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

SEPTEMBER 2 4 1982

MEMORANDUM FOR: Licensing Project Managers

FROM: Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

SUBJECT: BOARD NOTIFICATION - SEMISCALE TEST RESULTS
(BN 82-93)

The ASLB, ASLAB, or Commission (as appropriate) were notified of the subject report for the following plants:

Callaway, Unit 1	Waterford 3
Comanche Peak, Units 1 & 2	Diablo Canyon, Units 1 & 2
Palo Verde, Units 1, 2 & 3	Summer
Midland, Units 1 & 2	Floating Nuclear Plants 1-8

For those plants which are contested but for which the hearing has not yet started, please ensure that the substance of BN 82-93 is addressed in your SER or SSER. Attached is a summary of the subject report for your information.

T M Novak

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc: LBCs

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XA

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

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AUG 3 0 1982

MEMORANDUM FOR: Darrell Eisenhut, Director, Division of Licensing

FROM: Roger J. Mattson, Director, Division of Systems Integration

SUBJECT: BOARD NOTIFICATION CONCERNING RECENT SEMISCALE TEST RESULTS

SUMMARY

The purpose of this memorandum is to request that you notify all PWR licensing boards of the results of a recent Semiscale "feed and bleed" test.

During a recent test in the Semiscale facility* in which the "feed and bleed" mode of core cooling** was being tested, uncovery of the core simulator occurred, causing the test to be prematurely terminated to prevent core simulator overheating. The relevancy of this result is that core simulator uncovery was not expected to occur.

BACKGROUND

Recent licensing proceedings (in particular TMI-1 restart hearing) have focused on the ability of PWRs to remove decay heat using "feed and bleed" cooling in the event of loss of all feedwater.

Although neither the staff nor the licensees or applicants have ever relied upon feed and bleed in order to meet the Commission's regulations, and although the staff has never concluded that all plants with installed HPI and safety-relief systems can successfully "feed and bleed," we believe that there is an inherent margin of safety attributable to a feed and bleed capability.

CONTACT: M. Keane
X28957

*Semiscale is a test facility approximately 1/1500th volume-scaled to a typical Westinghouse 4-loop PWR.

**"Feed and Bleed" refers to a mode of core cooling in which all feedwater (main and auxiliary) is not available, and decay heat removal is accomplished by adding coolant inventory with the HPI system, and removing decay heat energy through the safety or relief valves.

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SEMISCALE RESULTS AND RELIANCE

The Semiscale test simulated a loss of all feedwater with a complete dryout of the steam generator secondary side. The scaled PORV was opened as the recommended action to depressurize the system to below the HPI pump shutoff head to allow the HPI flow to restore primary coolant inventory. Prior to achieving an equilibrium thermal hydraulic condition for core cooling, the core simulator rods began to heat up excessively. This caused the test to be prematurely terminated to protect the core simulator rods.

The relevance of this result is that core simulator uncovery was not expected to occur. Pretest predictions were not performed for this particular test, and it is not known if any new phenomena occurred that were not capable of being predicted by current analysis computer codes. (The expectation that no core simulator uncovery would occur was based on engineering judgment and not on detailed calculations.) Thus, the applicability of these results to the feed and bleed capability of large PWRs is unknown. Further information is presented in the RES memorandum from Bassett to Speis covering this topic which is attached.

A related test has been run in LOFT, which is approximately 1/60th in volume compared to a typical Westinghouse 4-loop plant. In this test, the PORV was latched open and the system depressurized to below the HPI shutoff head (the HPI was not allowed to inject for other testing purposes). There was no indication of core uncovery.

Westinghouse has also performed an analysis that indicates that with low-head HPI, core uncovery would occur if feed and bleed is not initiated before the steam generators have dried out. An analysis of a PWR at the Semiscale test conditions is part of the resolution plan.

RESOLUTION PLAN

To fully understand the relevance of the test, the following resolution plan will be pursued by the Office of Nuclear Regulatory Research:

1. Analysis of the Semiscale test, including study of the data and RELAP 5 computer code calculation of the experiment.

2. Analysis of the atypicality of Semiscale as compared to the PWR for this type of operation.

3. Analysis of a PWR for the same conditions that existed during the Semiscale test with the RELAP5 code.

CONCLUSIONS

Based on our assessment of the results to date and on the criteria of Office Letter Number 19, we do not believe that a board notification is warranted. However, due to the interest in feed and bleed cooling in recent licensing proceedings, we believe it is in the best interest of the regulatory process to inform the licensing boards of this recent

AUG 3 0 1982

test result. We do not believe that sufficient information is available yet to draw any conclusion from the results. We also do not believe that these results adversely impact our present staff position regarding reliance on feed and bleed cooling. We intend to pursue resolution of the issue with RES. We expect this resolution by approximately September 30, 1982 and we will inform the boards of our conclusions at that time.

Roger J. Mattson

Roger J. Mattson, Director
Division of Systems Integration

Enclosure: As Stated

cc:

H. Denton	E. Case
S. Hanauer	G. Knighton
R. Minogue, RES	D. Ross, RES
O. Bassett, RES	H. Sullivan, RES
R. Landry, RES	G. D. McPherson
N. Lauben	T. Marsh
W. Hodges	G. Mazetis
W. Lyon	R. Barrett
M. Keane	T. Novak
G. Lainas	W. Jensen

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

AUG 23 1982

MEMORANDUM FOR: Themis P. Speis, Assistant Director for Reactor Safety
Division of Systems Integration

FROM: O. E. Bassett, Director
Division of Accident Evaluation
Office of Nuclear Regulatory Research

SUBJECT: FEED AND BLEED EXPERIMENTS IN SEMISCALE

As you are aware, RES has performed a Semiscale feed and bleed experiment (S-SR-2) at the request of NRR. Results of this test indicate that difficulty was experienced in maintaining a steady-state feed and bleed condition without uncovering the heater rod bundles. Several members of your staff have had questions as to how this relates to the PWR feed and bleed operations being purposed for several plants. These questions are now being addressed by EG&G Idaho, Inc., while taking into account the atypicalities of Semiscale, as they might affect the feed and bleed behavior.

As a start, EG&G have provided a letter report of several steady-state calculations they have performed in an effort to help understand the Semiscale data (see enclosure). These calculations included a study of the sensitivity of the results to core power, break flow quality, surge line and pressurizer geometry, and availability of equipment, and were extended to parametric values typical of a commercial PWR. The report concludes that there are large uncertainties in predicting the satisfactory performance of feed and bleed in the steady-state, but that the Semiscale experiment does not point to the existence of a definite problem regarding the satisfactory performance of feed and bleed in a PWR. The Semiscale results still must be analyzed to determine the extent that the atypicalities effect the results and to put them in proper perspective. This analysis work is currently in progress and should be completed in September 1982.

The Division of Accident Evaluation has concluded that the Semiscale results have not produced new and unique results that indicate a PWR would have a definite problem regarding feed and bleed. Therefore, we do not recommend a board notification. We have concluded that the Semiscale results should be carefully analyzed to determine that relevance to PWR feed and bleed transients and this work is now in progress. We

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should be able to provide you with a more complete answer to the relevance of the Semiscale experiment and our investigation into the feed and bleed transient in a PWR by late September 1982.

Charles Kelber, for
O. E. Bassett, Director
Division of Accident Evaluation
Office of Nuclear Regulatory Research

Enclosure: Ltr fm North to
Tiller dtd 08/06/82

cc w/encl:
L. H. Sullivan, EPB
R. R. Landry, EPB
B. Sheron, NRR

EG&G Idaho, Inc.
P.O. BOX 1625, IDAHO FALLS, IDAHO 83415

August 6, 1982

Mr. R. E. Tiller, Director
Reactor Operations and Programs Division
Idaho Operations Office - DOE
Idaho Falls, ID 83415

PRIMARY COOLANT SYSTEM FEED AND BLEED - PM-137-82

Dear Mr. Tiller:

At the request of the Nuclear Regulatory Commission the Semiscale Program recently conducted experiments designed to investigate the feasibility of primary coolant system (PCS) feed and bleed as a means of rejecting decay heat in the absence of steam generator heat removal. The results and preliminary analysis of the experiments suggested that a reasonable uncertainty may exist in the ability to effect stable PCS feed and bleed. Since current pressurized water reactor emergency operating guidelines call for primary feed and bleed under certain abnormal conditions, it was considered of some importance that the general subject of feed and bleed be studied in some depth and that the Semiscale results be carefully analyzed so that they might be interpreted in the proper perspective. To this end, the Semiscale Program is currently engaged in an extensive analysis effort involving both full-scale plants and experimental results (i.e., Semiscale and LOFT). The purpose of this letter is to provide a brief overview of our analysis of feed and bleed to date, including the recent Semiscale results.

INTRODUCTION

Primary coolant system feed and bleed in a pressurized water reactor becomes a necessary decay heat removal mechanism in the unlikely event that all secondary heat removal capability is lost. While there exist numerous scenarios that could lead to this situation, the focus of the present analyses is the feasibility of achieving a favorable coolant and energy balance within the primary coolant system under conditions in which:

1. the reactor has scrammed

2. the steam generator secondaries are completely depleted of coolant

3. the high pressure injection system (ECCS) is operative in an undegraded condition

4. the pressurizer heaters are inactive

5. the pressure-operated relief valves(s) (PORVs) are operative

6. primary recirculation pumps are off

Feed and bleed would commence when the PORV(s) were opened (bleed) and high pressure injection began (feed). The passage of steam out the PORV(s) provides for the rejection of decay heat while the introduction of ECCS coolant provides makeup for the resultant coolant loss.

The remainder of this letter examines the general aspects of primary feed and bleed operation and the thermal-hydraulic phenomena that govern it. The predicted system response is outlined first and the effects of uncertainties are illustrated with an example from actual plant parameters. Next the key thermal-hydraulic phenomena that influence the uncertainties are discussed. Finally, the data from Semiscale experiments is briefly presented to demonstrate system signatures and response during an actual experiment.

THEORETICAL FEED AND BLEED OPERATING PRESSURE RANGE

A simple examination of the mass and energy transfer pathways associated with feed and bleed results in the conclusion that feed and bleed is theoretically possible within a certain band of pressure (see Figure 1). The governing parameters which determine this pressure band are decay heat level, HPIS flow rate, and PORV flow rate (and enthalpy flow rate). Except for the core decay heat level the remaining parameters are functions of primary systed pressure. The lower bound of the operating band represents the minimum pressure at which the PORV can pass enough steam[a] (with the coolant replaced by ambient temperature water) to remove sufficient energy from the system. Steady-state operation below this pressure without additional energy removal paths is not possible. Operation at a pressure above the lower bound may be accomplished by cycling the PORV open and closed within a desired pressure band.

An upper pressure bound to the steady-state operating band is defined by a balance between the PORV average coolant removal rate and the HPIS coolant injection rate. The average PORV coolant removal rate is simply defined as the core power divided by the difference between inlet and outlet enthalpies:

a. Base case conditions assume that 100% quality steam is discharged through the PORV. The affect of varied quality is examined later.

R. E. Tiller
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$$\dot{m}_{AVG} = \dot{Q}_{core}/(h_{out} - h_{in})$$

(This assumes that the coolant removed through the PORV is replaced with ambient temperature water. Actually a coolant deficit exists at pressures higher than the upper bound and a steady-state condition cannot exist due to a continual loss of system coolant inventory.) Below the upper bound the system mass inventory can theoretically be maintained within a desired operating range by either throttling the HPIS or cycling it on and off.

UNCERTAINTIES ASSOCIATED WITH STEADY-STATE OPERATING PRESSURE BAND

In practice the curves discussed above are not well defined due to several uncertainties. Subject to the greatest uncertainty are the PORV mass removal and energy removal curves. The mass flow through the PORV is dependent on the fluid conditions at the top of the pressurizer. If the pressurizer is near liquid full the flow through the PORV will be a mixture of liquid and vapor. At a given system pressure this results in greater mass flow than for saturated steam flow through the PORV. The result of having two-phase flow through the PORV is therefore to lower the upper bound pressure.

Uncertainty also arises in the PORV energy removal curve due to two-phase flow. With decreasing quality the energy removal per unit mass decreases while the mass discharge rate increases. Depending upon the quality the energy removal rate at a given pressure may be less than or greater than that for saturated steam. The lower bound of the operating band will vary accordingly.

Another significant variable that affects the width of the operating band concerns the actual heat load that must be rejected through the PORV. Core decay heat decreases continually with time after shutdown. Also, if some additional heat sink exists, such as environmental heat loss or residual water in the steam generator secondaries, the heat load required to be rejected through the PORV will decrease. Referring to Figure 1, this results in lowering the core power line and thus lowering the bottom end of the operating band, and also lowering the PORV average mass flow curve, thus raising the upper end of the operating band.

The final uncertainty addressed here is the uncertainty associated with the HPIS injection curve. The effects here are rather clear; a lower injection rate will lower the upper end of the operating band.

The quantitative effects of the uncertainties and/or variances discussed above are illustrated in Figures 2 through 6. For these examples the curves were generated using data obtained from the Zion 1 nuclear generating plant,[1,2,3] a 3411 MW(t) pressurized water reactor. Figure 2 shows a primary feed and bleed map for a 2% decay heat power level. A steady-state operating band is seen to exist between 7.5 and 14 MPa. A decay heat level of 2% of full power is typical of the time period from about 10 min to 20 min after shutdown. Figure 3 is a similar curve, but here no makeup pump injection is assumed; only the HPIS pumps were assumed to be operating. The HPIS pumps are shown to deadhead at about 10.3 MPa. For this case no operating band exists, since at the minimum pressure where the PORV can remove the energy there is a mass deficit between the PORV coolant removal and the HPIS injection capacity.

Figure 4 shows the primary feed and bleed map for 1-1/2% full power, a decay heat level typical of the period from 1/2 to 1 hr after shutdown, and for only HPIS injection. Comparison to Figure 3 shows that the reduction in core power and corresponding PORV average mass flow both act to establish a steady-state operating band.

The above curves are based upon the assumption that 100% quality steam exists at the PORV. Figure 5 shows the sensitivity of the PORV energy removal curve to lower qualities as determined with the HTM flow model. Since the energy removal per unit mass decreases while the mass flow rate increases the energy removal rate initially decreases with decreasing quality. However, since the mass flow rate increases substantially with decreasing quality the energy removal rate eventually increases. The effect on the lower operating bound pressure is not large; however the large increase in PORV mass flow rapidly lowers the upper end of the band. As an example, for the conditions used in Figure 4 the operating band does not exist at qualities below approximately 75% (see Figure 6).

The foregoing analysis is useful, in that it provides a basis for examining the feasibility of feed and bleed and for quantitatively assessing the effects of uncertainties or variations in the bounding parameters. However, it does not address transient behavior that may have an important bearing on the ultimate viability of primary feed and bleed. In particular, it should be evident that there exists some uncertainty regarding the ability to safely bring the primary coolant system to within the "feasible" operating pressure band without sustaining unacceptable coolant loss in the process. Factors which bear on this transient process include the primary coolant system state at the initiation of an attempt to feed and bleed, and the nature of the coolant discharged through the PORV(s) in depressurizing the system to within the operating band. These questions can only be addressed through experimentation and the use of computer code analyses.

FACTORS AFFECTING PORV DISCHARGE

Of the factors previously discussed the largest uncertainty affecting the feed and bleed operating band arises from the influence of two-phase PORV flow. The mass flow through the PORV is dependent on upstream fluid conditions at the top of the pressurizer. Several factors contribute to establishing pressurizer fluid conditions. The ones discussed here are: transient vs steady-state behavior, primary coolant system conditions, pressurizer/surge line geometry, and surge line orientation.

Transient vs Steady-State Conditions and Primary Inventory

If feed and bleed is not initiated soon after loosing the secondary heat sink the primary liquid swell will fill the pressurizer and collapse the steam bubble. Several conditions may form or sustain a vapor bubble at the top of the pressurizer. A vapor bubble can be produced by loss of pressurizer liquid inventory, heating of the fluid to saturation, and/or depressurization. In the present study the pressurizer heaters are assumed to be non-operational and direct heating is therefore precluded. In a transient depressurization, liquid flashing in the pressurizer will tend to create a high quality region near the top as long as the fluid in the pressurizer is the hottest in the system. However, the liquid swell that accompanies bulk flashing will tend to decrease the quality at the top of the pressurizer. For either a quasi-steady-state situation, or in a transient once the original pressurizer inventory has been replaced with coolant from the hot leg, the PORV fluid conditions are dependent upon the conditions in the hot leg. If the coolant lost through the PORV is replaced by low quality fluid the mass discharge out the PORV will remain fairly high. This will occur until the primary system inventory is reduced enough to cause significant voiding in the hot leg. Once significant hot leg voiding occurs pressurizer/ surge line geometry and orientation come into play as described below.

Pressurizer/Surge Line Geometry

For a given vapor volume a pressurizer with a large length-to-diameter ratio would have a "tall" void height relative to a pressurizer with a smaller ratio, in addition to also having a smaller cross-section. A steam bubble of greater height would tend to enhance separation from the vapor of liquid droplets created by bubbles breaking through the liquid surface, due to the greater wall surface area and reduced potential for droplets being thrown upward into the high vapor velocity area near the PORV line entrance[a]. However, since vapor must by necessity pass through the pressurizer liquid from the surge line to the PORV a large L/D would tend to promote liquid swell and droplet entrainment due to the smaller cross-sectional area.

In any case, the influence of the pressurizer geometry may be minimized by the preclusion of counter-current flow in the surge line. Even if a liquid/vapor separation mechanism did exist in the pressurizer, typical surge line velocities are well above flooding limits. Therefore, the liquid could not drain back to the loop and would continue to be stored in the pressurizer until the PORV discharge quality self-adjusted to accommodate removal of the mass. It therefore appears necessary to have high quality steam supplied from the hot leg in order to have high quality PORV discharge.

Surge Line Orientation

If hot leg voiding does occur, the orientation of the surge line would influence the primary system inventory at which high quality steam entered the pressurizer. Surge line-to-hot leg connections of various orientations, from horizontal side entrance to vertical top entrance, are used in current PWR's. With the top entrance line, and quiescent hot leg conditions, minimal hot leg voiding is necessary to allow high quality surge line flow. With a side entrance line the hot leg pipe liquid level must drop much lower before high quality flow begins. In either case the surge line flow may still be varied significantly if non-quiescent conditions exist that disrupt stratified flow, such as when primary recirculation pumps are turned on, or a transient depressurization is occurring.

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a. For typical PWR pressurizer dimensions the vapor velocity (due to an open PORV) in a vapor filled cross-section is on the order of 1 ft/s which presents little chance of droplet entrainment.

CONCLUSIONS BASED ON SIMPLIFIED ANALYSIS

Based on the foregoing discussion it is concluded that a simplified approach to determining the feasibility of primary feed and bleed in a pressurized water reactor lies in the mapping of energy and mass flows. Moreover, this technique can be used to quantitatively assess the sensitivity of the operating pressure band to variations in the boundary conditions of ECCS flow, PORV flow, and decay heat. It is evident that plausible variations and uncertainties in these parameters can lead to the elimination of a steady-state operating pressure range. Principal among these uncertainties is the coolant discharge through the PORV. The predictability of this single parameter is subject to much greater uncertainty than either decay heat or ECCS flow.

Irrespective of the existence of a theoretically feasible operating pressure band, there remains the question as to whether the reactor system can be safely maneuvered into this pressure range. In this regard it is clear that a dependence must be placed on computer code analyses (with suitable verification) and adequate supporting experimental data. Such analyses and/or experiments should examine the plausible scenarios which lead the operator to commence primary feed and bleed, since the initial condition of the primary coolant system (particularly inventory) will have a significant effect on the outcome.

Furthermore, it would appear to be a useful exercise to examine the operating map that results for each set of individual PWR plant parameters. The operating map represents an ultimate statement as to whether feed and bleed is possible, and is the starting point for examining specific design features that bear on the operating bounds.

RESULTS FROM SEMISCALE EXPERIMENTS

An experiment was conducted in the Semiscale Mod-2A facility to evaluate system behavior during primary feed and bleed operations. Figure 7 shows the primary feed and bleed operating map representing boundary conditions used for the experiment. It is seen that these parameters define a steady-state operating band between 7.1 and 8.2 MPa. Several attempts were made to establish steady-state feed and bleed within the operating band. While it was possible to maintain pressure control by cycling the PORV, measurements showed a continuous loss of primary coolant inventory due to a low quality discharge out the PORV. The phenomena that led to this behavior are described below.

Single-phase, liquid full conditions were established at a pressure of 15.2 MPa. At initial conditions the pressurizer heaters were used to control pressure with a small steam bubble in the pressurizer. Some subcooling existed in parts of the loops due to the lack of natural circulation resulting from having empty secondaries. At 15000 seconds (test time) pressurizer heaters were turned off and the PORV was latched open. Figure 8 shows that the system rapidly depressurized down to approximately 8 MPa. This corresponded closely to the saturation pressure of the coldest fluid in the loops. As seen from the pressurizer collapsed liquid level curve, flashing of the hot pressurizer fluid initially resulted in substantial voiding of the pressurizer. Referring to Figure 9 it is seen that this is reflected in the PORV mass discharge rate. Following a brief initial mass flow surge the flow out the PORV agreed closely with the predicted steam flow rate for 100% quality. The steam bubble depleted after approximately 250 s (Figure 8) and the PORV mass flow rate increased to approximately 5 times the steam flow rate (Figure 9). As seen in Figure 9, the mass flow rate out the PORV appeared to be dependent upon the conditions in the hot leg. Once substantial voiding of the hot leg occurred the flow out the PORV began to agree with the predicted steam flow rate, in spite of the fact that the pressurizer remained nearly liquid full (Figure 8). (The pressurizer surge line in the Mod-2A system is connected to the side of the hot leg.)

At the time when sufficient primary coolant inventory was finally lost so as to void the hot leg the core was still adequately covered and cooled. As seen in Figure 10, there was still a small deficit in the mass injected into the system with the HPIS relative to the PORV mass discharge rate. The result was then a very slow continued loss of mass which led to eventual uncovery of the core at about 17000 seconds.

The importance of the Semiscale results lies in demonstrating the dominance of the PORV discharge rate on primary feed and bleed capability and the dependence of the PORV discharge on hot leg conditions, and consequently system inventory. The inability to maintain system inventory once the PORV mass flow rate dropped to reflect steam flow is subject to experimental uncertainties, since the steady-state operating band of Figure 7 is rather narrow. Uncertainties exist in the actual PORV orifice characteristics, HPIS injection rate and the measurement thereof, system heat loss[a], and fluid leakage.

The observed PORV discharge relation to hot leg conditions however, lies outside the effects of uncertainties discussed above. The questions that need to be addressed in interpreting and extrapolating the results are largely related to the geometry effects. The Mod-2A system has a short pressurizer relative to the desired scaling of L/D. The L/D effects on liquid vapor separation must be analyzed. The surge line needs to be evaluated also, mainly with regard to the influence of the side entry to the hot leg as opposed to other designs.

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a. Core power was augmented to compensate for the best estimate system heat loss.

CONCLUSIONS BASED ON SEMISCALE EXPERIMENTS

In and of themselves, the results from the Semiscale experiments do not point to the existence of a definite problem regarding primary feed and bleed. But they do tend to support a concern about the relative tenuousness of the process. Further analysis attempting to quantify the potential experimental distortions and their effect on the results is now in progress. These analyses, along with the results of analysis of existent LOFT data, and computer code calculations of the Semiscale experiment and a full-scale plant, will be documented in September.

Very truly yours,

Paul North

P. North, Manager
Water Reactor Research
Test Facilities Division

DJS:td

Attachments:
As stated

cc: R. W. Barber, DOE - 2
R. R. Landry, NRC - 2
W. R. Young, DOE-ID
R. W. Kiehn, EG&G Idaho

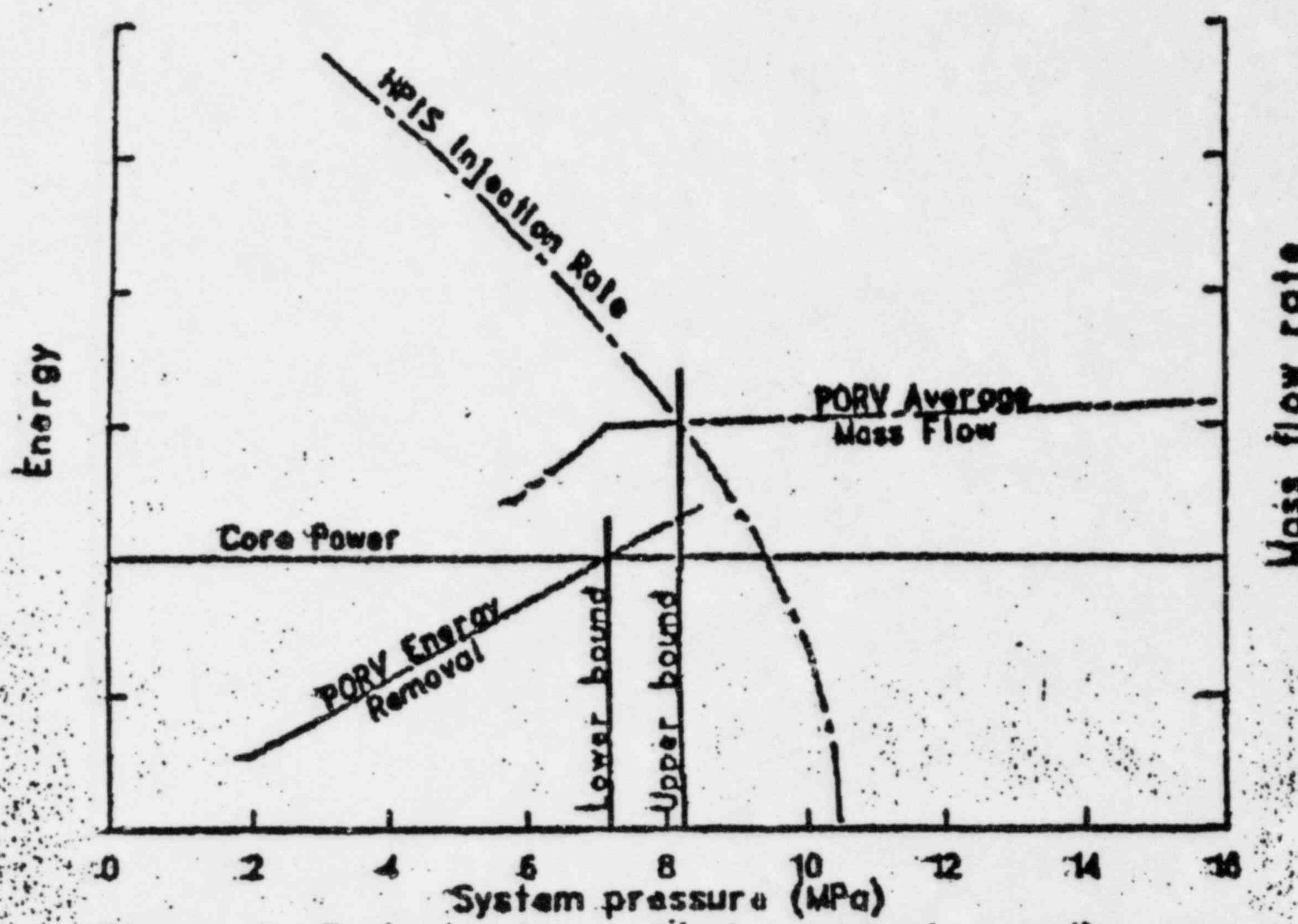
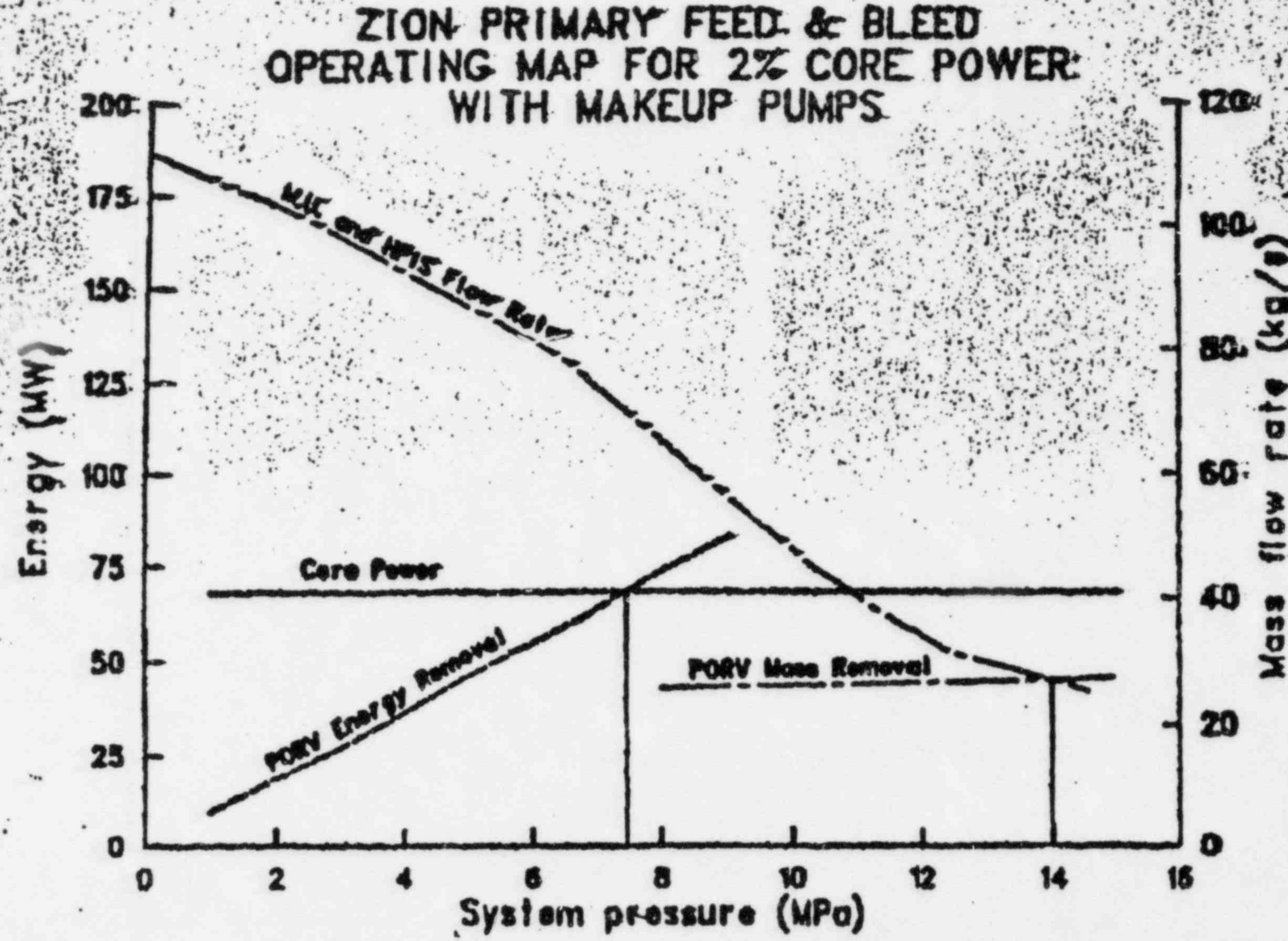


Figure 1. Typical primary feed and bleed operating map.



ZION PRIMARY FEED & BLEED
OPERATING MAP FOR 2% CORE POWER
WITH MAKEUP PUMPS
Energy (MW)
200
175
150
125
100
75
50
25
0
Mass flow rate (kg/s)
120
100
80
60
40
20
0
System pressure (MPa)
0
2
4
6
8
10
12
14
16
Core Power
PORV Energy Removal
PORV Mass Removal

Figure 2.

ZION PRIMARY FEED & BLEED OPERATING MAP FOR 2% CORE POWER

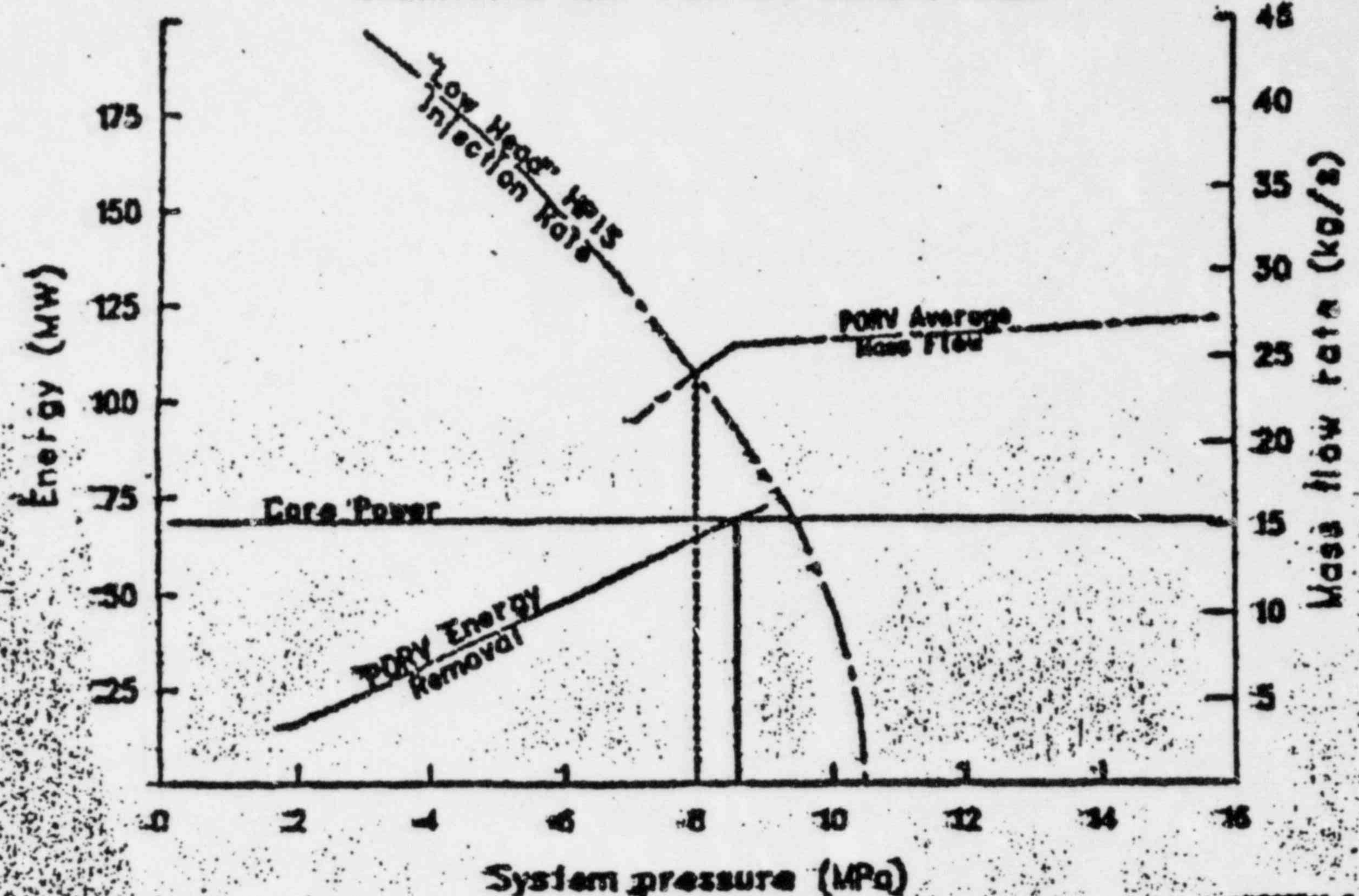


Figure 3.

ZION PRIMARY FEED & BLEED
OPERATING MAP FOR 1.5% CORE POWER

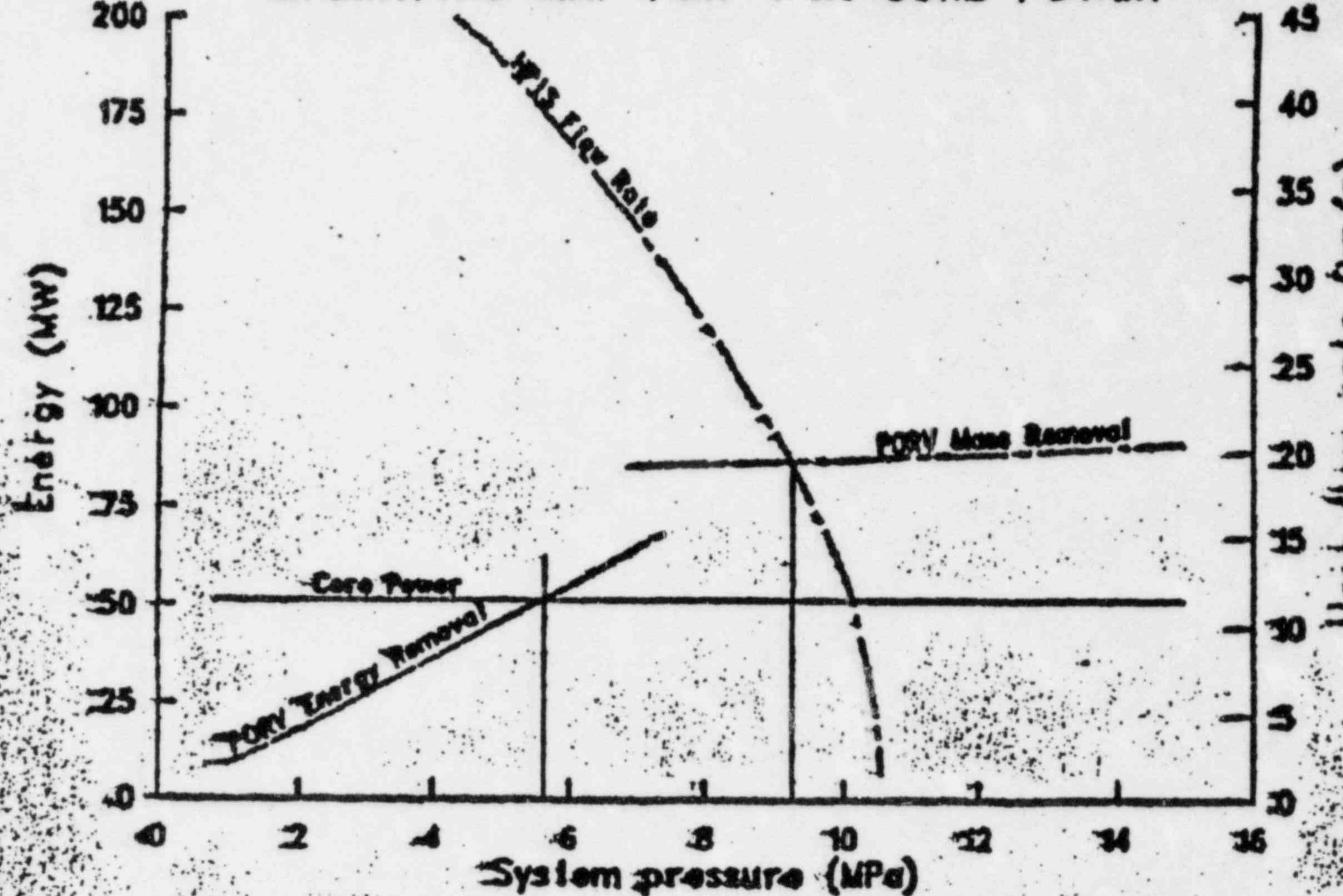


Figure 4.

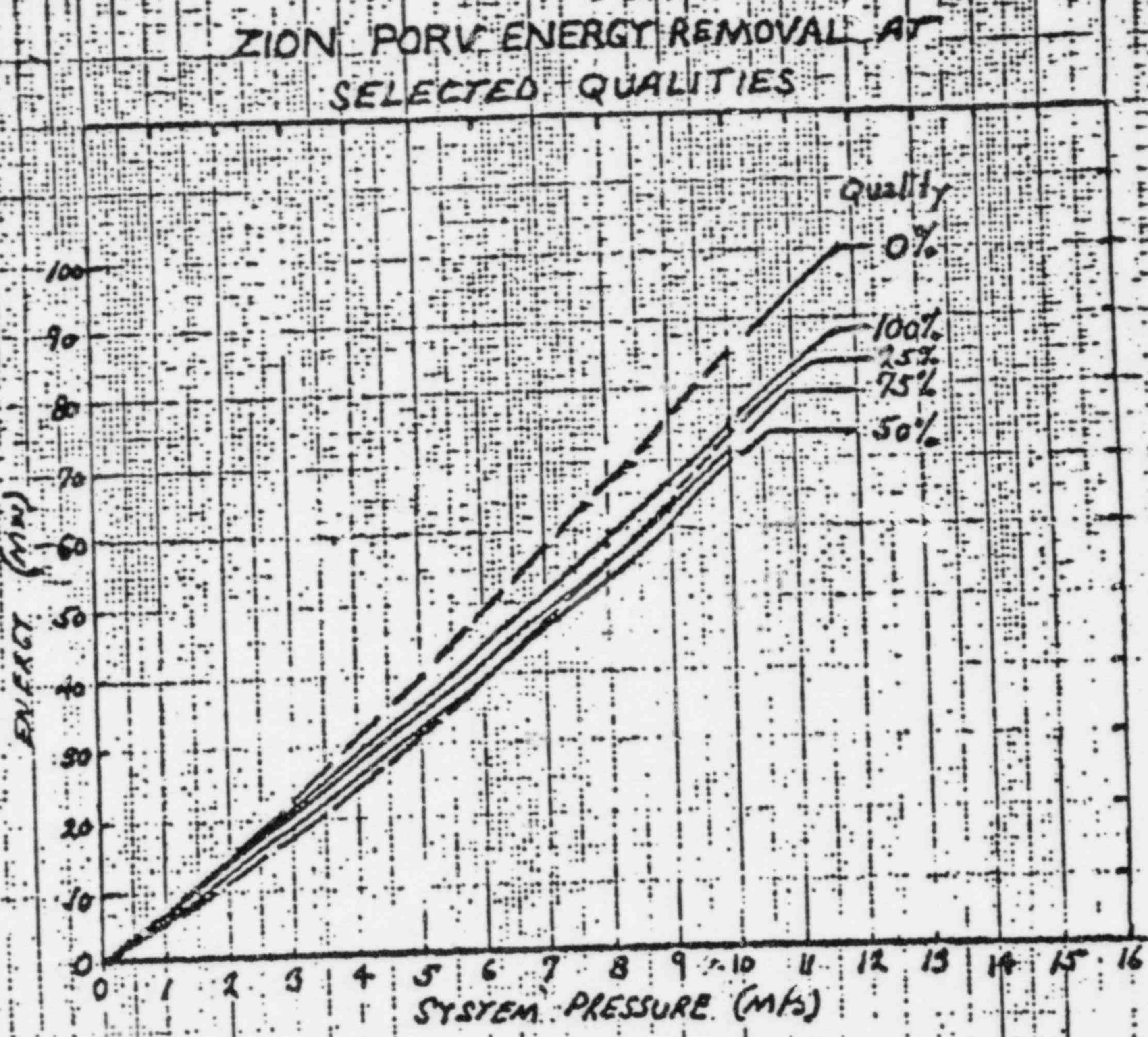


Figure 5.

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ZION PRIMARY FEED & BLEED OPERATING MAP FOR 1.5% CORE POWER AND 75% QU..LITY PORV STEAM FLOW

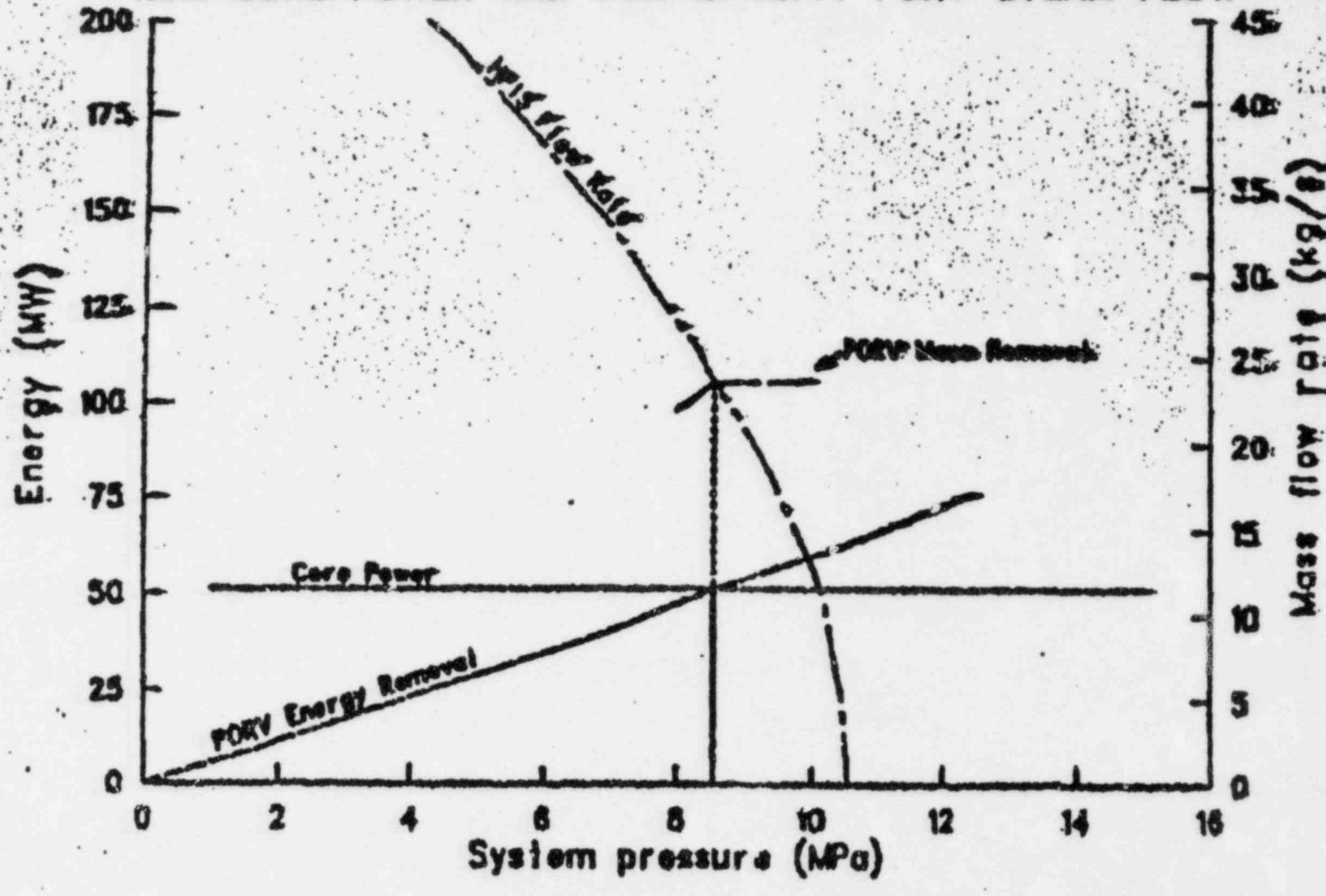
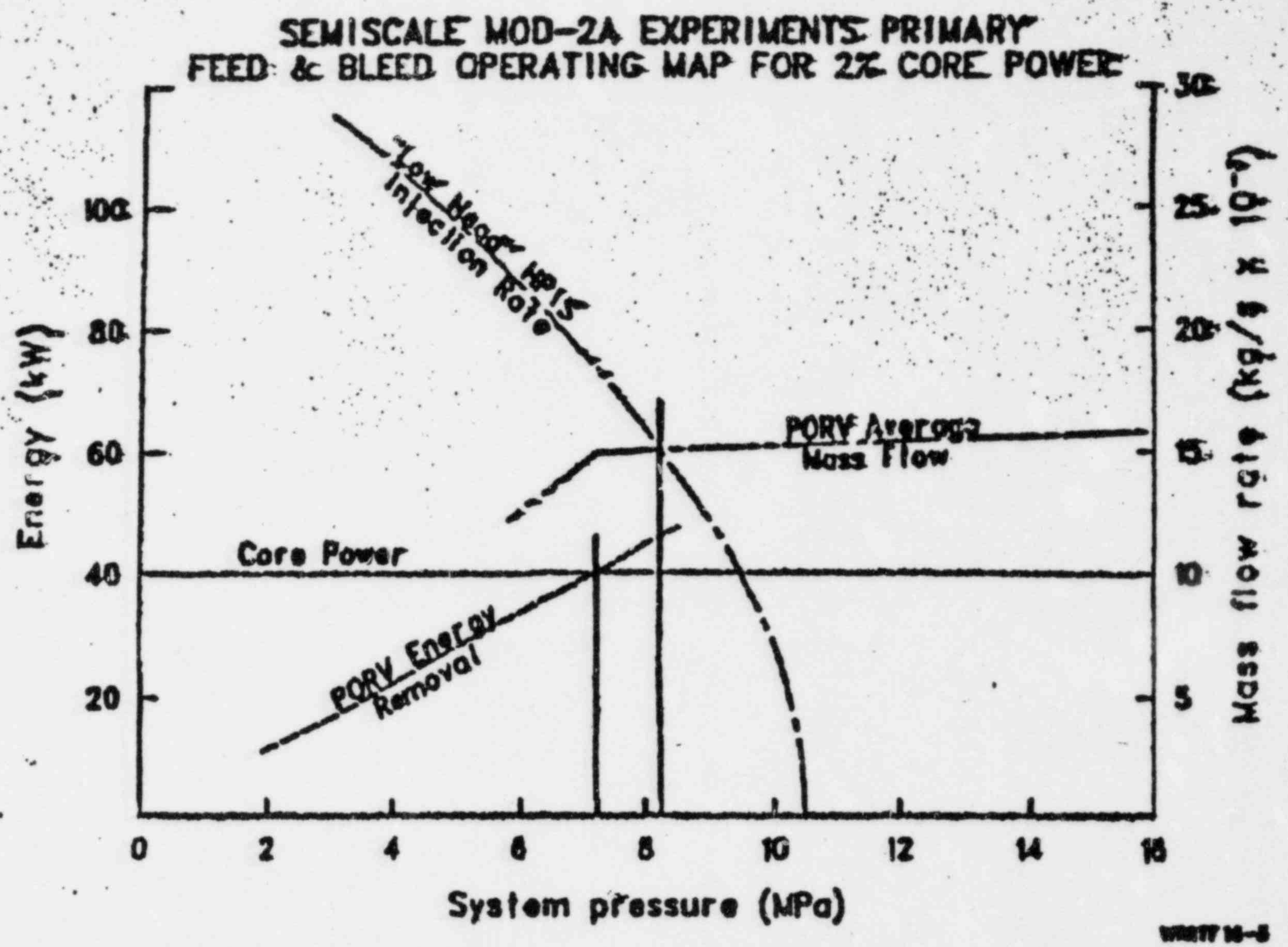


Figure 6.



SEMISCALE MOD-2A EXPERIMENTS PRIMARY
FEED & BLEED OPERATING MAP FOR 2% CORE POWER
Low Head HPIS
Injection Rate
PORV Average
Mass Flow
Core Power
PORV Energy
Removal
Energy (kW)
Mass flow rate (kg/s x 10⁻³)
System pressure (MPa)
0
2
4
6
8
10
12
14
16
20
40
60
80
100
5
10
15
20
25
30

Figure 7.

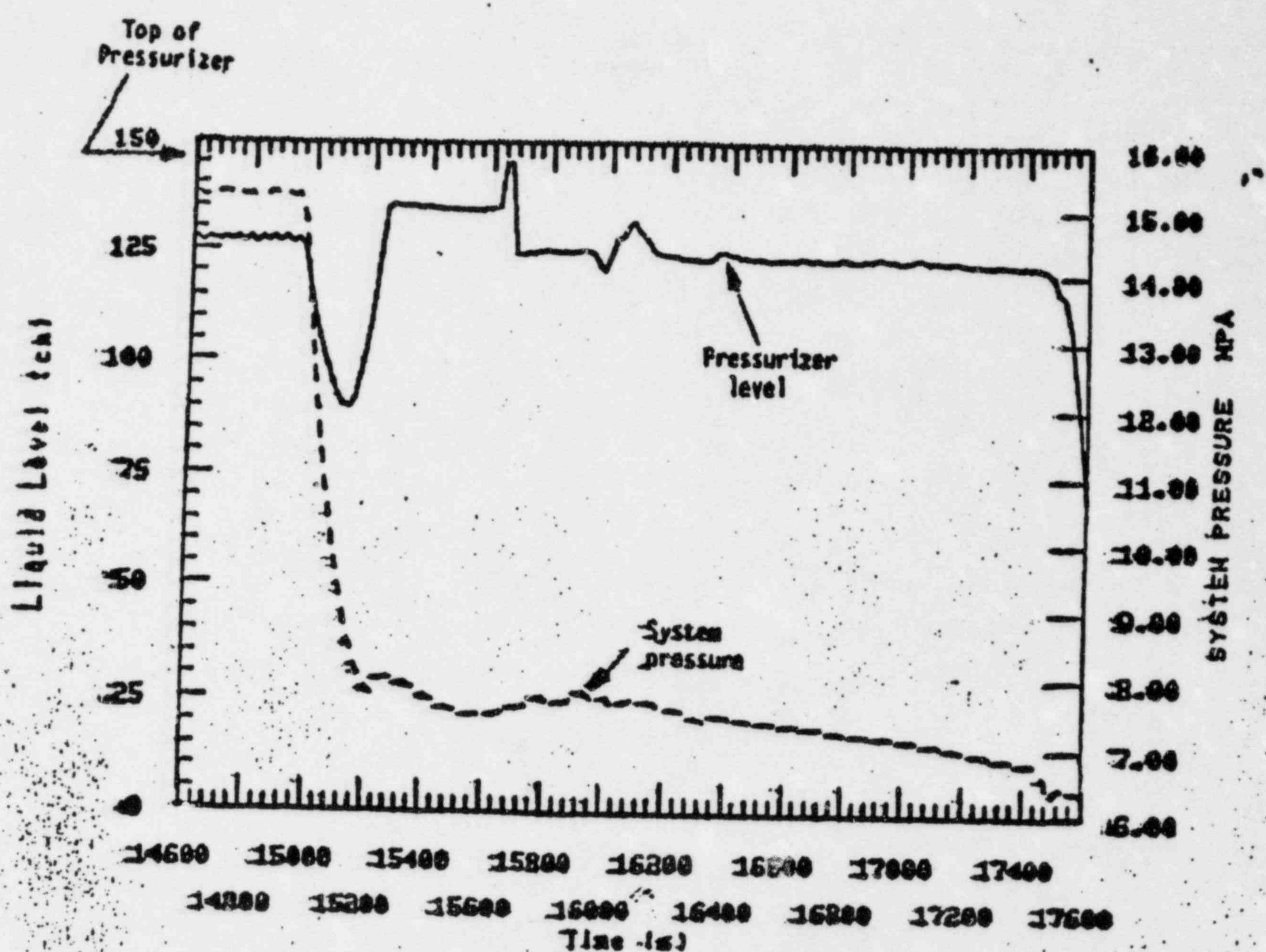


Figure 8. Comparison of pressurizer collapsed liquid level and system pressure for Test S-SR-2 depressurization.

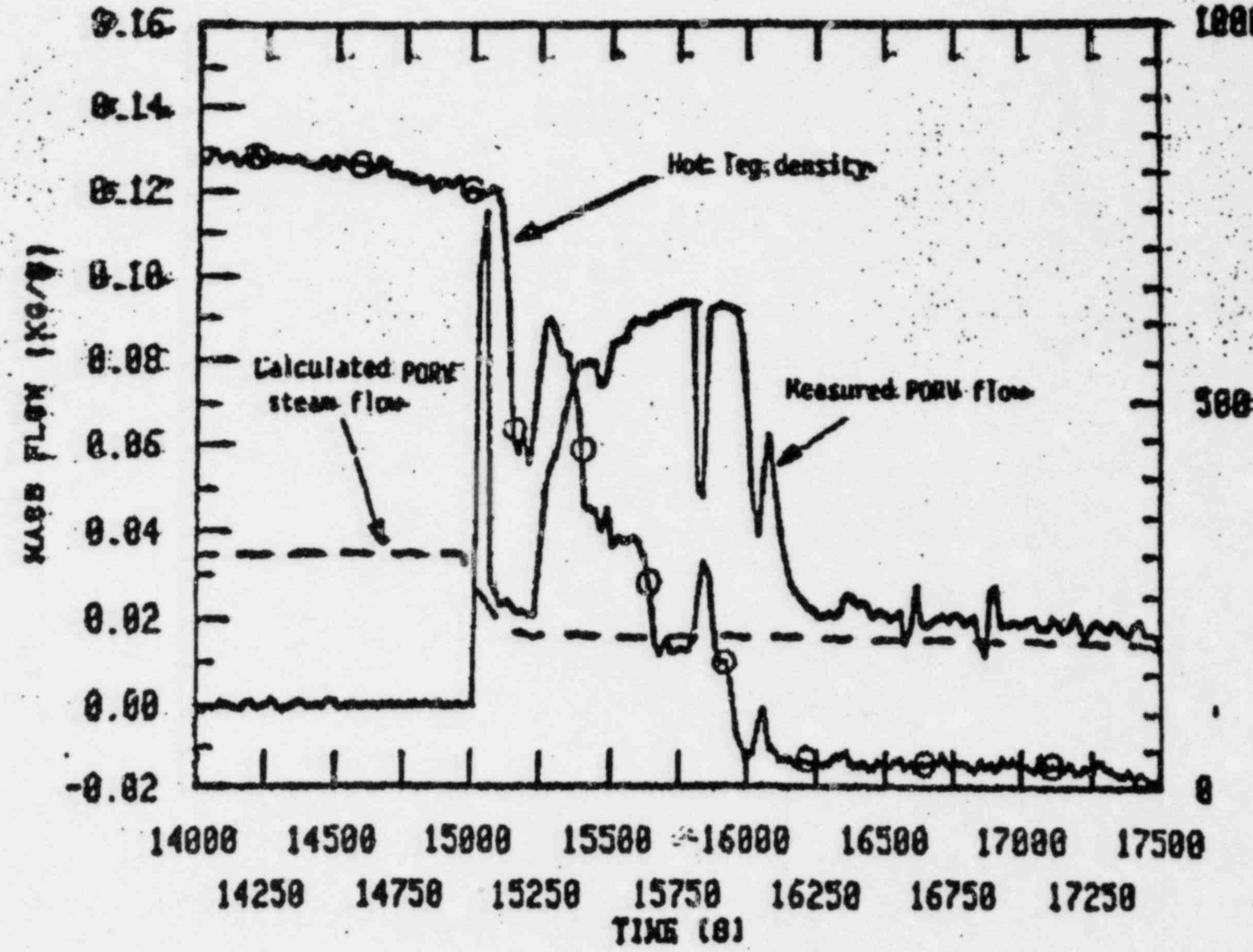


Figure 9. Influence of hot leg density on PORV flow rate.

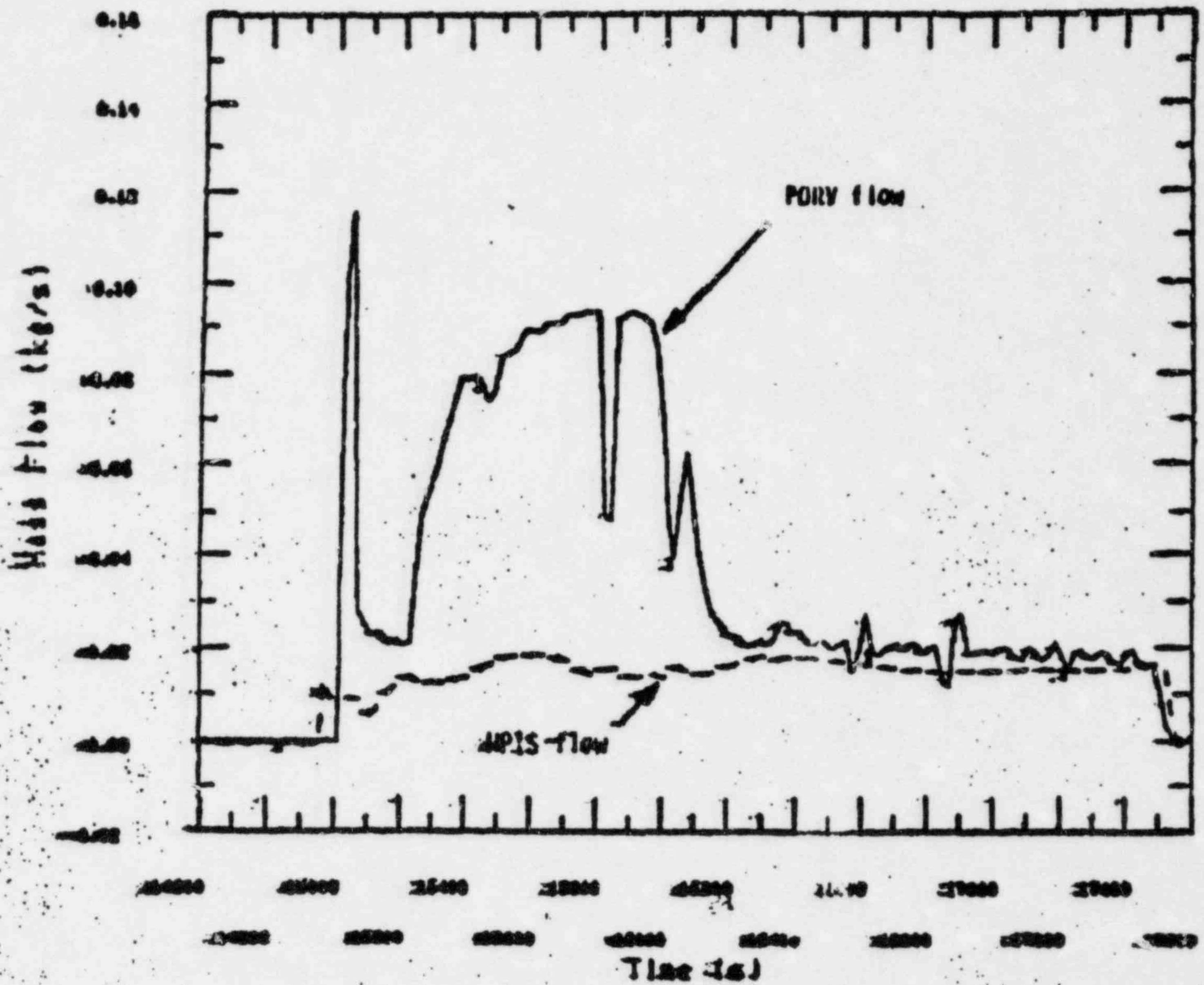


Figure 30. Comparison of PORV and HPIS flow for Test S-SR-2 depressurization.

REFERENCES

1. "Zion Station Final Safety Analysis Report, Commonwealth Edison Company, 1973.
2. Zion System Descriptions, Commonwealth Edison Company.
3. Information obtained from the reference file from the BE/EM study at the Idaho National Engineering Laboratory.
4. G. B. Wallis, One-dimensional Two-Phase Flow, McGraw-Hill Book Company, 1969.