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LA-UR -88-3348

LA-UR--88-3348

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TITLE: MODELING OF THE TMI-2 ACCIDENT WITH MELPROG/TRAC AND CALCULATION RESULTS FOR PHASES 1 AND 2

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SUBMITTED TO Sidney Langer
Technical Program Chairman
American Nuclear Society Topical Meeting
The TMI-2 Accident: Materials Behavior and
Plant Recovery Technology
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MODELING OF THE TMI-2 ACCIDENT WITH MELPROG/TRAC AND CALCULATION RESULTS FOR PHASES 1 AND 2[a]

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Total number of pages: 8 text, 2 tables, 10 figures

ABSTRACT

Work has been performed to develop a Three Mile Island Unit-2 (TMI-2) simulation model for MELPROG/TRAC capable of predicting the observed plant behavior that took place during the accident of March 1979. A description of the TMI-2 plant model is presented and calculation results through 174 min of the accident are discussed. Using the ICBC boundary conditions, the calculation predicts pressurizer draining and core recovering prior to fuel-rod damage. A parametric calculation (reduced makeup flow) is currently underway and is in better agreement with the observed plant behavior. Efforts are underway to resolve current discrepancies and proceed with an accurate simulation through Phases 3 and 4 of the accident (174–227 min and 227–300 min, respectively).

INTRODUCTION

This paper describes the MELPROG/TRAC Three Mile Island Unit-2 (TMI-2) input model and calculation results for Phases 1 and 2 of the TMI-2 accident (to 0–100 min and 100–174 min. respectively). An in-depth model of the TMI-2 nuclear power station has been developed with the objective of calculating all of the relevant phenomena believed to have occurred during the March 1979 accident. With this plant model, we are now directing our efforts toward obtaining good agreement between observed and calculated behavior. This task is difficult because of the uncertainty in boundary conditions and timing of major events.

These efforts support the US Nuclear Regulatory Commission (NRC) and the Organization for Economic Cooperation and Development (OECD)-sponsored TMI-2 Analysis Exercise by assessing the integrated MELPROG/TRAC code through all phases of the accident progression.(1-5) Numerous nuclear power plant simulation studies have been performed with the TRAC computer code in the past.[b] Some of these addressed the TMI-2 accident or other hypothetical events at the TMI plant.(6-8) In addition, studies have been performed with the MELPROG/TRAC code using Oconee-1 and Surry plant models.(9-12) This current work represents an ongoing analysis of an actual severe accident with a state-of-the-art code and provides important code assessment for code model improvements.

MELPROG/TRAC TMI-2 PLANT MODEL

In-vessel Modeling

Reactor pressure vessel modeling is done with a separate input deck that describes the initial geometry, material composition, and thermal and hydraulic conditions. The TRAC code models plant equipment as separate components (VESSEL, PIPE, TEE, PUMP, etc) and treats this separate vessel input model as a special component, "MELVSL." The MELVSL component replaces the usual TRAC "VESSEL" component in the calculation. The MELVSL model is shown in Fig. 1 and consists of the following:

- vessel bottom head
- in-core instrumentation tubes
- lower grid support posts
- lower grid rib section
- control rod assembly guide tubes
- vessel top head
- core barrel
- formers
- core support shield
- core
- flow distributor
- lower grid forging and shell
- lower grid distributor plate
- upper grid rib section and ring
- upper support plate
- thin metal in upper head
- baffle plate
- plenum cylinder
- vessel wall + thermal shield

The behavior of these structural components is handled primarily by the STRUCTURES module of MELPROG, which calculates the mechanical and thermal behavior for a wide variety of structure types. The structures of the vessel are modeled, so that comparison of temperatures and damage can be made between the calculation and observations. Core behavior is determined primarily by the CORE module in MELPROG, which calculates heat-up and melting of the fuel, cladding, and other core materials. Models to calculate candling, freezing, and crusting of core materials are included in CORE. The CORE module also treats energy generation by decay heat and chemical reactions and calculates fuel and cladding oxidation. In addition, CORE allows interaction of the melted material and can calculate debris bed behavior that will be important in analysis of Phases 3 and 4 of the accident.

Two-dimensional modeling with MELPROG/TRAC allowed accurate representation of the core and other structural geometry with 5 radial rings and 15 axial levels. The three inner rings[c] model the lower-inlet plenum, core, and upper-outlet plenum regions, ring 4 models the core bypass, and ring 5 models the downcomer region. Axial levels were chosen to separate major structural components in the upper and lower plenum/head and to divide the core region into six axial regions.

The core model consists of fuel-rod and control-rod specifications of material, geometry, and location. The fuel-rod model has two radial regions (fuel and clad) and two axial regions (active fuel and fission gas plenum). The control-rod model has two radial regions (absorber and clad) and one axial region. The inner radial rings (rings 1 to 3) of the vessel model were placed to allow the same number of fuel and control rods in each ring (12272 and 698, respectively).

Balance of Primary and Secondary Modeling

The balance of the primary is composed of TRAC components, as indicated in Fig. 2, including two hot legs, four cold legs, four reactor coolant pumps, and the pressurizer. In addition, both steam generators as well as control systems are modeled to simulate operator and equipment actions during the accident. The secondary system includes the secondary side of the steam generators and boundary conditions for main and auxiliary feedwater (AFW) flows and steamline pressures.

CALCULATION RESULTS

Initial Conditions

Initial conditions are shown in Table I for a generalized steady-state calculation for 100 s to provide consistent initial and boundary conditions for the transient calculation. The apparent difference in the initial pressurizer liquid level is because the observed value is based on a pressure differential between two pressure taps, whereas the calculated value is the collapsed liquid level in the pressurizer tank, hence an offset. The initial and boundary conditions are consistent with the data base that has been provided as part of the Initial Condition/Boundary Condition (ICBC) software to support the TMI-2 Analysis Exercise (13) Matching the secondary pressure in the steam-generator model produced primary temperatures 7 K higher than observed. The apparent imbalance in initial loop-A cold-leg temperatures and the initial small power-operated relief valve (PORV) flow were not modeled.

Transient Simulation

The sequence of events is shown in Table II, which compares observed with calculated event times. The accident initiator was loss of main feedwater followed by turbine trip and activation of AFW pumps and rapid primary pressurization. The calculated time to opening of PORV compared well with the observed value of approximately 4 s. The reactor tripped at 8 s in the accident, which corresponds to the time at which the primary pressure reached the high-pressure setpoint of 16.2711 MPa. The same pressure was reached at 8 s in the calculation, and the power was tripped.

Steam generators boiled dry at approximately the same time as observed. The times for primary system repressurization, pressurizer filling, and subsequent primary depressurization after 300 s compare moderately well.

Two calculations are discussed in the remainder of this report. A base case calculation, which used the boundary conditions recommended by Idaho National Engineering Laboratory (INEL) in the ICBC, and a parametric calculation, which reduced the makeup flow between 720 and 6000 s to 1 kg/s from 6.5 kg/s. The parametric calculation was undertaken because using ICBC boundary conditions did not lead to predicted damage of the core.

Superheated vapor is not observed for the base case calculation until approximately 7482 s, as compared to the 6720 s observed. The superheated vapor is an indication that the core has uncovered and that core damage is underway. Delay of core uncovery leads to reduced clad temperatures and prevents cladding oxidation and failure in the base case calculation. Subsequent draining of the liquid in the pressurizer through the surge line and hot leg back into the upper core cools the fuel rods further and prevents repressurization. The makeup flow chosen for the parametric calculation was based on previous calculations at Los Alamos and INEL, which uncovered the core at approximately the correct time.

This behavior is shown in Fig. 3, which plots observed vs calculated primary pressures. The initial calculated pressure response tracks the observed behavior quite well. However, after pressurizer block valve closure at 8340 s, the base case calculated pressure falls off rapidly. The reason for this is that delayed core uncovery prevents early and sufficient hydrogen generation to maintain system pressure increase and prevents the pressurizer liquid from draining into the core. By the time the core becomes uncovered, water is already flowing from the pressurizer back to the core, which prevents core temperatures from exceeding 1100 K and generating hydrogen. The parametric calculation begins system repressurization before the block valve closes. The parametric calculation is not complete at this time but does show signs of repressurization.

Figure 4 compares temperatures in the hot leg. The parametric calculation is in good agreement for the timing of hot-leg temperature increase at approximately 6400 s. The magnitude of the temperature increase is not in good agreement because the hot-leg instrument can not measure the vapor temperature. The pressurizer liquid level is shown in Fig. 5, and the draining in the base case calculation is clearly indicated after 7400 s. The draining is accelerated further after block valve closure at 8340 s. The parametric calculation has a larger decrease in pressurizer level when the pump is stopped at 6000 s. This is believed to be caused by the makeup flow being too low. The correct makeup flow is probably greater than 1 kg/s but considerably below the ICBC value of 6.5 kg/s.

The core liquid fraction for the base case calculation is shown in Fig. 6. Core uncovery begins at approximately 7500 s, although this is too late to allow early and sufficient hydrogen generation. Liquid draining back into the core from the pressurizer is seen to occur after approximately 9400 s. Figure 7 shows the core liquid fraction for the parametric calculation. Core uncovery begins at 6600 s. The fuel-rod temperature in the inner ring at level 9 is shown in Fig. 8. The temperature turnover in both calculations is because the small liquid drain backs from the pressurizer, which is also shown by unsteady hot-leg temperatures

(Fig. 4). The calculated total hydrogen generated is shown in Fig. 9. Primary-to-secondary heat transfer for the base calculation is shown in Fig. 10 for both steam generators. Steam generator B showed significant heat transfer after 4500 s, and heat continued to be removed from the primary throughout the duration of Phase 2. Figure 11 shows the steam-generator heat transfer for the parametric calculation. The core uncovery and heat-up with consequent hydrogen generation caused steam-generator heat transfer in the parametric calculation to be reduced. This eventually decoupled the secondary pressure from the primary pressure, allowing the primary repressurization necessary to sustain the pressurizer liquid level.

DISCUSSION

We are currently looking at some revisions to the code models, which should improve agreement between observed and calculated plant behavior. The effect of noncondensables on wall-to-vapor heat transfer has been added to the code. The same model for the effect of noncondensables used in TRAC for interphase heat transfer has been added to the wall heat transfer calculations. The previous model did not take this into account and, hence, allowed too much heat transfer after hydrogen generation began.

The modeling of the TMI-2 accident presents a challenge, and requires accurate establishment of initial and boundary conditions, plant geometry, and operator and equipment actions. Current efforts are directed at ensuring accurate initial and boundary conditions and revising code models, as needed, to provide more accurate calculation of the severe-accident phenomena.

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FOOTNOTES

[a]This work was funded by the US Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Division of Accident Evaluation.

[b]More than 125 full-scale plant calculations have been performed at Los Alamos in the last several years. These calculations covered numerous hypothetical accidents (SGTR, LOFW, SBLOCA, Boron Dilution, ATWS, LOSP, etc.) for most types of nuclear steam supply systems (B&W, Westinghouse, and CE). A listing of these calculations and associated report numbers is available by writing to the authors at MS K555, Los Alamos National Laboratory, Los Alamos, NM 87545.

[c] Rings are axi-symmetric (showing only one-half of the model) with the centerline of the reactor pressure vessel at the inner cell boundary of radial ring 1 (0.0 meters).

FIGURES

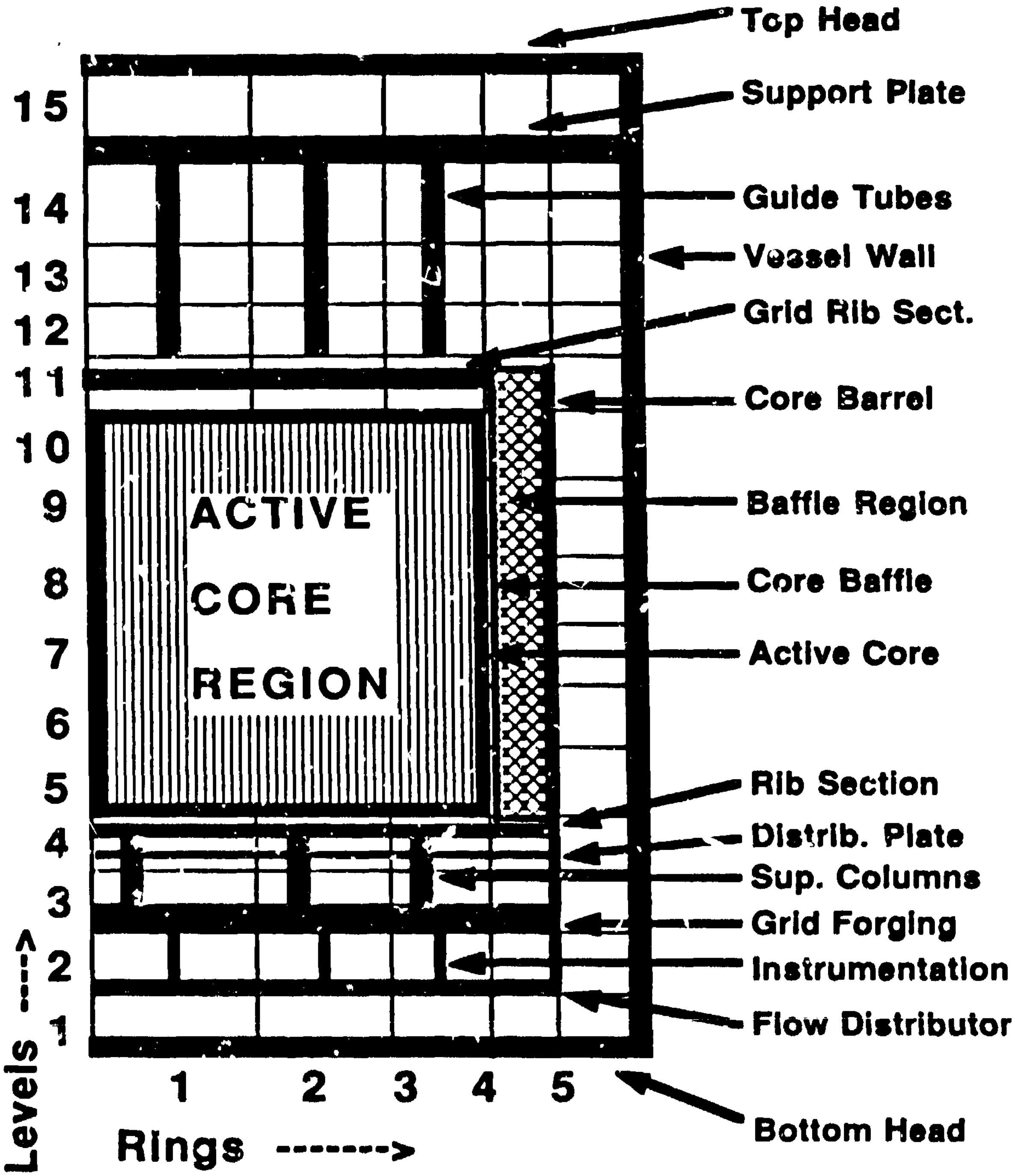
TABLE I

TMI-2 PLANT CONDITIONS: INITIAL

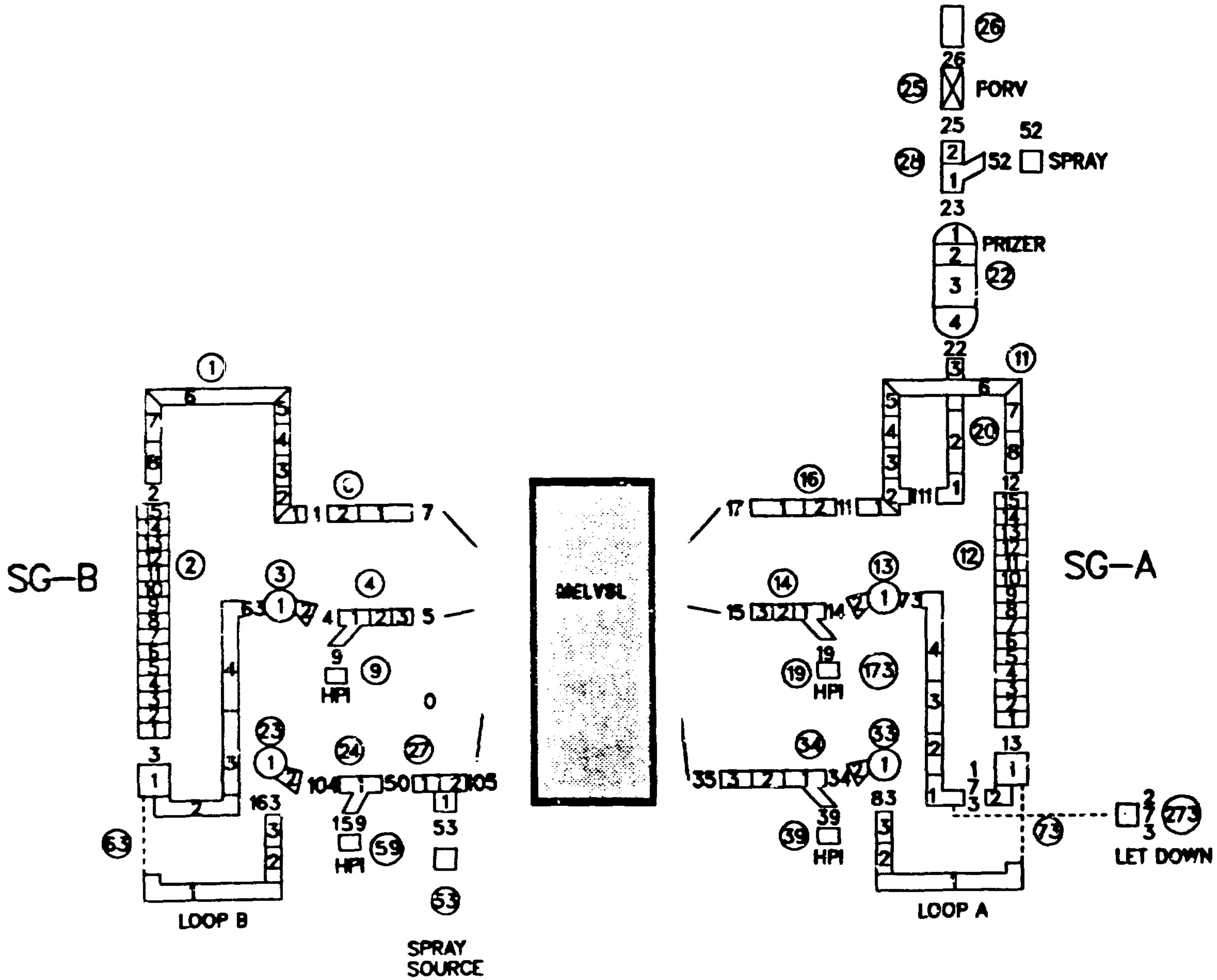
Parameter	Observed	Calculated
Aux feedwater injection SG A (kg/s)	0.0	0.0
Aux feedwater injection SG B (kg/s)	0.0	0.0
Calculated PORV flow rate (kg/s)	2.59 ± 0.517	0.0
Cold-leg temperature 1A (K)	561 ± 1.06	572.4
Cold-leg temperature 1B (K)	565 ± 1.06	573.2
Cold-leg temperature 2A (K)	548	572.4
Cold-leg temperature 2B (K)	565	572.3
Hot-leg temperature A Loop (K)	592 ± 0.633	599.7
Hot-leg temperature B Loop (K)	592 ± 0.633	599.7
HPI/make-up based on expected results (kg/s)	5.44	5.44
Letdown flow (kg/s)	4.18 ± 0.835	4.11
Main steam temperature A (K)	586 ± 1.17	590.4
Main steam temperature B (K)	586 ± 1.17	592.1
Pressure primary (MPa)	15.2 ± 0.0752	15.22
Pressure level (m)	5.77 ± 0.61	6.98
RC flow rate loop A (kg/s)	8280 ± 178	8285
RC flow rate loop B (kg/s)	8560 ± 184	8564
Reactor power (MW)	2700 ± 39	2689
Steam generator A feedwater flow (kg/s)	723 ± 13.4	723
Steam generator B feedwater flow (kg/s)	717 ± 13.4	717
Steam generator feedwater temperature (K)	513 ± 0.989	513
Steam generator A pressure (MPa)	7.31 ± 0.112	7.34
Steam generator B pressure (MPa)	7.24 ± 0.112	7.27
Top pressurizer heater group power (MW)	1.39	124

TABLE II
TMI-2 SEQUENCE OF EVENTS

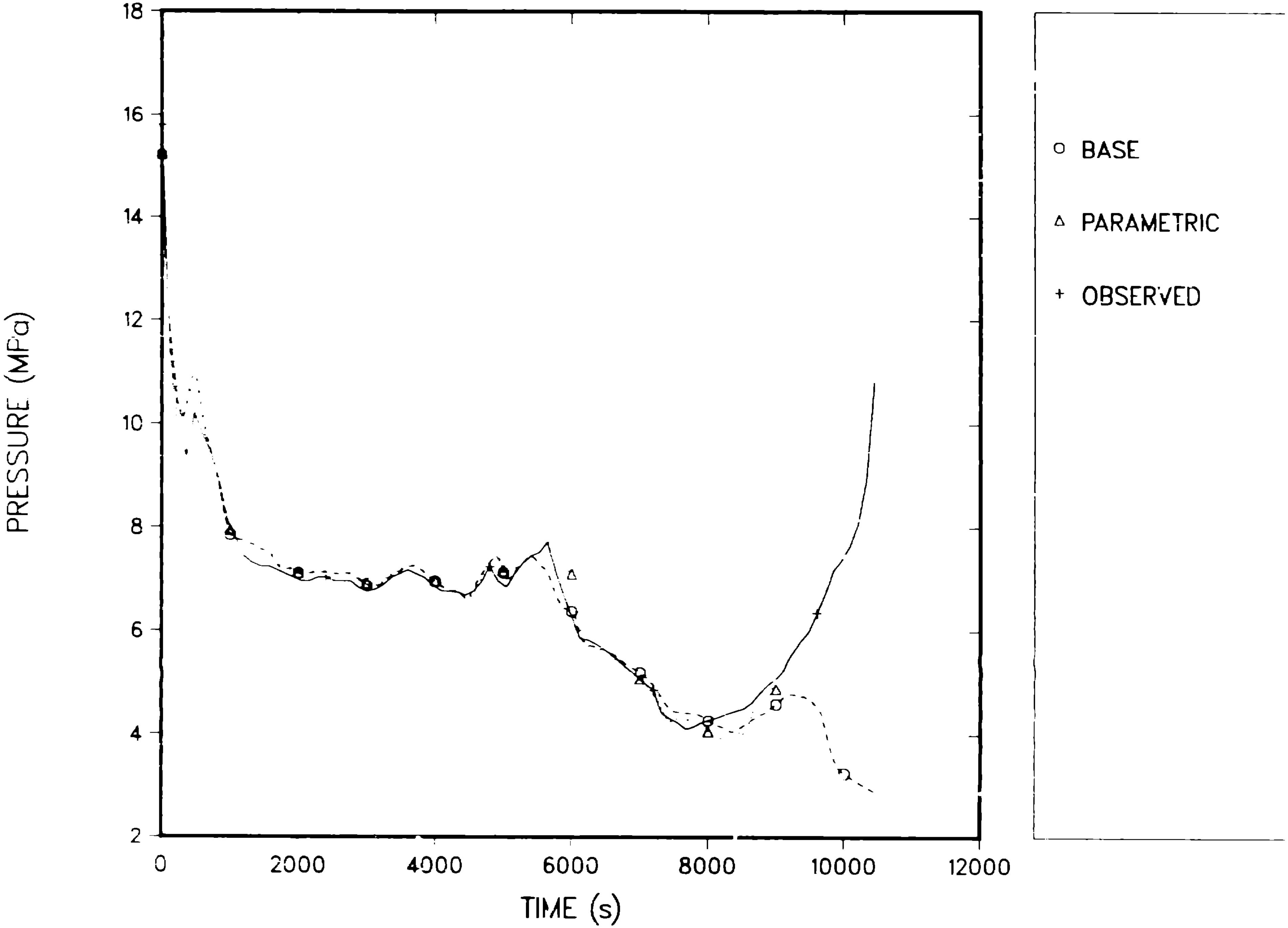
Time(s) OBSERVED	BASE	PARAMETRIC	Event
0	0	0	Loss of main feedwater
0	0	0	Turbine trip
0	0	0	Steam stop valves in steam chest close
0	0	0	AFW pumps start
4	5.2	5.2	Pressure >15.65 MPa: PORV opens
5	- -	- -	Turbine bypass valves open
8	8	8	Reactor trips
31	- -	- -	AFW valves open
41	41	41	HPI on
90	93	93	Steam generators boil dry
320	326	326	Primary system repressurization
450	500	500	Pressurizer full
480	480	480	Primary pressure decreases (AFW on)
4380	4380	4380	Loop B RCPs trip, Loop A voiding
6000	6000	6000	Trip A loop RCPs (end Phase 1)
6720	7440	6600	Superheat in Loop A hot-leg core uncovery
7500	8400	8300	Primary system repressurization starts again
7800	no	8000	Cladding failures (1100 K) followed by Zirc oxidation (1800 K) and fuel liquid faction
8340	8340	8340	Pressurizer PORV block valve closed
10440	10400		Restart of Loop B-2 RCP (end Phase 2) and end of calculation

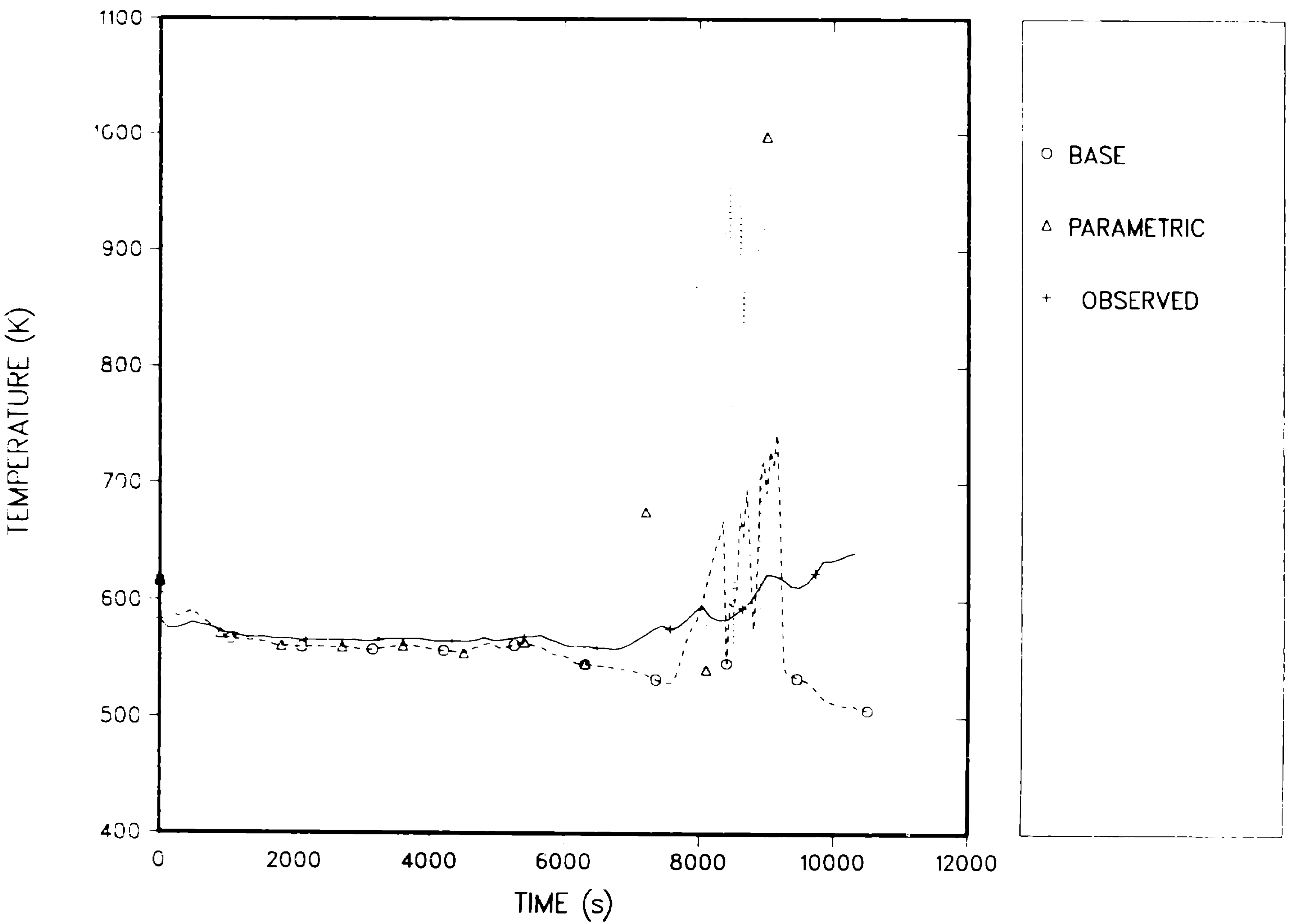

Top Head
Support Plate
Guide Tubes
Vessel Wall
Grid Rib Sect.
Core Barrel
Baffle Region
Core Baffle
Active Core
Rib Section
Distrib. Plate
Sup. Columns
Grid Forging
Instrumentation
Flow Distributor
Bottom Head
ACTIVE
CORE
REGION
15
14
13
12
11
10
9
8
7
6
5
4
3
2
1
Levels ---->
1
2
3
4
5
Rings ------->

2.


FORV
SPRAY
PRIZER
SG-B
SG-A
MELVSL
HPI
HPI
HPI
HPI
LOOP B
LOOP A
LET DOWN
SPRAY
SOURCE

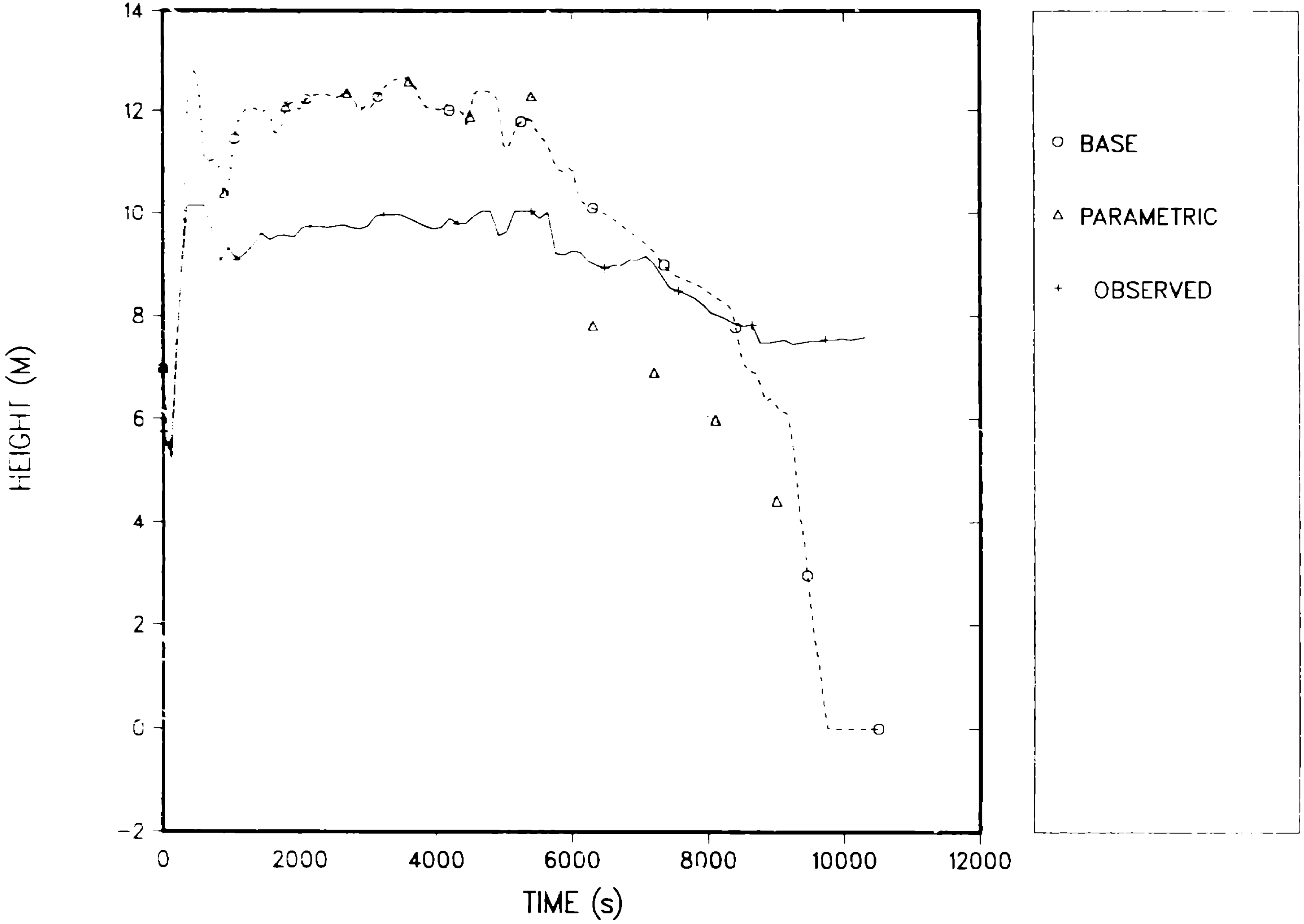
3


PRESSURE (MPa)
18
16
14
12
10
8
6
4
2
0
2000
4000
6000
8000
10000
12000
TIME (s)
BASE
PARAMETRIC
OBSERVED

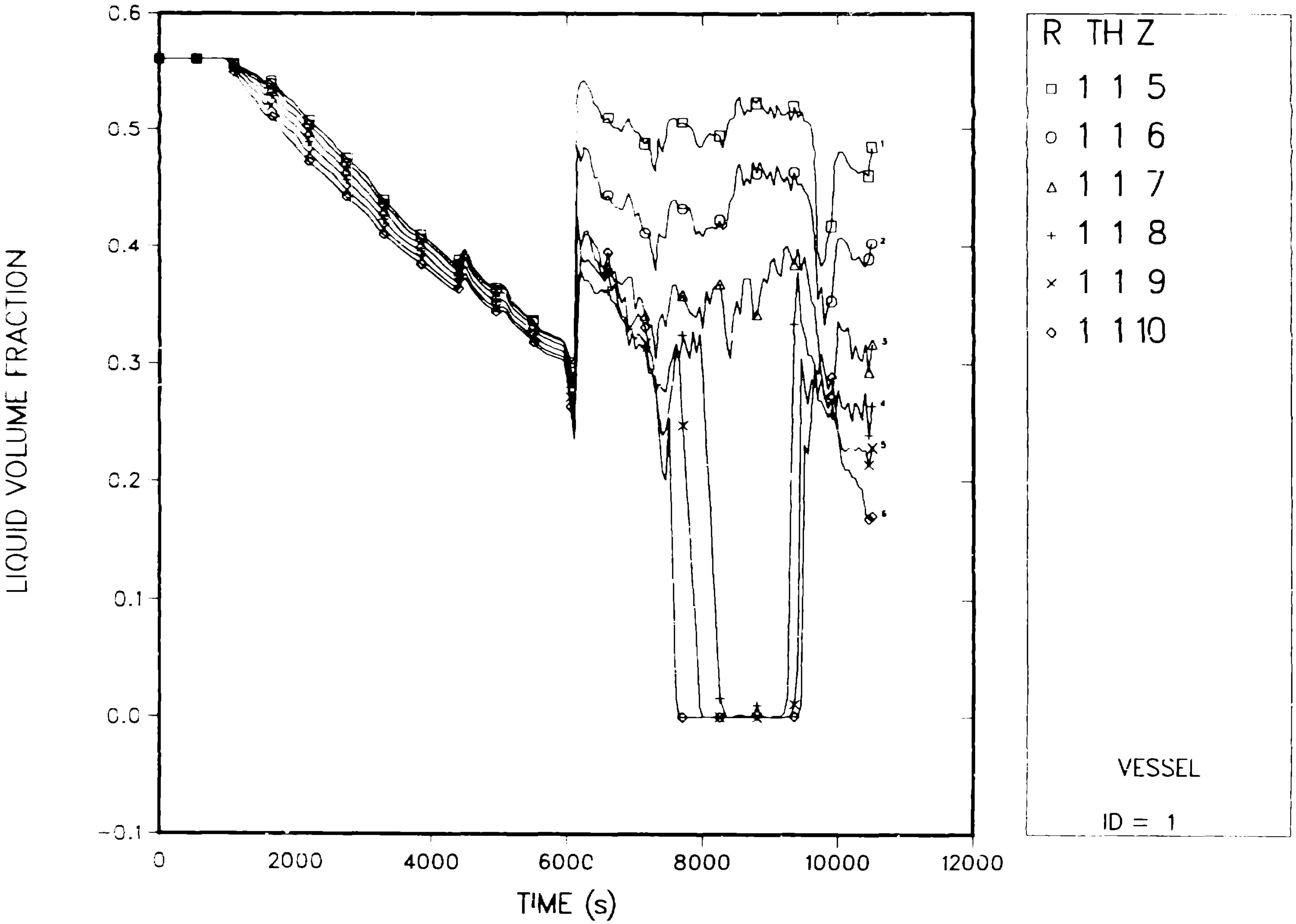


1100
1000
900
800
700
600
500
400
0
2000
4000
6000
8000
10000
12000
TIME (s)
TEMPERATURE (K)
BASE
PARAMETRIC
OBSERVED

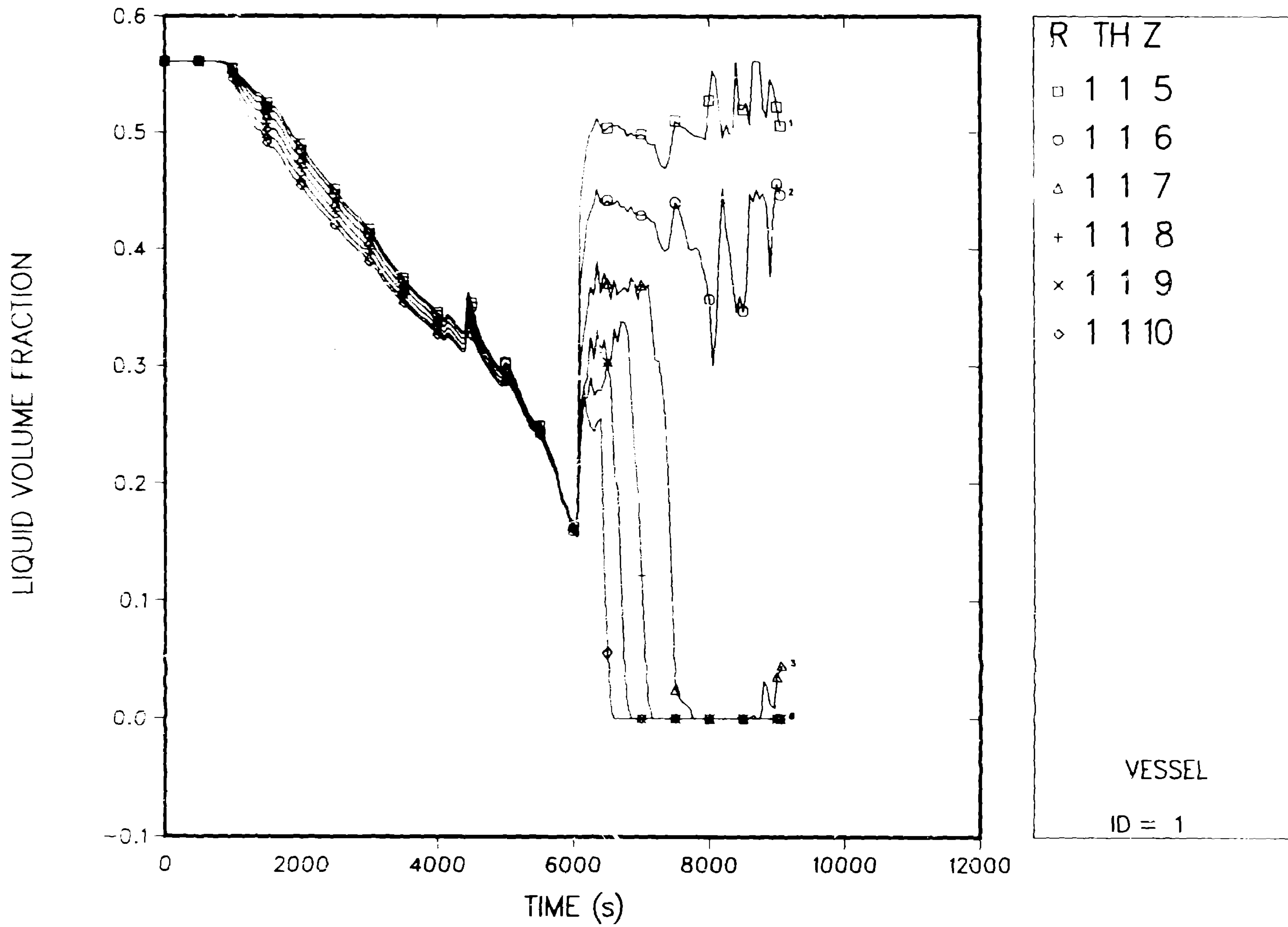
5



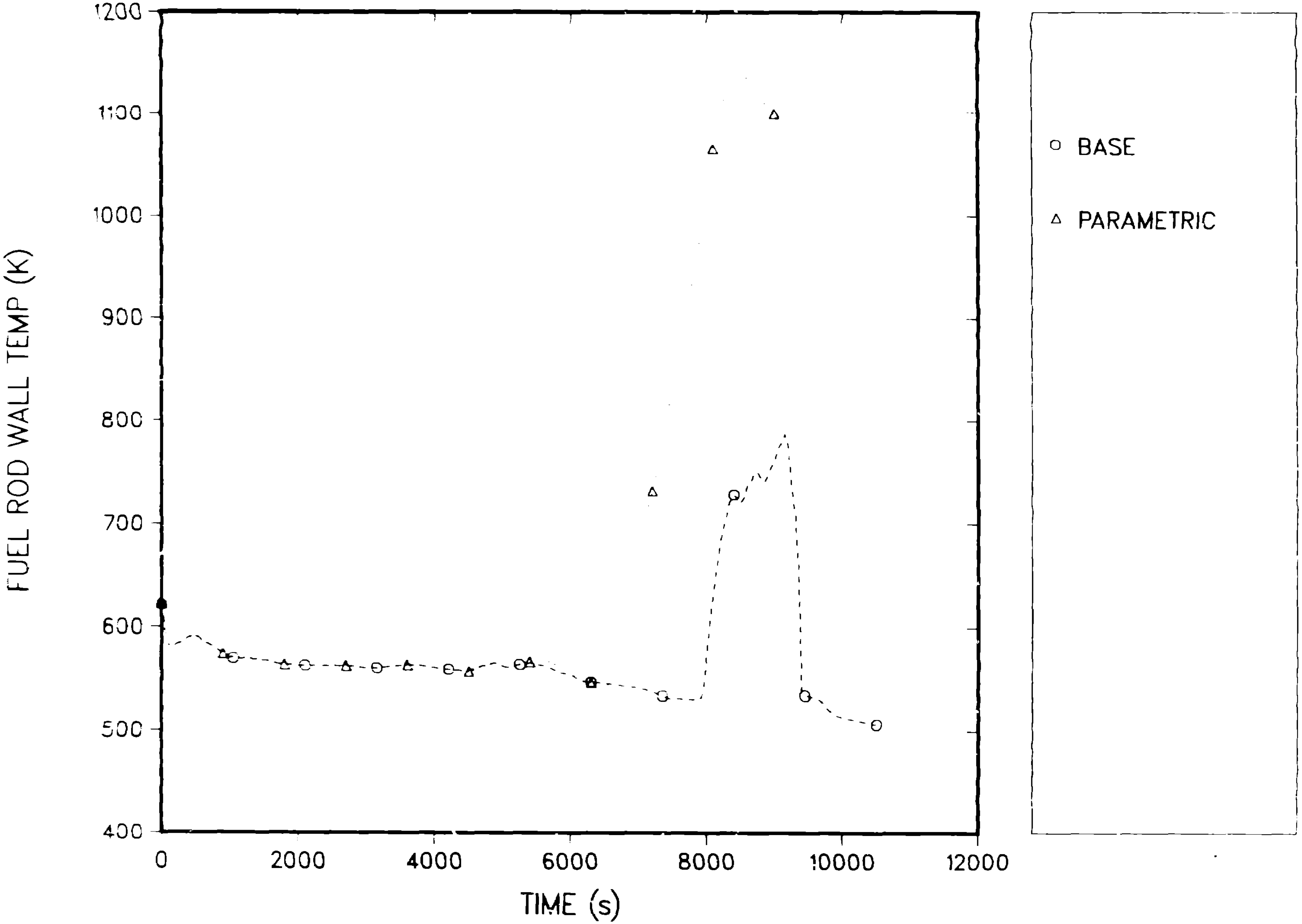
14
12
10
8
6
4
2
0
-2
HEIGHT (M)
0
2000
4000
6000
8000
10000
12000
TIME (s)
BASE
PARAMETRIC
OBSERVED



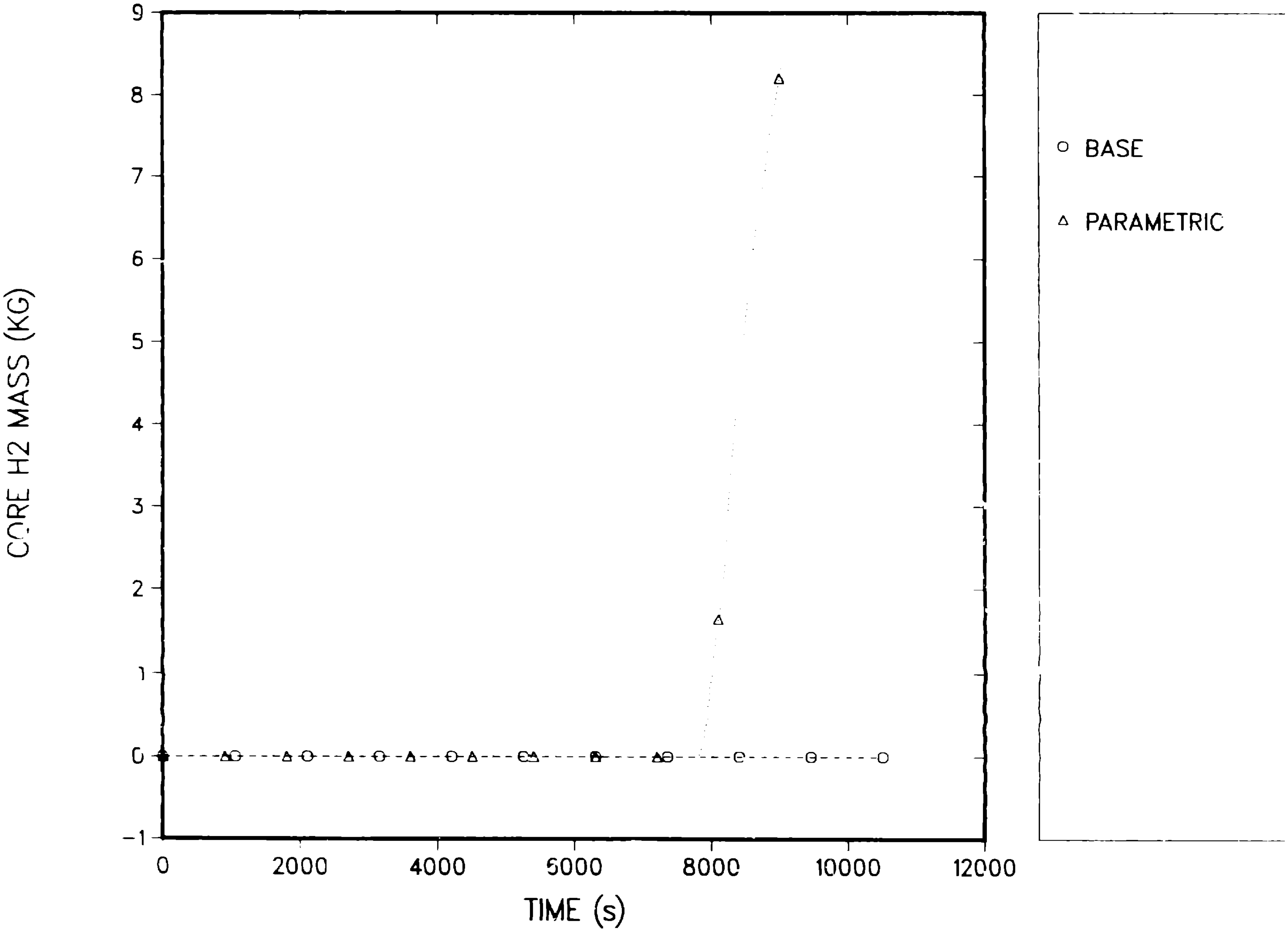
LIQUID VOLUME FRACTION
TIME (s)
0.6
0.5
0.4
0.3
0.2
0.1
0.0
-0.1
0
2000
4000
6000
8000
10000
12000
R TH Z
1 1 5
1 1 6
1 1 7
1 1 8
1 1 9
1 1 10
VESSEL
ID = 1



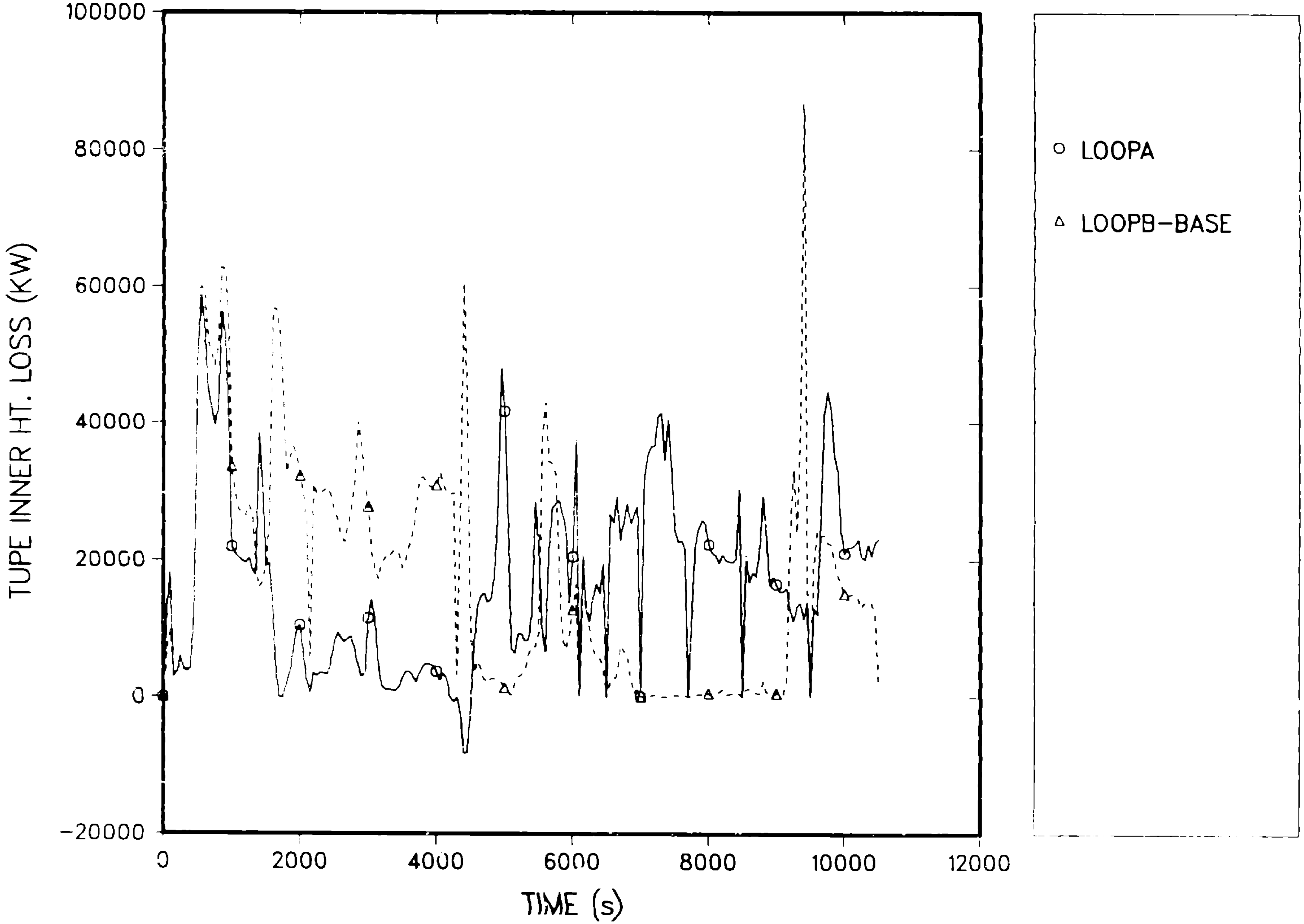
LIQUID VOLUME FRACTION
0.6
0.5
0.4
0.3
0.2
0.1
0.0
-0.1
0
2000
4000
6000
8000
10000
12000
TIME (s)
R TH Z
1 1 5
1 1 6
1 1 7
1 1 8
1 1 9
1 1 10
VESSEL
ID = 1



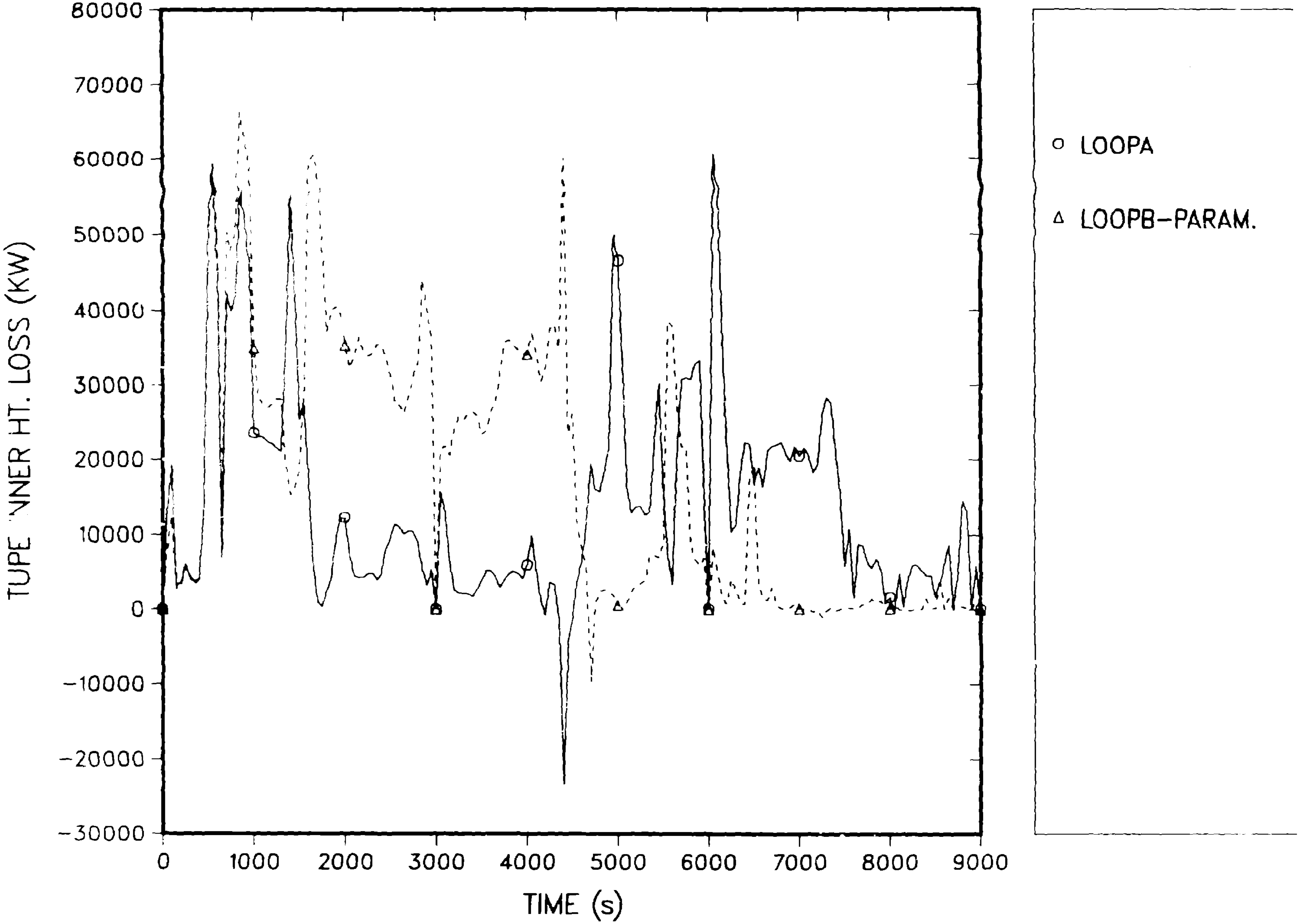
FUEL ROD WALL TEMP (K)
1200
1100
1000
900
800
700
600
500
400
0
2000
4000
6000
8000
10000
12000
TIME (s)
BASE
PARAMETRIC



CORE H2 MASS (KG)
TIME (s)
BASE
PARAMETRIC



100000
80000
60000
40000
20000
0
-20000
TUPE INNER HT. LOSS (KW)
0
2000
4000
6000
8000
10000
12000
TIME (s)
LOOPA
LOOPB-BASE



80000
70000
60000
50000
40000
30000
20000
10000
0
−10000
−20000
−30000
TUPE 'NNER HT. LOSS (KW)
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
TIME (s)
LOOPA
LOOPB−PARAM.