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International Agreement Report

Development of a Vandellos II NPP Model using the TRACE Code: Application to an Actual Transient of Main Coolant Pumps Trip and Start-up

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ABSTRACT

The Thermal-Hydraulics Studies Group (GET) of the Technical University of Catalonia (UPC) has developed a model of Vandellòs II NPP (CNV II) for TRACE using as a previous RELAP5 model of the plant. The model simulates the components of the primary and secondary circuits, along with the main heat structures (nuclear fuel, SG tubes and pressurizer heaters). Passive heat structures and neutron kinetics are not simulated in the model. The model includes some of the plant control systems.

The main predicted steady state nominal parameters successfully match plant and design values. A real loss of off-site power with the subsequent Reactor Coolant Pumps start-up sequence (occurred in August 24th of 1993) has been simulated. The asymmetric behaviour of the flows in the vessel has been useful in qualifying the 3D capabilities of the TRACE VESSEL component.

Mass flows of primary loops are among the most important parameters in the analyzed transient. The simulation results are very similar to the data recorded at the plant. The similarity of calculated and measured values is a guarantee of the validity of the model.

FOREWORD

This report represents one of the assessment/application calculations submitted in fulfillment of the bilateral agreement for cooperation in thermal hydraulic activities between the Consejo de Seguridad Nuclear (CSN) and the US Nuclear Regulatory Commission (USNRC) in the form of Spanish contribution to the Code Assessment and Management Program (CAMP) of the USNRC, whose main purpose is the validation of TRACE code.

The CSN and UNESA (the association of the Spanish utilities), together with some relevant universities, have set up a coordinated framework (CAMP-Spain), whose main objectives are the fulfillment of the formal CAMP requirements and the improvement of the quality of the technical support groups that provide services to the Spanish utilities, the CSN, the research centers and the engineering companies

This report is one of the Spanish utilities contributions to the above mentioned CAMP-Spain program and has been reviewed by the AP-28 Project Coordination Committee for the submission to the CSN.

UNESA
December 2009

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EXECUTIVE SUMMARY

Vandellòs II is a three-looped PWR of Westinghouse design located in the northeast coast of Spain. It started its commercial operation in March 1988. Its nominal power is currently of 1,087 MWe (2,940.6 MWt).

The Thermal-Hydraulics Studies Group (GET) of the Technical University of Catalonia (UPC) has developed a model of Vandellòs II NPP (CNV II) for TRACE using as a previous RELAP5 model of the plant. The TRACE model does not include all the scope of the RELAP5 model and is still pending of the usual and complete qualification process that the RELAP5 model has undertaken.

TRACE code version used is 5.0RC3. A direct translation to TRACE has been attempted (excluding the reactor vessel and control blocks) using the plug-in "R5 to TRACE" of the SNAP program. It has been found that the resulting translation had some errors that have been corrected by "hand work".

The model simulates the components of the primary and secondary circuits, along with the main heat structures (nuclear fuel, SG tubes and pressurizer heaters). Passive heat structures are not simulated in the model, but probably they will be incorporated in a near future. Neutron kinetics has not been modelled at the moment. The model includes some of the plant control systems (Pressurizer level and pressure, steam generator level and main feed water flow, turbine ...)

The first calculation performed with the model has been that of the nominal steady state. The main predicted parameters successfully match plant and design values.

New TRACE model for CNV II has an advantage compared to the previous RELAP5/Mod3.2 as it simulates the reactor pressure vessel using a three dimensional component (the VESSEL component of TRACE).

A real loss of off-site power (occurred in August 24th of 1993) has been simulated using the TRACE model. The event produced the Reactor Coolant Pump (RCP) trip and, after this, the reactor and turbine trips. The subsequent RCP start-up sequence produced an asymmetric behaviour of the flows in the vessel which has been useful in qualifying the 3D capabilities of the TRACE VESSEL component.

Transient data (with a 2 seconds record frequency) are available from a total of 95 variables.

The TRACE calculation of the transient has been done imposing, on the one hand, the thermal power in the core and, on the other hand, the Auxiliary Feed Water mass flow rate (different in each of the steam generators), in order to reproduce the plant conditions.

Mass flows of primary loops after the shutdown and subsequent start-up of the pumps are among the most important parameters in the analyzed transient. The simulation results are very similar to the data recorded at the plant.

The signs and the magnitudes obtained for azimuthal mass flows in the downcomer of the vessel show significant consistency.

The results that best fit the data recorded at the plant are SG pressures, SG levels and primary average temperature. This shows a certain quality of the developed model.

Regarding average temperature, TRACE model values are higher than those of the plant, which is explained by the fact that the average temperature control system is, by now, not included in the model.

The similarity of calculated and measured values is a guarantee of the validity of the model.

ABBREVIATIONS

AFW	Auxiliary Feed Water
ANAV	Asociación Nuclear Ascó-Vandellòs
CNV II	Vandellos II NPP
GET	Thermal-Hydraulic Studies Group (UPC)
MFW	Main Feed Water
NPP	Nuclear Power Plant
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pumps
RELAP	Reactor Excursion and Leak Analysis Program
SG	Steam Generator
UPC	Universitat Politècnica de Catalunya (Technical University of Catalonia)
UNESA	Asociación Española de la Industria Eléctrica (Association of the Spanish Utilities)

ACKNOWLEDGEMENTS

Authors are grateful to Association of the Spanish Utilities (UNESA) for funding the work presented in this report and to Asociación Nuclear Ascó-Vandellós (ANAV) for their continuous help and cooperation. Specially, authors recognize the valuable advices from Mr. Ismael Sol (ANAV).

1 INTRODUCTION

The Thermal-Hydraulics Studies Group (GET) of the Technical University of Catalonia (UPC) collaborates since 1991 with the operation responsible of Catalan Nuclear Power Plants in different tasks related to operation support 78. Among them, UPC's GET takes care of maintaining the thermal-hydraulic models of Ascó (two units) and Vandellòs II NPP that have been prepared with the aim of simulating a wide variety of transients using RELAP5/Mod3.2.

GET has also participated, in the past, in several activities related to the development of the so-called Consolidated Thermal-Hydraulic Code that culminated in the current status of the TRACE code 10.

Recently, due to an agreement with the "Asociación de la Industria Eléctrica Española (UNESA)", GET has developed a model of Vandellòs II NPP (CNV II) using TRACE. The model is considered preliminary because it does not include all the scope of the RELAP5 model and also because it is still pending of the usual and complete qualification process 9.

The TRACE version used is 5.0RC3. In preparing the model, use has been made of SNAP program in its version 0.26.6. The "plug-in" R5 to TRACE, version 0.1.3., and the AptPlot, version 6.1.1., for post-process are other used pieces of software.

The first step has been to predict a nominal steady state using the preliminary model of CNV II. The main predicted parameters successfully match plant and design values.

TRACE model of CNV II has an advantage compared to that of RELAP5/Mod3.2, due to the fact that TRACE code allows simulating the vessel using a three dimensional component.

A loss of off-site power (occurred in August 24th of 1993) is the transient simulated. The event produced the Reactor Coolant Pump (RCP) trip and, after this, the reactor and turbine trips. The system went to hot zero power and, afterwards, the RCP start-up sequence took place. The asymmetric nature of the transient helps qualifying 3D capabilities of the TRACE code.

2 PLANT DESCRIPTION

Vandellòs II is a three looped PWR NPP of Westinghouse design owned by Endesa (72%) and Iberdrola (28%), and operated by ANAV 1. It is located near Tarragona, in the northeast of Spain, and uses the Mediterranean Sea water as heat sink. It started its commercial operation in March 1988. Its nominal power is currently of 1,087 MWe (2,940.6 MWt).

The reactor vessel is Westinghouse designed. The plant has three Westinghouse (model F) steam generators of U-tubes kind without pre-heaters. Feed water enters directly the upper part of the downcomer through J-shaped nozzles.

The main features of the plant are shown in Table I:

Table I. Main features of Vandellòs II NPP

Reactor thermal power (MWt)	2940.6
Electrical power (MWe)	1087
Fuel	UO_2
Number of fuel bundles	157
Number of cooling loops	3
Reactor operating pressure (MPa)	15,4
Mean coolant temperature (K):	
Hot zero power	564,8
Full power	582,3
Steam generator (SG)	Westinghouse tipo F
Number of tubes in one SG	5626
Total tubes length in one SG (m)	98759
Tubes inner diameter (m)	0.0156
Tubes wall material	INCONEL
Coolant Recirculation Pumps	Westinghouse D 100
Volume of the primary (m^3)	106.19
Volume of pressurizer (PZR) (m^3)	39.65
PZR heaters power (kW)	1400

Some of these general features depend on the plant configuration. The table refers to the current one. The analyzed transient occurred before different technological improvements took place in the plant; related changes in the model will be presented in the appropriate section.

The most relevant features of main equipment of the Nuclear Steam Supply System (NSSS) are presented below 12.

2.1 Reactor vessel

The reactor vessel of Vandellòs II NPP is that of a three looped PWR. It is a cylindrical vessel with a hemispherical lower plenum and removable bolted cover which ensures water tightness. It contains the core, the structure supporting it, the control rod banks, the thermal shielding and other components directly related to the core.

The three inlet and the three outlet nozzles are located between the main flange and the upper part of the core. The vessel is designed and manufactured following the specific requirements of ASME code for nuclear equipment.

The vessel is made of carbon steel. The inner surfaces in contact with the cooling water are coated with a layer of austenitic stainless steel in order to minimize corrosion.

FigureFigure 1 shows an outline of the inner part of the reactor vessel.

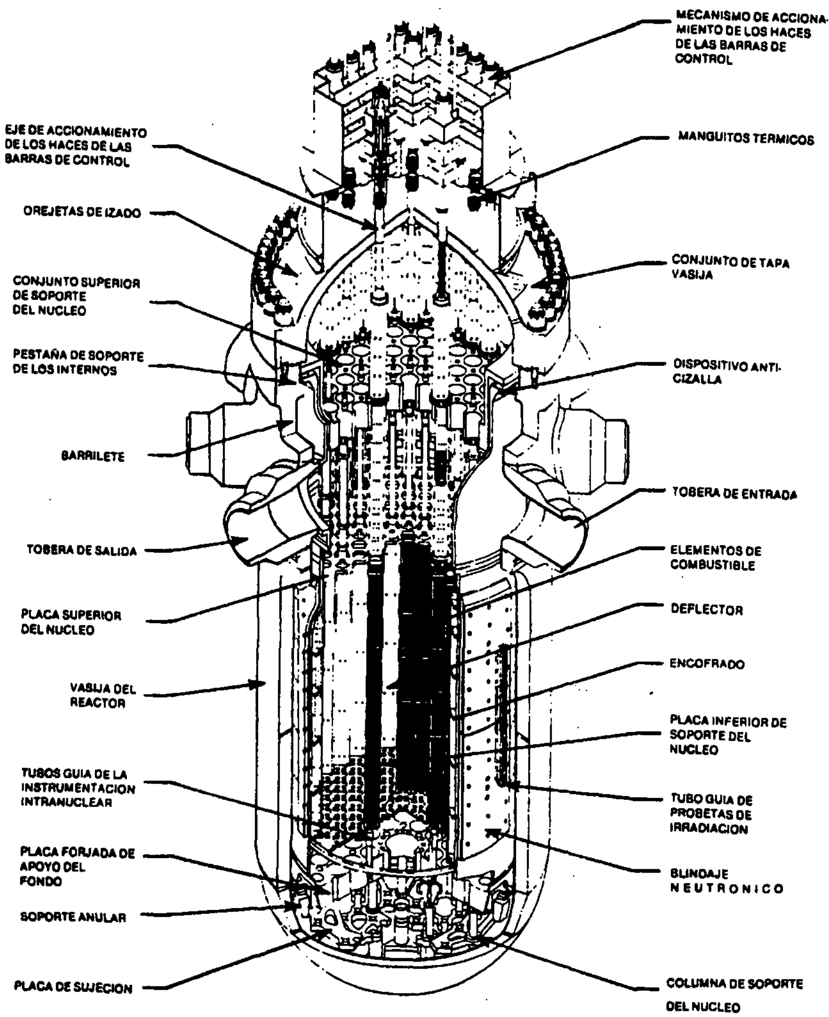


Figure 1. Outline of the inner part of the reactor vessel.

2.2 Reactor core

The reactor core consists of 157 fuel assemblies of different enrichment and burnup. Each fuel assembly has 264 active fuel rods, with the corresponding Zircaloy-4 cladding, which contain the UO_2 pellets.

2.3 Primary system

Primary pipes (hot, intermediate and cold legs) are made of steel. Their diameter is 74 cm. Hot legs length is 7.3 m.

Each loop of Vandellòs II NPP Reactor Coolant System has a one stage centrifugal vertical pump, with a system controlling seal leakage. The main pumps supply the water flow needed to cool the reactor.

The pressurizer is the component which controls the pressure in the Reactor Coolant System during normal operation and helps smoothing pressure changes during transients. Vandellòs II NPP pressurizer houses 78 vertical type heaters, with a total power of 1400 kW.

2.4 Steam generators

Steam Generators (SG) are the components which perform the heat transfer from primary to secondary circuits and produce the steam needed in the turbine. Each of the three SG 11 has 5,626 tubes made of INCONEL. The tubes are inverted U type, with an inner diameter of 1.56 cm and a total length of 98,759 m.

Figure 2 shows the outline of the SG.

The lowest part of SG primary side is hemispherical and split in two by a plate. The reactor coolant comes in through one of the mentioned two halves, then through the Inconel tubes and finally comes out through the other half. The heat from the reactor coolant is transferred through the wall of the tubes.

The secondary side is a shell that acts as a pressure barrier around the tube bundle. Separators and driers are located in the upper part. The water supply comes through the feed water nozzle and the feed ring designed to prevent water hammer.

The heat transferred by the reactor cooling water through U tubes raises the temperature of the secondary side to saturation, producing steam with high moisture content. After this, the steam goes through turbo separators and dryers that remove liquid water from vapour flow to the turbine. The water separated by the separators mixes with the main feed water and comes through a downcomer to the inlet of the tube bundle.

2.5 Secondary system

The main feed water (MFW) is driven by two turbo pumps. Pressure and flow of MFW system is controlled through the speed of turbo pumps and the opening of MFW valves.

The Auxiliary Feed Water (AFW) system is driven by one turbo pump and two motor pumps. The flow is manually controlled through AFW valves.

The current turbine of Vandellòs II NPP consists of one high pressure and three low pressure sections mechanically coupled by a shaft to the electrical generator.

Moisture Separators Reheaters are located between the high and low pressure turbine components. The steam from the header enters the high pressure turbine component through 4 inlet valves, which control the steam flow. In series, the stop valves are designed to isolate and trip the turbine.

The role of the steam dump system is to discharge directly to the condenser the power balance between nuclear and turbine. Its normal operation is in temperature mode (its performance is determined by changes in the reactor average temperature). It can also work in pressure mode.

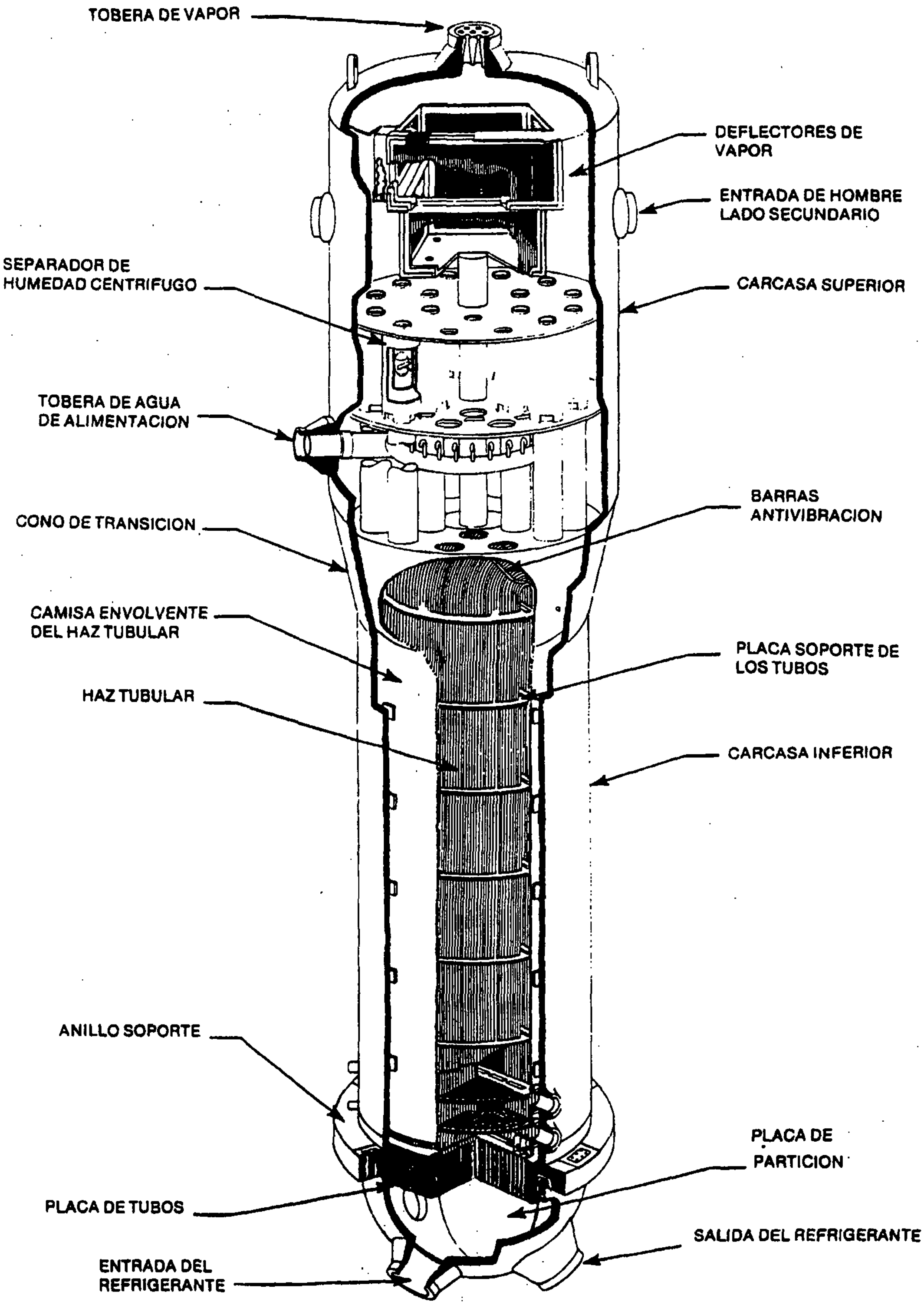


Figure 2. General view of a steam generator.

2.6 Control and protection systems

The plant has, among others, the following control and protection systems 4, 5, 13:

- Reactor control system (by average temperature and power mismatch)
- Primary system pressure control
- Pressurizer pressure control system
- Steam dump control system
- Feed water control system
- Reactor protection system
- Pressurizer relief and safety valves system
- Steam generators relief and safety valves system

3 TRANSIENT DESCRIPTION

The analyzed transient occurred on August 24, 1993 in Vandellòs II NPP, when a momentary loss of offsite power caused the shutdown of the main reactor coolant pumps (RCP) and subsequently the trip of the reactor and the turbine. The system went to hot stop and the operators proceed to the sequential RCP's start-up.

Data from the transient are available. The available records cover the period of time between 16:15:00 on the 24th until 17:30:00 on the same day, which means a total of 4500 seconds, of which 618 seconds belong to the steady-state period (before the loss of supply) and the rest to the transient.

The data come from the data acquisition system of CNV II 3. Measurements are available from a total of 95 variables. The frequency of recording information available is 2 seconds and so a total of 2251 values for each studied variable are considered. The records of the data acquisition system do not include the start-up of the last coolant pump, in particular that of the first loop, which was started after 17:30:00.

The transient began at 16:25:18. Henceforth, and to facilitate the understanding of the timing, the origin of time (t = 0s) is fixed at 16:15:00, the time at which the recorded data are available.

During the time between t = 0s and t = 618s, the plant was operating at full power (2775 MW on the date of the transient) in steady-state condition and at nominal levels.

At time 618 s a sudden and temporary loss of offsite power, caused a rapid loss of tension in the main and auxiliary transformers that connect the plant with the outside electrical grid. This loss of electricity caused the shutdown of the turbine, the RCPs and the reactor.

Also, the pressurizer heaters and the steam-dump system became unavailable.

Following the shutdown of the coolant pumps, the water flow in the three primary loops decreased and natural circulation was established.

After the turbine trip, the reference temperature becomes that of zero power and the average temperature tends to that value. This decrease in coolant temperature leads to an increase in density and, therefore, to a depressurization of the primary system.

The turbine and reactor trip caused the isolation of the main feed water and the start of the two motor pumps and the turbo pump of the AFW system which, from now on, provides water to steam generators. Modulation of AFW is done manually.

At time 1522 seconds, RCP-C (the coolant pump of the third loop) is manually started, reaching a flow rate in the 3rd loop around 107%. The start of this pump causes the appearance of reverse flow in the other two loops.

At time 2320 seconds, operators manually isolate the main steam. Secondary pressure and temperature are recovered.

The start-up of the second coolant pump (RCP-B) is carried out manually at 2584 seconds. This brings water flows in loops 2 and 3 to 106% of the nominal value and increases the reversed flow in the first loop.

From then until the end of the available data (4500 s), the system evolves towards its nominal values, recovering the SG levels.

Table II shows the events of the transient:

Table II. Principal events recorded during the transient of 24/08/93 in CNV II

Time (s)	Event
0	Start of data recording (16:15:00)
618	Momentary loss of offsite power (16:25:18)
620	RCPs trip
620	Turbine and reactor trip
635	Manual Control of the Auxiliary Feed Water system
1522	Start-up of RCP C (third loop)
2320	Main Steam isolation
2584	Start-up of RCP B (second loop)
4500	End of data (17:30:00)

4 MODEL DESCRIPTION

The following are the main features of the model of Vandellòs II NPP using the code TRACE. This model has been developed from the existing for RELAP5, which has proven its performance. Both models share technical information and the corresponding descriptive document 2.

A direct translation to TRACE has been tried using the plug-in "R5 to TRACE" of the SNAP program. This has been done excluding the reactor vessel, as it is three-dimensional in TRACE and one-dimensional in RELAP5. Control blocks have also been excluded because the translation does not perform correctly for this particular.

This way, a TRACE model was obtained for the primary and the secondary systems. This model, a priori, only needed to be checked and corrected for the elevations of the different components to be correct. It was found that the resulting translation had many additional errors. The errors (and the solution applied) are the following:

- Some junctions appeared twice after translation; so, some of them had to be deleted.
- The friction coefficient obtained by translation was zero on all sides of the nodalization volumes of each component. The present value of these coefficients has been introduced based on the available information of the RELAP5 model.
- The translation does not cover the difference between RELAP5 and TRACE in terms of homologous curves of pump components. Overlapping or null values were found after translation, as well as inconsistencies in the figures. Such curves have been implemented from scratch according to the values given in the descriptive document 2.
- The correspondence between the types of valves in both codes is not direct. For this reason some of them have been changed.

To describe in more detail the model, the following parts will be considered:

- Primary hydrodynamic volumes
- Secondary hydrodynamic volumes
- Heat structures and power components
- Control systems

4.1 <u>Primary hydrodynamic volumes</u>

The model of the primary circuit for TRACE includes the following main parts:

- Reactor vessel
- Primary loops, including SG U-tubes
- Reactor Coolant Pumps
- Pressurizer

Figure 3 shows an outline of the general layout of the components of the primary circuit.

The main features of the model for the primary circuit of CNV II are presented below.

To simulate the reactor pressure vessel the TRACE VESSEL component has been used. This component allows the simulation of a 3-dimensional region. For the model of CNV II, 12 axial levels, 2 radial and 3 azimuthal nodes have been considered (Figure 4).

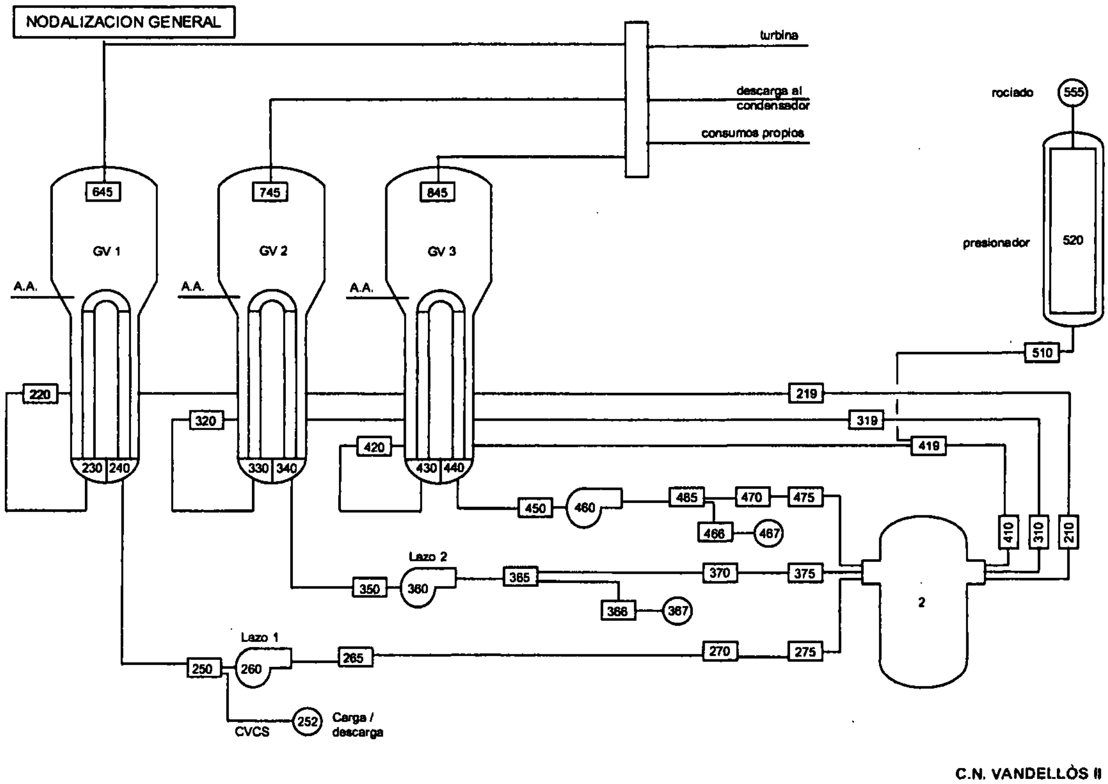


Figure 3. General layout of the primary circuit components in the model of CNV II

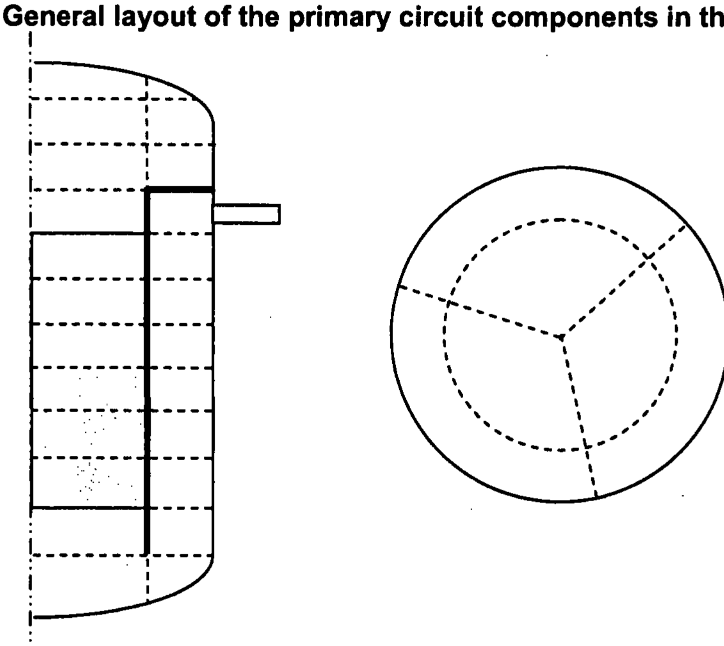


Figure 4. Sketch of the vessel model

The downcomer is located in the outer ring from axial level number 2 to level 9. The core takes axial levels from 3 to 8 of the inner part of the vessel.

The primary circuit of CNV II includes three loops. Each loop consists of the following parts and components:

1. Vessel outlet
2. Hot leg
3. SG inlet
4. SG tubes
5. SG outlet
6. Intermediate leg
7. RCP
8. Cold leg
9. Vessel inlet

RCPs are modelled using the PUMP component of TRACE. The values of the RCP technical features come from design information (nominal speed, friction torque, torque, inertia ...). The homologous curves correspond also to the design.

The pressurizer is connected to hot leg 3 by the TEE 419, as can be seen in Figure 3. The specific component PRESSURIZER available in the code has not been used. The two components used (510 and 520) are PIPEs.

The pressurizer spray system has been simulated in a simplified way using components FILL of TRACE.

4.2 Secondary hydrodynamic volumes

The model of the secondary circuit for TRACE includes the following main parts:

- SG
- MFW and AFW systems
- Steam lines and relief valves
- Turbine
- Steam-Dump
- Steam consumptions

The secondary of the steam generators includes the following parts and components:

1) FW feed ring
2) downcomer
3) boiler
4) turbo separators
5) driers and steam dome

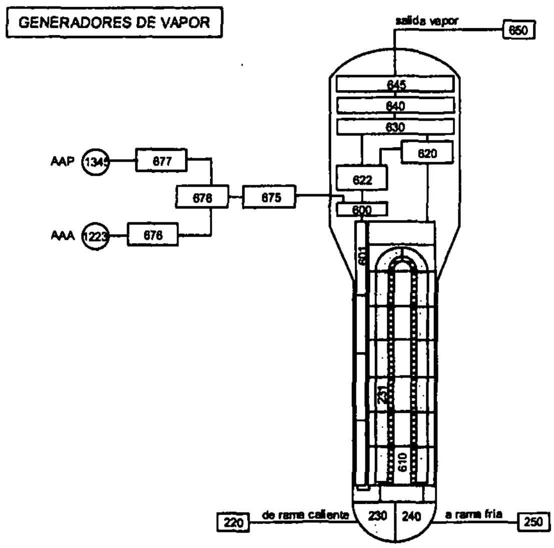


Figure 5. Represents the hydrodynamic model of one SG. The three steam generators are identical.

The mass of vapour and liquid water inside of SG, as well as the recirculation rate have been checked and they agree with design information.

MFW and AFW are simulated as boundary conditions.

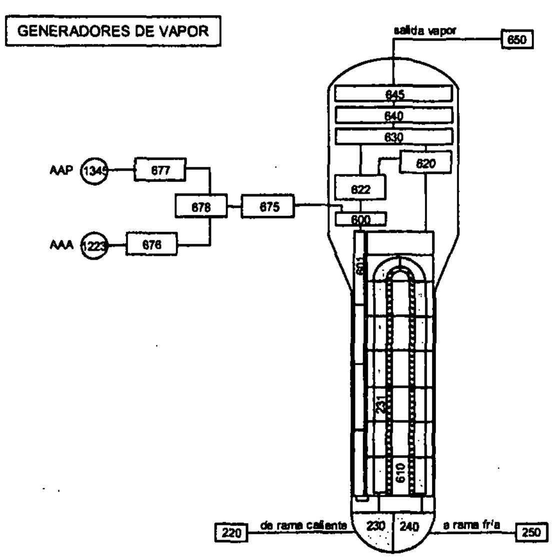


Figure 6. Drawing of the hydrodynamic model of the steam generators

The model includes the steam lines, the steam isolation valves, the steam header, the turbine valves (for control and for stop), the steam-dump (simplified form) and steam consumptions.

Figure 7 shows a nodalization diagram of the TRACE model.

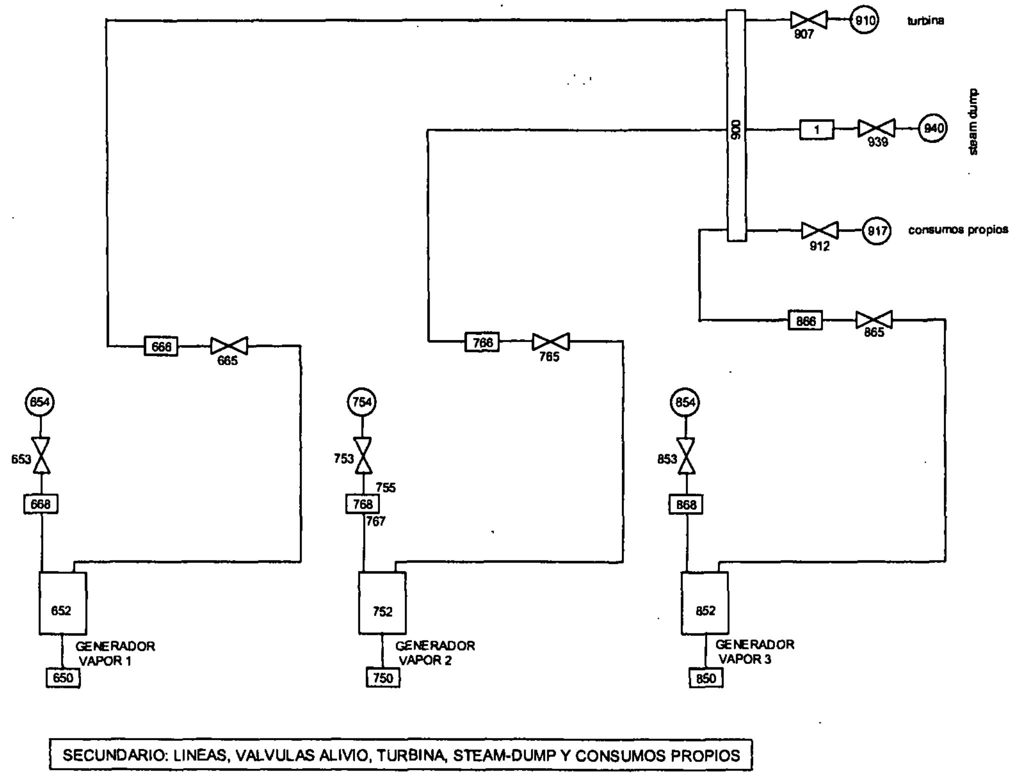


Figure 7. Sketch of the simulation of secondary system lines in the CNV II model.

4.3 Heat structures and power components

Next, the main features of the different heat structures and power components used in the TRACE model are described. The main heat structures modelled are:

- Heat transfer from primary to secondary sides through SG tubes
- Heat transfer from nuclear fuel to the primary coolant
- Pressurizer heaters

Other components such as high and low pressure MFW heaters are not simulated.

The heat structures that could represent the heat losses from the different components (vessel, steam generators, piping, etc.) to the environment are not simulated. The passive internal heat structures are not part of the model. The former have a limited impact on the transient analysis. The latter are awaiting the results of other studies, carried out in parallel, with RELAP5 and will probably be implemented in the near future.

The U-tubes are simulated by 3 heat structures, one for each loop and steam generator. Each one consists of 14 nodes: nodes 1 to 6 correspond to the ascending part of the tubes; nodes 7 and 8 correspond to the upper curve of the tube; and nodes 9 to 14 to the descending part.

The material of the pipes is INCONEL 600. The table of properties of this material has been obtained from the RELAP5 model.

The heat structures and power components modelling heat transfer between the fuel and primary circuit are justified below. Taking into account that the 3D vessel has 3 azimuthal sectors, 3 cylindrical heat structures have been modelled. Each heat structure corresponds to a power component. These heat structures simulate fuel rods. Each structure is divided in 6 axial nodes.

Finally, there is a power component to provide the heating of the 3 heat structures that simulate the fuel rods. Power has been introduced as a boundary condition. The reactor kinetics has not been modelled in this preliminary model of CNV II because in the transient discussed below kinetics plays a minor role. For this particular analysis, the residual power has been introduced as a boundary condition using a table.

Backup and control heaters are simulated as a single heat structure, associated with a power component. The pressure control system determines the power of this component.

4.4 Control systems

The model includes the following control systems 6:

- Pressurizer level
- Pressurizer pressure
- SG level and MFW flow
- Turbine control
- Secondary relief
- Steam-dump

As an example the diagram of the pressurizer pressure control system is shown in Figure 8.

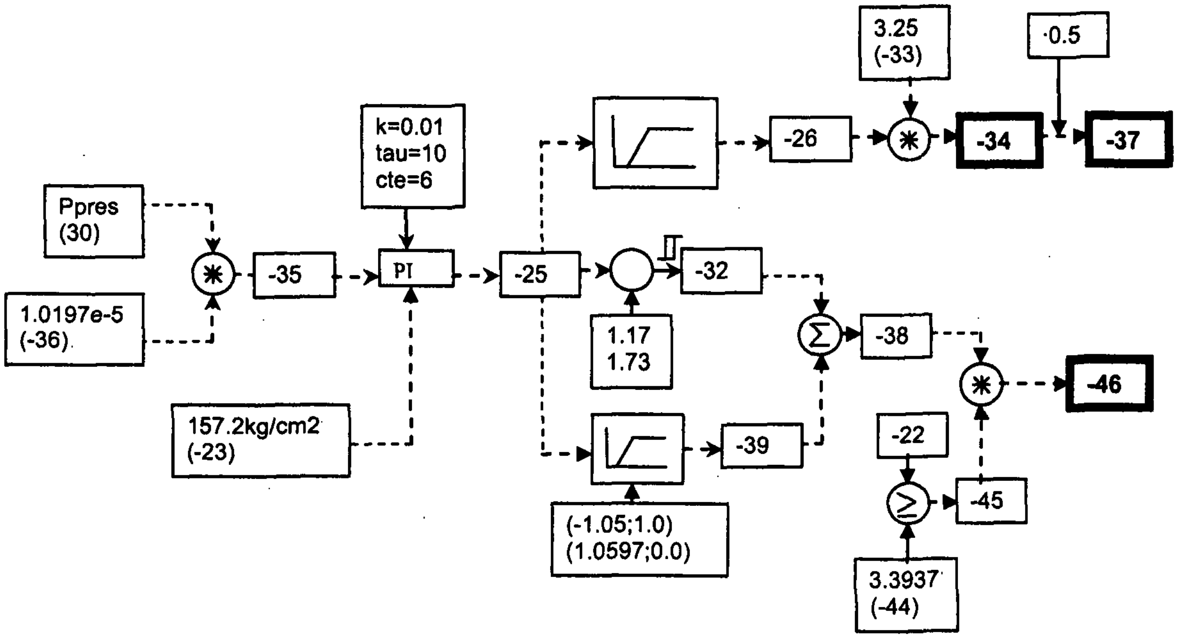


Figure 8. Diagram of the pressurizer pressure control

5 SIMULATION

5.1 Steady state calculation

The model described above has been used for calculating a steady-state.

Table III shows the comparison of the values of key parameters in the model with those of the plant (collected by instrumentation or established in the design).

5.2 Transient features

In Table II the chronology of events corresponding to the analyzed transient has been shown. Main boundary conditions have to do with the following areas:

- Reactor power
- Trip and start-up of RCP
- Turbine trip
- Steam isolation
- MFW and AFW
- Steam-dump

Table III. Comparison of key steady state parameters

	Plant	TRACE	Absolute Error	Relative Error (%)
Nuclear power (%)	100.56	99.43	-1.13	-1.12
Temperature hot leg 1 (K)	598.55	603.04	4.50	not applicable (na)
Temperature hot leg 2 (K)	598.12	602.33	4.21	na
Temperature hot leg 3 (K)	599.29	602.39	3.10	na
Mean temperature loop 1 (K)	581.92	585.22	3.31	na
Mean temperature loop 2 (K)	581.29	584.75	3.46	na
Mean temperature loop 3 (K)	581.85	584.86	3.01	na
Temperature cold leg 1 (K)	565.29	567.41	2.12	na
Temperature cold leg 2 (K)	564.46	567.16	2.71	na
Temperature cold leg 3 (K)	564.40	567.32	2.92	na
Pressurizer pressure (kPa)	15518.26	15513.50	-4.75	-0.03
Pressurizer level (%)	59.10	59.97	0.87	1.47
Pressure SG 1 (kPa)	6747.78	6770.45	22.66	0.34
Pressure SG 2 (kPa)	6726.34	6742.43	16.09	0.24
Pressure SG 3 (kPa)	6949.33	6745.42	-203.91	-2.93
Level SG 1 (%)	51.10	50.00	-1.10	-2.14
Level SG 2 (%)	49.79	50.00	0.21	0.41
Level SG 3 (%)	49.37	50.00	0.63	1.27
Steam flow 1 (kg/s)	521.21	514.13	-7.08	-1.36
Steam flow 2 (kg/s)	526.54	512.49	-14.05	-2.67
Steam flow 3 (kg/s)	535.04	505.54	-29.50	-5.51
Recirculation ratio*	3.27	3.13	-0.14	-4.23

* Design

- *Reactor power*
  In the steady-state, the reactor is at full power (100%). The value of power used is 2798 MWth. This value was used in a preliminary RELAP5 calculation and is maintained for possible comparisons RELAP5-TRACE, despite of being slightly higher than the other two values referenced below:

  - 2775 MWth (nominal value for the configuration of the plant when the transient took place)
  - 2791.2 MWth (value obtained performing an energy balance calculation)

Relative errors, respectively 0.8% and 0.2%, are considered low enough.

Power values after the reactor trip are obtained from a RELAP5 calculation.

- *Trip and start-up of RCP*
  In order to simulate the trip and start-up of RCPs, the following logical variables haven been defined:
  - variable 1: it stops RCP 260 (loop 1) at time t=620 s.
  - variable 2: it stops RCP 360 (loop 2) at time t=620 s and starts it at time t=2584 s.
  - variable 3: it stops RCP 460 (loop 3) at time t=620 s and starts it at time t=1522 s.
- *Turbine trip*
  The turbine trips at time 620s.
- *Pressurizer heaters control*
  Pressurizer heaters are disconnected at time 620 s and they connected again at time 700 s.
- *Steam isolation*
  Steam isolation valves remain open until time t=2320 s and close after this time.
- *MFW*
  MFW stops at time t=620 s.

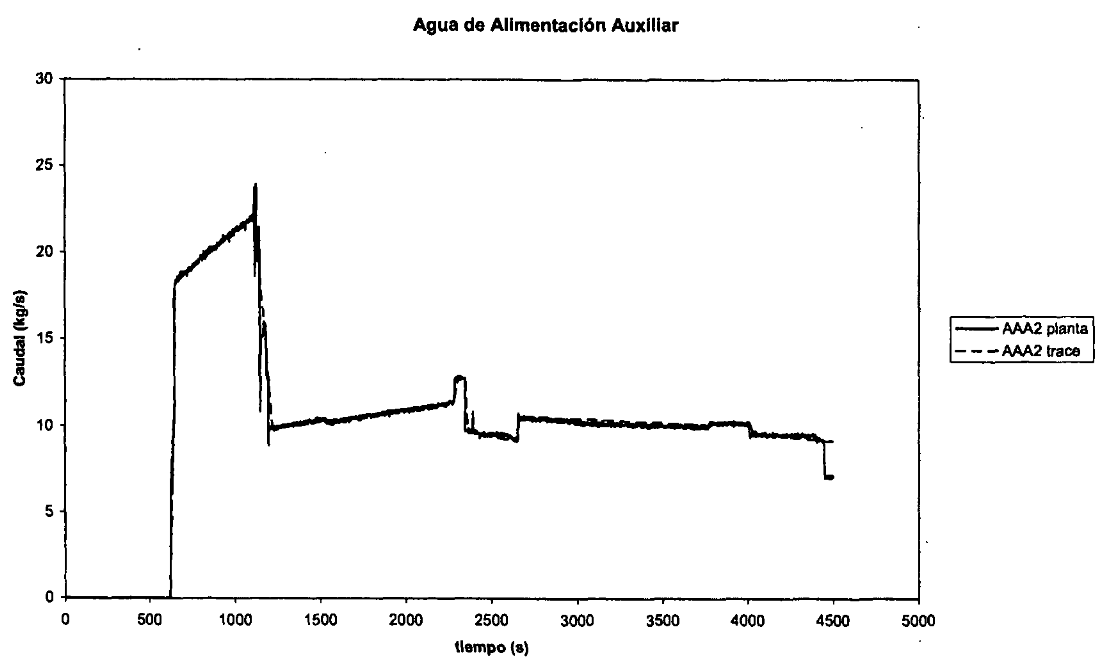


Figure 9. Auxiliary Feed Water mass flow rate

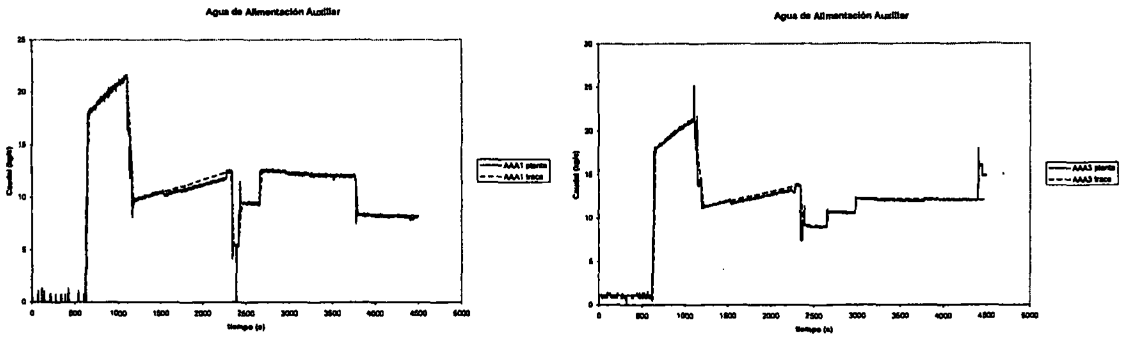


Figure 9. Auxiliary Feed Water mass flow rate (Continued)

- *AFW*
  In order to control AFW flow, plant data have been introduced using TRACE FILL components which simulate the AFW inlet. Figure 10 shows the corresponding curves.

- *Steam dump*
  The steam dump is not available due to the loss of offsite power.

6 RESULTS

In this last section, the main results of the simulation are presented and discussed.

As stated above, nuclear power has been introduced according to the results obtained with RELAP5 and it substantially conforms to that of the plant.

Figure 10 shows the plant nuclear power compared with that introduced in the TRACE model. It has been assumed that the calculated steady-state power fits the average power of the reactor. In the long term, a difference exists between registered data and TRACE values, due to the fact that the former are estimated from neutron detectors in the power range and are not accurate enough.

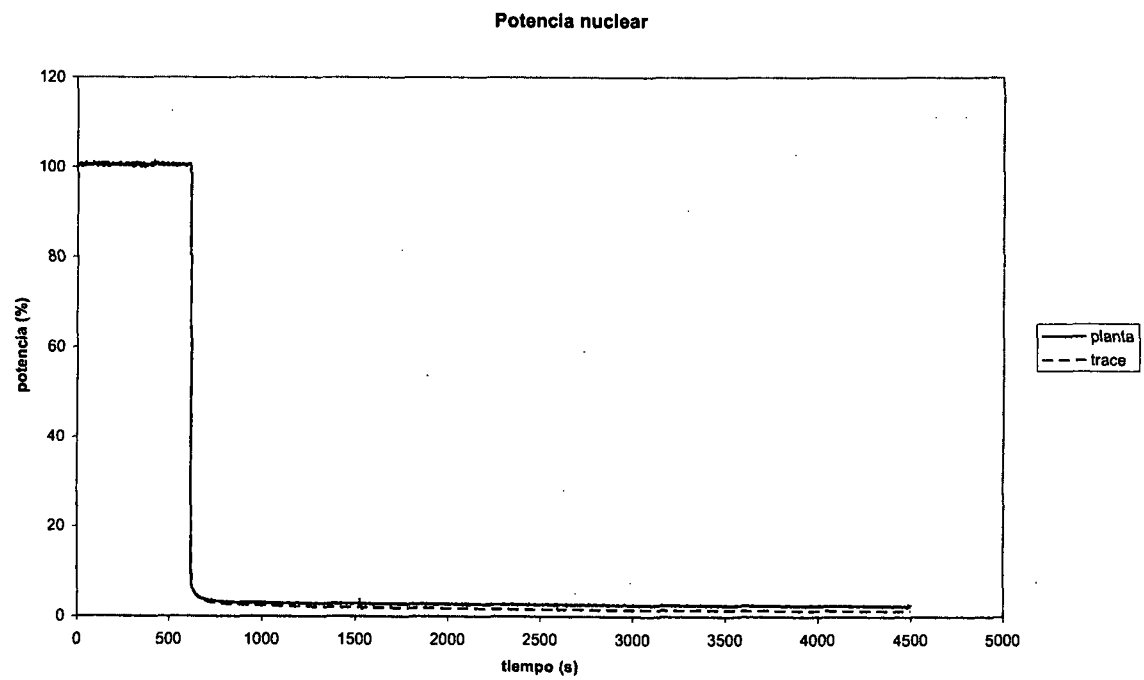
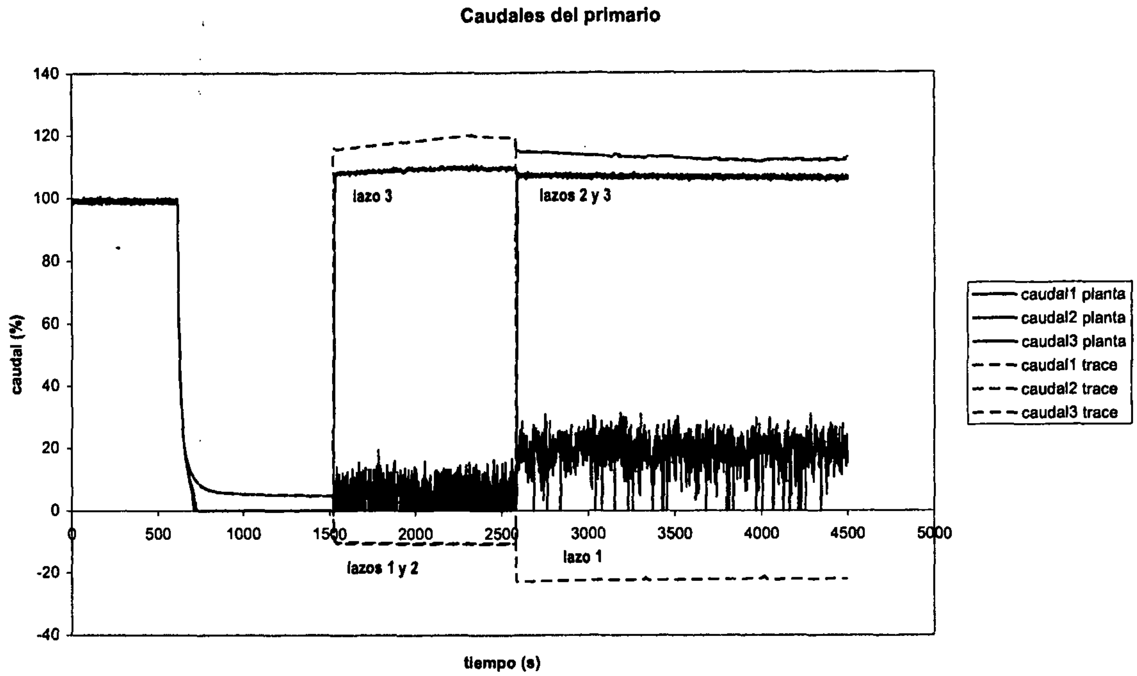


Figure 10. Nuclear power



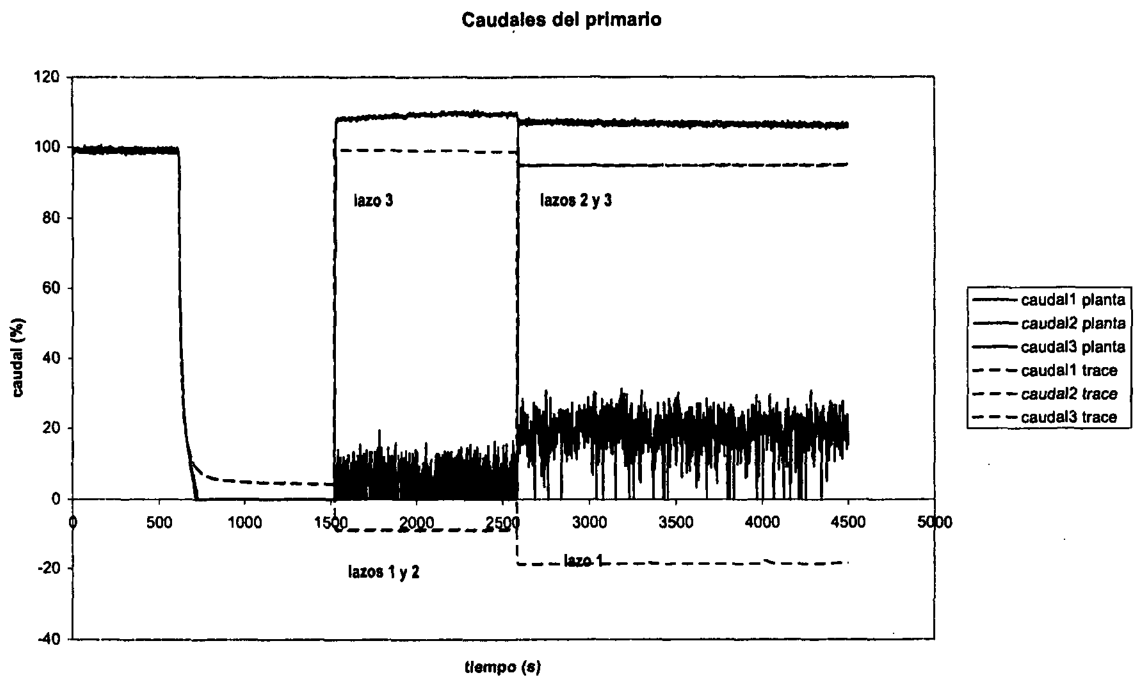


Figure 11. Primary (mass and volume) flow rates

Figure 11 compares the evolution of the predicted flow in the three loops with data collected in the plant.

It should be noted that the plant flow measurement is based on Δp at the elbows and the results are given in percentages. Just as with nuclear power, it has been assumed that the average flow (in kg/s) of the first 618 seconds is the average flow in the plant for each loop.

It should be also noted that the measurements at the plant for low flows are unreliable and that the flow meters always give positive values. With these considerations, in view of Figure 11, it can be concluded that the simulation results are significantly similar to the measured values.

Regarding average temperature, TRACE model values are higher than those of the plant. This deviation will not be treated as part of the scope of this study since the average temperature control system is, by now, not included in the model.

As shown in Figure 12, at the beginning of the transient, a sharp drop in average temperature takes place as a result of tripping the reactor. It is noted that the magnitude of the fall obtained by TRACE is similar to that recorded at the plant.

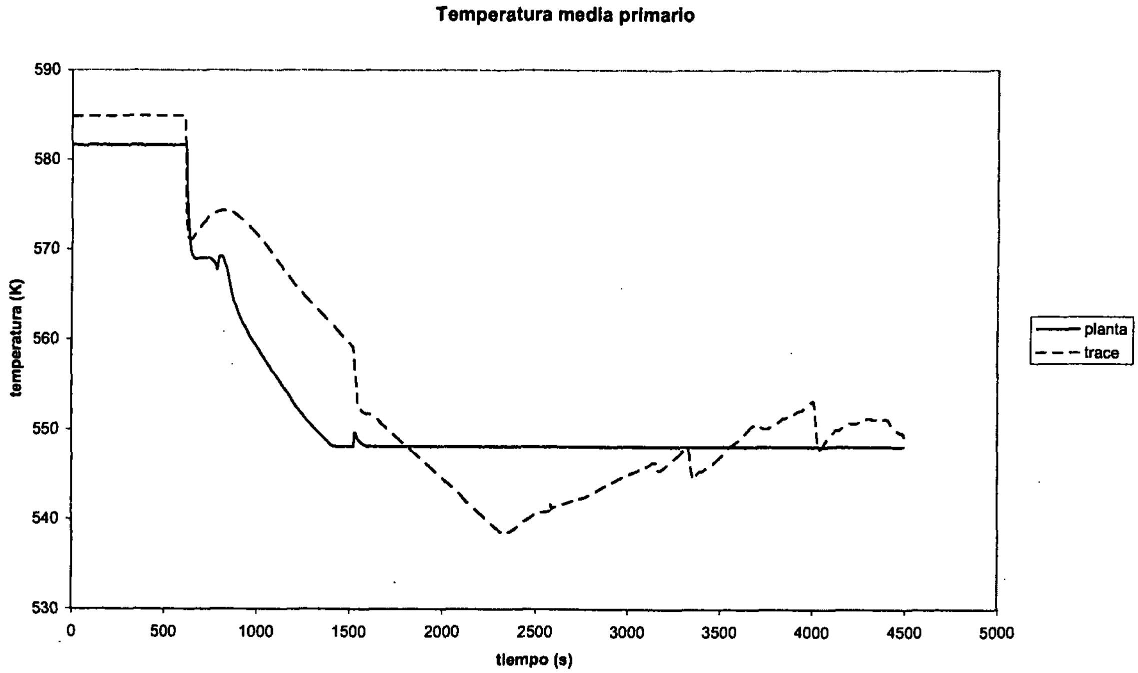


Figure 12. Primary mean temperature

At the time the transient took place, temperature was measured in the plant using thermocouples located in small pipes derived from the main loops. If forced circulation was lost, actual temperature was not reflected in the thermocouples. This explains the discrepancy between the results and the plant data. When forced circulation is recovered temperature values become similar.

Finally, it should be noted that Figure 12 shows the only values of average temperature that were available. Those values were collected by an instrument that was not able to record temperatures below 275 ° C.

Figure 13 shows the sharp decrease of primary pressure at the beginning of the transient. The simulation reproduces it, but slightly more pronounced.

In addition to the first pressure drop (1), Figure 13 shows the effect of the restart of RCP-C. The event (2) is the isolation of steam which, in TRACE calculation, causes a pressure increase more significant than in the plant. Finally, event (3) is the restart of RCP-B.

Pressurizer level follows pressure and temperature time trend.

The pressurizer level initially shows a sharp decrease and then evolves as a function of average temperature as well as of the flow of the chemical and volume control system, as shown in Figure 14.

The simulation results are similar to those recorded in the plant. The model shows a larger decrease in the level, as it also happens with the primary pressure.

Pressurizer level follows pressure and temperature time trend.

The pressurizer level initially shows a sharp decrease and then evolves as a function of average temperature as well as of the flow of the chemical and volume control system, as shown in Figure 14.

The simulation results are similar to those recorded in the plant. The model shows a larger decrease in the level, as it also happens with the primary pressure.

Figure 15 shows that, at the beginning of the transient, after the loss of offsite power, a sudden increase of secondary pressure occurs, which is due to the closure of turbine inlet valves. Later on, this pressure slowly decreases due to the AFW contribution and to the extraction of steam consumptions.

When RCP C restarts there is a small pressure increase with the improvement of heat transfer from primary to secondary.

When the steam is manually isolated, the pressure is recovered as there is no extraction of steam.

The simulation offers for SG pressure time trend is very similar to that recorded at the plant.

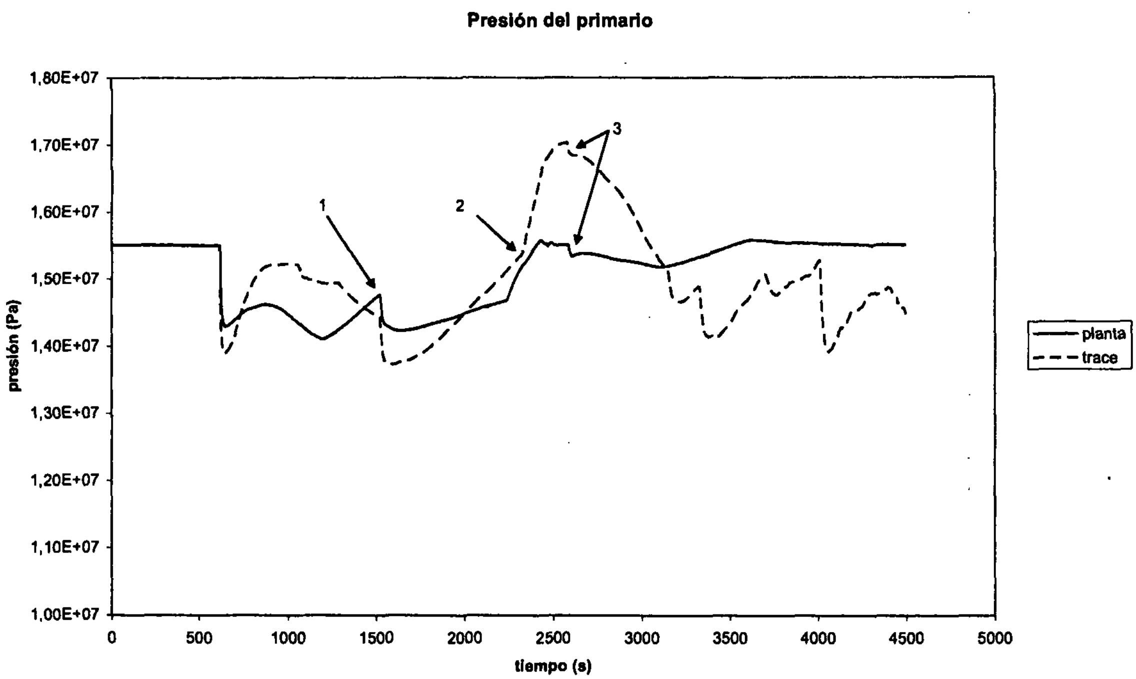


Figure 13. Primary pressure

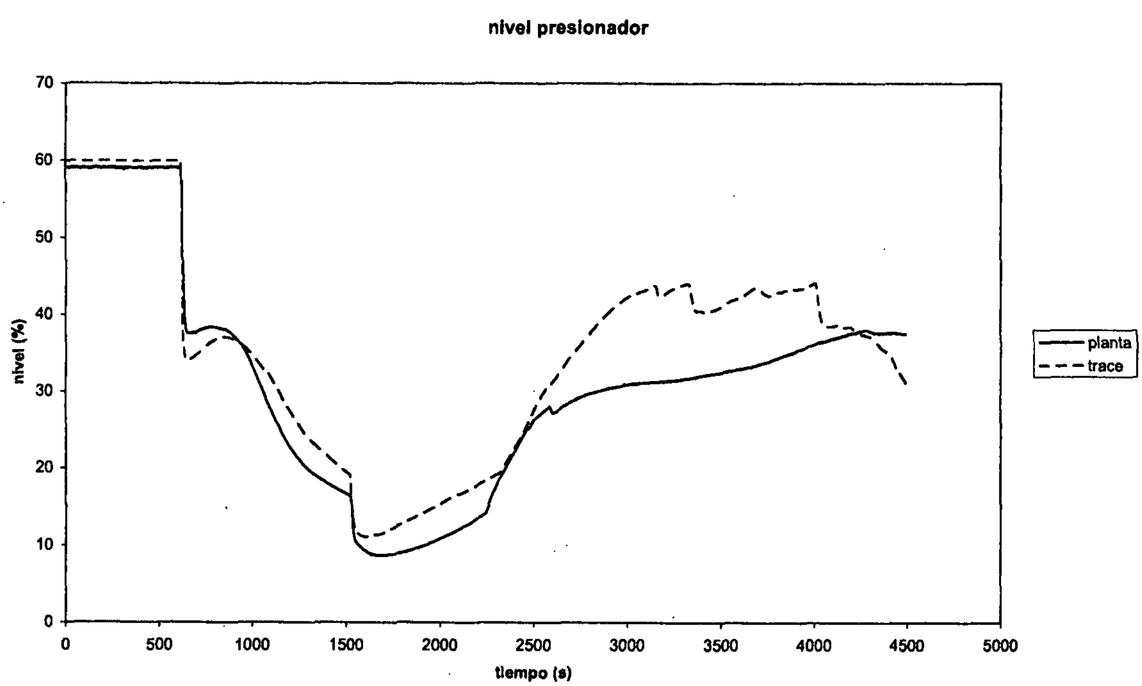


Figure 14. Pressurizer level

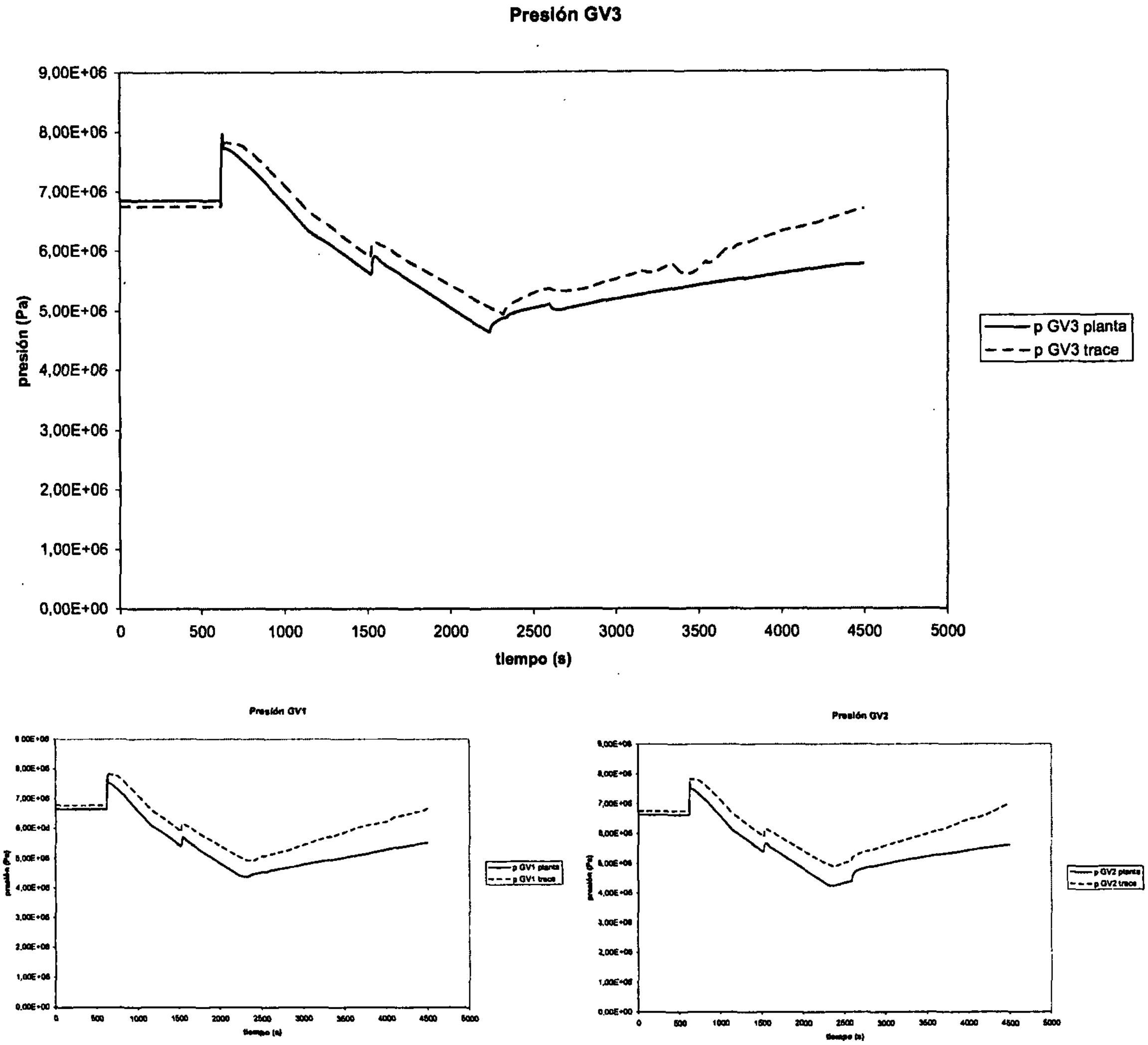


Figure 15. Secondary pressure (SG)

As can be seen in
Figure , the SG level (measured in narrow range) shows a sharp decrease at the beginning of the transient due to the reactor trip, the increase of secondary pressure, and the consequent void collapse.

In the long term the AFW systems allows the SG levels recovery.

The evolution of the SG level obtained with the TRACE model is essentially the same as that recorded at the plant. This similarity is especially remarkable when one considers that the measurement range of this level is relatively small.

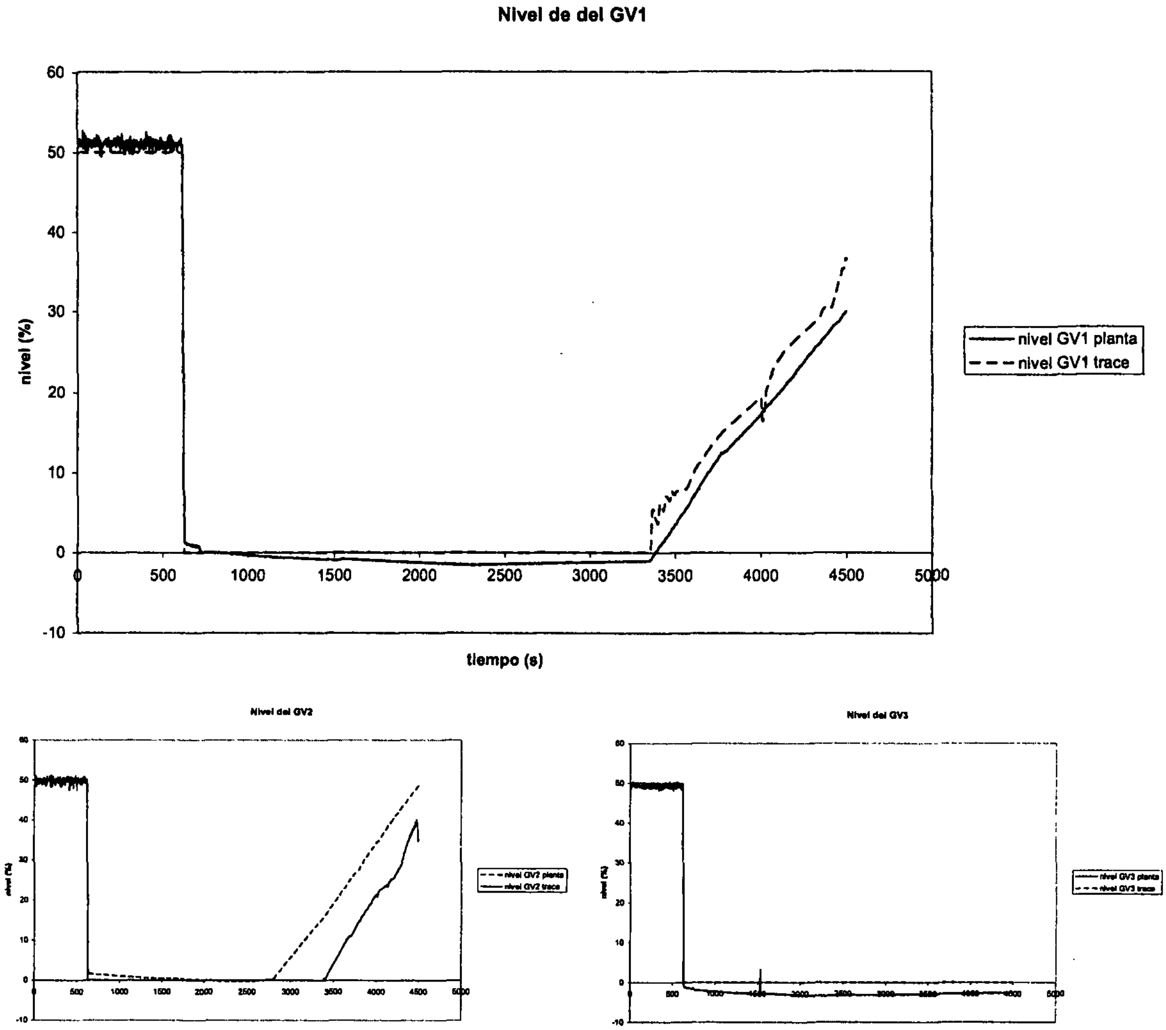


Figure 16. Steam generators level

Figures 16, 17 and 18 show the azimuthal mass flows at different downcomer axial levels. In these figures, "sector n" mass flow is the mass flow from "sector n" to "sector n+1", with the direction "n" to "n+1" positively defined (see Figure 19).

When the situation is symmetrical, either at full power (period 0-600 seconds) or when the three RCPs are stopped (period 600-1500 seconds), the azimuthal flows are virtually null.

Between 1500 and 2600 seconds, RCP-3 is running while the other two are stopped. There is inverted flow in loops 1 and 2 and therefore the nozzle 3 drives cold water through the junction from 3 to 1 (named "sector 3" in the figures) with a positive flow and through the junction from 3 to 2 (named "sector 2") with a negative flow. The flow at the junction from 1 to 2 (named "sector 1") is less visible and near zero.

caudales azimutales del downcomer (nivel axial 5)

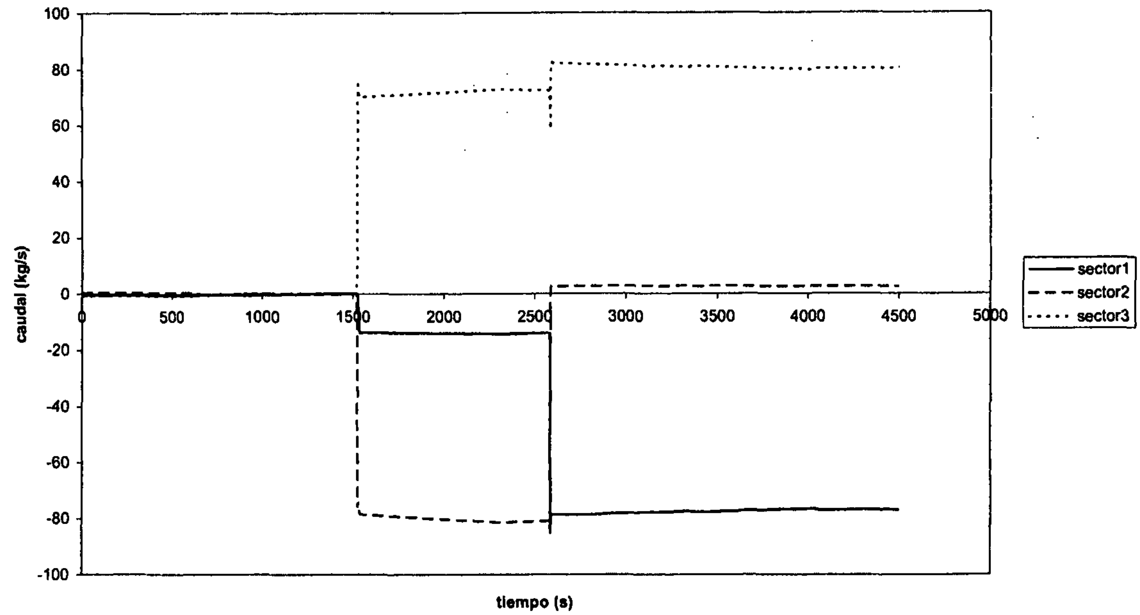


Figure 17. Azimuthal downcomer flows in axial level 5

caudales azimutales del downcomer (nivel axial 7)

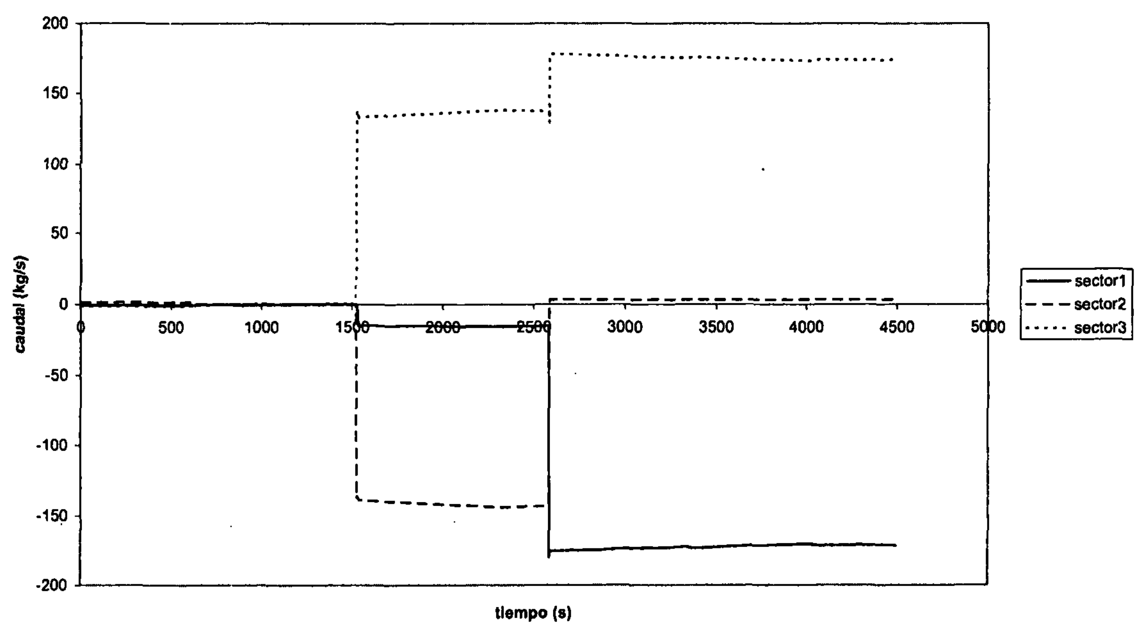


Figure 18. Azimuthal downcomer flows in axial level 7

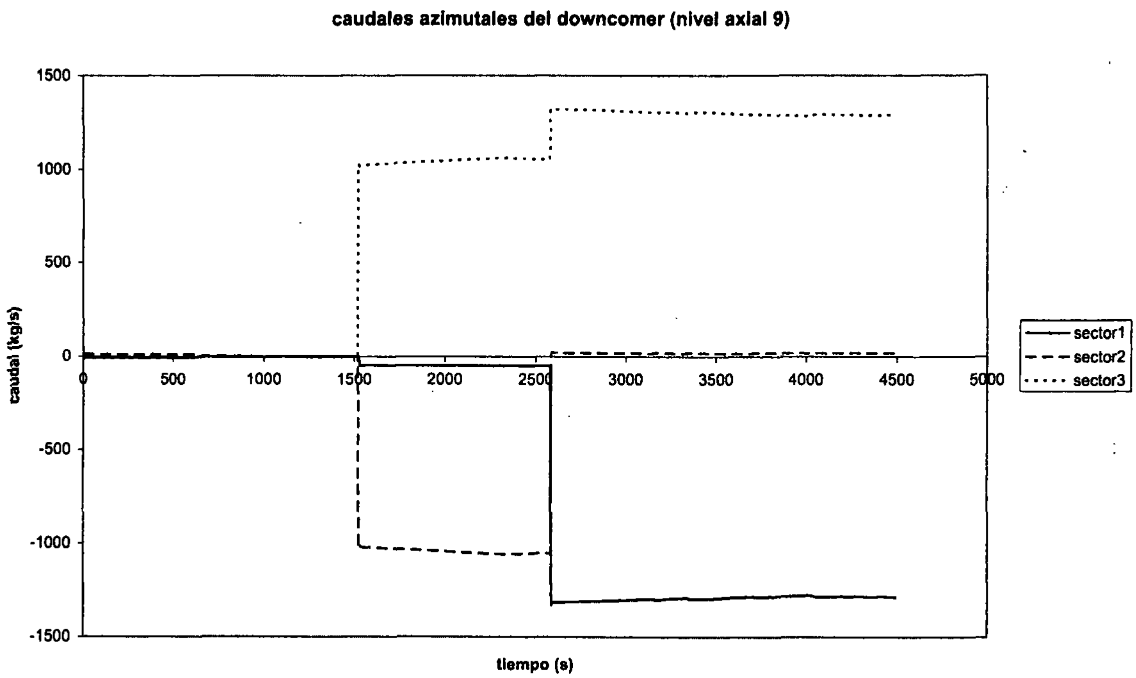


Figure 19. Azimuthal downcomer flows in axial level 9

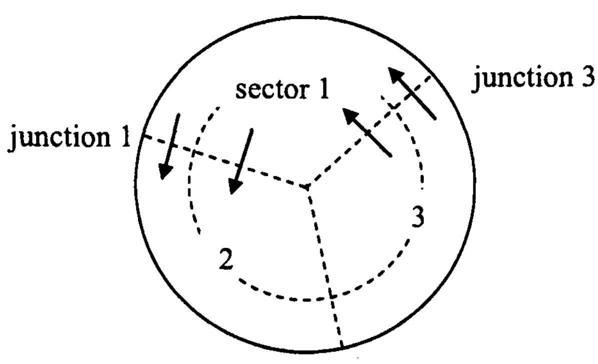


Figure 20. Sketch of the nomenclature followed in the VESSEL component (arrows represent positive directions)

Comparing the different axial levels, it can be verified that, in all cases, the azimuthal flows are higher when they are closer to nozzles.

This follow-up to the hydrodynamic behaviour of the downcomer in symmetry and asymmetry condition may appear irrelevant but it connects with the interest of accident scenarios in which the effect of core bypass should be properly assessed.

7 CONCLUSIONS

This report reflects how the TRACE model of CNV II has been developed, how the steady state has been checked and how the transient has been calculated and its results compared with data recorded in the plant.

The model includes the main systems of Vandellòs II NPP. The TRACE version used is 5.0RC3.

Most of the information of the plant comes from an existing RELAP5 model. The only brand new area of development is the 3D vessel. A direct translation to TRACE has been tried using the plug-in "R5 to TRACE" of the SNAP program but some errors have been detected that have been fixed:

- Some junctions appeared twice after translation
- The friction coefficient obtained by translation was zero on all sides of the nodalization volumes of each component.
- The translation of homologous curves of pump components was not done correctly.
- The correspondence between the types of valves in both codes is not direct.

It should be noted that among the most important parameters in the analyzed transient there are the mass flows of primary loops, after the shutdown and subsequent start-up of the pumps. The simulation results are very similar to the data recorded at the plant.

The signs and the magnitudes obtained for azimuthal mass flows in the downcomer of the vessel show significant consistency.

The results that best fit the data recorded at the plant are SG pressures, SG levels and primary average temperature. This shows a certain quality of the developed model. It should be noted that SG level obtained is very similar to that of the plant. Since the narrow range for the level measurement is small, the results are relevant.

The similarity of results obtained basically for average temperature and primary and secondary pressure guarantee the validity of the model.

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10. SUPPLEMENTARY NOTES

A. Calvo, NRC CAMP Program Manager

11. ABSTRACT *(200 words or less)*

The Thermal-Hydraulics Studies Group (GET) of the Technical University of Catalonia (UPC) has developed a model of Vandellòs II NPP (CNV II) for TRACE using as a previous RELAP5 model of the plant. The model simulates the components of the primary and secondary circuits, along with the main heat structures (nuclear fuel, SG tubes and pressurizer heaters). Passive heat structures and neutron kinetics are not simulated in the model. The model includes some of the plant control systems.
The main predicted steady state nominal parameters successfully match plant and design values. A real loss of off-site power with the subsequent Reactor Coolant Pumps start-up sequence (occurred in August 24th of 1993) has been simulated. The asymmetric behavior of the flows in the vessel has been useful in qualifying the 3D capabilities of the TRACE VESSEL component.

Mass flows of primary loops are among the most important parameters in the analyzed transient. The simulation results are very similar to the data recorded at the plant. The similarity of calculated and measured values is a guarantee of the validity of the model.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	
Thermal-Hydraulics Studies Group (GET) Technical University of Catalonia (UPC) Vandellòs II NPP (CNV II) Technical University of Catalonia Consejo de Seguridad Nuclear (CSN) Code Assessment and Management Program (CAMP) Reactor Coolant Pumps start-up sequence UNESA (the association of the Spanish utilities) U.S. Nuclear Regulatory Commission TRACE VESSEL component	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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