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Assessment of a Reactor Coolant Pump Trip for TRILLO NPP with RELAP5/MOD3.2

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ABSTRACT

This document presents the comparison between the simulation results and the plant measurements during a Reactor Coolant Pump Trip in C. Trillo Nuclear Power Plant in different burnup conditions.

This work is part of the Spanish contribution to the Code Assessment and Maintenance Program (CAMP) as a member of UNIDAD ELÉCTRICA, S.A. (UNESA).

C. Trillo is a 3010 MWth three loop PWR Nuclear Power Plant designed by Siemens-KWU. Commercial operation started in August 1988, currently the Plant is running the 11th annual cycle.

The simulation has been carried out with the RELAP5/MOD3.2 code, running on a HP 715/50 work-station.

The transient main parameters have been correctly assessed, however predictions at the beginning of transient are more difficult as the reactor remains at a high power level after pump trip.

EXECUTIVE SUMMARY

C. Trillo is a 3010 MWth three loop PWR Nuclear Power Plant of Siemens-KWU design located on the Alcarria region (Central Spain) on the Tajo river.

The Thermal-hydraulic Analysis Group of C. Trillo has prepared a model of the plant using RELAP5/MOD3.2. This model, that has been checked by a wide process of model validation against start-up test data and transients includes the following characteristics:

- Three loops
- Detailed nodalization for Reactor Pressure Vessel
- Reactor point kinetic model with actual data of C. Trillo core for both BOC and EOC conditions (Two inputs set)
- Auxiliary systems simulation.
- Control, Limitation and Protection System.
- Secondary side from Feedwater Tank to Turbine Inlet including Bypass, Isolation, Relief and Safety Valves.

The areas of application of Relap5 model for C. Trillo are the following:

- Analysis of transients occurred during operation
- Analysis of design modifications before their plant implementation
- Answer to specifics questions of different C. Trillo departments
- PSA project, including large, medium and small leaks, transients and Anticipated Transients Without Scram
- Answer to CSN (Spanish Nuclear Safety Council) questions
- Operators training: calculations of operation manual instructions.

The transients calculated for this assessment are two Reactor Coolant Pump Trip at different core burnup conditions. Such transients, according to plant design did not result in reactor trip, instead power operation (at part load level) is continued due to the actuation of the limitation system that cut-back the permitted reactor and generator power to values compatible with two loop operation.

The main conclusion was that all significant plant parameters have been correctly assessed, although predictions at the beginning of transients are more difficult as the reactor does not trip to zero power.

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LIST OF ABREVIATIONS

ATWS	Anticipated Transient Without Scram
BOC	Beginning of Cycle
CSN	Consejo de Seguridad Nuclear
EOC	End of Cycle
KWU	Kraftwerk Union
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
SG	Steam Generator

I. Introduction

The objective of present work is to evaluate the predictability of the Relap5/mod3.2 /1/ on major thermal-hydraulic behavior during a Reactor Coolant Pump Trip in C. Trillo. Such a transient behavior includes the following phenomena: asymmetric loop behavior, control and limitation system actuation, short-term reactor power behavior owing to different moderator temperature coefficient, etc. To do this the calculation results are assessed and compared with the start-up test data D-100-303 /2/ (BOC conditions, first cycle) and with the RCP trip transient occurred on July 18, 1990 /3/ (near EOC conditions, 2nd cycle).

According to KWU design the Upset Condition "Failure of one main coolant pump" does not result in reactor trip, but power operation (at part load level) is continued instead.

Comparing the calculations with real data, overall transient behavior predicted by using Relap5/mod3.2 was in a general good agreement with plant data, and the following conclusions were obtained:

- Relap5/mod3.2 predicted well the sequence of events and the major phenomena during the transient.
- There is increased difficulty on plant assessment calculation when reactor scram does not occur and plant remains at a high power level.
- Short-term behavior depends on core burn-up conditions, as reactivity feedback depends on core burn-up too.

II.-Plant Description

C. Trillo is a Three Loop PWR Nuclear Power Plant designed by KWU-Siemens. Commercial operation started in August 1988 and Plant Dynamic Start-Up Test took place in the spring and summer 1988.

C. Trillo has a nominal reactor power of 3010 MWth at an operating pressure of 158 bar and an average coolant temperature of 310 °C. The reactor core consists of 177 fuel assemblies, each containing 236 fuel rods in a 16×16 square array. All major plant components are similar to those of other PWR plants. The values of main parameters are listed in table 1.

The reactor pressure vessel is the fixed point of the reactor coolant loops whilst the reactor coolant pumps and steam generators are flexibly mounted. The reactor pressure vessel internals are basically composed of the lower core structure with flow skirt for flow distribution, and the upper core structure containing the control rod guide assemblies.

The three steam generators are designed as vertical U-tube bundle heat exchangers with preheaters.

The reactor coolant pumps are single-stage centrifugal pumps with overhung impeller and vertical shaft. They are driven by an asynchronous motor through a gear coupling. A flywheel prolongs pump coastdown time.

The pressurizer consists of a cylindrical shell, an upper and a lower hemispherical head. Its lower section is filled with water, the upper part with steam. The steam volume can be increased or reduced by heating the water or by spraying water into the steam cushion as necessary to regulate the operating pressure. The heating elements are flanged into the lower head. The nozzles for the spray lines are located in the upper head together with piping connections for the safety valves, the relief valve, and the connecting line to the RPV vent system.

A too high pressure rise in the reactor coolant system is prevented by the two pilot operated safety valves which discharge into the pressurizer safety tank. A relief valve, electrically controlled, is provided in addition to the safety valves as a means of limiting pressure increase. Its actuating pressure is lower than that of the safety valves. Safety valves pressure is only reached in case of a highly improbable transient (e.g. ATWS).

The reactor coolant system is supported by the auxiliary systems. A number of the nuclear auxiliary systems are directly involved in reactor operation for coolant injection and extraction, clean-up, degasification, adjustment of coolant boric acid concentration for reactivity control and admixture of chemicals to the coolant.

The emergency core cooling and residual heat removal system is of a three plus one train design connected to both cold and hot RCS legs. Each subsystem consists of:

- One safety injection pump that takes borated water from the storage tanks and deliver it via a logic circuit to the corresponding hot leg of primary side. A rupture in the hot side causes the actuation of logic circuit to feed via the intact cold side line.
- Two accumulators connected each to the hot ad cold side injection lines.
- One residual heat removal pump that sucks water parallel to the safety injection pump from the water storage tanks or, after these are empties, from the containment sump. The injection into the primary in also through the hot and cold side lines.
- Two borated water storage tanks.

In addition to those, there is a fourth train with one safety injection pump and one residual heat removal pump with the corresponding water storage tanks, that works as a spare system for any of the other three subsystems.

The steam generator is a standing U-tube bundle heat exchanger. The heating medium is the reactor coolant which enters the hot inlet chamber, passes through the U tubes and transfers the heat generated to the feedwater/steam cycle.

Three 50% capacity feedwater pump units, one as standby, deliver the feedwater through the high pressure heater system to the steam generators. The feedwater inlet temperature to the SGs is 220 °C at full load.

Each steam generator is assigned a separate feed station comprising low and full-load control valves. Each one of these main feedwater lines is divided on reaching each steam generator into three lines of 40%, 50% and 10%. The 10% line leads to a ring of sprinklers in the steam zone of the steam generator. The 40% and 50% lines lead to the economizer where feedwater is heated to just below boiling temperature, after leaving the economizer it mixes with the circulating water. Approximately 50% of the feedwater flows upwards through the preheater in a direction opposite to the primary coolant, and 40% flows downward. This so-called "split flow" design protects the tube sheet against cold feedwater shocks and, on the other hand, grants a high thermal efficiency of the steam generators. The steam generation occurs under natural circulation. For this purpose the tube bundle is surrounded by a guide shell extending above tube bundle and connecting to moisture separators. At full power, the secondary main steam pressure is 68.5 bar. The water level in the steam generators is kept constant by the feedwater control valve station.

The steam and feedwater system are designed such that the residual heat transported to the steam generators can be transferred to the circulating water in the condenser via the main steam bypass station after completed shutdown of the turbine generator unit. The main steam bypass station is capable of dumping approximately 60 percent of the full load steam flow to the turbine condenser in a controlled manner so that on turbine trip the reactor does not trip as well; in this case only controlled reactor power reduction is necessary. When main steam bypass station is not available, the main steam pressure rises are limited by main steam relief and safety valves.

The start-up/shutdown feedwater system comprises two electrically driven pumps connected to the feedwater tank which supply the steam generators via the low load control valves during start-up and shutdown operation. Those pumps start up automatically if all the main feedwater pumps fail and, being connected to the emergency diesel, are also capable of shutting down the plant on loss of auxiliary power without resort to the emergency feedwater pumps.

The reactor control systems are designed in such a way that the reactor process operates at its optimum operating point. The reactor control system actuates on control rods and boron/water injection.

The core protection system initiates actions to prevent accident conditions when deviations occur which the process controls can not correct. The trip system monitors the process variables which are significant to the safety of the reactor plant and the environment for the purpose of prompt detection of accident conditions, measurement of data processing and automatic initiation of protective actions to keep plant conditions within safe limits. If accident conditions occur, measures to mitigate their consequences are initiated automatically.

A significant feature of Siemens-KWU PWR plants is the so called *Limitation System*, which plays an intermediate role between Control and Protection Systems.

The limitation system has the following tasks:

- Holding of reactor power below set limits by limiting integral reactor power.
- Prevention of unacceptable local loading on fuel or fuel cladding by appropriate control of power distribution and limitation or reduction of reactor total power.
- Assurance of an adequate shutdown reactivity margin by limitation of control rod bank movement.
- Supervision of performance of control rod dropping after actuation of reactor scram.
- Prevention of draining or overfilling the pressurizer by limitation of coolant inventory and maintenance of coolant pressure within set limits actuating heaters and spray valves.

The limit settings and the countermeasures associated with these are selected on the basis of the related safety requirements such that conditions are no more severe than the initial conditions assumed in the safety analyses and such that process upset variables are returned within their normal operating values.

III.- Model Description.

The C. Trillo model is shown in figures 1, 2 and 3. It consists of 303 volumes, 351 junctions, 163 heat structures and 944 control variables. The three loops are simulated with actual data specific for each loop in order to have the possibility for simulating non symmetric transients and to understand the different loop behavior.

III.a.- Primary System.

The model of the primary system includes all the main components of the system. In the reactor vessel elements the volumes corresponding to the downcomer, lower and upper plenum, the core, core bypass and the upper head are defined. In the nodalization of the upper part of downcomer (volumes 510 and 540), special care was taken to correctly simulate the steady-state mass flow rates in the connection with the hot legs (bypass) as well as with the upper plenum and upper head (unintentional leak paths), so that to obtain the requested values. The downcomer is represented by the components 530 to 538. The lower plenum consists of the nodes 560, 570 and 571, while the core bypass is represented by node 578.

The core was subdivided to two parallel channels that are interconnected by junctions to allow cross flow exchange. The external channel (15% of the core volume) is represented by nodes 581 to 588, while the larger channel (85%) consists of nodes 591 to 598. The power and fuel are distributed in the same proportion. The exit of the core is represented by nodes 500 and 501 and upper plenum by volumes 511 and 512. Above the reactor coolant loops, the upper plenum consists of nodes 513 and 520, the vessel head of nodes 548, 549 plus the pipe 550.

The fuel rods have been simulated as a cylindrical heat structure with an internal heat source whose power derives from point kinetics calculation. The total reactor power is calculated as the sum of prompt fission power plus decay power from fission fragments.

The built-in data for fission products and actinides have been used, but data for delayed neutrons have been entered from specific kinetics calculation.

The reactivity feedback model used assumes separability of feedback effects. Two tables, one defining reactivity as a function of coolant density, the other as a function of volumetric average fuel temperature, have been supplied using Trillo-specific (BOC and EOC) data.

For the first case, simulating the start-up test, data from KWU input of transient analysis corresponding to the first core has been used /5/.

KWU supplies, each reload, nuclear results for the final pattern calculated by the code MEDIUM. The data of end of second cycle (89-90) has been used for the calculation of case 2.

Control variables simulating the control rods reactivity contribute to reactivity feedback calculations.

The control rods reactivity is calculated by a simulation of control rods movements caused by temperature or power deviations, assuming that reactivity is a function of control rod position.

The three loops are analogously simulated. The hot legs are modeled by volumes 100, 200 and 300. The steam generator inlet chambers are simulated by nodes 110, 210 and 310. The tubes volumes (120, 220 and 320) are divided into 8 cells each. Proper heat structures are used to link the primary to the secondary side of steam generator where real data are used.

Steam generator outlet is modeled with volumes 130, 230 and 330. Nodes 140, 240 and 340 simulate the intermediate leg between seam generator and reactor coolant pump.

The reactor coolant pumps are modeled by volumes 150, 250 and 350 by using specific homologous curves supplied by the vendor. Finally, the cold legs are modeled by volumes 160, 260 and 360.

The pressurizer has been modeled by means of volume 410 divided into 3 cells, the upper head is the node 415 and bottom zone is represented by volume 402. Safety and relief valves are connected to the steam filled space (volume 415) and discharge into the relief tank dome (468). Volume 450 models the spray line, while the mass flow is controlled by the valve 451. The pressurizer heaters are modeled by means of a heat structure governed by control system.

The following auxiliary systems are connected to the primary system:

- Volume Control System (TA). The volume control system is connected via three charging lines each to the cold leg. The extraction point is located between the steam generator and the reactor coolant pump. In addition, the volume control system is connected to a pressurizer spray line for the purpose of auxiliary spraying if necessary.
- Residual Heat Removal System (TH). The TH system is divided into four parallel subsystems. Three of them are directly assigned to the reactor coolant piping loops and connected via a hot and cold injection lines. Each of the three subsystem consists of 1 safety injection pump, 2 accumulators, 1 residual heat removal pump and 2 borated water storage tanks. In addition there are one safety injection pump and one residual heat removal pump with the corresponding water storage tanks, as a spare system for any of the other three.

Extra borating system (TW). The TW system is connected to the reactor coolant piping with lines from the storage tanks of TH system and own borating tanks.

III.b.- Secondary side

The three steam generator secondary sides are simulated identically, so only loop 1 is briefly described in the following :

The downcomer (volume 600), modeled as an ANNULUS component, divided into 3 cells is connected to volume 640 and to the bottom tubes volumes (602 and 604). The steam generator preheater is simulated by volumes 604 and 606. In its turn volume 604 is connected to the 40% feedwater junction and located below volume 606, which is itself connected to the 50% feedwater valve.

The tube area is modeled by means of volumes 602 and 610. The first one is divided into two nodes and connects the tubes plate to the top of the preheater. It is thermally connected to the primary side by heat structures. Volume 610 covers from the top of volumes 606 and 602 to the steam separator inlet.

Volume 620 models both steam separator and steam dryers. It is defined as a separator component and is connected to the riser, steam generator dome and downcomer. Volumes 635 and 636 simulate the area outside the separators and dryers.

Volumes 660 and 670 account for the steam line lying from the outlet of steam generator to the inlet of the steam header (volume 910). A line containing the turbine stop and control valves connects volume 910 and 940, which represents the turbine. There is also a line that connects volume 910 to a time dependent volume which represents the condenser (920) through the bypass valve 915 (SF system).

Each steam generator is provided with its own main steam , feedwater, emergency feed and blowdown lines (figure 2). The following secondary systems are associated with the steam generator:

- Main Steam System (RA). The steam generated is passed through three parallel main steam pipes (660, 760, 860) to the main steam header and from there to the turbine valves (912). The isolation (661, 761 and 861), relief (662, 762 and 862) and safety (663, 763 and 863) valves are simulated with their corresponding control system.
- Main Feedwater System (RL). The main feedwater pumps (32, 33, 56) deliver the feedwater from the tank to the steam generators via headers through the high pressure heaters. Each feedwater line is provided with a full load and low load

control valve. The required control, check and isolation valves are simulated (Figure 3).

- Startup and Shutdown System (RR). The startup and shutdown pumps supply the steam generator via the low load valves of feedwater system (62, 72, 82) bypassing the high pressure heaters.
- Emergency Feedwater System (RS). This system supplies the steam generators in the event of malfunctions accompanied by failure of the main feedwater and startup and shutdown system.
- Steam Generator Blowdown (RZ). This system is used to maintain a certain water quality in each steam generator and to detect heating tube leakage as well.

III.c.- Control, Limitation and Protection Systems

The more important systems regarding the dynamic behavior of the plant have been simulated. A brief description of each of them follows: (for a detailed description see Ref /4/)

- Control System
  - Average Coolant Temperature Control (KMT) . In steady-state operation the setpoint for average coolant temperature traces the appropriate characteristic on the part-load diagram. The average coolant temperature is measured in all reactor control loops and processed by a controller. The control deviation is formed by comparing with the setpoint and regulates control rods movements.
  - Secondary Maximum Pressure (PSOMAX). Its mission is to limit pressure increases in secondary side. When pressure gets larger than setpoint (not constant but function of plant conditions) the condenser bypass opens keeping pressure below a specific value avoiding safety and relief valves actuation.
  - Turbine Control. The turbine control valve is governed by the demanded power and the secondary minimum pressure setpoint.
  - Feedwater. This system controls feedwater flow as a function of both steam flow and steam generator level.

- Coolant Pressure Control. This system maintains the controlled variable, coolant pressure, within specified limits of the setpoint by the use of the pressurizer heaters and spray valves to control the water/steam volume in the pressurizer.
- Pressurizer Water Level Control. The task of this system is to maintain pressurizer water level at a preset setpoint under all operating conditions by using the charging pumps and the HP reducing station.

- Limitation System.
  - Permitted Reactor Power (PERL). The "permitted reactor power" signal sets the level to which reactor power is limited depending on the instantaneous plant condition. The following events are taken into account explicitly: Fast shutdown of the reactor, steam generator tube leak, malfunction of one reactor coolant pump, faults in feedwater supply, power density limitation and faulty control rod position measurement.
  - Permitted Generator Power (PERG). This setpoint is intended for limiting generator power and is itself a function of PERL.
  - Reactor Power Limitation Module (L-RELEB). The momentary reactor power level is continuously compared with the permitted reactor power level. The difference signal is compared with a series of limit signals with step-raised settings that actuates countermeasures of increasing intensity.
  - Reactor Power Limitation on Reactor Coolant Pump Malfunction (PUMA-RELEB). A few seconds after malfunction of one reactor coolant pump the permitted reactor and generator power levels are decreased and control rod insertion (STEW-PUMA) is actuated.
  - Reactor Power Limitation on Basis of Feedwater Mass Flow (SPEISE-RELEB). In the event of malfunction of main feedwater pumps in operation at high reactor power levels, permitted reactor and generator power are quickly reduced to prevent reactor trip caused by low steam generator level or high primary pressure.
  - Primary- Secondary Overpower (KOL-RELEB). This system prevents any increase in power which would otherwise lead to steam dump to the atmosphere. Limit signal results in an increase block for permitted reactor

power level and in the actuation of the countermeasures derived from this.

- Average Coolant Temperature Limitation System (KMT-RELEB). Its mission is to limit temperature excursions that the control system can not handle. A series of three limit signals with step-raised settings actuates countermeasures of increasing intensity where required.
- Reactor Power Limitation on High Energy Content in Primary Loop (LOOP-RELEB). For comparison with the preset limits for this system the pressurizer water level, coolant pressure and positive coolant pressure gradient are summated, appropriately weighted. On actuation of LOOP-RELEB control rod movement command are issued.

- Protection System. A list of Reactor Protection System actuation is presented bellow:
    - Reactor Trip (RESA). Reactor trip actuation and turbine trip actuation are automatically effected by the reactor protection system on the basis of the same criteria. Trip is effected if any of the limit values is fulfilled.
    - Cut out of the Reactor Coolant Pumps.
    - Feedwater Delivery Interruption.
    - Closing of Main Steam Isolation Valve.
    - Actuation Criteria for Emergency Core Cooling System (injection).
    - Automatic Cooldown by 100 K/h.
    - Closing of Isolation Valve upstream of Main Steam Safety Valve
    - Main Steam Safety and Relief Valves (opening/closing)
    - Pressurizer Relief and Safety Valves (opening/closing)

IV. Transient Description

The event "failure of one main coolant pump" is detected by the decreasing speed of the pump motor. As soon as pump speed falls below 94%, the reactor and generator power are reduced to about 40 and 30% resp. by the limitation system. The reduction of reactor power is performed by means of the so-called STEW-synchronous, i.e. a predetermined number of control rod elements fall into the core in order to reduce the reactor power to values near 40%. In the long term the reactor power remains at a level of about 30%, corresponding to the turbine power.

The decreasing pump speed causes a corresponding flow reduction that results in flow reversal in the affected loop, this also results in a reduction in the core flow rate leading to a decrease of pressure loss over the vessel, and as a consequence to a higher delivery rate of the running pumps.

After complete coastdown of the affected pump a new steady state condition is established, whereas the elevated flow in the other loops is distributed at vessel inlet into a major part for core flow and a minor part streaming backward through the idle loop.

The heat removal from primary side mainly takes place in the steam generators of the intact loops whereas the heat transferred in the affected loop is small in accordance with the new conditions.

The main steam flow rates in the intact steam generators follow the trend of the heat transfer rates. The main steam bypass station opens only for a short time (20 s) until a new equilibrium between reactor and generator power is reached.

The primary coolant average temperature is reduced in correspondence with a temperature setpoint which is automatically adapted to a 2-loop operation condition.

The control rod injection initiated automatically warrants the continuation of power operation in the event of a main coolant pump failure. This automatic power reduction proves as sufficiently fast and efficient with respect to both time and magnitude of power cut. With exception of the control rod injection due to limitation system actuation only control system intervenes during the event and manages the plant to attain a new part load steady state condition. The reactor protection system does not respond at all.

The general transient behavior, described above, is similar both at BOC and EOC cases except for the earlier phase before final stabilization.

The different reactivity coefficients have an influence on the first seconds of the transient when the control rods fall into the core. At EOC conditions the higher and more negative value of moderator temperature coefficient implies a stronger rise of reactor power, and consequently also in other plant variables (pressure, temperature...). This behavior is not observed when the transient occurs at BOC conditions with a moderator coefficient near zero.

Figures 4 to 10 present the comparison of different plant variables at BOC and EOC conditions.

V.-Relap5/Mod3.2 Simulation

The transient has been simulated with the C. Trillo model presented on figure 1. All actions, except the initiating event (loop 3 RCP trip) are due to automatic actions of control and limitation systems.

The initial steady-state conditions were obtained from new transient run up to 200 seconds each. Two steady-state calculations were performed for BOC and EOC conditions to simulate the particular conditions of the plant when the pump trips were initiated. The main difference between the initial states was the primary average temperature, so the calculated steady-states have to be adapted, by the modification of temperature setpoint to the real conditions measured in plant.

The results obtained from the steady-state calculations are compared with the measured initial conditions in Table 3.

Two set of figures are presented. The first corresponds to BOC conditions, with data of the start-up test D-100-303 and the second compares the transient simulation at EOC conditions with the data of the transient occurred on July, the 18$^{\underline{th}}$ 1990.

Comparing the calculations results with the plant data, it was found that the Relap5/Mod3.2 code predicts well the RCP trip transient behavior and the timing of sequence of events.

V.a .- Reactor Coolant Pump Trip at BOC.

Figure 1.1 presents the initiating event, trip and coastdown of loop 3 reactor coolant pump.

The insertion of control rod elements is actuated by the limitation system which detects the signal "pump speed $< 94\%$ and reactor power $> 65\%$". Selected pairs of control rod elements drop into the core reducing the reactor power to approximately 40%, as can be seen in figure 1.3.

The subsequent behavior of reactor power depends on primary average temperature, because control system is active during the whole transient. Figure 1.6 presents the comparison between calculated and measured temperatures: general behavior is similar, but some minor differences appear when temperature oscillates near to the setpoint. In the final part of the transient both temperatures converge to a common value defined by the two loop part load diagram.

The calculated and measured temperatures in loops 1 and 3 are presented in figures 1.4 and 1.5, an overall good agreement was obtained for these variables.

The transients of pressurizer water level (figure 1.8) and coolant pressure (figure 1.2) reflect the temperature transients, the general behavior is very similar and in the final stabilization converge to the same value limited by the control system.

Figure 1.3 presents values of thermal corrected reactor power, this variable is not measured directly in the plant but calculated as a function of thermal reactor power (sum of the temperature rises of the three loops) and neutron flux signal. This signal approaches the thermal reactor power during steady state and the neutron flux signal during fast power transient.

The Relap calculated thermal corrected power uses the thermal power calculated as sum of temperature rises and the fission power calculated by the code. The results do not compare well because of plant treatment of data for this case with one loop inactive, is different to Relap variables.

V.b .- Reactor Coolant Pump Trip at EOC.

The simulation corresponds to a real transient occurred in the plant on July 18, 1990, the core conditions coincide to a near EOC state.

At end of cycle (EOC) the reactor behavior is significantly modified by coolant temperature changes since the temperature reactivity coefficient is more negative than at beginning of cycle, then the influence of coolant temperature changes on reactor power is larger. The more negative moderator temperature coefficient leads to a reactor power rise after the first reduction (Figure 2.3), due to average temperature decrease (figure 2.6), that reaches the limiting value of permitted reactor power (not observed in the previous case) causing a new insertion of control rods. Unfortunately plant records for reactor power or control rods movements are not available and it is not possible to know exactly the sequence and the precise chronology of events, but by looking at the average temperature behavior it can be deduced that a second control rod insertion has occurred in plant transient.

Figure 2.2 presents the evolution of primary and secondary pressure and their correspondent Relap calculations that show a coherence with the temperatures behavior (figures 2.4, 2.5 and 2.6). The largest difference between measured and calculated data occurs after the transition to reversal flow in the affected loop.

VI.-Conclusions.

The Relap5/mod3.2 was assessed using C. Trillo real data for two different reactor coolant pump trips. An overall good agreement was obtained although some minor differences were observed.

The results demonstrated that the C. Trillo nodalization with the control, limitation and protection system could successfully describe the widespread behavior of plant transients and the following conclusions were obtained:

- Relap5/mod3.2 predicted well the transient behavior during the pump trip, including variables such as the primary and secondary pressure, secondary mass flow rate, and primary temperatures. The calculation results also predicted well the sequence of events and the major phenomena during the transient, such as the asymmetric loop behavior, the pump coastdown and the primary to secondary heat transfer.

- C. Trillo model for Relap5/Mod3.2 has shown to be a valuable tool to analyze Plant transients and to explain the behavior of main variables as in the 2nd case that a new control rod insertion can be deduced by the code results.
- There is increased difficulty on plant assessment calculation when reactor scram does not occur and plant remains at a high power level.

VII.- Run statistics.

The calculations were carried out with the RELAP5/MOD3.2 code, running on a HP 715/50 work-station.

The run statistics are presented in the table 4.

VIII.- References

/1/ Relap5/Mod3 Code Manual - Code Structure, System Models and Solution Methods. NUREG/CR-5535, EGG-25596, Idaho Falls, Idaho, June 1995.

/2/ Commissioning Result Report D-100-303

/3/ Informe de Incidente Operativo: Disparo de una bomba de refrigeración principal. Julio 1990.

/4/ Limitation Systems in the Reaktor-Leittechnik. R 193-82/e 11 h

/5/ Description of the Computer Model LOOP 7 for Calculation of the "Anticipated Transients without Scram" (ATWS) R11/103/75 E

Table 1.-Technical Data of the Reactor Coolant and Pressurizer System. (1/4)

Principal Data	
Thermal Reactor Power	3010 MW
Power transferred by the steam generators	3028 MW
Number of reactor coolant pumps	3
Total coolant flow through RPV at full load	16602 kg/s
Normal operating pressure at the RPV outlet	158 bar
Coolant temperature at zero load hot	296.7 °C
Coolant temperature between RPV and SG at full load	325.8 °C
Coolant temperature between RCP and RPV at full load	294.5 °C
Coolant temperature between SG and RCP at full load	294.3 °C
Mean coolant temperature of full load	310.0 °C
Pressure loss in the reactor coolant system at full load	6.6 bar
Water volume incl. full load water volume of pressurizer	300 m^3
Test pressure for hydrostatic pressure testing	227.5 bar_g
Temperature for hydrostatic pressure testing	
- min.	45 °C
- max.	55 °C
Reactor Pressure Vessel	
Design pressure	176 bar
Design temperature	350 °C
Normal operating pressure	
- Inlet	161.1 bar
- Outlet	157.3 bar
Operating temperature	
- Inlet	294.5 °C
- Outlet	325.8 °C
Water volume	123.4 m^3
Pressure loss across RPV at full load	2.8 bar
Fuel	Sintered UO_2

Table 1.-Technical Data of the Reactor Coolant and Pressurizer System. (2/4)

Number of fuel assemblies	177
Number of fuel assemblies with control assemblies	52
Total number of fuel rods	41772
Number of guide thimbles per fuel assembly	20
Arrangement	Square lattice
Overall length of fuel rods	4185 mm
Active length of one fuel rod at full load	3414.6 mm
Outside diameter of fuel rods	10.75 mm
Reactor Coolant Piping System	
Operating pressure on discharge side of RCP at full load	161.2 bar
Design temperature	350 °C
Design pressure	176 bar
Inside diameter	750 mm
Water Volume (three loops)	29.6 m^3
Flow velocity between RPV-SG at full load	18.8 m/s
Flow velocity SG-RCP and RCP-RPV	17.0 m/s
Coolant flow in cold leg of RCL	7.43 m^3/s
Steam Generators	
Number	3
Power transferred per SG at full load	1009.3 MW
Heat surface of the heating tubes per SG approx.	5400 m^2
Number of heating tubes per steam generator	4086
Primary Side	
Flow per steam generator at full load	5534 kg/s
Operating pressure of medium approx.	160 bar
Inlet temperature at full load	325.8 °C
Outlet temperature at full load	294.3 °C
Design temperature	350 °C
Design pressure	176 bar
Pressure loss at full load	2.4 bar

Table 1.-Technical Data of the Reactor Coolant and Pressurizer System. (3/4)

Volume of one steam generator	36.8 m^3
Inner diameter of tubes	19.6 mm
Secondary side	
Flow per steam generator at full load	552.8 kg/s
Main steam pressure in dome at full load	67.5 bar
Max. steam moisture at steam generator outlet at full load	0.25 %
Feedwater inlet temperature at full load	220.1 °C
Design temperature	350 °C
Design pressure	88.3 bar
Normal (controlled) water level	12. m
Volume of one steam generator	175 m^3
Reactor Coolant Pumps	
Number	3
Flow per pump at full load	7.49 m^3/s
Pressure increase at full load	6.6 bar
Power consumption at 30 bar and 50 °C approx.	9518 kW
Power consumption at 157 bar and 297 °C approx.	7126 kW
Operating temperature at full load	294.4 °C
Design temperature	350 °C
Design pressure	176 bar
Inlet and outlet parts, inside diameter	750 mm
Speed	1480 min^{-1}
Water volume per pump	2.45 m^3
Pressurizer	
Total available volume	45.2 m^3
Water volume at zero load (hot)	13.8 m^3
Steam volume at zero load (hot)	31.4 m^3
Water volume at full load	27.5 m^3
Steam volume at full load	17.7 m^3

Table 1.-Technical Data of the Reactor Coolant and Pressurizer System. (4/4)

Operating temperature	346 °C
Design temperature	362 °C
Design pressure	176 bar
Operating pressure	157.5 bar
Internal diameter in the cylindrical section	2200 mm
Total nominal pressurizer heating capacity	1638 kW
Total number of heater rods	78
heating capacity per heater rod	21 kW

Table 2.- Variables identification in transient figures. (1/2)

Keyword	*Variable*
	PLANT DATA
YT00P511	Primary Pressure
YT00X111	Thermal Corrected Reactor Power
YA10T003	Hot Leg Temperature (Loop 1)
YA20T003	Hot Leg Temperature (Loop 2)
YA30T003	Hot Leg Temperature (Loop 3)
YA10T955	Cold Leg Temperature (Loop 1)
YA20T955	Cold Leg Temperature (Loop 2)
YA30T955	Cold Leg Temperature (Loop 3)
YA10T001	Average Temperature (Loop 1)
YA20T001	Average Temperature (Loop 2)
YA30T001	Average Temperature (Loop 3)
YT00T101	Primary Average Temperature
YT00L951	Pressurizer Level
SF10C110	Secondary Pressure (Header)
RA01F901	Main Steam Mass Flow Rate (Steam Generator 1)
RA02F901	Main Steam Mass Flow Rate (Steam Generator 2)
RA03F901	Main Steam Mass Flow Rate (Steam Generator 3)
RL21F002	Feedwater Mass Flow Rate (Steam Generator 1)
RL22F002	Feedwater Mass Flow Rate (Steam Generator 2)
RL23F002	Feedwater Mass Flow Rate (Steam Generator 3)
YB10L951	Steam Generator 1 Level
YB20L951	Steam Generator 2 Level
YB30L951	Steam Generator 3 Level
	RELAP DATA
P 200_01	Primary Pressure
CNTRL 925	Thermal Reactor Corrected Power
CNTRL 056	Permitted Reactor Power

Table 2.- Variables identification in transient figures. (2/2)

Keyword	*Variable*
CNTRL 010	Hot Leg Temperature (Loop 1)
CNTRL 012	Hot Leg Temperature (Loop 2)
CNTRL 014	Hot Leg Temperature (Loop 3)
CNTRL 011	Cold Leg Temperature (Loop 1)
CNTRL 013	Cold Leg Temperature (Loop 2)
CNTRL 015	Cold Leg Temperature (Loop 3)
CNTRL 467	Average Temperature (Loop 1)
CNTRL 468	Average Temperature (Loop 2)
CNTRL 469	Average Temperature (Loop 3)
CNTRL 484	Primary Average Temperature
CNTRL 140	Primary Average Temperature (Setpoint)
CNTRL 004	Pressurizer Level
MFLOWJ 160	Coolant Mass Flow Rate (Loop 1)
MFLOWJ 260	Coolant Mass Flow Rate (Loop 2)
MFLOWJ 360	Coolant Mass Flow Rate (Loop 3)
PMPVEL 150	Pump Speed (Loop 1)
PMPVEL 250	Pump Speed (Loop 2)
PMPVEL 350	Pump Speed (Loop 3)
P 910_01	Secondary Pressure
MFLOWJ 660	Main Steam Mass Flow Rate (Steam Generator 1)
MFLOWJ 760	Main Steam Mass Flow Rate (Steam Generator 2)
MFLOWJ 860	Main Steam Mass Flow Rate (Steam Generator 3)
CNTRL 061	Generator Power
CNTRL 2165	Feedwater Mass Flow Rate (Steam Generator 1)
CNTRL 2265	Feedwater Mass Flow Rate (Steam Generator 2)
CNTRL 2365	Feedwater Mass Flow Rate (Steam Generator 3)
CNTRL 724	Steam Generator 1 Level
CNTRL 729	Steam Generator 2 Level
CNTRL 734	Steam Generator 3 Level

Table 3.- Steady state parameters. (1/2)

	BOC PLANT	*BOC RELAP*	*EOC PLANT*	*EOC RELAP*
Primary pressure (MPa)	15.795	15.85	15.89	15.87
Average Temperature (°C)	308.65	308.0	310.05	310.3
Cold Leg Temperature, Loop 1 (°C)	293.01	292.34	294.75	294.72
Cold Leg Temperature, Loop 2 (°C)		292.34		294.72
Cold Leg Temperature, Loop 3 (°C)	292.77	292.45	295.13	294.83
Hot Leg Temperaure, Loop 1 (°C)	322.24	323.6	328.	325.83
Hot Leg Temperaure, Loop 2 (°C)		323.6		325.83
Hot Leg Temperaure, Loop 3 (°C)	322.16	323.6	326.16	325.82
Pressurizer Level (m)	7.60	7.59		7.78
Secondary Pressure (MPa)	6.72	6.61	6.967	6.775
S.G. 1 Mass Flow Rate (kg/s)	531.59	529.15		550.46
S.G. 2 Mass Flow Rate (kg/s)		530.8		550.71
S.G. 3 Mass Flow Rate (kg/s)	524.48	527.41		547.86
S.G. 1 Feedwater Mass Flow Rate (kg/s)	553.8	529.24		547.16
S.G. 2 Feedwater Mass Flow Rate (kg/s)	553.3	529.18		548.12

Table 3.- Steady state parameters. (1/2)

	BOC PLANT	*BOC RELAP*	*EOC PLANT*	*EOC RELAP*
S.G. 3 Feedwater Mass Flow Rate (kg/s)	546.0	527.16		547.16
S. G. 1 Level (m)	11.96	11.96		11.98
S. G. 1 Level (m)		11.96		11.98
S. G. 1 Level (m)	12.05	11.97		11.97

Table 4.- Run Statistics.

CASE 1, BOC Conditions	
Transient time (s)	500.
CPU time (s)	5405.8
Total number of time steps	10000
CPU time /Transient time	10.81
Number of active volumes	303
CASE 2, EOC Conditions	
Transient time (s)	500.
CPU time (s)	5482.8
Total number of time steps	10000
CPU time /Transient time	10.96
Number of active volumes	303

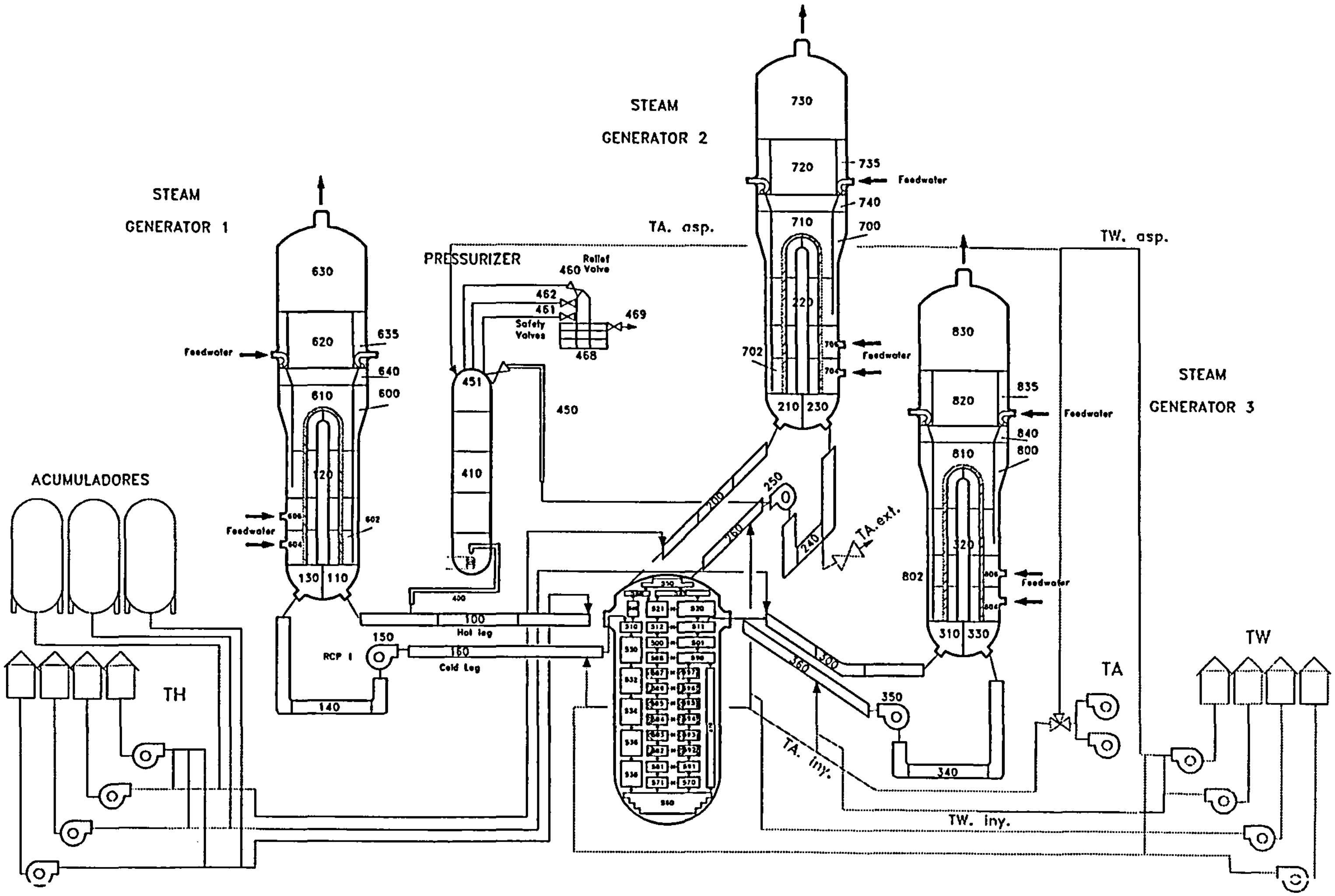

Figure 1.- C. Trillo Nodalization (Primary System)

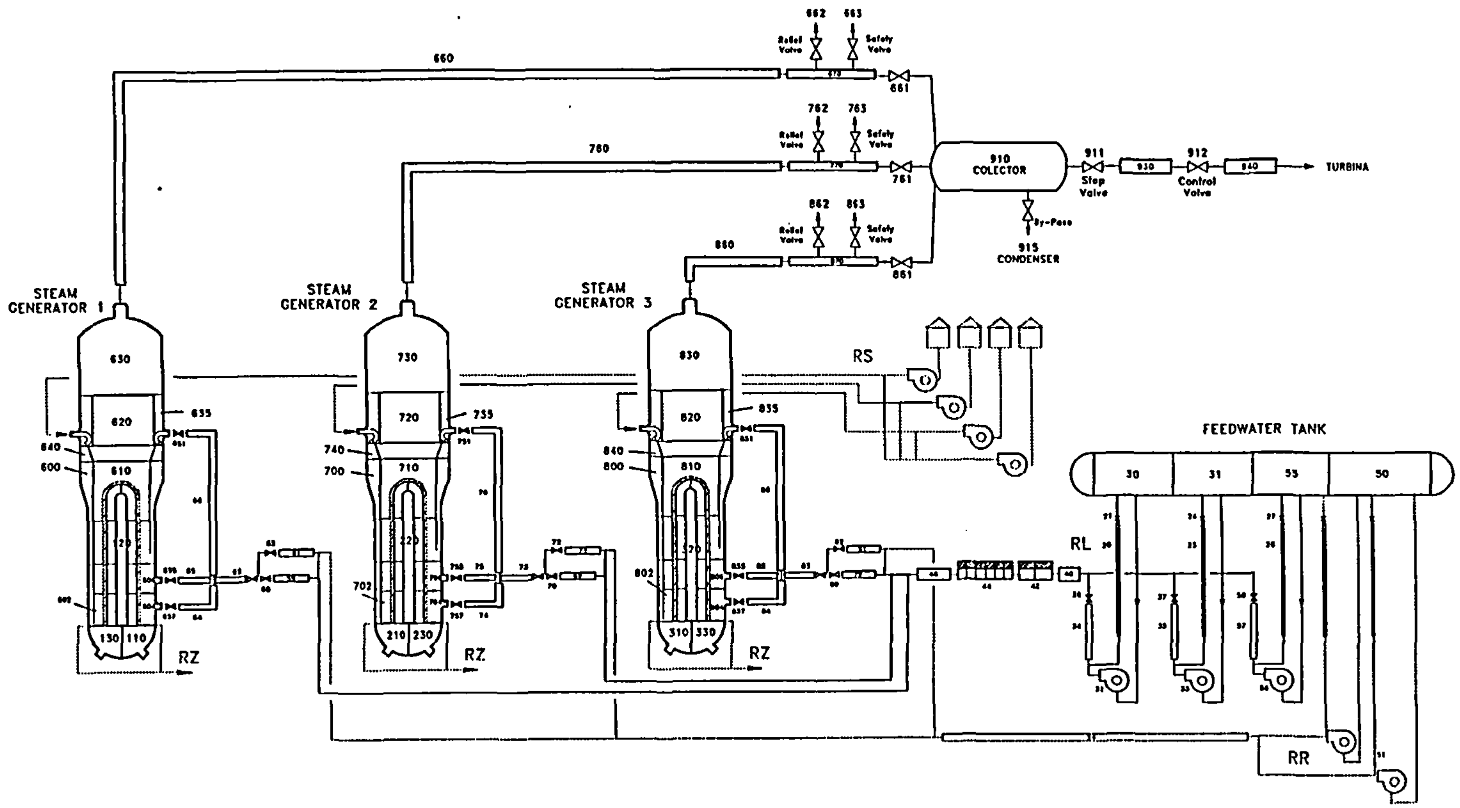

Figure 2.- C.Trillo Nodalization (Steam/Water cycle)

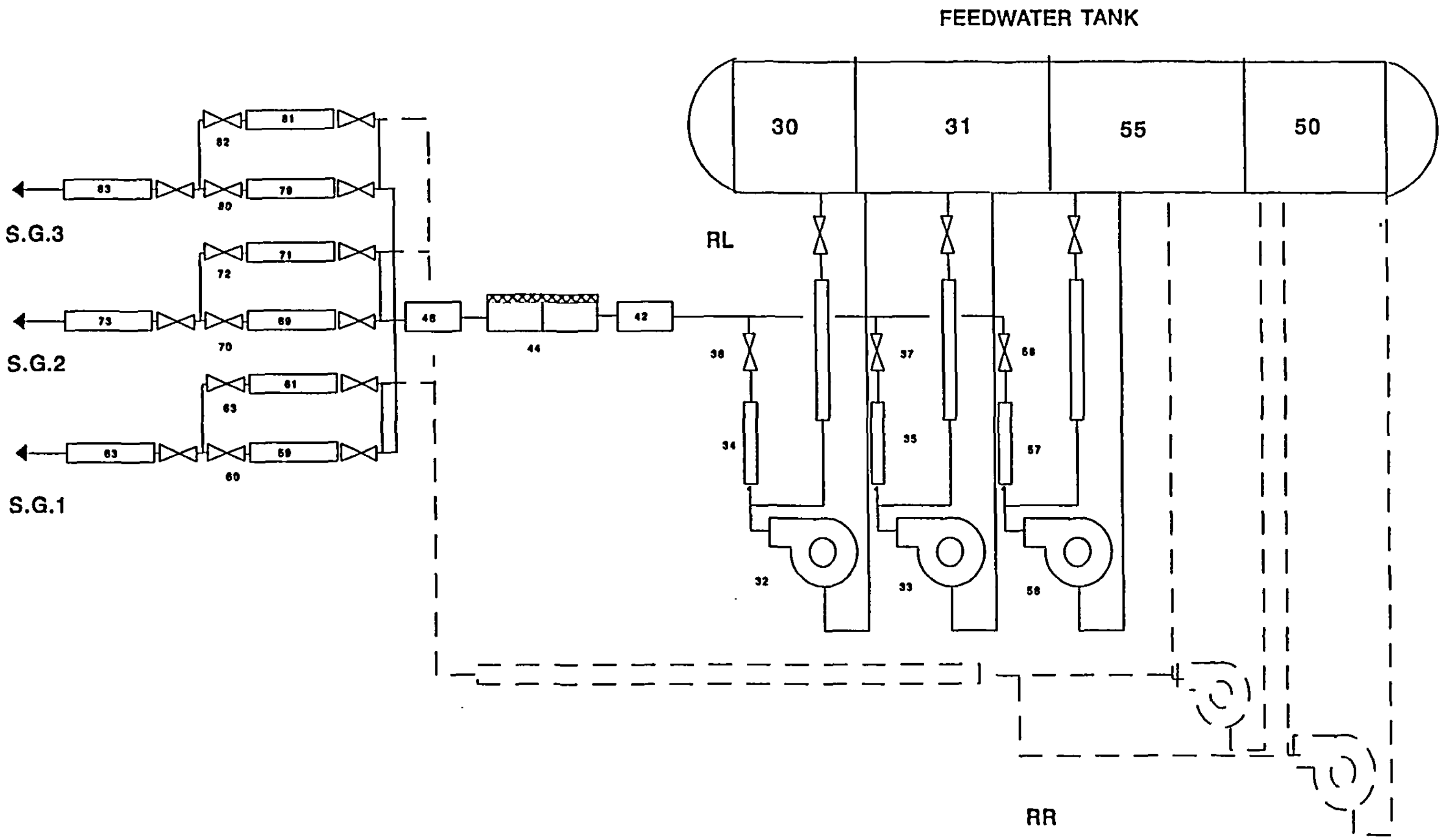

Figure 3.- C Trillo Nodalization (Feedwater System)

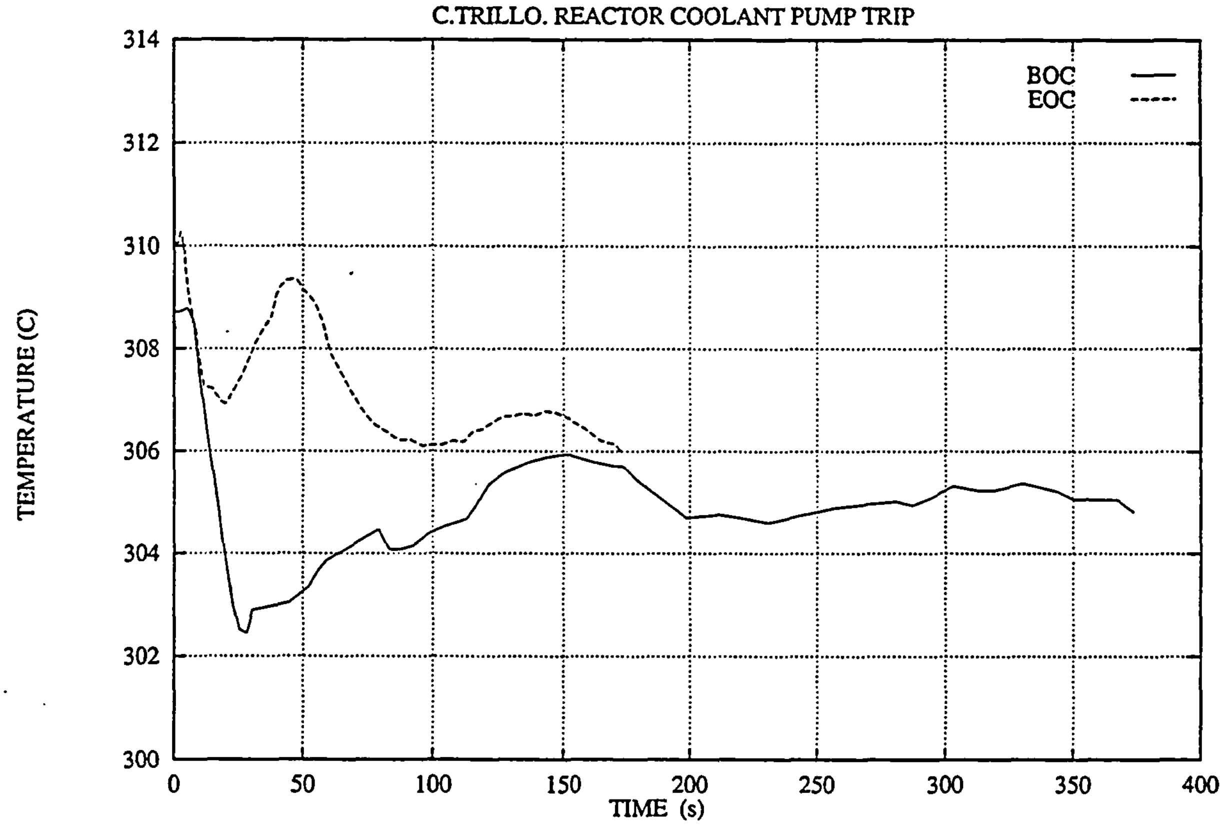

FIG.4.- PRIMARY AVERAGE TEMPERATURE, YT0

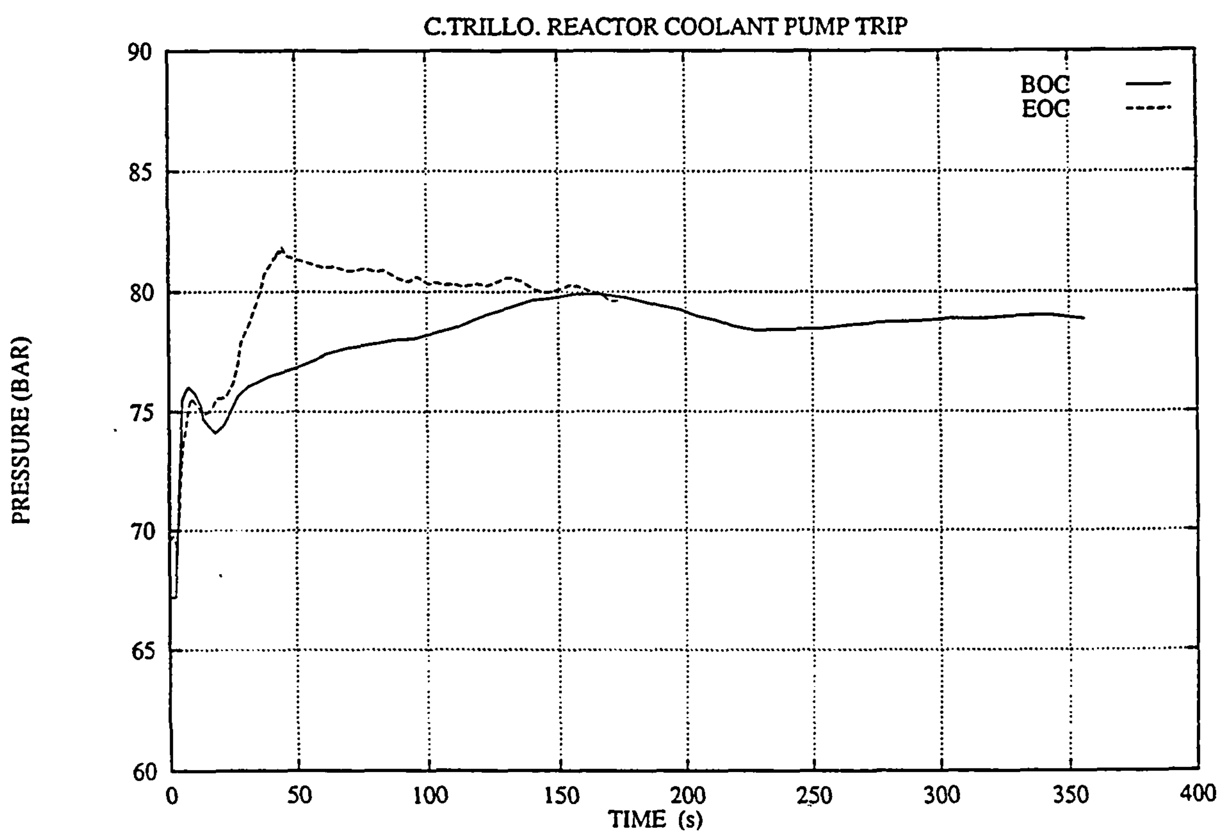

FIG 5.- SECONDARY PRESSURE, SF10C110

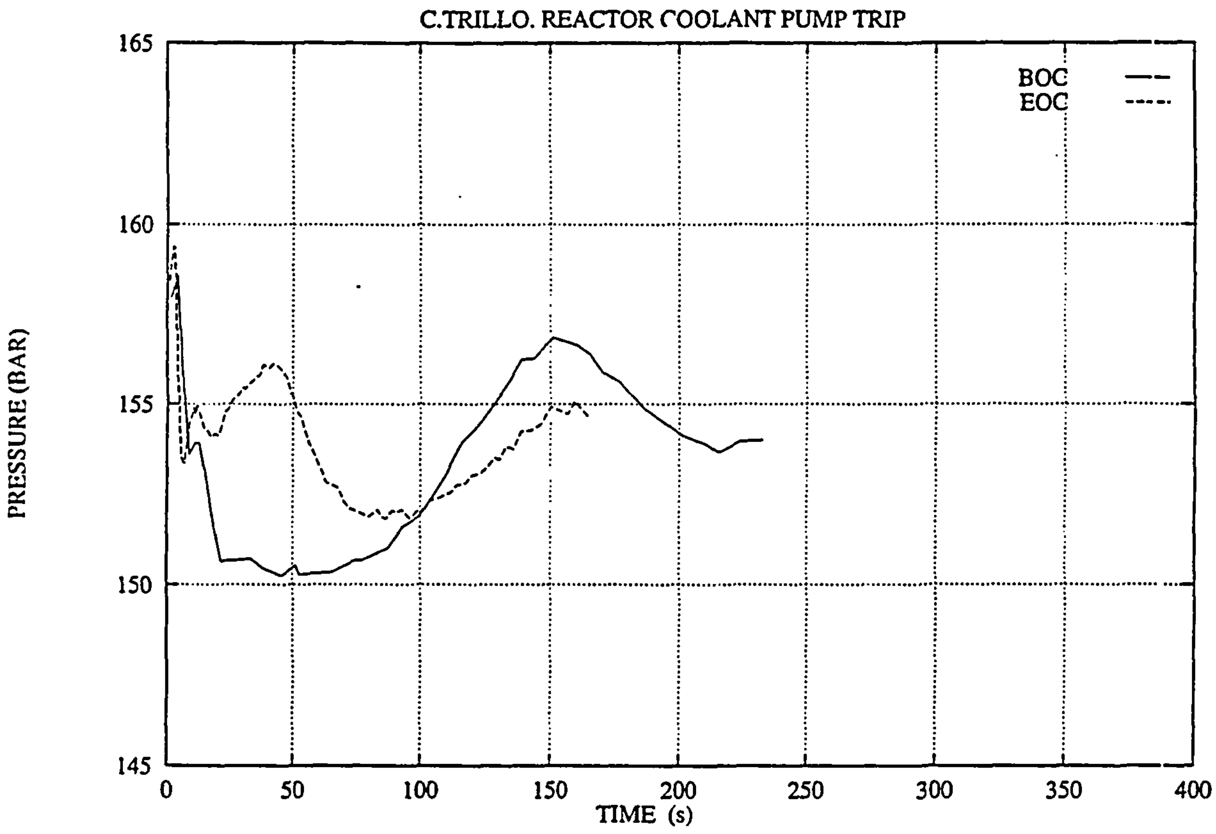

FIG.6.- PRIMARY PRESSURE, YT00P511

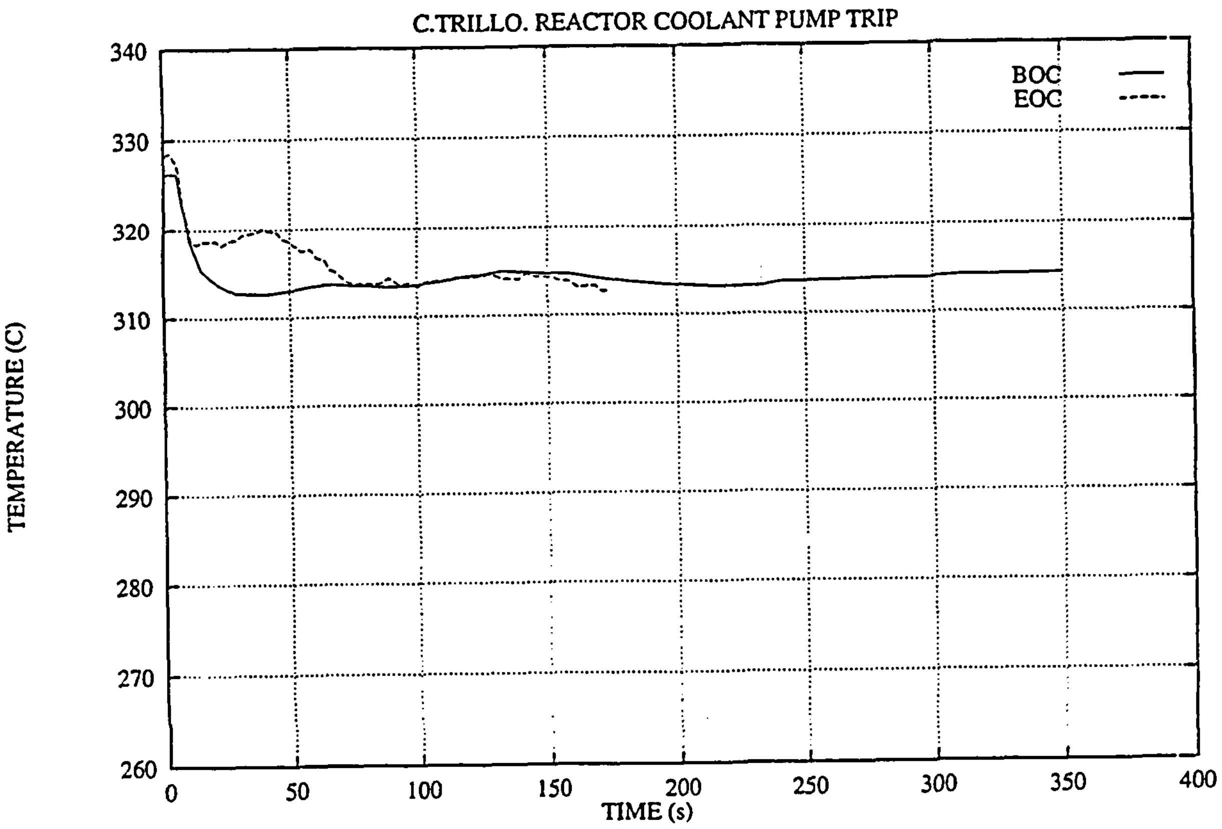

FIG.7.- HOT LEG TEMPERATURE, YA10T003

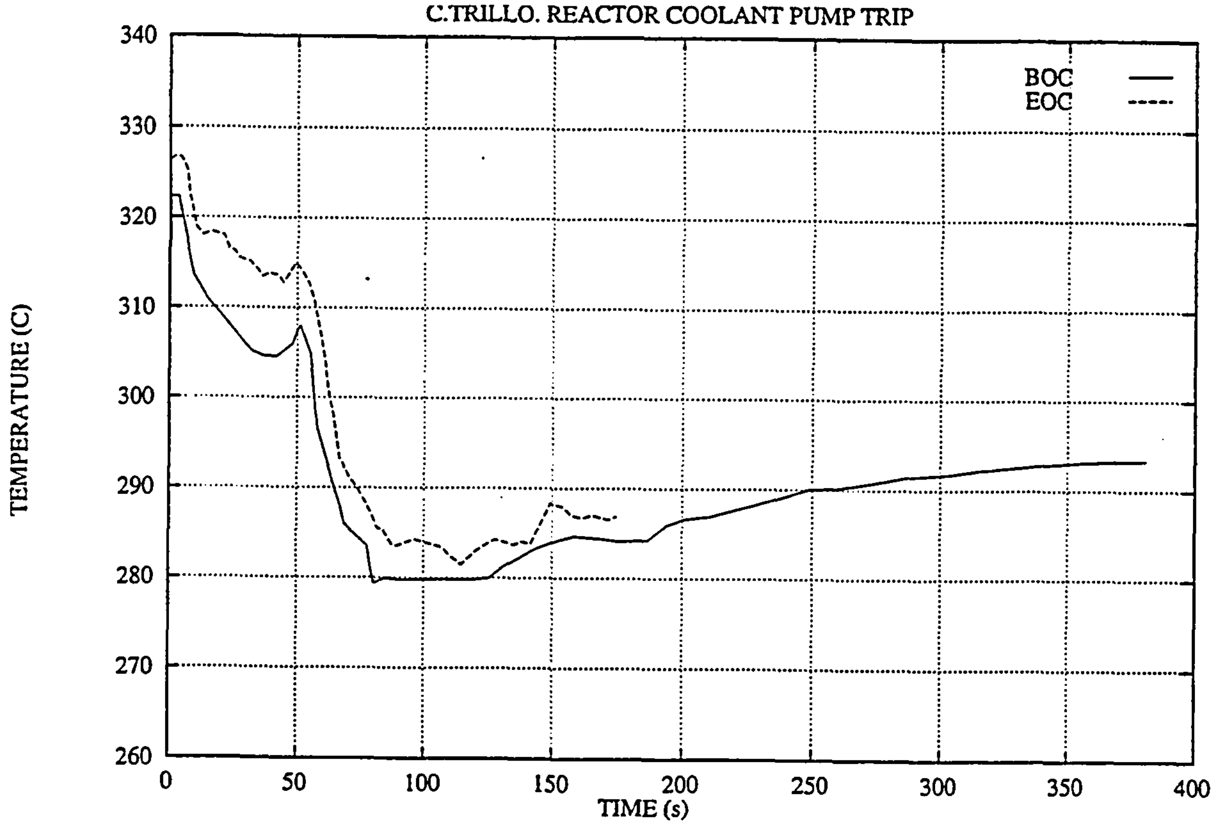

FIG.8.- HOT LEG TEMPERATURE, YA30T003

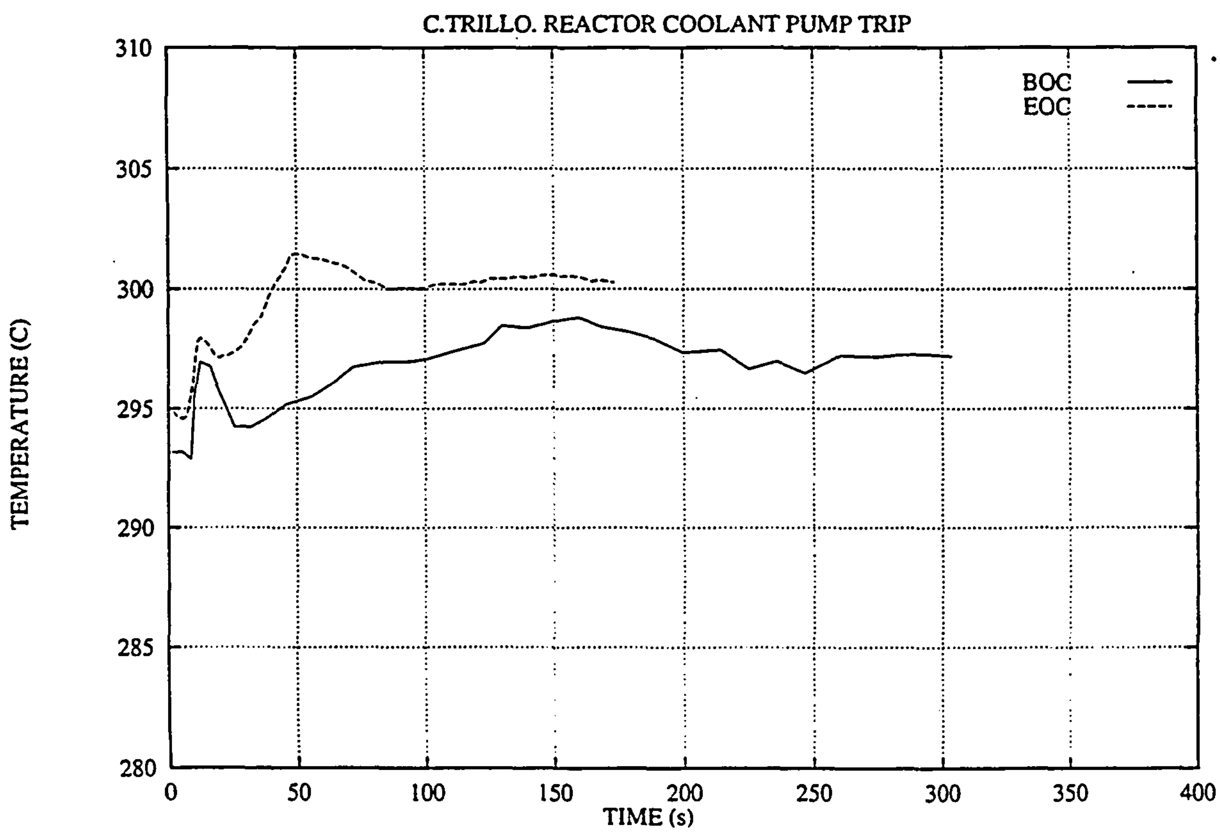

FIG.9.- COLD LEG TEMPERATURE, YA10T955

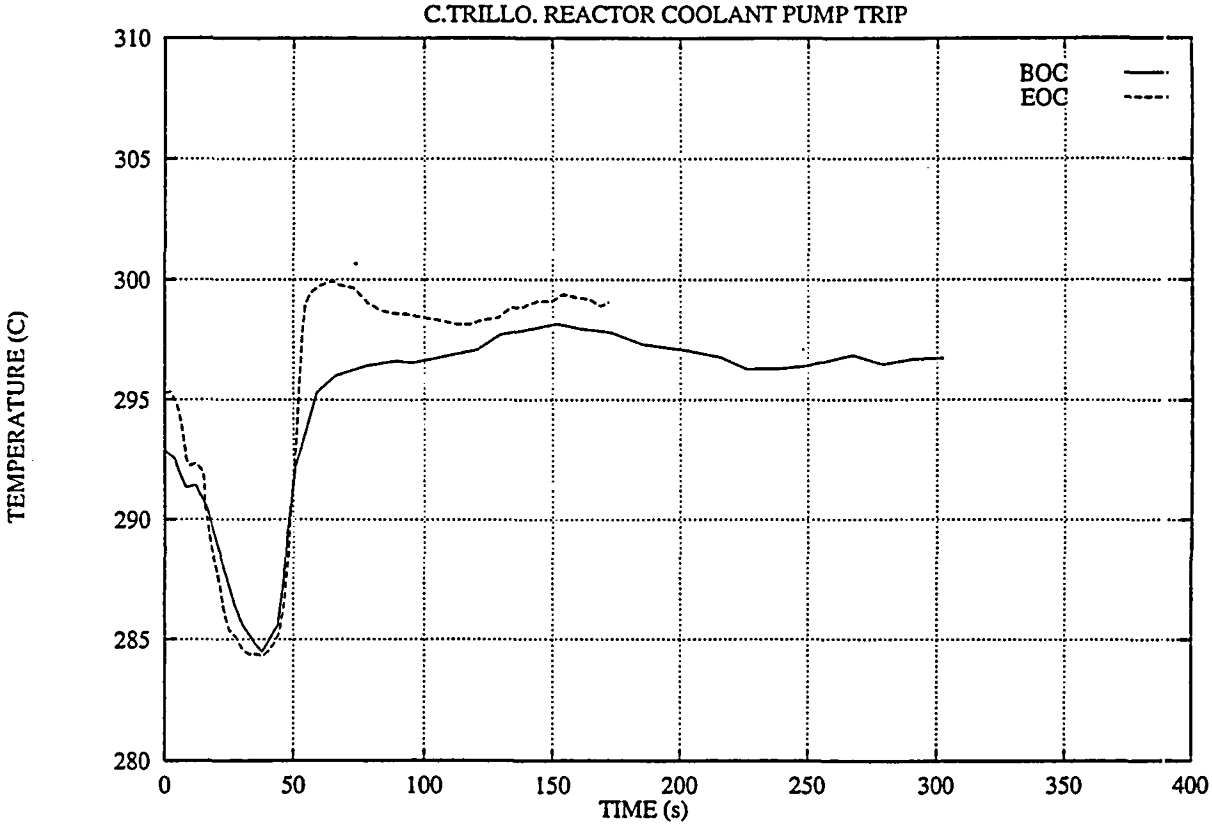

FIG.10.- COLD LEG TEMPERATURE, YA30T955

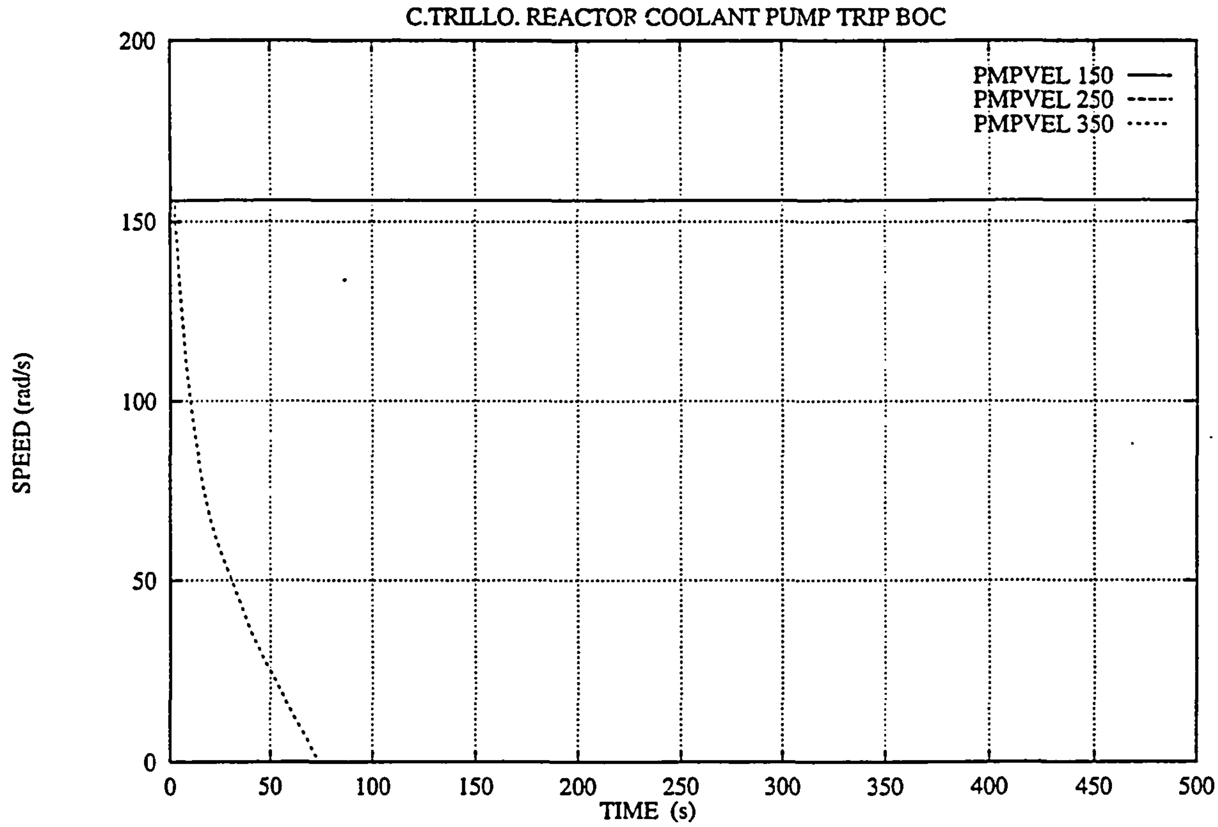

FIG.1.1.- REACTOR COOLANT PUMP SPEED

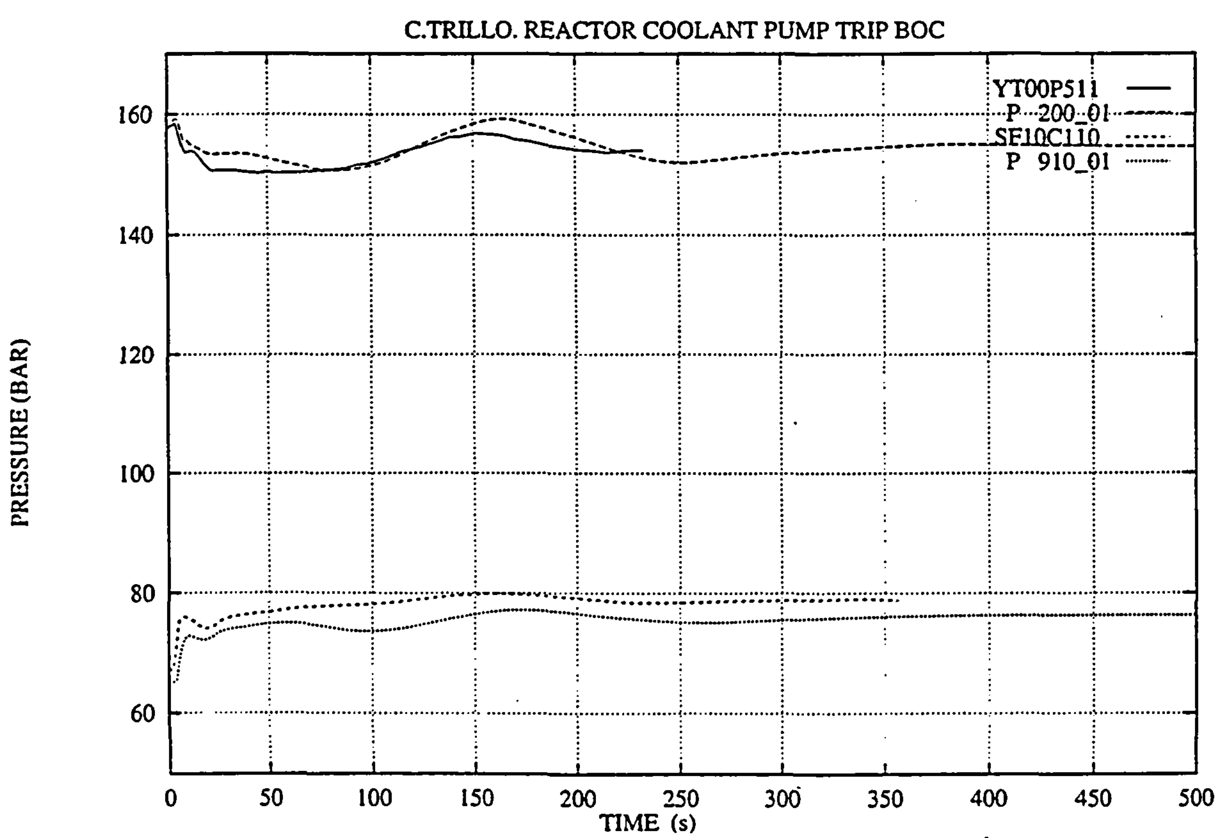

FIG. 1.2.- PRIMARY, SECONDARY PRESSURE

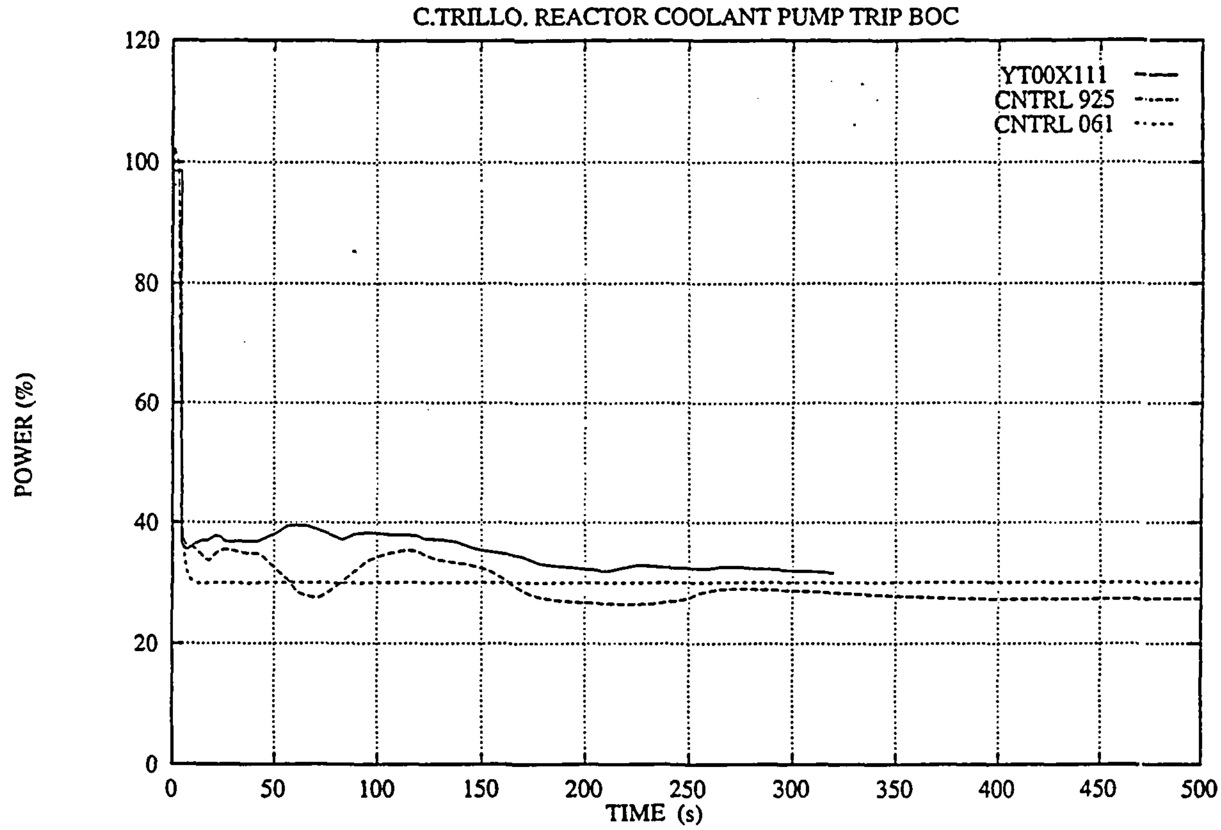

FIG.1.3.- REACTOR AND GENERATOR POWER

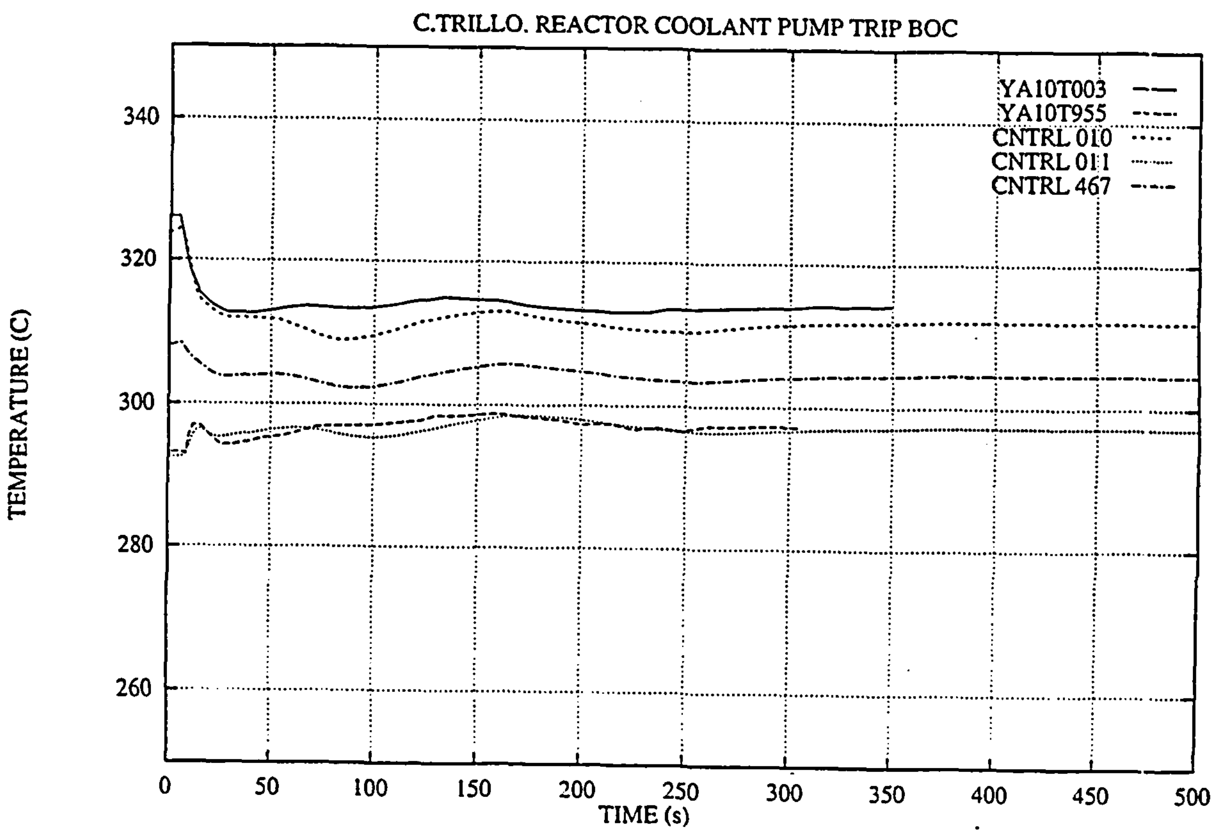

FIG.1.4.- LOOP 1 TEMPERATURES

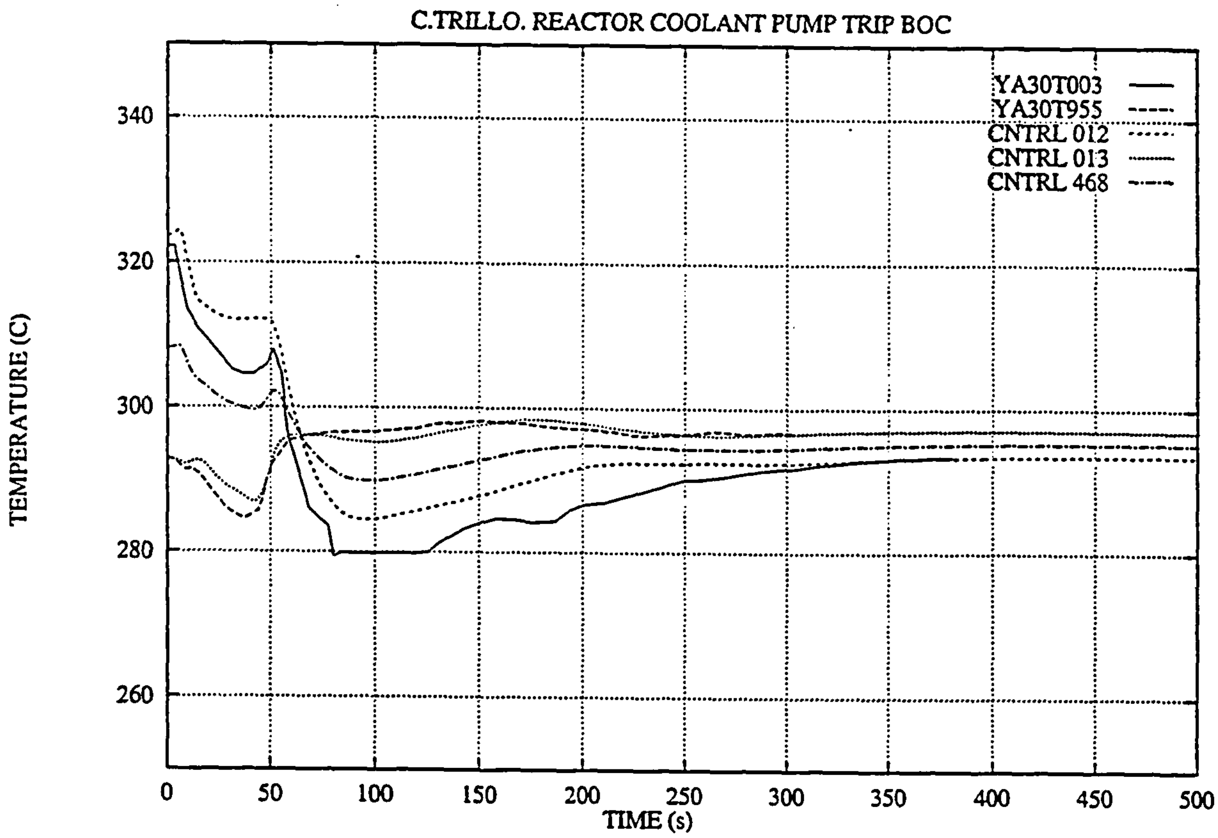

FIG.1.5.- LOOP 3 TEMPERATURES

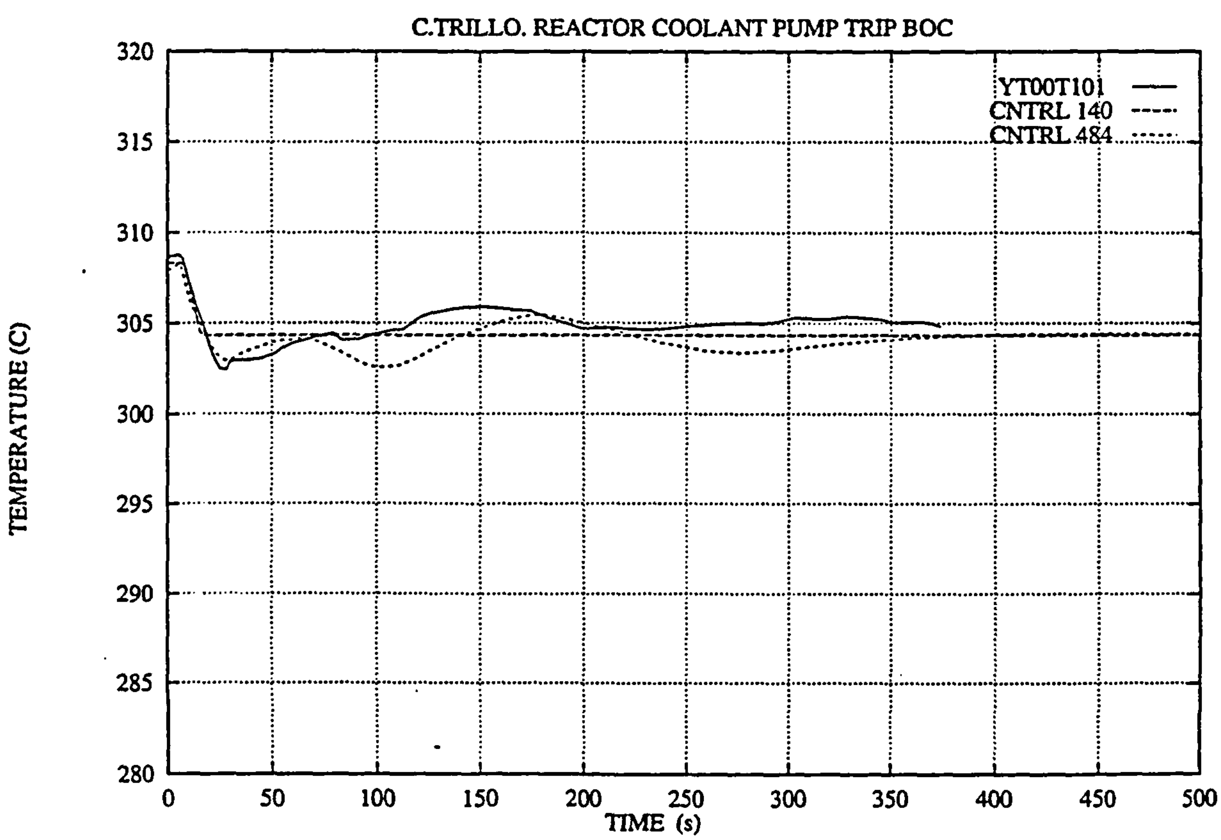

FIG. 1.6.- PRIMARY AVERAGE TEMPERATURE

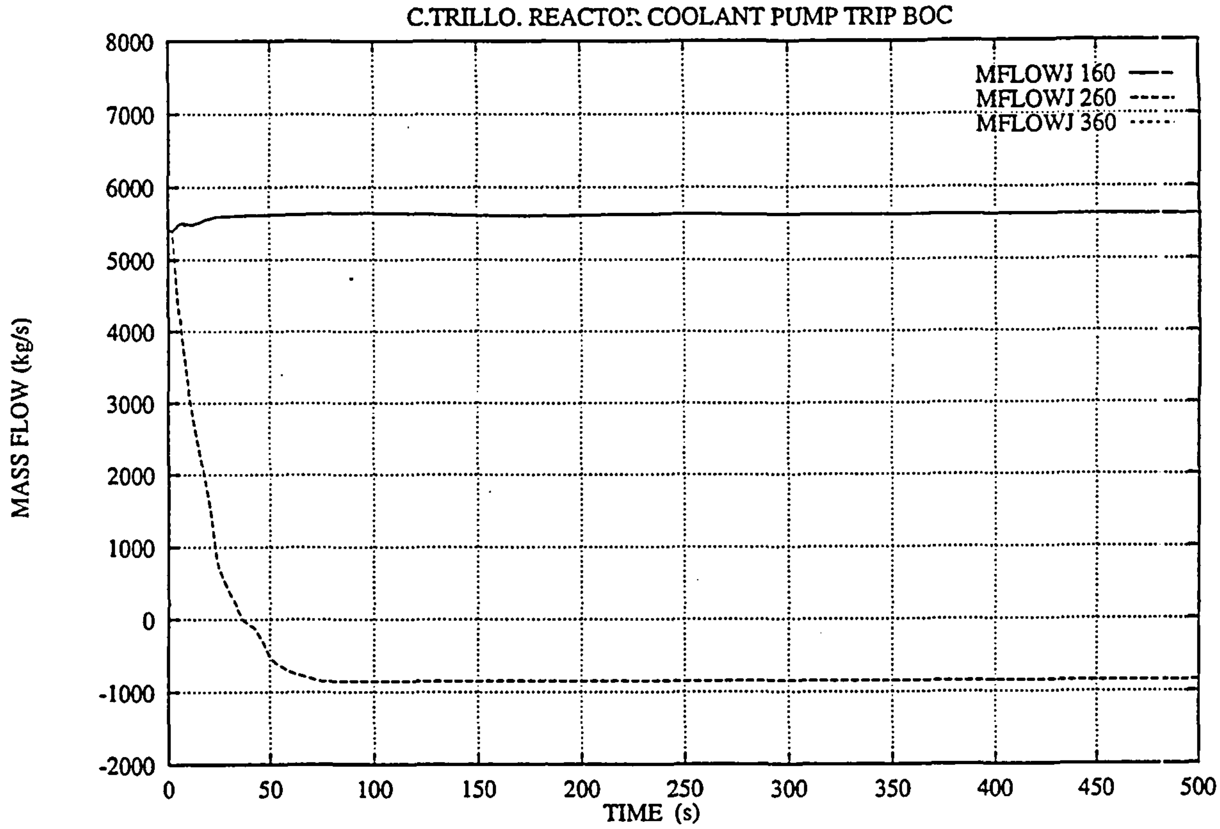

FIG.1.7.- COOLANT MASS FLOW RATE

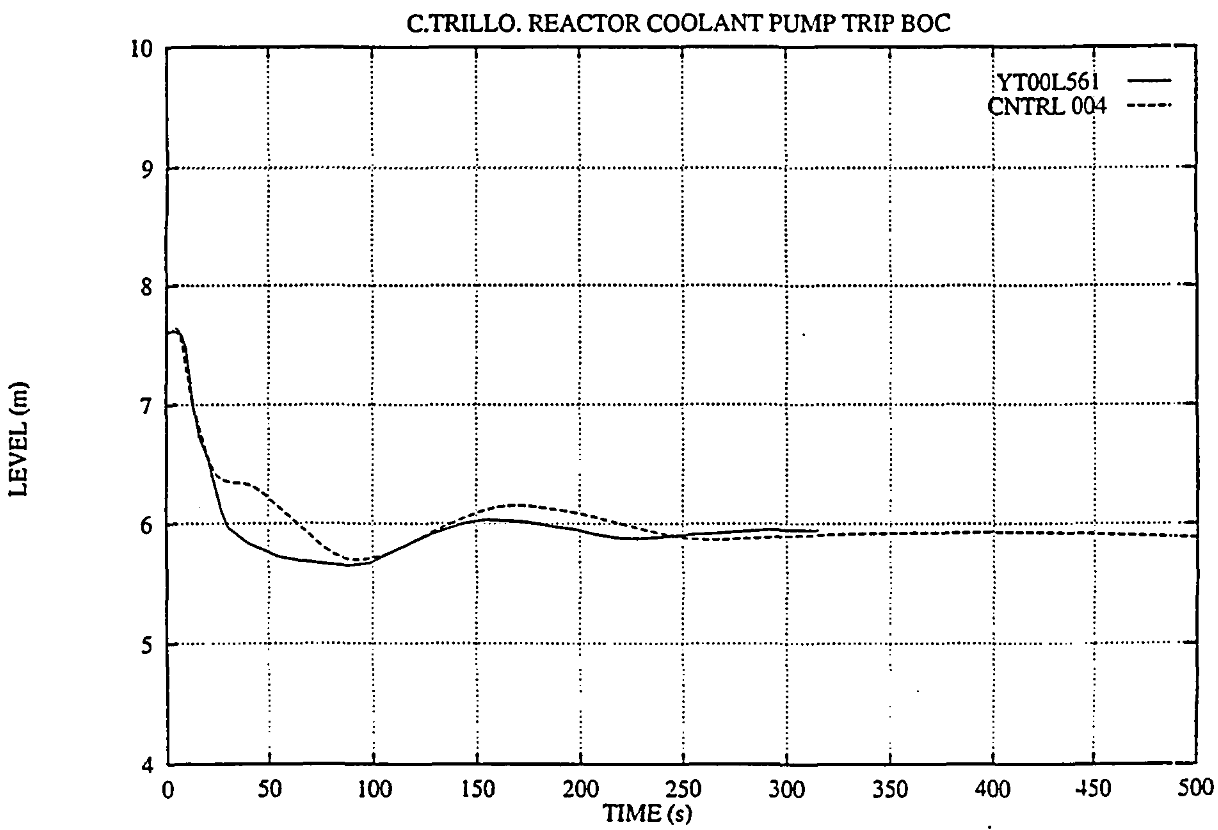

FIG.1.8.- PRESSURIZER LEVEL

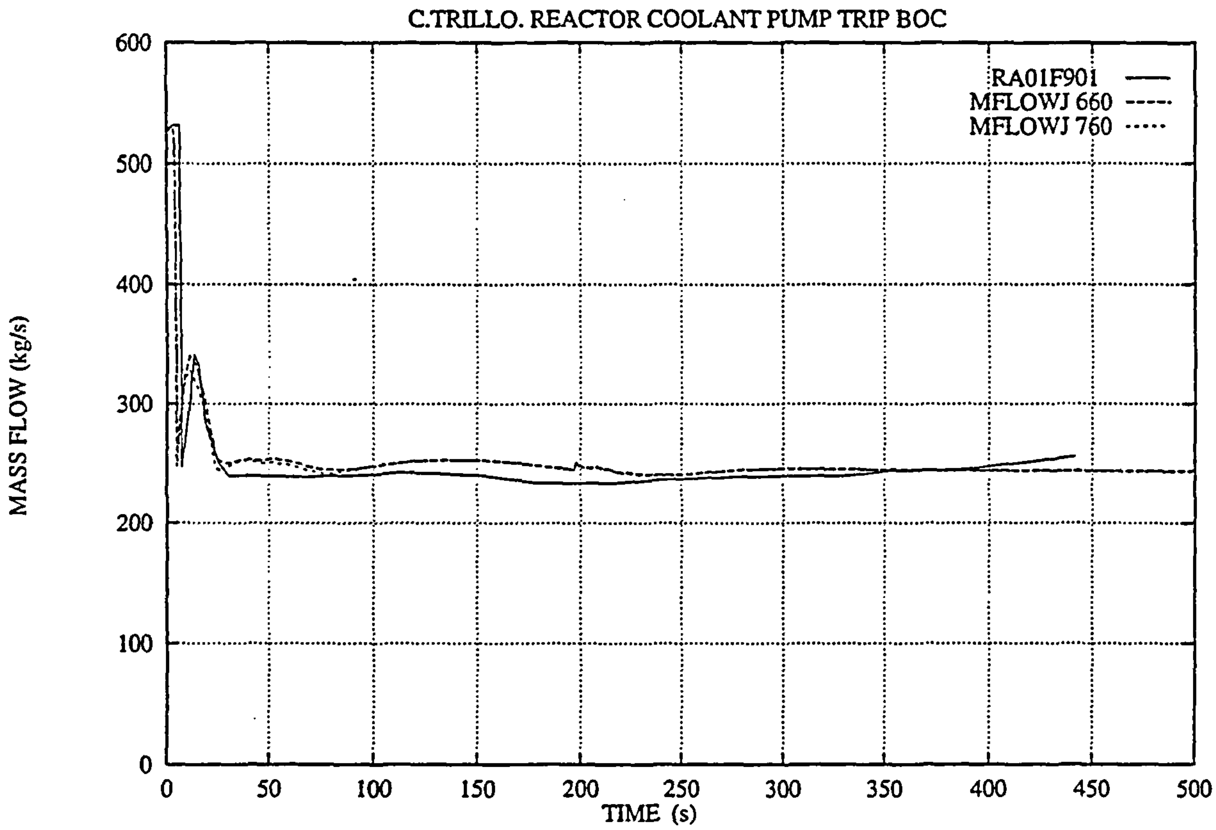

FIG.1.9.- S.G. MASS FLOW RATE (LOOP 1,2)

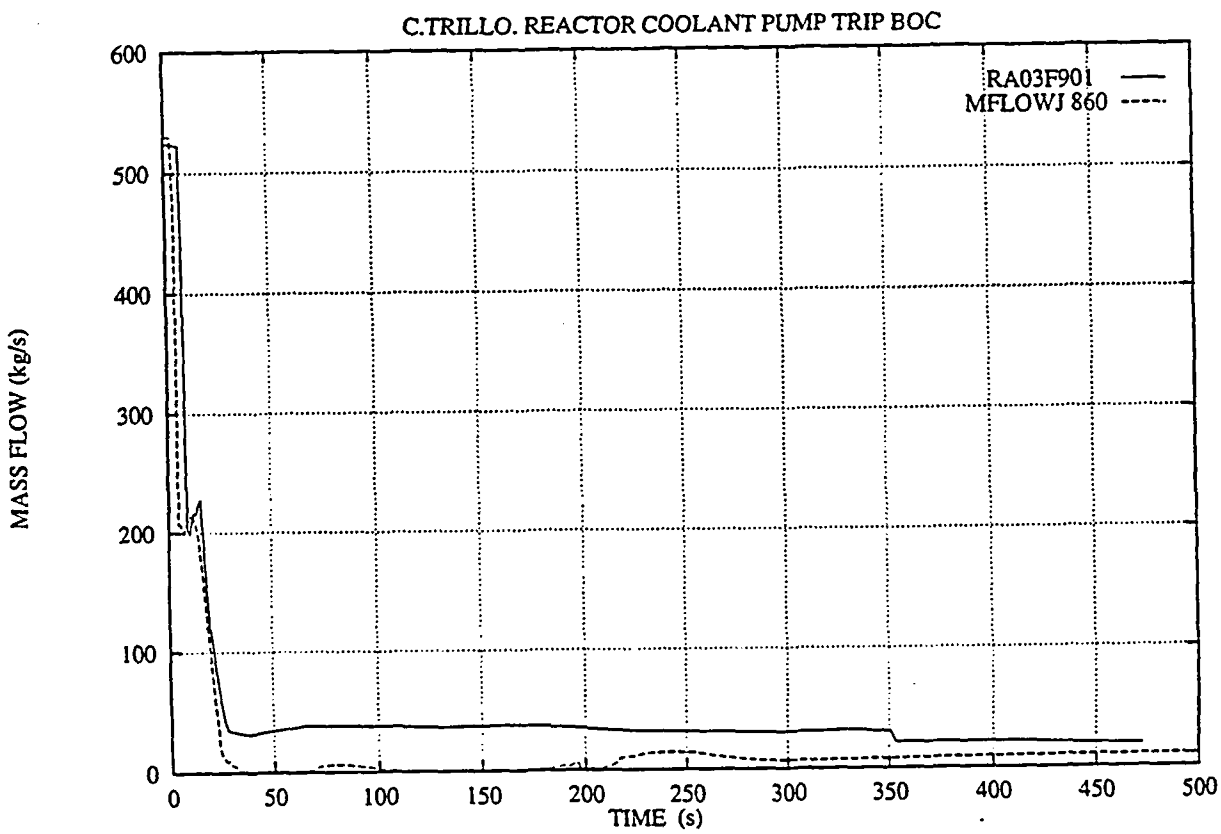

FIG.1.10.- S.G. MASS FLOW RATE (LOOP 3)

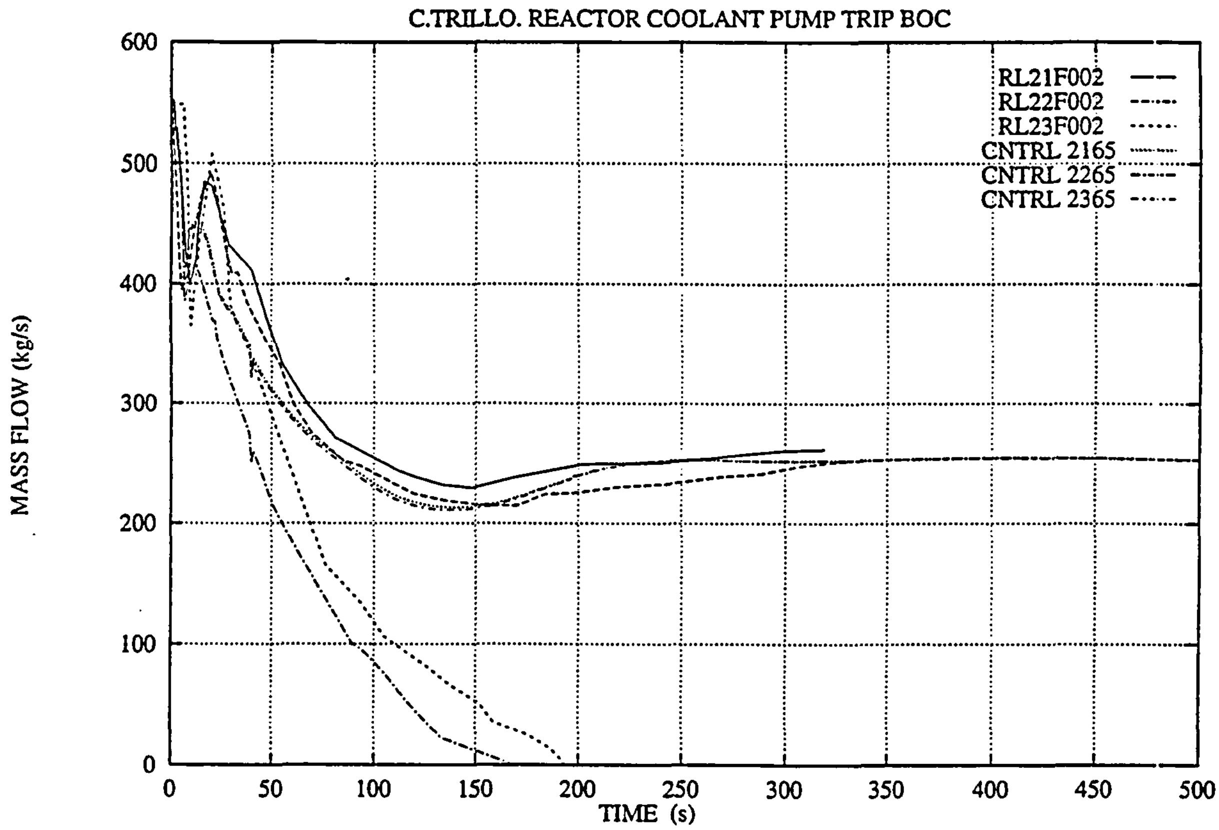

FIG.1.11.- FEEDWATER MASS FLOW RATE

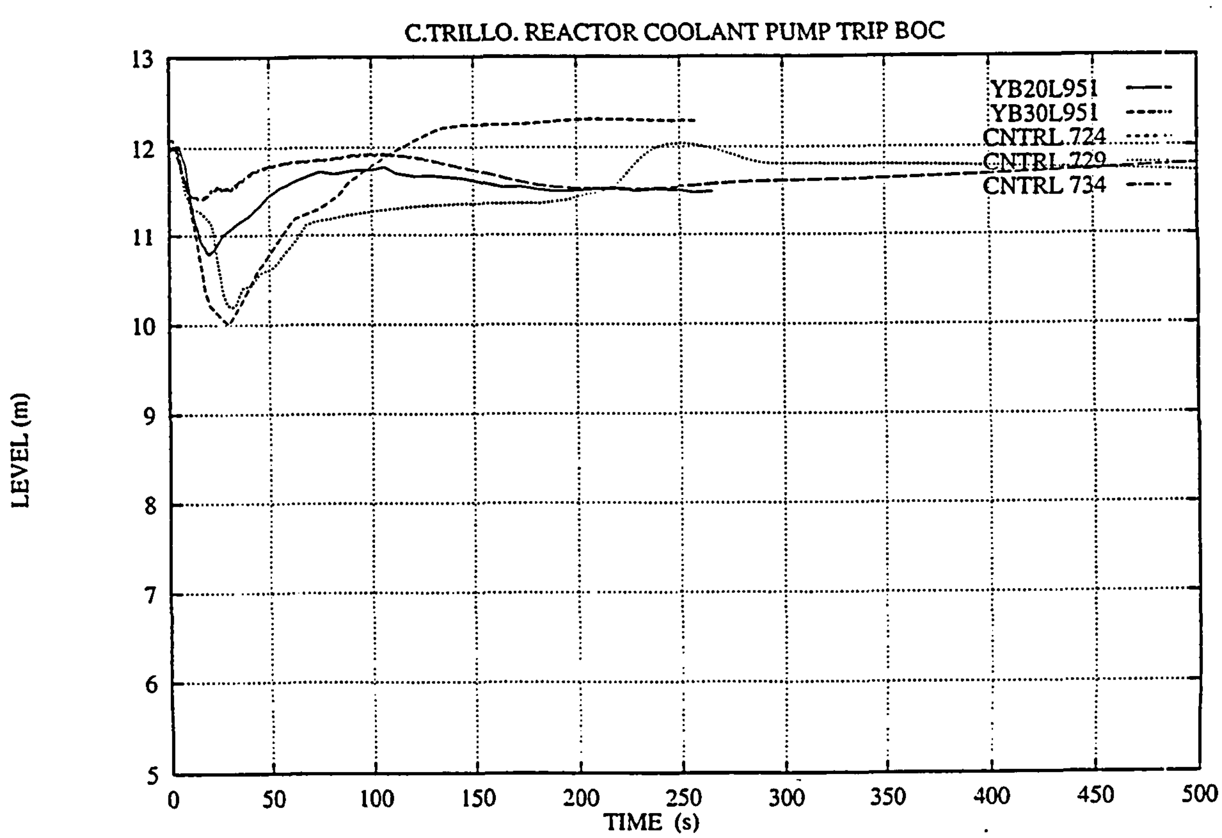

FIG. 1.12.- STEAM GENERATOR LEVEL

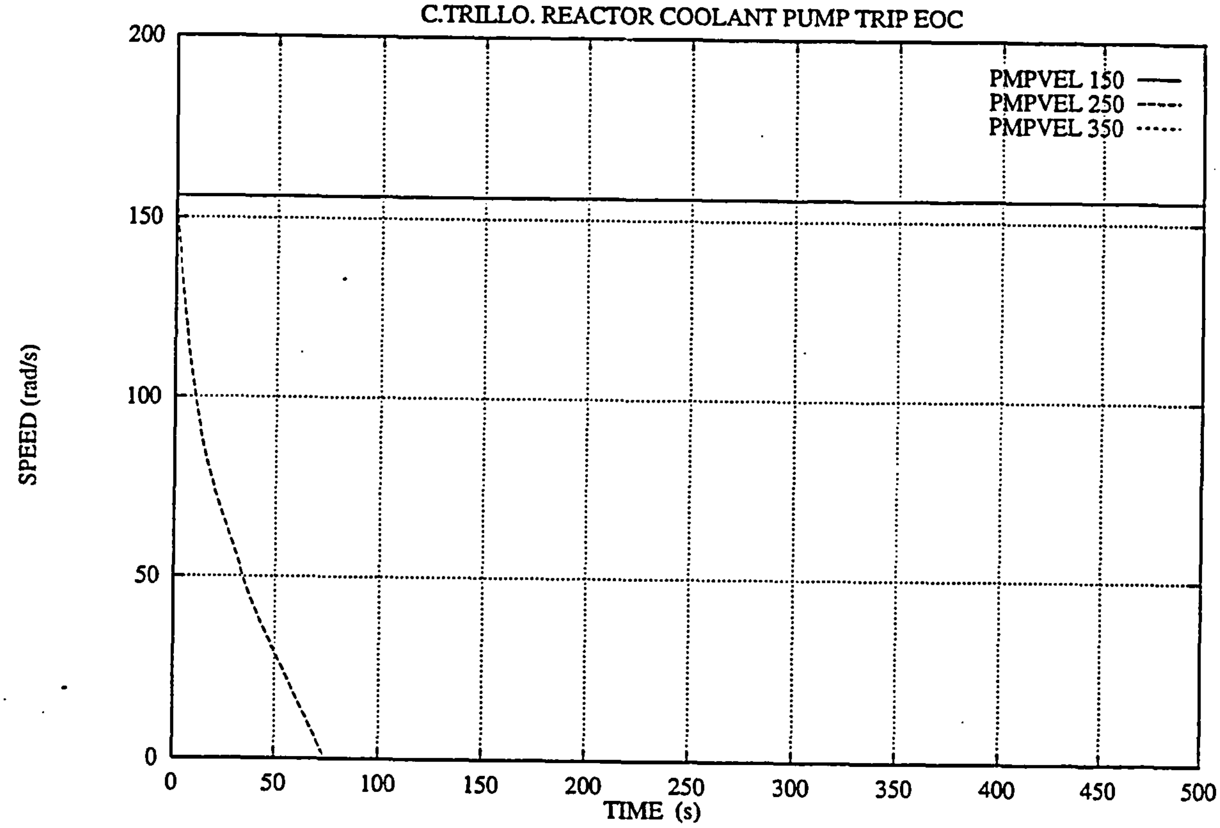

FIG.2.1.- REACTOR COOLANT PUMP SPEED

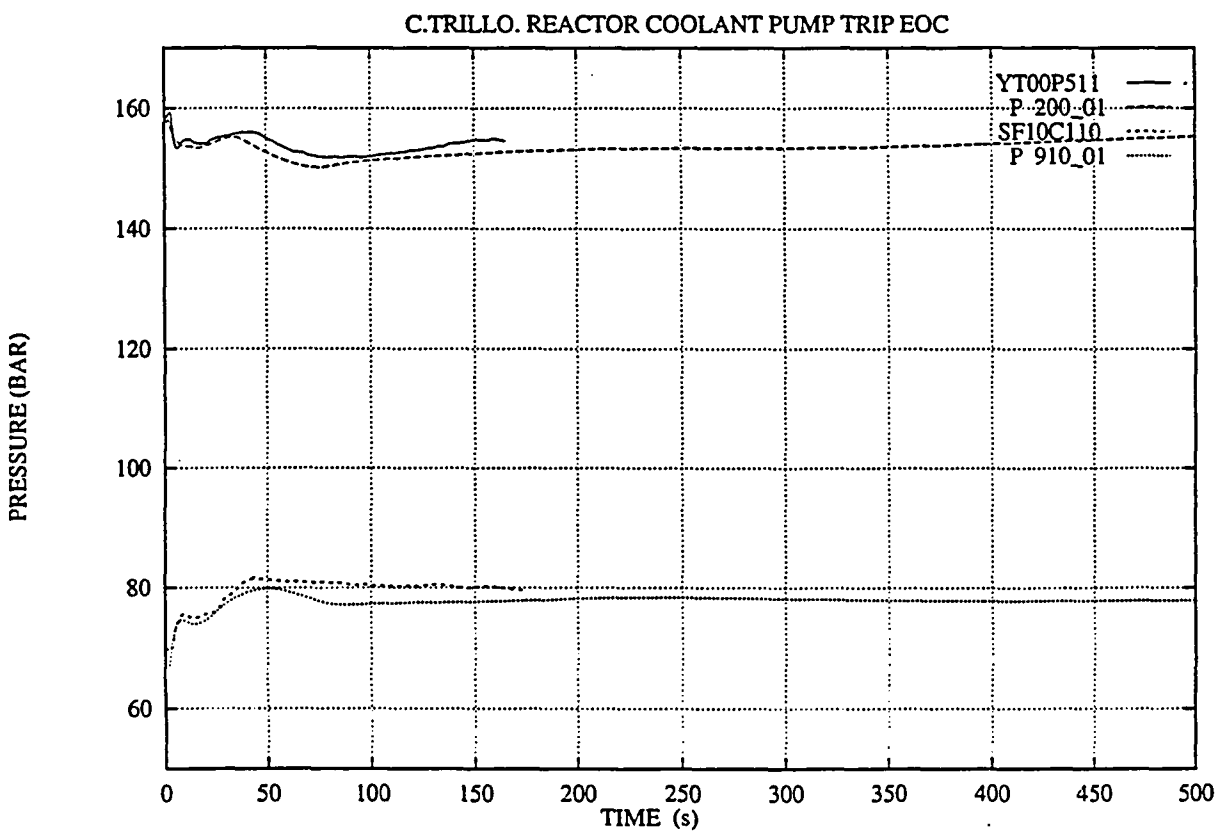

FIG.2.2.- PRIMARY, SECONDARY PRESSURE

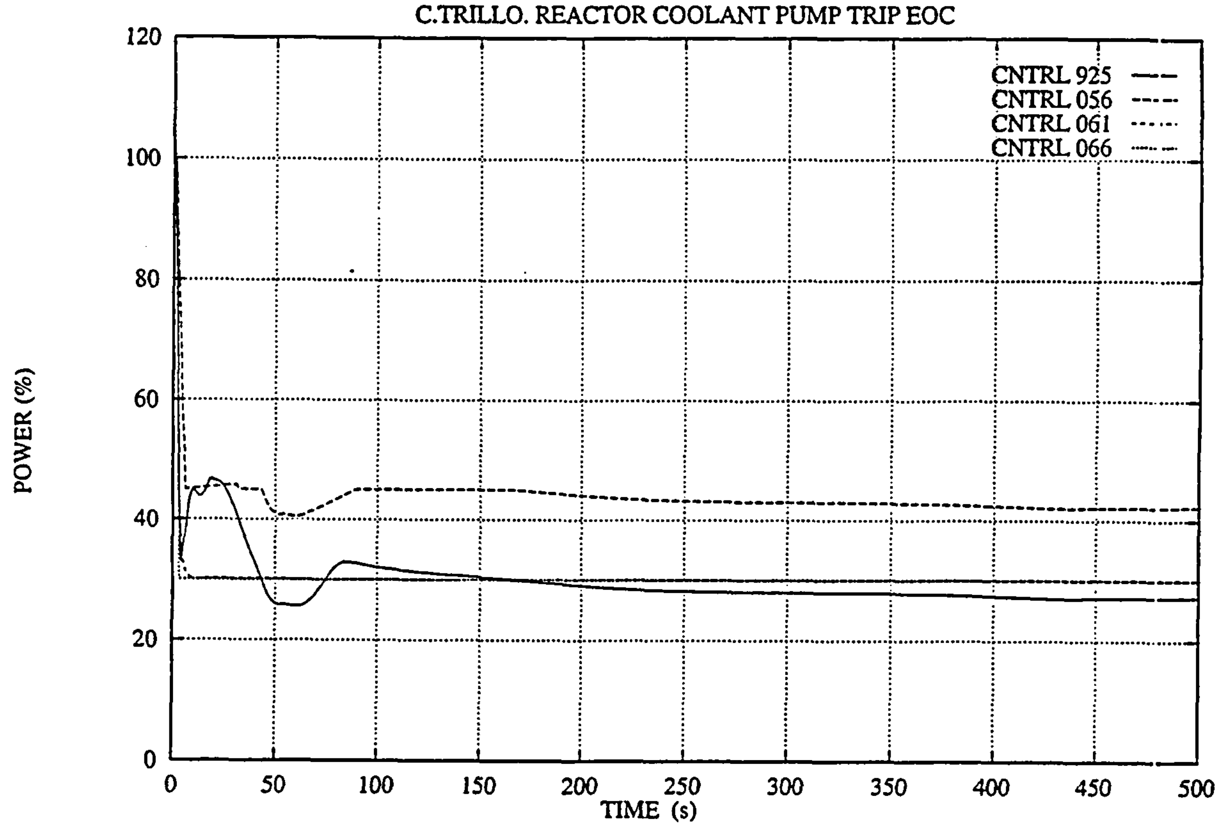

FIG.2.3.- REACTOR AND GENERATOR POWER

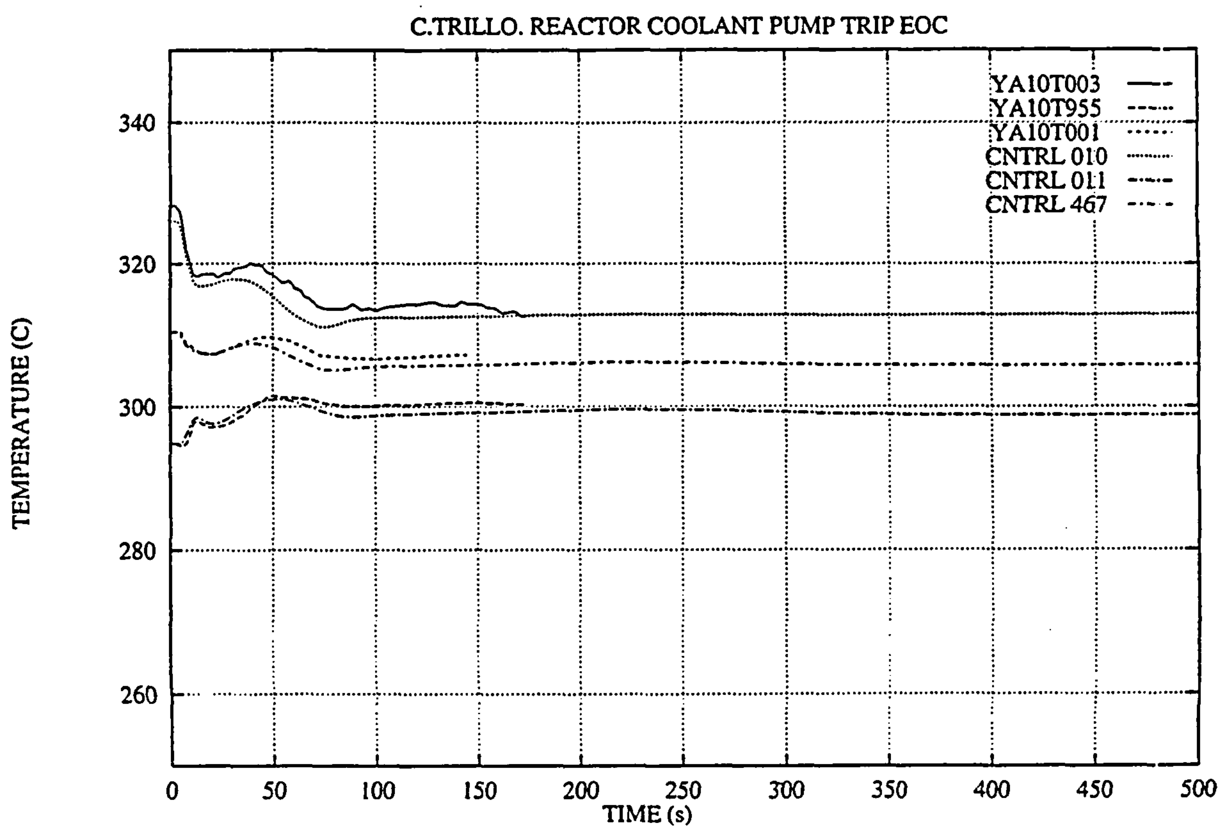

FIG.2.4.- LOOP 1 TEMPERATURES

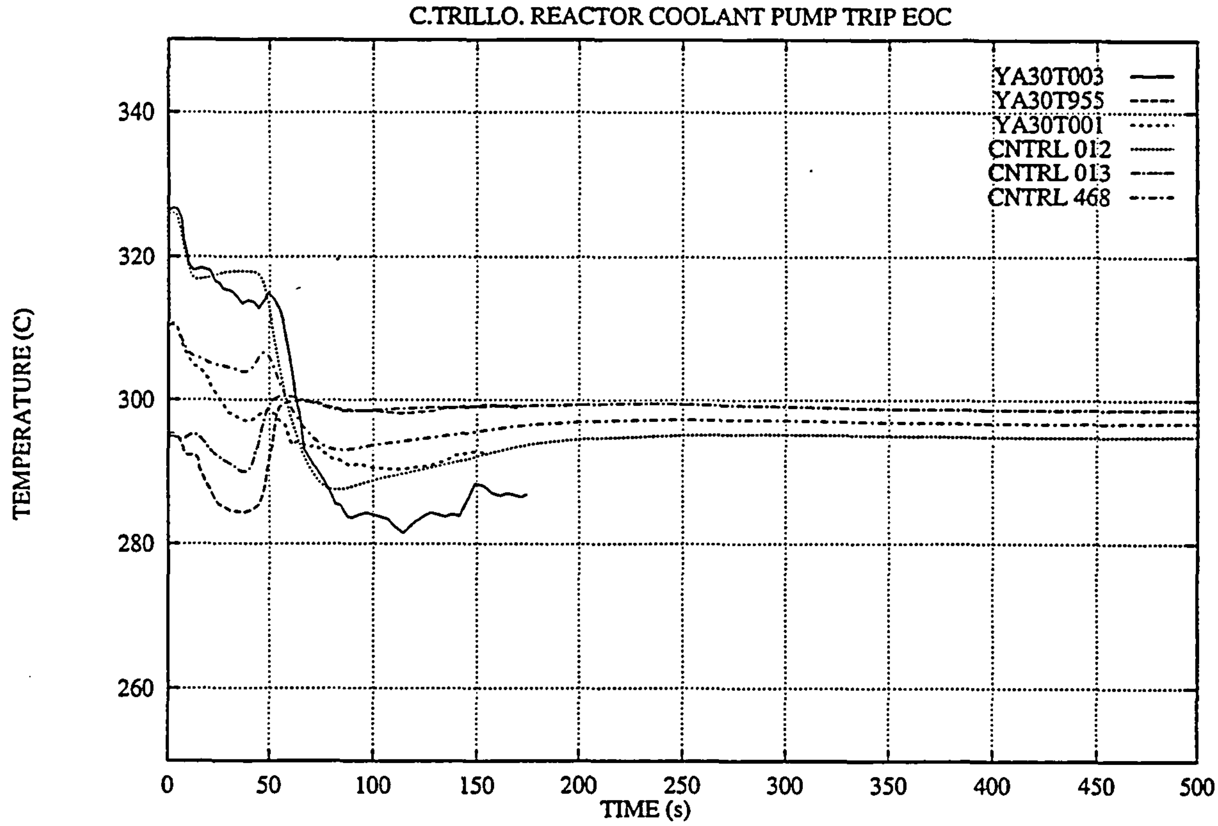

FIG.2.5.- LOOP 2 TEMPERATURES

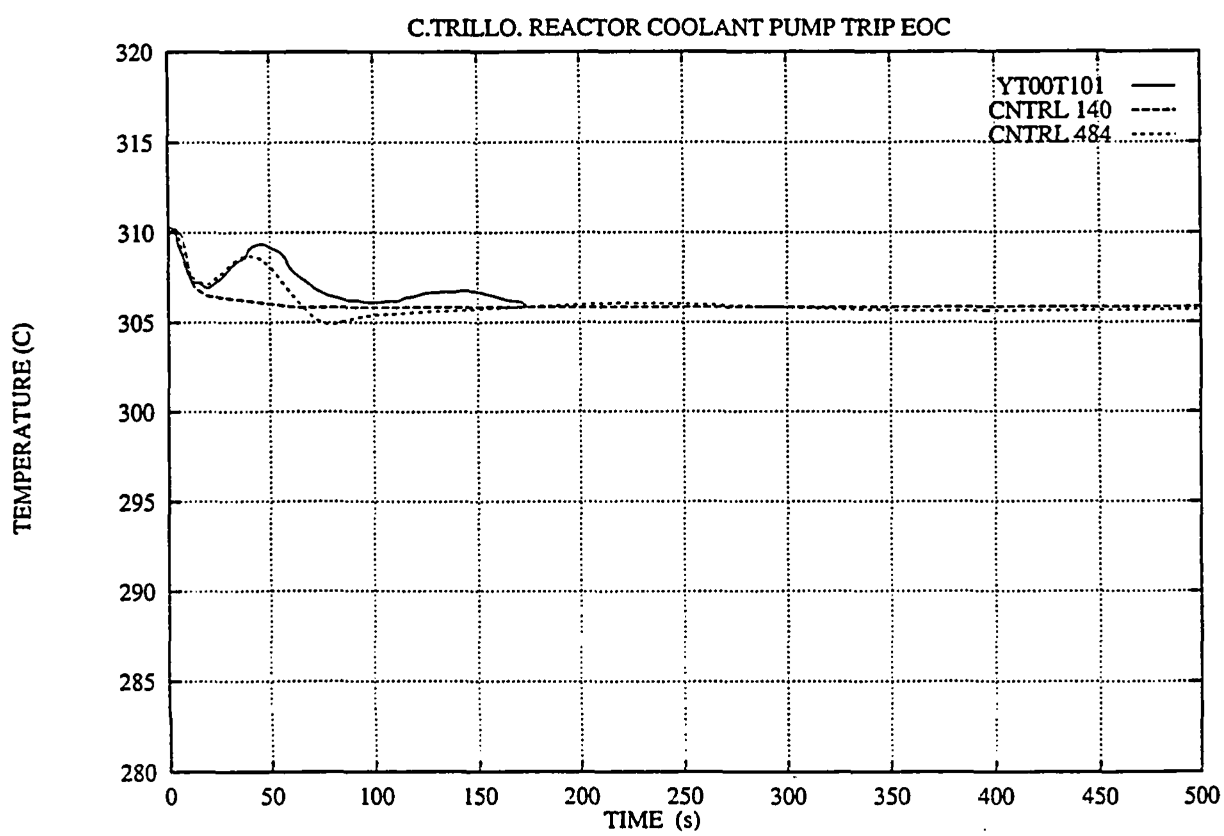

FIG. 2.6.- PRIMARY AVERAGE TEMPERATURE

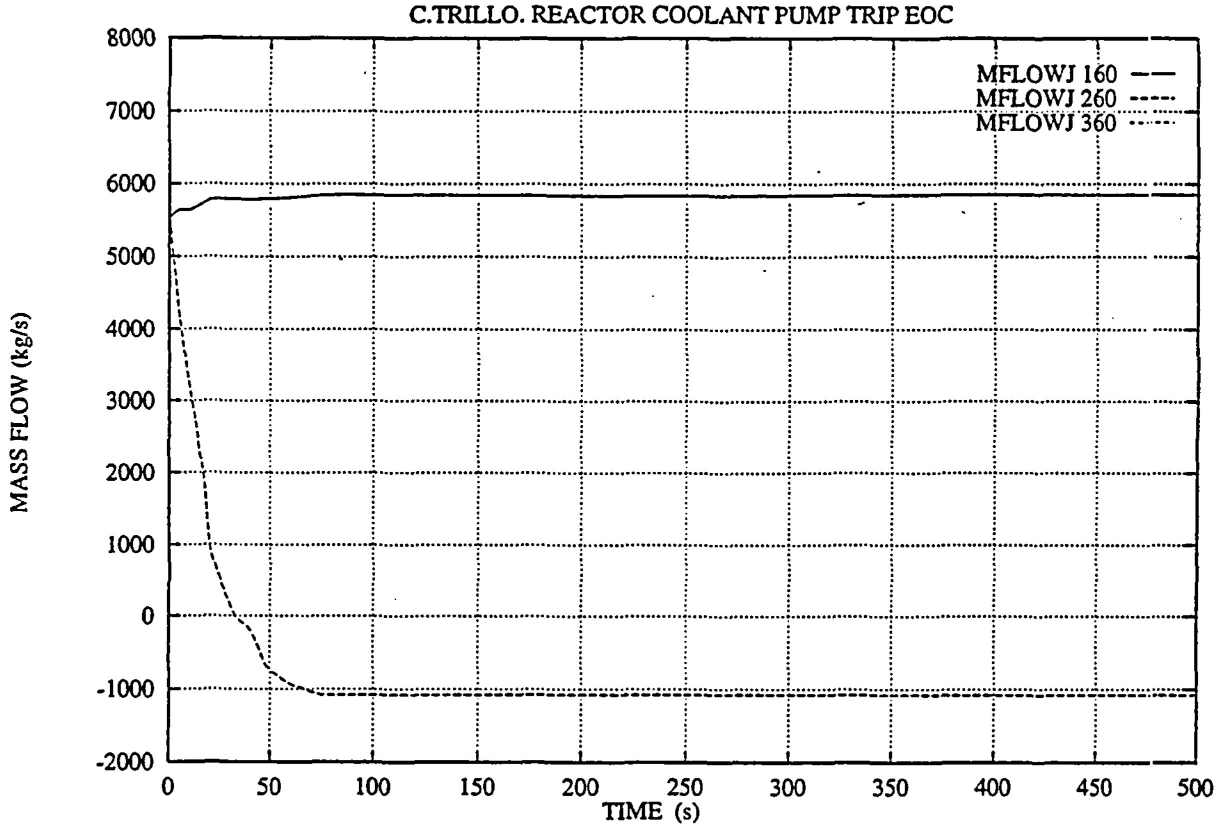

FIG.2.7.- COOLANT MASS FLOW RATE

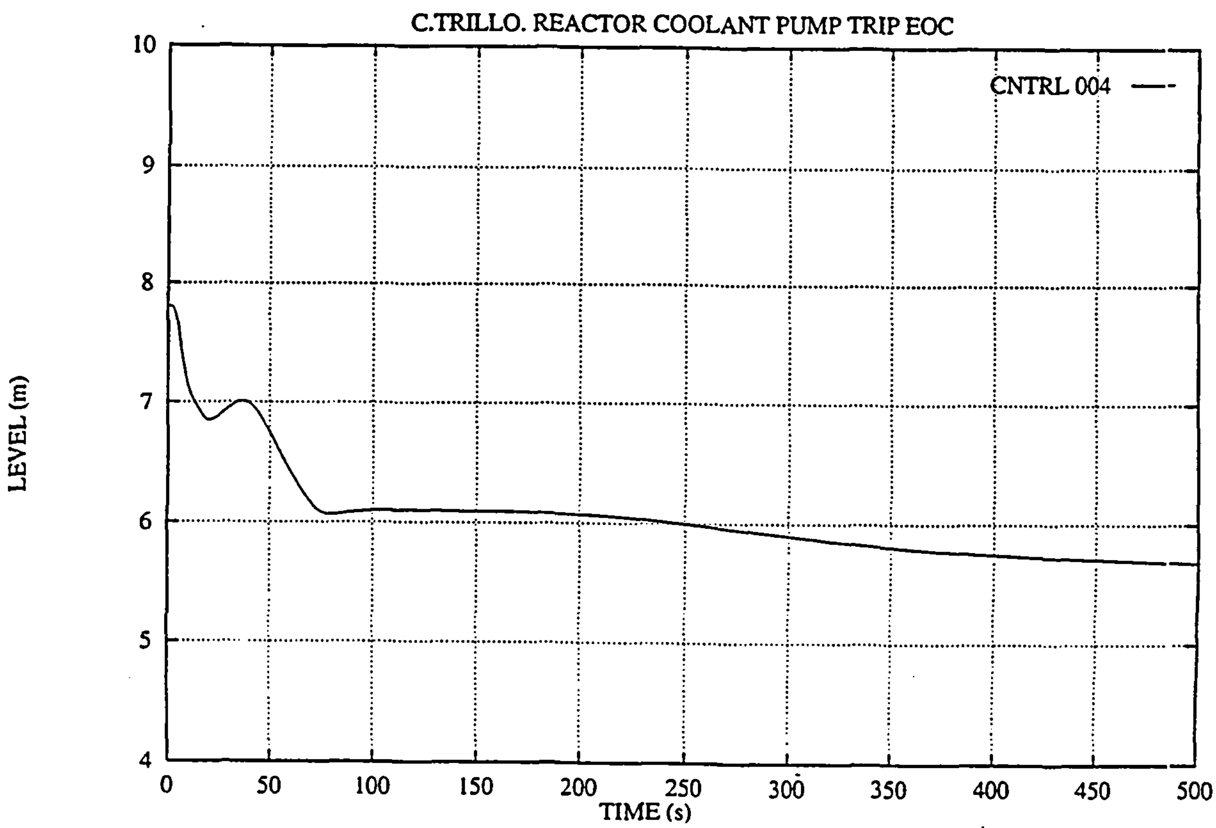

FIG.2.8.- PRESSURIZER LEVEL

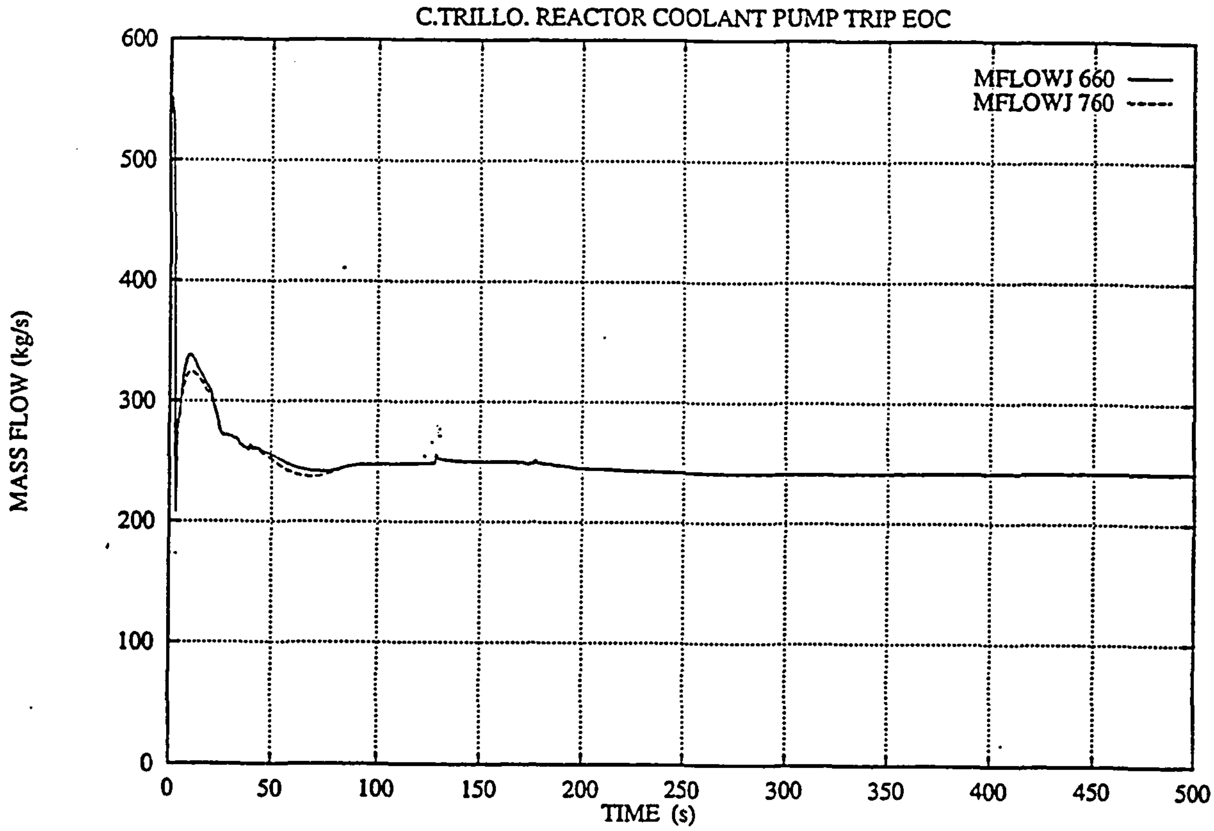

FIG.2.9.- S.G. MASS FLOW RATE (LOOP 1,2

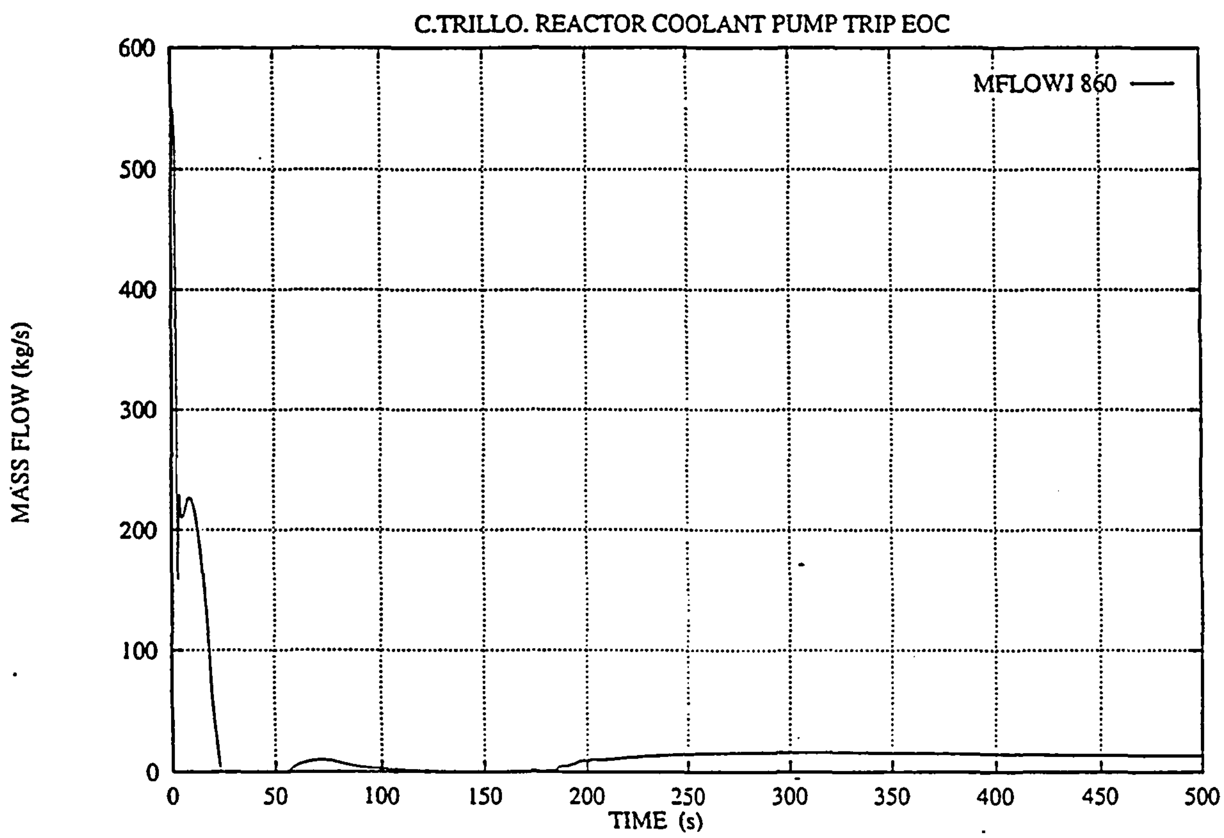

FIG.2.10.- S.G. MASS FLOW RATE (LOOP 3)

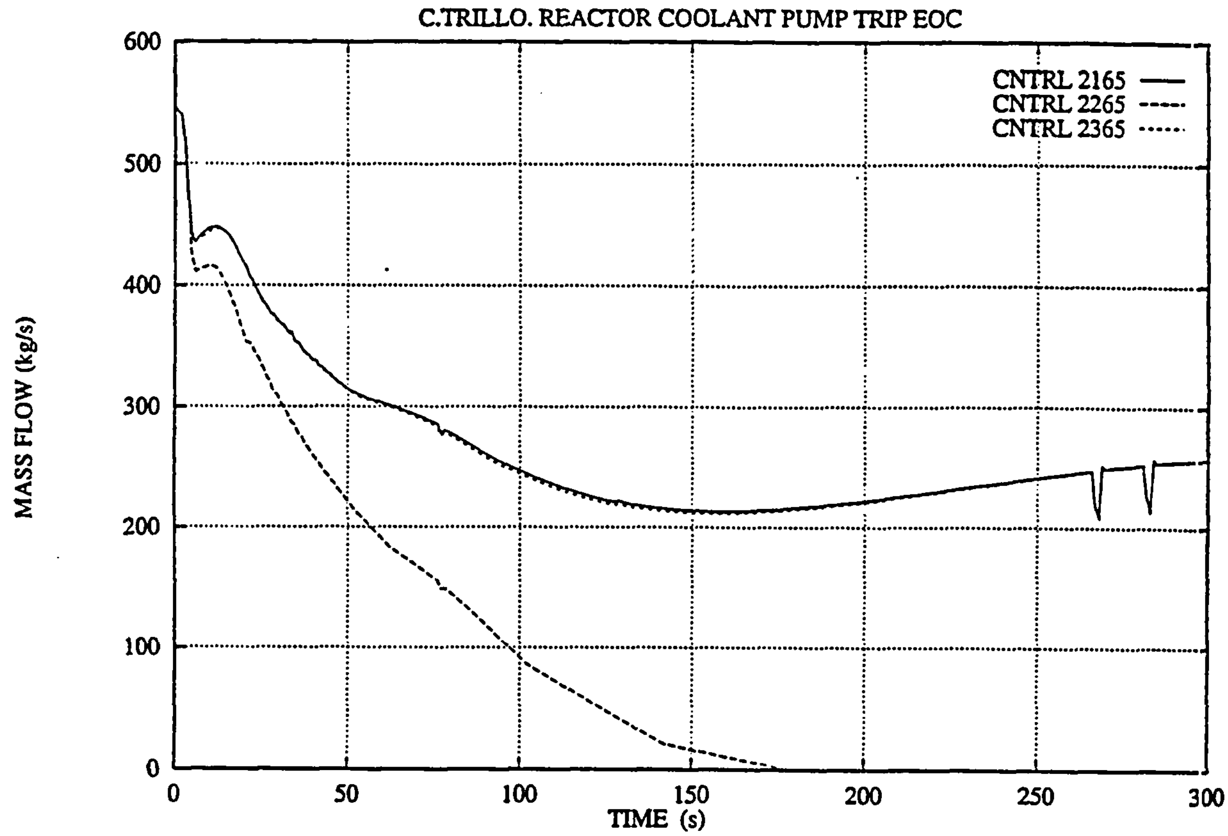

FIG.2.11.- FEEDWATER MASS FLOW RATE

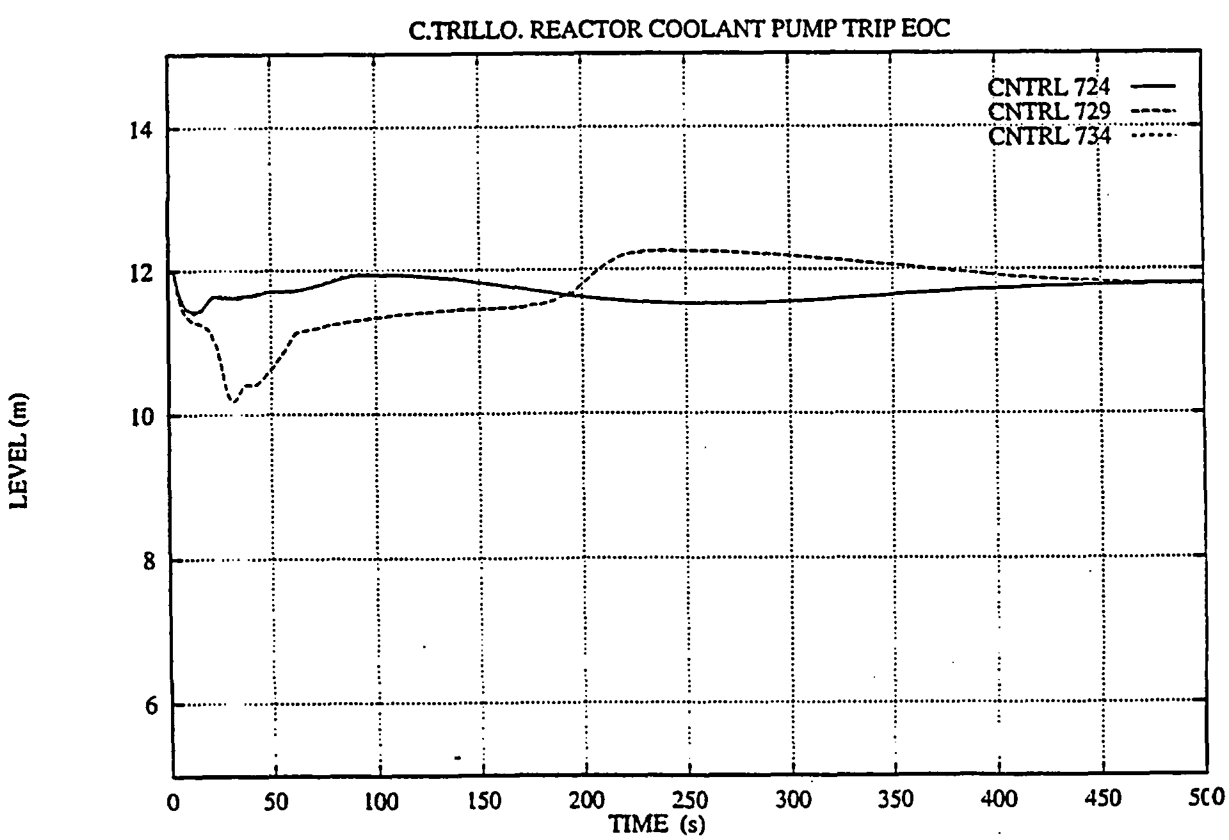

FIG.2.12.- STEAM GENERATOR LEVEL

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This document presents the comparison between the simulation results and the plant measurements during a Reactor Coolant Pump Trip in C. Trillo Nuclear Power Plant in different burnup conditions. This work is part of the Spanish contribution to the Code Assessment and Maintenance Program (CAMP) as a member of UNIDAD ELECTRICA, S.A. (UNESA). C. Trillo is a 3010 MWth three loop PWR Nuclear Power Plant designed by Siemens-KWU. Commercial operation started in August 1988, currently the plant is running the 11th annual cycle. The simulation has been carried out with the RELAP5/MOD3.2 code, running on a HP 715/50 work-station. The transient main parameters have been correctly assessed, however predictions at the beginning of transient are more difficult as the reactor remains at a high power level after pump trip.

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