UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF BRIAN W. SHERON
AND WALTON L. JENSEN, JR. CONCERNING
SEMISCALE TEST (S-SR-2) RESULTS

1. I, Brian W. Sheron, being duly sworn, state as follows:
I am Branch Chief, Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.
2. I, Walton L. Jensen, Jr., being duly sworn, state as follows:
I am a Senior Nuclear Engineer in the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. A copy of my professional qualifications is attached.
3. We are familiar with the Semiscale test (S-SR-2) which is the subject of BN-82-93 and BN-82-107, as well as the materials included with those two board notifications. We have also reviewed "Union of Concerned Scientists' Response To Board Notification BN-82-93 Concerning Semiscale Tests Of Feed And Bleed And Motion That Appeal Board Direct NRC Staff To Provide All Pertinent Documentation And Analyses." We make this affidavit in response to that UCS filing, which is referred to hereinafter as "UCS Response".

4. The "final report on the Semiscale feed and bleed tests" referred to at page 4 of the UCS Response is attached to BN-82-107.
5. The "request of NRR to RES to perform the Semiscale feed and bleed experiment" referred to at page 4 of the UCS Response was made orally as an outgrowth of a testing program discussed in a memorandum, dated April 4, 1981, from Paul S. Check to Harold R. Sullivan, which is Attachment 1 to this Affidavit. The Semiscale test (S-SR-2) which is the subject of BN-82-93 and BN-82-107 related to the heading "System Depressurization with Auxiliary Pressurizer Spray" in Attachment 1.
6. The "March 31, 1982" memorandum for Karl Kniel from Brian Sheron through Themis Speis referred to at page 4 of the UCS Response is Attachment 2 to this Affidavit. The proper date of the memorandum is March 31, 1981. UCS quotes from that document a passage which states "[f]eed and bleed, if performed, should be at a relatively low ($P \ll$ relief valve setpoints pressure)." This statement simply states the Staff's preference that feed and bleed be performed at low, rather than high, pressure in order to minimize the possibility of pressurized thermal shock. Feed and bleed can be performed successfully at TMI-1 at 2500 psi. The statement quoted is in no way inconsistent with the Staff's position in this proceeding that the safety valves can be relied upon during feed and bleed cooling.
7. An additional document related to the Semiscale test results is a September 7, 1982 note from Mary Ellen Keane and Walt Jensen to Brian W. Sheron entitled "Comparison of Westinghouse Feed and Bleed Analysis And Semiscale S-SR-2." This document is Attachment 3 to this Affidavit.

8. The time that transpired between EG&G's notice to the Staff of Semiscale test S-SR-2 and the issuance of BN-82-93 by the Staff was that necessary for obtaining the test results and associated analyses, conducting an evaluation of them and preparing, reviewing and issuing the board notification materials.
9. The Staff did not believe that a board notification was required concerning Semiscale test S-SR-2, since the test results did not adversely impact the Staff's position that feed and bleed capability provides an inherent margin of safety for defense in depth in the event of loss of all feedwater. (The Staff's position with respect to feed and bleed capability is set forth more fully in BN-82-71.) As stated in BN-82-93, however, a board notification was issued on the Semiscale test due to the interest in feed and bleed cooling in recent licensing proceedings, including this one.
10. The Staff attached to BN-82-93 those documents it considered most informative for the involved adjudicatory boards. The EG&G, Idaho, Inc. September 1982 report on Semiscale test S-SR-2 was not available to the Staff at the time Board Notification BN-82-93 was issued. The documents discussed at paragraphs 5 and 6 herein were not considered by the Staff to be of sufficient importance to warrant their inclusion among the informational matters voluntarily provided to the adjudicatory boards and the parties.
11. The Semiscale test results do not raise a significant safety issue. The relevance of the Semiscale test to feed and bleed capability was that core uncovery was not expected to occur. <u>See</u> BN-82-93. This expectation was not based on any pre-test

calculation. The Staff's analysis after the test demonstrates that a pre-test calculation of the Semiscale S-SR-2 test result would have predicted the phenomena. Had such a pre-test calculation been done, the test conditions for Semiscale test S-SR-2 might have been adjusted to better simulate those expected in a large PWR. The September 7, 1982 note from Keane and Jensen to Sheron discuss the possible atypicalities of the Semiscale test conditions for test S-SR-2.

12. The Semiscale test S-SR-2 results do not exhibit any new phenomena and can be adequately predicted by existing computer codes. Neither this Semiscale test nor the analyses of it conducted by EG&G and the Staff provide evidence that feed and bleed cooling will not work at TMI-1 or that its viability at TMI-1 is questionable.

13. The Staff is satisfied that the RELAP-5 code accurately calculates both the overall system response and local responses. Additional calculations were performed by the Staff in September 1982 utilizing the RELAP-5 computer code, which is the same code used by EG&G to correlate Semiscale. The staff did not write a report on its calculation; however, the curves drawn by the computer are attached. The staff concluded the following from these calculations.

    a. The core remained covered and cooled.

    b. The safety valves were only required to be opened for a fraction of the time. They opened and closed throughout the analysis thereby exhibiting excess relief capability.

    c. No excessive reactor vessel cooling which might produce pressurized thermal shock was calculated for the 5000 second duration of the analysis. This effect was attributed to the mixing action by the core barrel vent valves.

d. Less coolant loss from the reactor coolant system was calculated in the NRC calculation than in the B&W (Licensee Exhibit 9) calculation indicating that more water was available for core cooling in the NRC calculation than in the B&W calculation.

14. The UCS Response questions the adequacy of operator training and procedures for feed and bleed cooling and states that the operator actions required are complex. Operator action to initiate and maintain feed and bleed at TMI-1 was discussed by Staff witness Boger at the TMI-1 restart hearing in response to Board Question 6d. The required actions to initiate and maintain feed and bleed are to depress two push buttons on the main control board. Other actions such as opening the PORV and block valve, eventual throttling of ECCS and attempting to restore feedwater are recommended by the procedures but are not required for core cooling. Feed and bleed operation at TMI-1 involves little operator action and should require little additional training.

15. Contrary to the statements made on the bottom of page 7 and the top of page 8 of the UCS Response, analytical evidence is on the record that TMI-1 can feed and bleed at 2500 psig. This analysis is contained in Licensee's Exhibit No. 9. (B&W Document 86-1103585-00, "System Response to total loss of SG Heat Sink, August 7, 1979.")

Brian W. Sheron

Brian W. Sheron

Walton L. Jensen Jr.

Walten L. Jensen, Jr.

Subscribed and sworn to before me
this 25th day of October, 1982

Judy L. Butts

Notary Public

My Commission expires: July 1, 1986

STATEMENT OF PROFESSIONAL QUALIFICATIONS

BRIAN WALTER SHERON

My name is Brian Walter Sheron. I graduated from Duke University in Durham, North Carolina, in 1969, with a Bachelor of Science in Engineering (B.S.E.) majoring in electrical engineering. I received my Masters Degree (M.S.) in nuclear engineering in 1971 and my Doctor of Philosophy (Ph.D) degree in nuclear engineering in 1975, both from the Catholic University of America in Washington, D. C.

I joined the Atomic Energy Commission in 1973 in the Division of Reactor Development and Technology and worked on the LMFBR. I joined the Nuclear Regulatory Commission in 1976 as an engineer in the Analysis Branch in the Division of Systems Safety. In 1980, I was assigned to the Reactor Systems Branch, Division of Systems Integration, and was promoted to a Section Leader in the Branch that year. In February of 1982, I was promoted to Chief of the Reactor Systems Branch. In this capacity, I supervise the activities of approximately 33 engineers in the areas assigned to the Branch.

WALTON L. JENSEN, JR.

PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In this position I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems.

From June 1979 to December 1979, I was assigned to the Bulletins and Orders Task Force of the Nuclear Regulatory Commission. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

From 1972 to 1976, I was assigned to the Containment Systems Branch of the NRC/AEC, and from 1976 to 1979, I was assigned to the Analysis Branch of the NRC. In these positions I was responsible for the development and evaluation of computer programs and techniques to calculate the reactor system and containment system response to postulated loss-of-coolant accidents.

From 1967 to 1972, I was employed by the Babcock and Wilcox Company at Lynchburg, Virginia. There I was lead engineer for the development of loss-of-coolant computer programs and the qualification of these programs by comparison with experimental data.

From 1963 to 1967, I was employed by the Atomic Energy Commission in the Division of Reactor Licensing. I assisted in the safety reviews of large power reactors, and I led the reviews of several small research reactors.

I received an M.S. degree in Nuclear Engineering at the Catholic University of America in 1968 and a B.S. degree in Nuclear Engineering at Mississippi State University in 1963.

I am a graduate of the Oak Ridge School for Reactor Technology, 1963-1964.

I am a member of the American Nuclear Society.

I am the author of three scientific papers dealing with the response of B&W reactors to Loss-of-Coolant Accidents and have authored one scientific paper dealing with containment analysis.

Distribution:
Docket File
RSB R/F
PSCheck
TPSpeis
BWSheron

APR 04 1981

MEMORANDUM FOR: Harold Ru Sullivan, Acting Assistant Director, for NRSR

FROM: Paul S. Check, Assistant Director for Plant Systems, DSI

SUBJECT: SEMISCALE TESTING IN SUPPORT OF NRR.

Reference: Memorandum, Bassett to Ross, "Semiscale Testing in Support of NRR", Dated March 25, 1981

This memorandum is to acknowledge receipt of your reference memorandum and to identify some longer range testing needs for the Semiscale program.

With regard to the reference memorandum, we find the planned testing program responsive to our needs. We wish to complement both your staff and the Semiscale staff at EG&G, Idaho, for continually being responsive to the demanding priorities NRR has placed on the Semiscale facility. It has aided us greatly in understanding accident behavior in PWRs and has continually helped us in evaluating licensing issues.

In a recent meeting with your staff (W. Lyon), and the EG&G Semiscale staff, we (NRR) were asked to identify and future testing needs so that they could be properly factored into longer-term test planning for the facility. At the meeting, we informally identified potential areas for further testing. The enclosure to this memorandum documents our testing needs in these areas in detail.

We do not have any definite need date for item 2 in the enclosure at this time. Results from item 1, vessel head venting procedures, would be useful to us in our review of Inadequate Core Cooling Guidelines. For this reason, we would like you to consider including such tests as part of the natur natural circulation/noncondensable gas tests scheduled for later in FY81.

The above requests have been informally discussed with your staff and we understand they can be factored into the Semiscale test schedule in a timely manner without impact to the program.

My staff will remain in frequent contact with yours to work out testing details or supply additional guidance on data needs.

Paul S. Check, Assistant Director
for Plant Systems
Division of Systems Integration

cc: See next page

CONTACT: Brian W. Sheron
X 29450

DSI:RSB	DSI:RSB	DSI:PS
BWSheron:rem	TPSpeis	PSCheck

ATTACHMENT 1

cc: R. Minogue
R. Mattson
D. McPherson
W. Lyon
G. Knighton
R. Fraley
T. Murley
G. Knighton
S. Hanauer
D. Ziemann
D. Bechham
C. Michaelson, AEOD
J. Guttmann
E. Throm
M. Rubin
G. Mazetis
V. Pancéra
G. Alberthal

ENCLOSURE

SEMISCALE TESTING NEEDS

1. Confirmation of Vessel Head Venting Guidelines

Item II.B.1 of the TMI Action Plan requires the installation of vents in the reactor vessel head and reactor coolant system high points for the purpose of venting non-condensable gases. Included in the requirement is the need for procedures and supporting analyses for operator use of vents, including criteria to initiate and terminate venting.

Vent usage guidelines have recently been submitted by both Westinghouse and Combustion Engineering.

We request that RES first explore the feasibility of using the Semiscale facility to help confirm the acceptability of the submitted guidelines. If the results of the feasibility study are positive, we would like a series of tests scheduled in which the submitted guidelines will be evaluated for technical adequacy.

2. System Depressurization with Auxiliary Pressurizer Spray

Recent plants with NSSSs designed by Combustion Engineering do not have Power Operated Relief Valves (PORVs). Moreover, the High Pressure Injection (HPI) pumps have a shut off head of between 1250 and 1350 psi. Thus, PORVs are not available to depressurize the primary system to below the HPI shutoff head for Safety Injection in the event of a loss of secondary heat sink. A recent license applicant (San Onofre 2 and 3) stated that depressurization could be accomplished with auxiliary pressurizer spray which comes from the charging pump discharge.

The capability of the auxiliary spray to depressurize the system in a timely manner has not been confirmed either with analysis or test.

Our request for the analysis of this capability in a large PWR is being sent under separate cover, and this request pertains to the experimental verification and quantification of this capability.

Specifically, we would like a test series in which the time of auxiliary spray actuation after loss of all feedwater is varied. We are particularly interested in (a) activation prior to SG dryout, (b) activation with during the water solid phase of the event, and (c) activation with high primary system steam void (mixture level below hot legs).

The purpose of this test series would be to determine the ability of auxiliary spray to depressurize the primary system through verification of present analysis codes. We would specifically like a pretest prediction performed not only with RELAP5 but with RELAP4/MOD7 in order to determine the degree to which potential non-equilibrium behavior affects equilibrium code predictions.

3. UHI SBLOCA TESTS

Recent information indicates that the worst case small break LOCA for a UHI plant is a 5% to 6% break in the cold leg. We would like Semiscale test data to assist in our evaluation of this case. It would also be valuable if a comparable test could be provided for the non-UHI case. We understand Semiscale tests for these cases can be provided as part of the present small break test series, and that they can be conducted by lengthening the test program by approximately three weeks with no change in FY 81 funding.

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

MAR 3 1 1981

MEMORANDUM FOR: Karl Kniel, Chief
Generic Issues Branch, DST

FROM: Brian W. Sheron, Section Leader
Section A, Reactor Systems Branch, DSI

THRU: Themis P. Speis, Chief
Reactor Systems Branch, DSI

SUBJECT: STATUS OF FEED AND BLEED FOR EMERGENCY DECAY HEAT REMOVAL

Per my discussions with A. Marchese of your staff, I am providing you with a status summary of feed and bleed as an emergency means of decay heat removal. This summary is provided in the enclosure and is intended for your use in developing the overall action plan for USI A-45, Decay Heat Removal Reliability. The status summary considers the present capability of all operating plants to remove decay heat by feed and bleed. It also addresses the relative risk reduction potential associated with a feed and bleed capability. The general conclusions reached are that:

- Feed and bleed, if performed, should be at a relatively low (P<< relief valve setpoints) pressure.

- Feed and bleed capability can be accomplished in all PWRs if a sufficient capability to depressurize the plant is available. For some plants, this would probably require additional PORV capacity.

- The probability of loss of all feedwater due to loss of all ac power is an uncertain but finite fraction of the total probability of losing all feedwater due to all causes. The ac power-dependence of feed and bleed makes the overall risk reduction questionable. This is because the risk dominant sequences result from a loss of all ac power. Thus, feed and bleed will not improve risk dominant sequences. However, substantial improvement in assuring core cooling might be realized with feed and bleed.

ATTACHMENT 2

- The costs associated with increased pressure relief capability may be acceptable when compared to other risk reducing modifications. Further study is probably warranted.

If you have any further questions, please contact me.

Brian W. Sheron

Brian W. Sheron, Section Leader
Section A, Reactor Systems Branch
Division of Systems Integration

Enclosure:
Status Summary of Feed and Bleed Capability in PWRs

cc: w/enclosure
D. Ross
P. Check
T. Murley
F. Schroeder
P. Norian
A. Marchese
A. Thadani
R. Bernero
P. Baranowsky
J. Ebersole, ACRS
M. Bender, ACRS
H. Etherington, ACRS
J. Ray, ACRS
M. Plessett, ACRS
D. Okrent, ACRS
P. Boehnert, ACRS
R. DiSalvo, RES
P. North, EG&G
G. Johnsen, EG&G
M. Taylor
L. Cave

Enclosure

STATUS SUMMARY OF FEED AND BLEED CAPABILITY IN PWRS

1.0 INTRODUCTION

The feed and bleed process refers to direct removal of decay heat from the primary system utilizing the high pressure injection system and the pressure relief system. The use of this process for decay heat removal is not a preferred method but rather an emergency method when the secondary heat removal path is not available (i.e., no main or auxiliary feedwater available).

The capability to successfully feed and bleed the primary system in order to remove decay heat varies among not only the PWR vendors, but among the various plants designed by the same vendor. This is described in more detail in the following sections.

2.0 BASIC REQUIREMENTS

Notwithstanding certain constraints and limitations of feed and bleed which will be discussed later, the two basic requirements needed to feed and bleed are (1) availability of AC power and (2) the capability to establish a system pressure which will support feed and bleed.

The first requirement, availability of electric power, is an obvious requirement since pumped flow is required, and all HPI pumps have electric drives. For some plants, electric power to operate certain valves is also necessary. The capability to meet the second requirement, to establish a system pressure which will support feed and bleed, is plant-specific and requires further discussion regarding plant capabilities.

3.0 PLANT CAPABILITIES

3.1 B&W-Designed Plants

B&W plants of the 177FA lowered-loop design have high pressure injection pumps (which do "double-duty" as the charging pumps for inventory control) with shutoff heads between 2700 and 3000 psi. One plant, Davis Besse 1, which is of the raised-loop design, has separate charging and HPI pumps. The HPI pumps have a shutoff head of 4000 ft, or about 1500 psi, while the charging pumps have a shutoff head of 6500 ft, or 2600 psi.

All of the B&W plants have one PORV and two safety valves. The set pressure on the safety valves is 2500 psig and the set pressure on the PORVs is 2255 psig.

For all B&W plants, except Davis-Besse 1, the HPI pumps have the capability to inject coolant and discharge it through either the PORVs or the safety valves. An estimate of the flow requirements is about 7 gpm/MWth (based on converting subcooled ($\sim$80°F) water to steam at about 2500 psi). The HPI capability in B&W plants is around 250 to 300 gpm per pump at 2500 psi. Therefore, the HPI pumps can remove all of the core decay heat within a few minutes after shutdown.

3.2 CE-Designed Plants

All CE plants presently licensed, except for ANO-2 have two PORVs. ANO-2 and San Onofre 2 and 3 (NTOL) do not have any PORVs. St. Lucie 2 (NTOL) will have PORVs, but all CESSAR (System-80) plants (Palo Verde) will not. The PORV setpoint on CE plants is about 2385 psi. CE plants have either two or three safety valves, with a setpoint of about 2485 psi.

All CE plants, except for Maine Yankee, have HPI pumps with shutoff heads between about 1250 and 1350 psig. Maine Yankee has an HPI shutoff head of 2471 psi. From these values, it can be seen that no CE plants have the capability for feed and bleed at high pressure, and must be depressurized in order to utilitize HPI flow for decay heat removal. While Maine Yankee has the capability to pump against (and open) the PORV (but not the safety valve), the flow would most likely be insufficient to cool the core.

In order for CE plants to feed and bleed, the primary system must be depressurized. Since the need to feed and bleed was based on losing all feedwater, depressurization can only be accomplished by blowdown or condensation of primary steam. Condensation of primary steam could be accomplished by auxiliary spray to the pressurizer. No analyses have been performed yet on the effectiveness of this depressurization method, so its capabilities are not known. Blowdown of the primary system using the PORVs (for those plants which have them) is questionable. In CEN-114, CE evaluated the ability to recover the plant after a loss of all feedwater. Their conclusions were that the system could be recovered without fuel damage if either (1) auxiliary feedwater was restored within one hour, or (2) both PORVs were opened within 10 minutes after the initiation of the event. For the latter, the PCT was predicted to be 2040°F. The capability to depressurize using the PORVs is questionable because of the uncertainty in the critical discharge from the valve, particularly in the two-phase regime. Data from the EPRI program will hopefully shed light on this.

3.3 <u>Westinghouse-Designed Plants</u>

There are 25 Westinghouse-designed plants that were in operation at the time of the TMI-2 accident. Of the 25 plants, 17 have 3 safety valves and 8 have 2 safety valves. All of the plants have 2 PORVs except for Beaver Valley 1 and D. C. Cook 1 & 2 which have 3 PORVs, and Yankee Rowe which has one PORV. The PORV setpoints vary between 2190 psi and 2400 psi, with the majority at 2335 psi. The safety valve setpoints vary between 2360 psi and 2500 psi, with the majority set at 2485 psi.

Thirteen of the plants have HPI pumps with shutoff heads above the safety valve setpoints. However, the shutoff heads are usually within a few hundred psi of the safety valve setpoint and continuous operation at the safety valve setpoints could possibly damage the pumps due to insufficient internal cooling flow. The remaining twelve plants have low shutoff heads, typically around 1500 psi.

In general, Westinghouse designed plants must depressurize using the PORVs or auxiliary pressurizer spray in order to effectively feed and bleed. Those plants with 3 PORVs will be able to depressurize more readily than those with one or two PORVs. Likewise, plants with high head HPI pumps can more readily inject coolant than plants with low head HPI pumps.

Westinghouse presented an evaluation of bleed and feed capability in Westinghouse plants at the Sequoyah ACRS Subcommittee meeting on June 2, 1980(1). These analyses looked at a 2411 MWt plant (similar to Sequoyah) and concluded if the PORVs were opened and kept open on or before 3000 seconds (approximately the SG dryout time), and the SI pumps were started, the results would be acceptable. They also analyzed the case in which the PORVs were not opened, but allowed to open and close normally at the setpoint. HPI was started and left on. The results indicated a net mass loss from the system, but substantially increased (beyond 10^4 sec), the time available for operator action.

3.4 Plant Capability Summary

The capabilities of present vendor designs to successfully remove decay heat by feed and bleed is summarized in Table 1. Plant-specific data related to feed and bleed capability in each operating PWR is provided in Tables 2, 3, and 4 and were taken from references 2, 3, and 4.

4.0 OTHER CONSIDERATIONS

Decay heat removal by coolant addition using HPI and removal using the safety valves (or PORVs in a normal mode) will result in a repressurization of the primary system. As the decay heat subsides, the primary system will cool down.

High pressure and low temperature in the primary system, and particularly the vessel, may produce unacceptable conditions regarding thermal shock. This concern has been previously identified and is being addressed in item II.K.2.13 of NUREG-0737[5].

Moreover, the Westinghouse analysis presented in reference 1 indicates that long-term feed and bleed at high pressure results in a net inventory loss and is not considered a stable mode of operation.

The above two considerations indicate that feed and bleed should preferably be performed at low pressures and therefore require a system depressurization capability. As was seen from the CE analyses in CEN-114, the existence of PORVs does not mean an adequate depressurization capability exists. In the CE case, the operators would be faced with making a decision within 10 minutes of either blowing down the plant using the PORVs, or waiting and hoping feedwater would be restored within 30 to 60 minutes.

5.0 RECOMMENDED CRITERIA

In the event feed and bleed is considered a viable and desirable method of emergency decay heat removal, it is recommended that the system be capable of depressurizing to below the shutoff level of the HPI pumps within a given period of time. This time should be established based on the following criteria:

(1). A specified amount of time be available to an operator to make the decision of whether or not it is necessary to open PORVs and depressurize the plant (e.g., 20 minutes).

(2) Given the amount of time between the initiation of a loss of all feedwater and the time at which the PORVs must be opened (from #1 above), the system should be capable of being depressurized to below the HPI pump shutoff head, and the HPI pumps should be capable of injecting sufficient coolant to prevent core uncovery.

Requirements regarding the degree to which equipment needed for feed and bleed should meet safety grade requirements (particularly the PORVs) would need to be established.

Although a detailed review has not been performed, it appears from the information in Tables 2, 3, and 4 that all PWRs can remove decay heat under emergency conditions using feed and bleed if they have the capability to depressurize. This can be accomplished with additional PORV capacity. The capability to either increase present PORV relieving capability or add PORV relieving capability may be difficultif not impossible on plants already in operation.

6.0 PROBABILISTIC CONSIDERATIONS

The purpose of a feed and bleed capability is to remove decay heat in the event both main and auxiliary feedwater are lost. A consideration that must be addressed is the relative benefit (or decrease in risk) that would be realized by a feed and bleed capability. One factor in this consideration is that feed and bleed requires electric power to operate. Thus, it is necessary to look at the relative decrease in risk that feed and bleed could provide. Pat Baranowsky of RES provided estimates for the sequences of interest, and are shown in Table 5.

As can be seen, the probability of loss of all feedwater due to loss of all AC power is most likely in the range of a few to a few tens of percent of the total probability of losing all feedwater. However, Baranowsky estimates that the risk from the loss of feedwater due to loss of all AC sequences is perhaps 2 to 200 times greater than the loss of all feedwater sequence (AC power still available).

This is because the AC power availability would allow containment heat removal capability. The loss of all AC case would ultimately result in containment failure due to lack of containment heat removal.

SUMMARY

The following points summarize feed and bleed status today:

- Most PWRs do not have high pressure feed and bleed capability. Moreover, high pressure feed and bleed is not recommended due to vessel structural consideration. Feed and bleed should be performed at lower pressures.

- In order to effectively feed and bleed at lower pressures, a depressurization capability is needed. Perhaps about half of the operating PWRs have a sufficient depressurization capability.

- A sufficient depressurizing capability can most readily be attained by increasing the PORV capacity on plants which already have PORVs. The feasibility of increasing PORV capacity or installing PORVs on plants that presently do not have PORVs would have to be investigated. Other methods of depressurizing, such as with auxiliary pressurizer spray, require further analysis.

- The relative risk reduction that might be achieved by additional PORV capacity is finite, but highly uncertain at this time. The costs of adding additional PORV capacity would have to be weighted against a more exact estimate of risk reduction.

REFERENCES

1. Transcript of the Advisory Committee on Reactor Safeguards Subcommittee on Sequoyah Nuclear Power Station, June 2, 1980, Pgs. 150-173.

2. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, dated January 1980.

3. "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company," NUREG-0560, dated May 1979.

4. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Desinged Operating Plants," NUREG-0635, dated January 1980.

5. "Clarification of TMI Action Plan Requirements," NUREG-0737, dated November 1980.

TABLE 1

GENERAL CAPABILITY OF PRESENT PWR DESIGNS TO FEED AND BLEED

B&W	All plants (except Davis Besse 1) can feed and bleed at high pressure ($P>P_{sv}$). Capability to depressurize uncertain and needs further evaluation.
CE	No capability to feed and bleed at high pressure. Plants with PORVs have questionable capability to depressurize adequately. Plants without PORVs must rely on auxiliary pressurizer spray to depressurize. This capability presently unknown.
W	No capability for extended high pressure feed and bleed. Pump damage potential. Plants with 3 PORVs capable of depressurizing. Plants with 2 PORVs need further evaluation.

TABLE 2

B&W PLANT DATA RELATED TO FEED AND BLEED CAPABILITY

(FROM NUREG 0560)

	PORV Capacity, lb/hr/MWt/ Setpoint, psi (to be revised per IE 79-05B)	PORV Manufacturer/Model No.	SV Combined Capacity, lb/hr/MWt/ Setpoint, psi	High Pressure Injection Pump Characteristics: Shut Off Head, ft	High Pressure Injection Pump Characteristics: Gpm @ 1000 psig	High Pressure Injection Pump Characteristics: GPM @ 1600 psig	High-Pressure Injection Pump Manufacturer
TMI-2	40.4 / 2255	Dresser 315 33VX	249 / 2450	6500	@ 500 ea	@ 450 ea	Bingham
Crystal River	40.8 / 2255	Dresser 315 33VX	254 / 2500	6500	@ 500 ea	@ 450 ea	Bingham
Oconee 1,2,3	41.7 / 2255	Dresser	254 / 2500	7000	@ 500 ea	@ 450 ea	Ingersall-Rand
Rancho Seco	40.4 / 2255	Dresser	249 / 2500	7000	@ 500 ea	@ 450 ea	B-J
Davis-Besse	40.4 / 2255	Crosby HPN-SN	242 / 2435	4000 MU 6500	@ 700 ea	@ 200 ea	B&W Canada
ANO-1	38.9 / 2255	Dresser	234 / 2500	7000	@ 500 ea	@ 450 ea	B-J
TMI-1	38.9 / 2255	Dresser	243 / 2435	6500	@ 300 ea	@ 450 ea	Bingham

TABLE 3

CE PLANT DATA RELATED TO FEED AND BLEED CAPABILITY
(FROM NUREG 0635)

PARAMETER / PLANT	ARKANSAS POWER AND LIGHT CO., ARKANSAS NUCLEAR ONE-2	BALTIMORE GAS AND ELECTRIC CO., CALVERT CLIFFS	OMAHA PUBLIC POWER DISTRICT, FT. CALHOUN	MAINE YANKEE ATOMIC POWER CO., MAINE YANKEE	NORTHEAST UTILITIES, MILLSTONE 2	CONSUMERS POWER CO., PALISADES	FLORIDA POWER AND LIGHT CO., ST. LUCIE
PORV Capacity, lb/hr/ MWt/Setpoint, psi	NONE	56.7 2385	69.7 2392	57.0 2385	59.8 2380	60.5 2385	59.8 2385
Number of PORVS, Manufacturer/Model No.	NONE	TWO DRESSER 31533VX	TWO DRESSER 31533VS	TWO DRESSER 31533VS-30	TWO DRESSER 31533VX-30	TWO DRESSER 31533VX	TWO DRESSER 31533VS-30
Number of Safety Valves, Total Capacity, lb/hr/ MWt/Setpoint, psi	TWO 280.6 2500	TWO 222.2 2485	TWC 281.7 2485	THREE 228.1 2485	TWO 231.3 2485	THREE 272.7 2485	THREE 234.4 2485
Shut off head, ft/psi	3500 1517	2900 1257	3200 1387	5700 2471	2800 1213	2900 1257	2900 1357
Gpm @ 1000 psig	500	400	280	715	475	400	425
Gpm @ 1600 psig	0	0	0	550	0	0	0
Positive displacement charging pump capacity gpm	128	132	120	450	132	133	132

TABLE 4

WESTINGHOUSE PLANT DATA RELATED TO FEED AND BLEED CAPABILITY
(FROM NUREG 0611)

Utility/Plant	PORV Capacity, lb/hr/Mwt/set-point, psig	Number of PORVs, Manufacturer Model No.	Number of Safety Valves, Total Capacity, lb/hr/Mwt/ Set Point, psig	HPI PUMP CHARACTERISTICS: Shut Off Head, ft/psig	HPI PUMP CHARACTERISTICS: gpm @ 1000 psig	HPI PUMP CHARACTERISTICS: gpm @ 1600 psig	HPI PUMP CHARACTERISTICS: gpm @ Safety PORV Set Pt.	Positive Displacement Changing Pump Capacity, gpm
Alabama Power Co., Farley 1	79.2 / 2335	two Copes-Vulcan D-100-160	three 130.1 / 2485	6000 / 2600	750	550	-	250 / 325
Carolina Power & Light, H. B. Robinson	95.5 / 2335	two Copes-Vulcan D-100-160	three 130.9 / 2485	3300 / 1430	350	0	231	N/A
Commonwealth Edison, Zion 1&2	64.6 / 2335	two Copes-Vulcan D-100-160	three 129.2 / 2485	6000** / 2600	490**	380**	-	165 / 230
Connecticut Yankee, Haddam Neck	115.1 / 2270	two Copes-Vulcan D-100-160	three 160.7 / 2485	6800** / 2948	~800**	575**	-	325 / 388
Consolidated Edison, Indian Pt. 2	61.3* / 2335	two Copes-Vulcan D-100-160	three 147.9 / 2485	3500** / 1517	485	0	294	N/A
Power Authority of State of New York Indian Pt. 3	61.3* / 2335	two Copes-Vulcan D-100-160	three 138.9 / 2485	3500** / 1517	485	0	294	N/A
Duquesne Light, Beaver Valley 1	79.9 / 2335	three Masonellan 38-20771	three 129.7 / 2485	5850 / 2536	495	380	-	175 / 225
Florida Power & Light, Turkey Pt. 3&4	95.1 / 2335	two Copes-Vulcan 5-131642	three 132.8 / 2485	3500 / 1517	410	0	231	N/A
Indiana & Mich. Electric, D.C. Cook 1	64.6 / 2335	three Masonellan 38-20721	three 129.2 / 2485	5800 / 2514	560	400	-	70 / 225

TABLE 4 (CONT.)

	PORV Capacity, lb/hr/MWt/set-point, psig	Number of PORVs, Manufacturer Model No.	Number of Safety Valves, Total Capacity, lb/hr/MWt/ Set Point, psig	HPI PUMP CHARACTERISTICS: Shut Off Head, ft/psig	gpm @ 1000 psig	gpm @ 1600 psig	gpm @ Safety PORV Set Pt.	Positive Displacement Charging Pump Capacity, gpm
D.C. Cook 2	61.8/2335	three Masoneilan 38-20721	three 123.5/2485	5800/2514	560	400	-	170/225
Northern States Power Prairie Is. 1&2	108.5/2335	two Copes-Vulcan D-100-160	two 209.1/2485	5000/2168	760	600	180	N/A
Portland Gen. Electric, Trojan	61.6/2350	two Copes-Vulcan D-100-160	three 123.1/2485	600/2600	495	380	-	180/230
Public Service Electric & Gas, Salem 1	63/2350	two Copes-Vulcan D-100-160	three 125.8/2485	6168/2670	490	380	-	150/210
Rochester Gas & Electric, Ginna	117.8/2335	two Copes-Vulcan D-100-160	two 169.5/2485	3426/1485	285	0	180	N/A
Southern Calif. Edison, San Onofre 1	29/2190	two ACF Industries 70-18-9 DRTX	Two 178.2/2500	6000/2600	570	460	-	165/300
Virginia Elec. Power, Surry 1&2	86/2335	two Copes-Vulcan IA58RGP	three 120.5/2360	6000/2600	520	420	-	255/250
North Anna 1	76/2335	two Masoneilan 38-20721	three 137/2485	5900/2550	650	550	-	250/325
Wisconsin Electric Power, Point Beach 1&2	117.9/2335	two Copes-Vulcan D-100-160	two 189.7/2485	3550/1539	900	0	180	N/A

TABLE 4 (CONT.)

Utility Plant	PORV Capacity, lb/hr/Mwt/set-point, psig	Number of PORVs, Manufacturer Model No.	Number of Safety Valves, Total Capacity, lb/hr/Mwt/ Set Point, psig	HPI PUMP CHARACTERISTICS: Shut Off Head, ft/psi	gpm @ 1000 psig	gpm @ 1600 psig	gpm @ Safety PORV Set Pt.	Positive Displacement Charging Pump Capacity, gpm
Wisc. Pub. Serv., Kewaunee	106/2335	Two Copes-Vulcan D-100-160	Two, 209.1/2485	500/2167	750	500	180	N/A
Yankee Atomic Electric, Yankee Rowe	118/2400	One Dresser 31533 VX	two 153/2485	1950/844	0	0	99	N/A

†Data not provided
*To be increased to 78.7 about June 1979
**Charging pump (centrifugal)

TABLE 5

Probability Estimates*

Loss of Offsite Power	0.2	Failure of Emergency AC	10^{-2} - 10^{-4}
Recovery of Offsite Power	0.5	Failure of AFW w/o AC	5×10^{-2} - 10^{-3}
Loss of MFW	0.3	Failure of AFW	10^{-3} - 5×10^{-5}
Recovery of MFW (1/2-1 hr)	0.1		

Sequence Probabilities*

(Loss of Offsite Power) X (Loss of Emergency AC) X (Loss of AFW) = $5 \times 10^{-5} - 10^{-6}$

(Loss of Main Feedwater) X (Loss of AFW) = $3 \times 10^{-4} - 1.5 \times 10^{-5}$

*Informally provided by P. Baranowsky, RES

PRESENT STAFF POSITION REGARDING "FEED AND BLEED" FOR DECAY HEAT REMOVAL

- THERE ARE PRESENTLY NO REGULATIONS OR OTHER REQUIREMENTS THAT WOULD SPECIFICALLY REQUIRE "FEED AND BLEED" CAPABILITY IN THE DESIGN BASIS OF THE PLANT

- GDC34 SETS FORTH THE REQUIREMENT FOR RESIDUAL HEAT REMOVAL

- ONE OBJECTIVE OF TASK A-45 IS TO DETERMINE IF SIGNIFICANT REDUCTION IN RISK DUE TO LOSS OF ALL FEEDWATER CAN BE ACHIEVED BY REQUIRING "FEED AND BLEED" CAPABILITY IN DESIGN BASES OF PLANTS

- IF SIGNIFICANT SAFETY IMPROVEMENT CAN BE REALIZED, STAFF WILL CONSIDER METHODS FOR UPGRADING EXISTING PLANTS SO FEED AND BLEED CAPABILITY CAN BE ACHIEVED

NOTWITHSTANDING PRESENT POSITION, WE BELIEVE "FEED AND BLEED" IS DESIRABLE FEATURE AND WE ARE EVALUATING EXISTING CAPABILITIES AND OPTIMUM OPERATIONAL METHODS.

- MEMO, CHECK (NRR) TO SULLIVAN (RES), DATED APRIL 2, 1981 REQUESTED RES (VIA SASA PROGRAM) TO EVALUATE DEPRESSURIZATION CAPABILITY OF CE PLANTS WITHOUT PORVs
  - EXAMINE PRESSURIZER SPRAY CAPABILITY
  - HEAD VENT SYSTEM
- IN CEN-114 CE SHOWED LOFW WITH 2 PORVs OPENED AT 10 MINUTES PRODUCED ACCEPTABLE RESULTS USING EM MODEL
- W PRESENTED ACCEPTABILITY OF FEED AND BLEED FOR SEQUOYAH TO ACRS PREVIOUSLY
- B&W, IN MAY 7, 1979 "BLUE BOOK," SHOWED LOFW WITH EITHER AFW RESTORATION OR HPI ACTUATION WITHIN 20 MINUTES PRODUCED ACCEPTABLE RESULTS
- PRESENT DESIGNS DO NOT MEET GENERAL REQUIREMENTS OF A "SAFETY GRADE" SYSTEM

  E.G., B&W PLANTS HAVE ONE PORV (NO REDUNDANCY)
- DESIGN REQUIREMENTS FOR "FEED AND BLEED" NOT ESTABLISHED

  E.G., DEPRESSURIZATION CAPABILITY

- EPRI VALVE TESTING PROGRAM EXPECTED TO QUALIFY RELIEVING CAPABILITIES OF SVs AND RVs UNDER SINGLE PHASE LIQUID AND TWO PHASE FLOW CONDITIONS

- THE EXTENT TO WHICH OPERATING PLANTS CAN "FEED AND BLEED" TODAY DEPENDS ON THE:

  - CHARGING/HPI CAPACITY
  - SHUTOFF HEAT OF PUMPS
  - NUMBER OF SVs AND RVs

- "FEED AND BLEED" SHOULD BE ACCOMPLISHED AT LOW PRESSURE

- GUIDELINES AND PROCEDURES INSTRUCT OPERATOR ON STEPS TO TAKE REGARDING WHEN AND HOW TO "FEED AND BLEED"

  - TRY TO RESTORE FEEDWATER

  - IF FEEDWATER CANNOT BE RESTORED, START HPIs, OPEN PORVs

SEP 0 7 1982

NOTE TO: Brian W. Sheron, Chief, Reactor Systems Branch, DSI

FROM: Mary Ellen Keane, Section A, RSB, DSI
Walt Jensen, Section C, RSB, DSI

SUBJECT: COMPARISON OF WESTINGHOUSE FEED AND BLEED ANALYSIS AND SEMISCALE S-SR-2

Westinghouse performed a plant calculation for conditions similar to those of the Semiscale test (IP-3 in WCAP-9744). The Westinghouse calculation kept the core covered while the test resulted in core heat-up.

	Analysis W (IP-3)	Test SS (Trojan/Zion)
Full power:	3025 MWt	based on 3411MWt
PORV Cap:	139 lb/hr/MWt	based on 129.2 lb/hr/MWt*
SI Shutoff:	1470 psi	1500 psi
PORVs Open:	1500 sec after loss of FW	Start of Test
Conditions at PORV Opening		
Power:	2%	2%
Pressure:	2200 psi	2250 psi
Temp:	566 F	533 F
SG Water:	5 ft. (4 min. before dryout)	empty
Conclusion:	No core uncovering	Core heat up @ 20-30 min.

The following remarks concern difference between the test and the calculation.

1) The initial conditions in the test and the calculation appear to be roughly equivalent. The major difference is that in the W analysis, the PORV opened when there was still 5 ft. left in the SG (four minutes before SG dryout). Semiscale began with empty Steam Generators. The W analysis had an initial fluid temperature of 566 F while Semiscale had a temperature of 533 F. It is not clear whether this difference is significant. W concluded that if the operator waited until the SG was empty, the core would eventually be uncovered.

2) The Westinghouse analysis was for Indian Point 3 which has only low head HPI whereas the Semiscale test was based on Trojan and Zion which have safety grade charging which was considered inoperable. The Westinghouse report pointed out that plants with non safety grade charging are at lower power (3025MWt) and have a larger PORV capacity. Semiscale is based on 3411MWt plant.

*We understand semiscale had an additional 20% PORV capacity over the reference plant. This should have provided additional depressurization and increased ECCS flow.

ATTACHMENT 3

3) The two tests differed in the quality of the PORV discharge. The two RCS pressure transients are similar. For the Westinghouse analysis the PORV discharges subcooled or two-phase fluid for the first 370 seconds of the transient. The Semiscale preliminary test data (Figure 9 of North to Tiller letter August 6, 1982) indicates that the subcooled or two phase period lasts for 1000 seconds. The difference may be due to atypicalities in Semiscale pressurizer.

4) Although Semiscale was stated to be based on flow from non-degraded ECCS other than inoperable charging, the total ECCS flow out of Semiscale was less than that of one ECCS pump for IP-3 based on a scaling factor of 2MW/3411MW. (see attached curve)

5) Semiscale used a constant power level of 2% of initial power throughout the test. This was about the decay heat ratio as used by Westinghouse at the time of PORV opening (1500 sec) but at the time of semiscale core heat up (+1500 sec) a value of 1.65 % should have been used. This is a mismatch in decay heat ratio of 21%.

Conclusion: At the time of core uncovery, 1500 sec after PORV opening, Semiscale had a 37% higher core power (initial power mismatch plus power decay mismatch). At that time, the ECCS flow rate was less than the W rate by approximately 5%. These conditions probably account for the differences between Semiscale and the Westinghouse analysis.

Original signed by:

Mary Ellen Keane, Section A, RSB

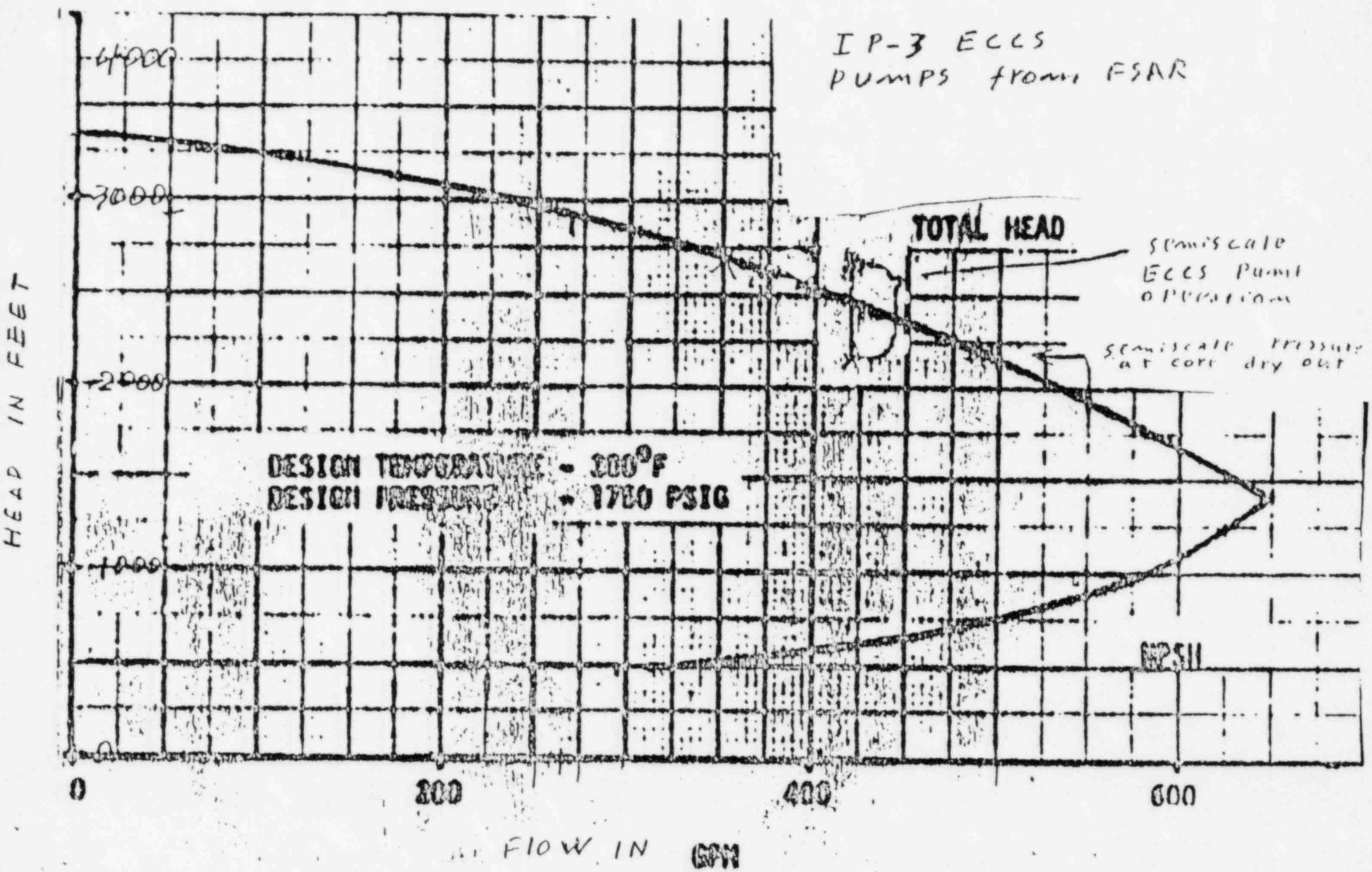
Original signed by:

Walt Jensen, Section C, RSB

cc: W. Lyon
T. Marsh
W. Hodges

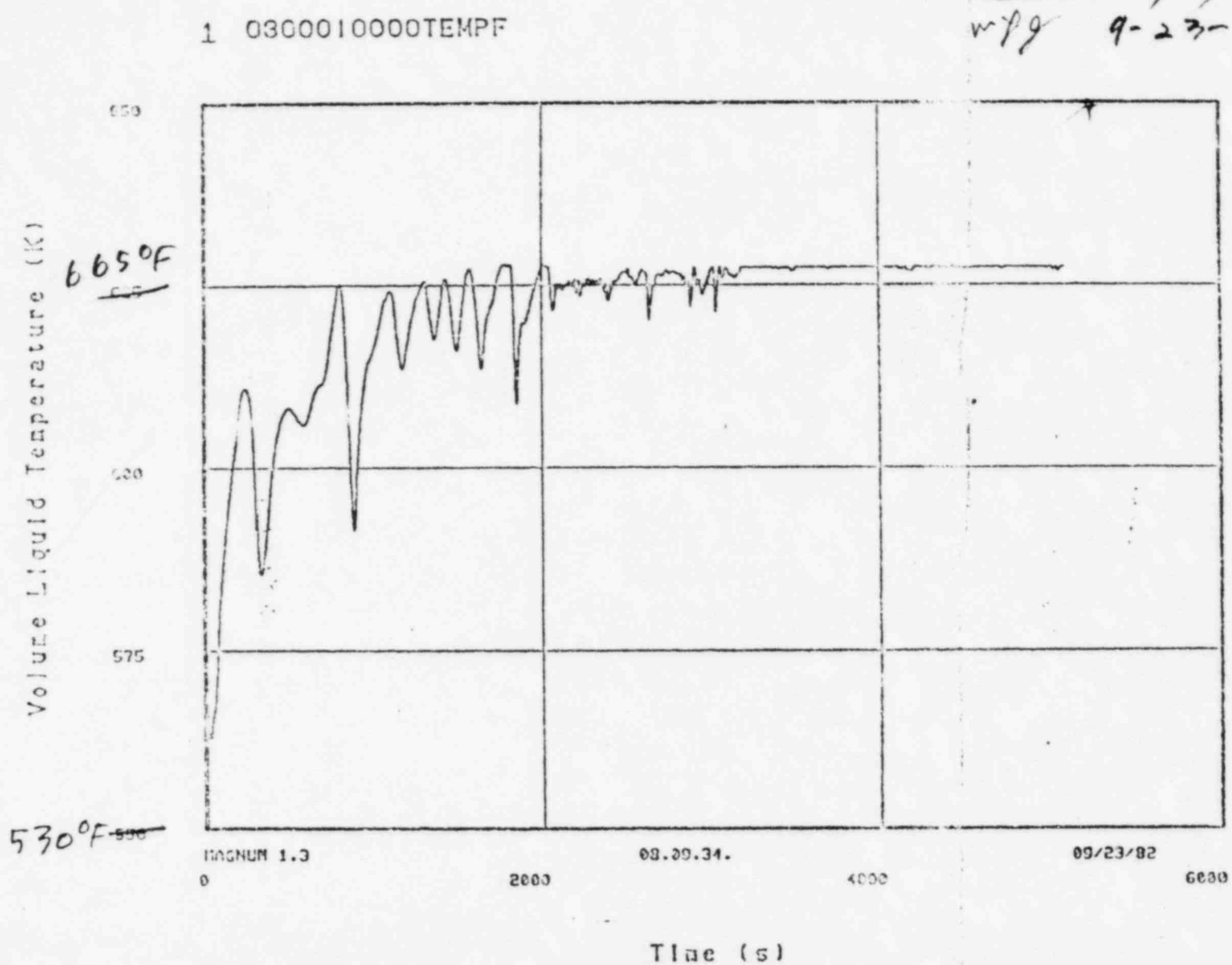
Distribution
Central File
RSB R/F
RSB S/F: Westinghouse
MKeane R/F
WJensen R/F
GMazetis
NLauben
MKeane
WJensen

DSI:RSB	DSI:RSB	DSI:RSB	DSI:RSB			
MKeane/bc	WJensen	GMazetis	NLauben			
9/7/82	9/7/82	9/7/82	9/7/82			

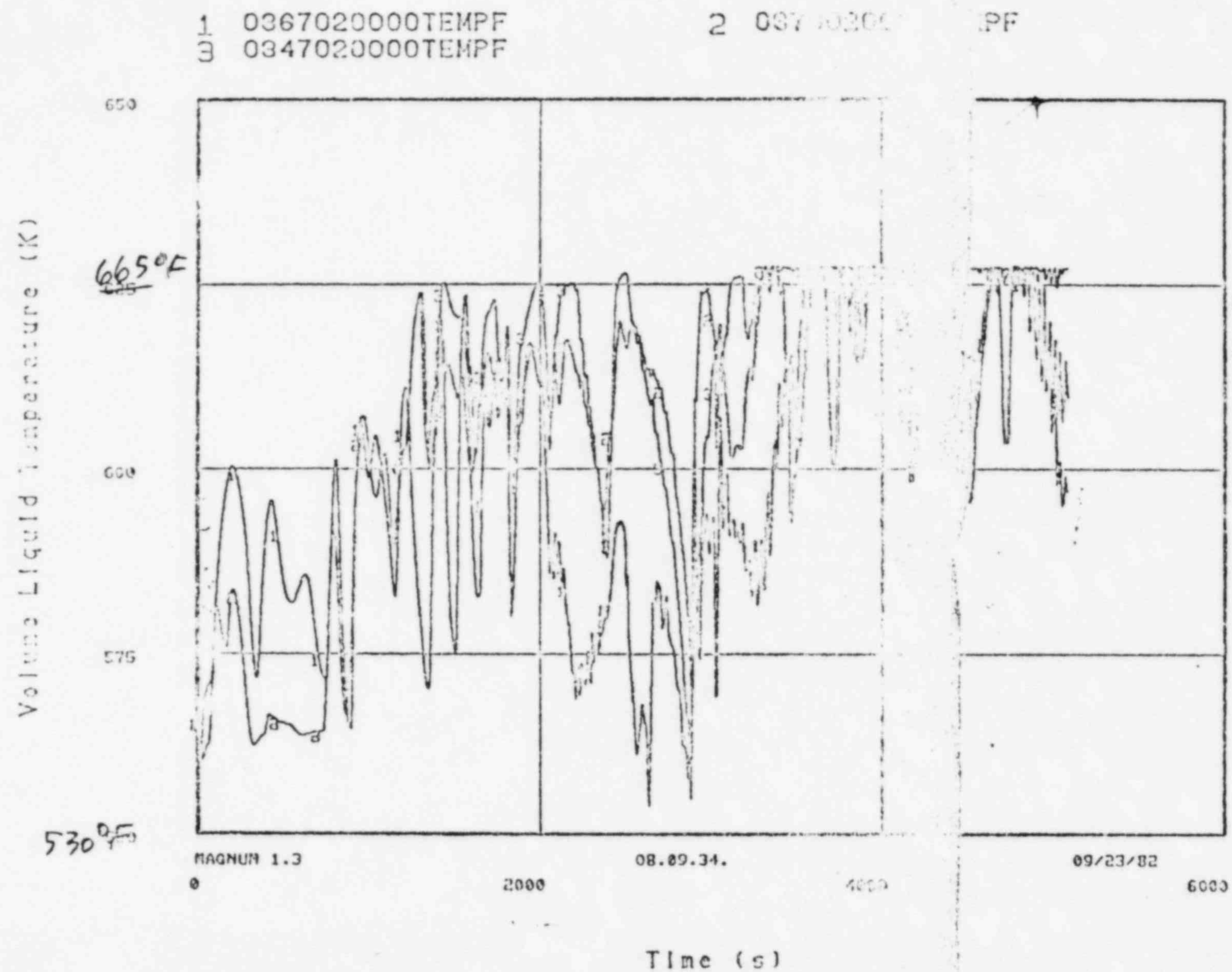

IP-3 ECCS
PUMPS from FSAR
TOTAL HEAD
Semiscale ECCS Pump operation
Semiscale Pressure at core dry out
DESIGN TEMPERATURE - 300°F
DESIGN PRESSURE - 1750 PSIG
NPSH
HEAD IN FEET
4000
3000
2000
1000
0
200
400
600
FLOW IN GPM

NRC STAFF AUDIT of B&W
Feed & Bleed Calculations NO core
uncovery / NO PTS
wfg 9-23-82

DOWNCOMMER
TEMP



ATTACHMENT 4



LOCAL
DOWNCOMMPT
TEMP @
COLD LEG
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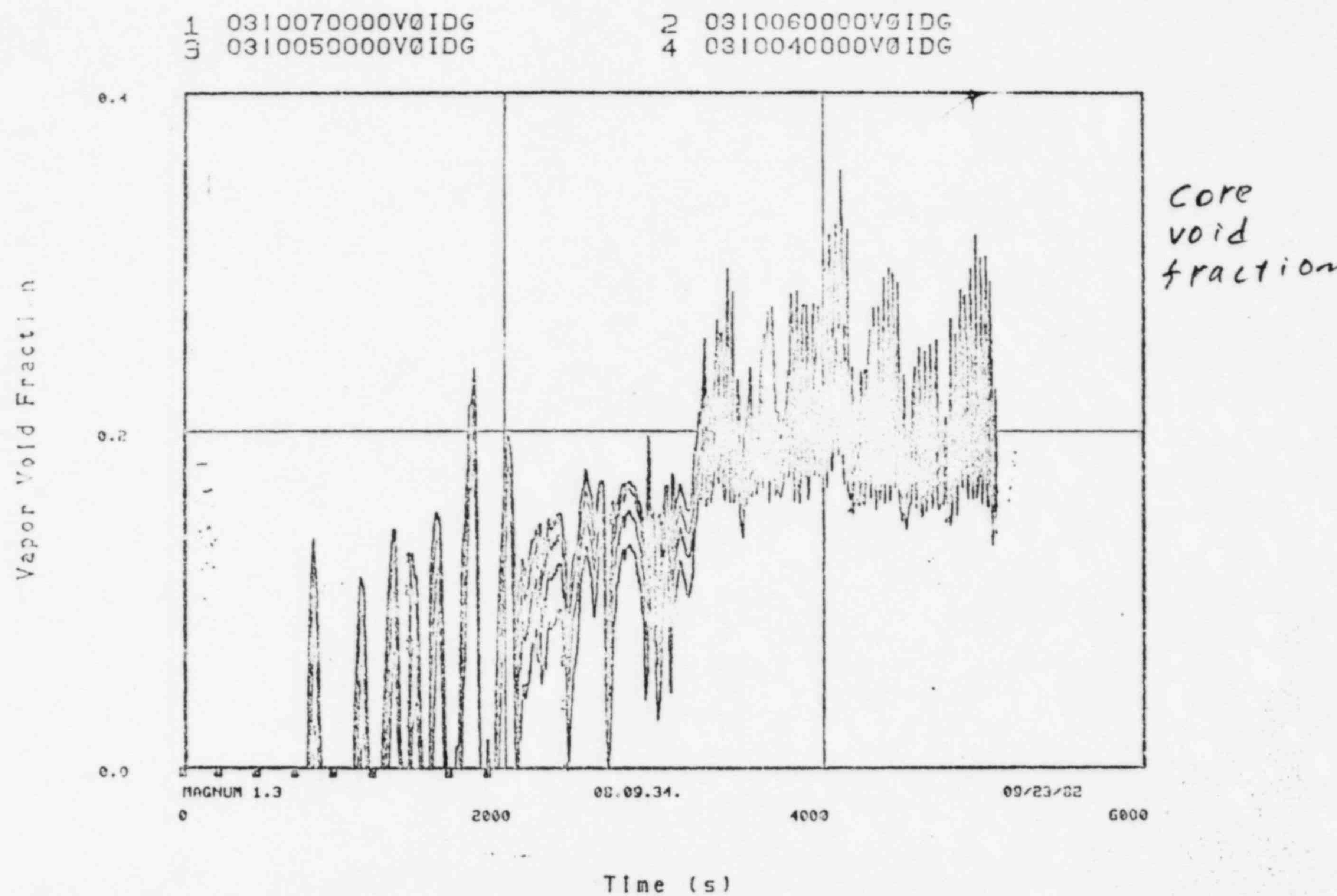
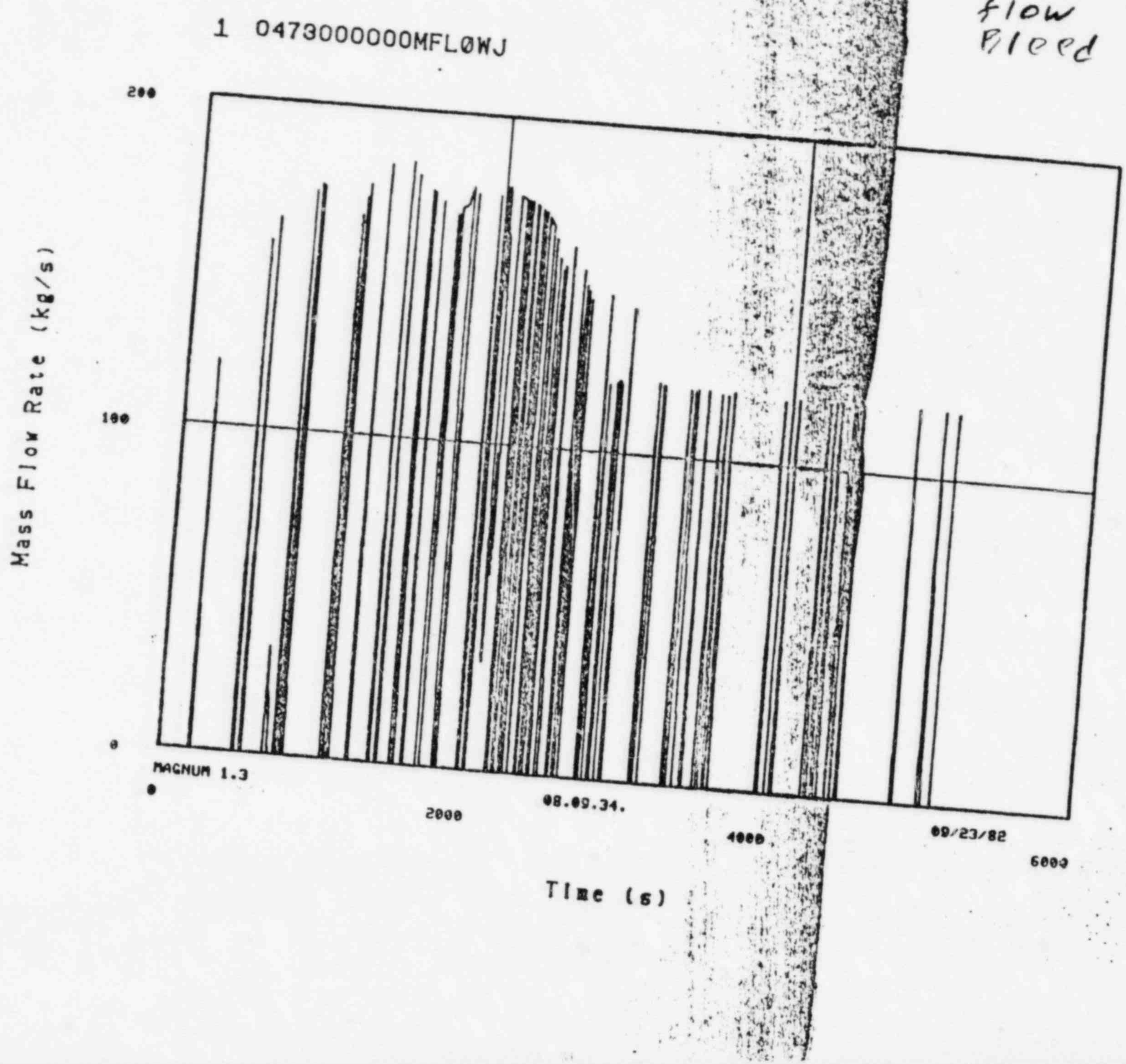


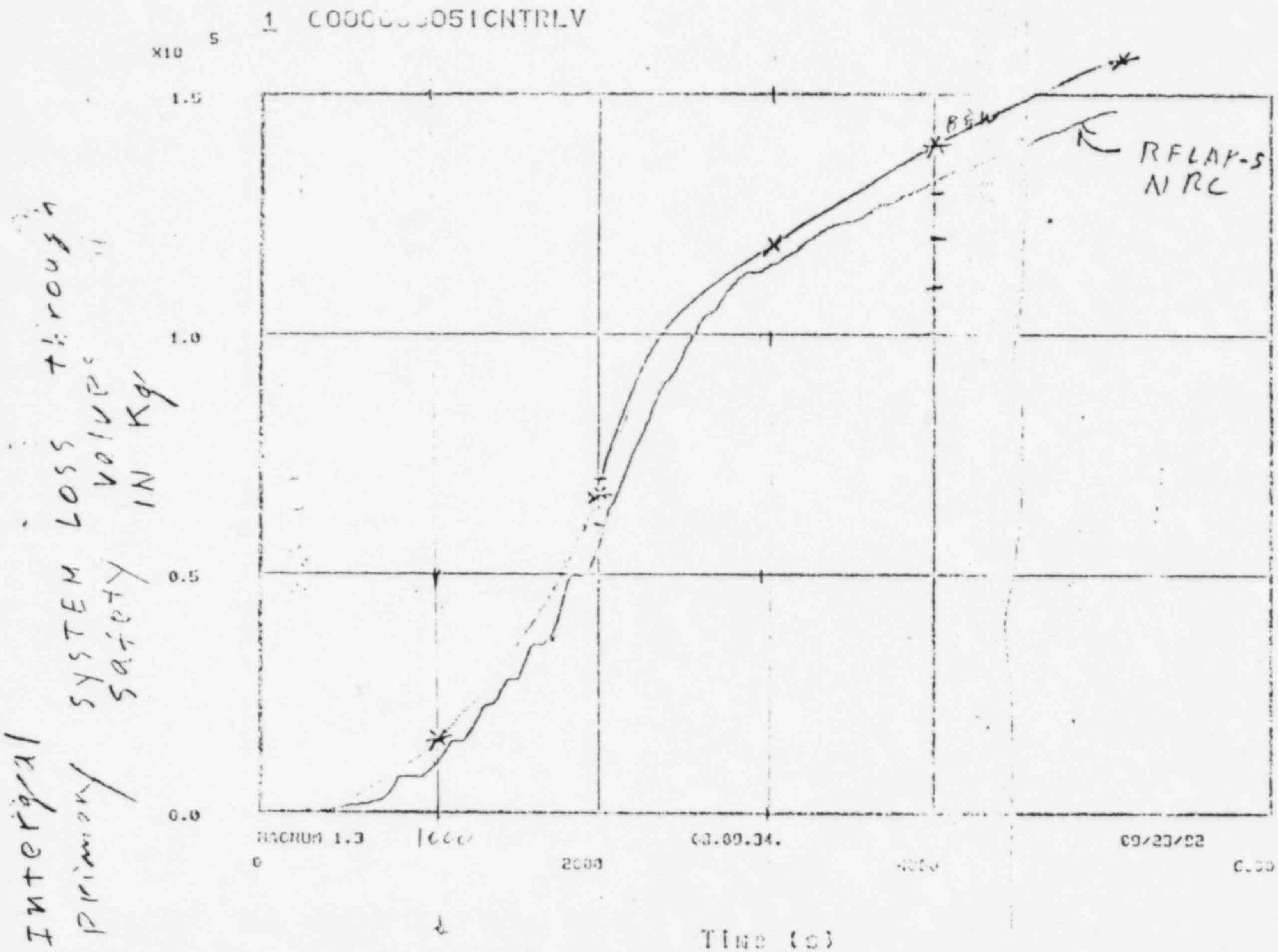
1 0310070000VØIDG
3 0310050000VØIDG
2 0310060000VØIDG
4 0310040000VØIDG
0.4
0.2
0.0
Vapor Void Fracti n
MAGNUM 1.3
08.09.34.
09/23/82
0
2000
4000
6000
Time (s)
Core
void
fraction

Safety valve flow Feed & Bleed



Feed & Bleed

WJG
10-1-82



UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of METROPOLITAN EDISON COMPANY, ET AL. (Three Mile Island, Unit 1)	)))))	Docket No. 50-289 (Restart)

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE TO APPEAL BOARD ORDER OF OCTOBER 15, 1982", in the above-captioned proceeding, have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, or, as indicated by double asterisks, by hand delivery, this 25th day of October, 1982:

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, ET AL.	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island, Unit 1)	)	

NRC STAFF RESPONSE TO APPEAL BOARD
ORDER OF OCTOBER 15, 1982

October 25, 1982

Richard J. Rawson
Counsel for NRC Staff

DESIGNATED ORIGINAL
Certified By DSO