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International Agreement Report

Assessment Study of RELAP5/MOD2 Cycle 36.05 Based on the DOEL 4 Reactor Trip of November 22, 1985

Prepared by
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

March 1992

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EXECUTIVE SUMMARY

As part of the first cycle testing program for the Belgian DOEL 4 plant, a turbine trip on high steam generator level followed by a reactor trip was performed on November 22nd, 1985. The DOEL 4 nuclear power plant is a 3000 MWth, 3 loop, WESTINGHOUSE designed pressurized water reactor located on the left bank of the river Schelde downstream of the city of Antwerpen. TRACTEBEL was the architect engineer for the plant which started commercial operation in April 1985 and is since operated by the EBES utility in Belgium.
The test was performed specifically to test the steam dump control systems operation.
A Data Acquisition System (DAS) was operational to record a number of plant parameters. These recorded data were used to evaluate the simulation results.
The computation was performed by means of the code RELAP-5/Mod-2/CYCLE 36.05 (frozen version) on a CYBER 180/825.
The scope of simulation includes the primary coolant system, the three 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 and letdown flows were taken as boundary conditions according to the DAS recorded values by means of time dependent junctions. Similarly, the auxiliary feed water flow was taken as a boundary condition. The depth of simulation was similar to the recommended nodalisation detail for a full plant, leading to 229 volumes and 248 junctions and an optimised Courant limit of 0.12 s.
The assessment of the code is based on nine runs among which one case (RUN 12) was taken as the reference simulation.
From these studies, the following conclusions can be drawn.

The simulation of the short term transient following a turbine and reactor trip on a commercial nuclear power plant highlights the rapid changes occuring in the secondary and primary systems due to the sharp transition from the nominal power to the no-load operation.

Specific efforts were undertaken in the following areas :

- The pressure and temperature evolution of both primary and secondary systems appeared to be very sensitive to the timing of the effective actions in relation with the power sources and sinks (turbine isolation time, reactor control rods penetration time, steam dump valves opening time). Sensitivity studies were conducted to adjust the instrumentation response times.

- Some boundary conditions adaptations were necessary : i.e. sensitivity studies were conducted to determine the steam dump valves capacity at partial opening positions.

The recorded transient (turbine trip due to high level in a steam generator) presents the particularity of displaying the level of the overfed steam generator within the narrow range level indication after the trip, which is a usefull indication of the water mass content in the steam generator.
To reproduce the S.G. level after the trip, it was necessary to increase its initial water content by several metric tons. Our interpretation is that the void fraction in the S.G. 's riser was probably overestimated.

By improving the boundary conditions (time delays, steam dump valve characteristics) the calculated parameters related to the primary system and the steam generator (except for the narrow range level) agree very well with the recorded plant data.

The run statistics illustrate that the code ran smoothly through the transient without changing the time step and with a negligible mass error. The ratio of computer time to transient time on the CYBER 180/825 is about 57.
The code performance as usually evaluated amounts to 24.6 ms/step/volume.

ABSTRACT

This report presents a code assessment study for RELAP-5 MOD-2/CYCLE 36.05 based on a plant transient that occured at the Belgian DOEL-4 nuclear power plant.
High level in steam generator G led to a turbine trip followed by reactor trip. This test was performed as part of the first cycle testing program on November 22th, 1985, 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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1. INTRODUCTION'

The objective of this RELAP-5 code assessment study is to evaluate its capability to simulate a specific nuclear power plant transient. The one that has been chosen for this study is a ıurbine trip followed by a reactor trip initiated on high level in one steam generator. This event occured at the Belgian DOEL-4 power plant on November 22 th, 1985. Results of the simulation analysis using RELAP-5 code have been compared with recorded data using the Data Acquisition System (DAS) which is standard equipment for each Belgian nuclear power plant.

This report is organized as follows :

Chapter 2 : provides a brief description of DOEL-4 nuclear power plant;

Chapter 3 : details the transient as recorded on the plant DAS;

Chapter 4 : presents the RELAP-5 model used to simulate the transient;

Chapter . provides the simulation numerical results (plots) and compares them with recorded data;

Chapter 6 : presents the parametric study performed before obtaining the base case model;

Chapter 7 : highlights some run time statistics;

Chapter 8 : presents the conclusions.

All figures are gathered .t the end of the report.

2. BRIEF DESCRIPTION OF THE DOEL-4 PLANT

DOEL-4 is a 3000 MWth (1000 MWe) pressurized water reactor located on the left bank of the river Schelde downstream of the city of Antwerpen (Belgium) and featuring a 3-loop, Westinghouse designed, Nuclear Steam Supply System.
The plant was connected to the grid in April 1985.
This plant was a lead plant for the preheater type steam generators (Model E-2).

2.1. Reactor Coolant System

The Reactor Coolant System consists of three similar primary loops connected to the reactor vessel, each loop containing a circulating pump and a steam generator.
The core of Doel-4 contains 157 fuel assemblies with 264 fuel rods per assembly, generating 2988 MW of thermal power under nominal operating conditions. The Reactor Coolant Pumps, rated at 4.5 MW each, circulate 6.4 m^3/s of coolant per loop with a net pump head of 95.1 m.

The primary coolant volume changes associated with the reactor load evolution are being accomodated by a 45.3 m^3 pressurizer connected to the hot leg of loop "B" through a 14" surge line. Control of the primary pressure also takes place within the pressurizer by adjustment of the heater rods power or the pressurizer spray flowrate.

2.2. Steam Generators feedwater system and steamlines

The DOEL-4 plant is equipped with three preheater type steam generators of the counterflow type (model E2), as shown on fig. 2.1.
The main feedwater (bottom feeding) with a nominal flowrate of 2000 t/hr per steam generator, enters the secondary side of the steam generator in the preheater section located above the tubesheet plate embracing the cold leg side of the inverted U-tube bundle.

The main feedwater flows downwards into the mixing plenum where most of the feedwater is deflected upwards through the preheater, where it emerges and mixes with the riser flow from the hot leg side. The water-vapour mixture enters the separator at a quality of about 37 % (recirculation ratio of 2.7 at full power). The separated water fraction flows downward through the steam generator downcomer annulus, of which about 83 % enters the riser section surrounding the hot leg, and the remainder is injected in the preheater mixing region.
When the power of the plant decreases below 20 %, the feedwater inlet is switched from bottom feeding to top feeding.

On the primary side, the inverted U-tube bundle, with a nominal heat transfer area of 6317 m^2 consists of 4864 Inconel tubes, with a 19.05 mm outer diameter and averaging 21.9 m in length.

The steam lines connect the three steam generator domes to a common steam header. To each of the steam lines are connected the steam generators safety valves (six per steam generator) and one power operated steam relief valve to the atmosphere with an individual capacity of 410 t/hr at 82.7 bar. The fast acting Main Steam Isolation Valves (2 per steam line) allow to isolate each steam generator from the common header located outside the containment.

2.3. Auxiliary feedwater system

The auxiliary feedwater system consists of 2 motor-driven feedpumps delivering each to two steam generators, and one steam driven turbopump, normally aligned with two steam generators, such that each steam generator is potentially fed by two auxiliary feedwater pumps. Their control valve system is designed such that in the automatic mode each steam generator is supplied by a fixed, metered flow of 91 t/hr regardless of the steam generator backpressure. The auxiliary feedwater enters the steam generators via dedicated lines. The inlet nozzle is located at the level of the separator cyclones (top feeding).

2.4. Steam dump to the condenser

The steam dump consists of a bypass of the main turbine, from the main steam header to the condenser. Its includes 16 valves (4 groups of 4 valves each) of identical capacity (totalizing 85 % of nominal steam flow) opening in sequence as instructed by a controlling program built around the maximum average primary temperature or, at low load, around the steam header pressure.

Within the considered sequence of events (see chapter 3) i.e. a turbine trip occuring at nominal power conditions, the steam dump dynamics is controlled by the mismatch between the no-load temperature and the auctioneered average reactor coolant temperature (the maximum value of the three loop average temperature, as measured in the RTD bypass lines).

Whenever the measured auctioneered average primary temperature exceeds the no-load reference temperature (297.2°C), the steam dump valves start to open, aiming at a capacity proportional to the error signal (9.5 % per °C). The time needed for each valve or group of valves to reach the full open position is 7 seconds. However, for large error signals, an accelerated opening process takes over, making available in 3 seconds the full capacity of the first group (4 valves) whenever of the signal exceeds 8.3°C and the full capacity of the groups 1 and 2 (8 valves) beyond 16.7°C. Capacity reductions follow the same path in reverse.

2.5. Data acquisition system and measurements uncertainties

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 logic signals 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°C for primary temperatures; 1.7 bar for pressurizer pressure; 3 % of the range for pressurizer level; 2 bar for steam generator pressure and 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

A part of a commissionning test, a high feed water flow to steam generator G was manually forced.
It induced a very high level in that steam generator which caused a turbine trip followed by reactor trip.

The Data Acquisition System (DAS) was triggered 108 s prior to the turbine trip and recorded the most important plant parameters for about 30 minutes.

A selection of the DAS plots can be found at the end of this chapter and are listed below. Time interval is given between brackets.

- Fig. 3.1. : Turbine data [1 min, 3 min]

  curve 1 : turbine speed (rpm)
  curve 2 : HP inlet control valve 1 position (%)
  curve 3 : HP inlet control valve 3 position (%)
  curve 4 : HP inlet control valve 2 position (%)

- Fig. 3.2. : Reactor data [0, 10 min]

  curve 1 : maximum nuclear power (%)

- Fig. 3.3. : Steam generators NR levels [0, 10 min]

  curve 1 : Steam generator R NR level (%)
  curve 2 : Steam generator G NR level (%)
  curve 3 : Steam generator B NR level (%)

- Fig. 3.4. : Pressurizer data [1.5 min, 2.5 min]

  curve 1 : pressurizer level (%)
  curve 2 : pressurizer pressure (bar)

- Fig. 3.5. : Reactor Coolant temperatures [1.5 min, 2.5 min]

  curve 1 : cold leg R temperature (°C)
  curve 2 : hot leg R temperature (°C)
  curve 3 : loop R delta T (°C)

- Fig 3.6. : Steam generator pressures [1 min, 3 min]

  curve 1 : Steam generator R pressure (bar)
  curve 2 : Steam generator G pressure (bar)
  curve 3 : Steam generator B pressure (bar)
  curve 4 : Steam header pressure (bar)

- Fig. 3.7. : Steam dump data [1 min, 3 min]

  curve 1 : Steam dump opening demand (%)
  curve 2 : Steam dump reference temperature (°C)
  curve 3 : Maximum average RC temperature (°C)
  curve 4 : HP inlet valve 3 position (°C)

- Fig. 3.8. : Feed water data [0, 10 min]

  curve 1 : feed water flow to SG R (t/hr)
  curve 2 : feed water flo\ to SG G (t/hr)
  curve 3 : feed water flow to SG B (t/hr)
  curve 4 : feed water temperature (°C)

For this study, we selected a time interval of 60 s, starting at 8s before the turbine trip as it highlights the most dynamic part of the transient suitable for code assessment.

3.1. Plant status prior to turbine trip (0 < t < 8 s)

The reactor was operating at near nominal power conditions.
However, some parameter recordings deviated slightly from nominal conditions such as :

- Neutron flux at 98 %;

- Primary coolant hot-cold leg temperature diffe.ence at 34.7°C, which is 97,8 % of nominal value (35.5°C);

- Primary coolant average temperature at 311.4°C, (within th. dead band centered around 311.9°C);

- Steam generator pressures below the nominal values :

  SG R : 75.3 bar

  SG G : 74.9 bar

  SG B : 74.8 bar

  for a nominal value of 76.4 bar

As feed water flow to steam generator G was manually forced at a high value (fig. 3.8, curve 1), narrow range level in that steam generator increased (fig. 3.3, curve 2) to reach the value that initiates a P14 signal (very high level in at least one steam generator).

3.2. Reactor trip and steam dump phase (8 s < t < 52 s)

At t = 8 s, a P14 signal is generated and this leads to the following automatic actions :

- turbine stop valves closure (fig 3.1) followed by a reactor trip (fig. 3.2);

- fast closure of all main feedwater regulating valves (fig. 3.8);

- start up of the auxiliary feedwater system.

The sudden closure of the turbine admission valves causes a secondary pressure increase which gives rise to an increase in the cold leg temperature (fig. 3.5.1.).

As a result of the reactor trip (fig. 3.2), a sudden reduction of the primary coolant temperature occurs (fig. 3.7, curve 3), resulting in a shrinking of the primary coolant volume and a drop of the pressurizer level and pressure (fig. 3.4). The error signal between the average primary coolant temperature and the no-load reference temperature activates a fast opening of a fraction of the first two steam dump banks (fig. 3.7, curve 1), which limits the steam generator pressure rise, trending towards the saturation pressure corresponding to the primary no-load temperature (fig. 3.6).

The sudden pressure increase in the steam generator leads to a steam bubbles collapse in the SG riser and thus to a reduction in the SG natural circulation driving force. This shows up a fast drop in the narrow range water level (fig. 3.3), which drops below the narrow range level taps for two SG's (fig. 3.3, curves 1 and 3), while for the SG which was subjected to an initial overfeed, the residual level after turbine trip remains within the narrow range level (fig. 3.3, curve 2).

The steam flow rate is now under control of the steam dump system, whose valves are closing gradually as the primary coolant temperature trends towards the no-load reference temperature.

4. CODE AND MODEL DESCRIPTION FOR PLANT SIMULATION

The simulation was carried-out with the RELAP-5 Mod.2 cycle 36.05 code (frozen version) on a CYBER 18O/825 computer, over a period of 60 s.

The reactor model was developed using the methods and procedures recommended in the code manual (*) . The primary circuit and secondary circuit (feedwater, steam generator, main steam) were both modelled explicity by control volumes and junctions respecting the true geometric and hydraulic features of the components.
The piping and component walls and internals in contact with the coolant were represented as heat structures.

On the other hand, auxiliary components and systems are being simulated functionally i.e by using control system packages reproducing the system effect either on the primary or on the secondary system, regardless of their particular components.
This applies to :

- the pressurizer relief (PORV's) and safety valves controls;
- the pressurizer spray and heaters control;
- the main feedwater system;
- the steam generator relief and safety valves controls;
- the steam dump to the condenser.

Finally, due to limitations in the scope of simulation (e.g. balance of plant not simulated) boundary conditions must be imposed to the explicitely modelled systems or components, this concerns :

- the charging and letdown flows;
- the control rods movement in the core;
- the main turbine admission valves.

The overall nodalization totals 229 volumes, 248 junctions and 197 heat structures (see fig. 4.1).
In annex 1, a microfiche of the input deck is included.
Annex 2 gives the restart input deck for the base case.

(*) VH. RANSOM et al. , "RELAP-5/MOD-2 Code Manual", NUREG/CR-4312, August 1985.

4.1. Explicitely modelled systems

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

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

The three steam generators of the preheater type, are modelled with sufficient detail to represent the preheater section, the mixing plenum, the recirculation flow and the separator region (25 volumes per steam generator).

As far as the Core power generation is concerned the RELAP-5 point kinetics model was used for the power generation, accounting for the Doppler and moderator reactivity terms for a boron concentration corresponding to middle of life fuel condition.
This option was preferred over a forced thermal input from the DAS recordings to evaluate the neutron flux variations during the initial phase of the transient and to benefit from the inherent negative feedback of the kinetics model on the variations of the moderator temperature.

4.2. Functionally modelled 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.

4.2.1. Pressurizer relief and safety valves

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

4.2.2. 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 5.1 bar above the pressure setpoint.
The constant flow is modelled as a time-dependent junction (TDJ 441), while the pressure-dependent variable flow is supplied by two servo-valves (VLV 435, VLV 445) inserted in the explicitely modelled spray lines (V 430, V 440) connecting the cold legs to the pressurizer vapour phase, and sized to deliver the nominal spray flow at full open position.

All pressurizer heaters are constructively identical. Functionally, however, they fall into two groups : the proportional heaters (308 kW) provide the standard regulation capability needed to keep the pressurizer pressure at the desired value; the back-up heaters (1294 kW) operate on an on/off basis to counter wider pressure variations that cannot be easily corrected with the first group alone or to cope with large water insurges into the pressurizer when the level rises significantly.

4.2.3. Auxiliary feedwater system

When activated, the auxiliary feedwater is being injected directly in the steam generators in the region surrounding the separator cyclones, using a time dependent volume (TDV X36) (*) and a time dependent junction (TDJ X35) (*).
For this study, the auxiliary feed water flow recorded by the DAS is fed into the steam generators as boundary condition (TDJ X35).
Without simulation of the balance of plant, the feedwater temperature has to be imposed as boundary condition (TDV X36).

4.2.4. Steam generators relief and safety valves

Each steam generator relief valve (VLV X41) (*) is modelled as a servo-valve operated by a proportional-integral controller tied to the steam line pressure. On the other hand, all six safety valves have been combined into a single servo-valve (VLV X44) (*) with a response similar to that of the overall system.

4.2.5. Steam dump to the condenser

The complex steam dump system is being reduced to a single control valve junction (VLV 925) from the main steam header to a low pressure volume (TDV 950).
All 16 steam dump valves have been lumped together in a single servo-valve junction whose critical area was calibrated on the basis of the total steam dump capacity at nominal pressure.
The control logic considers the load rejection mode : i.e. the steam dump demand signal is a function of the mismatch between the measured average temperature of the primary loops and the reference temperature derived from the turbine load. Upon turbine and reactor trip, the reference temperature corresponds to the no-load reference temperature of 297.2°C. The valve response inertia is modelled by two time constants for respectively the fast (trip open mode) and slow (throttling mode) actuations. Figures 4.2 illustrates the steam dump control logic used in this simulation.

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* (*) X= 6,7 or 8 for respectively steam generator R, G or B

4.2.6. Charging and letdown sys. ms

For this study, the charging flow system is disconnected and the charging flow recorded by the DAS is fed into the primary system as a boundary condition (TDJ 181).

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

5. BASE CASE SIMULATION NUMERICAL RESULTS

The simulation was performed over a period of 60 s. This period covers the most important phenomena that occur after turbine and reactor trip.

For those parameters where plant data were available, the DAS recordings are presented graphically (in dotted lines) together with the corresponding calculated RELAP data (in solid lines). The DAS data are identified by their channel label.

Figures 5.1. to 5.14 illustrate for the base case (RUN 12) a good agreement between RELAP-5 simulation data and plant data for most parameters.

5.1. Primary system simulation

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

The first figu.e (Fig. 5.1) shows the neutron and thermal power.

The primary pressure (Fig. 5.2) evolves correctly with the right slope at the end of the transient. It remains anyway about 0.8 bar maximum below the DAS recorded value at the end of the transient.

The pressurizer water level curve (Fig. 5.3.) follows the recorded value correctly.
It also remains about 1.5 % below the DAS curve at the end of the transient (Fig. 5.3).

The hot leg and cold leg temperatures evolve correctly with a maximum difference with DAS recorded values of 1°C. (Fig. 5.4 and 5.5. are given for loop R). Temperature difference and average temperature evolution can be examined for loop R on Fig. 5.6 and 5.7.

All the differences mentionned above are in the range of uncertainties of instrumentation (see paragraph 2.7).

A lead of 1 to 1.5 s in the computed primary coolant temperatures was noticed in earlier calculations, and corrected for the base case.
This was caused by the absence of the Resistance Temperature Detector (RTD) bypass loop in the RELAP nodalisation.
(see parametric study, paragraph 6.1).

5.2. Secondary system simulation

The calculated secondary pressure are also in good agreement with the recorded values. One has to take into account the fact that prior to the transient, steam generator pressures were not identical (see paragraph 3.1) and at 1.1 bar to 1.6 bar below the nominal value which was adopted for the simulation. Keeping this in mind, only negligible difference (< 0.2 bar) are noticed with the DAS recorded values for the three steam generators (Fig. 5.8, 5.9 and 5.10).
Pressure oscillations at 11 s are to be related to steam flow oscillations (Fig. 5.14) and are very likely due to local pressure waves detected by the pressure gunges.

As far as the steam generator levels are concerned a rather big discrepancy appears after 30 s in steam generator G. (Fig. 5.12). The RELAP two phase flow model seems to be involved (see parametric study, chapter 6). Anyway, the average slope of level curves is satisfactory (Fig. 5.11, 5.12 and 5.13).

Steam flow modelling is in good agreement with DAS recorded values (Fig. 5.14 for loop R). Anyway, one must be aware of the big uncertainty existing on this measurement for very low steam flowrates.

6. PARAMETRIC STUDIES

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

Table 6.1 summarises the various parameter changes for nine of the most important parametric studies in chronological order.

Main objectives of the various input data modifications starting from the first transient run (RUN 03) were the following :

1. To adjust the instrumentation delays in the primary temperature simulation (RUN 04)

2. To improve the steam generator pressure simulation (RUN 05)

3. To improve the steam generator G narrow range level response (RUNS 07, 08, 09, 10)

Table 6.2 presents an overview of the assessment study data including a history list of all runs performed.
For the plots presented in this chapter, the run number is shown in the figure label D4TTXX (second heading), where XX is the run number.

6.1. Instrumentation delays adjustment (RUN 04)

As already mentionned, a lead of 1 to 1.5 s in the computed primary coolant temperatures has been noticed (FIg. 6.3. and 6.4.). This effect is caused by the absence of Resistance Temperature Detector (RTD) bypass loops in the RELAP 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. However, this total lag of 3 seconds seems to be excessive and a lag of 1 s for the hot leg and 1.5 s for the cold leg has been introduced in RUN 04.
The impact of this lag can be examined on Fig. 6.5 and 6.6.
It must be noticed that this modification doesn't simply lead to a translation since the average primary temperature strongly intervenes in the steam dump control system.

6.2. Stream generator pressure simulation improvement (RUN 05 or 12) (*)

If we examine the secondary pressure on Fig. 6.7 (for RUN 03 loop R), we notice that it increases in a correct way during the first 10 s after trip.
But as soon as the steam dump is activated, the calculated pressure evolution first presents a large overshoot, followed by an abnormal low level compared to the measured pressure.
At the same time, primary pressure decreases in an abnormal way (Fig. 6.8).
These facts seem to point out that too much steam is released to the condensor. The RELAP simulation model lumps the 16 steam dump valves together as one servo valve (SRVVLV 925 controled by CNTRLVAR 947). The capacity of this valve is supposed to vary as a linear function of the steam dump demand, which is in turn a linear function as well of the temperature difference (T_{avg} - $T_{no\ load}$).

The latter relation is a well known linear control function. On the contrary, the former is not necessarily linear.
The valve opening response to a linear growing actuation signal does not seem to behave linearly. It does not seem to be so easy to assess the actual equivalent valve characteristic hence the steam flow delivered, for each of the steam dump valves at partial opening position. We tested a parabolic curve and got an excellent agreement with the recorded data for the primary pressure and level (Fig. 5.2 and 5.3), for the secondary pressure (Fig. 5.8, 5.9 and 5.10) and for the primary temperatures (Fig. 5.4 and 5.5).

Steam dump demand signal and valve position are shown on Fig. 6.9.
A consequence of the released steam flow reduction is anyway a decrease in the steam generator narrow range levels. This effect can be noticed if you compare Fig. 6.10 (before steam dump flow reduction) with Fig. 5.12 (after steam dump flow reduction). Both figures deal with steam generator G. The levels for the two other steam generators are out of range.

(*) RUNS 05 and 12 are identical except for some plots presentation

6.3. Steam generator G narrow range level response improvement (RUNS 07, 08, 09, 10)

To understand how levels behave after reactor trip, it is instructive to examine the collapsed levels evolution in the steam generator riser and downcomer (Fig. 6.11). Both levels tend towards a value that stabilizes at too low a level. The narrow range measurement interval extends from 10.80 m to 15.35 m.

6.3.1. RUN 07

We noticed on the DAS recordings that two out of three steam generator relief valves to the atmosphere opened partially during the transient. Their opening setpoint was obviously below the nominal value (86.2 bar abs).

In a first endeavour, we manually opened each steam generator relief valves at the same position as the DAS recorded data. By dumping steam at different rates out of the three steam generators, we hoped to swell preferentialy in steam generator G riser and consequently increase the level in the downcomer. Unfortunately, although the level slightly increases (Fig. 6.12), the secondary pressure decreases also as a consequence (Fig. 6.13).

6.3.2. RUN 08

A second endeavour concentrates on the narrow range level evaluation technique. The measured evolution of the NR level shows a variation in the slope when it reaches the value of 20 %. This effect is explained by the conic shape of the upper part of the downcomer extending from 0 to 20 % of narrow range level. The simulation model doesn't take this effect into account due to the cylindrical shape of all RELAP volumes.

Anyway, we tried to simulate this behaviour by using a fictive equivalent sawn-off conic volume to replace volume 714-01 in the NR level evaluation. This equivalent volume has the same volume and the same height as volume 714-01. It can be shown that the level in volume 714-01, that was previously evaluated by the following relation :

$$\ell = H \alpha$$

where ℓ is the collapsd water level in volume 714-01
H is the height of volume 714-01 (1.888 m)
α is the liquid volume fraction in volume 714-01

must be replaced by the solution of following equation :

$$0.096\ell^3 + 2.038\ell^2 + 14.455\ell - 35.202\alpha = 0$$

The steam generator G narrow range level computed by that way can be examined on Fig. 6.14 (solid line - previous computation method has been superimposed in dotted line).
This effect doesn't anyway help us explain the level discrepancy noticed after t = 30 s.

6.3.3. RUN 09

A third endeavour led us to try to reduce the condensation rate in volumes 714-01, 714-02 and 712-01 where auxiliary feed water is injected at 20°C. With this aim, the water temperature was increased till 120°C.

Condensation rates at the end of the transient are listed in Table 6.15 for both cases. The auxiliary feed water temperature increase reduces significantly condensation rates. Nevertheless, no sensibl. impact on the main thermal hydraulic plant simulated parameters (pressures, temperatures, SG levels) could be noticed.

6.3.4. RUN 10

It appears form the preceeding runs that the excessive decrease in narrow range level comes very likely from a lack of water mass in the steam generators. The origin of this mass deficit is probably due to the two phase interphase drag correlations used by the RELAP co. It seems to be well established that the RELAP code overestimates the quantity of water carried along in a two phase flow This effect leads to a void fraction in the riser that is higher than expected. It is meaningful to examine at this point the void fraction evolution in the riser (RUN 05) on Table 6.16.

It is possible to deduce from the collapsed level evolution (Fig. 6.11) the quantity of water that is lacking. It corresponds actually to about 20 % of the narrow range in the downcomer volume 714-01, which represents about 6 tons (10 % of the nominal SG water mass). In run 10 this water mass was added in volume 728-01 by increasing its section from 14.644 m^2 to 23.044 m^2. Its height (1 m) was kept unchanged (see Fig. 6.18).
This time, the steam generator G narrow range level is in very good agreement with the DAS recorded value. (Fig. 6.17). It decreases to the right minimum value (about 15 %) and then increases to reach the plateau at 20 % as expected.

The collapsed levels in the riser and in the downcomer show a reasonable evolution as well.
If we examine Table 6.16, we notice that this water mass increase reduces significantly the void fraction in the riser (RUN 10).
An auxiliary run has been performed to prove that if we increase the water mass of the two other steam generators by the same quantity, the narrow range level remains out of range, which is coherent with the DAS recorded values.

These results tend to prove that the excessive decrease in the steam generator narrow range levels is actually due to a lack of mass caused by the correlations used in the code. One must recognise however, that the real water content in the steam generator is not known with sufficient precision.

6.4. Effect of residual heat (RUN 11)

A last parametric study was performed to analyze the effect of the increase of residual heat after reactor trip.
With this aim, the ANS79-3 fission product type was replaced by the ANS73 type. This latter one generates a higher residual heat.

Only slight changes in the main thermal-hydraulic plant simulated parameters were noticed (plus 0.23°C at the end of the transient for the average temperature).

6.5. Reference run (RUN 12)

This run is identical to the RUN 05. Some plot presentation improvements have been made and is fully analysed in chapter 5.

7. RELAP-5 RUN STATISTICS

The study was performed on a CYBER 180/825 computer with a rated performance of 1.25 MIPS (0.24 MFLOPS LINPACK).

The requested time step for the whole calculation (base case) was 0.1 s (courant DT = 0.12 s) and only 2 repeated advances, of a total of 604 attempted advances, were required.

Fig. 7.1 illustrates the CPU time versus transient time, for which a constant performance is obtained of 56.5 CPU s/transient s. The code performance evaluated as follows :

$$p = \frac{1000\ CPU}{N\ .\ DT}$$

where p is the performance,
CPU is the CPU time,
N is the number of volumes in the nodalisation,
DT is the successful number of advances,

amounts to

$$p = \frac{1000 * 3391}{229 * 602} = 24.6\ \text{ms/step/volume}$$

Fig. 7.2 illustrates the evolution of the mass error, resulting in a maximum error of 93.5 kg, yielding a maximum mass error ratio of $1.8\ 10^{-4}$.
The source of mass error is mainly located in the surge line (volume 400), in the feed water inlet volumes to the SG's (volumes X08, X = 6,7,8) and in SG G downcomer (volume 714).

8. CONCLUSIONS

8.1. The simulation of the short term transient following a turbine and reactor trip on a commercial nuclear power plant highlights the rapid changes occuring in the secondary and primary systems due to the sharp transition from the nominal power to the no-load operation.

Specific efforts were undertaken in the following areas :

8.1.1. The pressure and temperature evolution of both primary and secondary systems appeared to be very sensitive to the timing of the effective actions in relation with the power sources and sinks (turbine isolation time, reactor control rods penetration time, steam dump valves opening time). Sensitivity studies were conducted to adjust the instrumentation response times.

8.1.2. Some boundary conditions adaptations were nec sary : i.e. sensitivity studies were conducted to determine the steam dump valves capacity at partial opening positions.

8.2. The recorded transient (turbine trip due to high level in a steam generator) presents the particularity of displaying the level of the overfed steam generator within the narrow range level indication after the trip, which is a usefull indication of the water mass content in the steam generator.
To reproduce the S.G. level after the trip, it was necessary to increase artificially its initial water content by several metric tons. Our interpretation is that the void fraction in the S.G. 's riser was probably overestimated.

8.3. By improving the boundary conditions (time delays, steam dump valve characteristics) the calculated parameters related to the primary system and the steam generator (except for the narrow range level) agree very well with the recorded plant data.

8.4. The run statistics illustrate that the code ran smoothly through the transient without changing the time step and with a negligible mass error. The ratio of computer time to transient time on the used system is about 57. The code performance as usually evaluated amounts to 24.6 ms./step/volume.

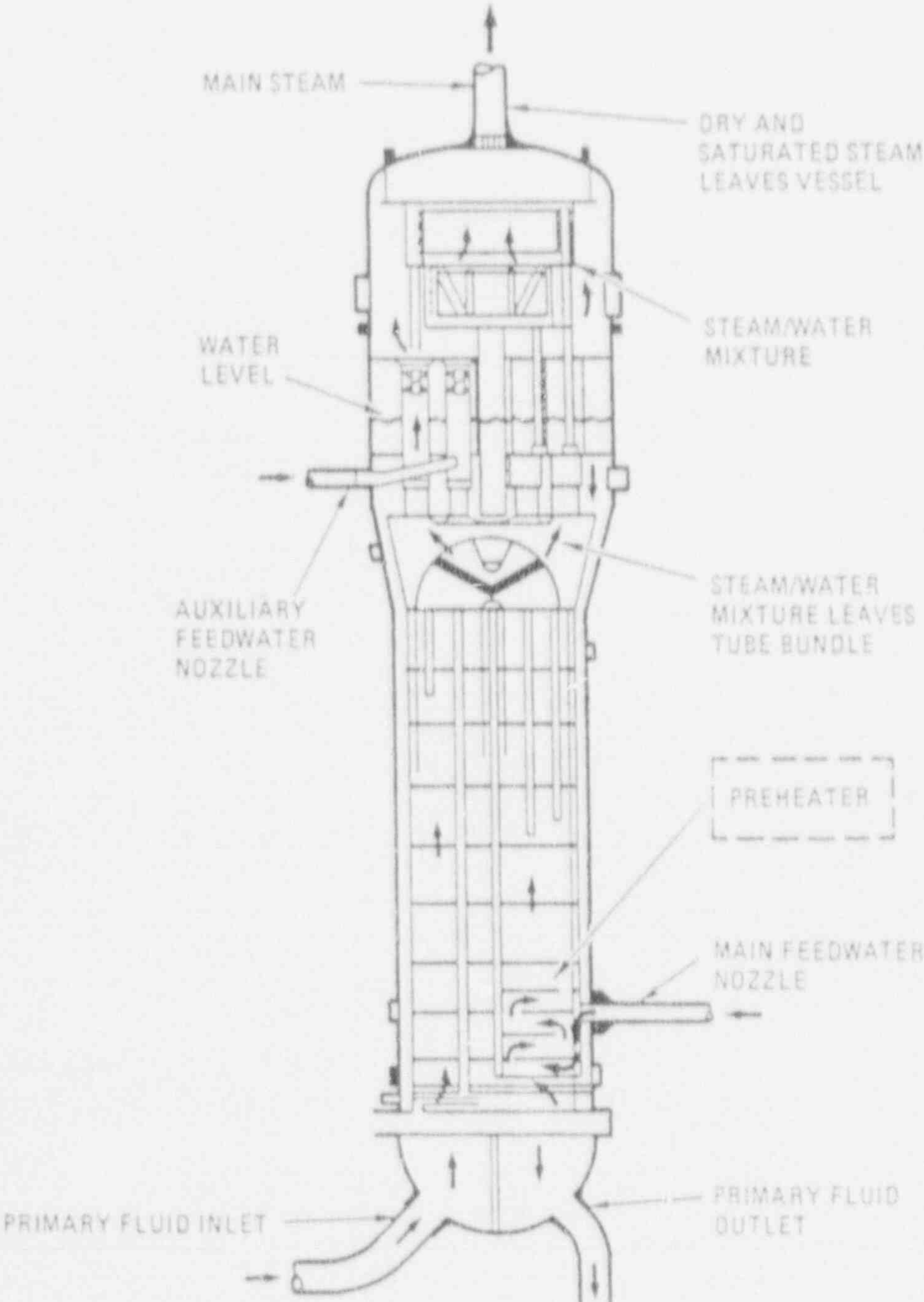

MAIN STEAM
DRY AND
SATURATED STEAM
LEAVES VESSEL
STEAM/WATER
MIXTURE
WATER
LEVEL
STEAM/WATER
MIXTURE LEAVES
TUBE BUNDLE
AUXILIARY
FEEDWATER
NOZZLE
PREHEATER
MAIN FEEDWATER
NOZZLE
PRIMARY FLUID INLET
PRIMARY FLUID
OUTLET

Fig. 3.1.

Fig. 3.1. RECORDED TURBINE DATA

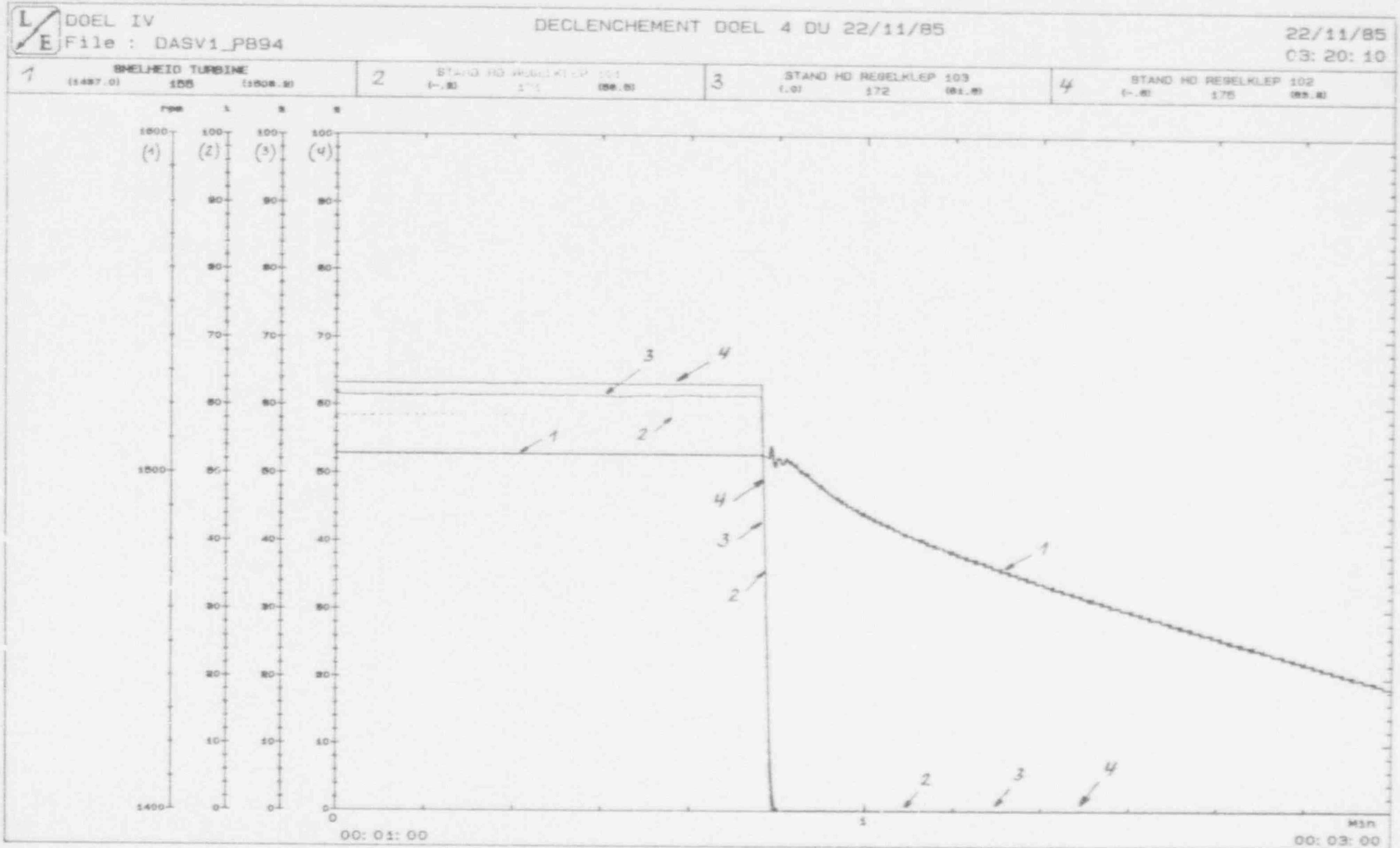


Fig. 3.2. RECORDED NUCLEAR POWER DATA

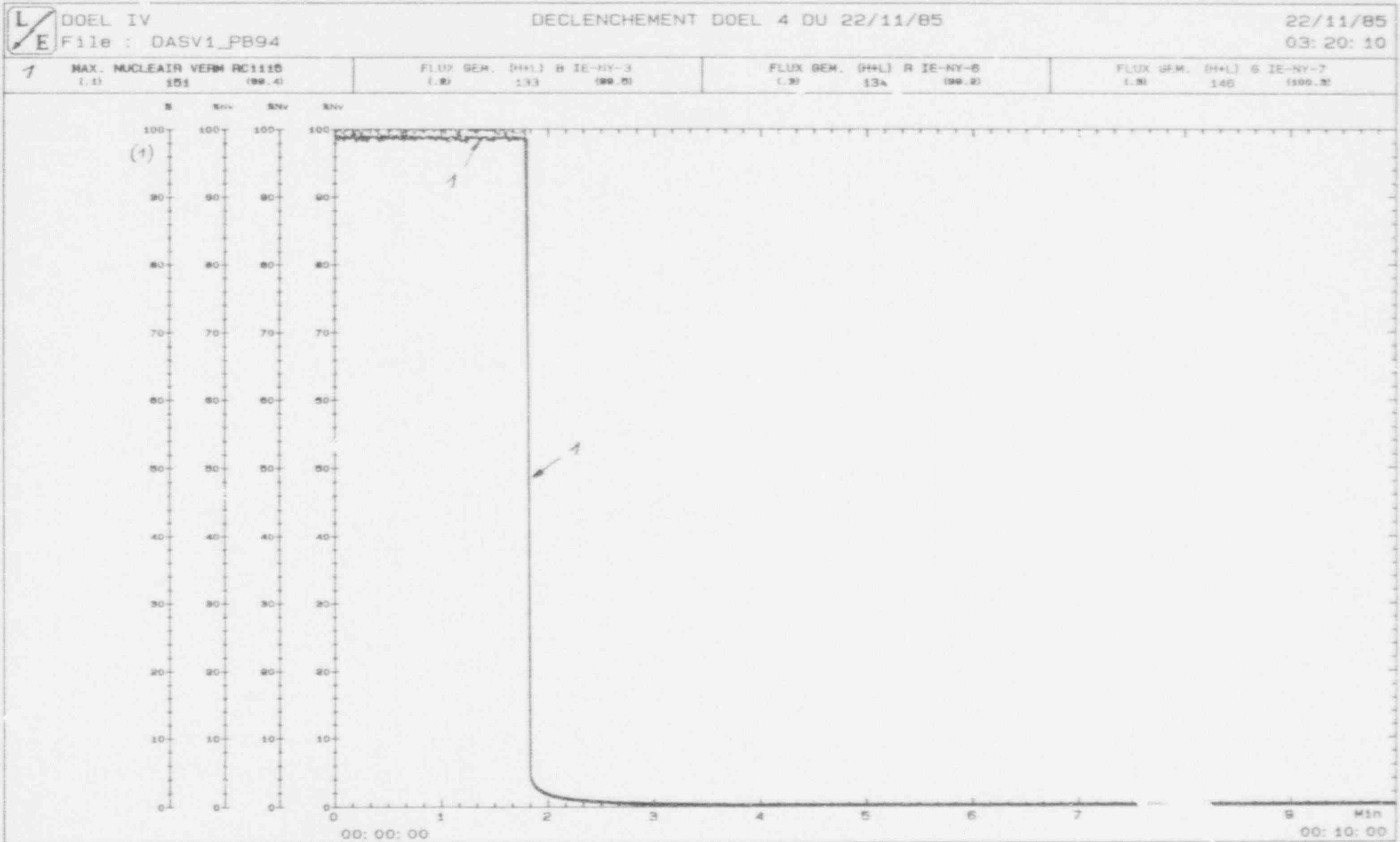


Fig. 3.3. RECORDED NARROW RANGE WATER LEVELS IN THE THREE SG'S

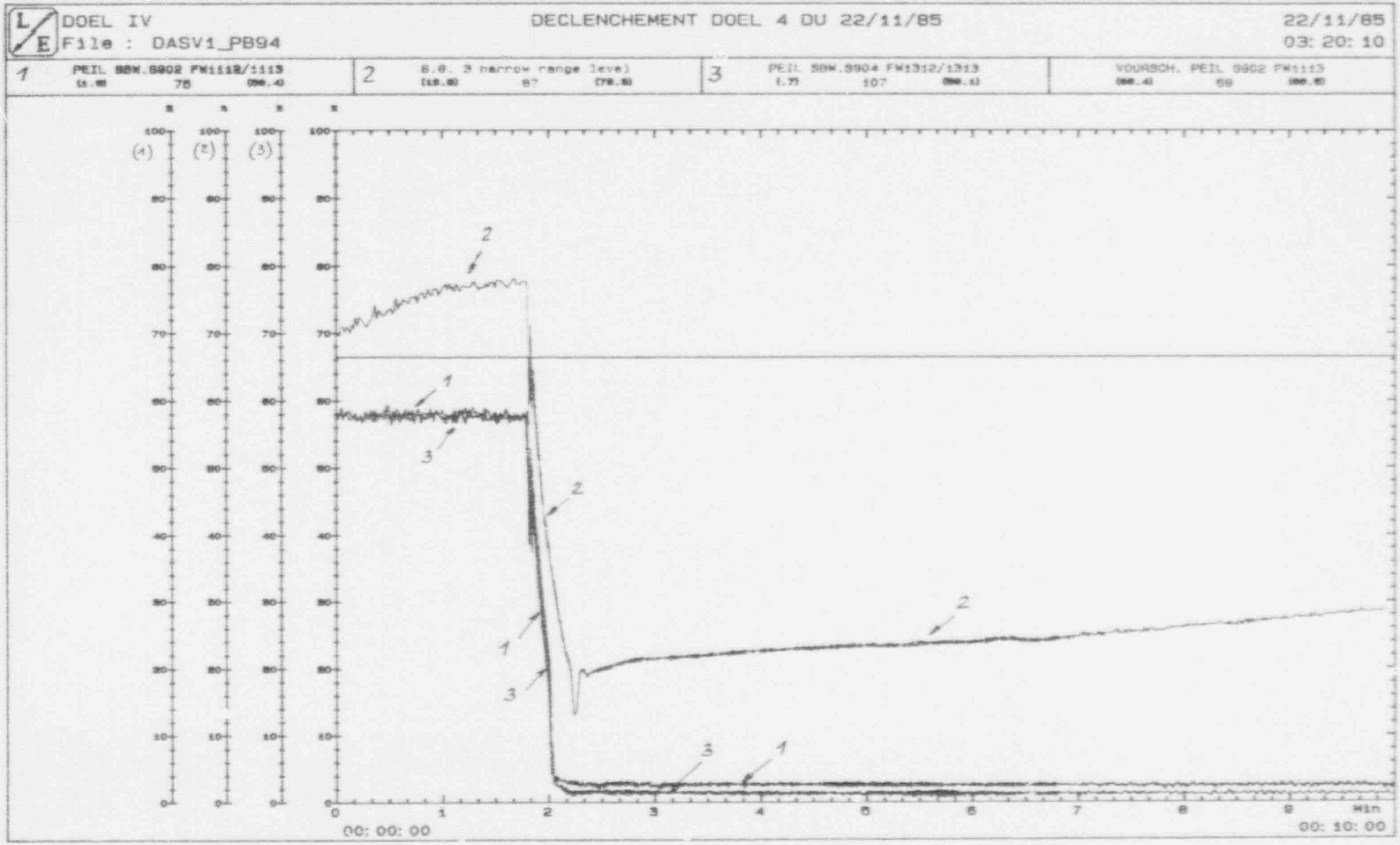


Fig. 3.4. RECORDED PRESSURIZER DATA

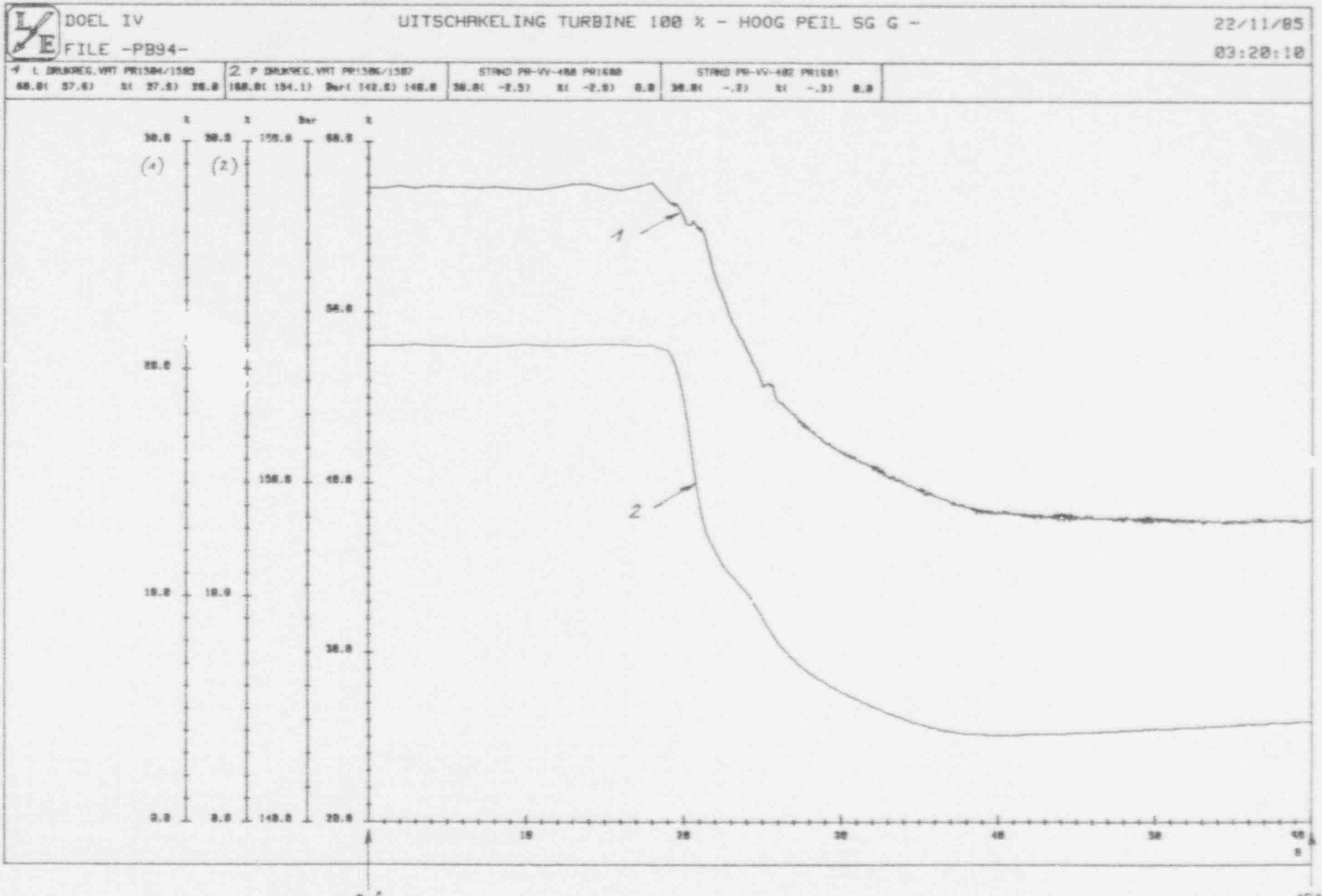


Fig. 3.5. RECORDED REACTOR COOLANT TEMPERATURES FOR LOOP R

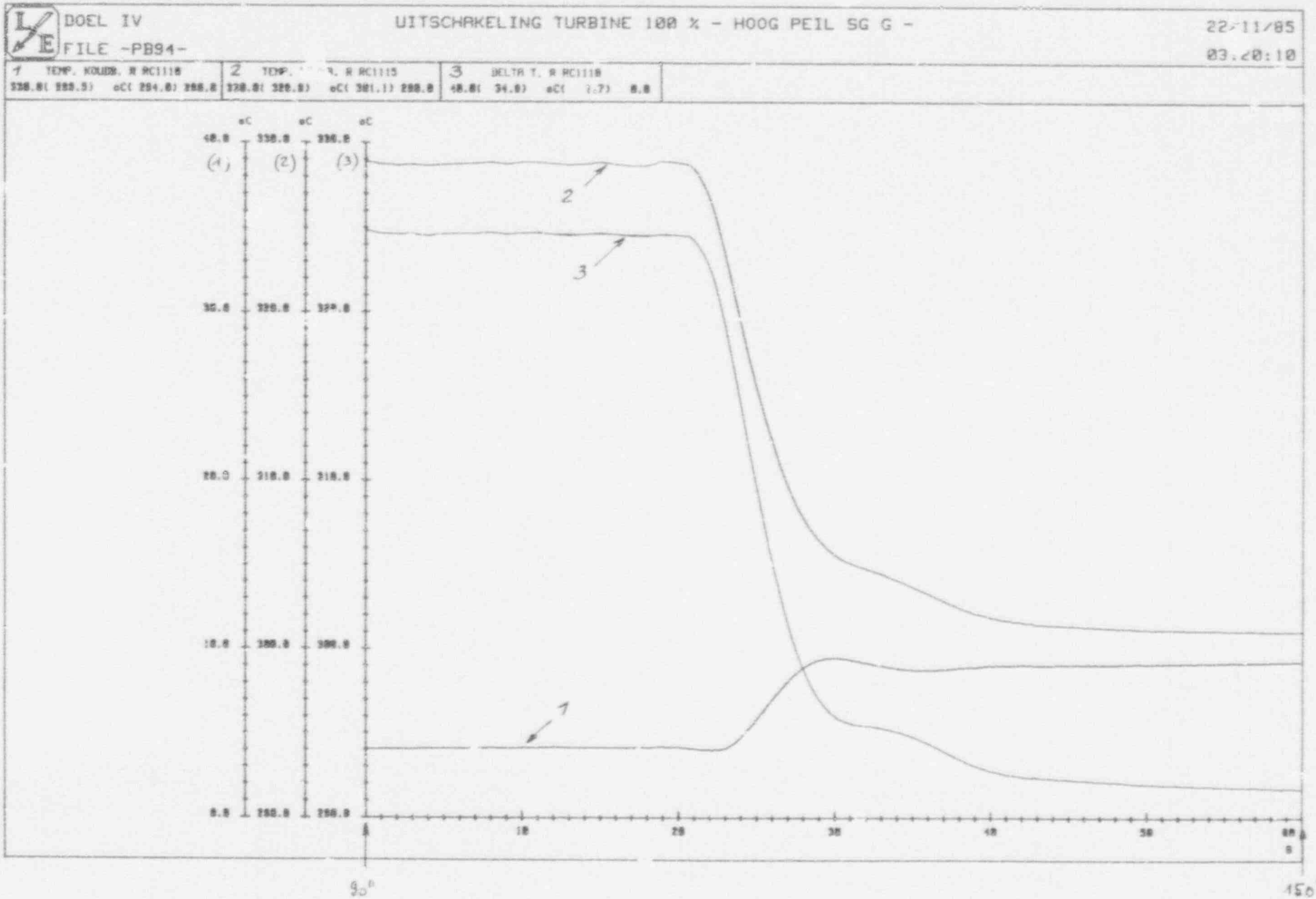


Fig. 3.6. RECORDED SG AND HEADER PRESSURES

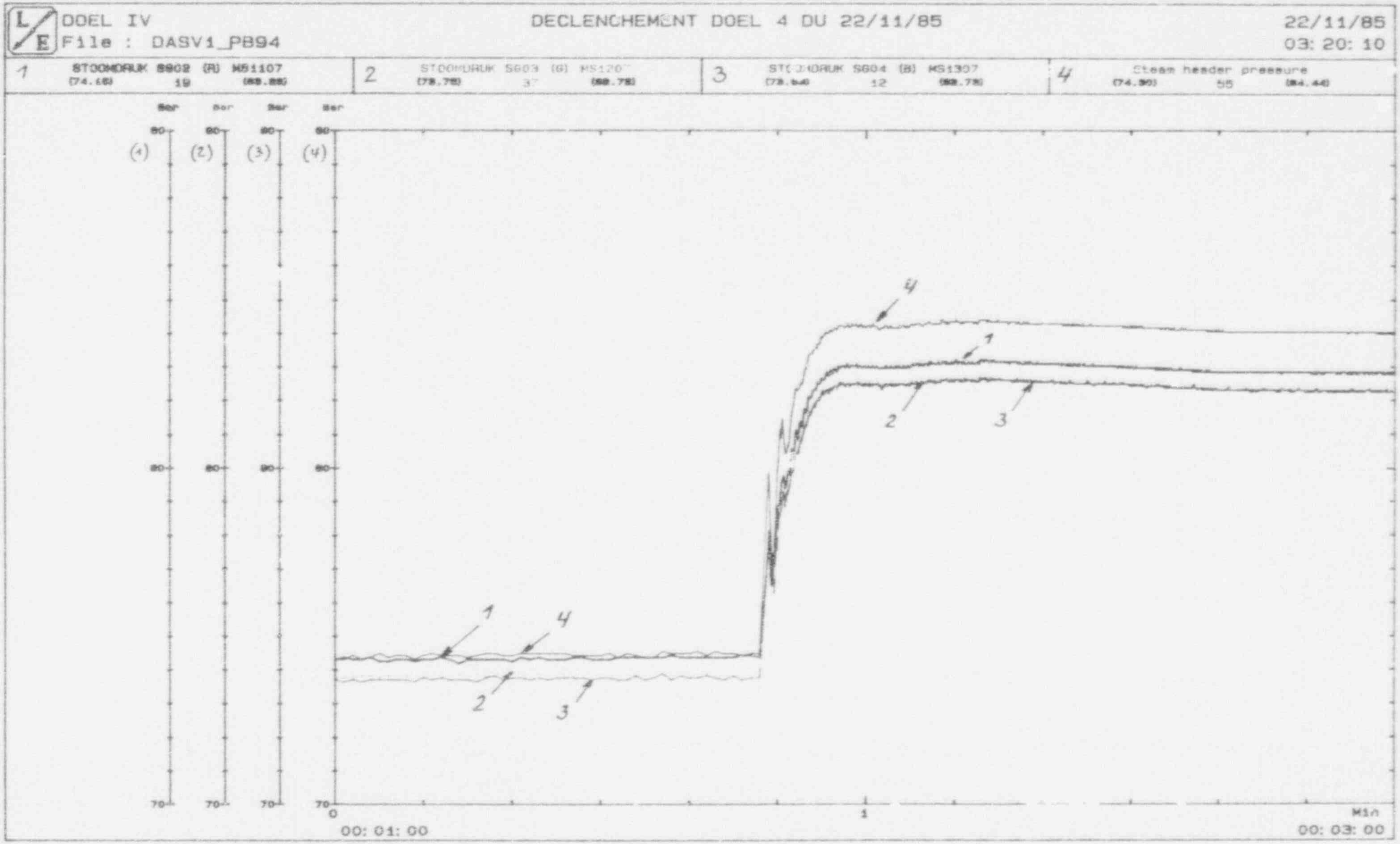


Fig. 3.7 RECORDED STEAM DUMP REFERENCE DATA

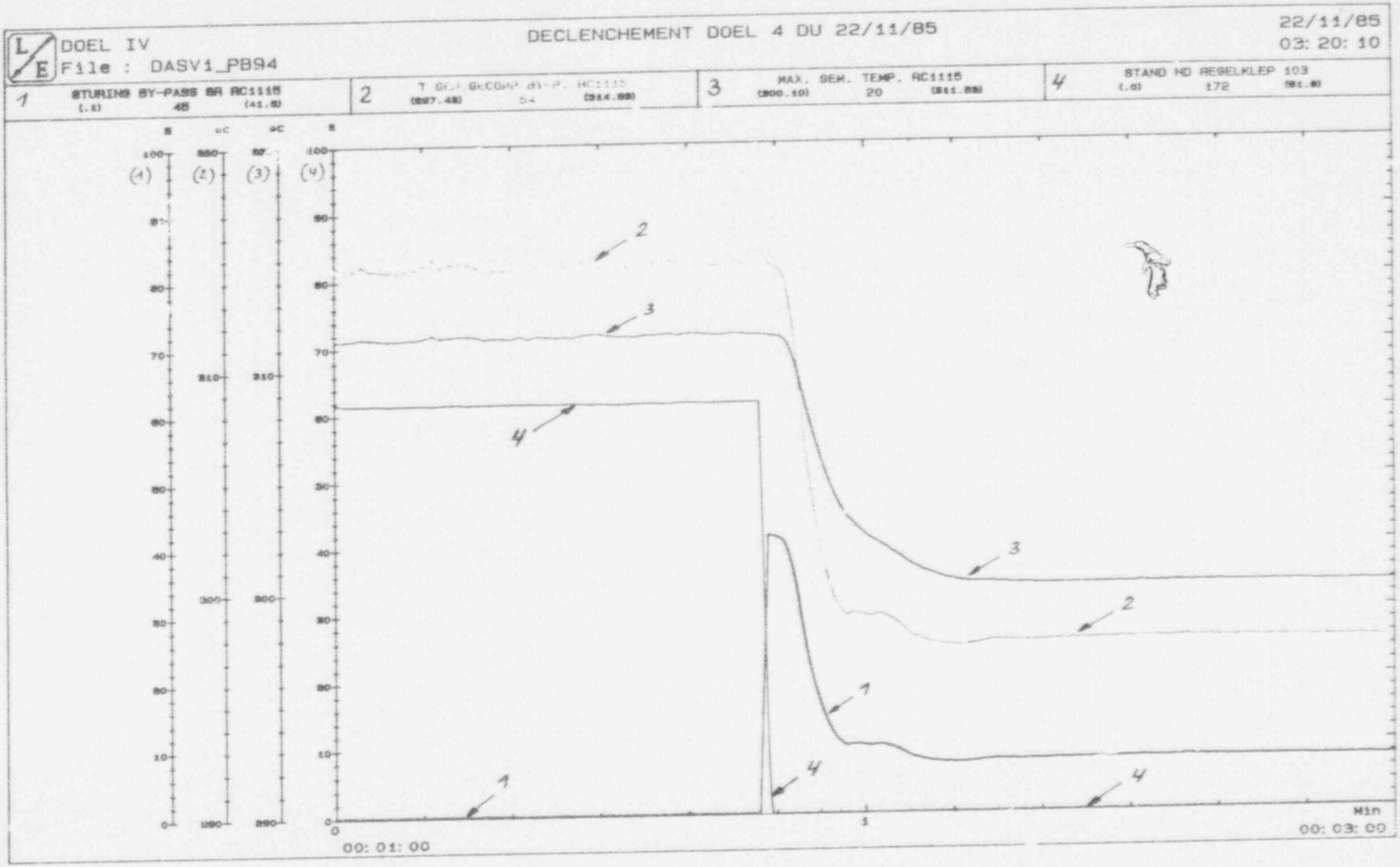
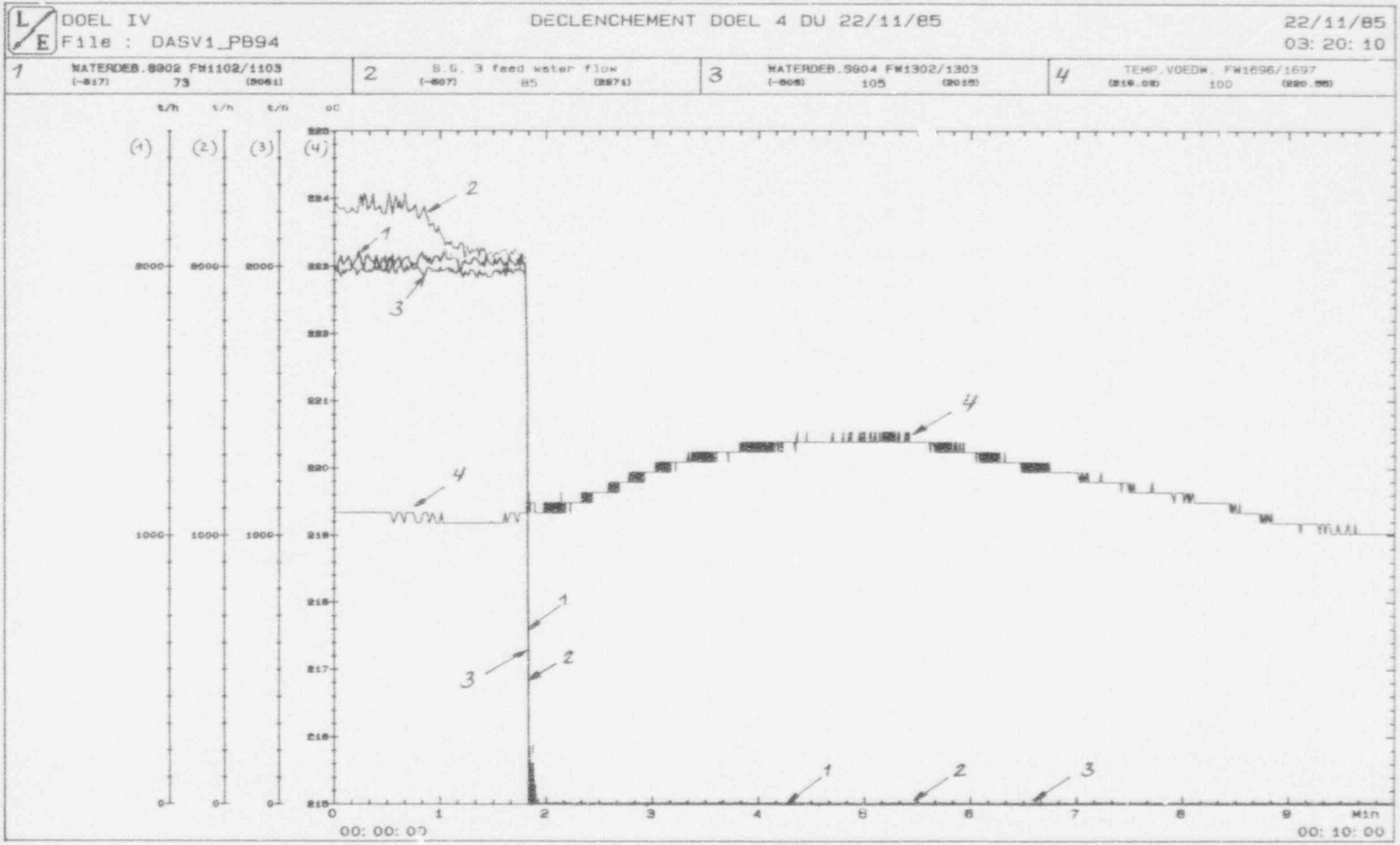


Fig. 3.8 RECORDED FEEDWATER DATA



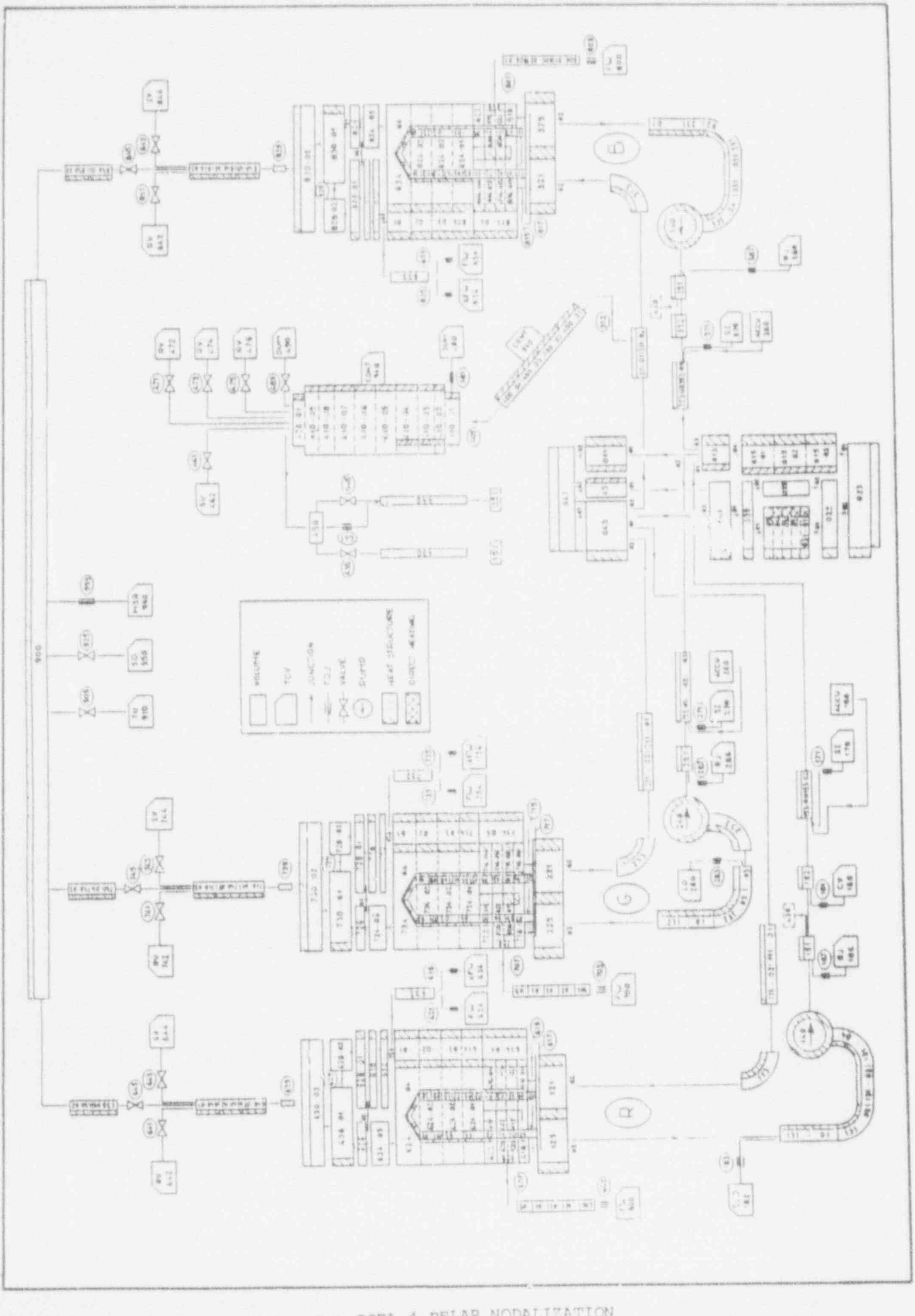
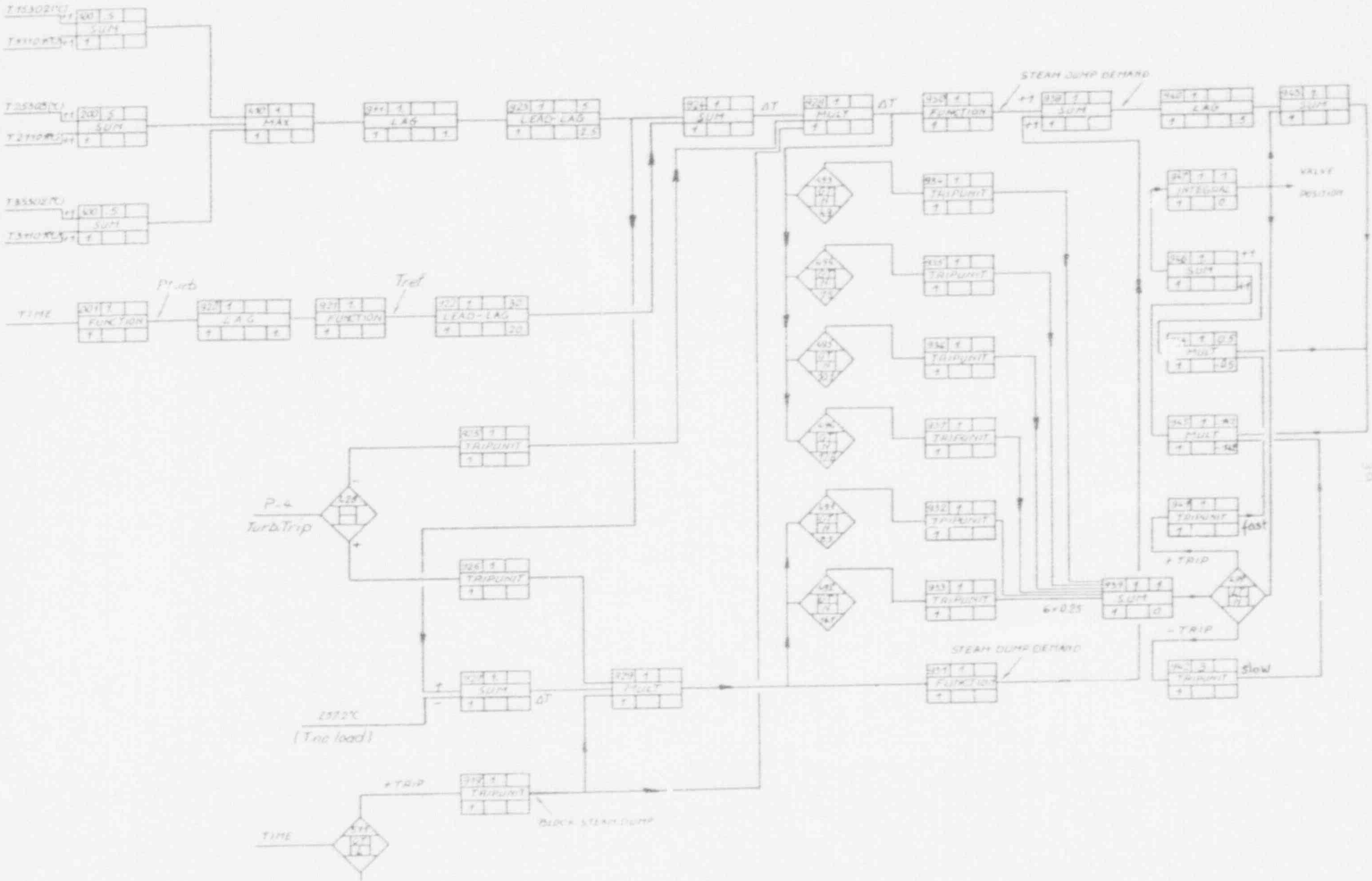
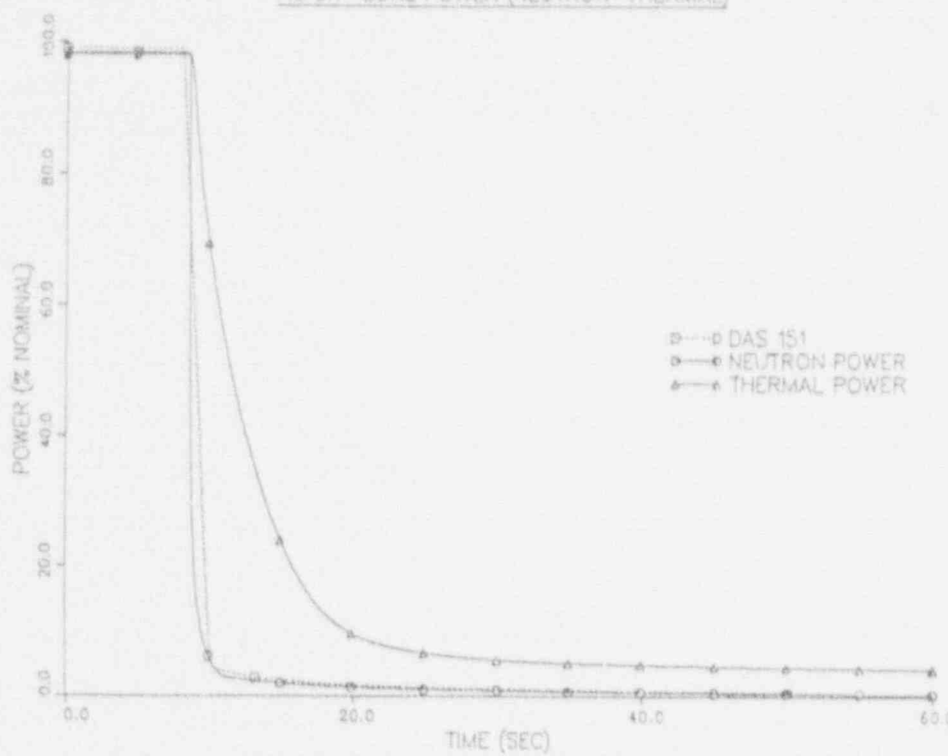


Fig. 4.1 DOEL 4 RELAP NODALIZATION

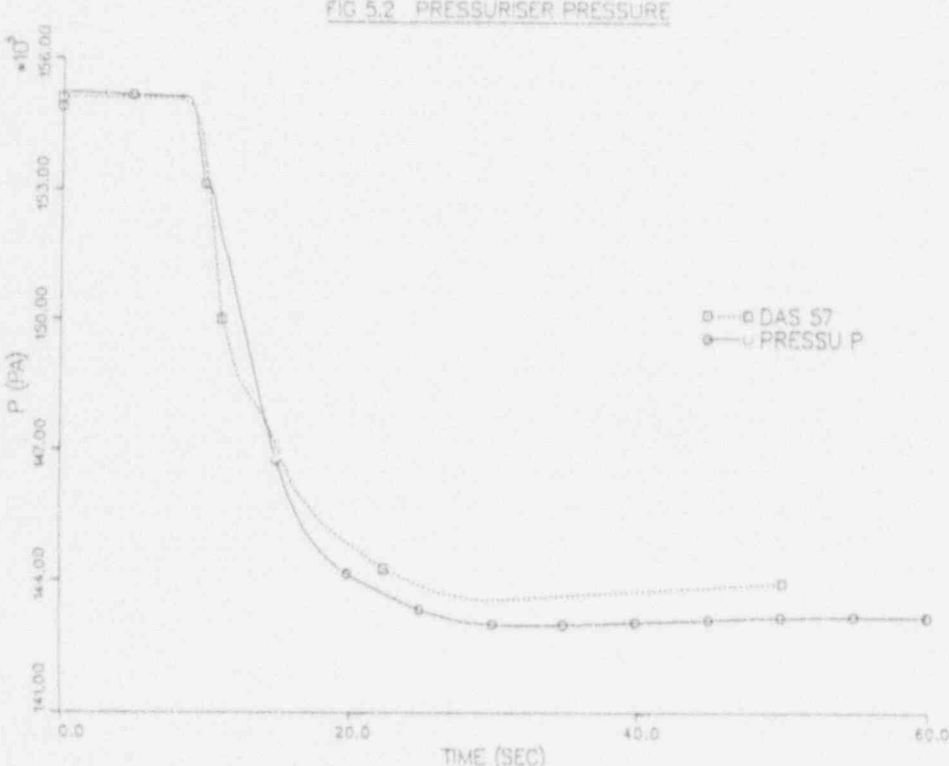
Fig. 4.2. SIMULATION OF THE STEAM DUMP CONTROL SYSTEM



DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP B (29/11/89)
FIG 5.1 CORE POWER (NEUTRON–THERMAL)



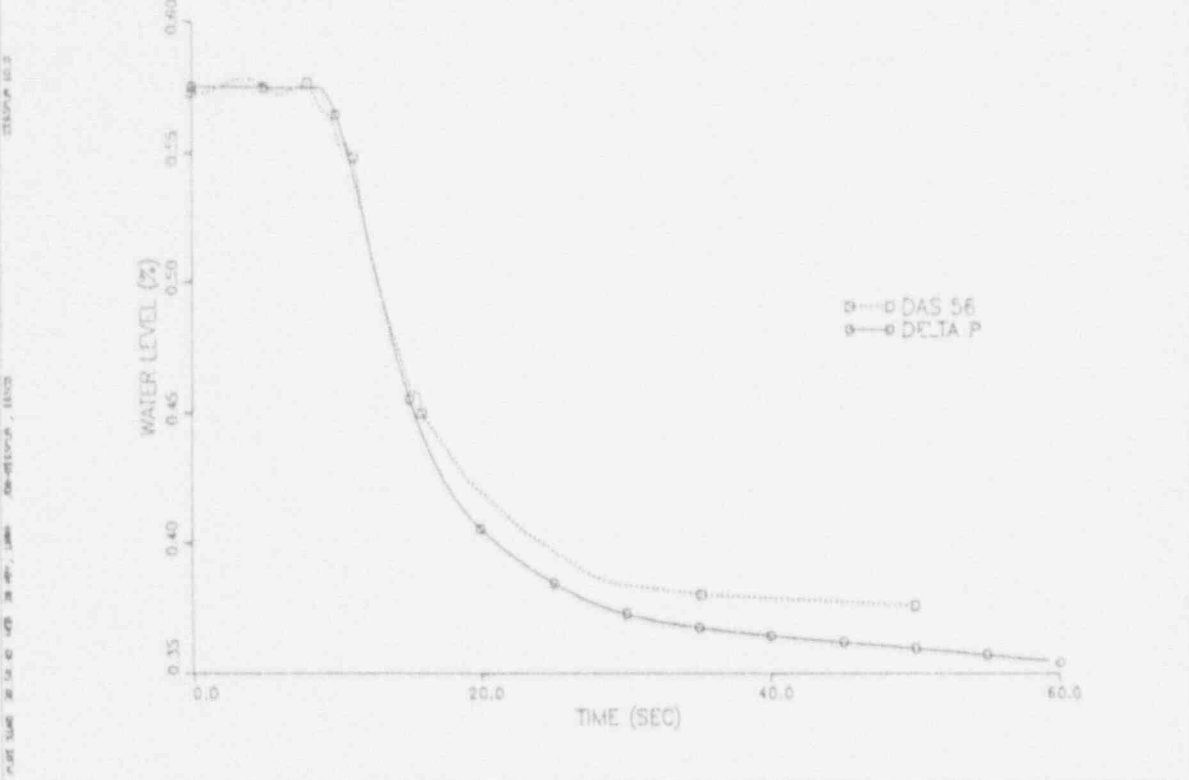
DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP B (29/11/89)
FIG 5.2 PRESSURISER PRESSURE



DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

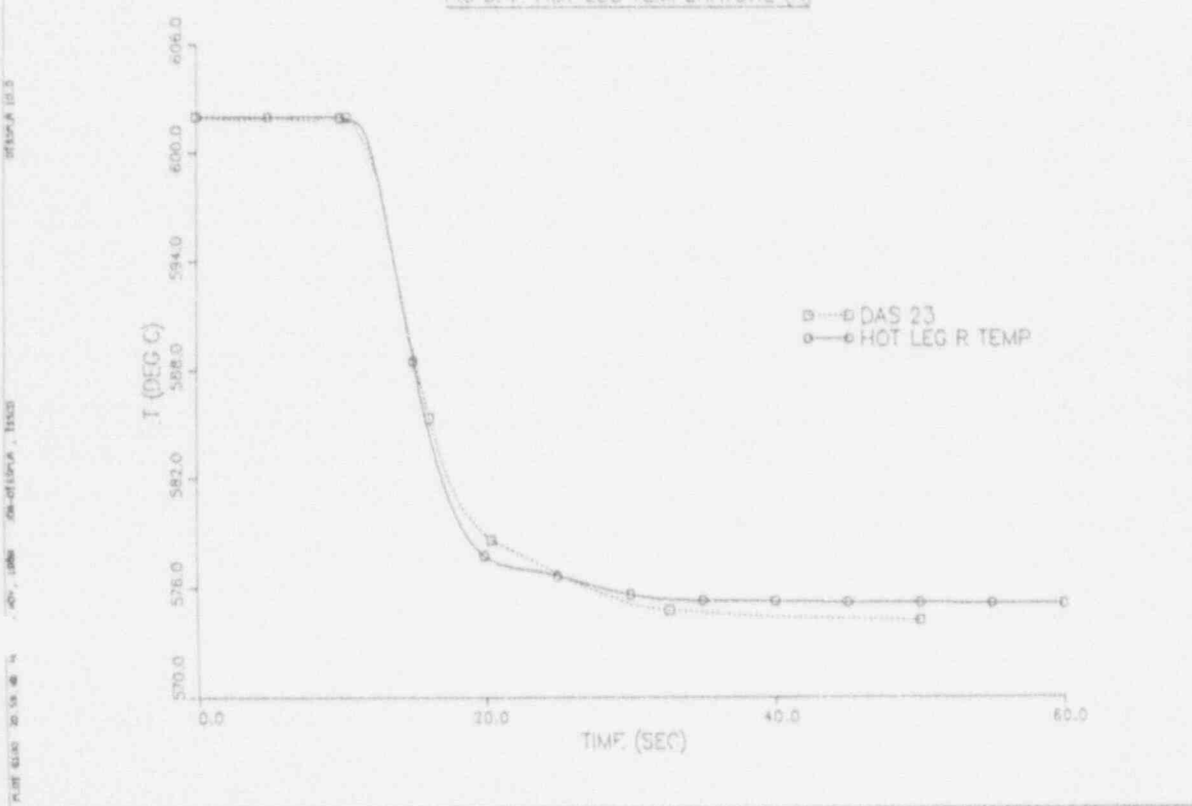
FIG 5.3 PRESSURISER WATER LEVEL

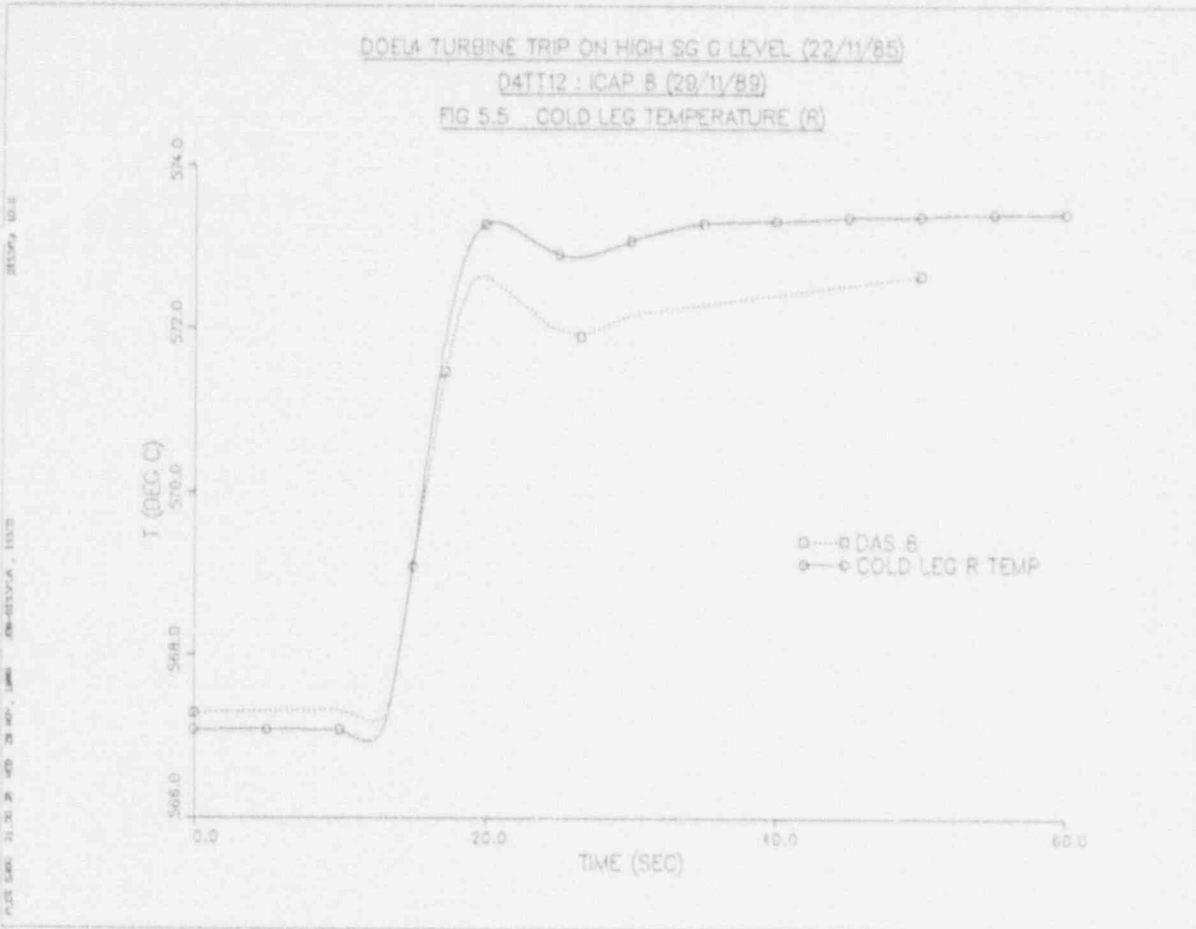


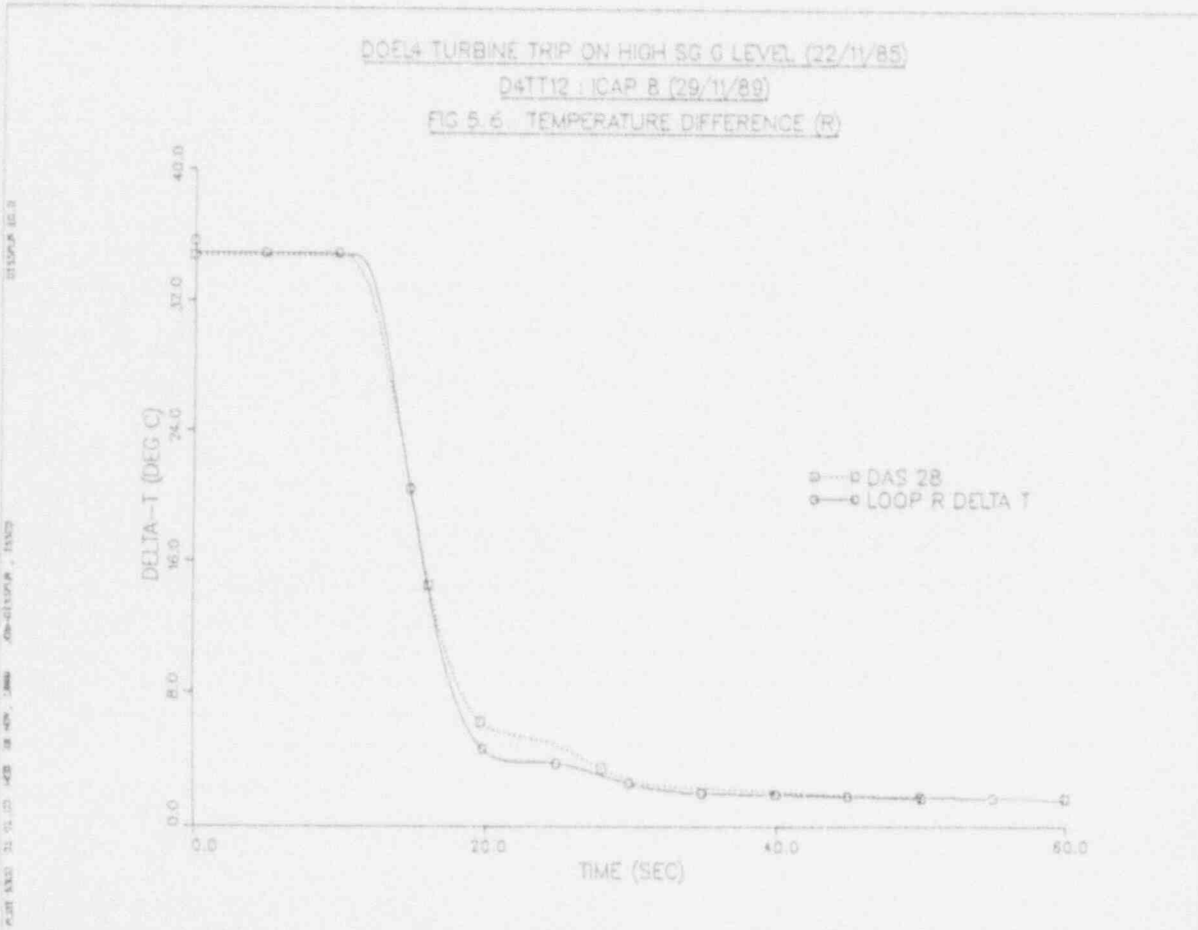
DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

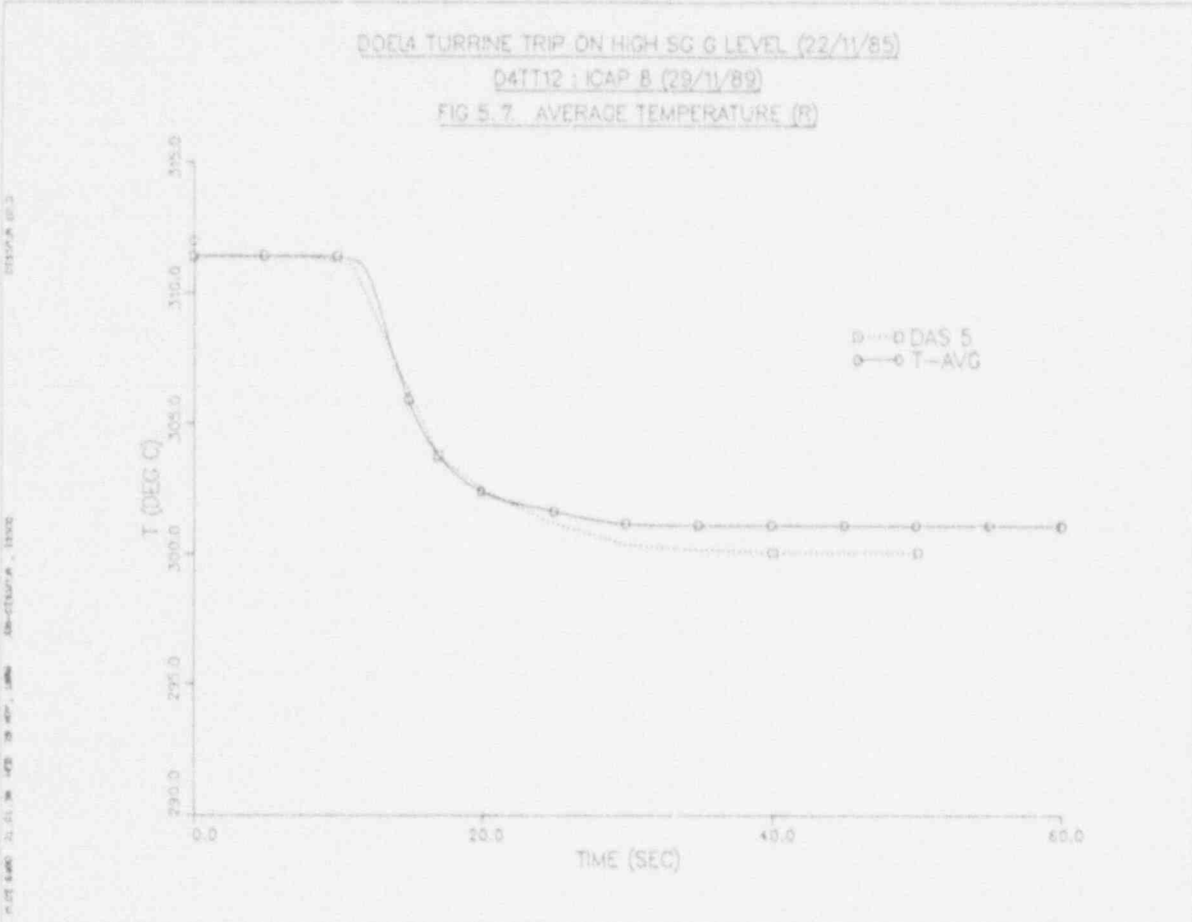
D4TT12 : ICAP 8 (29/11/89)

FIG 5.4 HOT LEG TEMPERATURE (R)

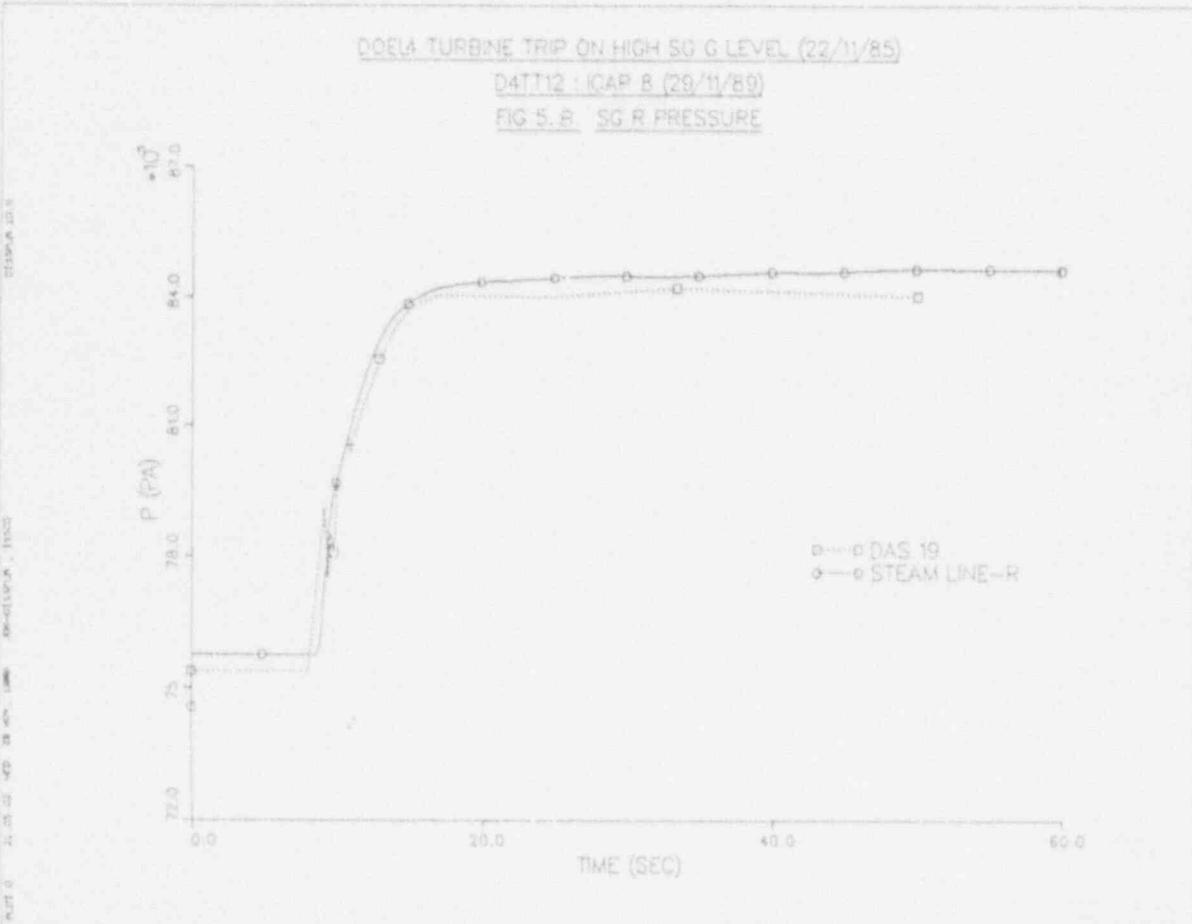



DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP 8 (29/11/89)
FIG 5.5 COLD LEG TEMPERATURE (R)
T (DEG C)
574.0
572.0
570.0
568.0
566.0
0.0
20.0
40.0
60.0
TIME (SEC)
DAS 6
COLD LEG R TEMP


DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP 8 (29/11/89)
FIG 5.6 TEMPERATURE DIFFERENCE (R)
DELTA-T (DEG C)
40.0
32.0
24.0
16.0
8.0
0.0
0.0
20.0
40.0
60.0
TIME (SEC)
DAS 28
LOOP R DELTA T



DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP 8 (29/11/89)
FIG 5.7. AVERAGE TEMPERATURE (R)
T (DEG C)
TIME (SEC)
DAS 5
T-AVG

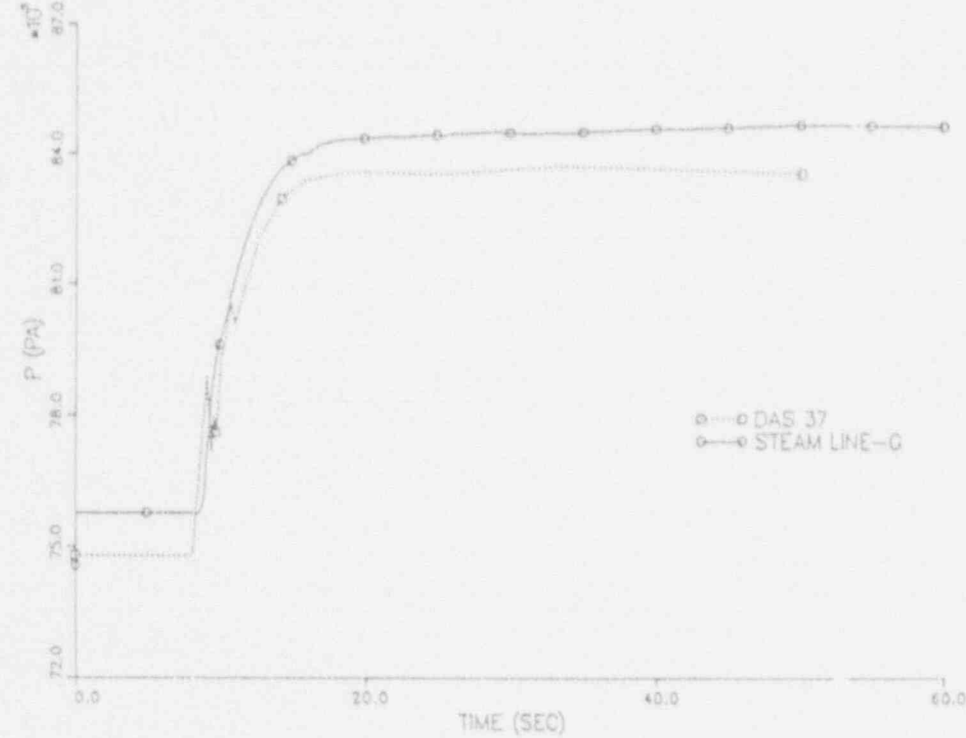


DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP 8 (29/11/89)
FIG 5.8. SG R PRESSURE
P (PA)
TIME (SEC)
DAS 19
STEAM LINE-R

DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

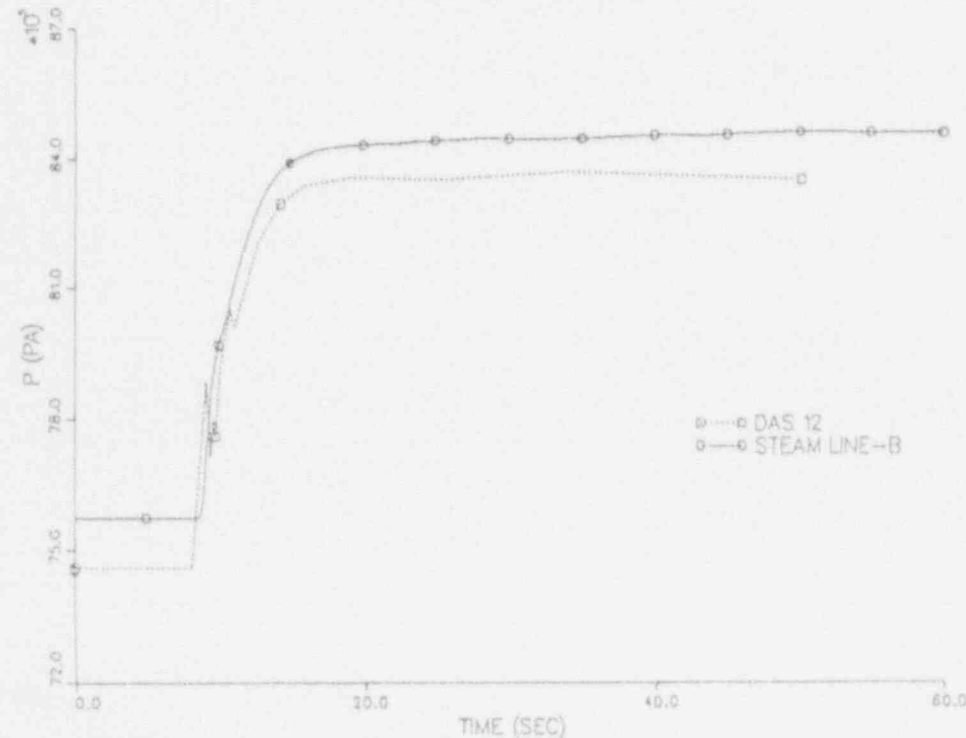
FIG 5.9 SG G PRESSURE

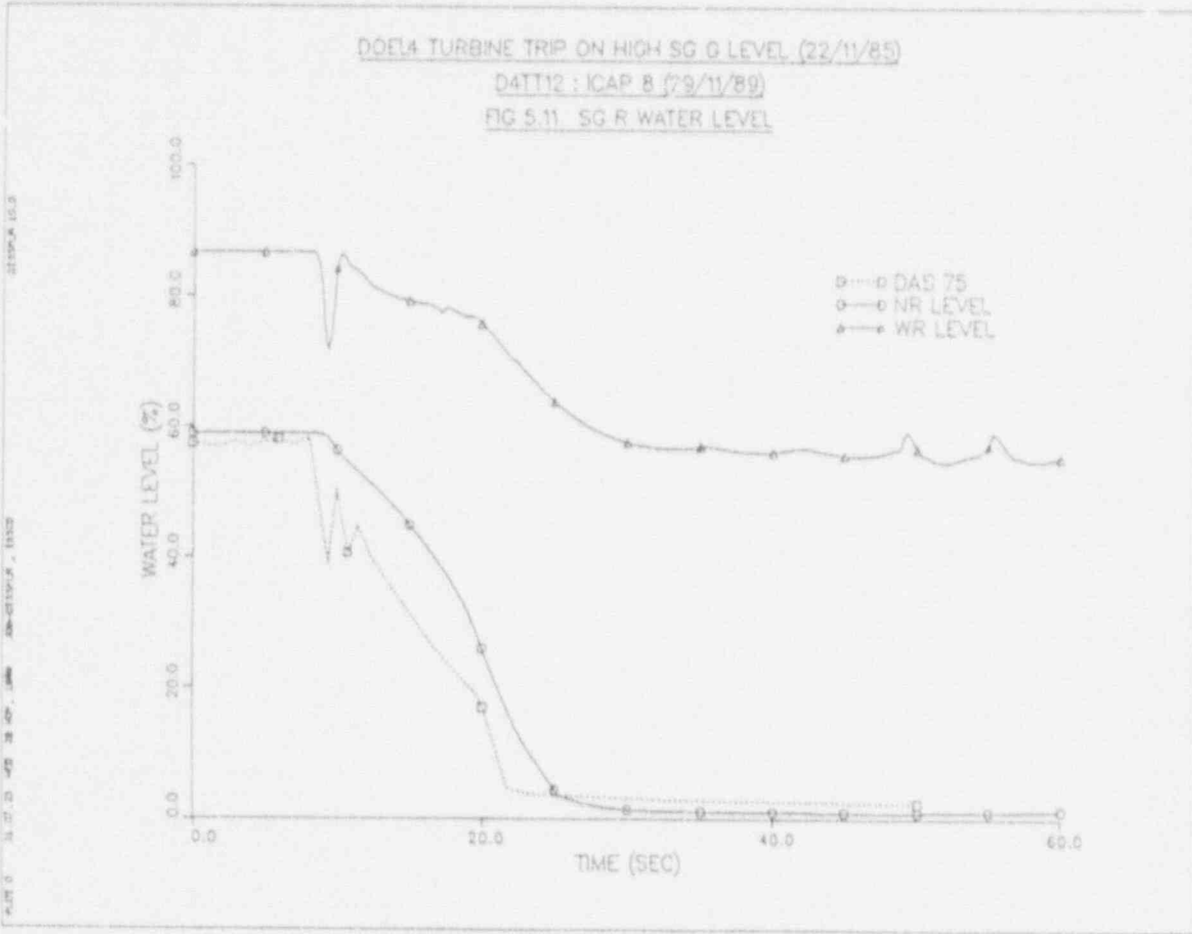


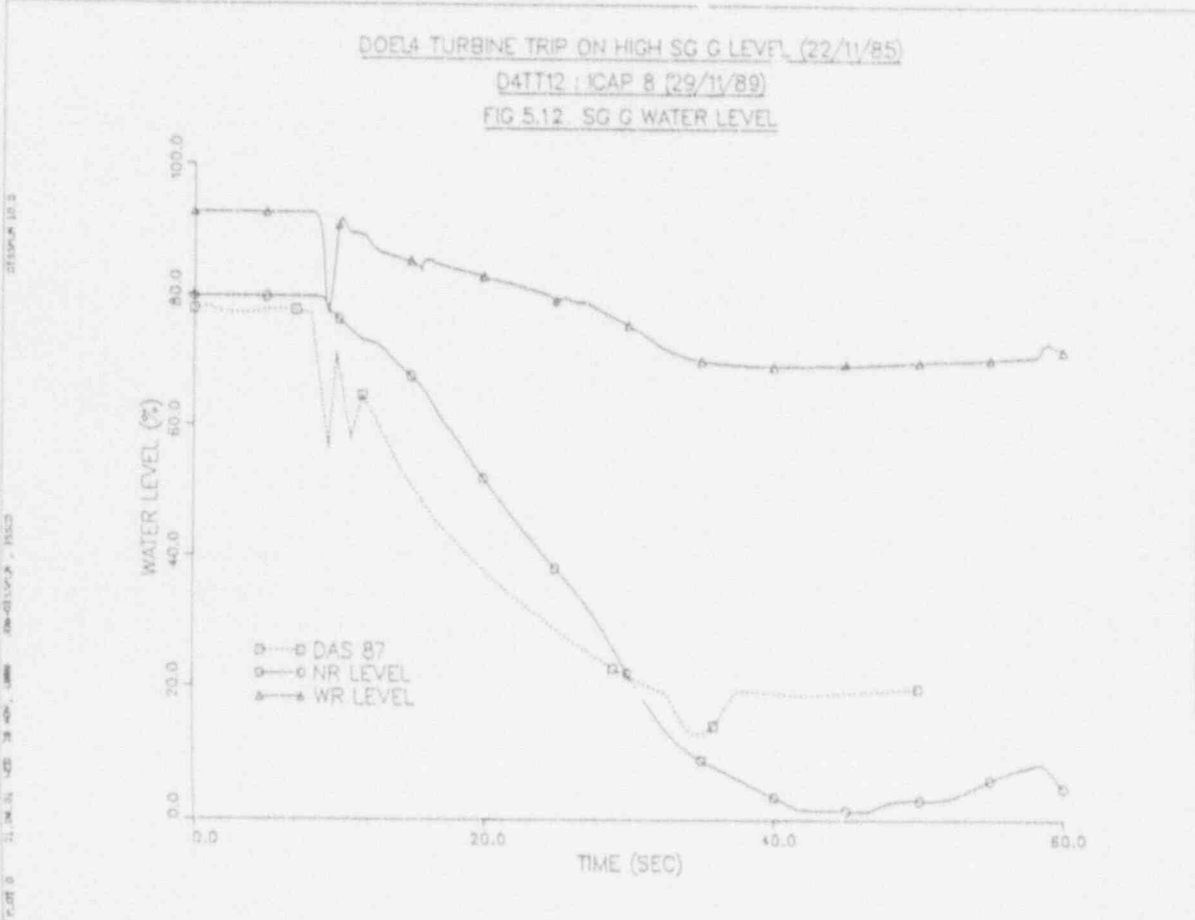
DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

FIG 5.10 SG B PRESSURE



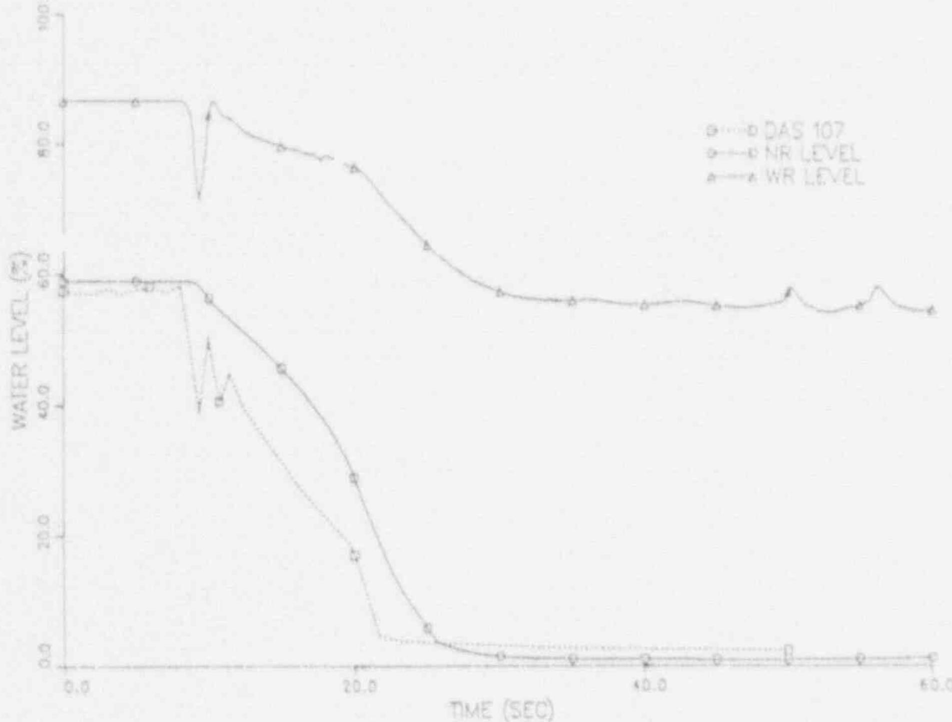

DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP 8 (29/11/89)
FIG 5.11 SG R WATER LEVEL
DAS 75
NR LEVEL
WR LEVEL
WATER LEVEL (%)
TIME (SEC)
0.0
20.0
40.0
60.0
80.0
100.0


DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP 8 (29/11/89)
FIG 5.12 SG G WATER LEVEL
DAS 87
NR LEVEL
WR LEVEL
WATER LEVEL (%)
TIME (SEC)
0.0
20.0
40.0
60.0
80.0
100.0

DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

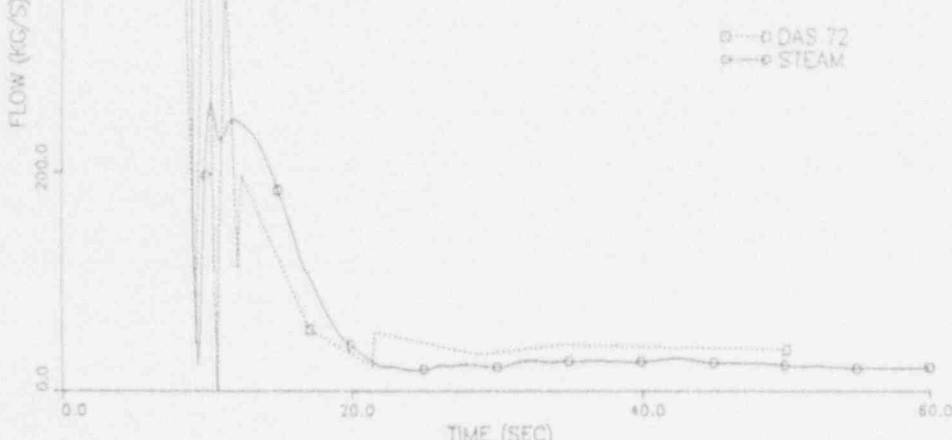
FIG 5.13 SG B WATER LEVEL



DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

FIG 5.14 SG R STEAM FLOW



RUN NUMBER	03	04	05	07	08	09	10	11	12
Delay on hot leg temperature	0 s	1 s	1 s	1 s	1 s	1 s	1 s	1 s	1 s
Delay on cold leg temperature	0 s	1,5 s	1.5 s	1.5 s	1.5 s	1.5 s	1.5 s	1.5 s	1.5 s
Steam dump curve	standard	standard	modified	modified	modified	modified	modifed	modified	modified
SG relief valves pos.	closed	closed	closed	part. open	closed	closed	closed	closed	closed
NR lev-l computation	standard	standard	standard	standard	modified	standard	standard	standard	standard
Aux. feed water temperature	20°C	20°C	20°C	20°C	20°C	120°C	20°C	20°C	20°C
Vol. 728.01 volume	14.644 m^3	14.644 m^3	14.644 m^3	14.644 m^3	14.644 m^3	14.644 m^3	23.044 m^3	23.044 m^3	14.644 m^3
Fission product type	ANS79-3	ANS79-3	ANS79-3	ANS79-3	ANS79-3	ANS79-3	ANS79-3	ANS23	ANS79-3
Remarks		-	-	-		-		-	plots presentations improvements

Table 6.1

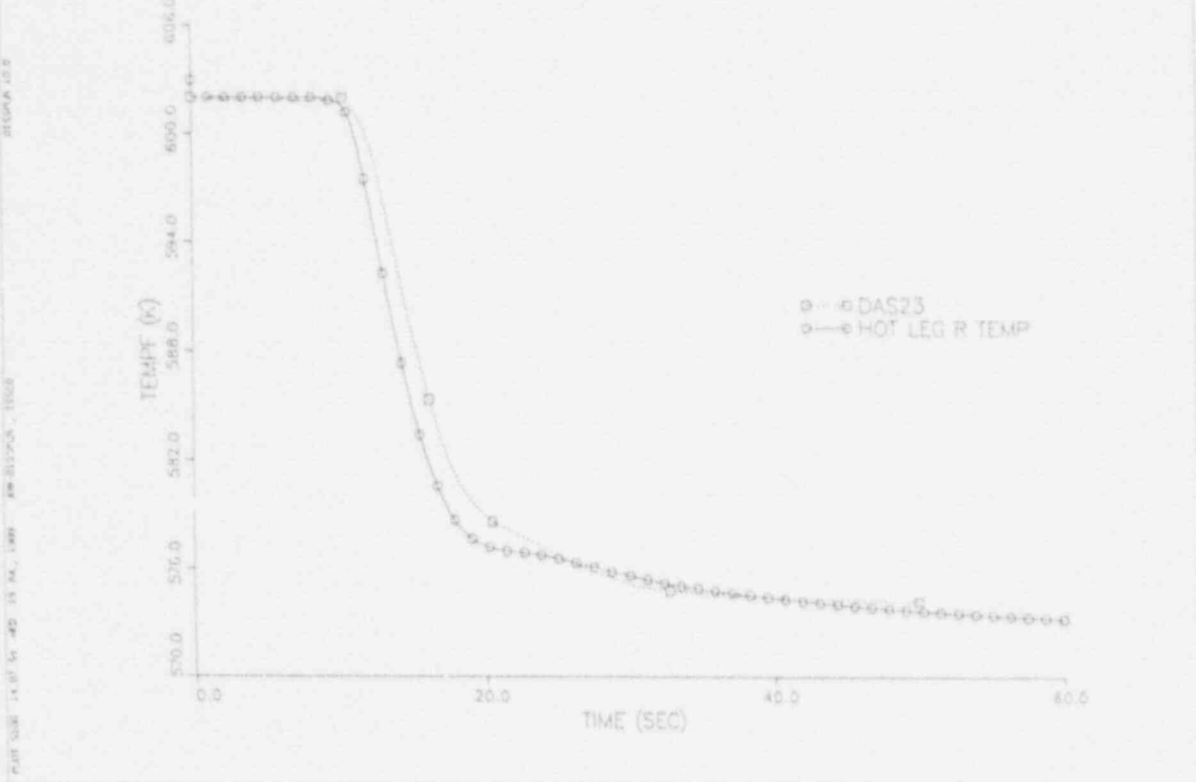
```
* ICAP 8  : TURBINE TRIP ON HIGH STEAM GENERATOR LEVEL
*
* PLANT    : DOEL 4  (BELGIUM)
*
* DATE OF TRANSIENT : 22/11/85
*
* DESCRIPTION OF TRANSIENT : TURBINE TRIP FOLLOWED BY REACTOR TRIP ON
*                            HIGH SG G LEVEL
*
* PROJECT NUMBER : 180048
*
* REFERENCE DOCUMENTS :
*
*            (1) BANQUE DE DONNEES DE L'UNITE DOEL 4 POUR LA SIMULATION
*                RELAP5/MOD2
*                TRACTEBEL,REF.71212/423/NT/0076/REO,10/12/88
*
*            (2) DAS RECORDINGS,LABORELEC
*
*********************************************************************
*
* RUN DATA : RUN 12
*            DATE       : 29/11/89
*            FILENAMES : INPUT FILE    : D4TT12
*                        RSTIN FILE    : D4TTR01
*                        RSTPLT FILE   : D4TTR12
*                        CALCOMP FILE  : D4TTC12
*                        OUTPUT FILE   : D4TTO12
*                        DAYFILE       : DAYTT12
*                        JCL FILE      : JOBTT
*
*********************************************************************
*
* R U N S  H I S T O R Y  L I S T
* *******************************
*
*
* REFERENCE MASTER DECK : D4MM (DEC. 88)
*
* RUN 01 : STEADY-STATE AT FOLLOWING DAS CONDITIONS :
*                 . NUCLEAR POWER AT 98% NOM ( 2928 MW )
*                 . AVERAGE TEMPERATURE AT 311.4 DEG.C
*                 . SG G LEVEL AT 78%
*          TO REACH THESE CONDITIONS,FOLLOWING CHANGES WERE MADE :
*                 . REFERENCE POWER SET AT 2928 MW IN REACTOR KINETICS MODEL
*                 . CNTRLVAR 901 SETPOINT SET AT 311.4 DEG.C
*                 . CNTRLVAR 723 SETPOINT SET AT 0.78
*                 . INHIBITION OF P14 BY INCREASING SG G LEVEL SETPOINT TO 1.
*                      (TRIP 406)
*
* RUN 02/03 : FIRST TRANSIENT RUNS WITHOUT ANY TUNING
*                 . STARTING INPUT DECK : RUN 01
*                 . PRESSURIZER PRESSURE AND LEVEL REGULATORS REMOVED
*                 . MANUAL HIGH SG G LEVEL AT 8 S
*                 . TURBINE TRIP ON P14 (TRIP 428)
*
* RUN 04 : HOT AND COLD LEGS TEMPERATURE TUNING
*                 . STARTING INPUT DECK : RUN 03
*                 . A 1.5 S DELAY WAS ADDED TO THE COLD LEG TEMPERAURE
*                 . A 1. S     "       "      "    HOT         "
*                 . STEAM DUMP VALVE POSITION PLOTS WERE ADDED
*
* RUN 05 : STEAM GENERATOR PRESSURE TUNING
*                 . STARTING INPUT DECK : RUN 04
*                 . THE STEAM DUMP FLOW CURVE WAS MODIFIED (FUNCTION 931)
*
*
* RUN 06/07 : STEAM GENERATOR NR LEVEL TUNING
*                 . STARTING INPUT DECK : RUN 05
*                 . MANUAL OPENING OF STEAM GENERATOR RELIEF VALVES
*
* RUN 08 : STEAM GENERATOR NR LEVEL TUNING
*                 . STARTING INPUT DECK : RUN 05
*                 . NR STEAM GENERATOR LEVEL WAS MODIFIED TO TAKE INTO ACCOUNT
*                      THE CONIC SHAPE OF THE DOWNCOMER UPPER PART
*
* RUN 09 : STEAM GENERATOR NR LEVEL TUNING
*                 . STARTING INPUT DECK : RUN 08
*                 . AUXILIARY FEED WATER TEMPERATURE SET TO 120 DEG.C TO
*                      REDUCE CONDENSATION IN VOLUME 712-01
*
* RUN 10 : STEAM GENERATOR NR LEVEL TUNING
*                 . STARTING INPUT DECK : RUN 08
*                 . MODIFICATION OF PLOTS TITLES AND SPECIFICATIONS
*                 . INCREASE OF VOLUME 728-01 SECTION IN ORDER TO INCREASE
*                      STEAM GENERATOR G WATER MASS
*
* RUN 11 : RESIDUAL HEAT TUNING
*                 . STARTING INPUT DECK : RUN 10
*                 . MODIFICATION OF FISSION PRODUCT TYPE : ANS79-3...ANS73
*                      (REAC BIAS IS FREEZED AFTER 6 S)
*
* RUN 12 : FINAL RUN (REFERENCE RUN)
*                 . STARTING INPUT DECK : RUN 05
*                 . FINAL MODIFICATION OF PLOTS TITLES AND SPECIFICATIONS
*
*********************************************************************
```

Table 6.2

DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)

D4TT03 : ICAP 8 (19/07/89)

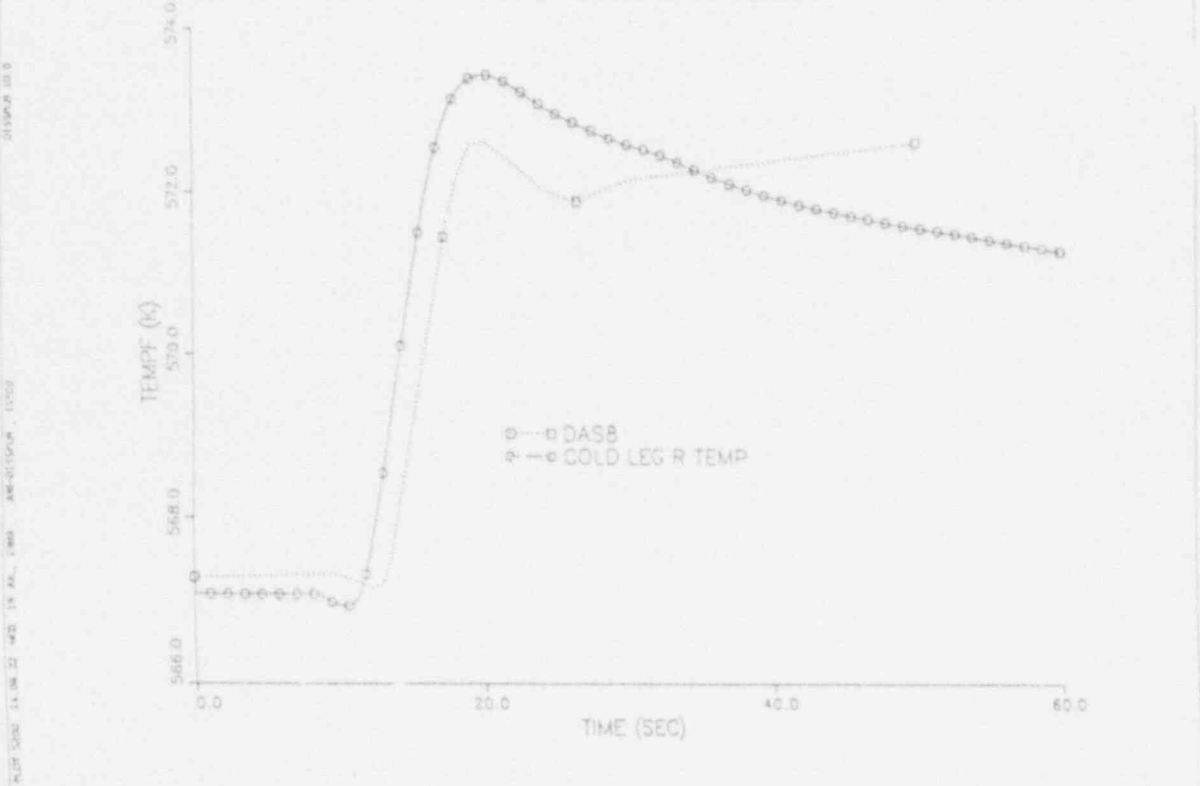
FIG 6.3 HOT LEG TEMPERATURE (R)

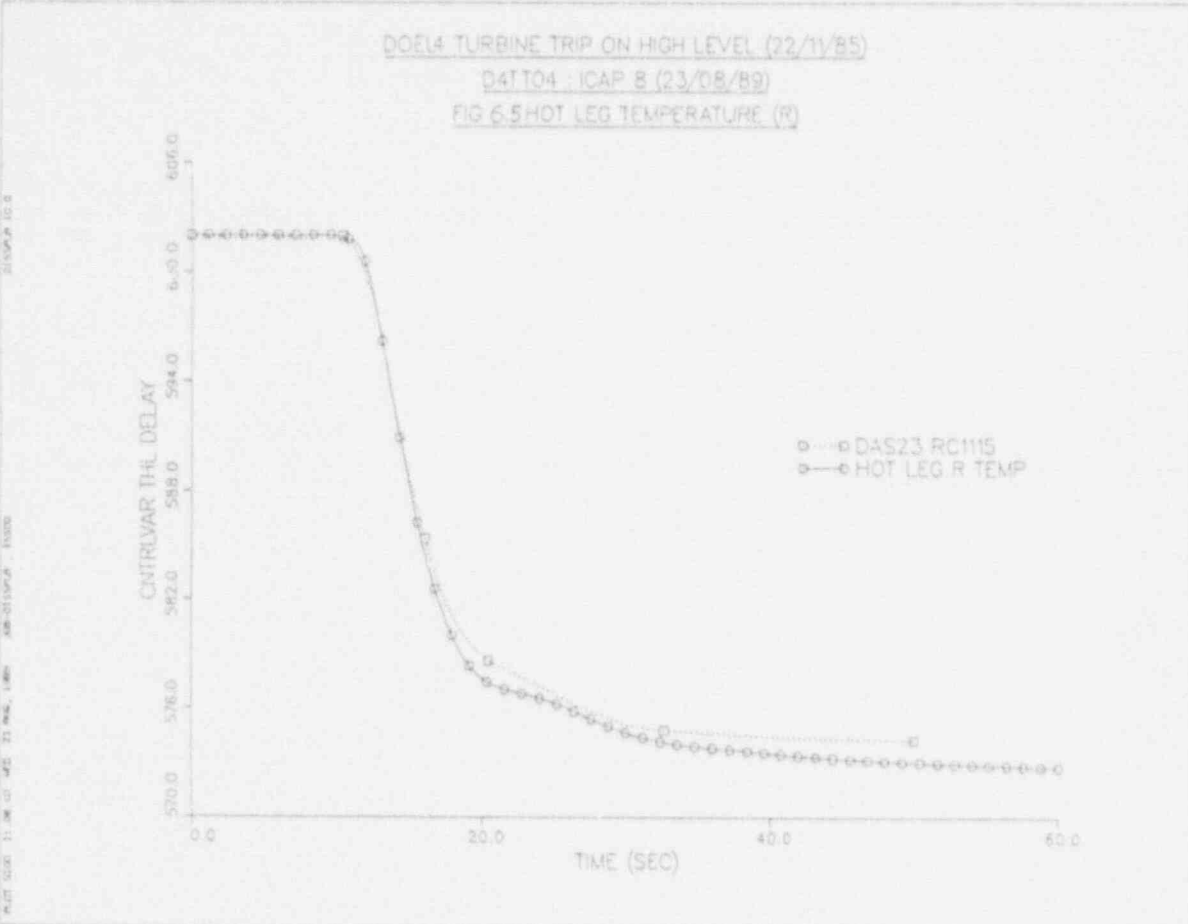


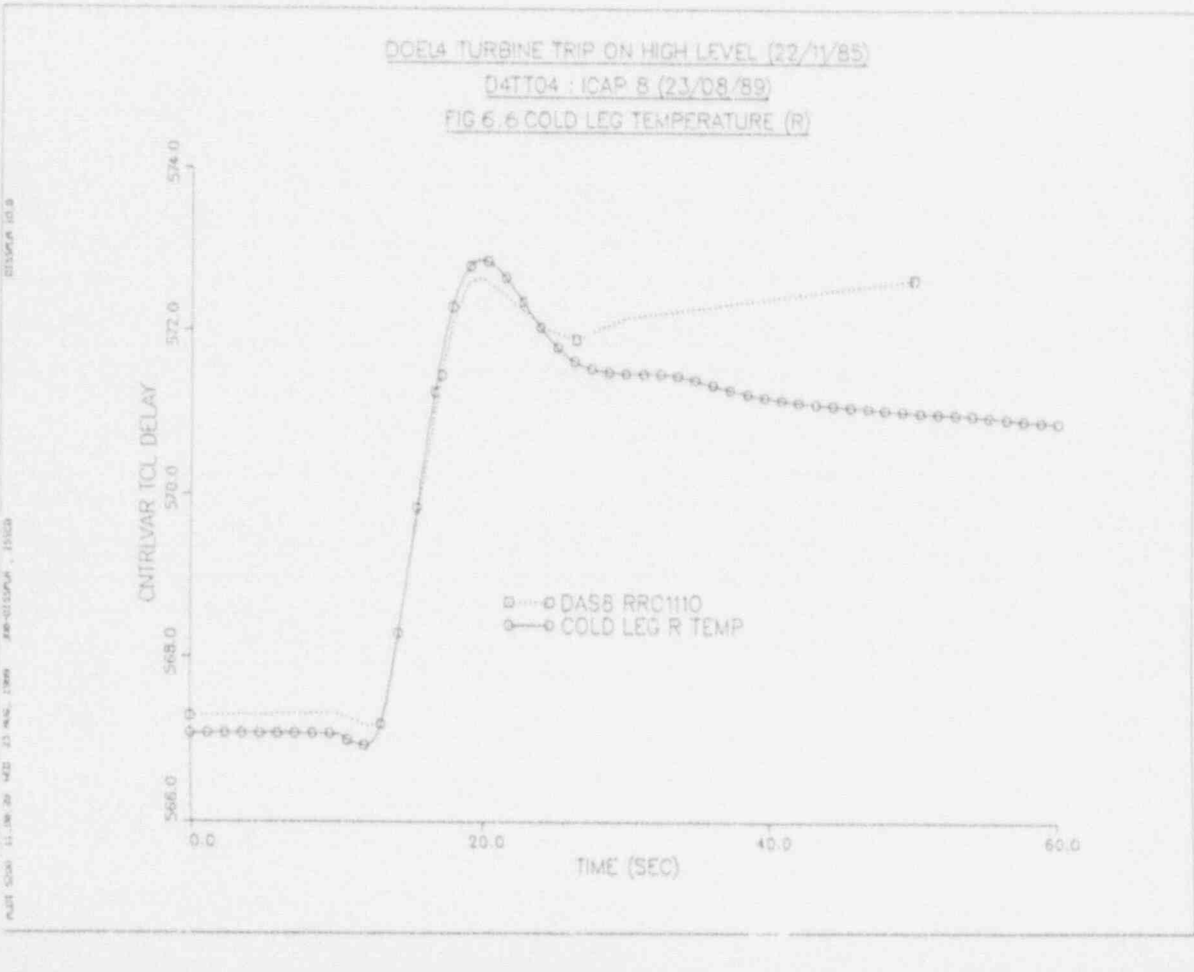
DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)

D4TT03 : ICAP 8 (19/07/89)

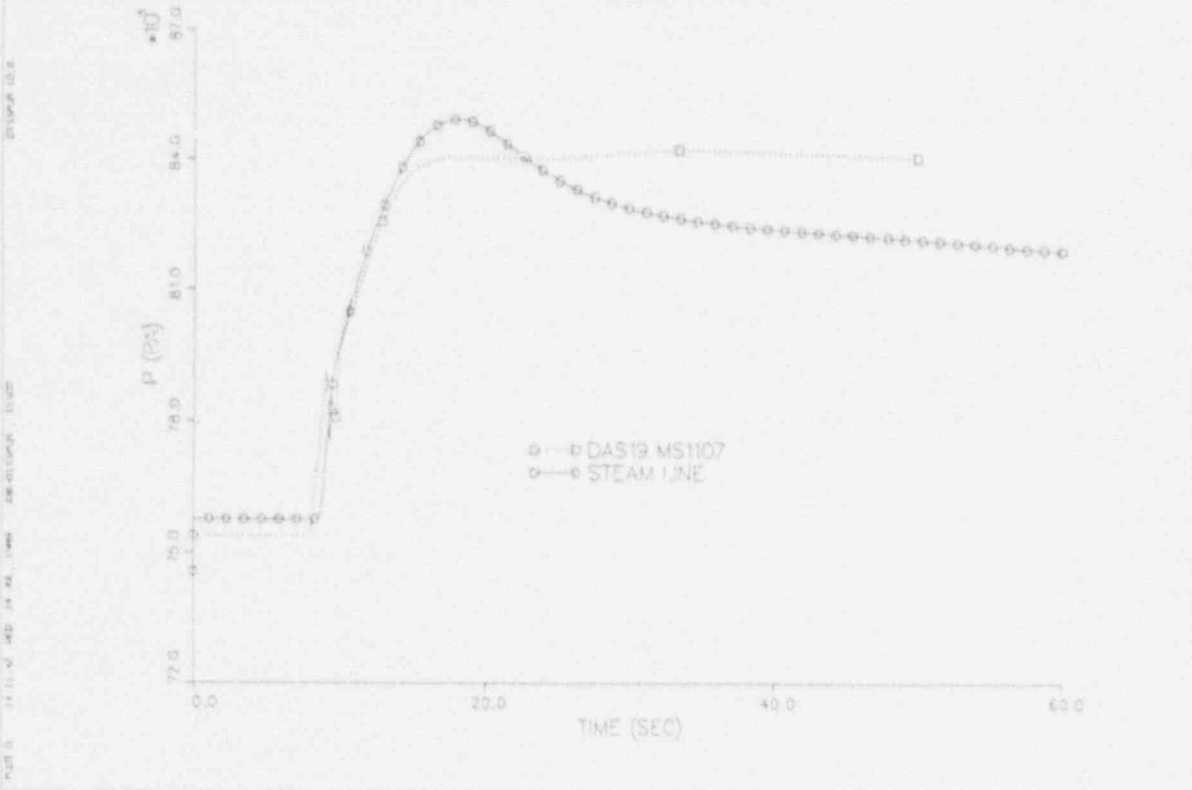
FIG 6.4 COLD LEG TEMPERATURE (R)



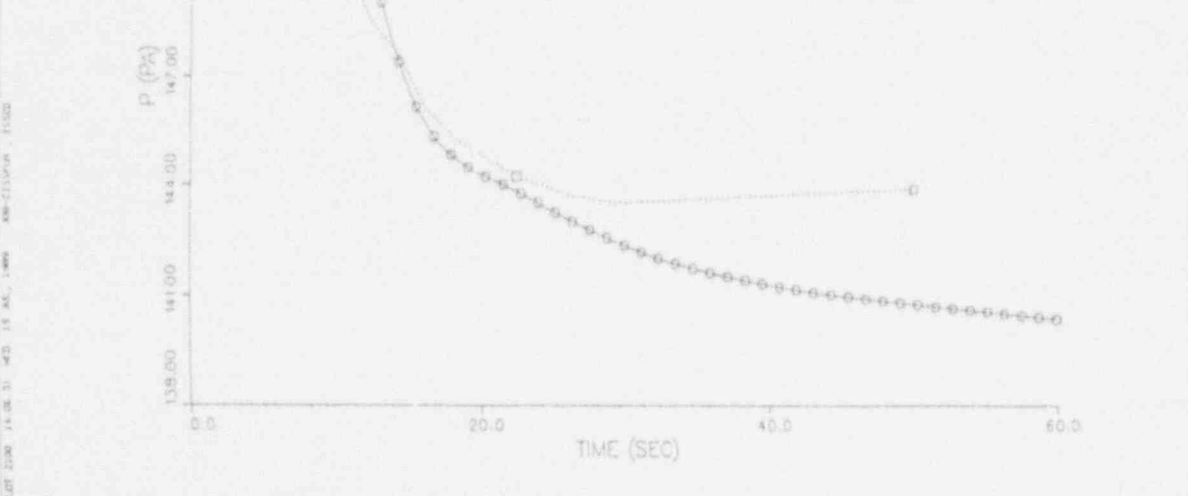

DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)
D4TT04 : ICAP 8 (23/08/89)
FIG 6.5 HOT LEG TEMPERATURE (R)
CNTRLVAR THL DELAY
606.0
600.0
594.0
588.0
582.0
576.0
570.0
0.0
20.0
40.0
60.0
TIME (SEC)
DAS23 RC1115
HOT LEG R TEMP


DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)
D4TT04 : ICAP 8 (23/08/89)
FIG 6.6 COLD LEG TEMPERATURE (R)
CNTRLVAR TCL DELAY
574.0
572.0
570.0
568.0
566.0
0.0
20.0
40.0
60.0
TIME (SEC)
DAS8 RRC1110
COLD LEG R TEMP

DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)
D4TT03 : ICAP 8 (19/07/89)
FIG 6.7 SG R PRESSURE



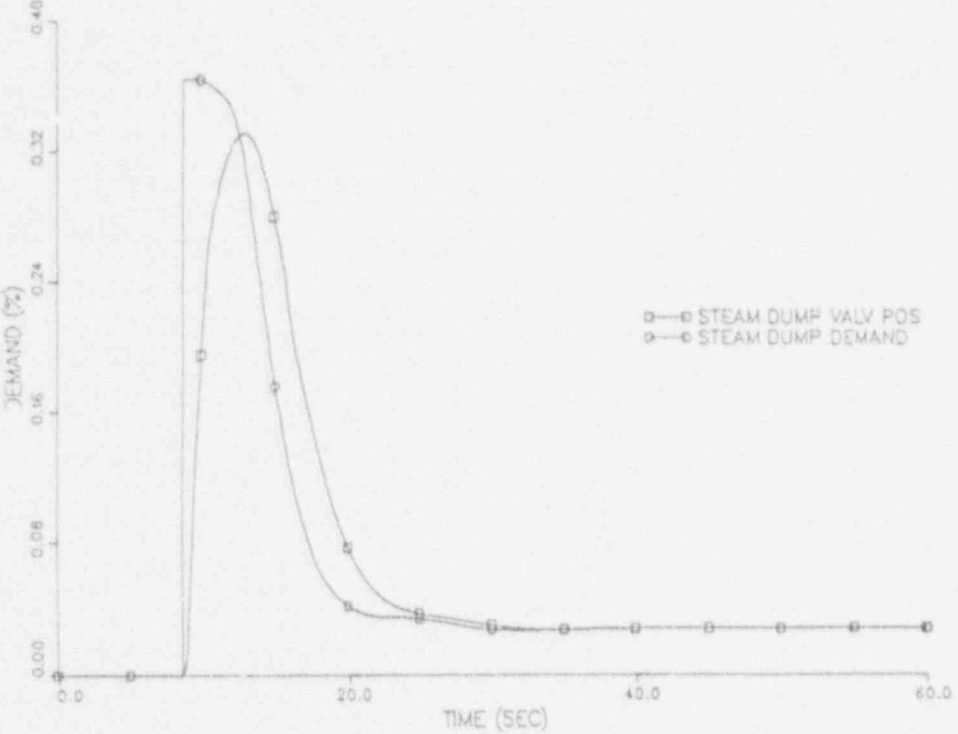
DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)
D4TT03 : ICAP 8 (19/07/89)
FIG 6.8 PRESSURISER PRESSURE



DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

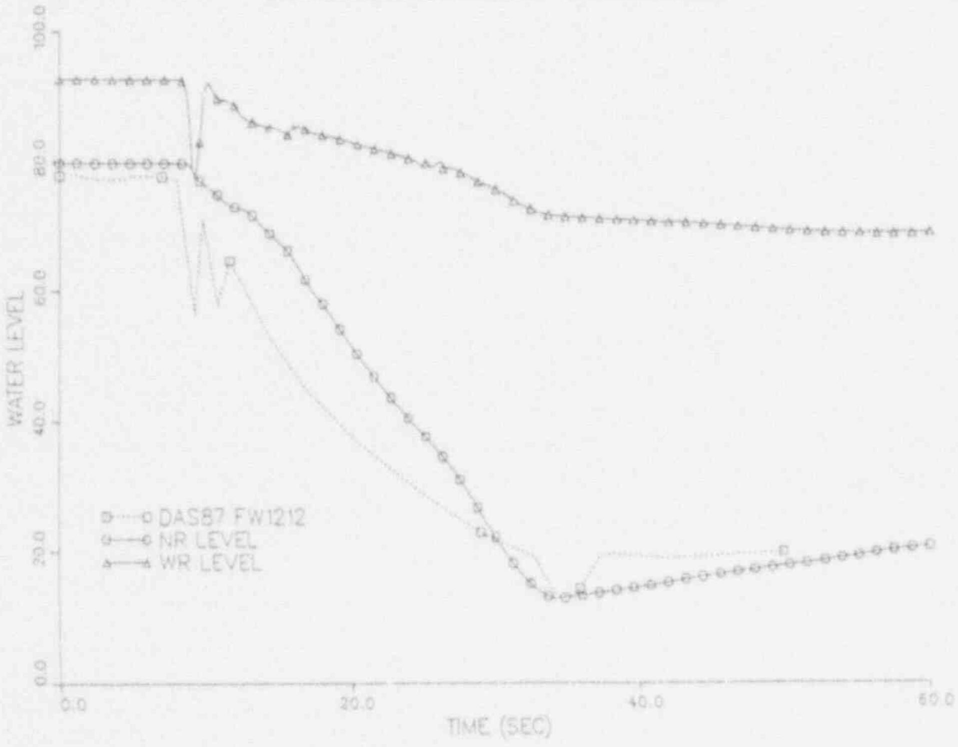
FIG 6.9 STEAM DUMP DEMAND AND VALVE POSITION

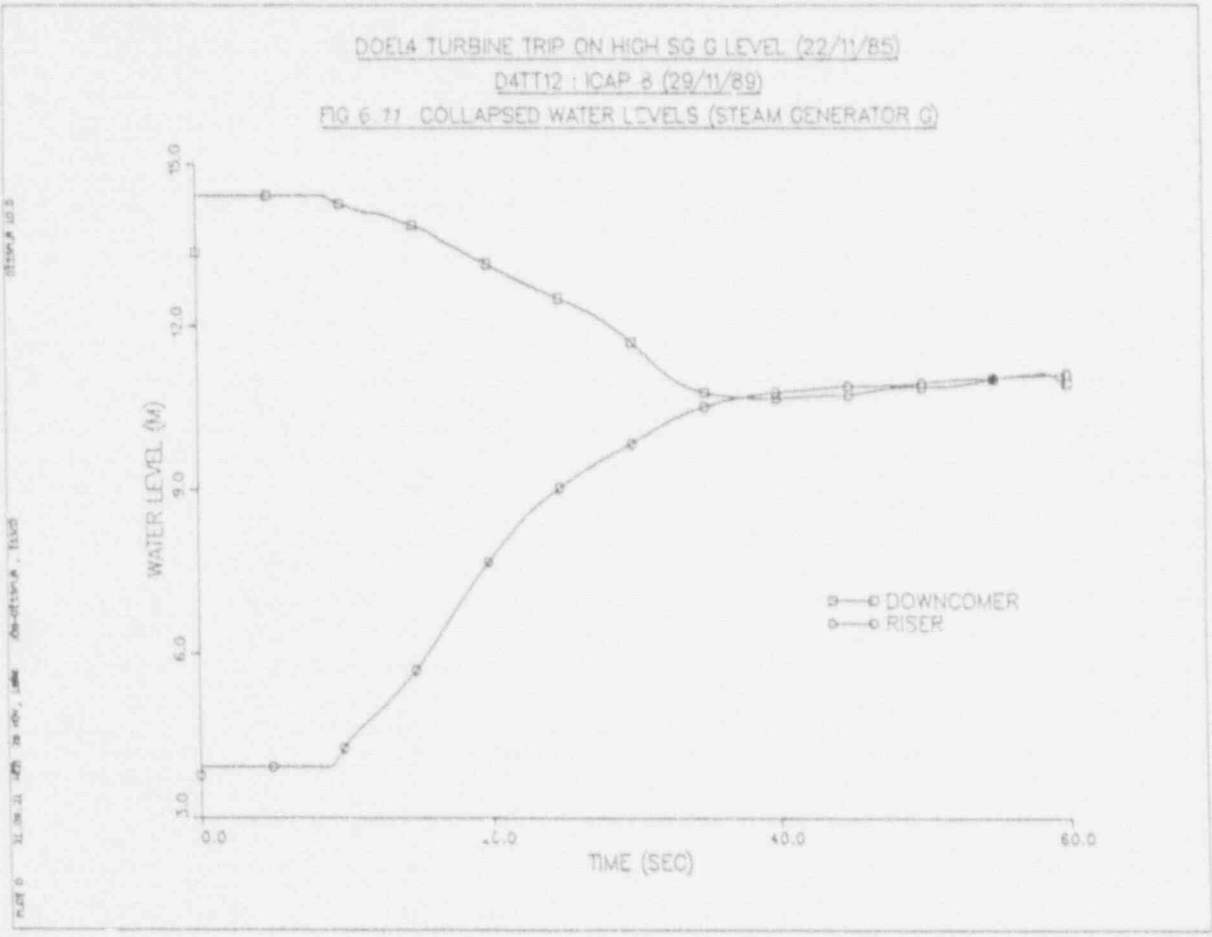


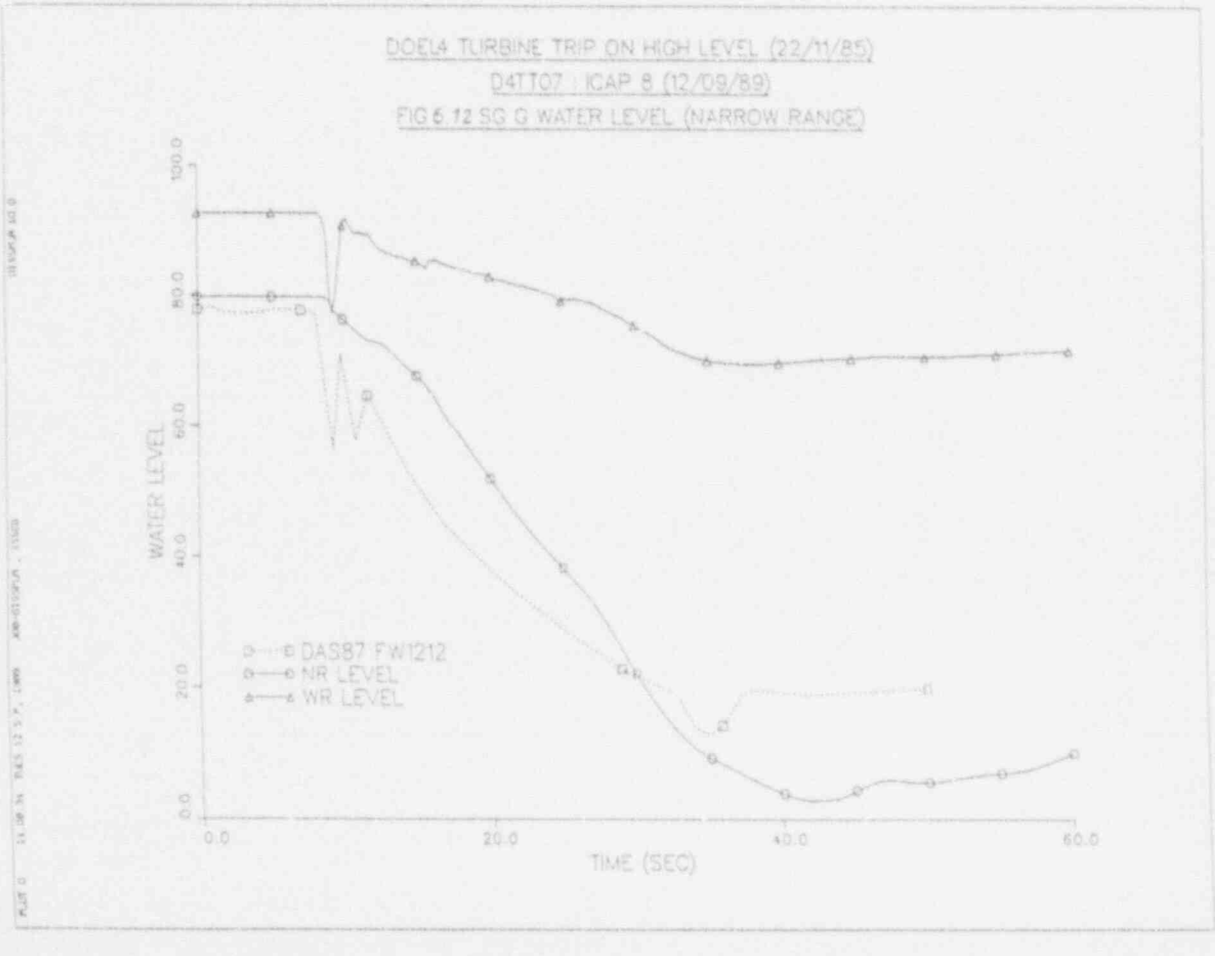
DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)

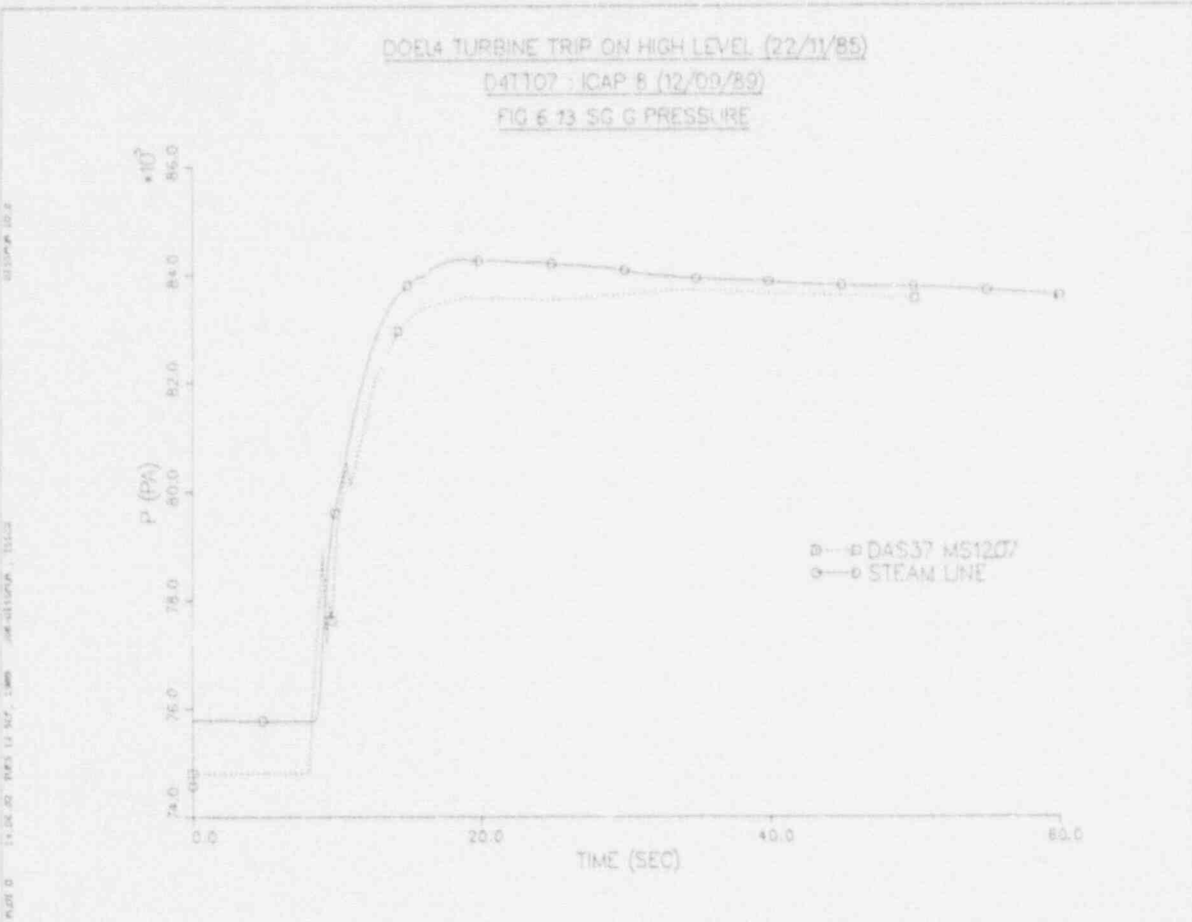
D4TT03 : ICAP 8 (19/07/89)

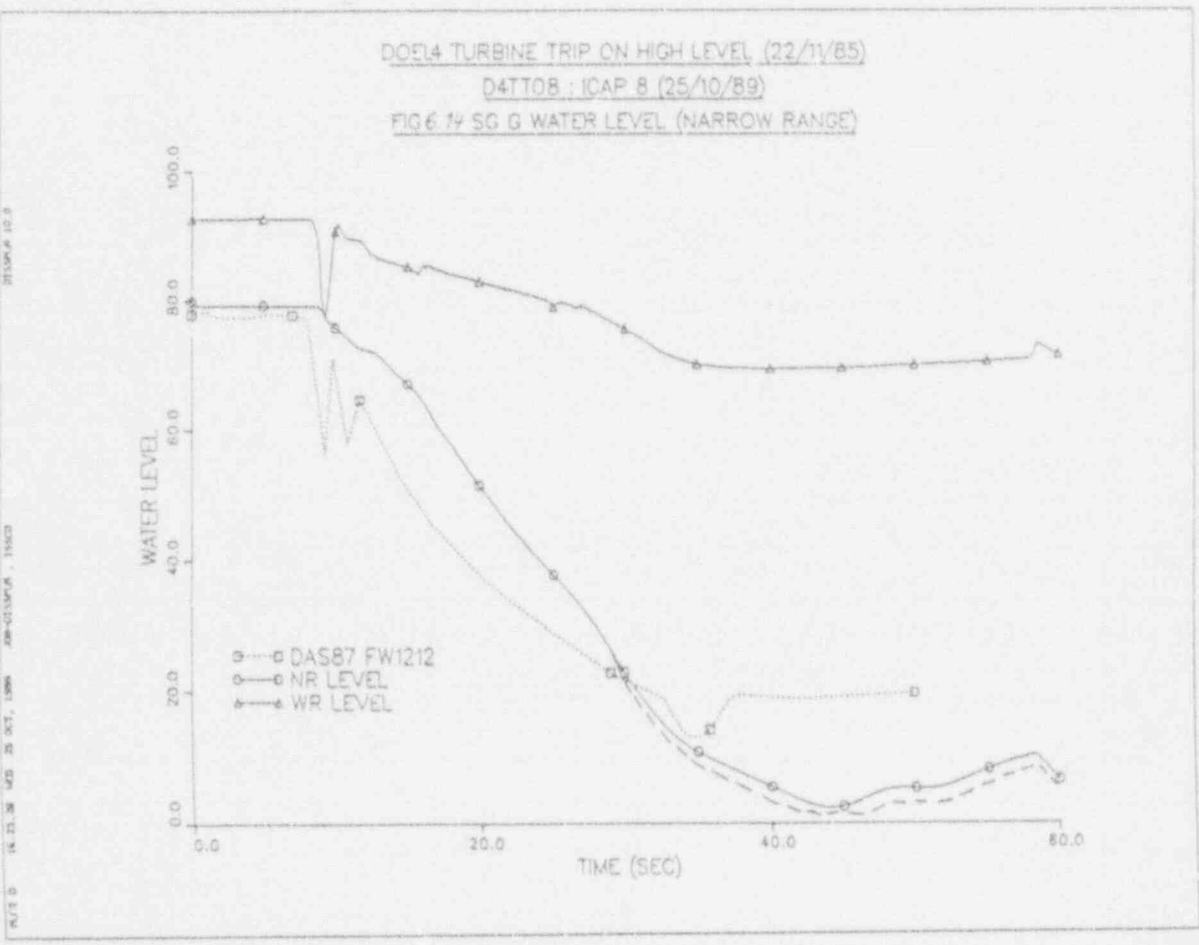
FIG 6.10 SG G WATER LEVEL (NARROW RANGE)




DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)
D4TT12 : ICAP 8 (29/11/89)
FIG 6.11 COLLAPSED WATER LEVELS (STEAM GENERATOR G)
WATER LEVEL (M)
TIME (SEC)
DOWNCOMER
RISER
3.0
6.0
9.0
12.0
15.0
0.0
40.0
60.0


DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)
D4TT07 : ICAP 8 (12/09/89)
FIG 6.12 SG G WATER LEVEL (NARROW RANGE)
WATER LEVEL
TIME (SEC)
DAS87 FW1212
NR LEVEL
WR LEVEL
0.0
20.0
40.0
60.0
80.0
100.0
0.0
20.0
40.0
60.0


DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)
D4TT07 : ICAP 8 (12/09/89)
FIG 6.13 SG G PRESSURE
P (PA)
*10^3
86.0
84.0
82.0
80.0
78.0
76.0
74.0
0.0
20.0
40.0
60.0
TIME (SEC)
DAS37 MS1207
STEAM LINE


DOEL4 TURBINE TRIP ON HIGH LEVEL (22/11/85)
D4TT08 : ICAP 8 (25/10/89)
FIG 6.14 SG G WATER LEVEL (NARROW RANGE)
WATER LEVEL
100.0
80.0
60.0
40.0
20.0
0.0
0.0
20.0
40.0
60.0
TIME (SEC)
DAS87 FW1212
NR LEVEL
WR LEVEL

	RUN 08	RUN 09
AF temp.	20°C	120°C
712-01	0.804	0.667
714-01	1.798	0.004
714-02	0.615	0.000

Table 6.15 Condensation rates in kg/m^3-s

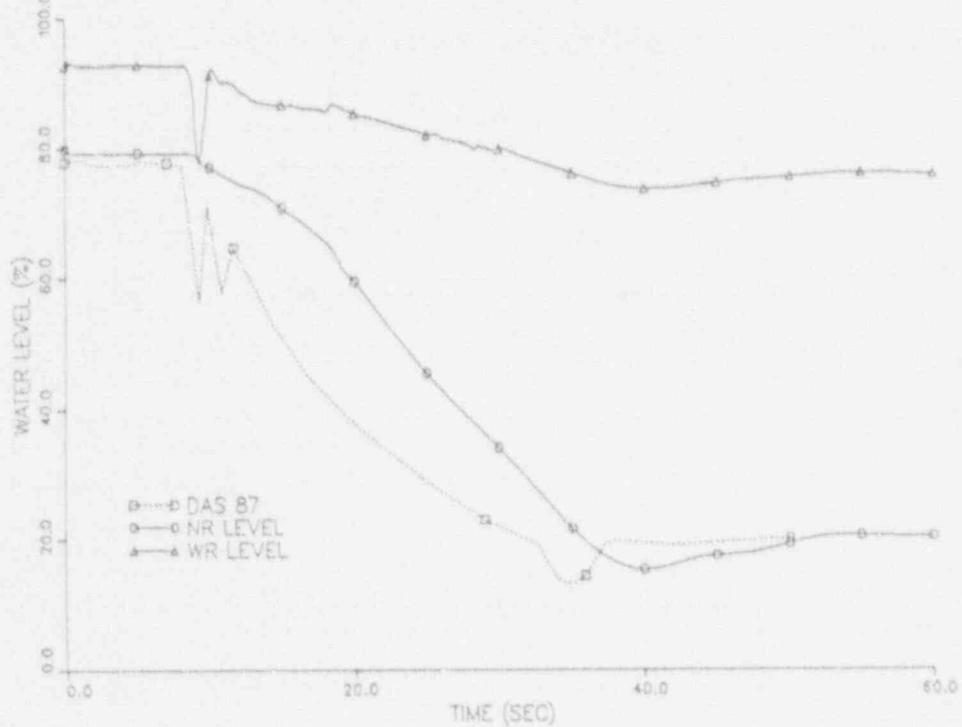
	RUN 05	RUN 10
716-01	0.038	[illegible]
716-02	0.086	0.037
716-03	0.142	0.083
716-04	0.142	0.096
724-01	0.151	0.118
724-02	0.194	0.164
724-03	[illegible]	0.163
724-04	0.217	0.205
724-05	0.360	0.281

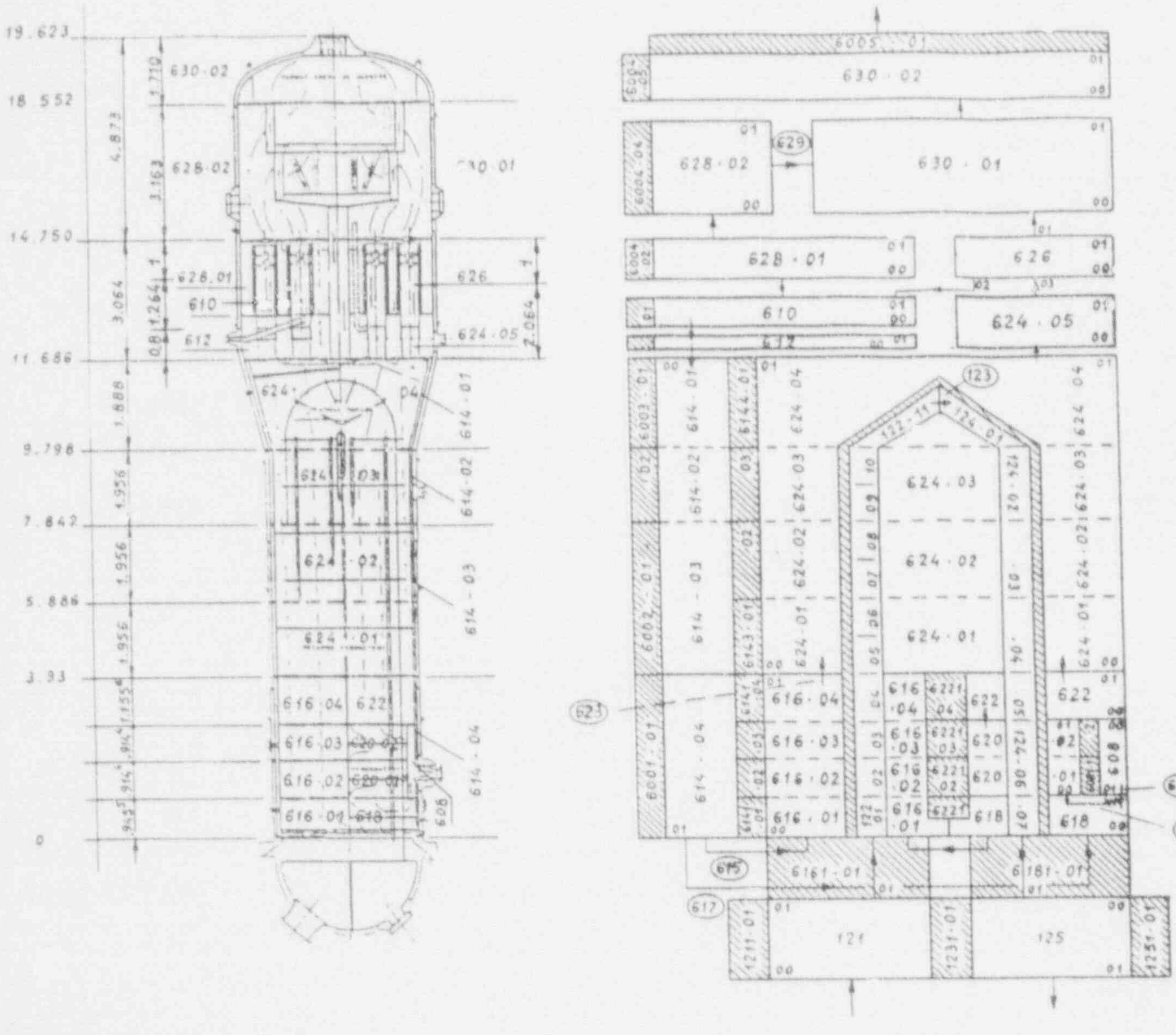
Table 6.16 Void fraction in the riser at the end of the transient

DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT10 : ICAP 8 (24/11/89)

FIG 6.17 SG G WATER LEVEL





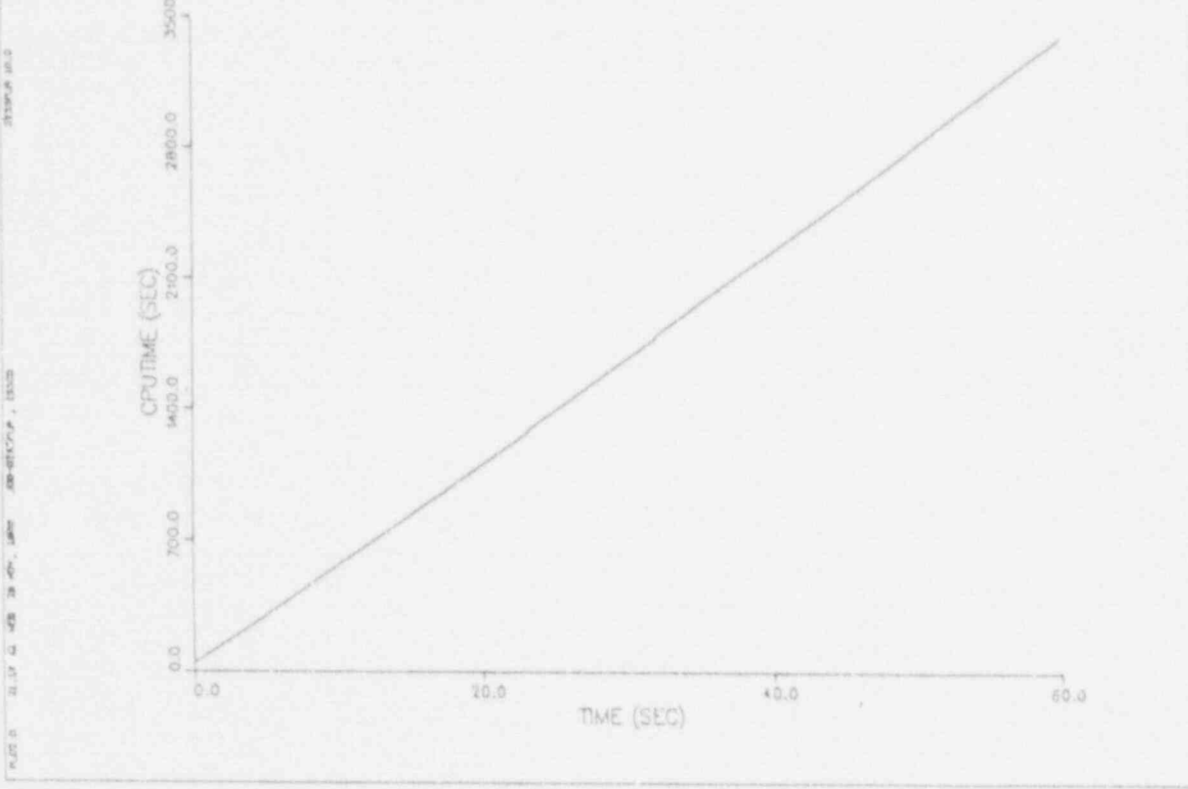
DOEL 4 STEAM GENERATOR NODALIZATION

Fig. 6.18

DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

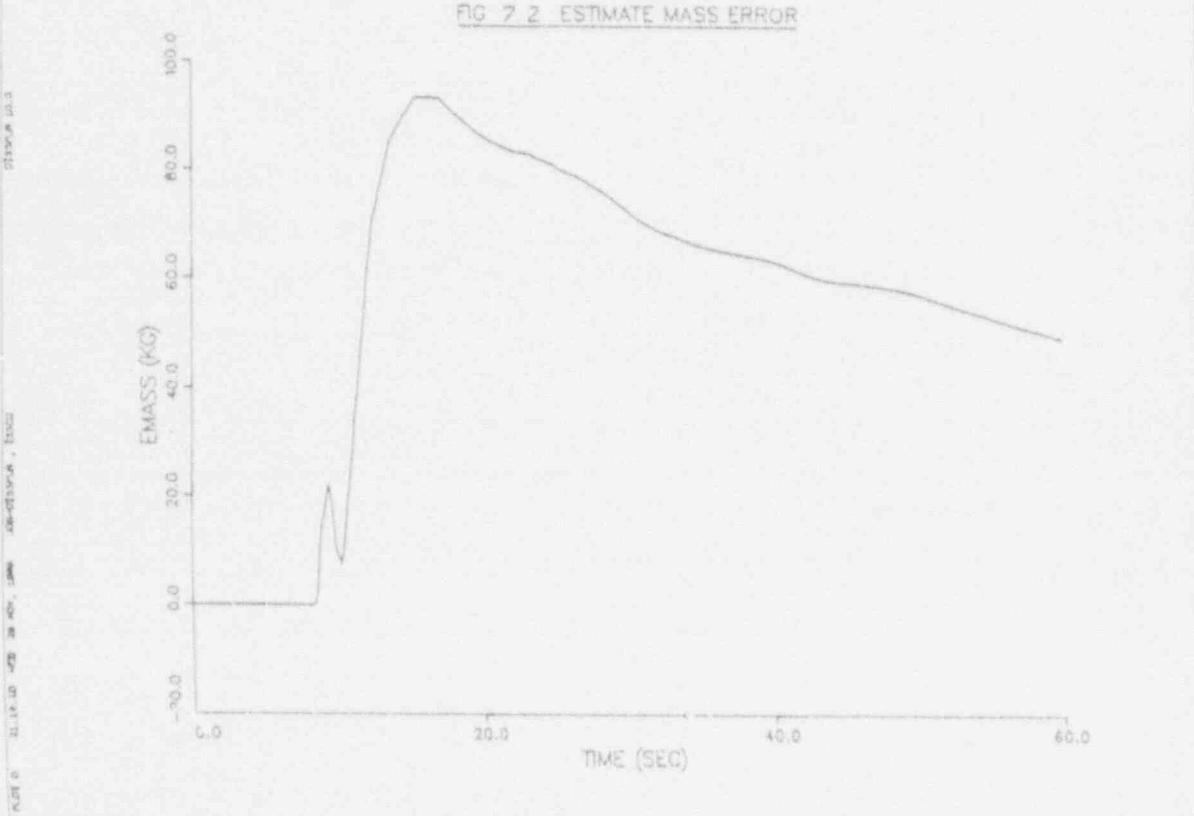
FIG 7.1 CPU TIME VERSUS TRANSIENT TIME



DOEL4 TURBINE TRIP ON HIGH SG G LEVEL (22/11/85)

D4TT12 : ICAP 8 (29/11/89)

FIG 7.2 ESTIMATE MASS ERROR



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11. ABSTRACT (200 words or less)

As part of the first cycle testing program for the Belgium DOEL 4 plant, a turbine trip on high steam generator level followed by a reactor trip was performed on November 22, 1985. Nine assessment runs were made using RELAP5/MOD2 Cycle 36.05 with the output compared to the data acquired during the test.

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