NUREG/IA-0110
ICSP-V2-TURFW-R

International Agreement Report

Assessment of RELAP5/MOD2 Against a Main Feedwater Turbopump Trip Transient in the Vandellos II Nuclear Power Plant

Prepared by
C. Llopis, A. Casals/A. N. V.
J. Perez, R. Mendizabal/C. S. N.

Asociacion Nuclear Vandellos (A. N. V.)
Consejo de Seguridad Nuclear (C. S. N.)
Madrid, Spain

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

December 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

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NOTICE

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 Eléctrica (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 Coordinación". This committee has approved the distribution of this document for ICAP purposes.

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ABSTRACT

The Consejo de Seguridad Nuclear (CSN) and the Asociacion Nuclear Vandellos (ANV) 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 feedwater turbopump trip transient simulation. This transient ocurred actually in plant on June 19, 1989.

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 colaborated with the CSN simulating and analyzing two of the four transients that 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.
- This transient causes the steam-dump to open, but does not cause either the relief or the safety valves to actuate, so that this allows analyzing the steam.dump behavior.

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 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 simulation of FSAR design accidents by means of a best estiamte model, in order to compare them to the results obtained using conservative codes.

- The comparison of the FSAR design accidents to the best estimate model.

- 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. This is the equipment used to record the parameters needed to assess this case.

The availability of such a great number of signals has allowed the use of RELAP to check the control blocks partial performances, specially the feedwater control block and the rod control block (verifications carried out during the load rejection from 75% to 65% case), and the steam-dump.

2.3 TRANSIENT DESCRIPTION

The test which is the subject of the current simulation ocurred on June 19, 1989, in Vandellos II, being the plant at the 99.2% power level.

The transient ocurred as a result of a maintenance operation consisting in the realignement of the turbopumps lubricating oil cooler. To realign it, an auxiliary three way valve is needed, and there is a position of this valve in which a pressure drop occurs. This pressure drop caused the main feedwater turbopump trip.

This trip of a main feedwater turbopump caused the runout in the other one, which has the capacity to supply the 85% of the total feedwater volume. In the same manner, this trip triggered the turbine runback from 100 % to 70% at 200% / min rate.

All the control systems in the plant were in automatic mode.

3. MODEL DESCRIPTION

Figure 2 shows the nodalization used to simulated the primary system of the plant. It consists of 117 volumes, 122 junctions, 78 heat structures and 155 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 the 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 flanges.
- 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 valves control have been simulated, but not the valves themselves. This allows verifying that in this transient these valves do not open.

Talking about kinetics, the Doppler coefficient value has been adjusted in such a way that for each rod position the plant nuclear flux level is reached. This adjustement has not been possible to be made with the Doppler coefficient design values, since RELAP5/MOD2 uses the punctual kinetics model, and during this transient a rod position shift occurs that implies a different core behavior depending on the axial height we consider.

3.2. SECONDARY SYSTEM

In the secondary side, the three steam generators and the three 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 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 the four groups. 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 settings during the test have been used as setpoints for the model.

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 response accurately. This and the steam flow are the main contributors to the RELAP5/MOD2 results and to the plant reponse mismatching in the main feedwater turbopump trip transient simulation.

In this case, however, the rod control system has been adjusted properly, since the turbine first stage impulse chamber pressure has been imposed as a boundary condition to represent the turbine power evolution. In the load rejection from 75% to 65% simulation case, this variable was not used because it is not calculated by RELAP model, but in this case it was expected to prove that using this procedure the control rod behavior is improved preceptibly. This point suggests extending the model downstream of the turbine valve.

4. INITIAL STEADY STATE CALCULATIONS

The new steady state has been reached starting from a 100% rated conditions steady state and modifying all the RELAP variables reinicialization until reaching a similar steady state to the plant one.

The main parameter 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 assessment is to check the model behavior, specially the control system (the rod control, the pressurizer level and pressure control and the steam-dump control) and the thermohydraulic evolution of the plant main parameters.

The simulation of this transient has been carried out starting from the initial steady state, imposing the feedwater flow, pressure and temperature conditions, and reducing the turbine flow. Another boundary condition that has been imposed is the turbine first stage impulse chamber pressure, which will be used by the rod control system as a turbine power reference.

This pressure is the plant actual pressure and is supplied by means of a table.

The transient starts with the turbine flow decrease, which causes a variation in the energy production and evacuation balance of the primary side. Owing to this variation, some changes in 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 mismatch and the average temperature evolution (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 plant and RELAP final level discrepancy (70% and 74% approximately) is attributable to the plant power measures error margin. A thermal balance at the end of the transient has been carried out, which seems to indicate that RELAP results are more trustworthy than plant data.

The cold leg, hot leg and average temperatures are shown in figures 9, 10, and 11. In the beginning, the average temperature increases because of the power produced by the reactor and the steam generators 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 12 has been included to show the primary delta temperature evolution, as a significant indicator of the primary power evolution.

This delta temperature has a close adjustement whith 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.

Figures 19, 20 and 21 show the steam dump valves opening, which operate in temperature mode. It can be observed that the steam dump opening time in the model is longer than in plant. Probably, attributable to the fact that the valves capacity has been underestimated.

Since the steam dump valves flow measure is not available in plant, it has not been possible to obtain a closer adjustement; however, the total mass evacuated is considered to be the correct one owing to the close adjust of the steam generator pressure and the primary side average temperature.

The primary pressure evolution is similar to the average temperature one, and is shown in figure 13. The same occurs with the pressurizer level (figure 14), which is modified esentially by the density variations in the primary side.

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

Figure 15 shows the feedwater flow evolution, which in this case has been imposed as a boundary condition. This figure shows that, after a sudden fall as a result of a turbopump trip, the flows tends to recover by means of the other turbopump, and it stabilizes at a different level because of the turbine run back.

The steam generators pressure (fig.17) increases initially because of the turbine valve closure and has a smooth decrease later, because of the heat transmission balance through the steam generator tubes, the evacuation through the turbine and steam dump, and the feedwater contribution.

The steam generators level (fig.18) has a sudden fall on starting the transient because of the steam binding, but it recovers later owing to the steam and feedwater flows balance (figures 15 and 16).

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 3.00, which is smaller than the ratio of the load rejection from 75% to 65% calculation for ICAP, due to the fact that during the time delay which existed between both calculations the computer capabilities were improved.

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

The run statistics are shown in Table V.

7. CONCLUSIONS

The control blocks models, which were individually assessed in the load rejection transient from 75% to 65%, the other case prepared for ICAP, have been used in this calculation without having introduced any modification. This means that a new validation of the models has been carried out, which proves again the outstanding performance of these models.

In this transient, which with regard to the primary side is in fact a load rejection from 100% to 70%, 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 lead us to conclude that the punctual kinetics is a conservative model that has to be corrected for this kind of transients.

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

The simulation of this transient has allowed analyzing the steam dump behavior with the RELAP5/MOD2 model. The opening and closure speeds of all the banks have been reproduced with accuracy; however, some differences exist with regard to the time the valves remain opened. Probably, this fact is due to the valves capacity in the model. A closer adjustement has not been possible since the plant steam flow measurement includes the turbine and steam dump consumptions and other services (MSR's, turbopumps,...).

Anyway, it has been considered that the value of the total mass evacuated through the steam dump is correct, since the steam generator pressure and the primary side average temperature have been closely adjusted.

This model, after being assessed with the calculations for ICAP:

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

is considered to be a valuable tool for transient simulation.

8. BIBLIOGRAPHY

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RELAP5/MOD2 Code Manual

Project Information Package

Steam Generator thermohydraulic analysis

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.	MAIN FEEDWATER TURBOPUMP TRIP TURBINE RUNBACK
3.0 SEC.	TURBINE VALVE AT A NEW POSITION
APROX. 400 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	(%)	99.2	99.2	(1)
COLD LEG TEMPERATURE	(°K)	564.8	564.8	
HOT LEG TEMPERATURE	(°K)	598.1	598.1	
AVERAGE TEMPERATURE	(°K)	581.5	581.4	
DELTA TEMPERATURE	(°K)	33.2	33.2	
PRESSURIZER PRESSURE	(MPa)	15.52	15.50	
PRESSURIZER LEVEL	(%)	59.9	59.8	
FEEDWATER MASS FLOW RATE	(Kg/s)	1527	1533	
STEAM GENERATOR PRESSURE	(MPa)	6.69	6.70	
STEAM GENERATOR LEVEL N.R.	(%)	50.6	50.4	
RECIRCULATIO RATIO		3.25	3.27	(2)

(1) CALCULATED DATA
(2) DESIGN DATA

TABLE IV

DESCRIPTION OF RELAP5/MOD2 VARIABLES		FIGURE
CNTRLVAR 340	ROD POSITION	7
CNTRLVAR 301	NUCLEAR POWER (PERCENT)	8
CNTRLVAR 328	TEMPERATURE AT THE COLD LEG	9
CNTRLVAR 327	TEMPERATURE AT THE HOT LEG	10
CNTRLVAR 330	AVERAGE TEMPERATURE	11
CNTRLVAR 947	DELTA TEMPERATURE	12
P 415090000	PRESSURIZER PRESSURE	13
CNTRLVAR 350	PRESSURIZER LEVEL	14
MFLOWJ 325000000	FEEDWATER MASS FLOW RATE	15
MFLOWJ 600010000	STEAM MASS FLOW RATE	16
P 600010000	STEAM GENERATOR PRESSURE	17
CNTRLVAR 203	STEAM GENERATOR LEVEL (N.R.)	18
CNTRLVAR 970	BANK 1 STEAM-DUMP DEMAND	19
CNTRLVAR 971	BANK 2 STEAM-DUMP DEMAND	20
CNTRLVAR 972	BANK 3 STEAM-DUMP DEMAND	21

TABLE V

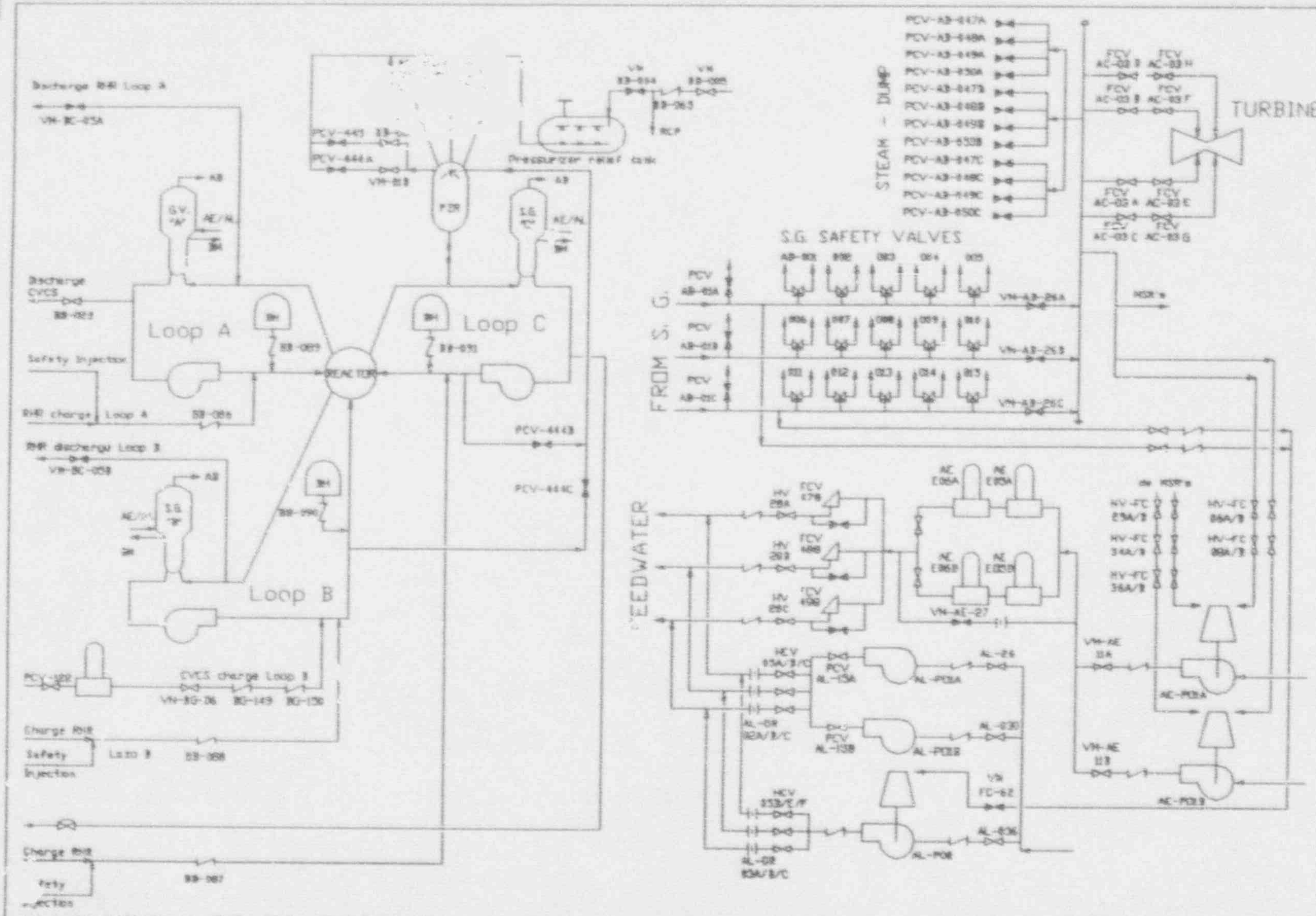
RUN STATISTICS

COMPUTER	IBM 3090
TRANSIENT TIME	650 sec
CPU TIME	1954 sec
C (TOTAL NUMBER OF ACTIVES VOLUMES)	117
DT (TOTAL NUMBER OF TIME STEPS)	13000

$$\frac{CPU * 1000}{C * DT} = 1.28$$

CPU TIME / TRANSIENT TIME	3.00

10. INDEX OF FIGURES



VANDELLOS II PRIMARY AND MAIN ASSOCIATED SYSTEMS FIG. 1

NODALIZATION
C.N. VANDELLOS II

FIGURE 2

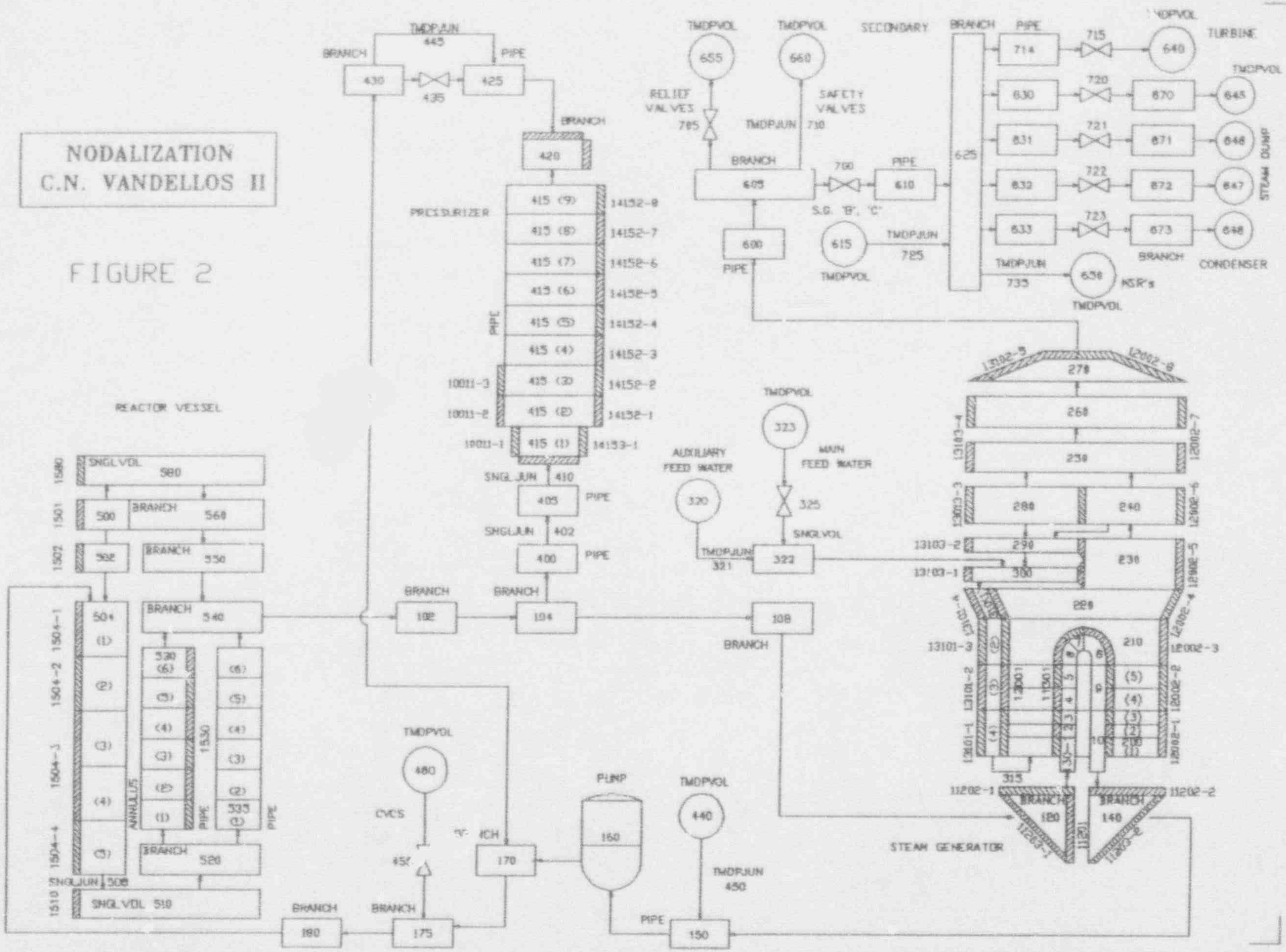
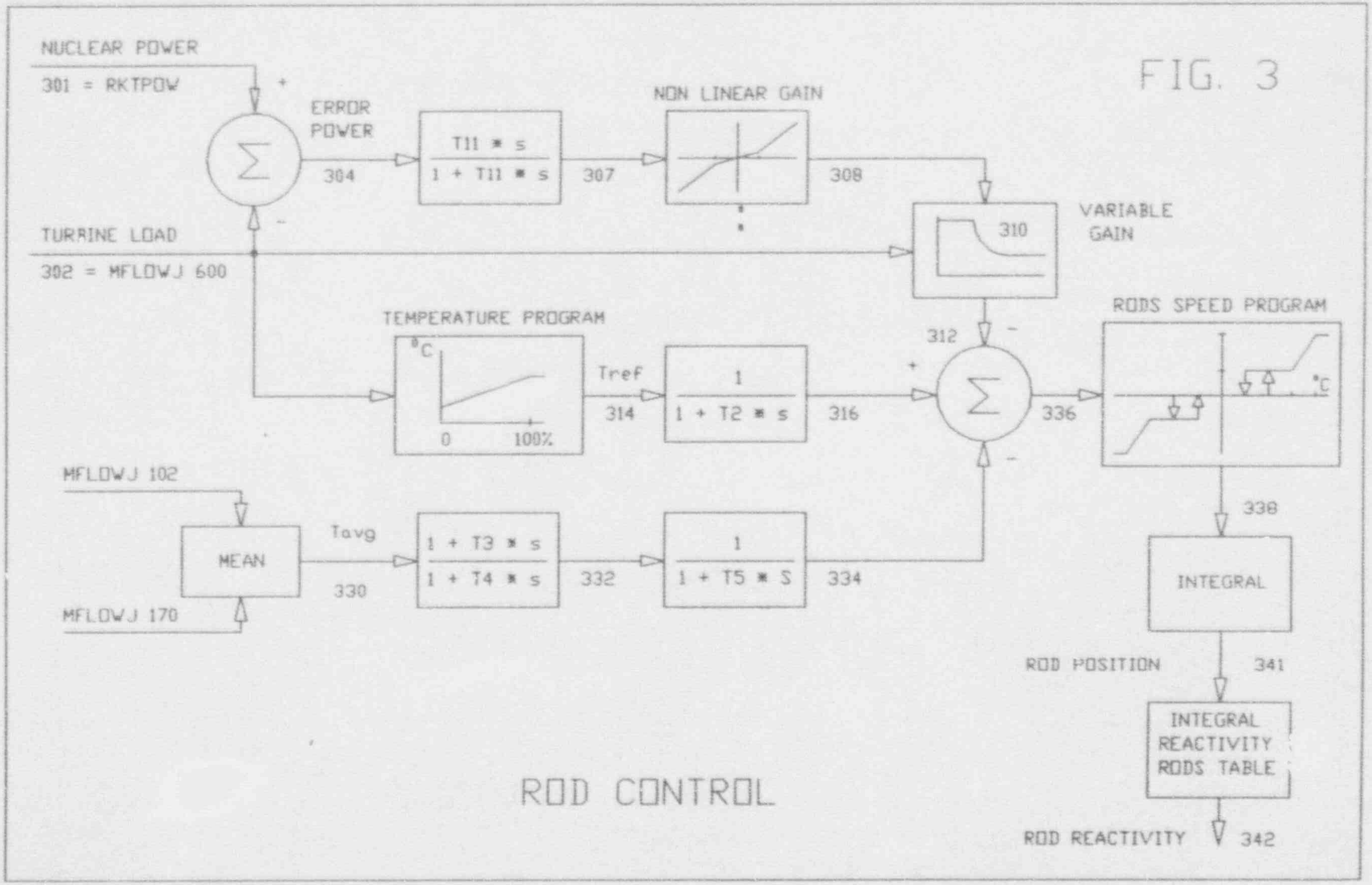
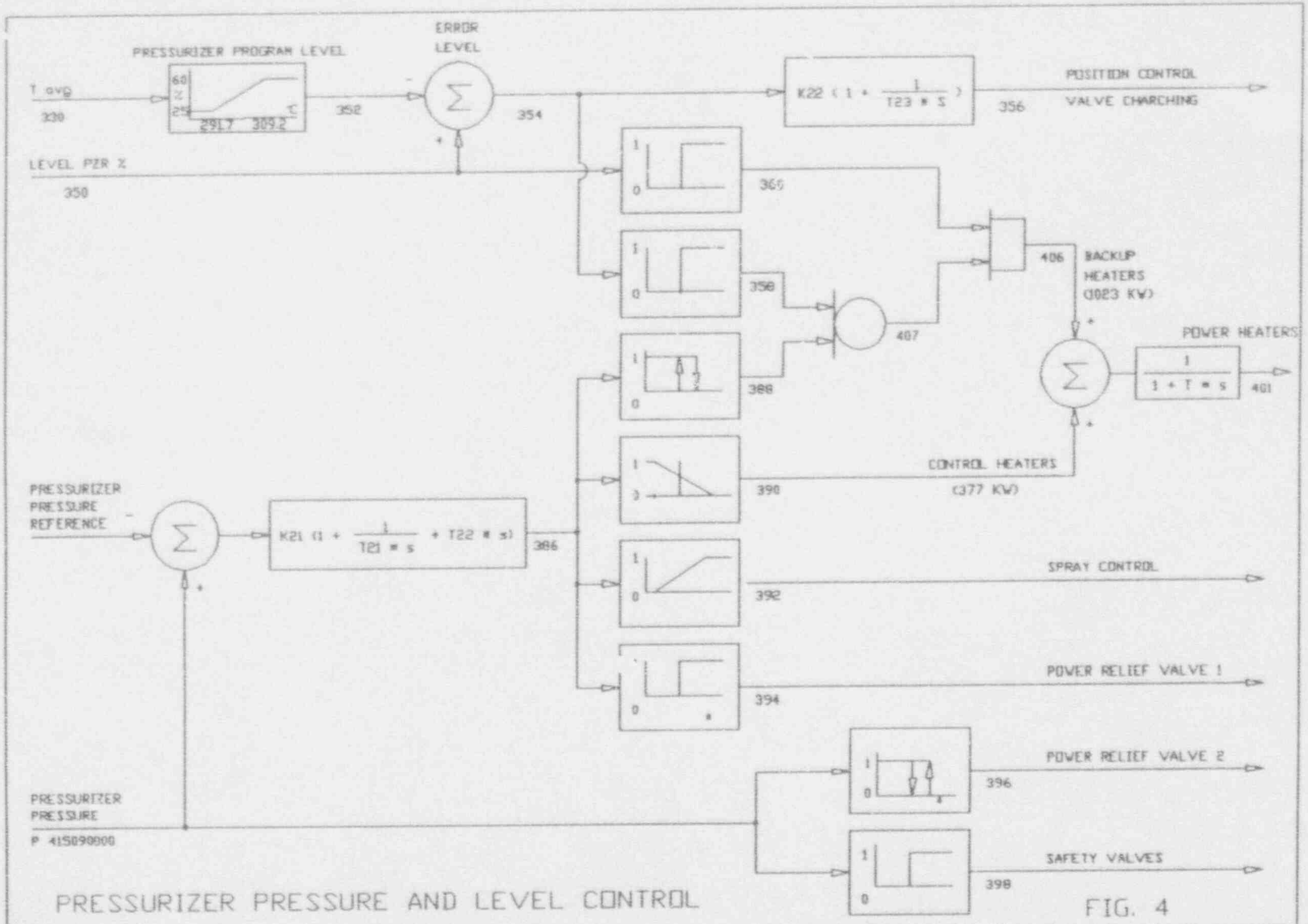


FIG. 3

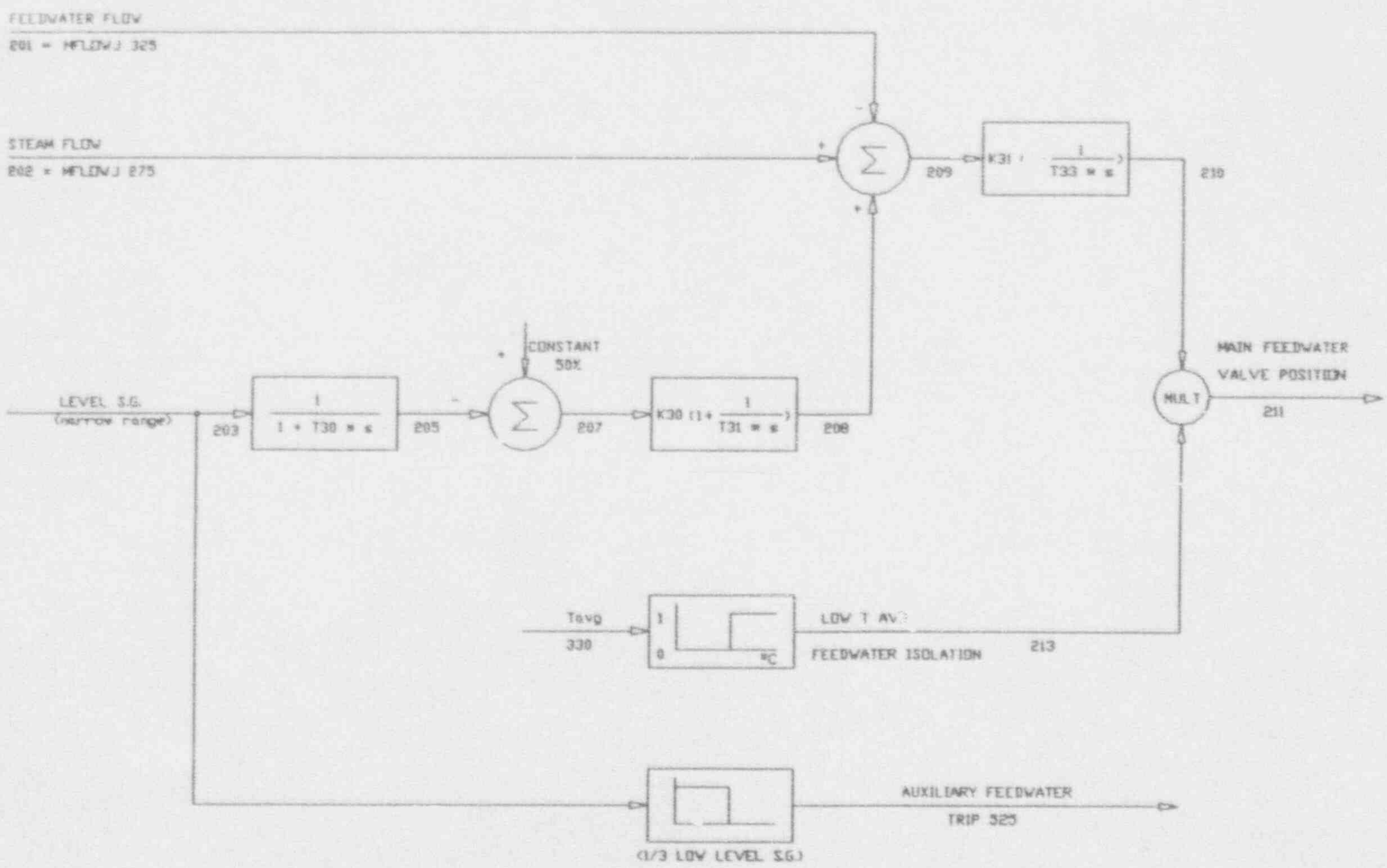


ROD CONTROL

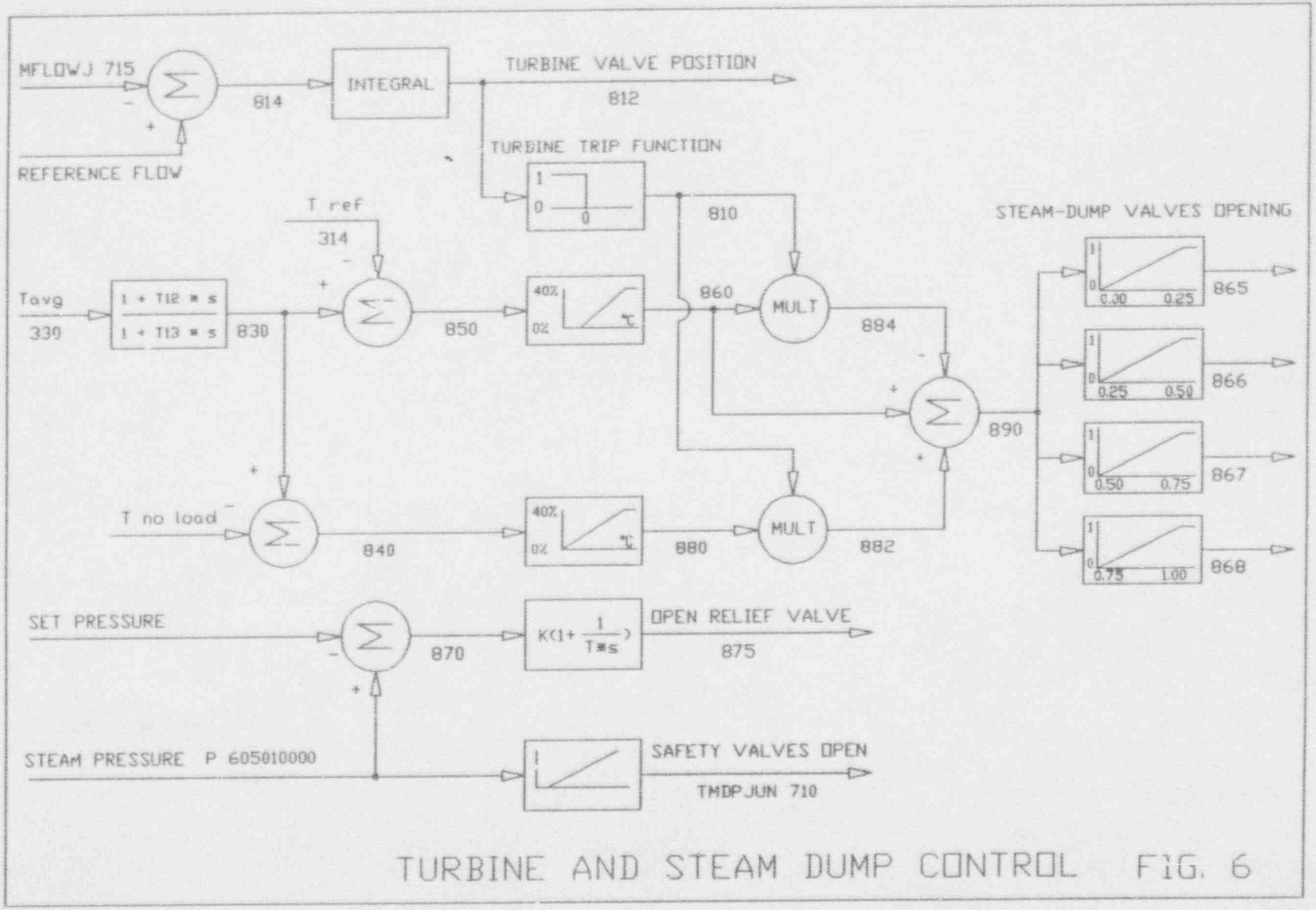


PRESSURIZER PRESSURE AND LEVEL CONTROL

FIG. 4



MAIN AND AUXILIARY FEEDWATER CONTROL FIG. 5



TURBINE AND STEAM DUMP CONTROL FIG. 6

FIG.7 ROD POSITION (BANK D)

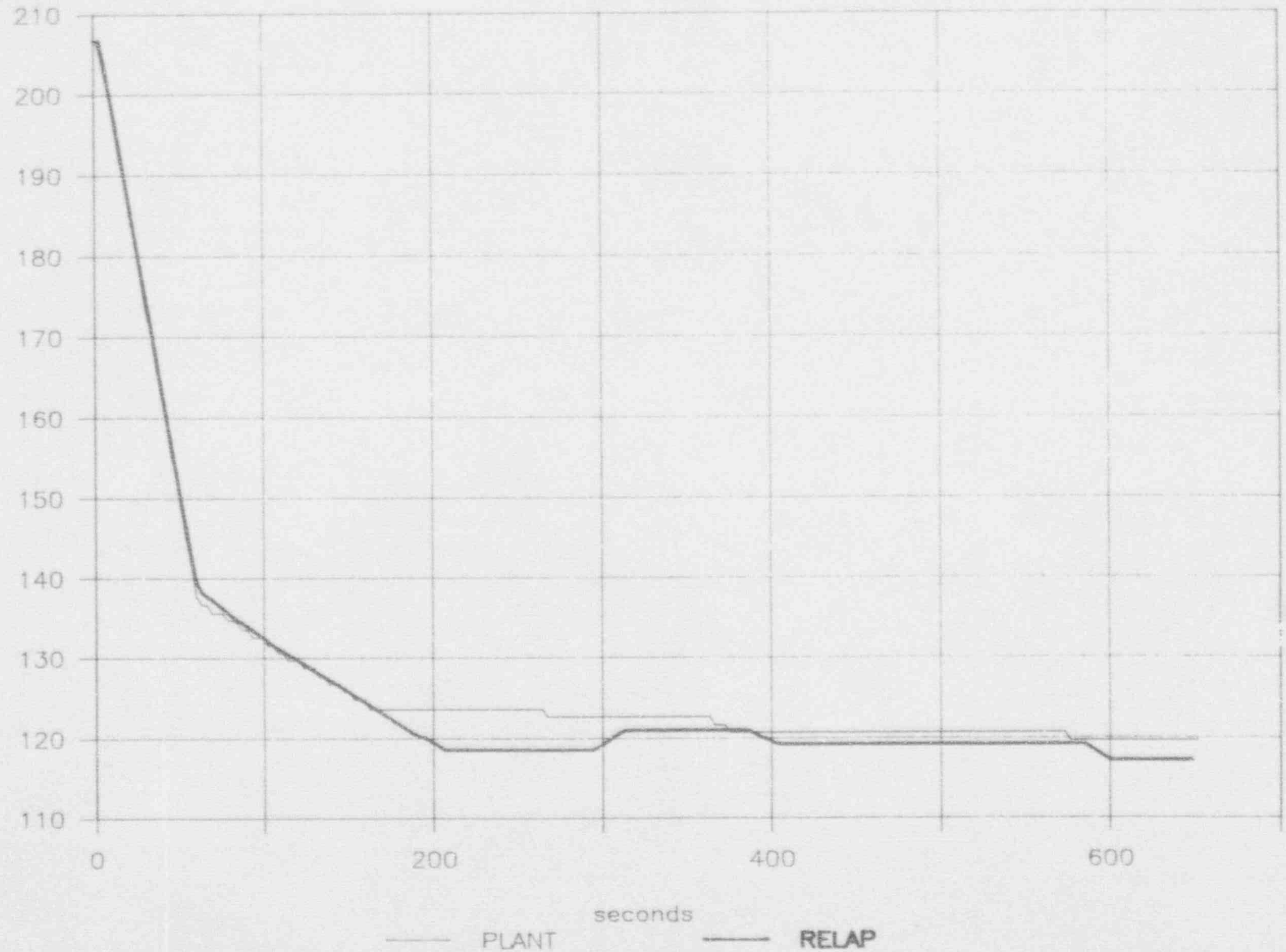
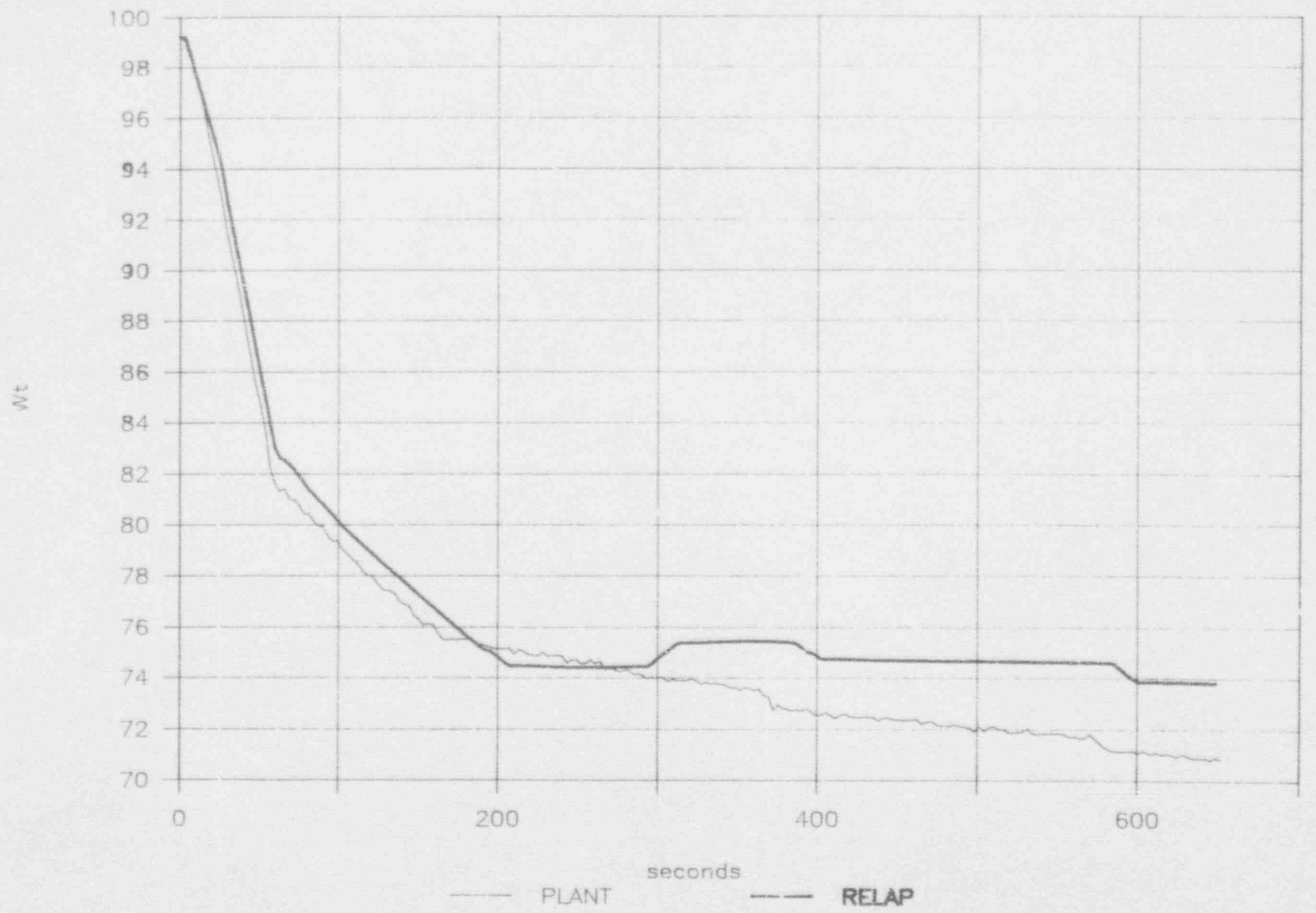


FIG.8 NUCLEAR POWER



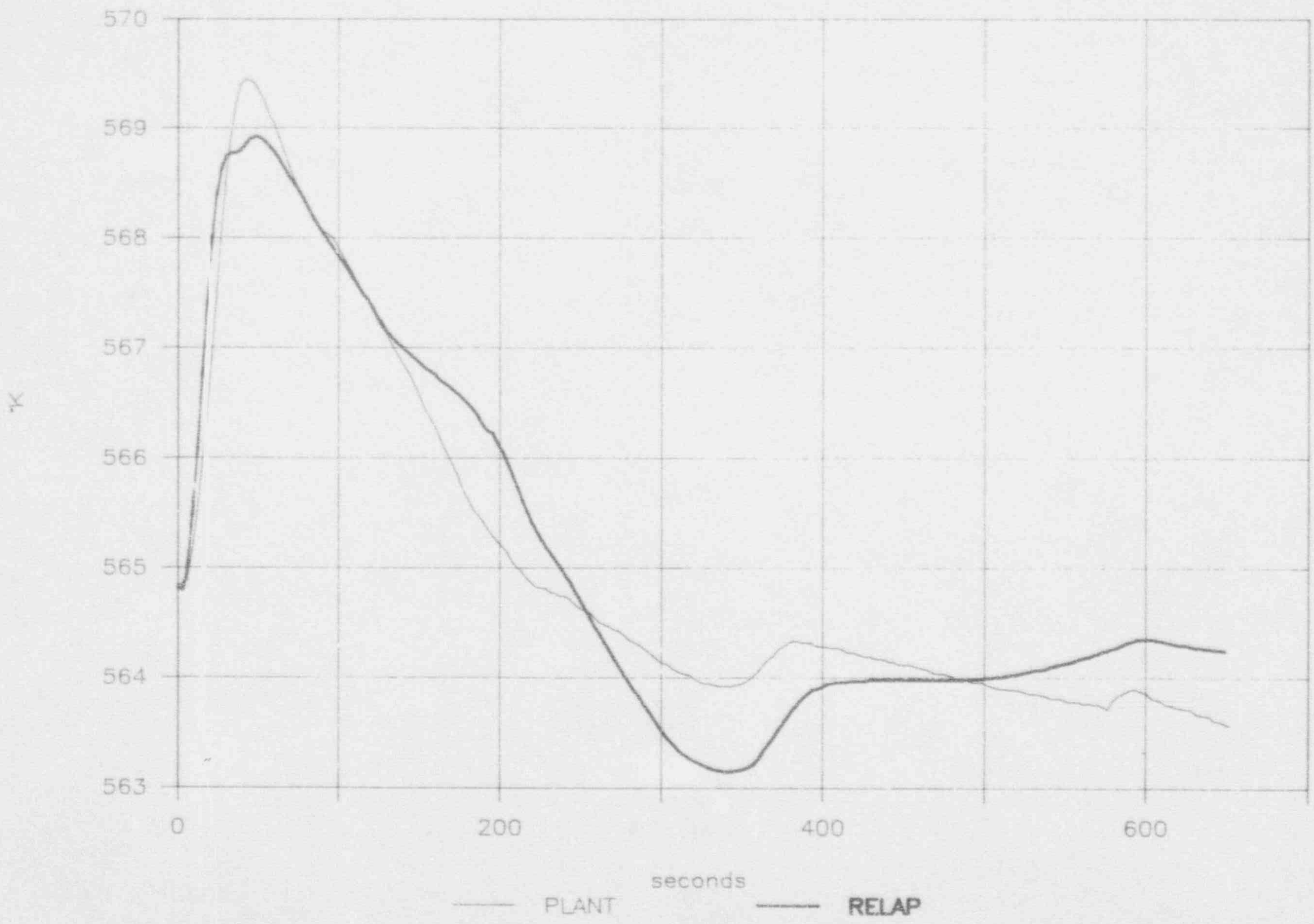


FIG.9 TEMPERATURE AT THE COLD LEG

FIG.10 TEMPERATURE AT THE HOT LEG

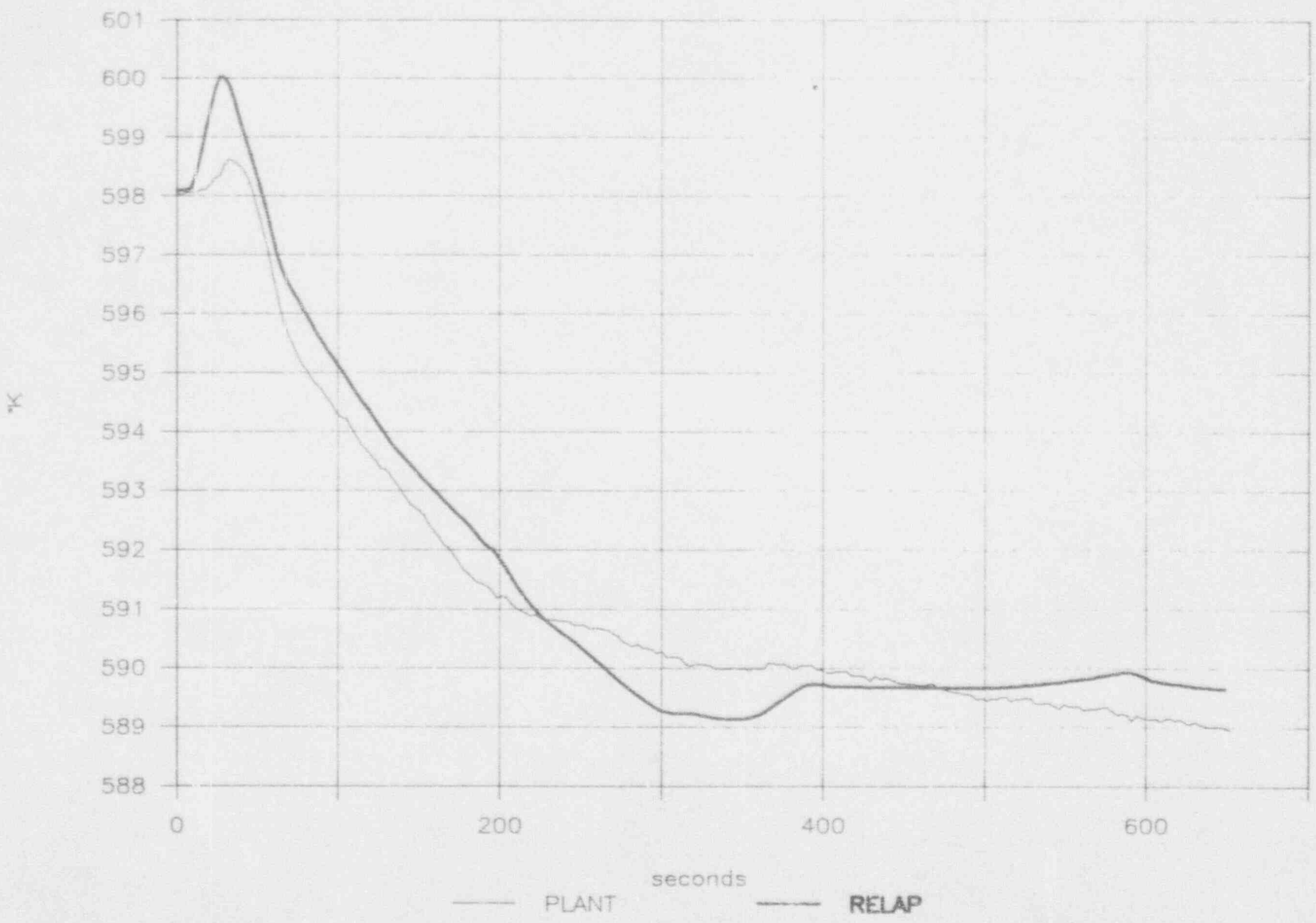


FIG.11 PRIMARY AVERAGE TEMPERATURE

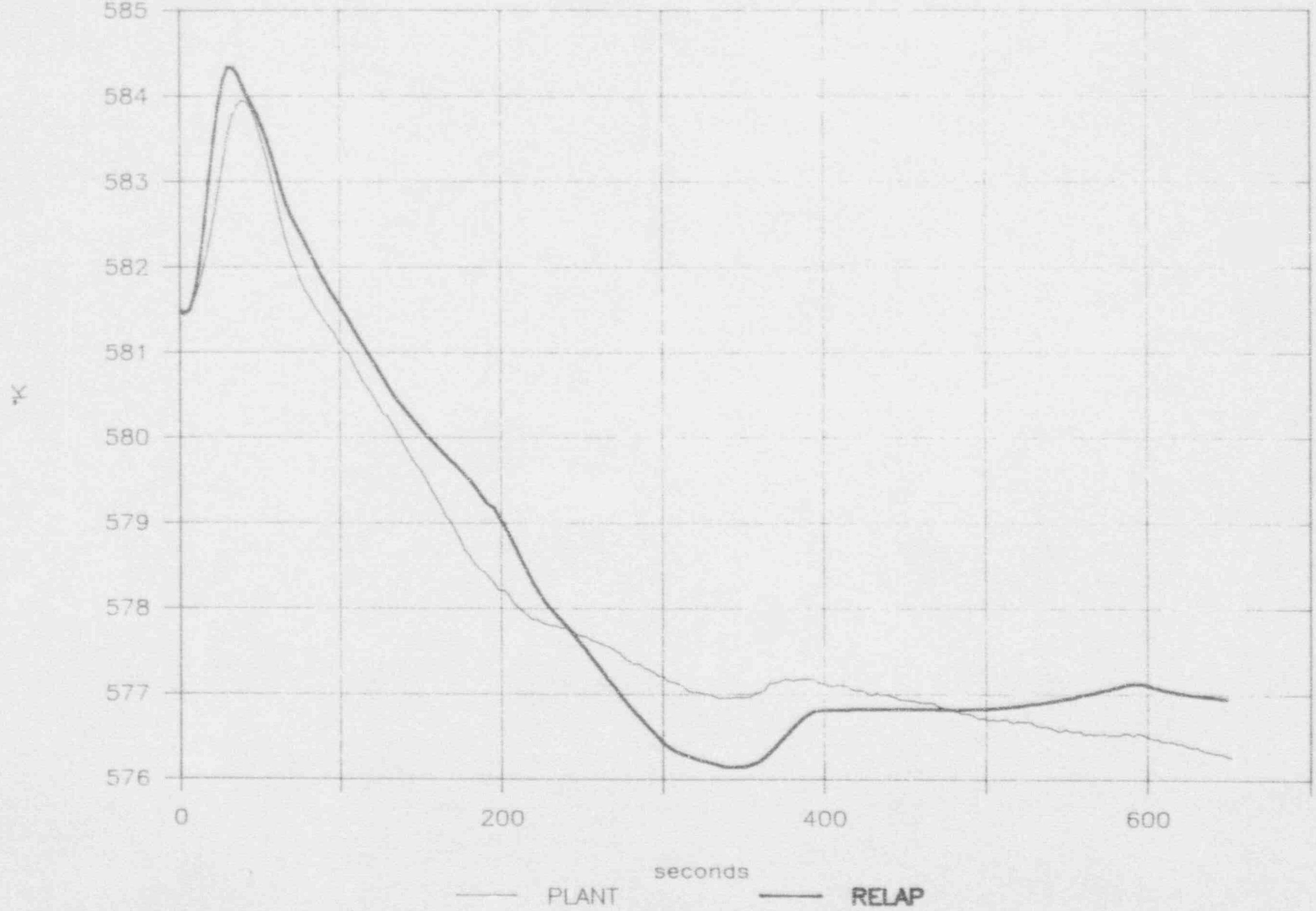


FIG.12 VESSEL DELTA TEMPERATURE

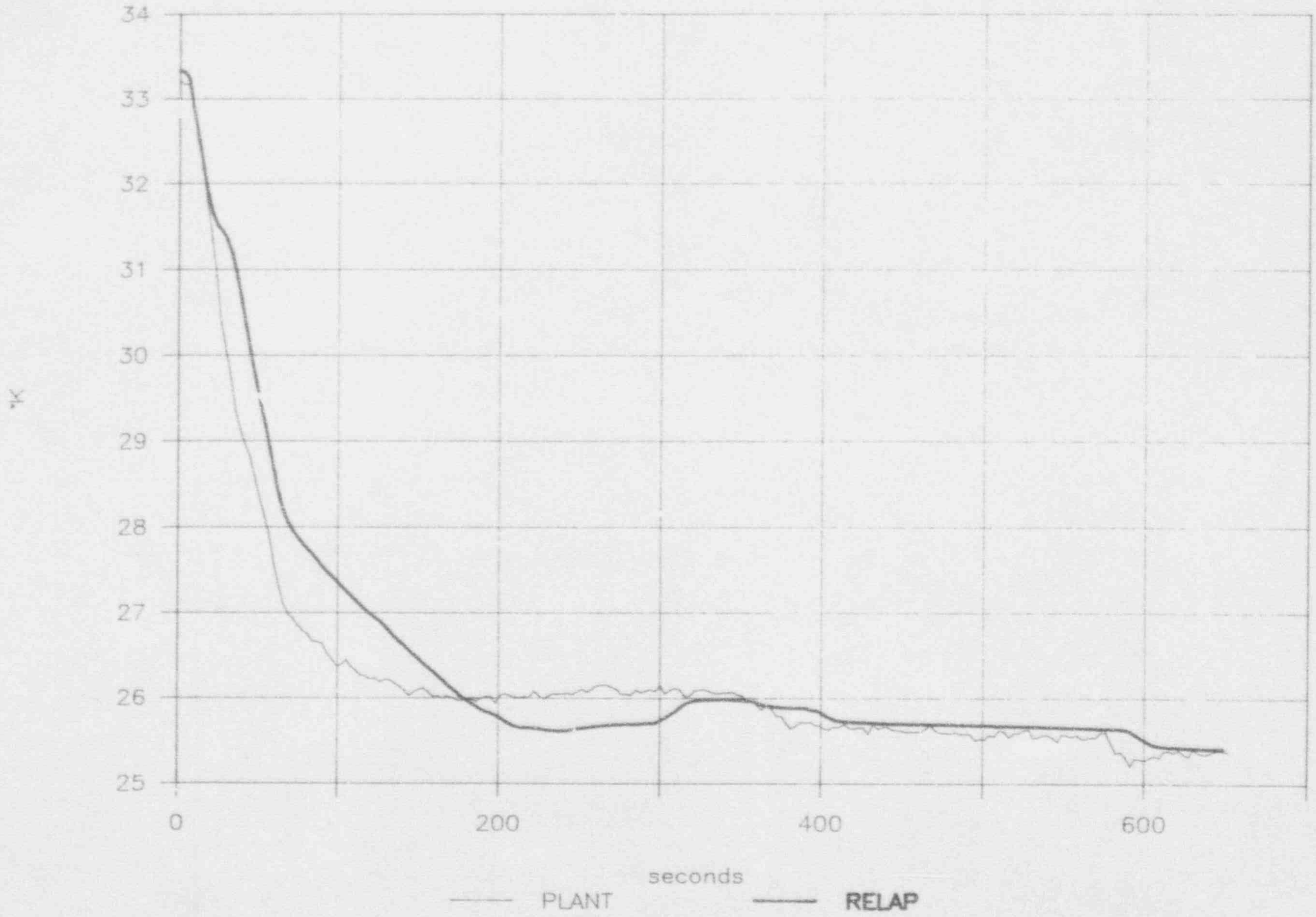


FIG.13 PRESSURIZER PRESSURE

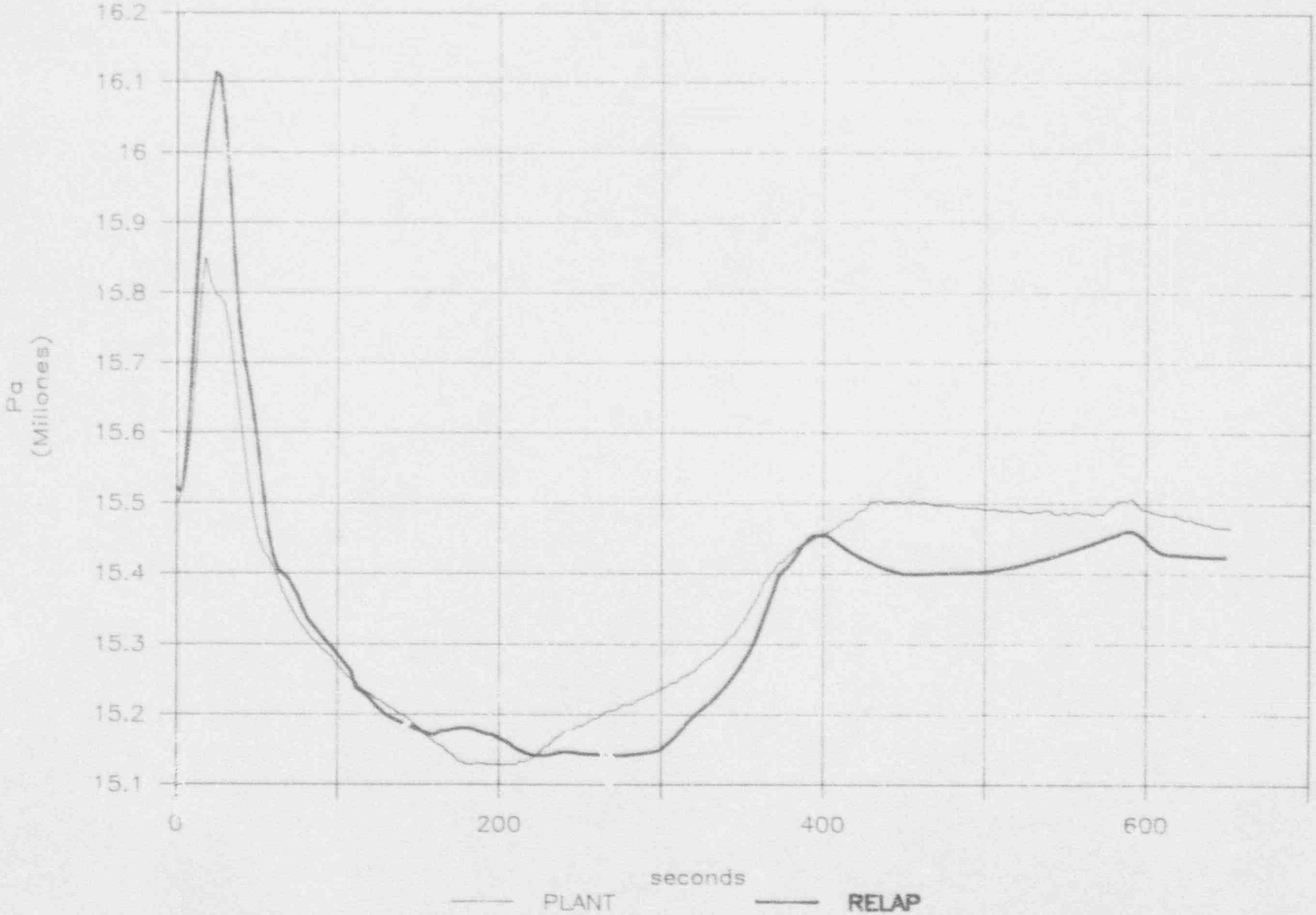


FIG.14 PRESSURIZER LEVEL

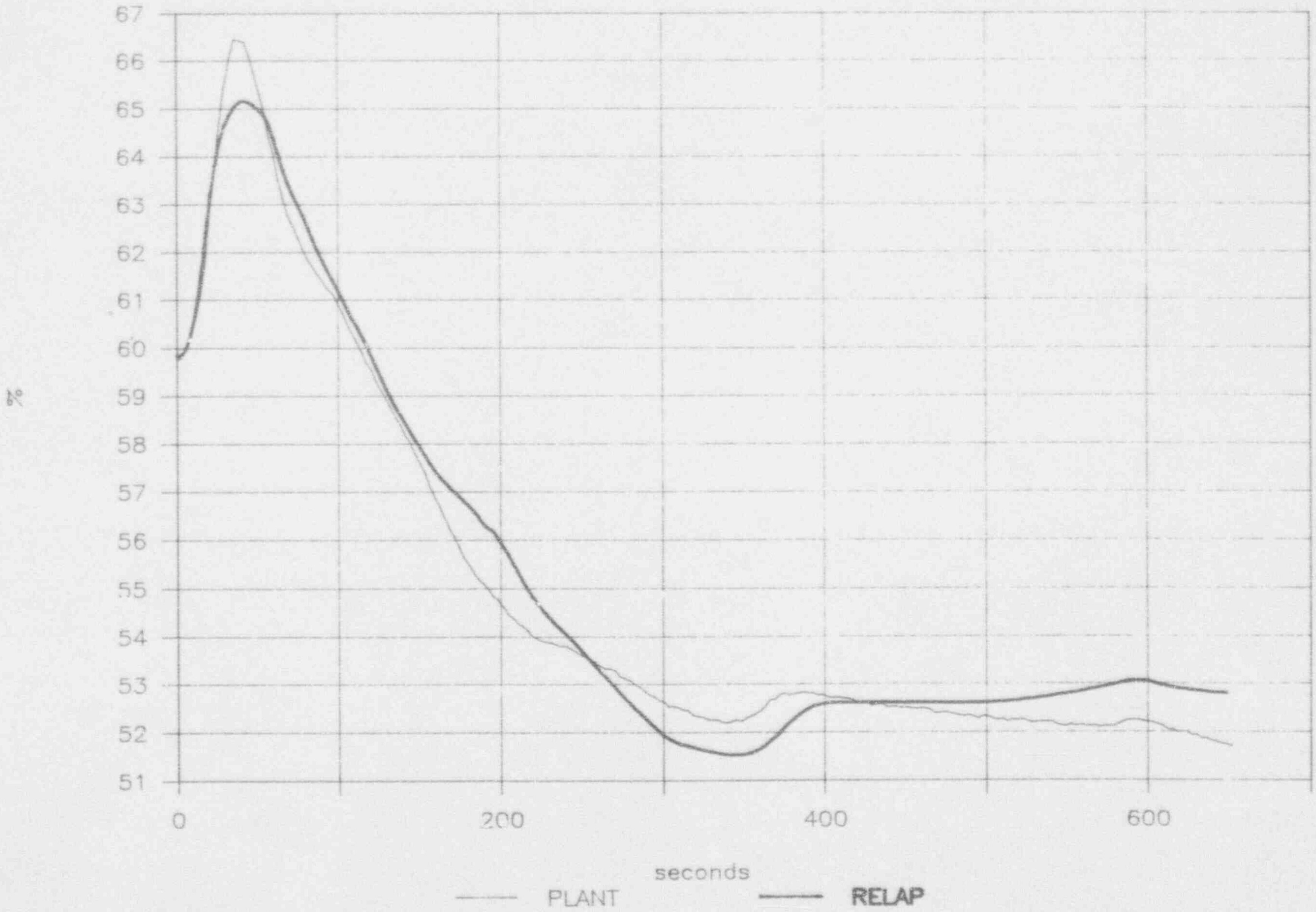


FIG.15 FEEDWATER MASS FLOW RATE

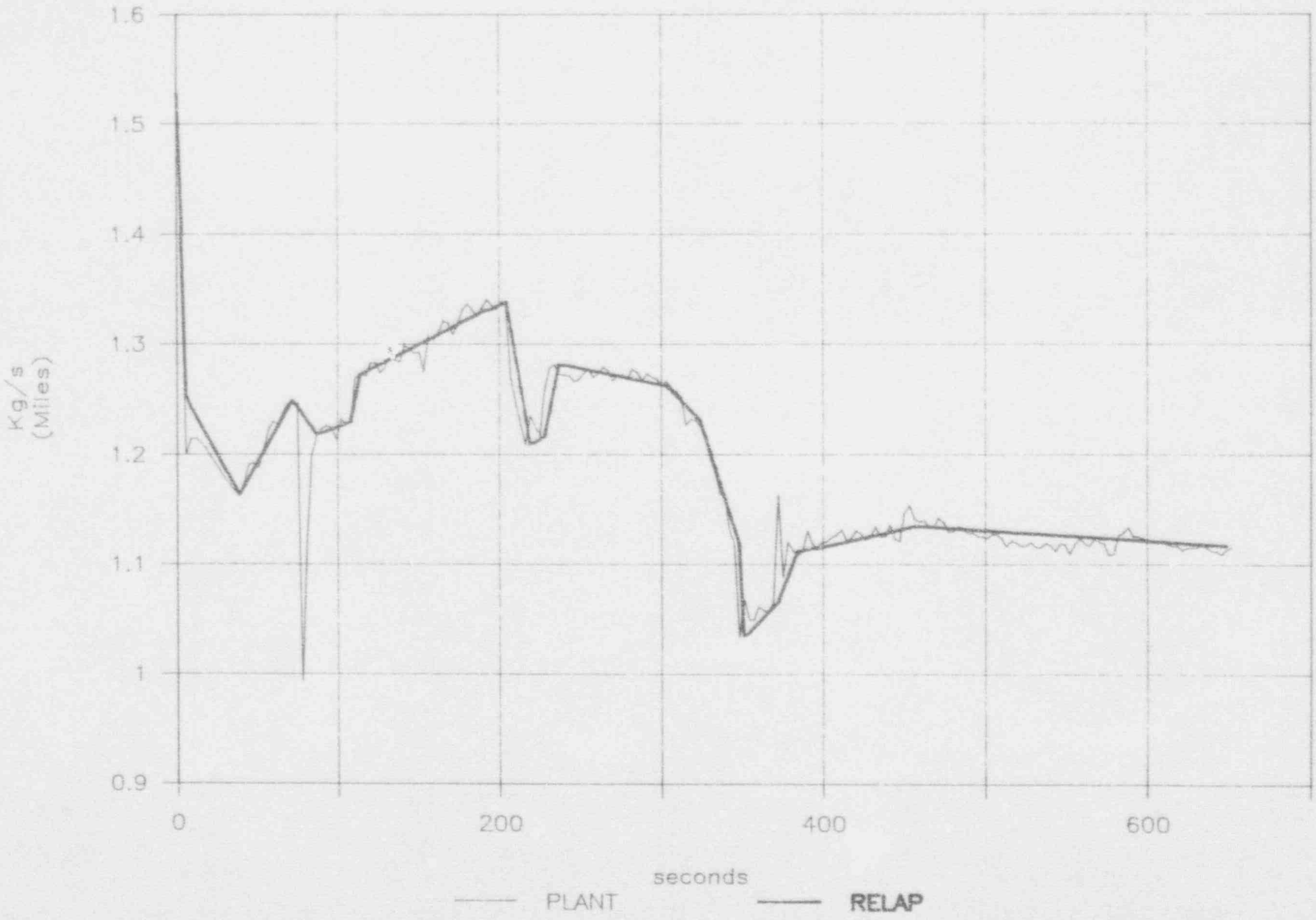


FIG.16 STEAM MASS FLOW RATE

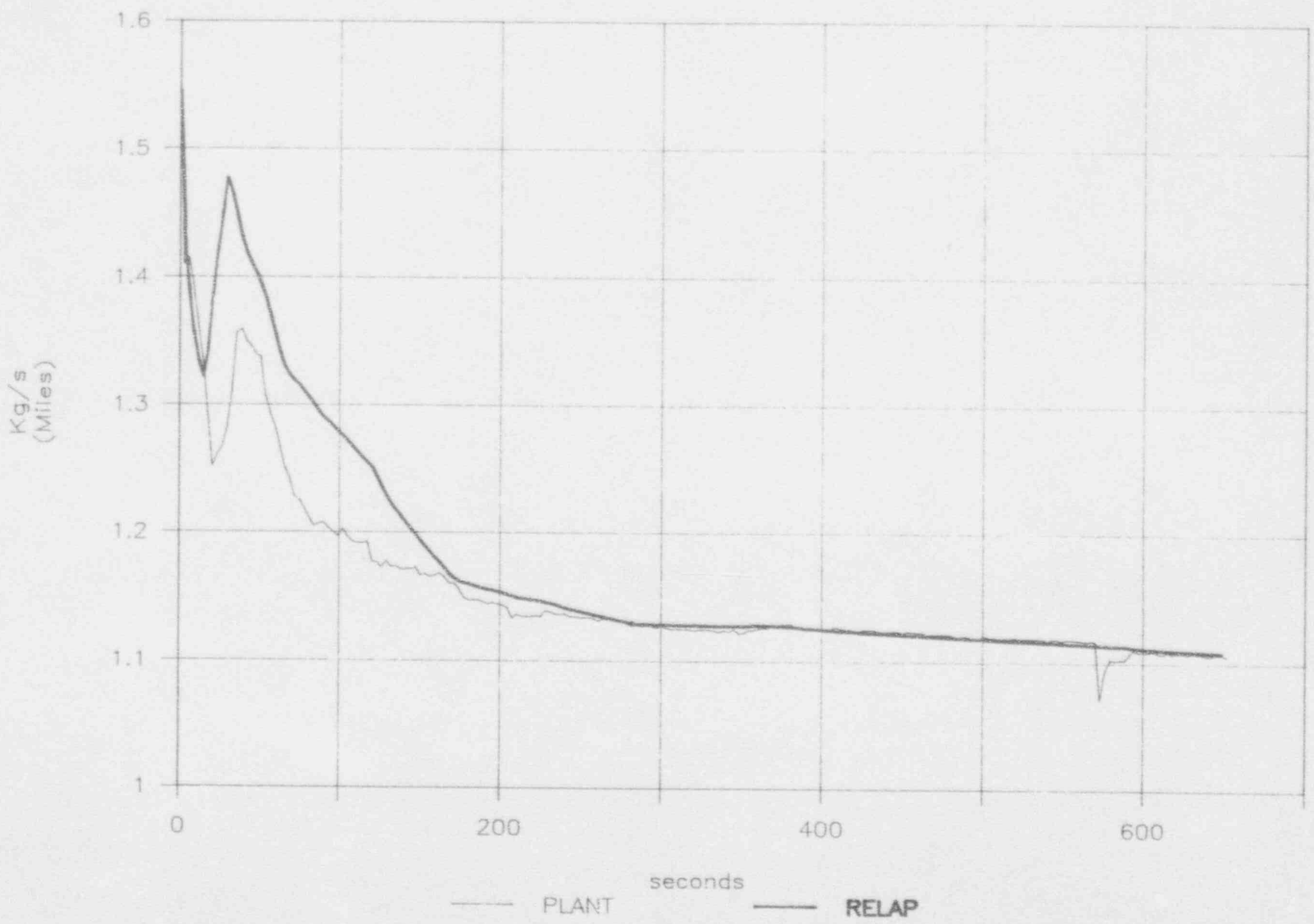


FIG.17 STEAM GENERATOR PRESSURE

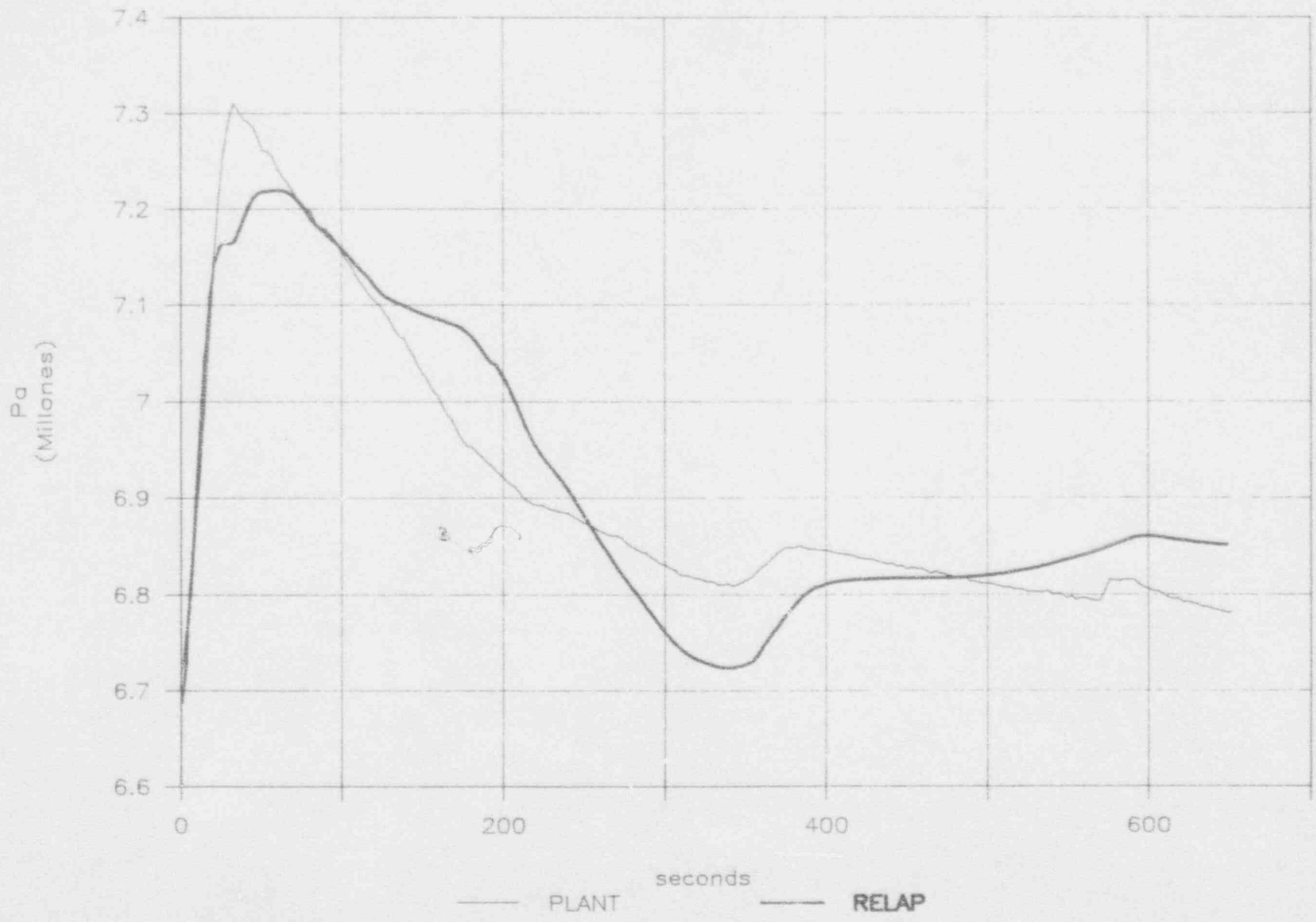


FIG.18 STEAM GENERATOR LEVEL

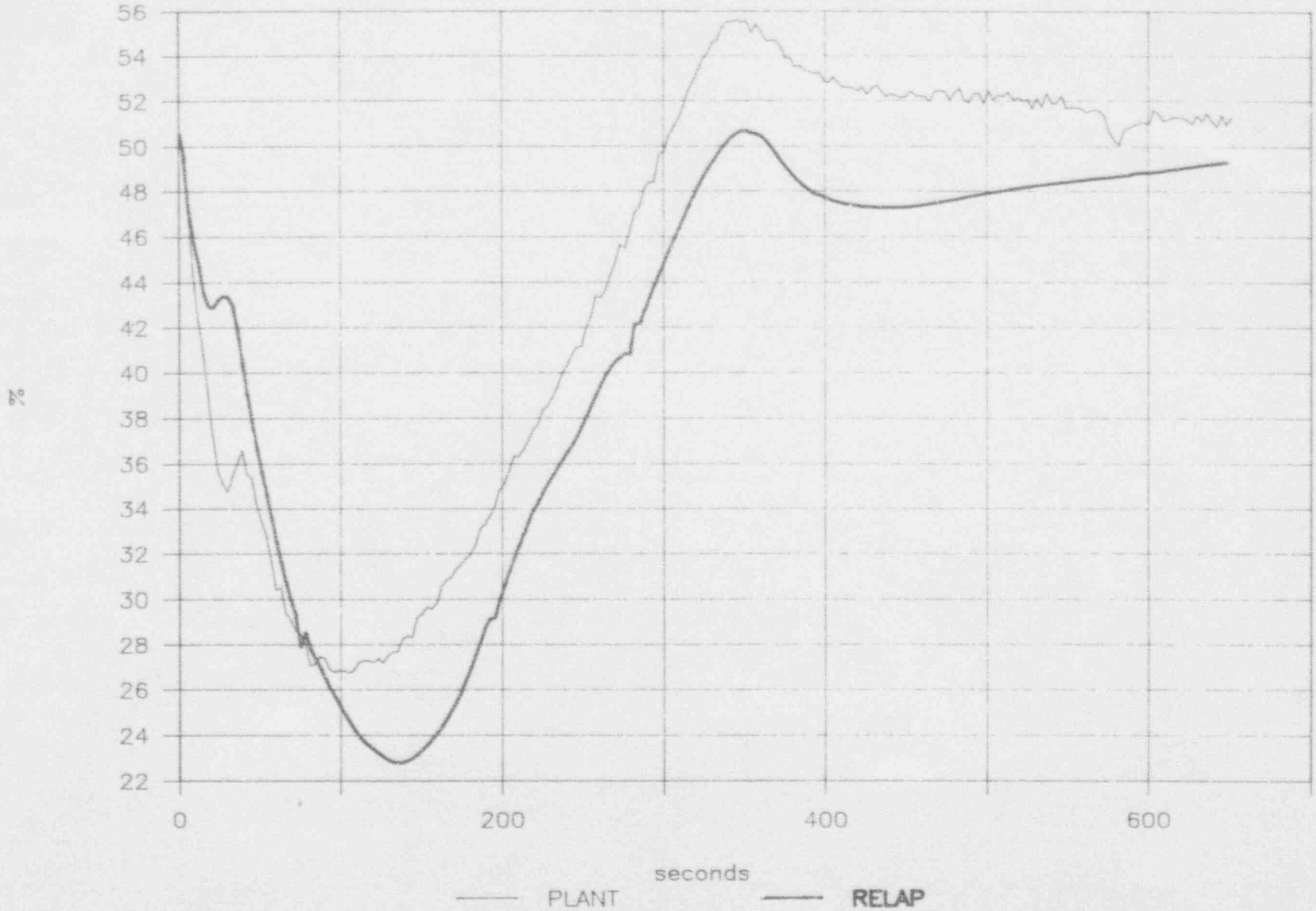


FIG.19 STEAM DUMP GR.1 VALVE POSITION

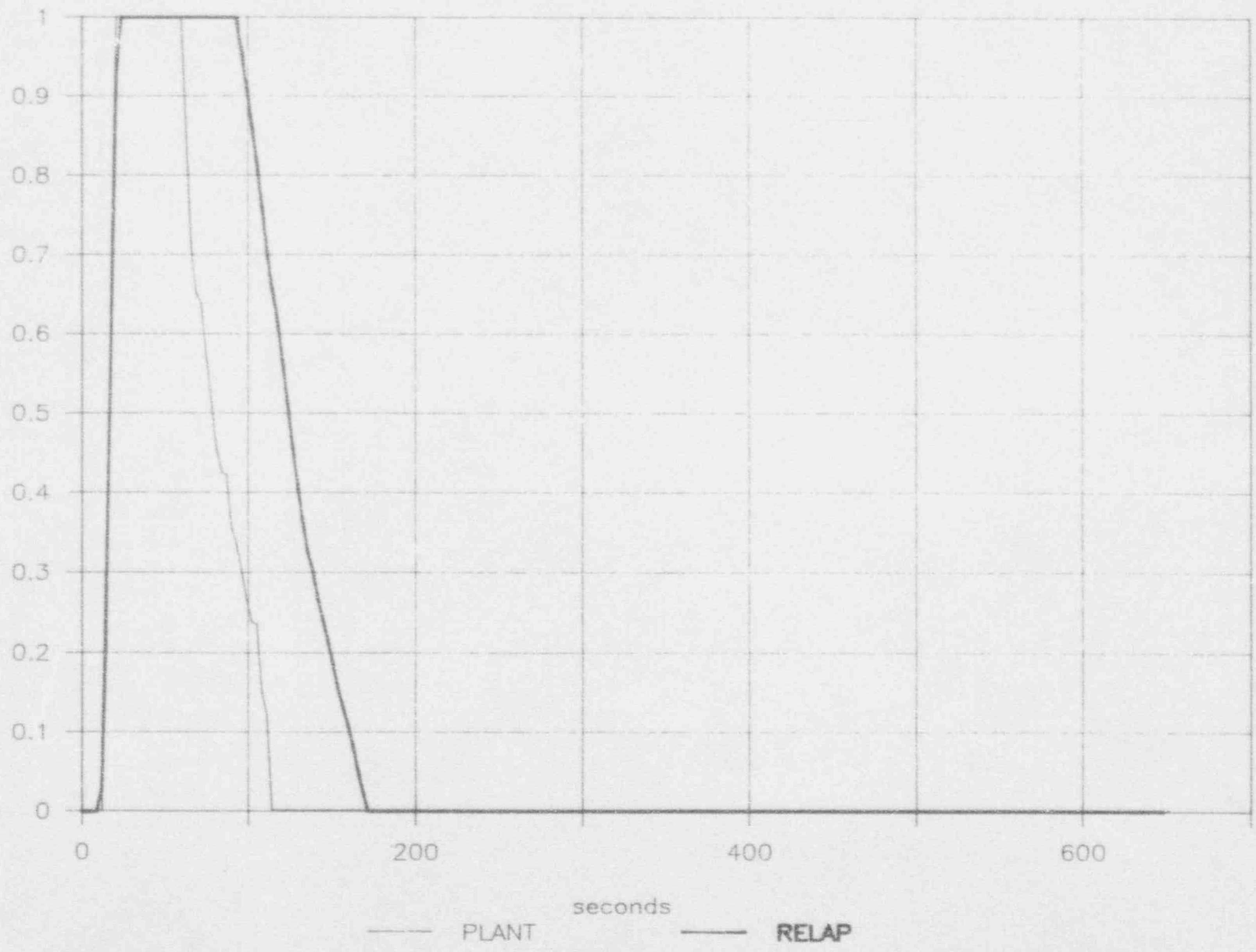


FIG.20 STEAM DUMP GR.2 VALVE POSITION

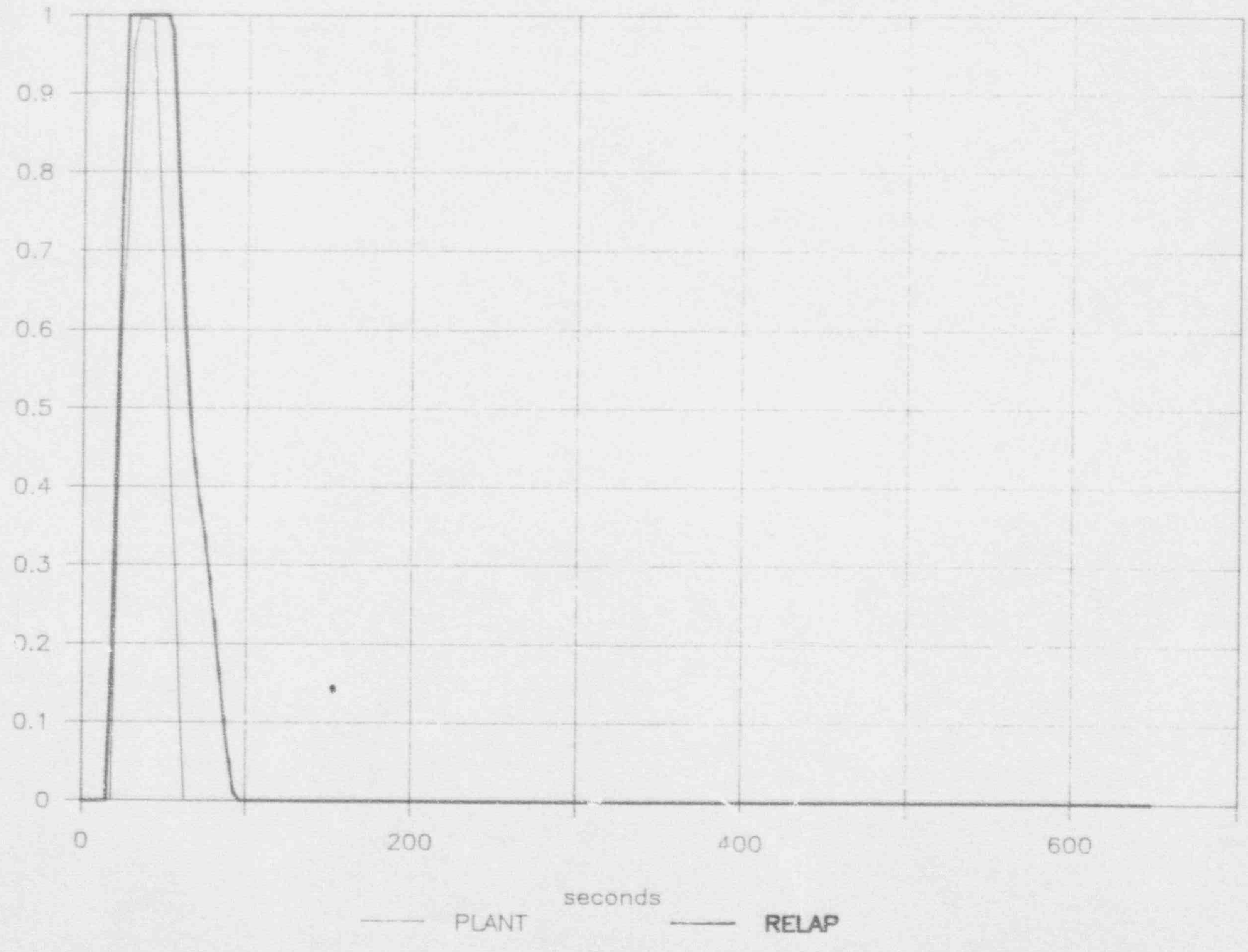
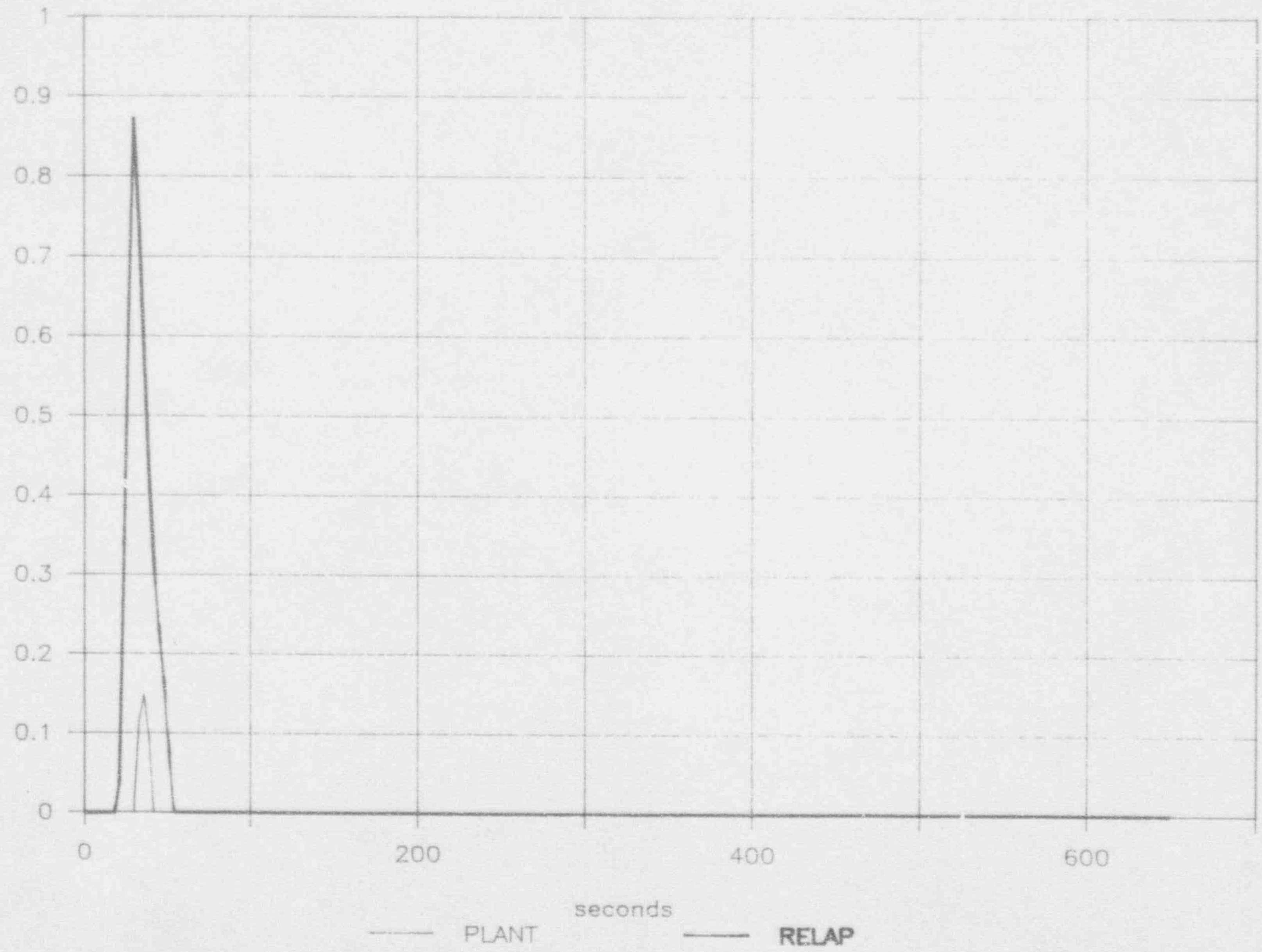


FIG.21 STEAM DUMP GR.3 VALVE POSITION



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-0110

ICAP00219

2. TITLE AND SUBTITLE

Assessment of RELAP5/MOD2 Against a Main Feedwater Turbopump Trip Transient in the Vandellós II Nuclear Power Plant

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	1993

4. FIN OR GRANT NUMBER

L2245

5. AUTHOR(S)

Carlos Llopis (ANV), Julio Perez (CSN), Albert Casals (ANV), and Rafael Mendizabal (CSN)

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 feedwater turbopump trip transient simulation. This transient occurred actually in plant on June 19, 1989.

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

ICAP.,
Vandellós II,
RELAP5,
Transient
Turbopump Trip

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

Printed on recycled paper

Federal Recycling Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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