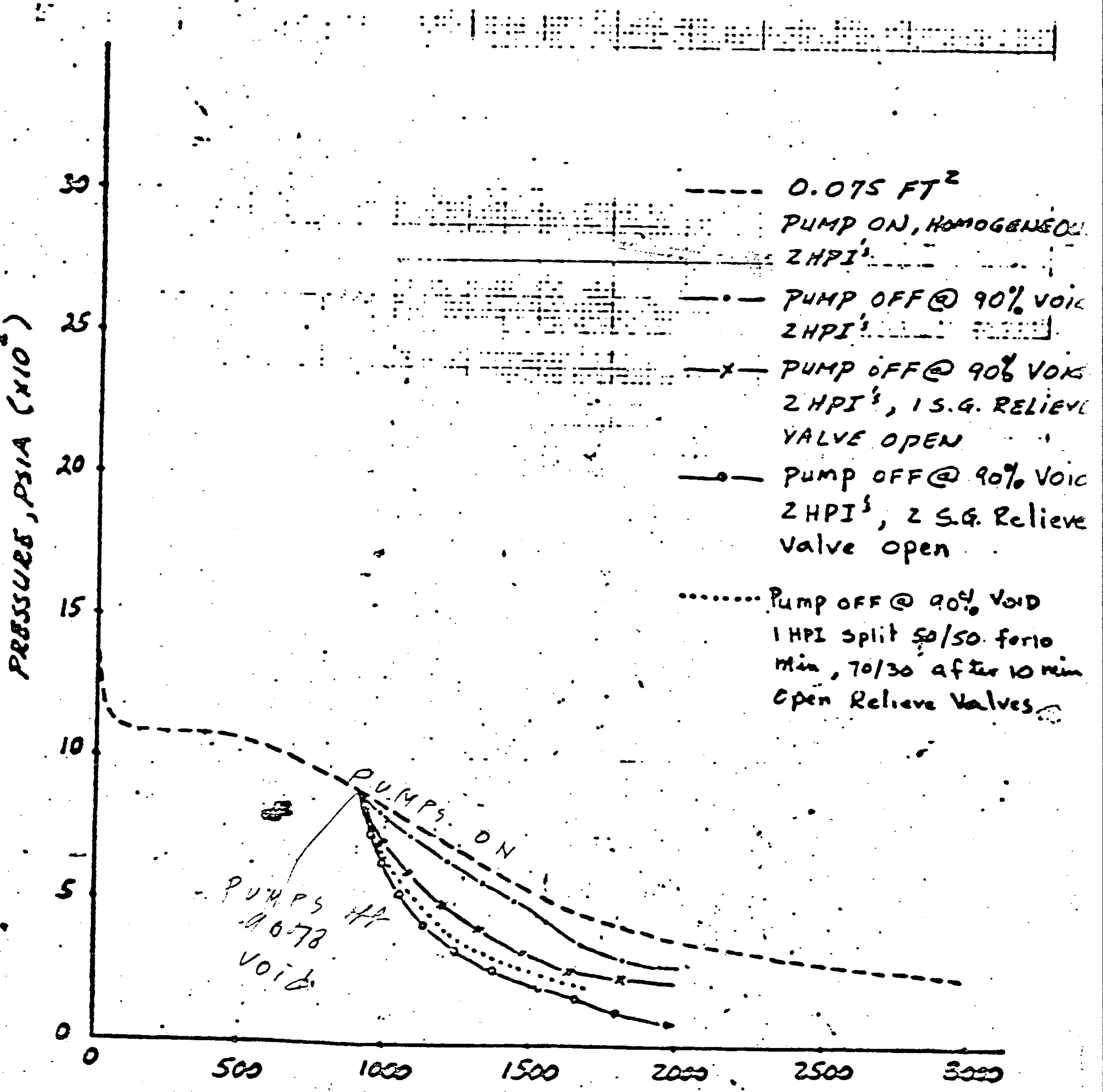
Enclosure 10

FIGURE 11: RC Pressure for 0.075 FT^2 Break Sensitivity study



1.9

FIGURE 15: Percent System Void fraction for 0.05 FT² Break with 1.0 and 1.2 ANS

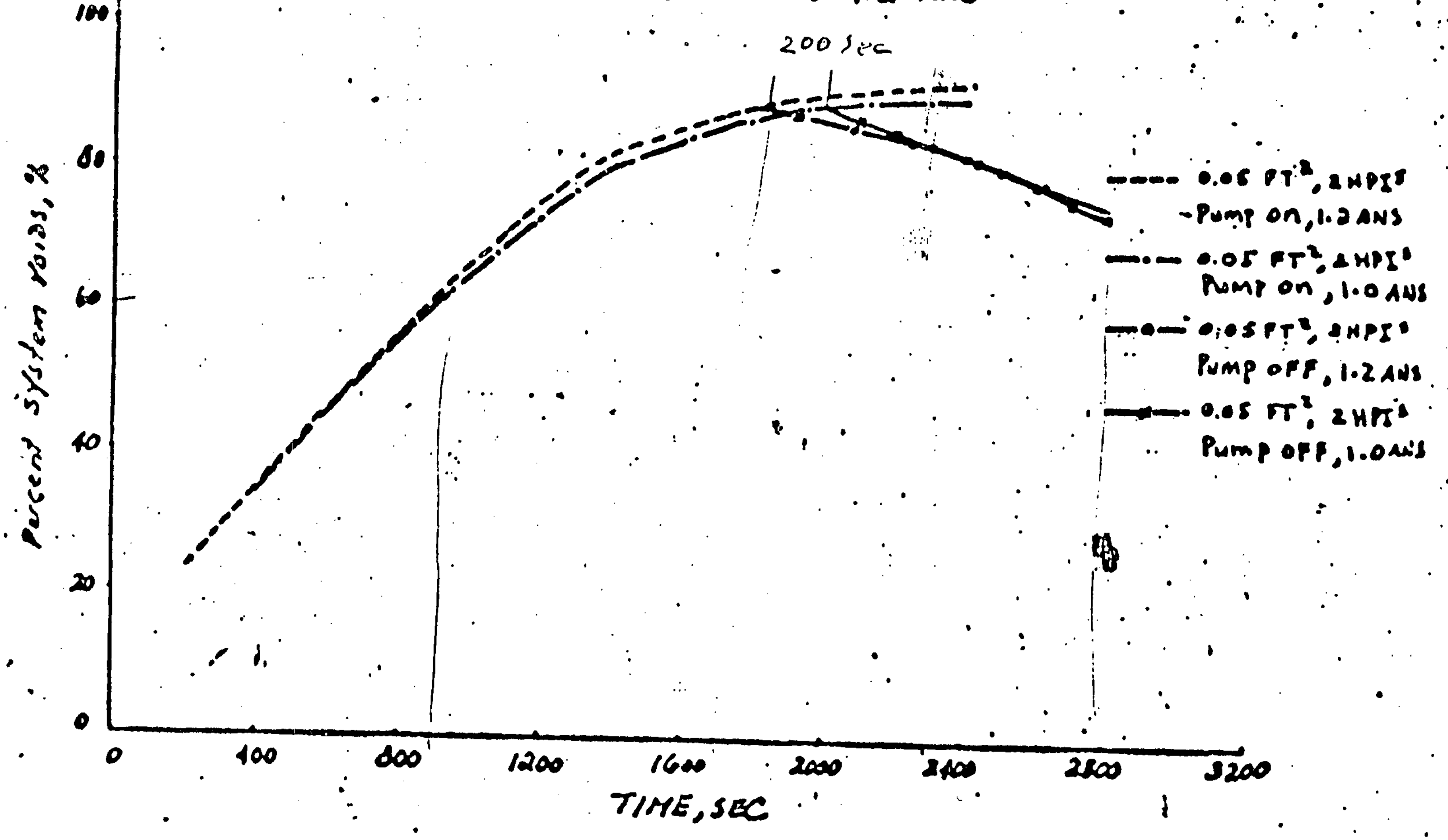
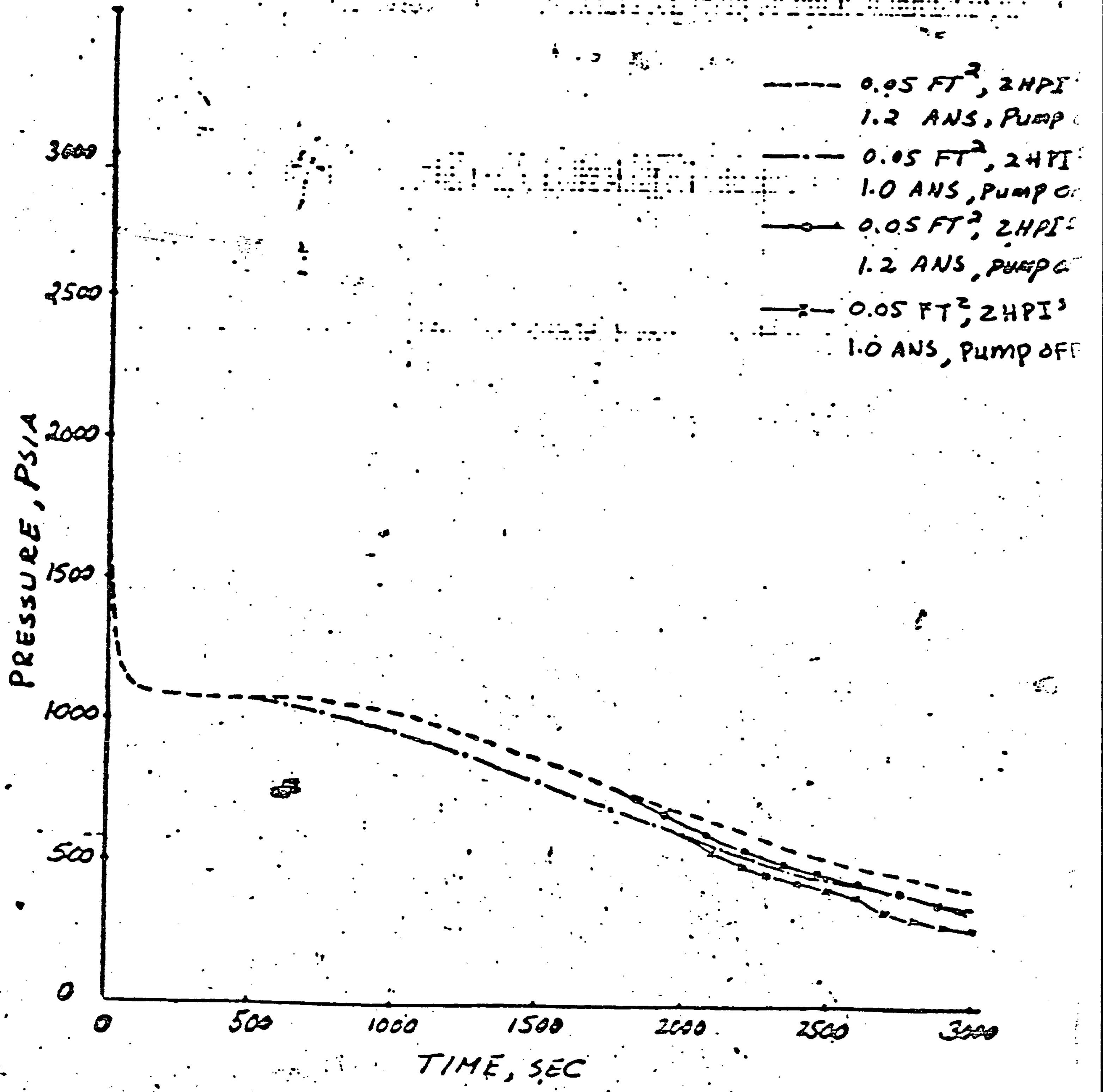
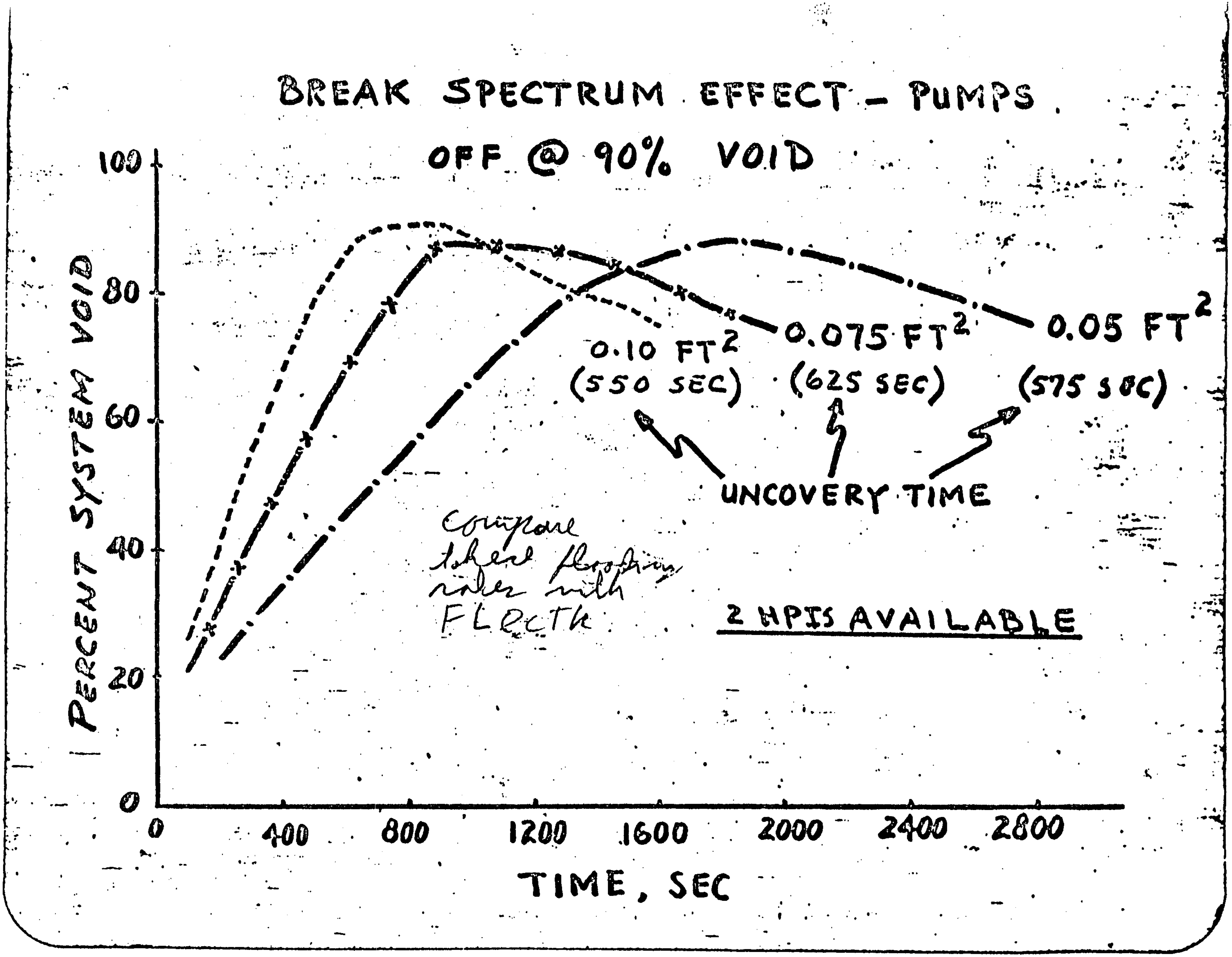
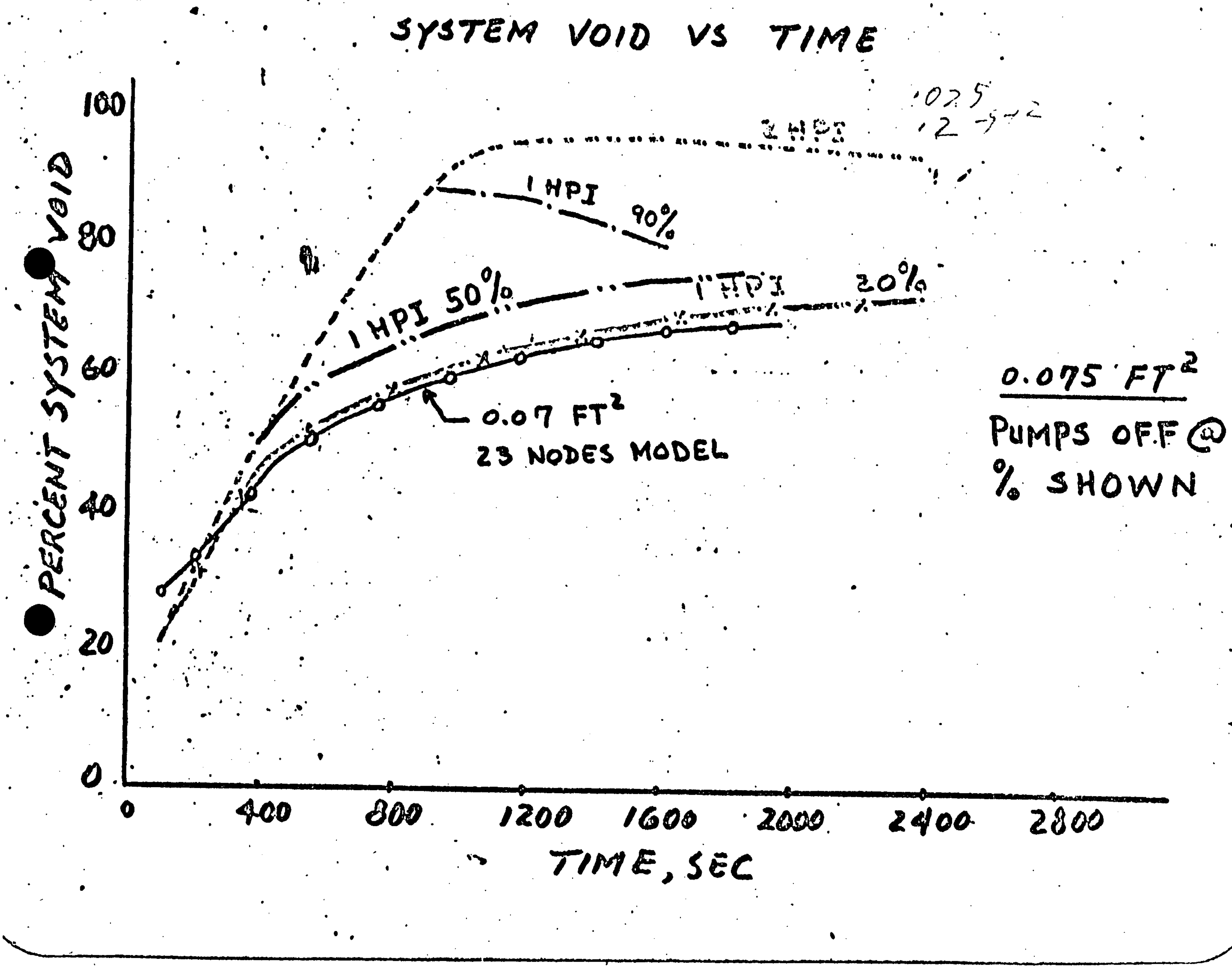


FIGURE 14: RC Pressure Vs Time for 0.05 FT^2 Break With 1.0 and 1.2 ANS





Enclosure 14



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Enclosure 15

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 26, 1979

IE Bulletin Nos. 79-05C & 79-06C

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Information has become available to the NRC, subsequent to the issuance of IE Bulletins 79-05, 79-05A, 79-05B, 79-06, 79-06A, 79-06A (Revison 1) and 79-06B, which requires modification to the "Action To Be Taken By Licensees" portion of IE Bulletins 79-05A, 79-06A and 79-06B, for all pressurized water reactors (PWRs).

Item 4.c of Bulletin 79-05A required all holders of operating licenses for Babcock & Wilcox designed PWRs to revise their operating procedures to specify that, in the event of high pressure injection (HPI) initiation with reactor coolant pumps (RCPs) operating, at least one RCP per loop would remain operating. Similar requirements, applicable to reactors designed by other PWR vendors, were contained in Item 7.c of Bulletin 79-06A (for Westinghouse designed plants) and in Item 6.c of Bulletin 79-06B (for Combustion Engineering designed plants).

Prior to the incident at Three Mile Island Unit 2 (TMI 2), Westinghouse and its licensees generally adopted the position that the operator should promptly trip all operating RCPs in the loss of coolant accident (LOCA) situation. This Westinghouse position, has led to a series of meetings between the NRC staff and Westinghouse, as well as with other PWR vendors, to discuss this issue. In addition, more detailed analyses concerning this matter were requested by the NRC. Recent preliminary calculations performed by Babcock & Wilcox, Westinghouse and Combustion Engineering indicate that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the RCPs can increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core.

The damage to the reactor core at TMI 2 followed tripping of the last operating RCP, when two phase fluid was being pumped through the reactor coolant system. It is our current understanding that all three of the nuclear steam system suppliers for PWRs now agree that an acceptable action under LOCA symptoms is to trip all operating RCPs immediately, before significant voiding in the reactor coolant system occurs.

Action To Be Taken By Licensees:

In order to alleviate the concern over delayed tripping of the RCPs after a LOCA, all holders of operating licenses for PWR facilities shall take the following actions:

Short-Term Actions

1. In the interim, until the design change required by the long-term action of this Bulletin has been incorporated, institute the following actions at your facilities:

   A. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

   B. Provide two licensed operators in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this Bulletin.

2. Perform and submit a report of LOCA analyses for your plants for a range of small break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters, determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 2200 degrees F is identified.

3. Based on the analyses done under Item 2 above, develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For Babcock & Wilcox designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a higher level, following RCP trip, to promote natural circulation flow.

4. Revise emergency procedures and train all licensed reactor operators and senior reactor operators based on the guidlines developed under Item 3 above.

5. Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed in Section 2.1.9 of NUREG-0578, "TMI 2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

Long-Term Action

1. Propose and submit a design which will assure automatic tripping of the operating RCPs under all circumstances in which this action may be needed.

Schedule

The schedule for the short-term actions of this Bulletin is:

Item 1: Effective upon receipt of this Bulletin,

Item 2: Within 30 days of receipt of this Bulletin,

Item 3: Within 30 days of receipt of this Bulletin,

Item 4: Within 45 days of receipt of this Bulletin,

Item 5: October 31, 1979 (as noted in Table B-2 of NUREG-0578, under Item 3).

A schedule for the long-term action required by this Bulletin should be developed and submitted within 30 days of receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office with copies forwarded to the Director, Office of Inspection and Enforcement and the Director, Office of Nuclear Reactor Regulation, Washington, D. C. 20555.

Approved by GAO (R0072): clearance expires 7/31/80. Approval was given under a blanket clearance specifically for generic problems.

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