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Assessment Study of RELAP5/MOD2 Cycle 36.05 Based on the Tihange–2 Reactor Trip of January 11, 1983

Prepared by
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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NOTICE

Abstract

This report presents a code assessment study for RELAP-5 Mod-2/CYCLE 36.05 based on a plant transient (TIHANGE 2 power plant following reactor trip).
The plant trip from full power was performed as part of a commissioning test series on January 11th, 1983, and most important plant parameters were recorded on a Data Acquisition System (DAS).
The analysis by means of the frozen version of the RELAP-5/MOD-2/CYCLE 36.05 code was performed to qualify the plant input data deck for this plant and assess the code potential for simulating such transient.
This work is performed by TRACTEBEL, which is the Architect-Engineer for all Belgian nuclear power plants and a member of ICAP.

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Executive Summary

As part of a commissioning test series, a plant trip from 100 % power was performed at the TIHANGE-2 nuclear power plant on Jan. 11th, 1983. The TIHANGE-2 power plant is a 2785 MWth, 3 loop, FRAMATOME designed pressurised water reactor located in the southern part of Belgium. TRACTEBEL was the architect enginneer for the plant which started commercial operation in june 1983 and is since operated by the INTERCOM utility in Belgium.

The test was performed to evaluate the dynamic behaviour of the plant, the steam dump control systems and the feedwater regulating valve response.

A high quality Data Acquisition System (DAS) was operational to record a large number of plan parameters, from which the dynamic behaviour of the various plant systems could be evaluated, and which is also the basis for comparison with the calculated plant response.
The simulation of this transient was performed by means of the code RELAP-5/MOD-2/CYCLE 36.05.

The scope of the simulation includes the primary coolant system, the 3 loops and steam generators (simulated explicitely), the feedwater and auxiliary feedwater systems, the steam lines, steam collector and steam dump systems with associated control logic.

The primary charging flow was taken as a boundary condition, with the DAS flow as a time dependent junction. Similarly, the feedwater flow,

being operated in manual mode was taken as a boundary condition.
The depth of simulation was similar to the recommended nodalisation detail for a full plant, leading to 264 volumes and 277 junctions (three loop plant) and an optimised Courant limit of 0.128 sec.

The assessment of the code is based on eight runs of which one case (run 12A) was taken as the reference calculation. The various runs were performed

a) to investigate the impact of uncertainties in the boundary conditions:
   - Charging and letdown flow on primary level and pressure (Runs 2A,3A)
   - Steam dump capacity on steam generator temperature and pressure gradient (Run 4A)
   - A delay in the response of the steam dump (Run 6A)
   - An adjustment of the steam dump closing time (manual mode) (Run 12A)
   - A change in the opening ramp of the steam dump (Run 14A)
   - An inversion of the initiating event (turbine trip versus reactor trip) (Run 15A).

b) to investigate the impact of modelisation of the steam generator:
   - Introducing an artificial trip valve in feedwater torus to reduce the condensation in the feedwater sparger (Run 9B) when the water level drops below the feedwater inlet.

From these studies, the following conclusions can be drawn:

The RELAP-5 Mod-2 code is able to simulate the basic thermal hydraulic phenomena that occur in a full scale nuclear power plant following a reactor trip.

A generic problem for code assessment based on real plant transients is the quality of the recorded plant data.
This test shows that, notwithstanding the high quality of the data acquisition system (DAS), the data are affected by a rather large uncertainty due to imprecision or offset of the many sensors,which are large compared to the high precision of the data obtained from separate effect tests or from integral scaled facilities.

The limited number of sensors available on a full scale plant, precludes one to improve the code constitutive equations. However, the basic merits of such assessment study should be:

- to gauge the scaling effect on the code models and correlations
- to uncover some code weaknesses which then should be improved on the basis of separate effect tests.

Agreement between recorded and calculated parameters is considerably better for the primary coolant system than for the steam generators. This is a general observation for most full scale plant transients when the reactor coolant system remains highly subcooled.
The basic reasons are:

- The single phase treatment does not pose any challenge to the code, while the highly two-phase nature of the fluid in the steam generators constitutes a real test of the delicate closure equations dealing with the interphase mass, momentum and heat transfer.
- The variations of the primary coolant parameters are highly buffered by the steam generators which take the largest share of the power mismatch upon sudden turbine valve closure and steam dump activation.

The parametric study clearly highlights the fundamental importance of boundary conditions for the plant model. The study shows that apparent minor changes to the timing and dynamics of the steam dump can induce rather large variations in the steam generator parameters (e.g. water level indication). Before deciding that the code manifests some weaknesses, one should filter out modeling uncertainties by performing a lot of sensitivity studies on the parameters of the systems which fix the boundary conditions of the modeled conponents. For any plant,these parameters are never known to very high accuracy. This concerns especially the secondary side steam and feedwater components and related systems such as:

- Atmospheric steam relief valves
- Steam dump systems
- Feedwater level control systems

This report underscores very well the need for the code users,not only to have a good understanding of the code and its limitations, but also to acquire a detailed knowledge of the plant and the functionning and the location of the plant sensors. An example is the temperature measurement for the primary coolant system which is obtained in the RTD bypass loops of the primary loops. The non negligible fluid transport time in these bypass loops does affect the timing of the steam dump behaviour, and should be accounted for by suitable delays if these loops are not simulated explicitely.

From this assessment study, one can conclude that the two phase code models do not tolerate high thermal desequilibrium conditions for the bubbly flow regime under fast pressurisation, and that, due to premature condensation, the temperature of the gas phase returns too soon to the quasi saturation conditions.This effect shows up as a temporary stagnation of the pressure and an abnormal water level response in the steam generators. One has to keep in mind that these 2 parameters are basically the only 2 parameters that the operator sees in the control room and upon which the control systems are based for plant protection and control.

The level swell observed in the recording following the manual opening of the steam dump (t > 142 sec) was not observed in the simulation. Since the recorded level swell was so small (L < 2 %), a deeper analysis of this discrepancy was not undertaken. Indeed, a small offset in the steam generator water inventory could well be the reason for such anomaly.

The run statistics illustrate that the code ran smoothly through the transient without changing the time step and with negligible mass error. However, in order to achieve real time for the given size of the model (264 volumes), the computer performance should be at least 50 MIPS equivalent.

1. INTRODUCTION

Predicting the behavior of full scale nuclear reactors is the ultimate goal of the RELAP-5 code and all the studies related to its assessment.

Reproducing the few simple transients obtained in a full scale plant adds confidence that the code can be used to predict some other more "difficult" transients for which data are not available. To reach this conclusion that the code can be used in such predictions, several points have to be investigated:
of course and above all the pertinence of the correlations used by the code, but also the suitability of the modelisation as well as the uncertainty of the measurements.

This report examines these points in relation with the TIHANGE-2 reactor trip of January 11th, 1983. It is organised as follows: Section 2 gives a brief description of the TIHANGE-2 power plant unit; section 3 describes the transient as recorded on the power plant Data Acquisition System; section 4 presents the RELAP-5 model used to simulate the transient; in section 5, the base case calculational results are discussed; in section 6, a parametric study shows the various steps taken before obtaining the base case ; section 7 highlights some run time statistics. The conclusions are presented in section 8.

2. BRIEF DESCRIPTION OF THE TIHANGE-2 PLANT

TIHANGE-2 is a 2785 MWth (941 MWe) pressurized water reactor located on the right bank of the river Meuse upstream of the city of Huy (Belgium) and featuring a 3-loop, FRAMATOME designed Nuclear Steam Supply System.

The plant was connected to the grid in June 1983.
Only the subsystems playing a significant role in the events following the simulated transient i.e. the January 11th, 1983 plant trip are described hereafter, namely the reactor coolant system, the main and auxiliary feedwater supply system, the main steam lines and the steam dump, and the associated control systems.

2.1. Reactor Coolant System

The Reactor Coolant System flow diagram (Fig. A.1) summarizes the functional and physical links between the main primary components : reactor vessel, steam generators, primary pumps and pressurizer.

The core of TIHANGE-2 contains 157 fuel assemblies with 264 fuel rods per assembly, generating 2775 MW of thermal power, under nominal operating conditions. The primary pumps, rated at 5.35 MW each, circulate 4590 kg/s of coolant per loop with a net pump head of 5.64 bar.

The primary coolant volume changes associated with the reactor load evolution are being accomodated by a 39.6 m3 (1400 ft3) pressurizer (see flow diagram Fig. A.2) connected to the hot leg of loop B through a 14" surge line.Control of the primary pressure also take place within the pressurizer by adjustment of the heater rod power and/or the pressurizer spray flowrate.

2.2. Steam Generators, Feedwater System and Steamlines

The TIHANGE-2 steam generators are of the inverted U-type design (series 51M) , with a nominal recirculation ratio of about 4.7. The U tube bundle, with a nominal heat transfer area of 4785 m2 consists of 3361 Inconel tubes with a 22.22 mm outer diameter.

The main feedwater (see flow diagram Fig. A.3) with a nominal flow rate of 1814 t/hr per steam generator,enters the secondary side of the steam generator in the separator region, slightly under the water free level; flowing from a 10", doughnut-shaped sparger that provides for a reasonably uniform distribution, it mixes with the recirculated water ejected from the separator. The flow diagram of the main steam system is shown on Fig. A.4. It features, among others, the atmospheric steam relief valves (one per steam generator) with an individual capacity of 200 t/hr at 70.3 bar, the steam generator safety valves (6 per steam generator) and the main steam isolation valves (2 per steam line).

2.3. Auxiliary Feedwater System

The Auxiliary Feedwater System consists of two motor driven auxiliary feedwater pumps feeding each two steam generators and one steam driven auxiliary feedwater turbopump , normally aligned with two steam generators such that each steam generator is normally fed by two auxiliary feedwater pumps. The system is designed such that in the automatic mode each steam generator is supplied by a fixed, metered flow of 80 t/hr regardless of the steam generator back pressure.

2.4. Steam Dump

The Steam Dump (see flow diagram on Fig. A.5) is the single most important piece of control equipment during the period following strong load variations such as reactor scram : at TIHANGE-2, it consists of sixteen valves of identical capacity (304 t/hr at 57.5 bar) opening in sequence as instructed by a controlling program built around the steam header pressure at low load and around the maximum average primary temperature at high load or after scram.

Within the considered sequence of events (see chapter 3) i.e. a stable operation followed by an operator-induced reactor trip, only the post-scram operating conditions lead to the steam dump activation : under such circumstances, eight of the sixteen valves (i.e. those belonging to the so-called groups 1 and 2) are allowed to open, freeing up to 50 % of the overall steam dump capacity.

In the plant, the instantaneous turbine load is derived from a pressure gauge in the first expansion stage of the turbine and is converted to a so-called turbine power reference temperature. The reactor power is obtained from a auctioneered average primary temperature signal.Any power mismatch is thus measured in terms of a discrepancy or error signal between the reference temperature and the auctioneered average primary temperature. Whenever the measured auctioneered average primary temperature exceeds the no-load reference temperature after scram (286 C) , the steam dump starts to open, aiming at a capacity proportional to the error signal (4.5 % per deg.C). The time needed for each valve or group of valves to reach the full open position is 7 sec. However, for large error signals, an accelerated opening sequence takes over, making available in three seconds the full capacity of the first group (4 valves) whenever the signal exceeds 6 deg.C and the full capacity of groups 1 and 2 (8 valves) beyond 11 deg.C.

Capacity reductions follow the same path in reverse, however without an accelerated sequence and with a closing time of 5 seconds instead of 7 seconds.

2.5. Measurements

The plant is equipped with a dedicated Data Acquisition System (DAS), enabling a high quality digital recording of 240 plant parameters. The on-line system is continuously recording and erasing data from the 240 channels, but stops erasing as soon as one of 24 important logical signal arrives, such as SCRAM, SI, etc. This enables the users to trace back the origin of plant disturbances when they lead to a serious plant transient. On the basis of such recorded data, displayed in graphical form, a comparison of the plant data and the simulation data is presented in this study.

The combined uncertainties affecting the sensor position, sensor response and signal handling have been estimated at:

9 % of nominal power for flux measurements ;

1.5 deg.C for primary temperatures ;

1.7 bar for pressurizer pressure ;

3 % of the range for pressurizer level ;

2 bar for steam generator pressure ;

2.5 % of the narrow range for steam generator level.

These figures are to be combined with an additional uncertainty estimated in all cases at 3 % of the range and accounting for the lack of recording accuracy.

3. DESCRIPTION OF THE PLANT TRANSIENT

As part of a commissioning program for the TIHANGE-2 plant, this plant trip test at nominal power was performed in order to evaluate the overall plant behavior,and more specifically to test the steam dump control systems and the closure time of the main feedwater regulating valves.

The test program called for manual opening of the feedwater control valve of steam generator B for about 5 sec. At this point, one forced 2 out of 3 water level indications at very high level in steam generator B which caused a turbine trip followed by a reactor scram. About two minutes later, the operator took the steam dump in a manual mode to force a cooldown below the no-load reference temperature.

The DAS was triggered manually 50 sec prior to the operator intervention and recorded the most important plant parameters for about 30 minutes. For this study only the first three minutes were selected as they highlight the most dynamic part of the transient,suitable for code assessment. All times indicated below refer to the time zero for the DAS recordings, which are shown in figures 3.1 to 3.6.
All figures illustrate the evolution of the most important plant parameters over a time interval of 8 minutes, except figure 3.6 which shows the evolution during 16 minutes.
The following parameters are shown:

Fig. 3.1 : For steam generator B (GV 02)
Curve 1 : Steam pressure

Curve 2 : Steam flow rate

Curve 3 : Feedwater flowrate

Curve 4 : Narrow range water level

Fig. 3.2 : For steam generator R (GV 03)

Curve 1 : Steam pressure

Curve 2 : Steam flow rate

Curve 3 : Feedwater flowrate

Curve 4 : Narrow range water level

Fig. 3.3 : For steam generator G (GV 04)

Curve 1 : Steam pressure

Curve 2 : Steam flow rate

Curve 3 : Feedwater flowrate

Curve 4 : Narrow range water level

Fig. 3.4 : Primary system parameters

Curve 1 : Demand signal for rod position of bank D

Curve 2 : Average nuclear power

Curve 3 : Primary system reference temperature

Curve 4 : Primary system average temperature

Curve 5 : Mismatch between reference and average temperature

Fig. 3.5 : Primary loop B parameters (RTD signals)

Curve 1 : Hot leg temperature

Curve 2 : Cold leg temperature

Curve 3 : Average loop temperature

Curve 4 : Reference temperature

Curve 5 : Temperature difference

Fig. 3.6 : Pressurizer parameters

Curve 1 : Primary pressure

Curve 2 : Programmed pressurizer level

Curve 3 : Pressurizer level program output signal

Curve 4 : Error between programmed and measured level

3.1. Plant status prior to transient (0 < t < 50 s)

The reactor was operating at nominal power conditions in a full automatic control mode for the primary coolant temperature, pressurizer pressure and level and steam generator level as a function of the turbine load. The recorded data point to some parameters which slightly deviate from the nominal conditions such as:

- Neutron flux detectors at 96.7 %
- Primary coolant hot-cold leg temperature difference of 38.5 deg.C, that is 108 % of nominal value (35.7 deg.C)
- Feedwater flow rate to steam generator G at 107 %
- Pressurizer pressure of 156.8 bar (nominal = 155 bar)

It was observed also that the temperature difference measurement (38.5 deg. C) is not consistent with the temperature measurements themselves: (322.8 - 285.66 = 37.14 deg.C)

3.2. Phase of excessive feedwater flow (50 < t < 55 s)

At t = 50 sec, the operator takes the feedwater flow to the steam generator B in manual mode and forces the feedwater control valve fully open to increase the flow from 1840 to 2300 t/hr. (Fig. 3.1)

This leads to a small increase in the narrow range level indication on

the affected steam generator. While the feedwater pumps are kept at a constant speed, the increased flow to steam generator B reduces the feedwater flow to the other steam generators because they are fed from a common header. (Figs. 3.2 & 3.3)
The impact on the primary system for this phase is minimal.

3.3. Reactor scram and steam dump phase (55 < t < 142 s)

At t = 55 sec, a false high level signal was generated by forcing two out of three water level indications on steam generator B above 75 %. A very high water level indication in one steam generator causes:

- Turbine trip followed by reactor scram,
- Fast closure of all main feedwater regulating valves,
- Start up of the auxiliary feedwater system.

Upon reactor scram, the no-load reference temperature (286 deg.C) is compared to the average primary coolant temperature (304.6 deg. C) and this error signal activates a fast opening of two out of four banks of the steam dump system to ensure a sufficient heat sink for the primary system once the turbine is tripped.

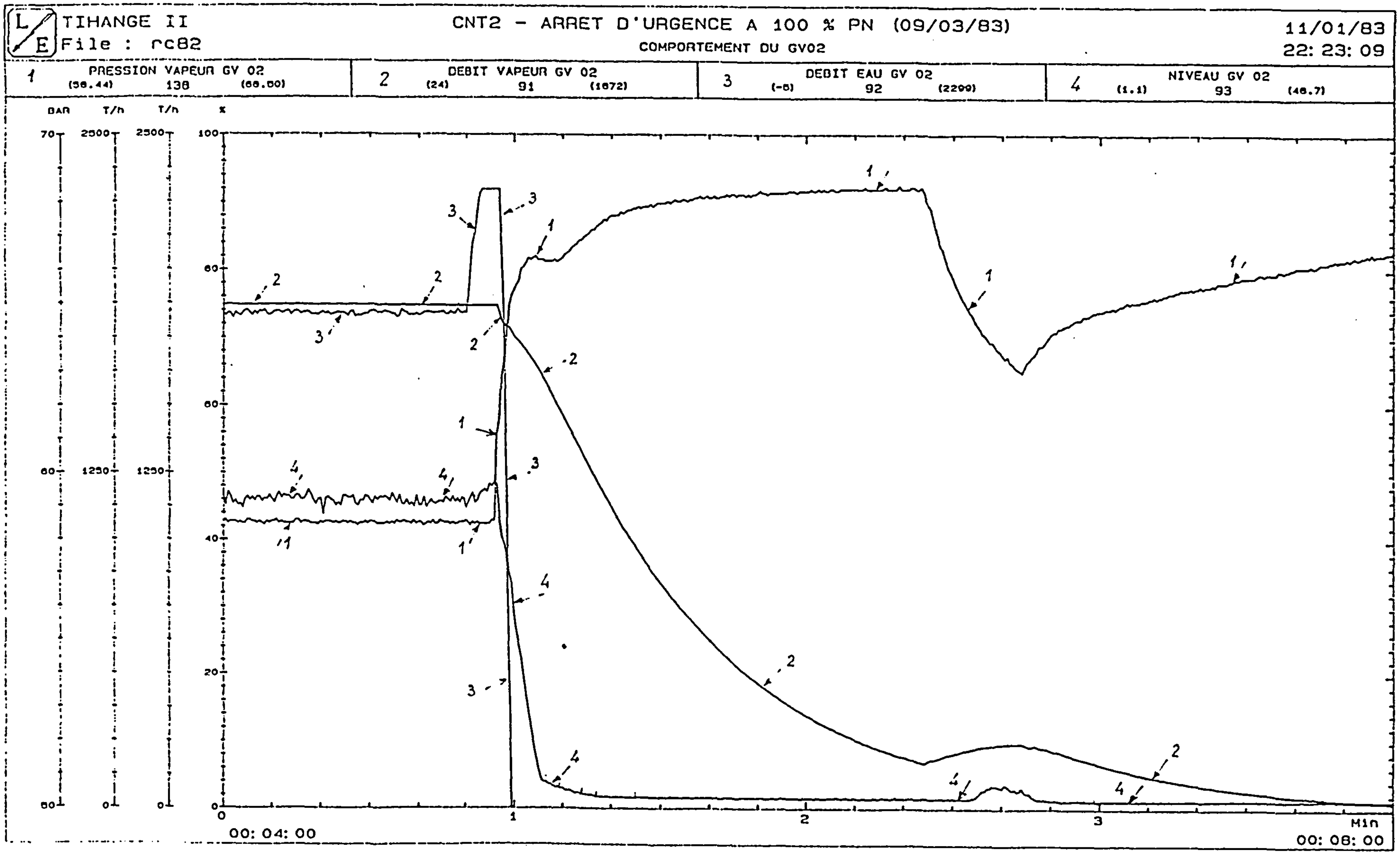
Figures 3.1 to 3.3 illustrate respectively for the three steam generators the evolution of the pressure, main steam and normal feedwater flow and the narrow range water level indication.
The closure of the turbine stop valves produces a fast pressure excursion in all the steam generators which is halted by the fast opening of the steam dump, preventing the opening setpoint (71.7 bar) of the atmospheric relief valves to be reached.

The increasing pressure causes a collapse of the steam bubbles in the steam generator riser section and thus a sudden rise in the two-phase mixture level, which is reflected by a sudden drop of the downcomer level and hence of the narrow range water level indication to off-scale low. The steam flow rate is now under control of the steam dump system, whose valves are closing gradually as the primary coolant temperature tends towards the no-load reference temperature (Fig. 3.4)
Fig. 3.5 illustrates the evolution of the primary coolant temperature in the hot and cold leg of loop B, the average temperature and the loop temperature difference.
Fig. 3.6 illustrates the evolution of the pressurizer pressure and the water level. (notice different time scale)

3.4. Manual control of the steam dump (t > 142 s)

At about 2'23" , the steam dump valves of the first bank are activated manually for about 22 secondes, which leads to a sudden pressure drop in all three steam generators. Although the water level indication dropped to zero at scram, the sudden pressure drop leads to a water level swell showing up as a very small water level increase in the narrow range water level gauges for all three steam generators.(Fig 3.1 to 3.3)
On the primary side,manual control of the steam dump leads to a further drop of the temperature (Fig. 3.5), the pressurizer water level and pressure (Fig. 3.6)

Figure 3.1



TIHANGE II
File : rc82
CNT2 - ARRET D'URGENCE A 100 % PN (09/03/83)
COMPORTEMENT DU GV02
11/01/83
22: 23: 09
1 PRESSION VAPEUR GV 02 (56.44) 138 (66.50)
2 DEBIT VAPEUR GV 02 (24) 91 (1672)
3 DEBIT EAU GV 02 (-5) 92 (2299)
4 NIVEAU GV 02 (1.1) 93 (46.7)
BAR
T/h
T/h
%
Min
00: 04: 00
00: 08: 00

Figure 3.2

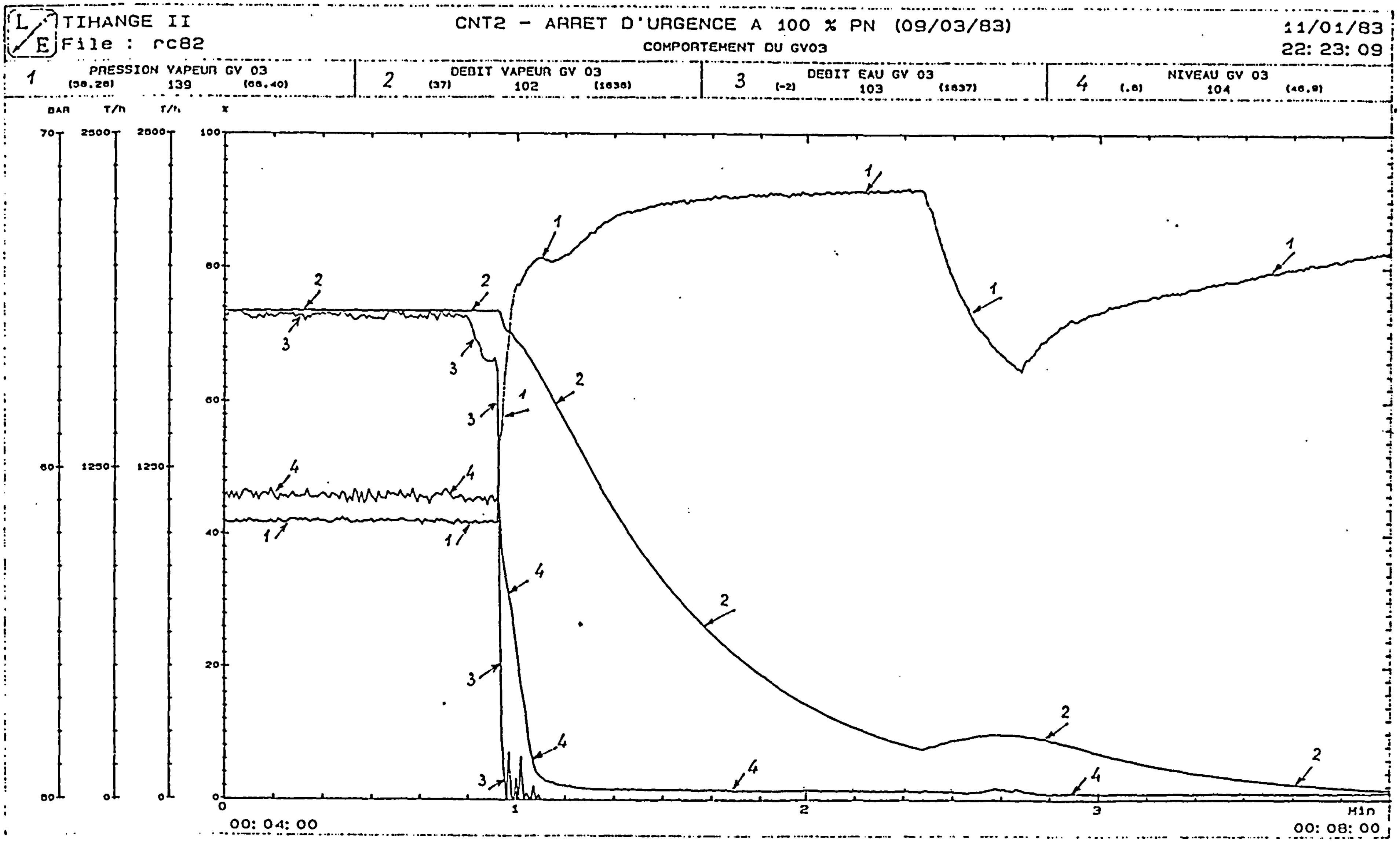
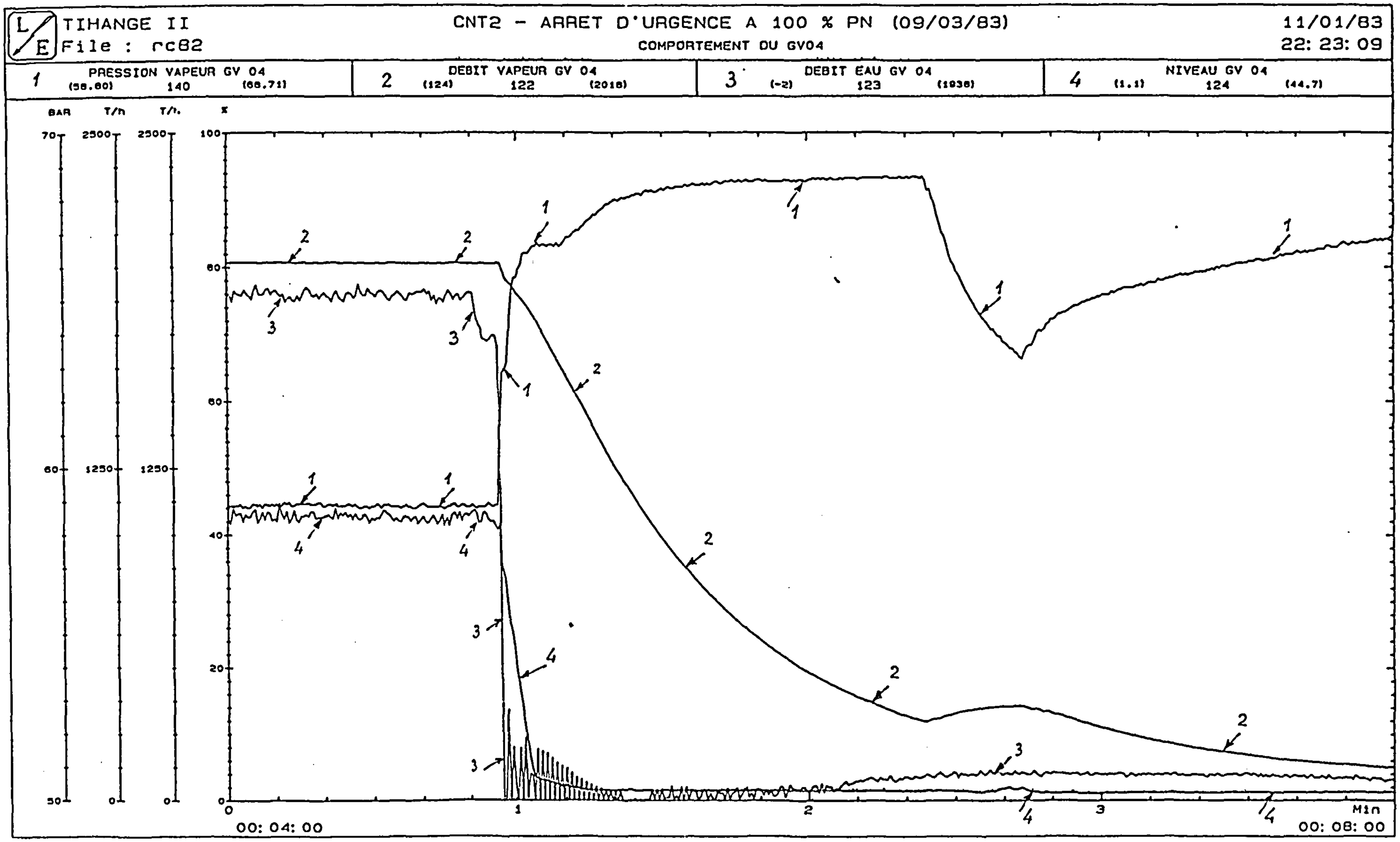

TIHANGE II
File : rc82
CNT2 - ARRET D'URGENCE A 100 % PN (09/03/83)
COMPORTEMENT DU GV03
11/01/83
22: 23: 09
1 PRESSION VAPEUR GV 03 139
2 DEBIT VAPEUR GV 03 102
3 DEBIT EAU GV 03 103
4 NIVEAU GV 03 104
BAR T/h T/h %
Min
00: 04: 00
00: 08: 00

Figure 3.3



TIHANGE II
File : rc82
CNT2 - ARRET D'URGENCE A 100 % PN (09/03/83)
COMPORTEMENT DU GV04
11/01/83
22: 23: 09
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2 DEBIT VAPEUR GV 04 (124) 122 (2018)
3 DEBIT EAU GV 04 (-2) 123 (1938)
4 NIVEAU GV 04 (1.1) 124 (44.7)
BAR
T/h
T/h
%
00: 04: 00
00: 08: 00
Min

Figure 3.4

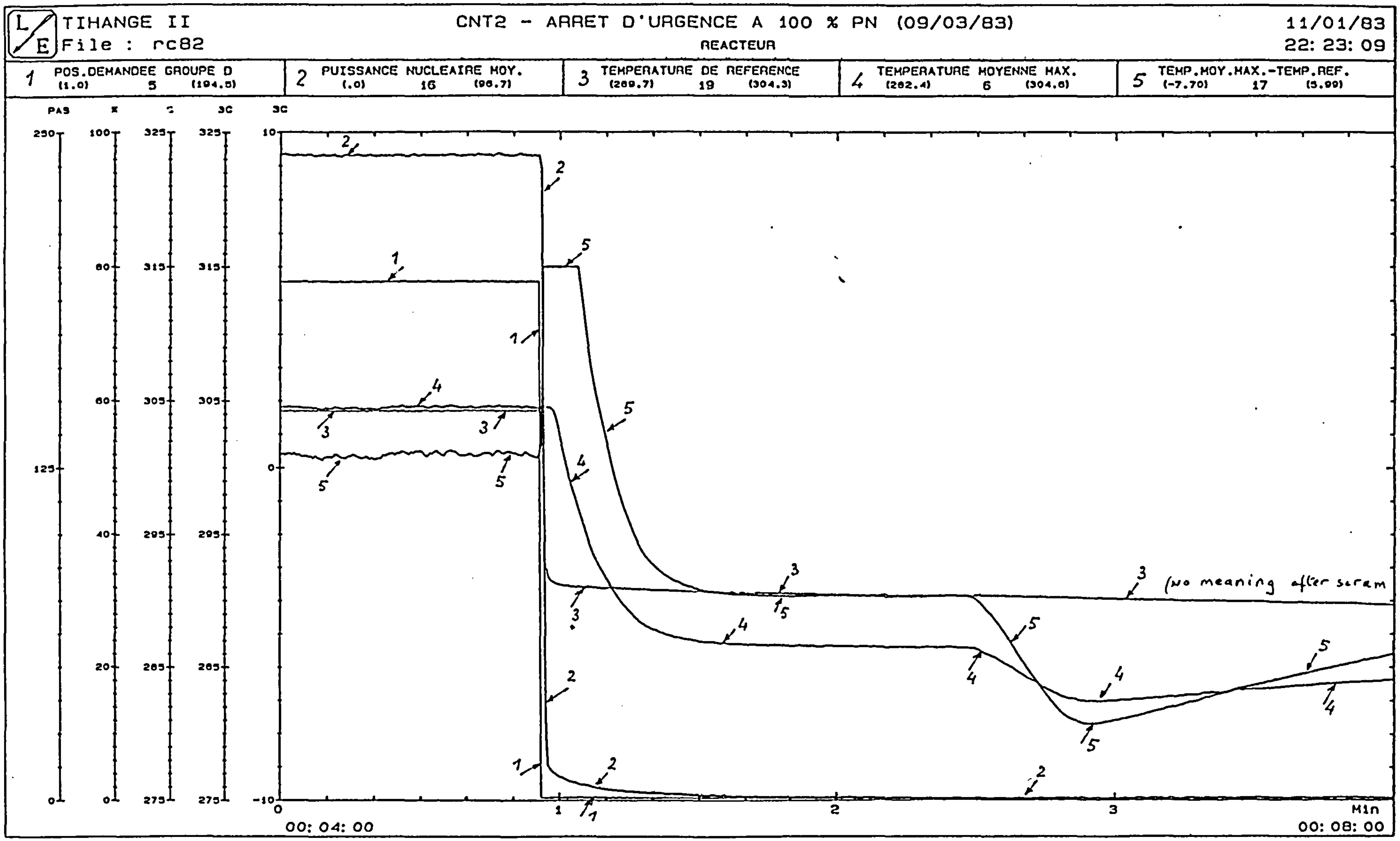

TIHANGE II
File : rc82
CNT2 - ARRET D'URGENCE A 100 % PN (09/03/83)
REACTEUR
11/01/83
22: 23: 09
1 POS.DEMANDEE GROUPE D (1.0) 5 (194.5)
2 PUISSANCE NUCLEAIRE MOY. (.0) 16 (96.7)
3 TEMPERATURE DE REFERENCE (289.7) 19 (304.3)
4 TEMPERATURE MOYENNE MAX. (282.4) 6 (304.6)
5 TEMP.MOY.MAX.-TEMP.REF. (-7.70) 17 (5.99)
PAS
%
3C
3C
(no meaning after scram
Min
00: 04: 00
00: 08: 00

Figure 3.5

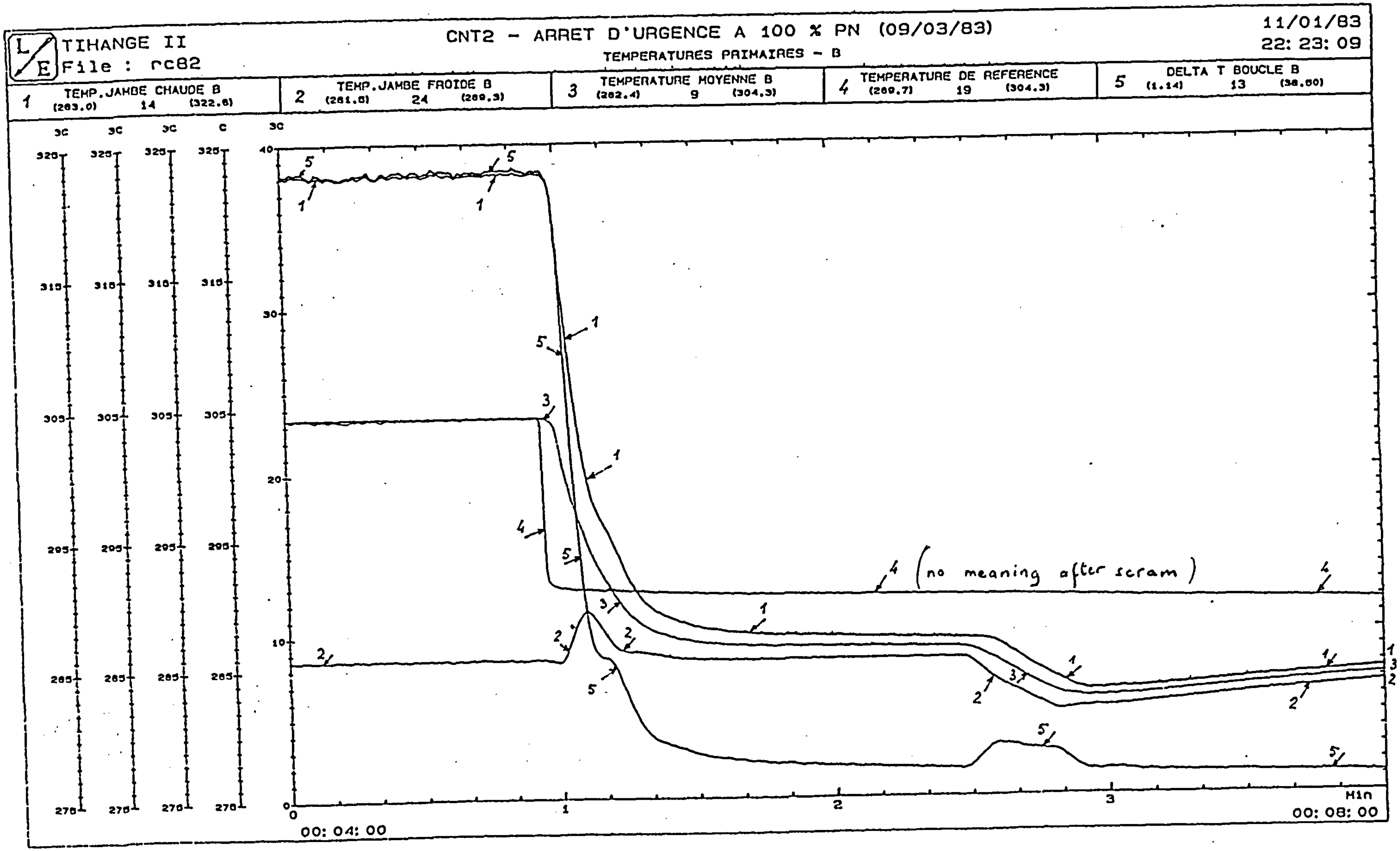
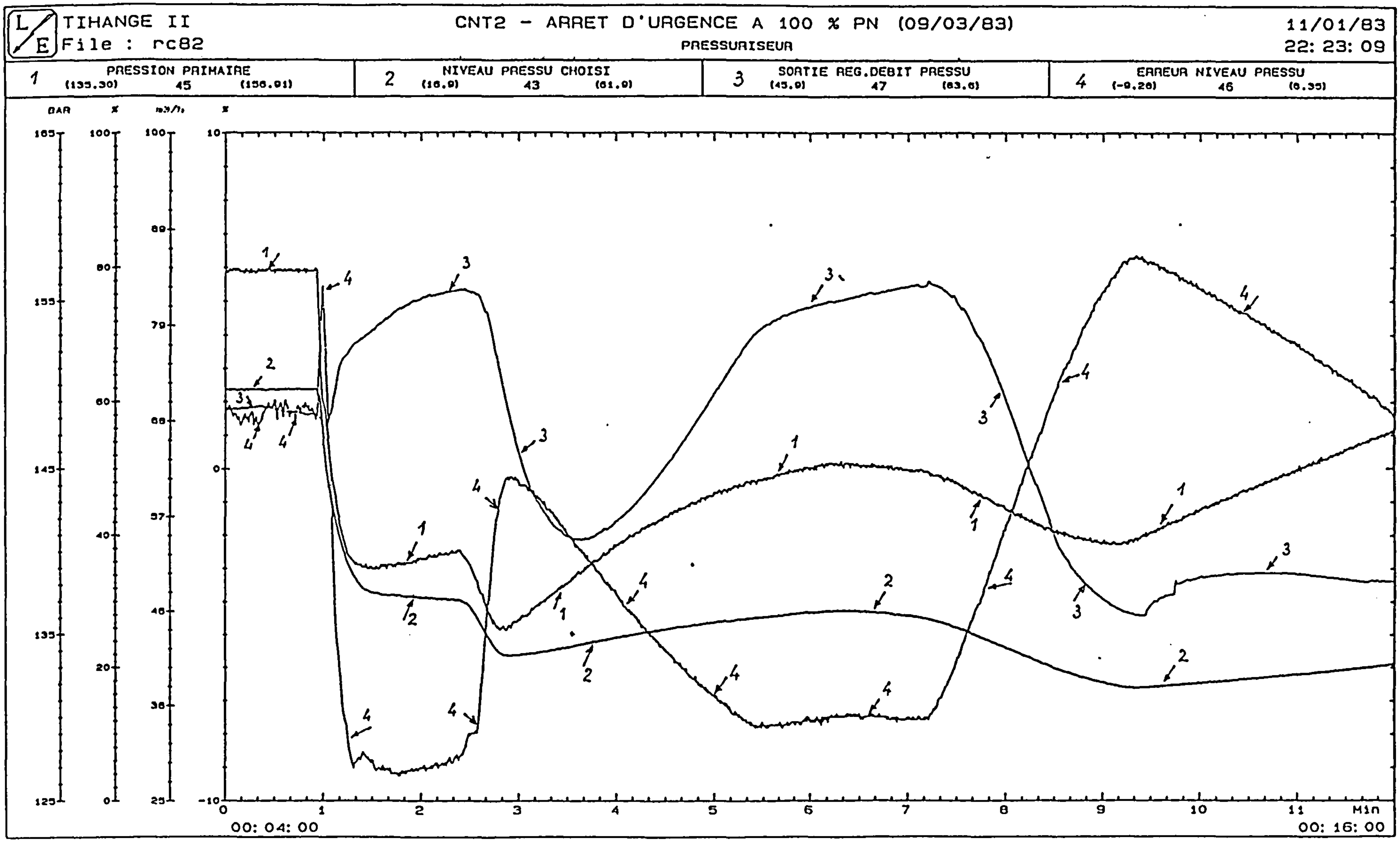

TIHANGE II
File : rc82
CNT2 - ARRET D'URGENCE A 100 % PN (09/03/83)
TEMPERATURES PRIMAIRES - B
11/01/83
22: 23: 09
1 TEMP.JAMBE CHAUDE B
2 TEMP.JAMBE FROIDE B
3 TEMPERATURE MOYENNE B
4 TEMPERATURE DE REFERENCE
5 DELTA T BOUCLE B
(no meaning after scram)
Min
00: 04: 00
00: 08: 00

Figure 3.6



TIHANGE II
File : rc82
CNT2 - ARRET D'URGENCE A 100 % PN (09/03/83)
PRESSURISEUR
11/01/83
22: 23: 09
1 PRESSION PRIMAIRE (135.30) 45 (156.91)
2 NIVEAU PRESSU CHOISI (16.9) 43 (61.9)
3 SORTIE REG.DEBIT PRESSU (45.9) 47 (83.6)
4 ERREUR NIVEAU PRESSU (-9.28) 46 (6.35)
BAR
%
m3/h
%
Min
00: 04: 00
00: 16: 00

4. CODE AND MODEL DESCRIPTION FOR PLANT SIMULATION

The simulation was carried-out with the RELAP-5/MOD-2/CYCLE 36.05 code on a CYBER 180/825 computer, over a period of 180 sec. starting at time zero of the recording sequence. It explores thus the most significant aspects of the chain of events, primarily the 85 seconds immediately following the scram and the 40 seconds associated with manual reopening of the steam dump.

The reactor model was developed using the methods and procedures recommended in the code manual (Ref. 1). The primary and three steam generators (feedwater / steam generator / main steam) were modeled explicitely, meaning that the various components are reduced to a series of "volumes", each described by its true geometric features,and connecting with one another through "junctions" incorporating frictional head losses.

All volumes are also exposed to "heat structures" describing the hardware (walls, internals,...) they are in contact with, for heat exchange purposes.

The overall nodalisation totals 264 volumes, 273 junctions and 263 heat structures (see Fig. 4.1)

REF. 1 : V.H. RANSOM et al., "RELAP-5/MOD-2 Code Manual" NUREG/CR-4312 August, 1985.

On the other hand, the auxiliary systems are being simulated " functionally ", i.e. by using the control system package of RELAP-5.

This applies to :

- the control and scram rods ;
- the safety injection system (high and low pressure) ;
- the charging and letdown system ;
- the pressurizer relief (PORV's) and safety valves ;
- the pressurizer spray and heaters ;
- the auxiliary feedwater system ;
- the steam generators relief and safety valves ;
- the turbine ;
- the steam dump .

4.1. Explicitely modeled systems

The primary and secondary systems are split into nine major components identified as follows :

- reactor vessel	: volumes 010 to 099
- primary loop "G"	: volumes 100 to 199
- primary loop "R"	: volumes 200 to 299
- primary loop "B"	: volumes 300 to 399
- pressurizer	: volumes 400 to 499
- feedwater/S.G./steam line "G"	: volumes 600 to 699
- feedwater/S.G./steam line "R"	: volumes 700 to 799
- feedwater/S.G./steam line "B"	: volumes 800 to 899
- steam header	: volumes 900 to 999

Their most significant geometrical and hydrodynamic features are summarized in the annexes :

reactor vessel	:	table A 1 (summary of nodalisation) and
		table A 2 (summary of the pressure loss data)
primary loops	:	table A 3 (summary of nodalisation) and
		table A 4 (summary of the pressure loss data)
pressurizer	:	table A 5 (summary of nodalisation)
steam generator	:	table A 6 (summary of nodalisation) and
		table A 7 (summary of the pressure loss data)

4.2. Functionally modeled systems

While the RELAP-5 control system package is a powerful tool to simulate hydraulic systems from a functional point of view,one should be careful and aware when applying this simulation capability that thermal and mechanical inertia effects are not accounted for unless suitable delay times are introduced, and that the numbering of trip and control variables should respect the physical sequence of events.

4.2.1. Control and scram rods

Control rods displacements are being simulated as anti-reactivity injections whose amplitude and time-dependence are controlled by the error on the average primary temperature. Scram rods are treated the same way, although the anti-reactivity injection rate and amplitude only depend on the time elapsed since the scram signal occured. The finite rod drop time should be accounted for if the anti-reactivity insertion rate is important.

4.2.2. Safety injection systems

The TIHANGE-2 plant has three completely independent and separated trains for safety injection (high and low pressure). The functional simulation of each train can thus be reduced to a time dependent volume (supplying the ECCS water at a given temperature) and a time dependent junction (forcing the ECCS water into the cold legs at a determined flow rate). (TDV X70 and TDJ X71 where X = 1,2,3)
The pressure-dependence of the flow has been tabulated on basis of the pump curves (high and low pressure pumps), of the pressure losses in the connecting lines and the status of the pump minimum flow bypass. Notice that for this study, no safety injection occured.

4.2.3. Charging and letdown systems

The charging flow, a function of the pressurizer level, is being fed into the primary system from a time-dependent junction according to a flowrate versus level table. However, for this analysis, the charging flow system is disconnected and the charging flow recorded by the DAS is fed into the primary system as a boundary condition. (TDV 280, TDJ 281).

The letdown flow, on the other hand, drains the primary loops through a calibrated orifice. It is simulated as a square-root function of the pressure in the originating volume.

4.2.4. Pressurizer relief and safety valves

The pressurizer relief valves (junctions 471,473,475) are represented as motor valves, featuring an "open" and a "close" trip operating at the same pressure.
The safety valves are being handled as servo-valves (junction 461) controlled by a control variable that simulates their pressure cycle.

4.2.5. Pressurizer spray and heaters

A small, constant spray flow - the "residual spray" - is supplied to the pressurizer whenever the primary pumps are operating.
At high pressures, it is complemented by a variable flow starting at 1.7 bar and peaking at 4 bar above the pressure set point. The constant flow is modeled as two time-dependent junctions from loops R and B (junctions 441 and 442) tripping with the primary pumps, while the pressure-dependent variable flow is supplied by two adequately sized servo-valves (junctions 425 and 435) that are part of the explicitely modeled spray lines.

All pressurizer heaters are constructively identical. Functionally, however, they fall into two groups: the proportional heaters (433 kW) provide the standard regulation capability needed to keep the pressurizer pressure within the desired range; the back-up heaters (988 kW) operate on an on/off basis to counter the low pressures and/or high levels that cannot be easily corrected with the first group alone.

4.2.6. Auxiliary feedwater system

The auxiliary feedwater system of TIHANGE-2 has been designed to supply cold water at a constant flow of 22 kg/s to each steam generator. When activated, the auxiliary coolant is being forced into the normal feedwater line close to the containment penetration, using a time dependent volume (TDV X00 for temperature) and a time dependent junction (TDJ X01 for flow) wherein X = 6, 7 or 8 for steam generators G, R or B respectively.

4.2.7. Steam generator relief and safety valves

Each steam generator relief valve (Junction X41) is modeled as a servo-valve. On the other hand, all six safety valves have been combined into a single servo-valve (Junction X43) with a response closely similar to that of the overall system.

4.2.8. Turbine

The turbine admission valves and stop valves have been simulated as a servo-valve (Junction 905) whose capacity closely parallels the turbine flow, using a calibrated critical valve area. The turbine trip logics are also included.

4.2.9. Steam dump

The complexe steam dump system is reduced to a single time dependent volume (V950) and a single time dependent valve junction (J925). All 16 steam dump valves have been lumped in a single servo-valve junction whose critical area was calibrated on the basis of the total steam dump capacity of 4864 t/hr at a pressure of 57.5 bar abs . The control logic has been made to incorporate all significant parameters contributing to the evaluation of the dump capacity at any given time or circumstance:

Reactor status (pre or post-scram) ;

Current availability of the dump ;

Amplitude of the controlling error ;

Signal valves opening rates (slow/fast/reverse) ;

A one second delay between fast opening signal and steam dump valve opening was accounted for. Figure 4.2 illustrates the steam dump control logic used in this simulation.

4.3. Systems not simulated

If the scope of the simulation is limited to the systems described above, it is essential to impose suitable boundary conditions in the RELAP input model for those parameters which are normally derived from non-simulated components :

- Without simulation of the balance of plant, the feedwater temperature has to be imposed as a boundary conditions. (TDV X03)
- Since the feedwater turbopumps are not modeled and the feedwater level control model is not used, the feedwater flowrate has been imposed as a boundary condition, from the DAS data. (TDV X04)
- The charging flow has also been imposed as a boundary condition from the DAS data (TDV 280,TDJ 281) and the letdown flow orifice (TDJ 181)

has been calibrated to reproduce the steady state letdown flow given by the DAS.

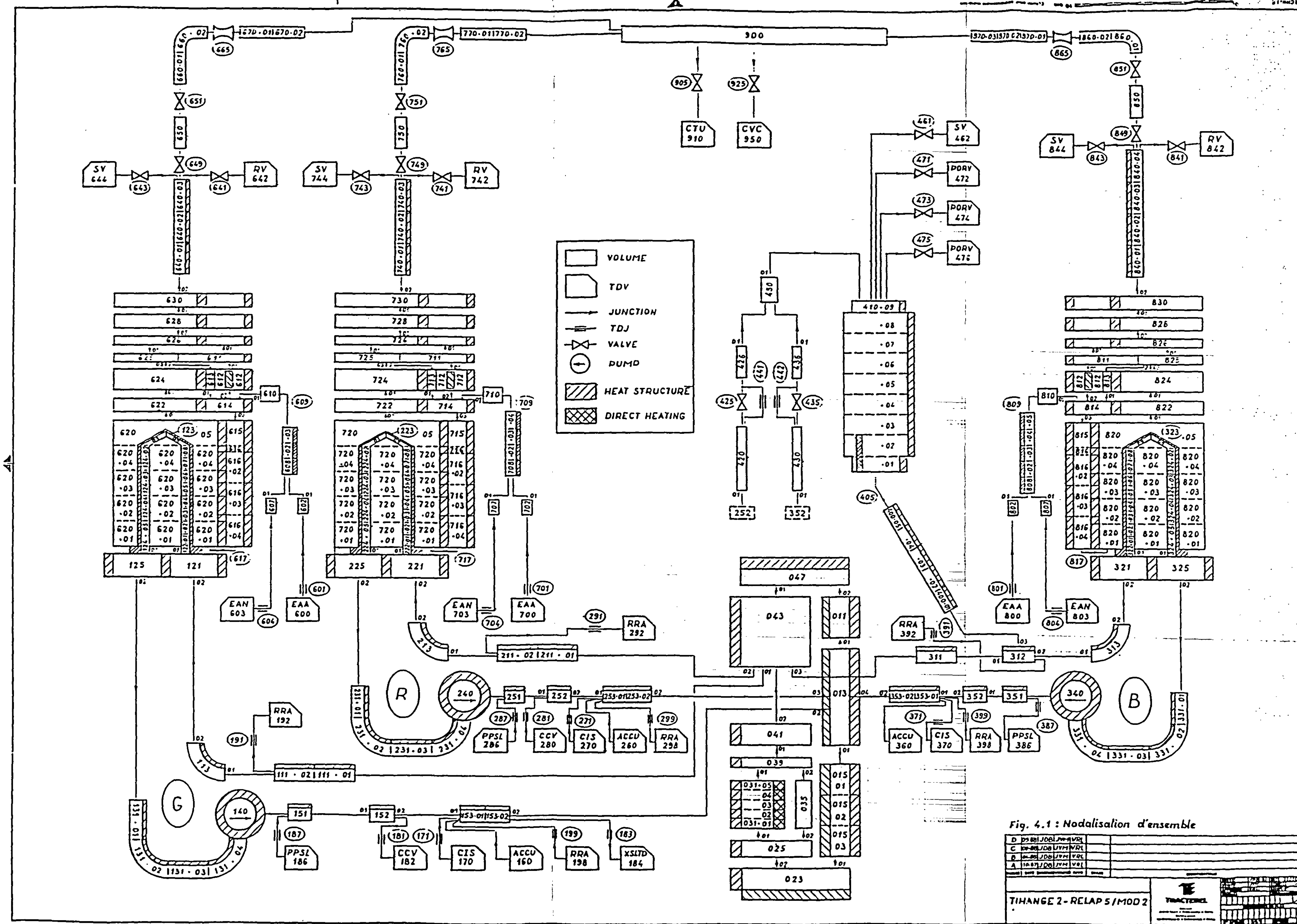
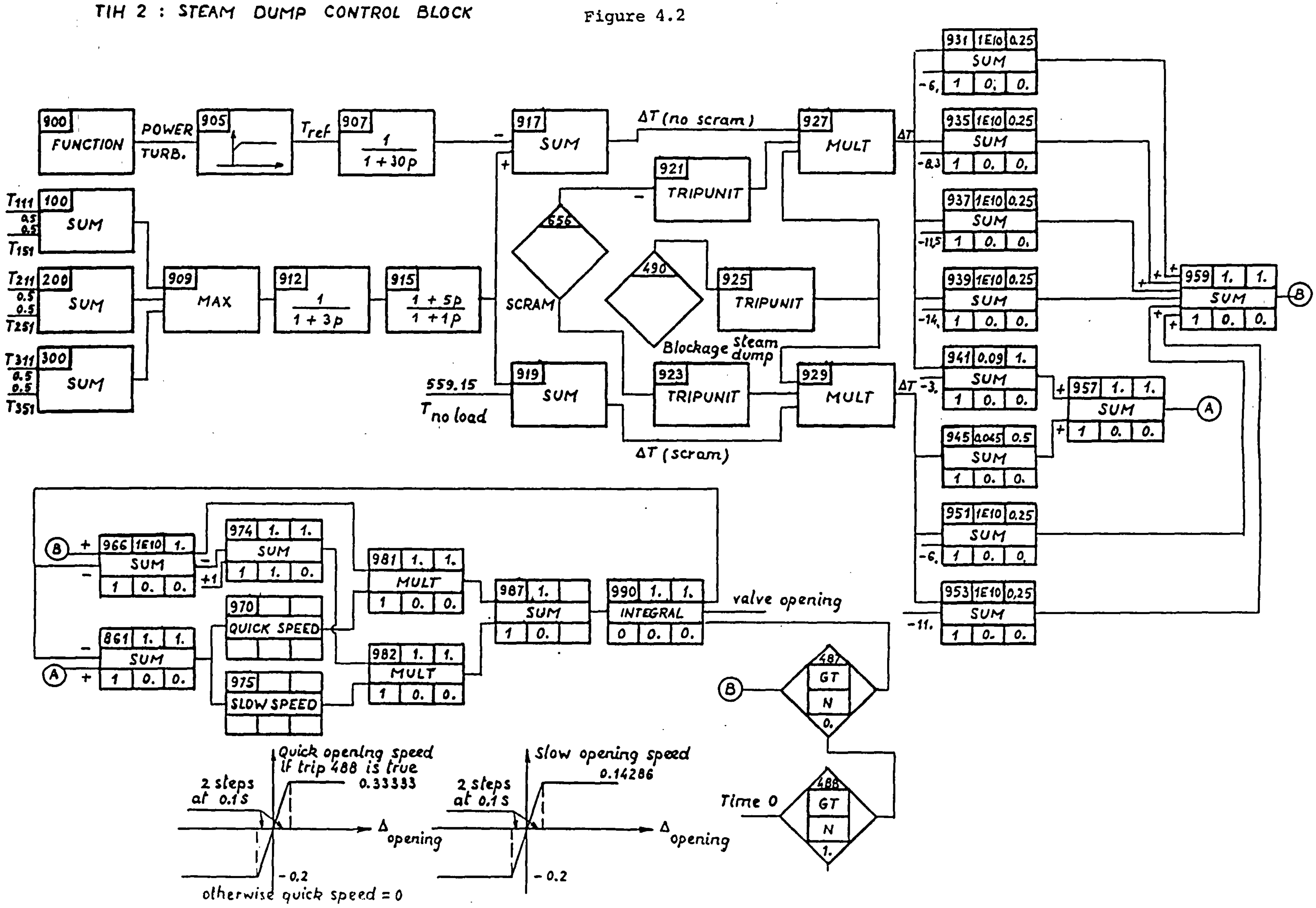


Fig. 4.1 : Nodalisation d'ensemble

TIH 2 : STEAM DUMP CONTROL BLOCK

Figure 4.2



5. BASE CASE CALCULATIONAL RESULTS

The test was simulated over a 3 minutes time span with an initial time which corresponds to the starting time of the DAS curves.Concerning the nature of the initiating event (reactor scram or turbine trip), there appears to be an uncertainty that could not be easily resolved. This is why a manual scram at t = 56 sec rather than a manual turbine trip at t = 55 sec has been used. The impact of this anomaly is examined in chapter 6 (Run 15).

For those parameters where plant data were available,the DAS recordings are presented graphically (in dotted lines) together with the corresponding calculated RELAP-5 data (in solid lines).
The DAS data (identified by DAS XXX, where XXX is the DAS channel identifier) were represented by a limited number of points, such that the appearance on the RELAP-5 plots (transferred via tables) represents the average DAS data.

Figs. 5.1 to 5.16 illustrate for the base case (Run 12A) a good to excellent agreement between RELAP-5 simulation data and plant data, and make us believe that the code is capable to simulate such a plant transient.

Some discrepancies however do appear and merit further discussion :

5.1. Comparison of steady-state conditions prior to the transient

As mentionned in chapter 3, some anomalies are observed in the nominal steady state parameters derived from the plant sensors. For this study, we have assumed best estimate conditions which are physically coherent:

- 100 % power level (Fig 5.1).
- pressurizer pressure of 156.8 bar (Fig. 5.2).
- nominal delta T between hot and cold leg of 35.7 deg.C (loops G & R) and 35.6 deg.C (loop B, Fig. 5.5).
- For the steam generators, the steady state water levels were changed from 44 % (nominal setpoint) to 43 % (SG G, Fig. 5.13) and to 46 % (SGs B & R ,Figs. 5.12 & 5.14) as indicated by the DAS values.
- The resulting calculated pressures are 59.4 bar (SG B, Fig. 5.8) and 59.2 bar (SGs G & R, Figs. 5.9 & 5.10)
- The initial steam flows are around the nominal value of 1814 t/hr (SG B, Fig. 5.15).

5.2. Comparison with data for the early transient (50 sec < t < 142 sec)

In general, the calculated system parameters related to the primary system agree rather well with the recorded plant data (pressure, level, temperatures).

A systematic lead of about 3 seconds in the calculated primary coolant temperatures was noticed in earlier calculations and was caused by the absence of the Resistence Temperature Detector (RTD) bypass loops in the RELAP-5 nodalisation. In the plant, there exists a finite transport time of about 1 second between the RTD bypass connections to the primary loops and the temperature sensors in the bypass. Furthermore, a

RTD filter time constant of 1 second is used to avoid abnormal signals in a high noise environment. In addition, a 1 second instrumentation response time is considered and therefore a lag of 3 seconds is introduced both in the calculation of temperature difference used for the steam dump logic and the temperature of the primary coolant loops. (Fig. 5.5). However this total lag of 3 seconds seems to be excessive for the hot leg temperature and a lag of 1 to 2 seconds should be sufficient to make the two curves coincide in this time period (Fig. 5.4).

The average primary coolant temperature in the calculation stagnates at a value about 0.4 deg.C above the measured value (Fig. 5.6).
The pressurizer level in the calculation stagnates at 2 % above the measured value (Fig. 5.3). The level is controlled by the charging flow and this last parameter is uncertain. (Fig. 5.7).
The calculated primary pressure curve deviates progressively from the measured pressure curve between 80 and 140 sec, to reach 0.7 bar difference, that is 0.5 % (Fig. 5.2).
All these values are in the range of uncertainties of instrumentation.
The average temperature excess (0.4 deg.C) accounts for 0.6 % of the pressurizer level excess. Therefore, the discrepancy between the calculated and measured levels in the pressurizer cannot be a consequence of the average temperature excess only.

On the secondary side, the agreement is also rather satisfactory:

- The calculated steam generator pressure reaches a final plateau very close to the measured pressure, the distance between the two curves being less than 0.2 bar (Figs. 5.8 to 5.10). However, in the first part of the period (50 to 80 sec) during which the pressure is rapidely increasing, the calculated pressure departs significantly from the measured pressure above 63 bar. Two possible reasons have been found which could explain this discrepancy:

a) At the time when the calculated pressure rise is temporarily reduced (around 63 bar), a rather large condensation is observed in the regions where cold feedwater was injected (compare Figs. 5.9 and 5.18). The dependence of pressure evolution on condensation will be further analysed in paragraph 6.7.

b) The quick opening time of the steam dump valve is 3 sec. When this time is increased to 5 sec, the pressure builds up more rapidely and again the two curves come closer to each other during this time period. (paragraph 6.7).

- The calculated steam collector pressure (Fig. 5.11) is about 0.6 bar above the measured pressure (1 %) during the "plateau phase". This is within the uncertainty interval and could reflect a difference in the instrumentation offset.

- Concerning the steam generator levels, (Figs. 5.12 to 5.14) the calculated value appears to follow closely the measured value. During the initial transient period, (50 to 80 sec), the curve for steam generator B shows an increase which reflects the feed water flow increase. In the following phase when the scram has occured, the level falls rapidely and the agreement between the two curves deteriorates due to the presence of a spike, which is probably related to the condensation process described above;its amplitude reaches 15 % of the narrow range for SG B (Fig. 5.12), but does not exceed 5 % of the narrow range for the other two steam generators.(Figs. 5.13 and 5.14). Around 80 sec, the measured level has dropped to 1.7 % whereas the calculated level has dropped to .7 %. Here again, these values are within the range of the instrumentation uncertainties.

The DAS curve 91 on Fig. 5.15 represents the steam flow. There is obviously an important time lag on the measurement signal; this time lag is estimated to 20-30 sec. Since it is known that steam flow rate measurements for low steam flow rates are not reliable, no conclusion can be drawn from the comparison between the simulated and recorded steam flow.

5.3. Comparison for the period with manual steam dump control (t > 142 s)

The boundary condition for this phase constitutes a manual increase in steam dump demand of 33 % of full capacity (dotted line in Fig. 5.16 taken from the DAS recording; a 2 % reduction is applied to this demand and represents the drawing inaccuracy). This leads to an increase of the steam flow rate (Fig. 5.15) and a sharp reduction in the steam generator pressure (Figs. 5.8 to 5.10). Although there is a discrepancy in the timing, the pressure jumps are comparable.
The calculated steam generator levels remain below the calculated values, but the difference is not significant (sensor offset).

On the primary side, the agreement during this phase is quite satisfactory for all the recorded parameters : pressurizer pressure and level (Figs. 5.2 and 5.3), hot and cold leg temperatures, delta T and average temperature (Figs. 5.4 to 5.6).

5.4. RELAP-5 models assessment

It is evident that such tests on full scale plants do not yield sufficient insight in the applicability of the code constitutive equations to full scale plants.

However, such simulation excercises may manifest some shortcomings in the modalisation scheme. Indeed this test shows that the timing of the steam dump opening may be affected, in the early stage, by the absence of an explicit modeling of the RTD bypass loops which introduce a delay time of about 1 sec (Fig. 5.4 : delay not simulated or Fig. 5.6: delay simulated).

An adequate simulation of the steam generators is of utmost importance, as they constitute an important buffer between the balance of plant and the primary system. Although imprecisions in the boundary conditions for the steam generators, such as steam dump timing and capacity, are partly responsible for the observed discrepancies,a proper nodalisation is essential to simulate the complicated phenomena occuring in the secondary side following a scram.
For example, a short lived spike in the steam collector pressure occuring after the scram (Fig. 5.11) is observed in both the plant and the simulation. It is caused by the interaction of fast turbine trip and fast initiation of the steam dump to the condensor. The present nodalisation of the steam generators and steam lines allows this phenomenon to be properly simulated.

The anomalies observed in the steam generators during the highly dynamic phase (around t = 56 sec) such as:

- excessive low pressure rise (Fig. 5.9)
- abnormal level swell (Fig. 5.13)

may be caused by excessive condensation as shown in Figs. 5.17 to 5.20. These figures illustrate for steam generator G, the vapour condensation (transformed to -100*VAPGEN by means of the control system) and the liquid and vapour temperatures for the

- separator water fall back volume 612 (Fig. 5.17)
- auxiliary feed water injection volume 614 (Fig. 5.18)
- top of downcomer volume 615 (Fig. 5.19)
- middle volume (616.01) of the downcomer (Fig. 5.20).

The condensation observed in these figures is also manifested by the abnormal rise in the narrow range water level indication of Fig. 5.13. Figure 5.21 illustrates the evolution of the void fraction in some two-phase regions in steam generator G, and clearly shows the reduction in vapour void when condensation sets in. This condensation drives the vapour and liquid temperatures to saturation temperatures and leads to a stagnation in the pressure evolution.
Since the measured data for the pressure history also manifests an intermediate pressure plateau, but at a higher pressure level (around 66 bars), this may suggest that thermal equilibrium is reached by nature of vapour condensation but at higher pressures.

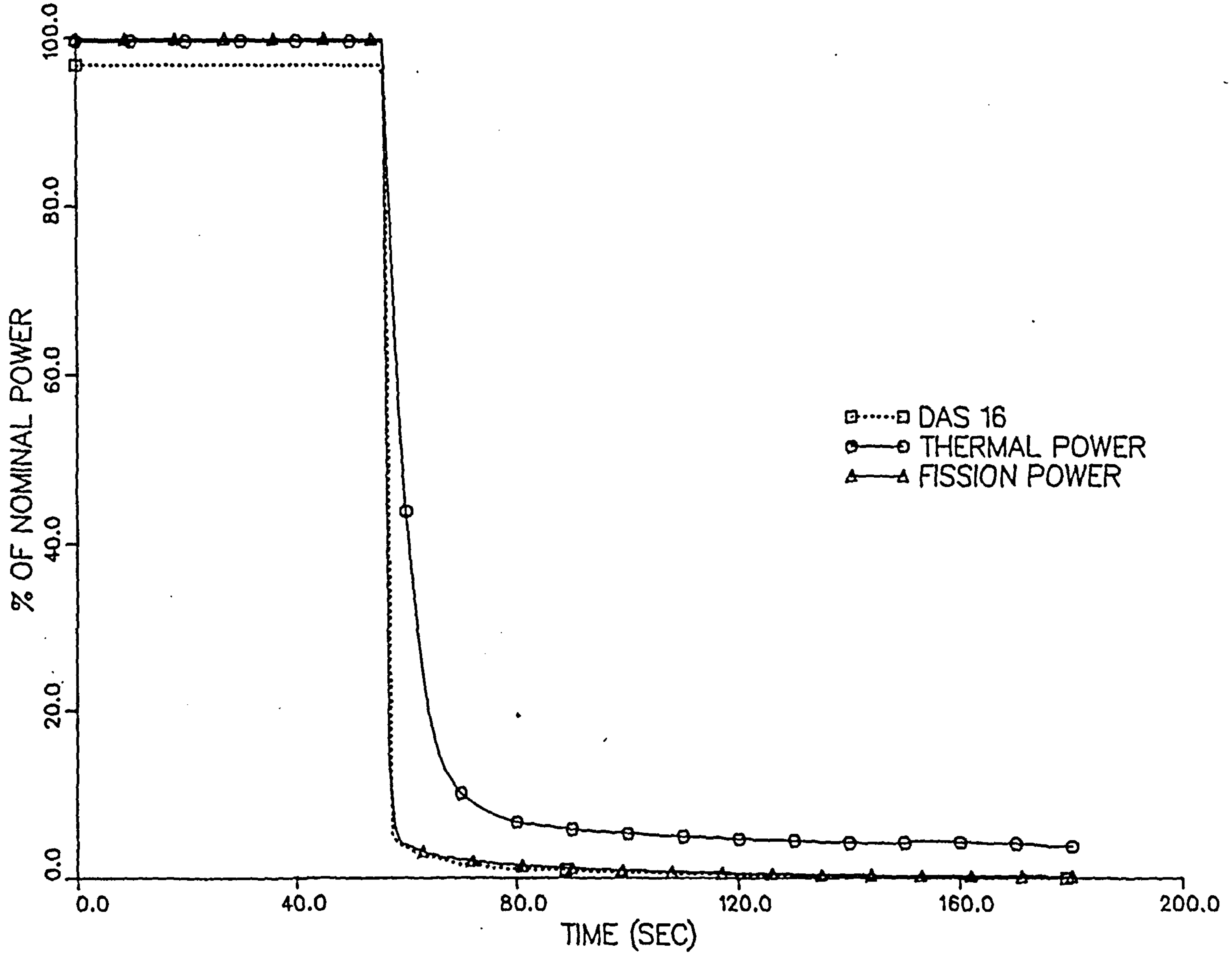
This test also manifests an important feature of dynamic level swell which is observed on the plant narrow range level recorders but not in the RELAP-5 simulation (Figs. 5.12 to 5.14) at the onset of the last phase (t = 142 sec). About 15 sec after manually increasing the steam dump capacity (at t = 157 sec) a slight increase in the water level is observed in the recording, while in the simulation, a similar level increase is not observed.The amplitude of the level swell is however very small (from 2 to 4 %).

The residual water level reading (0.4 %) in Figs. 5.12 to 5.14 is the weight of the steam column located between the narrow range water level taps. If the water inventory in the calculation is not exact, a lower steam generator inventory in the calculation could well lead to a water level below the lower gauge level tap, such that a small level swell is not sufficient to enter the measurement range.

TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

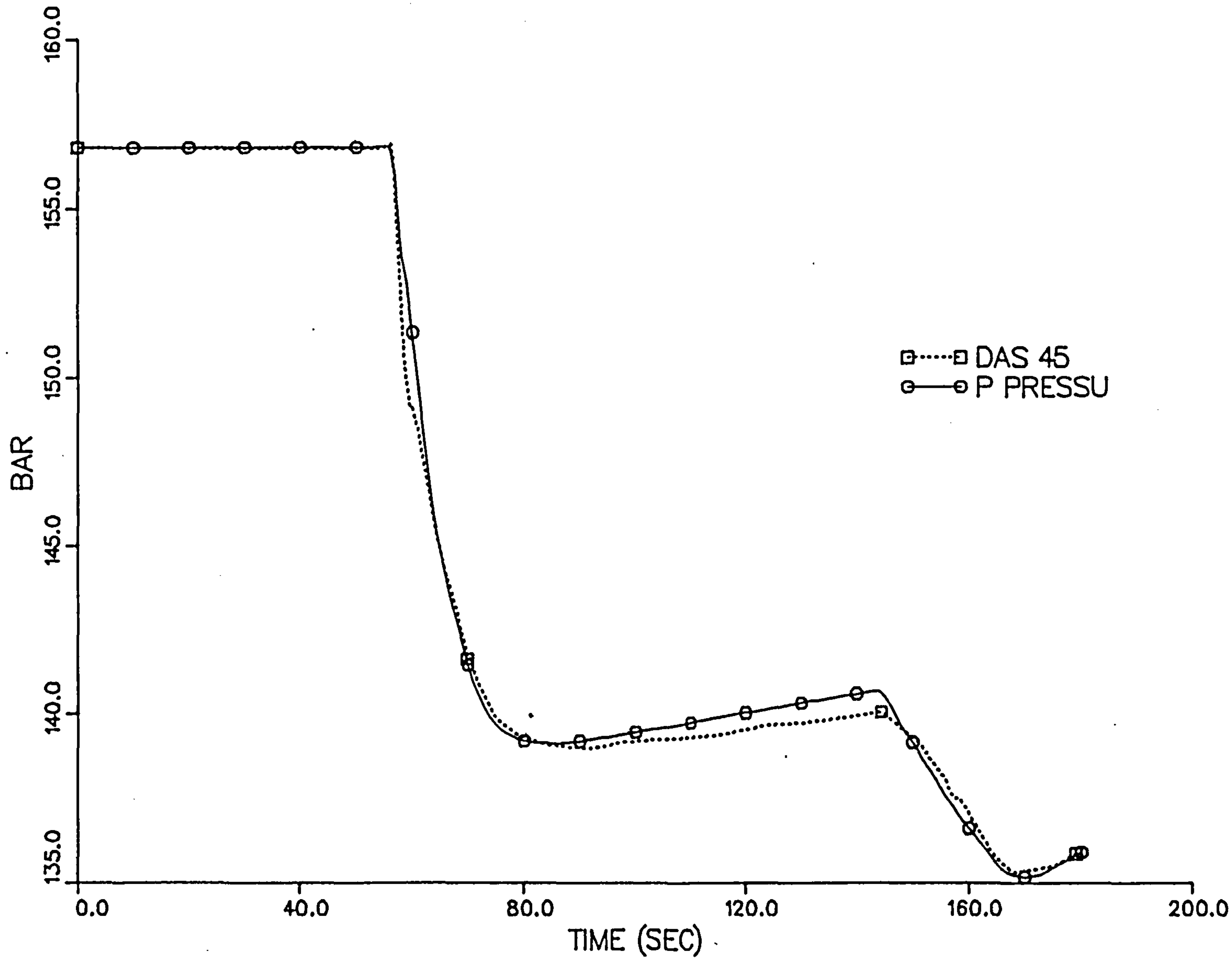
FIG 5.1: FISS./THERMAL POWER



TIHANGE–2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.2: PRESSURIZER PRESSURE

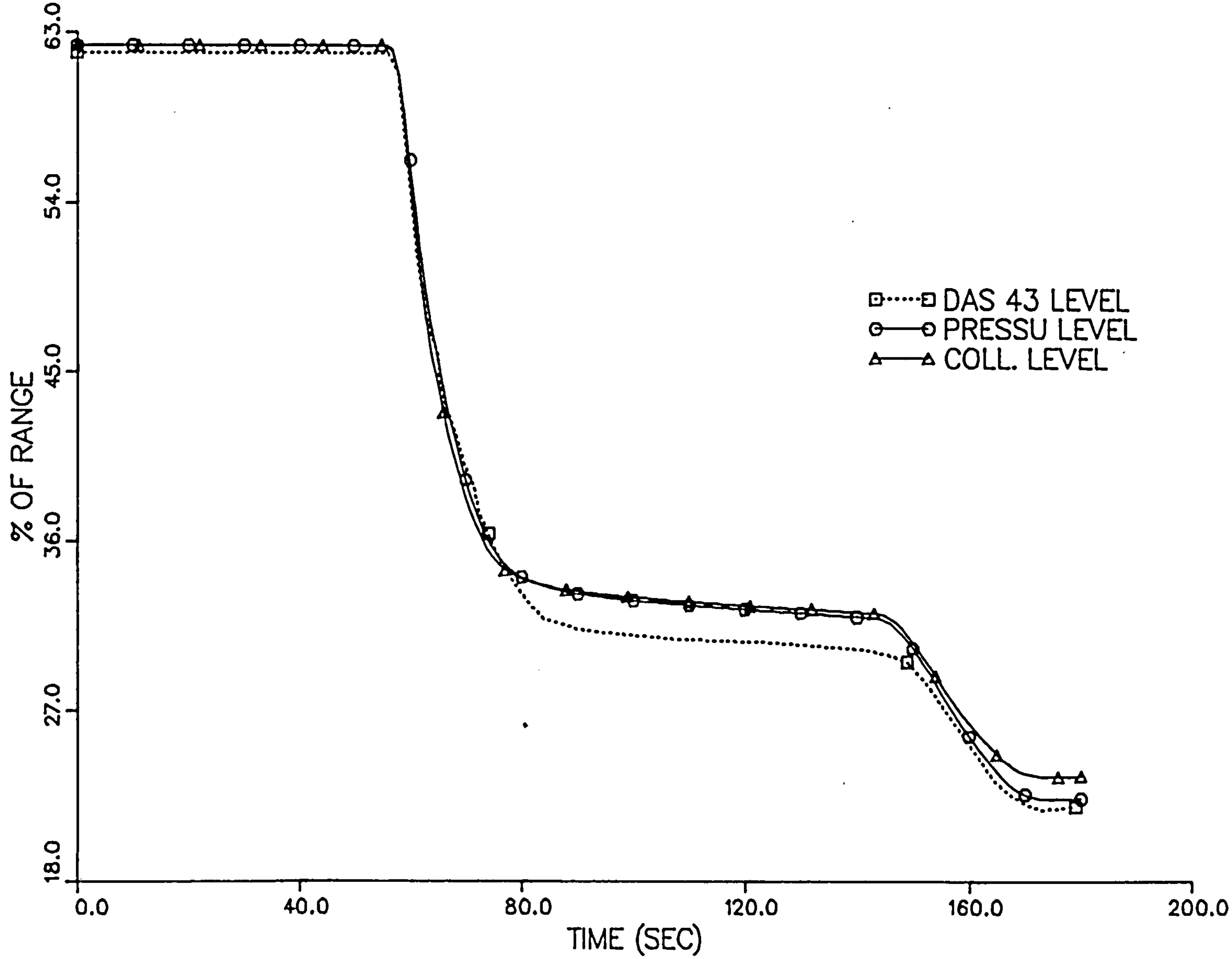


PLOT 0 10.30.32 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO DISSPLA 10.0

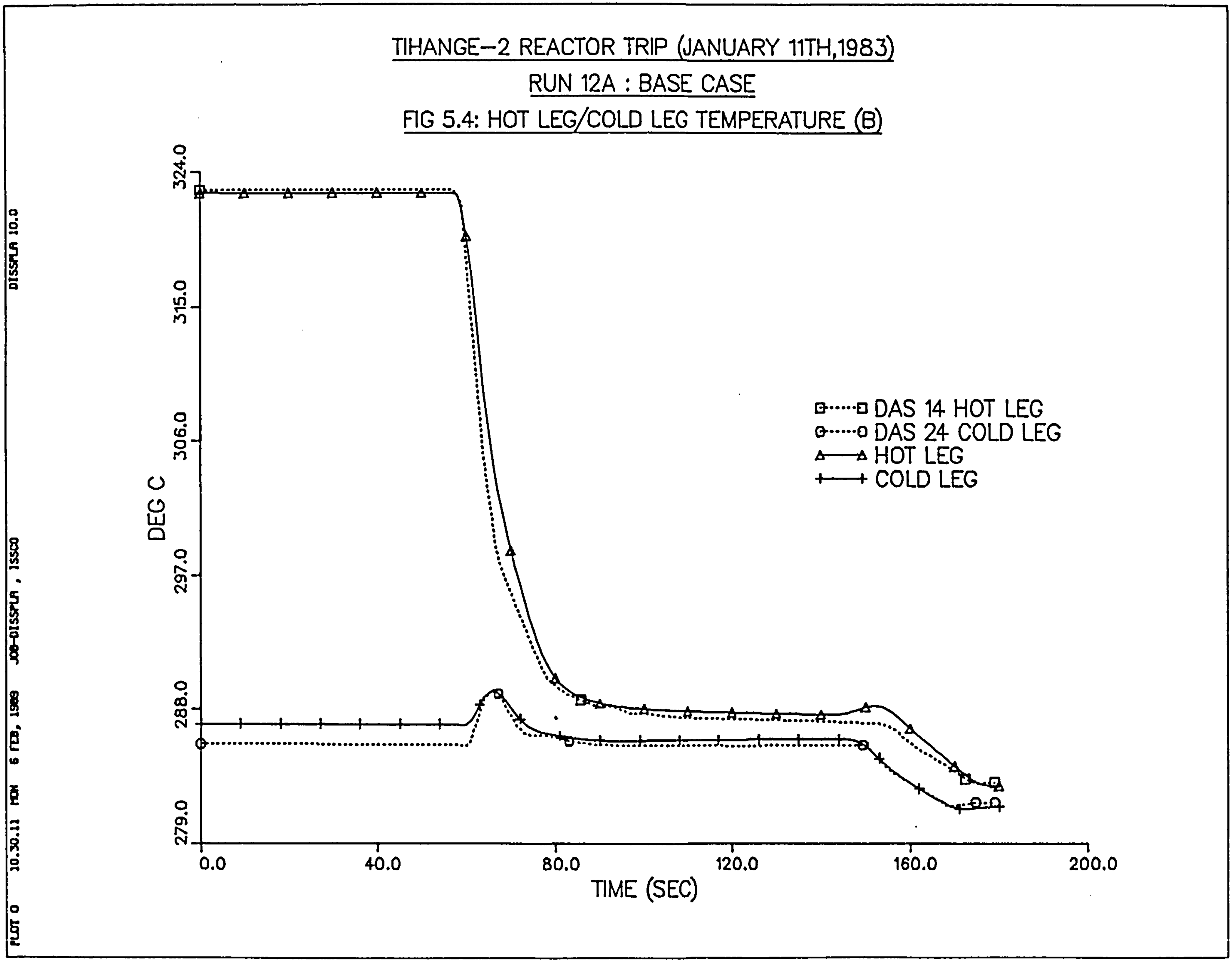
TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.3: PRESSURIZER LEVEL/COLLAPSED LEVEL



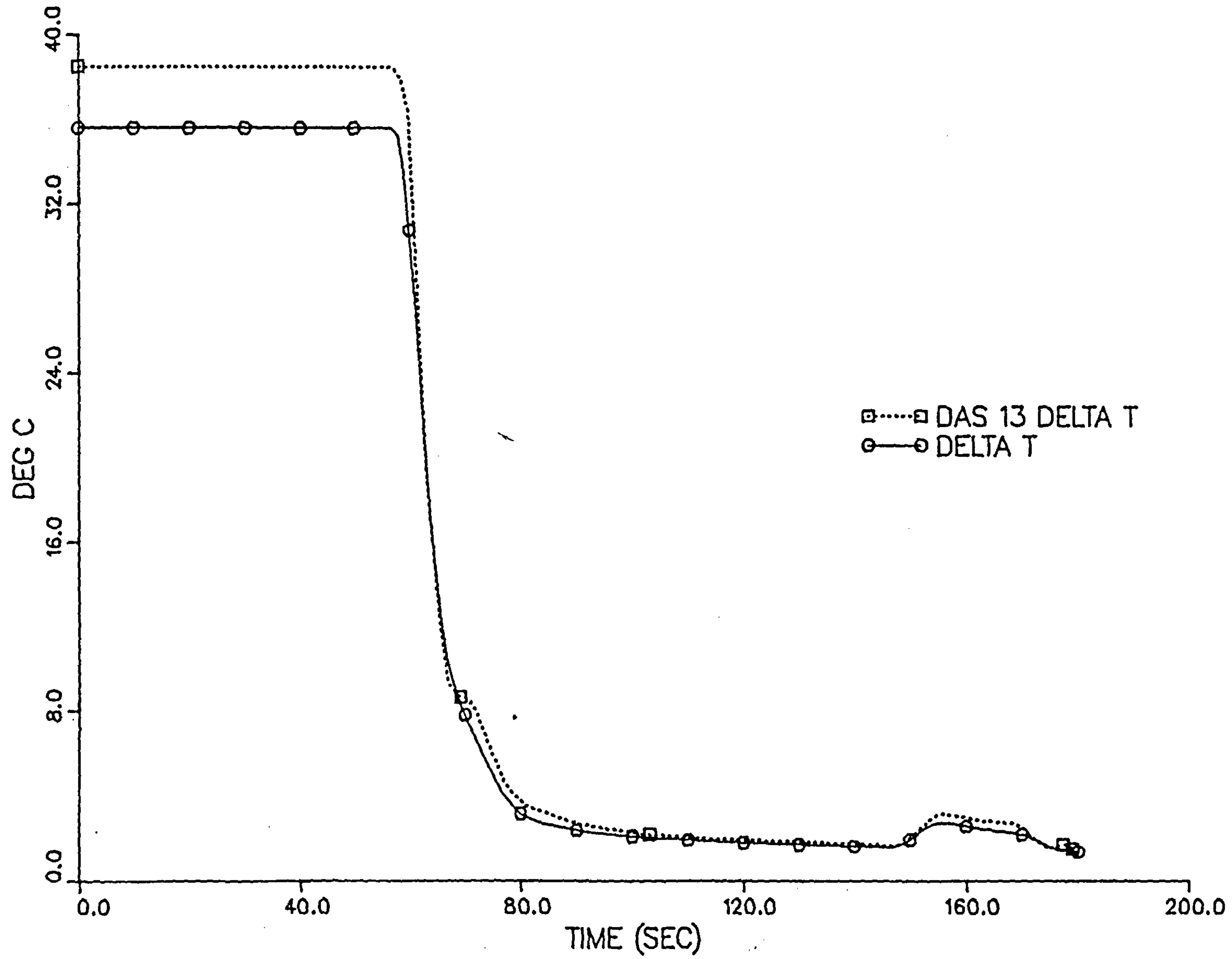
PLOT 0 10.30.43 MON 6 FEB, 1989 JOB-DISSPLA , ISSCO DISSPLA 10.0


TIHANGE–2 REACTOR TRIP (JANUARY 11TH,1983)
RUN 12A : BASE CASE
FIG 5.4: HOT LEG/COLD LEG TEMPERATURE (B)
DEG C
TIME (SEC)
DAS 14 HOT LEG
DAS 24 COLD LEG
HOT LEG
COLD LEG
324.0
315.0
306.0
297.0
288.0
279.0
0.0
40.0
80.0
120.0
160.0
200.0
DISSPLA 10.0
PLOT 0 10.30.11 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO

TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

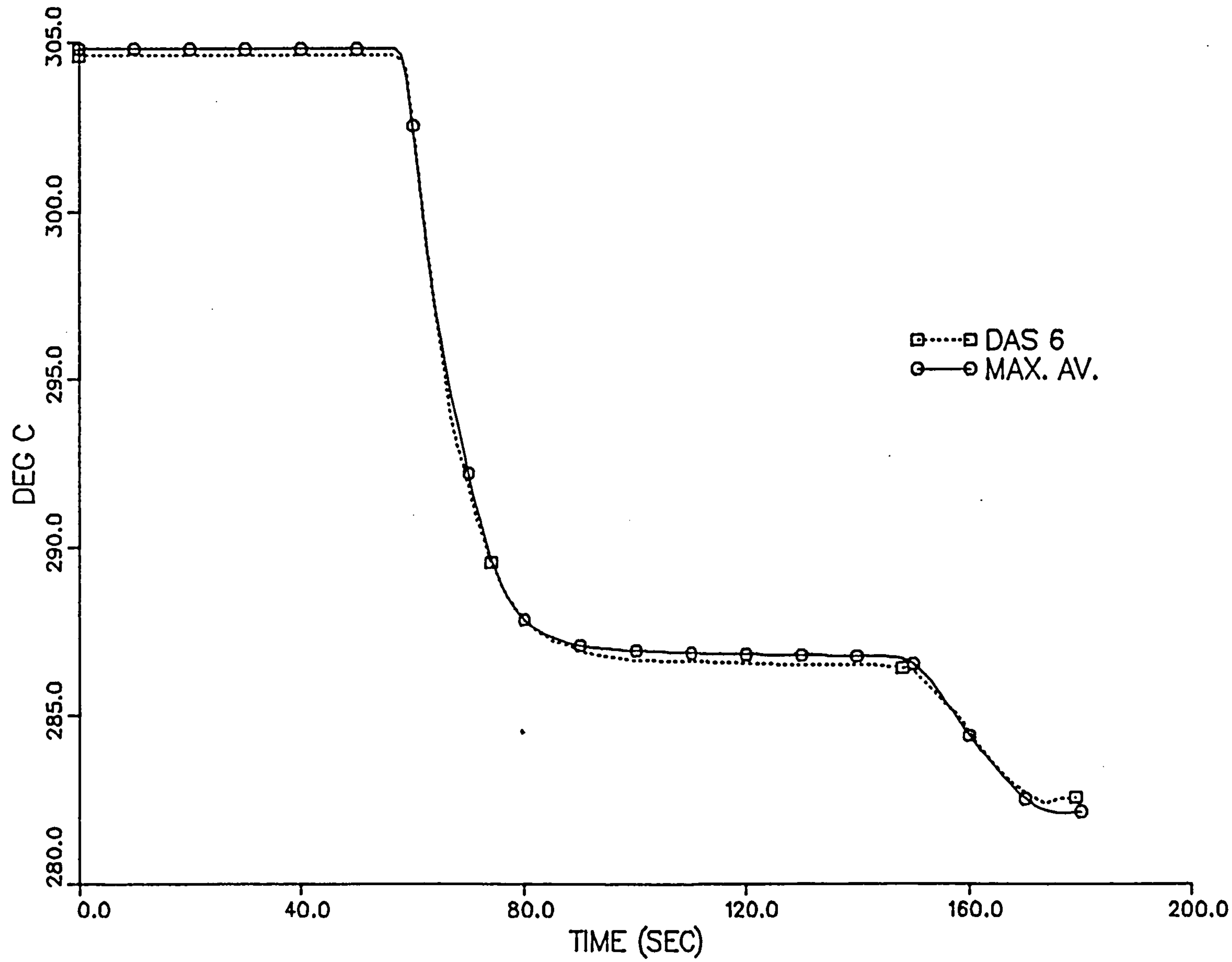
FIG 5.5: DELTA T (B)

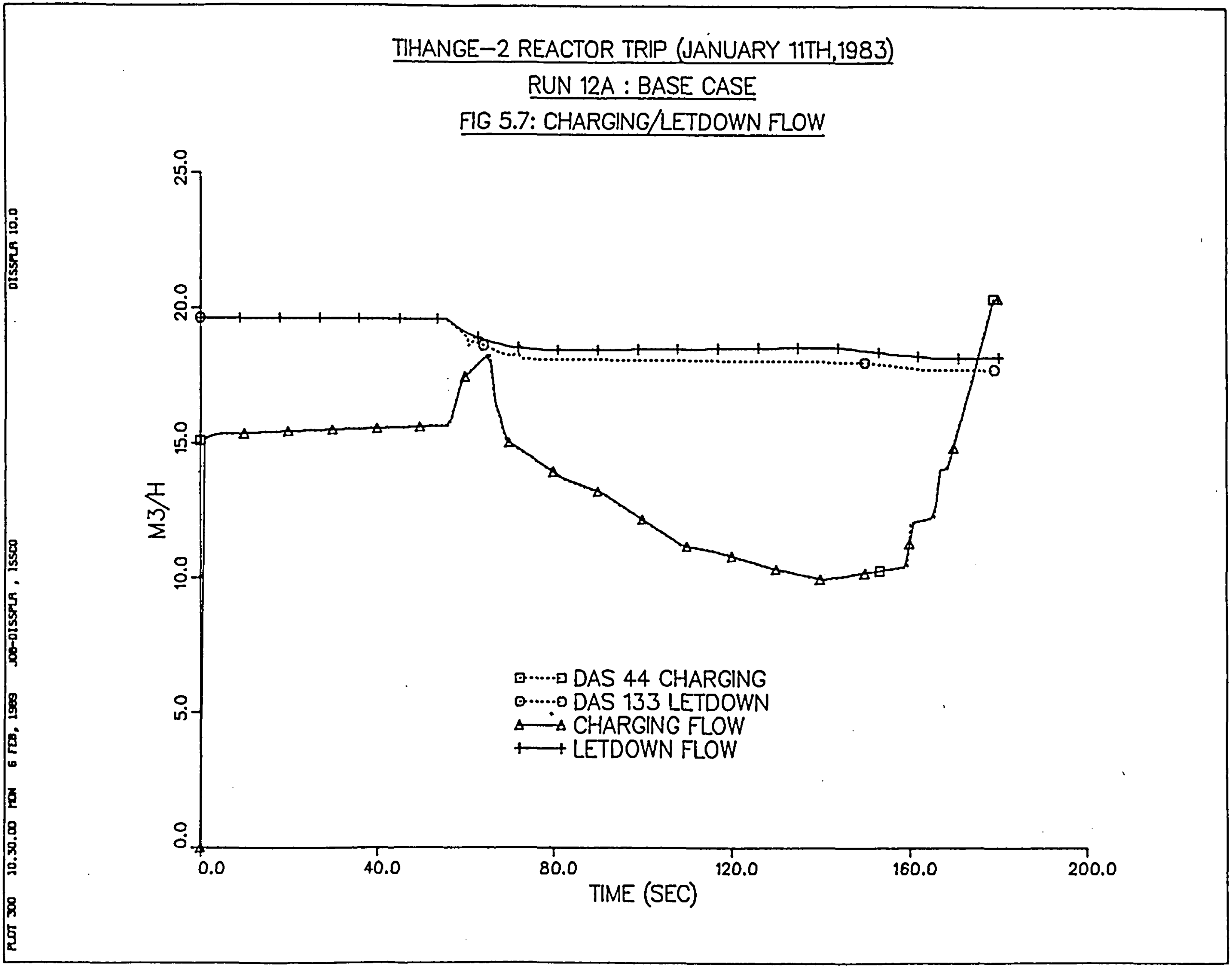


TIHANGE–2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.6: MAX.AVERAGE TEMPERATURE



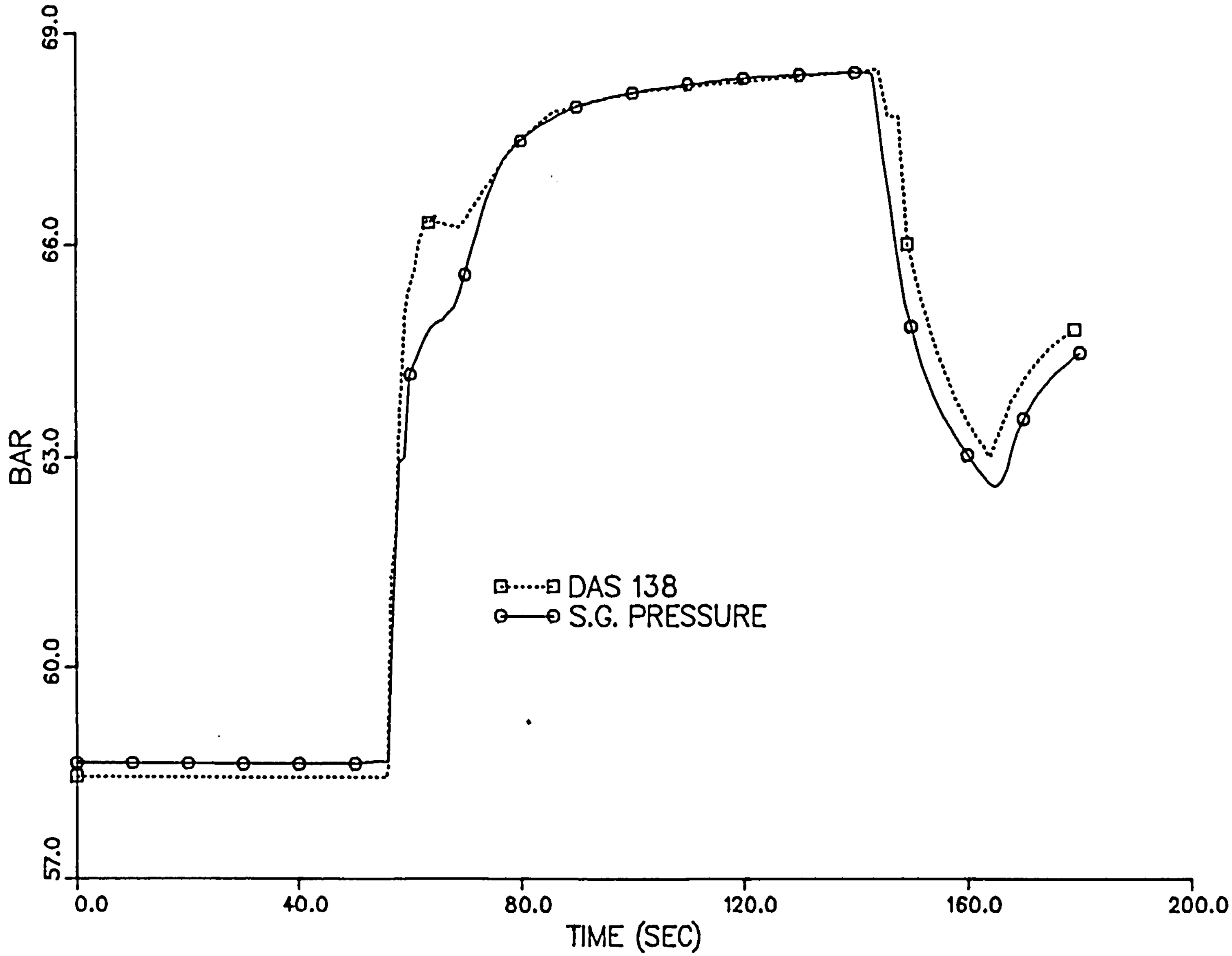


TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)
RUN 12A : BASE CASE
FIG 5.7: CHARGING/LETDOWN FLOW
M3/H
0.0
5.0
10.0
15.0
20.0
25.0
0.0
40.0
80.0
120.0
160.0
200.0
TIME (SEC)
DAS 44 CHARGING
DAS 133 LETDOWN
CHARGING FLOW
LETDOWN FLOW
DISSPLA 10.0
PLOT 300 10.30.00 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO

TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

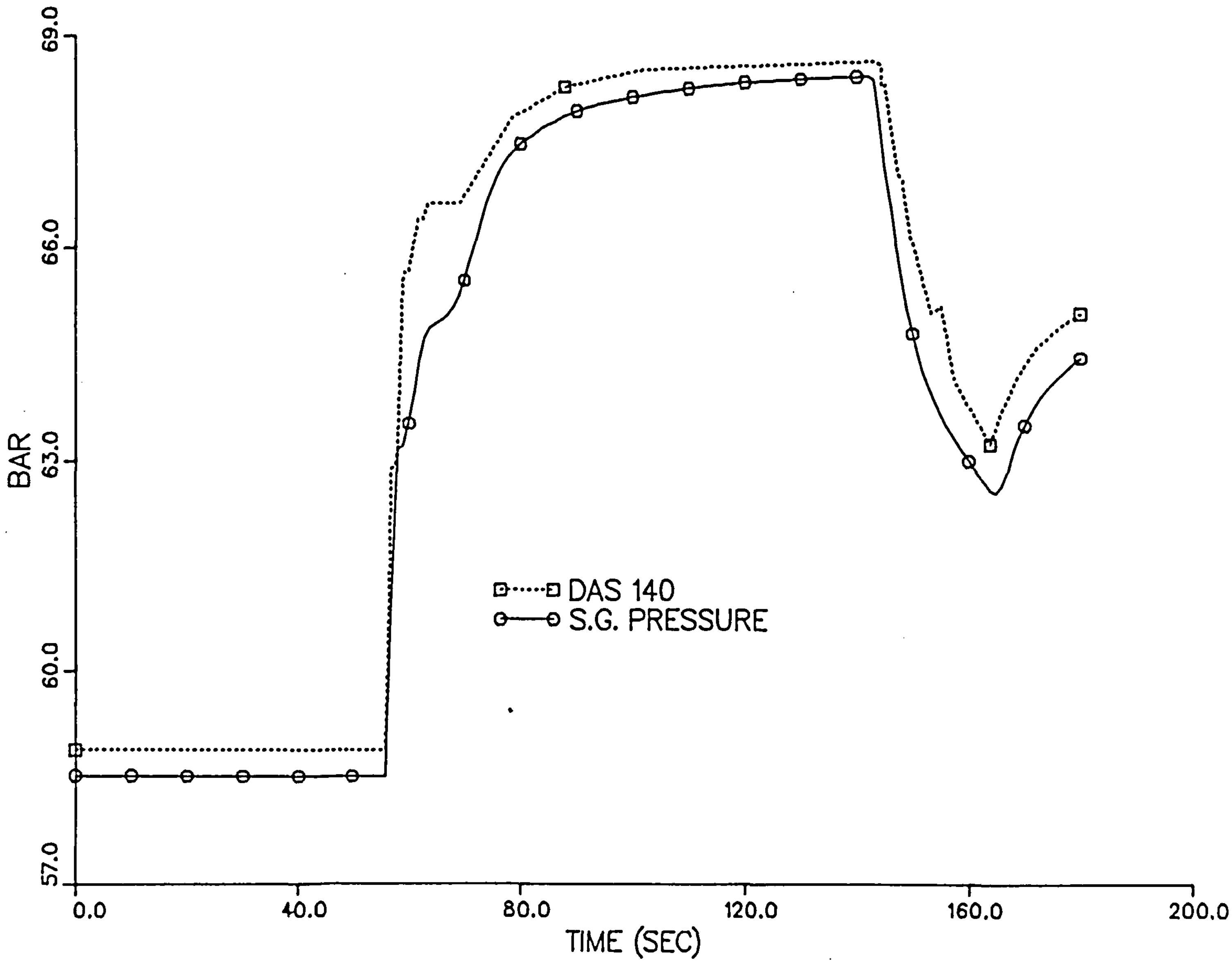
FIG 5.8: STEAM GENERATOR (B) PRESSURE



TIHANGE–2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.9: STEAM GENERATOR (G) PRESSURE

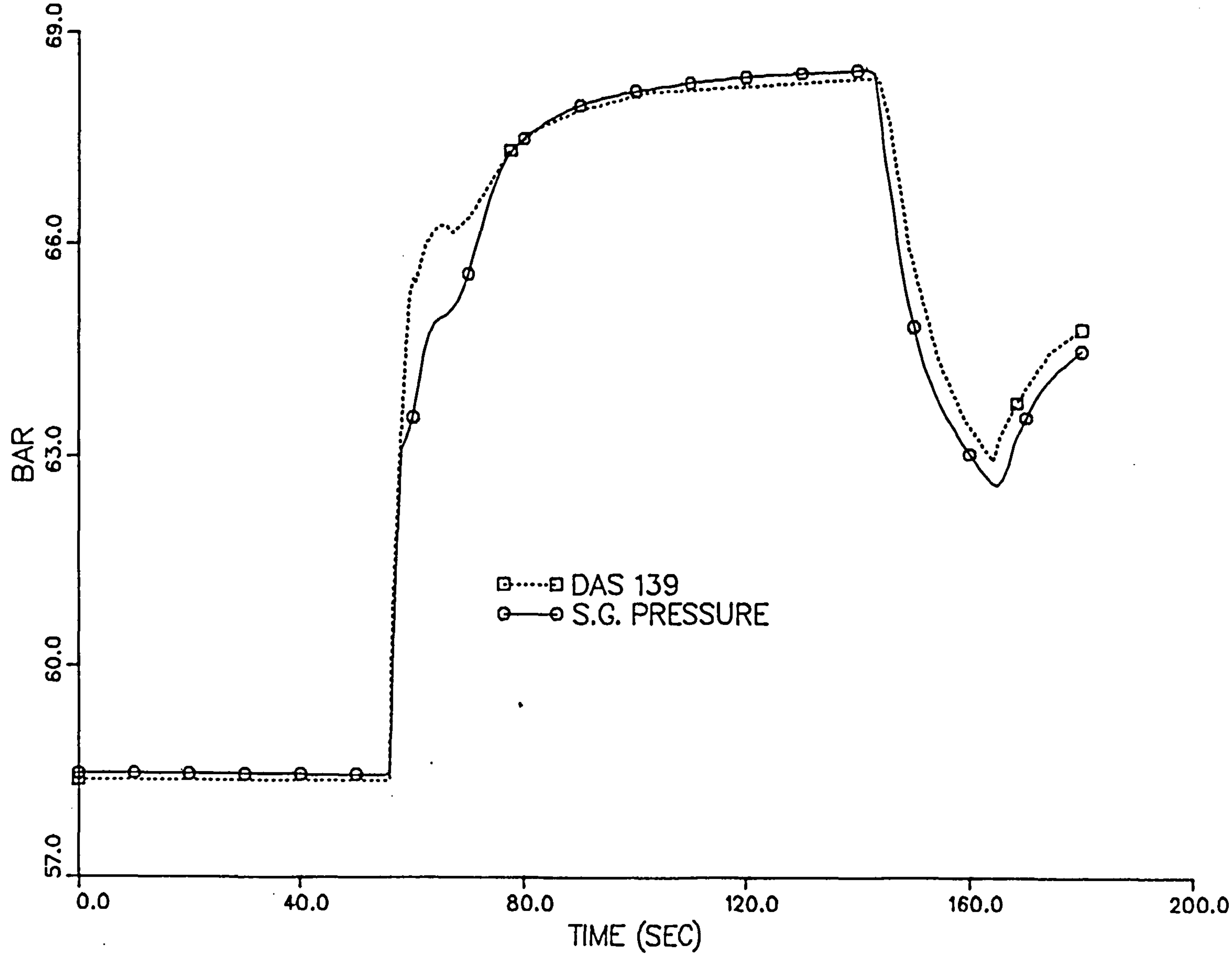


PLOT 0 10.31.29 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO DISSPLA 10.0

TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.10: STEAM GENERATOR (R) PRESSURE

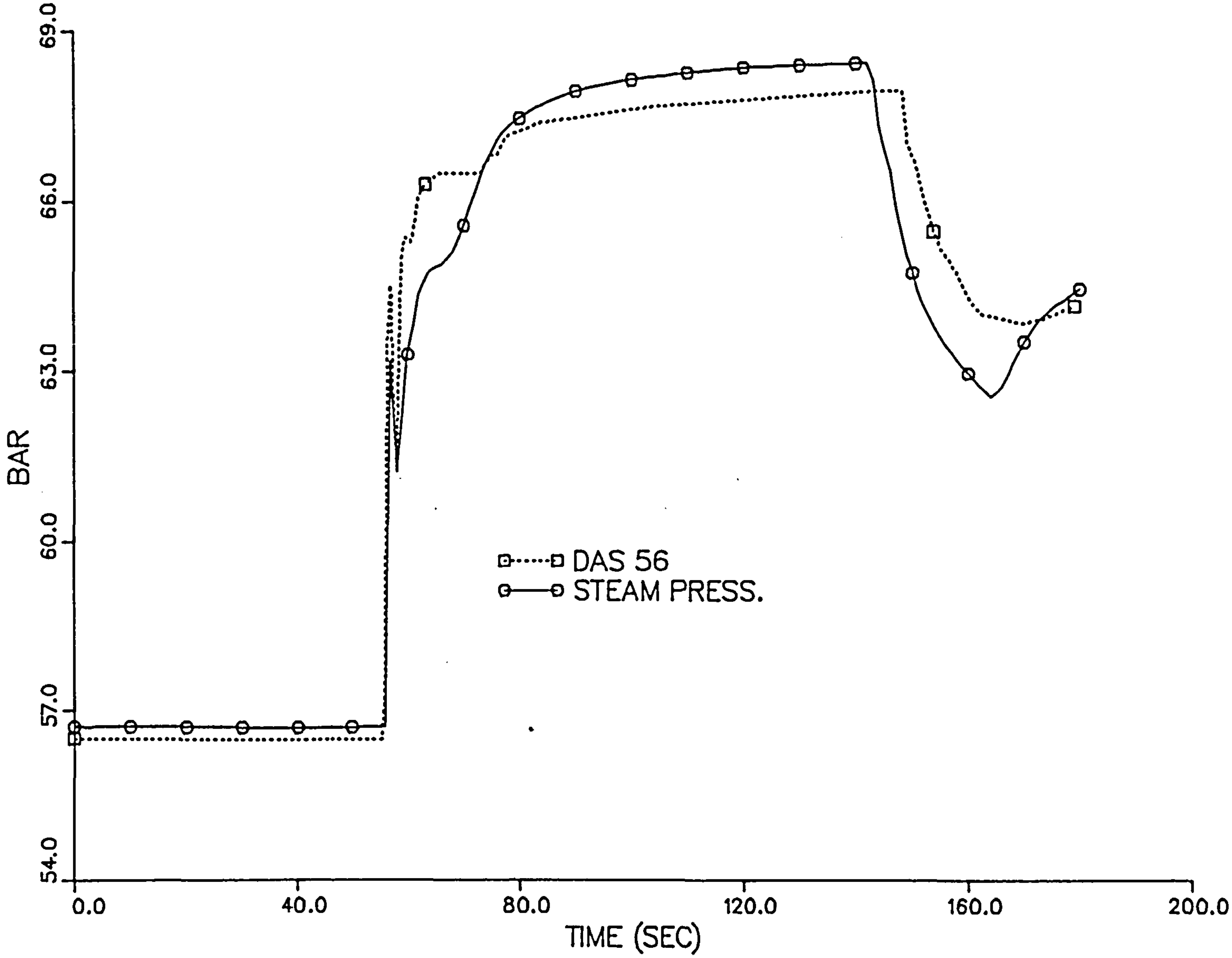


PLOT 0 10.31.41 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO DISSPLA 10.0

TIHANGE–2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.11: STEAM COLLECTOR PRESSURE

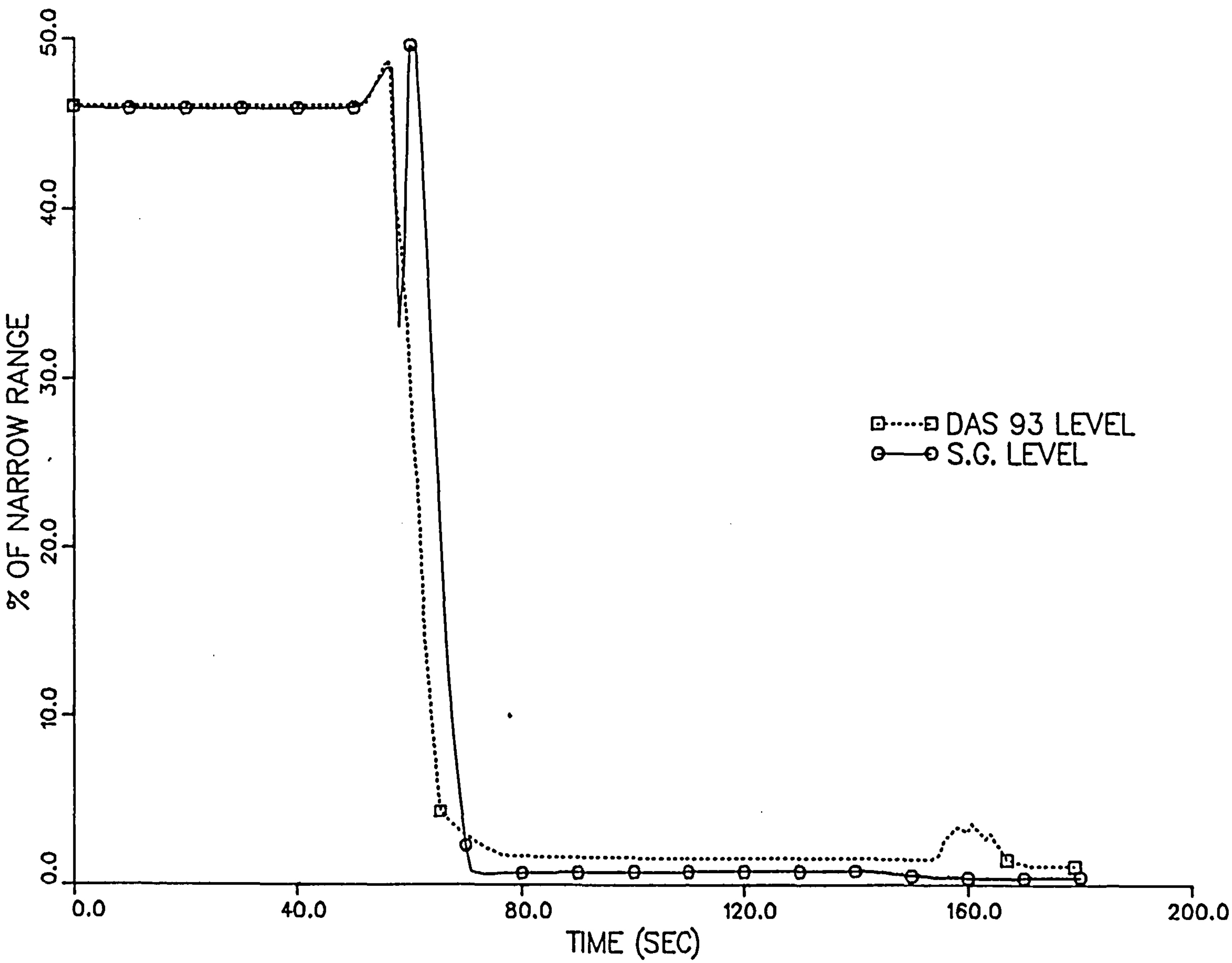


PLOT 0 10.32.51 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO DISSPLA 10.0

TIHANGE—2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.12: STEAM GENERATOR (B) LEVEL

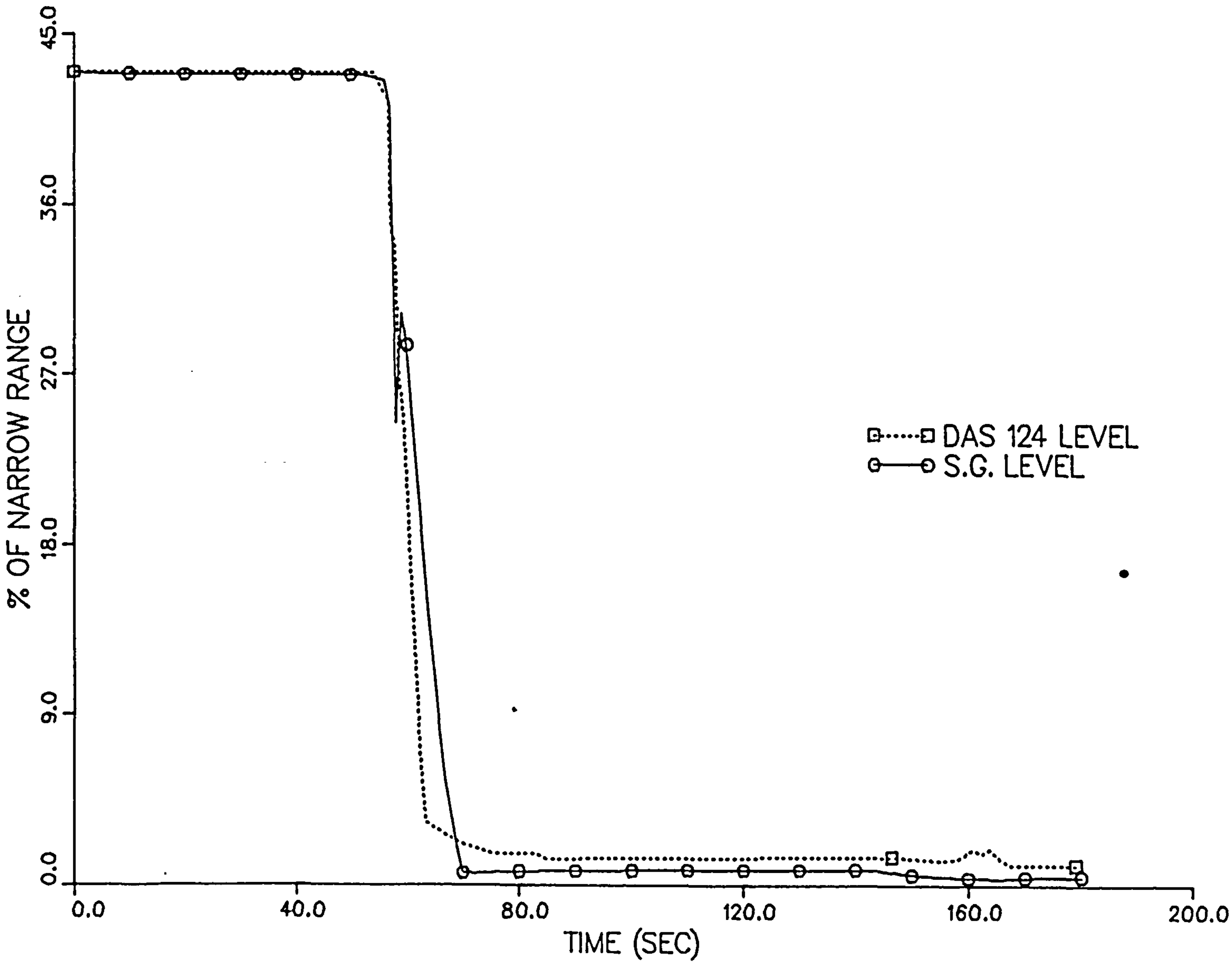


PLOT 0 10.32.04 MON 6 FEB, 1989 JOB-DISSPLA , ISSCO DISSPLA 10.0

TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.13: STEAM GENERATOR (G) LEVEL

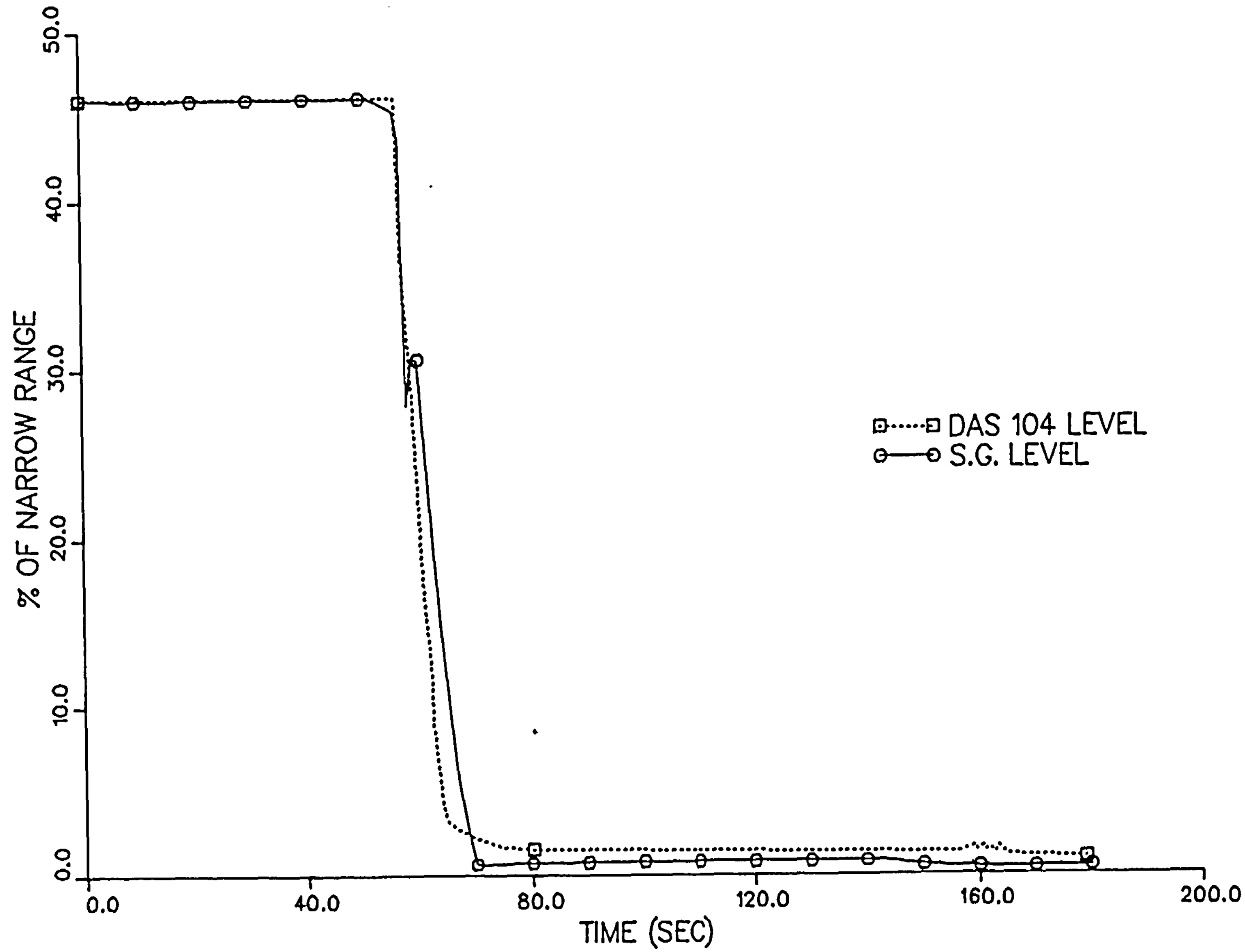


PLOT 0 10.30.55 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO DISSPLA 10.0

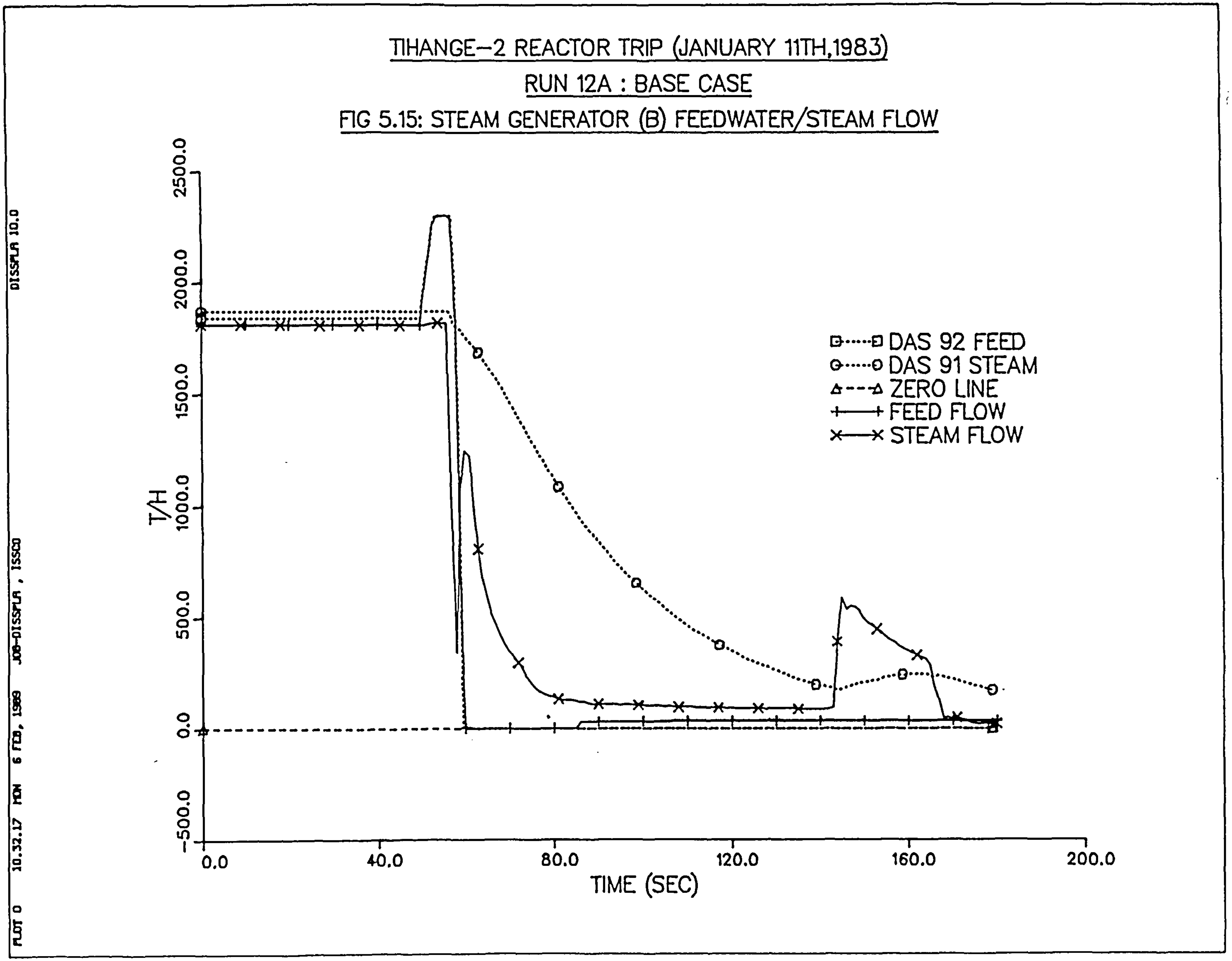
TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.14: STEAM GENERATOR (R) LEVEL



PLOT 0 10.31.05 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO DISSPLA 10.0

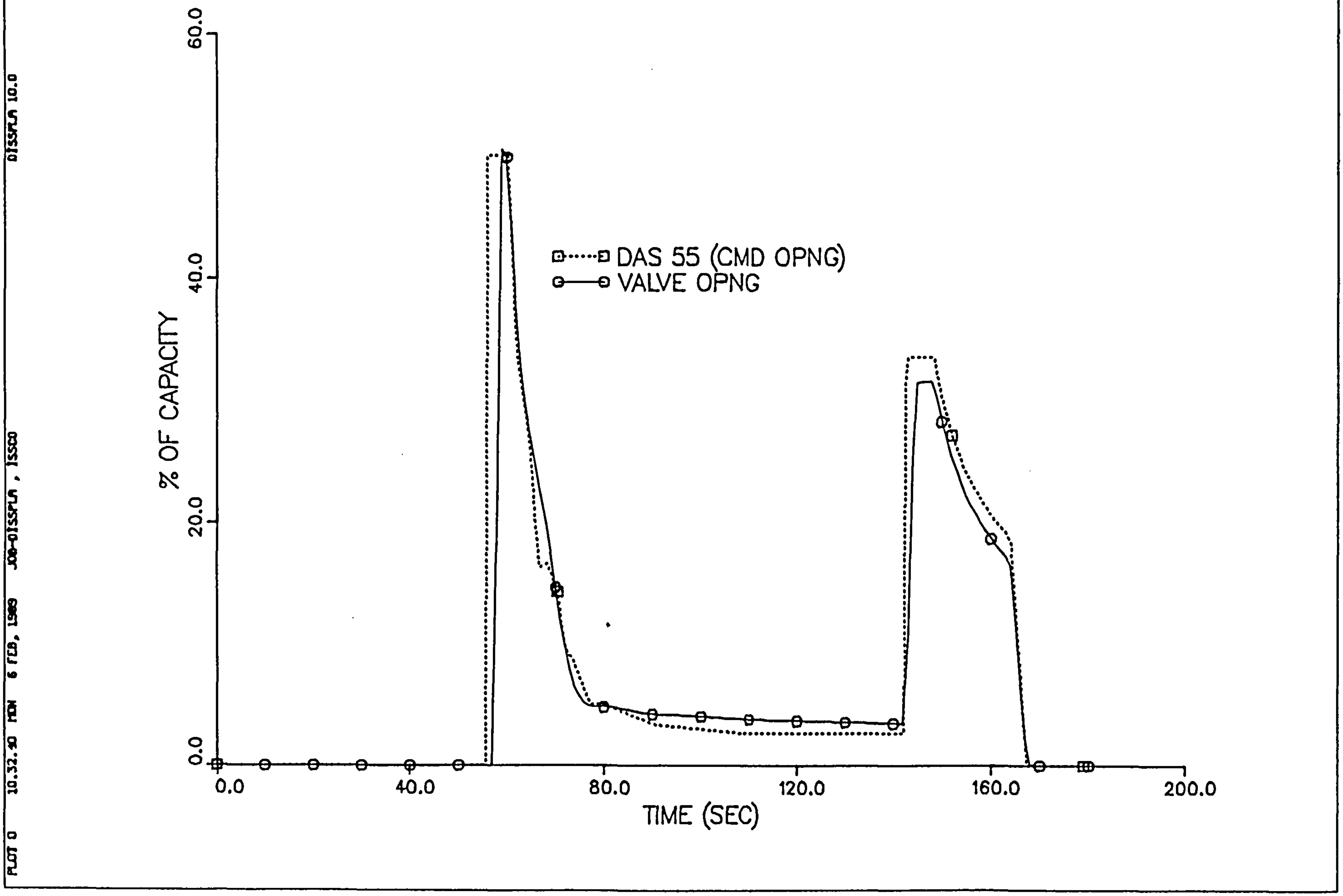


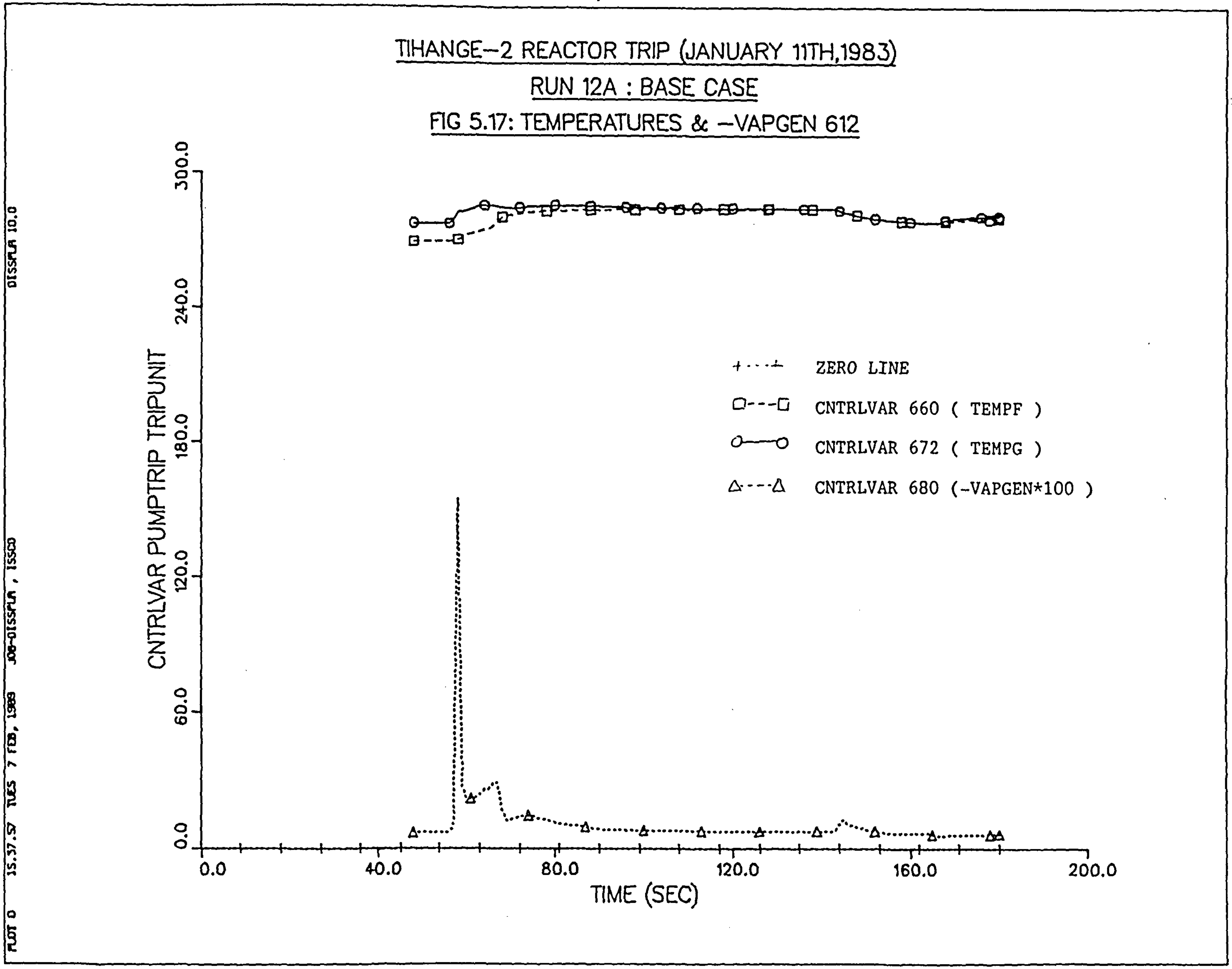
TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)
RUN 12A : BASE CASE
FIG 5.15: STEAM GENERATOR (B) FEEDWATER/STEAM FLOW
DAS 92 FEED
DAS 91 STEAM
ZERO LINE
FEED FLOW
STEAM FLOW
T/H
TIME (SEC)
2500.0
2000.0
1500.0
1000.0
500.0
0.0
-500.0
0.0
40.0
80.0
120.0
160.0
200.0
DISSPLA 10.0
JOB-DISSPLA , ISSCO
PLOT 0 10.32.17 MON 6 FEB, 1989

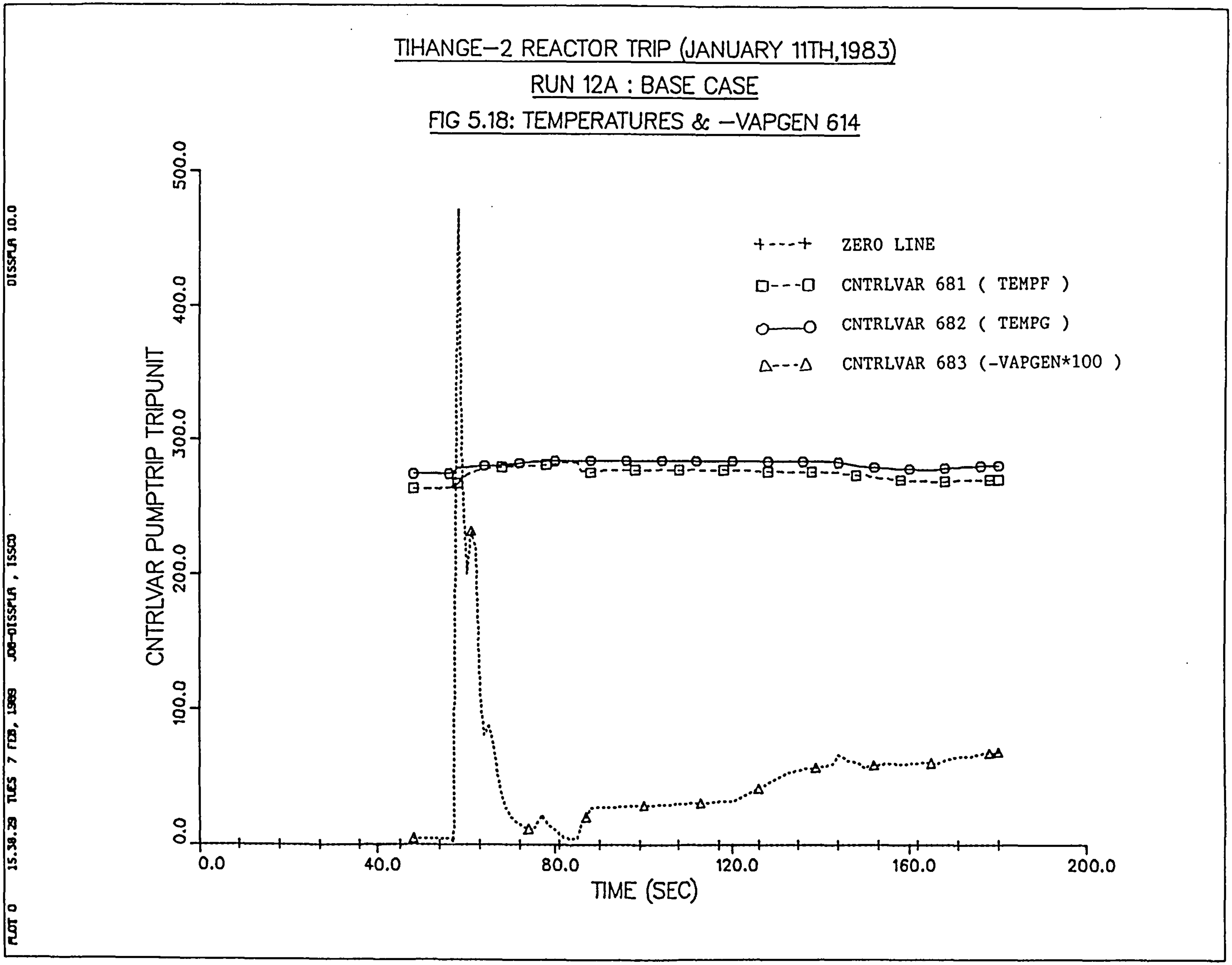
TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.16: STEAM DUMP CAPACITY




TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)
RUN 12A : BASE CASE
FIG 5.17: TEMPERATURES & -VAPGEN 612
CNTRLVAR PUMPTRIP TRIPUNIT
TIME (SEC)
ZERO LINE
CNTRLVAR 660 (TEMPF)
CNTRLVAR 672 (TEMPG)
CNTRLVAR 680 (-VAPGEN*100)
0.0
60.0
120.0
180.0
240.0
300.0
0.0
40.0
80.0
120.0
160.0
200.0
PLOT 0 15.37.57 TUES 7 FEB, 1989 JOB=DISSPLA , ISSCD DISSPLA 10.0

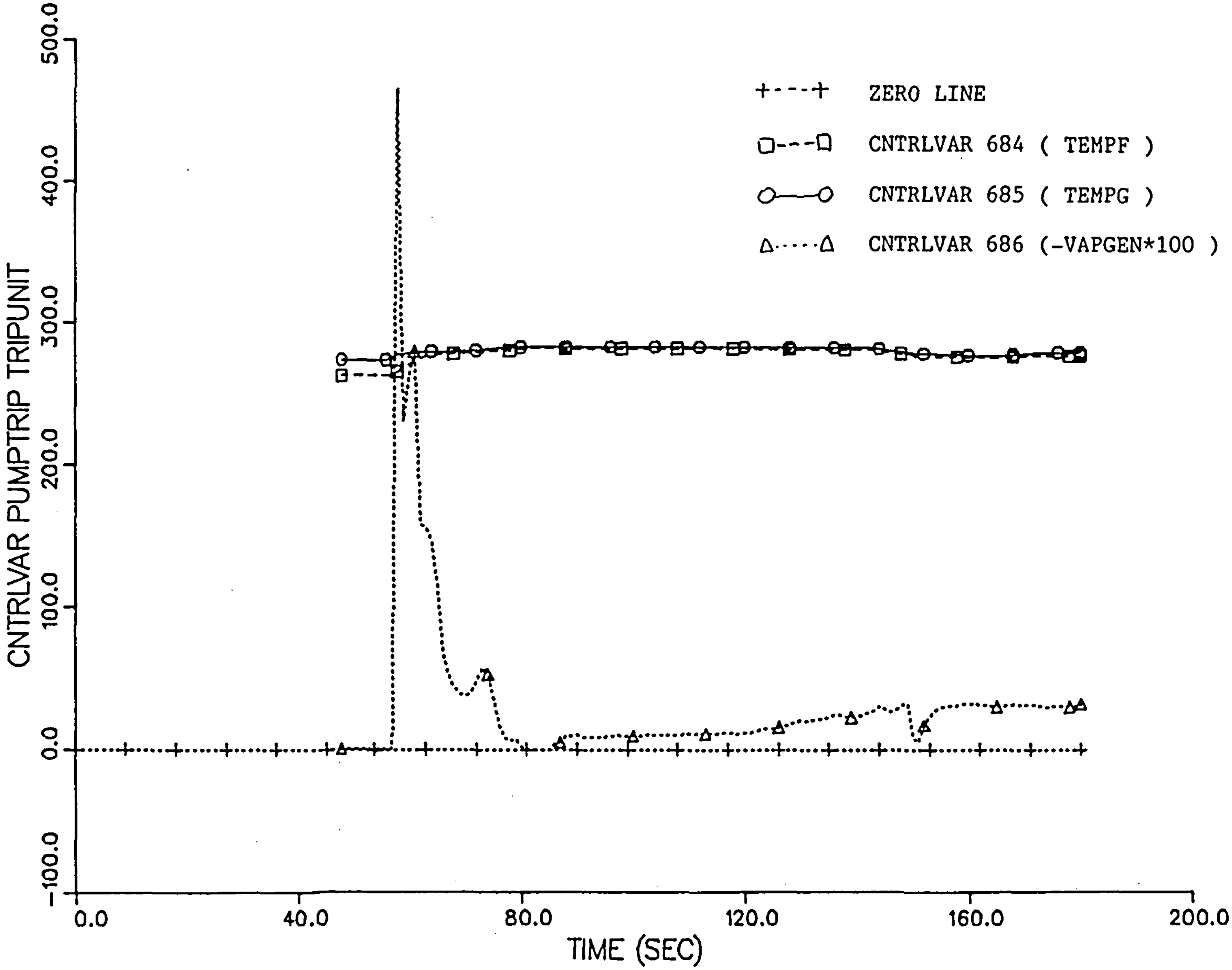


TIHANGE–2 REACTOR TRIP (JANUARY 11TH,1983)
RUN 12A : BASE CASE
FIG 5.18: TEMPERATURES & –VAPGEN 614
CNTRLVAR PUMPTRIP TRIPUNIT
TIME (SEC)
ZERO LINE
CNTRLVAR 681 (TEMPF)
CNTRLVAR 682 (TEMPG)
CNTRLVAR 683 (-VAPGEN*100)
0.0
100.0
200.0
300.0
400.0
500.0
40.0
80.0
120.0
160.0
200.0
PLOT 0
15.38.29 TUES 7 FEB, 1989
JOB=DISSPLA , ISSCO
DISSPLA 10.0

TIHANGE—2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

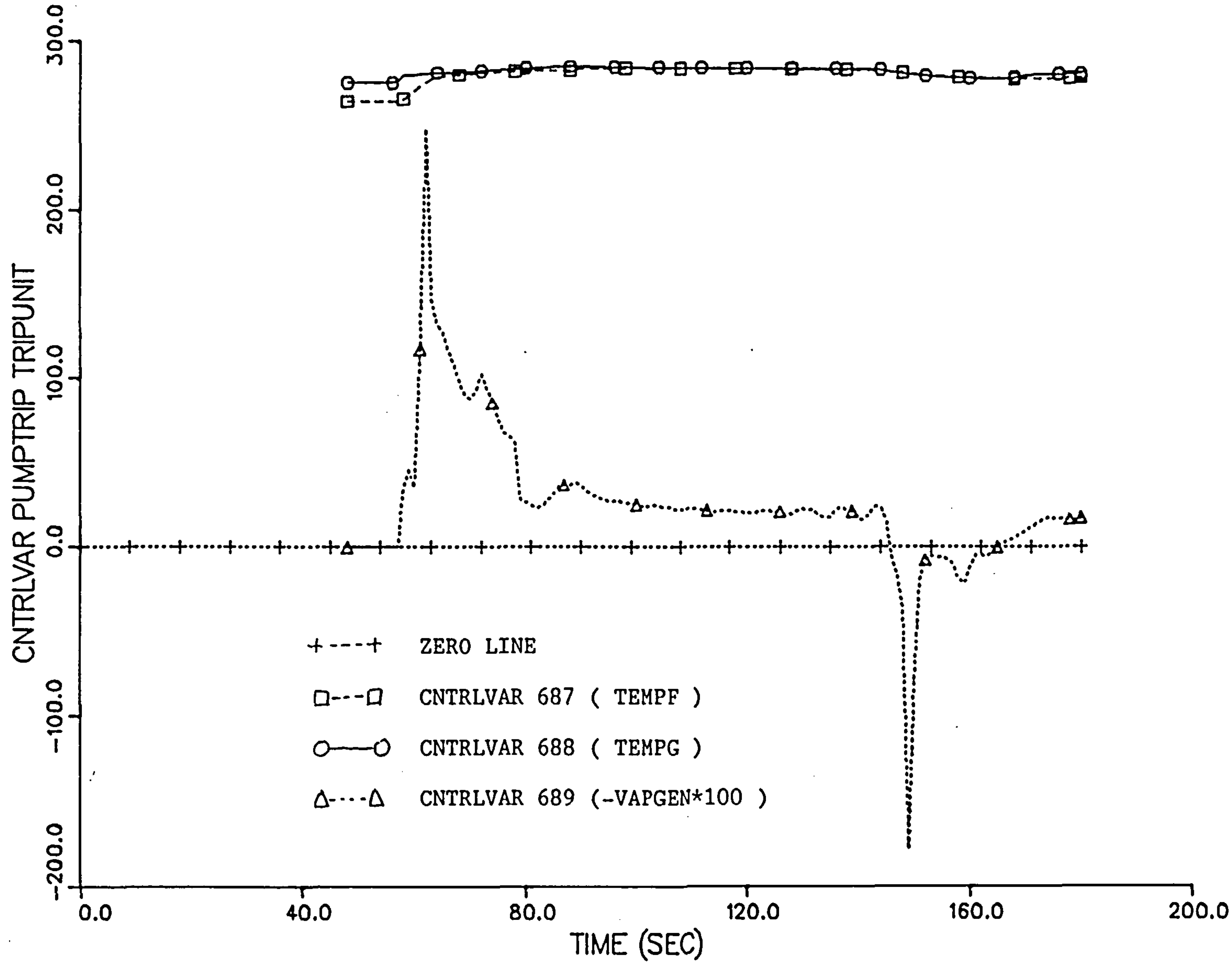
FIG 5.19: TEMPERATURES & —VAPGEN 615



TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

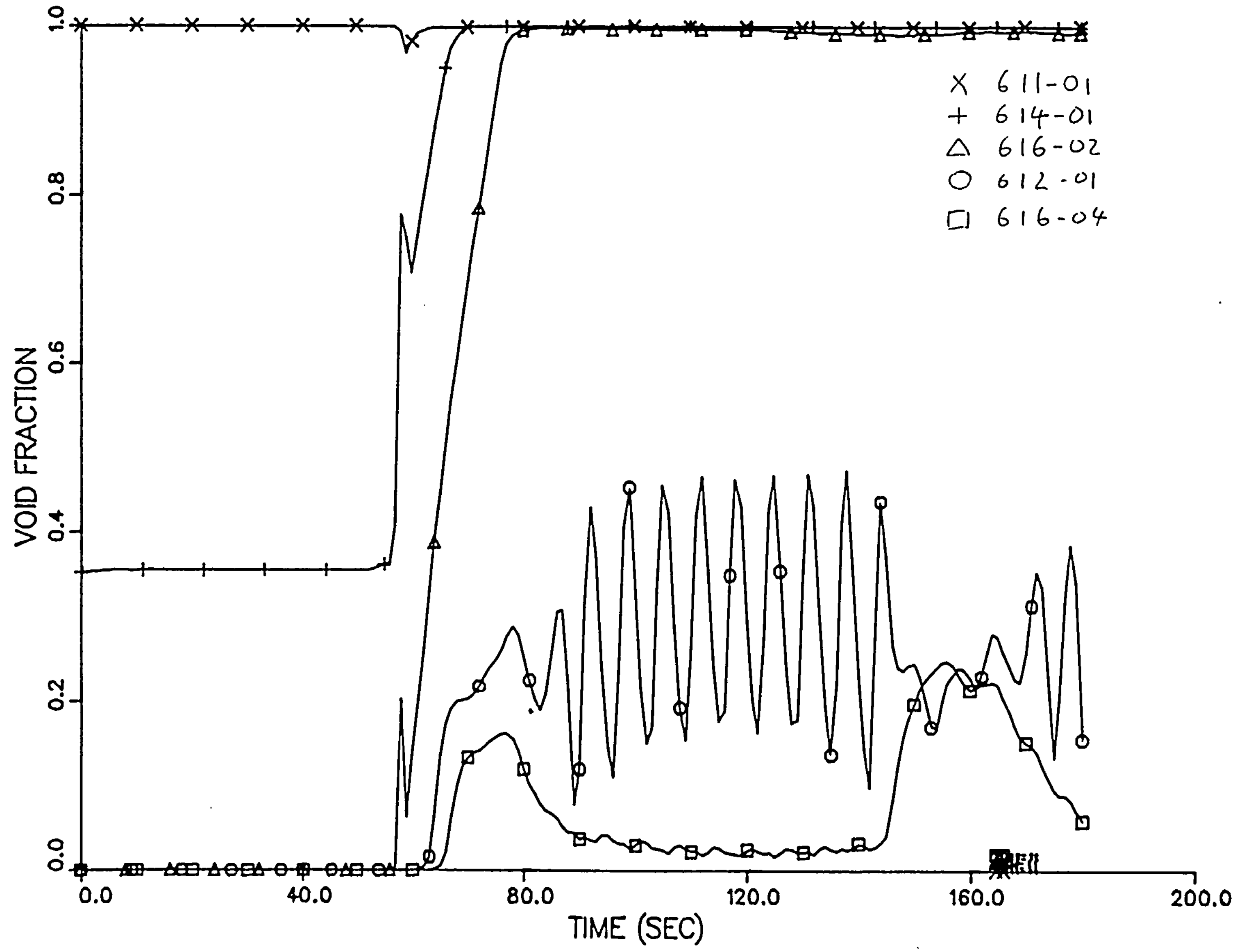
FIG 5.20: TEMPERATURES & -VAPGEN 616



TIHANGE–2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 5.21: S.G. (G) VOID FRACTION DOWNCOMER/SEPAR. BYPASS



PLOT 0 15.33.21 TUES 7 FEB, 1989 JOB-DISSPLA , ISSCO DISSPLA 10.0

6. PARAMETRIC STUDY

This chapter describes the impact of various changes in the input data leading to the base case (Run 12A) and further refinements.

Table 6.1 summarises the various parameter changes for eight runs in chronological order.

6.1. Run 2A

Figure 6.2 illustrates an average primary coolant temperature drop of about 18 deg.C ,whereas the primary pressure drop is too low.(Fig 6.1) If the pressurizer steam bubble expansion, caused by shrinking of the primary coolant is correct, but the pressure is too small, this points to either an initial pressurizer level which is too small,or to an incorrect letdown flow. The charging flow rate was imposed as a time dependent junction based on the DAS recorded data (Fig. 5.7).

6.2. Run 3A

The letdown flow was adjusted by a calibrated orifice to match the letdown flow as recorded by the DAS. While the average temperature remains unchanged, the pressurizer pressure history (Fig. 6.3) shows a better agreement.This test illustrates the impact of a boundary condition such as the letdown system on the primary system.

6.3. Run 4A

At this point, the steam dump valve was re-evaluated, and the global valve area was increased from 0.12 m2 to a best estimate value of 0.16 m2 based on the design data.

This resulted, as expected, in a faster temperature drop (Fig. 6.5) and hence a faster pressure drop (Fig. 6.4) in the primary coolant system.

This run illustrates the impact of the steam dump capacity on the temperature and pressure gradients during the steam dump activation.

While the primary system parameters exhibit a very good agreement,discrepancies on the secondary system remained as shown in fig. 6.6 which displays the pressure evolution in steam generator B.

6.4. Run 6A

Recognising the importance of the timing of the steam dump system, a 1 second delay was introduced between the steam dump demand (Fig. 5.16) and the effective opening of the steam dump valve. (Variable trips 487 and 488 on figure 4.2).

A slight improvement is observed in the pressurizer pressure (Fig. 6.7) and the primary coolant average temperature (Fig. 6.8) , while the agreement on the secondary side (Fig. 6.9) remains unsatisfactory during the fast transient phase between 60 and 80 seconds and also during the last phase after 140 seconds.

6.5. Run 9B

While focusing the attention on the steam generator pressure evolution following scram, several minor changes were attempted to increase the intermediate pressure plateau observed in the calculations (Fig. 6.9) between 60 and 80 seconds. Retarding the effective scram by 1 sec did not improve the simulation. Slight variations in the steam dump tuning did not resolve the discrepancy and the attention turned to the impact of the condensation which was observed in the steam generator regions where temperature desequilibrium existed between the vapour and liquid phase.

Since the inverted J-tubes on the feedwater sparger are not simulated explicitely, and since condensation was observed there, an artificial trip valve was simulated to avoid vapour backflow into the sparger,and hence to reduce the condensation of vapour on the colder feedwater in the sparger.

Fig. 6.10 illustrates that, by avoiding the steam condensation in the sparger, the pressure evolution in the steam generator is hardly improved. Indeed one observed that for such modification, the total vapour mass condensing on colder water in the sparger region was reduced from 316 to 300 kg. This means that excessive condensation in the other regions of the steam generator could be responsible to a larger extent for the pressure discrepancy during the fast transient phase, than the inacuracies in the steam generator boundary conditions.

6.6. Run 12A (Base Case)

For this run, the artificial trip valve at the sparger outlet was removed as its effect was minimal.

In an attempt to improve the steam generator pressure evolution during the last phase (t > 142), wherein the steam dump was under manual control, the closing time of the steam dump was advanced from 170 sec to 167 sec in order to match the DAS recorded steam dump timing.

Fig. 6.11 illustrates the comparison between the recorded and calculated steam dump capacity for this run, wherein the steam dump command is under automatic control (described in chapter 4) until t = 142 sec, and under manual control (i.e. imposed by data input limited to 31 %) for the last phase.The results of this run are discussed in chapter 5.

6.7. Run 14A

This test highlights the sensitivity of the quick opening time of the steam dump on the steam generator parameters.

In a first step, the quick opening time was increased from the design value of 3 seconds to 5 seconds and this resulted in an increase of the intermediate pressure plateau (between 60 and 80 sec.) from 64 to 65 bars. For run 14A, the quick opening time was increased from 7 to 9 sec.

The impact of this modification is shown for the steam generator B pressure response in Fig. 6.12 and the narrow range level response in Fig. 6.13. Comparing Figs. 5.8 and 6.12, one notices an improvement in the pressure response during the dynamic phase between 60 and 80 sec. Comparing Figs.5.12 and 6.13 also shows a considerable improvement for the water level response.

This run once again highlights the extreme sensitivity of the steam generator response on the steam dump dynamic behaviour. However, it is very unlikely that the quick opening time in the plant should be more than twice the design value. Hence,one should attribute the major part of the discrepancy to the condensation model in the code.

6.8. Run 15A

As mentionned in chapter 5, the timing of the operator intervention

could not be reconstructed exactly. For instance, it is not certain if the reactor scram was caused by turbine trip (on high steam generator water level) as is the normal sequence, or if the reactor was tripped manually whereupon a turbine trip follows in 0.1 sec.
In all former cases, reactor scram was considered to be the initiating event, while for this run, a turbine trip was programmed as the triggering event. This is the only change with respect to the base case.
Figs. 6.14 to 6.17 illustrates respectively the evolution of the primary pressure,average primary coolant temperature, the steam generator B pressure and water level.
This run shows that this modification hardly influence the parameters in the primary system, while there is a shift in the pressure history and the appearance of a double spike for the water level indication during the dynamic phase of the transient.
For this run, special attention was focussed on possible condensation phenomena which could affect the pressure and water level evolution during the dynamic phase of the transient.
Table 6.2 represents some important minor edit variables related to the water fallback volume 814-01 for steam generator B over a short time period from 55 to 63 seconds.
Between 57 and 58 seconds, the pressure increase levels-off while the narrow range water level (LVLGE) reverses in trend. These 2 parameters are global parameters and their variations may be explained in terms of the local parameters in volume 814-01. The reversal in the water level trend is also seen as a reversal in the vapour void fraction and is caused by the relatively high condensation occuring at t = 57 sec. (At this time, VAPGEN = -9.805). While the fluid temperature increases continuously but remains subcooled during this period, the vapour temperature experiences a short peak of high superheat (about 28 deg. C)

but is quickly reduced to quasi saturated temperature due to high condensation occuring at this moment.

From 57 seconds on, the condensation stays at a high level,and is most probably responsible for the intermediate pressure plateau and for the fluctuation in the water level. This would point to the fact that in the flow regime 4 (i.e. bubbly flow regime), RELAP-5 cannot maintain strong thermal desequilibrium and hence, tends to be too homogeneous.

TABLE 6.1

RUN NUMBER DATE	02A 88/10/28	03A 88/11/04	04A 88/11/05	06A 88/11/09	09B 88/11/14	12A 89/01/12 BASE CASE	14A 89/01/18	15A 89/01/19
CHARGE SYSTEM	DAS value	DAS value	DAS value	DAS value	DAS value	DAS value	DAS value	DAS value
LETDOWN ORIFICE	coded	new	new	new	new	new	new	new
ST. DUMP AREA	0.12 M2	0.12 M2	0.16 M2	0.16 M2	0.16 M2	.16154 M2	.16154 M2	.16154 M2
ST. DUMP DELAY	0 s	0 s	0 s	1 s	1 s	1 s	1 s	1 s
FW TORUS	hrznt'l	hrznt'l	hrznt'l	hrznt'l	inclined	inclined	inclined	inclined
FW TORUS VALVE	no	no	no	no	trip vlv	no	no	no
MANUAL ST.DUMP CLOSING TIMING	t=170.33	t=170.33	t=170.33	t=170.33	t=170.33	t=167.33	t=167.33	t=167.33
ST.DUMP QUICK OPENING IN	3 s	3 s	3 s	3 s	3 s	3 s	7 s	3 s
ST.DUMP SLOW OPENING IN	7 s	7 s	7 s	7 s	7 s	7 s	9 s	7 s
INITIATING EVENT	manual scram	manual scram	manual scram	manual scram	manual scram	manual scram	manual scram	turbine trip

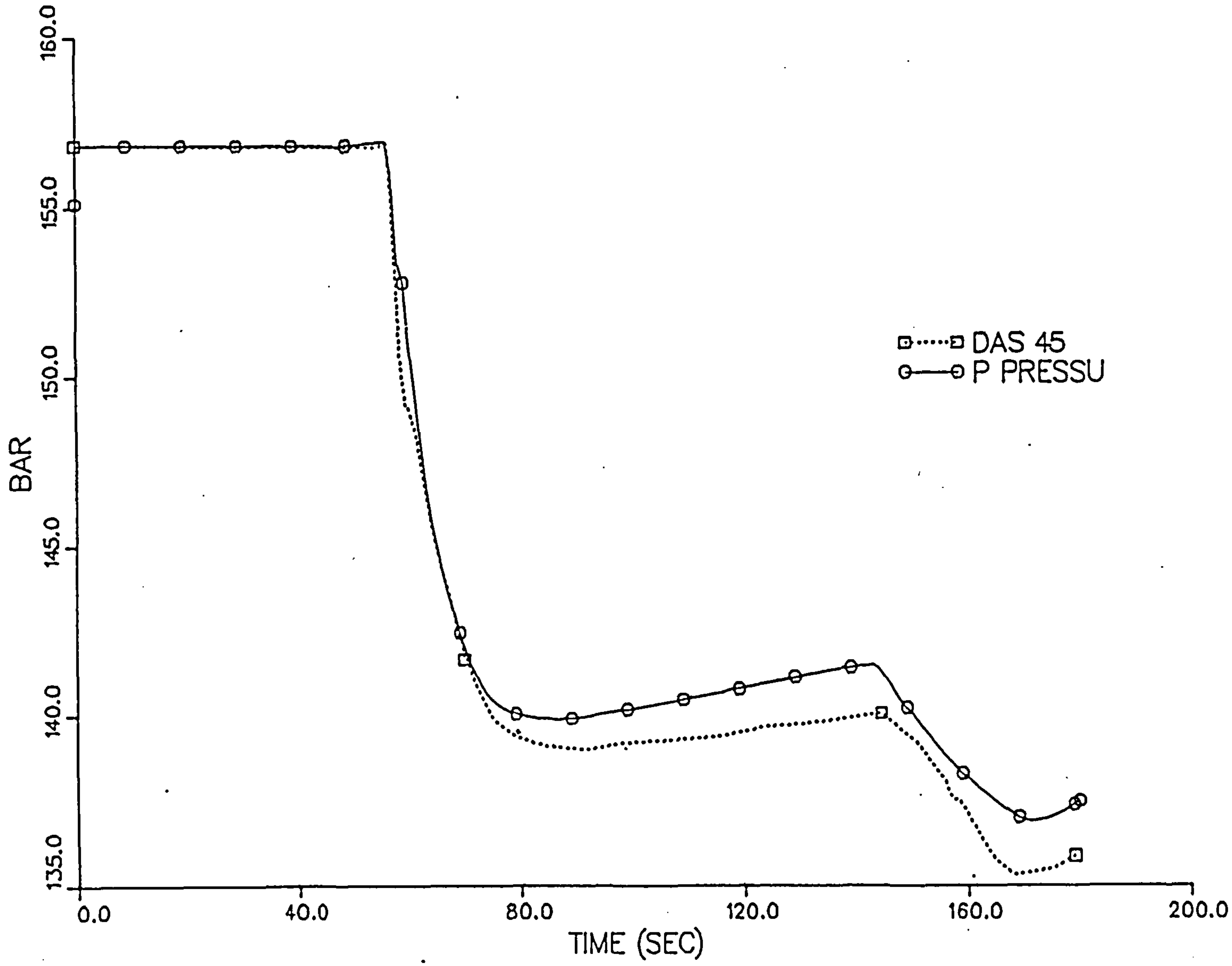
TABLE 6.2

MINOR EDIT VARIABLES TO ILLUSTRATE THERMAL NON-EQUILIBRIUM

T SEC	PRESSURE 630-01 (BAR)	CV 813 LVLGE (%)	VOIDG 814-01 (%)	TEMPF 814-01 DEG.K	TEMSAT 814-01 DEG.K (*)	TEMPG 814-01 DEG.K	VAPGEN 814-01 KG/S/M3	FLOW REG. -
55	59.24	47.81	0.013	535.7	548.	548.2	-0.029	4
56	60.62	43.13	1.684	536.1	549.	578.7	-3.6554	4
57	64.02	32.46	5.800	536.5	552.	552.8	-9.805	4
58	64.97	34.96	1.66	536.4	554.	555.5	-3.994	4
59	64.99	36.83	2.66	538.6	554.	554.5	-2.275	4
60	65.05	32.56	5.23	542.0	554.	553.8	-2.603	4
61	65.05	35.69	0.163	545.3	554.	554.5	-0.397	4
62	65.18	35.13	5.62	547.8	554.	554.6	-1.249	4
63	65.32	29.48	9.21	549.0	554.	554.4	-1.726	4

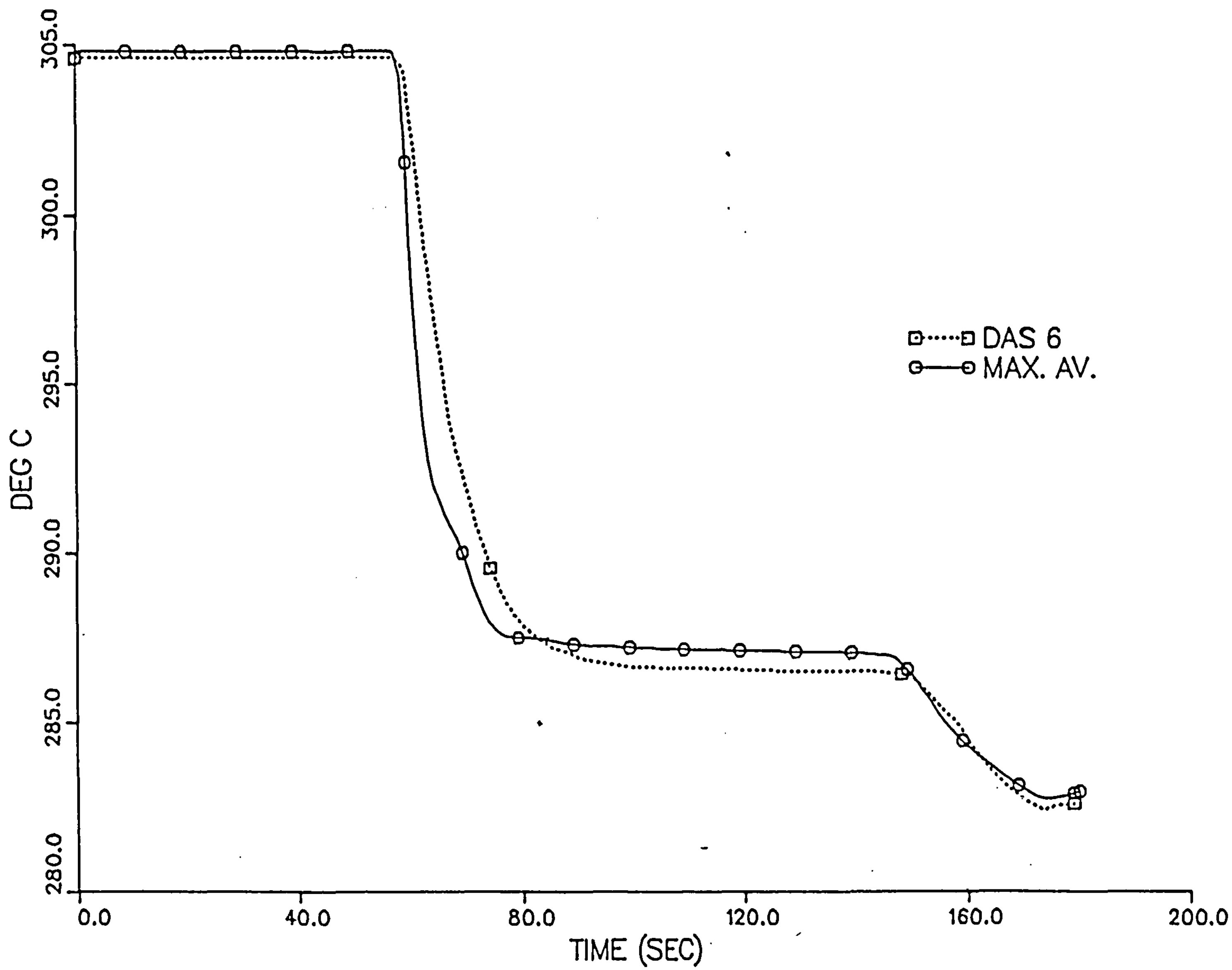
* : values for TEMSAT were evaluated manually at the pressure calculated in volume 630-01.

TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 2A
FIG. 6.1 : PRESSURIZER PRESSURE

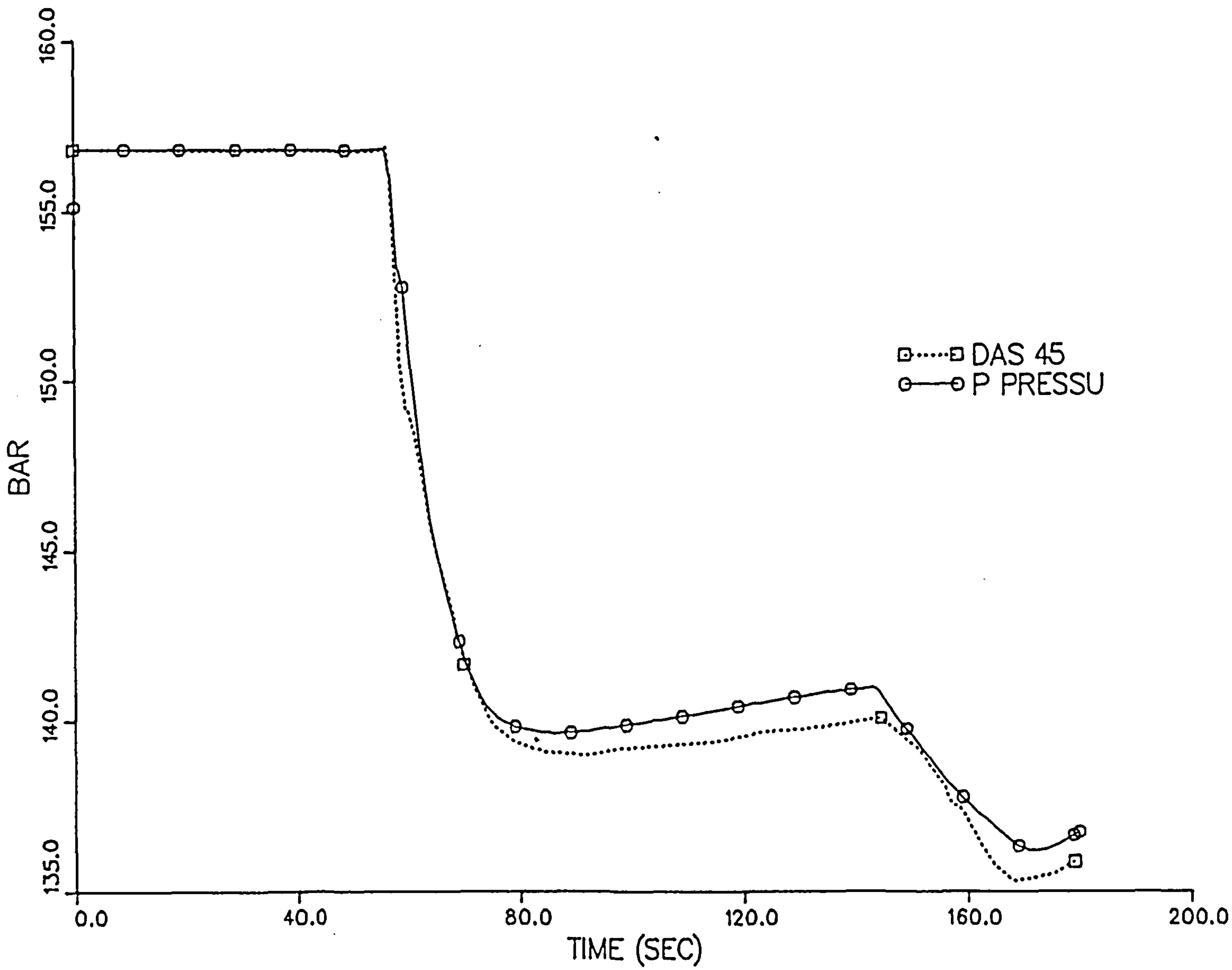


PLOT 0 11.46.36 MON 31 OCT, 1988 JOB=DISSPLA , ISSCO DISSPLA 10.0

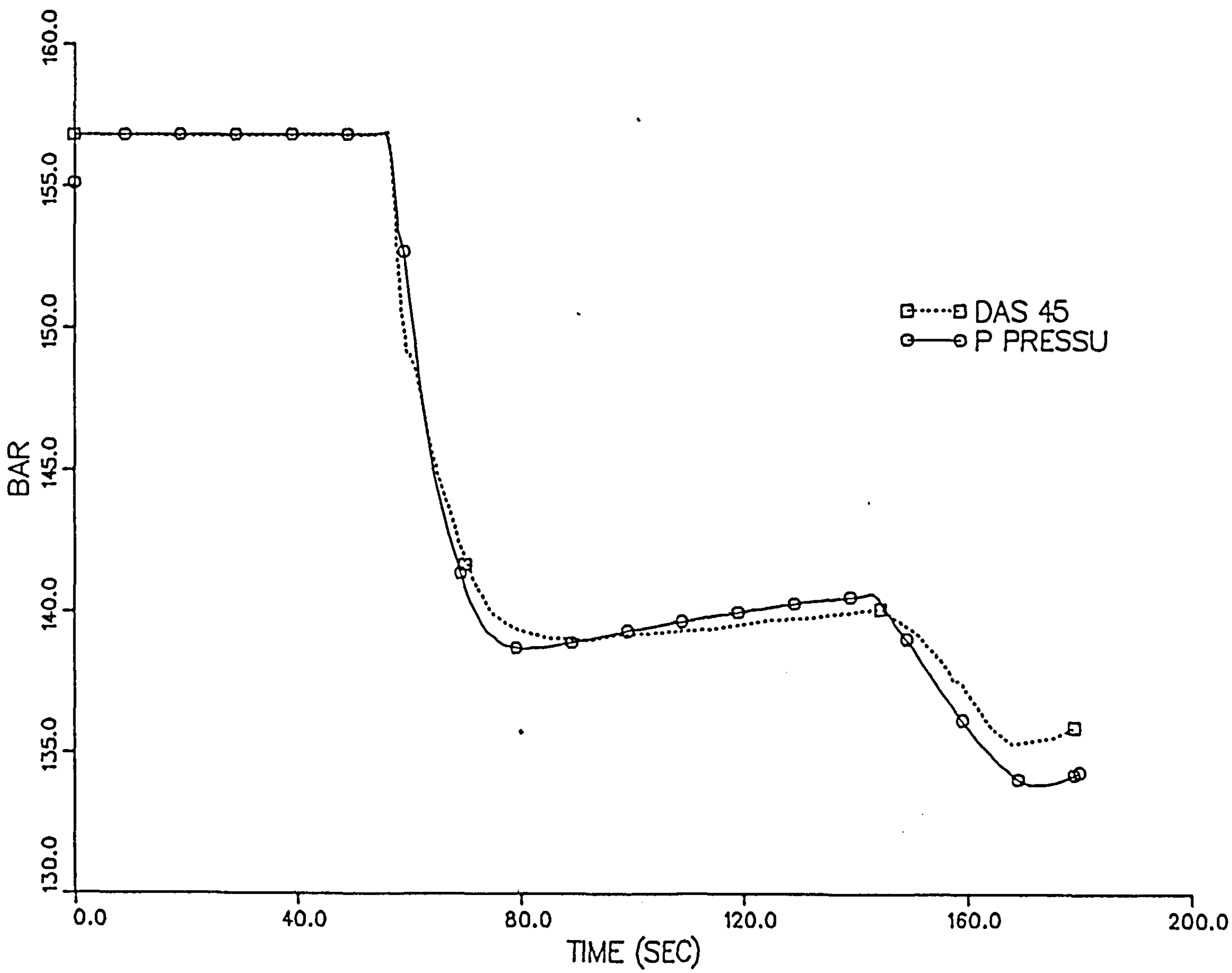
TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 2A
FIG. 6.2 : MAX.AVERAGE TEMPERATURE

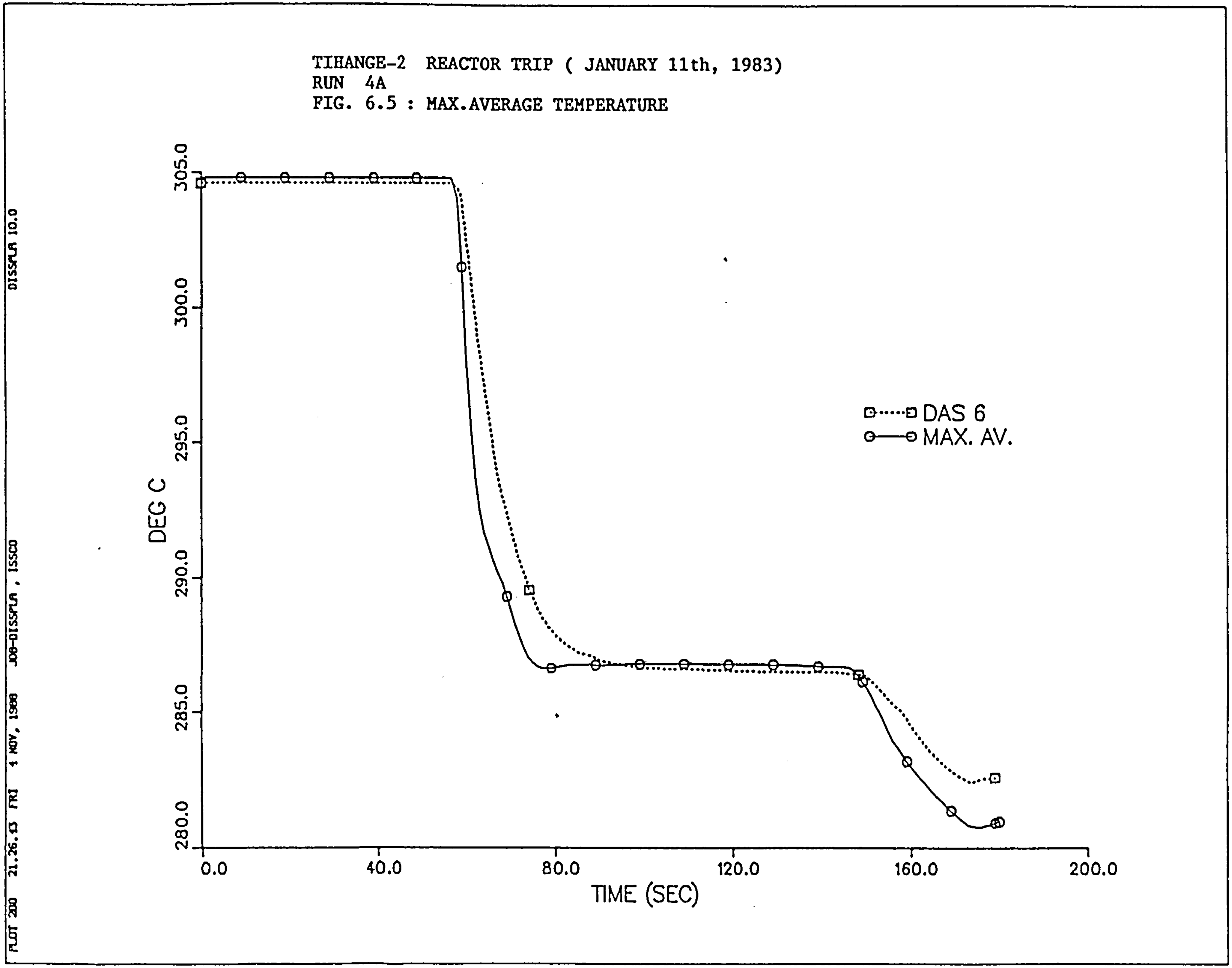


TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 3A
FIG. 6.3 : PRESSURIZER PRESSURE



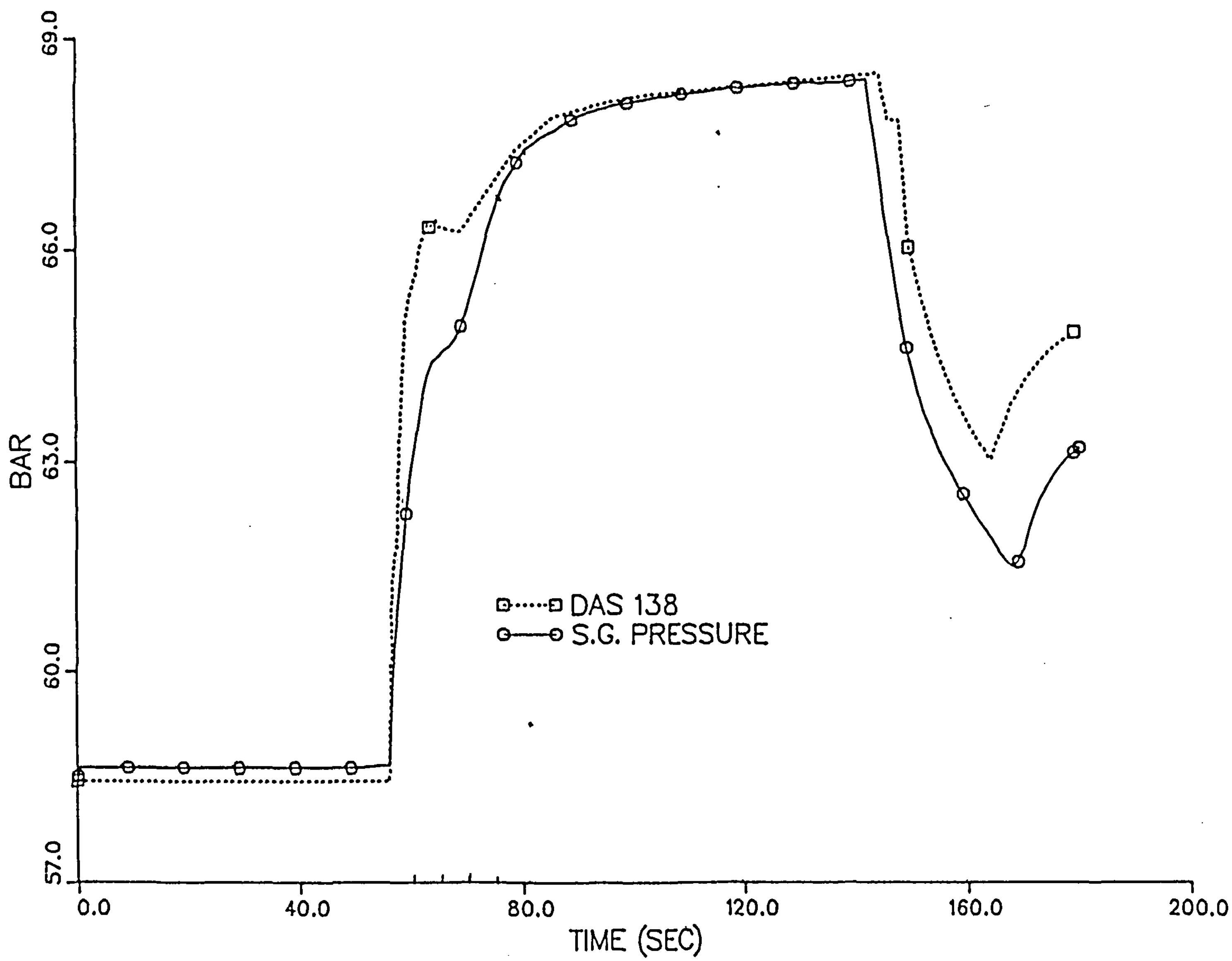
TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 4A
FIG. 6.4 : PRESSURIZER PRESSURE



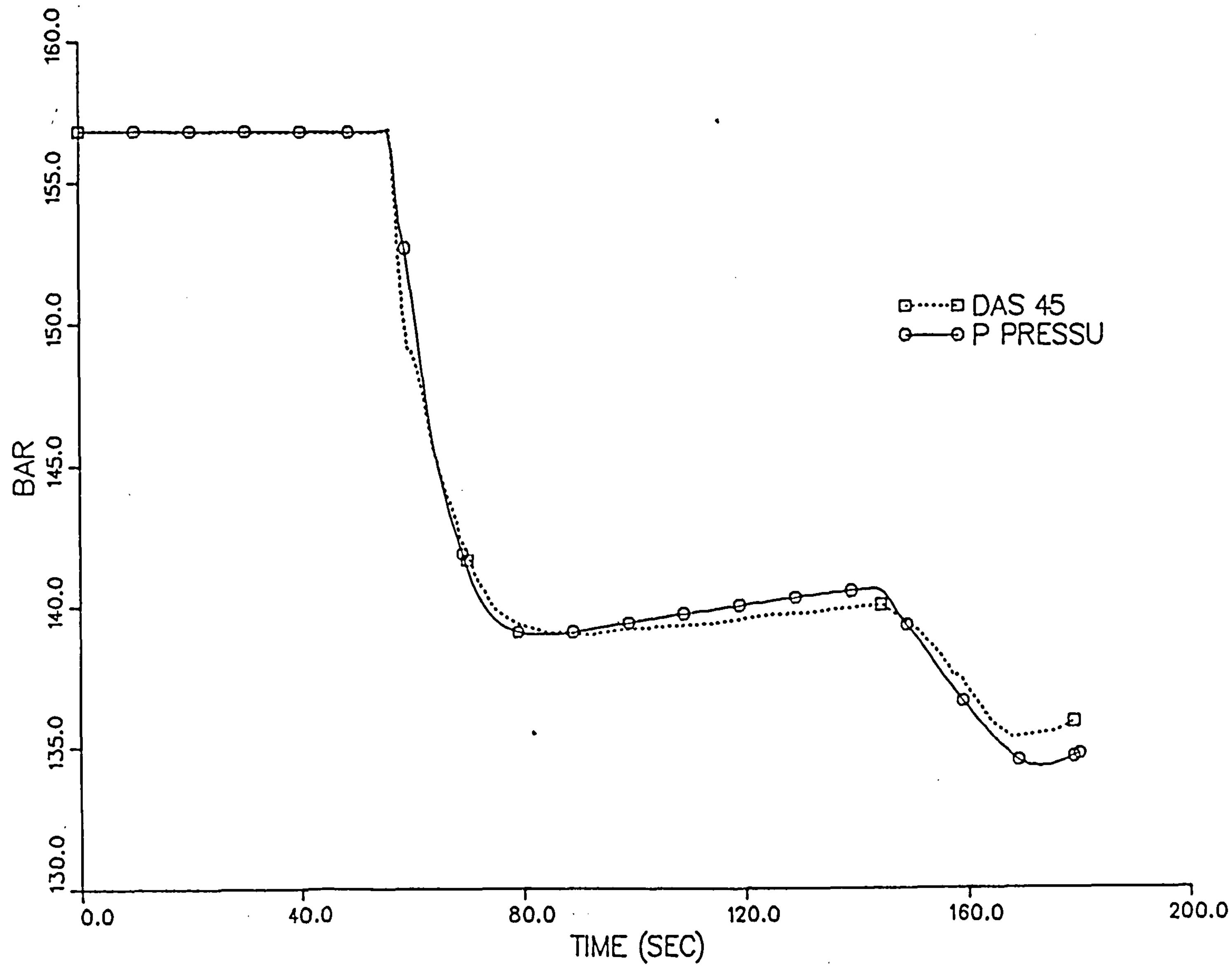


TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 4A
FIG. 6.5 : MAX.AVERAGE TEMPERATURE
DEG C
305.0
300.0
295.0
290.0
285.0
280.0
0.0
40.0
80.0
120.0
160.0
200.0
TIME (SEC)
DAS 6
MAX. AV.
DISSPLA 10.0
PLOT 200 21.26.13 FRI 4 NOV, 1988 JOB-DISSPLA , ISSCO

TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 4A
FIG. 6.6 : STEAM GENERATOR (B) PRESSURE

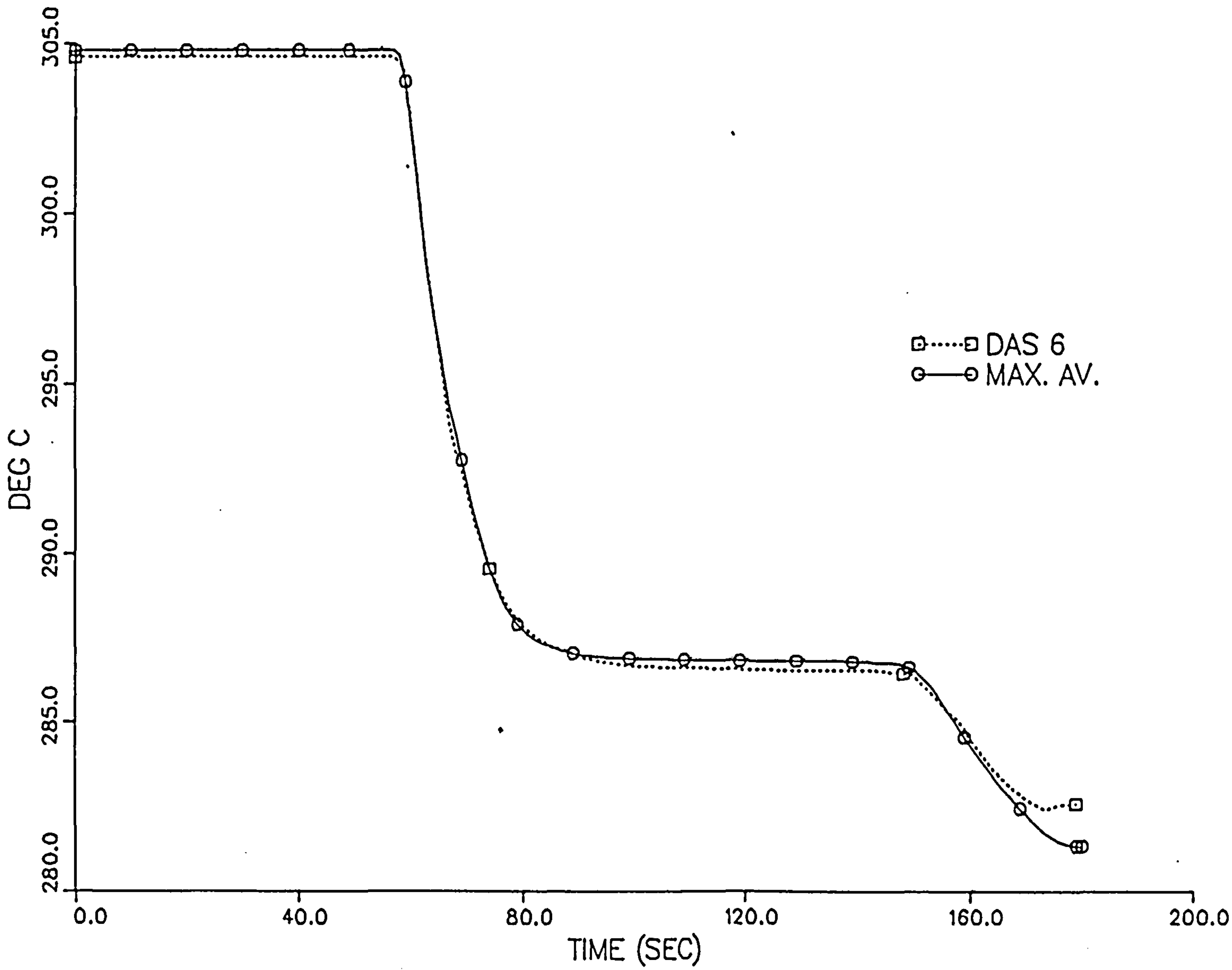


TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 6A
FIG. 6.7 : PRESSURIZER PRESSURE



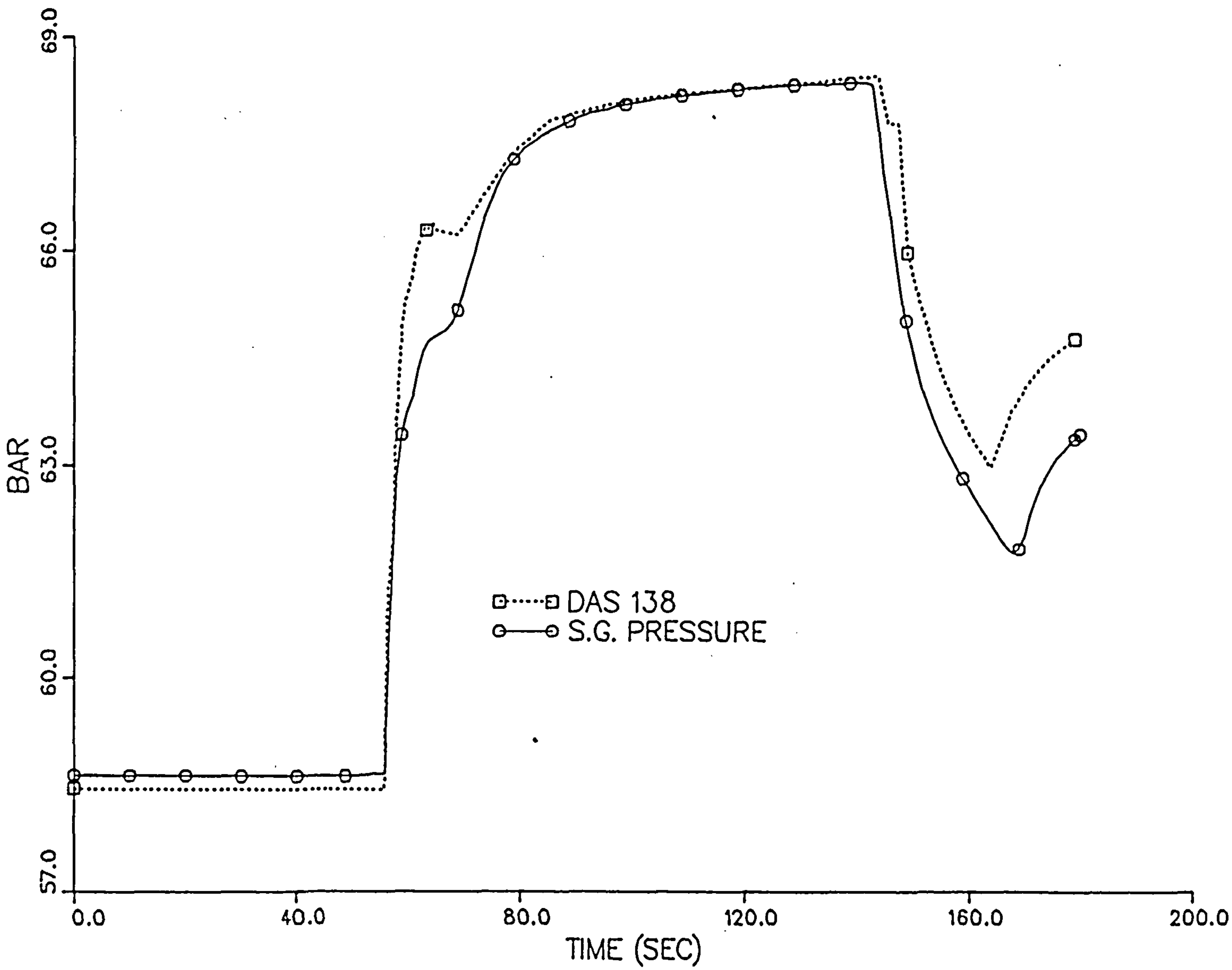
PLOT 0 21.11.02 WED 9 NOV, 1988 JOB-DISSPLA , ISSCO DISSPLA 10.0

TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 6A
FIG. 6.8 : MAX.AVERAGE TEMPERATURE

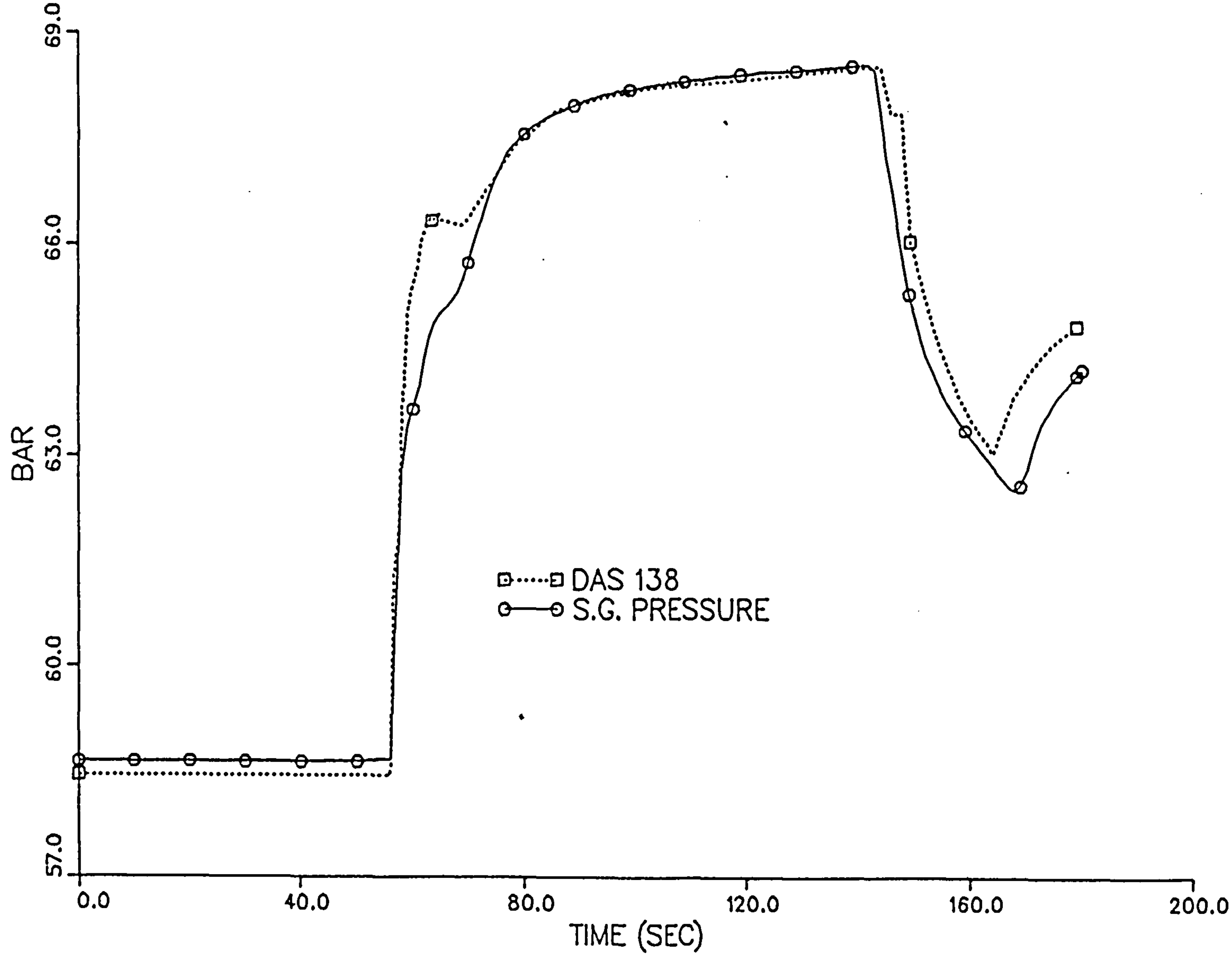


PLOT 200 21.38.54 WED 9 NOV, 1988 JOB-DISSPLA , ISSCO DISSPLA 10.0

TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 6A
FIG. 6.9 : STEAM GENERATOR (B) PRESSURE

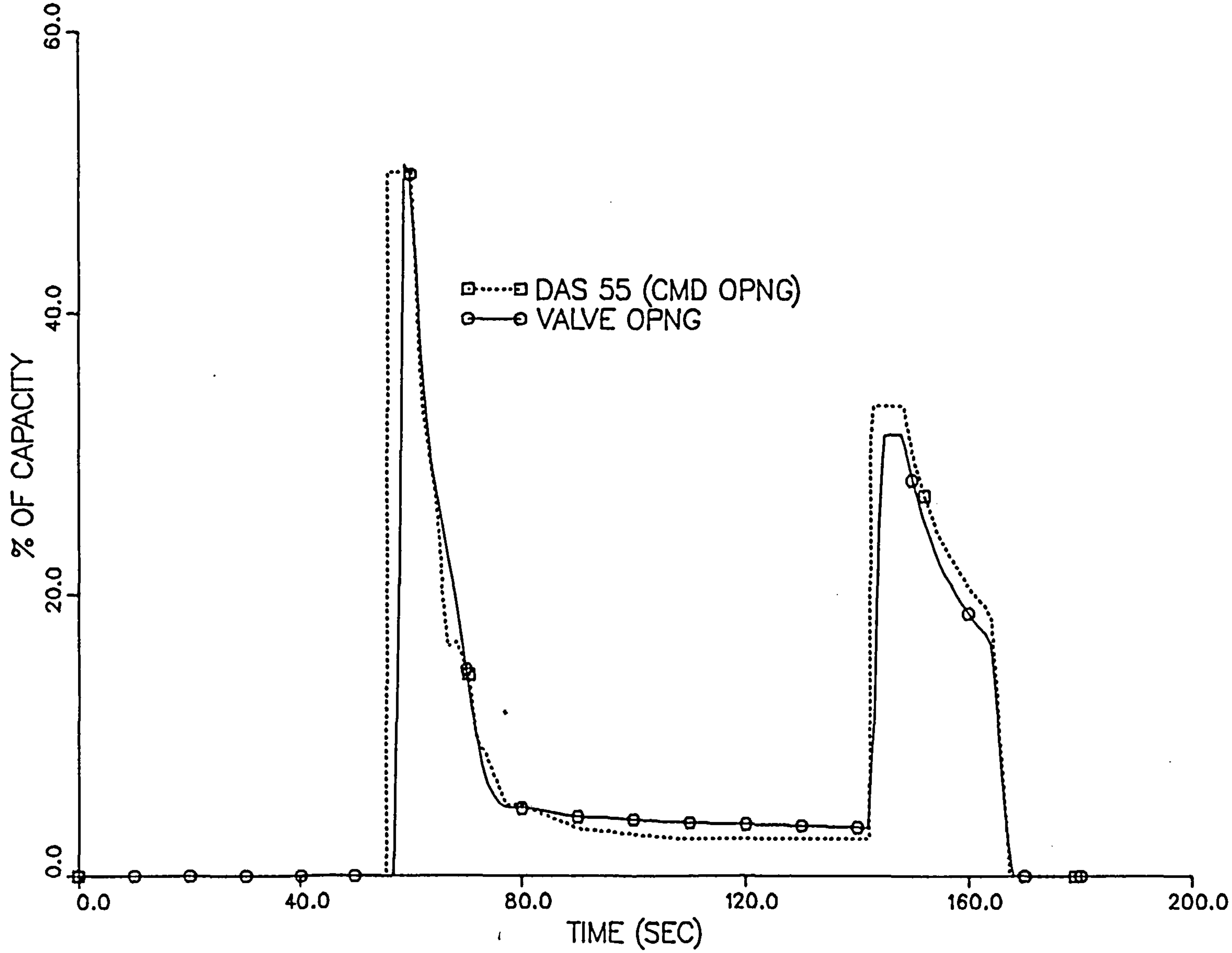


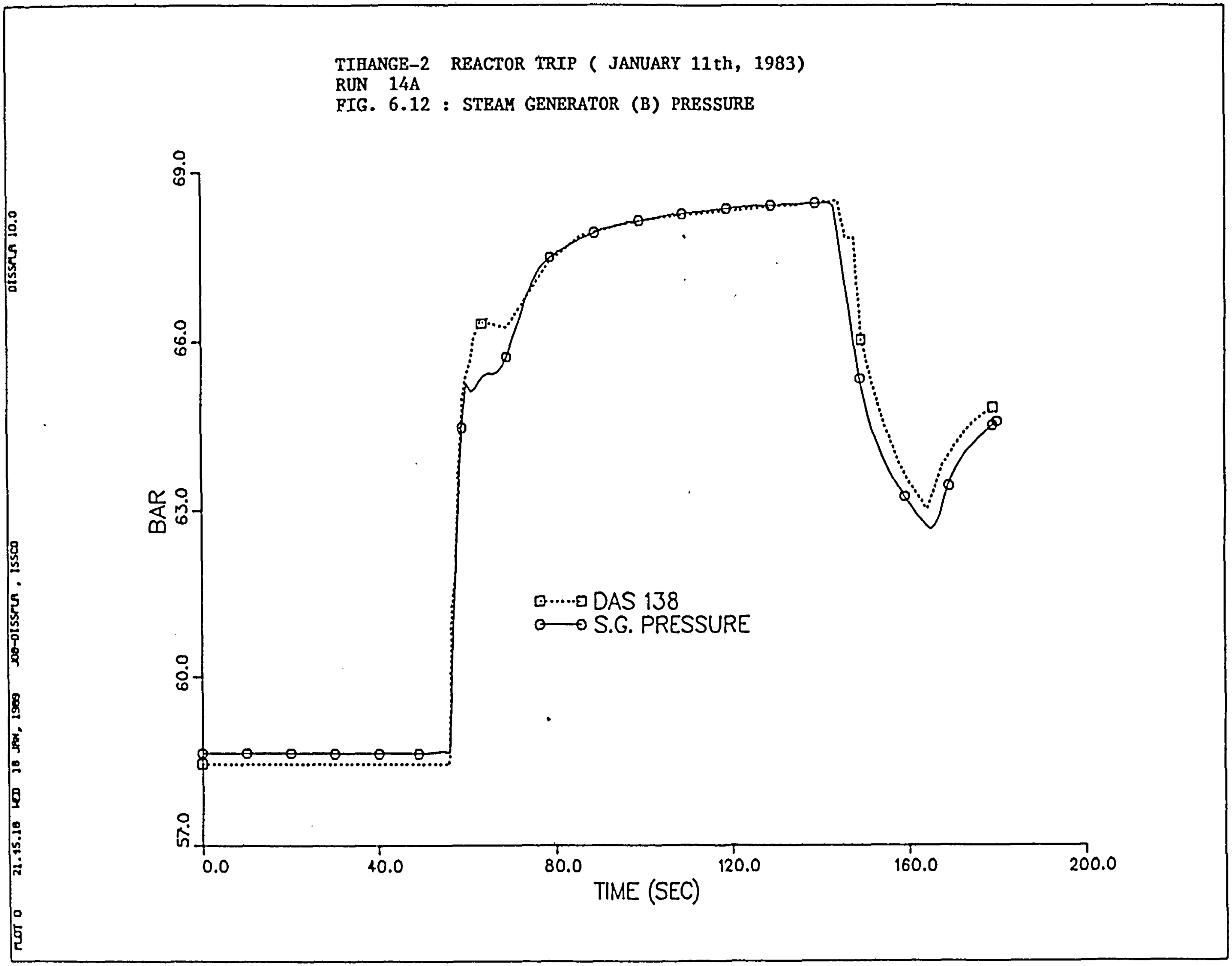
TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 9B
FIG. 6.10: STEAM GENERATOR (B) PRESSURE



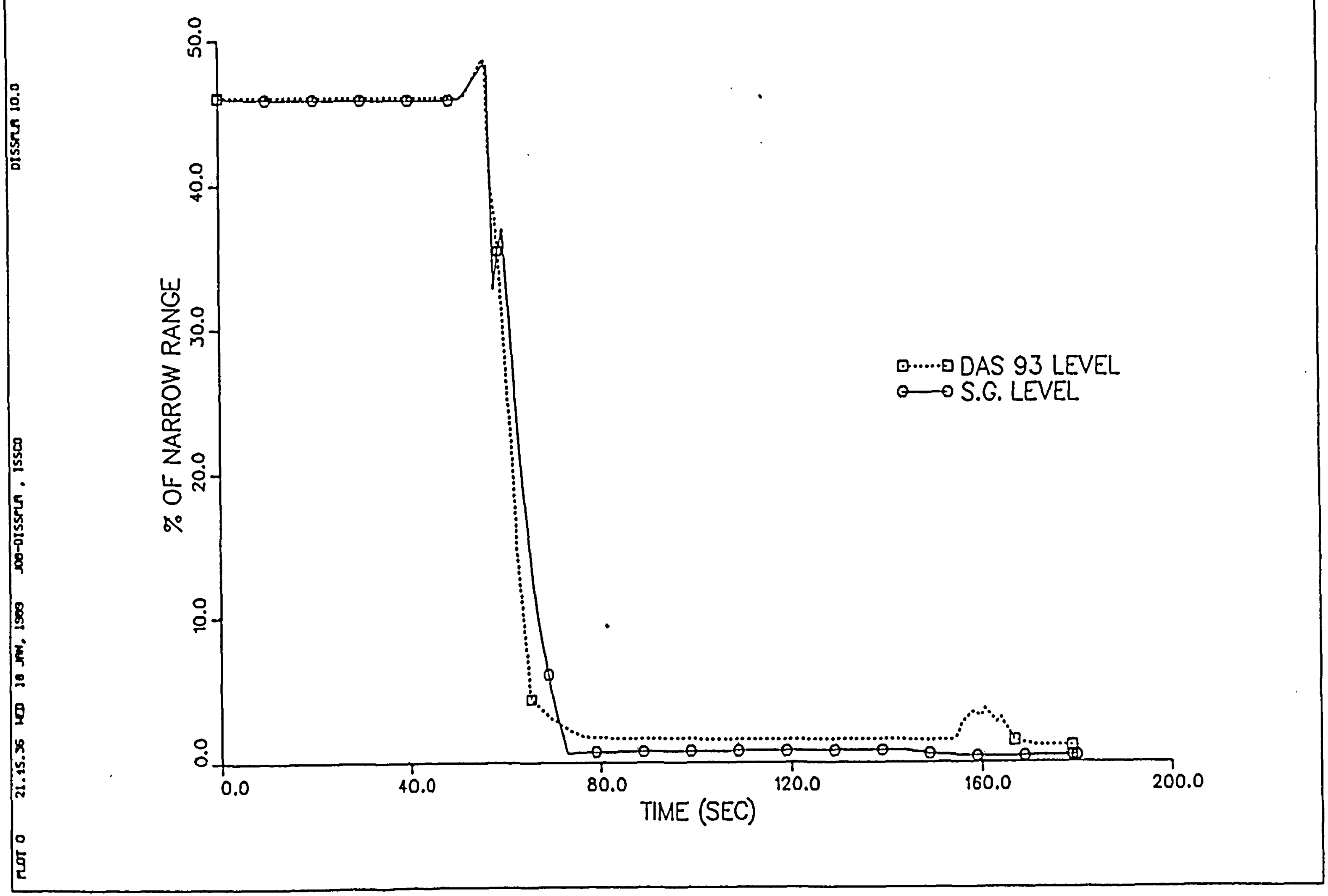
DISSPLA 10.0
18.29.55 TUES 15 NOV, 1988 JOB=DISSPLA , ISSCO
PLOT 0

TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 12A
FIG. 6.11: STEAM DUMP CAPACITY

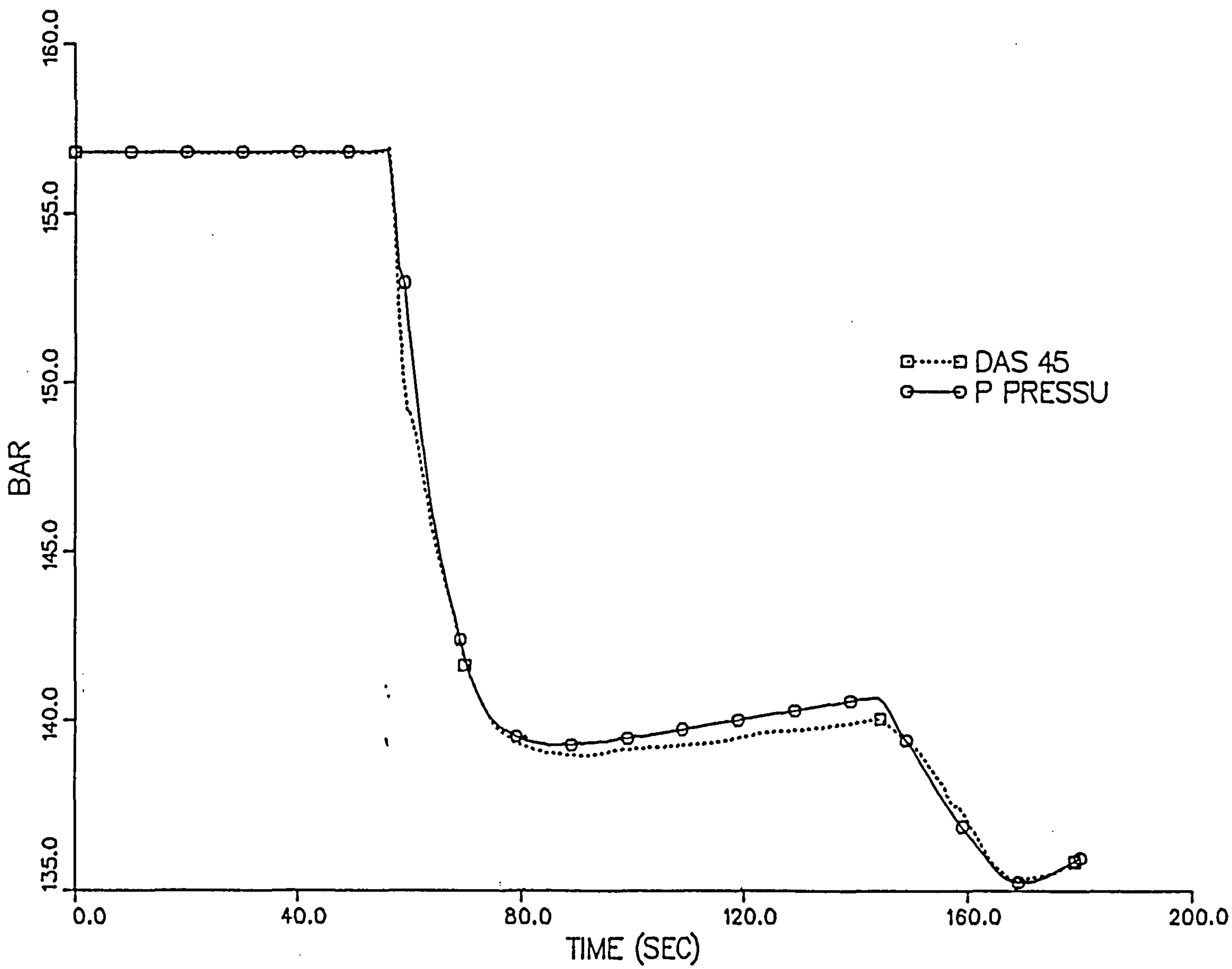



TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 14A
FIG. 6.12 : STEAM GENERATOR (B) PRESSURE
BAR
69.0
66.0
63.0
60.0
57.0
0.0
40.0
80.0
120.0
160.0
200.0
TIME (SEC)
DAS 138
S.G. PRESSURE
DISSPLA 10.0
21.15.18 WED 18 JAN, 1989 JOB-DISSPLA , ISSCO
PLOT 0

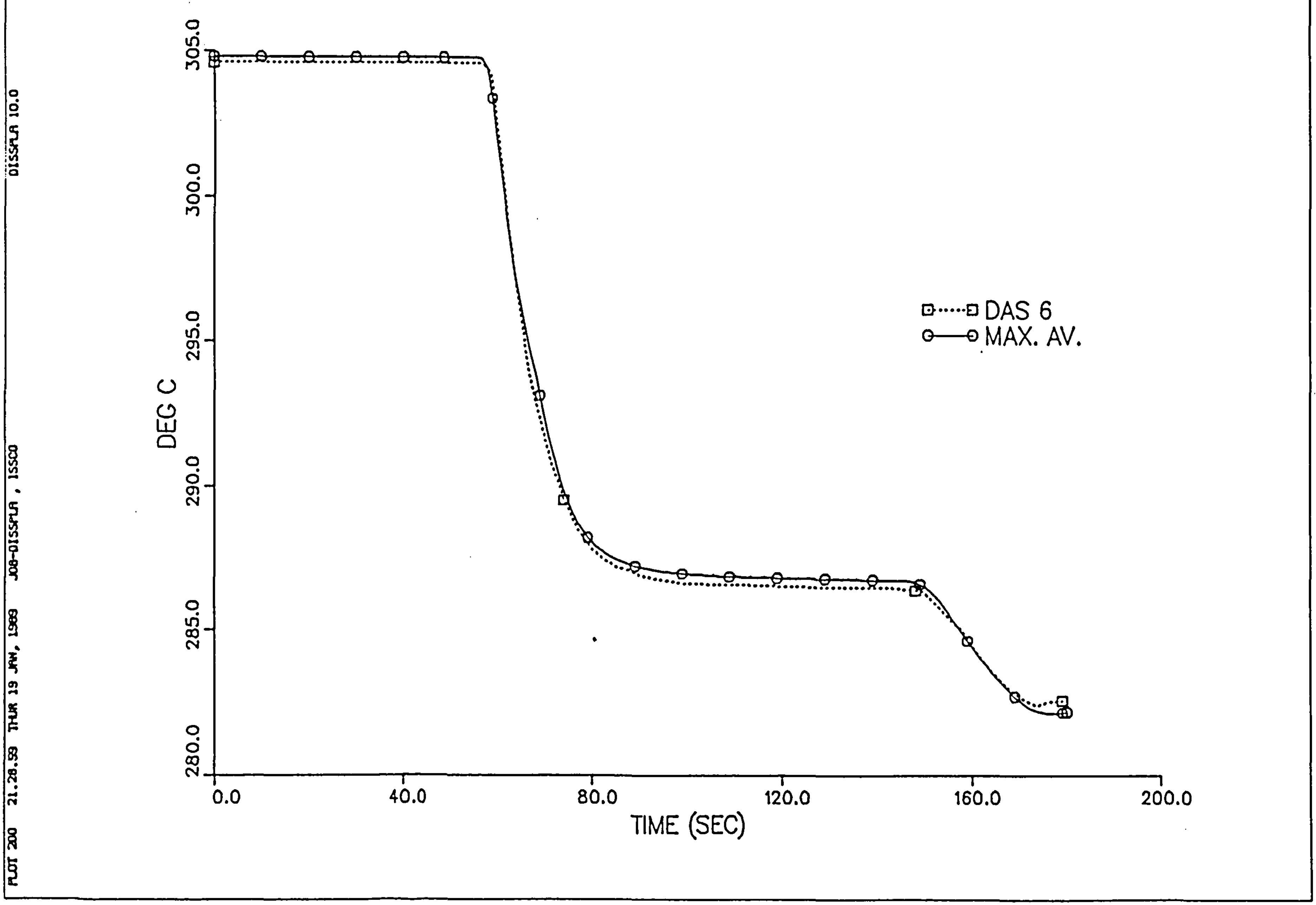
TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 14A
FIG. 6.13: STEAM GENERATOR (B) LEVEL



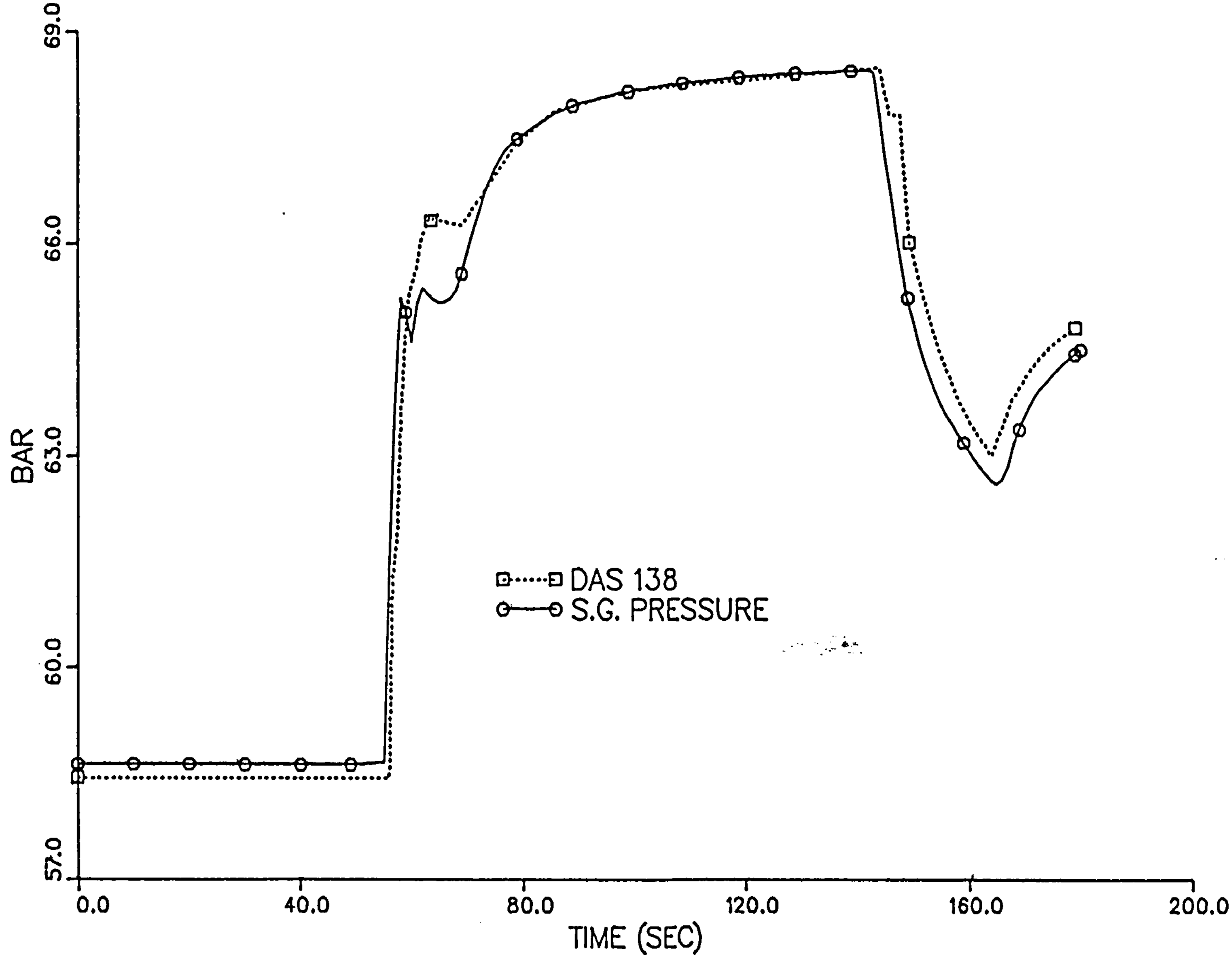
TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 15A
FIG. 6.14: PRESSURIZER PRESSURE



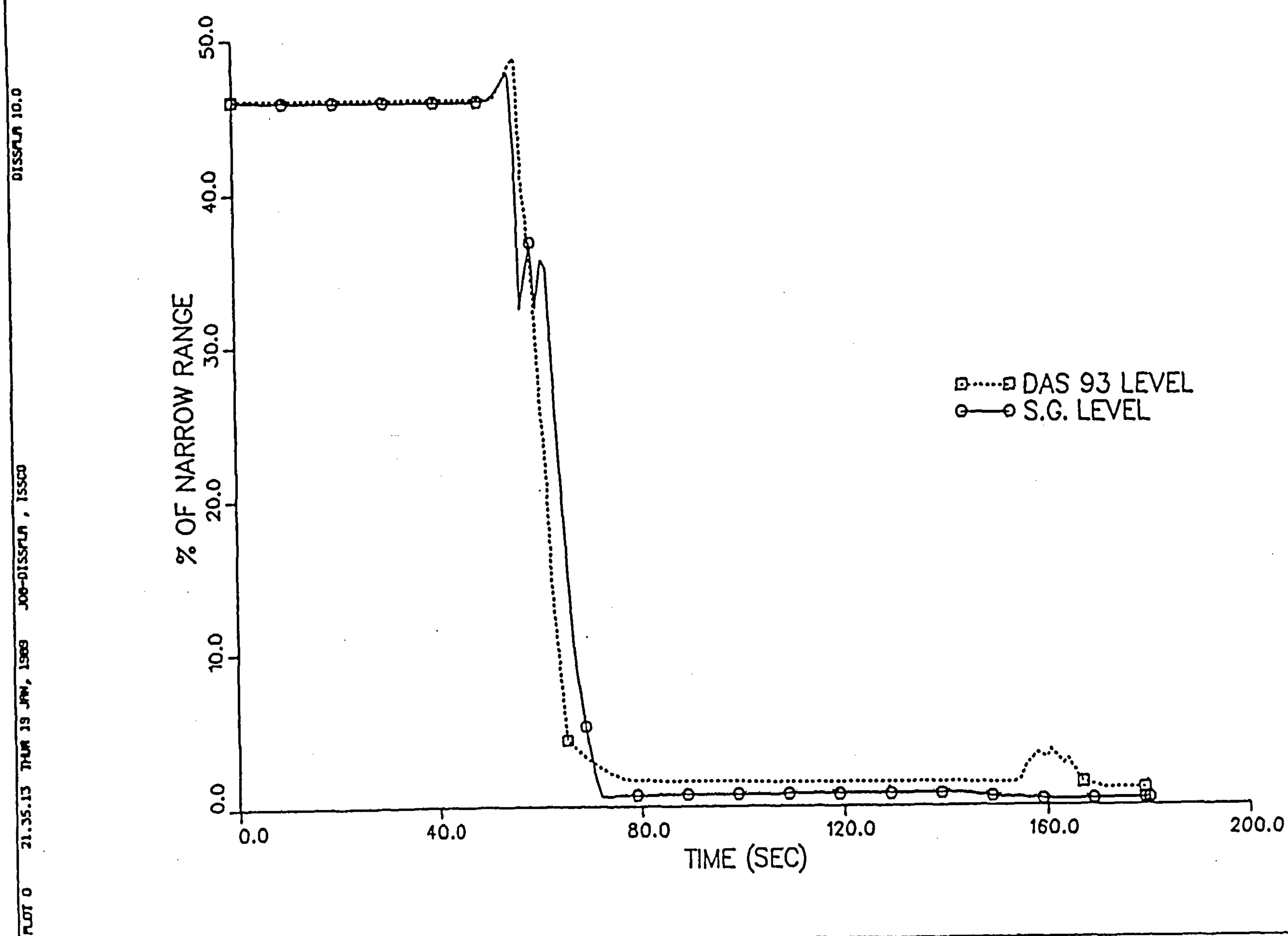
TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 15A
FIG. 6.15: MAX.AVERAGE TEMPERATURE



TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 15A
FIG. 6.16: STEAM GENERATOR (B) PRESSURE



TIHANGE-2 REACTOR TRIP (JANUARY 11th, 1983)
RUN 15A
FIG. 6.17: STEAM GENERATOR (B) LEVEL



7. RELAP-5 RUN STATISTICS

The study was performed on a CYBER 180/835 computer with a rated performance of 1.25 MIPS.
The requested time step for the whole calculation (Base Case) was 0.125 sec (Courant DT = 0.128 sec) and only 8 repeated advances, of a total of 1459 attempted advances, were required.

Fig. 7.1 illustrates the CPU time versus transient time, for which a constant performance is obtained of about 45 CPU Sec/Transient Sec.
The code performance PF [=(1000*CPU)/(N*DT)] amounts to :

(1000*5971)/(264* 1067) = 21.2 ms/step/volume

At time t = 48 sec, a restart was made to disable the steady state controllers such as:

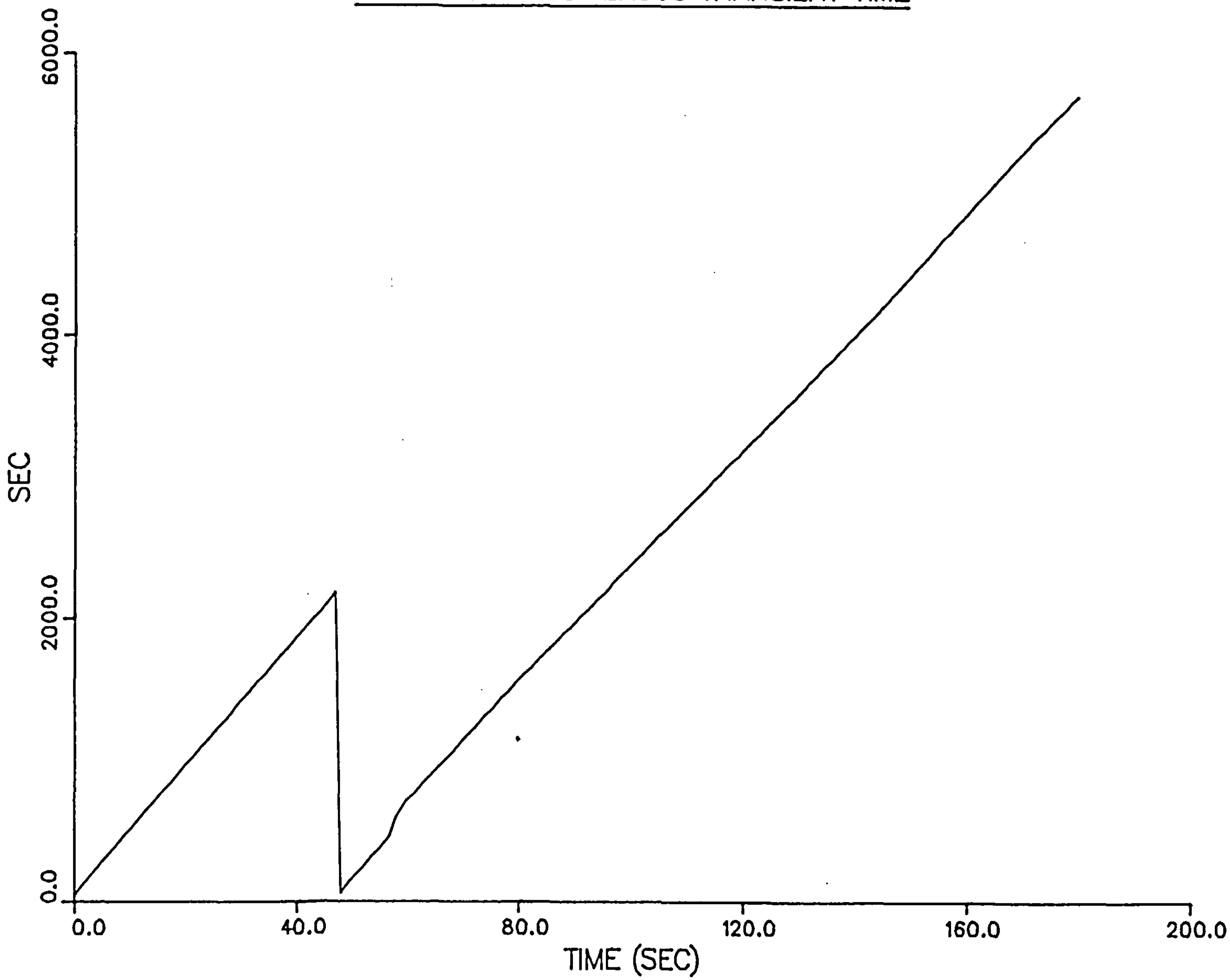
- Time dependent volume on pressurizer ;
- Fill and leak junctions on pressurizer and steam generators to obtain the requested water levels ;
- Steam generator pressure controllers ;

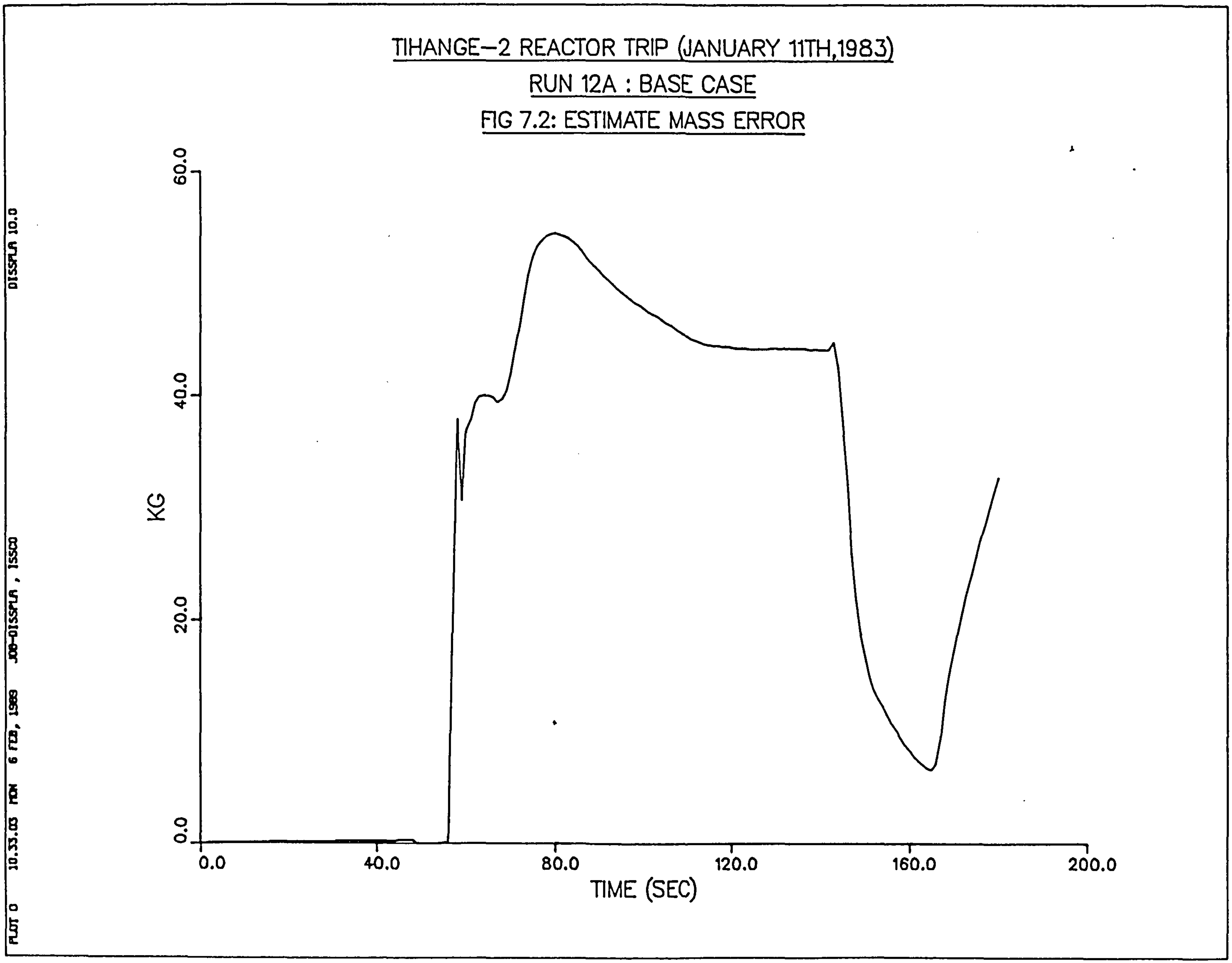
Fig. 7.2 illustrates the evolution of the mass error, resulting in a maximum mass error of 54.5 kg, yielding a global mass error ratio of 1.22*(10**-4). The main sources of mass error were located in the surge line (Volume 400) and the pressurizer (Volume 410).

TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)

RUN 12A : BASE CASE

FIG 7.1: CPU TIME VERSUS TRANSIENT TIME




TIHANGE-2 REACTOR TRIP (JANUARY 11TH,1983)
RUN 12A : BASE CASE
FIG 7.2: ESTIMATE MASS ERROR
KG
60.0
40.0
20.0
0.0
0.0
40.0
80.0
120.0
160.0
200.0
TIME (SEC)
PLOT 0 10.33.03 MON 6 FEB, 1989 JOB=DISSPLA , ISSCO DISSPLA 10.0

8. CONCLUSIONS

8.1 This assessment study illustrates that the RELAP-5 Mod-2 code is able to simulate the basic thermal hydraulic phenomena that occur in a full scale nuclear power plant following reactor trip.

8.2 A generic problem for code assessment based on real plant transients is the quality of the recorded plant data.
This test shows that, notwithstanding the high quality of the data acquisition system (DAS), the data are affected by a rather large uncertainty due to imprecision or offset of the many sensors,which are large compared to the high precision of the data obtained from separate effect tests or from integral scaled facilities.

8.3 The limited number of sensors available on a full scale plant, precludes one to improve the code constitutive equations. However, the basic merits of such assessment study should be:

- to gauge the scaling effect on the code models and correlations
- to uncover some code weaknesses which then should be improved on the basis of separate effect tests.

8.4 Agreement between recorded and calculated parameters is considerably better for the primary coolant system than for the steam generators. This is a general observation for most full scale plant transients when the reactor coolant system remains highly subcooled.
The basic reasons are:

- The single phase treatment does not pose any challenge to the code, while the highly two-phase nature of the fluid in the steam generators constitutes a real test of the delicate closure equations dealing with the interphase mass,momentum and heat transfer.
- The variations of the primary coolant parameters are highly buffered by the steam generators which take the largest share of the power mismatch upon sudden turbine valve closure and steam dump activation.

8.5 The parametric study clearly highlights the fundamental importance of boundary conditions for the plant model. The study shows that apparent minor changes to the timing and dynamics of the steam dump can induce rather large variations in the steam generator parameters (e.g. water level indication). Before deciding that the code manifests some weaknesses, one should filter out modeling uncertainties by performing a lot of sensitivity studies on the parameters of the systems which fix the boundary conditions of the modeled conponents. For any plant, these parameters are never known to very high accuracy. This concerns especially the secondary side steam and feedwater components and related systems such as:

- Atmospheric steam relief valves
- Steam dump systems
- Feedwater level control systems

8.6 This report underscores very well the need for the code users,not only to have a good understanding of the code and its limitations, but also to acquire a detailed knowledge of the plant and the functionning and the location of the plant sensors. An example is the temperature measurement for the primary coolant system which is obtained in the RTD bypass loops of the primary loops. The non negligible fluid transport time in these bypass loops does affect the timing of the steam dump behaviour,and should be accounted for by suitable delay if these loops are not simulated explicitely.

8.7 From this assessment study, one can conclude that the two phase code models do not tolerate high thermal desequilibrium conditions for the bubbly flow regime under fast pressurisation, and that, due to premature condensation, the temperature of the gas phase returns too soon to the quasi saturation conditions.This effect shows up as a temporary stagnation of the pressure and an abnormal water level response in the steam generators. One has to keep in mind that these 2 parameters are basically the only 2 parameters that the operator sees in the control room and upon which the control systems are based for plant protection and control.

8.8 The level swell observed in the recording following the manual opening of the steam dump (t > 142 sec) was not observed in the simulation. Since the recorded level swell was so small (L < 2 %), a deeper analysis of this discrepancy was not undertaken. Indeed,a small offset in the steam generator water inventory could well be the reason for such anomaly.

8.9 The run statistics illustrate that the code ran smoothly through the transient without changing the time step and with negligible mass error. However, in order to achieve real time for the given size of the model (264 volumes), the computer performance should be at least 50 MIPS equivalent.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

As part of a commissioning test series, a plant trip from 100% power was performed at the Tihange-2 Nuclear Power Plant.

A simulation of this transient was preformed by means of the code RELAP5/MOD2/Cycle36.05.

The assessment of the code is based on eight (8) runs. The RELAP5/MOD2 code is able to simulate the basic thermal hydraulic phenomena that occur in a full scale nuclear power plant following a reactor trip.

Agreement between recorded and calculated parameters is considerably better for the primary coolant system than for the steam generators.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

ICAP Program, RELAP5/MOD2, Full Scale Plant Trip

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