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Author(s): G. M. Rogers

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

This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

for H. P. Pearson, Supervisor
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Prepared for
U.S. Nuclear Regulatory Commission
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INTERIM REPORT

NRC Research and Technical Assistance Report

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EG&G Idaho

INTEROFFICE CORRESPONDENCE

date April 26, 1979

to L. P. Leach

from D. J. Olson

subject TRANSMITTAL OF THE FINAL LETTER REPORT FOR THE MOD-3 VERSUS PWR SCALING STUDY (TASK NO. 50913) - DJO-53-79

Attached is the final letter report (SEMI-TR-005) documenting the results of the Mod-3 versus PWR scaling study. This report was prepared for LOFT with funding arranged through a cooperative agreement between the NRC and OSTERREICHISCHE STUDIENGESELLSCHAFT FUER ATOMENERGIE (SGAE), Vienna, Austria.

The report presents the results of a comprehensive investigation of scaling effects in the Semiscale Mod-3 system, which was designed to simulate the response of a pressurized water reactor (PWR) equipped with upper head injection (UHI). The investigation utilizes both calculations and experimental data to identify the potential influence of individual component scaling compromises on Mod-3 system performance during the simulation of a hypothesized 200% cold leg break loss-of-coolant accident (LOCA) in a PWR. The results presented indicate that, while Semiscale Mod-3 will not entirely duplicate the thermal-hydraulic behavior of a PWR with UHI, the results should be sufficiently representative to provide information on important phenomena expected to occur in a PWR.

Transmittal of this letter report completes the scope of work authorized under LOFT Task No. 50913.

EAH:ln

Attachment:
As stated

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FORM EG&G-954
(Rev. 2-77)

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SEMI-TR-005

Analysis of Scaling the Semiscale Mod-3 System to a Pressurized Water Reactor

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IDAHO OPERATIONS OFFICE UNDER CONTRACT EY-76-C-07-1570

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EG&G Idaho, Inc.

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INTERIM REPORT

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Funding for this document was provided to the LOFT Experimental Program by a cooperative agreement between NRC and OSTERREICHISCHE STUDIENGESELLSCHAFT FLLER ATOMENERGIE (SGAE), Vienna, Austria.

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ABSTRACT

Scaling concepts and approaches used in designing the Semiscale MOD-3 system to provide scaled simulation of the thermal and hydraulic performance of a pressurized water reactor (PWR) equipped with upper head injection (UHI) are discussed. Calculations and experimental data were used to identify the potential influence of scaling compromises on Semiscale MOD-3 system behavior for each system component. The overall capability of the Semiscale MOD-3 system for use in assessing the UHI process is not expected to be altered by these compromises; thus providing a data base for the development of codes that can be used to calculate UHI behavior. The results obtained indicate that, while Semiscale MOD-3 will not entirely duplicate the thermal hydraulic behavior of a PWR with UHI, the results should be sufficiently representative to provide information on important phenomena expected to occur in a PWR.

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SUMMARY

The Semiscale Mod-3 Experimental Program is part of the overall Semiscale Blowdown and Emergency Core Cooling (ECC) Project conducted by EG&G Idaho, Inc., to investigate the thermal and hydraulic phenomena accompanying an hypothesized loss-of-coolant accident (LOCA) in a pressurized-water reactor (PWR) system. The Semiscale Mod-3 Program provides data for developing and verifying analytical models used to predict the performance of PWR systems during a LOCA. An additional objective is to provide data for the assessment of the upper head injection (UHI) concept.

Many scaling philosophies were considered in designing Semiscale Mod-3. In order to maintain the ratio of energy input to the total system volume the same in Semiscale Mod-3 as in a PWR, the volume scaling approach was selected. The system characteristics that resulted from this scaling approach are discussed and then related to each specific component in the Semiscale Mod-3 system. Specific areas where scaling was not achieved or where compromises in expected PWR behavior were necessary are also identified.

Some of the major design features of Semiscale Mod-3 include the capability for upper head injection (UHI), an external downcomer design which allows increased measurement capability in the core, and active pumps and steam generators in the broken and intact loops. These active loop components establish initial conditions in Mod-3

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which are typical of PWR initial conditions. Also included in the Semiscale Mod-3 design are a variety of support systems, including ECC subsystems, that provide conditions in Semiscale similar to those expected in a PWR during a hypothesized LOCA.

The intent in the Semiscale Mod-3 Program has been to design a high level of flexibility into the ECC subsystems such that injection locations, configurations, flow rates, subcooling, and pressures can be varied to experimentally investigate important parameters related to ECC performance. This added flexibility in the Semiscale Mod-3 ECC systems will expand the understanding of ECC performance characteristics, which can influence important system phenomena in a PWR. Although the Semiscale Mod-3 ECC systems cannot provide the simulation of every complex phenomena that occurs in a PWR, the ECC systems are expected to simulate overall behavior typical of a PWR.

Specific conclusions relative to system capability and limitations can be summarized as follows:

(1) Previous experience with integral-type loss-of-coolant experiments (LOCE) tend to confirm that the Semiscale system will preserve all major LOCE thermal-hydraulic behavior expected to occur in the PWR system in an appropriate time frame. The Semiscale Mod-3 system is expected to reproduce the magnitude of phenomena occurring in the PWR such as saturated blowdown decompression rates, fluid density, flow rates, and pressure drops in the operating and blowdown

lo ›rovided the PWR pump head degradation with void fraction is similar to that measured for the Semiscale Mod-3 intact loop pump.

(2) Differences in performance between a PWR and Semiscale Mod-3 are expected where two- and three-dimensional system effects influence controlling phenomena. Two particularly important phenomena influenced by two- and three-dimensional effects are core thermal performance and vessel lower plenum liquid level. The Semiscale Mod-3 system can not simulate the two- and three-dimensional temperature and flow distributions of the PWR core or the two and three-dimensional velocity distributions of the PWR vessel downcomer and plenum regions.

(3) The stored energy in the metal structures of Semiscale Mod-3 system is greater per unit volume of system fluid than that in a PWR, and as a result the released energy may cause adverse effects in system response. Various forms of insulation and materials with low thermal capacitance and low thermal conductivity have been incorporated in Semiscale Mod-3 system to reduce these effects.

The material presented in this document indicates that, although Semiscale Mod-3 will not entirely duplicate the thermal-hydraulic behavior of a PWR with UHI, the results should be sufficiently representative to provide information about operational parameters and

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system interaction characteristics that may be important in a PWR. Therefore, despite some scaling compromises, inherent in any small scale system, it is expected that the Mod-3 will provide basic insight into the UHI concept In addition, the data obtained from the Semiscale Mod-3 will be of significant importance in the development and assessment of codes used to predict UHI behavior.

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SCALING ANALYSIS BETWEEN THE SEMISCALE MOD-3 SYSTEM
AND A PRESSURIZED WATER REACTOR

I. INTRODUCTION

This document examines the application of the thermal-hydraulic scaling concepts used in designing the Semiscale Mod-3 test facility[1]. It also attempts to identify compromises that occur as a result of applying these scaling concepts and to assess the influence of these compromises on overall system behavior. Since the emphasis of the Mod-3 scaling approach has been to obtain results representative of those expected in a hypothesized, large cold leg break, loss-of-coolant accident (LOCA) in a large Pressurized Water Reactor (PWR), the analysis presented herein is limited to phenomena associated with this type of accident simulation. The objective of this report is to place the Semiscale Mod-3 system capabilities and limitations in perspective for those who will ultimately use the experimental data for light-water reactor safety evaluations.

From the inception of the Semiscale Program[2], it was recognized that traditional or classical thermal-hydraulic scaling laws could not be utilized successfully to design a small experimental model that would have the capability to reproduce the complex thermal-hydraulic two-phase response of a PWR during a LOCA. The impracticality of building full-scale experiments for reactor safety studies and the difficulty of designing reduced scale experiments to provide

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demonstration type data have been key considerations in the development of the U.S. Nuclear Regulatory Commission's water reactor safety program. Therefore, from the beginning, the basic approach has been to utilize data from experimental programs which incorporate both separate and coupled (or scaled integral) effects experiments to develop and assess analytical models that can be used to calculate the behavior of full-scale reactor systems. The Semiscale Mod-3 system has been designed as a scaled integral experiment system.

The Semiscale Mod-3 system was designed to simulate the primary features of a PWR. As a result, the Mod-3 system features a simulated full length PWR core with an active steam generator and pump in both the intact and broken loops, which is in contrast to earlier Semiscale systems[2] which had a shorter core and included a simulated steam generator and pump in the broken loop. Scaling emphasis in the Semiscale Mod-3 system has been concentrated in the upper head, since appropriate simulation of upper head injection (UHI) behavior is a major objective of the Semiscale Mod-3 Experimental Program. Proper scaling of the distribution of internals and their elevations in the upper head, upper plenum, and vessel was strictly adhered to, to insure that the thermal-hydraulic behavior would be similar to that expected in a PWR.

The concept of UHI involves injecting ambient temperature emergency core coolant (ECC) water into the upper head of the vessel very early in the blowdown sequence. As the system depressurizes, this fluid is expected to be drawn down through the heated core and

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provide early core cooling. The UHI process has introduced new modeling requirements to be incorporated into existing computer codes to calculate the thermal-hydraulic behavior of a PWR with UHI capabilities. The objective of the Semiscale Mod-3 testing program is to provide data to develop and assess computer codes designed to calculate thermal-hydraulic behavior for postulated LOCAs involving a PWR with UHI and to provide a basic experimental understanding of the UHI process under LOCA conditions.

The scaling concepts and philosphies applied to the Semiscale Mod-3 system are presented in Section II. Section III is a discussion of the system components, including their design, and compromises that may have resulted from conflicting scaling requirements. Where possible, analytical calculations are presented in Section III to identify specific characteristics which could influence the overall behavior of the Semiscale Mod-3 system. However, a detailed analysis of integral system effects has been limited because no computer codes are currently available that adequately calculate UHI behavior in an integral system. Therefore, final assessment of many expected or calculated integral effects can only be achieved through future tests to be conducted in the Semiscale Mod-3 system. The data obtained from these tests will then contribute to the experimental data base used in the development and assessment of current and future analytical codes.

II. SCALING CONCEPTS

Over the past decade, much experimental effort has been devoted to developing scaling approaches used in designing small scale LOCA experiments. This task has been complicated by the transient nature of the LOCA phenomena, especially when comparing complex and sometimes poorly understood two-phase flow characteristics from separate effects test results to the much more complex system behavior of an integral experiment. Although it is believed that the integral system is capable of modeling the expected PWR system behavior, the complex interactions between the many components that make up an integral system increase the difficulty in selecting the most appropriate scaling approach.

An indication of the complexity of the scaling task and the difficulties in applying strict scaling principles to an integral test facility is provided by Dr. L. J. Ybarrondo in the introduction to a paper examining scaling effects in the Loss-of-Fluid Test (LOFT) experimental facility[1] which states:

"It is generally recognized that for steady state, single-phase, thermal-hydraulic systems the application of traditional or classical thermal-hydraulic scaling laws, although valuable, are very limited when applied to complex thermal-hydraulic systems involving multiple flow paths of different sizes and energy transfer in pumps and heat exchangers. When the system being scaled is not only complex, but

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proceeds from a steady to transient state while incorporating interrelated single- and two-phase thermal-hydraulic, nuclear, and mechanical phenomena, the scaling task must of necessity involve selected compromises."

The basic conclusions reached by Ybarrondo were also found to be applicable to the design of the Semiscale Mod-3 system. Specifically, no single scaling approach was found to be entirely acceptable, in that, scaling approaches which adequately modeled important phenomena during one phase of a LOCA transient were inadequate for modeling important phenomena during other phases of the transient. For example, linear scaling, which is the maintenance of length-to-diameter ratios, would assure the correct timing and magnitude of pressure changes throughout the Mod-3 system during the very short subcooled blowdown period. However, application of this same scaling approach to the much longer saturated blowdown period would result in a large reduction in the time scale and a substantial distortion of the energy redistribution process.

The basic scaling approach for the Mod-3 system was developed after evaluating each phase of the LOCA (subcooled blowdown, saturated blowdown, lower plenum refill, and core reflood) to determine which thermal-hydraulic phenomena are most important to the overall LOCA simulation. The selected design approach utilized volumetric scaling principles in the sizing of individual components, while, in most cases, preserving full-scale elevation effects. The most significant advantage of using volumetric scaling principles was that by maintaining the same ratio of core power to system volume the relative amount of fluid energy exchange would be the same in the experimental system as in the full-scale system. This approach also allowed the preservation of time scale in the model by scaling of the break area. The decision to preserve full-scale elevation effects in the volume-scaled Mod-3 system was made because hydrostatic and dynamic fluid head characteristics have a strong influence on the upper head drain characteristics and the subsequent core thermal response during the simulation of a LOCA transient with UHI.

The overall Semiscale Mod-3 system scaling approach, which emphasized the preservation of relative fluid-energy exchange, time scale, and fluid elevation effects, was based on the recognition that a mismatch of these parameters between Semiscale and a full sized PWR could result in significant distortions in important hydraulic effects such as break flow, two-phase pressure drops, and pump performance. These effects are all influenced by steam generation and fluid quality. However, steam generation and fluid quality are dependent on rate-controlled phenomena such as energy transfer from the core heater rods to the fluid in the core and between the steam generator secondary side and primary side fluids. Therefore, the requirement for time-scaled energy transfer processes also identifies the need for geometric and dynamic similarity (application of classical scaling laws) in components such as the core and steam generator to insure that the heat transfer surface areas are scaled to a typical four-loop PWR.

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The application of volumetric scaling principles, the preservation of full-scale elevation effects, and the requirement to maintain geometric and dynamic similarity in the core and in the steam generators provide the following specific criteria, or characteristics that should be similar to a PWR. In the Semiscale Mod-3 system design (a) the ratio of core power to system volume, (b) the volume distribution for various regions of the system and (c) the relative hydraulic resistance distribution throughout the system, are all similar to a PWR.

The above mentioned scaling approaches also require some characteristics of small scale systems to be the same as those of the larger model. Therefore, the following characteristics should be maintained the same in Semiscale Mod-3 as in a PWR:

(1) The relative elevations between major volumes in the system

(2) The heat transfer surfaces in the core and steam generator (that is, full-length rods, typical rod pitch, and typical rod/tube diameter)

(3) The core length and axial power distribution

(4) The ratio of core flow area to system volume.

The above characteristics are designed into the Mod-3 system to give the best representation of overall LOCA behavior in a small system. However, compromises in the design of individual components were necessary since the overall scaling concepts could not assure the experimental system response would be similar to full-scale system response where two- and three-dimensional effects were significant. Specific areas where two- and three-dimensional effects in a PWR are expected to contribute to differences between Semiscale Mod-3 system and PWR hydraulic phenomena are:

(1) Subcooled decompression stress loads

(2) Two-phase flow regimes and pressure drops when other than a homogeneous regime exists in either system

(3) Liquid entrainment, phase separation, and mixing in piping and plenums

(4) Radial flow in the core, vessel upper and lower plenums, and downcomer

(5) Countercurrent flow in the downcomer.

With the exception of subcooled decompression loads, which are of concern primarily because of their importance to the structural design of the Mod-3 system, the capability of the Mod-3 system to duplicate the PWR hydraulic phenomena is limited. Consequently, the Mod-3

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system has been designed to include flexibility in various systems to bound important phenomena as well as a vast amount of instrumentation to determine the magnitude of the phenomena. The effects from differences in hydraulic phenomena, however, are expected to be localized and should not alter the capability of the Mod-3 system to simulate overall PWR system response.

The following section discusses the scaling of individual components in terms of their ability to meet specified design objectives, compares key features of the Mod-3 system with those in a PWR, and describes the expected effect of component scaling compromises on overall system behavior.

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III. SCALING CHARACTERISTICS AND COMPROMISES OF DESIGN FEATURES IN THE SEMISCALE MOD-3 SYSTEM

The Semiscale Mod-3 system, which is shown in Figure 1, has been designed to simulate the major features of a full-scale PWR but is much smaller in volume. A typical four-loop PWR was used for the scaling of the intact and broken loops, and the simulated reactor pressure vessel design includes features representative of a Westinghouse plant with UHI. Because each of the loops in a four-loop PWR are identical, the Semiscale Mod-3 system represents the three unbroken loops in a PWR by a single intact loop and the one ruptured PWR loop by a broken loop fitted with a break apparatus which will allow changes in break size, configuration (communicative or noncommunicative), and location, as shown in Figure 2.

The Semiscale Mod-3 system design utilized the scaling concepts discussed previously; however, in many cases scaling in one specific area created compromises in another area. Areas of primary importance were selected to be scaled while secondary areas were necessarily compromised. The major differences or design compromises between the Semiscale Mod-3 and PWR systems are:

(1) The use of electrical heater rods in the Semiscale Mod-3 system to simulate the nuclear fuel rods in the PWR

(2) The axial length of the Semiscale Mod-3 intact loop steam generator

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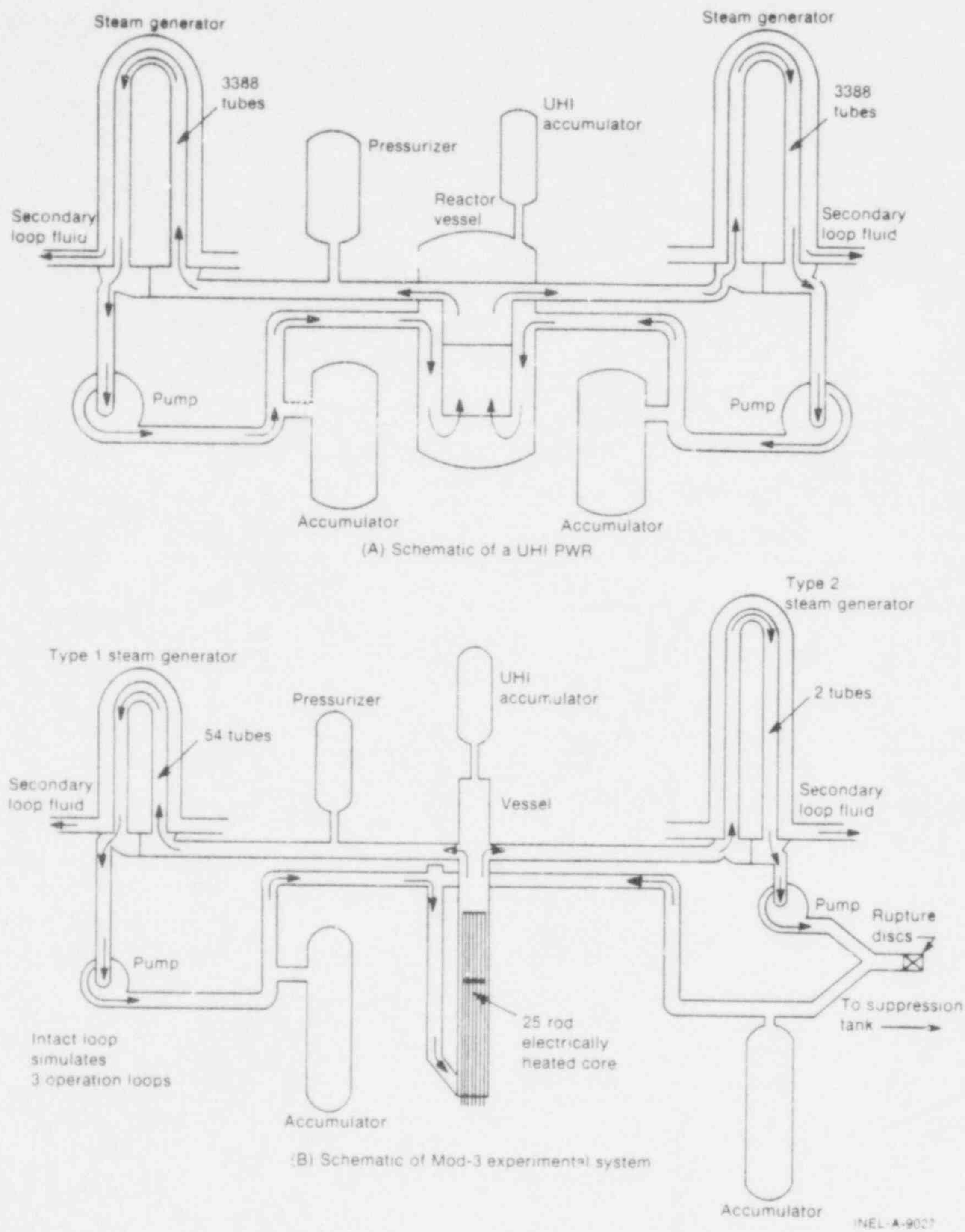

Fig. 1 Comparison of PWR with UHI and Semiscale Mod-3 system schematics.

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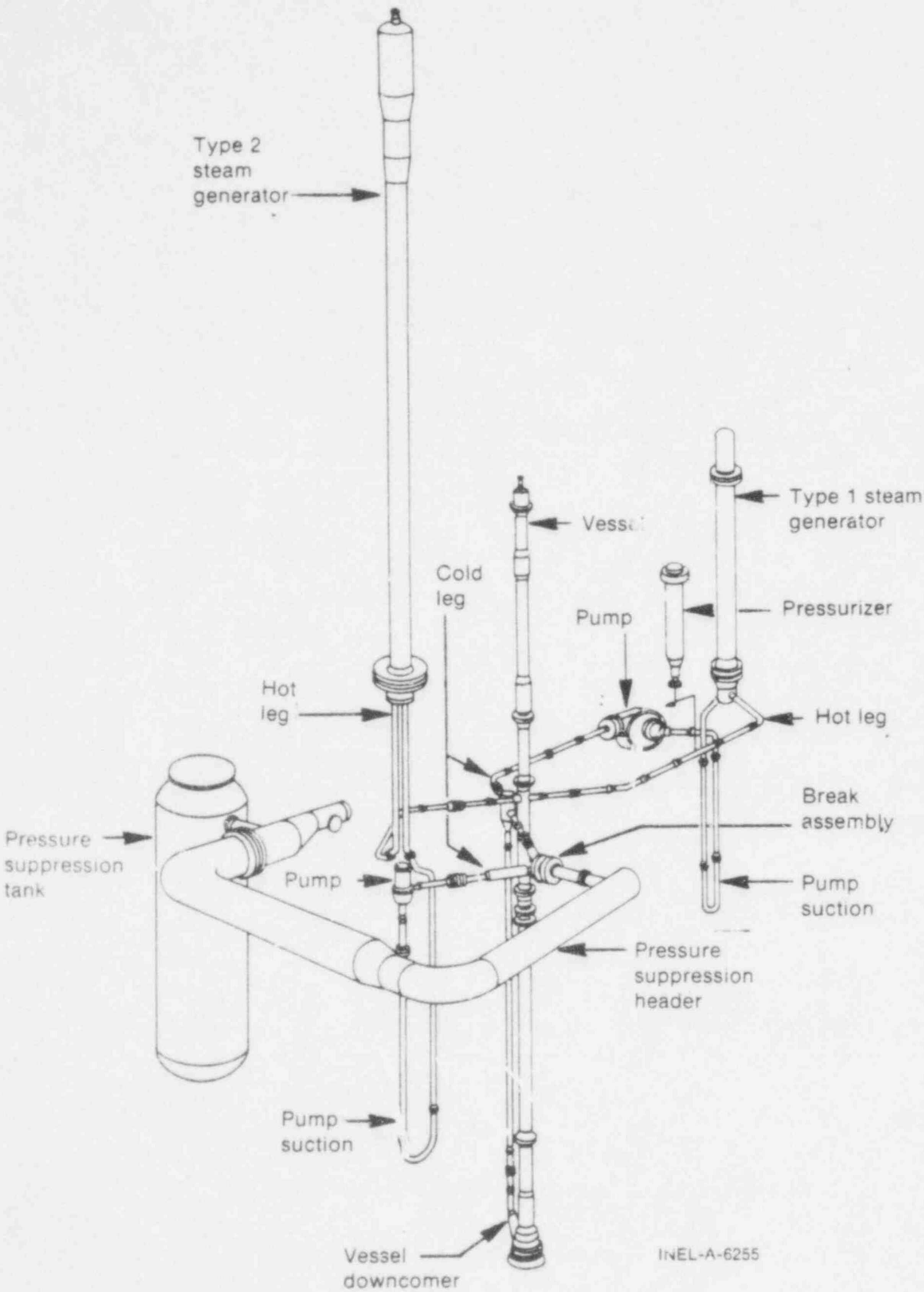

Fig. 2 Semiscale Mod-3 system configuration for noncommunicative cold leg break.

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(3) The one-dimensional characteristic of the pressure vessel in the Semiscale Mod-3 system

(4) The upper head structure with simulated core support columns (two) and a control rod guide tube

(5) The external single pipe downcomer design

(6) The lower plenum design with heater rods penetrating through the bottom of the pressure vessel.

Each of these areas of differences or design compormises are addressed in the following sections, which also include discussions of the function and operation of the various components in the Semiscale Mod-3 system. Section 1 discusses the simulated reactor pressure vessel; Section 2, the intact and broken loop; and Section 3, the ECC injection systems. In addition to volumetric scaling considerations, these sections deal with the effects of component elevation, surface area, flow area, and component pressure losses on overall system behavior. Therefore, for convenience, Tables I and II, which compare corresponding values of these parameters in the Semiscale Mod-3 system with desired values scaled from a typical PWR, have been included and are referenced in subsequent discussions of individual components.

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TABLE I

COMPARISON OF PWR AND SEMISCALE MOD-3 COMPONENT ELEVATION[a]

Location	Semiscale (cm)	PWR (cm)
Cold leg nozzle spillover level	-2.41	-34.93
Top of heated core	-127.00	-158.75
Bottom of heated core	-496.00	-520.00
Pump inlet (casing interface)	-260.77	-177.14
Pump discharge pipe centerline	0	0
Pump suction leg low point centerline	-272.77	-314.30
Steam generator bottom of tube sheet:		
Type I (intact loop)	+97.54	+207.26
Type II (broken loop)	+346.71	
Steam generator low tube spill-over:		
Type I (intact loop)	+346.71	+1114.04
Type II (broken loop)	+1109.78	
Top of core to top of pipe inside diameter in pump suction leg trap:	+139.196	+144.78[b]

a. Elevations are relative to cold leg nozzle centerline, which is the zero reference elevation point; + indicates above nozzle centerline.

b. Top of core is above pump trap.

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TABLE II

VOLUME AND VOLUME DISTRIBUTION FOR PWR VERSUS SEMISCALE MOD-3 (COLD LEG BREAK)

	Desired Volume Scaled from PWR m^3 (1)	Calculated MOD-3 Volume m^3	% Diff
1.0 Vessel			
1.1 Downcomer Region			
Distribution annulus	0.00709	0.00767	8.2
Downcomer pipe	0.00740	0.01447	95.4
Core bypass	0.00389	- (2)	-
TOTAL DOWNCOMER VOLUME	0.1838	0.02214	20.5
1.2 Upper head region			
Above top of guide tube	0.00445	0.00464	4.4
Below top of guide tube	0.00875	0.00937	7.1
TOTAL UPPER HEAD VOLUME	0.01320	0.01401	6.2
1.3 Upper plenum 0.01024	0.01119	9.2	
1.4 Core region0.01073		0.01056	-1.5
1.5 Lower plenum 0.01612	0.01566	-2.9	
1.6 Control rod guide tube	0.00495	0.00153	-69.
1.7 Core support tubes	0.00070	0.00040	-43.
TOTAL VESSEL VOLUME	0.07432	0.07549	1.6
2.0 Intact Loop			
2.1 Hot leg	0.00393	0.01025	161.
2.2 Pressurizer (liquid volume)	0.01793	0.01370 (3)	-23.6
2.3 Surge line	0.00076	0.00037	-52.
2.4 Steam generator	0.05365	0.04205	-21.6
2.5 Pump suction leg	0.00628	0.02430	287.
2.6 Pump 0.00399	0.00408	2.3	
2.7 Cold leg	0.00423	0.00883	109.
TOTAL INTACT LOOP VOLUME	0.09077	0.10358	14.1
3.0 Broken loop			
3.1 Hot leg	0.00131	0.00255	94.
3.2 Steam generator	0.01788	0.01756	-1.8
3.3 Pump suction leg	0.00209	0.00558	167.
3.4 Pump	0.00133	0.00133	0.
3.5 Cold leg	0.00141	0.00164	16.
TOTAL BROKEN LOOP VOLUME	0.02402	0.02866	19.3
TOTAL SYSTEM LIQUID VOLUME	0.18911	0.20773	9.8

(1) The reference plant for the operating and broken loop is Trojan. The vessel reference plant is a Westinghouse PWR with upper head injection system.

(2) The scaled reference system core bypass volume has been included in the Mod-3 downcomer volume.

(3) The total pressurizer and surge line volume is 0.034 m^3, total liquid volume is 0.014 m^3.

1. PRESSURE VESSEL

The Semiscale Mod-3 pressure vessel is comprised of various components. To allow a basic understanding of the general configuration of the pressure vessel, a brief description is first given, followed by a detailed discussion of each component.

The Semiscale Mod-3 vessel is a multisection pressure vessel, which consists of an upper head, upper plenum, heated core region, and lower plenum with an external inlet annulus and downcomer pipe attached. The general arrangement of the pressure vessel, together with the external downcomer, is shown in Figure 3. The upper head region is contained within approximately the top 25% of the pressure vessel. Internal to the upper head region are ports for upper head ECC injection, a filler piece to provide the proper upper head internal volume, an insulator designed to provide a 0.127-cm steam gap between the filler inside diameter and the insulator outside diameter, and a simulated control rod guide tube. An upper core support plate simulator forms the boundary between the upper head and upper plenum regions. This upper core support plate provides support for the simulated guide tube and for the upper ends of the two simulated core support columns, which extend down through the upper plenum region.

The upper plenum region extends from the upper core support plate to the top of the heated core region and is approximately 2.5 m long. Two hot leg nozzles extend from the vessel upper plenum to provide connections for the intact and broken loop hot leg piping. The volume

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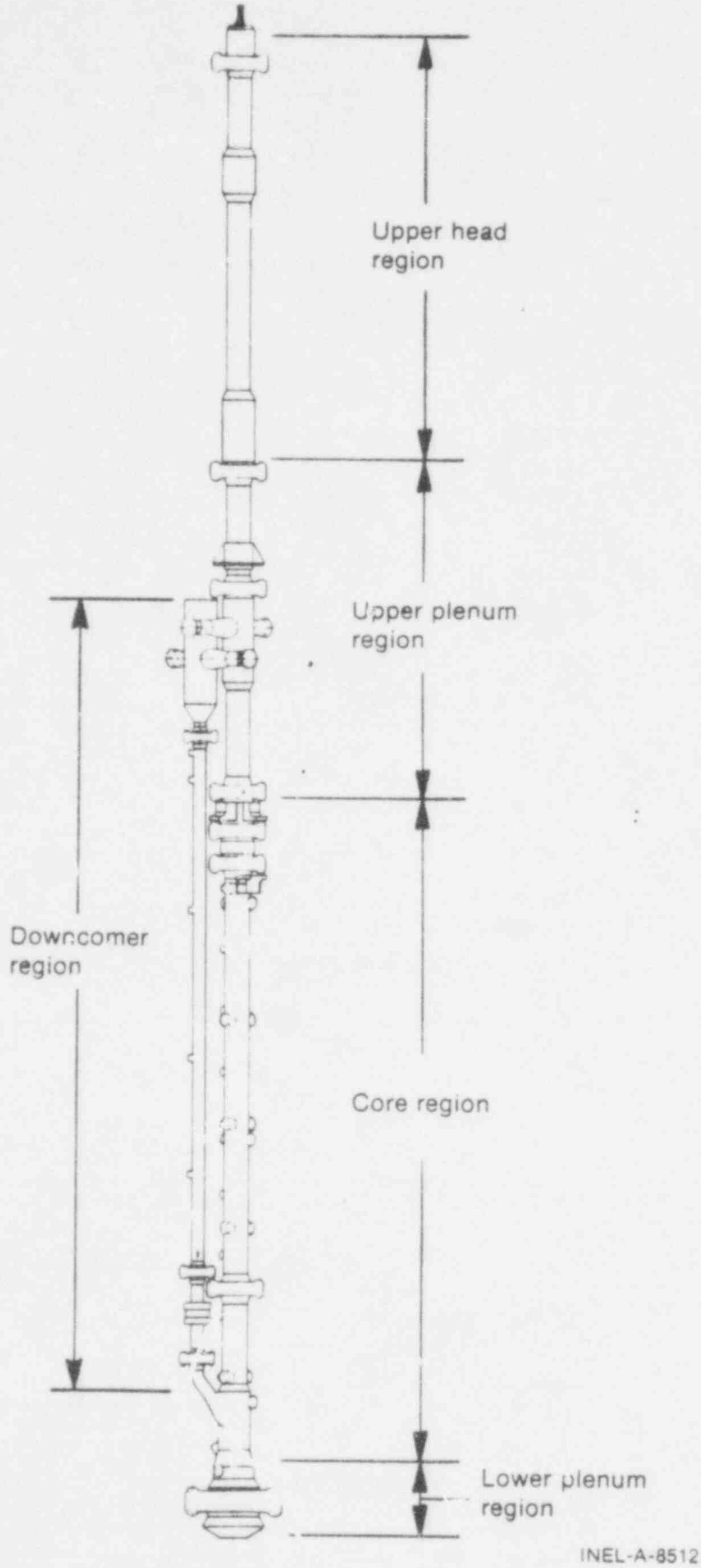

Fig. 3 Major regions of the Semiscale Mod-3 pressure vessel.

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within the upper plenum is divided into upper and lower sections by a core flow measurement station located at approximately the cold leg inlet elevation[a]. The upper and lower sections of the upper plenum contain fillers and insulators similar to those in the upper head. A flow restrictor assembly is located between the two hot leg nozzles which simulates the flow restriction in a PWR caused by control rod guide tubes and core support columns. The simulated control rod guide tube and core support columns extend from the entrance in the upper head through the upper plenum and terminate (open-ended) in the upper core plate located in the heater ground hub which forms the boundary between the upper plenum and the top of the electrically heated core region.

The electrically heated core consists of 23 powered heater rods, one unpowered rod, and one rod location reserved for a liquid level probe. The heater rods, 1.07 cm in diameter, are positioned and held in the core with 10 grid spacers, which maintain the heater rods on a typical PWR pitch (1.43 cm). The nine center rods can be powered independently of the remaining peripheral rods to simulate radial power peaking of rods within a PWR. The liquid level probe and unpowered rod are located at corner locations in the core bundle. The 3.7-m heated length of the heater rods extends from the heater rod ground hub, which provides support for the rods, to the top of the flow mixer box, which separates the core and lower plenum regions and is located approximately 496 cm below the cold leg centerline.

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a. The centerline of the cold leg inlet nozzle is the zero reference elevation point (refer to Table I).

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The lower plenum consists of an annular region between the flow mixer box and the pressure vessel, which serves to distribute flow from the downcomer pipe around the vessel periphery, and a chamber region below the mixer box which approximates the scaled volume of a PWR lower plenum. An insulator is provided inside the lower plenum chamber section to maintain a steam gap between the outer vessel wall and the fluid in the lower plenum. The bottom head serves as the lower section of the lower plenum chamber and provides penetration for the 24 heater rods and the core liquid level probe.

The external downcomer consists of an inlet annulus assembly, a downcomer pipe, and instrumented spool piece. The three sections are joined together and connected to the downcomer nozzle, extending from the lower plenum region at the lower end of the pressure vessel, by Grayloc seal rings and clamps. The total length of the downcomer assembly is approximately 5.5 m.

The inlet annulus assembly contains the cold leg nozzles and is designed to provide an annular inlet geometry similar to that of a PWR. Both surfaces of the inlet annulus are provided with insulators that maintain a steam gap to isolate the fluid from the hot walls of the assembly. The lower end of the inlet annulus contains a transition section that funnels the flow into the downcomer pipe.

The downcomer pipe is fabricated from 3-in. Schedule 160 pipe, and thermal insulation is provided on the inner surface to isolate the fluid from the hot walls. The instrumented spool piece provides the

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connection between the lower end of the downcomer pipe and the downcomer nozzle. The downcomer assembly is secured to the pressure vessel at approximately the hot leg elevation by a connection arrangement which allows differential thermal expansion between the downcomer and the vessel.

The overall design of the pressure vessel was based on applying scaling principles described earlier and on practical limitations relating to hardware and structural requirements and constraints. Individual regions of the vessel also conform to the scaling principles established for the Semiscale Mod-3 system design, however, in some instances scaling compromises were necessary. The following sections discuss scaling influences on the expected thermal-hydraulic behavior of individual regions in the pressure vessel, beginning with the upper head region.

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1.1 Upper head

The Semiscale Mod-3 upper head, shown in Figure 4, is scaled from a typical PWR with UHI. During a LOCA, the upper head receives injected ECC, which is then delivered to the upper plenum and the top of the heated core through the simulated control rod guide tube and core support columns. For a UHI plant, the volume of the upper head is increased by lowering the core support plate and consequently decreasing the volume of the upper plenum. However, the combined total volume of both upper head and upper plenum have remained almost unchanged from that of a non-UHI PWR. The enlarged volume of the upper head is maintained at the cold leg temperature by allowing about 4% of the total primary fluid flow to be bypassed from the upper annulus into the upper head by way of a series of spray nozzles.

In considering the UHI concept, three distinct periods of fluid delivery to the heated core can be identified. These three periods are the injection period, the reheat period, and the drain period. During the injection period, accumulator water is injected into the upper head starting at a relatively high system pressure (about 8.27 to 9.64 MPa) and continues until a specified volume of water has been injected. Following the injection period, the subcooled upper head fluid approaches saturation through a combination of wall heat transfer, condensation of steam flowing up the guide tube, and system depressurization. This period is termed the reheat period. The condensed steam coupled with the system depressurization requires the displacement of a small amount of fluid from the upper head which

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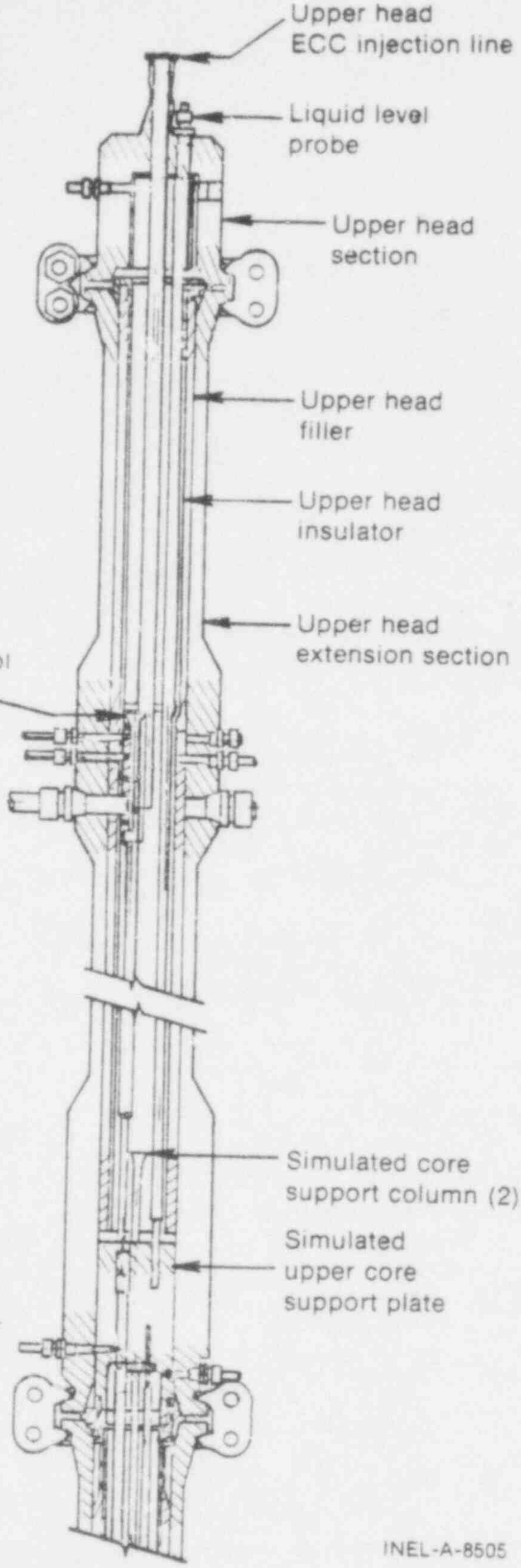

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Fig. 4 Semisca e Mod-3 pressure vessel upper head region.

flows through the support columns and into the region directly above the heated core. Once the upper head fluid reaches saturation, flashing occurs and rapid draining of the fluid out of the upper head begins. At first there is a large amount of fluid flowing through the guide tube and support columns due to flashing; however, as the guide tube uncovers and vents the steam to the break, the drain rate reduces to a gravity flow through the support columns. This draining condition continues until the tops of the support columns, located near the bottom of the upper head, have uncovered.

To insure proper simulation of the upper head thermal-hydraulic behavior, primary considerations of volume, elevation, and pressure loss across the component were scaled from PWR values. The volume of the Semiscale Mod-3 upper head is 7% higher than that in a PWR, with the elevations of the upper head internals at full scale and pressure loss across the upper head of Semiscale Mod-3 being scaled directly from PWR information. However, scaling of these considerations have resulted in compromises which include structural energy transfer, relative component elevations, flow resistance distribution, flow areas, and fluid conditions. Each of these compromises, which could cause distortion in the Semiscale Mod-3 upper head behavior, is discussed in the following paragraphs.

In subscaled systems where maintaining full-scale elevations is important, the surface area-to-volume ratio is always larger than the reference system, since this ratio varies as the inverse of the diamaters. In the case of the Semiscale Mod-3 upper head, the surface

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area-to-volume ratio (without internals) is approximately 25 times greater than that in a Westinghouse plant with UHI capability. This larger surface area results in more energy per unit volume being transferred from the upper head structures to the upper head fluid and can cause increased fluid temperatures and early flashing of the fluid in the upper head. To reduce the effects of metal heat transfer in the Semiscale Mod-3 upper head, an insulator and an insulating steam gap separate the fluid from the pressure vessel walls of the upper head.

To assess the effectiveness of the Semiscale Mod-3 insulator in reducing structural heat transfer to the fluid in the upper head, an analysis of the structural heating in the upper head of a PWR was performed and compared with similar calculations for the Semiscale Mod-3 system with and without the upper head insulator. The calculations were performed using a one-dimensional transient conduction code and the upper head fluid temperature response from Semiscale Mod-3 baseline Test S-07-1[4] as a boundary condition. Since the same fluid temperature boundary conditions were used in each calculation, comparison of the results provided a direct indication of the relative effectiveness of the Semiscale Mod-3 system insulation, although Test S-07-1 did not include UHI. To isolate the effects of the pressure vessel wall, no internal structures (guide tube and support columns) were included in any of the calculations. A very high heat transfer coefficient was placed on the fluid side of the upper head to simulate a conduction limited environment, which was

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determined to be a worst case situtation, and a very low heat transfer coefficient was placed on the outside of the upper head to represent the outside insulation.

The results of the upper head structural heating analysis are presented in Figure 5, which compares the predicted structural heat transfer rate per unit volume in the upper head of a PWR with that for the Semiscale Mod-3 system with and without upper head insulation. The comparion of Semiscale Mod-3 results shows that from 8 to 26 s after rupture, when flashing and fluid draining in the upper head occurred, the insulation in the upper head reduced the heat transfer rate to the upper head fluid by a factor of two or three below that without insulation. However, even with the insulator, Figure 5 shows structural heat transfer in the Semiscale Mod-3 upper head was still approximately five times higher than that calculated for a PWR.

While the calculations presented herein were for a non-UHI test and therefore not directly applicable to the evaluation of upper head behavior with UHI, the results do indicate the potential for greater upper head structural heat transfer in the Semiscale Mod-3 system than would be expected to occur in a PWR. The effect of structural heat transfer on upper head fluid conditions is addressed later in this section when the overall upper head fluid injection and drain characteristics are evaluated.

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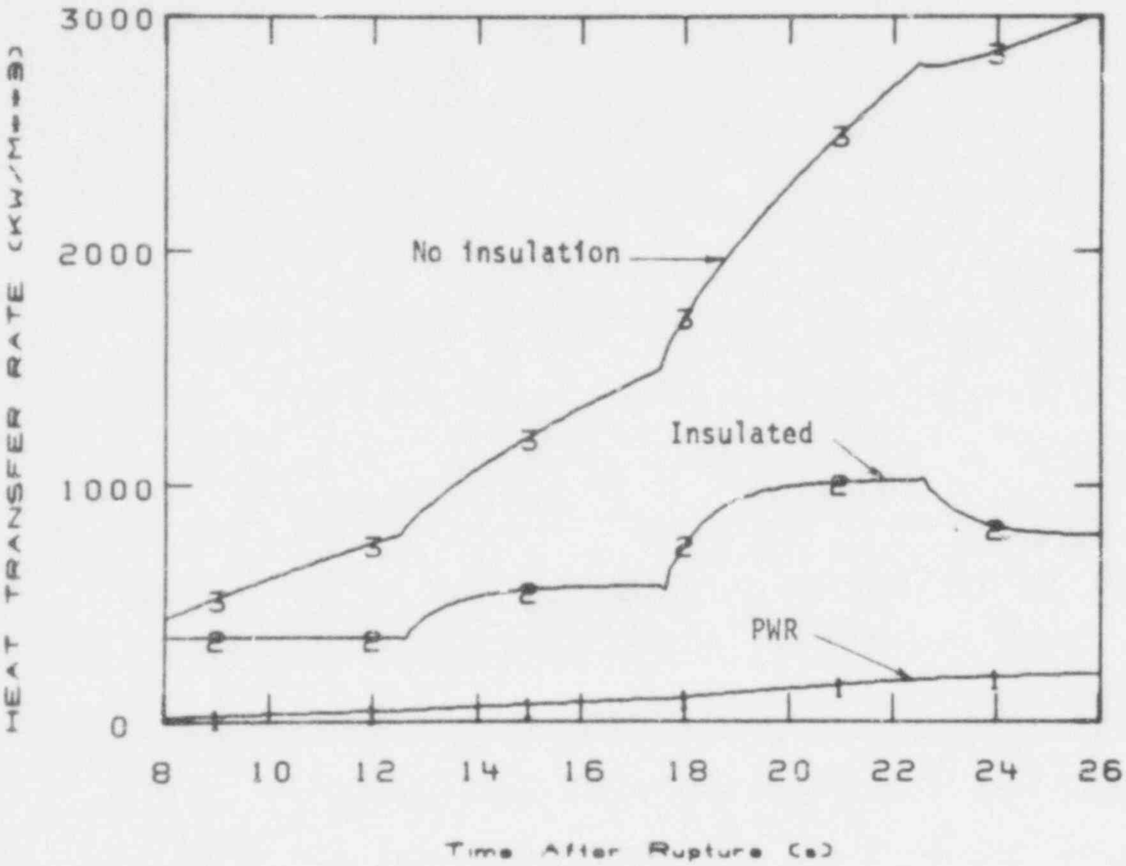

Fig. 5 Heat transfer rate curves from the upper head walls for Semiscale Mod-3 and a PWR.

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In addition to structural heating in the Semiscale Mod-3 upper head, elevation effects are also considered important, particularly with respect to the location of the guide tube and support column inlets. The Semiscale Mod-3 upper head was volume scaled with the height of the upper head being the average height of the hemispherical portion of a PWR upper head. This results in the Semiscale Mod-3 upper head height being 36.7 cm shorter than the maximum height of the PWR upper head. However, the distances from the top of the heated core to the top of the guide tube and support columns have been maintained identical to the UHI PWR. Thus, the elevation between guide tube, support columns, and heated core should provide similar conditions for static hydraulic behavior in the Semiscale Mod-3 and PWR upper heads.

The desire to volume scale the Semiscale Mod-3 upper head region while maintaining full-length elevation effects has resulted in a tall slender upper head design with a distorted length-to-diameter (L/D) ratio relative to that in a PWR. The L/D ratio in the Semiscale Mod-3 upper head is approximately 31, which is 44 times greater than the L/D ratio in a PWR. Because of this distortion in the L/D ratio, there exists the potential for different mixing characteristics and significant temperature stratification in the Semiscale Mod-3 upper head region, particularly during the ECC injection period. Since thermal stratification in the Semiscale Mod-3 upper head was recognized as a potential problem during the Mod-3 design phase, provisions have been included to allow ECC injection at different elevations in the upper head so that the influence of fluid mixing and

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temperature stratification on the upper head behavior can be evaluated. In addition, a perforated injection tube for injecting fluid uniformly over the full length of the upper head can be installed in the Semiscale Mod-3 system, permiting the evaluation of an injection condition approaching that of perfect mixing. Varying the injection location and configuration to evaluate the effects of thermal stratification of the fluid in the upper head is one of the objectives of the Upper Head Injection Test Series (Test Series 8), which should provide useful information relating to the potential effects of thermal stratification in a full-sized PWR as well as in the Semiscale Mod-3 system.

To evaluate the combined influence of upper head structural heating, volume scaling, and full-length elevation modeling, a study utilizing the RELAP4 computer code[5] to investigate upper head drain characteristics was performed. A RELAP4/MOD6[a] model of the computer code was used in this study which included a three volume upper head model with equal ECC injection into each of the volumes (simulating complete mixing with the perforated UHI tube). To bound the potential effects of upper head structural heating, two limiting structural heat transfer conditions were examined: one with water filling the gap between the insulator and the pressure vessel wall which represents the maximum potential for heat addition to the fluid in the upper head, and the second with all heat slabs removed from the upper head which would represent a perfect insulator. The calculations, which

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a. RELAP4/MOD6, Update 4, Idaho National Engineering Laboratory Configuration Control Number C0010006.

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extended through the injection and reheat periods, defined the potential upper head hydraulic behavior and provided an indication of the effect of structural heating on upper head drain characteristics. Figures 6 and 7 show the calculated mass flows through the support columns and guide tube respectively. Also indicated on each plot are the duration of the injection, reheat, and drain periods, which indicate structural heating in the Semiscale Mod-3 upper head does not have a significant effect on the upper head fluid behavior for the two limiting cases investigated.

To provide an indication of the validity of the RELAP4/MOD6 calculated upper head drain characteristics, the RELAP4/MOD6 calculations for Semiscale Mod-3 were compared with a Westinghouse prediction of the upper head drain characteristics for a full sized PWR[6]. Figure 8 shows a comparison of the calculated flow in the PWR support column with that calculated for the Semiscale Mod-3 support column with and without structural heating. So that a direct comparison of the results could be made, the Westinghouse flow values were first multiplied by the ratio of the core powers (1/1705) which is equivalent to volume scaling the results. The comparisons in Figure 8 show very good agreement in both the duration of the reheat period and the magnitude of the flows. In general, the RELAP4/MOD6 calculations for Semiscale Mod-3 indicated shorter reheat periods and slightly larger flows than those predicted by Westinghouse for a PWR. However, even with the maximum structural heating assumed, the reheat period in Semiscale Mod-3 was only about 3 s shorter than that calculated for a PWR.

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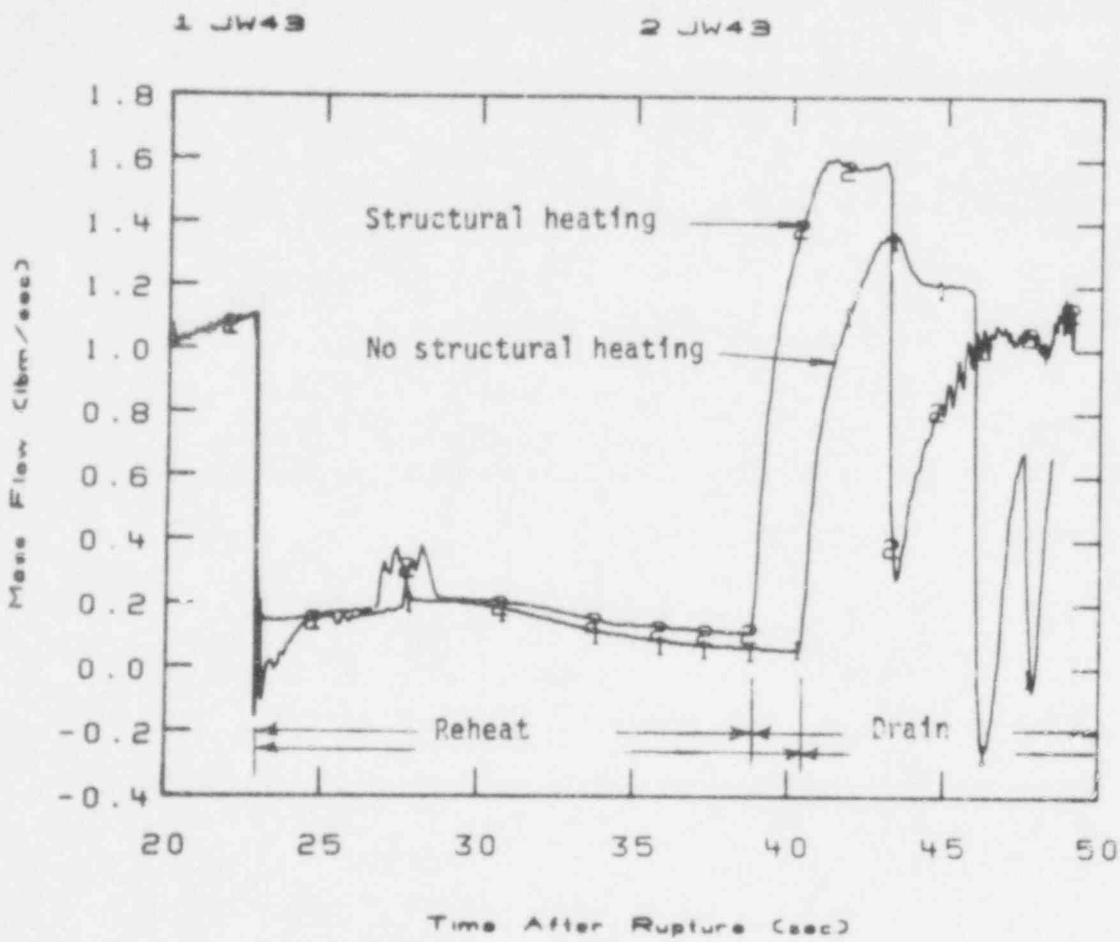

Fig. 6 RELAP4 calculated mass flow through the support column.

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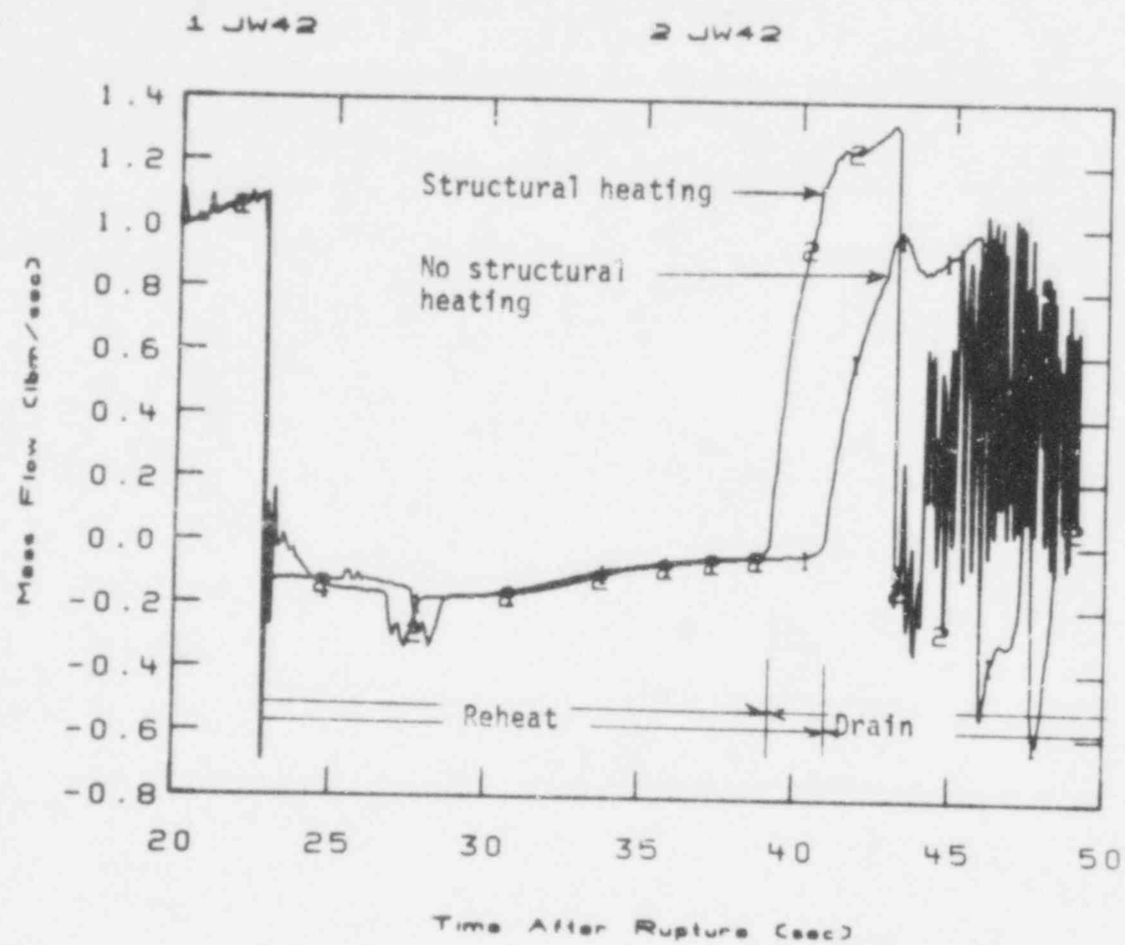

Fig. 7 RELAP4 calculated mass fow through the guide tube.

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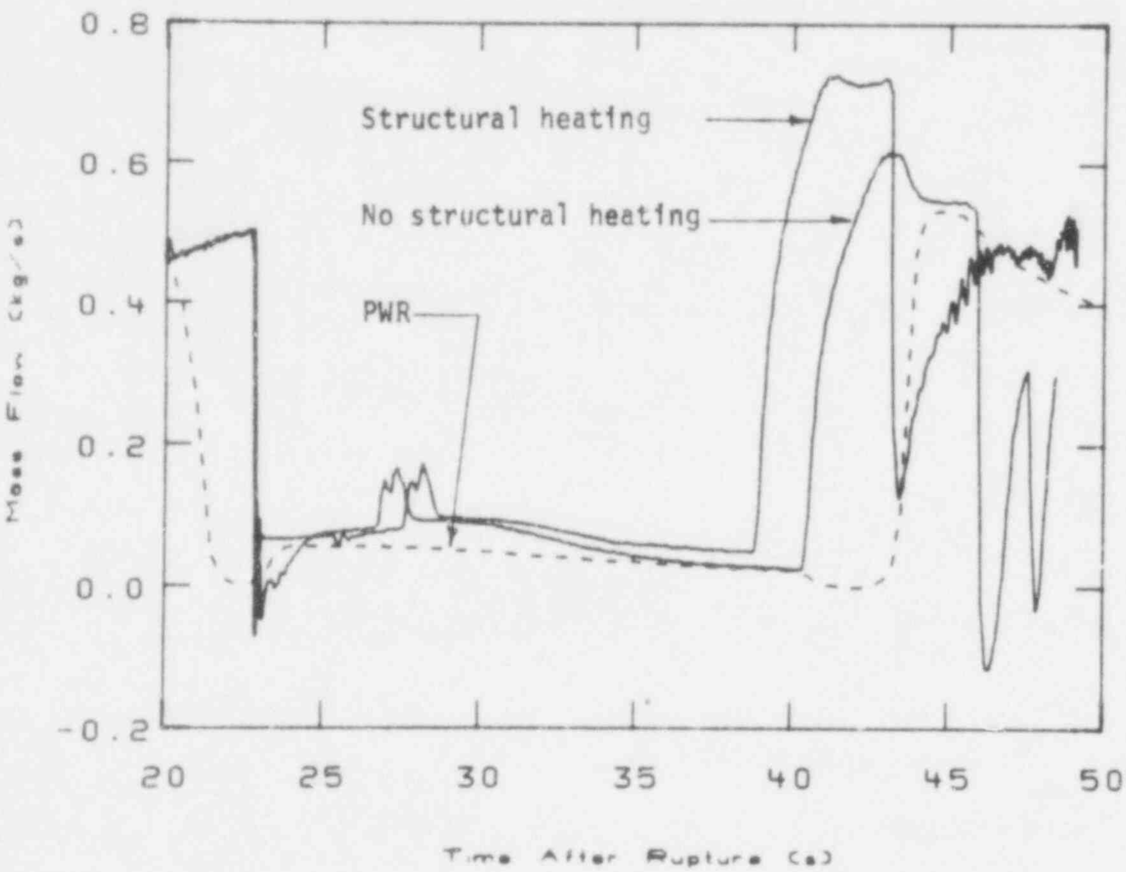

Fig. 8 RELAP4 support tube mass flow and PWR support tube mass flow.

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Although the RELAP4/MOD6 calculations were not carried out beyond the start of upper head draining, an estimate of the upper head drain time in Semiscale Mod-3 was made by utilizing results from the Westinghouse calculations. This was accomplished by dividing the scaled value of the average support column flow rate predicted by Westinghouse into the calculated volume of liquid remaining in the Semiscale Mod-3 upper head at the end of the reheat period. The results of this calculation indicate that the total upper head drain period in Semiscale Mod-3 is approximately 32 s, compared with the predicted PWR value of 35 s.

A final area of concern in the upper head design was the potential effects of two-phase flow on the hydraulic resistance of the Semiscale Mod-3 guide tube. Since scaled flow areas were not maintained for the full length of the Semiscale Mod-3 guide tube and support columns, the tube sizes and orificing were specified so that the total single-phase frictional and local losses would equal those in a PWR. However, this resulted in larger friction losses in the Semiscale Mod-3 guide tube and support columns than would occur in a PWR. Since frictional losses are more sensitive to two-phase flow effects than are local or form losses, there was a concern that two-phase flow up the guide tube during the reheat period would produce upper head hydraulic behavior different than that expected in a PWR.

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To assess hydraulic behavior in the upper head, the effects of two-phase flow on the hydraulic resistance of the guide tubes for both Semiscale Mod-3 and a PWR were calculated using fluid conditions from the RELAP4/MOD6 calculations discussed earlier. The calculations were made using the Baroczy correlations for friction losses[7] and the Chisholm correlation for orifice losses[8]. The results indicated that because of the higher guide tube frictional losses in Semiscale Mod-3 the total guide tube resistance could be as much as 40% higher than that calculated for a PWR over a short portion of the reheat period when a low quality fluid was present in the guide tube. However, this period of low quality flow only occurred during the last 6 to 7 s of the reheat period and therefore is expected to have little effect on the overall fluid behavior during the reheat or subsequent drain periods.

In summary, major concerns in the design of the upper head such as structural heating effects, fluid temperature stratification and fluid mixing, as well as the adequacy of the scaling approaches used have been addressed. The conclusion reached is that the design of the Semiscale Mod-3 upper head is acceptable in terms of meeting overall test objectives. While the Semiscale Mod-3 upper head thermal-hydraulic behavior may not entirely duplicate that expected in a PWR, RELAP4/MOD6 comparisons indicate that important upper head drain characteristics agree reasonably well in both timing and magnitude with those predicted for a PWR.

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1.2 Upper Plemum

The Semiscale Mod-3 upper plenum connects the upper head region to the heated core. It houses the simulated control rod guide tube and core support columns, through which upper head fluid flows into the heated core as well as into the upper plenum. The internal design of the Semiscale Mod-3 upper plenum was limited both by conflicting scaling requirements and the necessity to obtain an accurate measurement of the flow exiting the core. However, despite these design limitations, major design features such as the desired scaled volume, elevation effects, and hydraulic resistance characteristics have been preserved. The following paragraphs present a description of the important Semiscale Mod-3 upper plenum design features and discuss the important design characteristics in terms of their effects on overall system behavior.

The Semiscale Mod-3 upper plenum, shown in Figure 9, was volume and elevation scaled from a full-sized PWR. The elevation scaling was important to model full-length flow paths, hydrostatic effects, and internals which represent as closely as possible those in a PWR. However, because of design constraints and the need for flow measurements, a turbine meter and drag screen were installed in the middle of the upper plenum. The guide tube and support columns leaving the upper head pass through the upper plenum, terminating at the top of the heated core in the end box. Slots in the guide tube have been provided to allow upper head fluid as well as core fluid to enter the upper plenum. To simulate the guide tube and support column

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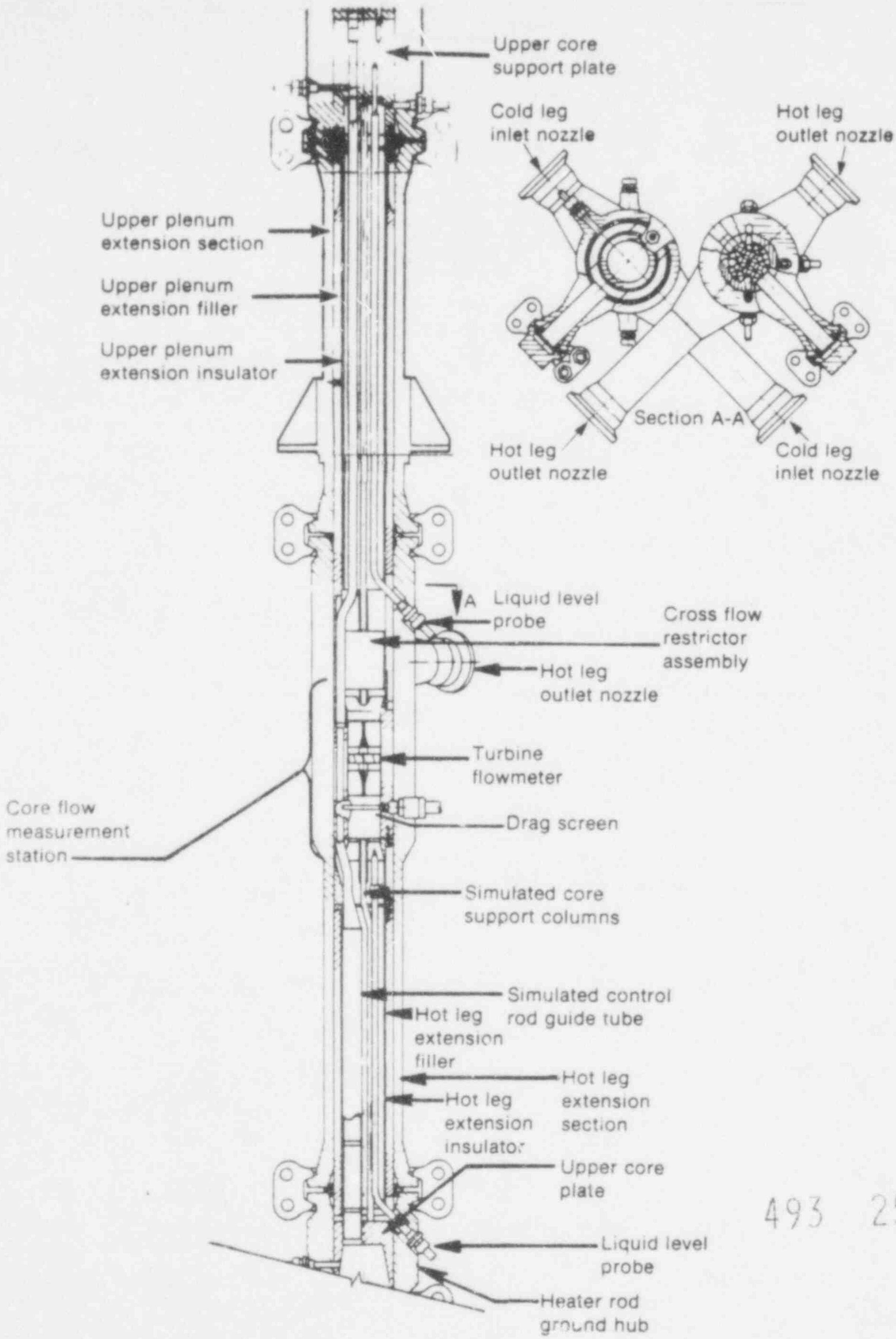

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Fig. 9 Semiscale Mod-3 pressure vessel upper plenum region.

resistance to cross-flow between the two hot legs in a PWR, a cross flow restrictor has been installed between the hot leg nozzles directly above the instrument station. A grounding hub located at the bottom of the upper plenum serves a twofold purpose in the Semiscale Mod-3 system by grounding the electrical heater rods to the vessel pressure boundary and by simulating the hydraulic resistance characteristics of the upper core support plate in a PWR. It is important that the upper plenum flow resistance be similar to that of a PWR during blowdown in order to provide adequate simulation of the UHI drain process; also, during reflood, the upper plenum flow resistance can influence deentrainment behavior.

The upper plenum was designed to preserve the desired volume scaling and full-length elevation effects. The volume is 7% larger than the scaled PWR value while its length is identical to that of a PWR. With this scaling, the average flow area of the upper plenum is very close to that of a PWR in most cases. However, as a result of selecting these primary factors for scaling, some secondary factors were compromised. The secondary factors include two- and three-dimensional flow effects and deentrainment characteristics. The effect of these compromises on the UHI phase of the blowdown is undetermined at the present time and will be further evaluated after Test Series 8 has been completed. Therefore, compromises in the upper plenum design are discussed primarily as they relate to the reflood portion of the blowdown.

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One of the scaling compromises was the sacrificing of the two and three-dimensional flow effects which are expected in a PWR. The one-dimensional nature of the slender Semiscale Mod-3 upper plenum design does not lend itself to an evaluation of the complex radial flow patterns expected in the upper plenum of a full sized PWR. The PWR flow patterns, as discussed in Reference 9, and depicted in Figure 10, are extremely complex and indicate that radial trajectories are not easily calculated or analyzed, even in relatively simple models which do not include internal structures. The degree of difficulty increases when the internal structures are added, thus modifying uniform flow paths into irregular geometric paths. The ability for the Semiscale Mod-3 system upper plenum to simulate this type of phenomena is extremely limited; however, it was determined during the design phase that axial flow length was the primary characteristic to be scaled and the radial flow behavior, therefore, had to be compromised.

Another result of the one-dimensional nature of the Semiscale Mod-3 upper plenum will be differences in the deentrainment characteristics. Deentrainment in the upper plenum is caused by gravity effects (fallback) and by impingement on internal structures. Since flow areas in the upper plenum have a great deal of influence on deentrainment and impingement characteristics, these flow areas have been calculated and are plotted as a function of elevation in Figure 11, which compares flow areas in the Semiscale Mod-3 upper plenum with the ideally scaled flow areas in a PWR. The complex flow

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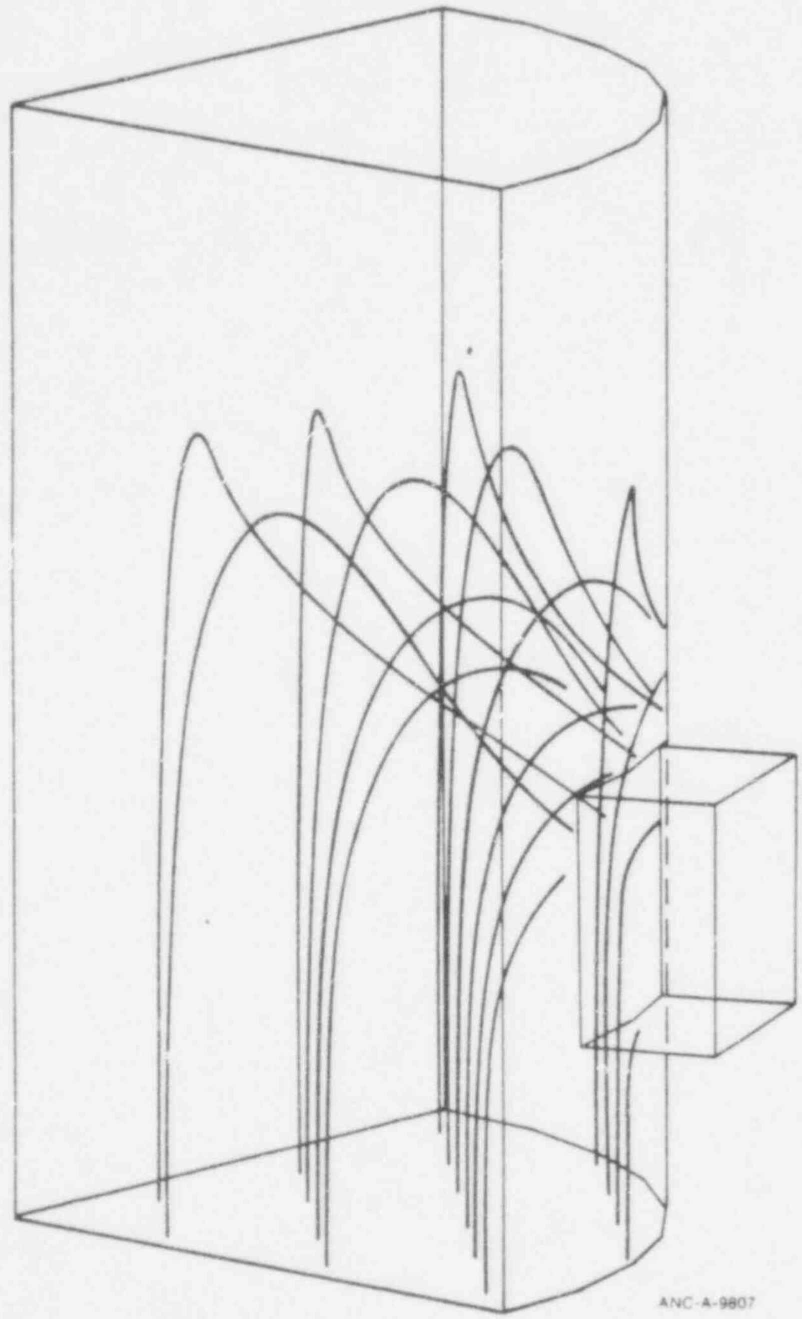

Fig. 10 Calculated droplet trajectories in the upper plenum of a PWR.

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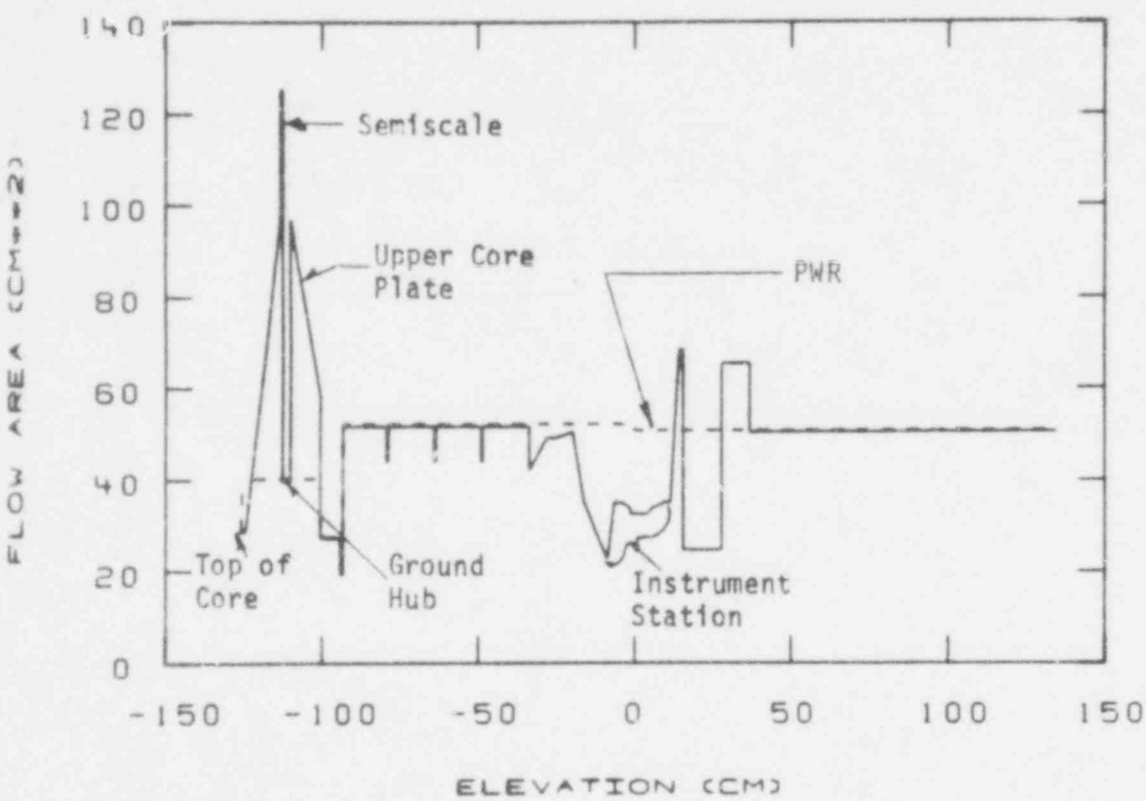

Fig. 11 Comparison of upper plenum flow areas as a function of elevation for Semiscale Mod-3 and a typical PWR.

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geometry in the upper plenum makes evaluation of the deentrainment mechanisms of fallback and impingement difficult; however, a few experimental reports have been published on upper plenum behavior, one of which used a small scale plexiglas model with air and water as the system fluids[10]. In this report the gravity mechanism was analyzed with the following observations. Irregularly shaped globules of water were propelled upwards in the upper plenum and generally fell back into an air-water froth above the heated core. This type behavior in the experiment was present above the upper core plate region where the flow area was larger. As shown in Figure 11, with the exception of the region immediately below the ground hub and the regions around the flow measurement station, the Semiscale Mod-3 upper plenum flow areas closely match the desired values scaled from the PWR. Therefore, it is expected that the potential for fallback in the Semiscale Mod-3 upper plenum will be similar to that expected in a PWR.

Impingement, on the other hand, occurs when water droplets strike a surface and either adhere to it or splatter into smaller droplets. These smaller droplets are then reentrained and become more subject to the existing flow velocities. Reference 9 indicates impingement is high for water droplets that travel radially from the center of the upper plenum to the hot leg nozzles. As mentioned previously, the Semiscale Mod-3 upper plenum does not have the radial distance of the PWR upper plenum and, therefore, deentrainment resulting from impingement in the radial direction will be very low. However, the regions of smaller flow area around the instrument station and hot leg flow restrictor in Semiscale Mod-3 will produce high velocities and increase the possibility for carryover and greater impingement in the regions of the upper plenum beyond the hot leg nozzles above that expected in a PWR. The differences in radial impingement characteristics and carryover are difficult to analyze, however, it is believed that radial impingement and carryover are of secondary importance and the overall system response will not be significantly influenced by these differences.

The effects on system response due to three-dimensional flow behavior and deetrainment characteristics are an important consideration during the reflood portion of the LOCA, and with the introduction of UHI, become important because of their potential influence on the blowdown response. However, the flow and deentrainment characteristics in upper plenums of either the Semiscale Mod-3 system or a PWR are extremely complex and very difficult to analyze or calculate, even with sophisticated computer models. Thus, the ability to accurately predict upper plenum flow and deentrainment characteristics for a PWR or Semiscale Mod-3 is seriously limited. However, an important aspect of the upper plenum design is that the internals of the Semiscale Mod-3 upper plenum were designed to simulate as closely as possi that of the reference PWR with only necessary compromises to allow monitoring of system behavior.

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1.3 Core

The Semiscale Mod-3 core configuration, shown in Figure 12, includes 23 powered heater rods, one unpowered rod, and one location containing a liquid level probe. To provide the best approximation of the thermal response characteristics of a nuclear rod, and to insure characteristic hydraulic behavior within the core representative of that in a PWR core, the rods are geometric full scale replicas of their PWR counterpart. Therefore, the resulting heated length (365.75 cm), pitch (1.43 cm), and outside diameter (1.07 cm) are identical to the nuclear fuel rods in a full-sized PWR core employing fuel bundles with a 15 x 15 fuel rod array. The core flow area and volume are scaled to maintain geometric, kinematic, and dynamic similarity with the PWR. As indicated in Table II, the total volume of the Semiscale Mod-3 core is within 2% of the desired value scaled from a PWR.

The scaling of the Semiscale Mod-3 core has resulted in compromises which affect the overall operation of the core. The use of electrical heater rods in Semiscale Mod-3 has required that the power to the electrical rods be controlled in such a manner that the electric rod surface temperature response will approximate the expected response of a nuclear rod. The electrical rods also restrict the axial power profile to one specific configuration, which approximates a middle of life chopped cosine profile in a PWR. Also compromised in the core, although not related to the electrical rods, is the potential of Semiscale Mod-3 to simulate the three-dimensional

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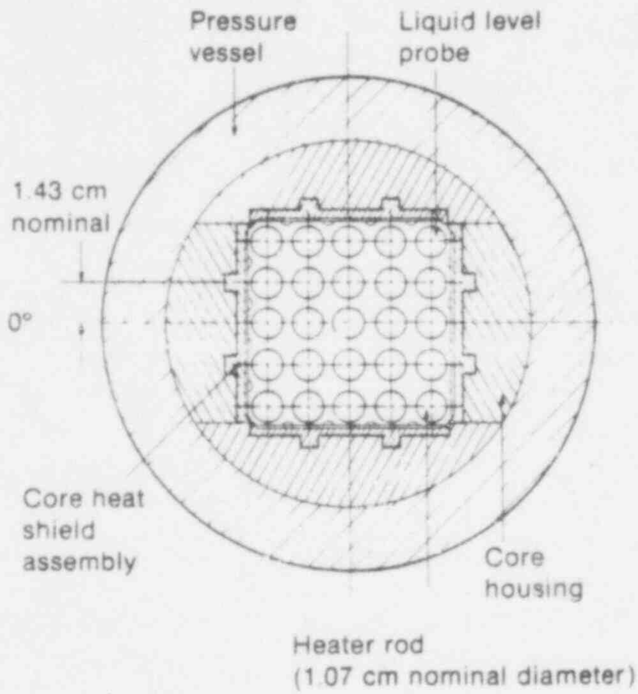

Fig. 12 Cross section of the Semiscale Mod-3 core region.

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flow distribution expected in a PWR. A discussion of these specific areas and the effect they have on the Semiscale Mod-3 system behavior is given in greater detail in the following paragraphs.

Despite the geometric similarity of the Semiscale Mod-3 electric heater rods to a PWR nuclear fuel rod, the electric rod thermal characteristics are not entirely representative of those of a nuclear fuel rod. The Semiscale Mod-3 electric heater rod construction, illustrated in Figure 13, uses boron nitride as an electrical insula about a helically wound constantan heater wire. The boron nitride, which represents about 2/3 of the total volume of the electrical heater rods, was selected because its thermal capacitance (ρC_p) closely approximates that of UO_2. However, because the boron nitride has a thermal conductivity appromately five times greater than that of UO_2, the steady state radial temperature gradient (and thus the total stored energy in the electric rod prior to the initiation of a LOCA experiment) is much less than that in a nuclear rod under the same conditions.

To make up for the difference in the stored energy in an electrical rod relative to that in a nuclear rod, the power input to the Semiscale Mod-3 electrical rods during the first few seconds of a blowdown experiment must, in general, be higher than the equivalent power from stored energy and decay heat calculated for a nuclear rod. The power input required to produce a surface temperature response for the electric rod representative of that in a typical PWR fuel pin can be obtained by one of two methods. In one case, a closed-loop control

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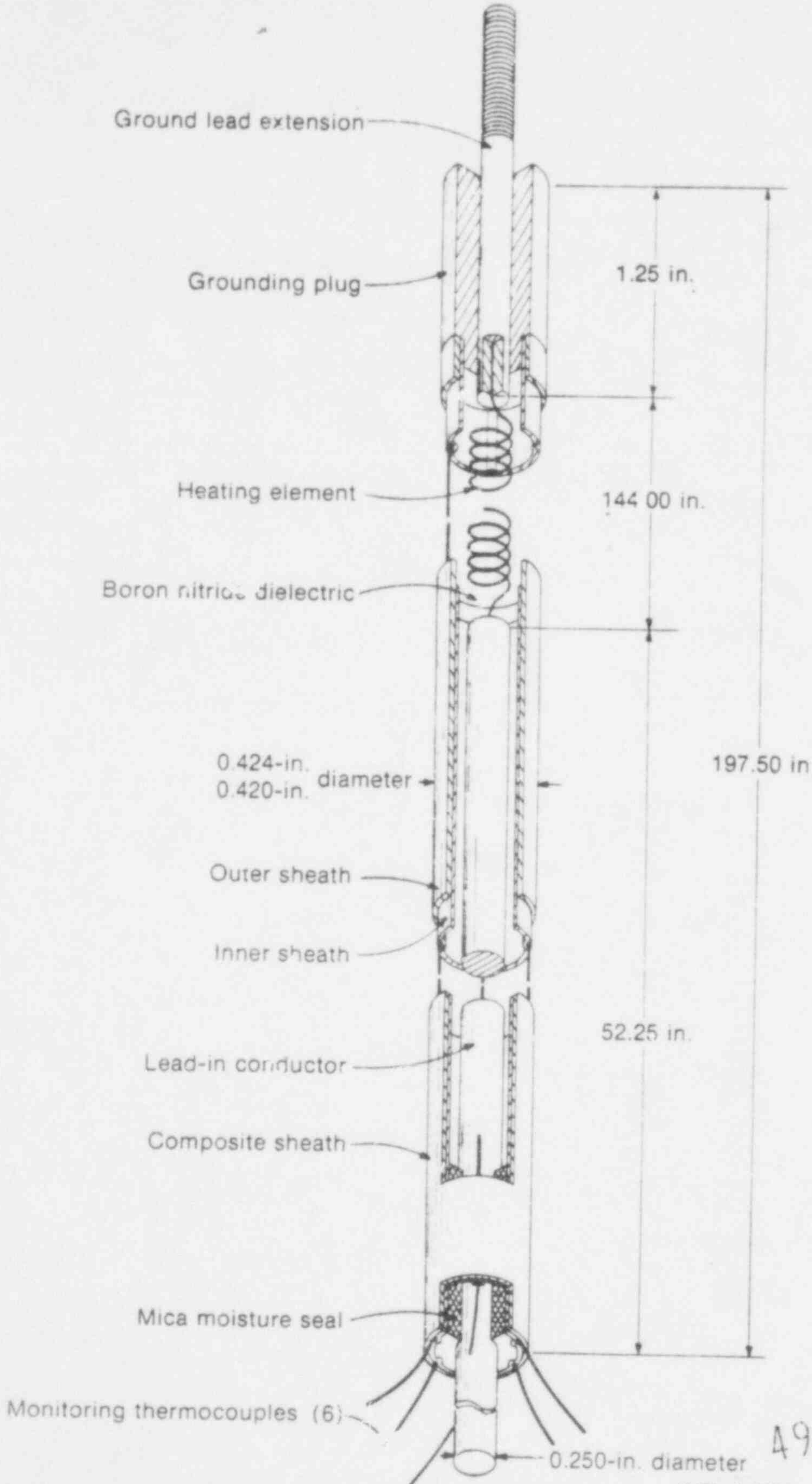

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Fig. 13 Mod-3 heater rod construction.

system can be used to calculate a real-time power decay based on conditions measured in the core during the test, and in the other case, a predetermined transient power input can be calculated based on calculated or experimental conditions believed to be representative of the actual test conditions. Each of these methods has certain advantages and disadvantages, which are described in the following discussion.

Under appropriate conditions, the use of the on-line power controller to calculate a real-time electrical rod power decay is believed to provide a more accurate representation of a nuclear rod in the actual test environment and will eliminate the uncertainty of defining the expected thermal-hydraulic conditions in the core prior to a given test. Closed-loop power control is advantageous because it eliminates the need for predetermining the core hydraulic behavior and then core electrical power prior to each Semiscale Mod-3 test. The controller used in Semiscale Mod-3, shown schematically in Figure 14 and discussed in detail in Appendix A, uses feedback control, which can regulate power such that the local surface heat flux of the electrical rod will match the surface heat flux of a hypothetical nuclear rod operating in the fluid environment present during a given Semiscale Mod-3 test. However, closed-loop power control is limited to applications where the thermal-hydraulic conditions over the length of the core remain relatively uniform. Since the on-line controller is designed to monitor the thermal response at a single axial location, large variations in the thermal-hydraulic conditions over

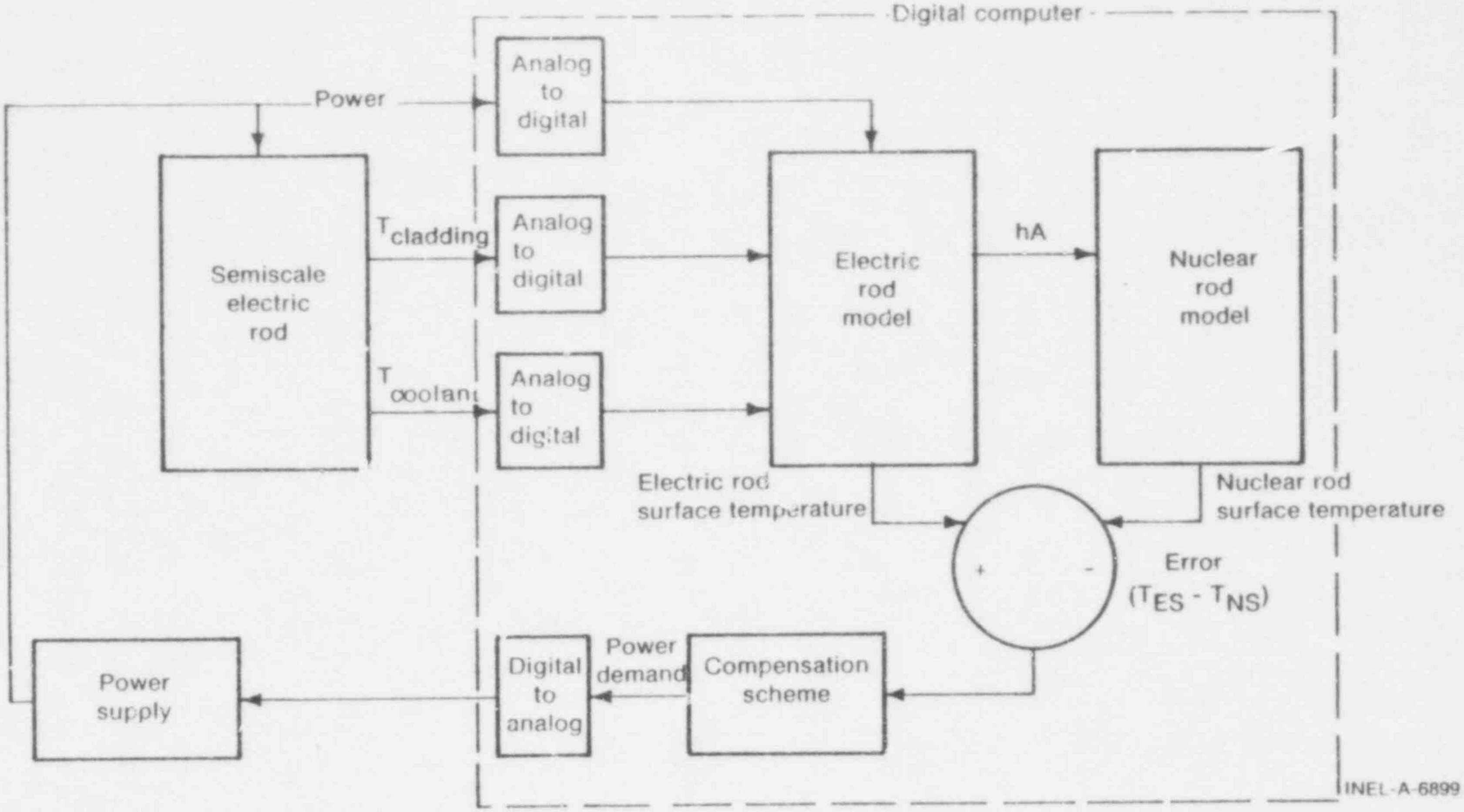

Fig. 14 Schematic of on-line core power control functions.

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the length of the core make it difficult to accurately represent a nuclear fuel rod by monitoring the temperature response at a single measurement location. For example, during the reflood period, the large variations in thermal-hydraulic conditions on either side of the core quench front as it progresses up the core would not be conducive to an accurate representation of the core power decay characteristics, particularly when the quench front propagated past the core loca being monitored by the controller. However, during reflood, the low constant core power and relatively flat radial temperature distribution in a nuclear fuel rod is similar to that in the electrical heater rod. Therefore, the closed-loop power controller is not adequate during reflood to calculate a real-time electrical rod power decay, and a predetermined power decay curve becomes more appropriate for the purpose.

The limitations on the use of the closed-loop power controller necessitates use of a predetermined power decay profile in specific instances. The criterion for selecting a predetermined electrical rod power profile is to approach as closely as practicable the surface temperature calculated for a nuclear rod. This criterion is met by matching the transient surface heat flux calculated for an electrical rod with the pretest 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 in all cases was the proposed standard power decay discussed in Reference 11. Since the Semiscale Mod-3 electrical

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heater rods have a fixed axial power profile, use of the technique described above to specify the predetermined core power control 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) is normally the point at which the nuclear and electrical fluxes are matched because the cladding temperature response at this location is of prime concern in most tests.

The ability of the Semiscale Mod-3 heater rods to accurately represent the temperature response of a nuclear rod by using either closed-loop power control or a predetermined electrical rod power decay curve was a major consideration in the evaluation of the Semiscale Mod-3 simulation of a LOCA. To calculate the nuclear rod temperature response, FRAP-T4,[12] which is a three-dimensional nuclear fuel rod code was used. FRAP-T4 includes the coupled effects of thermal, mechanical, internal gas pressure, and material properties in the analysis of fuel rod transient behavior, and the code calculates a variable gap conductance as a function of gap width. The results from the FRAP-T4 calculation of the nuclear fuel rod temperature response were used to evaluate the ability of the closed-loop power control and the predetermined power control to adequately represent the nuclear rod temperature response with the Semiscale Mod-3 electrical heater rods. The results of this evaluation are discussed in the following paragraphs.

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The closed-loop power controller employes a constant gap conductance (approximately 0.59 W/m·K) which, from past experience, provides a reasonable approximation of the variable gap conductance calculated by FRAP-T4. However, because the temperature response of a nuclear rod can be very sensitive to the gap conductance model, an attempt was made to evaluate the potential effect of the constant gap conductance assumed in the closed-loop controller model on calculated heater rod temperature response. To do this, a one-dimensional conduction code with a constant gap model (similar to that used in the power controller) was used to calculate the temperature response using core fluid conditions from Test S-07-1,[4] a non-UHI test performed in the Semiscale Mod-3 system. The temperature response was then compared with calculations using FRAP-T4 with a variable gap conductance. Figure 15 shows the constant gap calculations closely approximate the FRAP-T4 calculations. Figure 16 shows the constant gap calculations also provide a conservative and reasonable estimate of the peak core temperatures measured in Test S-07-1.

To evaluate the accuracy of the calculations used to generate a predetermined power decay curve for an electrical rod (discussed earlier), the electric rod temperature response for two different tests (Tests S-07-1 and S-07-6) in the Semiscale Mod-3 baseline test series (Test Series 7) were compared with the calculated nuclear rod temperature response from FRAP-T4. The calculated electrical rod power decay for the two tests, which had different core fluid conditions, are compared in Figure 17. The resulting measured temperature responses for Tests S-07-1 and S-07-6[13] are compared

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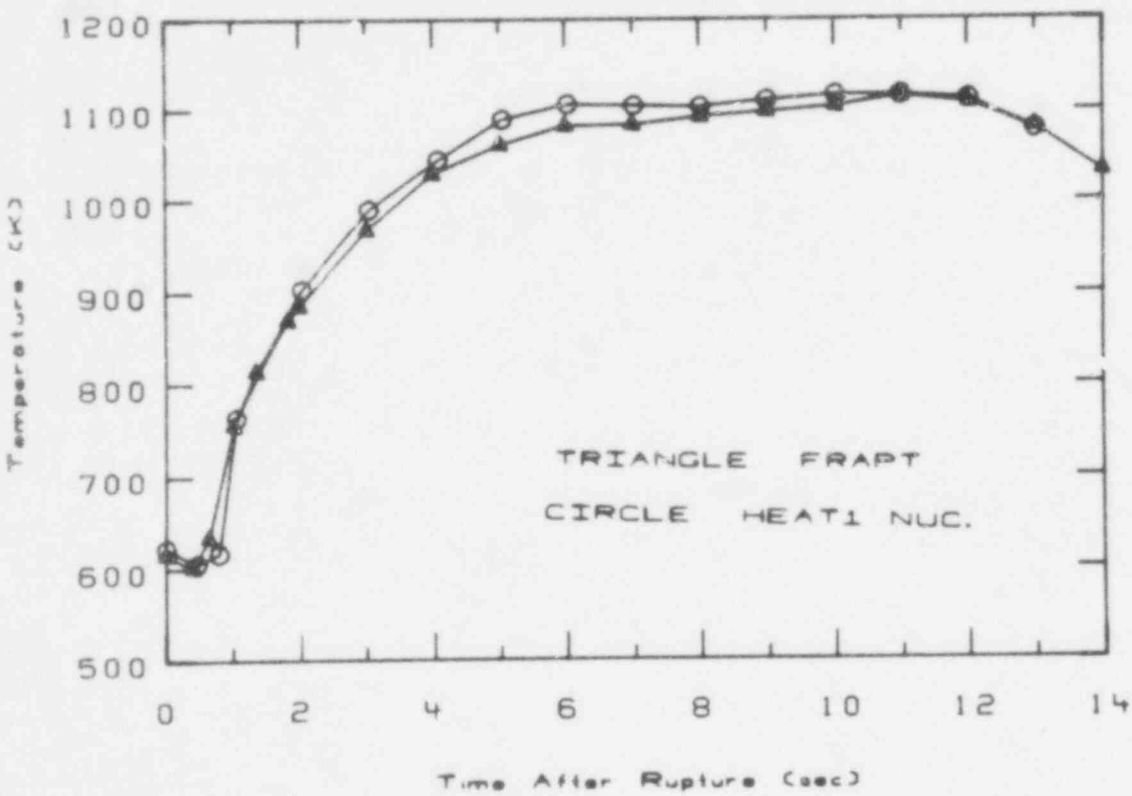

Fig. 15 FRAP-T4 calculated surface temperature versus constant one-dimensional surface temperature calculations.

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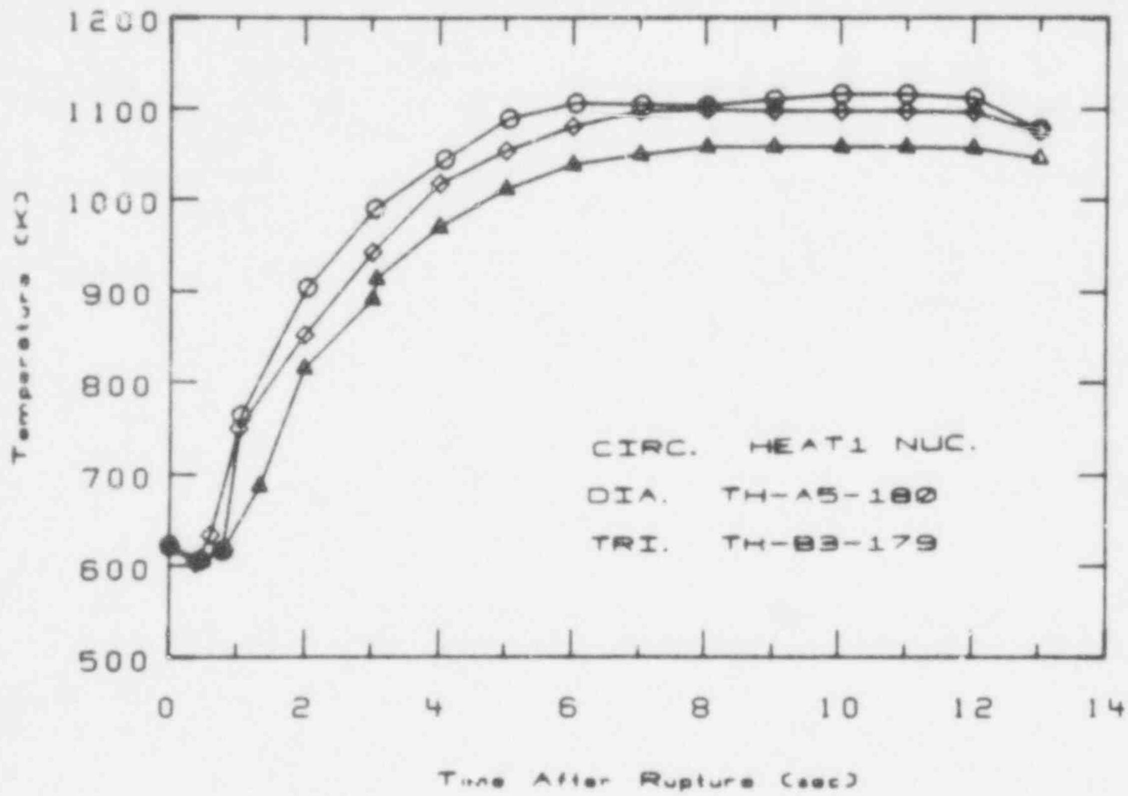

Fig. 16 One-dimensional calculated surface temperature versus S-07-1 data.

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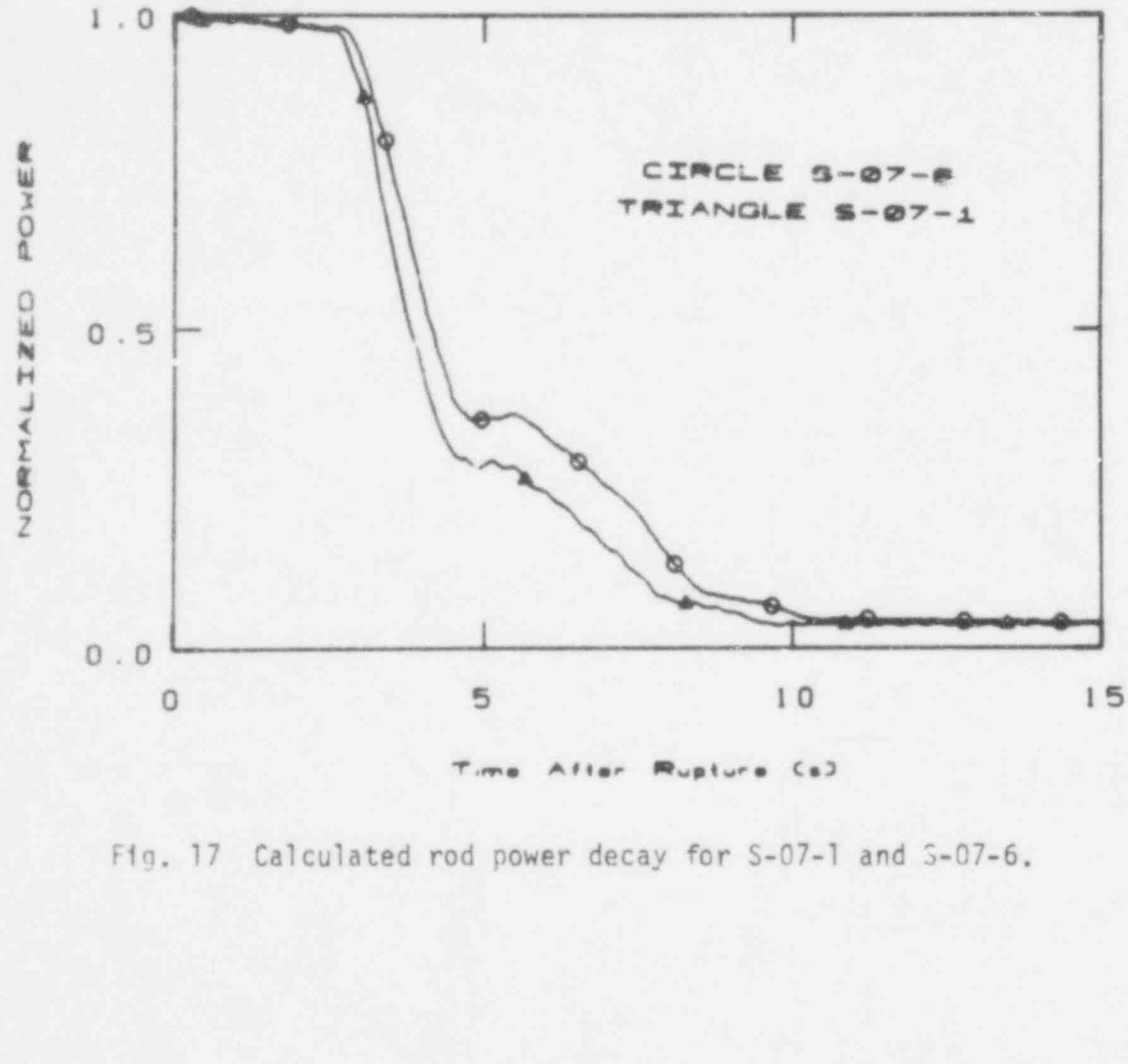

Fig. 17 Calculated rod power decay for S-07-1 and S-07-6.

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with the nuclear rod temperature response calculated by the FRAP-T4 code in Figures 18 and 19, respectively. The results show generally good agreement between the calculated nuclear rod temperature response and the measured electrical rod temperature response, indicating the capability of the one-dimensional conduction code to predict a power decay curve that will produce an electrical rod surface temperature response similar to that calculated for a nuclear fuel rod. Therefore, it can be expected that the surface temperatures of the Semiscale Mod-3 electrical rods should be reasonably representative of the cladding temperature response for PWR fuel rods.

The power peaking profiles of the Semiscale Mod-3 system are similar to a PWR if considered at a middle of life condition. At beginning of life the peak in the axial power profile of a PWR core would be skewed toward the bottom of the core while later in life the peak power would be located near the top of the core. In Semiscale Mod-3, the windings inside the core heater rods provide a fixed cosine axial power peaking profile representing the middle of life in a PWR core. The windings have a specified pitch for a designated length thus producing the profile shown in Figure 20. Semiscale attempts to incorporate the ratio of Q/Q_{avg} at a maximum value, with the axial peaked power value in Semiscale Mod-3 set at 1.55. The radial power profile in Mod-3 is variable, in that the nine center or high power rods can be powered higher than the 14 low powered rods; thus variable radial power peakings are available.

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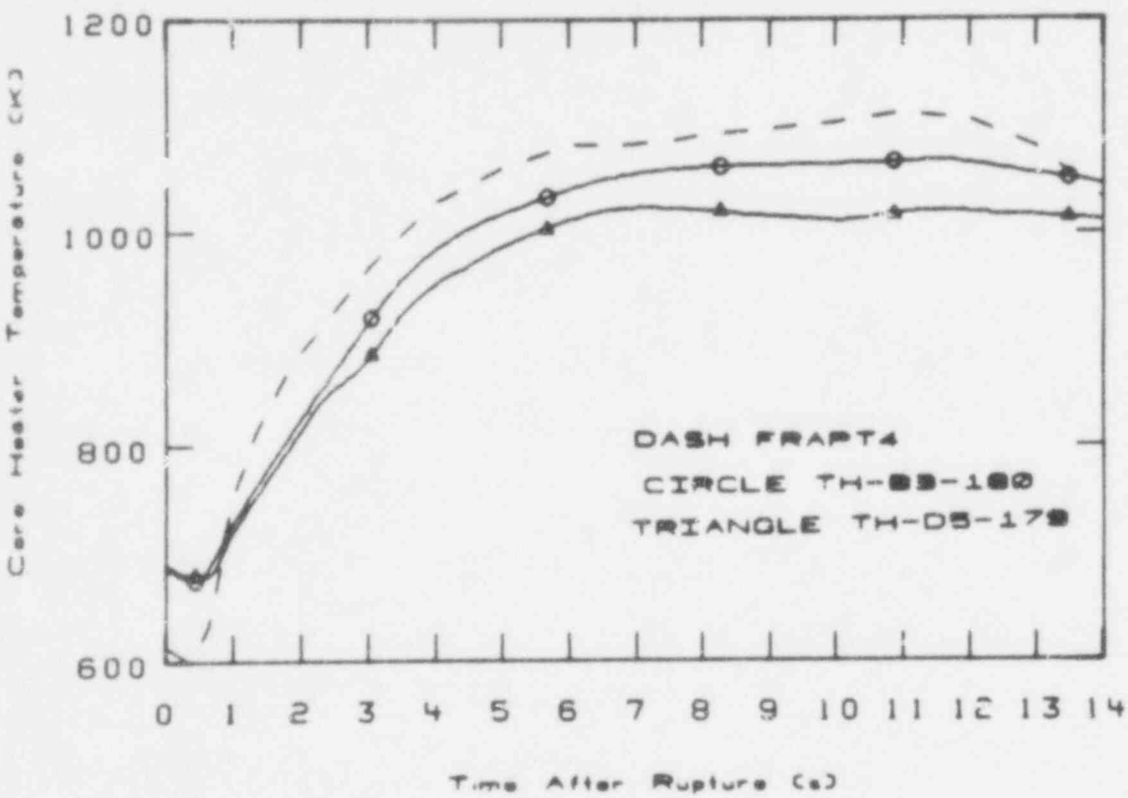

Fig. 18 FRAP-T4 surface temperature calculations versus S-07-1 data.

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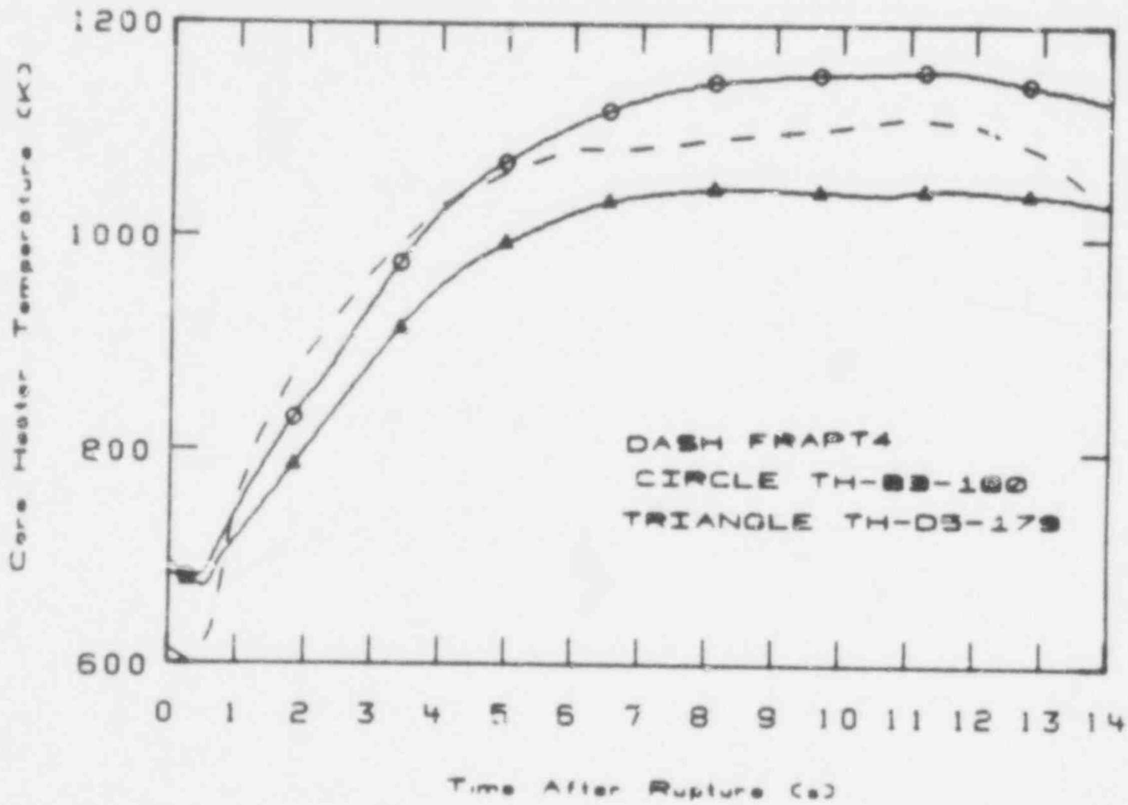

Fig. 19 FRAP-T4 calculated surface temperature versus S-07-6 data.

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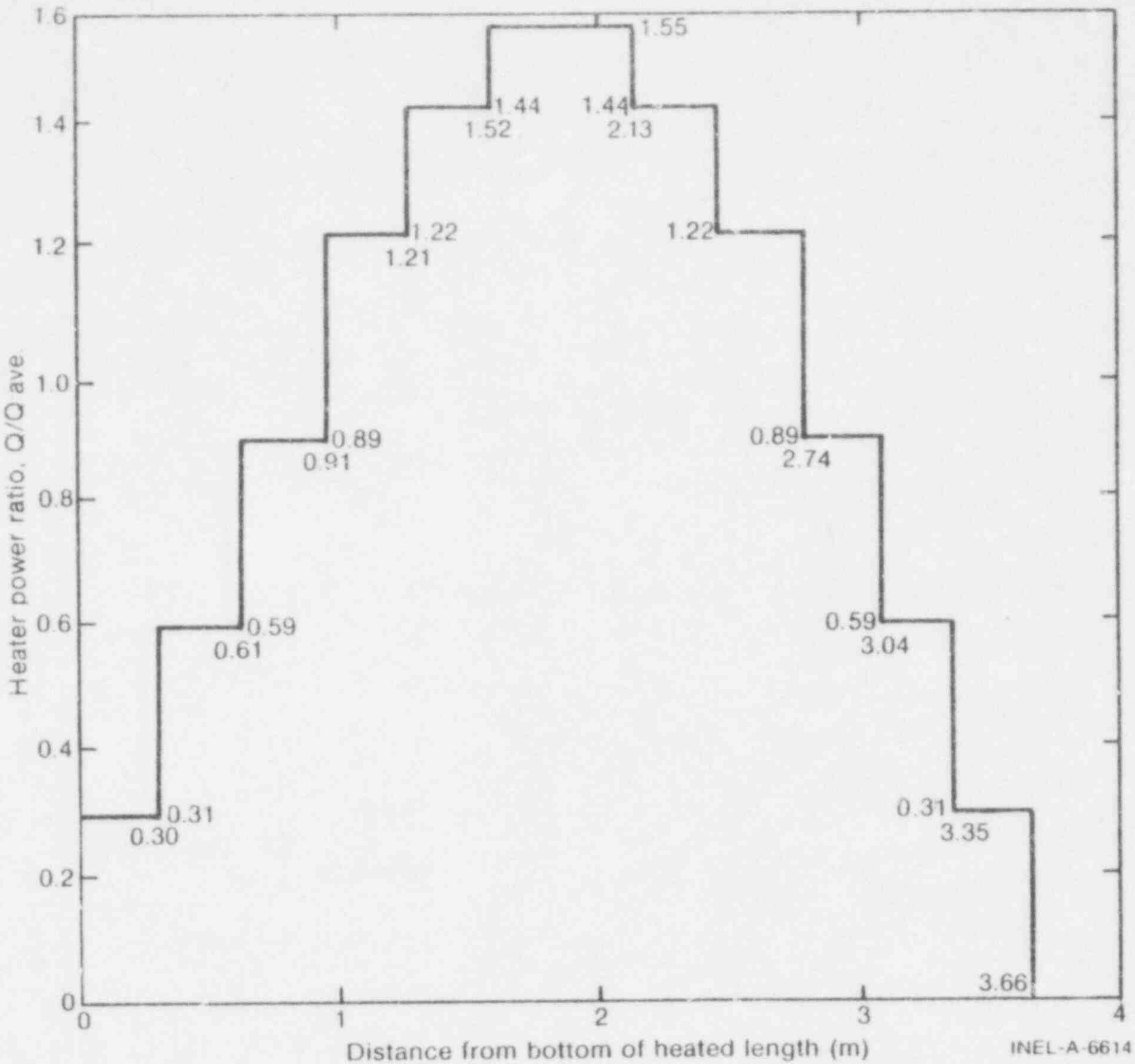

Fig. 20 Semiscale Mod-3 heater rod axial power distribution.

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In Semiscale Mod-3 the radial power profile was flat through most of the Baseline Test Series (Series 7) and will be flat through all of the UHI Test Series (Series 8). However, one of the tests in the Mod-3 Baseline Test Series (Test S-07-2)[14] was conducted with radial peaking, in which the nine center rods in the Semiscale Mod-3 core were powered at a 17.4% higher power than the remaining 14 pheripheral rods. The center nine rods experienced a 15 to 25 K higher temperature than the 14 outer rods; however, system behavior was very similar to other Semiscale Mod-3 tests which had a flat radial profile. This agrees with the results from Reference 9, which states that the influence of uneven axial and radial power distributions on PWR peak cladding temperatures during blowdown is insignificant, based on SCORE-EVET 3-D calculations. Thus three-dimensional effects resulting from uneven power distribution during blowdown are not expected to alter the ability of Semiscale Mod-3 to simulate average PWR core response. The degree that Semiscale Mod-3 can simulate average PWR behavior can be analyzed by comparing the calculated core flow from a non-UHI PWR versus the measured core flow from a Semiscale Mod-3 baseline test (Test S-07-1) scaled to PWR values, as shown in Figure 21. The good agreement between the two curves indicates that the Semiscale Mod-3 system should be able to simulate average PWR thermal-hydraulic behavior quite well.

The above comparisons were from blowdown tests only and did not consider the reflood portion. However, for Test S-07-4[15] and the Westinghouse FLECHT-SET Test 3105B[16], which were reflood tests, a qualitative comparison was made between the Semiscale Mod-3 system and

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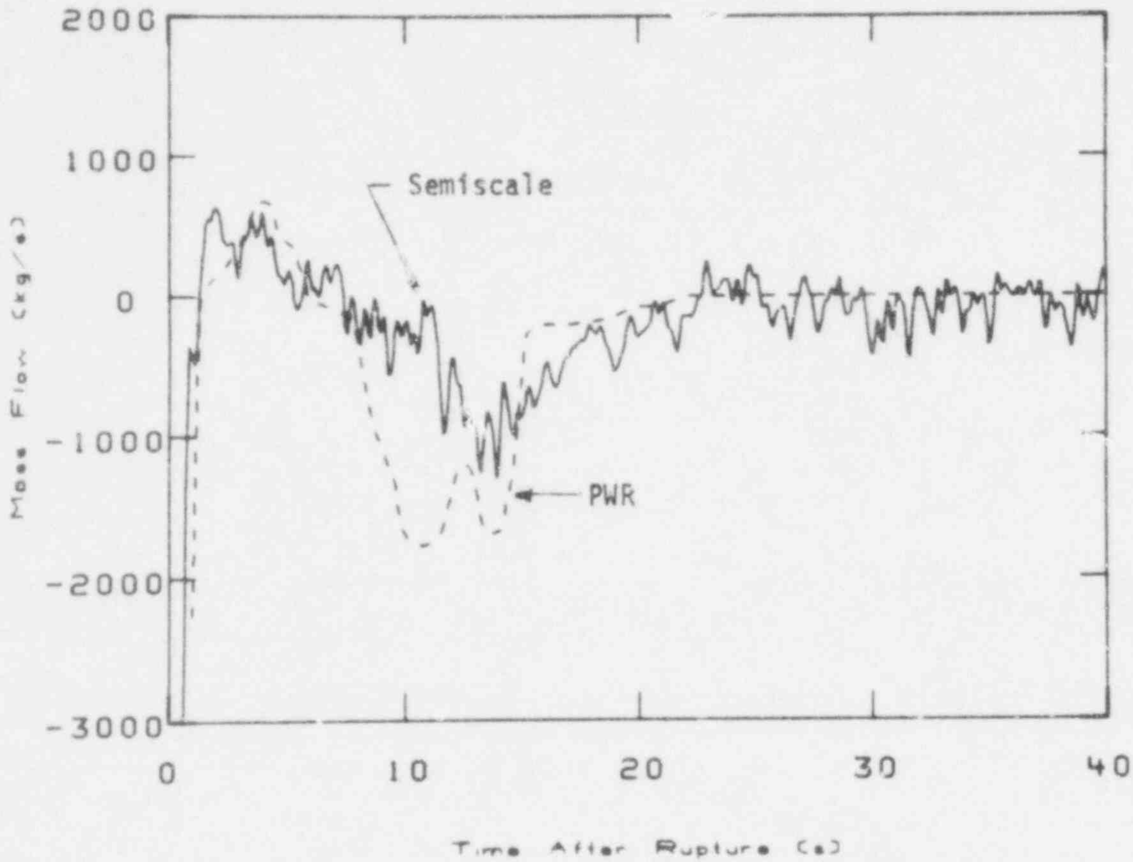

Fig. 21 PWR versus S-07-1 core inlet flow.

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the much larger Westinghouse system during the reflood period. FLECHT Test 3105B had localized peaking and some three-dimensional effects; however, overall quench behavior was very similar to Semiscale Mod-3 Test S-07-4, as shown in Figure 22. Selected elevations from FLECHT Test 3105B and Test S-07-4, shown in Figure 23, also indicate very similar temperature response characteristics. The overall good agreement between the core thermal response and core quench behavior for the two systems demonstrates that experimental facilities which utilize full-length core simulators and operate from similar initial and boundary conditions can produce similar reflood characteristics.

In conclusion, this section has attempted to address the major scaling considerations affecting the Semiscale Mod-3 core thermal-hydraulic behavior. However, there are some secondary considerations which should not alter the overall system behavior, but are mentioned for completeness. As discussed earlier, the Semiscale Mod-3 system has a core configuration representative of a 15 x 15 size nuclear rod fuel assembly. Although this is not typical of the 17 x 17 geometry of a PWR plant equipped with UHI, the overall system response should not be affected since surface temperatures are very close to those of the PWR. Also, the pressure drop across the core is a direct result of the geometry of the Semiscale Mod-3 core. However, since flow areas are scaled and heater rod geometry is similar to that of a PWR, the core pressure drop is not expected to be much different than in a PWR, except for the influence of the grid spacers, (there are 10 in the Mod-3 core), three more than in a PWR. Another

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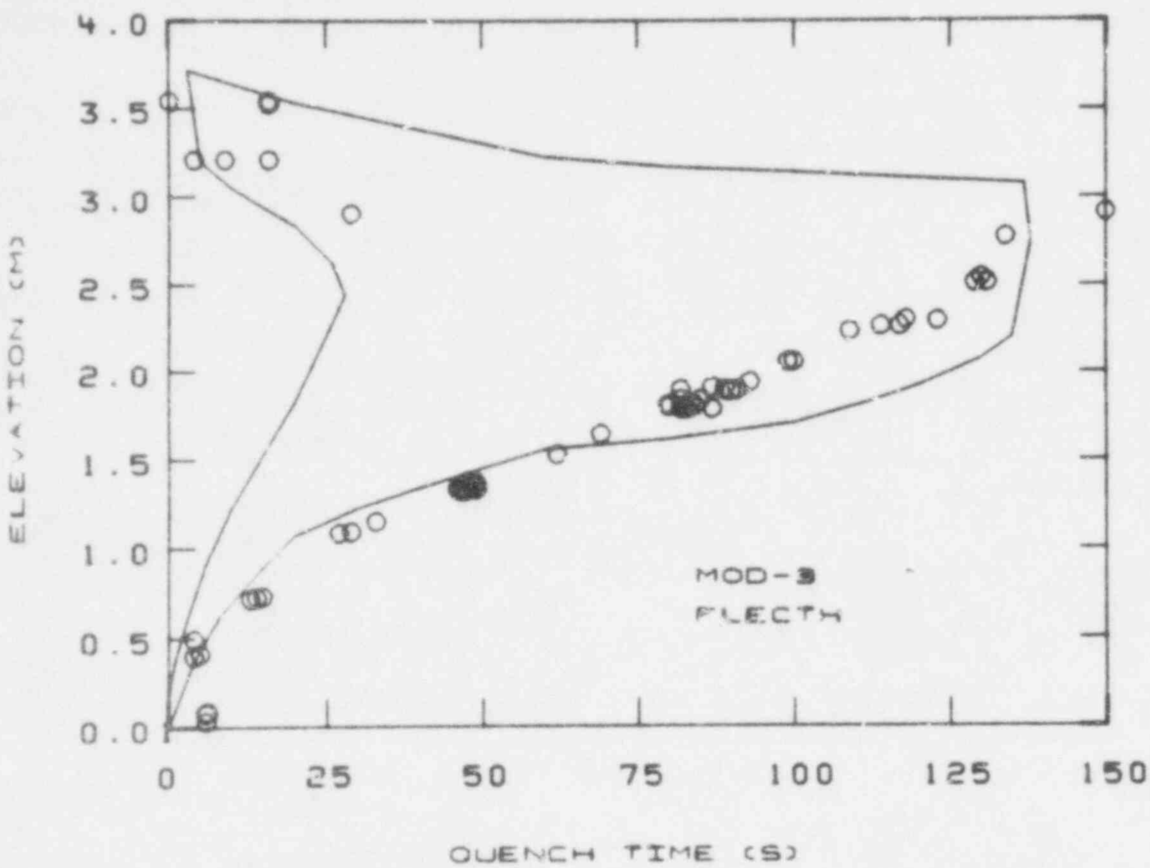

Fig. 22 Envelope of the quench front progression for FLECHT-SET Test 3105B and MOd-3 Test S-07-4 quench points.

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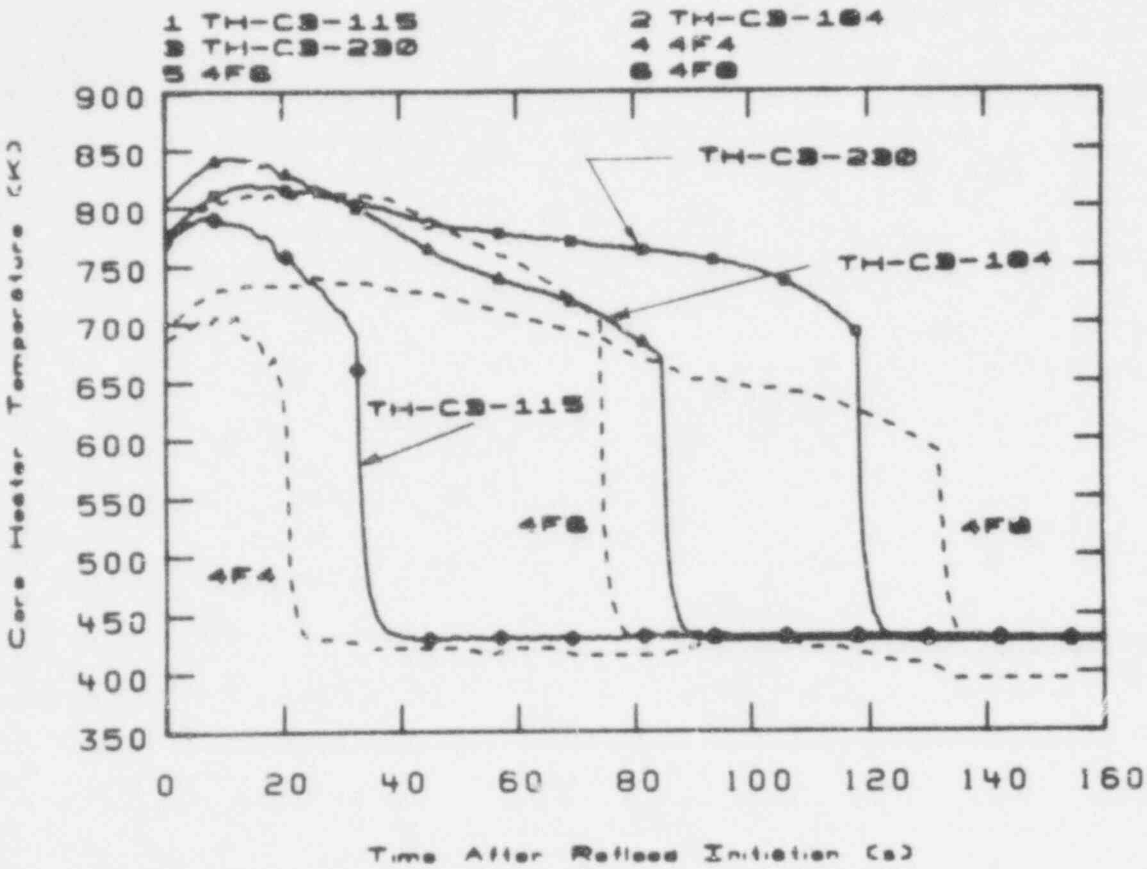

Fig. 23 Comparison of core rod thermocouple response for FLECHT-SET Test 3105B and Semiscale Mod-3 Test S-07-4 for saveral different elevations.

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secondary distortion in the Mod-3 core is the radiation effects of the hot electrical heater rods to the core barrel walls. Although the cross sectional flow area of Mod-3 is scaled from a PWR, the cross sectional size of the Mod-3 core relative to a PWR, places the hot center electrical heater rods much closer to the core barrel walls than in a PWR. This could result in the nine center high powered rods radiating excess energy to the lower temperature walls, which would result in lower peak temperatures on the high powered rods than expected. To reduce the potential for this type of heat transfer, Mod-3 was designed with a gold plated core shroud. The thermal properties of gold being such that most of the radiated energy would be reflected from the core barrel walls (assuming gray body behavior). Therefore radiation effects in the Mod-3 core are not expected to cause adverse system behavior. Finally, the surface area-to-volume ratio of the Mod-3 core barrel is 46 times greater than in a PWR. This compromise has the potential for generating large amounts of steam and effecting system response, expecially during refill and reflood. However, Mod-3 incorporates insulators on the core barrel walls to reduce the potential of this steam generation. Therefore, by installing effective insulators, the effect of core barrel steam generation is not expected to be significant.

The above discussions indicate that blowdown is basically one-dimensional in character and that the Semiscale Mod-3 system should provide adequate simulation of overall PWR behavior in terms of blowdown and reflood heat transfer. However, the implication that blowdown data from the Semiscale Mod-3 system can be directly related to a PWR should not be made as a result of the previous comparisons.

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1.4 Lower Plenum

The Semiscale Mod-3 system lower plenum design emphasized scaling considerations which would produce refill and sweepout behavior as typical of a PWR as possible. Also considered important was the maintenance of the proper amount of residual water in the lower plenum prior to initiation of refill. To effectively achieve these objectives, experimental data from previous Semiscale Mod-1 tests were used as a basis for designing the lower plenum. However, compromises in the lower plenum geometry were necessary to provide for passage of the core heater rod extensions through the lower plenum and out the bottom head. The following paragraphs discuss the lower plenum design and provide an indepth analysis of the lower plenum scaling approach.

The lower plenum region, shown in Figure 24, was defined as all the volume below the heated core including the downcomer annular inlet section. This region was volume scaled, thus allowing a refill rate similar to that of a PWR, with scaled ECC injection rates. Based on previous Semiscale Mod-1 testing[17], the L/D ratio was also considered to be important to properly simulate the sweepout behavior of the lower plenum. However, to model both of these conflicting requirements as closely as possible, the L/D ratio was slightly enlarged to produce a geometric configuration that would satisfy the volume requirement. Therefore, the L/D ratio of the Semiscale Mod-3 lower plenum is 1.57, whereas the PWR lower plenum L/D ratio is 0.67.

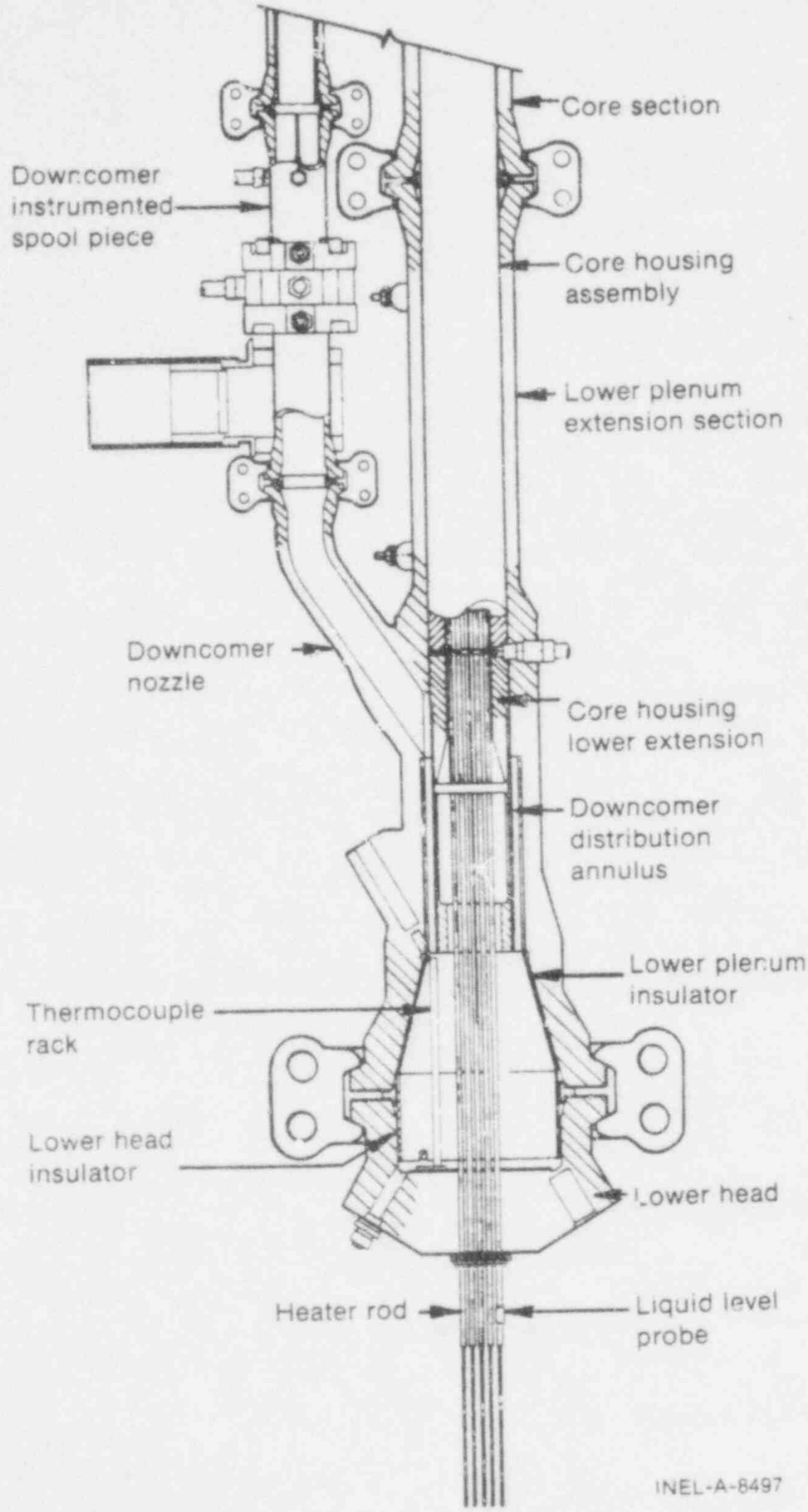

Fig. 24 Semiscale Mod-3 vessel lower plenum region and downcomer instrumented spool.

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Semiscale designers were aware that the presence of the core heater rod extensions would increase the amount of metal heat added to the lower plenum fluid and possibly cause deentrainment and affect lower plenum sweepout. However, because of the importance of the upper head design in the UHI Test Series, the rod penetrations were made through the lower plenum rather than through the upper head region as was done in the Semiscale Mod-1 design. While the total influence of these rod penetrations is difficult to analyze directly, experimental work by others indicates that the lower plenum geometry does not significantly influence the dominant phenomona during blowdown. Reference 18 discusses the results of an investigation evaluting different PWR lower plenum geometries in a test vessel which was a 1/15-scale replica of a four-loop PWR lower plenum. Injection flow rates and subcoolings were varied to determine the range of behavior; the results indicated little difference in sweepout for the different configurations investigated. While the results indicate PWR lower plenum structures did not affect lower plenum sweepout and liquid level depression significantly, the Semiscale heater rod penetrations are sufficiently different from the lower plenum structures in a PWR that this assumption may not be true in Semiscale. Therefore the effect of the rods on system behavior may not be negligble as suggested above, and will require further experimental investigation in the Semiscale Mod-3 system.

Another compromise resulting from the scaling criteria used in the Semiscale Mod-3 lower plenum design is the large surface area-to-volume ratio relative to that in a PWR. In Semiscale Mod-3

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this ratio is about eight times greater when no internals or insulation are considered. When internals are included, this ratio increases to about 16. However, insulation has been placed on the lower plenum walls to aid in reducing energy transfer from the heavy plenum walls. Since it was difficult to insulate heater rod extensions while maintaining a reasonably open flow path from the core region, there was still concern that the large surface area-to-volume ratio and the resulting heat transferred to the fluid could influence core flow with the potential for flow stagnation during the early blowdown period in Semiscale Mod-3. To address this concern, two calculations were performed using a one dimensional conduction code with heat transfer coefficients and fluid temperatures from the RELAP4/MOD6 pretest predictions of Test S-07-1 as boundary conditions. The first calculation used a model of the Semiscale Mod-3 lower plenum without internals while the second calculation incorporated a model of the PWR lower plenum also without internals. The comparison of the predicted structural heat transfer rate per unit volume of fluid, from a PWR and Semiscale Mod-3 lower plenum is given in Figure 25. It is apparent, based on the results in Figure 25, that the potential for excessive heat transfer (approximately 15 times greater) is present in the Semiscale Mod-3 lower plenum.

To further investigate the potential effects of lower plenum heat transfer on core flow behavior, RELAP4/MOD6 calculations were performed in which the lower plenum structural heating was varied to evaluate the effects on core flow behavior. The results from these calculations were then compared with the measured data from

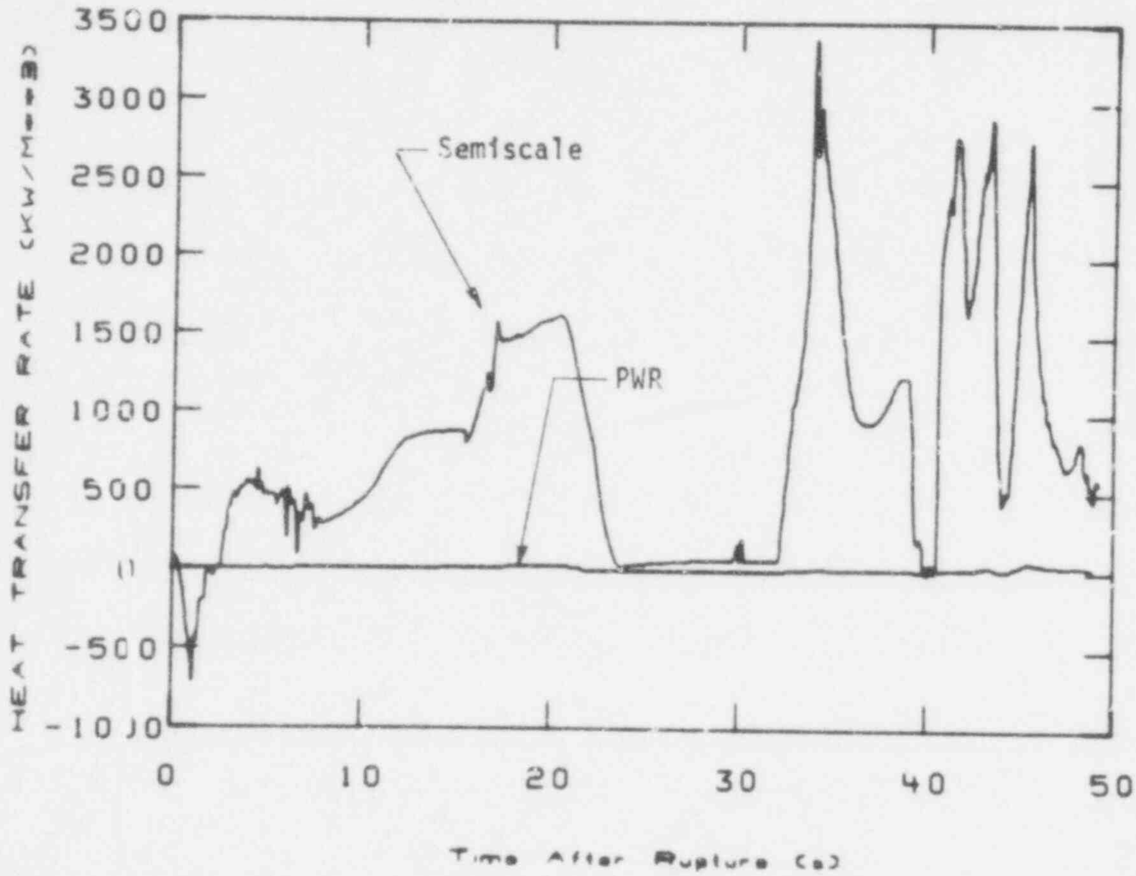

Fig. 25 Heat per unit volume added to lower plenum fluid from the PWR and Semiscale lower plenum walls.

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Test S-07-1 and with results from Westinghouse calculations. The measured core flow from Test S-07-1 is compared to the RELAP4/MOD6 prediction of core flow for Test S-07-1 in Figure 26. To provide a more meaningful interpretation of the results, the measured and calculated core mass flows in Figure 26 were multiplied by the PWR-to-Mod-3 core power ratio (3411/2), which is equivalent to the ratio of core flow areas, to obtain mass flows on a scale comparable to those that might occur in a PWR. The measured and calculated results in Figure 26 agree rather closely, indicating the RELAP4/MOD6 calculations predict the general trends and overall behavior of the Semiscale Mod-3 core reasonably well. This general similarity between the RELAP4/MOD6 predictions and the measured Semiscale Mod-3 results provided a confidence in the utilization of the RELAP4/MOD6 computer code for investigating lower plenum heat release. Two additional RELAP4/MOD6 calculations were performed which were identical to the original calculation for Test S-07-1 except for the variations in structural heat transfer. In the first calculation, all heat conductors in the RELAP4/MOD6 model were removed except those which represented the active core and steam generators. The second calculation had only the heat conductors in the lower plenum removed since this was an area where structural heat transfer was thought to be particularly critical. Since the latter two calculations were identical to the original calculation with the exception of the differences in structural heating, comparison of results from the three calculations which are discussed in the following paragraphs should provide a direct indication of the effect of structural heat transfer on predicted Semiscale Mod-3 system response.

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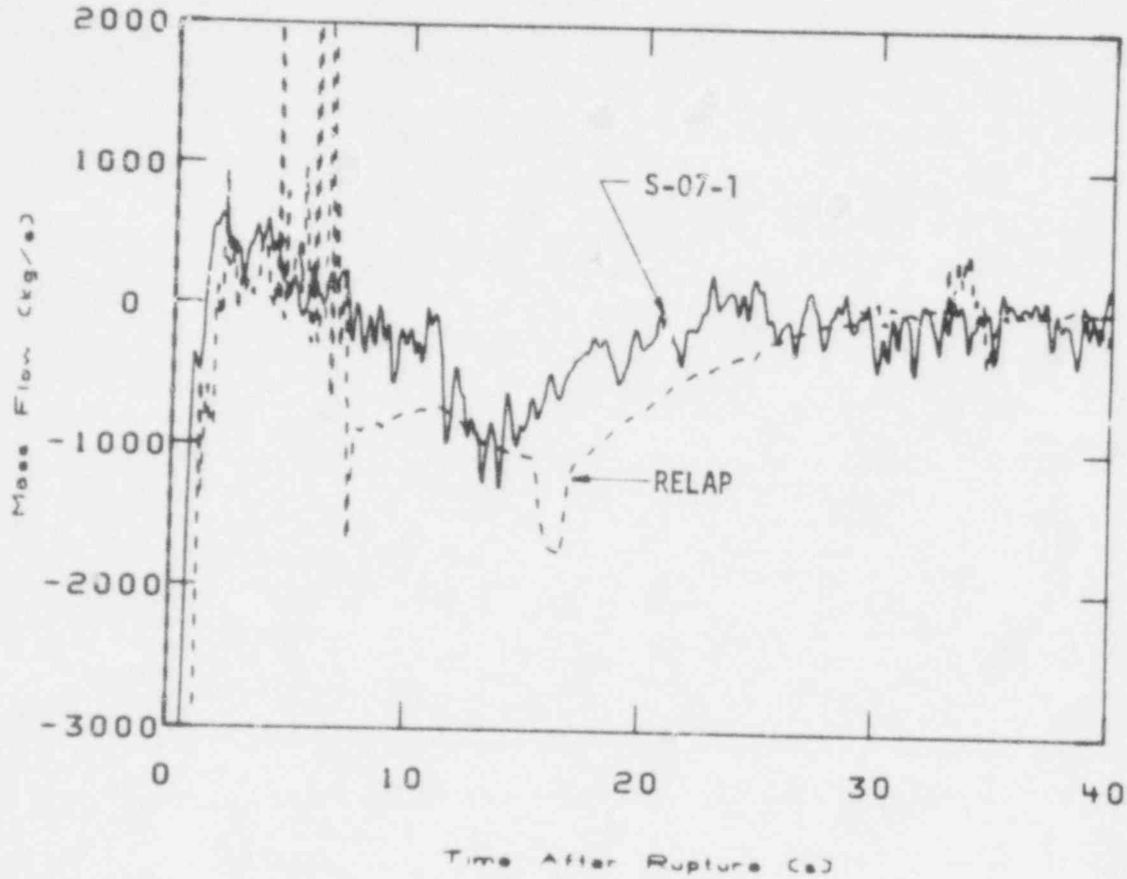

Fig. 26 RELAP4/MOD6 calculated core flow and Test S-07-1 core flow.

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Figure 27 compares the calculated Semiscale Mod-3 core mass flow with no structural heating to the RELAP4/MOD6 prediction for Test S-07-1, which included all structural heat conductors (again multiplied by the ratio of core powers, 3411/2). The comparison in Figure 27 demonstrates the sensitivity of the RELAP4/MOD6 calculations to changes in structural heating, and shows that with all structural heat conductors removed from the RELAP4/MOD6 model (piping, vessel, downcomer, and core barrel), the magnitude of the negative core flow peaked at a higher value and occurred slightly earlier than in the calculation with all structural heating included. However, in both calculations, the duration of the predicted Semiscale Mod-3 negative core flow was approximately the same, lasting until approximately 30 s after rupture.

Despite the demonstrated sensitivity of the RELAP4/MOD6 calculations to structural heat transfer, the effect of lower plenum heat transfer on the calculated Semiscale Mod-3 system response appears to be minimal. The comparison of predicted core mass flow with and without lower plenum heat transfer (again multiplied by the core power ratio), Figure 28, shows no significant difference in the core hydraulic behavior. Also, since the energy transfer from the heater rod extensions to the lower plenum fluid appears to be "conduction limited" the heat transfer rate should be relatively unaffected as long as sufficient water is delivered to the lower plenum to maintain a wet metal surface. Therefore, with UHI and the

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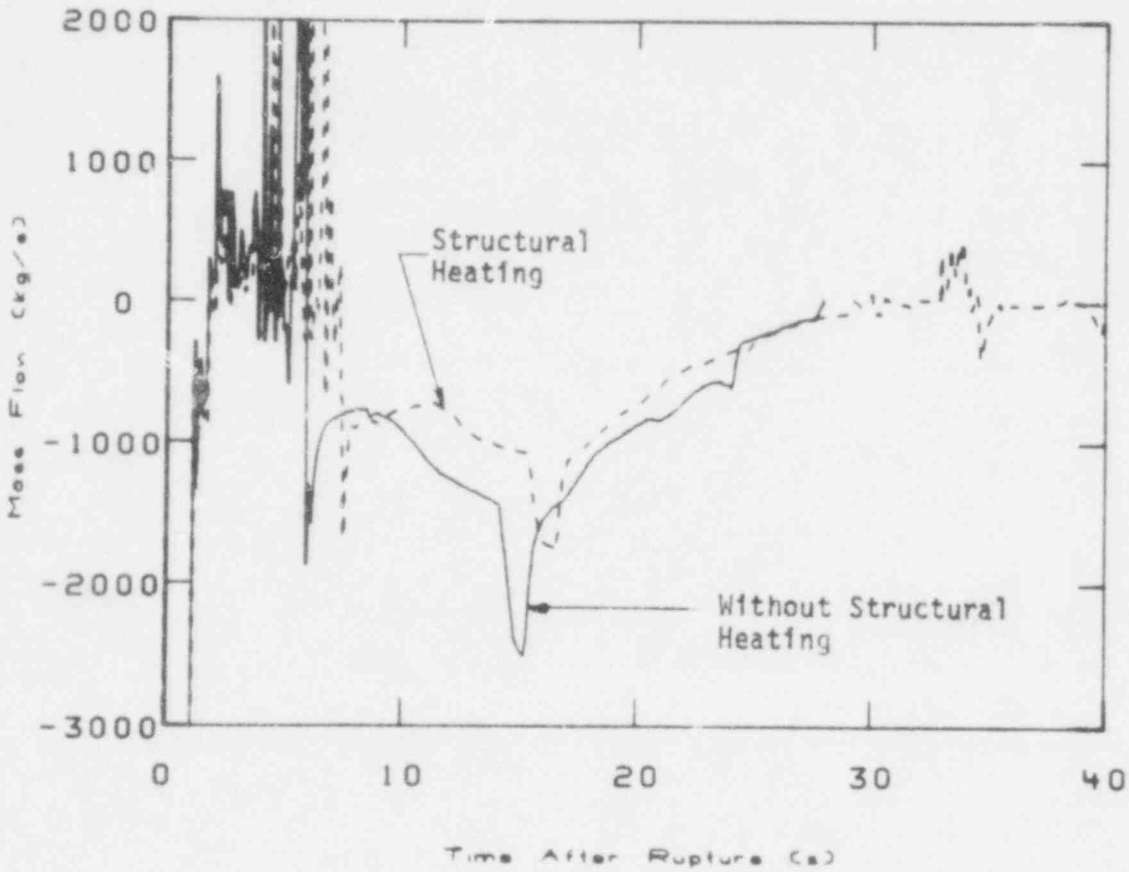

Fig. 27 Comparison of predicted core inlet mass flow with and without structural heating.

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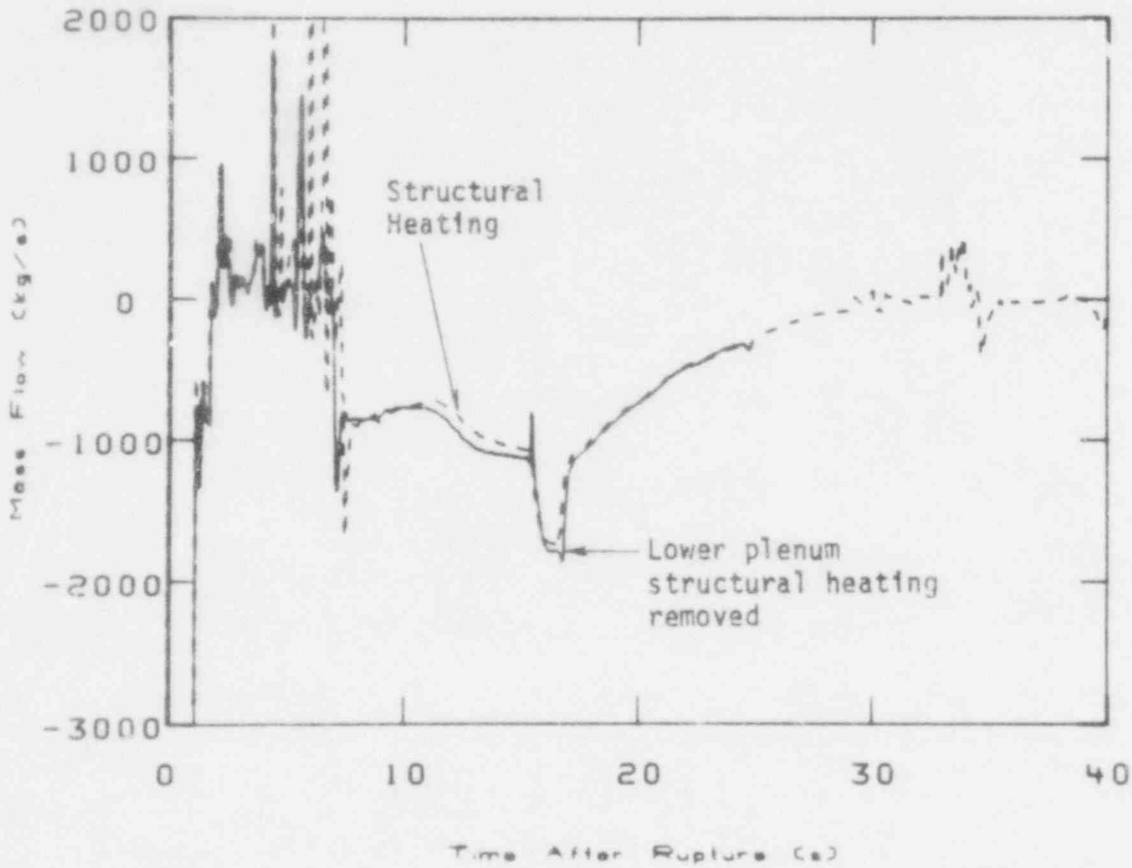

Fig. 28 Comparison of S-07-1 pretest prediction (with all heat slabs included) to calculation without structural heating in lower plenum.

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associated increase in negative core flows and entrained liquid, it is felt that structural heat transfer in the lower plenum is not likely to have a major influence on the calculated Semiscale Mod-3 system response.

Finally, Figure 29 compares the Semiscale Mod-3 calculations (with differing structural heat transfer) with the Westinghouse calculated transient (dashed line) for previously discussed in Section 1.3. Figure 29 shows that the peak in the calculated Westinghouse core mass flow occurred earlier than in the calculations for Semiscale Mod-3 and the duration of the negative core flow was not as long. However, considering that the calculations were performed for different systems using different modeling assumptions, the results are reasonably similar and indicate the same general trends in core hydraulic response.

In conclusion, RELAP4/MOD6 calculations indicate that lower plenum structural heat transfer will not significantly influence the calculated Semiscale Mod-3 system response. The effects of the L/D ratio on the Mod-3 sweepout characteristics will be further analyzed during Semiscale testing. Although the Semiscale Mod-3 core flow characteristics might not entirely duplicate the core flow in a plant with UHI, the results should be sufficiently representative to provide information about operational parameters and system interaction characteristics that may be important in a PWR.

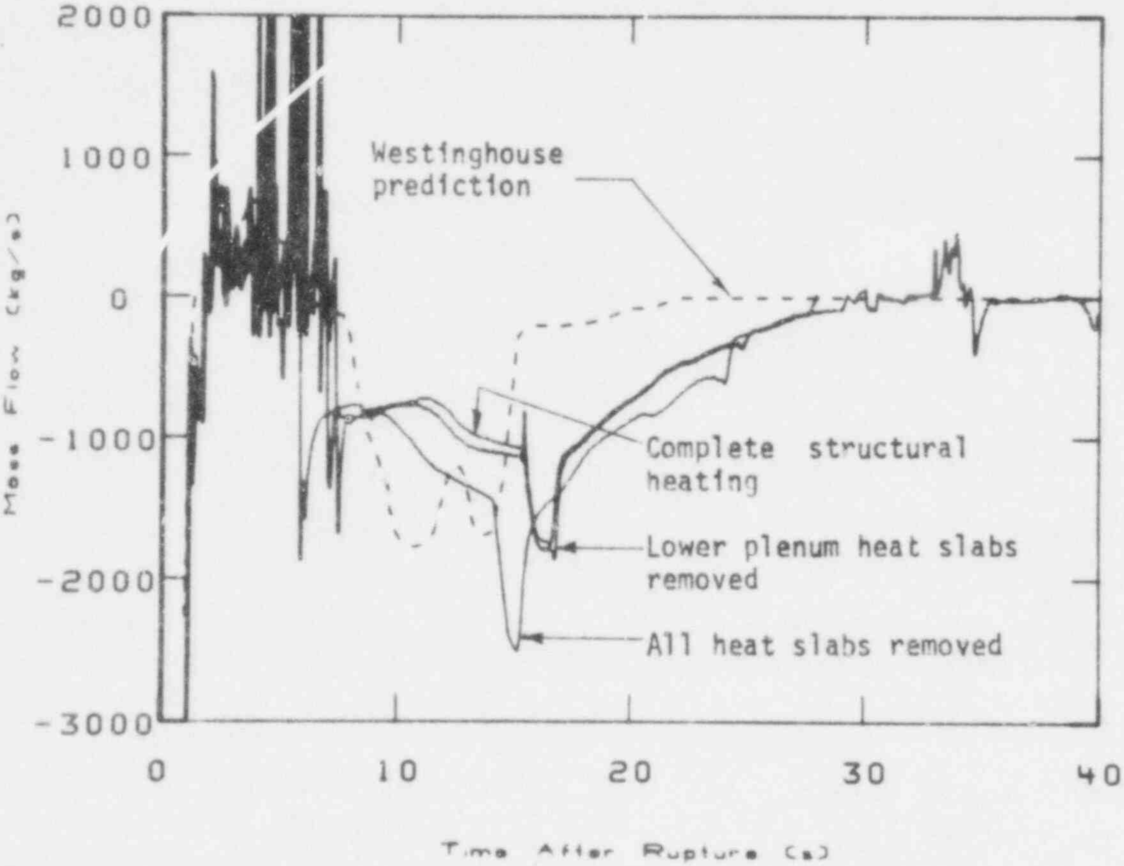

Fig. 29 Comparison of Mod-3 predicted core inlet mass flows with Westinghouse predicted transient.

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In conclusion, RELAP4/MOD6 calculations indicate that lower plenum structural heat transfer will not significantly influence the calculated Semiscale Mod-3 system response. The effects of the L/D ratio on the Mod-3 sweepout characteristics will be further analyzed during Semiscale testing. Although the Semiscale Mod-3 core flow characteristics might not entirely duplicate the core flow in a plant with UHI, the results should be sufficiently representative to provide information about operational parameters and system interaction characteristics that may be important in a PWR.

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1.5 Downcomer

One of the unique features of the Semiscale Mod-3 system is the inclusion of a single pipe external to the vessel to represent the downcomer in a PWR. The single pipe design, which is shown in Figure 3, consists of an inlet annulus, the downcomer pipe, and an instrument spool piece. The inlet annulus assembly contains the cold leg nozzles and is designed to provide an annular inlet geometry similar to that of a PWR. The inlet annulus then funnels into the downcomer pipe. The pipe has an insulating liner to reduce the amount of heat transferred from the outer pipe to the fluid flowing through the pipe. At the bottom of the downcomer pipe is a fully instrumented spool piece containing a full-flow turbine meter, drag screen, densitometer, and differential pressure port. Below the spool piece the downcomer angles into the vessel wall and enters the lower plenum through an annular flow skirt similar to that in a PWR.

The decision to utilize an external downcomer in the Semiscale Mod-3 system was based on several considerations. First, the need for more accurate in-core hydraulic information would be obtainable with the use of an external downcomer. Secondly, it was concluded that the one-dimensional behavior of a scaled annulus would differ little from that of a single pipe. Finally, the Semiscale Mod-1 annular downcomer experienced a hot wall delay during ECC penetration. The hot wall delay was attributed to the hot walls on both sides of the downcomer annulus generating large amounts of steam when the cold ECC fluid entered the inlet annulus. With a small annular gap in the Semiscale

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Mod-1 system, the steam could hold up the ECC fluid until sufficient heat was removed from the metal walls and the steam flow was reduced. By adopting a single external pipe, the surface area-to-volume ratio was lowered, reducing the potential for steam generation. In addition, it was potentially simpler to insulate the inner wall of a pipe.

Effective modeling of the annular entry of the PWR downcomer required an inlet annulus to be located at the top of the Semiscale Mod-3 single pipe downcomer. The Semiscale Mod-3 inlet annulus, shown in Figure 30, was sized to maintain the same fluid transit time (0.47 s) from the intact loop cold leg vessel penetration to the broken loop vessel penetration as expected in a PWR. The significance of flow transit time lies in the tendency of gravity to affect flow into the downcomer region during the refill and reflood portions of the LOCA. It is expected that the ECC injected during saturated blowdown will tend to flow around the distributor annulus and out the broken cold leg or fall down the downcomer according to the entering momentum flux and the transit time around the flow path. In maintaining transit time in the Semiscale Mod-3 upper annulus, the scaling of the upper annulus volume was somewhat compromised ($0.0077 m^3$ actual versus a desired volume of $0.0071 m^3$ from volumetric scaling, see Table II).

Some areas that were considered to be secondary as a result of scaling the inlet annulus volume and transit time included the shorter nozzle to nozzle distance and two- and three-dimensional effects at

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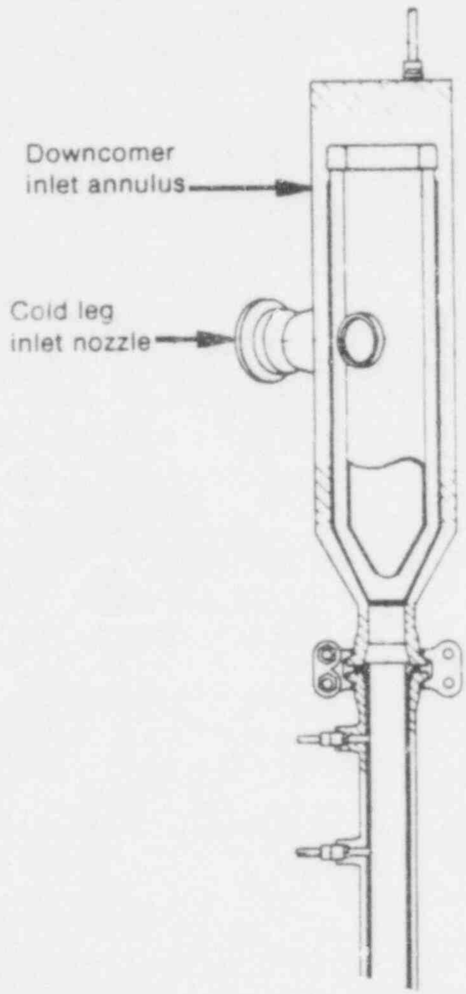

Fig. 30 Semiscale Mod-3 downcomer inlet annulus.

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the neck of the inlet annulus. The much shorter distance between the nozzle of Semiscale Mod-3, relative to a PWR, could allow the influence of fluid momentum to be present which may result in higher bypass flow rates. Also, the annulus funnels flow into the single pipe and restricts the potential for two- and three-dimensional effects in that region. The effects that these secondary considerations will have on the overall Semiscale Mod-3 system behavior were not believed to be as important as the transit time and volume scaling, and therefore were compromised.

Scaling of the downcomer was difficult due to an incomplete understanding of the two-phase countercurrent flow phenomena which dominate the behavior during the LOCA. Several methods of scaling were considered including use of the Kutateladze[19] number and use of a Wallis flooding correlation developed from Semiscale air-water countercurrent flow experiments[20]. Results from these methods of scaling were inconclusive, since it was difficult to characterize the PWR downcomer behavior. However, as stated previously, the secondary reason for using a single pipe design was based on the determination that flow through the Semiscale Mod-1 annulus behaved similar to flow in a pipe. The critical gas velocities for flooding or stagnation of falling fluid in the Semiscale Mod-1 annulus and the Semiscale Mod-3 single pipe, are shown in Figure 31. These velocities were determined by using the Wallis and Kutateladze correlations, the calculations of which are given in Appendix B. From this comparison, the expected countercurrent flow behavior for the annulus and the single pipe should be similar. Strict volumetric scaling of the downcomer was also considered so that downcomer volume and height would be

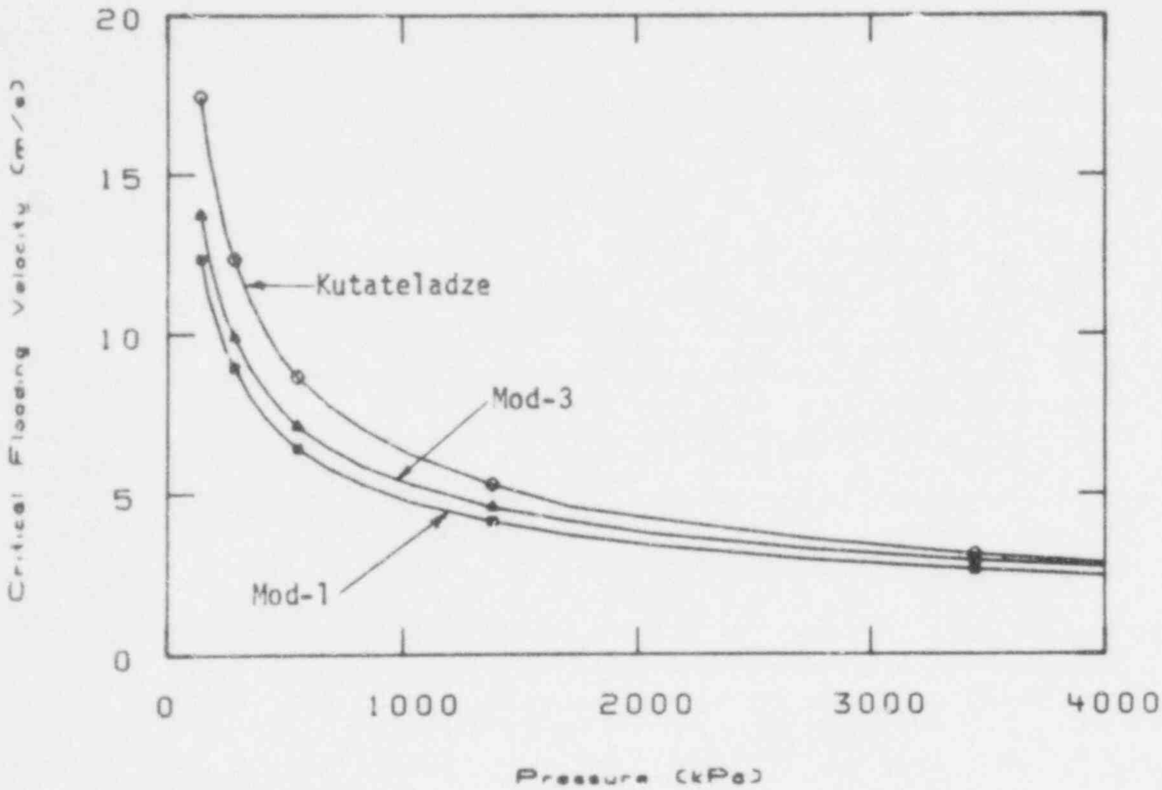

Fig. 31 Critical gas gelocities for Semiscale Mod-3 and Mod-1 downcomers.

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maintained as closely as possible to duplicate downcomer fluid elevation changes during the latter portion of refill and the reflood period. Since strict volumetric scaling and scaling based on the flooding correlations produced about the same diameter, downcomer volumetric scaling was selected.

The decision to scale the downcomer volumetrically, while maintaining full-length vertical distances, resulted in the downcomer being one-dimensional in nature. This allows gravitational effects to be modeled well, but results in a compromise in expected countercurrent flow phenomena. In a PWR, it is expected that ECC will establish two- and three-dimensional flow patterns where the water will channel and not interact directly with the steam. Lack of potential for two- and three-dimensional effects in a single pipe downcomer lead to development of a second downcomer design, which can be installed on the Semiscale Mod-3 vessel and will allow, for some seperate channeling of steam and water flows. This design, shown in Figure 32, is called the parallel pipe or two pipe downcomer.

Although this new downcomer design will improve the countercurrent flow potential of Semiscale Mod-3, it will also increase the surface area-to-volume ratio of the downcomer and, therefore, increase the effects of wall heat transfer characteristic to the previously mentioned single pipe downcomer. This heat transfer phenomena has been referred to as the hot wall effect.

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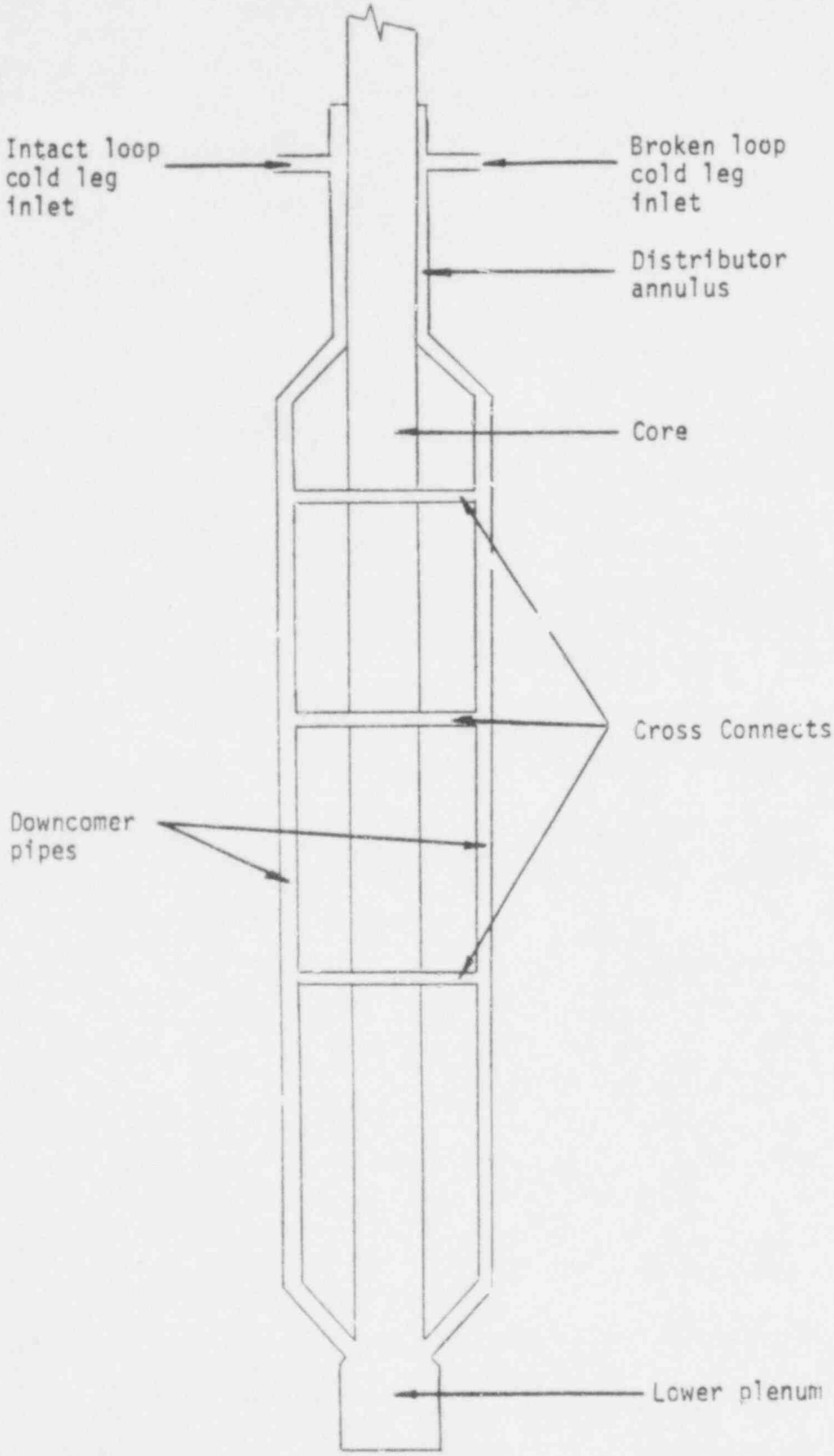

Fig. 32 Semiscale Mod-3 vessel and double pipe downcomer.

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Hot wall effects are expected to be minimal in a PWR because of the large gap size, thus allowing the potential of countercurrent flow. However, with a surface area-to-volume ratio in the Semiscale Mod-3 downcomer which is about nine times greater than the PWR downcomer, the potential steam generation due to heat transfer from the downcomer walls has greatly increased. The modification of PWR-like behavior in Semiscale Mod-3 as a result of these hot wall effects is of great importance. Therefore, a brief description of the nature of these hot wall effects and the way they effect Semiscale Mod-3 will be given, followed by a proposed solution to the hot wall problem.

The hot wall effect can be broken down into two specific areas: delay in ECC penetration to the lower plenum and mass depletion of water in the downcomer during core reflood. As ECC enters the downcomer at the beginning of injection, steam is generated and causes fluid to be held up in the downcomer and bypassed out of the downcomer inlet annulus to the cold leg break. This phenomena is termed hot wall delay, and was shown to occur in Test S-07-1[4]. Figure 33 shows the densities in the lower plenum and in the top of the downcomer and illustrates the time required for liquid to reach the lower plenum. The delay time from the start of penetration until refill initiation was observed to be 9 s. This also was observed in the remaining blowdown Tests S-07-2, S-07-3[21], and S-07-6[13].

Following the hot wall delay period, fluid fills the downcomer pipe, and then a second hot wall effect, unique to Semiscale Mod-3 occurs which is termed mass depletion. The fluid in the downcomer

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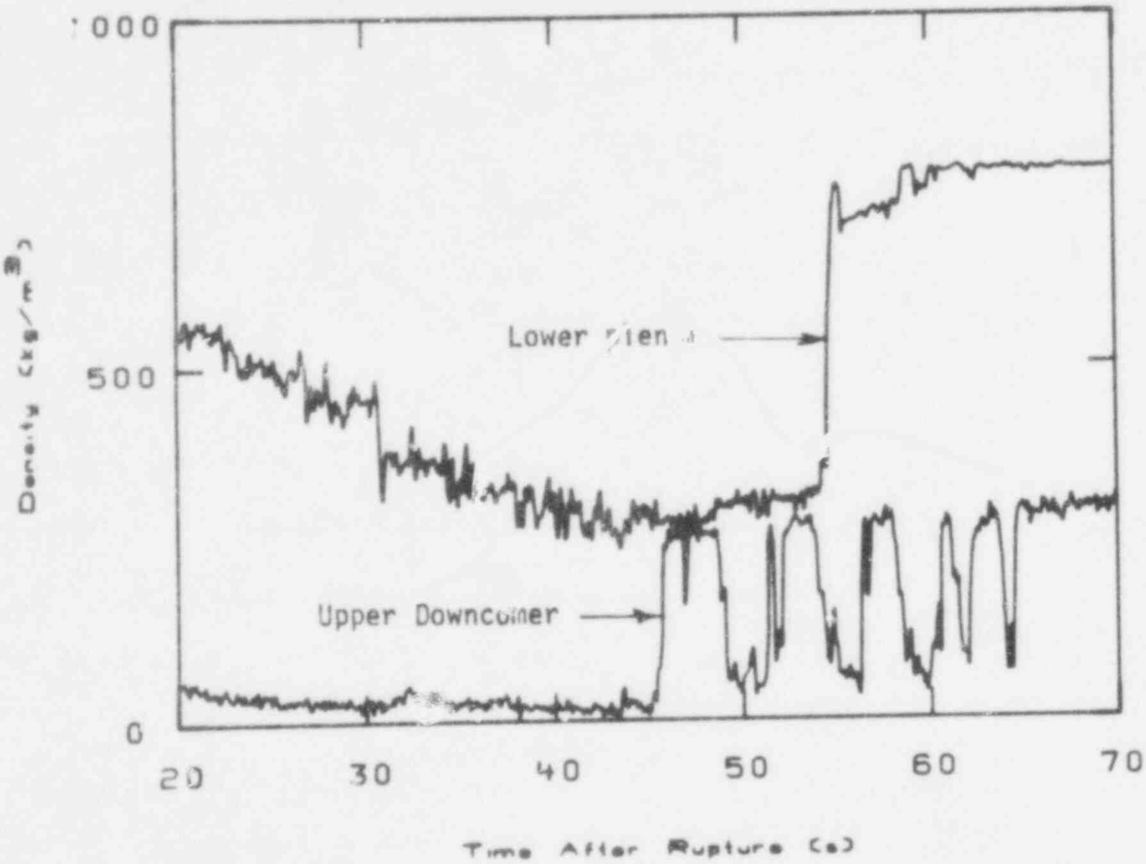

Fig. 33 Comparison of upper downcomer and lower plenum fluid density.

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pipe is heated by the stored energy from the pipe walls with the hottest fluid being at the bottom of the downcomer. When saturation is reached, the fluid expands as it flashes and causes an upward flow in the downcomer pipe. As the hydraulic head decreases, the expansion takes place more vigorously until the entire downcomer pipe has depleted of water. Test S-07-6A was an integral blowdown reflood test that exhibited the above mass depletion phenomena, see Figure 34. Following this mass depletion, the system behavior was oscillatory in nature. This resulted because of the gradual refilling of the downcomer by the low-pressure injection system (LPIS) fluid and then depletion as the fluid was heated to saturation. Notice that the time between oscillations is much greater than manometer oscillations, thus negating the effects of manometer-type phenomena. The effect that the hot wall phenomena will have on UHI will be determined during the course of Series 8 testing in Semiscale.

Several tests have been conducted in the Semiscale Mod-3 system to analyze the mechanism by which the stored energy in the outer downcomer pipe is transferred to the system fluid. A cross section of the present Semiscale Mod-3 downcomer design is shown in Figure 35. A complete description of these tests as well as a discussion of the downcomer mass depletion is contained in Appendix C. From the results of these tests, it was concluded that a new design for the insulator pipe was needed to reduce the amount of stored energy transferred to the system fluid.

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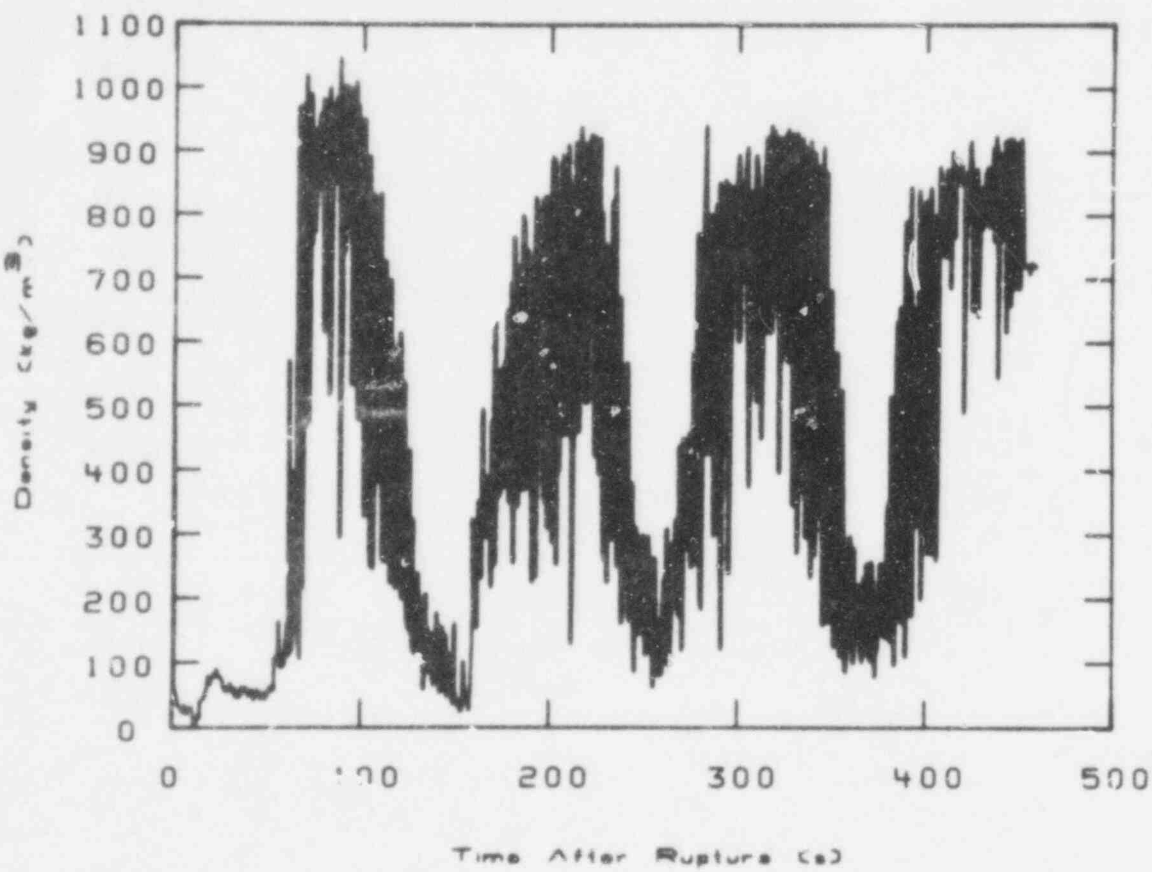

Fig. 34 Core inlet fluid density for Test S-07-6.

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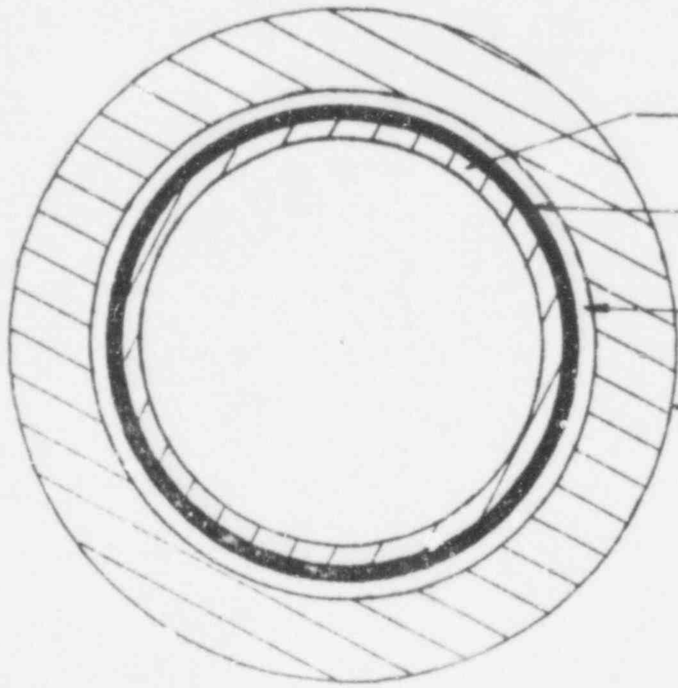

Fig. 35 Downcomer cross section.

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At present, a honeycomb structure is being considered for the insulator pipe and is currently being evaluated for thermal characteristics. The honeycomb structure is made up of a small, thin walled pipe, sized to the desired downcomer flow area, placed inside a larger thin walled pipe. The honeycomb material is sandwiched between the two pipes with the cells being evacuated of gases and sealed. The structure acts like a "thermos bottle," isolating the system fluid from the outer downcomer walls. Preliminary calculations for this material are encouraging as indicated in Figure 36, which is a comparison of heat transfer rates from the present Semiscale Mod-3 downcomer, a PWR downcomer, and the Semiscale Mod-3 downcomer incorporating the honeycomb insulator. It is expected that the effects of stored metal heat can be significantly reduced with the use of the honeycomb insulator.

As mentioned earlier, the single pipe downcomer of Semiscale Mod-3, incorporates an instrumented spool piece installed above the nozzle that enters the lower plenum. This instrumentation was necessary to measure system behavior as close to the lower plenum as possible. As a result of the presence of this instrumentation, flow behavior could possibly be altered. The turbine meter and drag screen could homogenize an annular flow behavior which is possible in a single pipe arrangement, and also deentrain fluid from the negative core flow. However, this compromise was accepted in trade for the vital hydraulic information gained.

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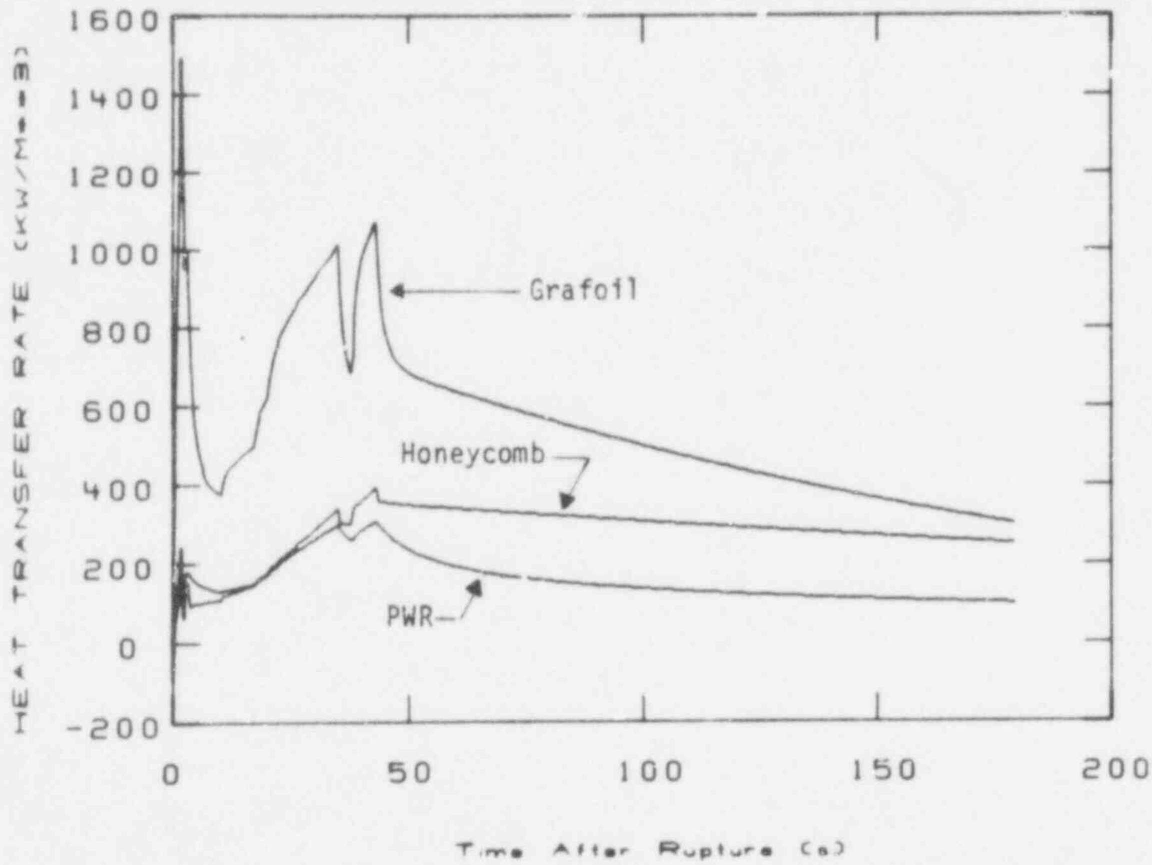

Fig. 36 Heat transfer rate comparisons for different downcomer insulators.

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Concluding the discussion of this section, the Semiscale Mod-3 downcomer has been designed and scaled to increase the flexibility of the Semiscale Mod-3 vessel to monitor the system response during a LOCA experiment. The external configuration, although atypical in appearance, should function similarly to the annular Semiscale Mod-1 downcomer in terms of countercurrent flow behavior. The hot wall effects encountered are not insurmountable, and upon modification to the insulator, the downcomer mass depletion problem should be solved.

In summary, the Semiscale Mod-3 vessel design has been discussed by identifying its individual components. There are scaling compromises which make its behavior not directly applicable to a PWR. However, important phenomena will be simulated, and, therefore Semiscale Mod-3 will be useful for code assessment. In addition, energy storage in the metal structure throughout the Semiscale Mod-3 system, coupled with the increased surface area-to-volume ratio, has the potential for generating large amounts of steam which could alter the effectiveness of UHI. Although Semiscale Mod-3 does not model the surface area-to-volume ratio of a PWR well, insulation have been incorporated to reduce the effects that the added heat flux will have on system fluid, and the addition of a new downcomer insulator design is expected to be even more effective in minimizing structural heat transfer. Semiscale Mod-3 required full length volumes, where possible, resulting in a vessel that is one-dimensional in nature and as a result two- and three-dimensional effects should not be expected

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because of the small scale. These characteristics, although not entirely representative of a PWR, are not expected to alter system behavior enough that the capability of Semiscale Mod-3 to determine effects of UHI will be hampered. However, the vessel behavior of Semiscale Mod-3 it typical of a small integral test facility and further extrapolation of Semiscale Mod-3 data should be done judiciously.

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2. INTACT AND BROKEN LOOPS

The intact and broken loops of Semiscale Mod-3 were designed to represent the operating and ruptured loops of a PWR. The intact loop represents three operating loops of a PWR, and the broken loop incorporates a break apparatus to simulate the ruptured PWR loop. Also, to insure a more accurate representation of the PWR loops, active steam generators and pumps were installed in the intact loop as well as in the broken loop.

The components that comprise each loop generally required application of specialized scaling principles. A combination of scaling considerations including volumtric, elevation, hydraulic resistance and many others were applied. The scaling approach for each of the components will be examined in more detail beginning with the loop piping.

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2.1 Loop Piping

The Semiscale Mod-3 loop piping was scaled with the primary considerations being maintenance of the desired resistance and scaled volume. The desired loop piping volume and the volume distribution, defined in Table II, could be implemented by long lengths of small diameter pipe or shorter lengths of large diameter pipe. However, if the small pipe was used, the pressure drop through the pipe would be larger than that in the PWR pipes. Therefore, the selection of larger diameter pipe (inside diameters of 6.65-cm versus 2.93-cm for intact loop and 3.40 cm versus 1.69 cm for the broken loop) allowed some degree of control of system resistance, whereas with full-length ideally scaled piping, the friction loss (proportional to f L/D) increases disproportionately. Use of the larger pipe also resulted in a more favorable surface area-to-volume ratio from the viewpoint of heat input to the fluid from the structure walls. The surface area-to-volume ratio varies inversely with the pipe diameter (= 4/D), and the use of the larger pipe (6.65-cm inside diameter) results in a 50% reduction of this ratio over that resulting from the use of the ideally scaled 2.93-cm inside diameter piping (2.93 cm represents 3 PWR loops combined and volumetrically scaled down to Semiscale Mod-3 values). However, even with the current pipe sizes the surface area-to-volume ratio is approximately 19 times greater.

The resistance of the piping in the Semiscale Mod-3 intact loop is lower than that required by scaling from a PWR because of the relatively large pipe diameter as discussed previously. Therefore, provisions have been made for the installation of orifices in the

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Semiscale Mod-3 system to obtain the required intact loop hydraulic resistance. Since 85% of the losses in a large PWR operating loop occur across the steam generator, and the rest occur in the piping, the required orifices in the Semiscale Mod-3 intact loop piping are located as close as possible to the steam generator. The resistance in the broken loop is very similar to the PWR value and did not require the use of orifices.

The decision to scale the loop piping with resistance and volume as the main criteria resulted in some secondary compromises. The piping used in Semiscale Mod-3 is larger in diameter and shorter in length than the scaled PWR values. As a result of the shorter length, the acoustic wave transit times, which are primarily important during the subcooled decompression period, are not the same as would occur in a PWR. However, the subcooled decompression process is relatively well understood, and acoustic wave transit time does not have a significant influence on core cooling and the ECC injection process, which are major concerns in the Semiscale Mod-3 Program.

The smaller pipes of Mod-3, relative to a PWR, can influence the duration of flow regimes that effect system behavior. The typicality of flow regimes occuring in the MOD-3 piping relative to those expected in a PWR have not been evaluated since PWR piping flow regimes have not been fully identified. However, Mod-3 does have the available instrumentation to allow determination of flow regimes from Semiscale data. An example of the flow regimes in the Semiscale Mod-3

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intact loop, calculated from measurements obtained during the UHI baseline test series, is shown in Figure 37. These calculations were made using a method suggested by Govier and Omer[22] as outlined in a report written by R. T. French[23]. The description of these calculations is given in Appendix D. The calculated flow regimes, identified in Figure 39, show the duration of the dispersed bubble and stratified flow regimes, which included wavy flow. While the measurement of flow regimes is important to the understanding of hydraulic phenomena occurring in Semiscale Mod-3, it should not be construed that the same two-phase flow phenomena will occur in the larger PWR piping systems. Especially if the orifices cause a "daming" of the water in the pipes. However, as more information becomes available from larger facilities, such as LOFT, the effects of pipe size on flow regimes and the potential influence of these flow regimes on system response in a full sized PWR will be better understood.

In conclusion, the Semiscale Mod-3 piping was scaled from resistance and volume values in a PWR. Full-scale loop resistance was maintained typical of a PWR while the L/D ratio was compromised in order to use larger than scaled diameter pipes to provide flexibility in resistance placement. The capability of the Semiscale Mod-3 loop piping to simulate the PWR piping is expected to be good; however, the influence of flow regimes behavior is, as yet, still hard to determine because of the lack of flow regime data from larger systems.

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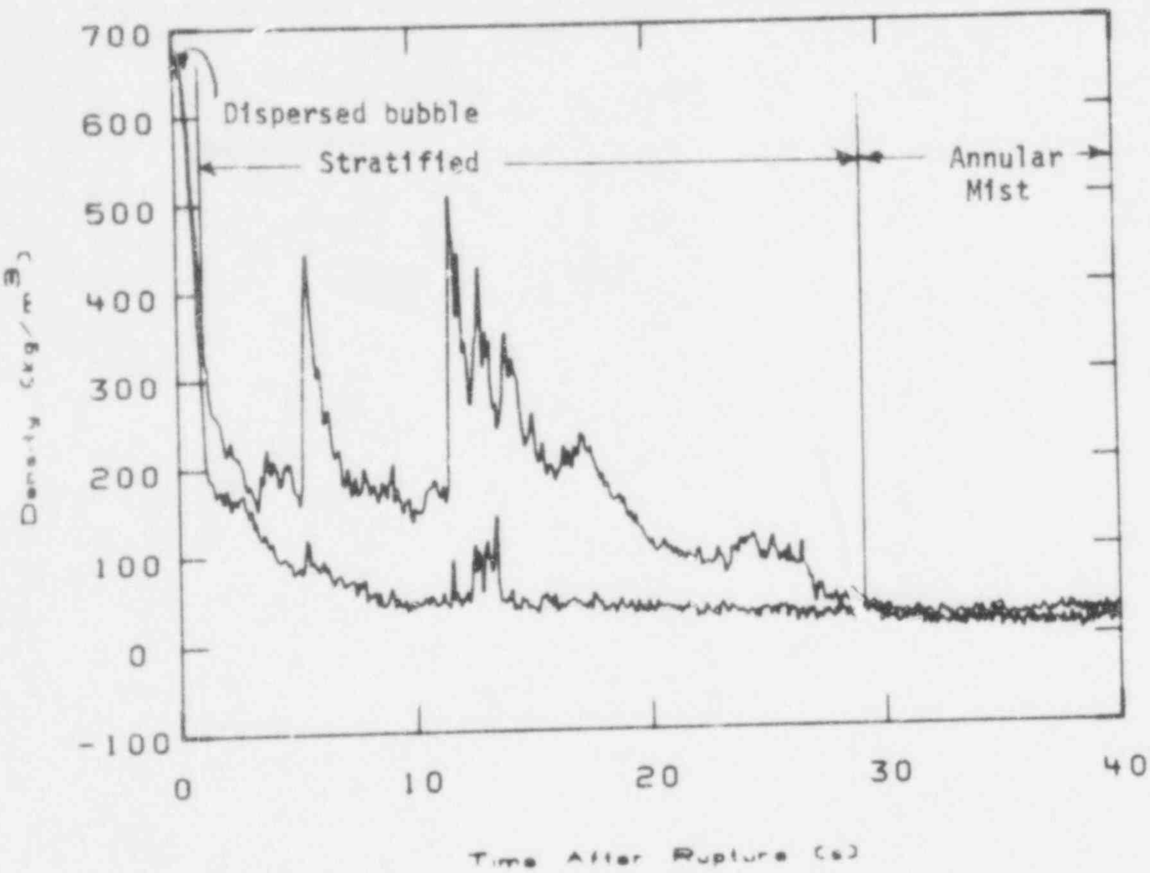

Fig. 37 Semiscale intact loop piping flow regimes.

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2.2 Steam Generators

The steam generators in the Semiscale Mod-3 system have the potential to directly influence the transient during both the blowdown and reflood portions of a LOCA experiment. Since this influence can occur in either the intact or broken loop, the Semiscale Mod-3 system includes active steam generators in both. To provide maximum flexibility in the Semiscale Mod-3 system, a steam generator referred to as the Type II steam generator (see Figure 38), has been designed to be interchangeable between the two loops. This flexibility has been achieved by sizing the unit to remove 100% of the core power (2 MW) during steady state operation. In this way, the Type II steam generator can be used in the intact loop with hydraulic resistance simulators in the broken loop, or by plugging the appropriate number of tubes, the steam generator can be modified for use in the broken loop with a second steam generator in the intact loop.

Although the Type II steam generator has been designed for installation in either the intact or broken loop, fabrication of a Type II steam generator for the intact loop is not yet complete. However, to facilitate the earliest possible testing schedule, a Type II steam generator has been installed in the broken loop with the existing Type I steam generator remaining in the intact loop. The Type I and Type II steam generators are similar in concept except that the Type I steam generator, shown in Figure 39, is scaled after the LOFT steam generator and, therefore, is much shorter in length than the Type II steam generator, which is scaled to match elevation

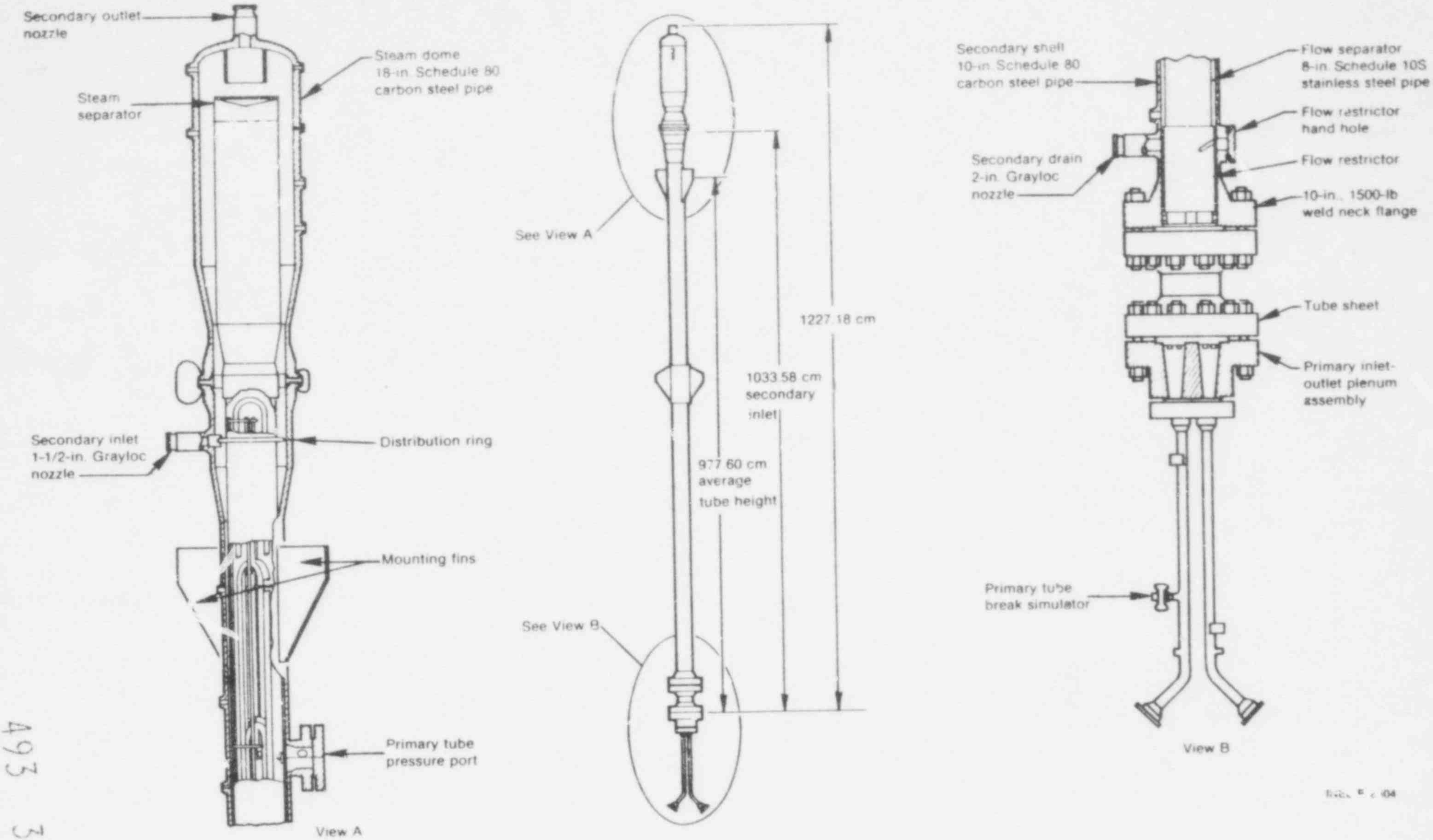

Fig. 38 Semiscale Mod-3 broken loop steam generator.

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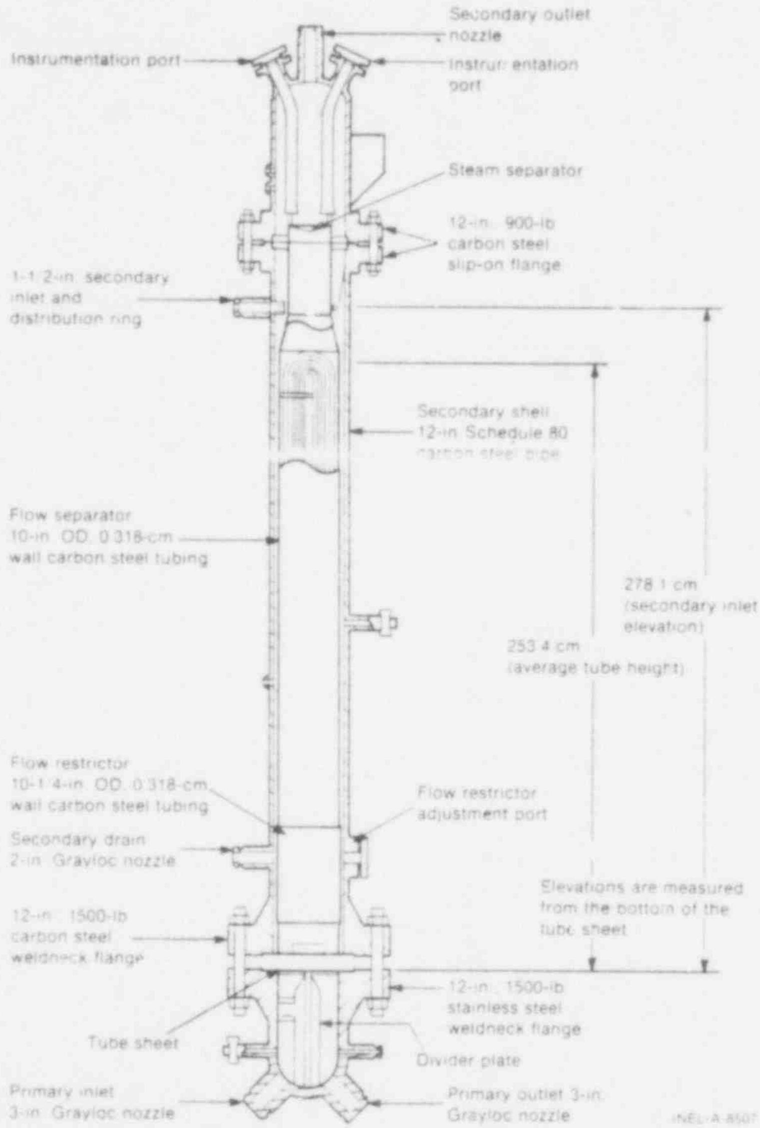

Fig. 39 Semiscale Mod-3 intact loop steam generator.

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effects in a PWR steam generator. Both steam generators are tube and shell designs in which the primary fluid passes through vertical "U" shaped tubes and the secondary coolant with resultant steam generation passes through the shell side. While the differences in the two steam generator designs will influence the loop hydraulics somewhat, this influence is expected to be secondary in nature. Therefore, the remainder of this section will deal specifically with design considerations for the Type II steam generator. For further information relating to the design of the Type I steam generator, which is scaled to the Loss-of-Fluid Test (LOFT) system[24], the reader is referred to Reference 2.

The scaling rationale used in deriving the Type II steam generator requirements is based on producing heat removal during the blowdown and heat addition during the reflood phases of a LOCA transient representative of those expected in a PWR. To accomplish this objective, the number and overall geometry of the U-tubes were selected to provide the best possible representation of specified elevations and scaled surface areas and volumes. Installation of the Type II steam generator nozzles, plenums, tube sheets, and U-tubes in either the intact or broken loop was, therefore, such that elevations would be preserved so that full-scale hydrostatic modeling effects in the Semiscale Mod-3 system would be similar to those in a PWR, see Table I. In addition, the same tube stock (2.22-cm outside diameter by 0.124-cm wall thickness) and tube spacing (3.175-cm triangular pitch) used for PWR U-tubes was used in the Type II steam generator.

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Since the heat transfer surface area was based on the ratio of PWR to Semiscale Mod-3 primary side system volumes, he number of tubes was fixed by the specified tube diameters and lengths.

The incorporation of the Type II steam generator into the broken loop required 9 of the 11 tubes to be plugged to reduce the primary system volume to as close to the scaled value as possible. By using the PWR length tubes, the volume is 2% lower than ideal and the heat transfer surface area is 10% smaller than the desired value. The pressure difference across the steam generator will remain close to that expected in a PWR since the tube size and length used in the Type II steam generator are identical to those in a PWR steam generator. However, since the steam generator secondary side was sized to remove 100% of the Semiscale Mod-3 system heat load, excess secondary volume will result when tubes are plugged. Therefore, when used in the broken loop, the Type II steam generator secondary volume is 90% greater than desired. As indicated from the analysis in Reference 19, this should not effect the Semiscale Mod-3 blowdown transient.

If the Type II steam generator were installed in the intact loop in conjunction with a second Type II steam generator in the broken loop, all but six tubes in the intact loop steam generator would be plugged. This would result in scaled parameters very close to the PWR values, with the volume being approximately 5% larger than the PWR value and with the heat tranfer surface area being 3% lower than desired. The pressure loss across the steam generator in this case would be nearly identical to the PWR value. This above discussion indicates the success that can be achieved by scaling from PWR dimensions. However, the secondary side volume would still be approximately 40% greater than desired.

In conclusion, the design of the Type II steam generator is scaled from PWR dimensions, while the Type I is scaled from LOFT. The differences between the two steam generators as well as the compromises inherent to each one are considered to be secondary in nature and therefore not expected to influence system response during blowdown. The steam generator characteristics during reflood are as yet to be determined and require further testing to gain the desired data for comparison to expected PWR behavior.

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2.3 PUMPS

The Semiscale Mod-3 system incorporates active pumps in both the intact and broken loops to establish an initial flow distribution typical of that in a PWR. As a result, the design for each pump was in part determined by steady state operating requirements to insure that desired initial conditions would be met. However, pump operating characteristics are also important during both the blowdown and reflood portions of a LOCA.

The Semiscale Mod-3 intact loop pump is the same pump used in the Semiscale Mod-1 system, but a larger impeller was installed to develop the hydraulic head needed to overcome the increased resistance of the Semiscale Mod-3 system. The pump is rated for a nominal flow of 22.6 l/s at a total head of 1336 kPa when operated at a rated speed of 367 rad/s. The pump is designed for a maximum pressure and temperature of 17240 kPa and 616 K, respectively, and has a specific speed of 930. The broken loop pump was designed for operation at very high speeds. The pump operating conditions are 2390 kPa head and 4.43 l/s flow at 2095 rad/s with a specific speed of 1550. In designing the intact and broken loop pumps for operation in the Semiscale Mod-3 system, specific speed, pump flow capacity, head, locked rotor resistance, minimum flow area, and two-phase flow characteristics were all important considerations in the final design selection. These design considerations will be discussed in the following paragraphs.

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During blowdown, the cold leg break demand is made up of flow from both the core and the intact loop. Therefore, initial pump operating characteristics can affect the amount of intact loop fluid delivered to the break and, as a result, influence core hydraulics and overall core thermal response early in time. However, during reflood the simulation of a locked rotor pump becomes the biggest resistance to flow in both the intact and broken loops and can, therefore, influence overall reflood behavior.

To simulate overall pump behavior, it is normally desireable to design the scaled pump to be geometric similar to the pump being modeled. A necessary requirement of geometrically similar pumps is that their specific speeds must match. In a system such as Semiscale Mod-3, in which full-scale differential pressures are maintained but flow rates are scaled, the design requirements for the scaled pump are difficult to obtain. This is evident upon inspection of the formula for specific speed, where if pump head (H) was held constant and the flow rate (Q) was decreased, the pump speed (N) would have to increase to keep specific speeds (N_s) identical as follows:

$$N_s = \frac{NQ^{1/2}}{H^{3/4}} .$$

This is the situation in both the intact and broken loop Semiscale Mod-3 pumps, where the head was required to be full scale while the flow rate was scaled down. As a result of this, the required speed to maintain identical specific speed was beyond the

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present material limits of the pumps. Therefore, specific speed and pump similarity were compromised in order to provide the head and flow rate needed to establish initial prerupture conditions.

The pump head and flow rate are system-imposed requirements, necessary to match initial steady state flow conditions prior to rupture. Semiscale Mod-3 tests utilizing a 3.66-m heated core have approximately the same pressure distribution as that expected in a PWR, hence the pump head is the same as that expected in PWR pumps. Therefore, the specified pump head and flow capacity curve shape should be similar to that in a PWR to preserve the relationship between shutoff head and runout flow as exists in the PWR pump. This criteria preserves the relationship of pump head and of design flow rates that occur in PWR systems. The pump H-Q curve shape (that is, zero head and zero flow intercepts) is of significance in the intact loop pump during the blowdown phase of a LOCA and may be important during reflood with pump suction leg ECC injection. As the system blows down, the quality of the fluid passing through the pump increases. Therefore, the ability of the pump to generate head decreases until the pump head degrades to zero and no longer influences loop flow. This characteristics decrease in pump head with increasing pump suction inlet fluid quality is demonstrated in Figure 40, which compares the calculated fluid quality and measured broken loop pump head for one of the tests in the Semiscale Mod-3 baseline test series. This figure indicates that pump operating characteristics and their effect on loop hydraulics are primarily important during the initial 30 s of the blowdown transient when the

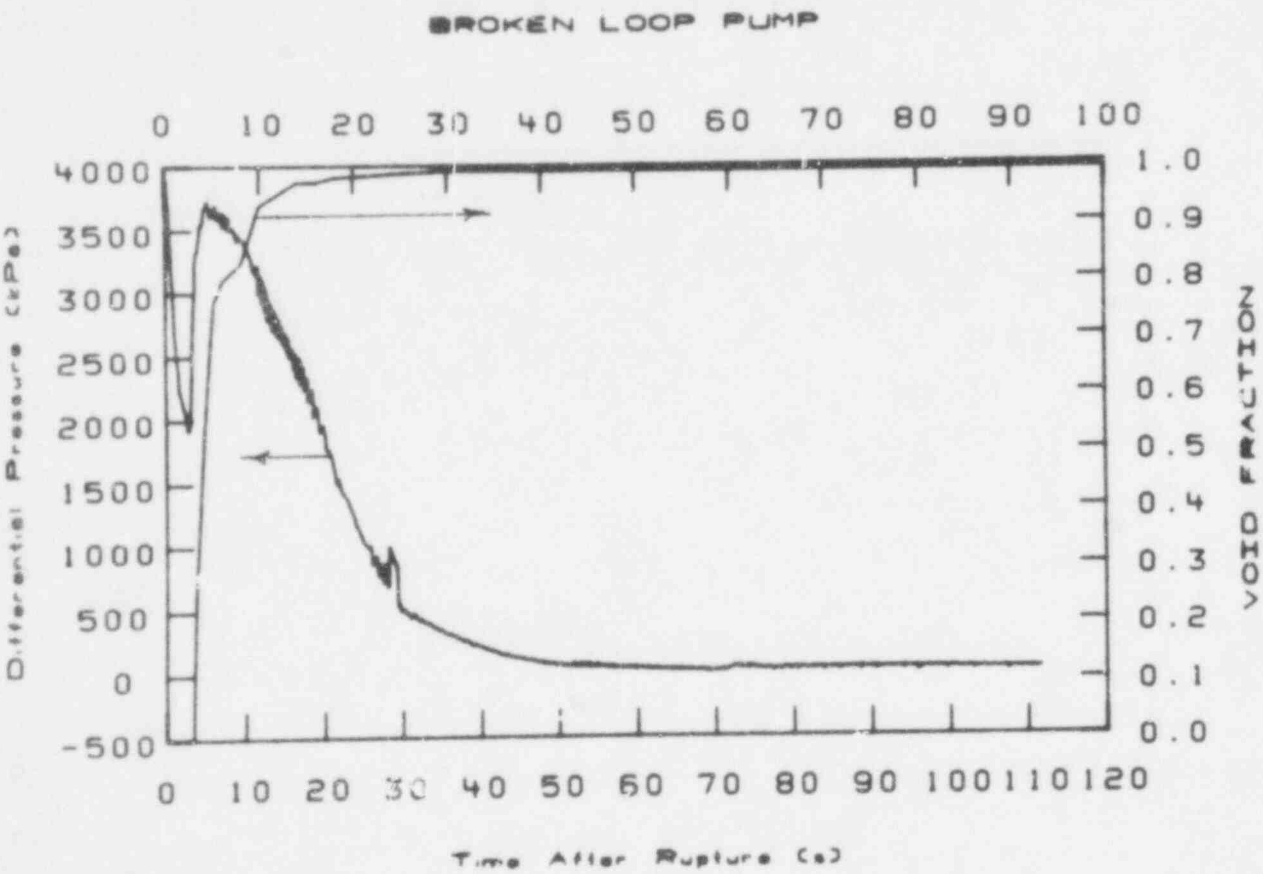

Fig. 40 Pump head and calculated fluid void fraction for broken loop pump.

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pump head has not yet degraded to zero. During this period, the Semiscale Mod-3 pumps should behave similar to their PWR counterparts. However, because of excessive frictional torque in the small Semiscale Mod-3 pumps, power controllers are required to control pump speed to insure that proper coastdown characteristics are achieved. The intact and broken loop pump speeds during blowdown have been specified to approximate a reference plant coastdown resulting from pump power trip simultaneous with break initiation. Also the pump speed controllers can provide various speeds, within design limitations, to simulate the different operating conditions of a PWR pump, thus increasing the flexibility of the Semiscale Mod-3 pumps.

During the reflood period, the locked rotor flow resistance in both the intact and broken loop pumps is a significant source of back pressure governing the core reflood rate. The scaling rationale in this case was to provide a locked rotor resistance that produces the PWR pump locked rotor pressure differential at the nominal (scaled) system flow rate. The flow rate is core-area scaled which is also power-to-volume scaled in the case of the 3.66-m Semiscale Mod-3 core. The ratio of PWR pump minimum flow area to core flow area should also be maintained to preserve possible flow choking phenomena during reflood. Therefore, the minimum flow area in the broken loop pump (0.97 cm^2) was calculated to give the same ratio of pump flow area to core flow area as occurs in a typical PWR plant. The

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calculation is based on a minimum PWR pump flow area of 0.164 m^2 and was ratioed using PWR and Semiscale Mod-3 core flow areas of 4.84 m^2 and 28.56 cm^2, respectively.

The minimum flow area of the intact loop pump (7.92 cm^2) is 2.73 times bigger than the scaled flow area of three PWR pumps. As mentioned previously, this enlarged flow area could alter the system reflood behavior somewhat. However, variations in the intact loop resistances have been shown not to have significant effects on reflood behavior and almost no effect on blowdown behavior[25] Therefore these slight compromises in the intact loop pump flow area are not expected to effect the Semiscale Mod-3 system capability to simulate the UHI process.

In conclusion, the active Semsicale Mod-3 pumps were designed to give the head and flow needed to meet desired PWR initial conditions. As a result, specific speed scaling was not able to be achieved. Each pump has a locked rotor resistance to provide reflood back pressures representative of those expected in a PWR. However, due to high frictional torque of the small pumps, power has to be applied to the Semiscale Mod-3 pumps to simulate PWR behavior during blowdown. By controlling pump speed during blowdown and providing locked rotor resistance during reflrod, Semiscale Mod-3 pump behavior is expected to produce operating characteristics similar to those expected in PWR pumps.

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3. ECC INJECTION SYSTEMS

The intent in the Semiscale Mod-3 Program has been to design a high level of flexibility into the ECC subsystems such that injection locations, configurations, flow rates, subcooling, and pressures can be varied to experimentally investigate important parameters related to ECC performance. This flexibility helps evaluate philosophies used in ECC scaling.

The ECC systems included in Semiscale Mod-3 are the UHI accumulator system, the intact and broken loop accumulator injection systems, and the high-pressure injection system (HPIS) and low-pressure injection system (LPIS), which are pumped injection systems. The HPIS and LPIS are initiated at 12.4 and 1.03 MPa, respectively, and continue providing constant average scaled flow rates until terminated. The more complex UHI and accumulator ECC systems provide variable injection rates, which are dependent on system depressurization characteristics, and the injection time is determined by scaling the injected fluid volume from a PWR.

The total volume of accumulator water injected into the intact loop should be approximately three times the amount injected into the broken loop to represent the scaled injection into the three unbroken loops of a PWR. However, results from the Semsicale Mod-3 baseline test series (Series 7) have indicated that an average hot wall delay of 8 s will occur in the Semiscale Mod-3 system. To allow for accumulator ECC bypass out the cold leg break and insure that the desired volume of ECC was available for lower plenum refill and core 493 334 reflood, an additional amount of water (12.5 l) was placed in the intact loop ECC accumulator to account for this delay. The combined total injected volume from the Semiscale Mod-3 intact loop accumulator is therefore 57.5 l $\pm$ 5% . After all the ECC fluid has been depleted from the intact and broken loop accumulators, a scaled volume of nitrogen is allowed to be injected into the system. However, with UHI, fluid injection is stopped after a specified volume of liquid has been injected into the vessel, and no nitrogen is allowed to enter the upper head region. To obtain average accumulator ECC injection rates in Semiscale Mod-3 representative of the expected values scaled from a PWR, the injection line resistances are scaled by the ratio of the square of the core powers:

$$R'_{SS} = R'_{PWR} \left[\frac{P_{PWR}}{P_{SS}}\right]^2.$$

The Mod-3 system resistances are calculated in terms of the parameter, R', which is defined:

$$R' = \frac{\rho \Delta P}{\dot{m}^2}$$

where

ρ = Fluid density, kg/m^3

$\Delta \rho$ = Pressure drop, Pa = $\left(\frac{kg\text{-}m}{s^2} \middle/ m^2\right)$ 493 335

$\dot{m}$ = Mass flow rate, kg/s

The accumulators are then pressurized to operating pressures corresponding to the accumulator operating pressures in a PWR. The scaled accumulator injection rates will then be similar to the expected injection rates in a PWR, if the relative pressure differentials between the accumulators and the primary systems remain the same as the systems depressurize. However, differences in the accumulator gas expansion characteristics or in the system depressurization characteristics can influence the accumulator injection rate characteristics. To address this concern, a computer code was used to calculate the expected accumulator injection rates into the Semiscale Mod-3 upper head and intact loop cold leg. The calculation was performed using scaled accumulator line resistances and a system depressurization characteristic representative of that expected in the Semiscale Mod-3 system. The results of this investigation are compared with a Westinghouse prediction[26] of the UHI and cold leg accumulator injection rates for a full sized PWR in Figure 41. So that a direct comparison of the injection rates could be made, the calculated Semiscale Mod-3 results have been multiplied by the ratio of core powers (3411/2). The comparison of injection rates in Figure 41 shows generally good agreement between the Semiscale Mod-3 and Westinghouse results. The differences in magnitude and timing which did occur are believed to be primarily due to slight differences in system depressurization characteristics or to differences in the assumed expansion characteristics of the accumulator Nitrogen (the Semiscale Mod-3 calculations assumed an isentropic expansion). However, the completion of accumulator ECC

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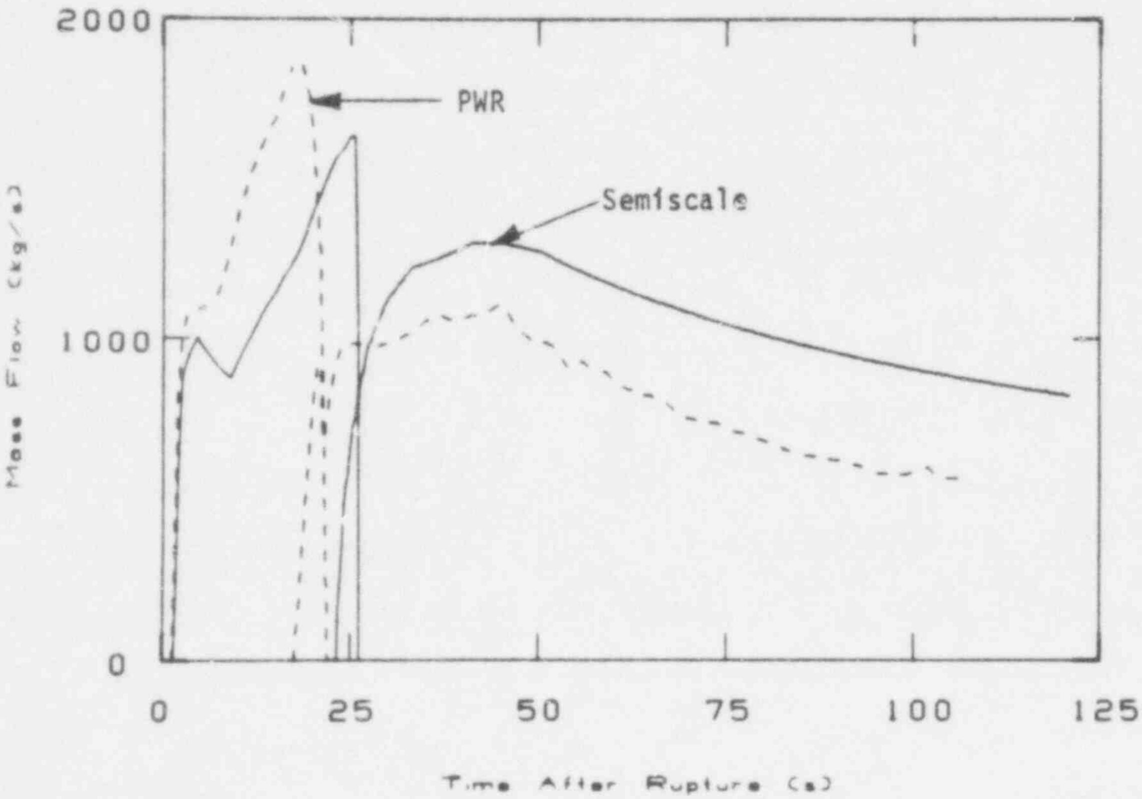

Fig. 41 Scaled Mod-3 accumulator flowrates and PWR accumulator flow rates.

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injection into the upper head and the start of cold leg ECC injection in Semiscale Mod-3 were within 4 s of those predicted by Westinghouse, and the sequence of events (with the start of cold leg injection occurring slightly before the completion of UHI) were very similar for the two calculations. Based on the results of these calculations, it appears that a reasonable representation of the accumulator injection characteristics in a PWR can be obtained with the present Semiscale Mod-3 accumulator ECC subsystems.

The Semiscale Mod-3 pumped injection systems, unlike the accumulator systems, are forced injection and therefore deliver a constant injection rate which is independent of pressure. The Semiscale Mod-3 pumped injection systems differ from their PWR counterpart in that the Semiscale Mod-3 pumps are constant speed positive displacement gear pumps while in the PWR the pumps are centrifugal and therefore are sensitive to system depressurization characteristics. As a result, the Semiscale Mod-3 pumped injection systems will not simulate the mass flow versus time behavior of the injection systems in a PWR, but instead, inject at a constant rate of approximately the average scaled mass flow. However, because the Semiscale Mod-3 pumped injection rates are small relative to the accumulator injection rates, the differences in mass flow may cause some distortions in the condensation process but probably not a significant amount. Therefore, the effects on system behavior caused by atypicalities in the Semiscale Mod-3 pumped ECC injection characteristics are expected to be secondary influences.

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In summary, the ECC injection systems in the Semiscale Mod-3 system can supply water to the vessel and piping at a variety of flow rates, water temperatures and injection locations. The volume of water injected is scaled from PWR values, and the majority of the ECC flow rates are also scaled from PWR values; however, the pumped ECC injection rates are average rates scaled from the PWR flow rates and may introduce some secondary distortions in system behavior. With the flexibility designed into the Semiscale Mod-3 ECC systems, a more comprehensive analysis of ECC behavior can be evaluated which will benefit not only the Semiscale Mod-3 Program but the nuclear safety community also.

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VI. CONCLUSIONS

The basis for scaling the Semiscale Mod-3 system has been presented and the system capabilities and limitations have been placed in perspective relative to a pressurized water reactor with upper head injection. The basic scaling principles have been identified and the influence of compromises made in component selection has been qualitatively evaluated. Specific conclusions relative to system capability and limitations include:

(1) Previous experience with integral-type loss-of-coolant experiments (LOCE) tend to confirm that the Semiscale Mod-3 system will preserve major LOCE thermal-hydraulic events expected to occur in the PWR system in an appropriate time frame. The Semiscale Mod-3 system is expected to reproduce the phenomena occurring in a PWR during a postulated LOCA such as saturated blowdown decompression rates, fluid density, flow rates, and pressure drops in the operating and blowdown loops, provided the PWR pump head degradation with void fraction is similar to that measured for the Semiscale Mod-3 intact loop pump.

(2) Differences in performance between a PWR and Semiscale Mod-3 are expected where two- and three-dimensional system effects influence controlling phenomena. Some particularly important phenomena influenced by two- and three-dimensional effects are core thermal performance and vessel lower plenum

liquid level, upper head mixing, and downcomer countercurrent flow. The Semiscale Mod-3 system can not simulate the two- and three-dimensional temperature and flow distributions of the PWR core or the two- and three-dimensional velocity distributions of the PWR vessel downcomer and plenum regions.

(3) The stored energy in the metal structure of Semiscale Mod-3 is greater per unit volume of system fluid than that in a PWR, and as a result, the released energy may adversely effect system response to some degree. However, various types of insulation with low thermal capacitance and low thermal conductivity have been incorporated to reduce these effects.

The material presented in this document indicates that Semiscale Mod-3 data are of primary value for model development and code verification activities and will provide basic insight into the UHI process in a PWR so equipped. However, the scaling compromises associated with any small-scale design, such as Semiscale Mod-3, makes direct extrapolation to a full-scale PWR design difficult. Therefore, the results obtained from Semiscale Mod-3 relate specifically to the Semiscale Mod-3 system and should be considered in that context.

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APPENDIX A

DESCRIPTION OF SEMISCALE MOD-3 ON-LINE

CORE POWER CONTROL SYSTEM

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APPENDIX A

DESCRIPTION OF SEMISCALE MOD-3 ON-LINE CORE POWER CONTROL SYSTEM

1. INTRODUCTION

This appendix describes a new method of core power control developed for use in the Semiscale Mod-3 system. The new method of power control, referred to as on-line core power control, attempts to simulate the temperature response characteristics of a nuclear fuel rod by calculating the required real-time power decay of an electrically heated rod using thermal-hydraulic conditions measured in the core during a test.

The on-line core power control system was developed to eliminate the need for predetermining the core power profile before each Semiscale Mod-3 test. In the past, the power profile has been predetermined, using computer calculation techniques, to determine the electrically heated rod power required to give the same surface heat flux as that calculated for a nuclear fuel rod operating in the same test environment. This open-loop control of core power was time consuming and assumed the computer calculation precisely emulated the behavior of the Semiscale facility during the test. The new on-line core power controller provides feedback control that functions

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independently of plant operating conditions. Therefore, the on-line core power control system will drive the Semiscale Mod-3 electrical heated rods in a manner that produces a local surface heat flux which matches the surface heat flux of a hypothetical nuclear fuel rod operating in the same thermal-hydraulic environment established in the Semiscale Mod-3 tests.

The following section describes the on-line power control system and discusses some of its important features.

2. CONTROL SYSTEM DESCRIPTION

The on-line core power control system software includes control algorithms, which are programmed on a PDP11/55 digital computer. The computer is equipped with 32 analog-to-digital converters and 4 digital-to-analog converters, which are used to accept inputs from the Semiscale Mod-3 electrical heated core and to provide the required driving signals. The computer has a 16-bit word length with 32,000 memory locations; in addition it has two removable disk packs with 1.2×10^6 word storage locations each. Access to the computer is provided through an LA-36 terminal or a 300 card/min card reader.

The computational philosophy of the on-line core power control program can be followed on the block diagram shown in Figure A-1. Figure A-1 shows the control program receives input from the analog-to-digital converters and the measured values of electrical heated rod power, electrical heated rod cladding temperature, and

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coolant temperature. The control program then calculates the surface heat transfer coefficient, the surface heat flux, and the radial temperature profile for the electrically heated rod. The calculated heat transfer coefficient is then applied to a model of a nuclear fuel rod, and in conjunction with a nuclear decay heat curve, the resulting surface heat flux and radial temperature profile are calculated. By taking the difference between the electrically heated rod and the nuclear fuel rod surface temperatures, a surface temperature error is determined and used as the input to the digital compensation scheme. The resulting output of the compensation algorithm is proportional to the desired core power level. The power supply is driven with an analog voltage, which is the result of a digital-to-analog conversion of the power demand produced in the digital program.

The finite-element method is used to model both the electrical heated rod and the nuclear fuel rod[A-1]. Up to 15 radial nodes can be selected for the electrically heated rod and up to 10 for the nuclear fuel rod. However, care must be taken to ensure integration stability by choosing a sufficiently small integration interval. The present model of the electrically heated rod has nine radial nodes. The location of the nodes and the thermal properties used in the model are shown in Figure A-2. The nuclear fuel rod is represented by seven radial nodes and is pictorially shown along with its corresponding thermal properties in Figure A-3.

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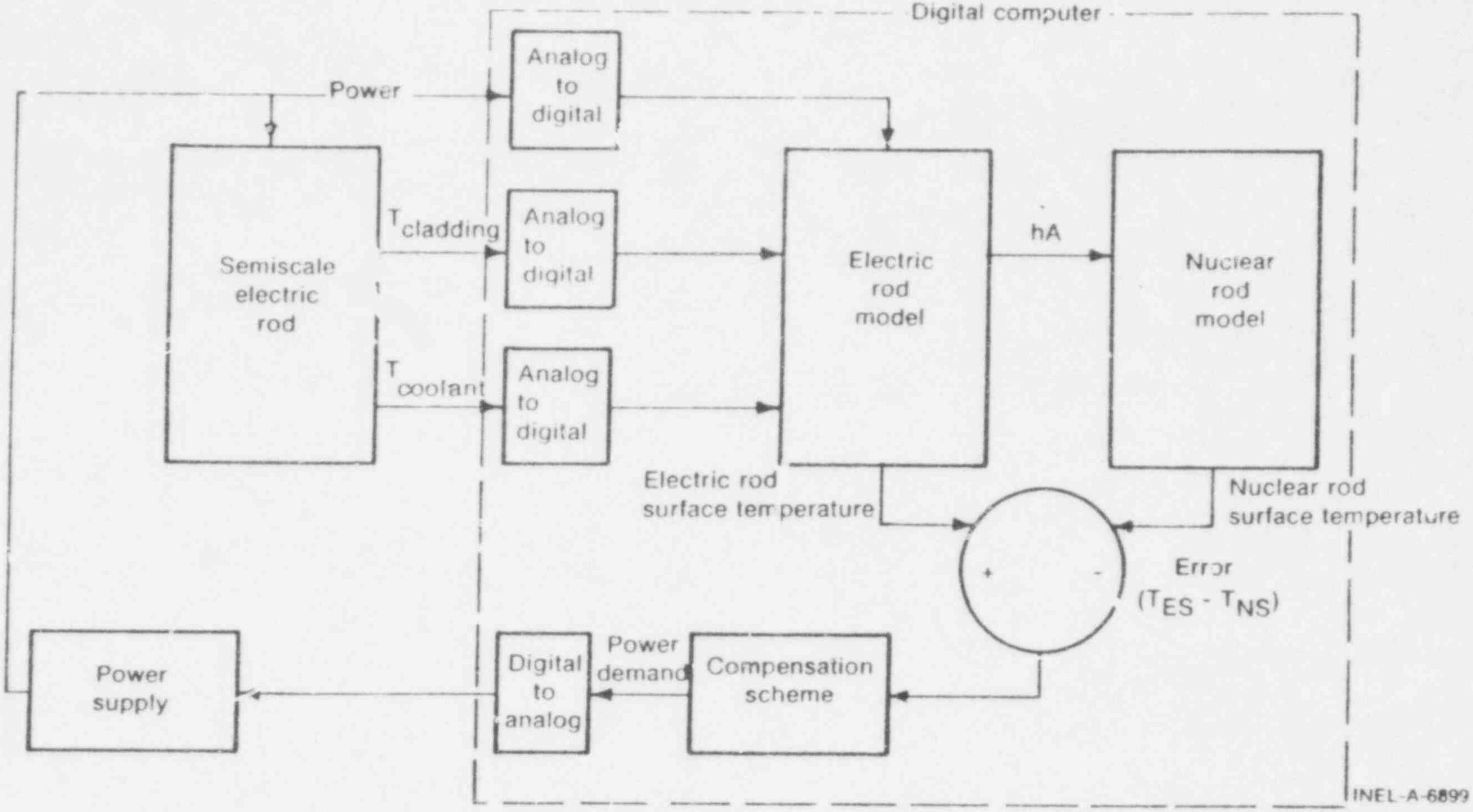

Fig. A-1 Schematic of on-line core power control functions.

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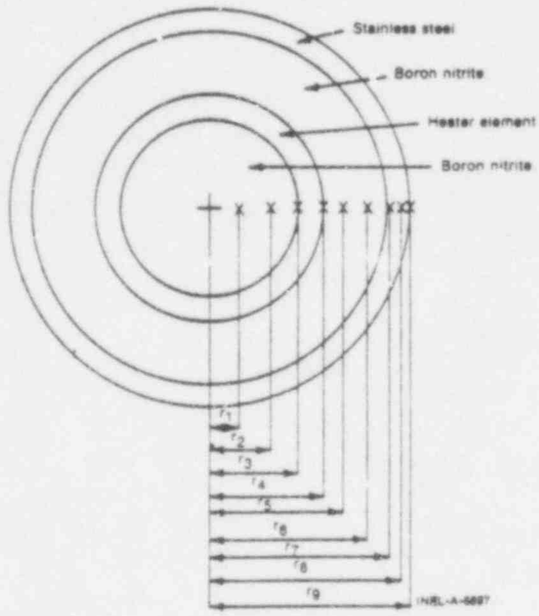
r_1=0.053 cm
r_2=0.1058 cm
r_3=0.1588 cm
r_4=0.24 cm
r_5=0.3056 cm
r_6=0.3713 cm
r_7=0.4369 cm
r_8=0.4699 cm
r_9=0.5359 cm



<u>Conductivity and Specific Heat (Staniless Steel):</u>

$$K_{ss}=13. + (1.5 \times 10^{-2})\ T \quad \left[W/_{m\text{-}K}\right]$$

$$C_{ss}=3.89 \times 10^{6} + (1.41 \times 10^{3})\ T \quad \left[W\text{-}s/_{m^3\text{-}K}\right]$$

<u>Conductivity and Specific Heat (Heater Element):</u>

$$K_{con}=24.9 + (1.66 \times 10^{-2})\ T \left[W/_{m\text{-}K}\right]$$

$$C_{con}=3.595 \times 10^{6} + (1.46 \times 10^{3})\ \ T \quad \left[W\text{-}s/_{m^3\text{-}K}\right]$$

<u>Conductivity and Specific Heat (Boron Nitride):</u>

$$K_{BN}=3.24 - (6.51 \times 10^{-3})\ T \quad \left[W/_{m\text{-}K}\right]$$

$$C_{BN}= 1.63 \times 10^{6} + (6.27 \times 10^{3}\ T - (5.52)\ T^{2} + (1.85 \times 10^{-3})\ T^{3} \quad \left[W\text{-}s/_{m^3\text{-}K}\right]$$

T=TEMPERATURE (^{o}C)

Fig. A-2 Electrical heater rod model.

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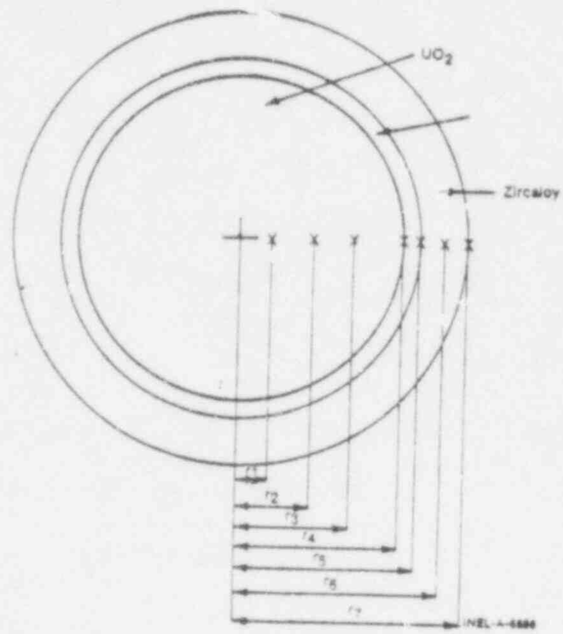
r_1=0.1173 cm
r_2=0.2346 cm
r_3=0.3519 cm
r_4=0.4692 cm
r_5=0.4742 cm
r_6=0.5051 cm
r_7=0.5359 cm



<u>Conducting and Specific Heat ($^{UO}2$):</u>

$$K_{UO_2} = \frac{3824}{(T + 129.6)} + 6.124 \times 10^{-11} T^3 \qquad \left[W/_{m\text{-}K} \right]$$

$$C_{UO_2} = 3.5 \times 10^6 \ \ W\text{-}s/_{m^3}\text{-}K$$

<u>Conductivity and Specific Heat (Zircaloy):</u>

$$K_{Zr} = 7.8476 + (2 \times 10^{-2})\ T - (1.676 \times 10^{-5})\ T^2 + (8.7 \times 10^{-9})\ T^3 \left[W/_{m\text{-}K} \right]$$

$$C_{Zr} = 1.57 \times 10^6 + (1.09 \times 10^3)\ T - (4.36 \times 10^{-2})\ T^2 \left[W\text{-}s/_{m^3}\text{-}K \right]$$

<u>Conductivity of Gap:</u>

$$K_{GAP} = 1 \times 10^4 \quad \left[W/_{m^2}\text{-}K \right]$$

T=TEMPERATURE (K)

Fig. A-3 Nuclear fuel rod model.

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3. SUMMARY

A computerized core power control system has been tested at the Semiscale facility and has been shown to be stable and functionally operational. The system is limited in response by the power supply time constant, the thermal time response of the electrical heated rods (that is, the heat capacity and conductance of the materials making up the electrical heated rods), and the computer program computational and sample interval (presently 100 ms).

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APPENDIX B

FLOODING CALCULATIONS FOR SEMISCALE MOD-3 AND MOD-1 DOWNCOMERS

APPENDIX B

FLOODING CALCULATIONS FOR SEMISCALE MOD-3 AND MOD-1 DOWNCOMERS

This appendix compares the counter-current flow limiting (flooding) behavior of the Semiscale Mod-3 downcomer pipe with an annular configuration such as was used in Semiscale Mod-1. The flooding correlations developed by Wallis and Makkenchery and discussed in Reference B-1 were used in this comparison.

The Wallis and Makkenchery correlations are based on the j* parameter, given by

$$j_g^* = j_g \rho_g^{1/2} \left[gD(\rho_f - \rho_g)\right]^{-1/2}$$

where

j_g = gas volumetric flux

ρ_g = gas density

ρ_f = liquid density

D = hydraulic diameter.

The preceding equation represents a ratio of inertial and gravitational forces and includes a characteristic dimension. Other correlations use the Kutateladze number, given by

$$K = j_g \rho_g^{1/2} \left[g \, g_c \, (\rho_f - \rho_g) \right]^{-1/4}$$

which represents a balance of inertial, gravitional, and surface tension forces and contains no characteristic dimension. K and j_g^* are related by

$$K = j_g^* \; D^{*1/2}$$

where D* is the dimensionless tube diameter, given by

$$D^* = D \left[g(\rho_f - \rho_g)/(g_c \, \sigma) \right]^{1/2}$$

where

σ = surface tension.

For small tube diameters (D* less than about 20), j_g^* is the controlling parameter; for the larger D*, K is the controlling parameter.

The point of interest here is the critical flooding condition, which occurs when the minimum gas flow prevents downflow of liquid in the downcomer. The critical flooding condition determines if emergency core coolant (ECC) penetration of the downcomer will occur. For small pipes, j_g^* is approximately constant at the critical condition. In air-water experiments with 1.27 and 2.54 cm tubes, Hagi et al[B-2] found that the critical conditions occurred at $j_g^{*1/2}$ = 0.725. This is in agreement with previous work by

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Wallis[B-1]. For larger diameter pipes, the Kutateladze number appears to represent a limit to the critical flooding velocity, beyond which the critical velocity does not increase with increased pipe diameter. The critical value is K = 3.2. In Semiscale counter-current flow tests conducted with an annular geometry, flooding was found to be independent of annulus gap size[B-3]. The Semiscale data was correlated by multiplying $j_g^{*1/2}$ by $D^{1/4}$ to remove the dimensional dependence. The critical value was found to be

$$j_g^{*1/2} D^{1/4} = 0.14\ m^{1/4}.$$

The Wallis critical flooding velocity for the Semiscale Mod-1 and Mod-3 downcomers is shown in Figure B-1 as a function of pressure for saturated steam and cold (300 K) ECC water. The Kutateladze critical flooding velocity is shown for comparison. The values used were

$j_g^{*1/2} D^{1/4} = 0.14\ m^{1/4}$ for Semiscale Mod-1

$j_g^{*1/2} = 0.725$ for Semiscale Mod-3

K = 3.2 for the Kutateladze curve.

The calculation indicates that the Semiscale Mod-3 downcomer pipe has approximately the same critical flooding velocity as the Semiscale Mod-1 downcomer annulus.

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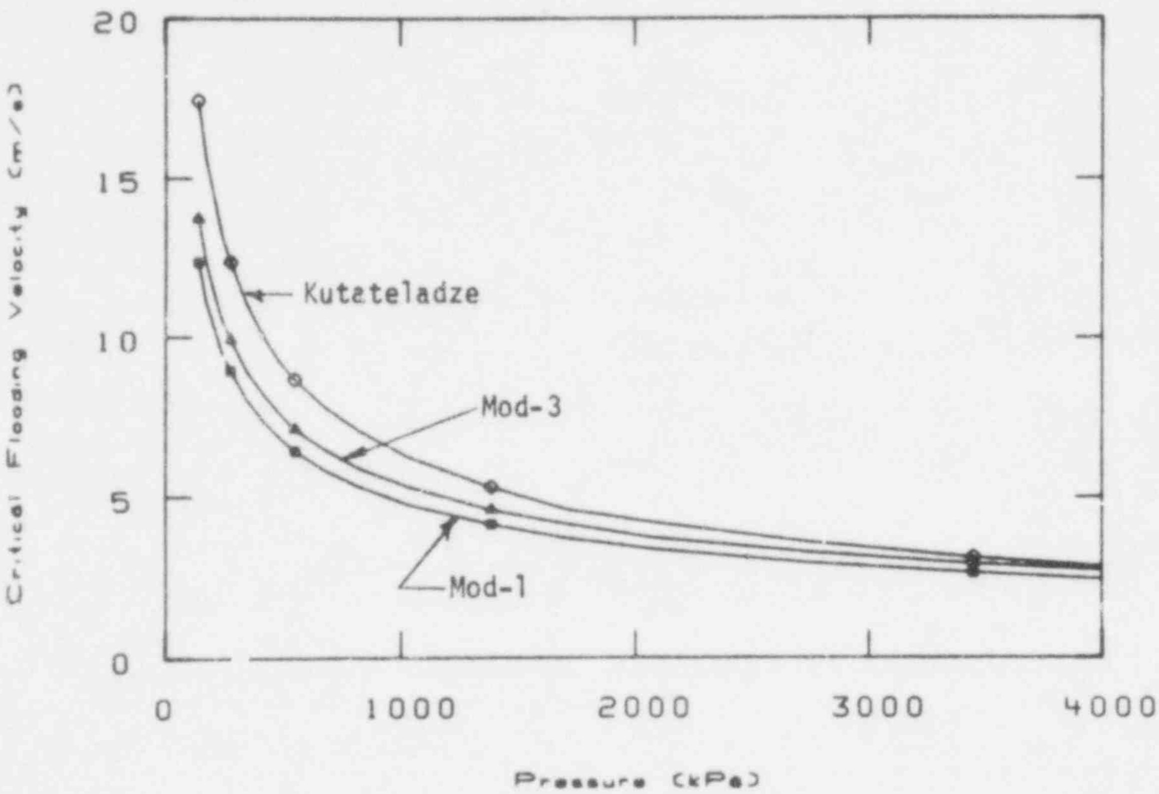

Fig. B-1 Critical flooding velocities in Semiscale Mod-1 and Mod-3 downcomers.

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APPENDIX C

INVESTIGATION OF DOWNCOMER MASS DEPLETION IN SEMISCALE MOD-3

APPENDIX C

INVESTIGATION OF DOWNCOMER MASS DEPLETION IN SEMISCALE MOD-3

This appendix discusses the expected causes of mass depletion in the Semiscale Mod-3 downcomer. Mass depletion has been defined as the explusion of water from the downcomer as a result of heat transfer to the downcomer fluid. In studying this phenomena, first observed in baseline test S-07-6[C-1], several tests were conducted that va[illegible] the downcomer insulator geometry in an effort to select the m[illegible]anism by which heat was being transferred from the downcomer walls to the system fluid.

This study began with the analyzing of Test S-07-6 data and deriving several suspected causes of the downcomer mass depletion phenomena. These suspected causes include: heat transfer to the downcomer liquid from hot downcomer metal structures, backflow of steam from the core into the downcomer, and the one-dimensional nature of the Semiscale Mod-3 downcomer. Each of these causes were then examined, and where possible, an integral test in the Semiscale Mod-3 system was performed to further evaluate the ability of each cause to contribute to downcomer mass depletion.

The results of this study indicated that heat transfer from the downcomer walls was the main contributor to downcomer mass depletion and recommended that a new insulator for the downcomer be installed to

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inhibit the heat transfer. As a result of this recommendation, a new insulator has been designed and is presently being built. Consequently, the new insulator has not been tested in the Semiscale Mod-3 system; therefore, the causes stated in this appendix for the mass depletion phenomenon have not been studied experimentally, and thus are still considered suspected causes.

1. EVALUATION OF TEST S-07-6 RESULTS

Test S-07-6 was the first integral blowdown-reflood test performed in the Semiscale Mod-3 system and was conducted from an initial system pressure of 15.6 MPa, a core inlet temperature of 557 K, and a core temperature rise of 37 K. The steady-state core power was 2 MW, and 23 of the 25 electrical heated rods in the core were powered. To simulate radial power peaking, the nine center rods were powered 13% higher than the remaining rods, resulting in high- and low-power rod peak power densities of 39.7 and 35.0 kW/m, respectively. Ambient temperature ECC fluid was injected into the intact loop cold leg using an accumulator, high-pressure injection system (HPIS), and low-pressure injection system (LPIS).

An evaluation of the results from Test S-07-6 indicates that the blowdown and refill response was similar to the respons[illegible]d during previous blowdown-refill tests conducted in the [illegible]e Mod-3 system. However, based on results of separate effects re[illegible]ood tests conducted in the Semiscale Mod-3 system and on results of integral blowdown-reflood tests conducted in the Semiscale Mod-1 system, the reflood behavior was considerably different than expected. The system

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hydraulic behavior during the reflood portion of Test S-07-6 was characterized by several periods in which refill of the downcomer and partial reflooding of the core was followed by a rapid reduction in both the downcomer and core liquid inventories. The filling and depletion process for Test S-07-6 is illustrated in Figure C-1, which compares the downcomer and core collapsed liquid levels (obtained from differential pressure measurements). Each decrease in the downcomer liquid level (and the corresponding drop in the core liquid level) coincided with an increase in fluid density in the vessel inlet side of the broken loop which indicates that the downcomer liquid was entrained and carried toward the break. Since a continuous reflooding of the core was not maintained, the core electrical heated rod cladding temperatures remained relatively high until late in the test. These high cladding temperatures are indicated in Figure C-2, which presents the cladding temperatures on electrical heated rod C3 (a high-power rod) at the 49-, 115-, 184-, and 230-cm elevations. A comparison of Figures C-1 and C-2 indicates that cladding temperatures tended to decrease during periods when the downcomer and core were refilling and increased during periods when the downcomer and core were emptying. The core peak power zone (144- to 213-cm elevations) eventually quenched at about 550 s after rupture.

The phenomena which led to the multiple depletions of the downcomer fluid during Test S-07-6 were not readily apparent based on an evaluation of the data from this test. However, several possible causes of the mass depletion were identified. These included: (a) heat transfer to the downcomer liquid from hot downcomer metal

IMAGE EVALUATION
TEST TARGET (MT-3)

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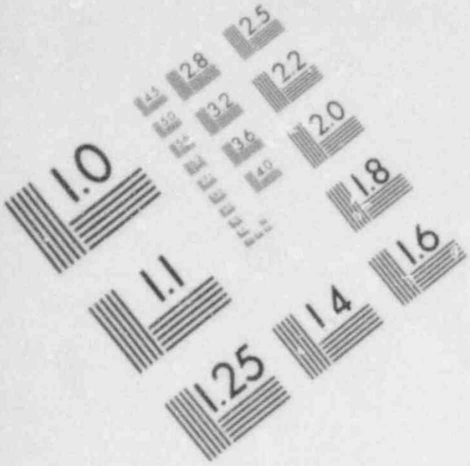
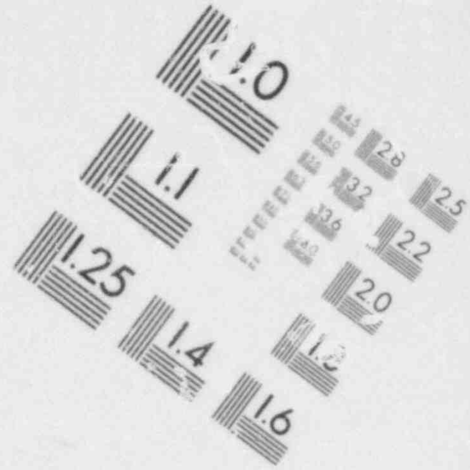
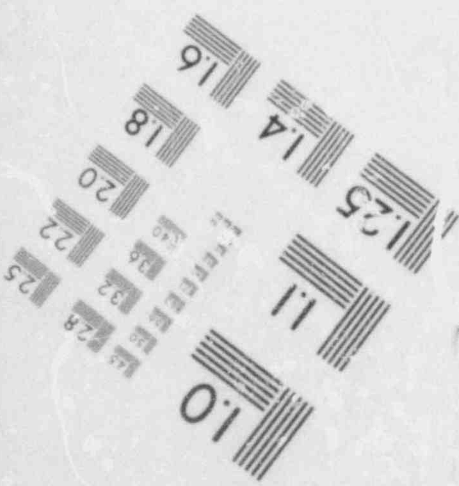
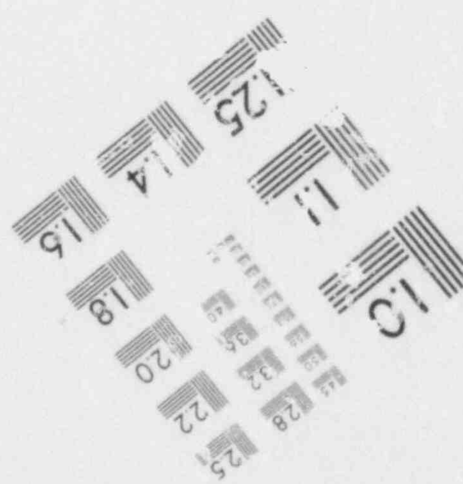



IMAGE EVALUATION TEST TARGET (MT-3)

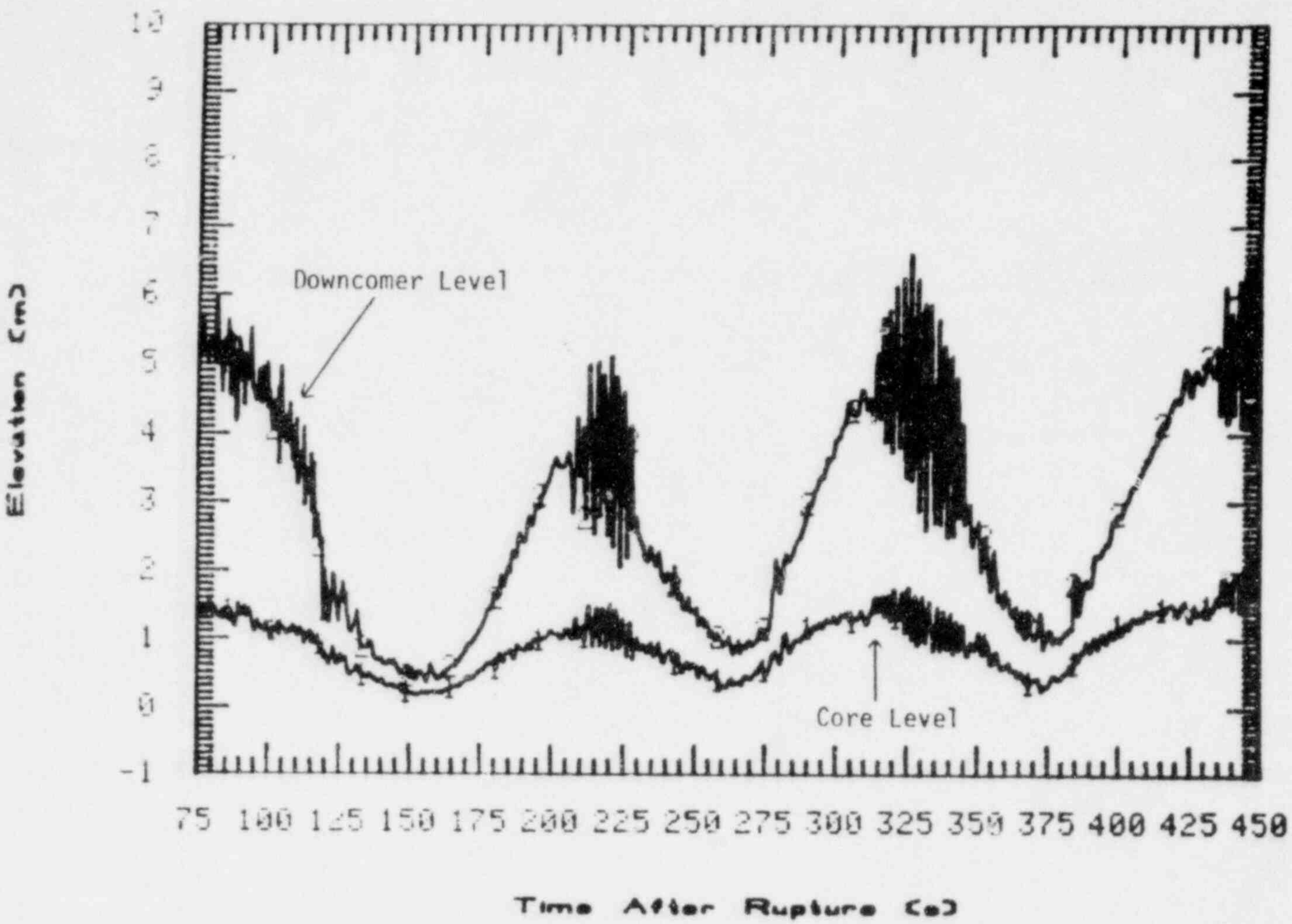


Fig. C-1 Comparison of the downcomer and core collapsed liquid levels for Test S-07-6.

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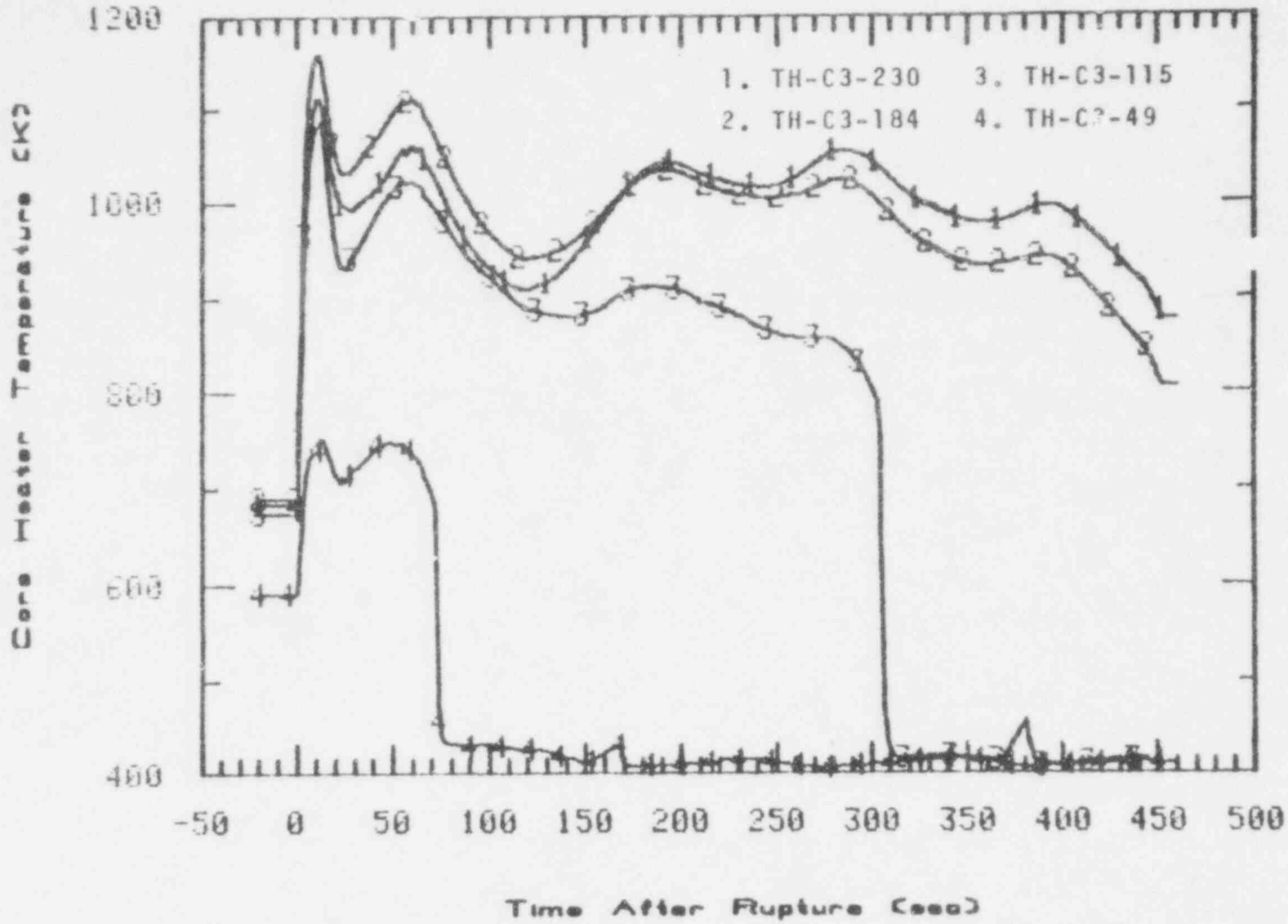

Fig. C-2 Axial temperature distribution in core for Test S-07-6.

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structures[a]; (b) backflow of steam from the core into the downcomer, and (c) the one-dimensional nature of the Semiscale Mod-3 downcomer. A discussion of each of these causes follows along with comparative data from other integral tests performed in Semiscale Mod-3 to help understand the mass depletion behavior.

Heat transfer to the downcomer liquid from the hot metal walls could result in downcomer mass depletion by causing the fluid to heat up and eventually boil. The resulting swell of the liquid-vapor mixture would then force liquid out the top of the downcomer causing a decrease in the effective downcomer pressure head. The potential for downcomer metal-to-fluid heat transfer in Test S-07-6 is illustrated in Figure C-3 which compares the downcomer metal temperatures and the fluid temperature at the 364-cm elevation (similar results were observed at other elevations throughout the downcomer). Although the change in stored energy of the downcomer metal, corresponding to a metal temperature decrease of the magnitude indicated in Figure C-3

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a. In the Semiscale Mod-3 system, the downcomer (which is external to the vessel) was designed to minimize hot wall effects and metal-to-fluid heat transfer. The Mod-3 downcomer is comprised of an outer heavy wall pipe (which provides the pressure boundary between the system and the atmosphere) and an inner thin wall pipe (which acts as a liner to provide the proper flow area). A gap between the outer pipe and the downcomer liner can be filled with an insulating material to alter the heat transfer characteristics. In Test S-07-6, the downcomer liner was wrapped with a commercially available insulating material (Grafoil), which is capable of withstanding the severe thermal-hydraulic conditions in the experimental environment. However, although the conductivity of Grafoil is quite low (3.46 W/m·k), the heat transfer rate from the outer downcomer pipe to the downcomer fluid was a factor of two to three times higher (on a surface area to volume ratio basis) than would be expected in a PWR downcomer under similar conditions.

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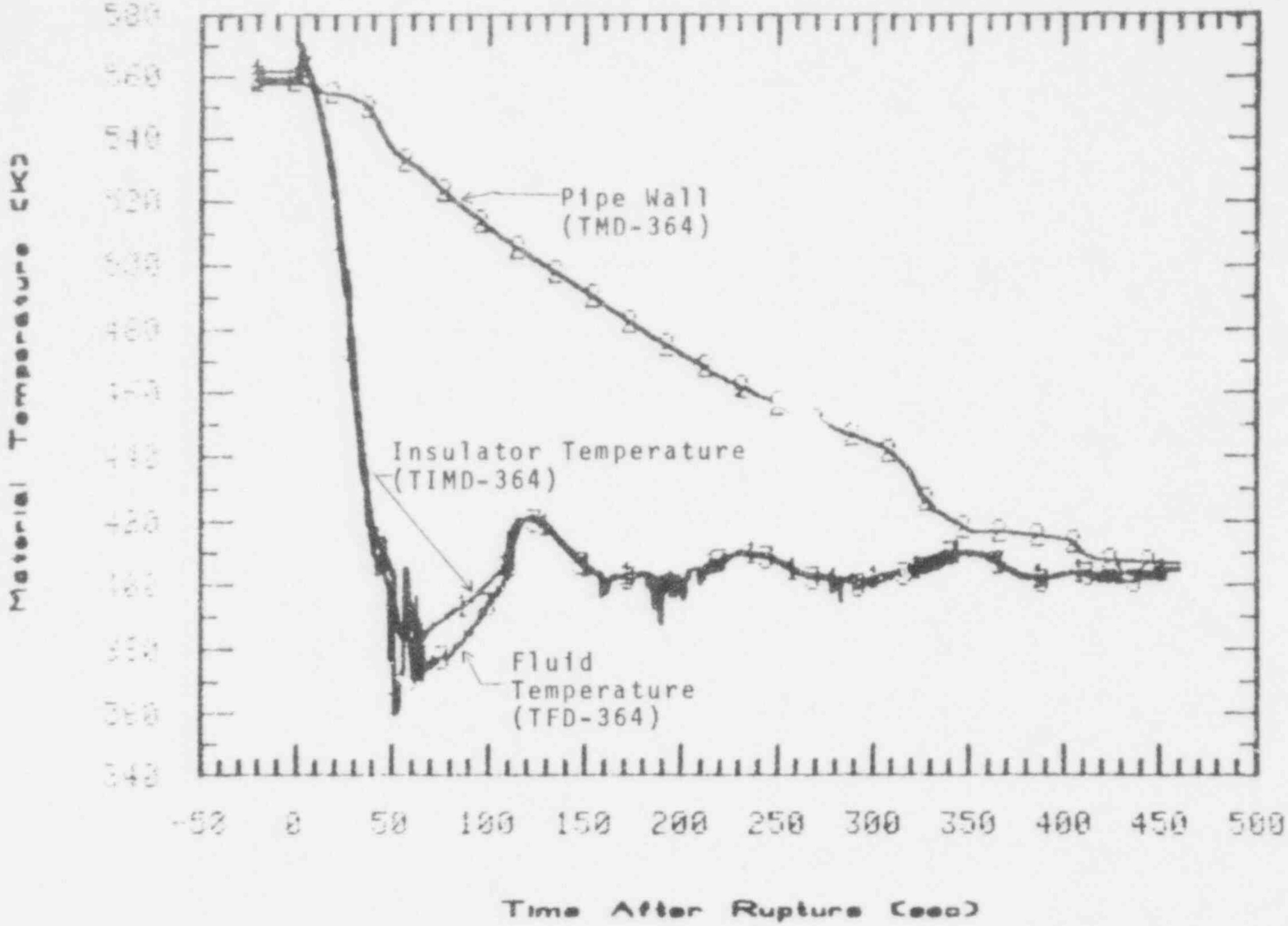

Fig. C-3 Temperature gradient across downcomer pipe during Test S-07-6.

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would not be sufficient to completely vaporize the entire downcomer liquid inventory, the resulting swell due to vaporization of even a relatively small percentage of the downcomer fluid could expel much of the remaining liquid. To examine the effects that this heat transfer would have on the system response, an integral blowdown-reflood test (Test S-B7-6) was conducted. In this test the downcomer was modified to decrease the potential for heat transfer to the system fluid. A discussion of this test and the results follow.

2. EVALUATION OF TESTS S-B7-6 RESULTS

Test S-B7-6 was performed with essentially the same initial boundary conditions as Test S-07-6, with the exception that the downcomer geometry was modified to reduce the downcomer metal-to-fluid heat transfer. As indicated previously, the downcomer metal-to-fluid heat transfer rate in Test S-07-6 was excessively high. Thus, in Test S-B7-6 an attempt was made to reduce the downcomer metal-to-fluid heat transfer. This reduction was to be accomplished by removing the Grafoil insulation which would provide space for a larger steam gap to be generated between the outer pipe and the downcomer liner[a]. A cross section of the downcomer before modification is shown in Figure C-4. Since the conductivity of steam is significantly lower than the conductivity of the Grafoil, it was expected that the heat transfer rate would be considerably lower during much of the reflood portion of the test than occurred during Test S-07-6. However, during

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a. A similar steam gap downcomer insulator technique was used with some success in the Semiscale Mod-1 system.

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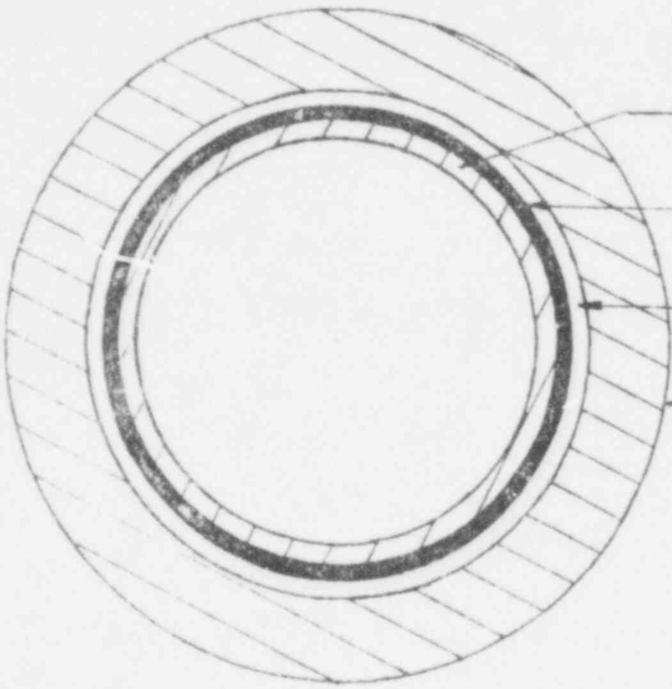

Fig. C-4 Downcomer cross section.

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the period immediately following the initiation of downcomer refill when considerable subcooling existed in the downcomer fluid, it was expected that steam in the insulator gap could condense, thus causing an order of magnitude increase in the metal-to-fluid heat transfer rate. The high heat transfer rate could continue until the downcomer fluid reached the saturation temperature. As a result, it was considered likely that depletion of the downcomer fluid would occur once. Following the original depletion, however, it was expected that the downcomer steam gap would be regenerated and would be maintained for the remainder of the test[a]. Thus, once the downcomer refilled under the influence of the LPIS (as occurred in Test S-07-6 after the original depletion), it was considered likely that the heat transfer to the downcomer fluid would be sufficiently low that a second depletion would not occur.

The refill-reflood behavior of Test S-B7-6 was characterized by an initial downcomer mass depletion response that occurred much as expected, followed by a refill response that was much slower than expected based on the LPIS flow rate. Figure C-5 presents the downcomer and core collapsed liquid levels for Test S-B7-6. As indicated in the figure, refill of the downcomer first occurred at about 45 s after rupture, and was followed by a rapid depletion at about 60 s after rupture. The initiation of nitrogen flow from the

a. In the Semiscale Mod-3 system, the flow of superheated steam in the intact loop during reflood is sufficiently high to remove most, if not all, of the subcooling from the LPIS fluid before it enters the downcomer. Thus, after the initial downcomer mass depletion, there would not be sufficient subcooled liquid entering the downcomer to cause the steam in the insulator gap to recondense.

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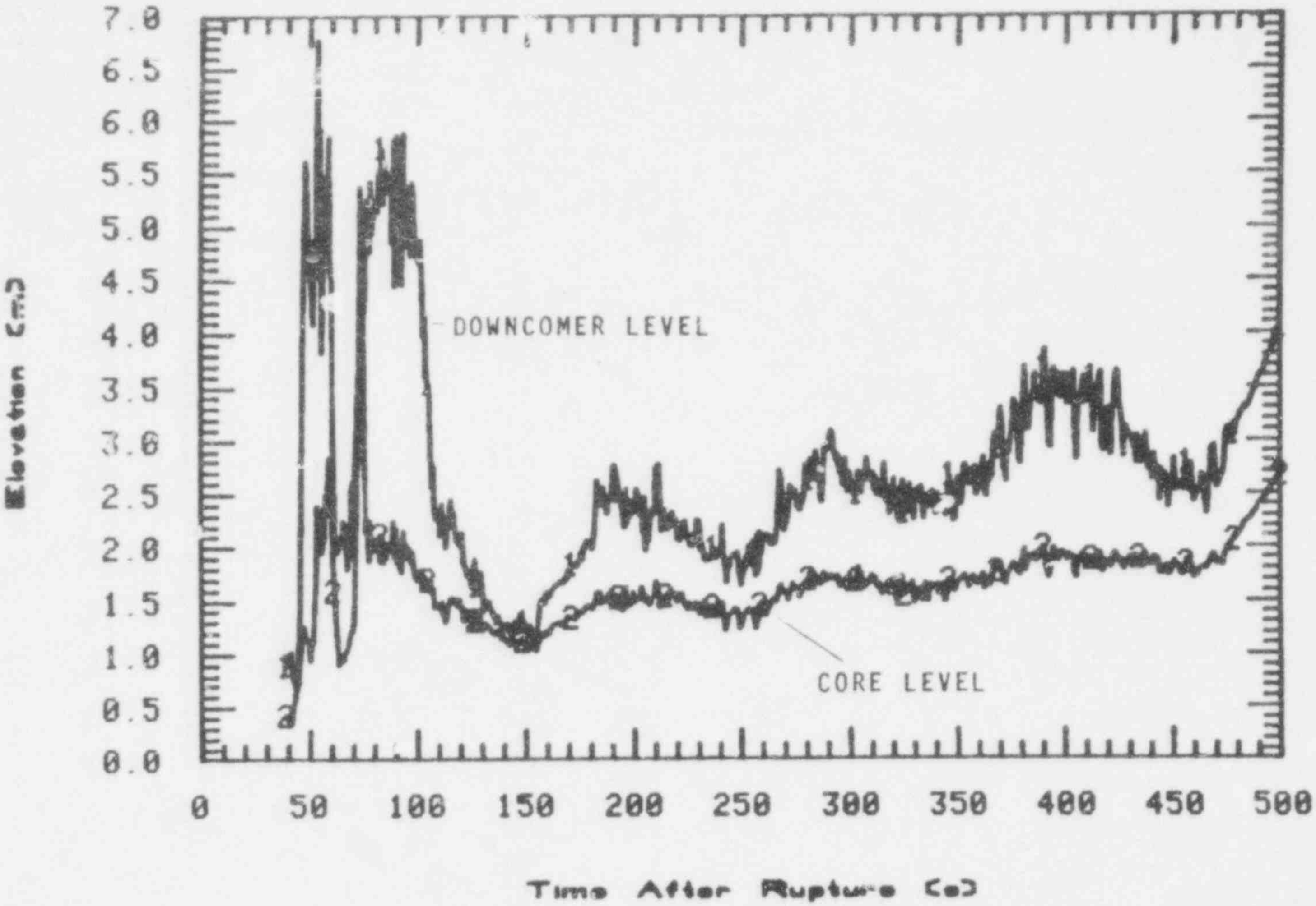

Fig. C-5 Core and downcomer collapsed liquid levels for Test S-B7-6.

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intact loop accumulator again forced liquid into the downcomer at about 70 s. A final rapid depletion of the downcomer fluid occurred at about 100 s after rupture. Between 150 and 475 s, a very slow refill of the downcomer was observed. Again, as was the case in Test S-07-6, the trend of the core liquid level response followed closely the trend of the downcomer liquid level response. An evaluation of the data from Test S-07-6 indicates that the downcomer mass depletion response that occurred prior to 120 s after rupture can be attributed directly to excessive downcomer heat transfer during this period and not to the effect of steam backflow from the core. Figure C-6 compares the downcomer metal temperatures and the fluid temperatures at the 364-cm elevation (similar results were observed at other elevations throughout the downcomer). The rapid decrease in the pipe wall temperature between approximately 45 and 100 s after rupture indicates that most of the energy stored in the outer downcomer pipe was transferred to the downcomer fluid during this period. (Recall that during this period it was expected that condensation of steam in the downcomer insulator gap would result in excessively high heat transfer rates.) The effect of this high rate of downcomer heat transfer on the mass depletion process is illustrated by comparing fluid temperatures in the downcomer and lower plenum with the fluid saturation temperature shown in Figure C-7 and by comparing the downcomer collapsed liquid level with the lower plenum diagonal density shown in Figure C-8. Several important points can be made using these figures. First, mass depletion began shortly after the fluid in the lower portion of the downcomer reached the saturation temperature. Figure C-7 shows that the fluid temperature in the lower

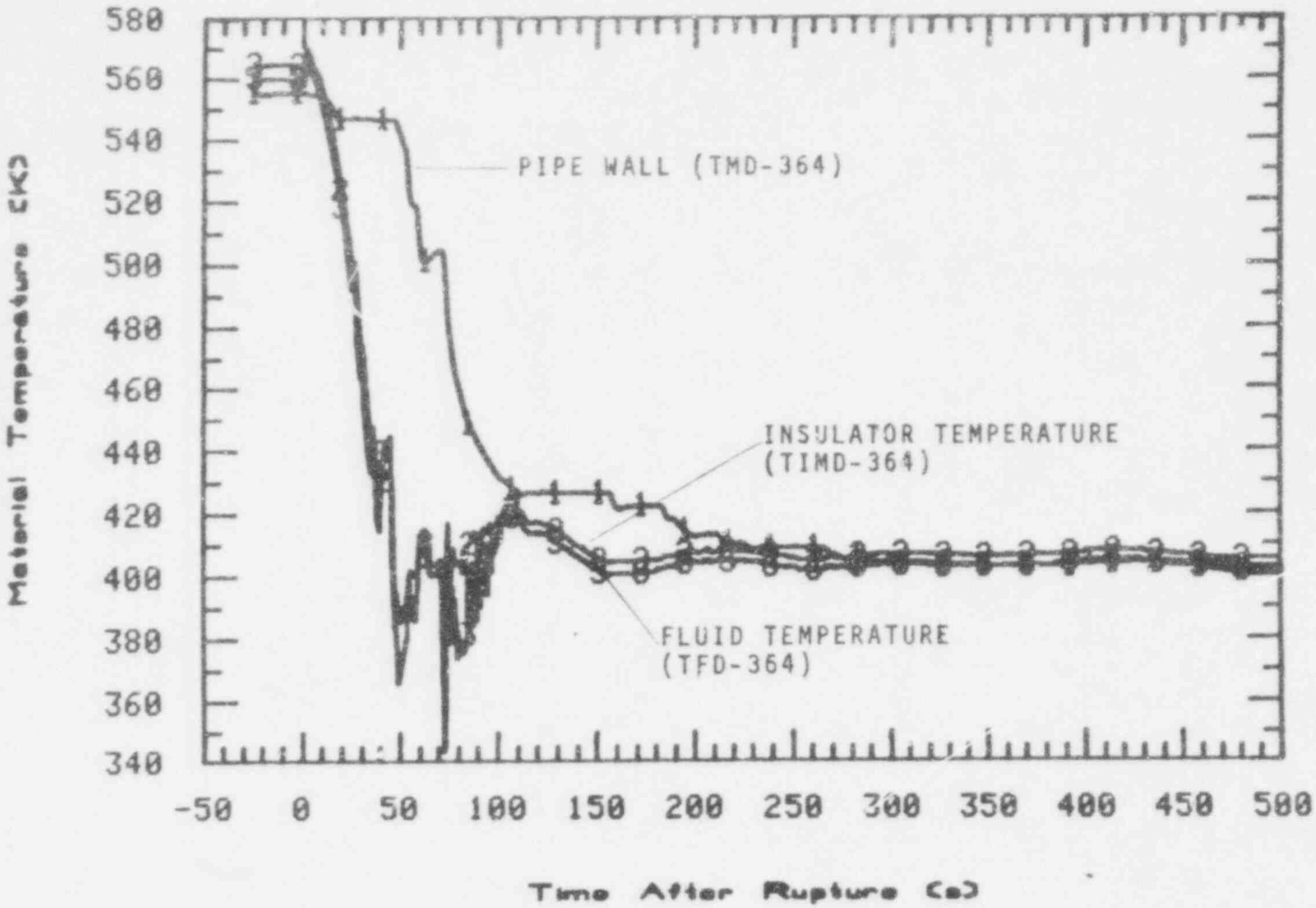

Fig. C-6 Temperature gradient across downcomer pipe during Test S-B7-6.

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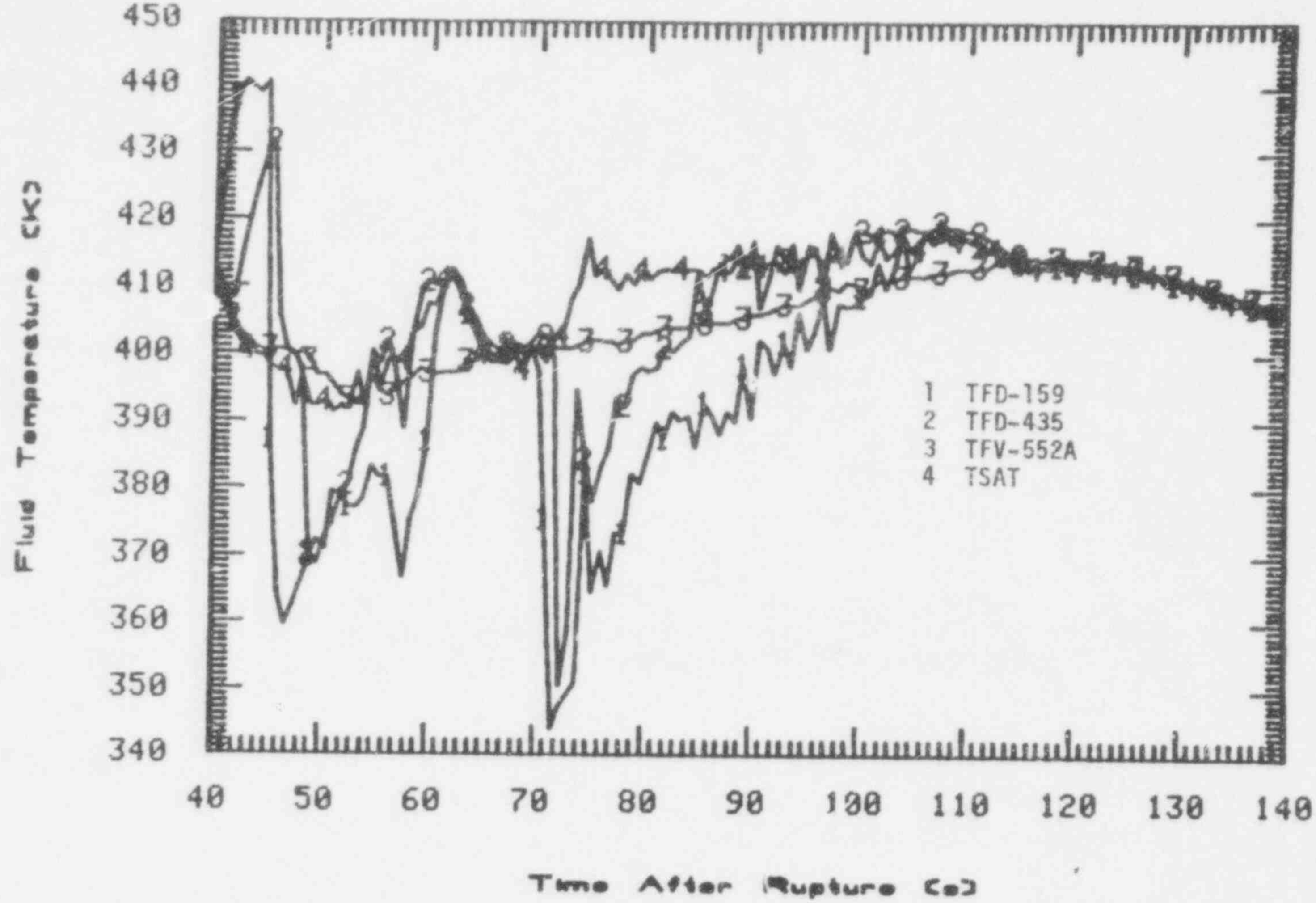

Fig. C-7 Downcomer and lower plenum fluid temperatures for Test S-B7-6.

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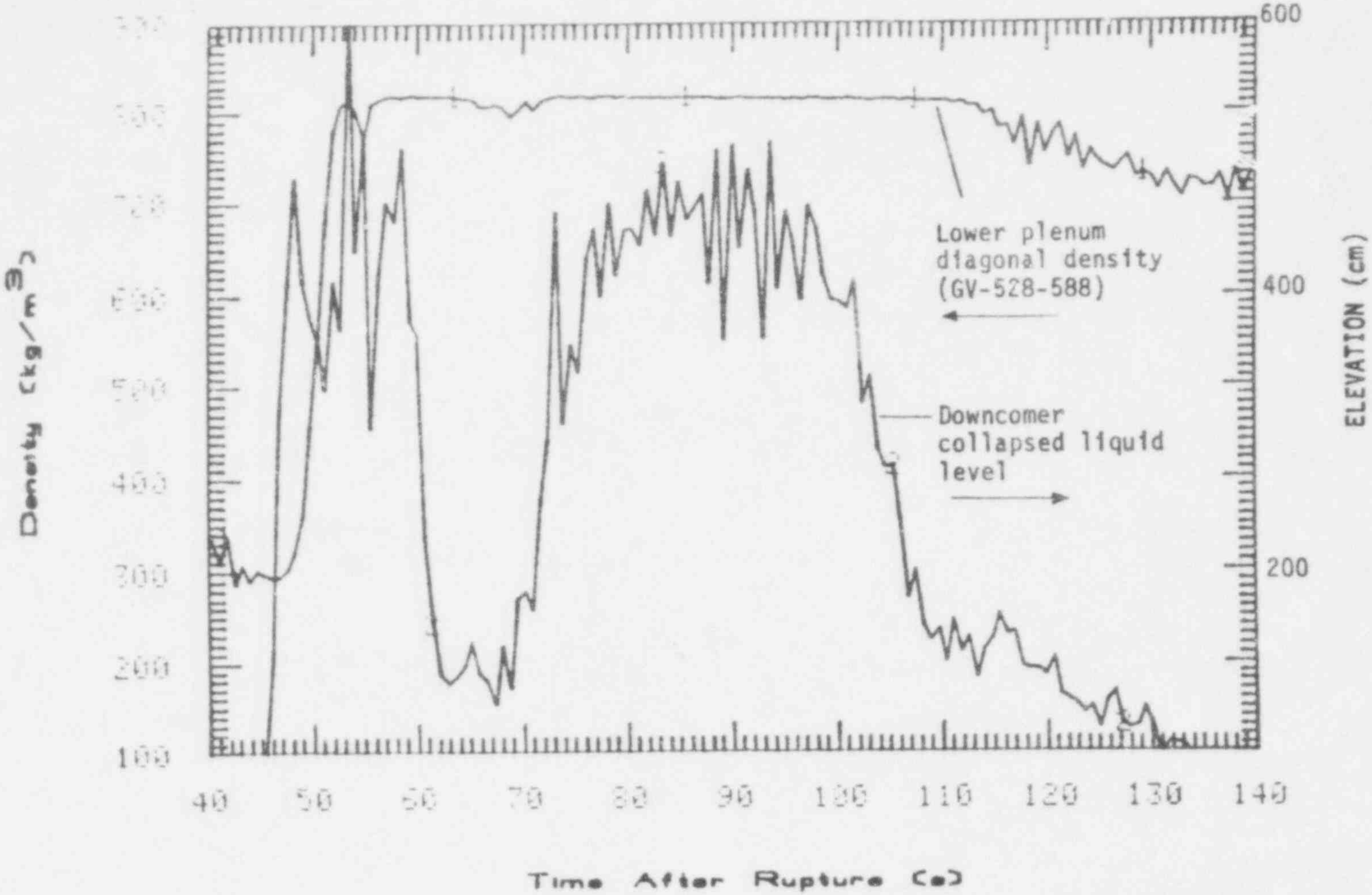

Fig. C-8 Downcomer collapsed liquid level and lower plenum diagonal density for Test S-B7-6.

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portion of the downcomer (TFD-435) reached saturation at about 55 s and again at about 88 s, and Figure C-8 shows downcomer depletion began at about 60 and 100 s, respectively. Second, fluid in the upper portion of the downcomer (TFD-153) reached the saturation temperature only after mass depletion had begun (that is, at about 61 and 104 s) which is an indication that saturated fluid from the lower portion of the downcomer moved upward past the upper thermocouple locations. Finally, the fluid temperature in the lower plenum (TFV-552A)[a] remained subcooled until the downcomer liquid was essentially depleted. The presence of subcooled liquid near the top of the lower plenum during the depletion process indicates that there was not substantial steam backflow from the core. In addition, the lower plenum diagonal density (GV-528-588) shown in Figure C-8 did not indicate the presence of steam backflow until the downcomer liquid depletion had been essentially completed.

However, the rate of downcomer refill following 150 s in Test S-B7-6 did not occur at the fairly rapid rate that was observed in Test S-07-6 (refer to Figures C-1 and C-5). In fact, refill of the downcomer to the level obtained at about 225 s in Test S-07-6 did not occur until about 425 s in Test S-B7-6. The extremely slow filling rate in Test S-B7-6 cannot be attributed to the excessively high downcomer heat transfer that was observed prior to 150 s. As indicated in Figure C-6, the temperature of the outer pipe of the

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a. This thermocouple (TFV-552A) is situated about 6 cm below the bottom of the core barrel.

downcomer was only slightly higher than the fluid temperature after about 200 s. Thus, the downcomer metal-to-fluid heat transfer rate was essentially zero and should not have affected the downcomer refill response. A possible explanation of the slow downcomer refill that occurred in Test S-B7-6, however, is that a relatively high steam generation rate in the core combined with a low-pressure head in the downcomer (refer to Figure C-5) resulted in considerable backflow of steam from the core region, which introduces the second suggested cause of downcomer mass depletion, steam backflow from the core to the downcomer.

The steam backflow and the corresponding countercurrent steam flow in the downcomer could limit the rate at which LPIS fluid entered the downcomer. Although the steam generation rate in the core cannot be measured directly, the core outlet volumetric flow rate is indicative of the steam generation rate (that is, a high volumetric flow rate out the top of the core corresponds to a high steam generation rate in the core region, and vice versa). Therefore, a comparison of the core outlet volumetric flow rates for Test S-07-6, shown in and S-B7-6 Figure C-9, provides an indication of the differences in the core steam generation rate, and thus gives some insight into the differences in the magnitude of the countercurrent steam flow in the downcomer[a]. In Test S-07-6, the volumetric flow

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a. The turbine flowmeter located near the bottom of the downcomer (FD-424) did not indicate strong upward flow when the steam generation rate in the core was high. However, densities near the turbine flow meter location indicate that a two-phase mixture was present. Thus, the turbine flow meter would not be expected to indicate a high steam flow rate even though a substantial steam flow may have existed.

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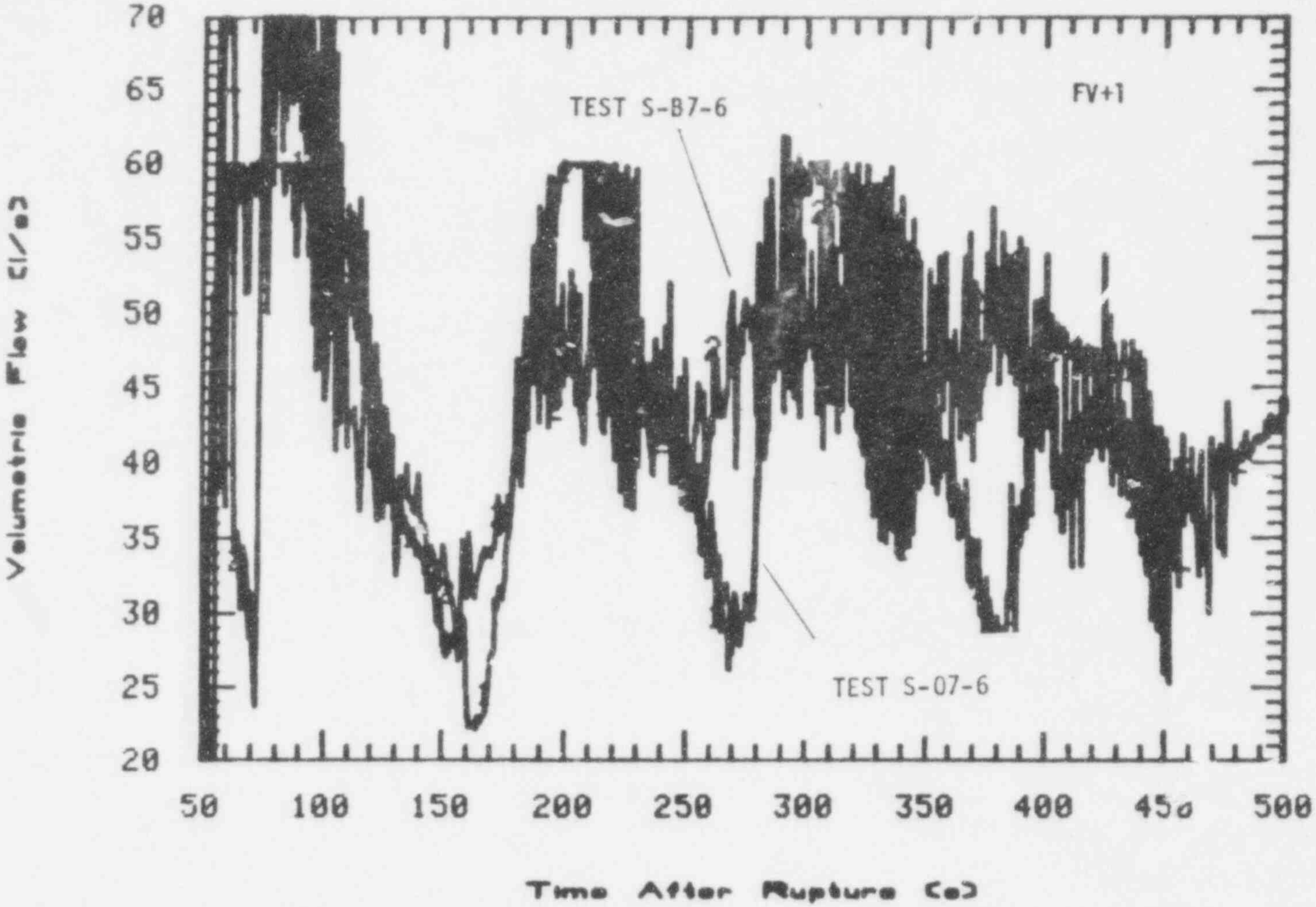

Fig. C-9 Core outlet volumetric flow for Tests S-07-6 and S-B7-6.

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rate out the top of the core dropped to a minimum (which is an indication that the countercurrent steam flow in the downcomer also dropped to a minimum) just prior to each time refill of the downcomer began. Thus it appears that steam generation in the core region and the corresponding upflow of steam in the downcomer had to be sufficiently low to allow rapid penetration of LPIS fluid into the downcomer. In Test S-B7-6, however, the volumetric flow out the top of the core, and thus the countercurrent steam flow in the downcomer, was continuously higher than the minimum rates observed in Test S-07-6. Thus, it is likely that the downcomer refill rate in Test S-B7-6 was limited by the relatively high steam generation rate in the core and the corresponding high countercurrent steam flow rate in the downcomer. The higher core steam generation rate in Test S-B7-6 corresponds to a core liquid level, which was continuously higher than the minimum liquid level observed in Test S-07-6. The core collapsed liquid levels for Tests S-07-6 and S-B7-6 are compared in Figure C-10.

3. EVALUATION OF TEST S-D7-6 RESULTS

To better understand the effects of core steam generation, an isothermal blowdown-reflood test (Test S-D7-6) was performed. The objective of this test was be to reduce steam generation in the core to a minimum which would provide further evidence that the downcomer mass depletion process was initiated by downcomer heat transfer rather than steam backflow from the core. Test S-D7-6 was, therefore,

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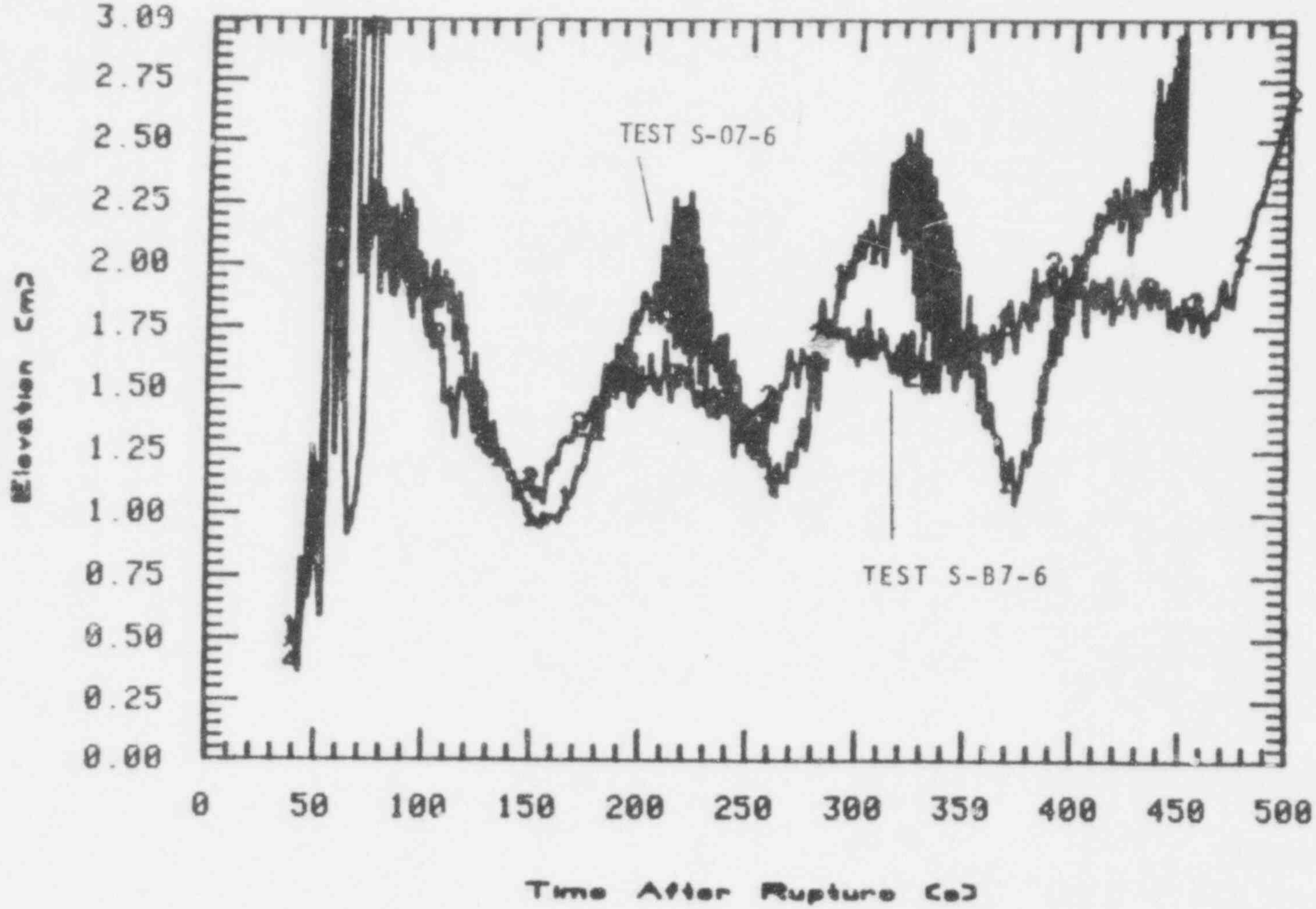

Fig. C-10 Core collapsed liquid levels for Tests S-07-6 and S-B7-6.

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conducted with an unpowered core and at an initial system fluid temperature of 557 K. The downcomer configuration was the same as that used in Test S-B7-6.

Even though the steam generation rate in the core was substantially reduced for Test S-D7-6, the vessel refill-reflood behavior was very similar to that observed in Test S-B7-6. The downcomer and core collapsed liquid levels for Test S-D7-6 are presented in Figure C-11. As indicated in the figure, the initial depletion of the downcomer fluid began at about 105 s after rupture. In addition, the comparison of the downcomer and lower plenum fluid temperatures, shown in Figure C-12, indicates that liquid in the lower part of the downcomer (TFD-435) reached the saturation temperature prior to the initiation of mass depletion, while the fluid temperature in the lower plenum (TFV-552) remained substantially subcooled until the depletion process was essentially completed. Thus, as was the case in Test S-B7-6, heat transfer from the downcomer walls was responsible for initiating the depletion process. The subcooling in the lower plenum during this depletion process again indicates that backflow from the core did not initiate the depletion process. The slow refill, which occurred between about 170 and 350 s after rupture, again was a result of steam generation in the core and a corresponding countercurrent steam flow in the downcomer. Although the core was unpowered for Test S-D7-6 and the steam generation rate was substantially reduced from that obtained in Test S-B7-6, the steam generation in the core region was still relatively high as indicated in Figure C-13, which compares the core outlet volumetric flow rates

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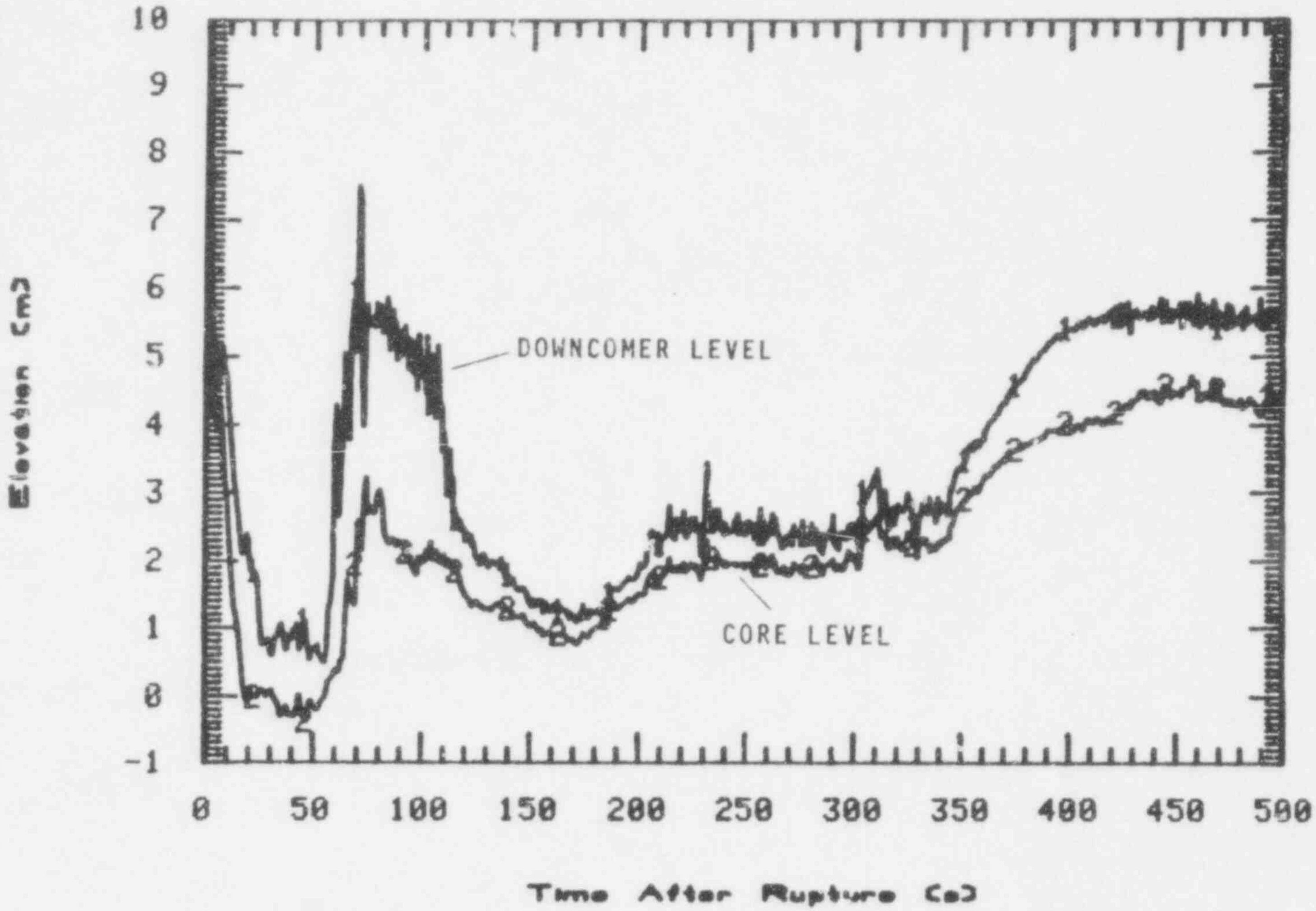

Fig. C-11 Core and downcomer collapsed liquid levels for Test S-D7-6.

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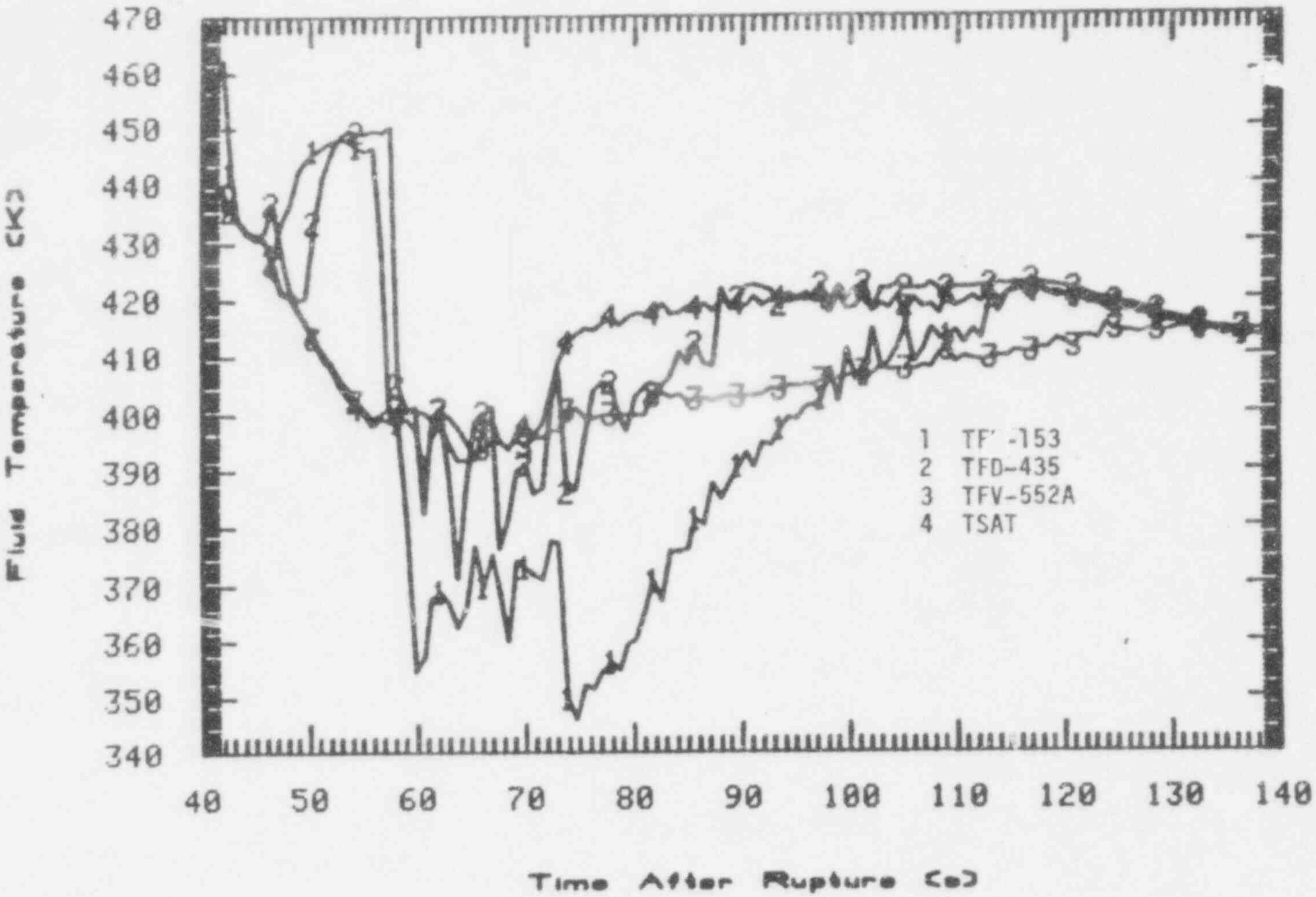

Fig. C-12 Downcomer and lower plenum fluid temperatures for Test S-D7-6.

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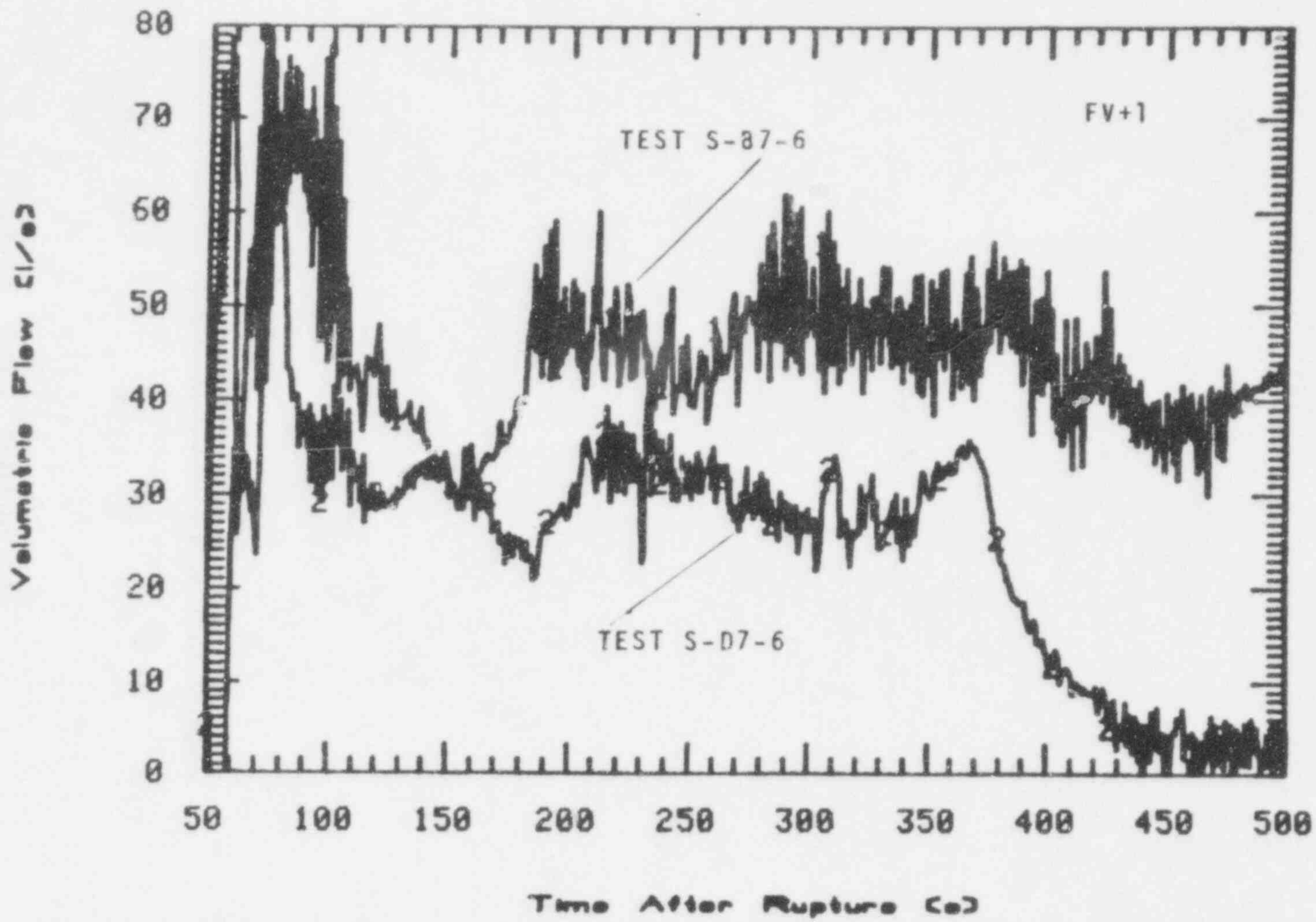

Fig. C-13 Core outlet volumetric flow for Tests S-B7-6 and S-D7-6.

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for Tests S-07-6 and S-B7-6. The relatively high steam generation rate for Test S-07-6 is attributed to excessive heat transfer from the hot metal structures in the core region. The metal structures in the core region in the Semiscale Mod-3 system are insulated using a steam gap system. However, based on the results of tests conducted to date, it appears that the steam gap insulators are not functioning as expected. Once the hot metal structures became quenched, as indicated by the rapid decrease in volumetric flow out the top of the core at about 350 s after rupture, a relatively rapid refilling of the downcomer and a corresponding rapid reflooding of the core occurred. Therefore, even with the large steam generation in the core, steam backflow is not a significant contributor to the initiation of downcomer mass depletion.

The effects of the one-dimensional downcomer, the third suspected cause of the mass depletion phenomenon, are difficult to assess based on available experimental data. However, a comparison of the reflood behavior for Tests S-07-6 and S-04-6[C-2] does provide some insight into the possible one-dimensional effects of the downcomer. Test S-04-6 was conducted in the Semiscale Mod-1 system which had an annular downcomer internal to the vessel. As was the case for Test S-07-6, results of Test S-04-6 also showed downcomer mass depletion. The initiation of mass depletion in Test S-04-6 occurred only after the downcomer fluid temperature reached the saturation temperature and as was the case for Test S-07-6, appears to have been due to a combination of heat transfer from the downcomer walls causing boiling and backflow from the core. However, unlike Test S-07-6, mass

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depletion of the downcomer fluid occurred only once in Test S-04-6. The initial mass depletion was then followed by a gradual refill of the downcomer which continued for the duration of the test. The fact that mass depletion occurred in both Tests S-07-6 and S-04-6, but was repeated several times in Test S-07-6, may be due to the different downcomer geometries in the two tests. The annular downcomer geometry in Test S-04-6 may have provided a path for steam generated on the downcomer walls or for steam backflow from the core to escape from the system, thus allowing the LPIS flow to gradually refill the downcomer. On the other hand, the relatively small inside diameter of the downcomer in Test S-07-6 most likely did not allow countercurrent steam flow upward and liquid flow downward in the downcomer. As a result, steam generated in the downcomer due to heat transfer from the walls or steam backflow from the core would have a much greater tendency to inhibit ECC flow into the downcomer. However, results from Tests S-07-6 and S-04-6 do not prove conclusively that the differences in downcomer hydraulics were due to the downcomer geometry. Other differences, such as differences related to the core lengths, may also have contributed to the different downcomer hydraulic behavior during reflood for the two tests.

4. SUMMARY OF EXPERIMENTAL RESULTS

Results of the analysis of data from Tests S-07-6, S-B7-6, and S-D7-6 have shown that the downcomer mass depletion process is initiated by metal-to-liquid heat transfer, which causes boiling of the downcomer fluid. The resulting swell of the liquid-vapor mixture forced liquid out the top of the downcomer, causing a decrease in the effective downcomer pressure head. Steam backflow from the core was not found to have a significant effect during most of the downcomer depletion process; however, after the depletion occurred, steam backflow effected the rate of LPIS fluid entering the lower plenum. Possible modifications to the downcomer design that would considerably reduce or eliminate the mass depletion problem include the addition of a system to externally cool the downcomer or the incorporation of a low conductivity-heat capacity honeycomb insulator with sealed gas spaces to replace the Grafoil liner insulator presently employed.

In addition to identifying the causes of the downcomer mass depletion in the Semiscale Mod-3 system, the current analysis effort has indicated that heat transfer from the metal structures in the core region, and thus the core steam generation rate, is excessively high. Although the metal structures in the core region in the Semiscale Mod-3 system are insulated, it appears that the insulators are not functioning as expected. Modification of the core metal insulators will be necessary to reduce the core steam generation rate to a value that would be more typical of what is expected in a PWR system.

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5. REFERENCES

C-1 V. Esparza, K. E. Sackett, K. and Stanger, Experiment Data Report for Semiscale Mod-3 integral Blowdown and Reflood Heat Transfer Test S-07-6 (Baseline Test Series), TREE-NUREG-1226 (January 1979).

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APPENDIX D

FLOW REGIMES IN INTACT LOOP PIPING

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APPENDIX D

FLOW REGIMES IN INTACT LOOP PIPING

The possibility that Semiscale Mod-3 piping can produce flow regimes similar to those expected in a pressurized water reactor (PWR) has been a matter of curiosity. To determine if flow regime similarities could be established, a study using Test S-07-1 as the data base was initiated. Calculations for the study were made using a method suggested by Govier and Aziz[D-1] as outlined in a report written by R. T. French[D-2], from which the remainder of this description is taken.

Two-phase blowdown mixture quality was related to the slip (or holdup) ratio using an equation supplied by Wallis[D-3]. The calculations required input of mass flow rate, liquid and steam phase densities, and the fluid void fraction. An iterative method was required to obtain the flow regime. The equations that are solved are as follows:

$$X = \left\{1 + \frac{1}{S}\ \frac{\rho_\ell}{\rho_g}\ \left(\frac{1-\alpha}{\alpha}\right)\right\}^{-1} \qquad \text{(D-1)}$$

$$Vs = \frac{\dot{m}\ (1-x)}{\rho_\ell\ A_T} \qquad \text{(D-2)}$$

$$Vs_g = \frac{\dot{m}}{\rho_g}\frac{x}{A_T} \qquad \text{(D-3)}$$

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where

X = mass quality, dimensionless

S = slip or holdup ratio, dimensionless

ρ_ℓ = liquid phase density at system pressure, kg/m^3

ρ_g = gas phase density at system pressure, kg/m^3

α = void fraction, dimensionless

Vs_ℓ = superficial liquid phase velocity, m/s

$\dot{m}$ = mass flow rate, kg/s

Vs_g = superficial gas phase velocity, m/s

A_T = total cross-sectional flow area, m^2.

Void fractions used in Equation (D-1) were obtained as a function of the measured density from a calculation for a typical Semiscale pipe section by assuming either a homogeneous, annular, or stratified flow regime. This information is presented in Figure D-1. The void fraction information presented in the figure was obtained from straight-forward geometric considerations of the Semiscale Mod-1 loop piping geometry. Annular and stratified void fractions were obtained

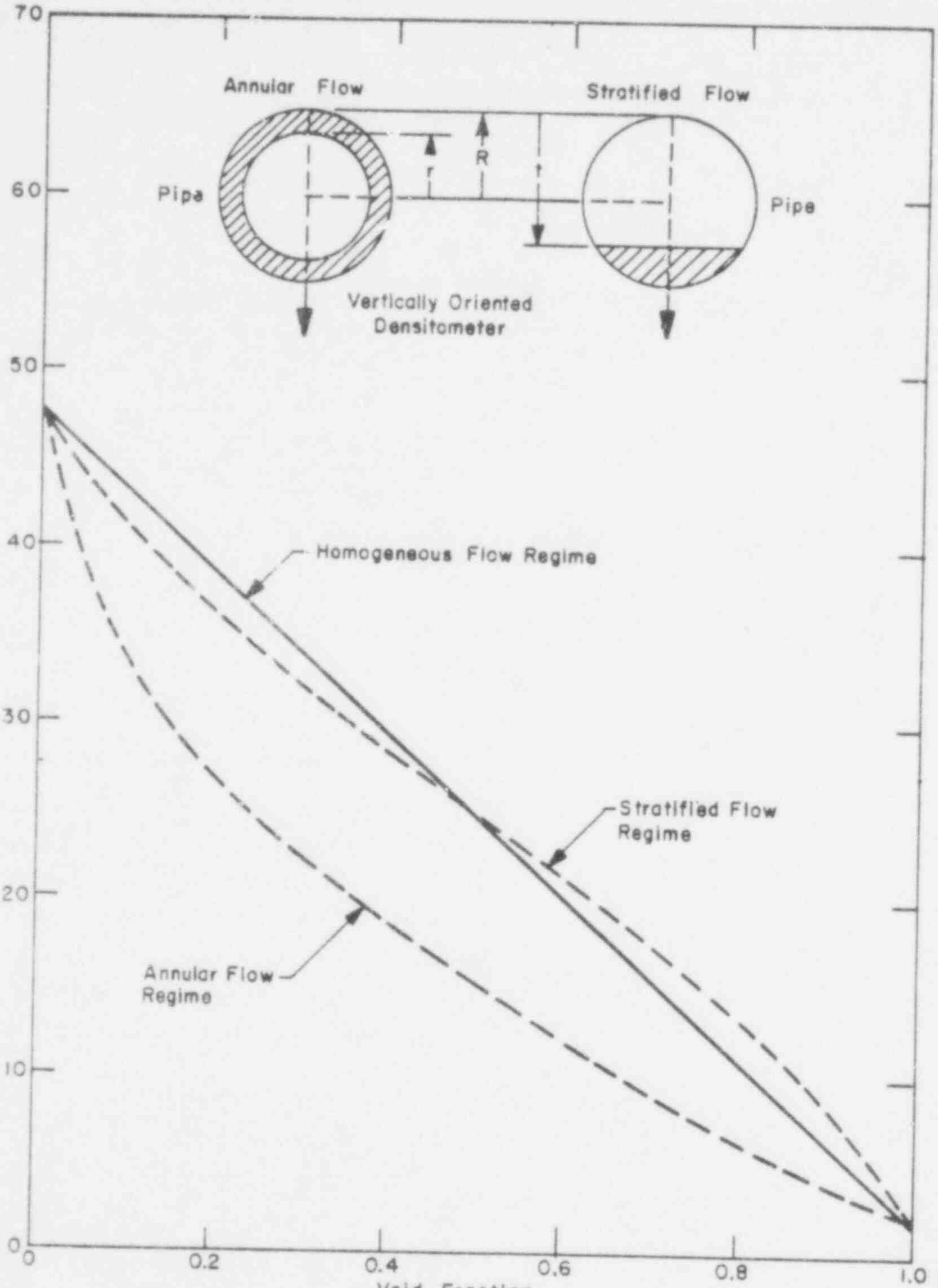

Fig. D-1 Void fraction as a function of measured density at 5515 kpa.

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assuming complete phase separation. The infomration presented in Figure D-1 was obtained using the analysis presented in Appendix E of Reference D-4.

The empirical data of Govier and Aziz[D-1], developed for air-water flow in a horizontal pipe, relating the holdup (slip ratio in terms of the superficial phase velocities) are presented in Figure D-2.

Equation (D-1), (D-2), and (D-3) were solved in the following manner: First, a flow regime was assumed, and a void fraction was obtained from Figure D-1. A homogeneous void fraction was used for the dispersed bubble flow regime. An annular void fraction was used for elongated bubble and annular flow regimes. A stratified void fraction was used for stratified, wavy, and slug flow regimes. A slip ratio was then assumed, and the three equations were solved. The calculated superficial phase velocities were then used in conjunction with Figure D-2 to obtain the flow regime and holdup ratio, which were then compared with initial flow regime and holdup ratio assumptions. If in error, the new information was then iterated on until convergence of both the flow regime and the holdup ratio was obtained.

Using the above procedure, the flow regimes, shown in Figure D-3, were calculated. The calculation determined the duration of the dispersed bubble and stratified flow regimes, which included wavy flow. The fact that these flow regimes are present is important; however, it should not be construed that these same flow regimes will

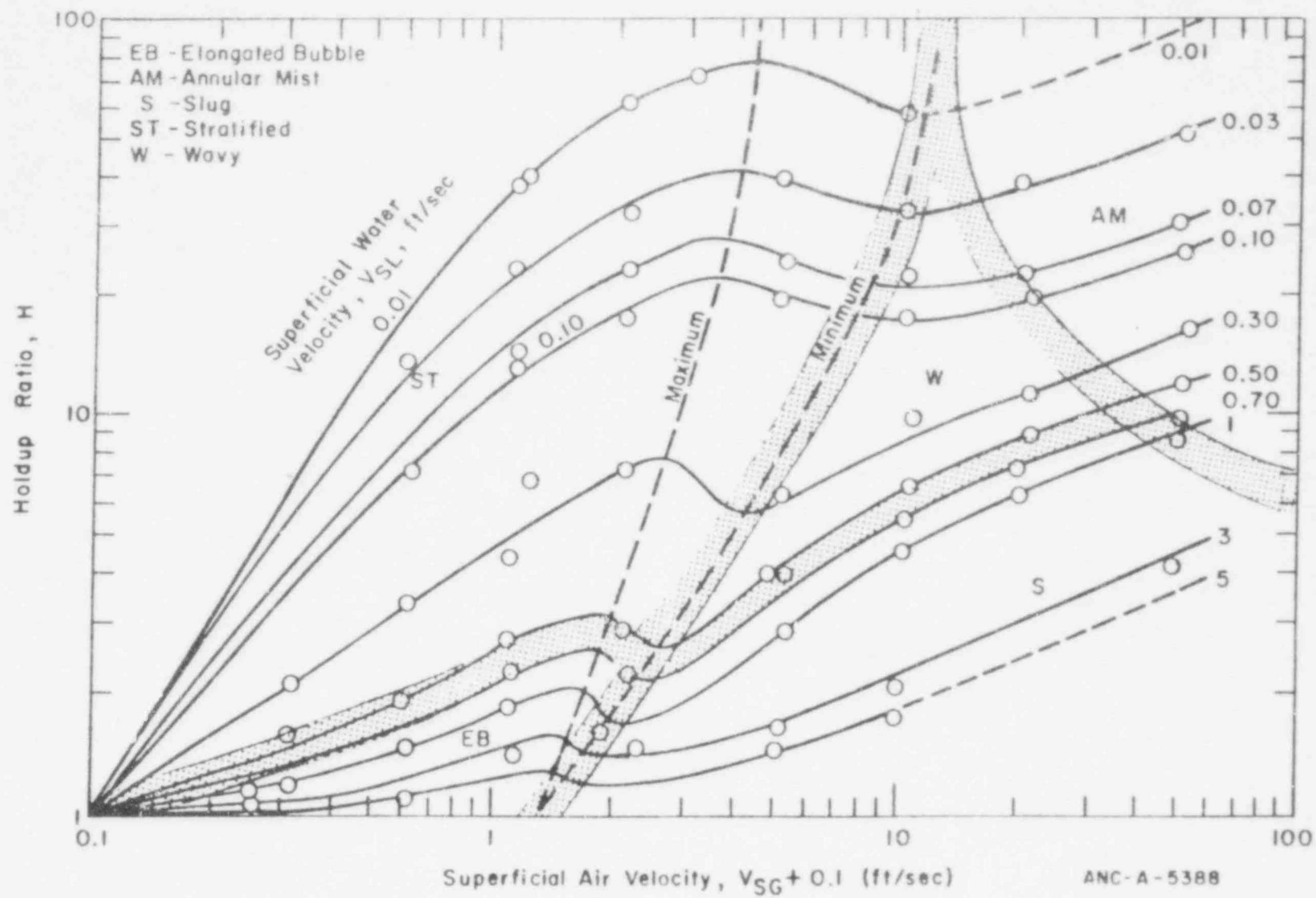

Fig. D-2 Holdup as a function of superficial phase velocities.

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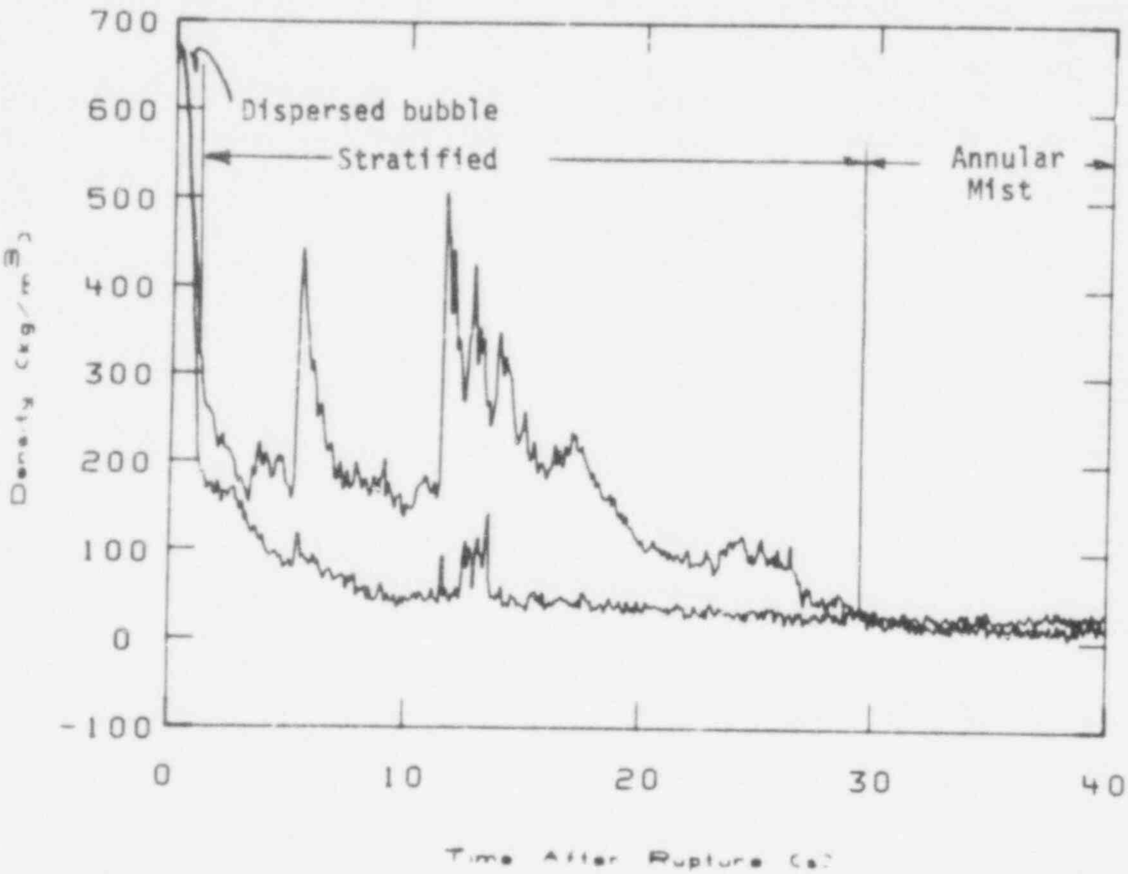

Fig. D-3 Flow regimes for Semiscale Mod-3 intact loop, hot leg piping.

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be present in a PWR for the same duration of time. Because of the small flow area and possible piping heat transfer of Semiscale Mod-3, the above calculations can only indicate that these flow regimes are typical of Semiscale Mod-3 piping.

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