INTERIM REPORT

8-22-79

Accession No. ______________

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Contract Program or Project Title: Semiscale Program

Subject of this Document: "Semiscale Stimulations of the Three Mile Island Transient - A Summary Report",

Type of Document:

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Date of Document: July 1979

Responsible NRC Individual and NRC Office or Division: W. D. Lanning

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Prepared for
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Washington, D.C. 20555

NRC Fin #A6038

INTERIM REPORT

NRC Research and Technical Assistance Report

927 335

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EG&G Idaho, Inc.

P. O. Box 1625
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July 13, 1979

Mr. R. E. Tiller, Director
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SEMISCALE SIMULATIONS OF THE THREE MILE ISLAND TRANSIENT - A SUMMARY REPORT (SEMI-TR-010) - DJO-84-79

Ref: D. J. Claflin and E. L. Wills (Eds.), Quarterly Technical Progress Report on Water Reactor Safety Program Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, April-June 1979, NUREG/CR-0871, TREE-1300 (to be published July 1979)

Dear Mr. Tiller:

Enclosed is a summary report on eight Semiscale Simulations of the Three Mile Island Unit 2 nuclear power generating station transient. A total of ten simulations of the Three Mile Island transient have been conducted in the Semiscale Mod-3 system. The first two tests were upper plenum vent tests that were run during the Three Mile Island transient on March 30 and 31, 1979 and are documented in the reference. The remaining eight tests are simulations of the sequence of events during the first few hours of the Three Mile Island transient as understood by the Semiscale Program. The objectives of the eight simulations were to (a) to gain a more fundamental understanding of the thermal-hydraulic phenomena which occured in the Three Mile Island reactor and (b) determine the Semiscale capability and problems associated with conducting extremely slow loss-of-coolant accident (LOCA) transients.

The overall thermal-hydraulic trends observed in the Semiscale simulations were similar to those observed in the TMI data available. For example, the Semiscale pressurizer level behavior indicated trends similar to those exhibited in TMI. Moreover, the Semiscale simulations showed that the pressurizer level was not an appropriate indication of the system mass inventory: core uncovery and core heatup occurred in the Semiscale simulations even though the pressurizer remained liquid full. Superheated steam was observed in the Semiscale system hot legs in the same time frame as was observed during the Three Mile Island transient, indicating the core heatup for Three Mile Island and Semiscale occurred at about the same time. An estimation of the Three Mile Island core heatup was made using Semiscale heat transfer data. Results indicate that significant core damage could have occurred above the 2.9 m elevation during the first few hours of the TMI transient.

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NRC Research and Technical Assistance Report

Several scaling distortions in the Semiscale system were identified during the simulations which require attention for future small break type experiments. Most notably, system external heat losses and pump seal leakage are physical distortions which must be eliminated or quantified to reduce potential limitations on the data from very small break tests in the Semiscale facility.

Very truly yours,

D. J. Olson, Manager
Semiscale Program

GGL:nt

Attachment:
As Stated

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SEMI-TR-010

SEMISCALE SIMULATIONS
OF THE
THREE MILE ISLAND TRANSIENT
A SUMMARY REPORT

SEMISCALE PROGRAM

JULY 1979

Prepared for the

U. S. Nuclear Regulatory Commission

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

DEPARTMENT OF ENERGY

IDAHO OPERATIONS OFFICE UNDER CONTRACT DE-AC07-76IDO1570

927 338

EG&G Idaho, Inc.

FORM EG&G 398
(Rev. 12-78)

INTERIM REPORT

Accession No. ______

Report No. SEMI-TR-010

Contract Program or Project Title:

Semiscale Program

Subject of this Document:

Simulation of the Three Mile Island Transient in Semiscale

Type of Document:

Test Results Summary

Author(s):

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Date of Document:

July 1979

Responsible NRC Individual and NRC Office or Division:

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Prepared for the
U.S. Nuclear Regulatory Commission
and the U.S. Department of Energy
Idaho Operations Office
Under contract No. DE-AC07-76ID01570
NRC FIN No. A6038

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INTERIM REPORT

NRC Research and Technical Assistance Report

SUMMARY REPORT ON SEMISCALE SIMULATIONS
OF THE THREE MILE ISLAND UNIT 2 NUCLEAR
POWER GENERATING STATION TRANSIENT

BY

R. W. Shumway
G. G. Loomis
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NRC Research and Technical
Assistance Report

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ACKNOWLEDGEMENT

Credit is due to those members of the Semiscale Program and its support organizations who contributed to this report. Special recognition is due to the members of the Semiscale Facilities and Test Operation Branch for conducting the experiments and tolerating our changing opinions concerning the experiment sequence. Thanks are due to the Design and Fabrication and Measurement Engineering and Experimental Instrumentation Branches for providing the new hardware and instrumentation required and making it function properly. Acknowledgement is made to J. E. Blakeley and M. King for assisting in the analysis and interpretation of data. Efforts extended by the Semiscale Data Processing Group in data reduction, processing, and production of data plots are greatly appreciated.

Thanks are also due to the Semiscale Text Processors, Denise Teuscher and Pamela Howard, who worked so hard and produced the various drafts (often from scattered bits and pieces) of this report. Credit for final preparation of figures and tables is due to Nancy Thornley, Linda Nelson, and Donna Rish.

SUMMARY

This report presents the results of a preliminary review and analysis of the data obtained from eight simulations of the Three Mile Island Unit 2 Nuclear Power Generating Station transient (March 28, 1979) that have been conducted in the Semiscale Mod-3 System.

The Semiscale simulations of the Three Mile Island (TMI) transient were basically conducted from the same sequence of events as those recorded in the plant. System initial conditions representative of those in the TMI system were established and the transient was initiated by terminating steam generator feedwater and steam valve flow. The steam generator secondaries were drained to control primary to secondary heat transfer. The pressurizer power operated relief valve, pressurizer code safety valve, and core power trip were operated on system pressure. High pressure safety injection was activated for about one minute during the Semiscale simulations. In addition, both primary loop coolant pumps were shut off in the Semiscale simulation at the same time that the Three Mile Island loop 2A pump was shut off. Results are briefly summarized in the following paragraphs.

During the first few hours of the TMI transient three periods were identified with distinct thermal-hydraulic events for each period. The first period included a rapid pressure transient. The second period was quasi-steady state with the primary pumps running. During the third period the pumps were turned off and core heatup occurred. The Semiscale simulations followed the basic thermal-hydraulic trends of the TMI transient for all three periods.

Early in time the rapid pressure transient was shown to be influenced greatly by steam generator heat transfer characteristics. Other system controls such as pressurizer and steam generator secondary side code safety valve openings and core decay power levels were also shown to influence timing and levels of system pressure maxima and minima early in time.

During the second time period voids in the system were increasing but the system pressure was fairly stable. Pump head degradation was more severe in Semiscale than it was in TMI but core temperatures remained low. The pressurizer level response was noted to be generally similar in trend to the measured plant pressurizer level behavior. Although there were shifts in the timing, the Semiscale level basically showed filling trends as the transient progressed It was clearly demonstrated that the pressurizer level was an inappropriate reflection of system mass inventory when the system was in a saturated two-phase state.

The third period began when the loop pumps were shut down which eventually led to core heatup. During this period the core mass inventory was noted to be decreasing and the cladding temperatures increasing even though the pressurizer remained liquid full. The cladding heatup rates were somewhat lower than those computed for adiabatic conditions. The exposed cladding surface heat transfer coefficients were shown to decrease as the core collapsed water level decreased. Application of the measured Semiscale heat transfer coefficients to the Three Mile Island Reactor core indicates that cladding temperatures in excess of 1500 K could have occurred at the 3.2 m elevation. Such temperatures are well into the temperature range where the Zirconium-water exothermic reaction is a dominant factor. The calculated results suggest that significant core damage could have occurred above the 2.9 m elevation in the reactor core.

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FIGURES

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FIGURES (contd)

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TABLES

I. INTRODUCTION

The occurrence of the Three Mile Island (TMI) Unit 2 Loss-of-Coolant (LOCA) incident has stimulated increased interest in the simulation, understanding, and calculation of thermal-hydraulic phenomena associated with extremely slow off-normal transients in nuclear power generating systems. To assist in the analysis of the TMI transient, and to investigate the capabilities of the Semiscale facility with regard to small break simulation, a series of experiments has been conducted in the Semiscale Mod-3 system. The experiments were conducted at the request of the United States Nuclear Regulatory Commission and were based on a test plan taken from the best available information regarding the sequence of events attendant to the TMI transient. The objectives of conducting these experiments were many. However, two main objectives can be cited. These are: (1) to gain a more fundamental understanding of the thermal-hydraulic phenomena which occurred in the TMI reactor, and (2) to determine the capability of Semiscale to duplicate the behavior measured in the TMI plant and to evaluate potential problems in conducting extremely slow LOCA transients.

The Semiscale facility is operated by EG&G Idaho, Inc., for the Department of Energy and the United States Nuclear Regulatory Commission. The facility is essentially a small-scale model of a typical four-loop nuclear reactor system and is used primarily to provide transient thermal-hydraulic data that can be used to assess and help develop computer codes used to calculate nuclear reactor response to large break LOCA's. The system contains most of the hardware found on large nuclear systems including a vessel, two active coolant loops with associated pumps and steam generators, and the associated peripheral systems such as the pressurizer and emergency core cooling (ECC) subsystems. The vessel includes a full-length upper plenum and upper head and an external downcomer. A full-length (3.65 m) electrically heated 25-rod bundle is contained within the vessel to provide a simulation of the nuclear core. The rods in the

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bundle have an axial chopped cosine power distribution similar to that found in nuclear fuel pins at middle-of-life. The vessel, loops, and core are extensively instrumented in order to provide information on pressure drop, volumetric flow, momentum flux, fluid density, and fluid and metal temperatures.

Two completely different classes of experiments were conducted by the Semiscale Program to support analysis of the TMI transient. Two experiments were conducted to investigate the behavior of a postulated noncondensible gas bubble in the vessel upper head and upper plenum during system depressurization late in the transient, and eight experiments were conducted to examine the thermal-hydraulics of the system for the early part of the transient (basically within the first two hour period). The experiments conducted to investigate the noncondensible gas behavior are discussed separately in Reference 1. The remaining eight experiments focused on simulation of the TMI transient, beginning with feedwater trip and culminating with core uncovery. The remainder of this report addresses these latter tests and is divided into five sections. Section II presents a more detailed description of the Semiscale facility and its associated hardware and a discussion of the scaling considerations and potential distortions in the system that may influence the results and their applicability to the TMI plant transient. Section III discusses the test plan used to conduct the Semiscale simulations and its relevance to the TMI sequence of events. The significant results of the numerous experiments conducted are presented and discussed in Section IV, followed by conclusions and recommendations in Section V.

II. SEMISCALE FACILITY DESCRIPTION

1. SYSTEM HARDWARE

The Semiscale Mod-3 facility is shown in an isometric view in Figure 1. The system consists of a high pressure vessel with associated internal hardware: upper head, upper plenum, core region, and external pipe downcomer. Two primary coolant loops are used to simulate the four loops in a typical Westinghouse commercial reactor system (reference plant). One Semiscale loop is scaled from the reference plant to simulate three circulating loops and the "broken" loop (also volume scaled) is used to simulate the single loop where a postulated double-ended break occurs. Both loops contain coolant circulation pumps and active U-tube in shell steam generators. The principal difference between the two loops, aside from the total fluid volume, is the relative elevations of the steam generators. Elevation relationships are maintained to match the reference plant in the broken loop whereas they are not maintained in the intact loop. Elevations of principal components in the vessel are generally maintained relative to the reference plant. The upper plenum and upper head contain the required hardware to represent a Westinghouse reactor equipped with upper head emergency core coolant injection.

The nuclear fuel rods in a reactor are simulated in the Mod-3 system using electrically heated rods. Each of the rods have dimensional and axial power generation characteristics typical of a nuclear fuel pin. Figures 2 and 3 depict the rod radial dimensions and the axial power generation distribution. The electrical power supplied to the rods is variable so that the power generated by a nuclear core during a given transient experiment can be approximated. The maximum Semiscale core power is 2 MW.

The Semiscale system is extensively instrumented both in the vessel and loop regions. Measurements taken in the loops generally include volumetric flow, momentum flux, fluid and metal temperatures,

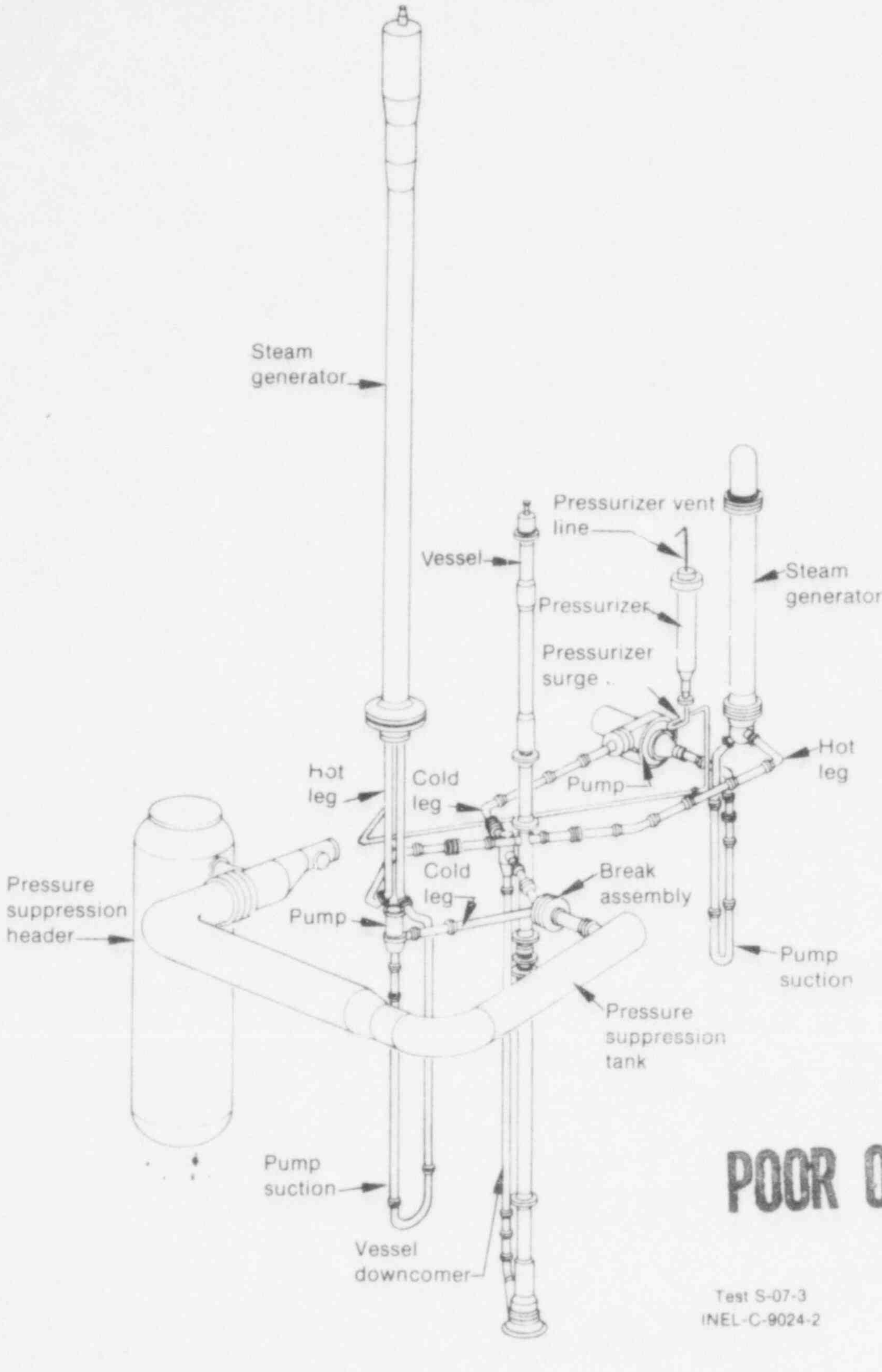

Fig. 1 Isometric of the Mod-3 system.

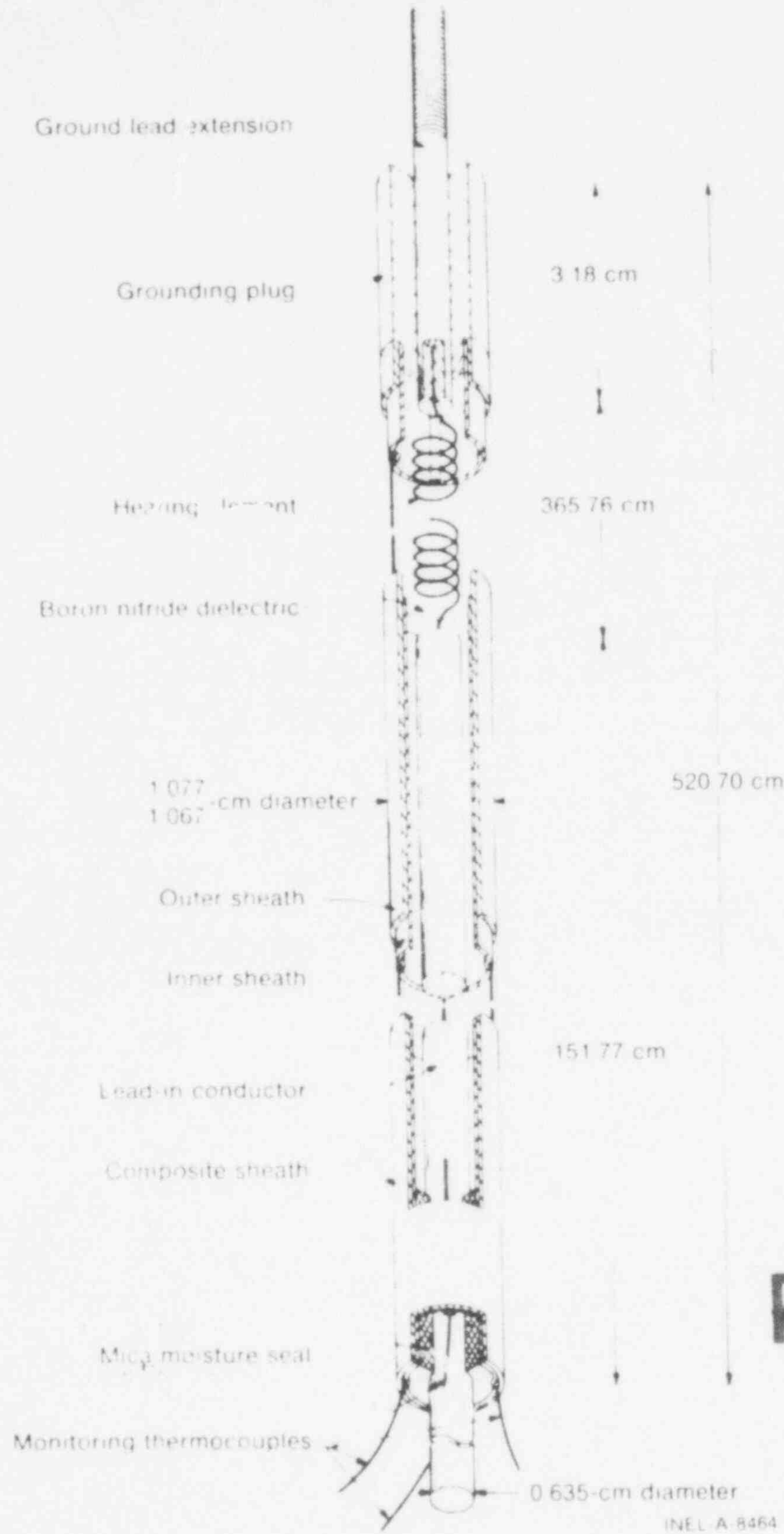

Fig. 2 Semiscale Mod-3 heater rod axial and radial dimensions.

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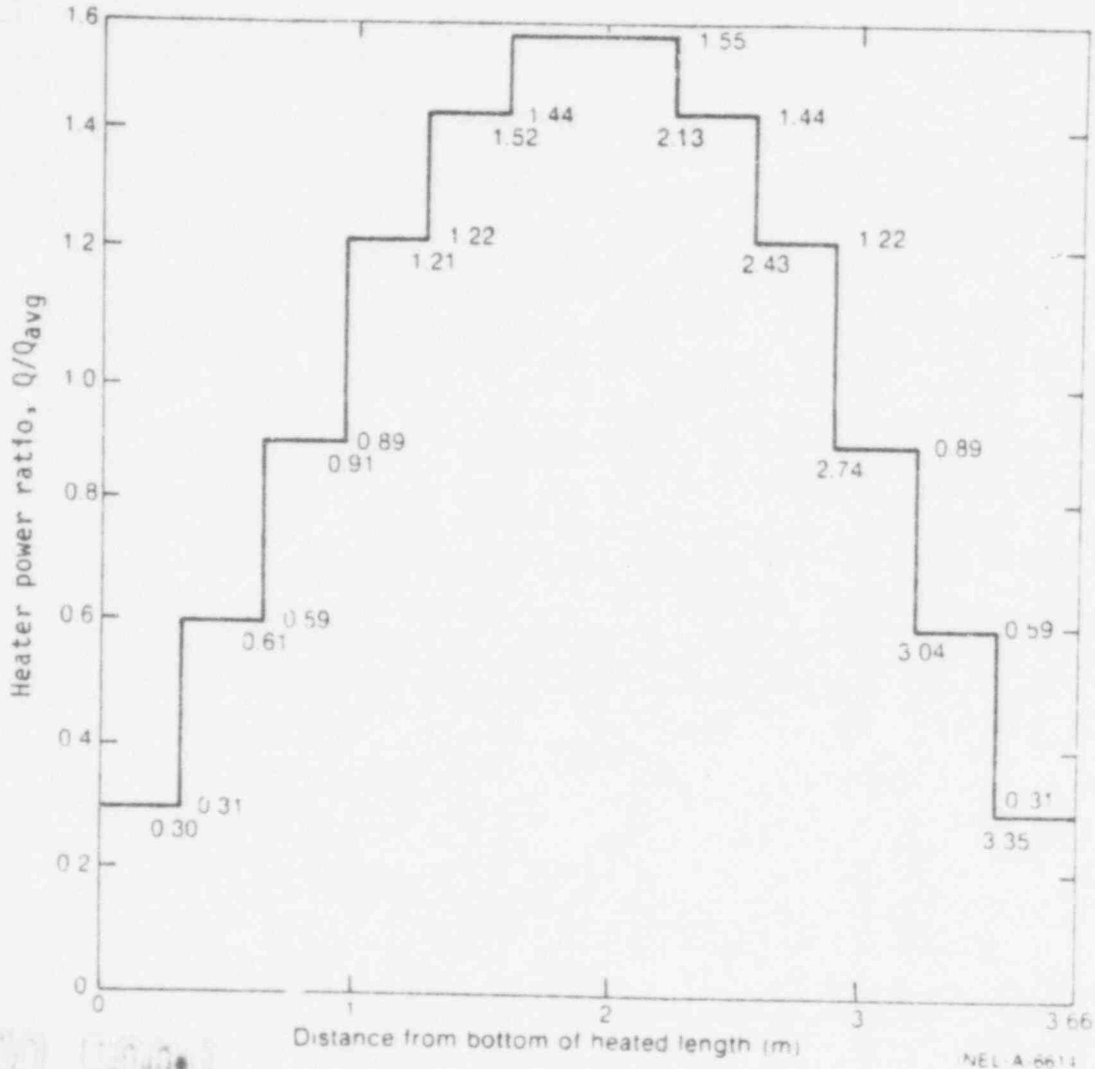

Fig. 3 Semiscale Mod-3 heater rod axial power distribution.

pressure, differential pressure, and fluid density. Figure 4 shows the general measurement locations in the Semiscale loops. The vessel measurements include essentially the same type of measurements as included in the loops. The external pipe downcomer is instrumented to obtain fluid density, metal and fluid temperature, and differential pressure at several different axial locations. The vessel is instrumented to obtain fluid temperatures, fluid density, and differential pressure at several different axial locations along the length of the heater rod bundle in addition to momentum flux and fluid density at the core inlet and outlet. The locations of in-core instrumentation (gamma densitometers and core inlet momentum flux device) are shown in Figure 5. Each of the individual heater rods is instrumented with six thermocouples that are located approximately 0.095 cm beneath the surface of the cladding. The thermocouples are located at various axial positions along the 3.66 m length of the rods. A plan view of the core showing the azimuthal (referenced to intact loop cold leg centerline) and axial (referenced to the bottom of the heated length) locations of the thermocouples is shown in Figure 6. Additional detailed information related to the Semiscale system design, hardware, and instrumentation can be found in Reference 2.

2. HARDWARE CHANGES REQUIRED TO SIMULATE TMI UNIT 2

As mentioned in the previous section, the Semiscale system was basically volume-scaled from a four-loop Westinghouse reactor design. In order to improve simulation of the TMI plant, which is a Babcock and Wilcox Company (B&W) 2 x 4 design (two hot legs, four coolant pumps and four cold legs), several significant changes to the Mod-3 hardware and operating procedures were made. These changes included the following:

(1) Addition of orifices, valves, and instrumentation to simulate the pressurizer code safety relief valves and the power operated relief valve (POV).

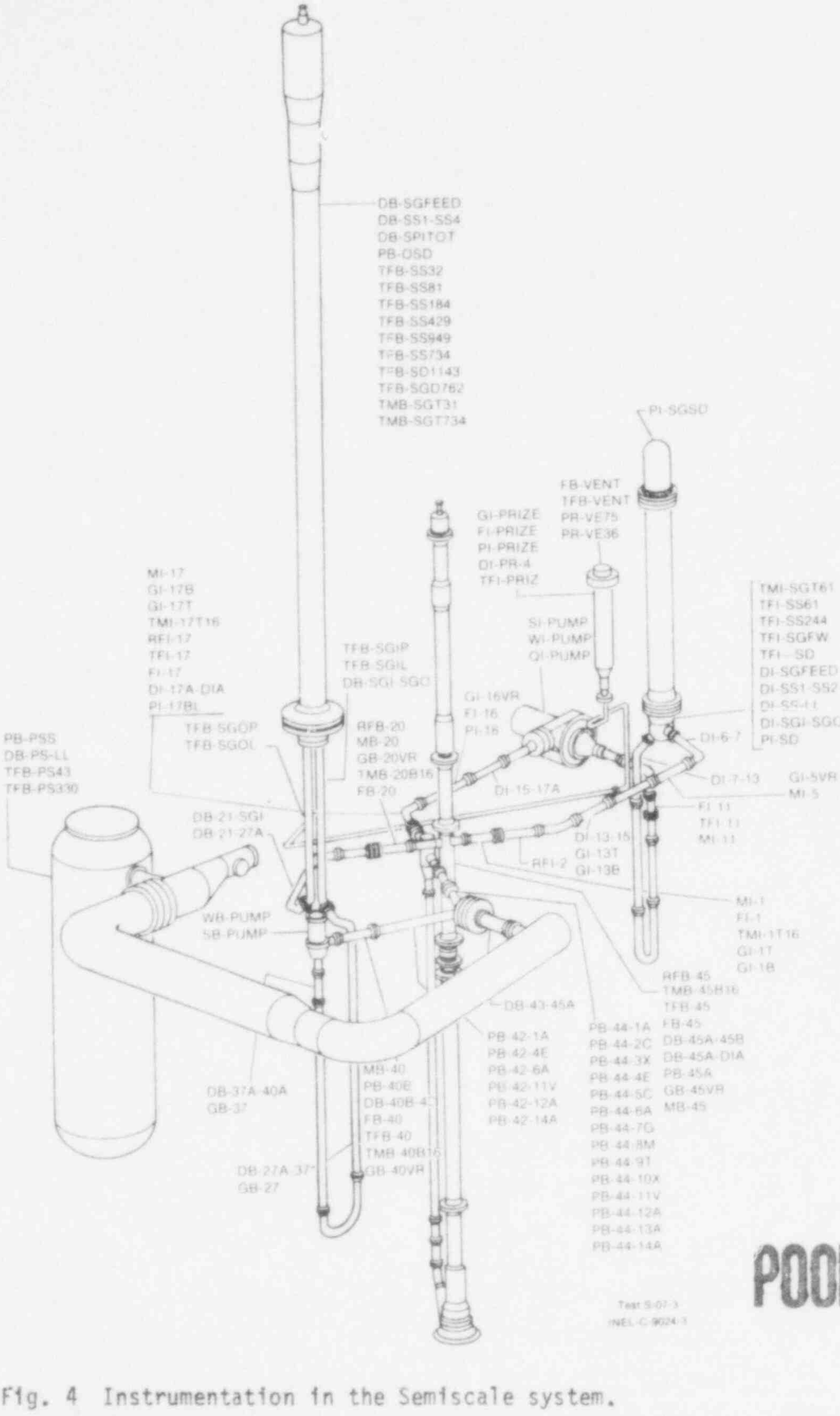

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Fig. 4 Instrumentation in the Semiscale system.

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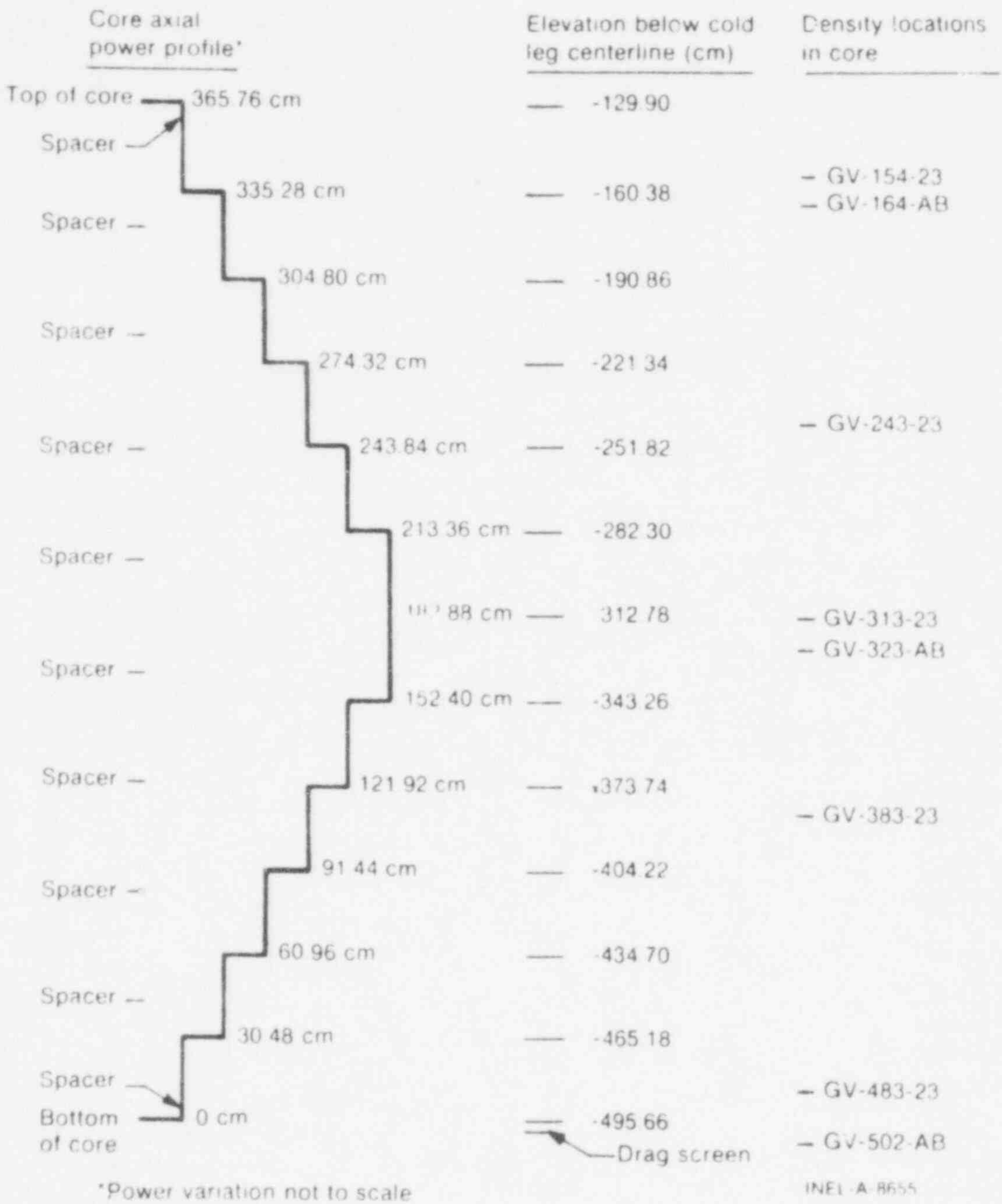

Fig. 5 Axial power profile in relation to vessel instrumentation.

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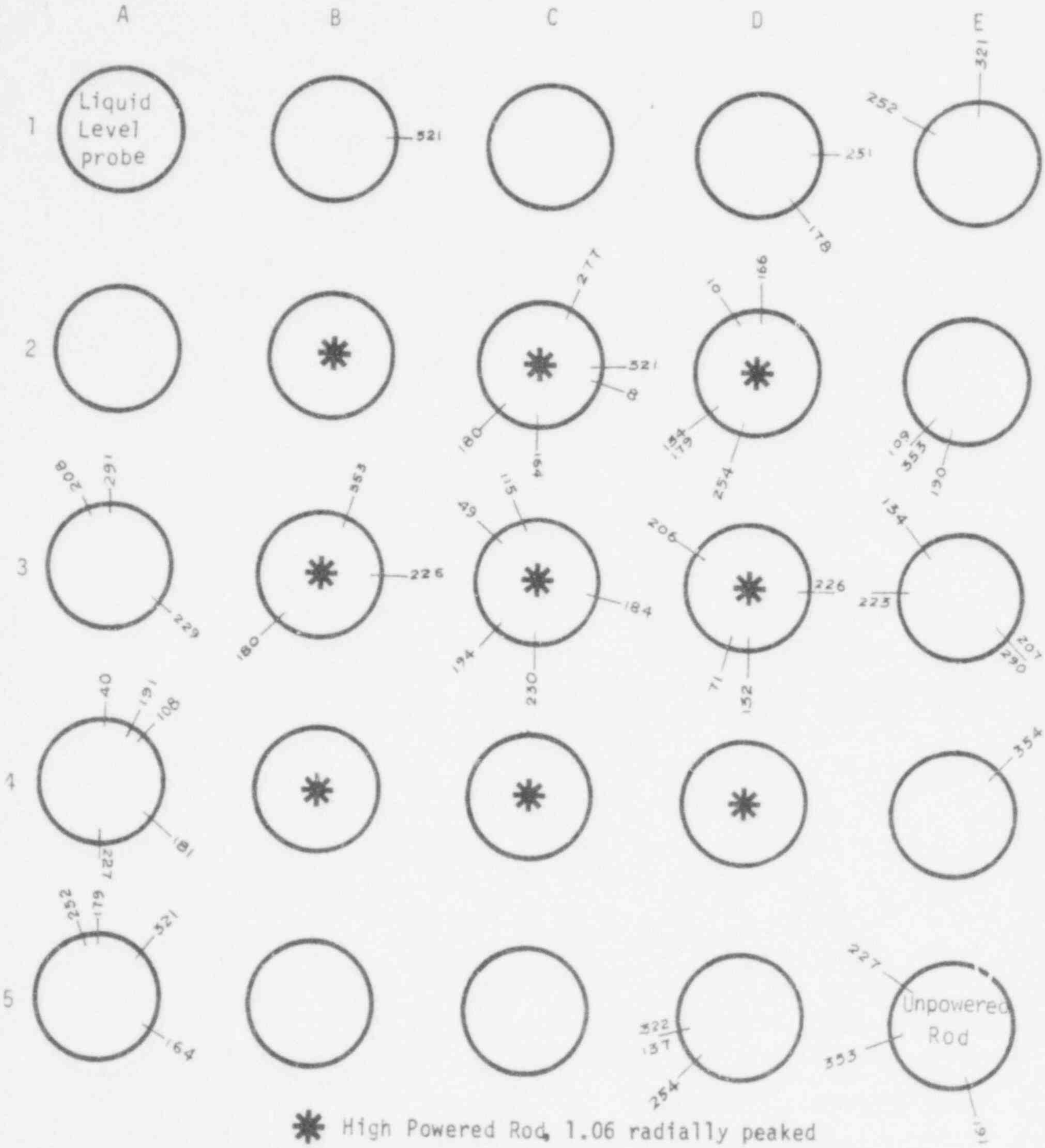

Fig. 6 Location of core heater rod thermocouples for TMI test.

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(2) Addition of piping, check valve, and instrumentation necessary to represent the B&W vent valve (connection between the vessel upper plenum and inlet annulus).

(3) The pressurizer surge line was modified to more closely simulate the hydraulic resistance, elevations, and point of connection to the loop hot leg in the TMI plant.

(4) Orifices were removed from the Semiscale intact loop steam generator and broken loop pump discharge in order to decrease loop hydraulic resistance so that increased loop flow rates could be obtained with the existing pumps.

(5) An orifice was installed in the vessel intact loop hot leg nozzle to achieve identical "top invert" elevations in the intact and broken loops.

The addition of the valves and instrumentation to the top of the pressurizer was required to simulate the normal venting capabilities and safety relief valves found in the B&W plant design. In the Semiscale system, the POV valve was simulated using an orifice with an area of 6.567E-3 cm^2 that was volume scaled from the B&W plant POV area (6.774 cm^2, 1.05 $in.^2$). The two pressurizer code safety relief valves in the B&W pressurizer were simulated in the Semiscale system using a single orifice with an area of 2.850E-2 cm^2. Additional details relative to the scaling of the orifices and the Semiscale hardware configuration can be found in Appendix A.

The B&W vent valves were simulated in the Semiscale system with a check valve and the piping required to connect the vessel upper plenum to the downcomer inlet annulus. The vent line hardware and instrumentation are addressed more fully in Appendix A.

The pressurizer surge line for the B&W plant is attached to a vertical section of pipe at the inlet side of the steam generator (see Appendix A for details). The surge line extends below the hot leg of the reactor system and then up to the pressurizer, thus creating a "loop seal" in the pressurizer surge line. The elevations of the surge line and the hydraulic resistance of the line simulated the TMI system.

To better simulate the TMI core fluid temperature differential (27.8 K, 50^{o}F) modifications to the Semiscale loop hydraulic resistances had to be made so that the current loop pumps could be used. These modifications included removal of the orifice at the inlet to the intact loop steam generator and removal of the orifice in the discharge of the broken loop pump (used to simulate locked rotor resistance).

The final significant hardware change made to the Semiscale facility was the addition of an orifice in the vessel hot leg outlet nozzle. This orifice was inserted to provide the same effective "top invert" elevation in the intact loop as the broken loop. This change was necessary because of the difference between the broken loop hot leg pipe and intact loop hot leg pipe areas (both pipes have the same centerline elevation) and the expected influence of this difference on draining of vessel fluid into the loops. Details of the orifice are discussed more fully in Appendix A.

3. OPERATING AND HARDWARE LIMITATIONS

Since the Semiscale Mod-3 system was designed primarily for the simulation of large break LOCAs, some of the operating and hardware limitations of the system that may adversely influence the results during slow transient testing are worthy of mention. Some of these limitations are primarily due to the fact that the system was scaled from a reference plant unlike the TMI system and others are problems inherent in small-scale systems such as Semiscale. The most significant limiting factors include:

(1) transient electrical core power control required to simulate a nuclear fuel rod,

(2) system external heat losses,

(3) steam generator design and elevation, and

(4) loop pump degradation characteristics.

The transient electrical core power and the techniques used in its derivation are discussed in detail in Appendix B. The primary system surface area to fluid volume ratio promotes atypical heat losses in Semiscale. The core power had to be increased to values above the scaled decay heat value in order to offset heat losses in the Semiscale system that are large relative to a PWR. Semiscale steam generator design differences result in secondary volumes that are oversized from a scaling standpoint* and, therefore, represent a heat sink (or source) that is disproportionately large. Steam generator secondaries were drained to help offset the effect of basic differences in the design of the Semiscale steam generators (U-tube in shell) relative to the TMI steam generators (once through design). In addition to the oversized secondary volumes in both loops, the steam generator elevations are significantly distorted (shorter) in the intact loop relative to the reference plant.

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* The Semiscale intact loop steam generator has a volume to surface area ratio that is a factor of 2.7 more than the TMI ratio. The broken loop steam generator volume to surface area is a factor of 5.3 larger than the TMI value.

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III. SEQUENCE OF EVENTS AND TEST PLAN

The transient at the TMI-Unit 2 plant was initiated by a loss of the condensate feed pumps in the steam generator secondary feedwater system. The loss of these pumps induced the main feedwater pumps to trip which in turn resulted in an electrical turbine trip and main steam valve isolation. Loss of turbine and feedwater pumps activated the auxiliary feedwater system pumps which should have started to refill the steam generator secondaries and thus maintain heat removal capability in the steam generators. The auxiliary feedwater pump isolation valves were closed, however, and auxiliary feedwater flow was not initiated until 8 minutes into the transient. Normally, it is expected that in a loss-of-feedwater transient system overpressure occurs and the pressurizer power operated relief valve (POV) opens to maintain safe pressure levels. Generally, a reactor scram does not occur since auxiliary feedwater and steam bypass flow are adequate to re-establish full heat removal capability.

In the case of TMI transient, the POV opened but the lack of auxiliary flow to the steam generators helped promote enough system overpressure to induce a reactor trip due to high pressure. In the events that immediately followed, it appears that the pressure may have continued to increase to a level at which the pressurizer code safety relief valves opened to decrease the system pressure. Current data is insufficient to establish whether this occurred. Unknown to the operators at this point in time, the POV had not reseated as it should have when system pressure subsided. An engineered safeguards (ES) activation occurred when the system pressure decreased to about 11 MPa (1600 psi) and the high pressure injection system (HPIS) started to inject water into the system. Activation of the HPIS in conjunction with system swell due to lack of steam generator heat rejection and the open POV resulted in an increase in the pressurizer level. To stem the increasing level (taken as an indication that the system was going water solid) the ES signal was bypassed and the HPI pumps were throttled. In addition, maximum letdown rates were established when the pressurizer level continued to increase. This chain of events, in addition to the fact that the POV remained open and continued to discharge mass from the system, eventually led to loop pump head degradation, core uncovery, and apparently a core temperature excursion as evidenced by radiation monitors, superheat in the hot legs, and the formation of noncondensible gas bubbles in the system (presumably from cladding-water reaction).

The scenario for this transient has been prepared based on discussions with the NRC and B&W. A sequence of events for the Semiscale simulations was assembled from these sources and is shown in Table I. Since the Semiscale system does not have closed loop secondary systems such events as the loss of condensate pumps and turbine trip were not simulated. The initiating event in the Semiscale simulations was the termination of feedwater flow, closure of the steam exit valves, and initiation of the steam generator drain. Generally, the experiments were conducted with actuating events that were similar to those believed to occur in the TMI plant. For example, opening of the POV and core scram were actuated on pressure setpoints of 15.55 MPa (2255 psig) and 16.24 MPa (2355 psig), respectively, which are the TMI setpoints. There are several unknown aspects relative to the actual TMI plant transient that required a certain amount of educated speculation in order to complete the Semiscale test plan, such as the value of the actual HPIS flow rate as a function of time and the letdown/makeup flow histories. Letdown flow was not simulated in the Semiscale experiments. Makeup flow was simulated to the extent required to account for Semiscale loop pump seal leakage rates. The HPIS flow characteristics were generally as listed in Table I except where noted otherwise.

Several attempts were made to conduct the most appropriate simulation of the TMI transient in the Semiscale facility. In some cases, the actual sequence of events deviated slightly from that listed in Table I because of unplanned actions or required changes due to results obtained during the course of the transient. Table II

TABLE I

PLANNED SEQUENCE OF EVENTS FOR SEMISCALE SIMULATIONS OF TMI

Time(s)	Event
0	Terminate feedwater, close steam valves, initiate draining of steam generator secondaries
5 (approximate)	Open pressurizer power operated relief valve when hot leg pressure reaches 15.55 MPa (2255 psig).
8	Turn pressurizer heaters off and set to automatic control.
12 (approximate)	Scram core power when hot leg pressure reaches 16.24 MPa (2355 psig).
60	Steam generator secondary level reduced to approximately 10% of initial level.
120	Initiate high pressure injection (HPIS) when hot leg pressure reaches 11.03 MPa (1600) psig. Flow rate will be 42 ml/s.
180	Terminate HPIS flow.
300	Steam generator secondary should be depleted.
4400	Throttle intact loop pump to decrease core flow to approximately one-half of original.
6000	Terminate power to loop pumps. Reinitiate HPIS flow at a rate of 20.8 ml/s.

TABLE II

SEQUENCE OF OCCURRENCE (SEC)

PARAMETERS & COMPONENTS	S-TMI-3A	S-TMI-3B	S-TMI-3C	S-TMI-3E	S-TMI-3F	S-TMI-3G	S-TMI-3H	S-TMI-3I
Feedwater off	0	0	0	0	0	0	0	0
POV opening (FB-VENT)	Not opened	0	12.5	8.2	7	6.9	8.5	38.2
POV close (FB-VENT)	N/A	1417	8870	Not closed	Not closed	Not closed	Not closed	Not closed
Core power trip (VH-HI - VH-LO)	29	N/A	5	21	18	20	17	12
CSV opening (FB-VENT)	30	Not opened	Not opened	21.4	Not opened	Not opened	Not opened	Not opened
CSV close (FB-VENT)	33.8	N/A	N/A	25.6	N/A	N/A	N/A	N/A
ISGRV open (PI-SD)	Uncertain	Not Opened	12.5	17	14	14	17	13.5
BSGRV open (PB-SD)	Not opened	Not opened	Not opened	Not opened	Not opened	Not opened	Not opened	Not opened
ISG drain complete (DI-SG-LL) (final liquid level, m)	20 (0.4)	41.5 (1.9)	23.3 (0.4)	17.5 (0.6)	25 (0.35)	31 (2.53)	17 (0.65)	13.5 (0.92)
BSG drain complete (DB-SS1-SS4) (final liquid level, m)	89 (0.7)	69 (0)	66.3 (0.4)	25 (0.5)	22 (0.4)	25 (3.2)	24 (0.55)	30.1 (0.61)
HPIS on-off (FI-HPIS)	259 - 343	Off	164.9 - 224.1 6039 - 7260	Off	Off	Off	Off	215 - 291 7078 - 7685.5
Core uncovery	5950	913	early 6351 late 7758	N/A	N/A	N/A	N/A	6462
First increase of core temperature due to uncovery	5980	965	early 6435 late 7839	N/A	N/A	N/A	N/A	6500

POV: Power Opening Valve

CSV: Core Safety Valve

ISGRV: Intact Loop Steam Generator Relief Valve

BSGRV: Broken Loop Steam Generator Relief Valve

HPIS: High Pressure Injection System

N/A: Not Applicable

lists the experiments conducted and some of the more important events that occurred in th tests. With the exception of one experiment (Test S-TMI-3B) the initial conditions from which the experiments were conducted were essentially the same (steam generator liquid level is an exception). Table III lists the initial conditions for all of the experiments.

TABLE III

INITIAL CONDITIONS FOR SEMISCALE TMI TRANSIENTS

PARAMETER	TMI-3A	TMI-3B	TMI-3C	TMI-DRIO(E)	TMI-3F	TMI-3G	TMI-3H	TMI-3I
Pressure (Mpa) (PI-PRIZE)	15.2	12.2	14.74	15.1	14.97	14.89	15.05	15.1
Core power (kW) (AH-HI & LO / VH-HI & LO)	2030.0	120	2120.0	1940.0	1943.0	1950.0	1950.0	1960.0
Core flow (ℓ/s) (FV+1)	15.6	12.7	16.04	15.68	15.94	15.91	15.89	15.94
Core differential temperature (K) (TFI-1 & TFI-17)	31.4	1	29.8	29.1	29.3	29.15	29.7	29.9
Hot leg temperature (K) (TFI-1)	596.9	559	595.0	594.5	594.5	593.85	594.4	594.1
Cold leg temperature (K) (TFI-17)	565.5	558	565.2	565.4	565.2	564.7	564.7	564.2
*SG level intact loop (m)	3.0	4.0	3.0	3.5	3.0	3.15	3.1	3.0
*SG level broken loop (m)	10.9	7.6	9.2	3.0	3.6	3.23	3.6	4.1
*Pressurizer level (m)	0.53	0	0.3	0.82	0.76	0.72	0.72	0.75

* The calculations were made from differential pressures.

IV. DISCUSSION OF RESULTS

The Three Mile Island simulations in the Semiscale system were made to gain greater insight into the actual thermal-hydraulic events that transpired during the first few hours of the Three Mile Island transient. In addition, the Semiscale simulations were used to determine the capability of the Semiscale system to follow the major thermal-hydraulic trends of a large scale system during a slow loss-of-coolant type transient. Specific thermal-hydraulic parameters that were examined for the Semiscale simulations include system pressures, core collapsed liquid level, core heater rod temperatures, steam generator and pressurizer liquid levels, loop fluid temperatures and loop flow rates.

The thermal-hydraulic events noted in the Semiscale simulations of the first few hours of the TMI transient can be divided into three distinct time periods. The first period consisted of a rapid system pressure transient, core power trip, and pressurizer POV opening following the steam generator feedwater trip. In the discussion that follows, special attention is given to the pressurizer water level and the various parameters that affect system pressure during the early part of the transient. This first section of the discussion considers the time period from feedwater trip (time zero) to about 60 s. The second period includes the time from approximately 60 s until the loop pumps were shutdown at 6000 s. This period was characterized by quasi steady-state system behavior during which the pressurizer filled, the loop pumps gradually cavitated, and the mass inventory in the system gradually decreased. The third period of interest in the transient occurred after the primary loop pumps were shut down and includes uncovering of the core heater rods, the resultant core heatup, and the increase in hot leg fluid temperatures in the Semiscale system. Each of these time periods is discussed more fully in the following subsections. In addition an estimate of the TMI-2 core heatup transient is made based on Semiscale core heat transfer measurements.

1. EARLY SYSTEM THERMAL-HYDRAULIC RESPONSE

In this section the early time response of the Semiscale system and its relation to the TMI plant response are discussed. Those conditions in the system that were found to have an influence on the thermal-hydraulics are delineated.

1.1 Semiscale System Response Relative to TMI Response.

As discussed in the sequence of events section, termination of steam generator feedwater flow during the TMI transient resulted in a system pressure transient because of the sudden decrease in heat removal rate from the system. Several Semiscale experiments were conducted in an attempt to simulate this pressure transient and associated thermal-hydraulic events. Figure 7 shows a comparison of the system pressure from the TMI plant and the result from Semiscale Test S-TMI-3E. The Semiscale simulation follows the trends of the TMI transient; however, several events are shifted in time. As indicated in the figure, the initial Semiscale pressure rise rate is not as high as the TMI pressure rise rate in the 0 to 8 s time period. The difference is attributed primarily to steam generator heat transfer. Steam generator design differences between Semiscale and TMI allow a higher energy removal rate in Semiscale as discussed in Section II.3. The excess heat removal potential effectively limited the rate of system pressure rise in Semiscale during the first 8 s. At 8 s there was an unexpected inflection in the Semiscale pressure rise rate when the POV opened. A simple reduction in rise rate was expected. A possible reason for the inflection is that the POV line geometry allowed condensate to collect upstream of the valve. The condensate could momentarily allow a higher sonic velocity at the valve than would have existed with steam only flow. The condensate induced moderation further shifted the Semiscale data relative to TMI. Another event that caused the Semiscale pressure rise to moderate was the opening of the relief valve on the secondary side of the steam generator (intact loop) at about 14 s. The secondary side blowdown

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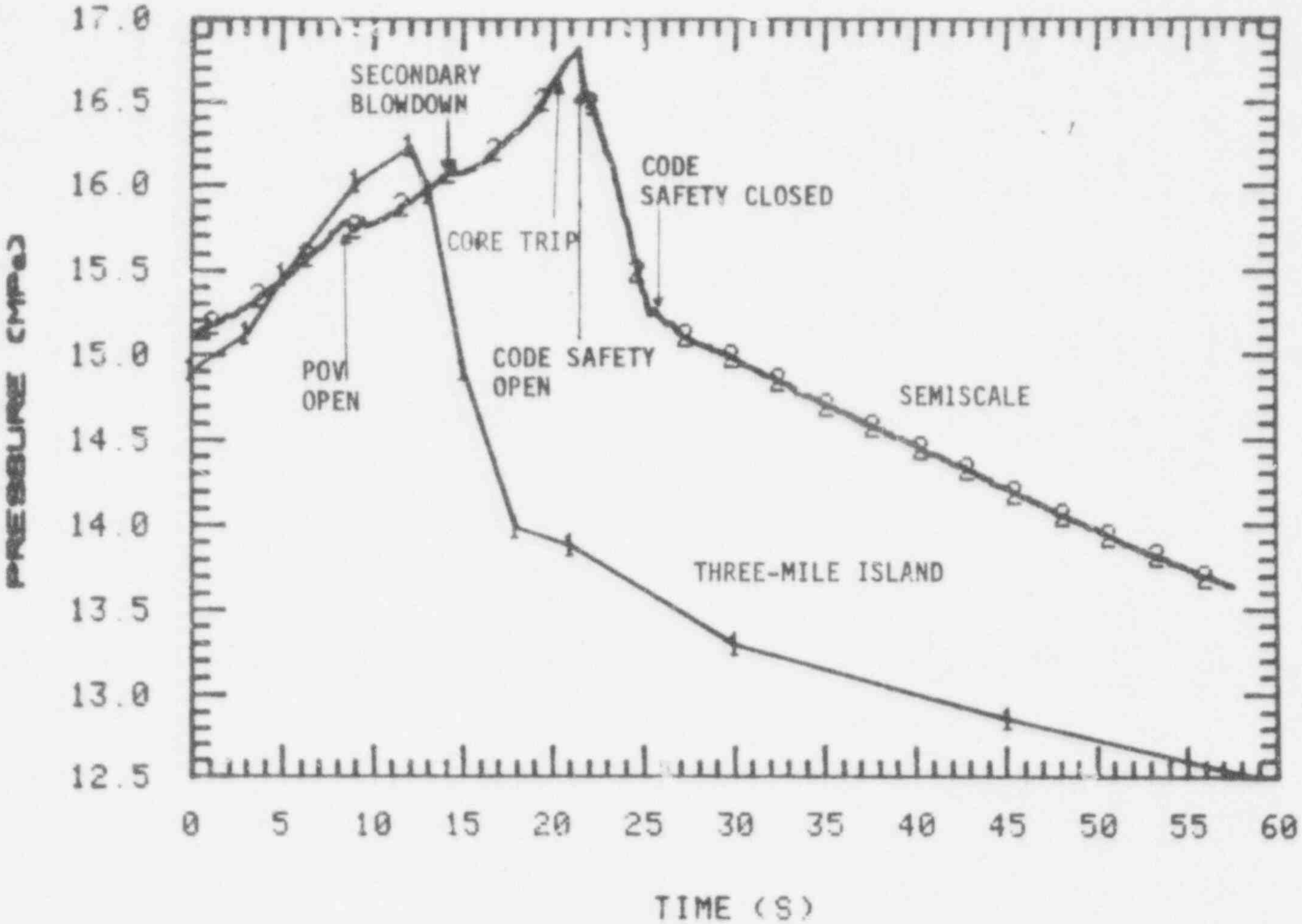

Fig. 7 Comparison of system pressures from Three Mile Island and Semiscale Test S-TMI-3E.

increased heat transfer between the primary and secondary because of the secondary side temperature decrease and increased turbulence. In addition, the Semiscale core scram was actuated about 4 s late in the Semiscale experiment. By then, the rate of primary system energy removal by the steam generators was low enough that the system pressure kept rising.

The pressurizer code safety valves opened and induced the rapid pressure reduction at approximately 21 s. The code safety valve closed in the Semiscale test at 25 s when the system pressure dropped below the valve closure setpoint (15.3 MPa). The rate of pressure reduction in TMI and the Semiscale systems (at 12 s in TMI and 21 s in Semiscale) was approximately the same. This suggests that the pressurizer code safety valves may have activated in the TMI transient. All available data sources from the plant indicate that the pressure did not reach the setpoint required (16.89 MPa) for code safety valve activation. However, the TMI valve set points could have been reduced considerably due to "weeping". For this particular test, the code safety valve closure contributed to the sudden reduction in depressurization rate at 25 s. However, flattening of the pressure trace also occurred during Semiscale tests in which the code safety valve had not actuated primarily due to flashing in the pressurizer as the pressure fell to the saturation value where flashing and void formation in the pressurizer liquid tended to reduce the rate of pressure decrease. The fact that the prominent TMI pressure slope change at 18 s occurs 1 MPa below the initial system pressure suggests several possibilities: (1) If the code safety valve were opened initially, it closed below the closure set point. (2) Pressurizer sprays reduced the saturation pressure. (3) Although the pressurizer began flashing at the initial pressure, steam generator energy removal was dominant. As will be shown later, the TMI average system temperature decreased much faster after core scram than it did in Semiscale. The reduction in system temperature allowed the hot leg pressure (shown in Figure 7) to be less than the pressurizer pressure and caused the pressurizer liquid level to fall. A more accurate TMI pressure curve is needed to determine whether or not the code safety valve opened*.

Following the knee in the pressure curves, the depressurization rates for TMI and Semiscale were similar, indicating that the POV exit mass flux values were similar. A comparison of the measured POV flow rate and the values predicted by the homogenous equilibrium critical flow model (HEM) for the time period between 25 and 60 s, is shown in Figure 8. The HEM values were calculated using the measured Semiscale pressurizer pressure and assuming saturated steam conditions. If the HEM values are multiplied by a factor of 0.84 the POV flow is predicted quite adequately. Use of the factor of 0.84 has been found necessary in previous work relative to large break critical flow(3). These results suggest that for computer code simulations of the TMI transient in which the HEM model is used to predict the POV flow, the 0.84 factor should be considered in order to promote an accurate calculation of the mass inventory and pressure response.

One of the items of interest in the Semiscale simulations was the pressurizer level response and its relation to the TMI-2 pressurizer level behavior. Figure 9 shows a comparison of the normalized pressurizer levels from the TMI-2 plant and one Semiscale simulation. In both facilities, the levels start to increase at time zero because of the decrease in secondary heat removal rate and the resultant increase in bulk temperature (Figure 10) and fluid swell (increase in fluid specific volume) of the fluid in the system. As can be seen from a comparison of the data in Figures 7, 9, and 10, the pressurizer levels basically follow the trends in the system pressures and bulk fluid temperature. Pressurizer level changes are correlated with the system bulk liquid specific volume changes due to the rise in bulk temperature. The differences in bulk temperature response between the two systems are primarily related to the steam generator heat removal

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* The TMI pressure data was extracted from 8 minute plots.

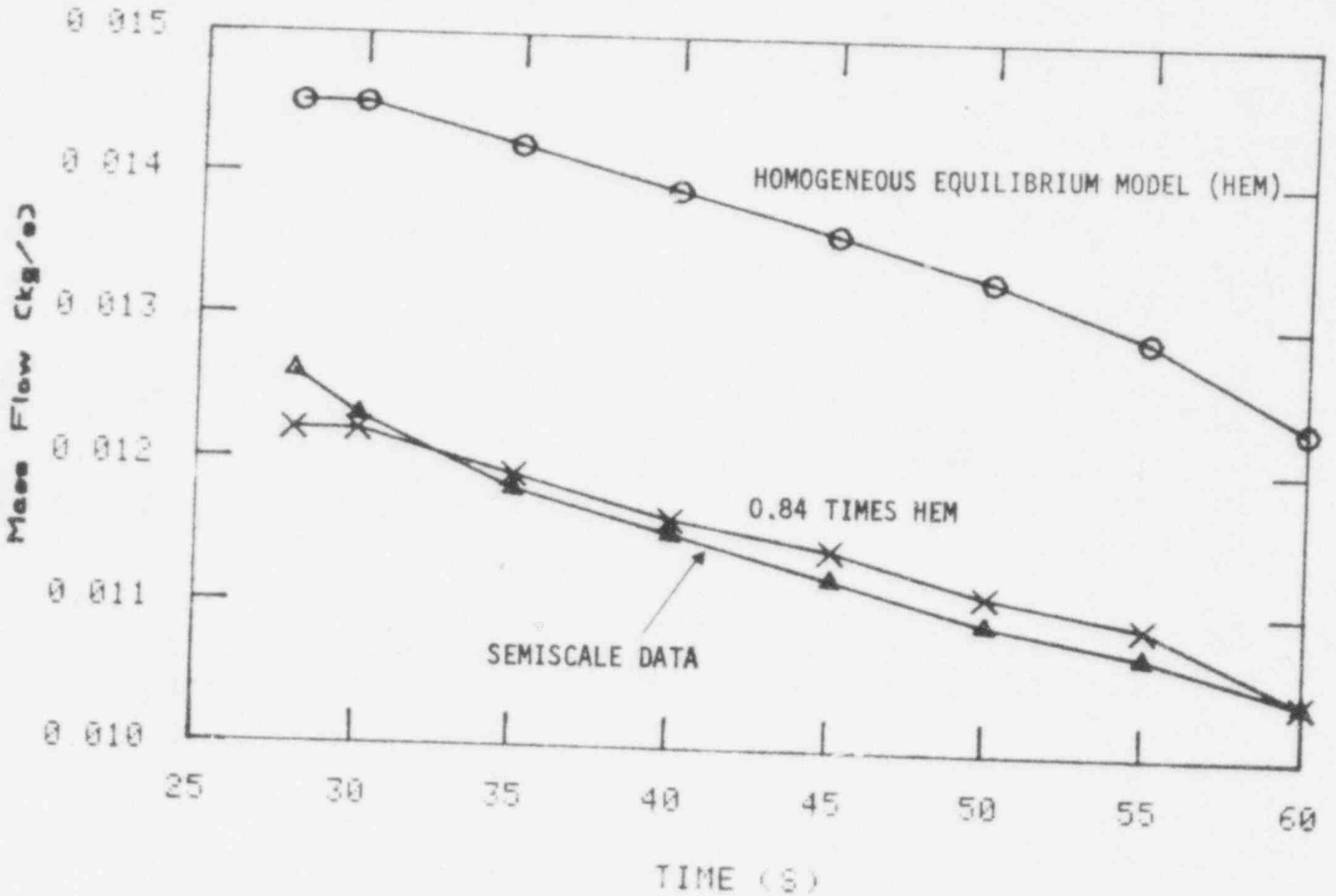

Fig. 8 Comparison of POV flow from Semiscale Test S-TMI-3E and flow predicted from homogeneous equilibrium model.

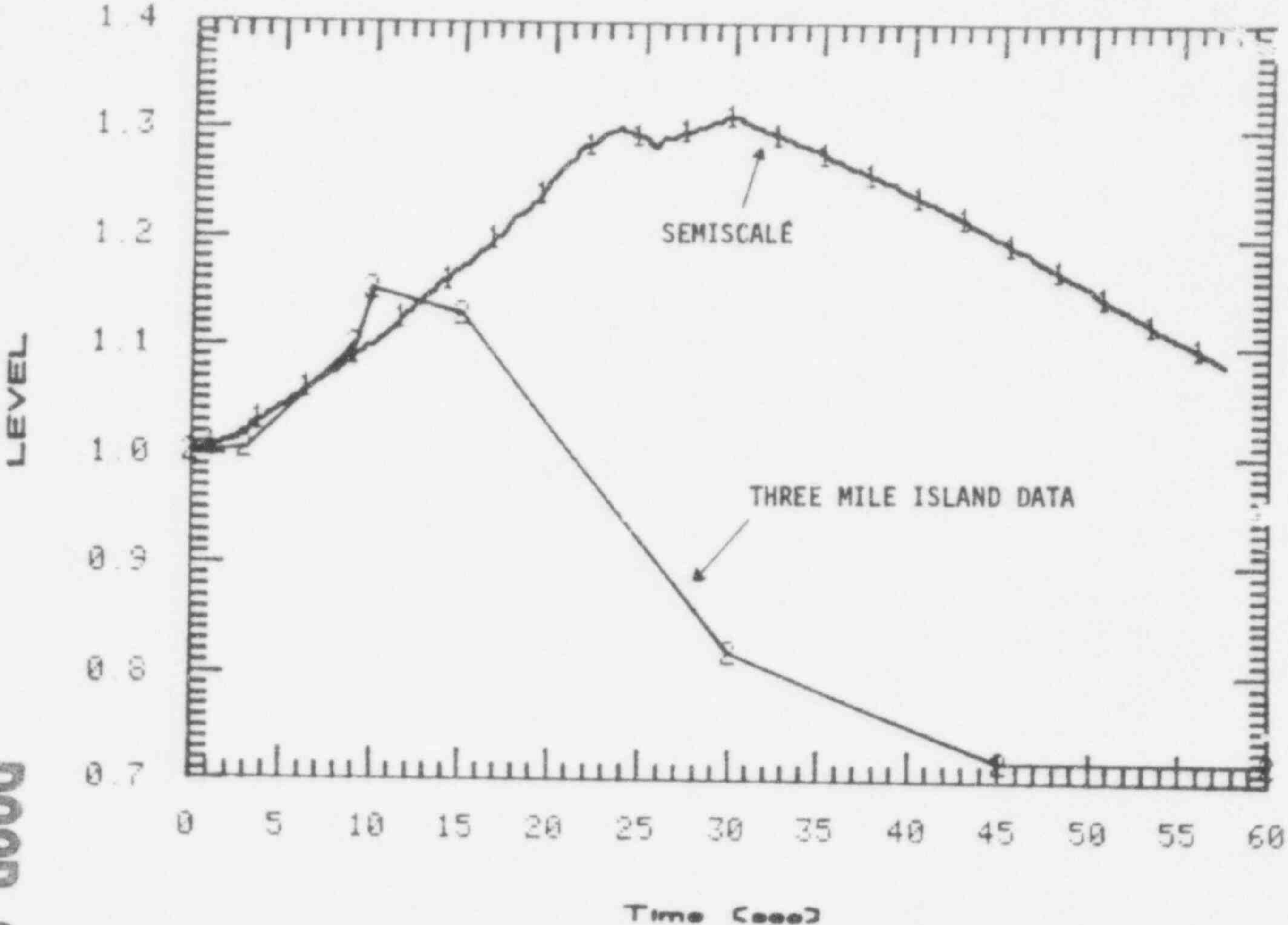

Fig. 9 Comparison of normalized pressurizer liquid levels from Three Mile Island data and from Semiscale Test S-TMI-3E.

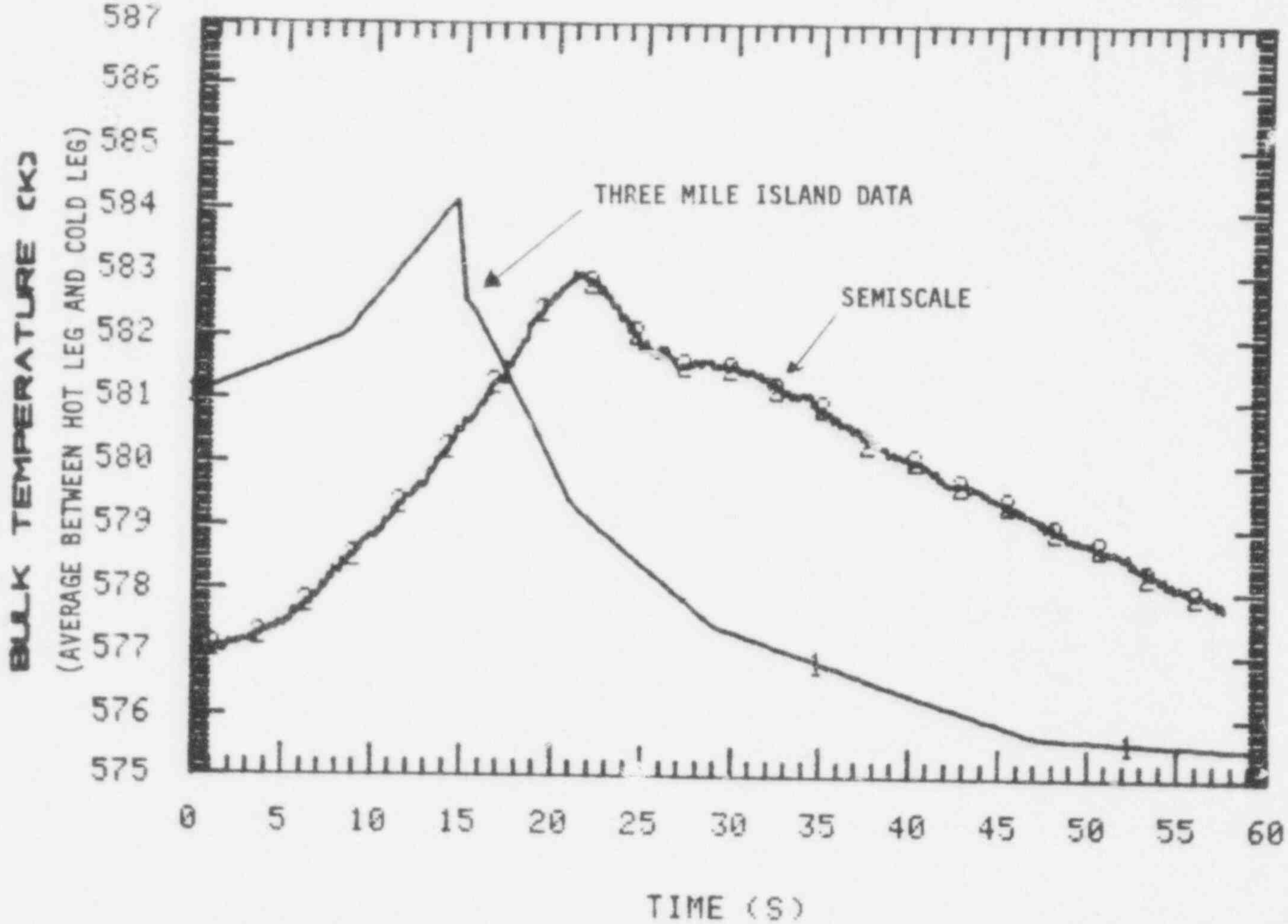

Fig. 10 Comparison of bulk fluid temperatures from Three Mile Island data and Semiscale Test S-TMI-3E.

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characteristics. Other Semiscale simulations show that the bulk temperature and thus the pressurizer level can be controlled by the steam generator secondary drain rate. The drain rate used for the test shown in Figure 9 was too fast to get a good match of the TMI pressurizer level drop following core scram.

The data comparisons presented so far have shown that the Semiscale system response to the TMI sequence of events showed trends similar to the TMI thermal-hydraulics. Although the exact TMI plant response was not simulated, the Semiscale results demonstrated that the data could be used to infer and help validate the plant thermal-hydraulic measurements.

1.2 Conditions Influencing Semiscale System Response

Several Semiscale experiments were conducted to investigate the effect of system hardware and controls on the early thermal-hydraulic response. Some of the variations investigated included POV opening time, steam generator secondary drain rates and relief valve actuation, and core decay power after scram. These tests were done in an effort to help establish whether or not the Semiscale pressurizer code safety valve had to open in order to simulate the TMI pressure response.

The influence of the POV valve opening time on the system pressure response is illustrated in Figure 11. In Test S-TMI-3I, the opening of the POV was delayed until 40 s to test the hypothesis that, in the TMI transient, the POV isolation valve was closed (so in effect there was no mass discharge from the system) until an operator action at 40 s opened the valve. For Test S-TMI-3F, the POV was opened at the normal set pressure of 15.51 MPa as is evidenced by the inflection in pressure oscillation at 9 s. As expected, failure to open the POV caused a more rapid pressure rise and, therefore, core scram at an earlier point in time relative to the case in which the POV was opened. With the exception of the time shift, the pressure decrease

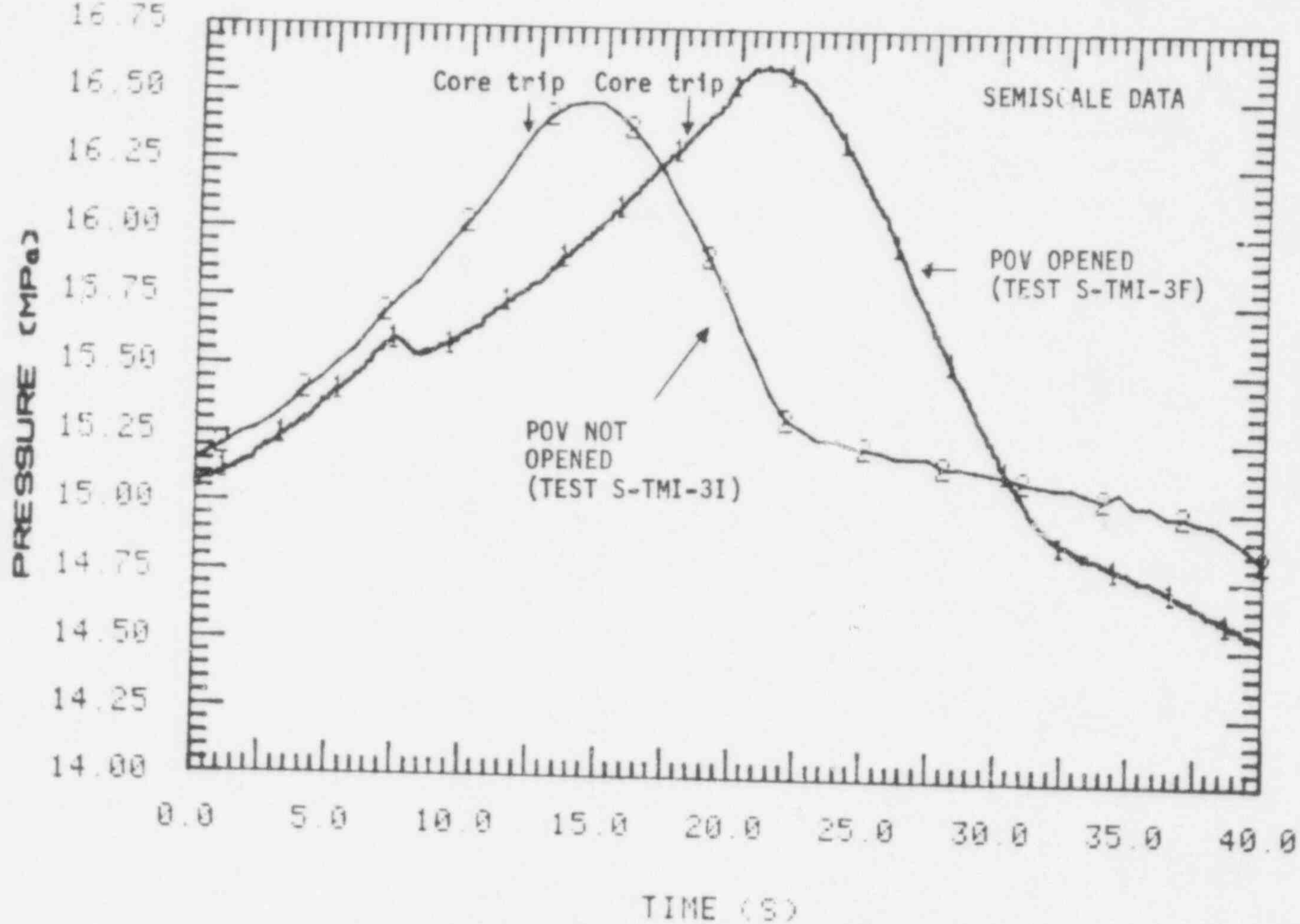

Fig. 11 Influence of POV status on system pressure response - Semiscale Tests S-TMI-3F and S-TMI-3I.

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that occurred shortly after core scram in each experiment was relatively unaffected by the POV status. The pressure response is dominated by steam generator heat transfer and core scram.

The influence of the steam generator secondary mass inventory on system pressure is shown in Figure 12. Figure 12 compares a Semiscale simulation in which there was no draining of the steam generator secondary to one in which draining was imposed. For both cases shown, the core power was decreased to zero after core scram. Although the differences are slight between the two simulations, a 60 kPa/s difference in the depressurization rate exists after peak pressure is recorded. The secondary liquid levels shown in Figure 13 reflect the different mass inventory in each steam generator for each simulation.

In addition to the secondary mass inventory, actuation of the secondary side relief valves was noted to have an effect on the primary system pressure response. Overpressurization of the secondary actuates relief valves which in turn causes a secondary pressure reduction and promotes boiling (high heat transfer rates). The pressure plateau in the Semiscale simulation shown in Figure 12 for Test S-TMI-3G is a result of secondary side blowdown. Actuation of the secondary relief valves would be expected to induce a change in slope of the primary pressure. This effect can be seen in Figure 14 which shows a comparison of the system pressure and the intact loop steam generator secondary pressure for one of the Semiscale simulations. The change in slope of the secondary pressure at 14 s is an indication that the secondary relief valves opened. At approximately the same time, the primary system pressure rise rate moderated, reflecting an increase in the secondary heat transfer rate. The secondary relief actuation caused the point at which the peak pressure was reached to be shifted in time.

Another parameter that influences the system pressure is the core power generation rate. Two different experiments were run with different decay heat values to examine the influence of the decay heat

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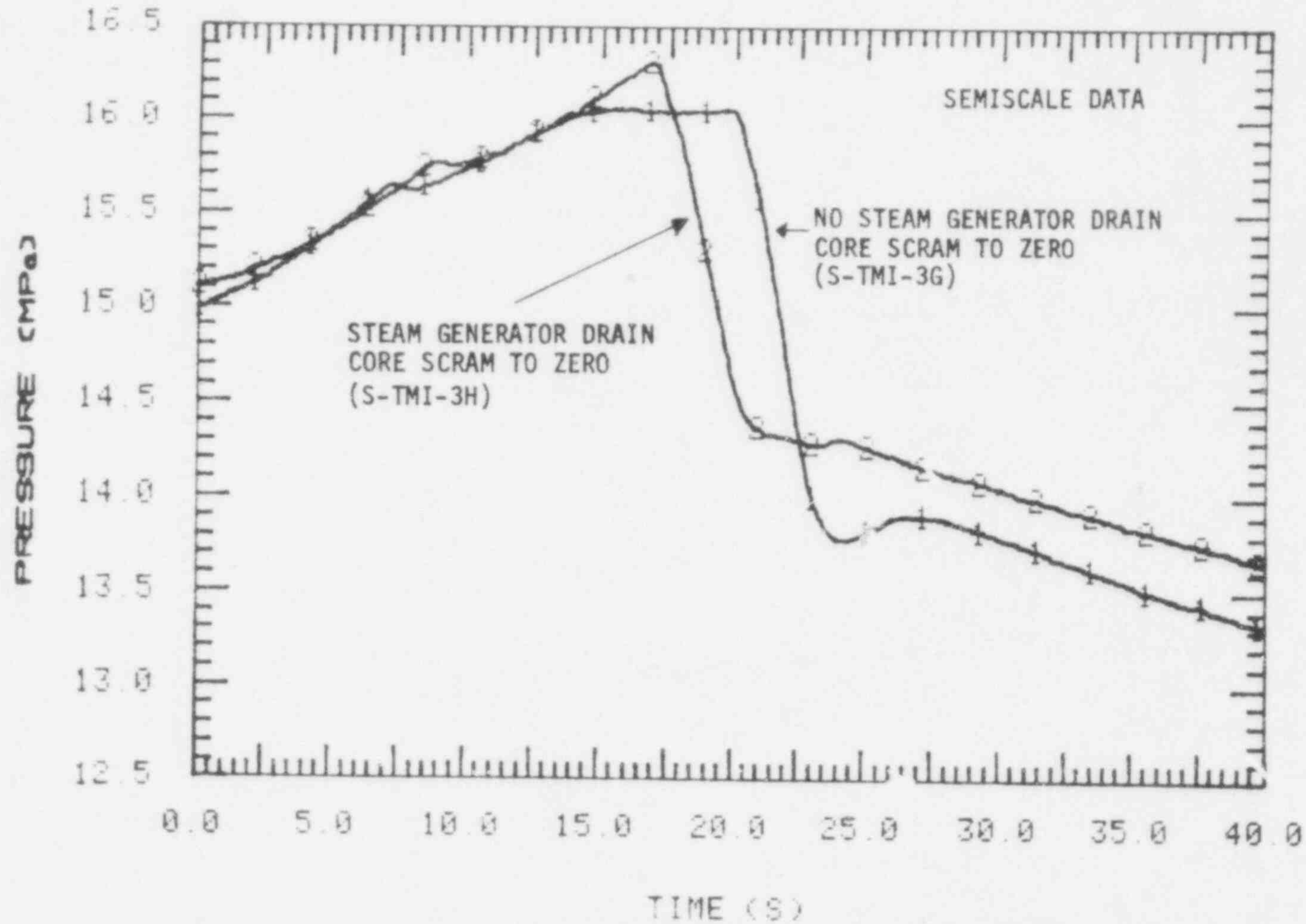

Fig. 12 Influence of steam generator secondary drain on system pressure for Semiscale Tests S-TMI-3G and S-TMI-3H.

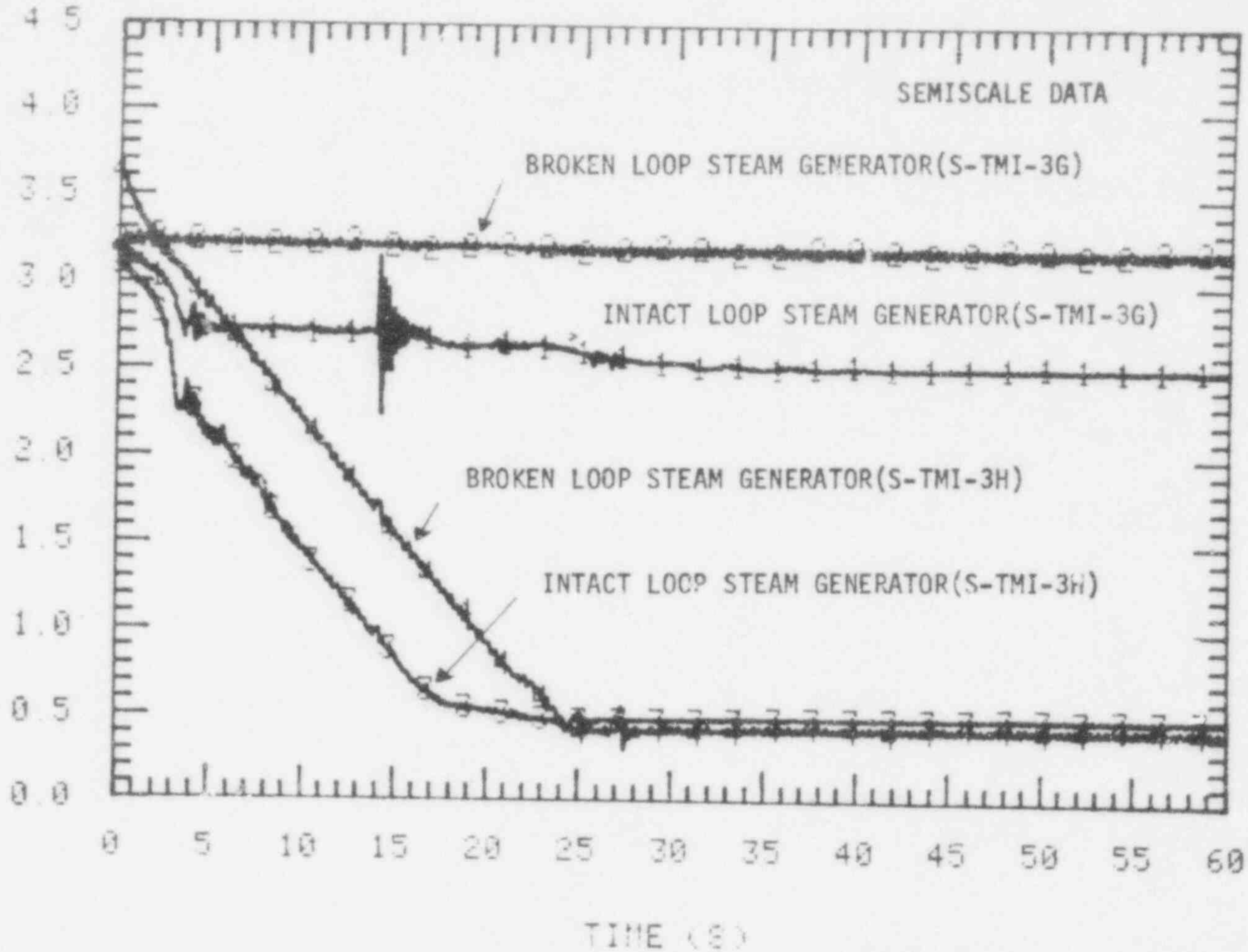

Fig. 13 Steam generator secondary liquid levels for Semiscale Tests S-TMI-3G and S-TMI-3H.

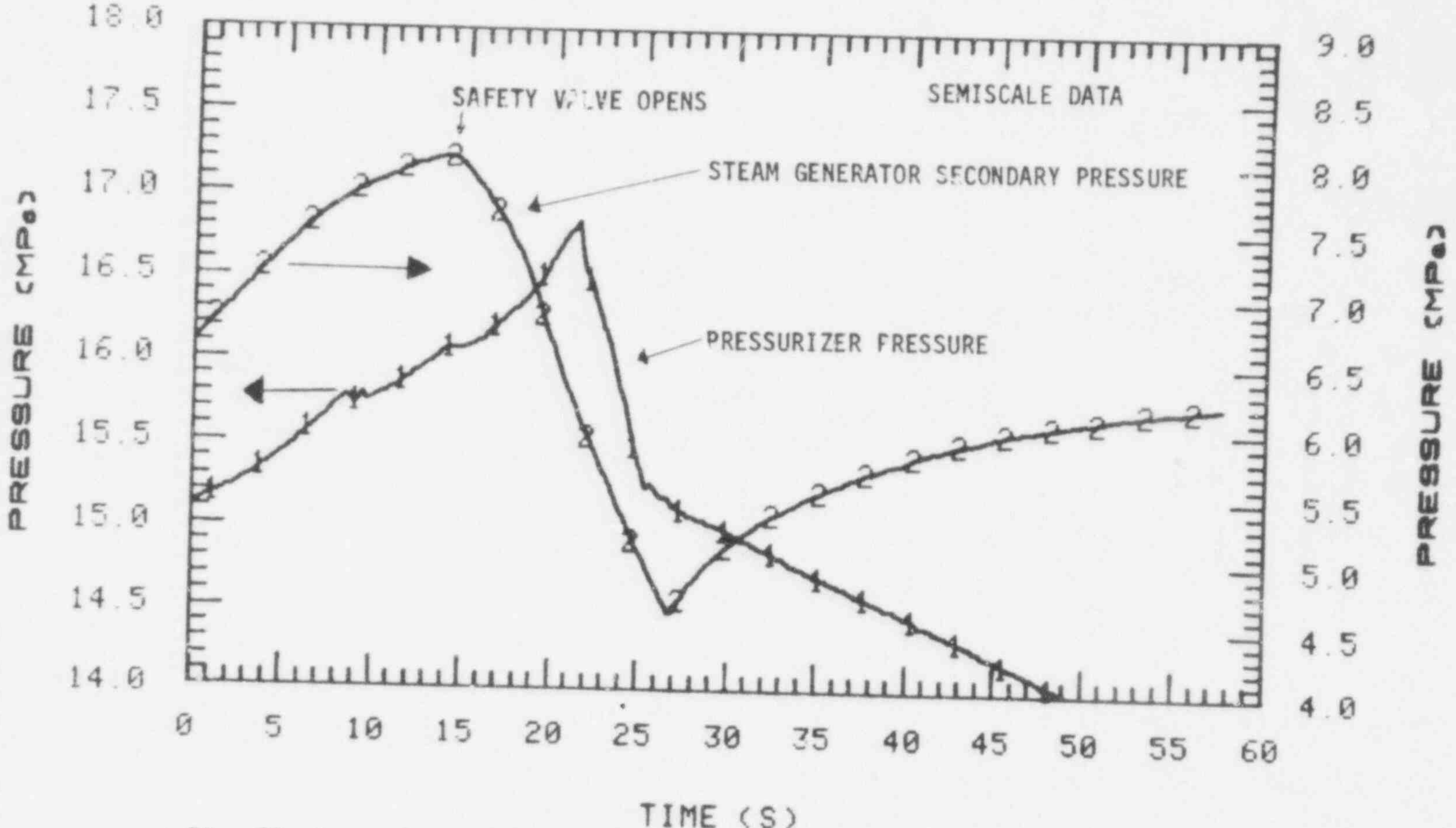

Fig. 14 Comparison of system pressure and intact loop steam generator secondary pressure for Semiscale Test S-TMI-3E.

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used in the Semiscale tests. In one of the experiments, the core power followed the basic curve discussed in Appendix B, and in the other simulation the power was set to zero at scram. The differences induced in the system pressure response are illustrated in Figure 15. Continued core heat generation results in a higher pressure and a pressure change rate that is considerably lower after core scram relative to the case where the core power was tripped to zero.

The results presented in this section have shown that the Semiscale system pressure response during the TMI simulations was sensitive to several parameters in the system. As expected, the steam generator heat transfer characteristics are an extremely important factor. The system pressure response was influenced by POV opening in that a shift in the point in time at which the peak pressure occurred resulted when the POV was not opened. The status of the POV did not strongly influence the system depressurization immediately after core scram. The closest duplication of the TMI depressurization rate after scram, was obtained in Semiscale when the pressurizer code safety valve opened. The complicated interactions between the steam generator heat transfer and the other system conditions along with the lack of detailed TMI data preclude a definitive statement concerning the status of the safety valves during the TMI transient. In general, the Semiscale results have shown that for modeling purposes all of these effects have to be considered.

2. SECOND PERIOD - QUASI STEADY-STATE OPERATION

Following the initial pressure transient during the TMI accident there existed a period of almost two hours in which the primary coolant pumps remained on with the pressure remaining fairly stable. During this period the core was in nucleate boiling and the vessel liquid levels gradually decreased as shown by the out-of-core neutron detectors. Discussed in this section are the significant thermal-hydraulic events that transpired during this period for both TMI and the Semiscale simulation and include: The pressurizer water

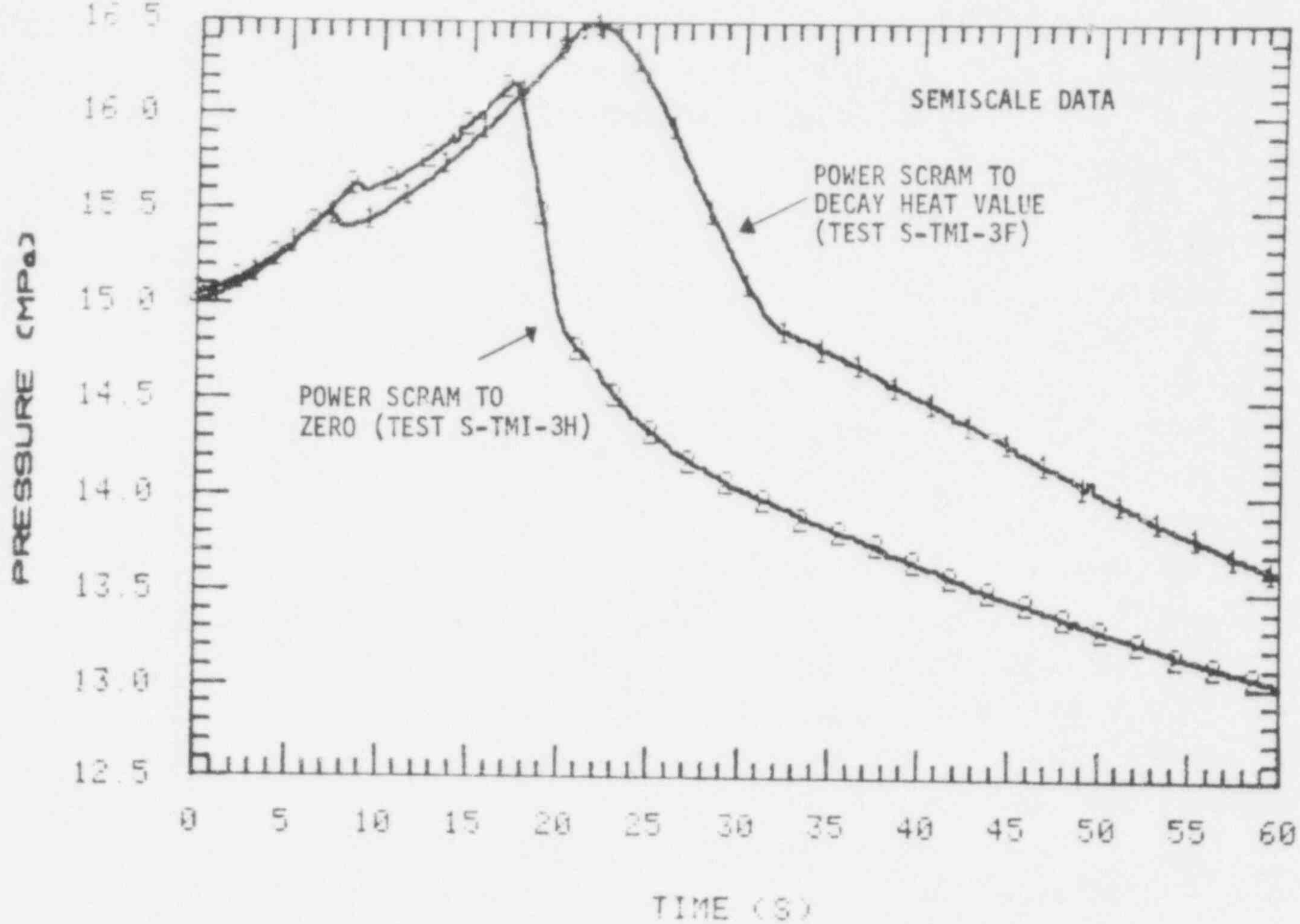

Fig. 15 Influence of core power decay on pressurizer pressure response for Semiscale Tests S-TMI-3F and S-TMI-3H.

level, the vessel water level, the formation of voids in the system and resulting pump cavitation, and critical flow from the pressurizer valve. This second period ended when the primary coolant pumps were shut off at 6000 s. As mentioned previously, several "long term" TMI simulations were run in Semiscale. Test S-TMI-3I is a reasonable representation of the sequence of events that occurred during the quasi steady-state period in the TMI transient; therefore, the discussion of this second period is limited to this test.

2.1 Pressurizer Water Level

Pressurizer water level is a parameter frequently monitored in normal reactor operations, and its performance during long transients is therefore of interest. The primary system mass discharge rate was low enough that HPIS injection caused the Semiscale pressurizer level to rise as shown in Figure 16. The Semiscale pressurizer was discharging steam through the POV at this time at about half the HPIS injection rate*. HPIS flow caused the Semiscale system pressure to decrease at a reduced rate, as shown in Figure 17. The net mass flow into the TMI system caused the pressurizer to rapidly fill which helped cause a pressure rise. The TMI pressure rise at 360 s, shown in Figure 18, occurred when the pressurizer filled. The TMI POV volumetric discharge flow rate dropped significantly when the fluid velocity in the valve was reduced from the sonic velocity of steam to the sonic velocity of a boiling liquid. The core exit liquid also approached the boiling point at this time and contributed to the TMI pressure rise.

Pressurizer level variations for TMI and Semiscale are illustrated on Figure 19. The Semiscale pressurizer filled at 1700 s and remained full. The TMI level dropped back into the range of the

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* HPIS was on for only 60 s to determine its influence on system parameters. A longer injection time was not used because the TMI HPIS injection history was unknown.

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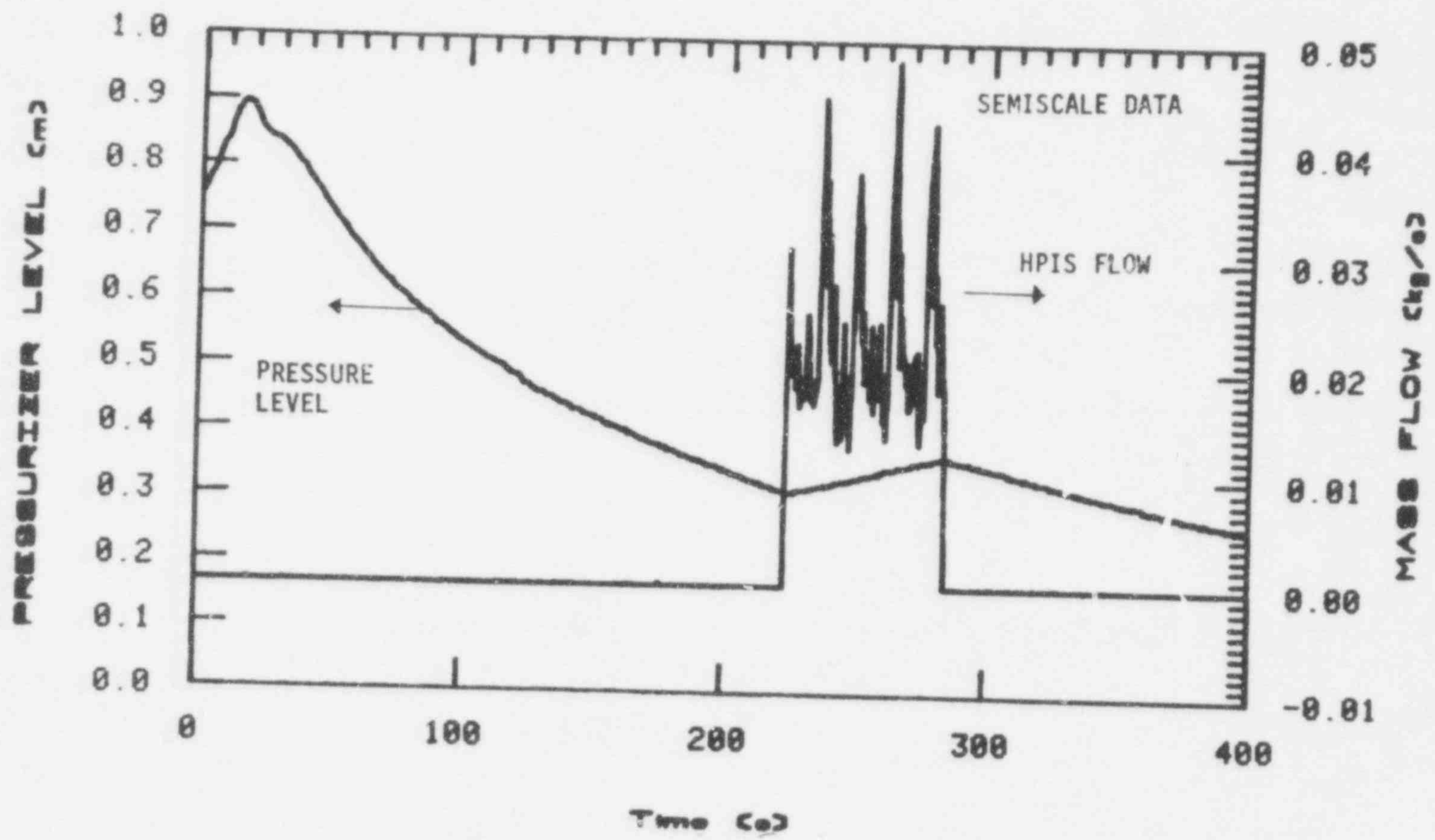

Fig. 16 Comparison of pressurizer level and HPIS flow for the Semiscale simulation of TMI (Test S-TMI-31).

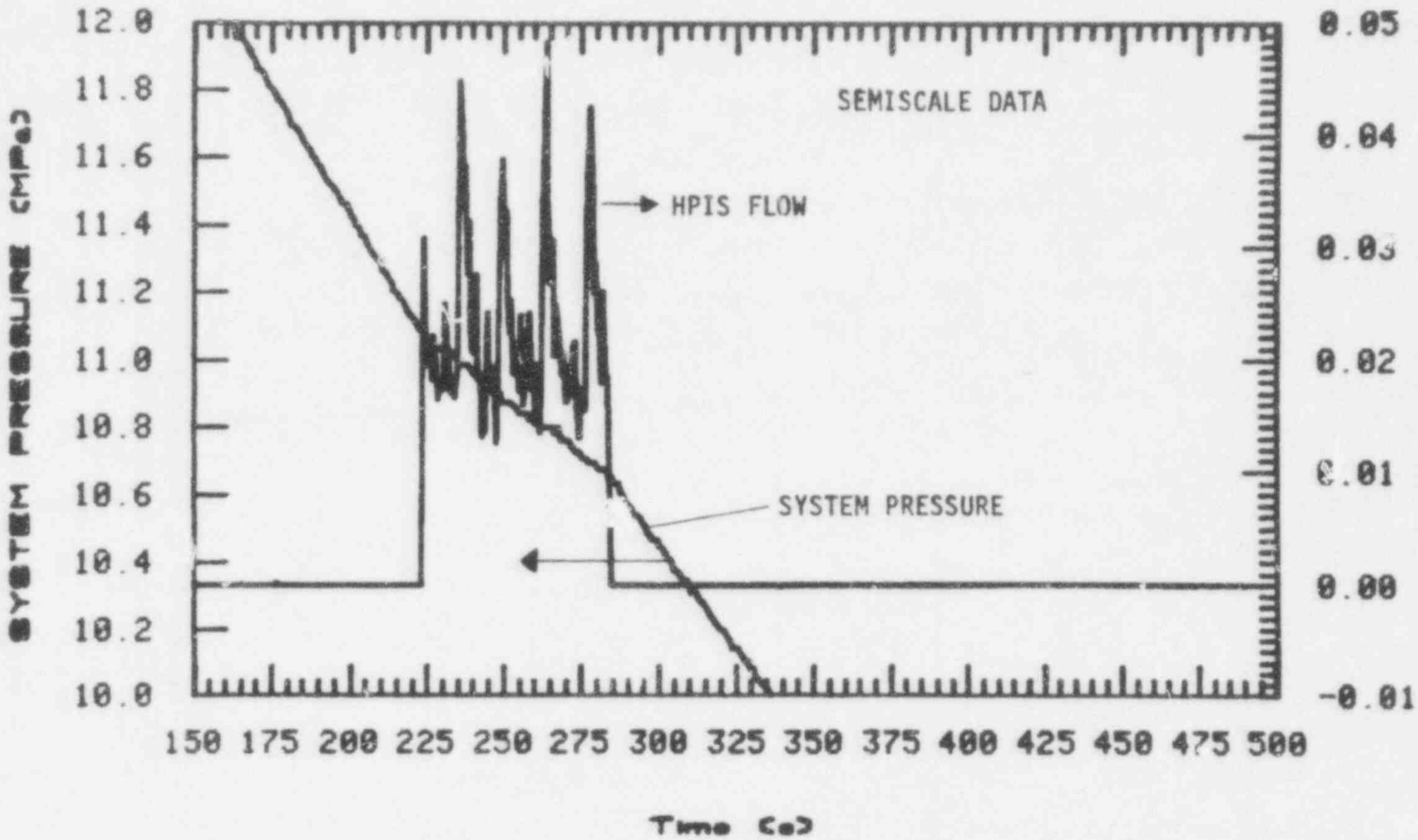

Fig. 17 Comparison of system pressure and HPIS flow for Semiscale simulation of TMI (Test S-TMI-3I).

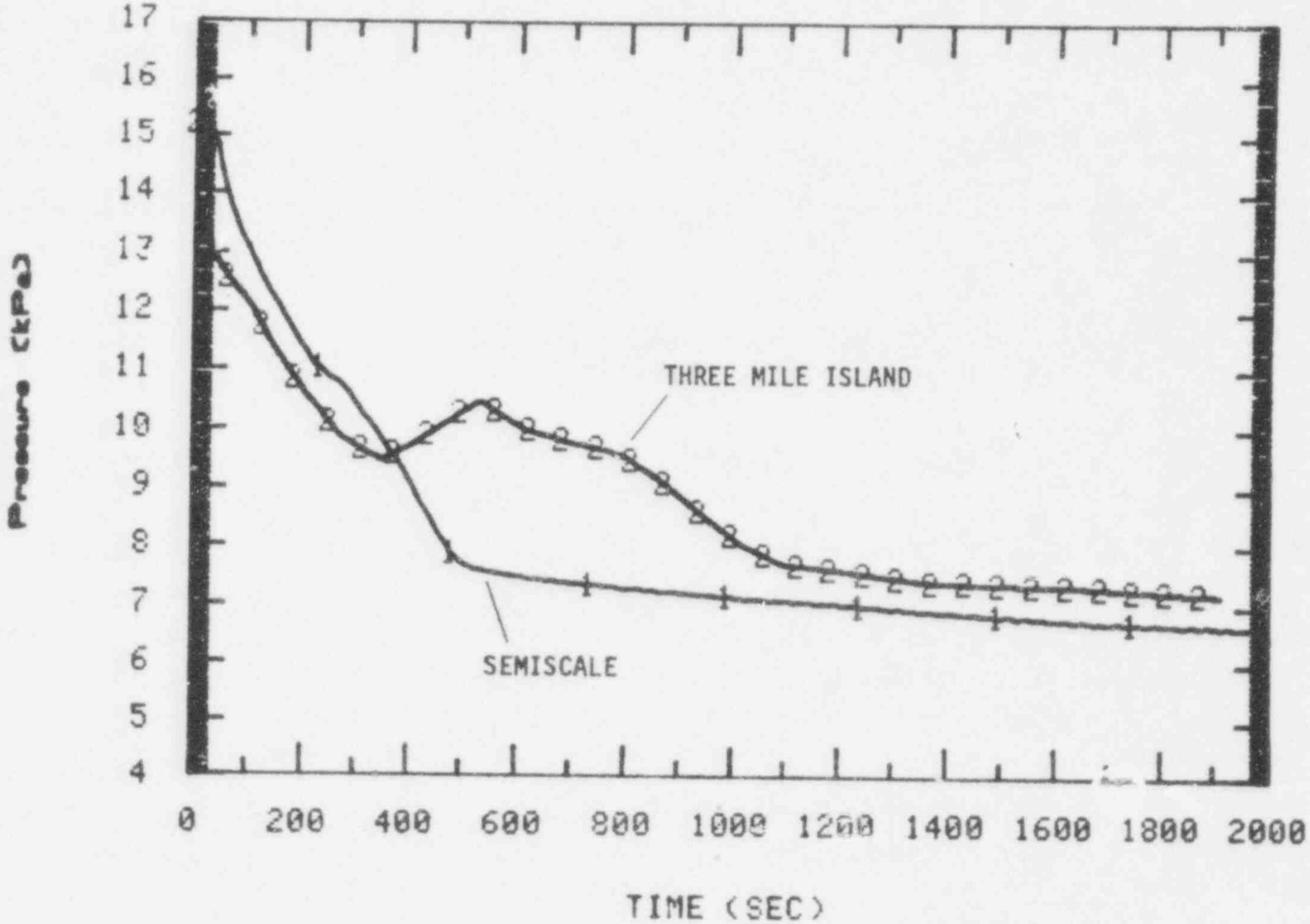

Fig. 18 Comparison of system pressure for TMI transient and Semiscale simulation (Test S-TMI-3I).

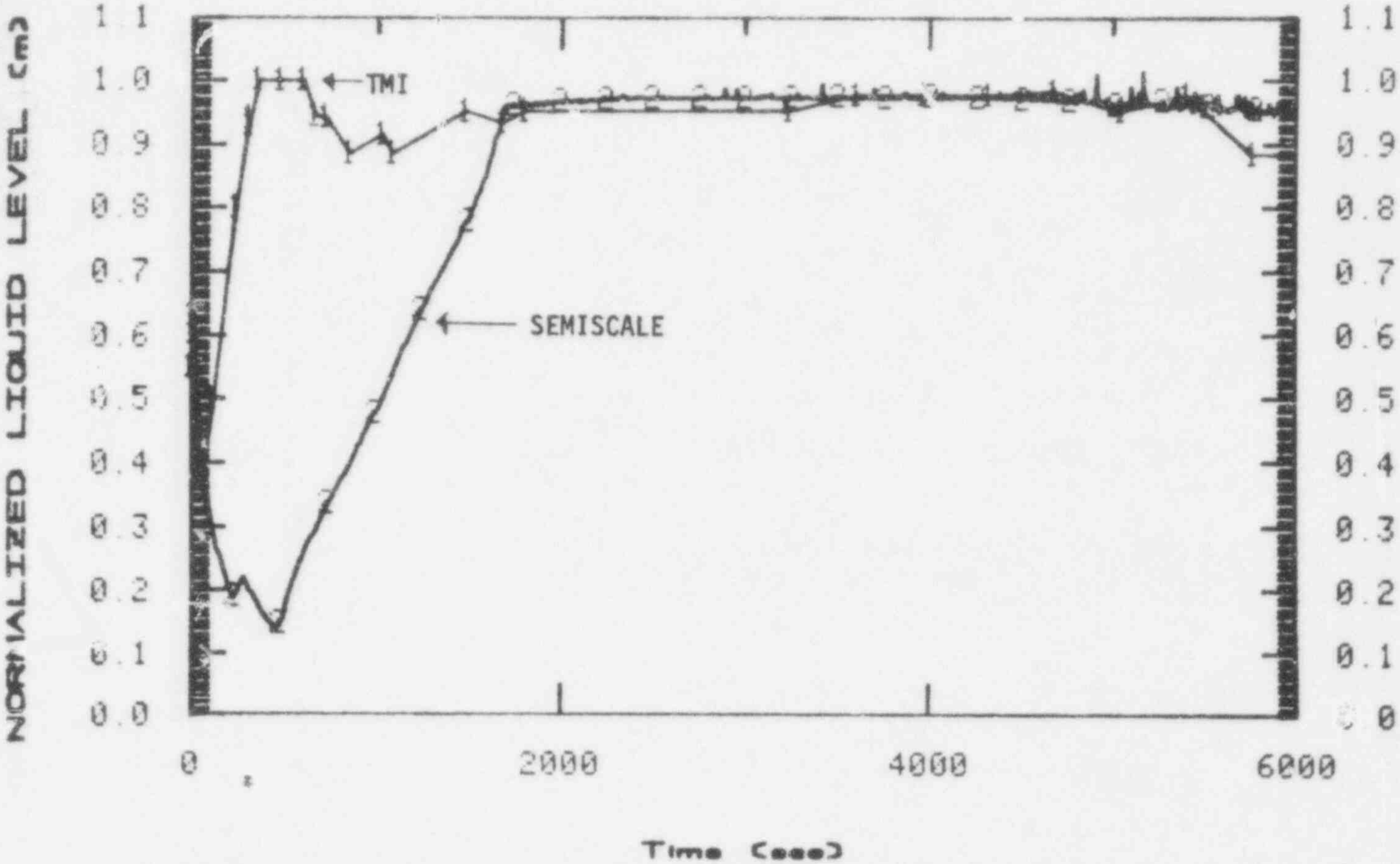

Fig. 19 Comparison of normalized level in the pressurizer for TMI and the Semiscale simulation (Test S-TMI-3I).

measurement devices at about 600 s due to both throttling the injected flow and reducing the average system temperature by turning on auxilary feedwater flow.

2.2 Upper Plenum Water Level

The Semiscale system utilizes gamma densitometers at various levels in the primary vessel to measure axial fluid density variations. These were used to obtain the position of the water level as it decreased during the transient. The densitometers showed a growing steam bubble at the top of the vessel. Below the steam bubble and above the hot legs was a decreasing mass of solid liquid. The position of the liquid/vapor interface within the vessel is depicted in Figure 20. Below the hot legs was a steam-liquid mixture region. The average density above the hot legs may have been less in TMI because steam generated in the core must pass through an internal vertical baffle before exiting through the hot legs. Semiscale has an upper plenum which simulates the Westinghouse UHI design and does not have a baffled barrel.

2.3 Pump Cavitation and Head Degradation

System mass depletion in Semiscale resulted in void formation throughout the system including the loop coolant pump inlet. Voids at the pump inlet cause pump cavitation and head degradation resulting in a reduction in loop flow. Figure 21 shows the pump inlet void fraction and the pressure rise across the pump for the Semiscale simulation. The pump head degraded significantly at a void fraction of about 20 percent and continued to degrade throughout the transient. Even though there was void formation and head degradation during the second time period of the Semiscale simulation, the core temperatures remained low until after the pumps were turned off. Comparison of normalized loop flow for TMI and Semiscale show that the Semiscale flows were considerably lower than TMI, as shown in Figure 22. Either the void formation at the pump inlet was not as

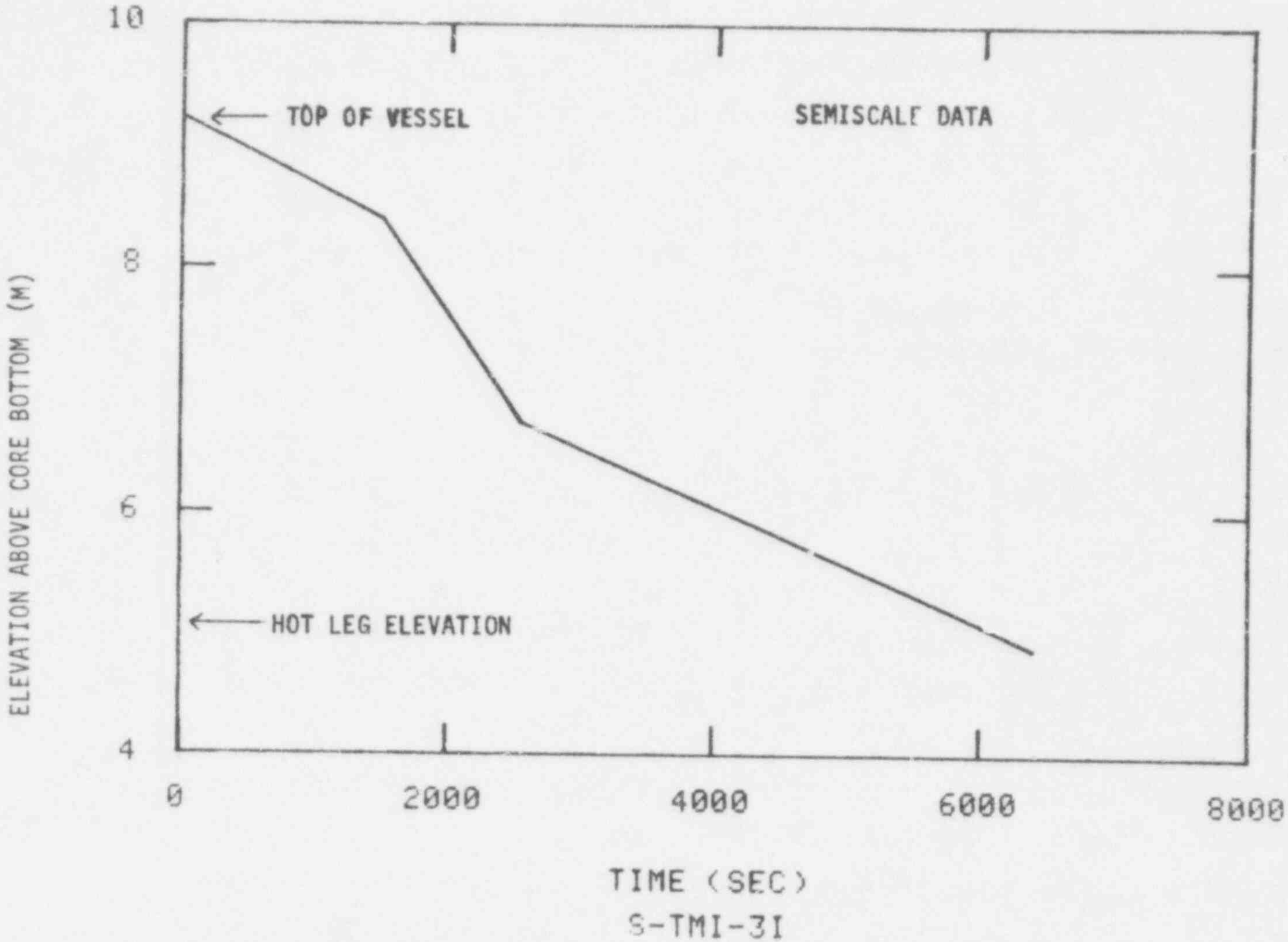

Fig. 20 Upper plenum water level, Semiscale Test S-TMI-3I.

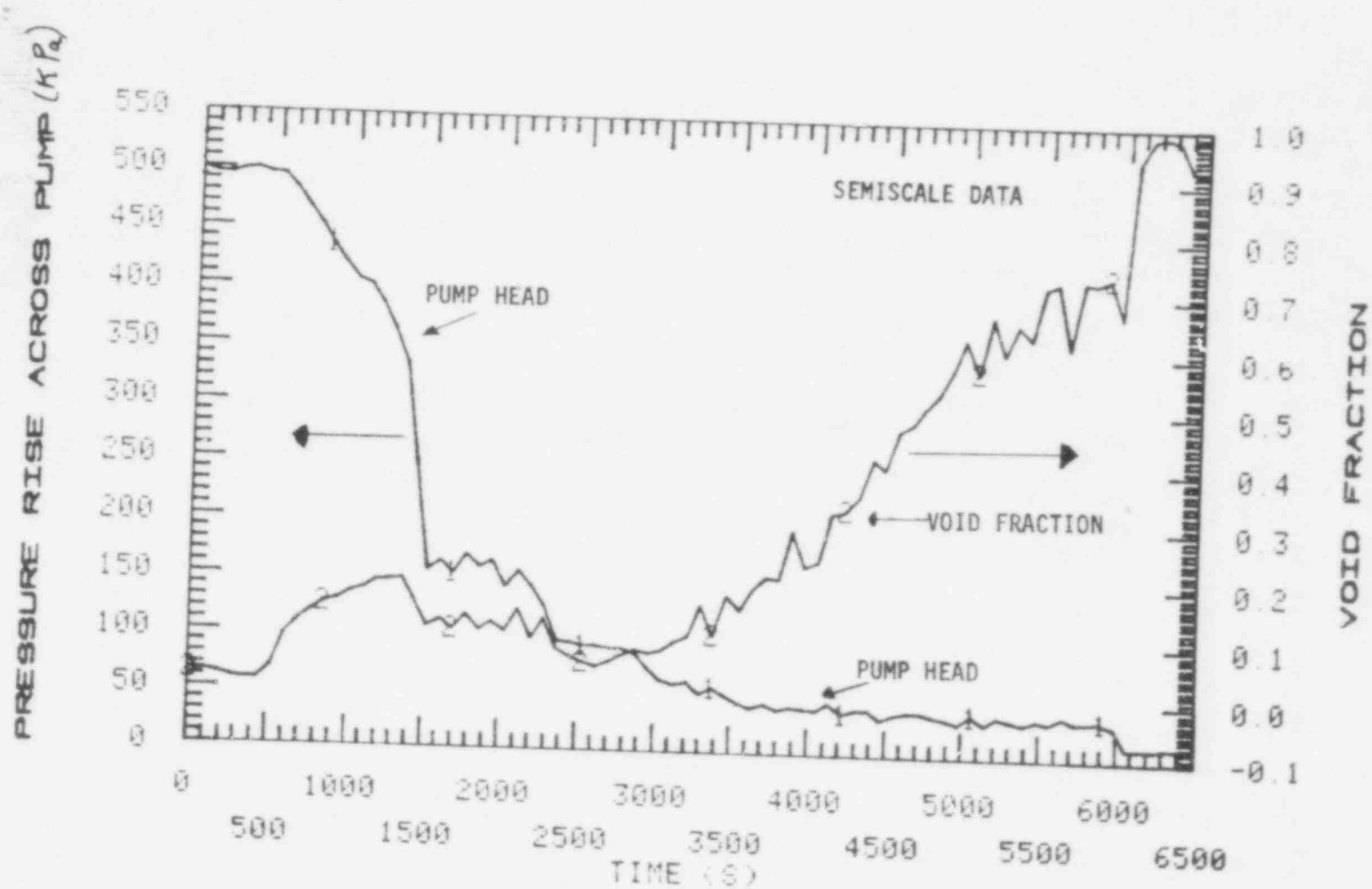

Fig. 21 Semiscale pump head and void fraction (Test S-TMI-3I).

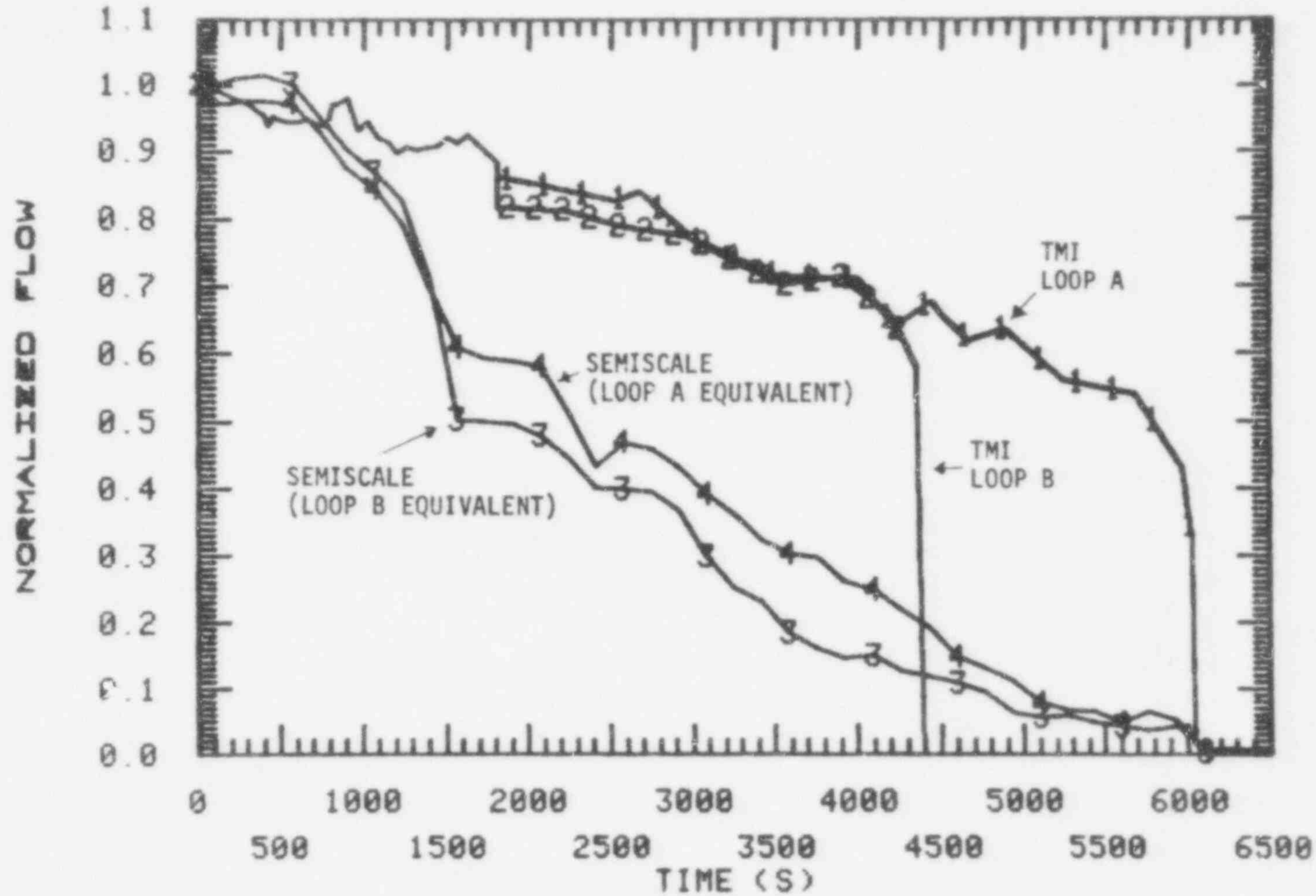

Fig. 22 Comparison of normalized loop flows for TMI and the Semiscale simulation (Test S-TMI-3I).

great in the TMI tranisent and/or the large TMI pumps did not degrade as much as the small Semiscale pumps. Since there is no loop density data from TMI it is uncertain whether TMI had lower loop void fractions than Semiscale. It is expected, however, that large pumps are effected less by density reductions than are small pumps.

2.4 Critical Flow

Loss of coolant accident codes such as RELAP4[4] use break flow models to describe the critical flow (maximum discharge rate) from a break during an accident. The homogeneous equilibrium model (HEM) used by RELAP4 describes critical flow in terms of stagnation enthalpy and pressure. The POV flow rate in Semiscale was estimated using the pressurizer pressure and vent line fluid temperature from Semiscale and the HEM model. Figure 23 shows the HEM calculated break flow. When the pressurizer filled at about 1700 s, there was an increase in mass flow as the calculated flow changed from saturated steam to saturated water. Multiplying the HEM break flow by 0.84* produces an average break flow that is comparable to the average break flow estimated by system mass balance. The average break flow from the Semiscale test was estimated by taking the difference between the initial and final mass in the system over a 6400 s period. The average flow rate using HEM was found by integrating the computed flow and dividing by the time period covere Using these methods, the average flow rate from HEM was 0.0146 k s and the average flow rate from the mass balance method was 0.0126 g/s.

In summary, the second period included quasi steady-state system pressure with a full pressurizer. **Voids formed in the system and pump** cavitation resulted. The Semiscale simulation demonstrated that HPIS

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* The 0.84 multiplier when applied to the HEM model has been found to compare favorably to break flow data from previous Semiscale tests[3].

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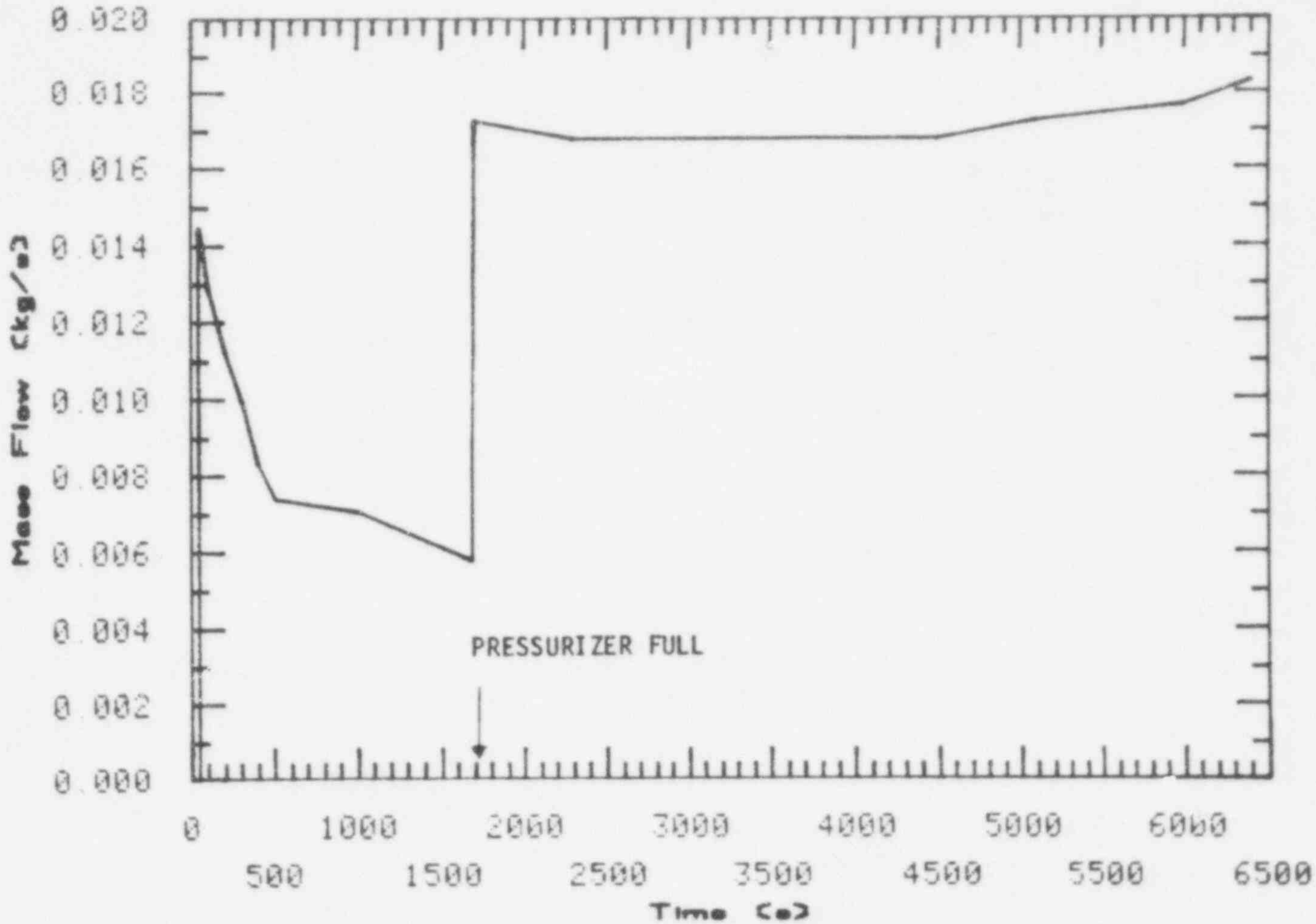

Fig. 23 HEM calculated break flow for the Semiscale simulation (Test S-TMI-3I).

injection affects both pressurizer level and system pressure. Integrating the flow calculated with the HEM model and multiplying by 0.84 agrees with a mass balance on the Semiscale system.

3. CORE UNCOVERY - CORE HEATUP

Following termination of power to the primary pumps at 6000 s, a period of core uncovery and eventual heatup of the core occurred in both the Semiscale and the TMI systems. Turning off the primary pumps eventually precipitated an increased core fluid depletion rate that lead to high temperatures in the core. This section discusses and compares the significant thermal-hydraulic occurrences that transpired after the pumps were turned off for both the Semiscale simulation and the TMI transient. In addition, TMI core temperatures that could have been achieved during the heatup period were estimated using heat transfer information from the Semiscale simulations and data from the TMI transient.

Prior to turning off the pumps at 6000 s in the simulation the liquid inventory in the core had been continually decreasing. This decrease in liquid inventory was caused by more mass escaping through the POV valve and let-down system than was being injected into the primary system. At 6000 s the Semiscale collapsed liquid level (the level that would exist if no steam was in the water) was near the top of the core. The pressurizer was nearly full during the entire period of core uncovery even though mass was leaving through the POV. Thus, an equivalent amount of mass was entering the surge line from the hot leg. The most likely source of the mass entering the surge line was steam produced in the core that eventually condensed in the pressurizer surge line or the pressurizer. Figure 24 compares the pressurizer level, core collapsed liquid level and core rod thermocouple response for the Semiscale simulation. When core power was terminated at 6830 s (due to high core temperatures) the pressurizer drained. Figure 24 shows that the core midplane rod temperature rapidly increased after the water level receded below the

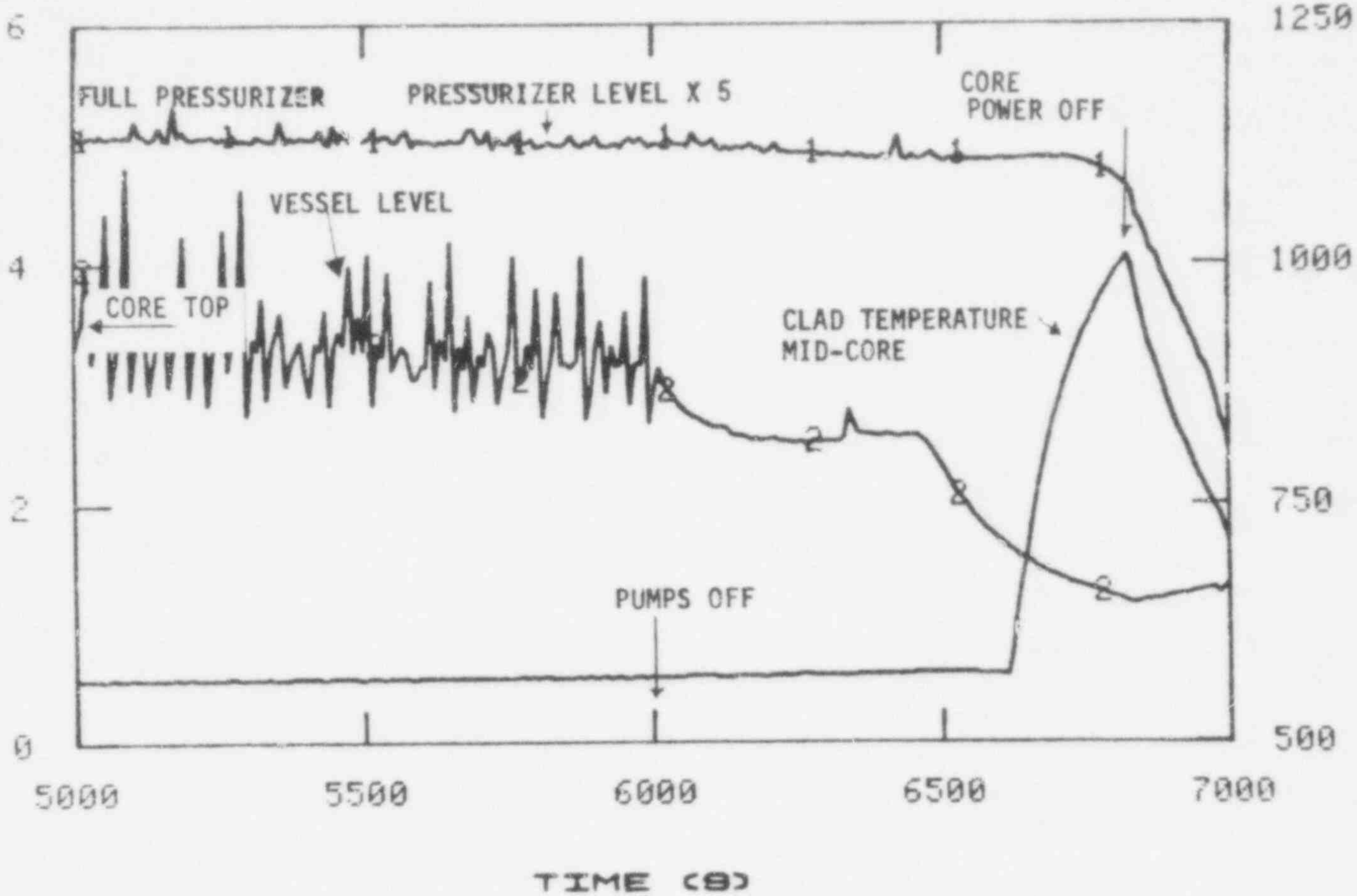

Fig. 24 Comparison of pressurizer and core liquid level with rod clad temperature for Semiscale Test S-TMI-31.

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midplane. The collapsed level is somewhat misleading in that it represents all the liquid in the core collapsed into a pool referenced to the bottom of the core and thus does not give information about the fluid density distribution. The in-core densitometer data from the Semiscale vessel are compared to the core rod thermocouple responses in Figure 25, indicating a definite fluid density stratification in the core. At about 6500 s the fluid density at the 3.32 m elevation decreases suddenly and similar dramatic decreases for lower positions (1.73 m, 1.13 m) in the core occur at later times. The temperature for the top rod position begins increasing about 60 s after the frothy mixture drops below that level, indicating a certain amount of entrainment induced cooling above the froth. The midplane temperature, however, begins increasing immediately after the froth level drops below the midplane (1.84 m). The slope of the temperature response at the midplane is much steeper than that at the top of core because of the higher power density at the midplane.

Figure 26 illustrates the progression of the froth front level (from the densitometer data) and the collapsed liquid level. The froth level and collapsed level approach similar values below the midplane. One reason the froth level collapses after dropping below the midplane of of the core is the presence of subcooled water in the lower core does not support the boiling necessary to maintain a froth level. For core positions above which saturated liquid is present, boiling produces steam that supports a froth level. However, this saturated region evidently boils off leaving the subcooled pool of water with a small froth level. As the steaming rate decreases, the rate of heat removal in the upper part of the core becomes smaller. Subcooling in the lower region of the Semiscale core is due to system heat losses in the downcomer. This is illustrated in Figure 27 which compares the fluid temperatures at various elevations in the downcomer to the saturation temperature. A definite stratification of fluid temperature exists with about 10 K subcooling in the lower plenum. If the TMI lower plenum liquid was less subcooled than in Semiscale, heat transfer in the core would be higher since the froth level would be higher. The available TMI graphical data can not be interpreted

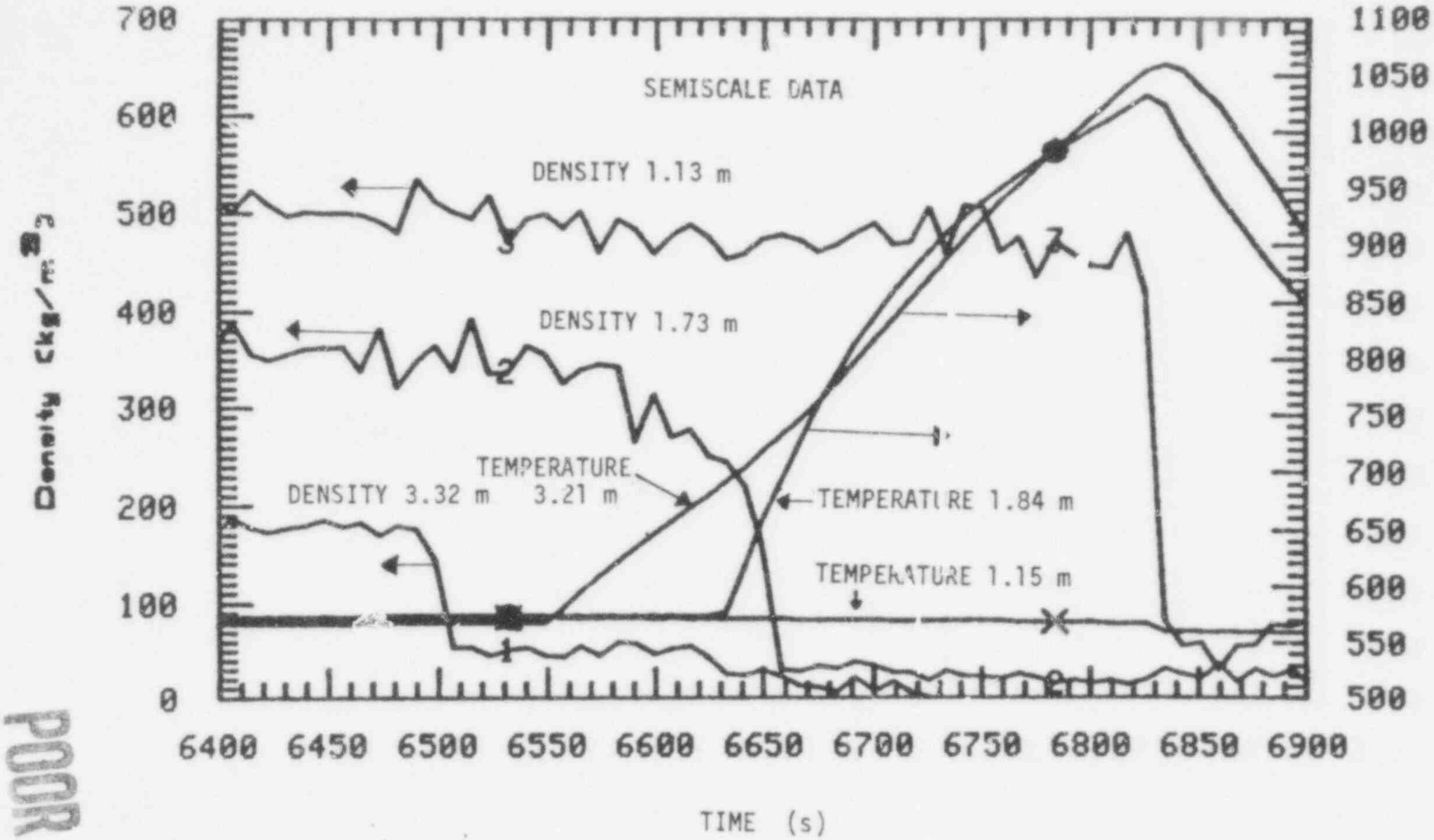

Fig. 25 Comparison of incore fluid density and core rod thermocouple response for the Semiscale simulation (Test I).

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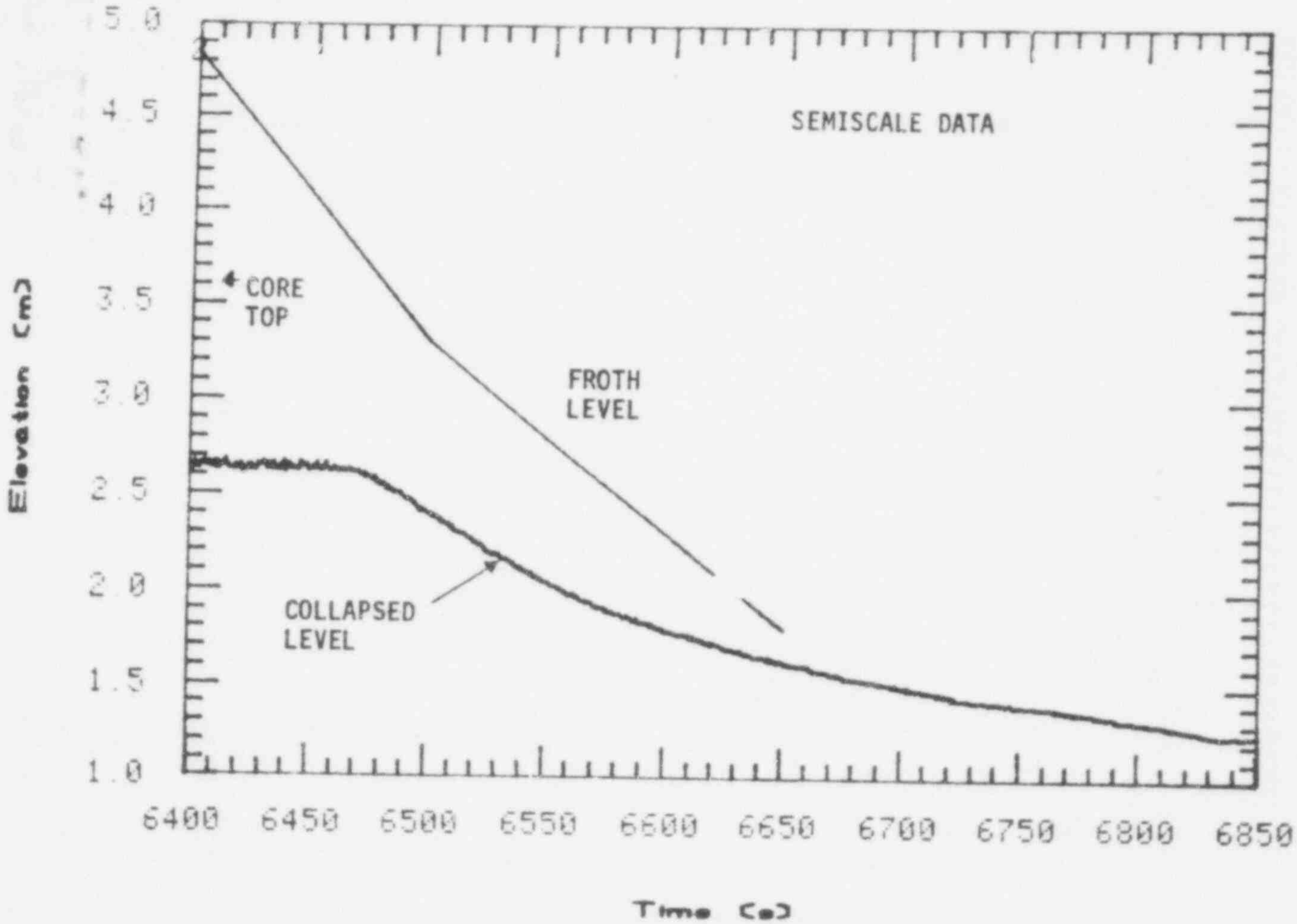

Fig. 26 Comparison of collapsed level and froth level in the core for the Semiscale simulation (Test S-TMI-3I).

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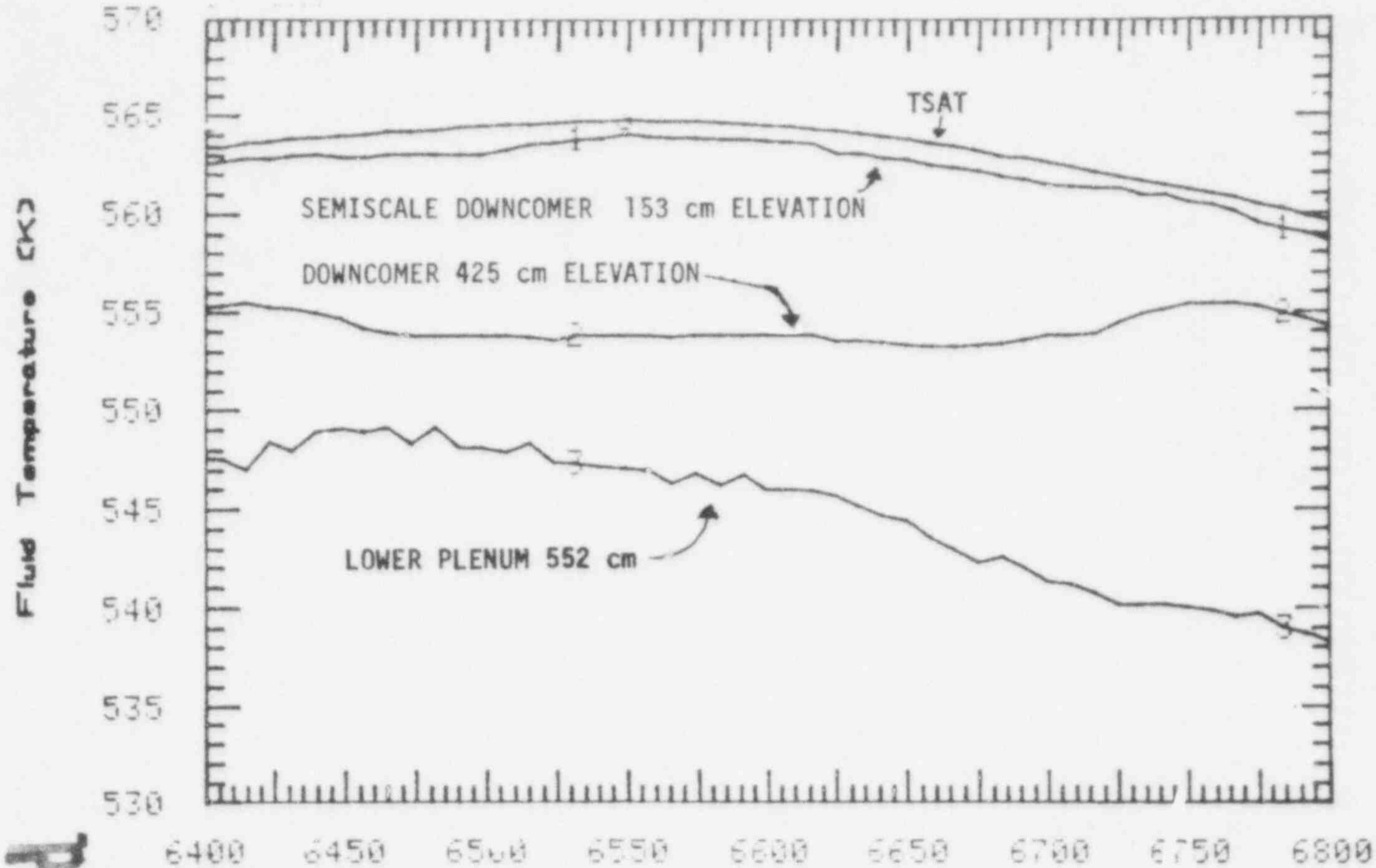

Fig. 27 Comparison of saturation temperature to fluid temperature in the downcomer at various elevations.

accurately enough to determine the degree of subcooling, however the subcooling in the TMI cold leg appears to be less than 10 K. Another reason the froth and collapsed level approach each other is because the integrated power below the froth level is continually decreasing resulting in fewer voids.

The sudden temperature increase at a particular core rod location (see Figure 25) is caused by a departure from nucleate boiling to film boiling. Predicting the time of this departure is a difficult and a significant challenge to analytical reactor models used to predict Semiscale or TMI transients. The rapid heatup of any rod position in the core is expected to be a function of system pressure, power density at the position, and the relative position of the liquid or froth level. However, for the Semiscale simulation the rod temperatures consistently increase rapidly at positions above the collapsed liquid level regardless of the value of other system variables. This is illustrated in Figure 28, which presents a comparison of the rod position at the time of temperature rise (departure from nucleate boiling) versus the collapsed level at that time for several Semiscale transients. The Semiscale simulations involved different bundle power levels (25 - 120 kW), system pressures (400 - 1100 psia) and loop hydraulics yet the relationship between the rod position at the time of temperature rise occurs at about the same collapsed liquid level for all these simulations. For the bottom three-fourths of the core, rod positions at the time of temperature rise are a linear function of the collapsed liquid level. However, for the top one-fourth of the core the data is considerably scattered, much like bottom up reflood data*. The variation in position of temperature rise versus collapsed level for the top one-fourth of the core is probably due to the interaction of entrained fluid with upper plenum structures.

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* Both Semiscale and Flecht[5] bottom reflood data show a linear narrow band for a rod position versus quench time cross-plot for the bottom part of the core and quite a scattered band of data at the top part of the core.

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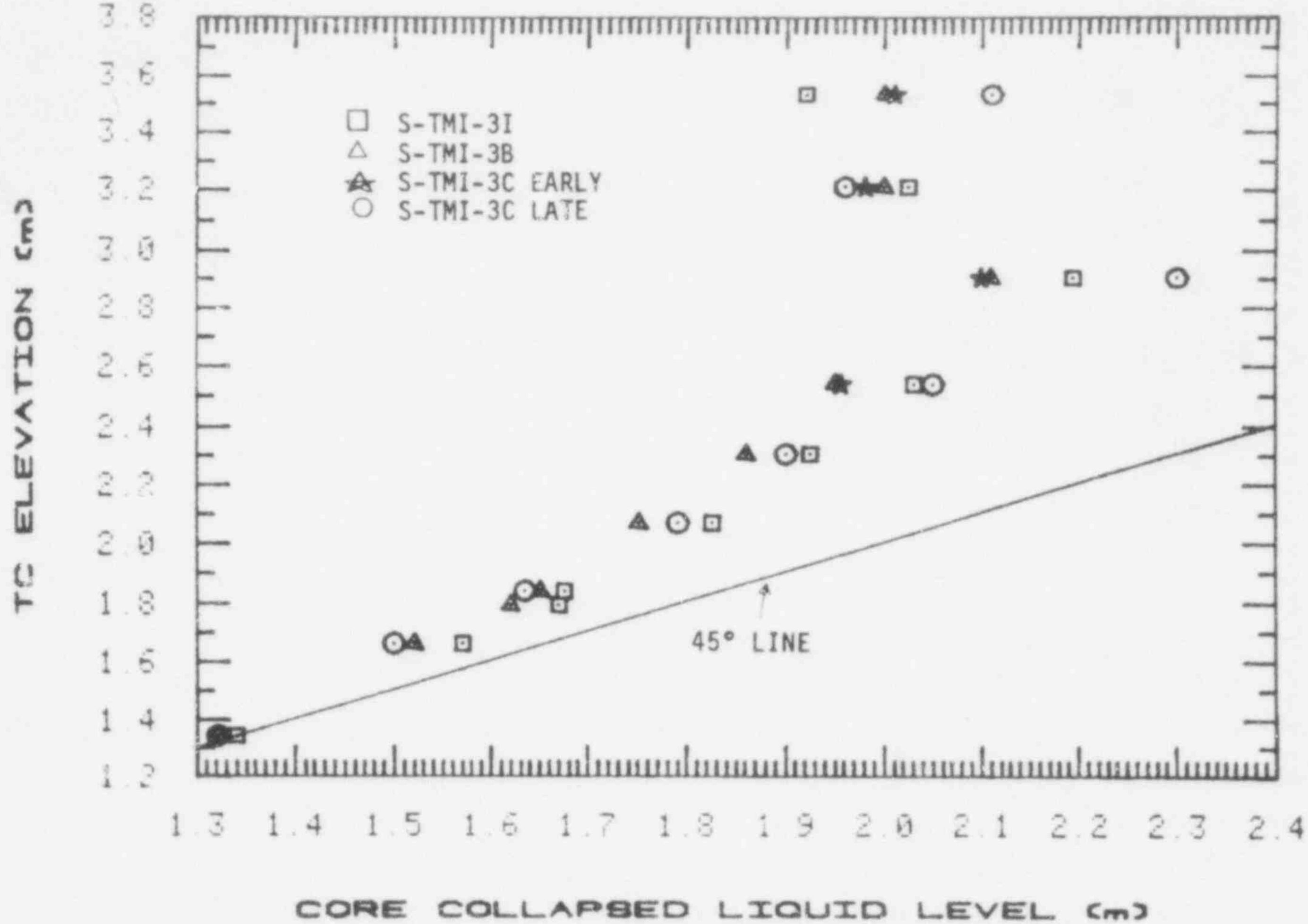

Fig. 28 Comparison of the rod position at the time of temperature take off versus the collapsed liquid level at the time of several Semiscale simulations.

The "external-core" detector (NI-1) information from TMI indicated that shortly after the primary pumps were shut off the vessel level increased dramatically as shown in Figure 29. When the pumps were shut off in the Semiscale simulation at about 6000 s, the resultant loss of flow allowed fluid in the primary side of the steam generators to drain back into the vessel (supported by steam generator level measurements and loop densitometers). As a result, the core liquid level remained fairly stable even as the downcomer level began decreasing.

An indication of core uncovery and core heatup in TMI is the observance of superheated fluid temperature in the hot leg. The superheated fluid could only have come from a core region that had uncovered and heated up to temperatures above saturation. The Semiscale simulation indicated the presence of superheated fluid in the hot legs at about the same time as occurred for the TMI transient, as shown in Figure 30. Despite hardware limitations such as heat losses and sequential variations, Semiscale simulated this most important event of the TMI transient fairly well. This agreement between Semiscale and TMI data is probably because the uncovery of the core and core heatup is principally controlled by mass discharge out the POV valve.

Core heatup in TMI, as discussed in this report, is intended to stimulate further thinking about the physical processes that occurred in the TMI transient and to illustrate what a direct extrapolation of Semiscale data would yield for the TMI peak cladding temperature. The assumptions used in extrapolating Semiscale data to TMI will be noted. The fact that superheating in the hot legs of both Semiscale and TMI occurred at about the same time suggests that the general trends between the two facilities were similar.

Heat transfer coefficients in the core are a function of the core water level. By assuming that the TMI loops drained to give the same collapsed water level in the core as in Semiscale, and that at similar

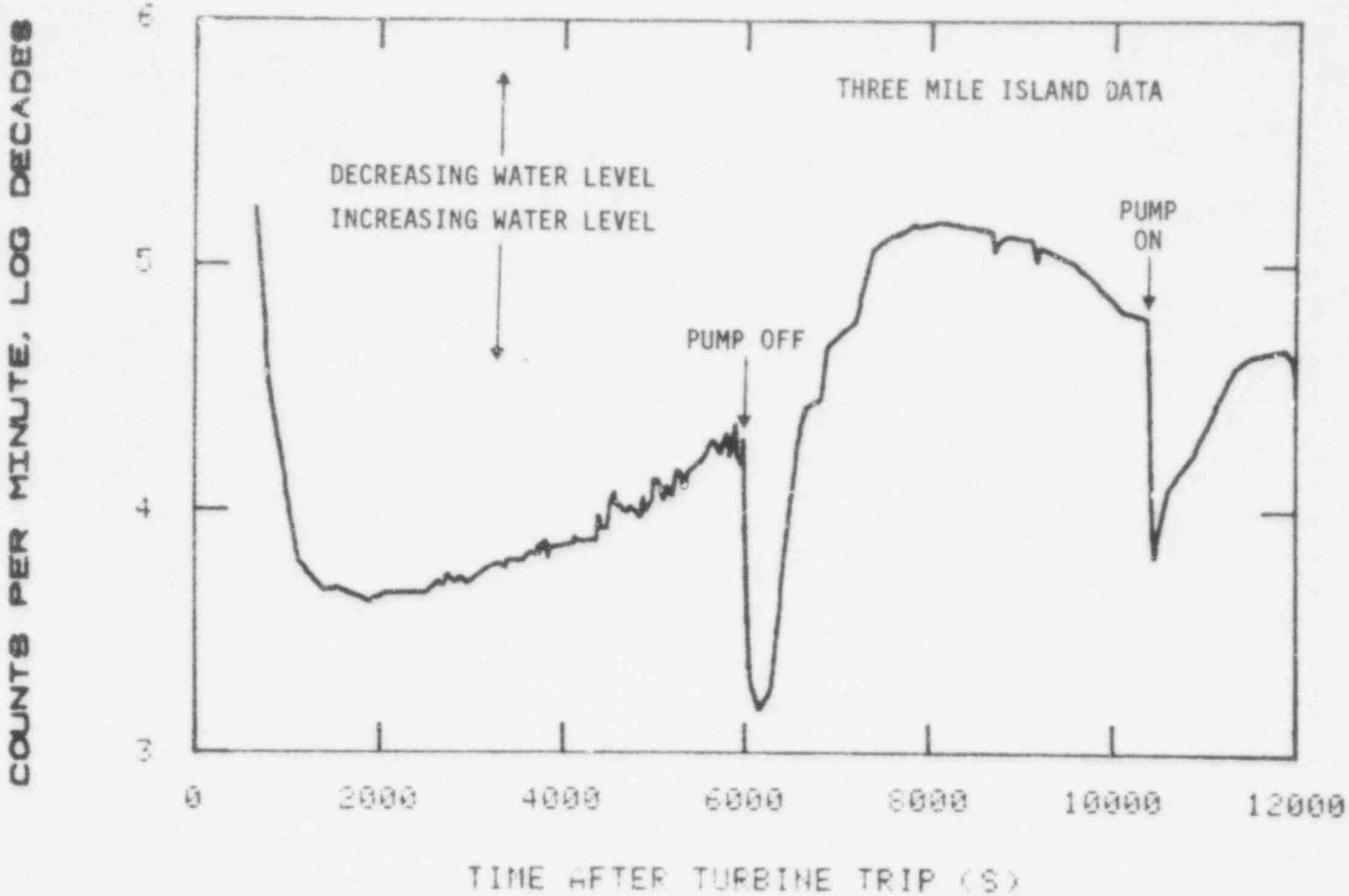

Fig. 29 Three Mile Island external core detector, source range channel NI-1.

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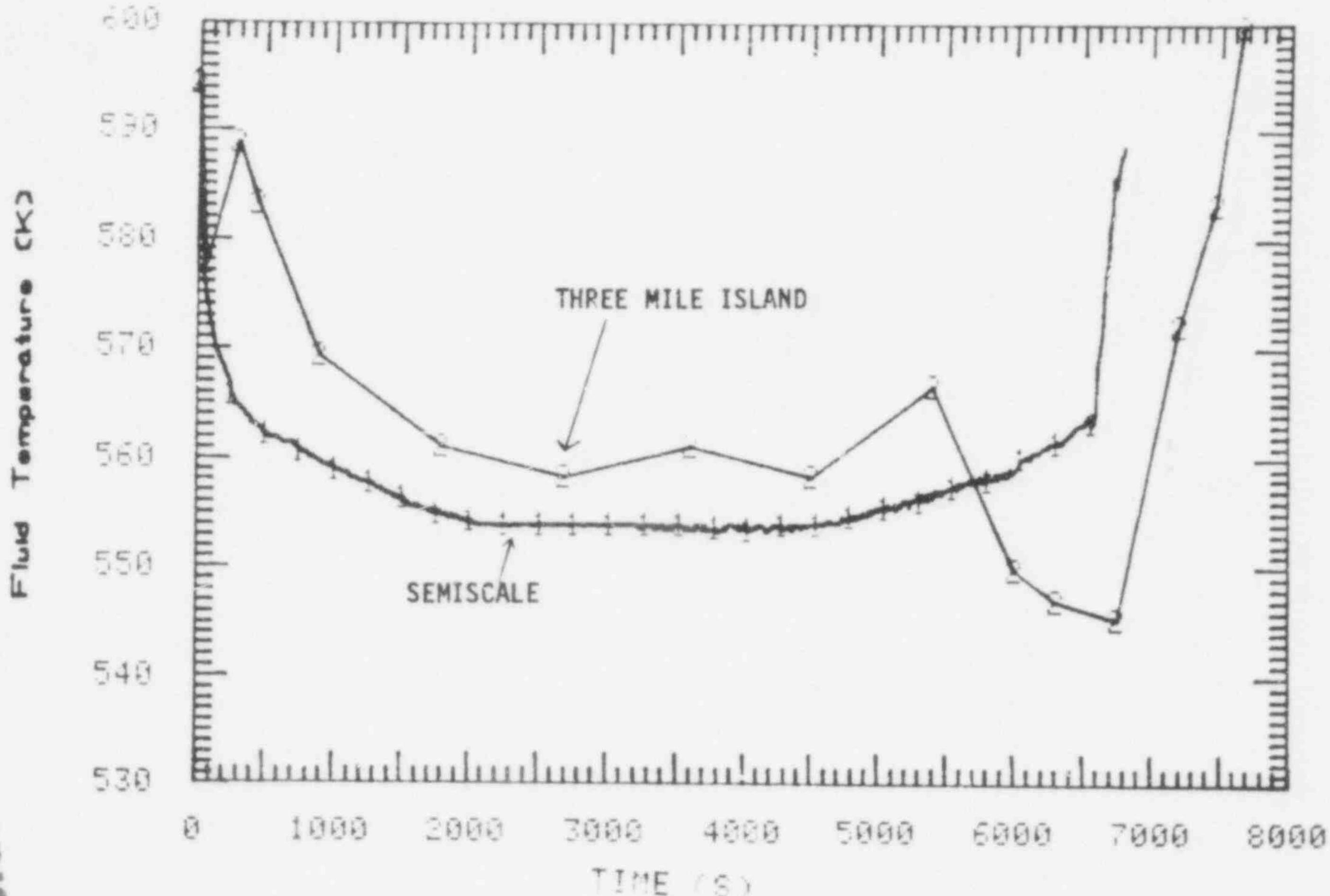

Fig. 30 Comparison of hot leg fluid temperature for TMI and the Semiscale simulation (Test S-TMI-3I).

rod power densities the boiloff rates were the same, the TMI core water level decrease was calculated. Figure 31 shows the estimated TMI core collapsed liquid level. At about 8300 s the downcomer level began to increase again, possibly due to increased steam condensation in the steam generators as the feedwater flow was increased. At 10449 s reactor coolant pump 2B was restarted. The pump pulled water out of the loop seal and forced it into the vessel causing core requenching.

Heat transfer coefficients for various elevations in the Semiscale core were obtained from the rod temperature response and an inversion calculation. These heat transfer coefficients are plotted versus collapsed level on Figure 32*. The TMI water level curve shown on Figure 31 was used with the Semiscale heat transfer coefficient versus water level data to obtain TMI heat transfer coefficients at various elevations. Estimated heat tranfer coefficients for TMI at four elevations are given in Figure 33. Because of oscillations in Semiscale of heat transfer coefficients at low values for the 3.2 m elevation, a nominal and a minimum value are given. These estimated TMI heat transfer coefficients were used in a rod conduction model to calculate TMI core temperatures. The rod conduction model included metal-water reaction. Additional assumptions used for the heat up calculations were that:

1. Decay power was 1.0 times the proposed ANS[6] value for infinite operating time.

2. Metal-water reaction rate was 0.6 times the Baker-Just[7] rate.

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* The oscillation at 1.4 meters resulted when the rod power was switched to a lower value. The high value was needed to maintain system pressure and the low value was needed to obtain boil off rate data at the proper power level.

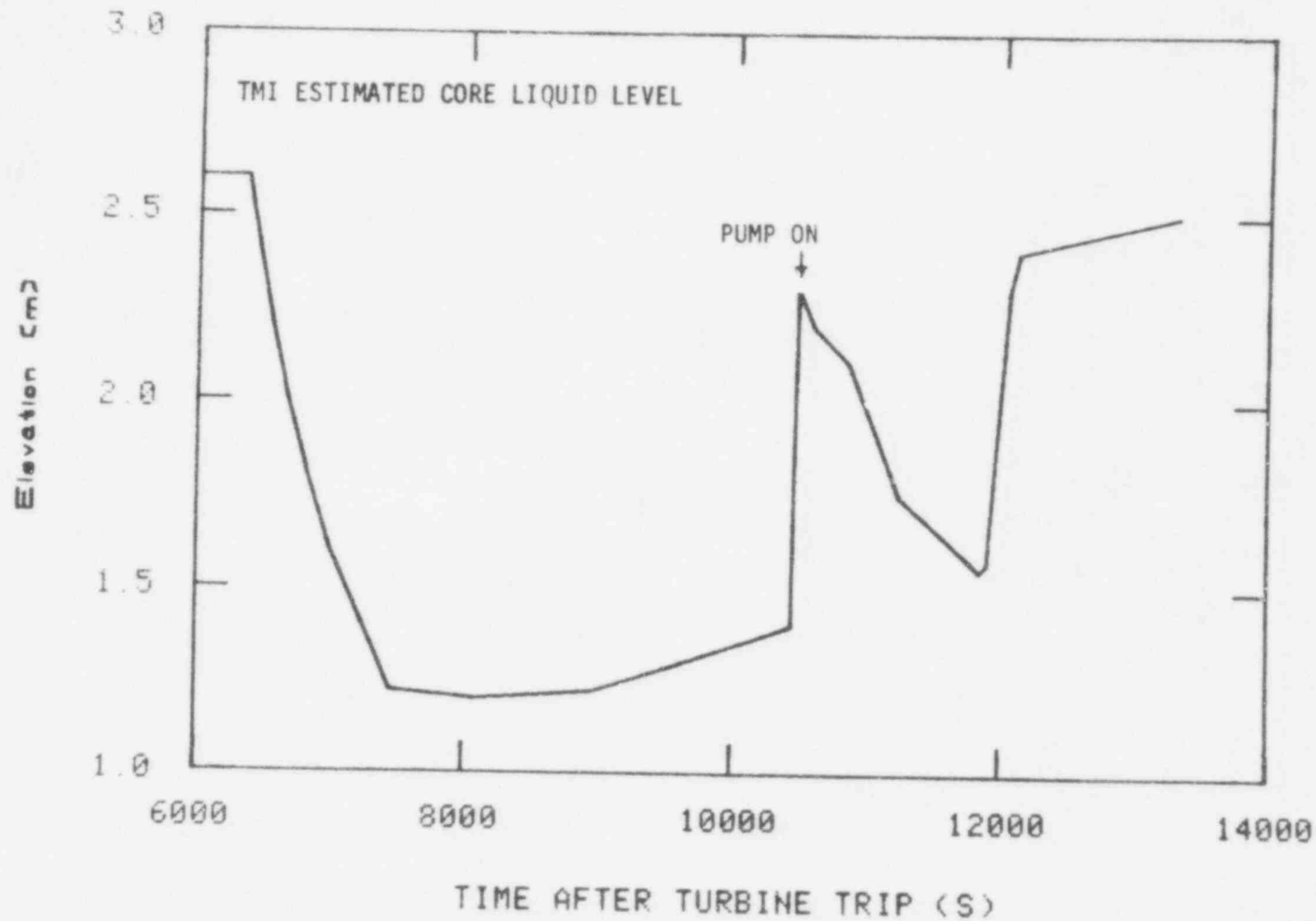

Fig. 31 TMI core liquid level estimated from Semiscale data and TMI external core detector.

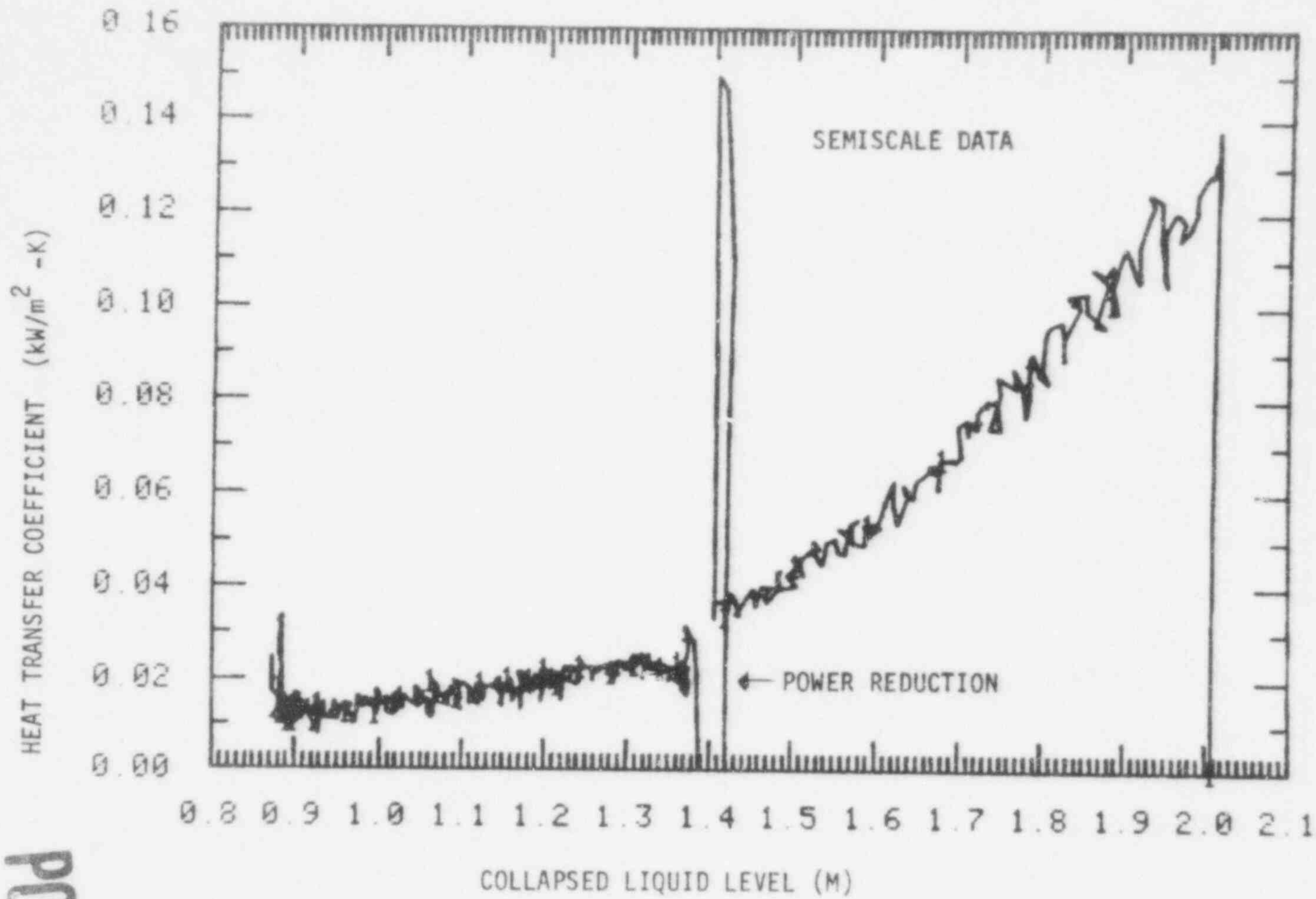

Fig. 32 Semiscale heat transfer coefficient versus liquid level (Test S-TMI-3C).

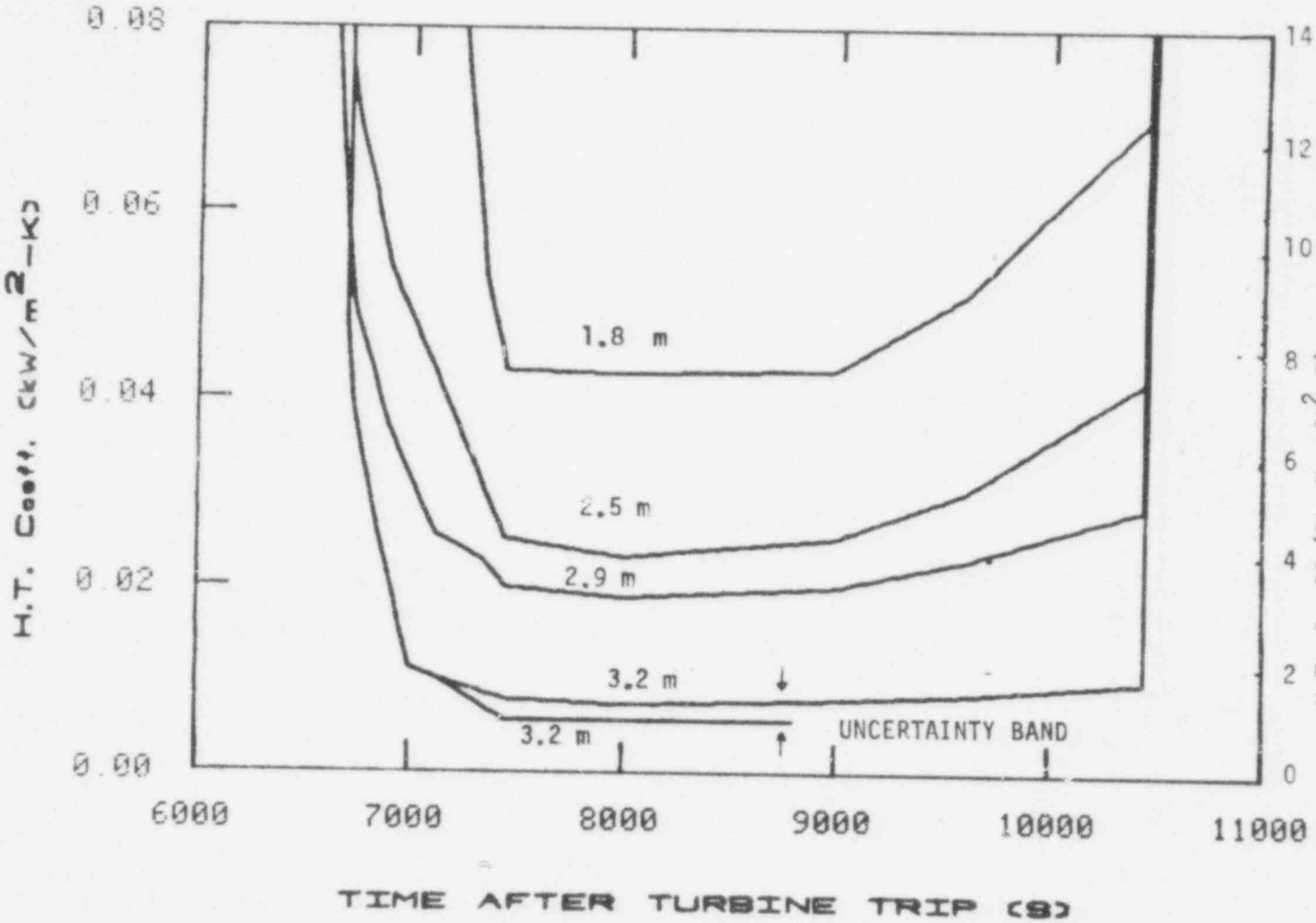

Fig. 33 Heat transfer coefficients estimated for TMI.

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3. The axial power shape was as defined by the GRASP code results from the SPND data taken on the center fuel assembly in TMI on March 19, 1979. Appendix C gives the GRASP results and compares the TMI power profiles with Semiscale.

The calculated TMI clad temperatures are shown in Figure 34. Using the estimated heat transfer coefficients and the above assumptions implies that severe rod damage may have occurred in the TMI core above the 2.9 m elevation. Elevations below 2.9 m remained below 1000 K. If the cladding at the 3.2 m elevation did get as hot as indicated, regions above and below could have suffered adverse effects due to flow channel blockage. The calculated oxide layer following cool down, for the lower 3.2 meter elevation calculation shown on Figure 34, was 40 percent of the initial cladding thickness. The alpha layer underneath ZrO_2 is also brittle and is similar in thickness. In addition, oxygen in the UO_2 diffuses into the zircaloy to form a brittle oxygen stabilized alpha layer. Thus the cladding at the 3.2 m elevation is expected to be completely embrittled as a minimum. The effects of eutectic reactions with the inconel grids were not considered.

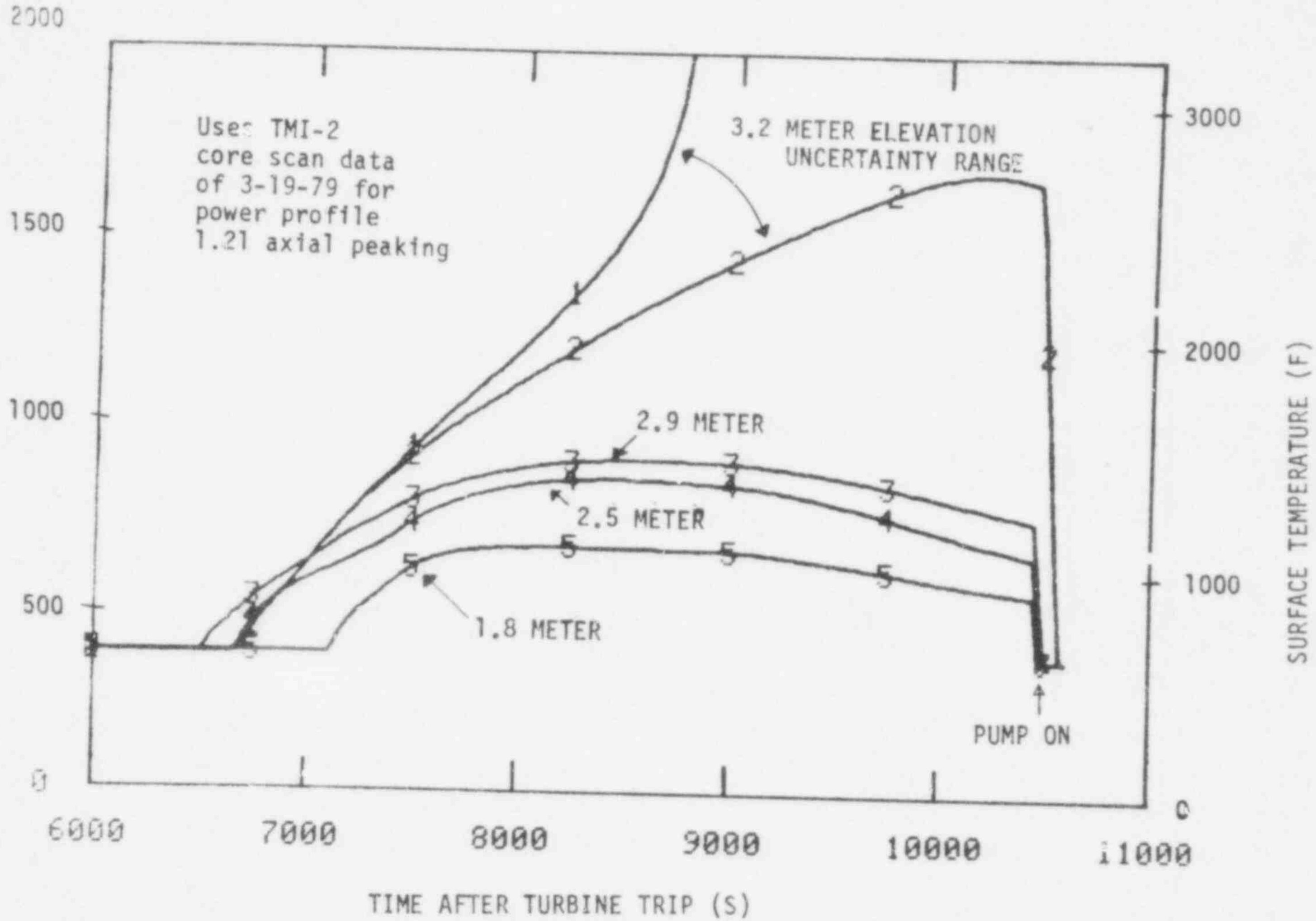

Fig. 34 Three Mile Island estimated core temperature.

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V. CONCLUSIONS AND RECOMMENDATIONS

The following conclusions were reached regarding the results of the Semiscale Three Mile Island simulations:

(1) The overall thermal-hydraulic trends observed in the Semiscale simulations were similar to those observed in the TMI data.

(2) Complex interaction between the steam generator secondary heat transfer, relief valve characteristics, core power decay and the pressurizer relief valve characteristics control the early time period pressure response.

(3) When modified with a multiplier of 0.84, the homogeneous equilibrium model appears to predict the flow rate through the Semiscale pressurizer POV reasonably well.

(4) As expected, the Semiscale loop coolant pumps degraded significantly earlier than the TMI reactor coolant pumps.

(5) The Semiscale pressurizer level showed trends similar to the TMI pressurizer level.

(6) Semiscale data showed that when the system was operating in a saturated two-phase condition, a liquid full pressurizer indication was not an appropriate indication of core liquid inventory. Core uncovery and heatup occurred even though the pressurizer remained full.

(7) Based on the Semiscale results, it would appear that the TMI pressurizer level instrumentation was a valid measurement of pressurizer liquid inventory for at least the first few hours of the transient.

(8) Superheated steam was observed in the Semiscale system hot legs in the same time frame as was observed in the Three Mile Island Reactor. This result suggests that core heatup in the TMI reactor occurred shortly after the loop pumps were shutoff, as occurred in Semiscale.

(9) After core uncovery occurred, the Semiscale core heat transfer coefficients were correlatable to the core collapsed liquid level.

(10) Assuming applicability of Semiscale core heat transfer data and assuming that the TMI core mass inventory was similar to that in Semiscale, results in calculated TMI cladding temperatures at the 3.2 m elevation in excess of 1500 K. In fact, the calculated cladding temperatures were high enough that the exothermic Zirconium-water reaction is important and severe cladding embrittlement could occur. Positions below the 2.9 m elevation did not show excessive calculated temperatures.

(11) Scaling distortions (most notably system heat losses) must be considered and accounted for when performing slow transients in the Semiscale facility.

In general, the Semiscale simulations conducted were successful in terms of representing the general thermal-hydraulic phenomena observed in the TMI plant transient. However, several recommendations can be made with respect to future simulations of slow transient in the Semiscale Facility.

One of the more significant distortions in the Semiscale system relative to a large reactor is heat losses to ambient. As was discussed earlier, this loss was accounted for in the simulations by increasing the core power above the required decay heat value. This complicates analysis efforts and is not a suitable long-term solution to the heat loss problem. For future slow transients performed in the Semiscale system alternatives methods of heat loss reduction need to be investigated. Viable alternates include system insulation upgrades and/or the use of electric heat tape in selected portions of the system.

Steam generator design and secondary surface heat transfer area per unit volume were found to be important considerations during the TMI simulations. To achieve proper secondary to primary heat transfer both manipulation of steam generator secondary levels and improved designs need to be examined.

In some cases during the TMI simulations, the Semiscale system fluid leakage rates were of a magnitude comparable to the flow rate through the pressurizer POV. Generally, the largest contributor to the system leaks were the loop pump seals, although small leaks also existed at some of the instrument port penetrations. Currently leaks are compensated for by makeup pump flow. Techniques for reducing the system leakage rates to an absolute minimum need to be reviewed.

VI. REFERENCES

1. D. J. Claflin and E. L. Wills (ed.), Quarterly Technical Progress Report on Water Recator Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, April-June 1979, NUREG/CR-0871, TREE-1300 (To Be Published July 1979).

2. M. L. Patton, Semiscale Mod-3 Test Program and System Description, NUREG/CR-0239, TREE-NUREG-1212 (July 1978).

3. D. G. Hall, A Study of Critical Flow Prediction for Semiscale Mod-1 Loss-of-Coolant Accident Experiments, TREE-NUREG-1006 (December 1976).

4. S. R. Fischer et al, RELAP4/MOD6, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems, Users Manual, EG&G Idaho Inc., CDAP TR003, (January 1978).

5. E. A. Harvego, Quick Look Report for Semiscale Mod-3 Test S-07-4 (Baseline Test Series), WR-S-78-011, EG&G Idaho, Inc. (June 1978).

6. Proposed ANS Standard, Decay Energy Release Following Shutdown of Uranium-Fueled Thermal Reactors, ANS-5-1, (October 1971).

7. L. R. Baker, Jr. and L. C. Just, Studies of Metal-Water Reactions at High Temperature, ANL 6548 (May 1962).

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APPENDIX A

SCALING RATIONALE AND HARDWARE CHANGES FOR THE TMI SIMULATIONS

APPENDIX A

SCALING RATIONALE AND HARDWARE CHANGES FOR THE TMI SIMULATIONS

1. INTRODUCTION

This appendix contains additional information that supplements the previous discussions on scaling and hardware changes made to the Semiscale Mod-3 system in order to perform Three Mile Island simulations. The first part of the appendix discusses the scaling criteria used to size the pressurizer power operated (POV) and safety relief valves and the hardware modifications made to the pressurizer. The second and third parts discuss the hardware modifications required to simulate the Babcock and Wilcox (B&W) vent valve and the loop resistance changes made to allow better simulation of the TMI plant initial conditions.

2. PRESSURIZER SCALING AND MODIFICATIONS

Several modifications had to be made to the Semiscale pressurizer in order to better represent the B&W plant hardware. These changes included adding components to represent the POV valve and the two code safety valves in addition to the installation of a new pressurizer surge line.

The scaling philosophy used to size the POV and code safety valves for the Semiscale simulations was that of attempting to maintain the ratio of valve full open area to total system liquid volume between the two systems. In equation form this can be stated as

$$\left(\frac{A}{V}\right)_{\text{Semiscale}} = \left(\frac{A}{V}\right)_{\text{Three Mile Island}} \qquad (1)$$

where:

A = valve area (full open)

V = total system liquid volume

The resulting values for the Semiscale valve areas using the above scaling technique are listed in Table A.I. For most of the simulations conducted in Semiscale, however, the POV area listed in Table A.I was not used. The area was increased to 6.567×10^{-3} cm^2 after initial experiments showed that the choked steam flow through the simulated valve was less than it should have been based on scaling from the quoted steam flow characteristics[A.1] for the TMI POV valve. The extremely small areas listed in Table A.I made it impossible to actually use valves in the simulations. Instead, in the Semiscale experiments, the desired areas were simulated with specially made sharp-edged orifices. Since the relief flows were expected to be choked, the use of the orifices was deemed to be appropriate.

The pressurizer surge line in the Semiscale system was modified to simulate both the resistance and elevation characteristics of the TMI pressurizer surge line. The desired surge line hydraulic resistance was scaled in order to maintain the same pressure drop as the reference plant using the following equation:

$$R'_{S.S.} = R'_{PWR} \frac{V^2_{PWR}}{V^2_{S.S}} \qquad (2)$$

where:

R'	=	hydraulic resistance
V_{PWR}	=	PWR system liquid volume
$V_{S.S.}$	=	Semiscale liquid volume

The resistance of the TMI surge line was calculated using the basic definition of hydraulic resistance and the geometry of the line as given in Reference A.1. By definition

$$R' = \frac{K_{entrance} + \frac{fL}{D} + K_{exit} + K_{bends}}{2\ A^2} \qquad (3)$$

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TABLE A.I

SEMISCALE AND THREE MILE ISLAND UNIT 2 PRESSURIZER VALVE AREAS

System	System Fluid Volume (m^3)	POV Area (cm^2)	Safety Area (cm^2)
Three Mile Island	314.556[a]	6.774[b]	43.110[b]
Semsicale	0.207[c]	4.458×10^{-3}	2.837×10^{-2}

(a) Data obtained from TMI-II Safety Analysis Report (Reference A.1). System fluid volume obtained by subtracting pressurizer gas volume from listed total primary volume.

(b) Data from Reference A.1.

(c) Data from Reference A.2.

where:

$K_{entrance}$ = line entrance losses (1.0)

K_{bends} = line turning losses (3.6 for four 90^{0} bends)

k_{exit} = line exit losses (0.5)

f = friction factor (estimated at 0.015)

L = surge line length (13.72 m)

D = surge line hydraulic diameter (22.22 cm)

A = surge line flow area (3.8807 cm^{2})

These values substituted into Equation (3) give an approximate PWR line resistance of $2.0391 \times 10^{-2}\ s^{2}/cm^{2}\text{-}m^{3}$). Substituting this value and the system fluid volume values into Equation (2) gives the desired Semiscale line resistance of $4.6146 \times 10^{-4}\ s^{2}/cm^{2}\text{-}m^{3}$. An iterative process was then required to size a new surge line for the Semiscale facility. Because of the limited time and hardware available, the new line was sized from 1.27 cm O.D., 0.165 cm wall thickness, drawn tubing. After accounting for pressure drops due to instrumentation, bends, line losses, and entrance and exit losses, the total resistance was calculated to be approximately $3.3797 \times 10^{-4}\ s^{2}/cm^{2}\text{-}m^{3}$. A schematic of the pressurizer and associated hardware is shown in Figure A-1.

3. MODIFICATIONS FOR VENT VALVE SIMULATION

Reactors manufactured by the Babcock and Wilcox Company are fitted with a series of "vent" valves which connect the vessel upper plenum to the downcomer inlet annulus. Under normal operation the valves remain closed since the inlet annulus pressure is higher than the upper plenum pressure. Situations in which the upper plenum pressure is higher will cause the valves to open and allow the pressure in the two parts of the system to equalize. Since the vent valves had the potential to play a role in the TMI transient, appropriate modifications were made to the Semiscale system to

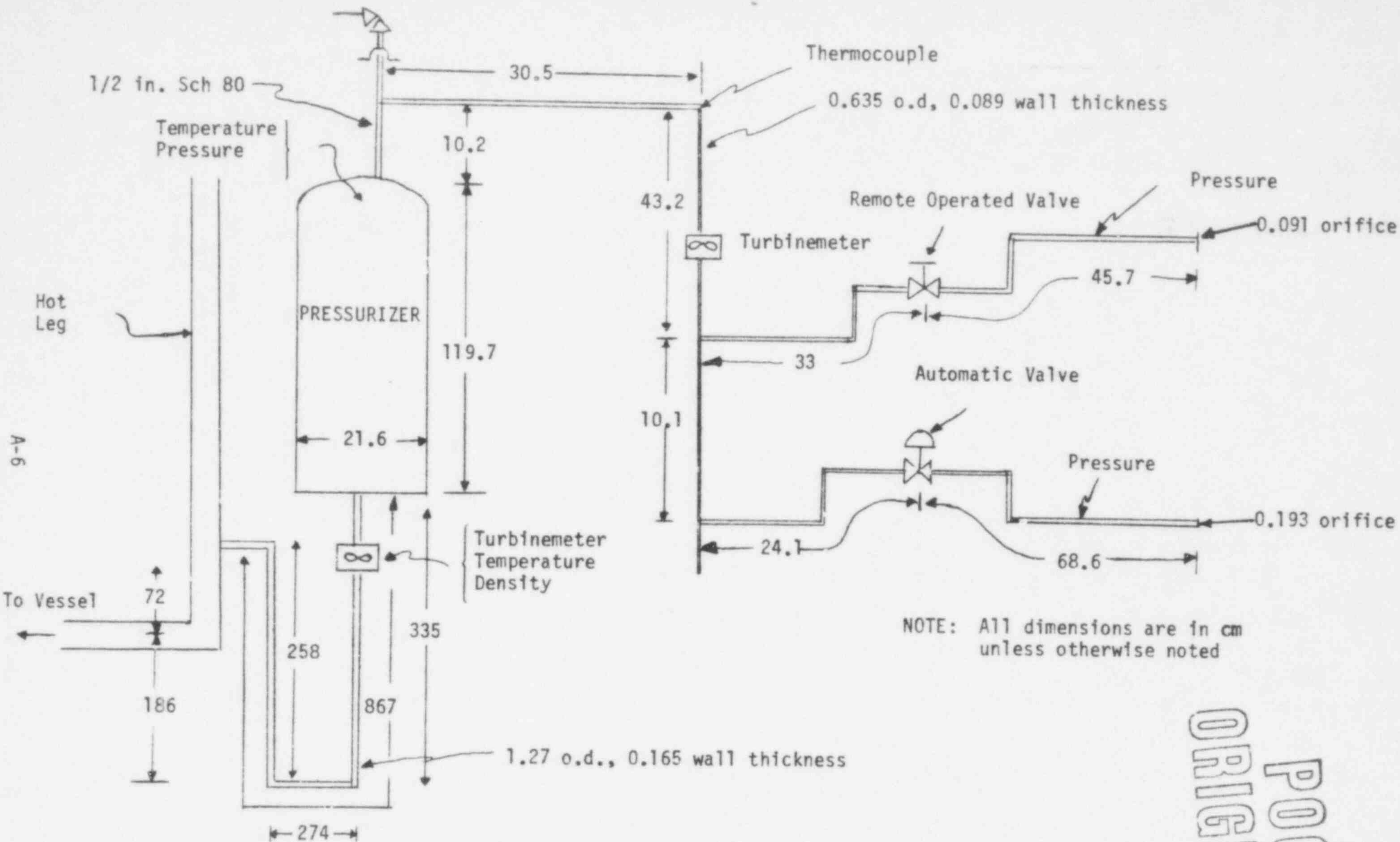

Fig. A-1 Schematic of Semiscale pressurizer in Three Mile Island simulation configuration.

simulate the vent flow path. The modifications consisted of the addition of piping to connect the Semiscale vessel upper plenum to the top of the inlet annulus. The pipe was fitted with a swing check valve and a turbine flowmeter. Line hydraulic resistance was scaled as closely as possible to the PWR value using the equations cited in the previous section. The calculated hydraulic resistance for the Semiscale vent line (including turbine meter, piping, and check valve) was 62 $s^2/cm^2\text{-}m^3$. Figure A-2 shows a schematic for the vent line as installed in the facility.

4. SEMISCALE COOLANT LOOP ORIFICE MODIFICATIONS

In order to achieve the initial conditions in the Semiscale system required to simulate the TMI plant initial conditions, modifications were made to reduce the hydraulic resistance in both coolant loops. These changes were made primarily so that the present loop coolant pumps could develop the head required to achieve core mass flow rates necessary to obtain a 27.8 K fluid temperature rise across the core. Changes to the intact loop consisted of the removal of an orifice on the steam generator inlet and the addition of a similar orifice at the vessel hot leg nozzle. The orifice installed in the vessel hot leg nozzle was intended to provide the same effective "top-of-pipe" elevation in the intact loop as exists in the broken loop* (see Figure A-3). In the broken loop, the only change made was to replace the pump discharge orifice with a venturi. The orfice was previously used to simulate pump locked rotor resistance during large break loss-of-coolant simulations and was not required in the TMI simulations. The venturi was installed in order to facilitate reassembly of the loop. Table A.II provides a summary of the Semiscale loop hydraulic resistance values.

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* The intact loop and broken loop pipes have the same centerline elevation but the intact loop pipe is of larger diameter. Therefore, the effective "top-of-pipe" elevation in the intact loop is above that in the broken loop. This difference was thought to be an important consideration during slow transients where upper plenum draining into the loops was of interest.

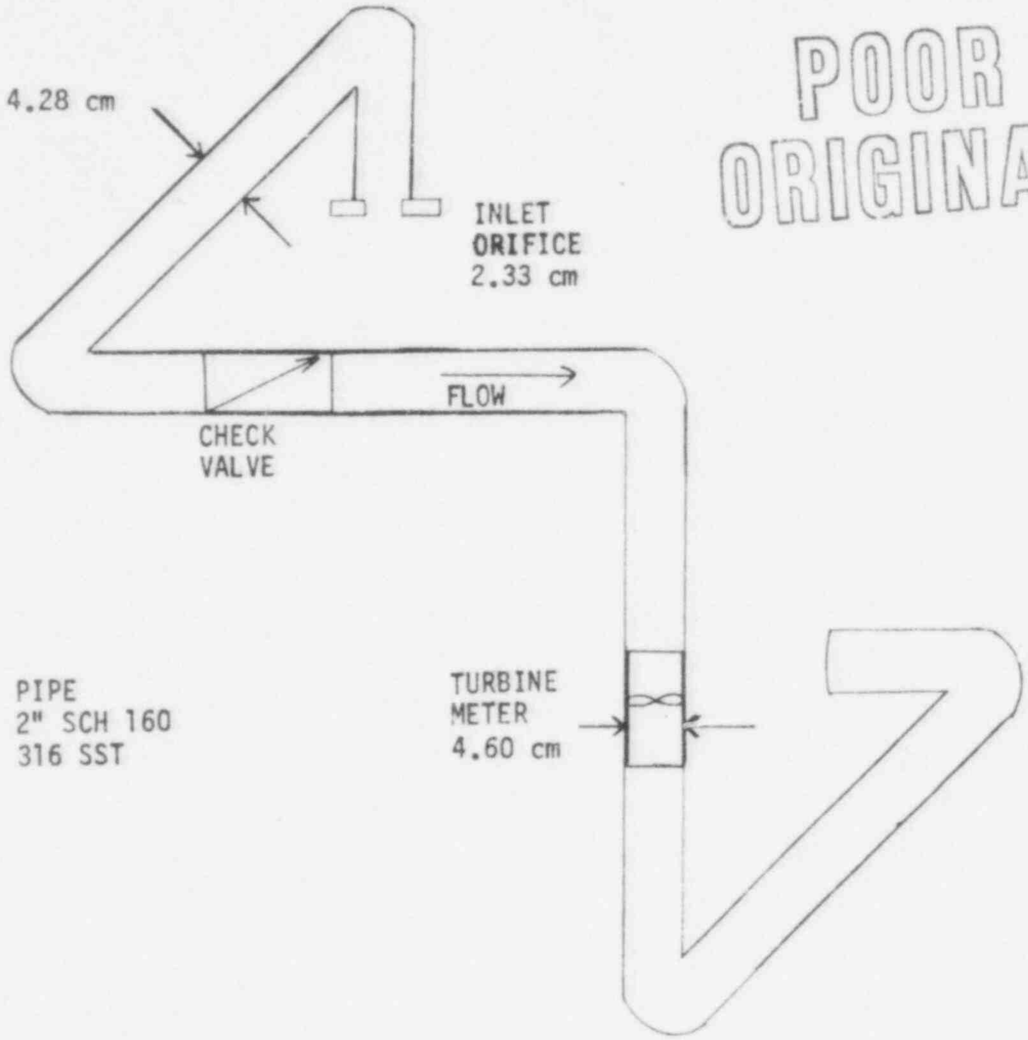

Fig. A-2 Schematic of Semiscale vent line.

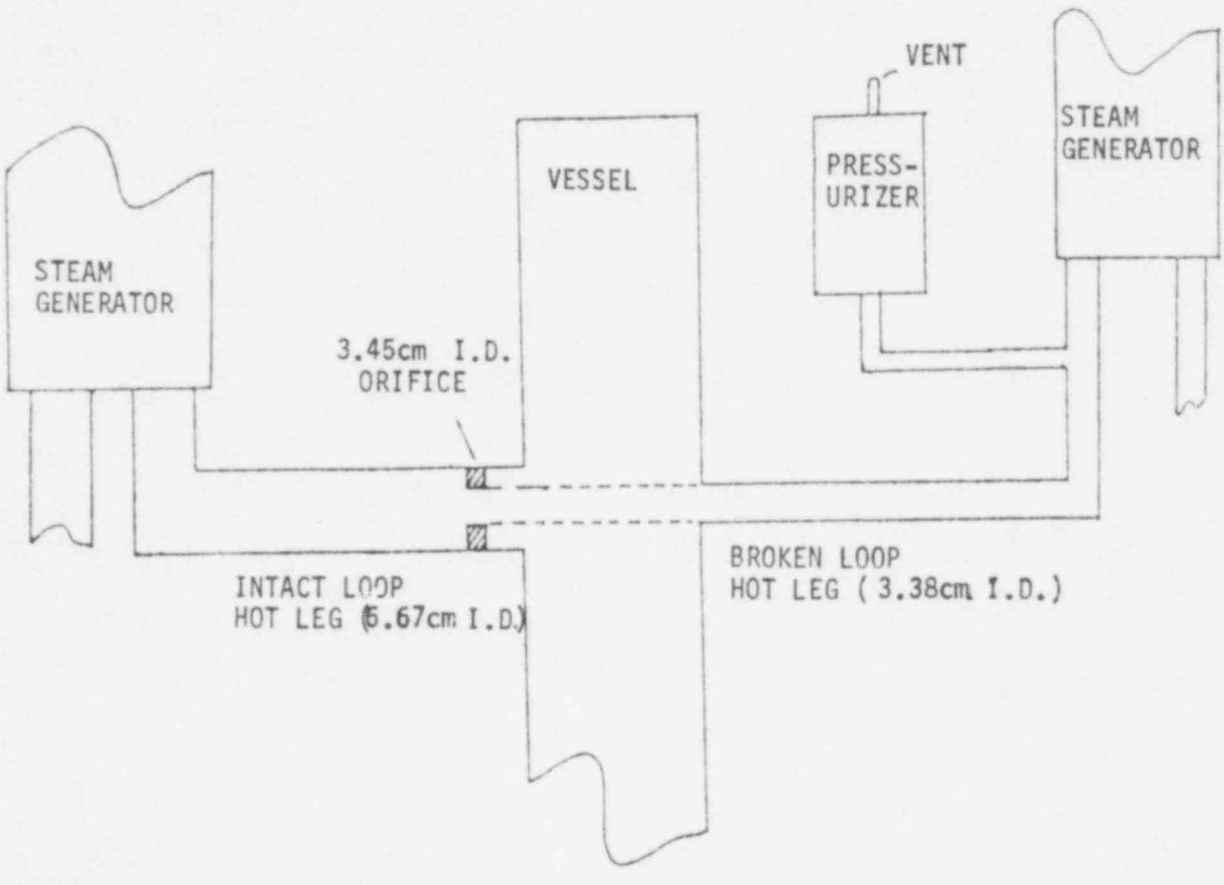

Fig. A-3 Schematic of the intact and broken loop hot legs.

TABLE A.II

SUMMARY OF SEMISCALE LOOP RESISTANCES

	Normal (s^2/cm^2-m^3)	TMI Simulations (s^2/cm^2-m^3)
Intact Loop	39.55	29.52
Broken Loop Pump Discharge	985	218.8

5. REFERENCES

A.1 Metropolitan Edison Company, Safety Analysis Report for Three Mile Island Nuclear Station Unit 2, Docket No. 50-320.

A.2 M. L. Patton, Semiscale Mod-3 Test Program and System Description, NUREG/CR-0239, TREE-NUREG-1212 (July 1978).

APPENDIX B

ELECTRICAL CORE POWER CONTROL

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APPENDIX B

ELECTRICAL CORE POWER CONTROL

The purpose of this appendix is to explain the need for use of a transient electrical core power control during the Semiscale Three Mile Island (TMI) simulations that is different from the standard nuclear core power d.cay curve[B.1] for the first 30 s. The analytical technique used to determine an appropriate electrical power control is also described.

The material property differences between the Semiscale electrical rod and a nuclear rod and the lower peak temperature limit on the electrical rod result in a somewhat different thermal performance. Since the thermal diffusivity of a UO_2 rod is much lower than that of boron nitride (principally composition by volume inside the cladding of the electrical rod), the nuclear rod will contain a greater amount of stored energy at a given temperature than will the electrical rod. Therefore, in order for the electrical rods to adequately model a nuclear core during a transient, the transient electrical power must be adjusted to account for differences in the stored energy of the rods.

The criterion for selecting an electrical rod power control is to cause the surface temperature of an electrical rod to approach as closely as possible the surface temperature calculated for a nuclear rod. This criterion was met by matching the transient surface heat flux calculated for an electrical rod with the transient surface heat flux calculated for a nuclear rod, assuming that both rods were subjected to the same transient boundary conditions. These calculations were performed using one-dimensional analytical heat conduction models of the electrical and nuclear rods. The power decay curve applied to the nuclear rod was the proposed standard power decay discussed in Reference B.1. Since the Semiscale electrical heater rods have a fixed axial peaking factor of 1.55, use of this technique

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allows the matching of electrical and nuclear rod surface heat fluxes at only one axial location. The rod axial location of peak power generation (the hot spot) was the point at which the nuclear and electrical fluxes were matched.

Figure B-1 shows a comparison of the electrical core power curve required to simulate nuclear rod decay heat for the Semiscale TMI simulations. The electrical power curve was developed assuming that the peak linear power generation rate of 39.36 kW/m and that the surface heat transfer mechanism was nucleate boiling. The boundary condition (pressure) required for use in the nucleate boiling heat transfer correlation was taken from data received from the Three Mile Island Unit 2 plant. The convergence of the two power traces at about 60 s indicates that the nuclear and electrical rod stored energy differences at that point in time are minimal and that the electrical power required to simulate nuclear rod thermal performance is equivalent to the nuclear rod decay value.

Although the curve shown in Figure B-1 represents the electrical power required in Semiscale to simulate the core decay heat characteristics for the TMI plant, in most of the experiments conducted, the actual core power used was higher than that shown. This was required to offset atypically large heat losses in the Semiscale facility.

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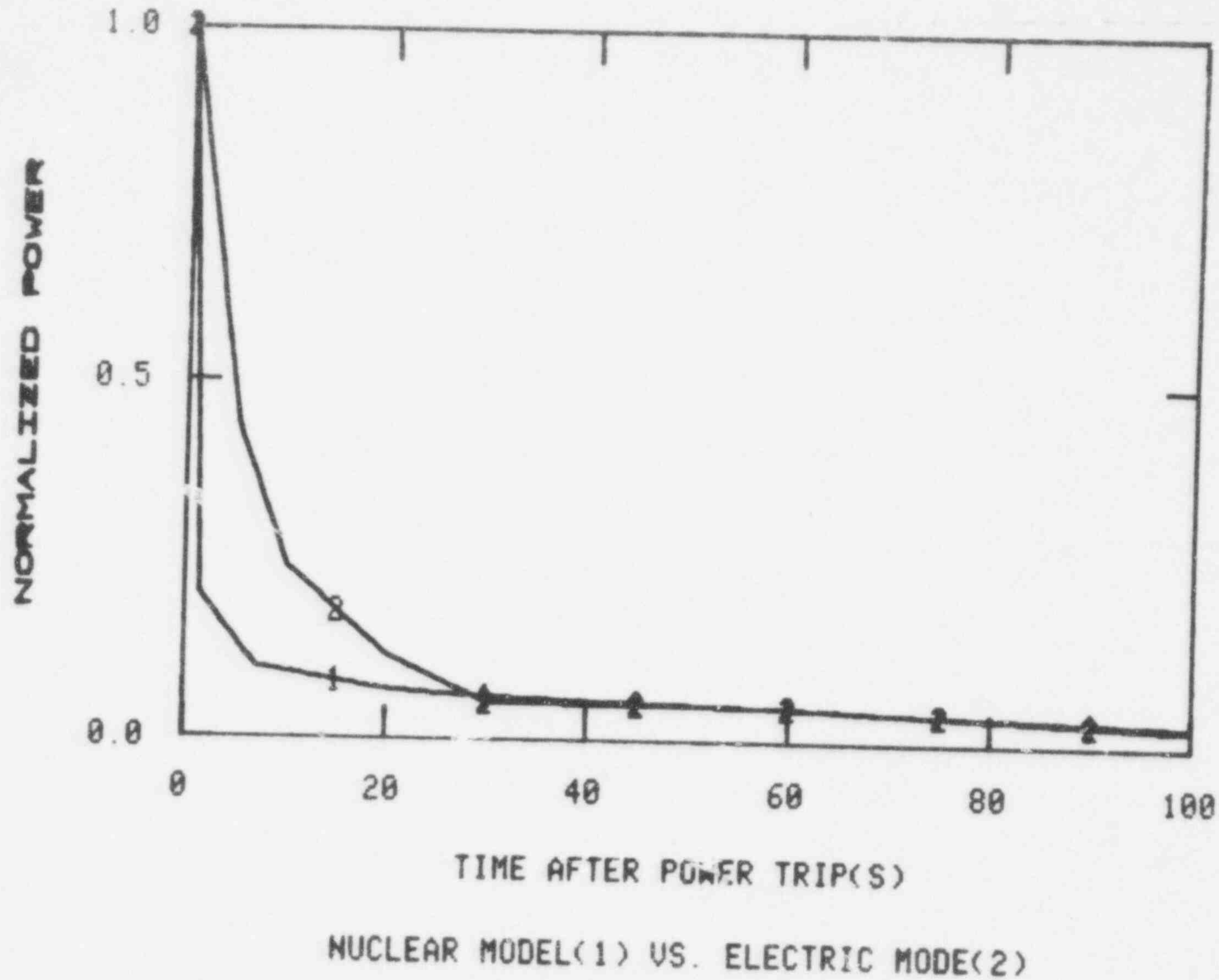

Fig. B-1. Comparison of nuclear rod decay heat and Semiscale core power required to simulate decay heat for Three Mile Island simulations.

REFERENCE

B.1. Proposed ANS Standard, Decay Energy Release Following Shutdown of Uranium-Fueled Thermal Reactors, ANS-5-1, (October 1971).

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APPENDIX C

CORE POWER PROFILE FOR SEMISCALE AND TMI

APPENDIX C

CORE POWER PROFILE FOR SEMISCALE AND TMI

The calculation of TMI core cladding temperatures based on heat transfer coefficients from the Semiscale simulation was made using only one axial power profile. This appendix presents a one-eigth segment power map and compares the power profile for the central assembly used in the TMI heat up calculations (Section 4) with profiles in other rod bundles and with the Semiscale profile. The TMI power profiles were calculated from output of the GRASP computer program. The GRASP program used self powered neutron detector data taken one week before the TMI transient. Table C.I gives the detector data.

A sample of TMI axial power profiles is presented on Figure C-1. For comparison, the high power Semiscale axial power density profile is also shown. Although heatup calculations for TMI were for assembly H-8 other radial positions could have had severe damage above 2.9 m elevation because the upper elevation power densities are similar for these different radial positions.

The Semiscale power density profile is slightly lower than the TMI profile for positions above about 2.5 m; however, the heat transfer coefficients at upper elevations are governed by fluid hydraulics rather than by local power density. The integral of the power from the bottom of the core to the point in question influences the hydraulics to a greater extent than the local power.

Table C.I TMI-2 GRASP Core Power Map *

CONDITIONAL CERTIFICATION * * CONDITIONAL CERTIFICATION * * CONDITIONAL CERTIFICATION * TM2FAHV

GRASP VERS.4.3 CREATED 12/ 6/78
BASE= 14 TMI-2 CYCLE 1 GRASP ANALYZE PDQ 03-19-79
03/22/79 PLANT: JERSEY CENTRAL P&L 3 MILE ISLAND 2
DATA TAKEN ON 03/19/79 8:10 EST ANALYZE

CYCLE BURNUP = 79.60 EFPD POWER = 2710.5 MWT
CORE BURNUP = 2662. MWD/MTU % FULL POWER = 98.%
BANKS 6/7/8 = 96/ 95/ 27 %WD BORON = 1037. PPM

SEGMENT MAP
PDO-ITEM:R

LEVELS USED: 1 THRU 7 LISTED TOP TO BOTTOM IN INCREASING ORDER

	8	9	10	11	12	13	14	15
H	2.3793	2.1162	1.8773	1.9392	1.6423	1.9313	2.1545	1.3866
	3.6383	3.1707	2.8145	2.9233	2.4290	2.8899	3.2433	2.1306
	3.8923	3.4167	2.9993	3.0562	2.5354	2.9910	3.3294	2.2257
	3.8013	3.4543	3.0094	3.0794	2.6263	3.0170	3.2949	2.2249
	3.6217	3.3994	2.9926	3.1901	2.7883	3.1320	3.3810	2.2734
	3.2327	3.0270	2.7345	3.0010	2.5793	2.8793	3.1253	2.0616
	1.9037	1.7144	1.5917	1.7034	1.3722	1.5414	1.6997	1.0870
K		1.8885	1.9872	1.8033	1.9189	1.7518	1.7196	1.2766
		2.8717	2.9781	2.6984	2.7764	2.5892	2.5841	1.9711
		3.0607	3.1580	2.7531	2.7233	2.6270	2.6589	2.0798
		3.0816	3.1982	2.7457	2.7129	2.6356	2.6435	2.0764
		3.1135	3.2577	2.8937	2.9714	2.7985	2.7184	2.0491
		2.8837	3.0210	2.7687	2.9159	2.6668	2.5082	1.7229
		1.6660	1.7363	1.5605	1.6203	1.4772	1.3584	.8041
L			1.8064	1.9365	1.7899	1.7769	1.7852	.9373
			2.7259	2.8053	2.4298	2.6107	2.6802	1.4730
			2.8050	2.7238	1.9789	2.5280	2.7383	1.5689
			2.8070	2.7084	1.7417	2.4891	2.7085	1.5919
			2.9552	2.9945	2.4271	2.7459	2.8065	1.6450
			2.8327	2.9315	2.7369	2.6882	2.6414	1.5095
			1.6094	1.5827	1.3535	1.4373	1.4689	.8081
M				1.7461	1.7458	1.4548	1.2989	
				2.5360	2.5596	2.1689	2.0100	
				2.4431	2.4366	2.1520	2.0503	
				2.4304	2.4006	2.1250	2.0124	
				2.7076	2.7313	2.2953	2.1090	
				2.6239	2.7507	2.2340	2.0199	
				1.3547	1.4844	1.2324	1.1246	
N					1.4246	1.3245	.8144	
					2.1226	1.9653	1.2581	
					2.1440	2.0209	1.2878	
					2.1411	2.0090	1.2972	
					2.2839	2.0690	1.3973	
					2.1873	1.9485	1.3313	
					1.2053	1.1118	.7082	
O						.8916		
						1.3941		
						1.4125		
						1.3692		
						1.4290		
						1.3534		
						.7558		

* All numbers listed are powers in MW per segment of a bundle. The segment powers from top to bottom on the page represent from bottom to top in the core. Each segment represents 0.52 m and each bundle has 208 rods.

Therefore, the kW/m of any position is:

$$(\mathrm{MW} / (208 \text{ rods} \times 0.52 \text{ m})) \times 1000 \frac{\mathrm{kW}}{\mathrm{MW}}$$

POOR ORIGINAL

928 072

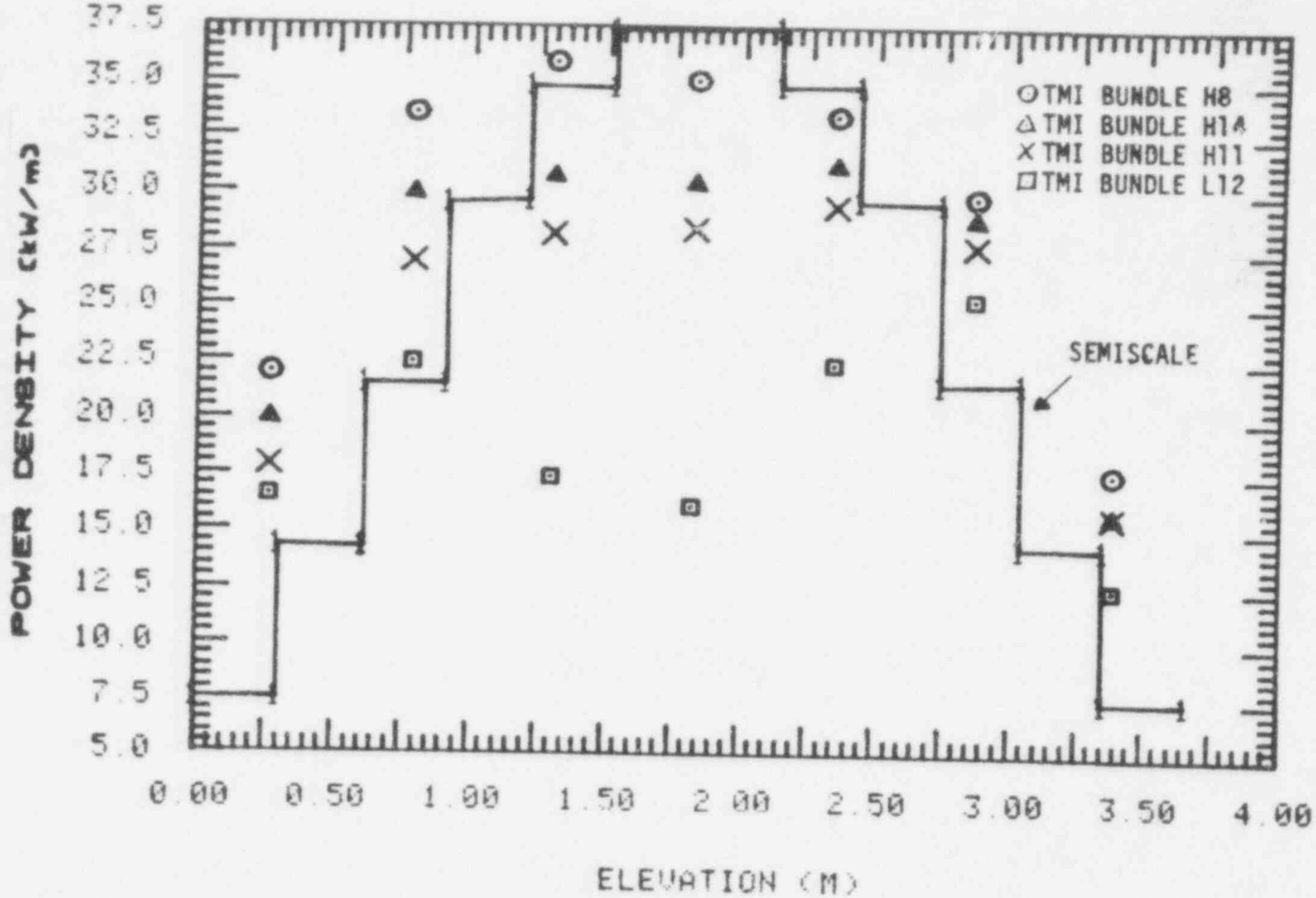

Fig. C-1 Comparison of Test S-TMI-2 axial profile for several radial positions to the Semiscale axial power profile.

928 073