NUREG/IA-0043

International Agreement Report

Assessment Study of RELAP5/MOD2 Cycle 36.04 Based on the DOEL-4 Manual Loss of Load Test of November 23, 1985

Prepared by
E. J. Stubbe, P. Deschutter

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Place du Trone, 1
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Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

March 1992

Prepared as part of
The Agreement on Research Participation and Technical Exchange
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and Application Program (ICAP)

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TABLE OF CONTENTS

Executive Summary

The loss of external load transient test conducted on the Doel-4 power plant has been analysed on the basis of a high quality data acquisition system.
A detailed numerical analysis of the transient by means of the best estimate code RELAP-5 MOD-2 is presented, covering the most important plant components and systems, and complemented by imposed boundary conditions, taken from the recordings, when necessary.
Comparison of recorded and calculated data show that

. The RELAP-5 code is capable to simulate the basic plant behaviour which allows a deeper insight in the physical phenomena.
. Some deficiencies affecting either the model or the code are observed, they could be attributed
  - to the absence of structural heat simulation of the steam generators
  - to acoustic phenomena which influence the steam generator level sensor
  - to excessive interphase drag in the steam generator at low void regimes.
. Typical characteristic features of the plant can better be quantified such as
  - strong thermal coupling between feedwater flow and cold leg temperature for a preheater type steam generator .
  - Delay between feedwater flow variations and level swell and shrink response.

1. INTRODUCTION

Power system failures can easily cascade into a blackout when not enough spinning reserve or external capacity can be added and if the load shedding cannot be established fast enough. Complete blackouts have occurred in the U.S.A., France (Dec. 1978), Belgium (Aug. 1982), Sweden (Dec. 1982), etc.

Although most nuclear power stations are operated in base load, some flexibility must be built-in to cope with grid transients (e.g. low voltage or low frequency) for reactor protection purposes, or with grid failures, in which case the turbine generator set should be quickly isolated from the grid, and the reactor be returned to house load power level (typically 5%) without scram, thereby assuring a quick return to power when grid conditions permit. All Belgian nuclear power plants are designed and tested to ride through such "islanding" transient.

This paper will discuss such test which was successfully performed at the DOEL-4 power station on November 23th, 1985, and compares the recorded data to a numerical analysis which was performed with the best estimate code Relap5-Mod2.
Full scale plant transients are highly desirable to complement the available data base for code assessment. Indeed, validating 1D codes (like RELAP), exclusively on the basis of essentially 1D test facilities (e.g. SEMISCALE, LOBI, PKL, etc...) questions the scalability of the code models and the capability for simulating hydraulic phenomena in highly 3D components.
This report is organised as follows : section 2 provides a brief description of the Doel 4 power plant unit ; section 3 details the transient as recorded on the plant data acquisition system ; section 4 describes the Relap5 model used to simulate the transient ; in section 5 a discussion of the numerical

results is provided ; section 6 discusses some of the possible hardware improvements. The conclusions are presented in section 7.

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 m3/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 (1600 ft^3) pressuriser connected to the hot leg of loop "B" through a 14" surge line. Control of the primary pressure also takes place within the pressuriser by adjustment of the heater rods power or the pressuriser spray flowrate.

2.2. Steam Generators feedwater system and steamlines

The DOEL-4 plant is equipped with preheater type steam generators of the counterflow type (model E2), as shown on the sketch fig. 2.1.
The main feedwater with a nominal flowrate of 2000t/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 (Bottom feed).
The main feedwater flows downward 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 m2 , 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 riven 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 sixteen 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 stable operation followed by a manually activated loss of load test, the steam dump dynamics is controlled by the mismatch between turbine power (derived from a pressure gauge in the first expansion stage in the turbine), converted to a so called reference 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 programmed reference temperature, 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 group 1, 2, 3 and 4 (4 valves each) whenever the signal exceeds respectively 4.[illegible]°C, 7.5°C, 10.1°C and 12.8°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 measurement physical process, sensor response and signal handling have been estimated at 1.1% of nominal power for flux measurements ; 1 °C for primary temperatures ; 1.2 bar for pressuriser pressure ; 4.5% of the range for pressuriser level ; 1 bar for steam generator pressure and 4.5% of the narrow range for steam generator level.

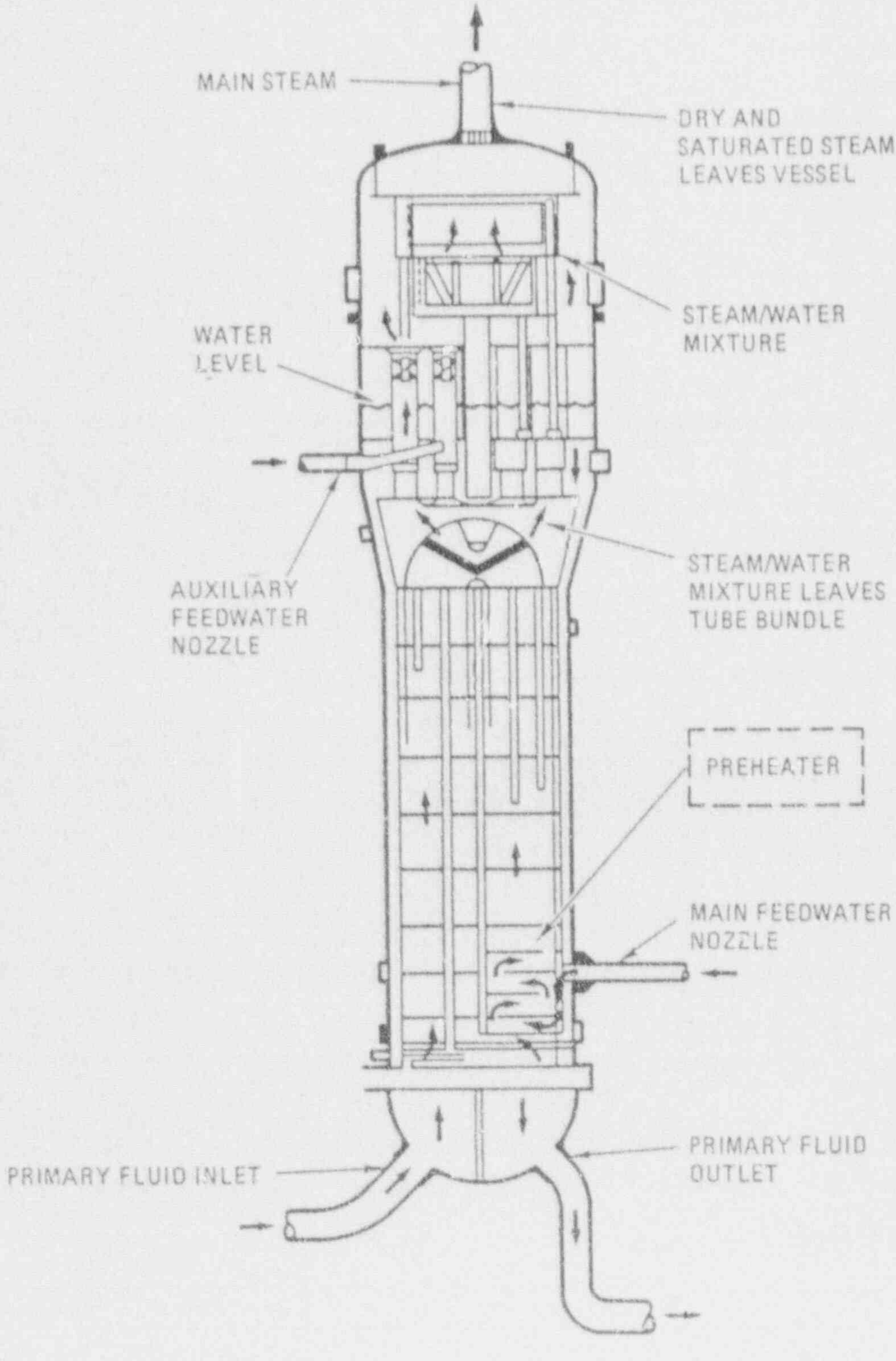


FIG. 2.1 . Preheat steam generator design

3. DESCRIPTION OF THE PLANT TRANSIENT.

As part of a commissioning program for the DOEL-4 plant, this loss of load test at nominal power was performed in order to evaluate the overall plant behaviour, such as the steam dump control systems, more specifically the return to house load without scram, and the switch-over control logic for the feedwater system from bottom feeding to top feeding below 20% reactor power.

The test was initiated by manually opening of the main high voltage breaker when the plant was at full power.

The plant phenomena are analysed on the basis of many DAS recordings, of which the most important are shown in figures 3.1 to 3.6. The origin of the time scale is selected at 13 seconds prior to the initiation of the transient.

All times indicated below refer to the time zero for the DAS recordings.

Figure 3.1 illustrates the fast flux variations and other related parameters over a 20 s time interval.

Figures 3.2 and 3.3 illustrate the long term (600 s) evolution of the most important measured parameters for respectively the steam generators and the primary coolant system.

Figures 3.4 and 3.5 illustrate, on an expanded scale (20 s) the evolution of the primary pressure, pressuriser water level, steam generator pressure and water level.

Figure 3.6 illustrates the steam dump demand over the 600s period.

3.1. Short term (0 - 100 s)

The test was initiated by manual opening of the main high voltage (380 kV) breaker when the plant was at nominal power. The sudden loss of external load results in an acceleration of the turbine-generator set (fig.3. 1 curve 1) and of the primary coolant pumps (fig.3. 1 curve 2), which are powered by the

generator. The resulting increase in the primary loop flow causes a drop in the average core coolant temperature which, through negative moderator feedback, produces a neutron power excursion (fig.3.1 curve 3). This excursion follows the trend of the primary pump speed until t = 18 s.
At this time, a moderator density decrease, resulting from an increase in the cold leg temperature (fig. 3.1 curve 5) initiates a strong neutron flux reduction, as observed between 18 and 22 seconds.
Thereafter, the antireactivity resulting from a fast control rod insertion, further reduces the neutron power. The magnitude of the overshoot and subsequent undershoot of the neutron flux rate, which is essentially caused by the negative moderator coefficient, increases towards the end of core life (EOL), as the boron concentration decreases. These fast flux variations during the initial 10 s of the transient, should be kept within limits to avoid reactor trip on excessive high or low flux variations. Since it is a Belgian requirement that the nuclear power plants must be able to return to house load for power system safety reasons, such tests are performed preferably at EOL core conditions to demonstrate such capability.

The turbine speed controller produces a fast closure of the turbine admission valves, to protect the turbine from overspeed upon loss of external load. This leads to a sudden decrease in the steam generator steam flow rate (fig. 3.2, curve 1), resulting in a sudden increase in the steam generator pressures, and temperatures. The plant reference temperature (T_{REF}), being an image of the turbine power, is monitored by the first stage turbine pressure and decreases suddenly upon turbine valve closure. The sudden difference between the reference temperature and the measured primary average temperature activates on one hand a full speed insertion of the controls rods (72 steps/min), and on the other hand a fast opening of the steamdump bypass valves to the condensor (85% of nominal flow capacity), to preclude overheating of the primary system.

The sudden increase in steam generator pressure creates a fast drop of the narrow range level (fig. 3.2, curve 3 and fig. 3.4), due to a sudden steam collapse in the riser section. The steam generator feedwater control system reacts immediately by steering the feedwater control valves wide open to restore the steam generator water level (fig. 3.2, curve 2). The sudden opening of the steam dump valves creates a bulk boiling in the steam generators thereby enhancing the water level increase between 20 and 30 seconds, which in turn reduces the feedwater flow rate temporarily. A gradual reduction in steam flow demand at 40 s depresses the level again below the reference level until 100 s. During this period, the feedwater control valves are fully opened again, thereby producing a large feedwater supply. It is of interest to note that during this period the cold leg temperature drops very fast which is a specific feature of preheater type steam generators. The direct injection of the feedwater around the cold side of the U tube bundle in the preheater leads to a strong thermal coupling between the feedwater flow/temperature variations and the cold leg temperature variation.

On the primary side (fig. 3.3), the sudden closure of the turbine admission valve causes a large power mismatch between primary and secondary side which gives rise to a sudden increase in the cold leg temperature, resulting in a fast excursion of the pressuriser pressure and water level (fig. 3.5). The quick opening of all the steam dump valves to the condenser precluded a reactor trip on high pressuriser pressure, as well as the opening of the pressuriser and steam generators relief valves. The still existing power mismatch between primary and secondary sides generates a fast insertion of the control rods. The combined effect of steam release to the condenser and of control rods insertion reverses the previous parameters trends, and a cooldown of the plant results.

3.2. Long term (100 - 600 s)

The long term behaviour is mainly dominated by the steam dump demand (fig. 3.6) and by the feedwater system behaviour. The steam generators level control system at normal power is an improved version of the classical three element control system, in that it involves two regulators in cascade, the upstream one based on level error, and the downstream one based on steam/feedwater flow mismatch. This second regulator is a circuit accounting for the static level swell and shrink characteristic of the steam generator type.

By lagging the feedwater flow over the steam flow, one compensates for the negative slope of the static secondary water mass versus steam generator power at fixed water level (Delay T_o). This is seen in figure 3.2 by observing that the feedwater flow variations are lagging the steam flow variations by 50 s. With this approach, only the static component of the shrink/swell phenomenon is compensated by the flow regulator. The transient component should be compensated by the level regulator (Ref. 1).

Indeed, a positive step change in feedwater leads to a level response showing initially a drop before the level rises due to the increased feedwater flow. The level does not increase immediately since the added feedwater (colder than the stem water mixture in the steam generator) suddenly condenses a number of steam bubbles, thus producing a decrease in the specific volume of the mixture (dynamic level shrink). The inverse is true for a sudden negative step change of the feedwater (dynamic level swell). The effect of those phenomena can be interpreted as a time delay T_1 between the flow

variation command and the effect on the level variation. This delay is specific for each steam generator de..gn and depends also on the feedwater temperature.
If the feedwater/steam flow lag (T_o) is of the same order as the time delay (T_1), and if the respective gains are not optimised, oscillations can occur, which are visible in this test in the long term.
When the feedwater flow is decreased below 20%, an automatic switch-over from preheater feeding to top feeding is initiated around 600 s. Finally the plant reaches an equilibrium at house load (5%).

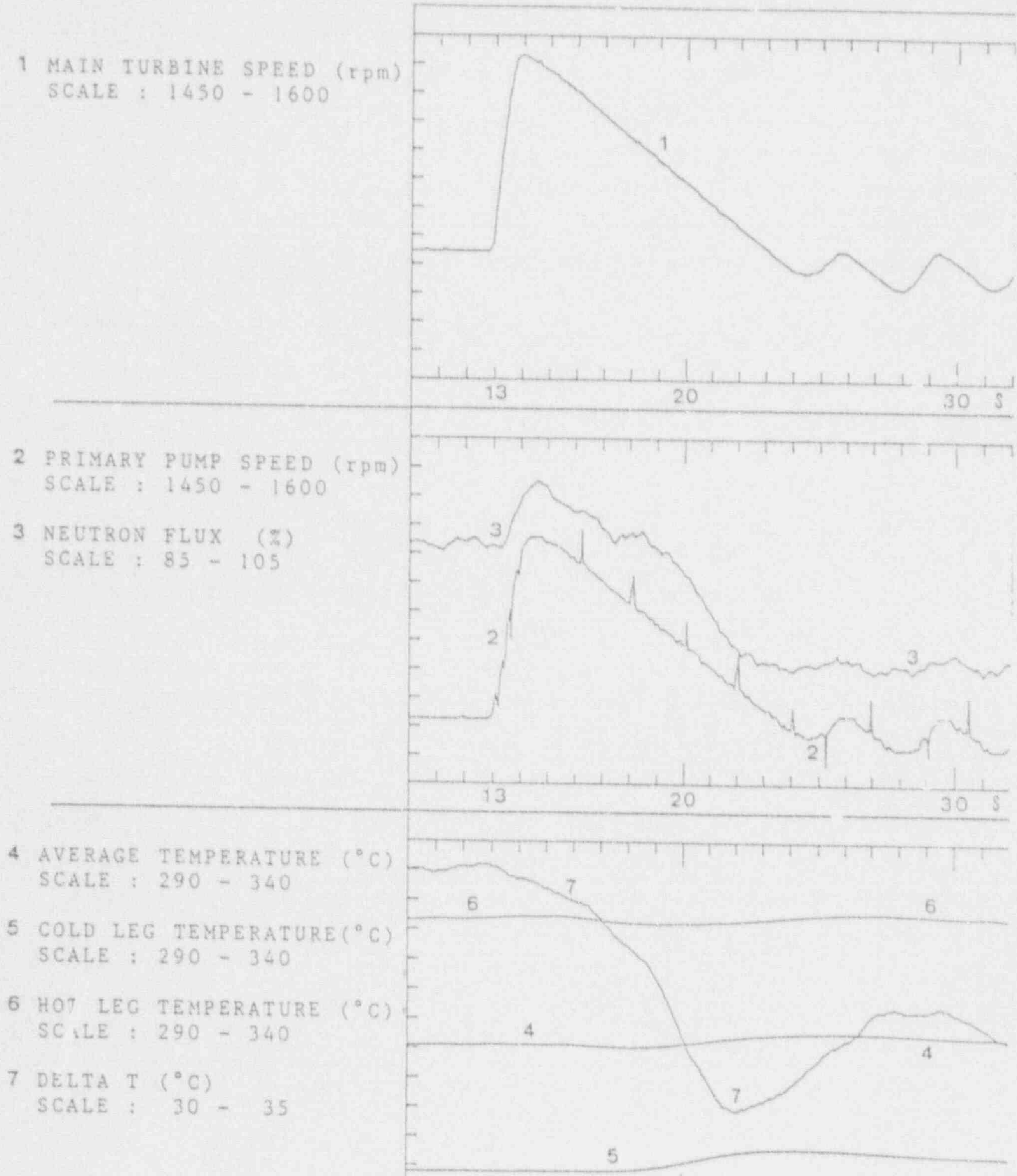


FIG. 3.1: FAST FLUX VARIATION INDUCED BY A LOSS OF EXTERNAL LOAD

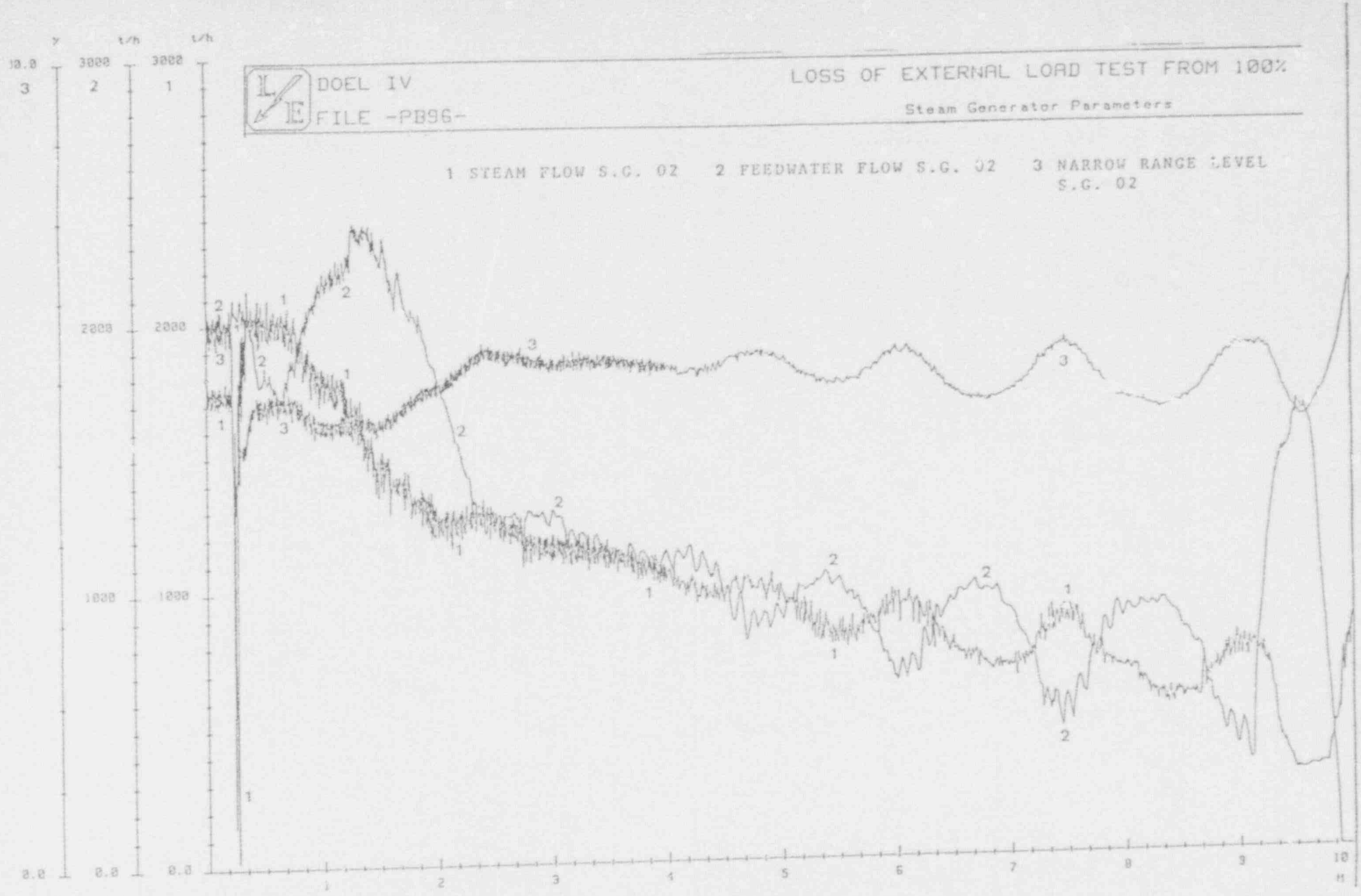


FIG. 3.2 : STEAM GENERATOR PARAMETERS

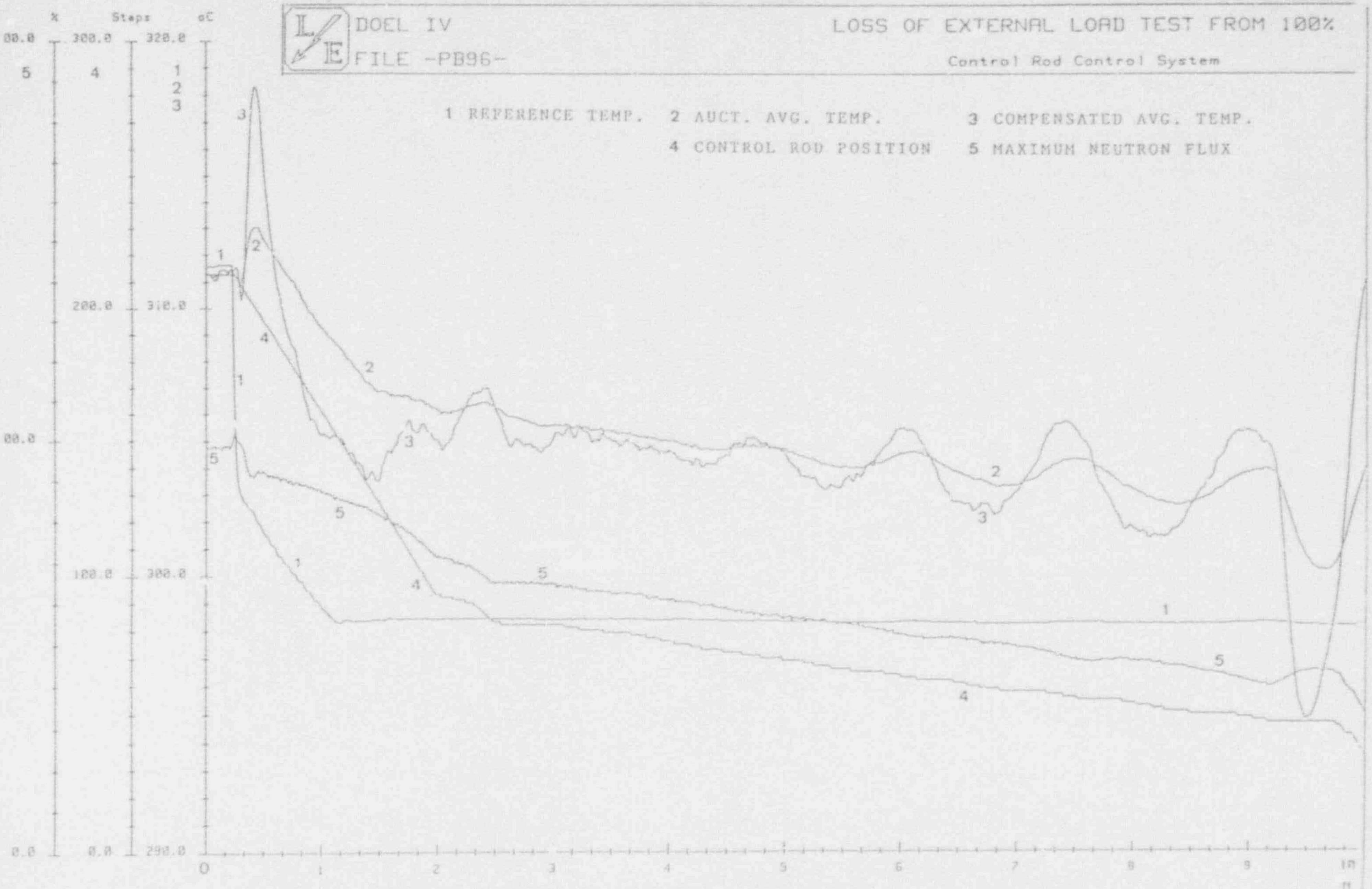


FIG. 3.3 : PRIMARY SYSTEM PARAMETERS

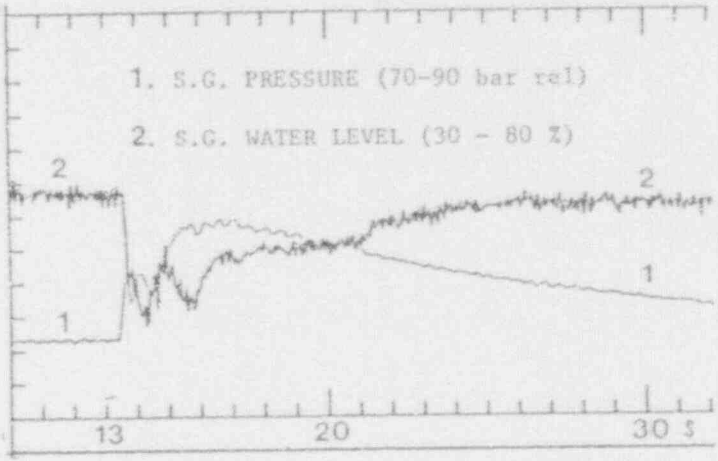


FIG. 3.4: S.G. PRESSURE AND LEVEL

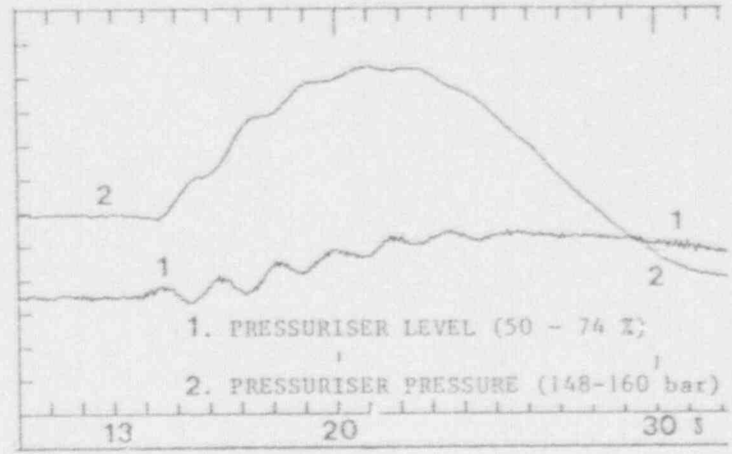


FIG. 3.5: PRESSURIZER PRESSURE AND LEVEL

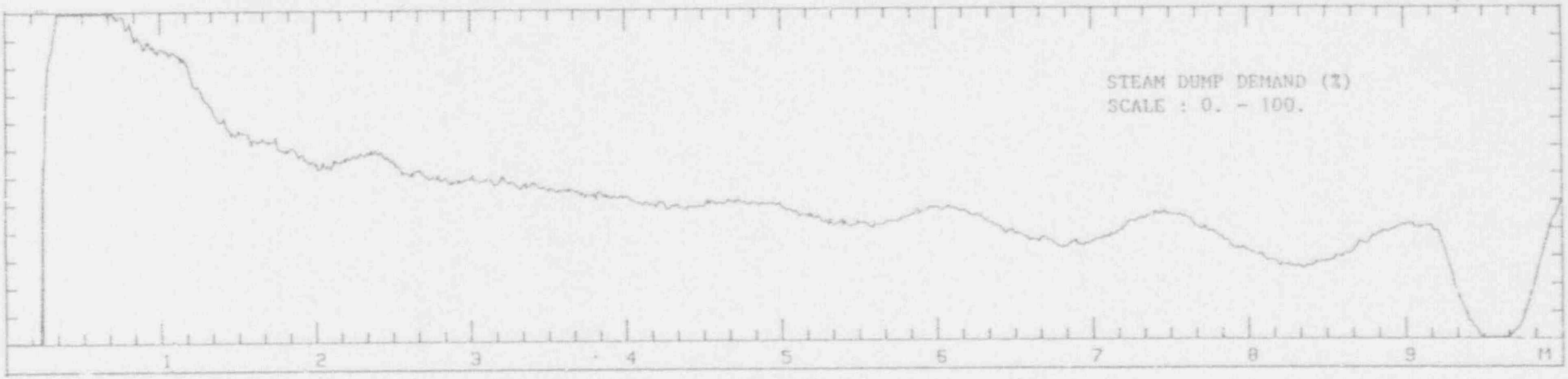


FIG. 3.6: STEAM DUMP DEMAND

4. CODE AND MODEL DESCRIPTION FOR PLANT SIMULATION

The simulation was carried-out with the RELAP-5 Mod.2 cycle 36.04 code on a CYBER 180/825 computer, over a period of 600 sec. starting at time zero of the recording sequence.

The reactor model was developed using the methods and procedures recommended in the code manual (Ref. 2). The primary and secondary systems (feedwater / steam generator / main steam) were both modeled <u>explicitly</u> 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, with the exception given in section 4.1 below.

On the other hand, auxiliary components and systems are being simulated <u>functionally</u> 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 pressuriser relief (PORV's) and safety valves controls ;
- the pressuriser spray and heaters control ;
- the main feedwater system ;
- the auxiliary 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 Reactor Coolant Pumps velocity ;
- the charging and letdown flows ;
- the control rods movement in the core ;

- the main turbine admission valves.

A schema of the control systems and boundary conditions is given on fig. 4.1.

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
- pressuriser : 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 overall nodalization totals 229 volumes, 248 junctions and 197 heat structures (see fig. 4.2).

In Annex 1, a listing of the input deck is included.

Annex 2 gives the restart input deck for the base case.

The 3 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).

Due to computer memory limitations, a reduction of the size of the complete model was required. A parametric study for a single steam generator with and without heat slabs produced identical results, except for sudden temperature variations occuring at the onset of the transient. The impact of the simulation of the heat slabs on the initial cold leg temperature rise was about 0.5 degrees C.

Thus it was decided not to simulate the structural heat slabs for the 3 steam generators.

Core power generation

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

Those control systems which tie together important physical phenomena and which may exhibit close feedback on the plant parameters have been simulated by the available control variables in the RELAP-5 code.

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 to apply this simulation capability for fluid systems where fluid transit times are large compared to the RELAP time step or where the inertia inherent to the system may be important.

4.2.1. Pressuriser relief and safety valves

The three pressuriser relief valves (junctions 471, 472 and 473) 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 (J461) operated by a control variable that simulates their pressure cycle hysteresis.

4.2.2. Pressuriser spray and heaters

A small, constant spray flow - the "residual spray" - is supplied to the pressuriser 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 (J441), while the pressure-dependent variable flow is supplied by two servo-valves (J435, J445) inserted in the explicitely modelled spray lines (V430, V440)connecting the cold legs to the pressuriser vapour phase, and sized to deliver the nominal spray flow at full open position.

All pressuriser heaters are constructively identical. Functionally, however, they fall into two groups : the proportional heaters (308 kW) provide the fine regulation capability needed to keep the pressuriser 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 pressuriser when the level rises significantly.

4.2.3. Main feedwater system

The three feedwater lines to the preheater (bottom feeding) are explicitely modelled from the steam generator down to the Feedwater Isolation Valves.
In the end volume of each line, a time dependent junction delivers a flowrate computed by a plant-like controller, whose input data are the ca[illegible] S.G. level, the steam line flowrates, the actual feedwater flow and the feedwater temperature. A sketch of the main feedwater controller is shown on the figure 4.3.

4.2.4. 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 for temperature wherein X = 6,7 or 8 for respectively steam generator R,G or B) and a time dependent junction (TDJ X35 for flow).

4.2.5. Steam generators relief and safety valves

Each steam generator relief valve is modelled as a servo-valve (JX41) 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 (JX44) with a response similar to that of the overall system.

4.2.6. Steam dump to the condenser

The complex steam dump system is being reduced to a single control valve junctions (J925) from the main steam header to a low pressure volume (V950).
All 16 steam dump valves have been globalised 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.
The valve response inertia is modelled by 2 time constants for respectively the fast (trip open mode) and slow (throttling mode) actuations. Figure 4.4 illustrates the steam dump control logic used in this simulation.

4.3. Systems or effects simulated as boundary conditions

By limiting the scope of simulation to the components as shown in Fig. 4.1, it is essential to impose suitable boundary conditions on the RELAP-5 model for those parameters which are derived from non-simulated components.
This is the case for the turbine generator set, which experiences a velocity transient starting when the grid demand drops sharply during the loss of external load test. As the reactor coolant pumps are powered by the turbo-generator set, their velocity was imposed as a time dependent boundary condition, as recorded on the plant.

The same approach was retained for the charging (J181) and letdown (J283) system, whose flows were imposed as function of the time, to and from the concerned volumes of the primary loops.

Control rods displacements versus the transient time, taken from the DAS recordings, were imposed as boundary conditions. Should a reactor trip signal be generated, scram rods would be treated as a antireactivity injection curve dependent on the time elapsed since the scram signal, accounting for the finite rod drop time.

As seen from the secondary system point of view, the main turbine, as well as the other live steam consumers (feedwater turbopumps, feedwater reheaters, air ejector,...) acts as a sink for the steam produced in the steam generators. Since the purpose of the simulation was not to address the quite complex turbine behaviour, it was modelled as a valve (J905) opening in function of the time, derived from the recordings of the turbine admission valves positions.

Table 4.1 summarises the list of imposed boundary conditions.

TABLE 4.1 - LIST OF SIMULATED CONTROL SYSTEMS AND IMPOSED BOUNDARY CONDITIONS

Label (Fig. 7)	Control system imposed B.C.	Simulated	Description/Impact
RT	Plant reference temperature	NO	DAS data used since turbine component was not simulated. No impact on calculation since Tref drops to constant value
PP	Pressuriser pressure control (Heaters, spray valves)	YES	Heater power controlled by pressuriser pressure and water level. Spray valves controlled by pressuriser pressure
PL	Pressuriser water level	NO	DAS data input for charging and letdown. Weak feedback.
RP	Reactor coolant pump speed	NO	DAS data used since turbine-generator not simulated. Impacts only the first 20 sec. of the transient.
CR	Control rod position	NO	DAS insertion rate used.
FT	Feedwater temperature	NO	DAS data used since the balance of plant was not simulated.
FW	Steam generator feedwater flow	YES	Feedwater flow rate controlled by calculated water level, steam and feedwater flows (very strong feedback). No FW pumps.
SD	Steam dump control system	YES	Bypass steamflow controlled by difference between reference temperature and calculated average loop temperature.
TU	Turbine admission valve control	NO	Preprogrammed position from best estimate calculation. No feedback on calculated data.

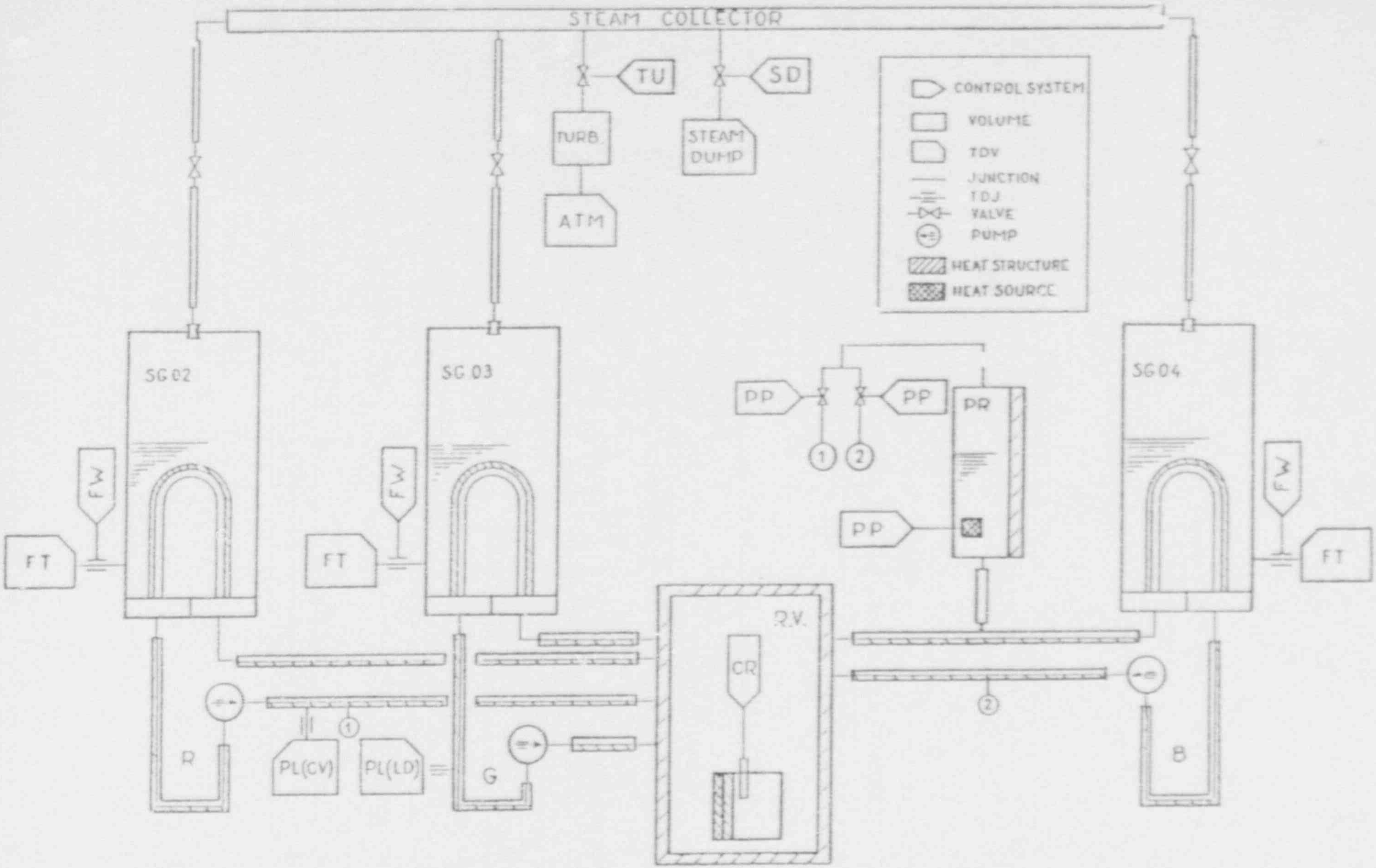


FIG. 4.1 : BLOCK NODALISATION FOR DOEL 4 LOSS OF LOAD TRANSIENT

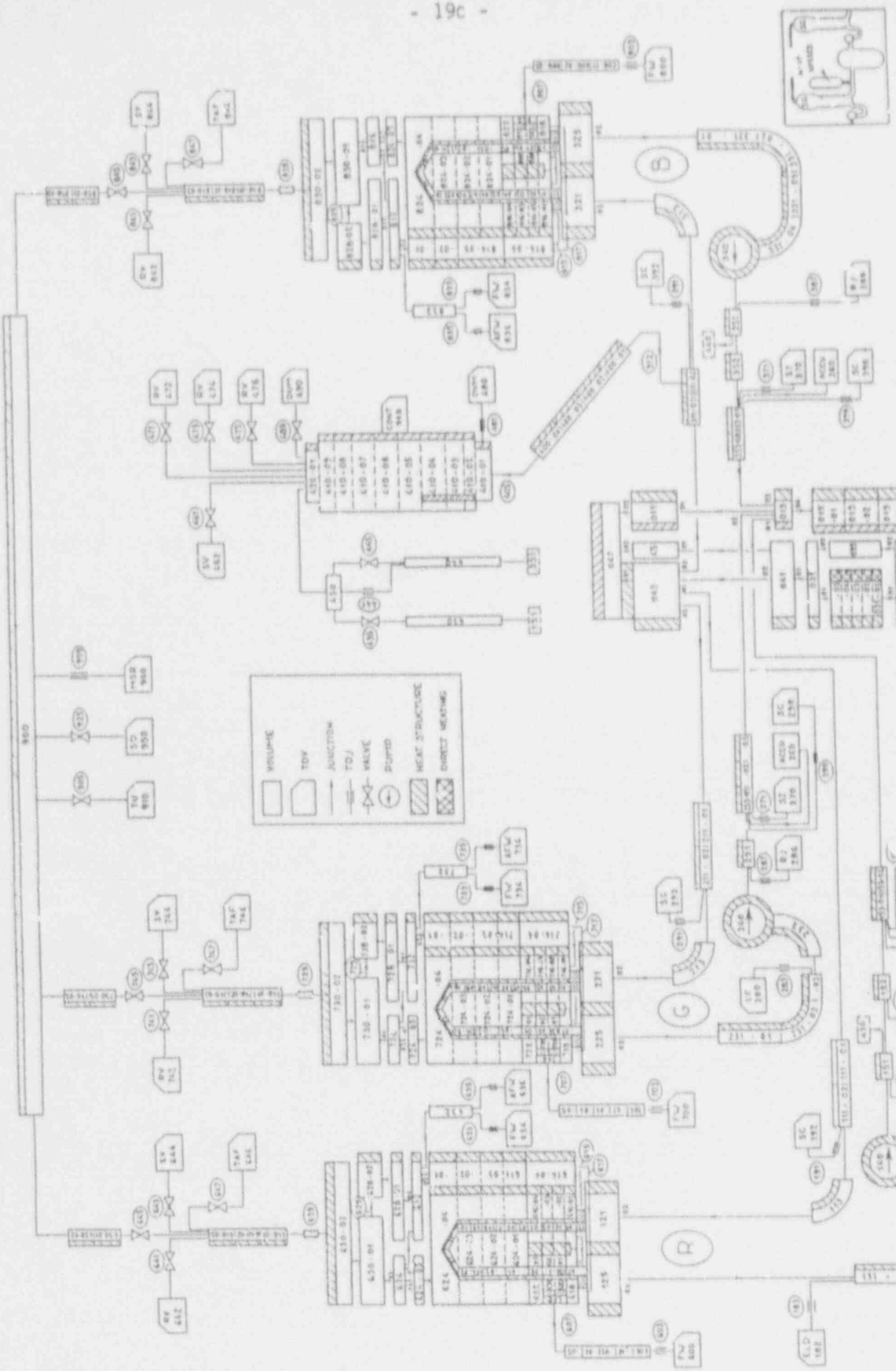

VOLUME
TDV
JUNCTION
TDJ
VALVE
PUMP
HEAT STRUCTURE
DIRECT HEATING

FIG. 4.2

Fig.4.3 : D4-Feedwater control schema block (1 SG)

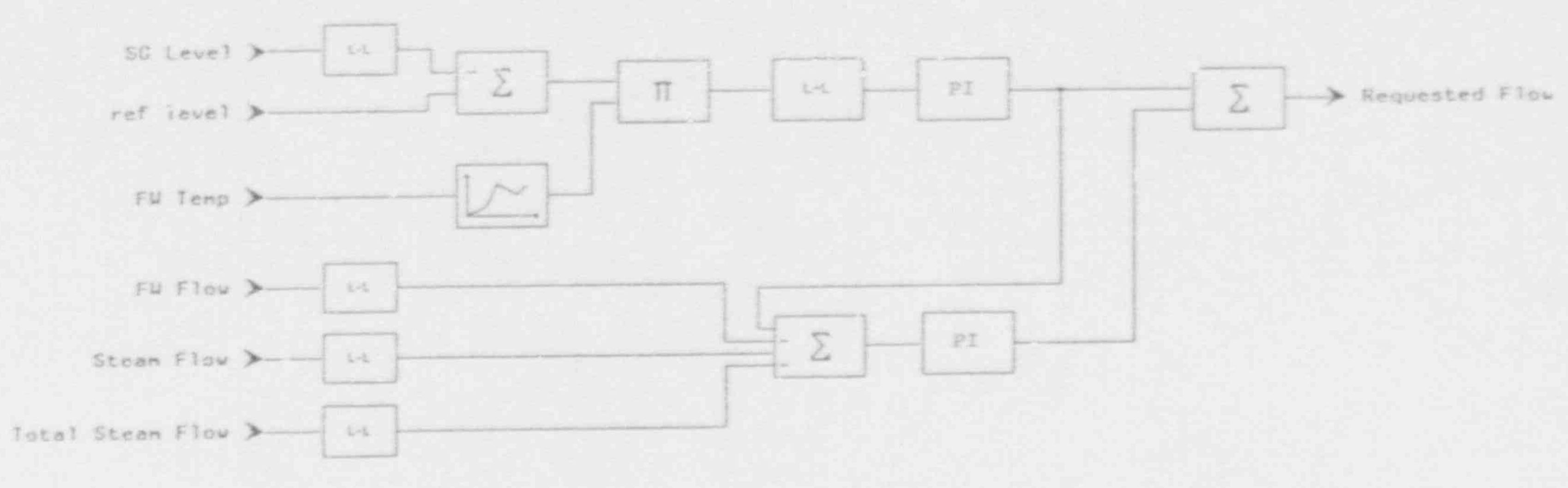
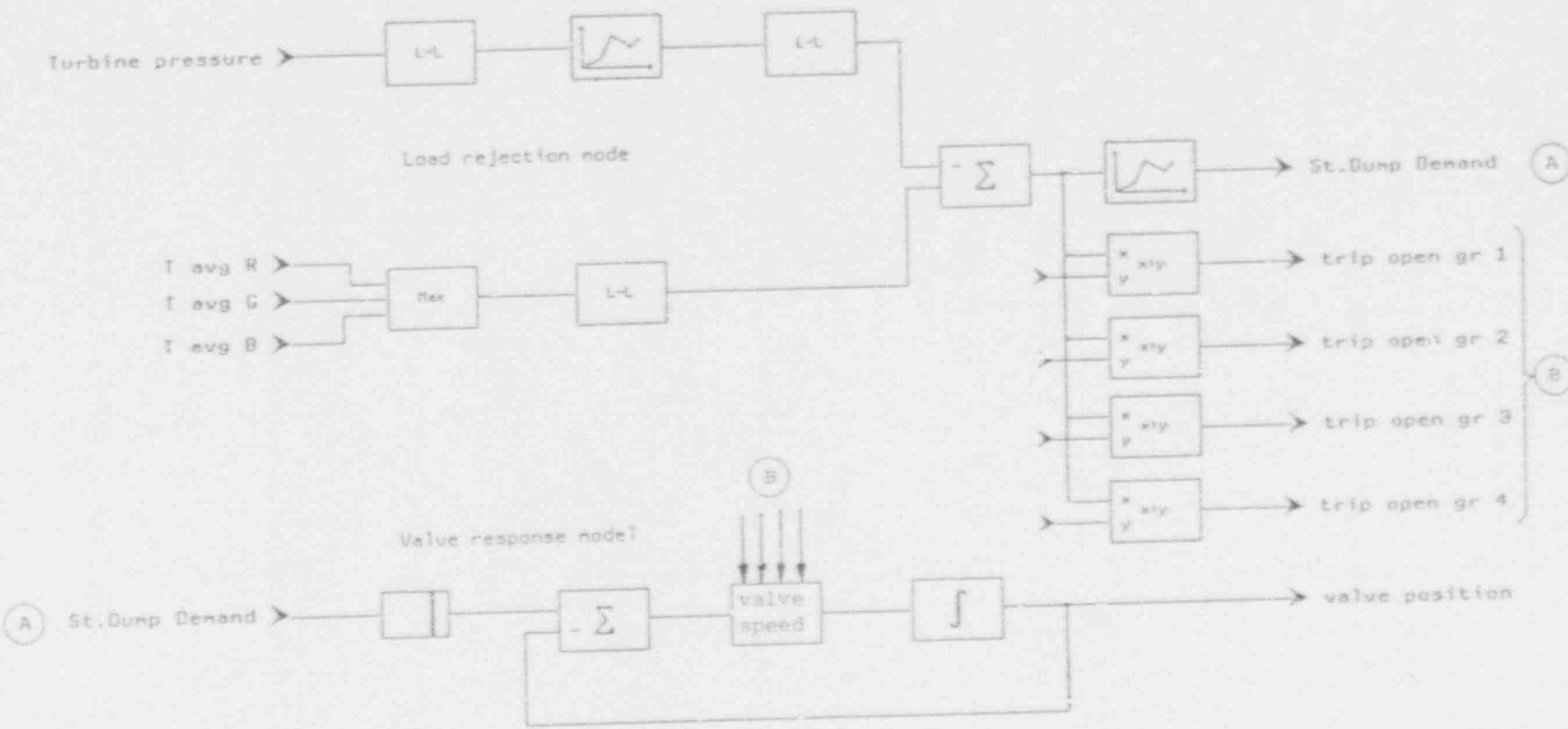


Fig. 4.4 RELAP Schema for steam dump control



D4-Dump

5. ANALYSIS AND DISCUSSION OF THE NUMERICAL RESULTS

The numerical simulation was performed over a period of 600 s, including a stabilisation period of 13 s prior to the initiation of the transient. This period of 600 s covers the most important phenomena which govern a successful transition from full power to house load. At 600 s, feeding of the steam generators switched over from bottom feed to top feed. This feature was not retained in the RELAP-5 model simulation. The most representative results are illustrated in figures 5.1 to 5.4 which compare the calculated data (solid lines) to the DAS recorded data (dash-dot lines) for some essential plant parameters, such as primary pressure (Fig.5.1), cold leg temperature (Fig. 5.2), main steam collector pressure (Fig. 5.3) and steam generator water level (Fig. 5.4).

An overall acceptable agreement is observed between calculated and measured data, which allows one to assert that the first two objectives are satisfied :

- RELAP-5 capability
- DOEL-4 model quality

However, some discrepancies are evident during the first 150 seconds, which need further investigation.

Between 0-13 s : Due to slight differences between the available steady state input deck for the plant, and the plant initial conditions, some fluctuations are observed in the numerical data which are believed to be of no importance for the remainder of the transient.

Between 13 and 30 s : An overshoot of about 2.5 bar (Fig. 5.1) in the primary pressure should be linked to an excessive rise in the cold leg temperature (Fig. 5.2) which is caused by the

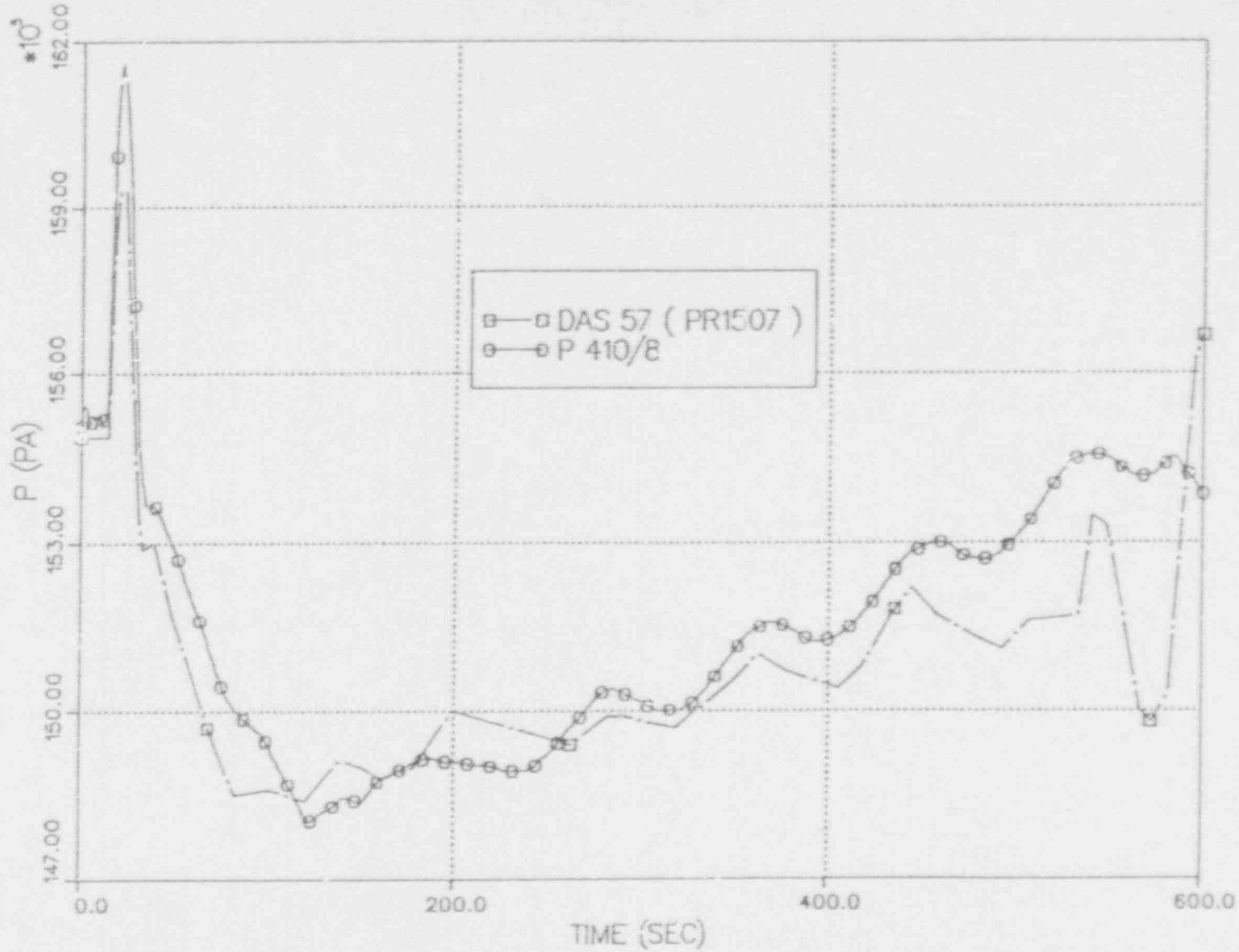
absence of structural heat absorption in the steam generator metal structures, when the pressure (and also the temperature) suddenly increases upon closure of the turbine admission valves. A good representation of the initial pressure overshoot is essential to justify the absence of reactor trip on high pressure (164 bar), or even the actuation of the pressuriser relief-valves.
For the same period, the calculated steam generator water level drop (Fig. 5.4) is much smaller than the recorded level fall. The location of the upper level tap (just above the upper deck plate) makes it very sensitive to acoustic pressure pulses which are generated in the main steam lines upon sudden closure of the turbine admission valves and reflected in the upper dome. Closer examination of the recorded data shows clearly sharp water level indication spikes which are just in opposite phase with the recorded pressure spikes (fig. 3.4). The crude nodalisation of the steamlines and the steam generator domes does not allow one to reproduce these acoustic phenomena. Discounting this effect, there still remains a level discrepancy of about 3.5% which should be attributed to the separator modeling deficiencies.

Fig. 5.4 also manifests an excessive level swell following the opening of the steamdump valves. This anomaly may be traced back to too strong a coupling between the water and vapour phase in the riser in the low void regimes, which causes excessive water entrainment into the separator region where the delta P level measurement is located. The deficiency of RELAP-5 to mimic the level swell phenomena correctly is attributed to the interfacial shear model (Ref. 3, 4). This anomaly feeds back, via the steam generator water level control system, to a reduction in the feedwater flow, which is reflected immediately by too high a cold leg temperature, as visible in Fig. 5.2 between 20 and 100 s. The design of the preheater section manifests a very tight thermal coupling between the feedwater flow rate and the cold leg temperature.

DOEL4 LOSS OF EXTERNAL LOAD TRANSIENT (23/11/85)

RELAP 5 SIMULATION (06/03/87)

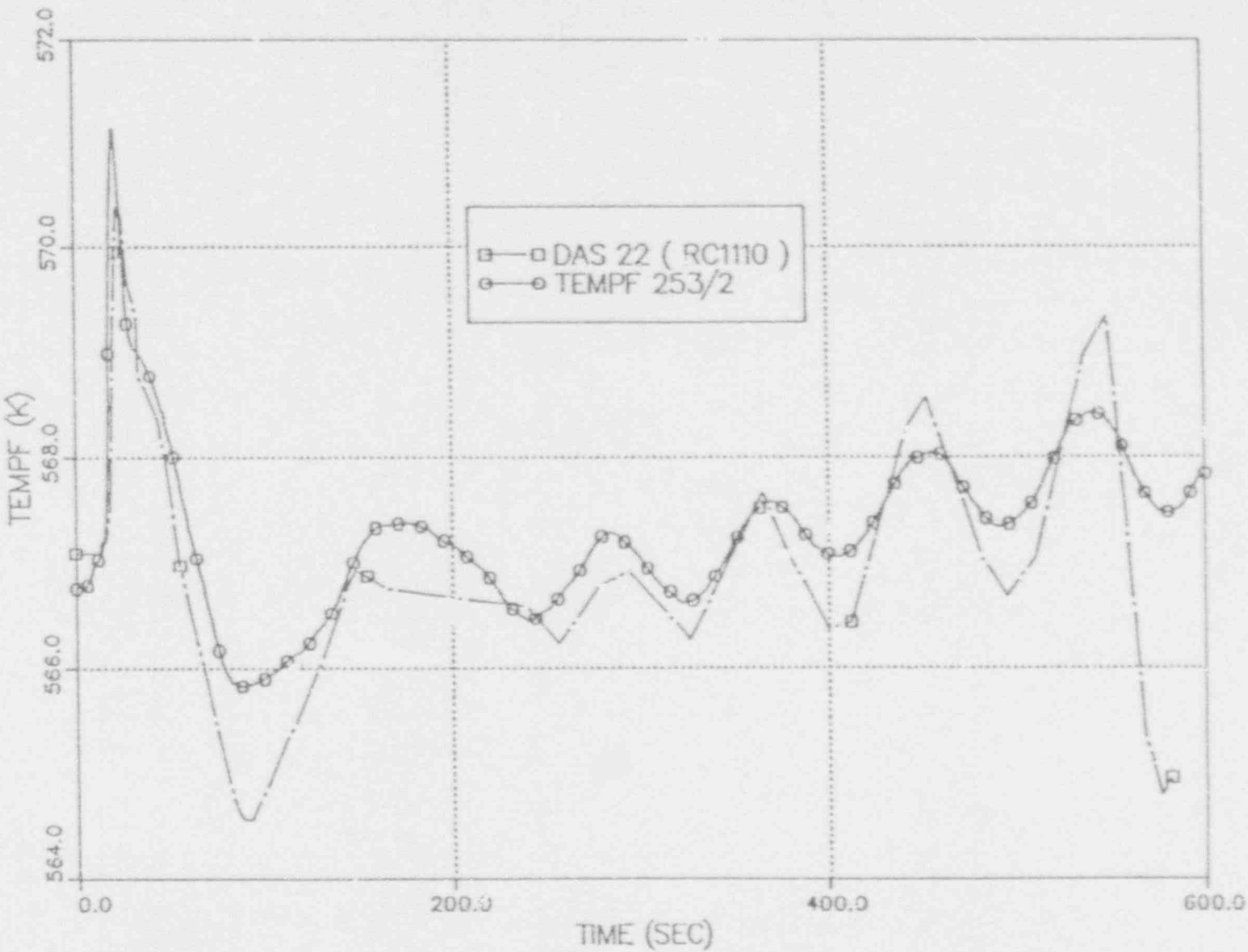
FIG. 5.1 PRESSURISER PRESSURE



DOEL4 LOSS OF EXTERNAL LOAD TRANSIENT (23/11/85)

RELAP 5 SIMULATION (0' /03/87)

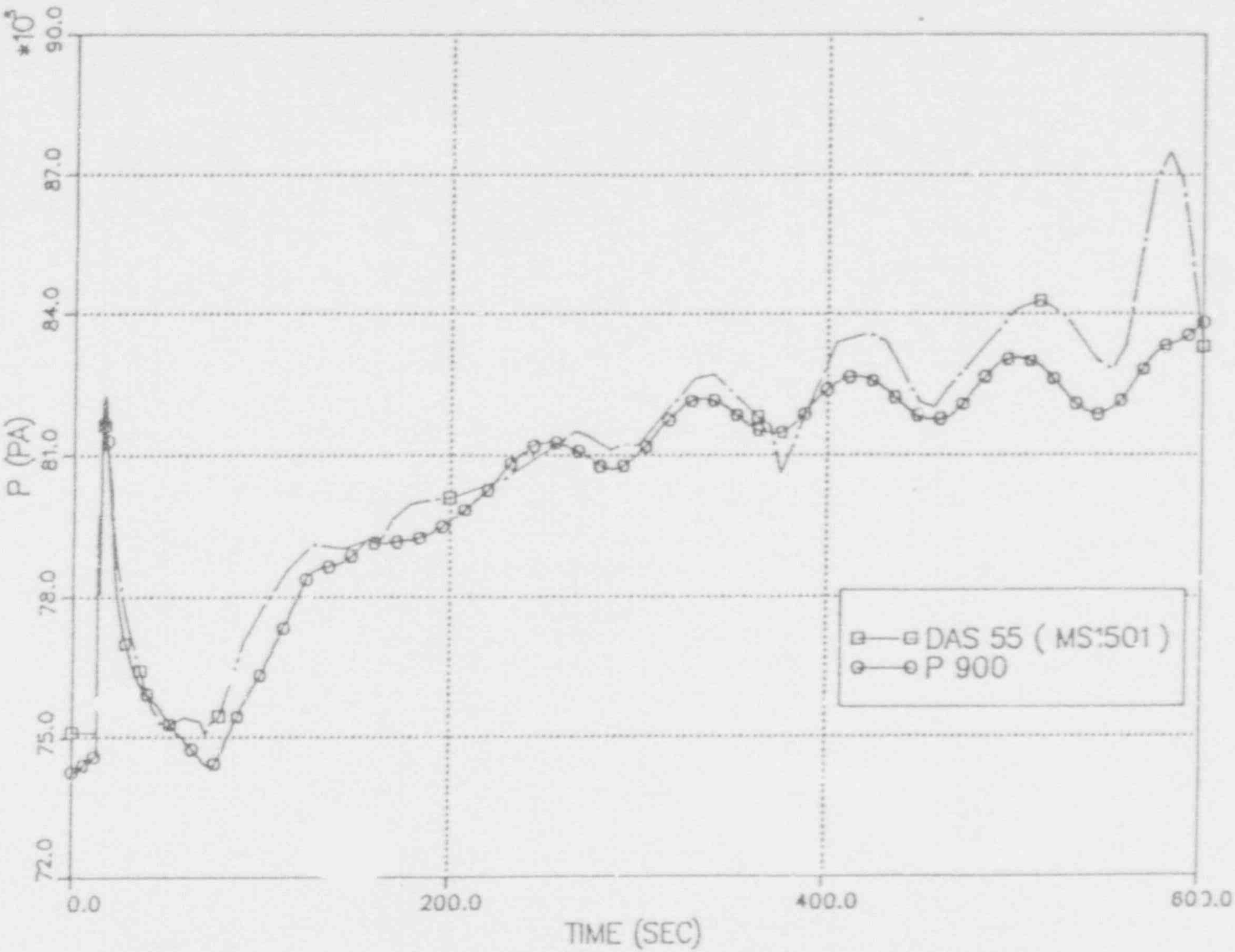
FIG. 5.2 COLD LEG TEMPERATURE



DOEL4 LOSS OF EXTERNAL LOAD TRANSIENT (23/11/85)

RELAP 5 SIMULATION ([illegible]/03/87)

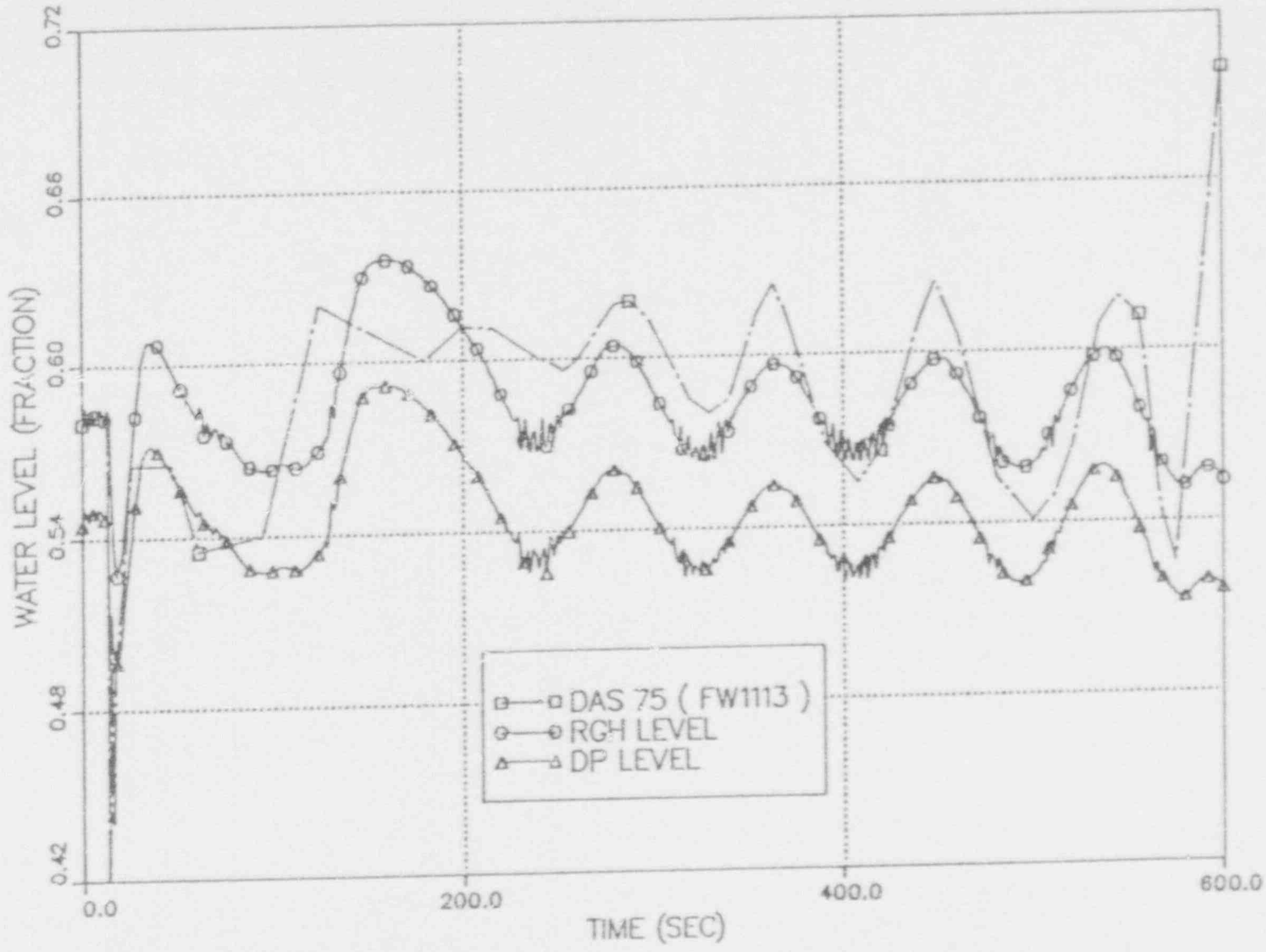
FIG. 5.3 STEAM COLLECTOR PRESSURE



DOEL4 LOSS OF EXTERNAL LOAD TRANSIENT (23/11/85)

RELAP 5 SIMULATION (06/03/87)

FIG. 5.4 STEAM GENERATOR WATER LEVEL (N.R)



6. ANALYSIS AND OPTIMISATION OF SOME CRITICAL PARAMETERS FOR "ISLANDING"

The most critical plant parameters liable to generate a reactor trip are

- Neutron flux variations :
  They are strongly influenced by the moderator negative reactivity coefficient, which aggravates the situation towards end of core life. To cope with this problem one could either relax the dϕ/dt limits or add compensation signals. (Ref. 5).
- Pressuriser pressure :
  To avoid reactor trip on high pressuriser pressure, or actuation of the pressuriser relief-valves which has become a Belgian utility requirement, a timely intervention of a large capacity steamdump is required, to reverse an initial pressurisation rate of the order of 1 bar/s.
- Steam generator water level :
  The measured and calculated data show oscillations building up during the first 10 minutes of the transient. These oscillations can be traced back to too strong a coupling of the requested feedwater flow on steam generator water level, combined with accumulating phase shifts due to either built-in delays, or natural response time of the system, which can give rise to a positive feedback mechanism as explained in chapter 3.

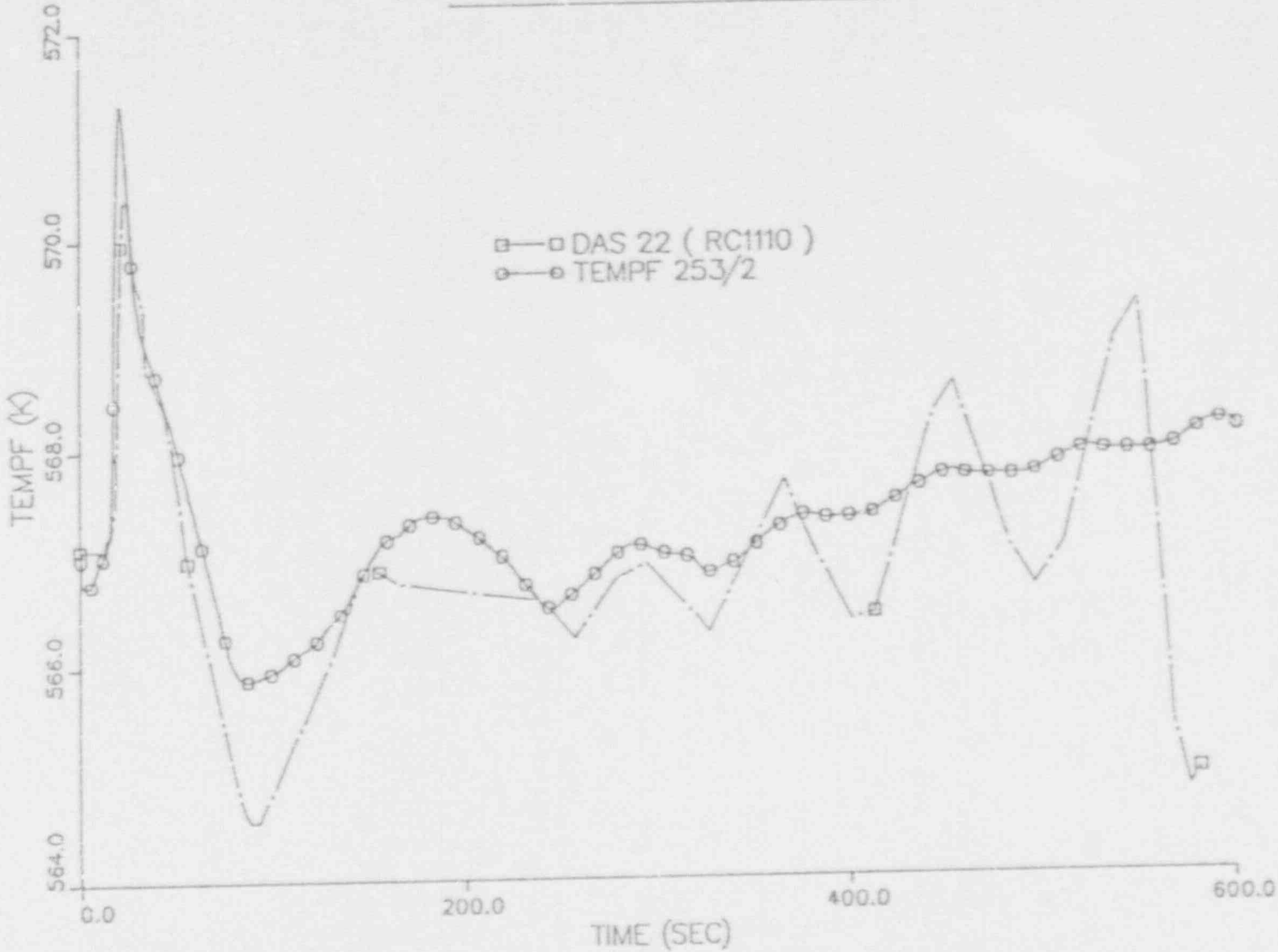
Due to the excessive level swell and shrink in the numerical results, especially in the short term, even stronger oscillations are observed in this period (see fig. 5.1 to 5.4). By reducing the gain by 20%, of the level controller on a level mismatch, the oscillations almost disappeared as is illustrated in fig. 6.1 (to be compared to fig. 5.2).

The level control system will be optimised to dampen out the level oscillations, conform to the numerical results, thus avoiding possible trip on steam generator level limits.

DOEL4 LOSS OF EXTERNAL LOAD TRANSIENT (23/11/85)

RELAP 5 SIMULATION (06/03/87)

FIG. 6.1 COLD LEG TEMPERATURE



7. CONCLUSION

A loss of external load transient test has been analysed on the basis of a high quality data acquisition system and a detailed numerical analysis by means of the code RELAP-5 MOD-2.

On the basis of some 240 recorded plant parameters, the analysis of the transient gives a clear picture of the behaviour of a great many components of the plant which are tied together either physically or by means of the plant control systems, and helps the designer in recognising the vital parameters which must be controlled to avoid reactor trip, within the limits of a safe operation of the plant.

A detailed numerical analysis of the transient by means of a best estimate analysis code RELAP-5 MOD-2 is presented, covering the most important plant components and systems, and complemented by imposed boundary conditions, taken from the recordings, when necessary.

This transient, as is usually the case for most plant transients in nuclear power plants, is highly conditionned by the secondary side steam and feedwater components and related systems such as the steam dump system and the feedwater level control system. Without a qualified data deck incorporating these components and control systems, it is practically excluded to simulate correctly the early part of most transients.

Comparison of recorded and calculated data show that

. The RELAP-5 code is capable to simulate the basic plant behaviour which allows a deeper insight in the physical phenomena.

. Some deficiencies are observed which can be explained

- by the absence of structural heat simulation of the steam generators (nodalisation problem)
- by acoustic phenomena which influence the steam generator level sensor (nodalisation problem)

- by excessive interphase drag in the steam generator at low void regimes (code weakness).

. Typical characteristic features of the plant can better be quantified such as

- strong thermal coupling between feedwater flow and cold leg temperature for a preheater type steam generator .
- Delay between feedwater flow variations and level swell and shrink response.

This study revealed some important feedback mechanisms which could lead to plant divergence, and hence reactor trip, and has shown that by optimising the feedwater flow controller gain, stability can be improved considerably.

One can also conclude that best estimate codes like RELAP-5, may be considered qualified to perform detailed optimisation studies of plant setpoints.

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The loss of external load test conducted on the DOEL-4 power plant has been analyzed on the basis of a high quality data acquisition system.

A detailed numerical analysis of the transient by means of the best estimate code RELAP5/MOD2 is presented.

The RELAP5 code is capable to simulate the basic plant behaviour.

Deficiences noted involved structural heat simulation, acoustic phenomena, and excessive interphase drag.

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