NUREG/IA-0226

International Agreement Report

Assessment of the Turbine Trip Transient in Santa María de Garoña Nuclear Power Plant with TRACE version 4.16

Prepared by:
P. Corcuera, E. Rodriguez, and G. Villescas

University of Cantabria
ETSIIT
Av. De los Castros s/n
39005 Santander, SPAIN

A. Calvo, NRC Project Manager

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

March 2010

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402-0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161-0002
   www.ntis.gov
   1-800-553-6847 or, locally, 703-605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: Office of the Chief Information Officer,
Reproduction and Distribution
Services Section
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301-415-2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
www.ansi.org
212-642-4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG-XXXX) or agency contractors (NUREG/CR-XXXX), (2) proceedings of conferences (NUREG/CP-XXXX), (3) reports resulting from international agreements (NUREG/IA-XXXX), (4) brochures (NUREG/BR-XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG-0750).

NUREG/IA-0226

Assessment of the Turbine Trip Transient in Santa María de Garoña Nuclear Power Plant with TRACE version 4.16

Manuscript Completed: June 2008
Date Published: April 2010

Prepared by:
P. Corcuera, E. Rodríguez, G. Villescas

University of Cantabria
ETSIIT
Av. De los Castros s/n
39005 Santander, SPAIN

Antony Calvo, NRC Project Manager

Prepared for:
Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ABSTRACT

This report presents the results of the assessment of the TRAC/RELAP Advanced Computational Engine (TRACE) code, version 4.16, using the model of the Santa María de Garoña Nuclear Power Plant to simulate the transient that occurred on June 23, 1992, when a false high-level signal in the moisture separator caused an automatic trip of the main turbine.

The steady state was adjusted by connecting submodels of portions of the system previously tuned to the desired conditions.

The results show a good agreement with data for all the compared variables. The results of the calculations were in reasonable agreement with plant measurements. The simulations were run on a Pentium IV 3.4 megahertz under Windows XP with 32 bits executable.

This report was prepared by the Computer Science and Intelligent Systems Group belonging to the Applied Mathematics and Computer Science Department of the University of Cantabria, which collaborates in the area of simulation with the company Nuclenor S.A., owner of the nuclear power station Santa María de Garoña. The Asociación Espanola de la Industria Eléctrica (Electric Industry Association of Spain) and Nuclenor S.A. sponsored this work.

FOREWORD

This report represents one of the assessment or application calculations submitted to fulfill the bilateral agreement for cooperation in thermal-hydraulic activities between the Consejo de Seguridad Nuclear (CSN) and the U.S. Nuclear Regulatory Commission (NRC) in the form of a Spanish contribution to the NRC's Code Assessment and Management Program (CAMP), the main purpose of which is to validate the TRAC/RELAP Advanced Computational Engine (TRACE) code.

CSN and the Asociación Española de la Industria Eléctrica (Electric Industry Association of Spain), together with some relevant universities, have established a coordinated framework (CAMP-Spain) with two main objectives: to fulfill the formal CAMP requirements and to improve the quality of the technical support groups that provide services to the Spanish utilities, CSN, research centers, and engineering companies.

The AP-28 Project Coordination Committee has reviewed this report, the contribution of one of the Spanish utilities to the above-mentioned CAMP-Spain program, for submission to CSN.

UNESA
December 2009

CONTENTS

Figures

Tables

EXECUTIVE SUMMARY

This report presents the results of the assessment of the TRAC/RELAP Advanced Computational Engine (TRACE) code, version 4.16, using the Santa María de Garoña Nuclear Power Plant (NPP) as a model to simulate the turbine trip transient that occurred there in June 1992.

The Santa María de Garoña NPP is a General Electric boiling-water reactor-3 plant, with a nominal core thermal power of 1,381 megawatts thermal, in commercial operation since 1971 and owned and operated by Nuclenor S.A.

The objective of this assessment is to generate a Garoña model for TRACE and compare data from the model with plant-recorded data during the above-mentioned transient. The model was developed with the aid of the Symbolic Nuclear Analysis Package (SNAP) code, version 0.24.1. Principal characteristics of the model include a four-ring, 11-axial-level vessel, two recirculation loops, and one representative steamline. The control systems and trips were also modeled.

The SNAP program was used to adjust a reference steady-state condition by connecting submodels of portions of the system previously tuned to the desired conditions. The final tuning of the input model was done by adjusting the flow area fraction of components in the lower plenum of the vessel and the loss coefficients of components near the input of the core.

As a result of this assessment, a model of the Santa María de Garoña NPP has been developed for TRACE that reproduces, in an acceptable manner, the operational transient behavior of the plant. Improvement of the recirculation loop model is an area identified for further work. Another area of potential improvement is the tuning of the control systems, such as feedwater, pressure, and recirculation.

ABBREVIATIONS

BWR	boiling-water reactor
CAMP	Code Assessment and Management Program
cm	centimeter(s)
CPU	central processing unit
CSIS-UC	Computer Science and Intelligent Systems Group—University of Cantabria
CSN	Consejo de Seguridad Nuclear (Spanish nuclear regulatory commission)
EPR	electric pressure regulator
FW	feedwater
GE	General Electric
kg	kilogram(s)
l/s	liter per second
m	meter(s)
mm	millimeter(s)
MPa	megapascal
kg/cm^2	kilogram per square centimeter
ºC	degrees Celsius
ºK	degrees Kelvin
MSIV	main steam isolation valve
MW	megawatt(s)
MWe	megawatt(s) electric
MWt	megawatt(s) thermal
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
Rel	pressure relative to the ambient pressure (gauge)
RV	relief valve
s	second(s)
SNAP	Symbolic Nuclear Analysis Package
SRV	safety/relief valve
SV	safety valve
T/h	tonne per hour
TRACE	TRAC/RELAP Advanced Computational Engine

1. INTRODUCTION

The Computer Science and Intelligent Systems Group of the University of Cantabria (CSIS-UC) worked with TRAC-BF1, in 1998–2000, to analyze some transients for the Santa María de Garoña Nuclear Power Plant (NPP) and to develop one graphical postprocessing tool for TRAC (Ref. 1) as a result of the participation in the CAMP project. In 2007, this group began working with the TRAC/RELAP Advanced Computational Engine (TRACE) code. The aim was to obtain the model of the Santa María de Garoña NPP and use it to analyze a turbine trip transient that occurred in 1992.

Nuclenor S.A. owns and operates the Santa María de Garoña NPP. This facility has a General Electric (GE) boiling-water reactor (BWR)-3, rated at 1,381 megawatts thermal (MWt) and connected to the grid in 1971. CSIS-UC has had a close collaboration with Nuclenor in the area of simulation with thermal-hydraulic codes.

The purpose of this report is to document the generation of the model code for TRACE and its use to simulate the 1992 turbine trip at the Santa María de Garoña NPP.

This report consists of the following sections:

- a brief description of the Santa Maria de Garoña plant, Section 2
- a description of the plant turbine trip transient, Section 3
- a description of the model developed for TRACE, Section 4
- a description of the steady-state calculations, Section 5
- an analysis and comparison of transient results with plant data, Section 6
- an analysis of run statistics, Section 7

The simulations were run on a Pentium 4 workstation, 3 gigahertz under the Windows XP Professional 64-bit operating system.

2. PLANT DESCRIPTION

The Santa María de Garoña NPP is owned by Nuclenor S.A., which is also responsible for its operation.

The plant is a BWR-3, with a Mark I primary containment designed by GE. The plant is rated at 1,381 MWt. It is located in the province of Burgos, Spain, and was connected to the grid in 1971.

The nuclear boiler assembly consists of the reactor pressure vessel and internal reactor components, such as the core structure, steam dryer assembly, fuel supports, and control guide tubes. The reactor core is made up of 400 fuel assemblies and 97 control rod blades, as well as the neutronics instrumentation. At present, it is loaded with GE14 (10x10) elements. However the transient that will be compared with a TRACE analysis took place in 1992 (Cycle 17) and the reactor core was loaded at that time with the following elements:

- GE7B (8x8) elements
- GE8B (8x8) elements
- GE10 (8x8) elements

Each control rod blade consists of a sheathed cruciform array of vertical absorber rods made of boron carbide. These rods penetrate the core from the bottom.

The recirculation system provides the hydraulic energy required to force coolant through the reactor core, providing it with forced convection cooling. The recirculation system consists essentially of two recirculation piping loops located outside the reactor pressure vessel, in the drywell area, and includes 20 jet pumps located inside the reactor pressure vessel, between the reactor pressure vessel wall and the core shroud. The flow from the recirculation pump is the driving force for the jet pump. The primary function of the reactor recirculation system is to permit reactor power level changes without changing the position of the reactor control rods.

Two centrifugal pumps, each driven by an electric motor, supply feedwater. The pumps are discharged through spargers located in a ring in the annulus between the core shroud and vessel wall. The primary purpose of the feedwater system is to maintain the water level in the reactor vessel within a programmed range during all modes of plant operation. In normal operation, the level of water in the reactor is controlled by a feedwater controller that receives inputs from the reactor vessel water level, steam-mass flow rate, and feedwater-mass flow rate transmitters. In turn, the feedwater control system generates signals that regulate the opening of the flow control valves.

The nuclear instrumentation to obtain the necessary information from the local thermal neutron flux of the core during full-power reactor operation consists of 22 sets of local power range monitors, located radially in the core. Four average power range monitors average signals from the 22 local power range monitors to collect information on average power generated in the core.

The main steam system consists of four lines that provide steam to the turbine from the reactor vessel. Steamlines run downward, parallel to the vertical axis of the vessel, until they reach the elevation at which they emerge from the containment. Two air-operated isolation valves are installed on each steamline, one inboard and one outboard of the primary containment penetration. A flow-restricting nozzle is included in each steamline as an additional engineered safeguard to protect against a rapid uncovering of the core in case of a main steamline break.

Three relief valves (RVs) and three safety/relief valves (SRVs) discharging into the suppression pool, and seven safety valves (SVs) discharging into the drywell, are installed on the steamlines. The main function of these valves is to protect against overpressure of the reactor primary system and the depressurization to allow actuation of low-pressure emergency systems in case of a loss-of-coolant accident.

The primary containment in the Santa María de Garoña NPP is a Mark I. The drywell component is a steel “light-bulb shaped” vessel with a spherical lower portion and an upper cylindrical portion with the minimum volume necessary to accommodate the reactor vessel and ancillary equipment and to allow necessary maintenance and inspection. A bolted head closes the top of the cylinder. Reinforced concrete encloses this vessel, providing additional shielding and resistance.

The pressure suppression chamber, or the wetwell, is a toroidal steel vessel that surrounds the lower portion of the drywell. Eight circular vent pipes interconnect the wetwell and the drywell. The reactor building encloses the containment and also encompasses the refueling area, fuel storage facilities, and other auxiliary systems.

The Santa María de Garoña NPP has the following safeguard systems:

- isolation condenser system
- core spray system
- automatic depressurization system
- low-pressure cooling injection system
- high-pressure cooling injection system

Figure 1 is a functional diagram of the Santa Maria de Garoña NPP obtained from the safety parameter display system.

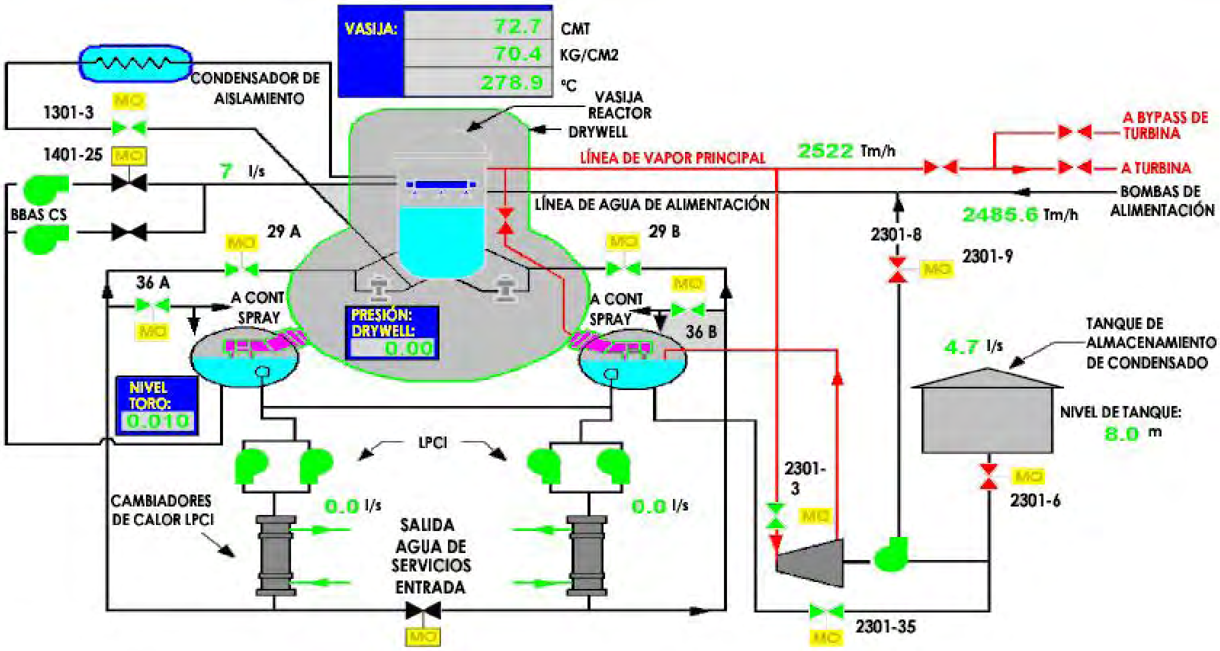

Figure 1 Santa María de Garoña NPP functional diagram

3. TRANSIENT DESCRIPTION AND PLANT RESPONSE

The analyzed transient corresponds to a turbine trip that occurred on June 23, 1992, at 10:43 pm (22h.43 min). The cause of the trip was the loss of tension in the drain valves to moisture condenser separators that created a false high water level signal in the moisture separator M1-3A (Ref. 9), when the power plant was operating at 99.5 percent of nominal power (1,375 MWt). This signal led to the turbine trip of the plant and the automatic reactor scram.

The sudden increase of reactor pressure, because of the closure of the stop valves, caused the opening of three RVs and two SRVs (one valve didn't open). Also, the core void fraction content was reduced, and the level dropped in a few seconds, reaching the low-level setting (+18 centimeters (cm)). Then, the pressure control system opened the bypass valve to the main condenser, which allowed it to dominate the pressure. The bypass valve was closed in less than 30 seconds from the start of the scram. The feedwater control system controlled the reactor level by injecting greater flow initially and then the high-level setting (+122 cm) tripped the pumps.

Table 1 summarizes the initial conditions of unity, and Table 2 describes the transient main events chronologically.

The transient's temporal evolution plots of the most important variables obtained from the computer process, some of which have been filtered to eliminate the noise signal plant, are shown below:

- reactor power (Figure 2)
- steam dome pressure (Figure 3)
- reactor level (Figure 4)
- steam flow (Figure 5)
- feedwater flow rate A (Figure 6)
- feedwater flow rate B (Figure 7)
- recirculation flow rate A (Figure 8)
- recirculation flow rate B (Figure 9)
- core flow (Figure 10)
- bypass valve position (Figure 11)

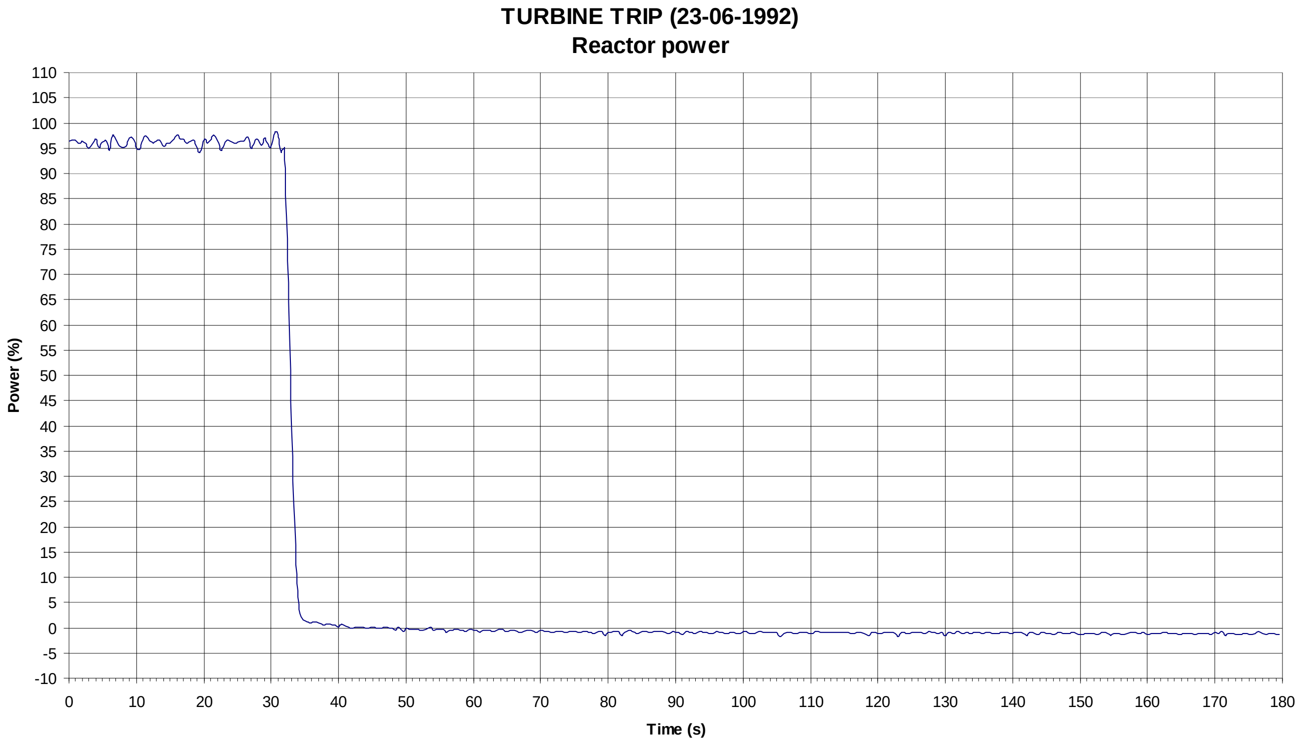

Figure 2 Reactor power (plant data)

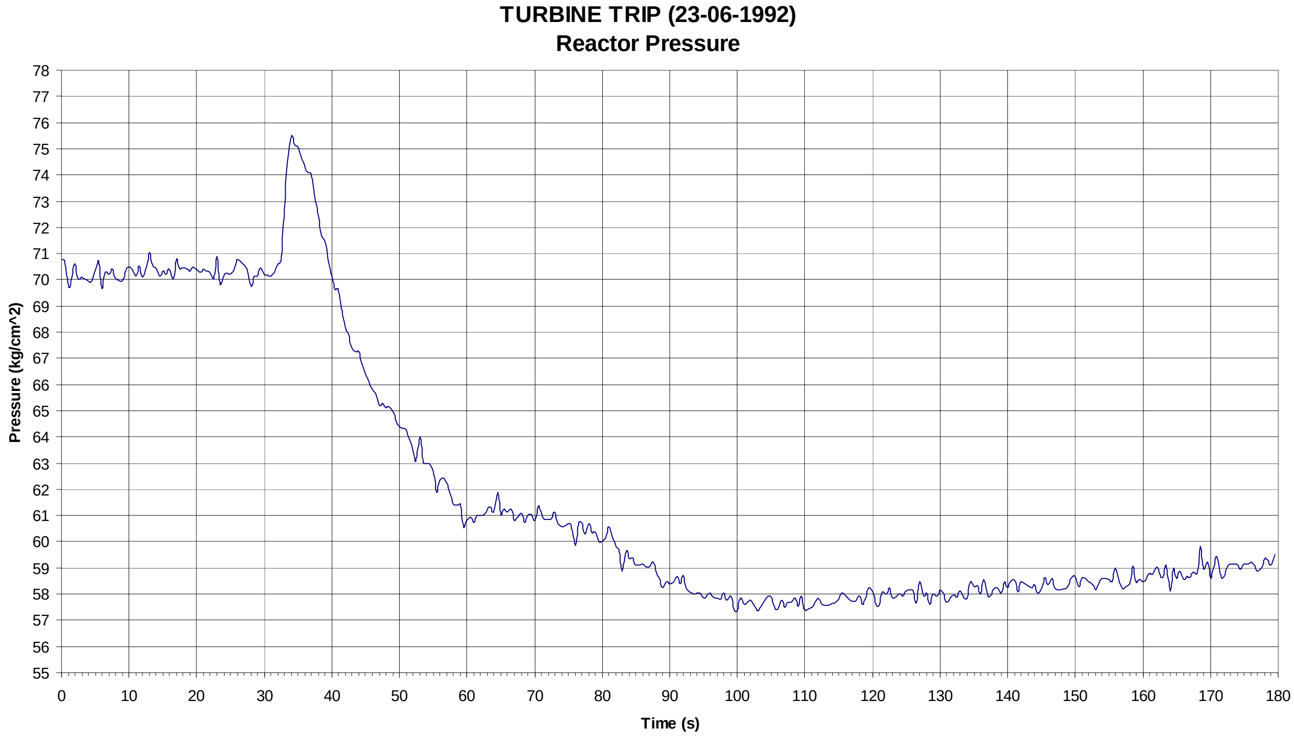

Figure 3 Steam dome pressure (plant data)

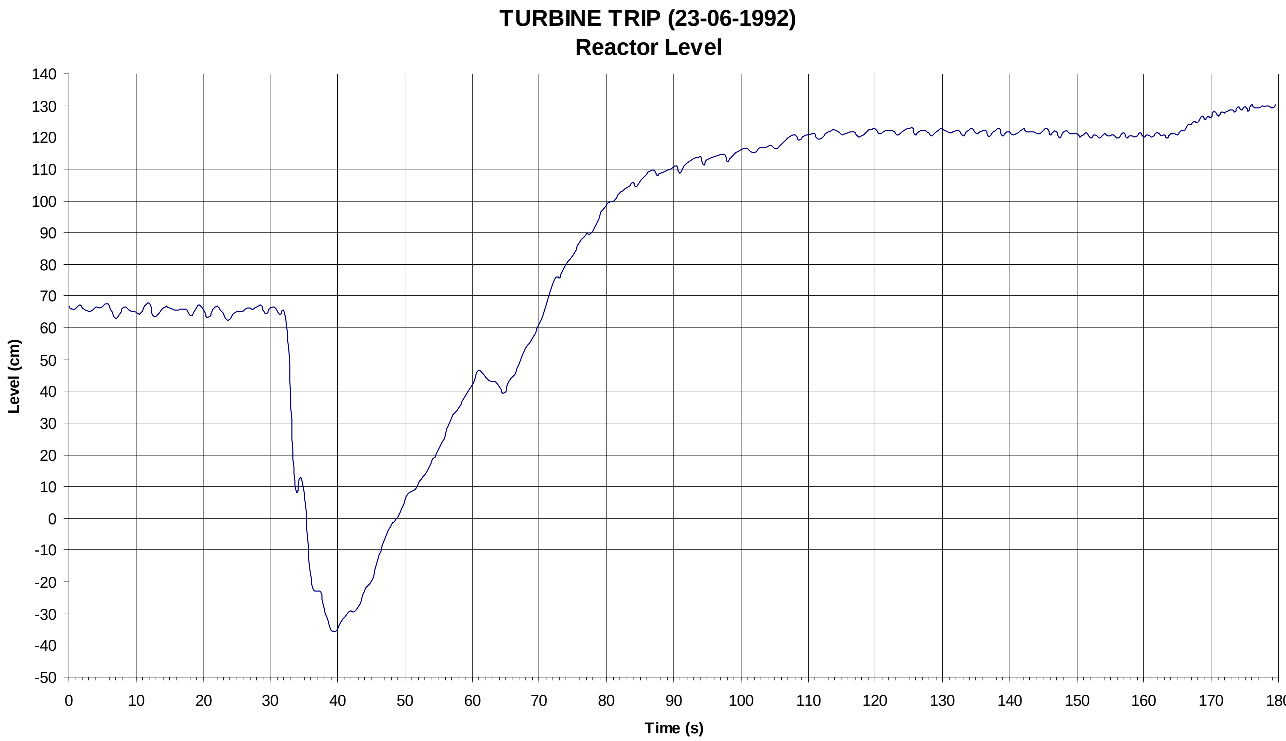

Figure 4 Reactor level (plant data)

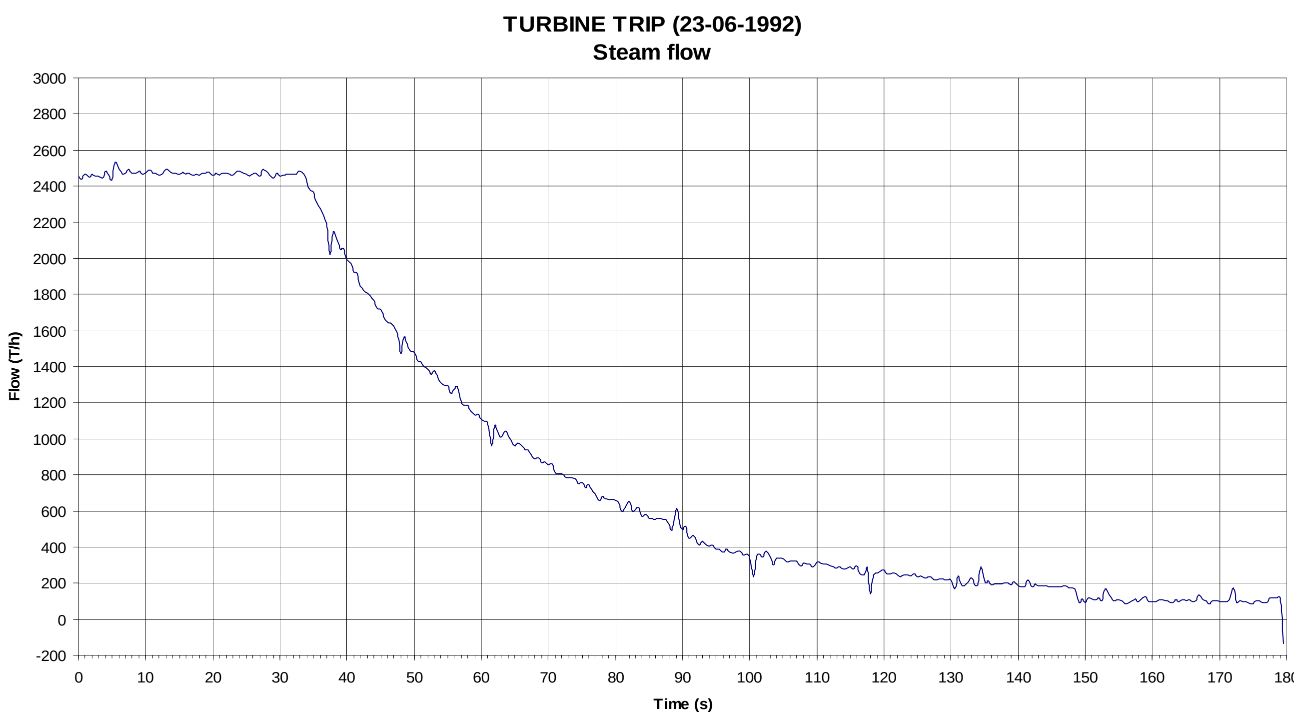

Figure 5 Steam flow (plant data)

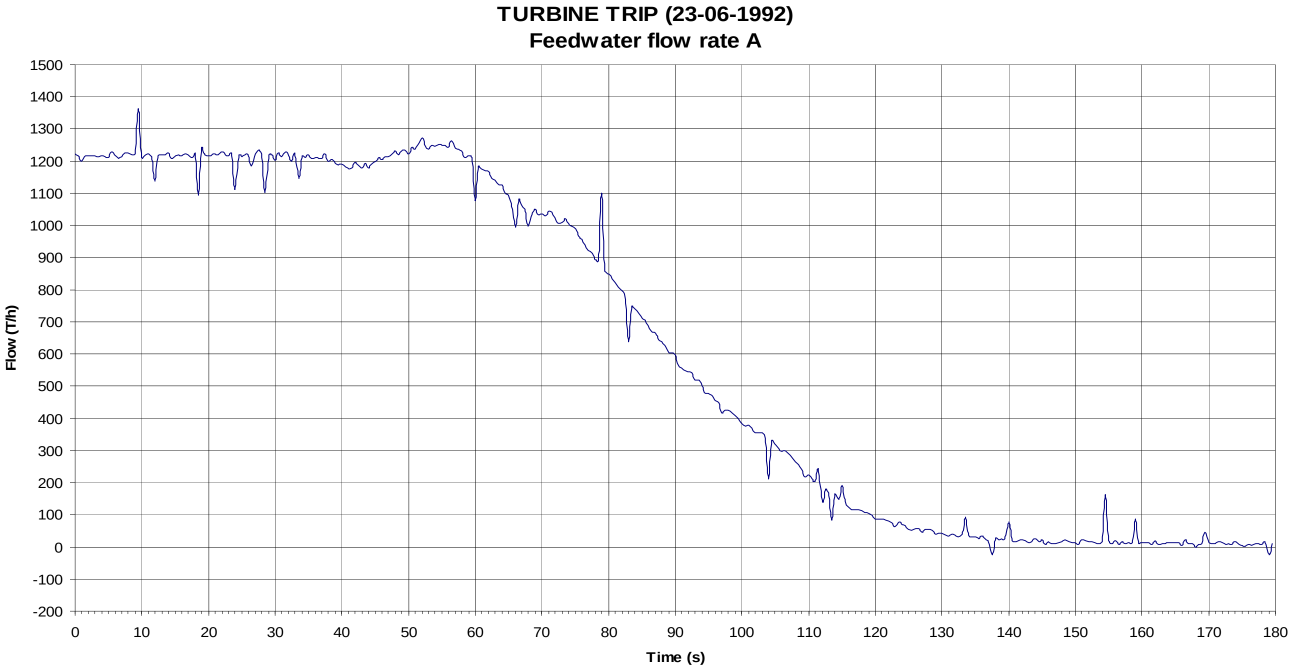

Figure 6 Feedwater flow rate A (plant data)

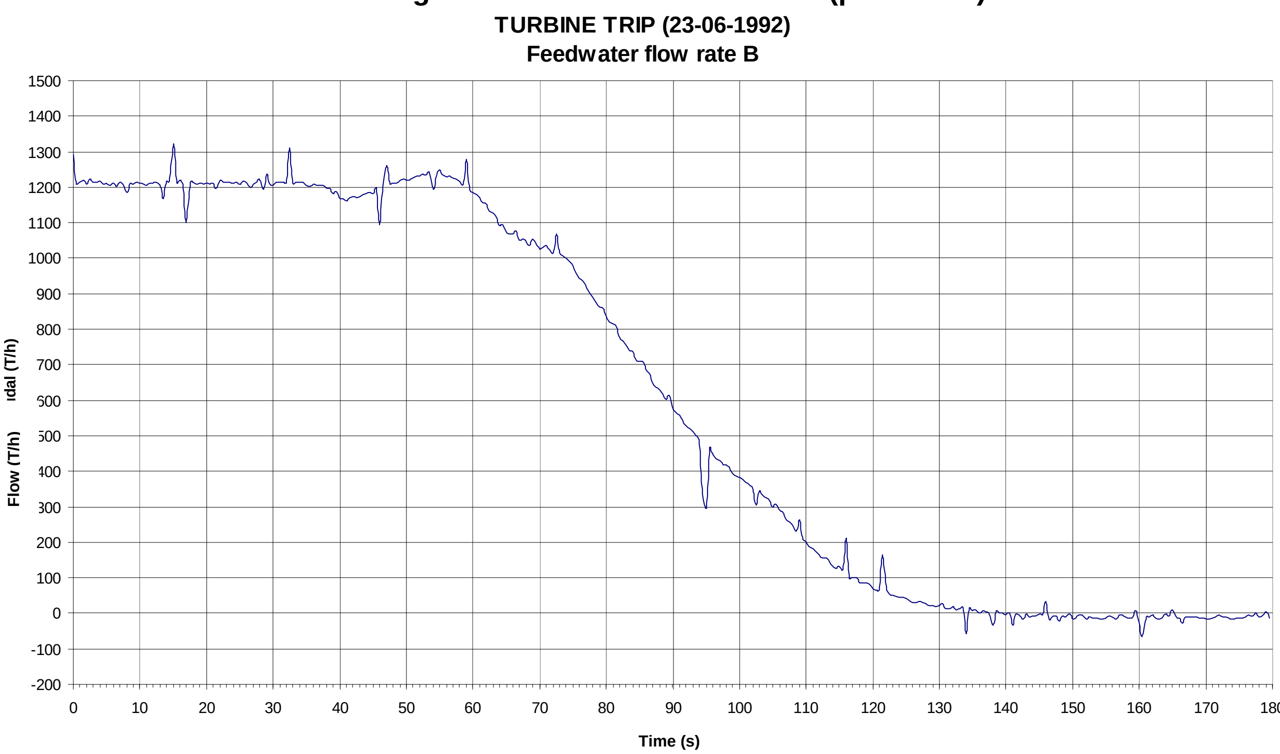

Figure 7 Feedwater flow rate B (plant data)

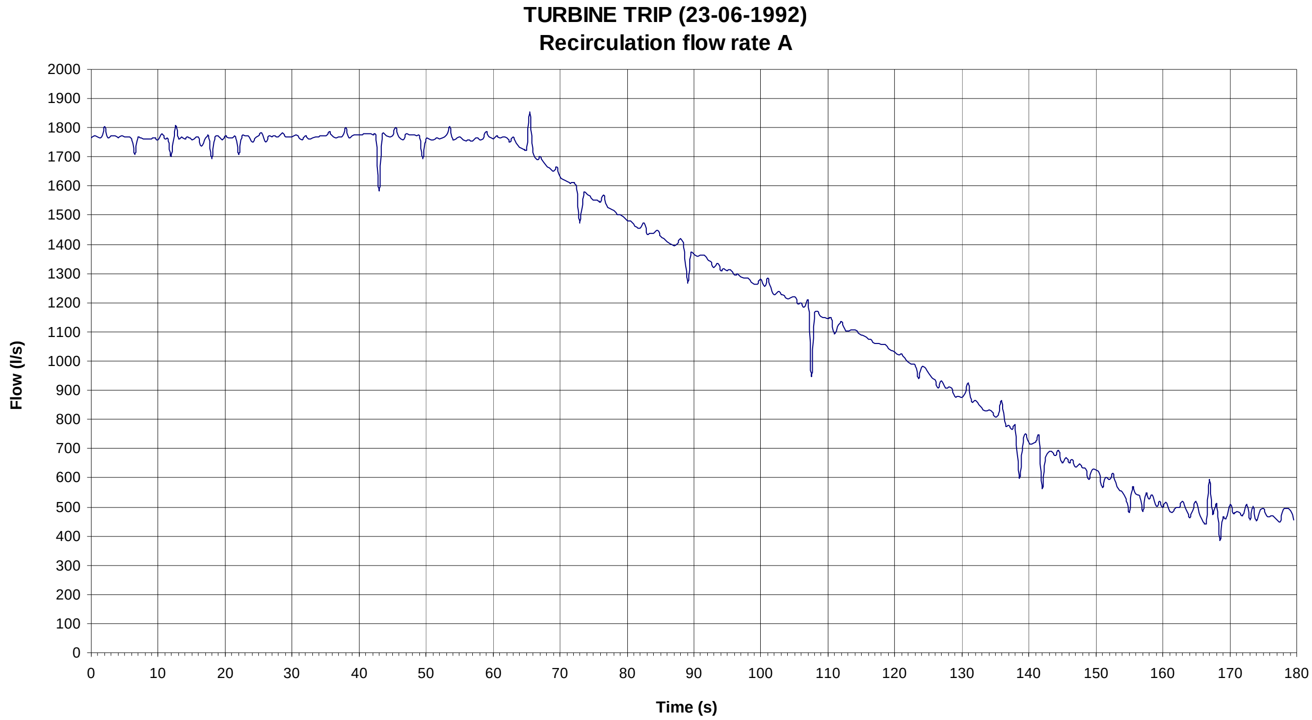

Figure 8 Recirculation flow rate A (plant data)

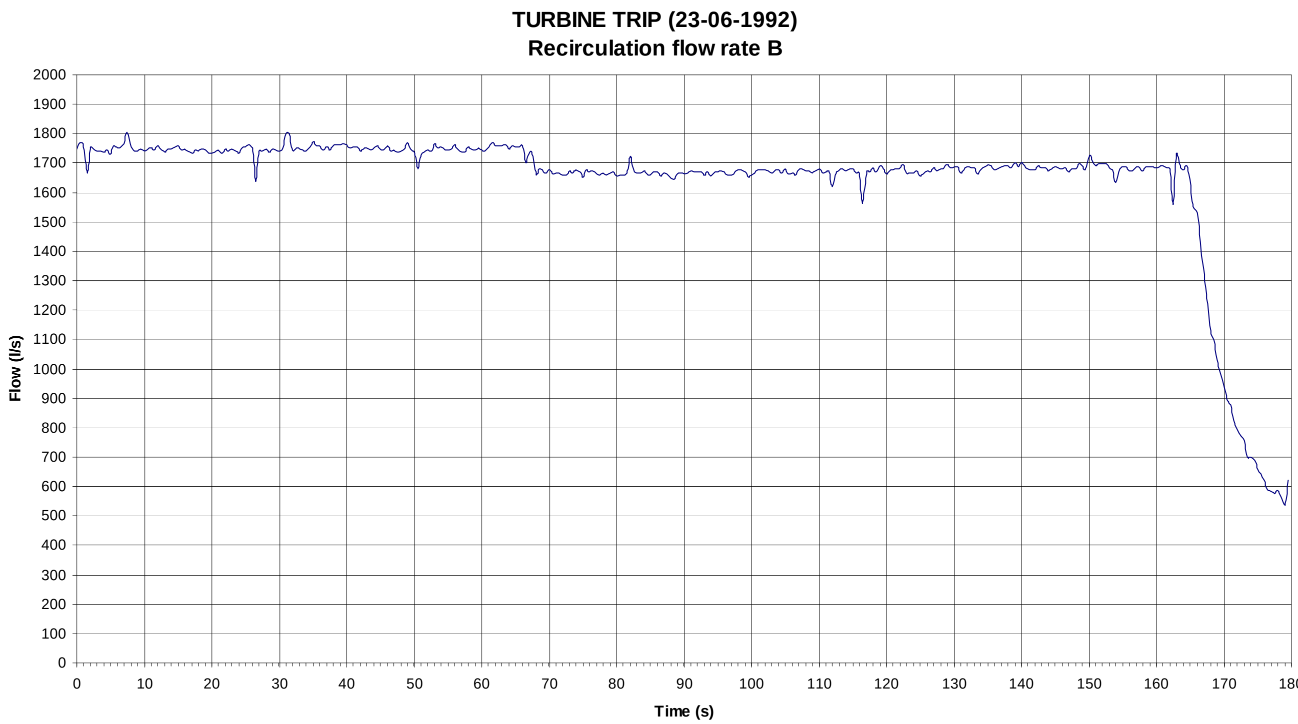

Figure 9 Recirculation flow rate B (plant data)

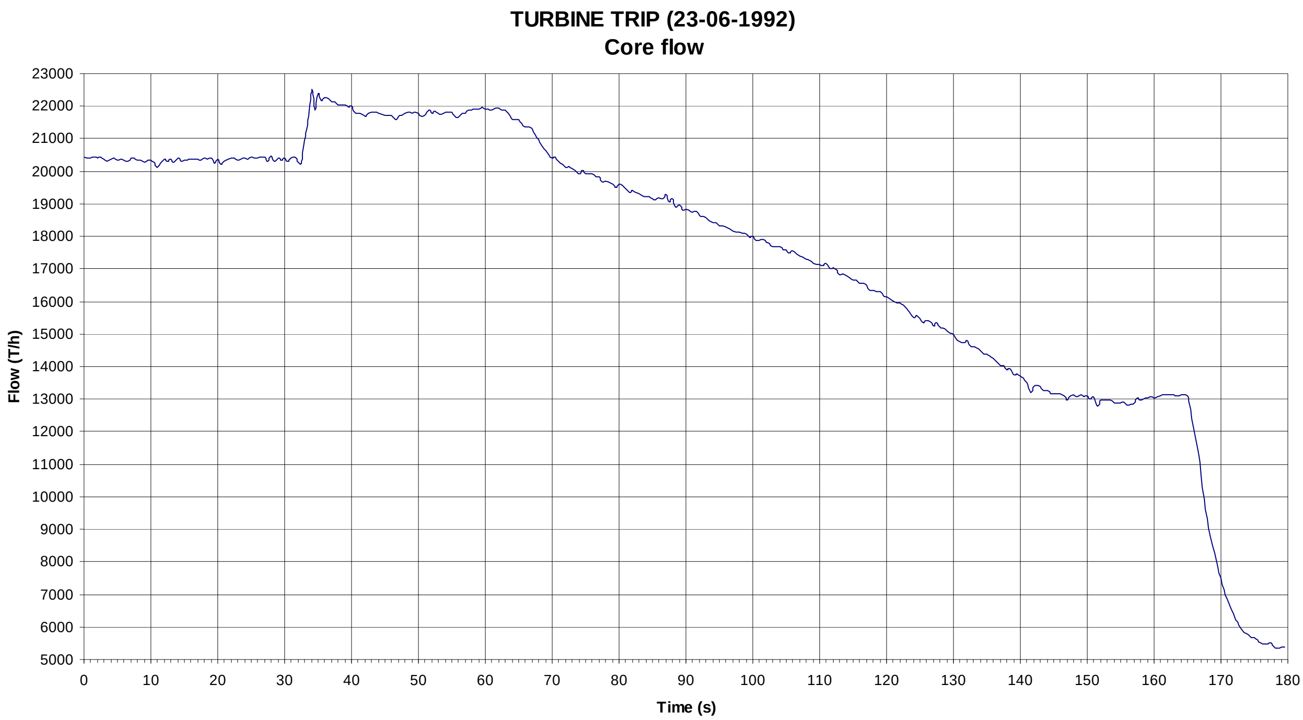

Figure 10 Core flow (plant data)

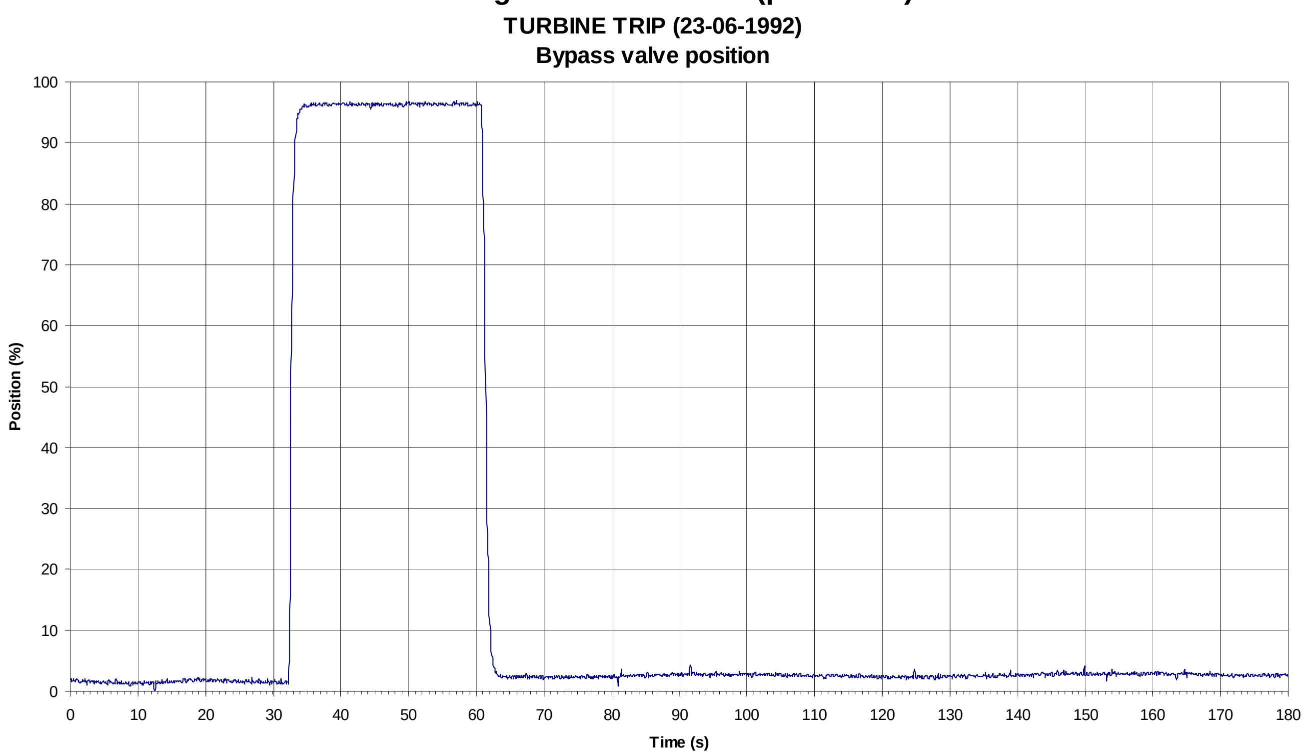

Figure 11 Bypass valve position (plant data)

Table 1 Initial Plant State

Parameter	Value
Core thermal power (MWt)	1375
Generator output (MWe)	445
Switch mode position	RUN
Reactor dome pressure (kg/cm^2)	70.3
Reactor level RPV (mm)	660
Core flow rate (T/h)	20.24×10^3
Feedwater pumps:	
Running	A and B
Selected	C
Selected to start by low level	B
High-level trip selector	Normal
Condensate pumps:	
Running	B and C
Selected	C
Recirculation pumps:	
Running	A and B
Scoop tube blocking by difference signal	Normal
Control settings:	
Reactor level	AUTO/A/3 elements
Recirculation pumps	Manual
Turbine/Generator:	
Pressure regulator	EPR
Pressure setting (kg/cm^2)	64.7 kg/cm^2
Load limiter	90
Amplidine	Yes
Local electric distribution	NORMAL

Table 2 Time Sequence for Events in the Turbine Trip Transient

Hour	Event	State
22:43:09	Level separator 3A	High
22:43:28	High level separator	Trip
	Stop turbine valves close	Trip
	General scram A–B	Trip
	Bypass valve	Open
	Target Rock Valve C	Open
22:43:29	Isolation Group 2–6	Trip
22:43:30	RV C	Open
	RV B	Open
	Target Rock Valve A	Open
	RV A	Open
	Target Rock Valve A	Closed
22:43:31	RV A	Closed
	Target Rock Valve C	Closed
22:43:33	RV B	Closed
22:43:34	Manual scram A–B	Trip
22:43:37	Inverse power relay	Trip
	Generator block relay 86/G	Trip
	RV C	Closed
22:43:46	Level separator 3A	Normal
22:43:57	Bypass valve	Closed
22:45:06	General scram A–B	Normal
22:45:07	Generator block 86/G	Normal
22:45:58	Isolation group 2–6	Normal

4. CODE INPUT MODEL DESCRIPTION

The development of the TRACE input deck file (Ref. 2, 3, 4, 5, 6) for this analysis was based on data taken from previous calculations for TRAC-BF1 code (Ref. 7). Figure 12 shows the main features of the Santa María de Garoña NPP model, which includes the reactor vessel and core, recirculation loops, and steamlines from the vessel to the turbine valves.

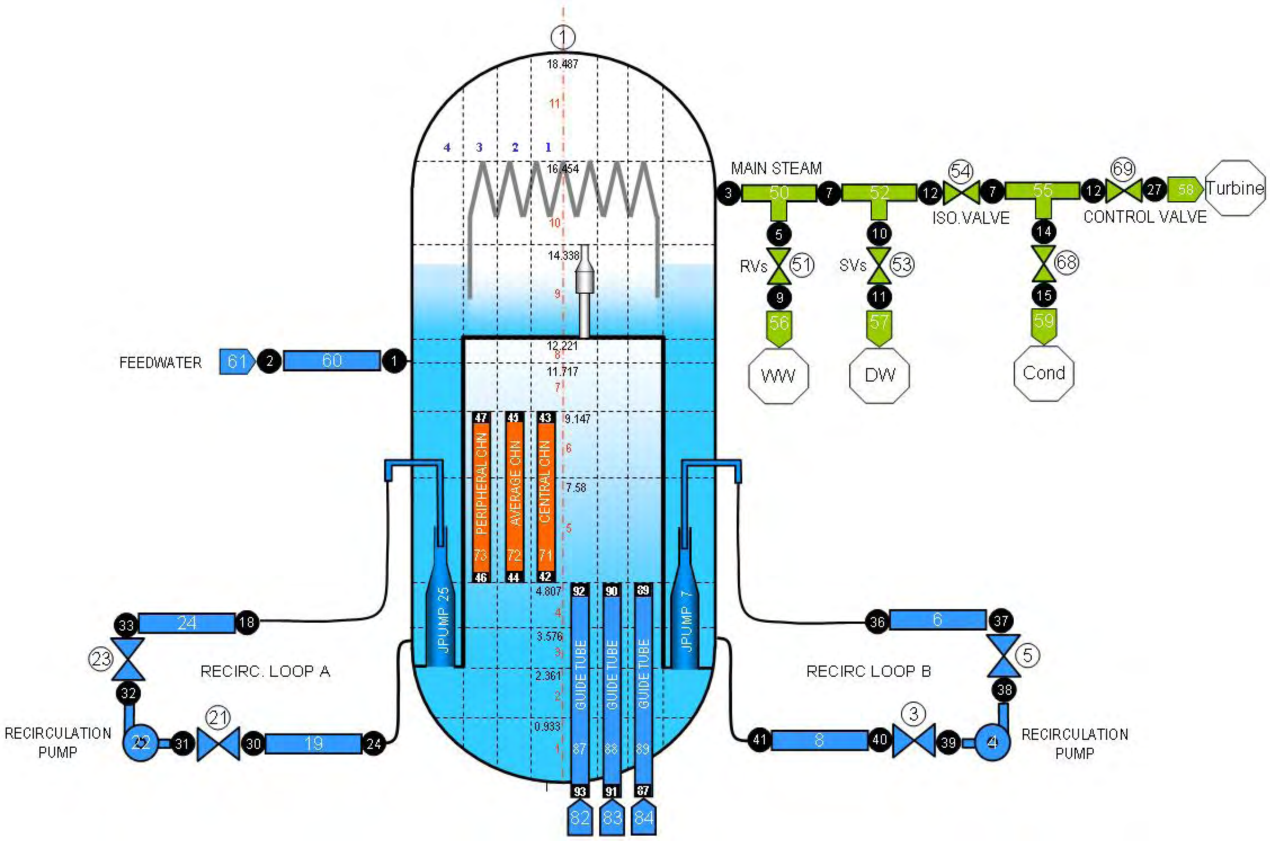

Figure 12 Santa María de Garoña NPP nodalization

Table 3 summarizes the components used. The following sections describe the main components of the model.

4.1 Vessel

The VESSEL component, which models the reactor vessel, has been divided into 11 axial levels and four radial rings. Figure 13 shows the main dimensions of the vessel geometry.

Table 3 Components of Garoña NPP Input Deck

Component	Id.	Description
VESSEL	1	Vessel
VALVE	3	Recirculation suction valve B
PUMP	4	Recirculation pump B
VALVE	5	Recirculation discharge valve B
PIPE	6	Recirculation discharge line B
JETP	7	Jet pump loop B
PIPE	8	Recirculation suction line B
PIPE	19	Recirculation suction line A
VALVE	21	Recirculation suction valve A
PUMP	22	Recirculation pump A
VALVE	23	Recirculation discharge valve A
PIPE	24	Recirculation discharge line A
JETP	25	Jet pump loop A
TEE	50	Main steamline from the vessel lines
VALVE	51	RVs
TEE	52	SVs branch
VALVE	53	SVs
VALVE	54	MSIV
TEE	55	Main steamline to the turbine
BREAK	56	Wetwell boundary condition
BREAK	57	Drywell boundary condition
BREAK	58	Turbine
BREAK	59	Bypass boundary condition
PIPE	60	Feedwater line
FILL	61	Feedwater fill
VALVE	68	Bypass valve
VALVE	69	Control valve
CHAN	71	Hot channels
CHAN	72	Average channels
CHAN	73	Peripheral channels
FILL	82	Guide tube entrance 1
FILL	83	Guide tube entrance 2
FILL	84	Guide tube entrance 3
PIPE	87	Guide tube hot channels
PIPE	88	Guide tube average channels
PIPE	89	Guide tube peripheral channels

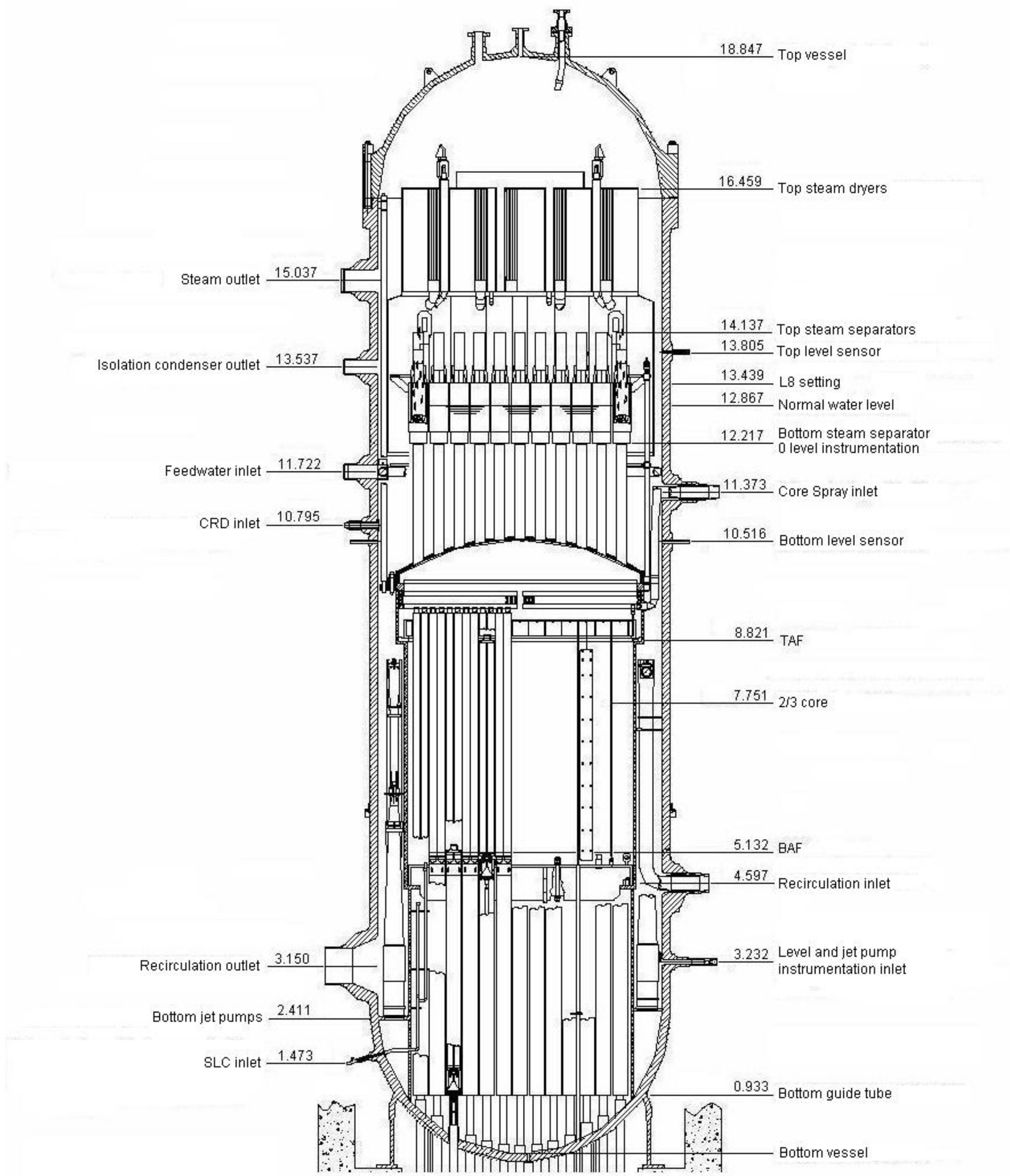

Figure 13 Dimensions of the vessel geometry

Each of the 11 axial levels is associated with a component or significant elevation in the vessel; the height above each level is defined by the following:

1. bottom elevation of the control rod drive housings

2. bottom elevation of jet pumps

3. bottom elevation of suction pipe recirculation loop

4. core support plate

5. upper elevation of throat of the jet pumps

6. upper elevation of upper core grid

7. upper elevation of feedwater sprays

8. bottom elevation of the skirt of steam separators

9. upper elevation of steam separators

10. upper elevation of steam dryers

11. bottom elevation of upper dome, assuming that the dome is cylindrical with a radius equal to the vessel, and the volume of the dome is the same as the cylinder

Figure 14 shows the distribution chart levels in the reactor vessel.

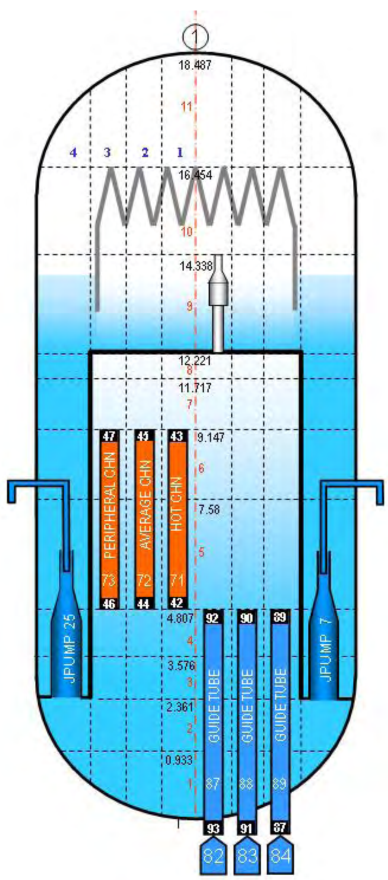

Figure 14 Level distribution in the reactor vessel

Four radial rings, as shown in Figure 14, are used to model the vessel, with three of them, the internals, to simulate the core, and the fourth as the downcomer. It is assumed that the reactor vessel has a cylindrical symmetry, as it uses only one azimuthal segment. The first ring contains 28 fuel channels; the second ring, 288 fuel channels; and the third ring, 84 fuel channels.

The perfect separator option is used for axial level 9. The vessel connections to other components include the feedwater injection, modelled as a fill governed by the level control system, discharging in the downcomer (level 8, ring 4). The steam outlet is located in level 10 (ring 4). Outlets to recirculation loops from the lower downcomer are located in level 3 (ring 4). Recirculation flow mixes in the jet pump with the driven flow from downcomer level 6, to discharge into the lower plenum. Channel components representing fuel bundles are connected to lower and upper plena, with leakage flow discharging to the bypass flow region (levels 5–7, rings 1, 2, and 3).

4.2 Fuel Elements

The 400 fuel elements (GE-8 type on the date of the transient) of the core were divided into three groups, corresponding to the three inner rings of the vessel model. The distribution took into account the elements of similar power. Thus, the distribution of the core elements in each type of channel is as follows:

- 28 central high power (hot channel)
- 288 central average power (average channel)
- 84 peripheral low power (peripheral channel)

Each channel was divided into 12 axial nodes, of which 9 are on the active side. Figure 15 shows the distribution of fuel elements in the core.

4.3 Recirculation Loops

Both recirculation loops have been modelled, each being divided into five components:

- PIPE 8-19, representing the suction pipe, from vessel downcomer loop B and A, respectively

- VALVE 3-21, representing recirculation suction valves loop B and A, respectively

- PUMP 4-22, representing centrifugal pumps loop B and A, respectively

- VALVE 5-23, representing the recirculation discharge valve loop B and A, respectively

- PIPE 6-24, representing the discharge pipe up to the jet pumps inlet loop B and A respectively

To simulate the recirculation pumps were used generic characteristic curves of the pumps while the moments of inertia were obtained from the data in data sheets of equipment.

4.4 Jet Pumps

Santa María de Garoña NPP has 20 jet pumps, 10 in each recirculation loop. Each jet pump has a single nozzle. The 10 jet pumps of one loop have been combined into one single component, JETP. The components used in the models are given below.

- JETP 7, modeling the 10 jet pumps in loop B
- JETP 25, modeling the 10 jet pumps in loop A

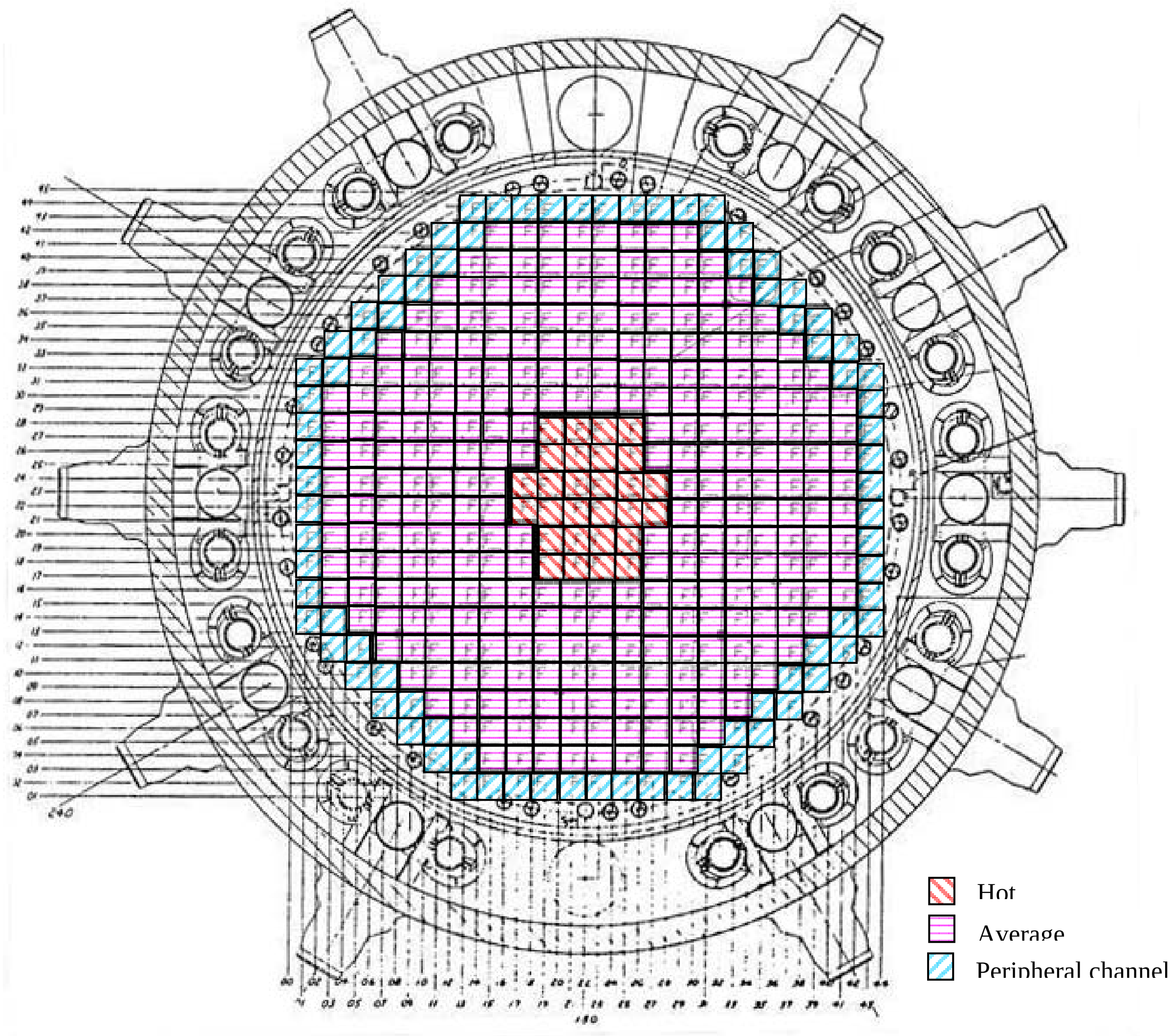

Figure 15 Vessel radial nodalization

4.5 Guide Tubes

The 97 guide tubes for the control rods are modelled with three PIPE components, each corresponding to an inner ring of the vessel, and the number of guide tubes is a function of the number of fuel elements assigned to each ring. The upper guide tube connects to the bottom of the core bypass. At the entrance of each tube guide, a null FILL is connected to close this boundary.

4.6 Main Steamlines

One single equivalent line has been used to model the four parallel steam pipes. This line includes the model of the pressure relief system (3 RVs, 3 SRVs, and 7 SVs), simulated by components VALVE. The following components were used for the model of this line from the dome of the vessel to the turbine:

- TEE 50 models the first part of pipes ranging from vessel exit to RVs and SRVs.
- VALVE 51 models the behaviour of the 3 RVs and 3 SRVs.
- BREAK 56 represents the pressure boundary condition at the discharge of the RVs and SRVs to the wetwell.
- TEE 52 models the portion of pipes ranging from the RVs and SRVs to the main steam isolation valves (MSIVs).
- VALVE 53 models the behaviour of the 7 SVs.
- BREAK 57 represents the pressure boundary condition where the SVs discharge to the drywell.
- VALVE 54 models the behaviour of the MSIVs.
- TEE 55 represents the pipes from the MSIVs to the turbine control/stop valves, and branching to the bypass valve.
- VALVE 68 models the bypass valve.
- VALVE 69 models the control/stop valves.
- BREAK 58 represents the pressure boundary condition at the turbine inlet.
- BREAK 59 represents the pressure boundary condition at the discharge to the main condenser.

Table 4 specifies the opening/closing setting pressure and areas for the valves.

4.7 Core Power

A reactor point kinetics model with trip-initiated reactivity feedback and trip-initiated scram reactivity insertion has been used to calculate the core power rate. The reactivity feedback model for void, moderator, and fuel temperature employs reactivity coefficients that have been calculated with the PANACEA code by polynomial approximations using core-averaged properties. A common axial power distribution is defined for the three types of fuel channel/bundles modelled.

Table 4 Pressure Setting Values and Areas for SRVs

Valve	Area (m^2)	Relative area (%)	Opening setpoint pressure (MPa)	Closing setpoint pressure (MPa)
RV1	0.0058	0.1306	7.277	7.080
RV2	0.0058	0.1306	7.345	7.149
RV3	0.0058	0.1306	7.414	7.218
SRV1	0.009	0.2027	7.683	7.297
SRV2	0.009	0.2027	7.722	7.336
SRV3	0.009	0.2027	7.761	7.375
SV1	0.0145	0.29	8.5	8.1
SV2	0.0145	0.29	8.6	8.2
SV3	0.0215	0.43	8.7	8.2

4.8 Feedwater

The first level extends from the vessel bottom to the top control rod drive housings. The second one ends at the jet pumps discharge support ring. The third and fourth ones go from this support ring up to the core bottom.

The feedwater flow is modelled by a constituent FILL, controlled by the feedwater control system, injecting water into the vessel downcomer at the temperature defined by the transient conditions, through a component PIPE.

4.9 Control Systems and Trips

The typical BWR control systems have been modelled: level control, pressure control, and recirculation control systems. Additionally, trips that represent the reactor protection system and other automatic actions have been developed. It is noted that, during the transients analyzed, some manual actions were initiated, such as the speed control of a group of recirculation pumps.

5. STEADY-STATE CALCULATIONS

A nominal steady state of the system was defined as a reference condition, which was obtained from the recorded data before the transient occurred. Table 5 shows the reference conditions and the values obtained with the TRACE code for this steady state.

Table 5 Reference Steady-State Condition

Parameter	Plant Data	TRACE
Thermal power 100% (MW)	1375	1375
Dome pressure (kg/cm^2 rel)	70.26	70.48
Reactor level (cm rel[1])	66.0	66.02
Recirculation flow (kg/s)	2617.0	2630.02
Core flow (T/h)	20240	18647.7
Feedwater flow (kg/s)	677.56	689.10
Steam flow (kg/s)	685.78	688.7
Feedwater temperature (K)	452.0	452.8

To adjust the steady state with TRACE, the approach was to adjust components or submodels separately, with the proper boundary conditions, and assemble them one by one to build up the entire model. For this purpose, the Symbolic Nuclear Analysis Package (SNAP) program, version 0.24.1 (Ref. 8), was used. Partial models were adjusted for recirculation, steamlines, channels, vessel, and control systems. The final tuning of the input model was obtained by adjusting the flow area fraction of components in the lower plenum of the vessel and the loss coefficients of components near the input to the core. Figure 16 shows the full model done with SNAP, and Figures 17–19 plot some relevant variables to reach the steady state.

Some problems were encountered with the nodalization generated with SNAP, because the properties of components were incomplete. These situations were resolved by consulting the user's manual and by using the input samples provided with TRACE.

Table 5 presents the final steady state reached for the most significant variables. It is noted that the conditions obtained by TRACE closely approximate the referenced steady state. The greatest error occurs in core flow (7.8 percent lower in TRACE). A null transient was run after the steady state was reached, to verify the stability of the steady-state conditions.

Figures 20–24 show the axial distribution of void fraction, pressure, liquid temperature, liquid velocity, and steam velocity for hot, average, and peripheral channels in a steady state.

[1] The centimeters relative to zero-scale (12.22 m from the bottom head) correspond to the bottom of the steam separators.

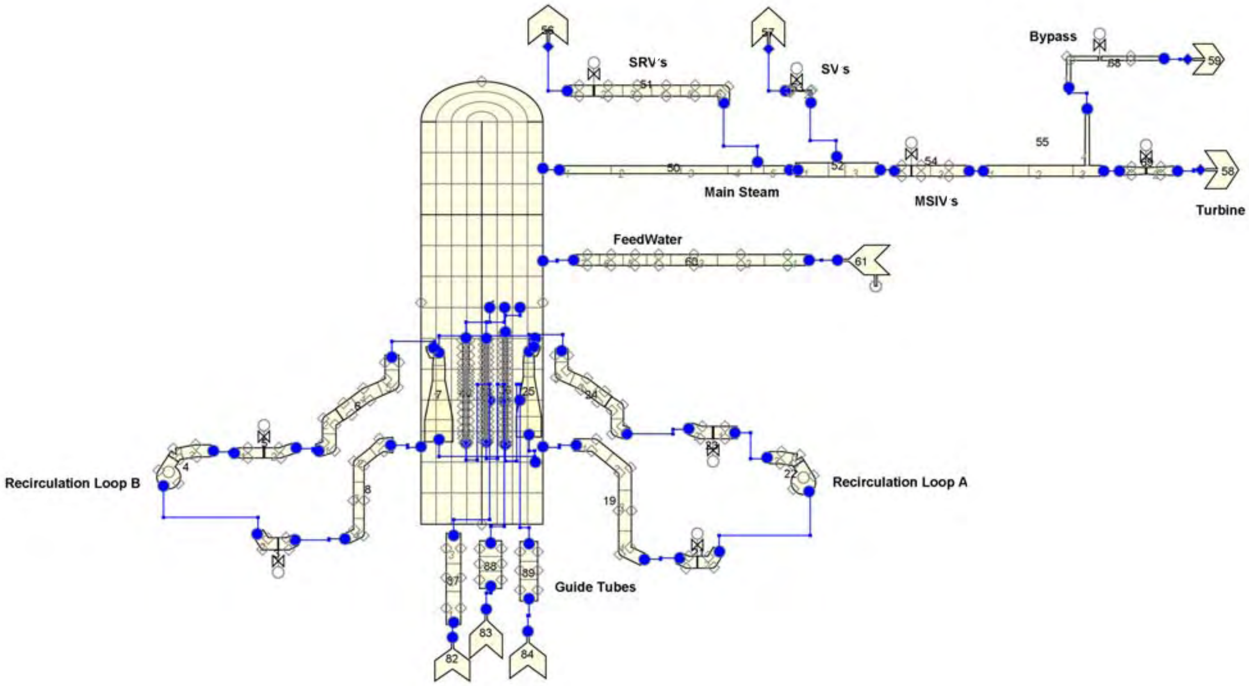

Figure 16 SNAP nodalization for Santa María de Garoña NPP

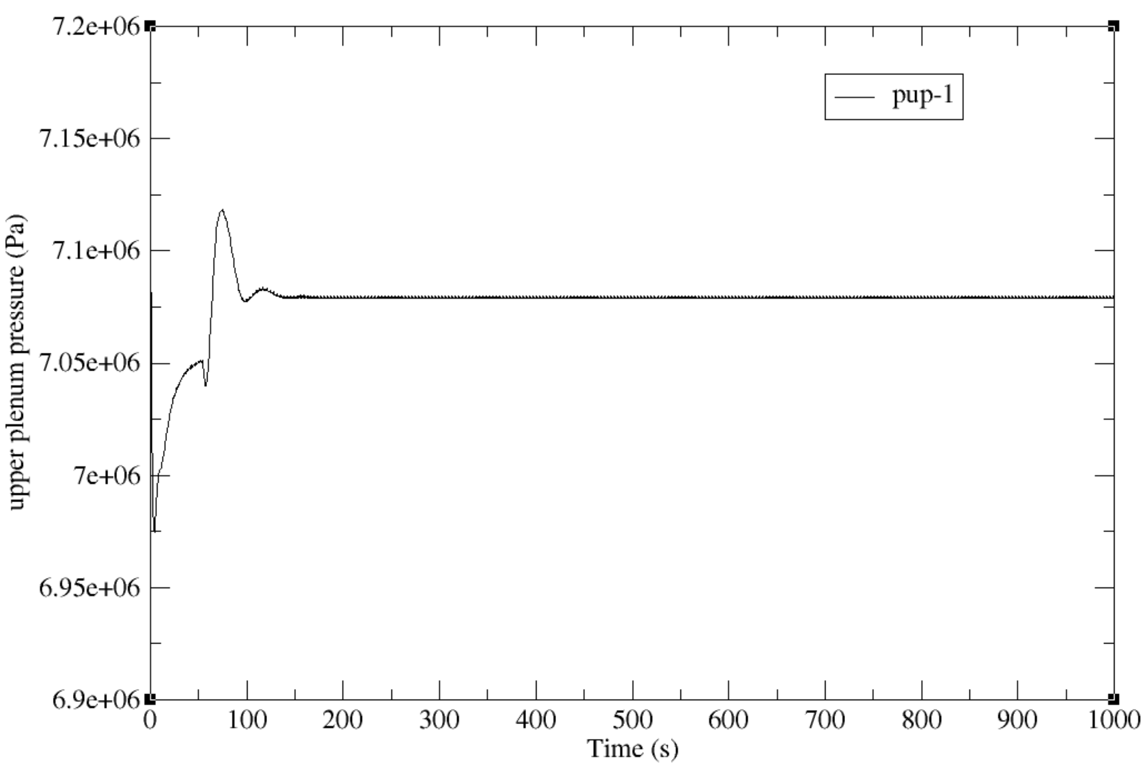

Figure 17 Steady-state dome pressure

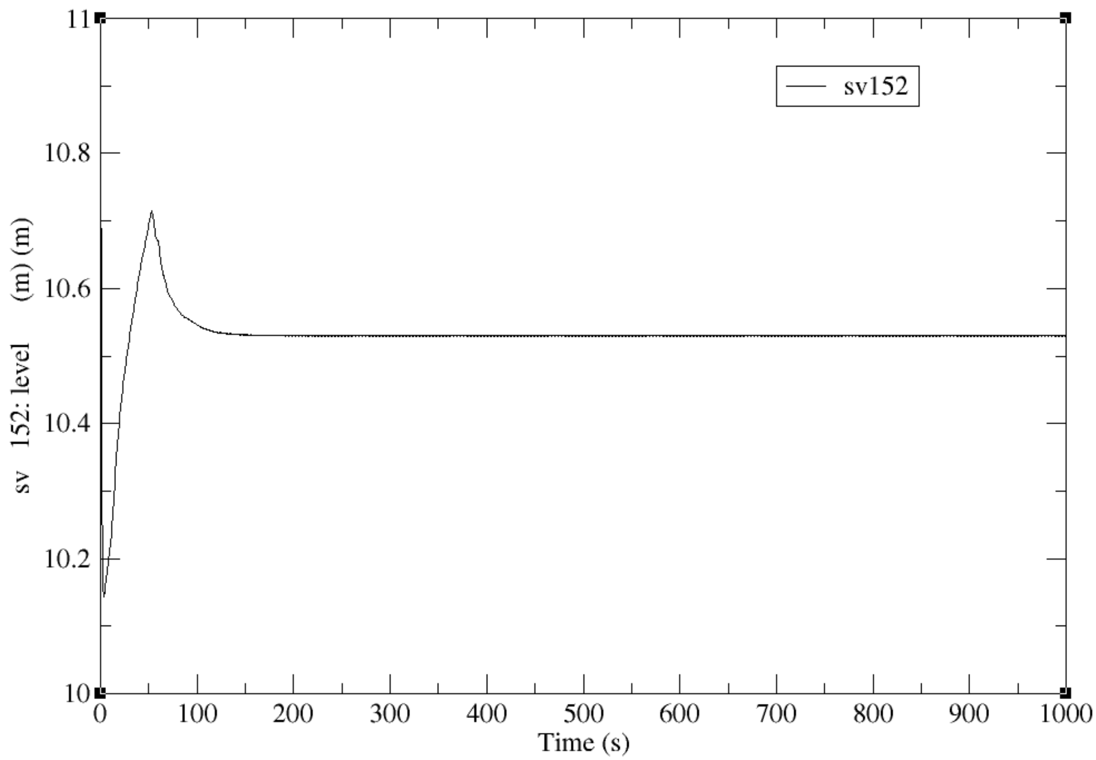

Figure 18 Steady-state reactor level

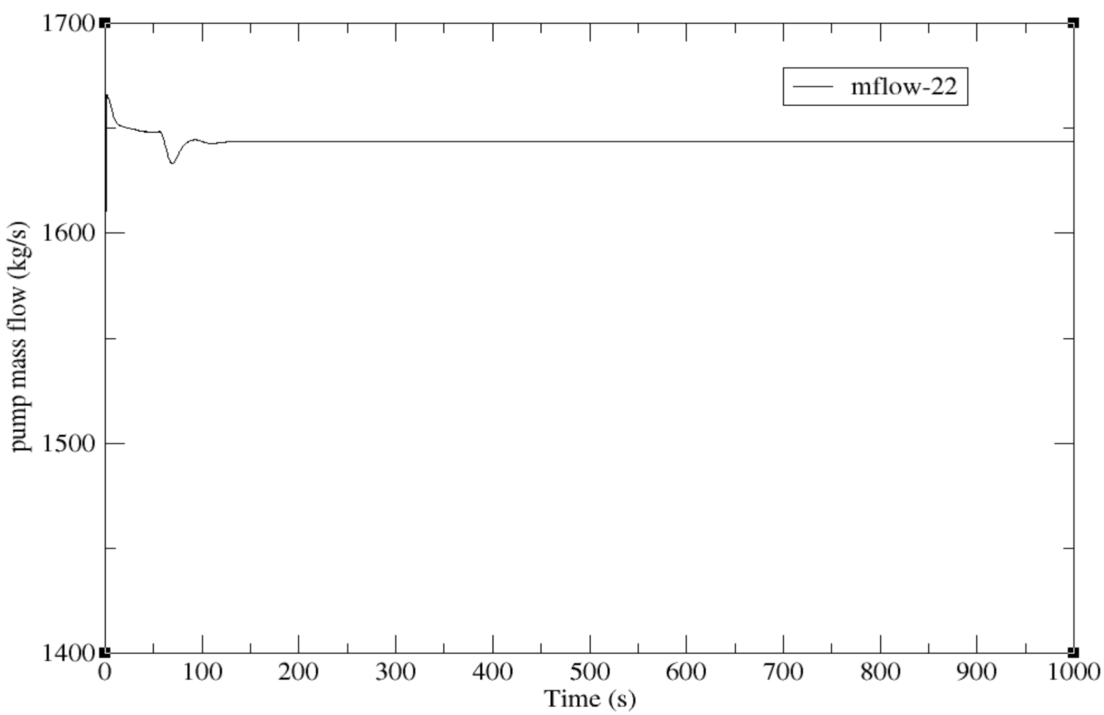

Figure 19 Steady-state recirculation flow rate A

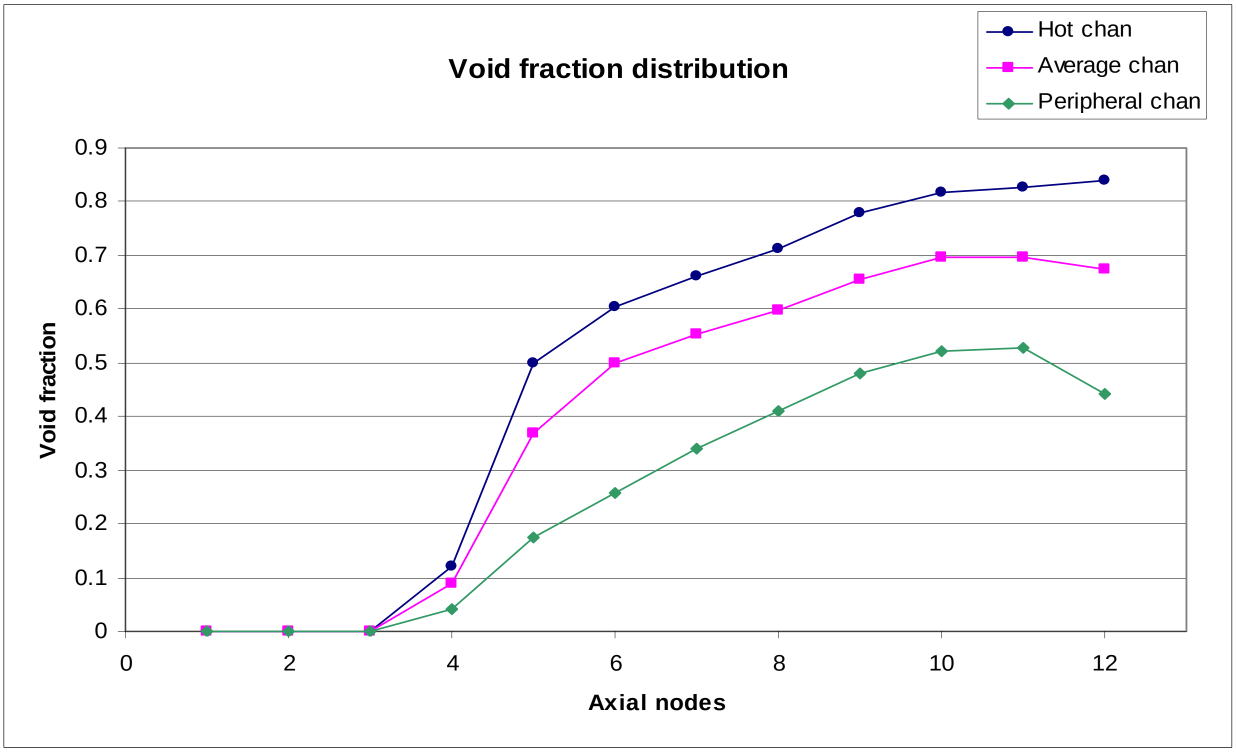

Figure 20 Axial void fraction distribution

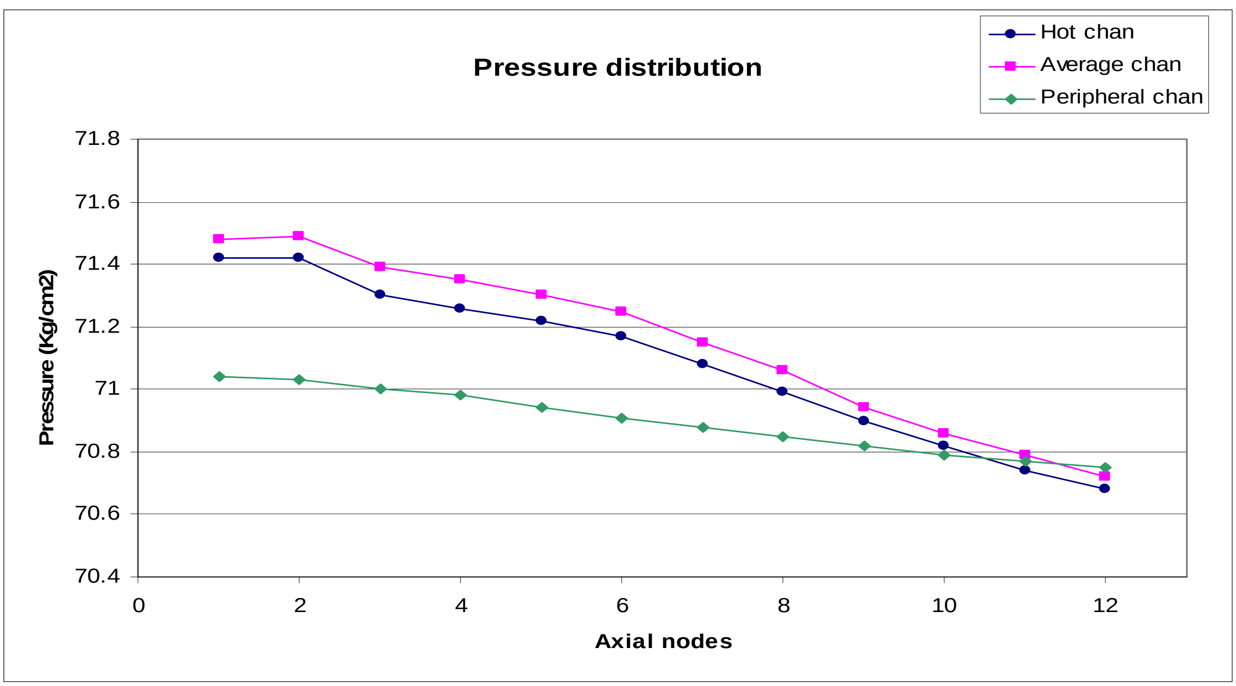

Figure 21 Axial pressure distribution

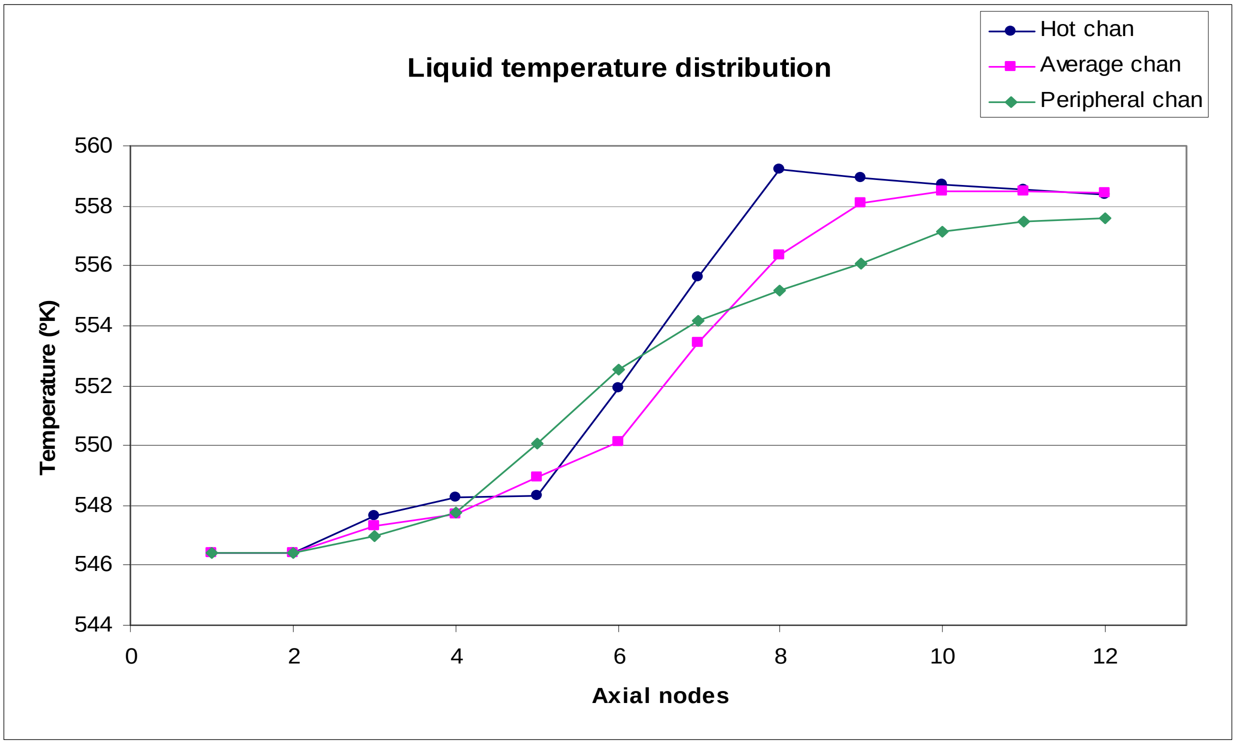

Figure 22 Axial liquid temperature distribution

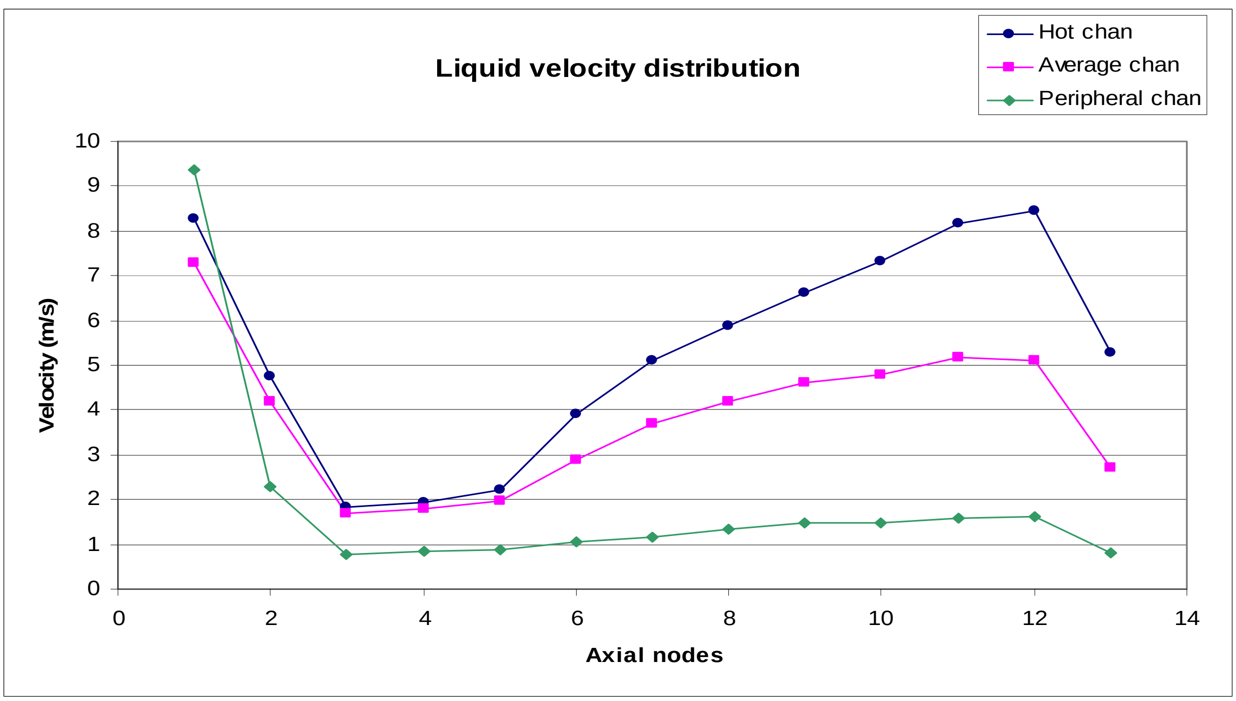

Figure 23 Axial liquid velocity distribution

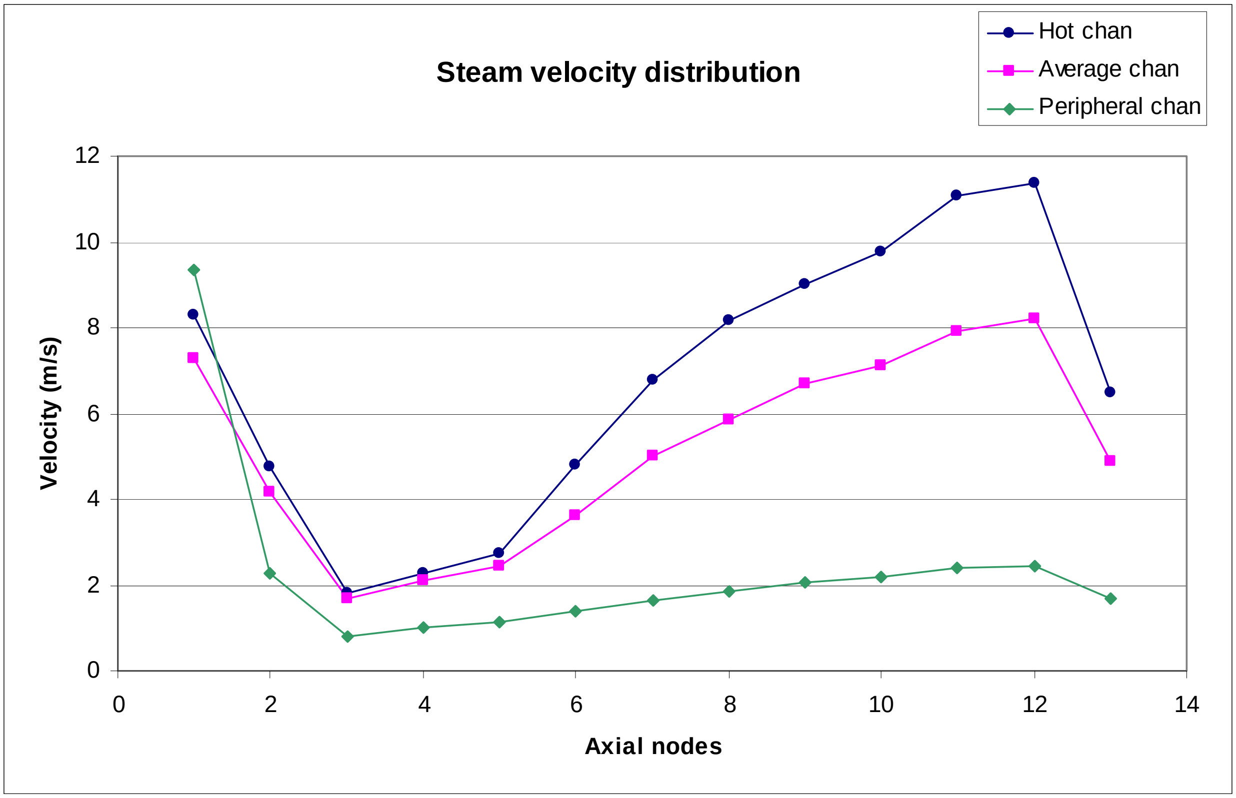

Figure 24 Axial steam velocity distribution

6. TRANSIENT RESULTS AND COMPARISON WITH PLANT DATA

The transient simulation using the TRACE code considered the following factors that occurred at the plant:

- The SRV B remained closed.

- The operator manually reduced the speed of the recirculation pumps (Figures 8, 9). During this time, the scoop tube of group B was blocking until an automatic run back occurred, caused by the feedwater flow falling below 20 percent of the nominal flow (Figures 6, 7).

After a null transient, the transient began with the closing of control valves in 3 seconds. For all other active components during the transient, a lookup table was added for the feedwater flow injection, speed of recirculation pumps, position of control valve, and core power. It has also been introduced by tabular method the feedwater temperature change caused by the loss of steam into the shell of heaters in the feedwater system.

Figures 25–27 show plot comparisons of calculated and measured values for the most relevant variables. All plots include 32 seconds of steady state before the initiation of the transient.

Figure 28 shows a combination plot of the area of RVs and SRVs and the reactor pressure. It shows the opening of three RVs and one SRV. Figure 29 shows another combination plot that represents the temperature feedwater, the plant data reactor pressure, and the calculated reactor pressure.

These figures show that the calculated variables with TRACE are in acceptable agreement with the measured values. For this purpose, it was necessary to adjust the values of the above-mentioned tables, especially those relating to temperature feedwater that greatly influence the results of the transient.

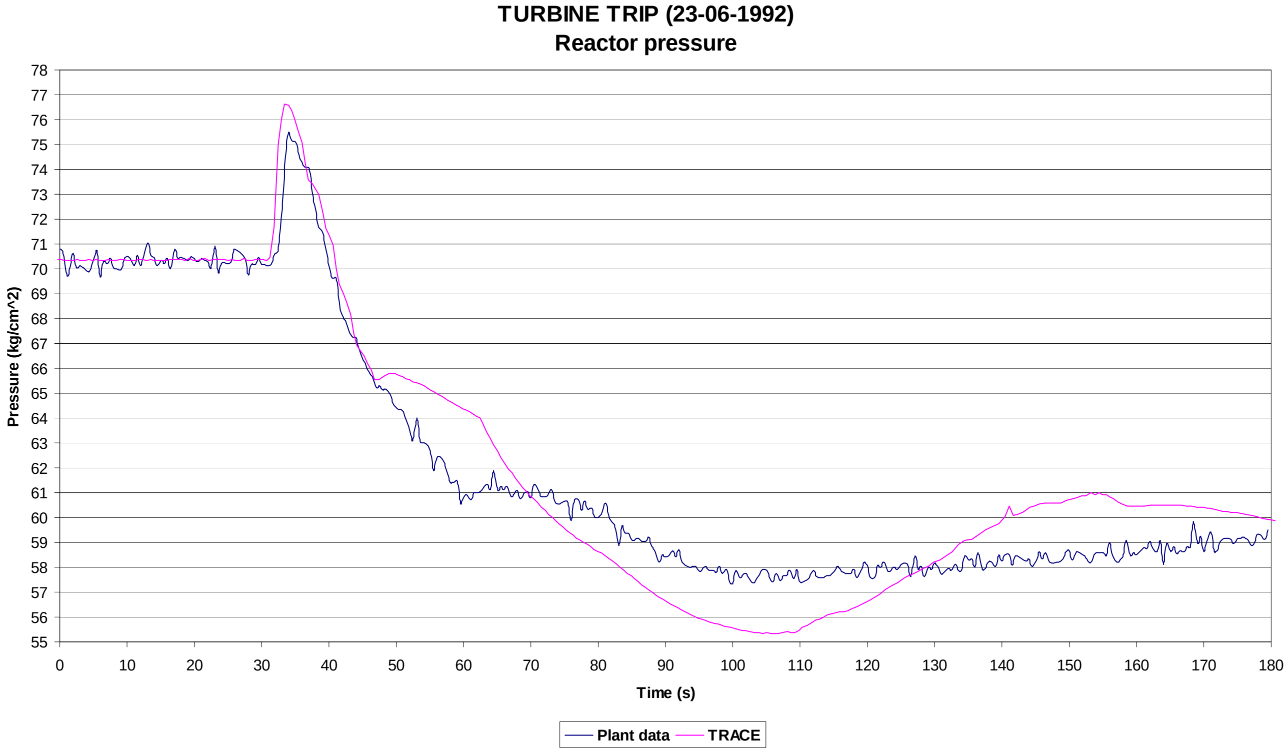

Figure 25 Steam dome pressure

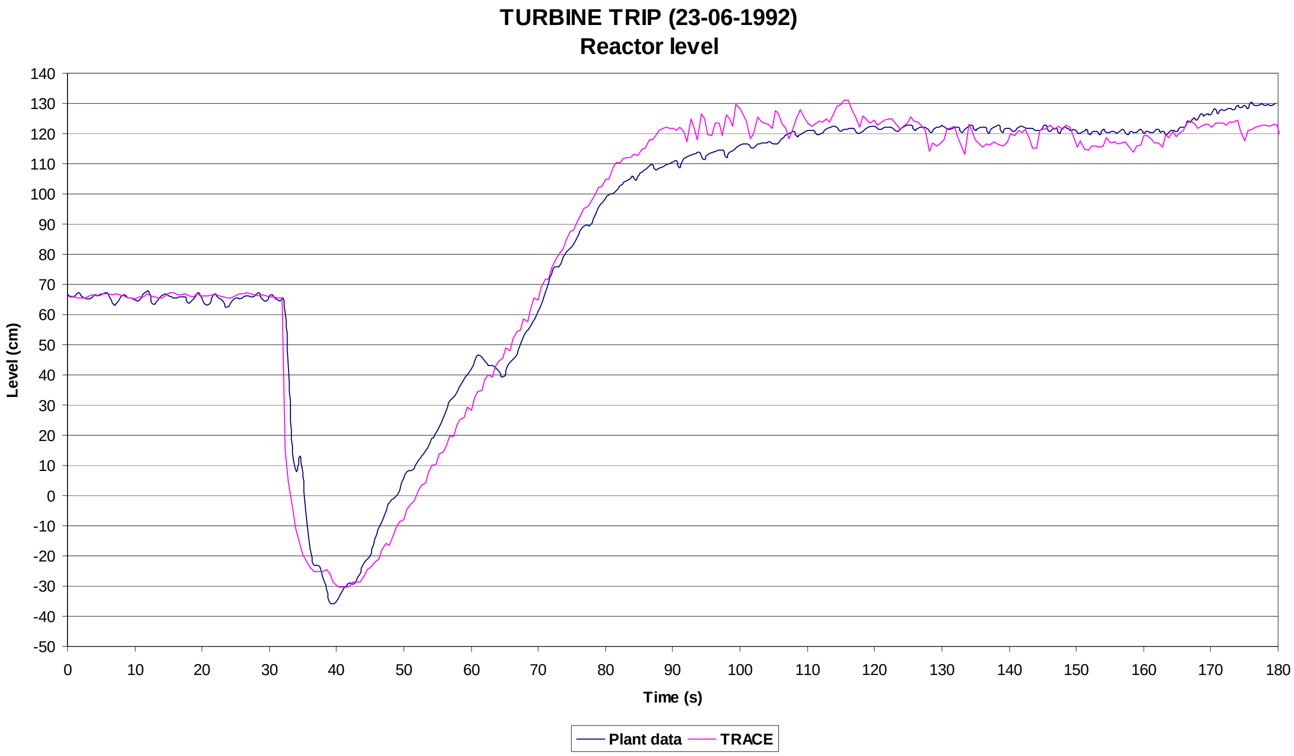

Figure 26 Reactor level

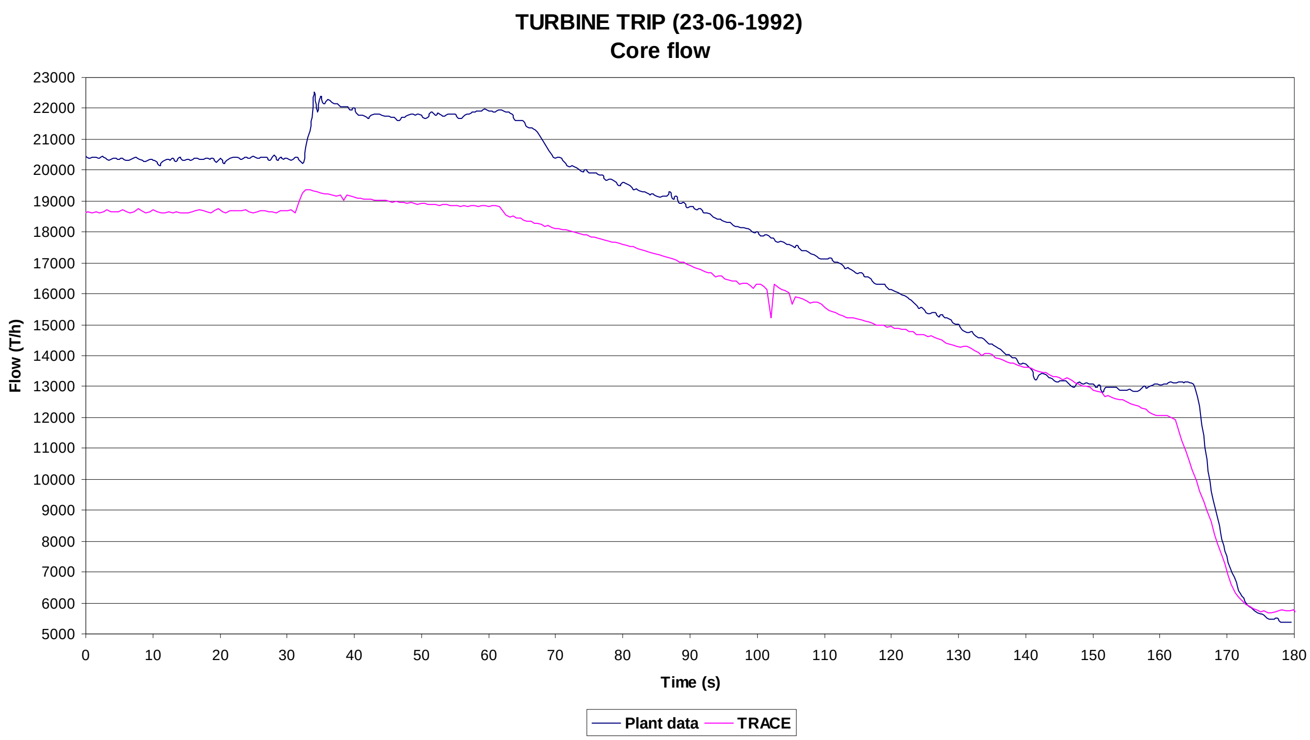

Figure 27 Core flow

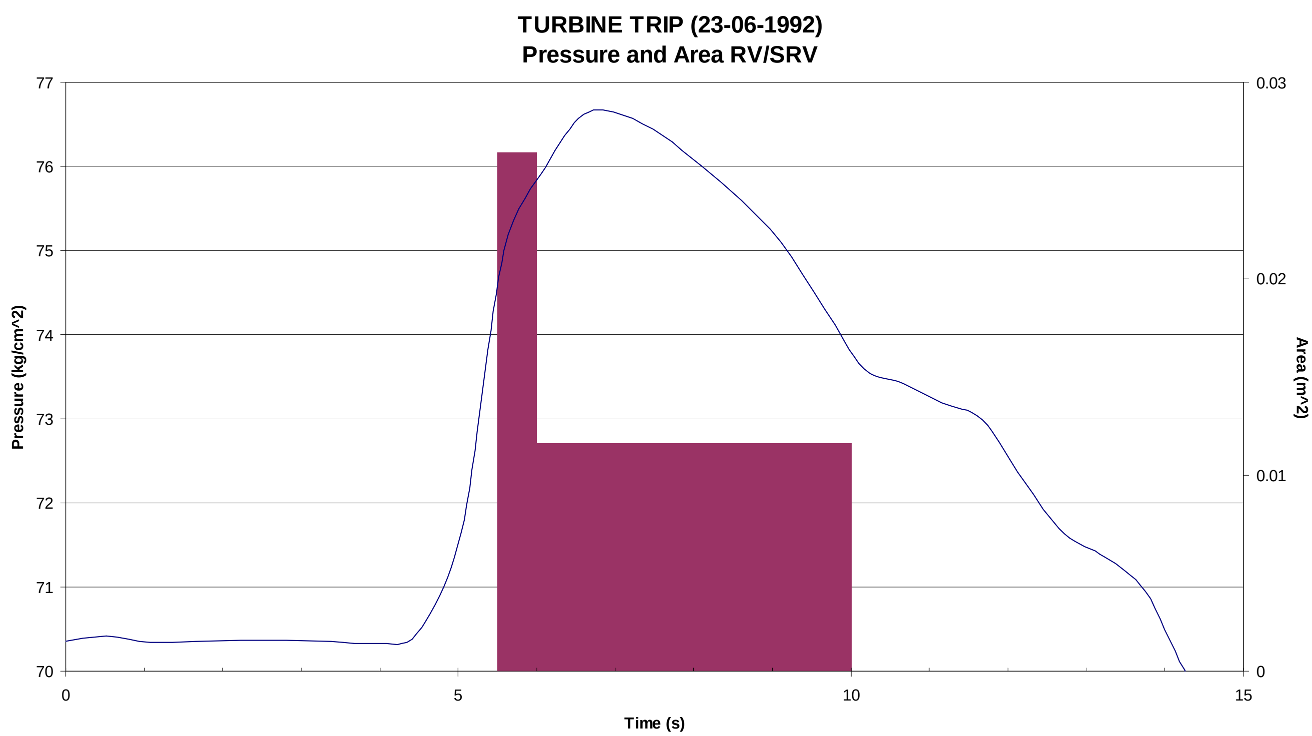

Figure 28 Combined graph of reactor pressure and area of RV/SRV

TURBINE TRIP (23-06-1992)

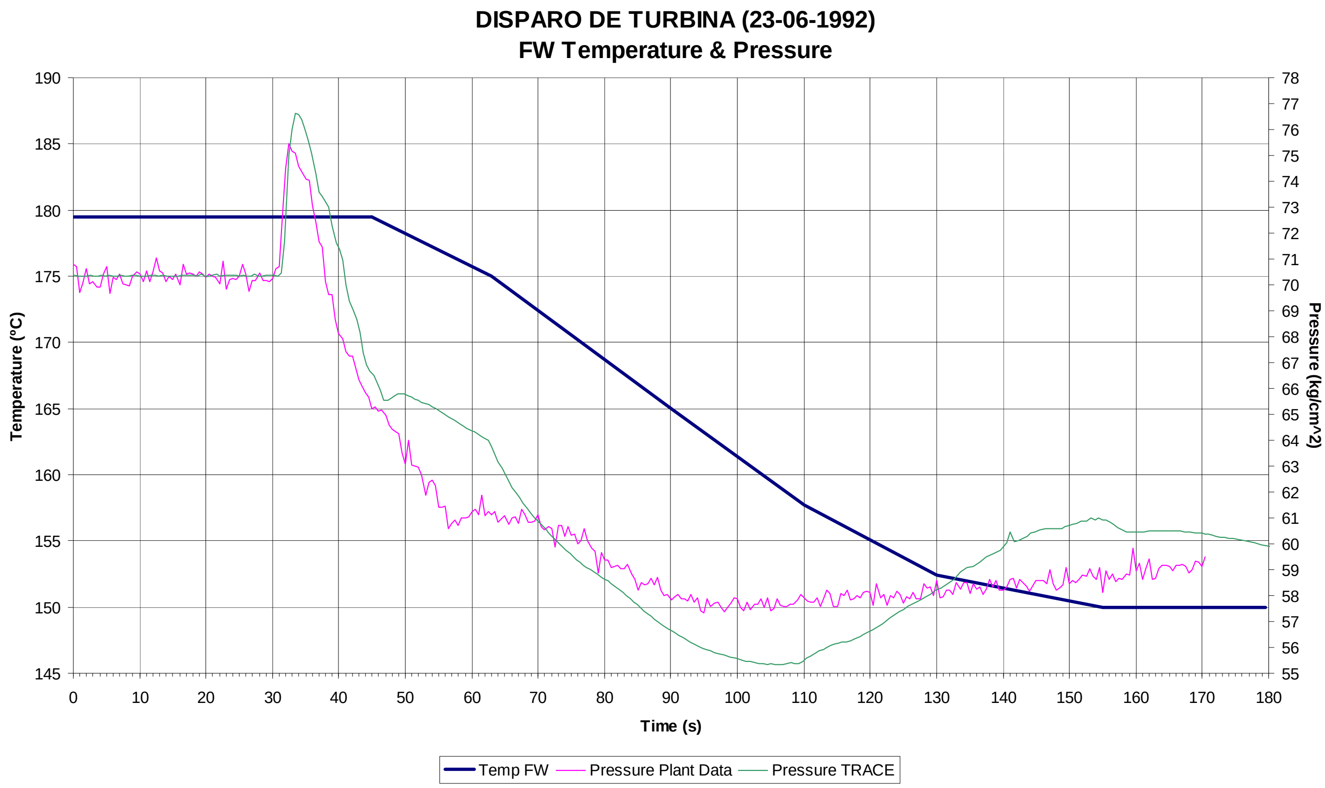

Figure 29 Feedwater temperature and reactor pressure (real and calculated)

7. RUN STATISTICS

The simulations were run on a Pentium 4 workstation, 3 gigahertz under Windows XP Professional 64-bits operating system.

Figure 30 is a plot of the total central processing unit (CPU) time, and Figure 31 plots the timestep size during the simulation. It shows that the simulated transient runs faster than real time, and there is a sudden increase in the CPU time when the transient starts, caused by a lower timestep size. Then the timestep size increases, and the total CPU decreases again.

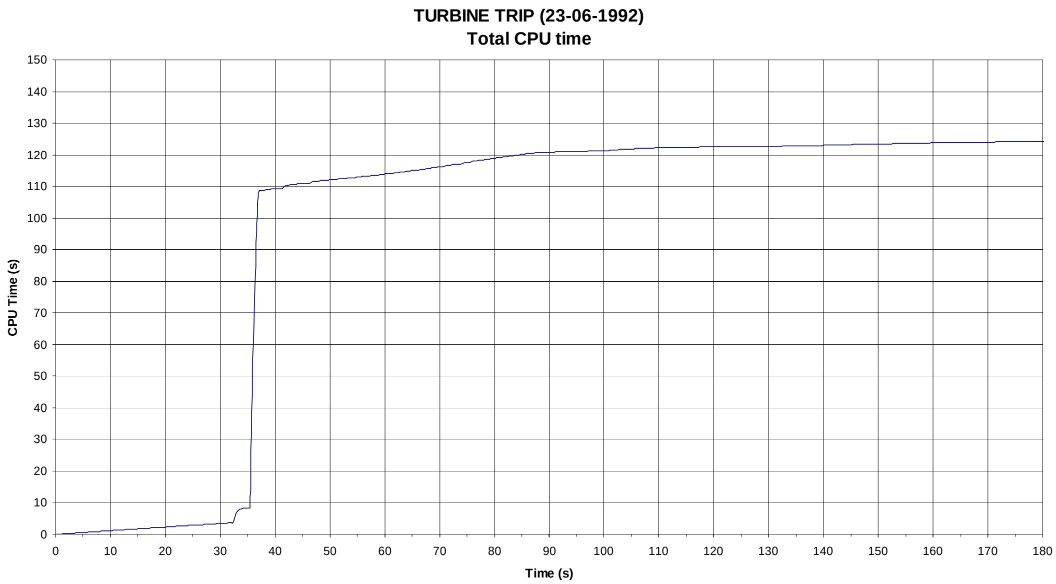

Figure 30 Total CPU time

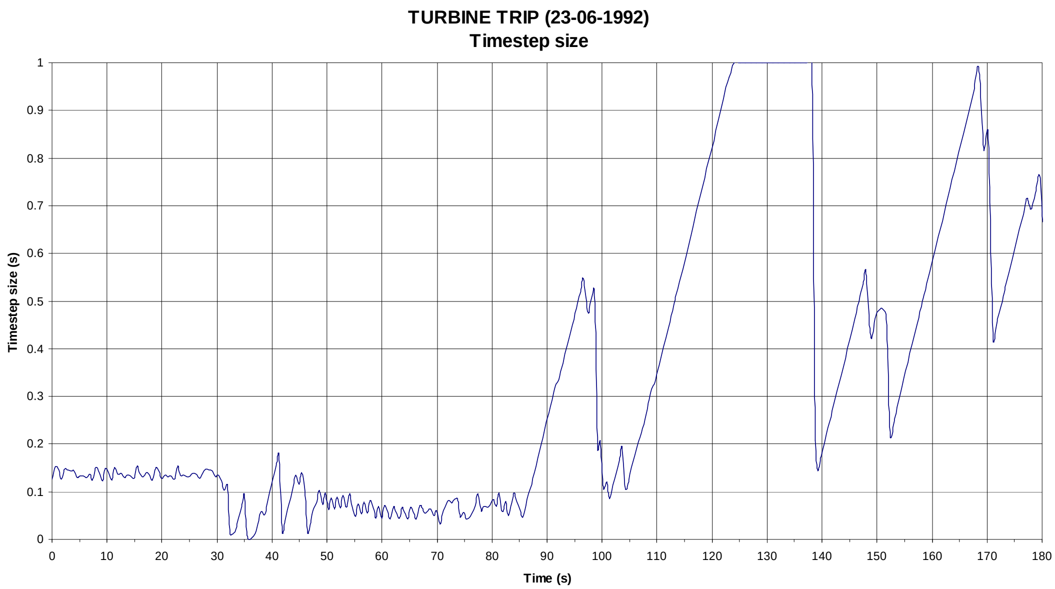

Figure 31 Timestep size

8. CONCLUSIONS

A model of the Santa María de Garoña NPP for the TRACE code, version 4.16, has been developed. The SNAP code, version 0.24.1, was used during the model nodalization and tuning. The model has been validated to obtain a steady state and, at the same time, a turbine trip transient. In both cases, the calculations were compared with plant data with acceptable results.

To achieve a steady state, the reference values were the ones used in the model for TRAC BF1, with adjustments of a number of parameters of the components regarding the flow areas and loss coefficients near the core inlet.

Core flow is the variable that is responsible for the greatest difference between measurements and calculations. Improvement of the recirculation loop model is an area identified for further work. Another area of potential improvement is tuning the control systems, such as feedwater, pressure, and recirculation.

9. REFERENCES

1. P. Corcuera, M. Garcés, “Nowadays Tools for Graphical Post-Processing of TRAC-BF1 Results,” NUREG/IA-0190, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, 2000.
2. “TRACE V4.0 User’s Manual,” U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, 2004.
3. “TRACE V4.16 User’s Manual Volume 1: Input Specification,” Division of Systems Analysis and Regulatory Effectiveness, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, 2004
4. J.W. Spore, J.S. Elson, S.J. Jolly-Woodruff, T.D. Knight, J.H. Mahaffy, C. Murray, “TRAC-M/FORTRAN 90 (Version 3.0), Theory Manual,” Report LA-UR-00-910, Los Alamos National Laboratory and Pennsylvania State University, 2000.
5. J.A. Borkowski, N.L. Wade, et al., “TRAC-BF1/MOD1: An Advanced Best-Estimate Computer Program for BWR Accident Analysis,” NUREG/CR-4356, EGG-2626, Idaho National Engineering Laboratory, August 1992.
6. J.A. Borkowski, N.L. Wade, et al., “TRAC-BF1/MOD1 Models and Correlations,” NUREG/CR-4391, EGG-2680, Idaho National Engineering Laboratory, August 1992.
7. “Cuaderno de Cálculo de la Nodalización de la CN de Santa María de Garoña con el código TRAC-BF1,” Sección de Sistemas, Nuclenor S.A., 1994.
8. “Symbolic Nuclear Analysis Package (SNAP), User’s Manual,” Applied Programming Technology, Inc., 2007.
9. “Informe SCRAM por disparo de turbina 23-06-92,” Rev. 1, Sección de apoyo técnico a operación, Nuclenor S.A., January 7, 1992.