NUREG/IA-0109
UNID-91-08

International Agreement Report

Assessment of RELAP5/MOD2 Against a 10% Load Rejection Transient from 75% Steady State in theVandellós II Nuclear Power Plant

Prepared by
C. Llopis, Asociación Nuclear Vandellós
J. Perez, Consejo de Seguridad Nuclear
A. Casals, Asociación Nuclear Vandellós
R. Mendizabal, Consejo de Seguridad Nuclear

Asociación Nuclear Vandellós
c/Travesera de Les Corts 39-43.-08028 Barcelona
Consejo de Seguridad Nuclear
c/Justo Dorado 11.-28040-Madrid

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

May 1993

Prepared as part of
The Agreement on Research Participation and Technical Exchange
under the International Thermal-Hydraulic Code Assessment
and Application Program (ICAP)

Published by
U.S. Nuclear Regulatory Commission

Available from

Superintendent of Documents
U.S. Government Printing Office
P.O. Box 37082
Washington, D.C. 20013-7082

and

National Technical Information Service
Springfield, VA 22161

NUREG/IA-0109
UNID-91-08

International Agreement Report

Assessment of RELAP5/MOD2 Against a 10% Load Rejection Transient from 75% Steady State in the Vandellós II Nuclear Power Plant

Prepared by
C. Llopis, Asociación Nuclear Vandellós
J. Perez, Consejo de Seguridad Nuclear
A. Casals, Asociación Nuclear Vandellós
R. Mendizabal, Consejo de Seguridad Nuclear

Asociación Nuclear Vandellós
c/Travesera de Les Corts 39-43.-08028 Barcelona
Consejo de Seguridad Nuclear
c/Justo Dorado 11.-28040-Madrid

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

May 1993

Prepared as part of
The Agreement on Research Participation and Technical Exchange
under the International Thermal-Hydraulic Code Assessment
and Application Program (ICAP)

Published by
U.S. Nuclear Regulatory Commission

NOTICE

This report documents work performed under the sponsorship of the Consejo De Seguridad Nuclear of Spain. The information in this report has been provided to the USNRC under the terms of an information exchange agreement between the United States and Spain (Technical Exchange and Cooperation Agreement Between the United States Nuclear Regulatory Commission and the Consejo De Seguridad Nuclear of Spain in the field of reactor safety research and development, November 1985). Spain has consented to the publication of this report as a USNRC document in order that it may receive the widest possible circulation among the reactor safety community. Neither the United States Government nor Spain or any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability of responsibility for any third party's use, or the results of such use, or any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

F O R E W O R D

This report has been prepared by A.N. Vandellós in the framework of the ICAP-UNESA Project.

The report represents one of the assessment calculations submitted in fulfilment of the bilateral agreement for cooperation in thermalhydraulic activities between the Consejo de Seguridad Nuclear of Spain (CSN) and the United States Nuclear Regulatory Commission (USNRC) in the form of Spanish contribution to the International Code Assessment and Applications Program (ICAP) of the USNRC whose main purpose is the validation of the TRAC and RELAP system codes.

The Consejo de Seguridad Nuclear has promoted a coordinated Spanish Nuclear Industry effort (ICAP-SPAIN) aiming to satisfy the requirements of this agreement and to improve the quality of the technical support groups at the Spanish Utilities, Spanish Research Establishments, Regulatory Staff and Engineering Companies, for safety purposes.

This ICAP-SPAIN national program includes agreements between CSN and each of the following organizations:

- Unidad Electrica (UNESA)
- Unión Iberoamericana de Tecnología Eléctrica (UITESA)
- Empresa Nacional del Uranio (ENUSA)
- TECNATOM
- EMPRESARIOS AGRUPADOS
- LOFT-ESPAÑA

The program is executed by 12 working groups and a generic code review group and is coordinated by the "Comité de Coordinacion" This committee has approved the distribution of this document for ICAP purposes.

TABLE OF CONTENTS

ABSTRACT

The Consejo de Seguridad Nuclear (CSN) and the Asociación Nuclear Vandellós have developed a model of Vandellos II Nuclear Power Plant. The ANV collaboration consisted in the supply of design and actual data, the cooperation in the simulation of the control systems and other model components, as well as in the results analysis.

The obtained model has been assessed against the following transients occurred in plant:

- A trip from the 100% power level (CSN)

- A load rejection from 100% to 50% (CSN)

- A load rejection from 75% to 65% (ANV)

- A feedwater turbopump trip (ANV)

This copy is a report of the load rejection from 75% to 65% transient simulation. This transient was one of the tests carried out in Vandellós II NPP during the startup tests.

EXECUTIVE SUMMARY

The Vandellos II NPP, owned by ENDESA (72 %) and HIDROELECTRICA ESPAÑOLA (28 %), is located in Tarragona (Spain), by the Mediterranean sea. Its commercial operation started on March 3, 1988.

The Vandellos II NPP obtained the code RELAP5/MOD2 through the ICAP project. Then, Vandellos II NPP collaborated with the CSN simulating and analyzing two of the four transients the CSN had prepared for ICAP. However, Vandellos II NPP had already some experience in the use of this code due to previous collaboration agreements with the CSN.

This transient has been selected because of these two reasons:

- Enough plant data were available to check the results.

- The initial steady state is 75 %, instead of the habitual 100 %, this allows checking the model behavior in this new power level, so that the 100%, 75%, 65%, and 50% levels have been simulated and analyzed.

The main conclusions of this analysis are the following:

- Close agreement between results and data.

- The RELAP5/MOD2 is a valuable tool to simulate the primary side behavior.

- Basically, the differences between the model results and the plant data are due to the secondary side behavior during the transient: high sensibility to steam flow fluctuations, the indeterminateness of plant data and the accuracy of the reactor kinetics calculations (specially, the Doppler effect calculations).

INTRODUCTION

The Asociación Nuclear Vandellos II (ANV) decided, at the beginning of the commercial operation, to promote efforts aiming to study the following topics related to the simulation:

- The analysis of plant actual transients.
- The preparation for future IPE (Individual Plant Examination) works.
- The support to Vandellos II Operation Department in simulations of transients and accidents.
- The simulation of FSAR design accidents by means of a best estimate model, in order to compare them to the results obtained using conservative codes.
- The colaboration in the ICAP project with the analysis of two transients.

This work is one of the contributions of Vandellos II NPP (inside the UNESA group) to the ICAP project.

Other works have been carried out in order to support Vandellós II NPP Emergency Operation Procedures Rewiev and in the near term the contribution to the IPE is expected to begin, the experiencie gained during the collaboration in the ICAP project is considered to be very valuable for this contribution.

2. PLANT AND TRANSIENT DESCRIPTION

2.1. PLANT DESCRIPTION

Vandellos II is a three-loop PWR Nuclear Power Plant, designed by Westinghouse, with a nominal thermal power of 2775 MWt. It is equipped with three Westinghouse U-tube steam generators (model F) without preheaters. The feedwater is fed through the upper portion via J-tubes. The vessel is cold head type. The nominal electrical power is at present 992 MW.

Plant features are shown in table I.

2.2 PLANT DATA ACQUISITION SYSTEM DESCRIPTION

To record the main parameters of the plant, during the startup tests period, a temporary data acquisition system was installed. It consisted of a digital system with an up to 0.05 seconds and 146 signals trail capacity.

The recorded parameters depended on the test carried out.

The use of this system permitted a better and faster review of the test results. Therefore, once the nuclear plant tests had finished, Vandellos II NPP decided to install a final similar data acquisition equipment in order to interprete the plant behavior.

The availability of such a great number of signals has allowed th use of RELAP to check the control blocks partial performances, specially the feedwater control block and the rod control block, which are the main contributors to the plant evolution in this transient.

2.3 TRANSIENT DESCRIPTION

The test which is the subject of interest of the current simulation was performed on January 31, 1988 in Vandellos II NPP.

The objective of this test was to check that the reactivity introduced by the rod system was enough to absorb up to a 10 % load variation.

At the beginning of the test the plant was aproximately at 75 % power level steady state: the examinated and recorded parameters were the following:

- Electrical power
- Nuclear power
- Average temperature
- Steam pressure
- Pressurizer pressure
- Pressurizer level
- Steam generators levels.
- Feedwater flow
- Steam flow
- Feedwater turbopumps speed

The control of the following systems was in automatic mode:

- Rod control
- Steam generators level control.
- Turbopumps speed control
- Pressurizer pressure Control
- Steam-dump control
- D.E.H.

Once the power level was checked to remain stable, a 10 % load rejection was forced to happen by means of the D.E.H. at a 200 %/min rate, and a new power level at approximately the 65 % rated conditions was reached. The plant stabilized at this new level.

The expected evolution of the main parameters of the plant were the following:

- Primary pressure variation: 3.5 Kg/cm2
- Steam generators levels variation: ± 5 %
- Steam pressure variation: ± 1.75 %

The test acceptance criteria were the following:

- Neither the reactor nor the turbine should trip.
- The safety injection should not actuate.
- Neither the pressurizer safety valves nor the pressurizer relief valves should open.
- The control system must lead the plant to a new steady state automatically (no hand operation is allowed).
- During the 10 % load rejection the steam dump should not open (the transient must be absorbed by the rods system.
- The plant variables - such as the average temperature, the steam pressure, the pressurizer pressure, the feedwater flow, the steam flow, the levels - should not oscillate either continuosly o divergently.

The test results were satisfactory and in close agreement with the results obtained with RELAP5/MOD2.

3. MODEL DESCRIPTION

Figure 2 shows the nodalization used to simulate the primary system of the plant. It consists of 117 volumes, 122 junctions, 78 heat structures and 107 control variables.

A single loop which simulates the three loops of the plant has been implemented, the reason of this simplification is the reduction in the computing time; however, inaccuracy is not introduced with this simplification. A three loops model has been developed, and some tests have been carried out in order to compare the results obtained with this model to the results obtained with the single loop model. This tests have susbstantiated the single loop model validity for symmetric transients.

3.1. PRIMARY SYSTEM AND STEAM GENERATORS

This model includes the vessel, the primary loops, the steam generators, the pumps and the pressurizer.

The single loop model requires triplicating the volumes, the surfaces and heat structures transmission surfaces of the primary loops and steam generators.

The components of this model have been ellaborated and checked singly. For example, the steam generator was tested separately from other components and with the plant calorimetric data. The objective of this test was to adjust the primary - secondary heat transfer and the steam generator pressure. Another example is the comparison of the pressurizer behavior versus the plant spray and heaters performance.

The main components of the vessel are the following:

- Volume 504: Downcommer
- Volume 510: lower plenum
- Volume 520: from lower core support forging to lower core plate.
- Volume 530: core
- Volume 535: Between internals core barrel and baffles, and other core by-pass
- Volume 540: from upper core plate to mid loop elevation.
- Volume 550: from mid loop elevation to upper support assembly.
- Volume 560: from upper support assembly to internals flange elevation.
- Volume 580: upper plenum.

The vessel by-pass design flow has been adjusted through the volume 535 (core by-pass) and the volumes 502, 500 and 580 (vessel head cooling) by means of the energy loss coefficients.

The main components of the steam generator are the following:

Secondary side:

- Volume 200: Boiler
- Volume 220: expansion zone in the boiler upper portion.
- Volume 310: downcommer.
- Volume 230: turboseparators tubes lower portion
- Volume 240: turboseparators.
- Volume 280: turboseparators external zone.

Primary side:

- Volume 120 and 140: water boxes.
- Volume 130: steam generator tubes.

The recirculation ratio at 100%, 75% and 65% power levels has been substantiated to fit the design values.

Besides, vessel loops, steam generators and pumps pressure drops have been successfully checked.

The pressurizer has been divided into 10 volumes; two of these divisions match the pressurizer levels at 0% and 100% power levels.

The pressurizer relief and safety valve controls have been simulated, but not the valves themselves. This allows verifying that in this transient these valves do not open.

Talking about kinetics, the moderator temperature coefficient has been considered to be equal to zero, since when the test was carried out the core was in the beginning of life. An important observation is that the rod position modifies this coefficient, but RELAP5/MOD2 does not allow simulating this variation.

3.2. SECONDARY SYSTEM

In the secondary side, the three steam generators and the lines to the steam header, have been simulated as a single steam generator and a single line. The lenghts of the lines have been averaged since the three lines are not exactly equal.

The steam generator relief valves have been simulated, and the safety ones have been simulated as a single TMDPJUN. However, in this transient they do not open.

Downstream of the header, the four turbine admission valves have been simulated as a single valve. The steam-dump valves, which in plant are 12 gathered into 4 groups and which discharge into the three condenser shells, have been simulated as four valves to simulate four benches. The MSR's, ejectors, and turbopumps consumptions have also been simulated.

3.3. CONTROL SYSTEM

The primary basic controls can be groupped into four groups:

- Rod control
- Pressurizer pressure and level control
- Feedwater control
- Turbine and steam dump control

The four groups have been simulated according to the plant design. The plant actual control setting values during the test have been used as a setpoints.

The control blocks diagrams are shown in figures 3, 4, 5, and 6.

The availability of the signals continous recording system through the data acquisition system, has allowed checking all the control systems, and it has been observed that plant data are in close agreement with RELAP5/MOD2 results.

It has not been possible, however, matching the reactor kinetics to the plant reponse accurately. This and the steam flow are the main contributors to the RELAP5/MOD2 results and to the plant response mismatching in the load rejection from 75% to 65% transient simulation.

Another item that has not been possible to adjust accurately is the rod control system, since the pressure in the turbine first stage impulse chamber is the signal taken to measure the turbine power, and this signal is not available with RELAP5/MOD2 since the turbine has not been simulated.

With RELAP5/MOD2 the steam flow has been used as the secondary power measure, since is a variable more related to the impulse chamber pressure than the valve position. For example in figure 20 a porcentual comparison of the following variables is shown:

- Turbine valve position
- Impulse chamber pressure
- Steam flow
- Electrical power

and the differences among them can be evaluated.

A transient calculation has been carried out considering the impulse chamber pressure as a boundary condition, and the results have been much satisfactory. Nevertheless, this method has not been used since it would be useful only in those cases which have occurred in plant and this signal is available.

4. INITIAL STEADY STATE CALCULATIONS

Starting from a 100% initial level and simulating a load rejection transient to the 75 %, by closing the turbine regulating valve, a new steady state has been reached by means of the control systems, which have led the model to the new conditions. The INPUT has been reinitialized with the obtained data. In fact, this means having simulated a new case of load rejection from 100 % to 75 %.

The main parameters values obtained with RELAP5/MOD2 have been compared to the plant actual values, as shown in Table III.

5. TRANSIENT CALCULATION AND COMPARISON VS ACTUAL DATA

The main purpose of this transient simulation is to check the model at different power levels. Moreover, being the transient developement governed by the controls - specially by the rod control, main feedwater control, and the pressurizer pressure and level control - another purpose is checking the behavior of these controls.

The simulation of this transient has been carried out starting from the initial steady state, reducing the turbine flow and allowing the control systems to perform automatically. In plant, this flow reduction is governed by the D.E.H., which commands a 200 % / min. load rejection, so that in 3 seconds the turbine admission valves are about to be at their new position.

As a result of the order given by the D.E.H. and the new turbine valve position, the steam flow diminishes (fig.17). The second immediate consequence is the reduction in the first stage impulse chamber pressure, which is being used as a turbine power reference by the rod control system.

In fact, a close relationship exists between the steam flow and the impulse chamber pressure, but it is not a linear relation during a transient. In our model, the turbine steam flow has been adopted as the turbine power measure.

Owing to the variation in the energy production and evacuation balance of the primary side, new variations of pressures and temperatures occur. The reactor will attempt to adapt the new power level, by means of the rod control system, which will move the rods as a result of the power error and the average temperature (fig.7).

The nuclear flux decreases quickly (fig.8) and, from there on, the reactor will adopt a new average temperature according to the temperature program.

The 63% and 65% power levels difference between plant and RELAP values, respectively at the end of the simulation, is caused by the error margin of the plant power measurement. A thermal balance carried out at the end of the transient points out that the RELAP value is more reliable than the plant value.

The cold leg, hot leg and average temperatures are shown in figures 10, 11 and 12. At the beginning, the average temperature increases because of the reactor power production and the steam generator power evacuation mismatch, later, once this mismatch has been overcome, the average temperature decreases down to the new level because of the nuclear flux reduction.

Figure 13 has been included to show the primary delta Temperature evolution as a significant indicator of the primary power evolution. This delta Temperature allows a close adjustement with RELAP, which indicates that the level reached is the same than the plant one. However, as it has been seen before, it has not been possible to adjust the nuclear flux to the plant values. These two points prove that the plant nuclear flux measurement has an error margin.

The primary pressure evolution is similar to the average temperature one, and is shown in figure 14. The same occurs with the pressurizer level (fig.15), which is modified essentially by the density variation in the primary side.

It can be observed an initial pressure peak which is higher in RELAP than in plant. This fact may be caused by the spray efficiency, since the RELAP code does not allow simulating the actual physical phenomenon.

The primary flow (fig.9) evolution is due to the density variation too. It can be observed that plant measures have a lot of noise, and this happens because it is calculated as elbow taps pressure difference, so that small variations in the fluide regime imply small (0.1%) variations in the measurements.

The steam generators pressure (fig. 18) increases initially because of the turbine valve closure, but it stabilizes when the heat transmission through the steam generator tubes and the evacuation through the turbine balance.

The steam generators level (fig.19) has a sudden fall on starting the transient because of the steam binding but it recovers on increasing the feedwater flow (fig.16).

The curling that can be observed in the steam generators level is supposed to be due to void fraction variations owing to any numerical inestability of the RELAP5/MOD2.

The outstanding adjustement of the feedwater control system with RELAP5/MOD2 has been achieved owing to the availability of intermediary recorded signals of the plant control systems, which have allowed working with partial models.

The sole boundary conditions which have been imposed to the model in this transient are the turbine steam flow and the pressure and temperature of the feedwater header, since the secondary side next to the valves has not been simulated by the moment.

6. RUN STATISTICS

This case has been simulated on an IBM 3090, owned by ENDESA, located in Madrid.

RELAP5/MOD2 cycle 36.04 has been used in the version adapted by ISPRA the 1st of November, 1987.

The CPU TIME / REACTOR TIME ratio has been 4.31.

The time step has been constant (0.05 sec.) during all the transient.

The run statistics are shown in Table IV.

7. CONCLUSIONS

The Dispose of such a great number of digital signals from the data acquisition system has allowed carrying out simulations, in which plant signals have been supplied to the control blocks in order to check each block separately. These checks have permitted and supported the individual control blocks assessment prior to assemble the complet model. This process has been shown to conduct to very accurate results predictions.

In this load rejection transient, in which a control rods insertion occurs, the reactor behaves in a different way depending on the axial zone we consider. To reproduce the final power level with RELAP correctly, it has been necessary to modify the Doppler coefficient design values. This fact lead us to conclude that the punctual kinetics is a conservative model that has to be corrected for this kind of transients.

The power initial level in this transient is the 75%, instead of the habitual 100%, and this has allowed checking the model behavior in this new power level. The obtained results are very satisfactory, as shown in Table III.

The evolution of most of the RELAP main variables in this transient are in close agreement with plant data. Besides, in these cases in which mismatches can be observed, the differences are within the plant instrumentation error margins.

After having assessed this model against the following transients:

- A trip from the 100% power level (CSN)

- A load rejection from 100% to 50% (CSN)

- A load rejection from 75% to 65% (ANV)

- A feedwater turbopump trip (ANV)

the RELAP5/MOD2 is considered to be a valuable tool for transient simulations.

8. BIBLIOGRAPHY

Precautions, Limitations and Setpoints of Vandellós II, rev. 6.

Estudio final de seguridad de Vandellos II

Setpoints Study Vandellos unit 2 (WENX/85/38)

RELAP5/MOD2 Code Manual

Project Information Package

Steam Generator thermohydraulic analysis

Copia oficial de prueba PAN 55

Copia oficial prueba eficacia calentadores y spray

Copia oficial prueba reactividad de barras

Datos sistema Adquisición de Datos de planta

9. INDEX OF TABLES

TABLE I

MAIN CHARACTERISTICS OF VANDELLOS II NPP

- THERMAL REACTOR POWER (MWt)	2775
- ELECTRICAL POWER (MWe)	992
- FUEL	UO2
- NUMBER OF ASSEMBLIES	157
- NUMBER OF COOLANT LOOPS	3
- CLADDING TUBE MATERIAL	ZIRCALOY 4
- ABSORBER MATERIAL	B4C + Ag-In-Cd
- REACTOR OPERATING PRESSURE (MPa)	15.4
- COOLANT TEMPERATURE AT NO LOAD (°K)	564.8
- COOLANT AVERAGE TEMPERATURE AT 100% (°K)	582.3
- STEAM GENERATOR	WESTINGHOUSE TIPE F
- NUMBER OF TUBES IN STEAM GENERATOR	5626
- TOTAL TUBE LENGHT (m.)	98759
- INNER DIAMETER TUBES (m.)	0.0156
- TUBE MATERIAL	INCONEL
- PUMPS TYPE	WESTINGHOUSE D 100
- DISCHARGE HEAD OF PUMPS (bar.)	18.8
- DESIGN FLOW RATE (m3/s)	6.156
- SPEED OF PUMPS (rad/s)	155

- PRIMARY VOLUME (m3) 106.19
- PRESSURIZER VOLUME (m3) 39.65
- HEATING POWER OF THE HEATERS RODS (KW) 1400
- MAXIMUM SPRAY FLOW (Kg/s) 44.2
- STEAM MASS FOOW RATE AT 100 % (Kg/s) 1515

TABLE II

MAIN EVENTS

TIME	EVENT
0.0 SEC.	BEGINNING OF REJECTION LOAD FROM 75 % TO 65 %
3.0 SEC.	TURBINE VALVE AT A NEW POSITION
APROX. 600 SEC.	REACHED NEW STEADY STATE OF 65 % OF POWER

TABLE III

COMPARISON BETWEEN RELAP5/MOD2 VALUES AND ACTUAL DATA

VARIABLE		RELAP5/MOD2	PLANT	
NUCLEAR POWER	(%)	73.9	73.8	
PRIMARY MASS FLOW RATE	(Kg/s)	14907	14907	(1)
COLD LEG TEMPERATURE	(°K)	564.3	564.5	
HOT LEG TEMPERATURE	(°K)	589.5	589.7	
AVERAGE TEMPERATURE	(°K)	576.9	577.1	
DELTA TEMPERATURE	(°K)	25.2	25.2	
PRESSURIZER PRESSURE	(MPa)	15.50	15.49	
PRESSURIZER LEVEL	(%)	50.0	49.9	
FEEDWATER MASS FLOW RATE	(Kg/s)	1120	1117	
STEAM GENERATOR PRESSURE	(MPa)	6.86	6.86	
STEAM GENERATOR LEVEL N.R.	(%)	50.0	50.3	
RECIRCULATIO RATIO		3.62	3.6	(2)

(1) CALCULATED DATA
(2) DESIGN DATA

TABLE IV

DESCRIPTION OF RELAP5/MOD2 VARIABLES		FIGURES
CNTRLVAR 340	ROD POSITION	7
CNTRLVAR 301	NUCLEAR POWER (PERCENT)	8
MFLOWJ 180010000	PRIMARY MASS FLOW RATE	9
CNTRLVAR 328	TEMPERATURE AT THE COLD LEG	10
CNTRLVAR 327	TEMPERATURE AT THE HOT LEG	11
CNTRLVAR 330	AVERAGE TEMPERATURE	12
CNTRLVAR 947	DELTA TEMPERATURE	13
P 415090000	PRESSURIZER PRESSURE	14
CNTRLVAR 350	PRESSURIZER LEVEL	15
MFLOWJ 325000000	FEEDWATER MASS FLOW RATE	16
MFLOWJ 715000000	TURBINE STEAM MASS FLOW RATE	17
P 600010000	STEAM GENERATOR PRESSURE	18
CNTRLVAR 203	STEAM GENERATOR LEVEL (N.R.)	19

TABLE V

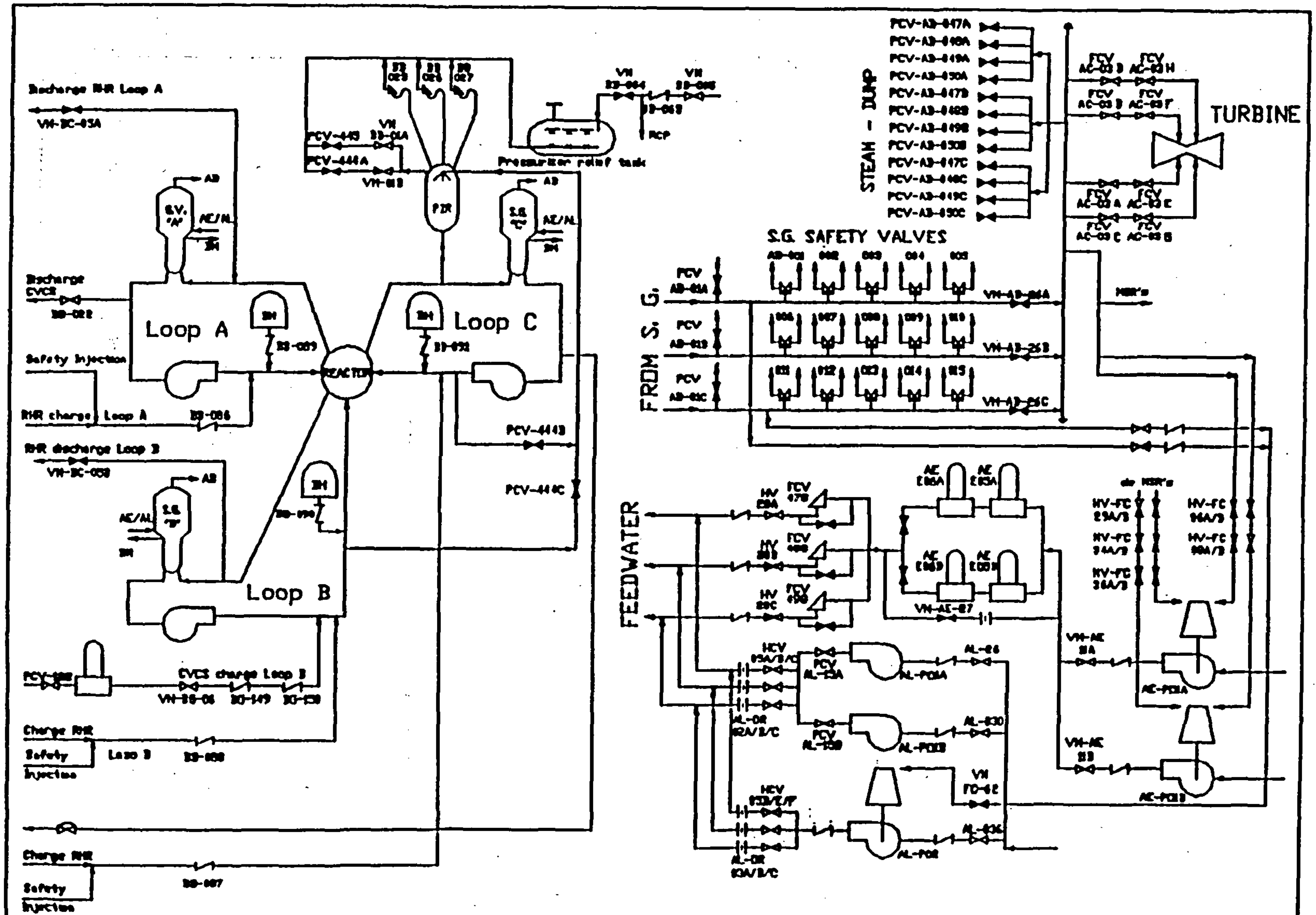
RUN STATISTICS

COMPUTER	IBM 3090
TRANSIENT TIME	800 sec
CPU TIME	3452 sec
C (TOTAL NUMBER OF ACTIVES VOLUMES)	117
DT (TOTAL NUMBER OF TIME STEPS)	16000

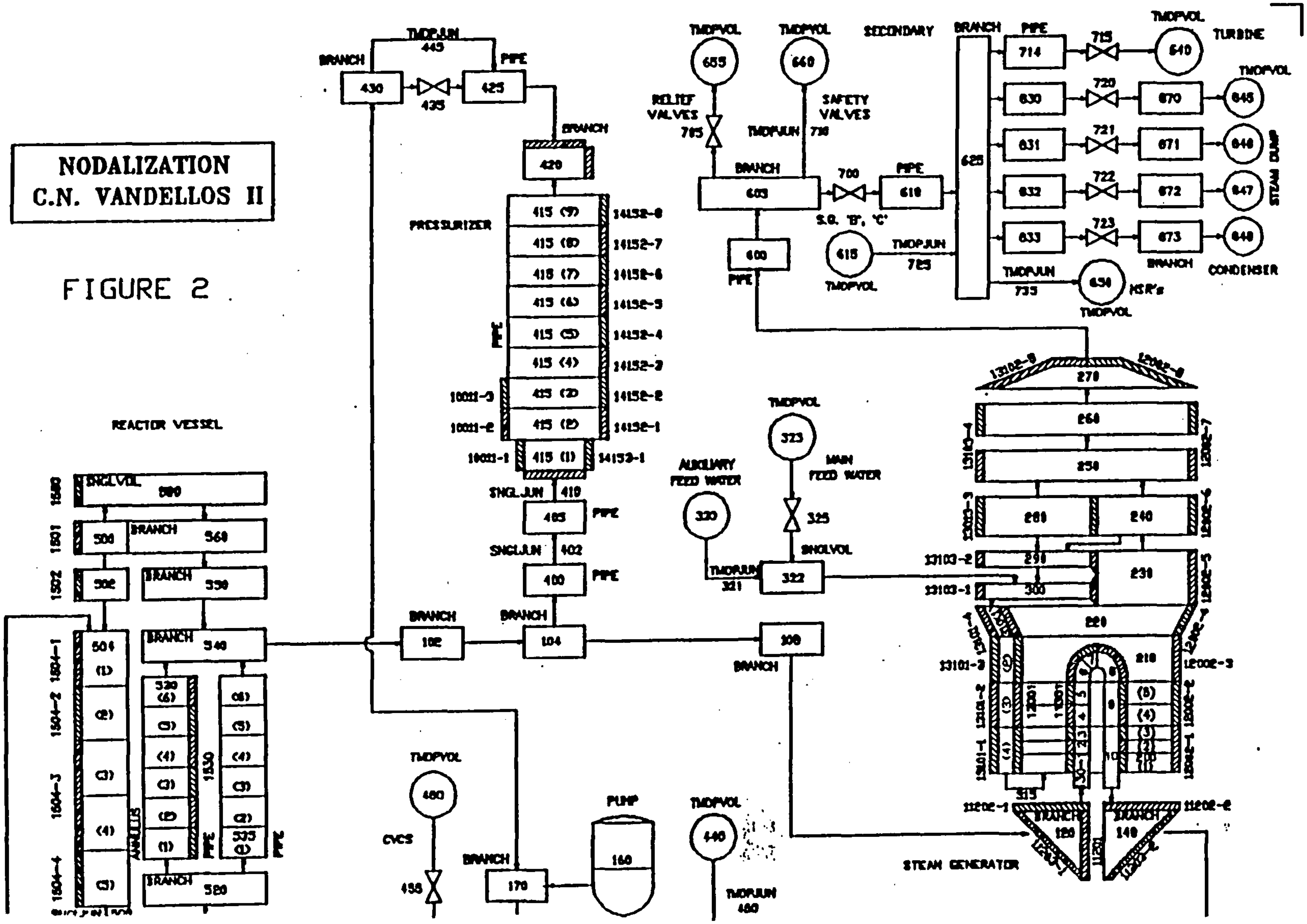
$$\frac{CPU * 1000}{C * DT} = 1.84$$

CPU TIME / TRANSIENT TIME	4.31

10. INDEX OF FIGURES



VANDELLOS II PRIMARY AND MAIN ASSOCIATED SYSTEMS FIG. 1



NODALIZATION
C.N. VANDELLOS II

FIGURE 2

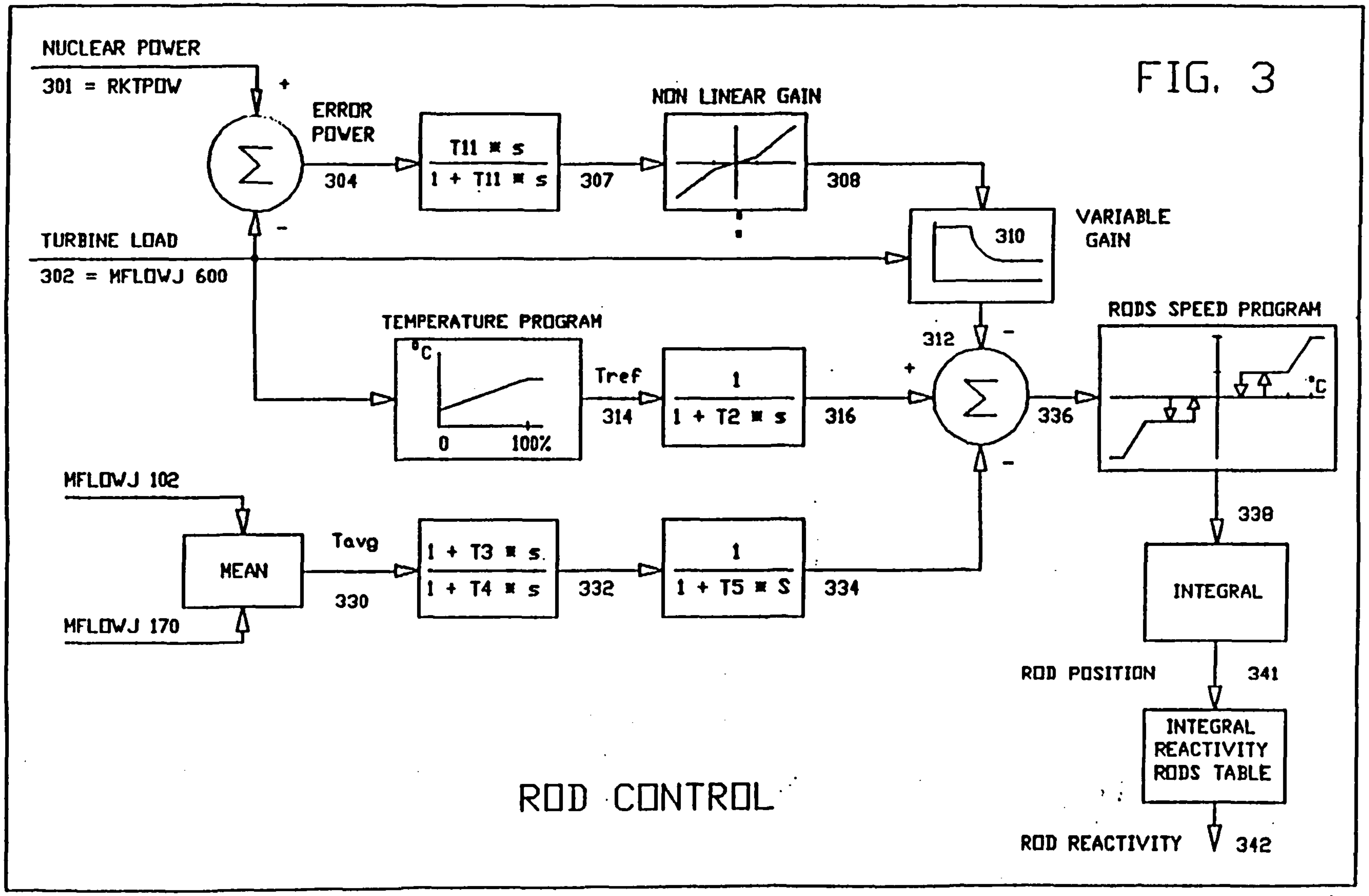
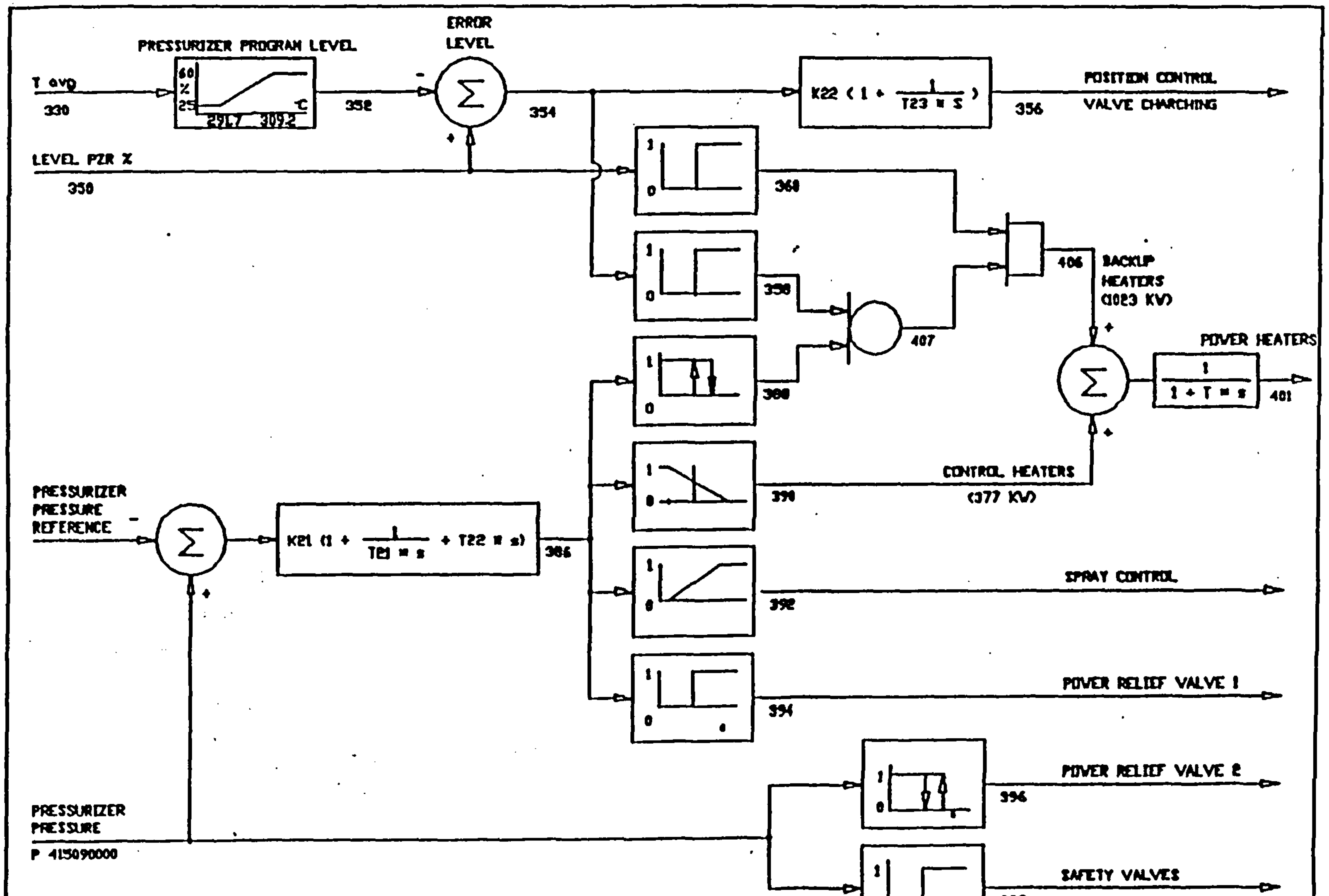


FIG. 3

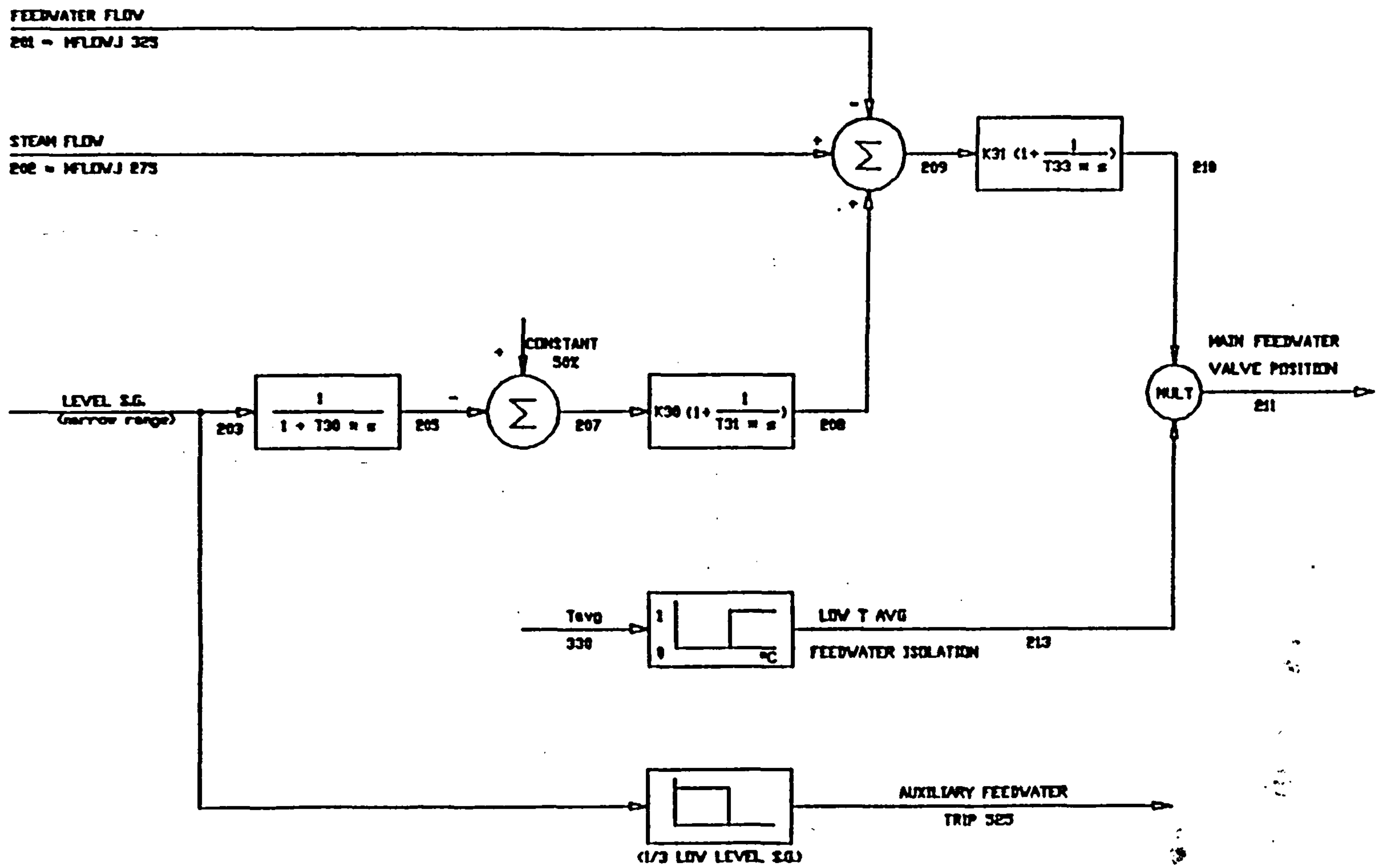
ROD CONTROL

34

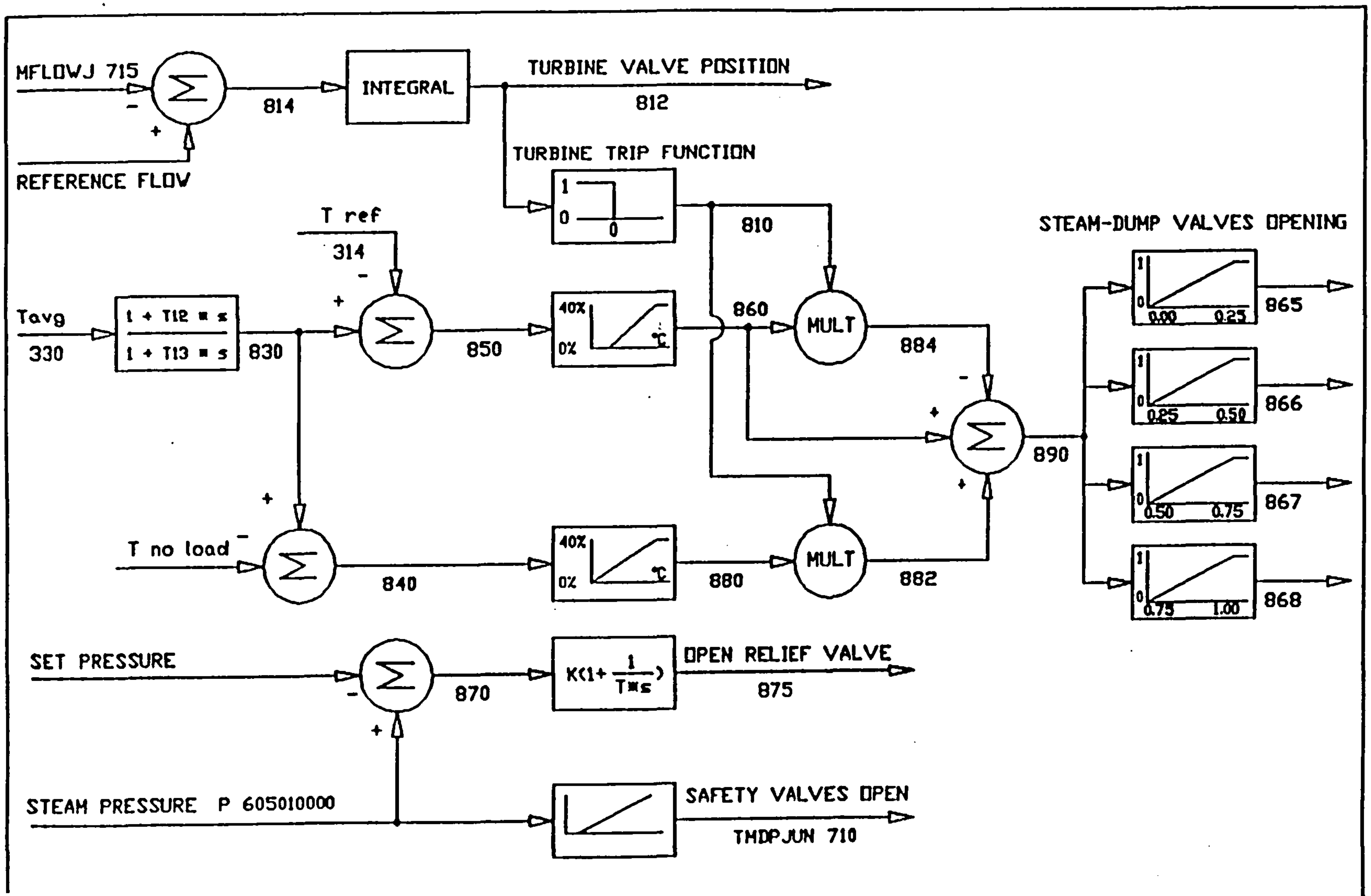


PRESSURIZER PRESSURE AND LEVEL CONTROL

FIG. 4



MAIN AND AUXILIARY FEEDWATER CONTROL FIG. 5



TURBINE AND STEAM DUMP CONTROL FIG. 6

FIG 7: ROD POSITION

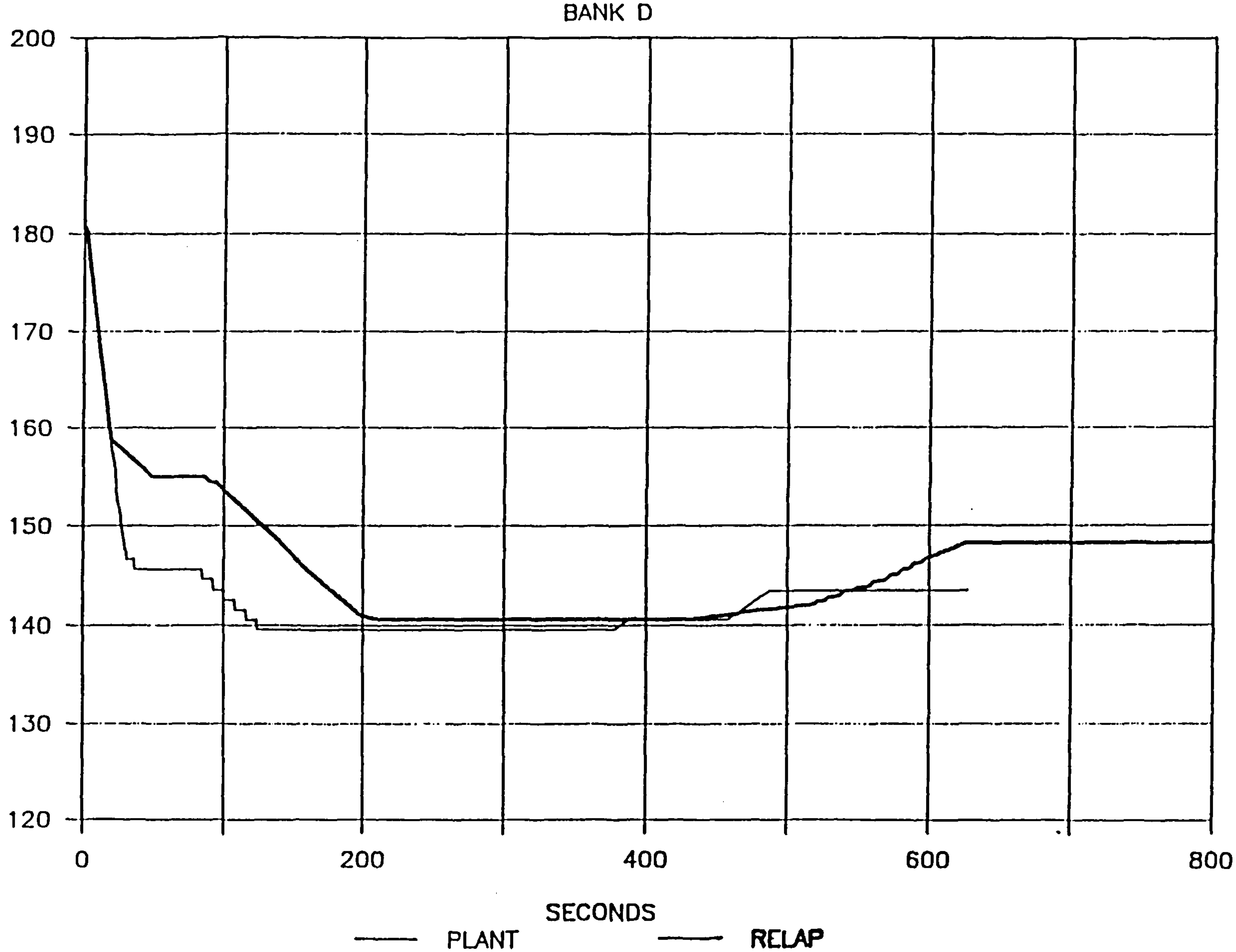


FIG 8: REACTOR POWER

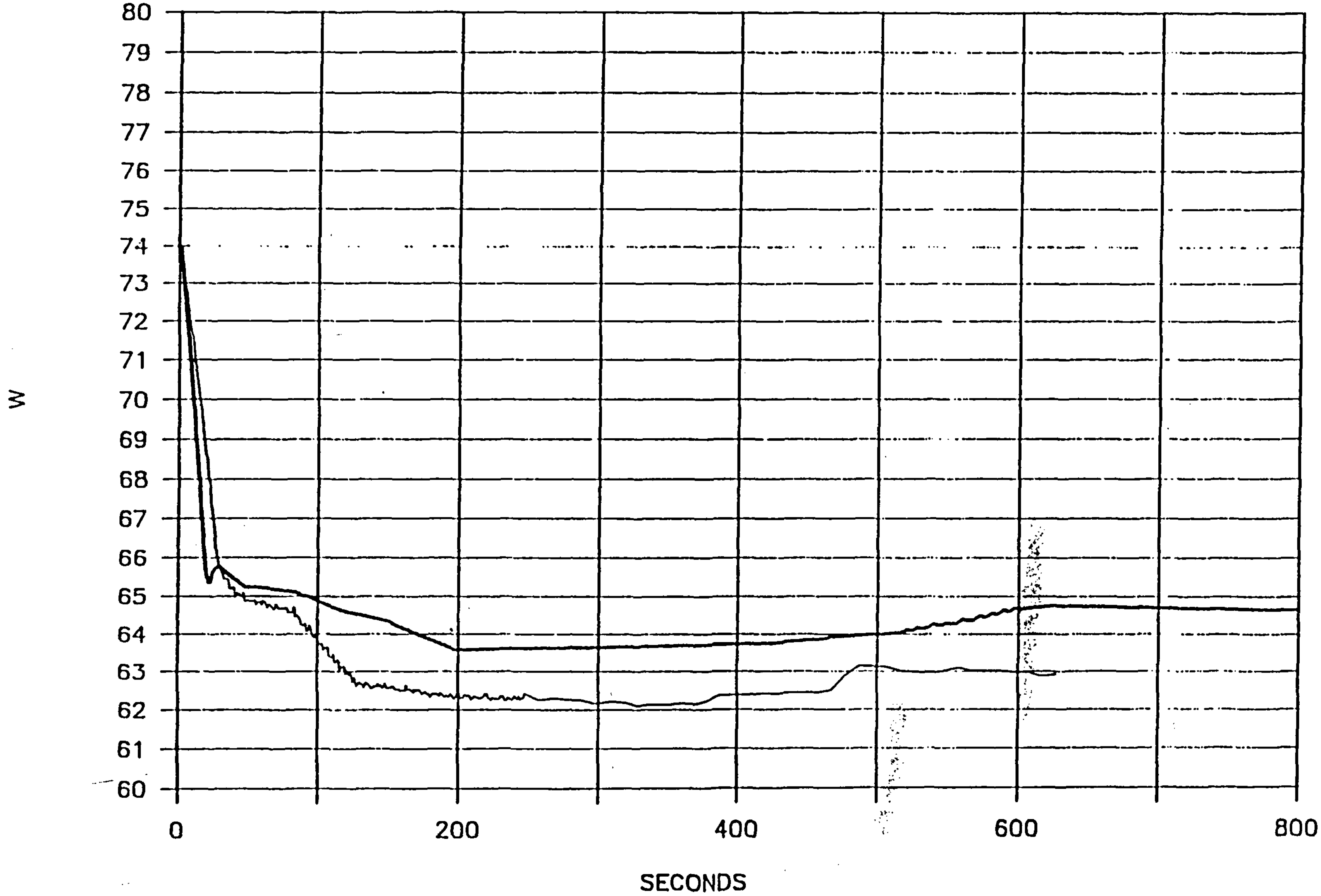


FIG 9: PRIMARY MASS FLOW RATE

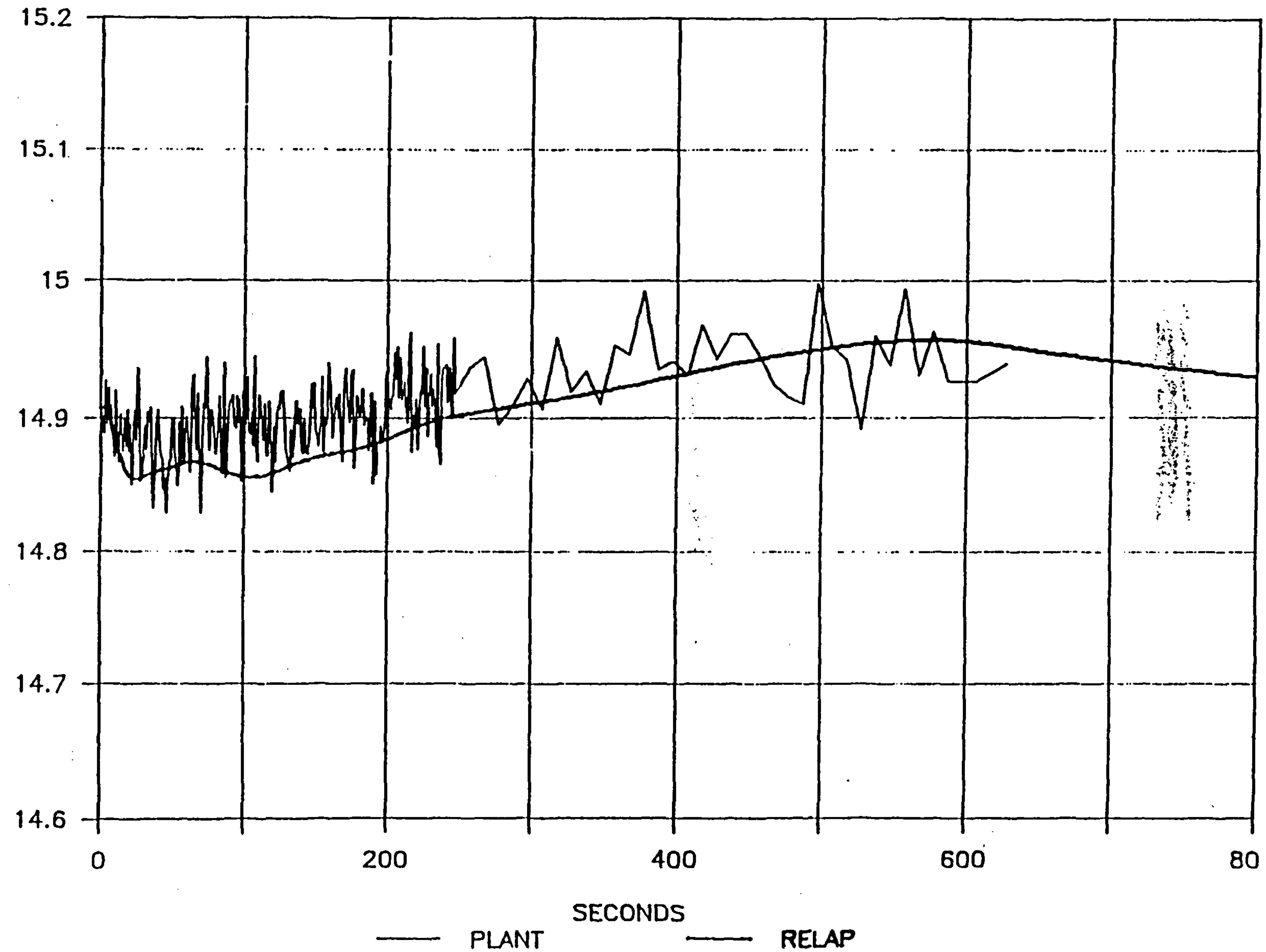


FIG 10: TEMPERATURE AT THE COLD LEG

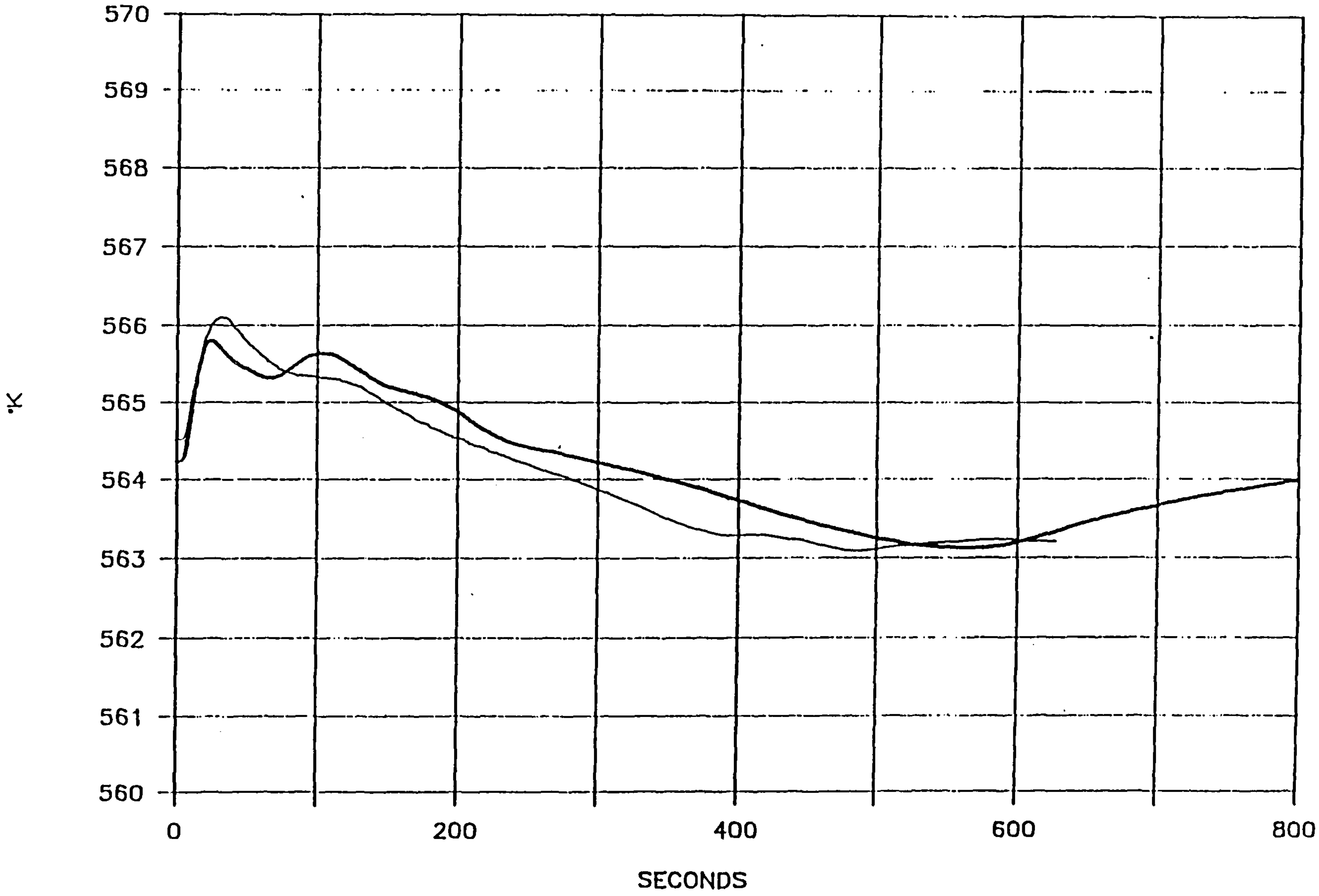


FIG 11: TEMPERATURE AT THE HOT LEG

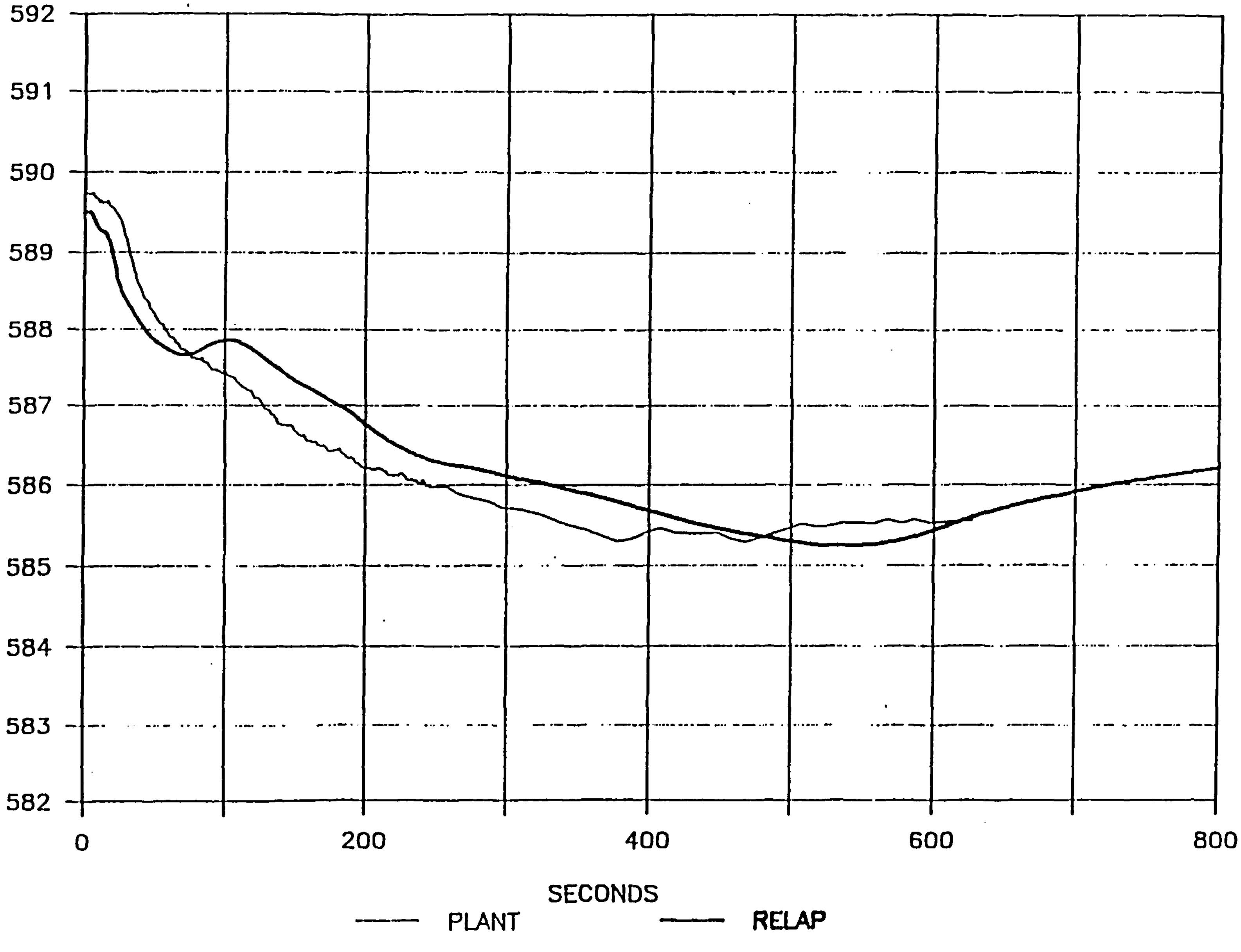


FIG 12: PRIMARY AVERAGE TEMPERATURE

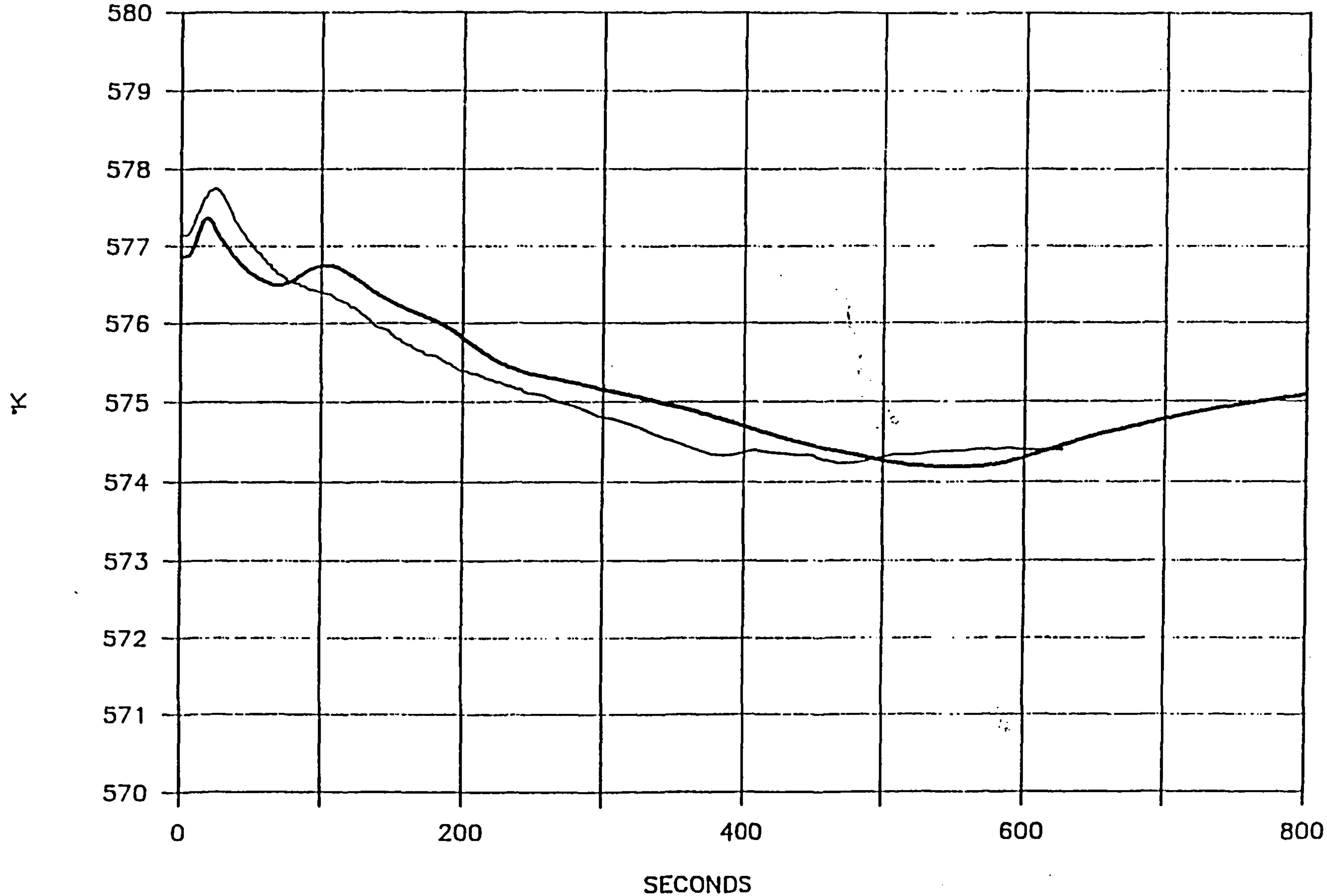
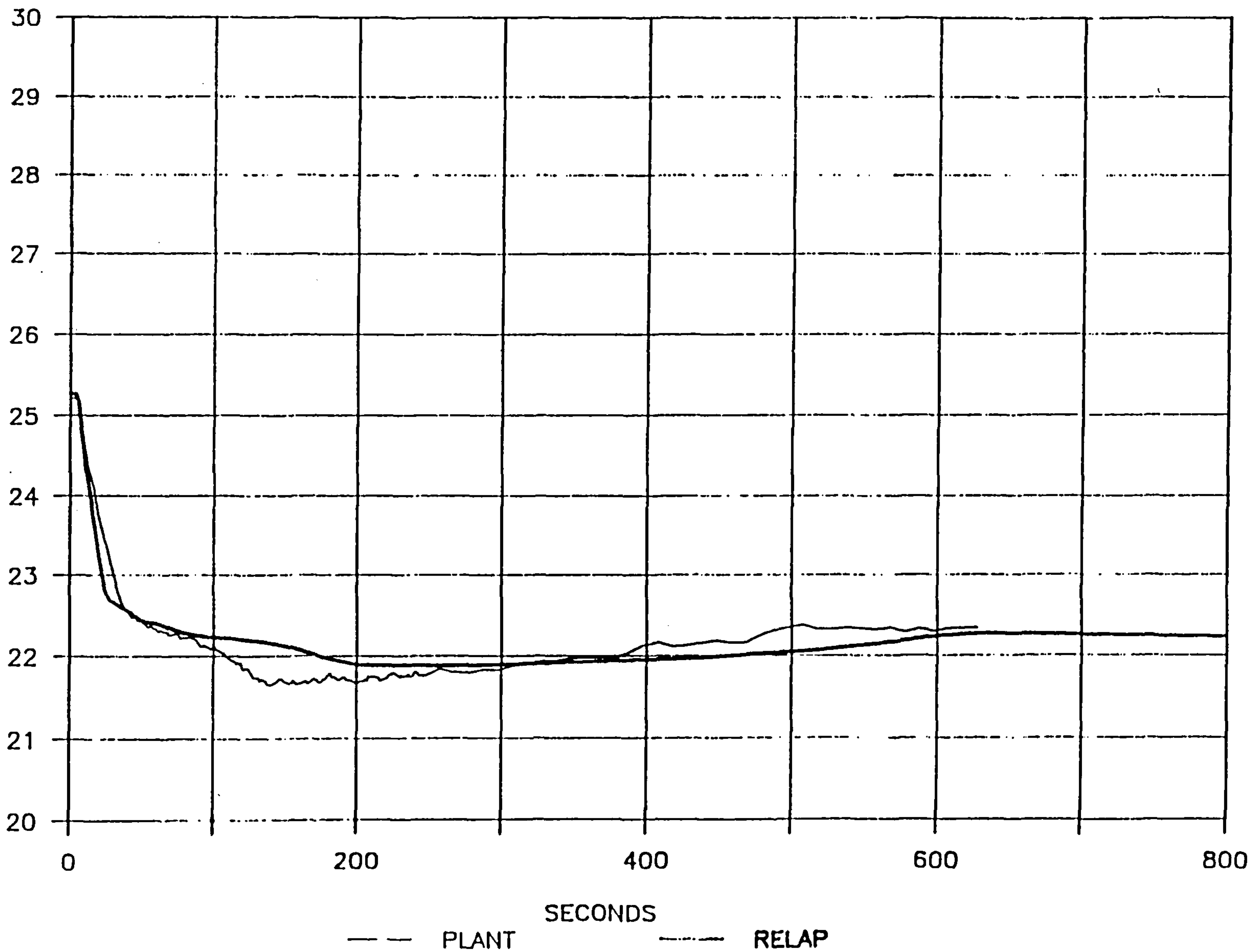


FIG 13: VESSEL DELTA TEMPERATURE



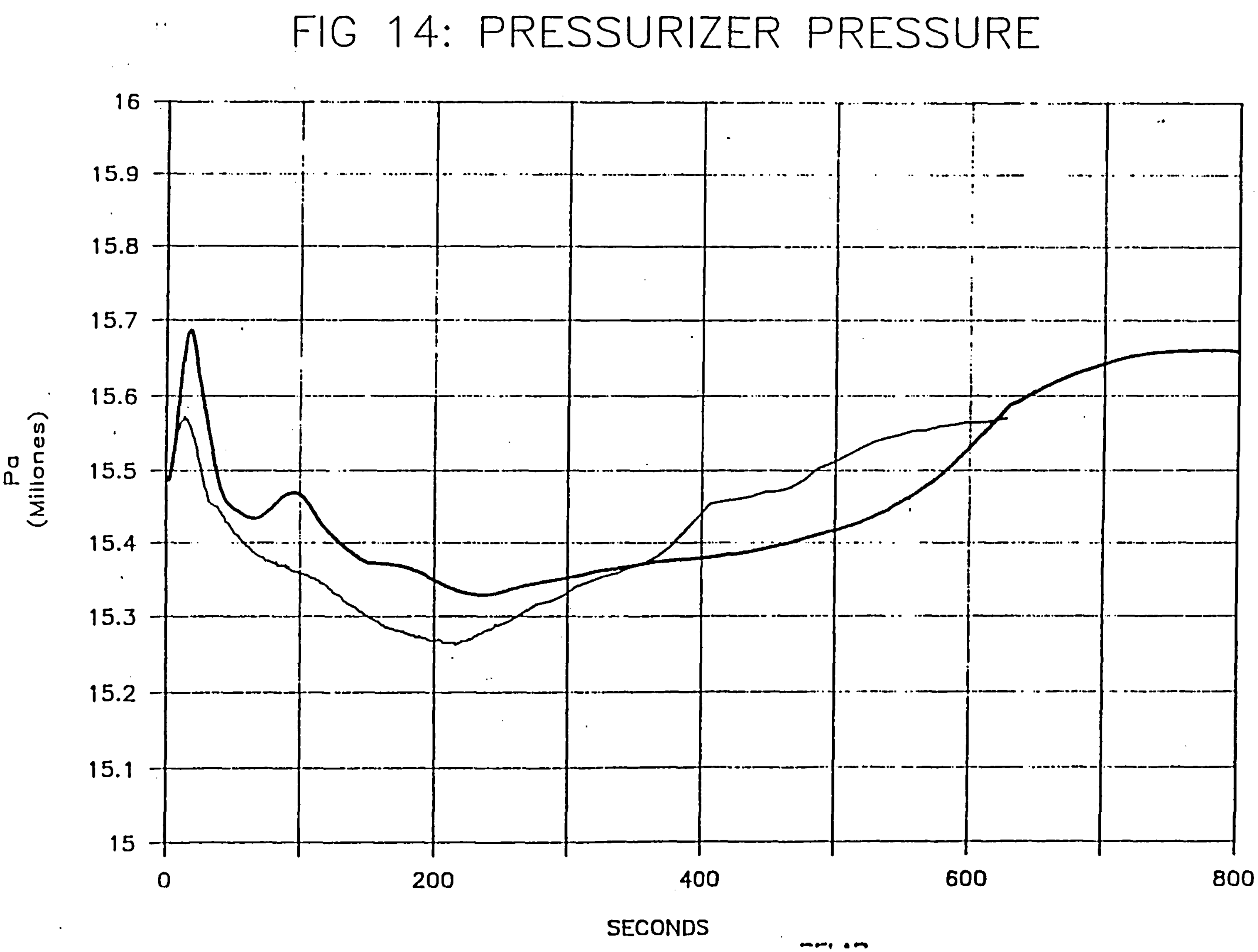
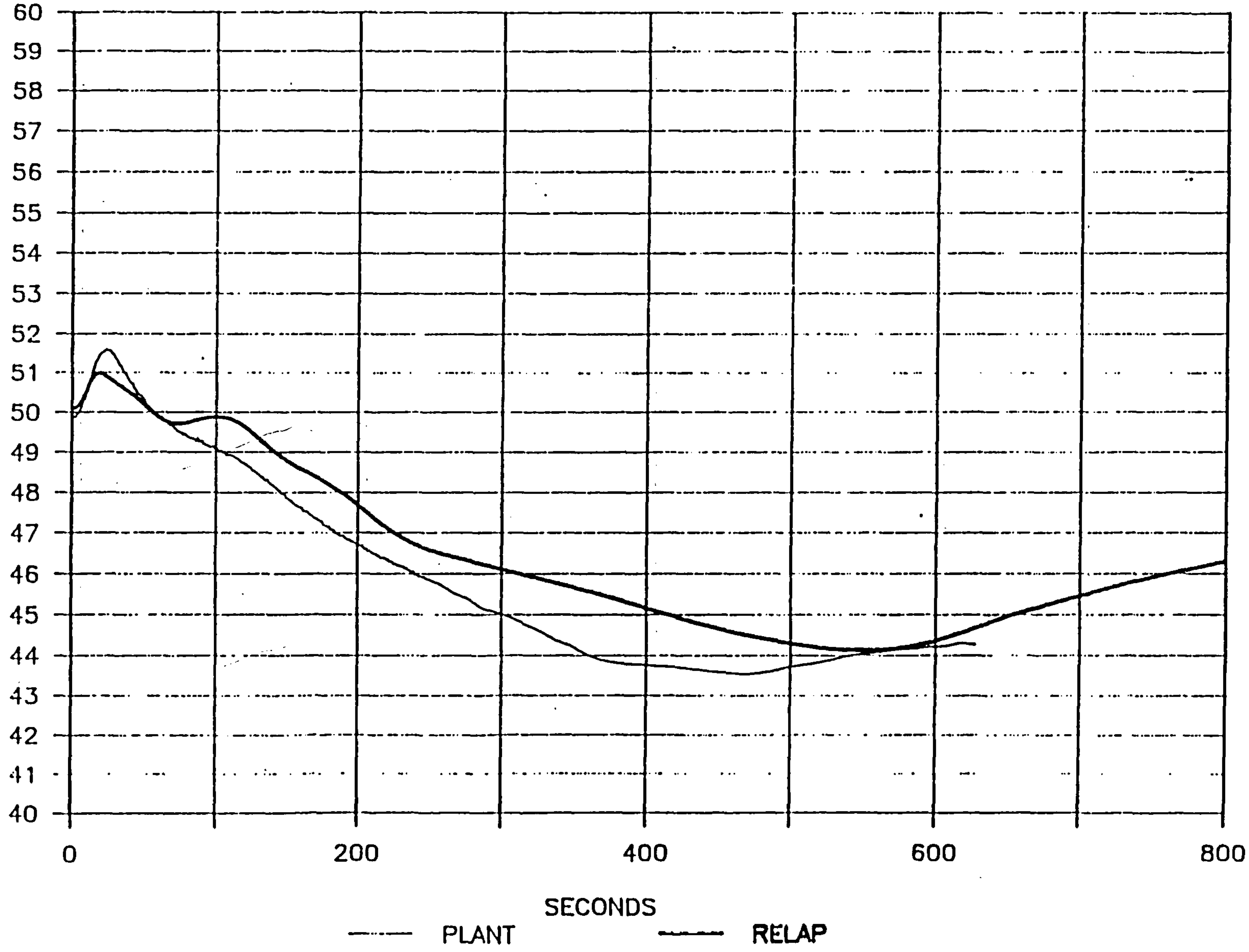


FIG 14: PRESSURIZER PRESSURE
Pa
(Millones)
16
15.9
15.8
15.7
15.6
15.5
15.4
15.3
15.2
15.1
15
0
200
400
600
800
SECONDS

FIG 15: PRESSURIZER LEVEL



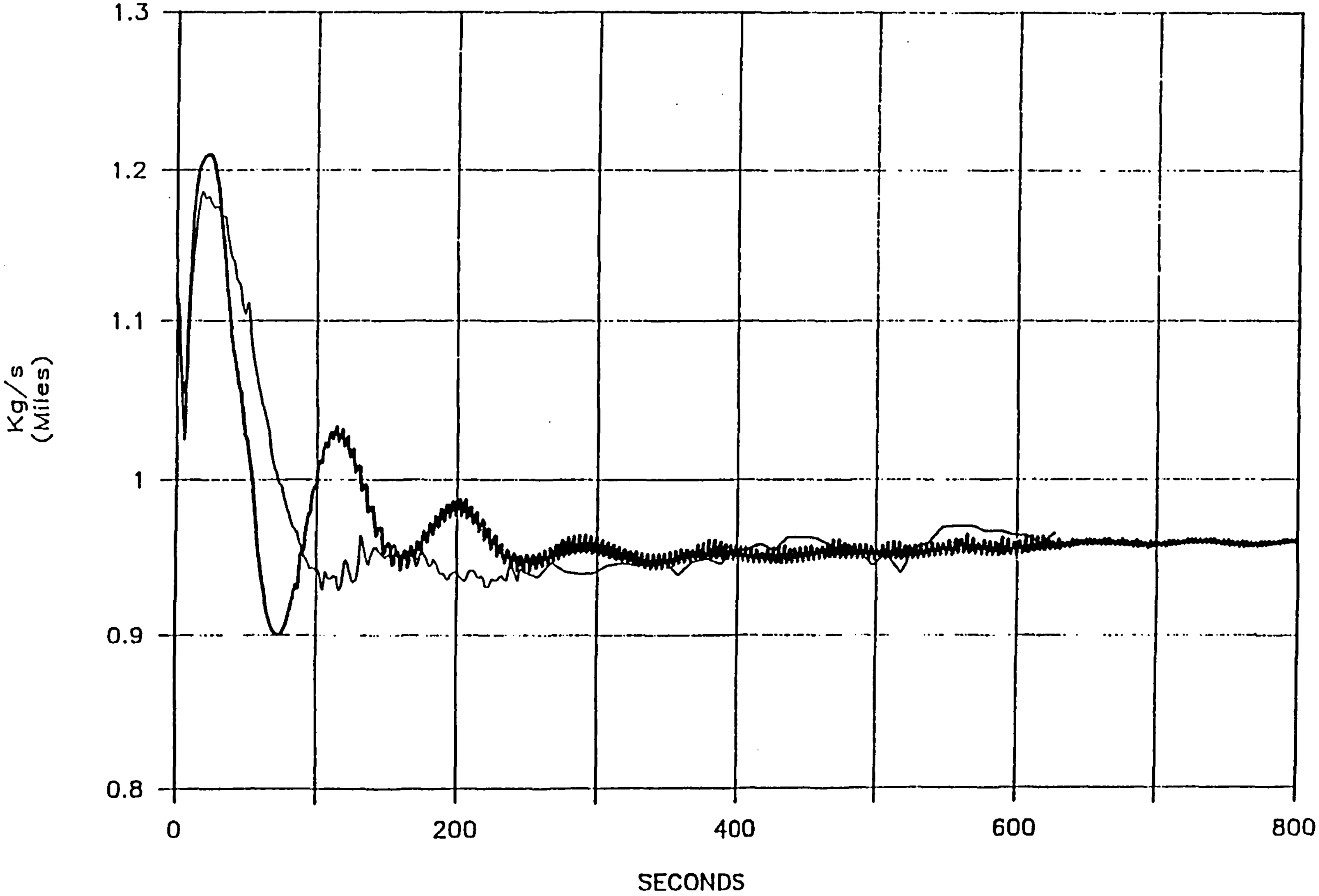


FIG 16: FEEDWATER MASS FLOW RATE

FIG 17: STEAM MASS FLOW RATE

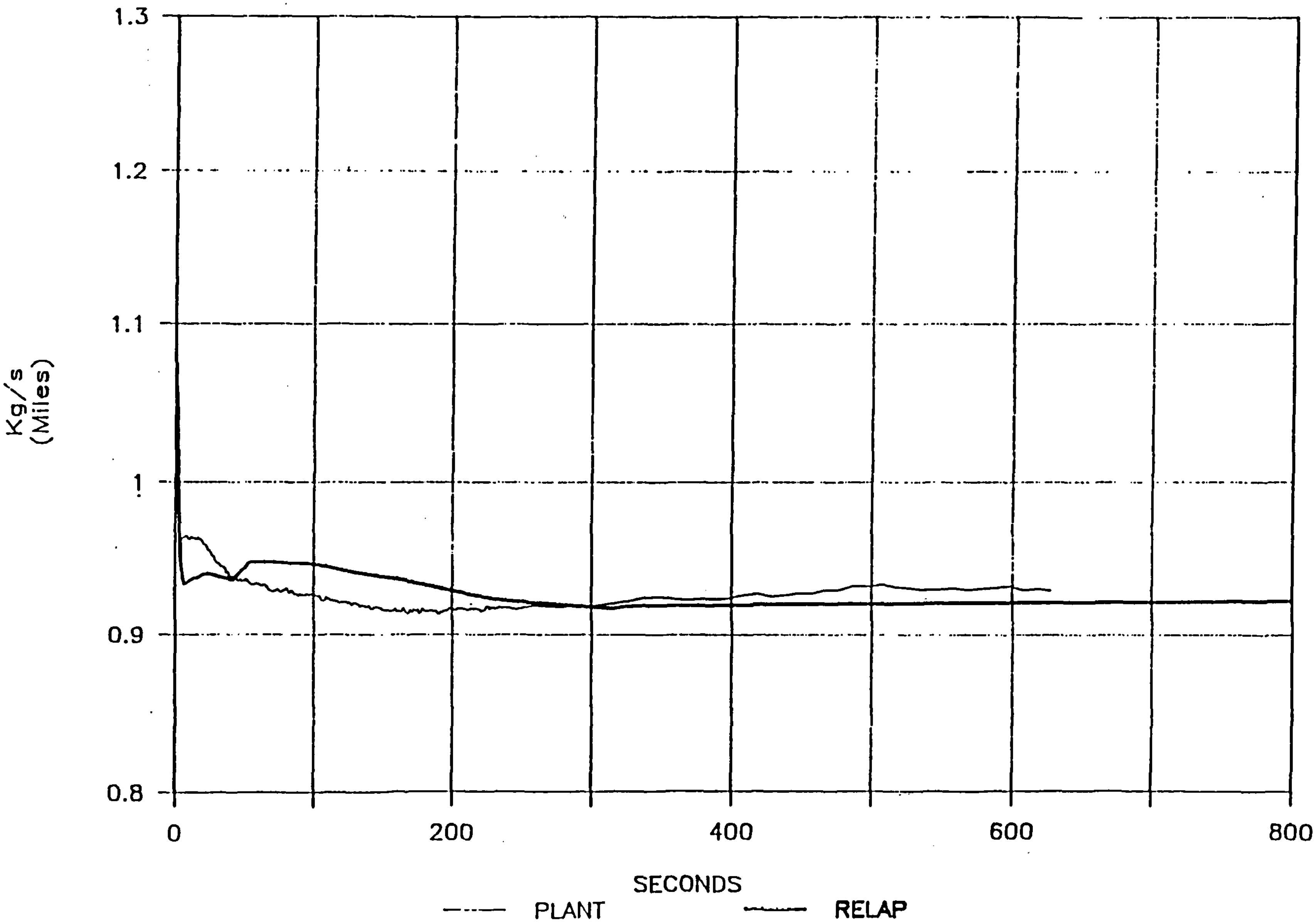


FIG 18: STEAM GENERATOR PRESSURE

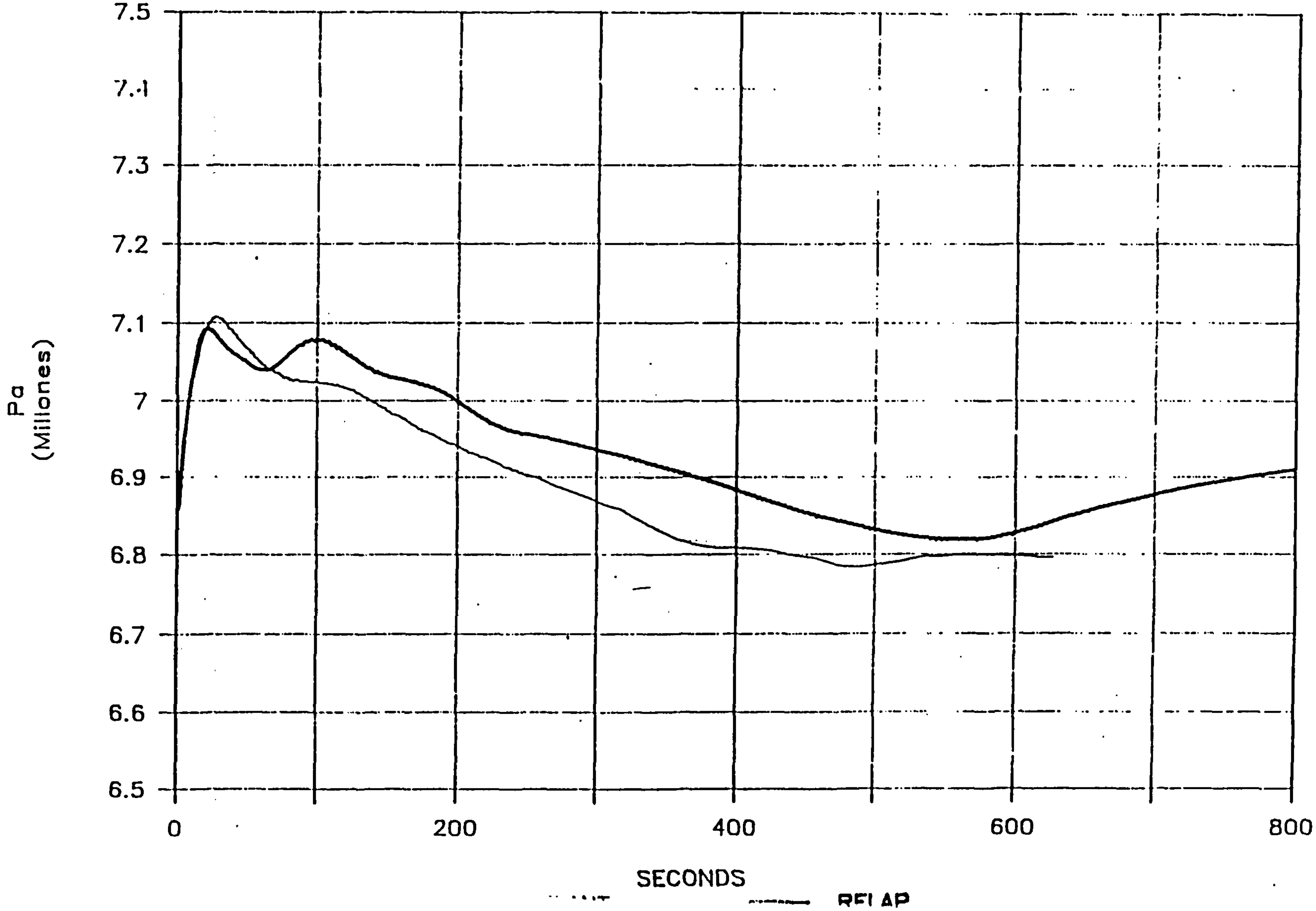


FIG 19: STEAM GENERATOR LEVEL

(NARROW RANGE)

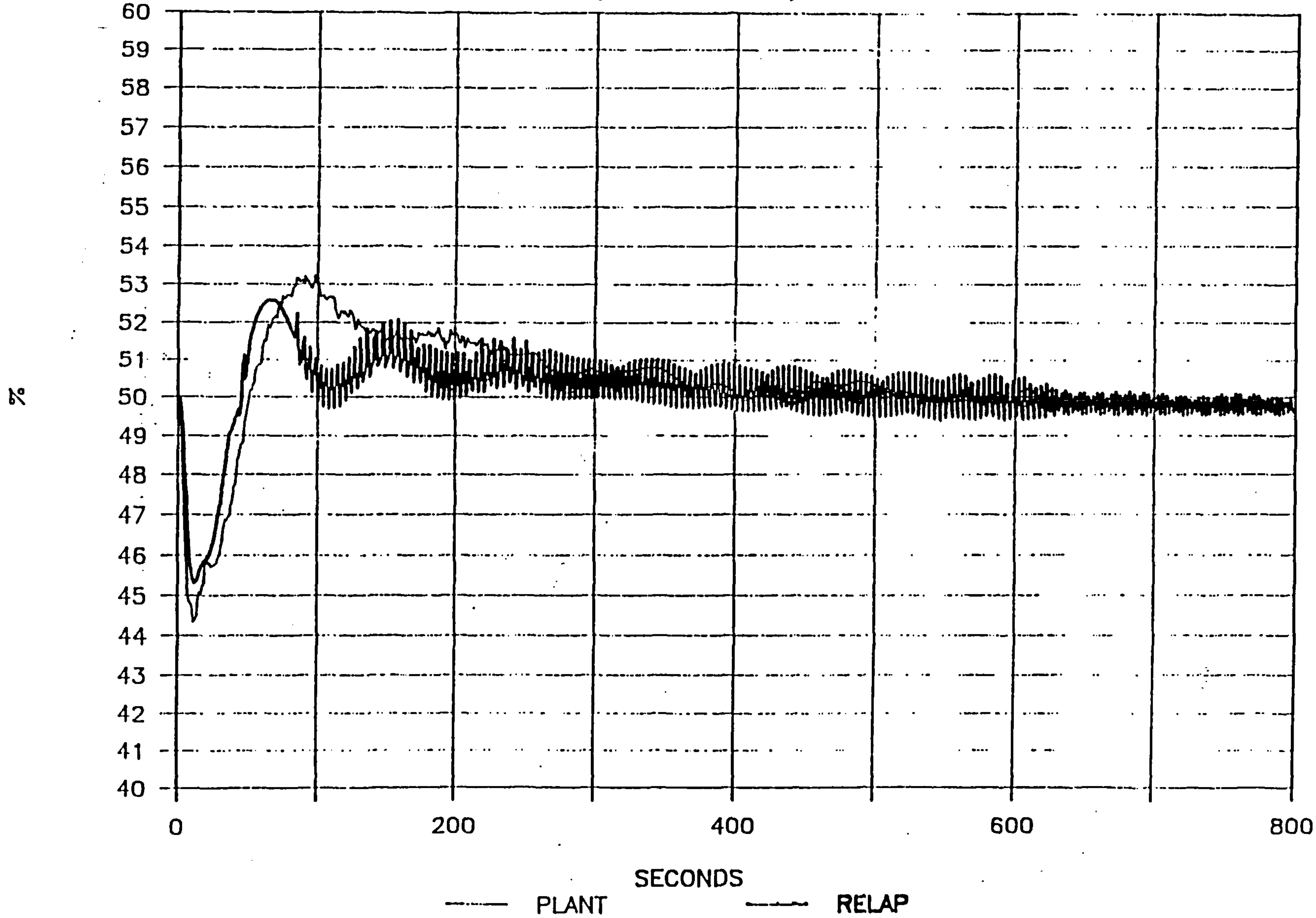
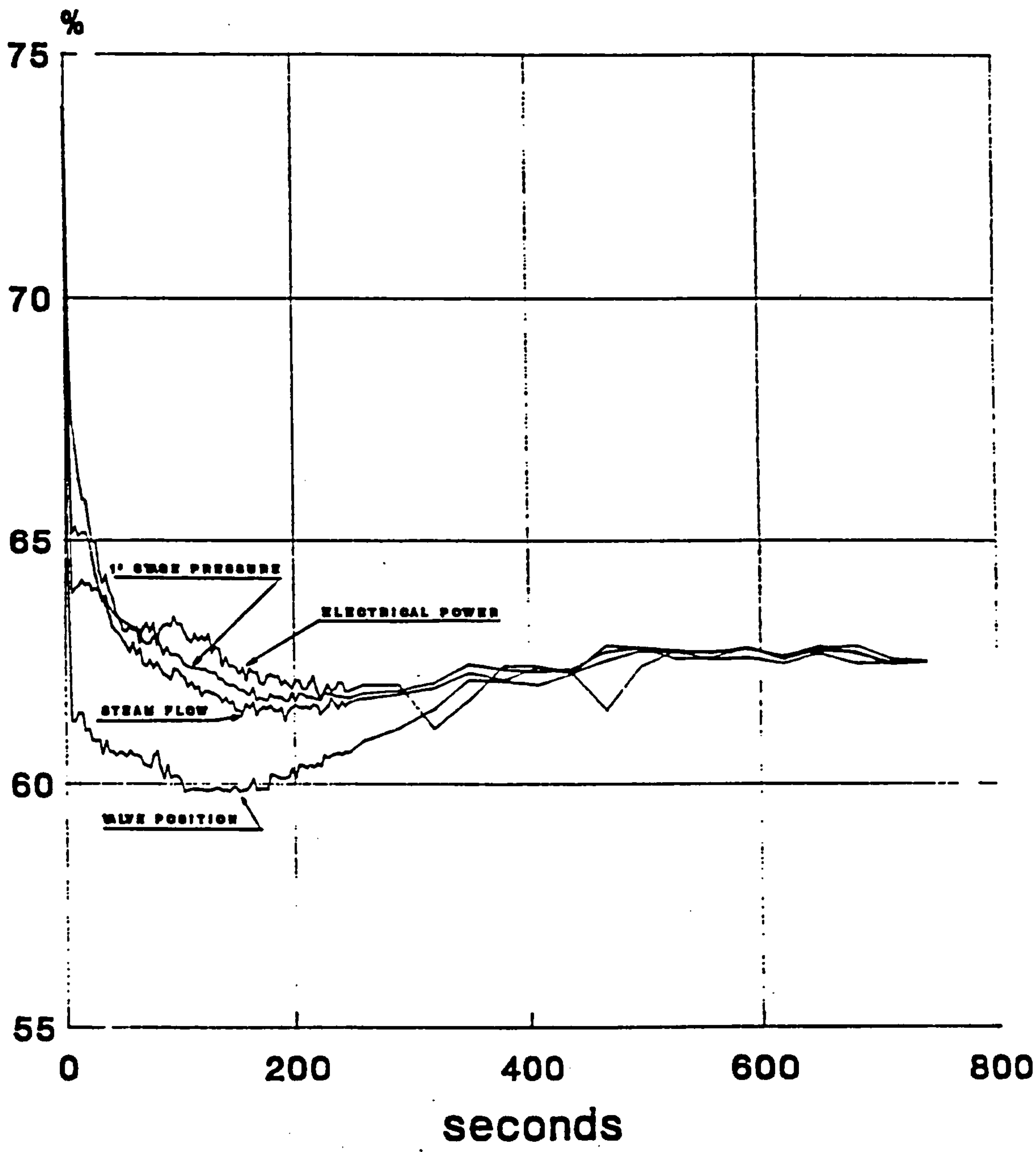


FIG 20: POWER COMPARISON
(percent)



NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/IA-0109
UNID-91-08

2. TITLE AND SUBTITLE

Assessment of RELAP5/MOD2 Against a 10% Load Rejection Transient from 75% Steady State in the Vandellós II Nuclear Power Plant

3. DATE REPORT PUBLISHED

MONTH	YEAR
May	1993

4. FIN OR GRANT NUMBER

L2245

5. AUTHOR(S)

C. Llopis, Asociación Nuclear Vandellós
J. Perez, Consejo de Seguridad Nuclear
A. Casals, Asociación Nuclear Vandellós
R. Mendizabal, Consejo de Seguridad Nuclear

6. TYPE OF REPORT

Technical Report

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION — NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Asociación Nuclear Vandellós
c/Travesera de Les Corts 39-43.-08028 Barcelona
Consejo de Seguridad Nuclear
c/Justo Dorado 11.-28040-Madrid

9. SPONSORING ORGANIZATION — NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

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

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Consejo de Seguridad Nuclear (CSN) and the Asociación Nuclear Vandellós (ANV) have developed a model of Vandellós II Nuclear Power Plant. The ANV collaboration consisted in the supply of design and actual data, the cooperation in the simulation of the control systems and other model components, as well as in the results analysis.

The obtained model has been assessed against the following transients occurred in plant:

- A trip from the 100% power level (CSN)
- A load rejection from 100% to 50% (CSN)
- A load rejection from 75% to 65% (ANV)
- A feedwater turbopump trip (ANV)

This copy is a report of the load rejection from 75% to 65% transient simulation. This transient was one of the tests carried out in Vandellós II NPP during the startup tests.

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

ICAP,
Vandellós II,
RELAP5,
Transient,
Load Rejection

13. AVAILABILITY STATEMENT

Unlimiited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

3.

Printed on recycled paper

Federal Recycling Program