NUREG/IA-0489

International Agreement Report

RELAP5 Model of a CANDU-6 (Embalse) Nuclear Power Plant: Application to a Turbine Trip Event

Prepared by:
A. I. Lazarte and M. V. Guala

Autoridad Regulatoria Nuclear (ARN)
Av. Del libertador 8250, CABA C1429PNB

Kirk Tien, NRC Project Manager

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

Manuscript Completed: May 2017
Date Published: January 2019

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 Library at 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 Publishing Office
Washington, DC 20402-0001
Internet: bookstore.gpo.gov
Telephone: (202) 512-1800
Fax: (202) 512-2104

2. **The National Technical Information Service**
5301 Shawnee Road
Alexandria, VA 22312-0002
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: **U.S. Nuclear Regulatory Commission**
Office of Administration
Multimedia, Graphics, and Storage &
Distribution Branch
Washington, DC 20555-0001
E-mail: distribution.resource@nrc.gov
Facsimile: (301) 415-2289

Some publications in the NUREG series that are posted at NRC's Web site address 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, 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-0489

International Agreement Report

RELAP5 Model of a CANDU-6 (Embalse) Nuclear Power Plant: Application to a Turbine Trip Event

Prepared by:
A. I. Lazarte and M. V. Guala

Autoridad Regulatoria Nuclear (ARN)
Av. Del libertador 8250, CABA C1429PNB

Kirk Tien, NRC Project Manager

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

Manuscript Completed: May 2017
Date Published: January 2019

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

ABSTRACT

As part of the Embalse Nuclear Power Plant (CANDU-6) refurbishment, the Nuclear Regulatory Authority of Argentina (ARN) has developed a full plant thermo-hydraulic model with point kinetics. Using the RELAP5 mod.3.3 Patch4 thermal-hydraulic code, the model simulates accidents within the design basis. This model is being used to perform an independent review of the safety analysis submitted by the designer and operator to the regulatory body. It would also be used as a tool for understanding plant dynamic behavior. The model developed by the ARN includes most of the hydraulic systems and control logic from the original design; the planned design changes for refurbishment are implemented in a more recent version of the model. To validate the model regarding its response to transients, a postulated initiating event (PIE) was modeled and simulated, and the results obtained were compared with recorded data of that particular event which took place in the NPP in 2007. The event consists of a spurious turbine trip leading to a rapid power reduction (stepback) to 49% of full power. This reduction was the only boundary condition imposed to the model. The purpose of this comparison is to verify consistency between important plant parameters, thermal-hydraulic model predictions and recorded data. Only the first 15 minutes of the transient were simulated. It is shown that the evolutions of some given variables are similar to those observed in the NPP. Also, the results obtained indicated that modifications should be performed to improve the model, especially regarding control components for getting better adjustment of plant data.

TABLE OF CONTENTS

Page

ABSTRACT iii

TABLE OF CONTENTS v

LIST OF FIGURES vii

LIST OF TABLES vii

ACKNOWLEDGMENTS ix

ABBREVIATIONS xi

1 INTRODUCTION 1

2 RELAP5 MODEL OF EMBALSE NUCLEAR POWER PLANT 3
- 2.1 Core and Feeders 4
- 2.2 Primary Heat Transport System 6
- 2.3 Secondary system 7
  - 2.3.1 Steam Generators 8
  - 2.3.2 Feedwater System 9
  - 2.3.3 Steam Line System 10
- 2.4 Emergency Core Cooling System 11
- 2.5 Control System and Safety Trips 13
- 2.6 Other Plant Systems 13
  - 2.6.1 Pressure and Inventory Control System 14

3 SIMULATION OF A SPURIOUS TURBINE TRIP 15
- 3.1 Steady Sate and Model of the Spurious Turbine Trip 15
- 3.2 Comparison of the Spurious Turbine Trip 16

4 CONCLUSIONS 23

5 REFERENCES 25

LIST OF FIGURES

Page

Figure 1 CANDU6's PHTS Schematic Drawing. ... 3

Figure 2 Diagram of Zone Division or Equivalent Fuel Channels Used in the Plant Model of the CNE. ... 4

Figure 3 Diagram of the Implemented Channel Model. ... 5

Figure 4 Channels Selected to Model the Geometry of the Inlet and Outlet Feeder. This is relevant as to the Axial Location of the Channel. ... 5

Figure 5 Nodalized Diagram of the PHTS in the CNE Model. ... 6

Figure 6 Nodalized Pressure Control System of the CNE Model. ... 7

Figure 7 Diagram of a SG of a CANDU6 Type Reactor. ... 8

Figure 8 (Left) SG Nodalized Diagram of the Secondary Side (Right) U-Tubes Bundle Nodalized Diagram of the PHTS. ... 9

Figure 9 Nodalized Diagram of the Main and Auxiliary Feedwater Pipelines ... 10

Figure 10 Steam Pipelines from SGs Domes to the Turbine ... 11

Figure 11 High Pressure ECCS Nodalized Diagram in RELAP5. ... 12

Figure 12 Diagram of the Medium and Low Pressure ECCS in RELAP5 ... 13

Figure 13 Comparison Between Mass Flow Rates of the ARN Plant Model and CNE Measurements for One of the Feedwater Lines. ... 17

Figure 14 Levels of the 4 Steam Generators (GV1/2/3/4) Obtained from the ARN Model and the CNE measurements ... 17

Figure 15 Steam Flow to the Condenser. For this comparison the results of the operator model were available [8] and the contributions of the flow to the condenser and the flow of preheaters were added ... 18

Figure 16 Comparison Between the Secondary System Pressure Measured in CNE and the Pressure Obtained by the ARN Model for SG4 ... 19

Figure 17 Comparison Between the Pressure Measured in the CNE (average) and the Pressure Obtained by the ARN Model for the HDR5. ... 20

Figure 18 Comparison Between the Temperatures of the Header HDR8 and the Temperature Obtained by the ARN Model for the HDR8. ... 20

Figure 19 Comparison Between the Pressurizer Levels of CNE and the ARN Model. The resulting curve if the initial condition of the pressurizer level were changed is also shown. ... 21

LIST OF TABLES

Page

Table 1 CNE steady state variables ... 15

ACKNOWLEDGMENTS

The authors acknowledge the technical support from the Research Center of Computational Methods (CIMEC for its Spanish acronym) and the information given by the operator (NA-SA) of Embalse Nuclear Power Plant.

ABBREVIATIONS

ARN	Nuclear Regulatory Authority of Argentina
ASDV	Atmospheric steam discharge valves
CNE	Embalse Nuclear Power Plant
CSDV	Condenser steam discharge valve
DBA	Design basis accidents
ECCS	Emergency core cooling system
EWS	Emergency water supply
F&B	Pressure and inventory control system
LRV	Liquid relief valves
MSSV	Main steam safety valve
PHTS	Primary heat transport system
PHWR	Pressurized heavy water reactor
RELAP	Reactor Excursion and Leak Analysis Program
SDS	Shutdown system
SEP	Shutdown cooling system
SGs	Steam generators
TSO	Technical support organization
TGV	Turbine governor valves

1 INTRODUCTION

The Embalse Nuclear Power Plant (CNE for its Spanish acronym) is a CANDU- 6 type reactor, which has completed its designed lifespan. A refurbishment project is in progress and several modifications are being implemented. Due to this, safety assessment has to be updated.

With the collaboration of a Technical support organization (TSO), the ARN developed a completely independent thermo-hydraulic model for performing different analyses and simulations of a selected set of accidents. For the development of the CNE model, RELAP5 mod3.3 Patch 4 code was used (RELAP5 hereinafter). Additionally, users have performed different validations of this code for CANDU6 type reactors; see references [1] and [3]. A code validation was also performed through an event of loss of forced cooling (total loss of Class IV and turbine trip) in Darlington NGS Unit 4 (CANDU9), reference [4]. On the other hand, a comparison between codes was held at the IAEA on a small loss of coolant accident in a Canadian test facility called RD14m, showing the code's ability to simulate a small break in a CANDU reactor in a satisfactory way. In this code inter-comparison, a very good correspondence between the experimental data and the model was observed, reference [5].

Documentation of the existing plant such as isometric drawings, data sheets of components, design manuals, drawings, flow charts, general arrangements and operating manuals were used as the main source for the RELAP5 model development. No generic input data was used.

The model's main objective is to improve knowledge on the CNE's response to design basis accidents and to provide an independent tool for transient calculations for the refurbishment project. It should be noted that the model is designed to evaluate accidents. Therefore, safety systems and simplifications of plant process systems are the main focus.

In this paper, a comparison is made between the model developed and a CNE event which occurred in 2007. The event's sequence consists of a turbine trip followed by a controlled power reduction to about 49% of full power. The results of the comparison were used to validate the model. In addition, the plant model is being conditioned and updated to include design changes.

The next section presents modeling criteria and the schematics of components and systems. The third section presents a comparison between model variables and real plant steady-state variables. Finally, several model process variables are shown and compared to plant recorded variables, reference [8].

2 RELAP5 MODEL OF EMBALSE NUCLEAR POWER PLANT

Embalse nuclear power plant is a pressurized heavy water reactor (PHWR), which at nominal conditions, generates and transfers approximately 2015 MWth to the secondary system. The CANDU-6 core consists of 380 fuel channels arranged horizontally. The primary heat transport system (PHTS) has two closed cooling circuits in a "figure 8" and uses D_2O as coolant. These circuits, although independent, are interconnected in multiple locations. An extensive description of CANDU-6 operation and the components can be found in reference [7]. To simplify, this information is not included in this paper.

The primary system contains four vertical steam generators (SGs), four main pumps, four inlet and four outlet headers. Between hot legs in each circuit, there are two interconnection lines that maintain the PHTS hydraulic stability. A schematic drawing of the PHTS is shown in Figure 1. The core inlet temperature range is between 260-267°C, depending on plant aging, whereas the coolant outlet temperature is approximately 309 °C. The fluid leaves the core as a two phase mixture with low steam quality. The mass flow of each loop is slightly higher than 2100 kg/s.

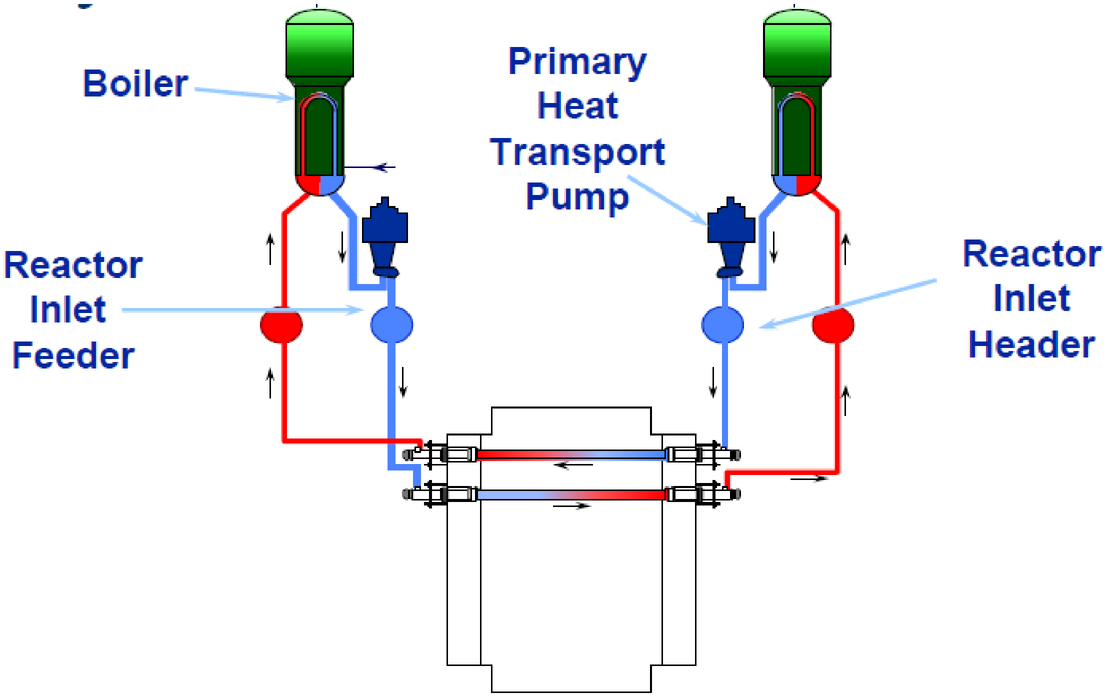

Figure 1 CANDU6's PHTS Schematic Drawing

Each fuel channel consists of a pressure tube and a calandria tube. The latter isolates the coolant from the moderator system by a non-conducted gas which is located between both tubes. The channel contains 12 fuel bundles with natural uranium pellets. Each fuel bundle measures half a meter length and contains 37 fuel rods.

There are 4 headers per coolant circuit. Two of those outlet headers collect the coolant coming out from 95 pressure tubes whereas the other two inlet headers distribute coolant to another 95 pressure tubes. The pipes connecting pressure tubes with headers are called feeders. Additionally, in the PHTS there are some connections to the pressure control, purification, inventory control (F&B hereinafter), emergency core cooling (ECCS) and other systems.

The model includes the following systems and some of them will be described below:

i. PHTS,
ii. Secondary system (primary and auxiliary feedwater and steam lines, turbine and condenser),
iii. Pressure and inventory control system,
iv. Emergency core cooling system (high, medium and low pressure stages)
v. Moderator system
vi. Emergency water supply (EWS)
vii. Shutdown systems (SDS N1 and N2)
viii. Point kinetics

and the corresponding control logic for each system. These models were independently developed and then integrated into a global plant model, providing steady state conditions similar to real plant parameters.

2.1 Core and Feeders

The core of the CNE was modeled with 5 hydraulic areas per circuit. Each area is modeled with two averaged channels, having a total of 20 channels. In order to understand the behavior of an individual channel, from one of the hydraulic areas of higher power (as shown in Figure 2) the channel with highest power per circuit was modeled separately, leading to a total of 24 channels.

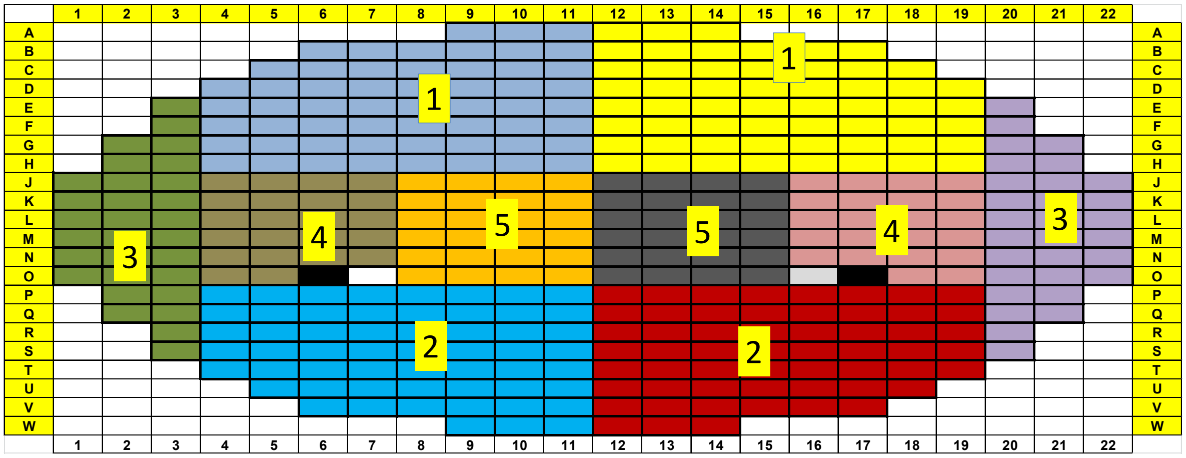

Figure 2 Diagram of Zone Division or Equivalent Fuel Channels Used in the Plant Model of the CNE

Each coolant channel is modeled by a set of components that represent its different parts (coolant channel, closure plug, annular space, etc.). In Figure 3 the outline of the channel model is shown.

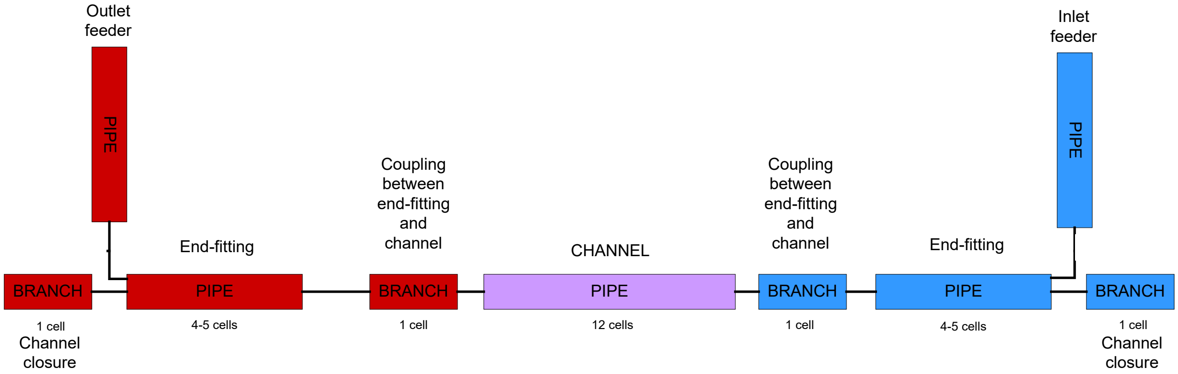

Figure 3 Diagram of the Implemented Channel Model

The channel itself is represented by a 12-cell pipe, where each cell represents a bundle of fuel elements.

From each core area, a channel was selected to represent its feeders for that particular area. The feeder/channel chosen corresponds to one located near the geometric mean within each area. Both inlet and outlet feeders have the same geometric arrangement except for the hydraulic diameter (flow area). This was applied to all the zones except the model of the individual channels, where the corresponding feeders were used. Figure 4 shows the selected and modeled channels as representatives within each zone, the white ones are the individual channels.

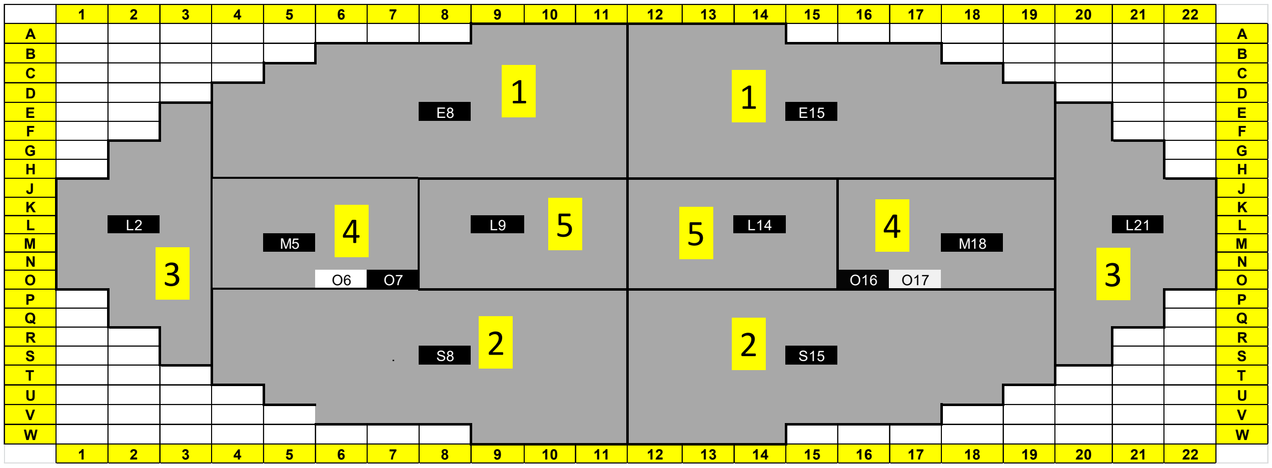

Figure 4 Channels Selected to Model the Geometry of the Inlet and Outlet Feeder. This is relevant as to the axial location of the channel

Due to the fact that some peripheral channels have a flow restrictor, a concentrated pressure loss was applied, so that the average channel mass flow would be close to the expected total mass flow for the set of channels of the respective zone. When considering CANDU-6 phenomenology, it is well known that each inlet and outlet feeder has an individual roughness. The developed plant model imposes an average roughness for all channel groups, except in the individual channels.

2.2 Primary Heat Transport System

Figure 5 shows a schematic model of the PHTS in RELAP5. In it, the four main pumps, the PHTS pipeline, primary side tubes of SGs, etc. can be distinguished.

The RELAP5 “Bingham” type pump was selected to model the four main pumps. Their characteristic parameters were adjusted to be consistent with experimental data. The experimental data used for the first quadrant came from operation and shutdown during plant commissioning.

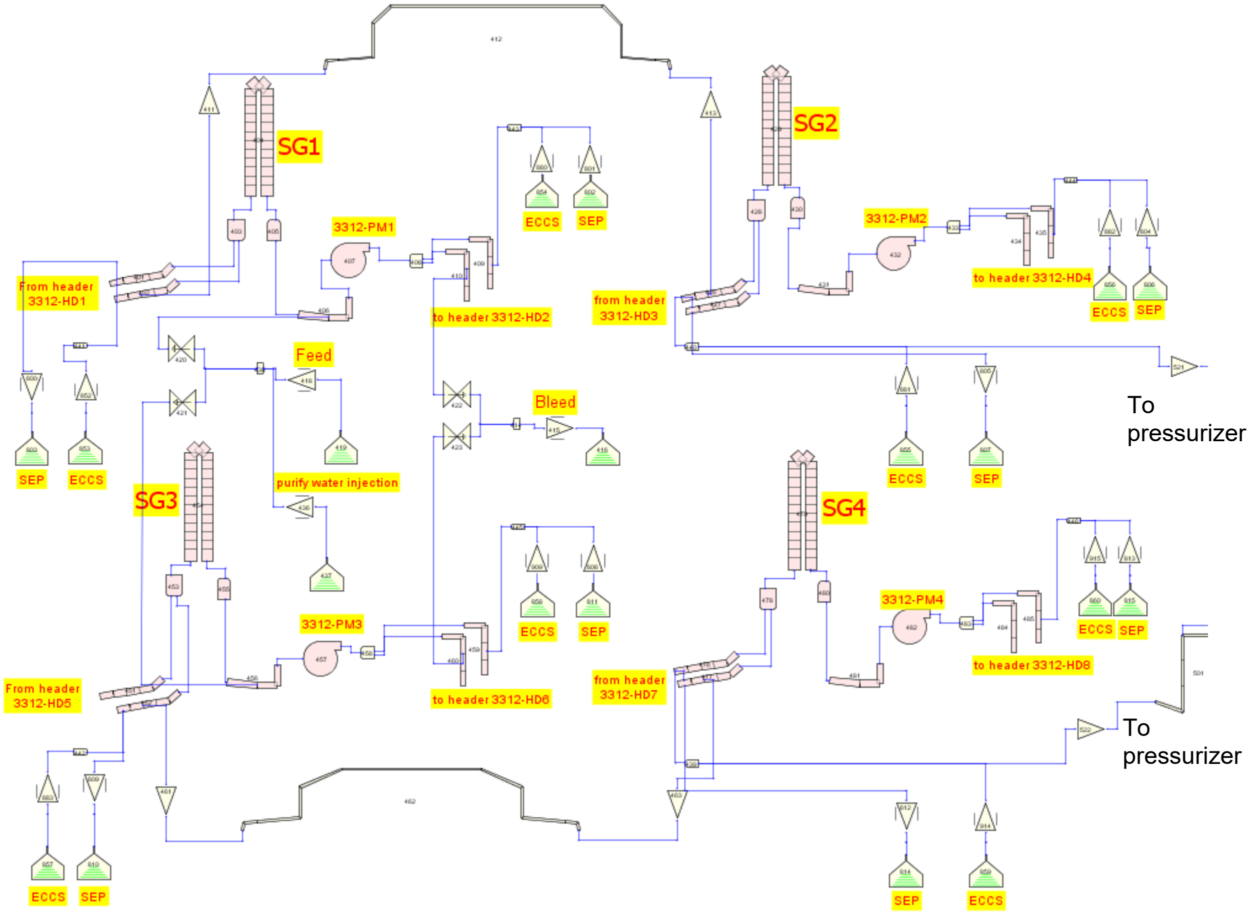

Figure 5 Nodalized Diagram of the PHTS in the CNE Model

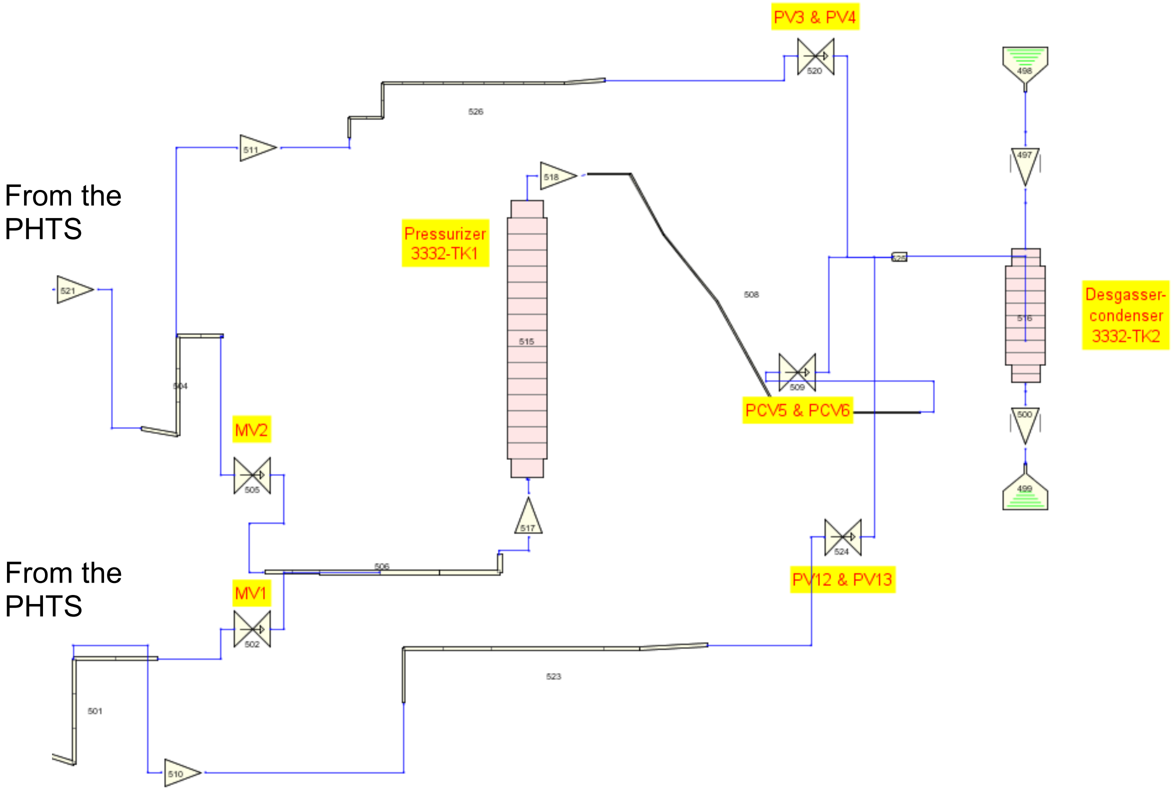

Figure 6 Nodalized Pressure Control System of the CNE Model

Figure 6 also shows the pipelines connecting the hot legs with the pressurizer and the degasser condenser. PHTS pressure is controlled by the pressurizer with the degasser-condenser. This maintains the pressure of two outlet headers at 100 bar. The implemented pressure control is rather simple but sufficient for plant control when required during steady state and transient operation. If the PHTS pressure increases further, the model contains two valves which represent the plant liquid relief valves (LRV) PV3, PV4, PV13 and PV12. Figure 6 shows the nodalized PHTS pressure control system of the CNE model.

The degasser-condenser level and pressure controls were simplified because it does not participate in design basis accidents (DBA).

2.3 <u>Secondary System</u>

The modeled secondary system consists of: the feedwater lines, SGs, main steam lines and SGs level control valves and condenser steam discharge valve (CSDV) as well as turbine governor valves (TGV). The main steam safety valve (MSSV) and atmospheric steam discharge valve (ASDV) are also included in the model. The turbine and the condenser are boundary conditions of constant pressure and do not participate during the evolution of the events to be simulated.

2.3.1 Steam Generators

The SGs generate steam from the heat exchanged with the PHTS. The steam is then transported to the turbine by means of the steam lines.

A schematic of a typical SG of a CANDU is shown in Figure 7. In the primary side, each SG has two inlet nozzles that distribute the coolant into a tube bundle of 3550 small Incoloy 800 tubes approximately 13 mm in diameter. In the secondary side, the feedwater (H_2O) enters the SGs at subcooled conditions, coming from 4 pre-heater stages, into an integrated preheater within the SG. The preheated water with temperature close to the saturation point mixes with the recirculation stream. The feedwater travels up the riser while heating and evaporating.

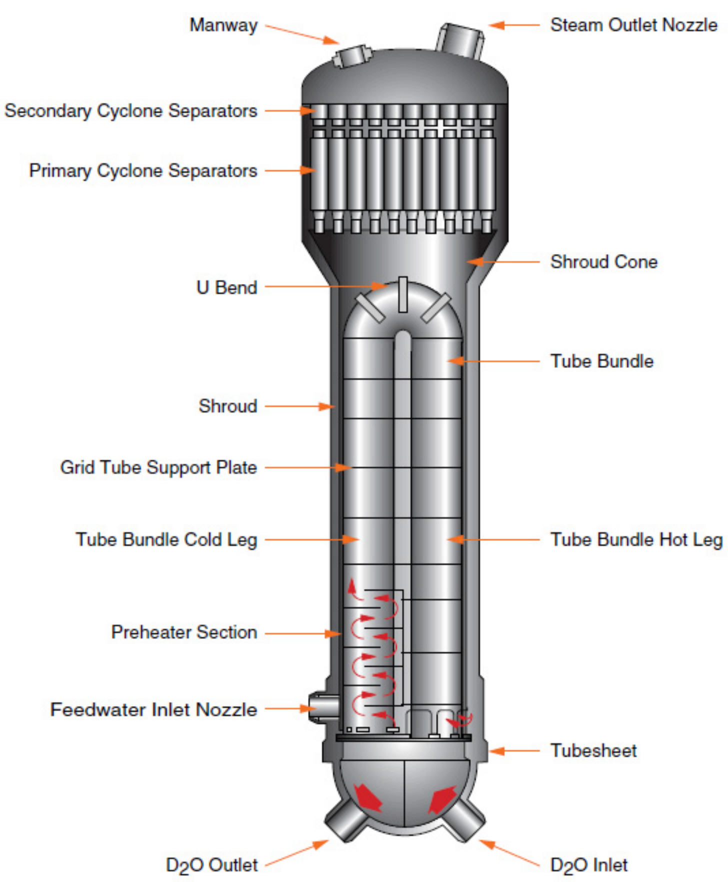

Figure 7 Diagram of a SG of a CANDU6 Type Reactor. Picture taken from reference [7]

The SG model is illustrated in Figure 8. The four SGs are identical and modeled following the recommendations given in reference [3]. The separator is modeled using a standard RELAP5 built-in separator with default options. The adjustment of the mass flow rates and pressure distributions in the SG is quite complex. Several restrictions must be satisfied simultaneously in 1D code to model a 3D phenomenon. The expected concentrated pressure losses and usage parameters of the separator and the pressure loss of the downcomer-riser connection were adjusted to match the SG recirculation ratio during normal operation.

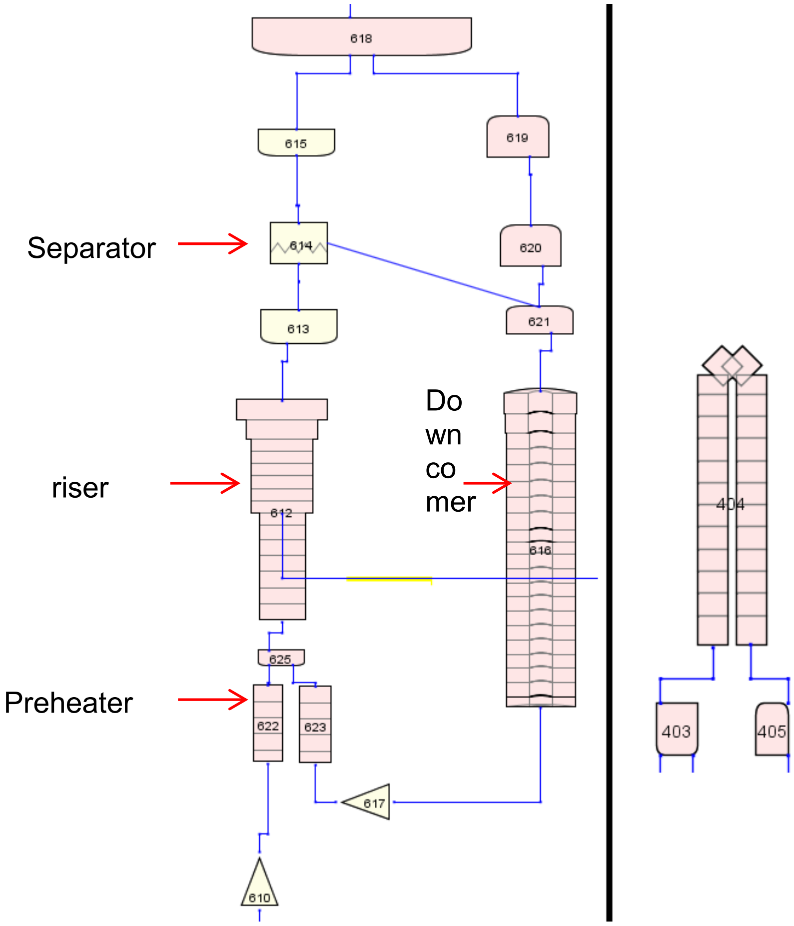

Figure 8 (Left) SG Nodalized Diagram of the Secondary Side (Right) Nodalized Diagram of the PHTS U-Tubes Bundle

The heat exchange area in the preheater and heat transfer conditions have been adjusted to meet an imposed constraint that feedwater should come out of the heater close to saturated conditions.

2.3.2 Feedwater System

The feedwater system consists of several pipelines which carry water from the condenser hotwell through the condensate extraction pump and from the preheating stages to the SGs. The feedwater enters the SGs through a set of control valves (three) in each feed line.

All pipelines between the feed pumps (designated P102 A/B/C) and SG inlet nozzles are modeled in RELAP5 as shown in Figure 9. The model was simplified to represent the set of three control valves with a single valve. The valve control was adapted to accomplish all required functions.

The two pumps of the auxiliary supply water system (designated P107 and P107A) were also modeled as a single pump. Its valves are included in the model.

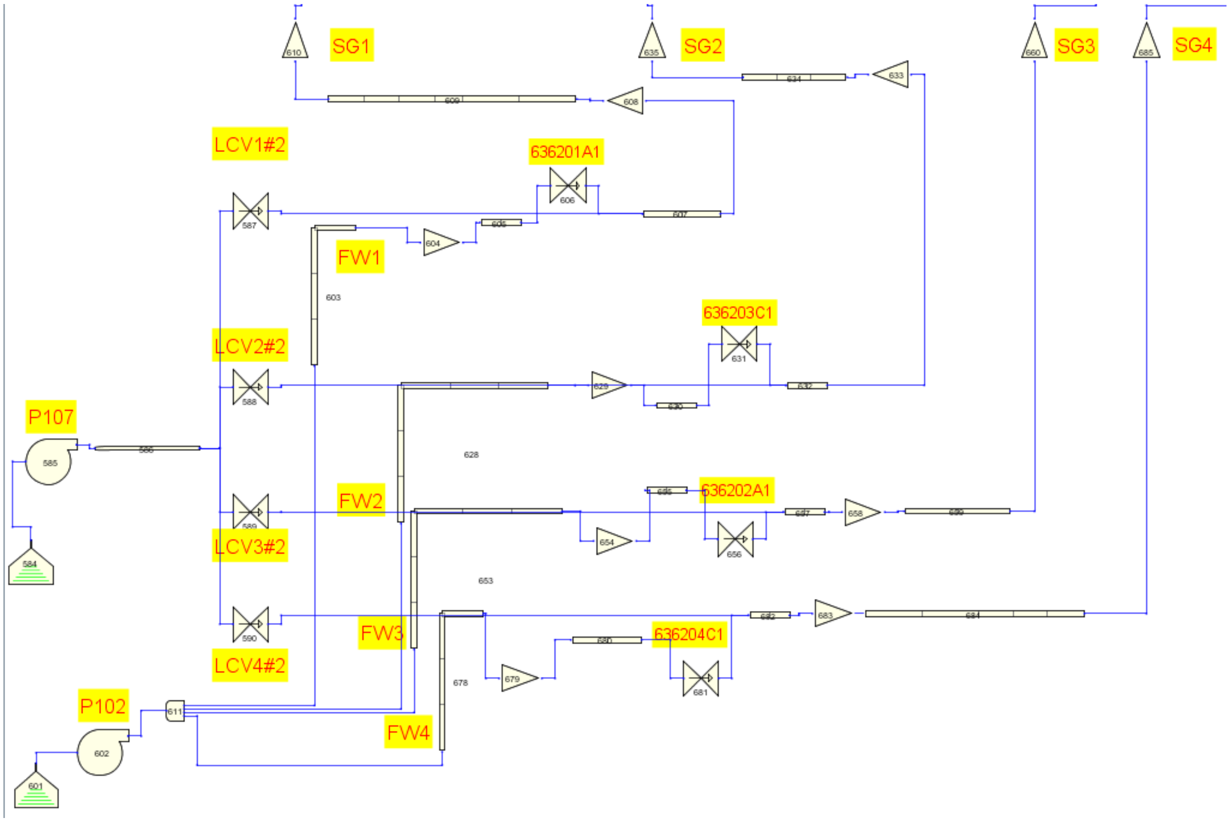

Figure 9 Nodalized Diagram of the Main and Auxiliary Feedwater Pipelines

The SGs level control logic is as closed as possible to that used by the process computer to determine control valve stem position [9].

2.3.3 Steam Line System

The steam is generated in the SGs and transported to the turbo-generator by means of four individual pipes that merge into a single manifold. From this manifold the steam feeds the turbine and the condenser, both of them are considered boundary conditions at constant pressure.

At a given location in the main steam lines, the MSSVs and the ASDVs are connected. Also, the CSDV and the TGVs are directly connected to the steam collector. It should be taken into account that the developed model does not consider the intake steam used in the feedwater preheating stages. A diagram of the steam lines in the model is shown in Figure 10.

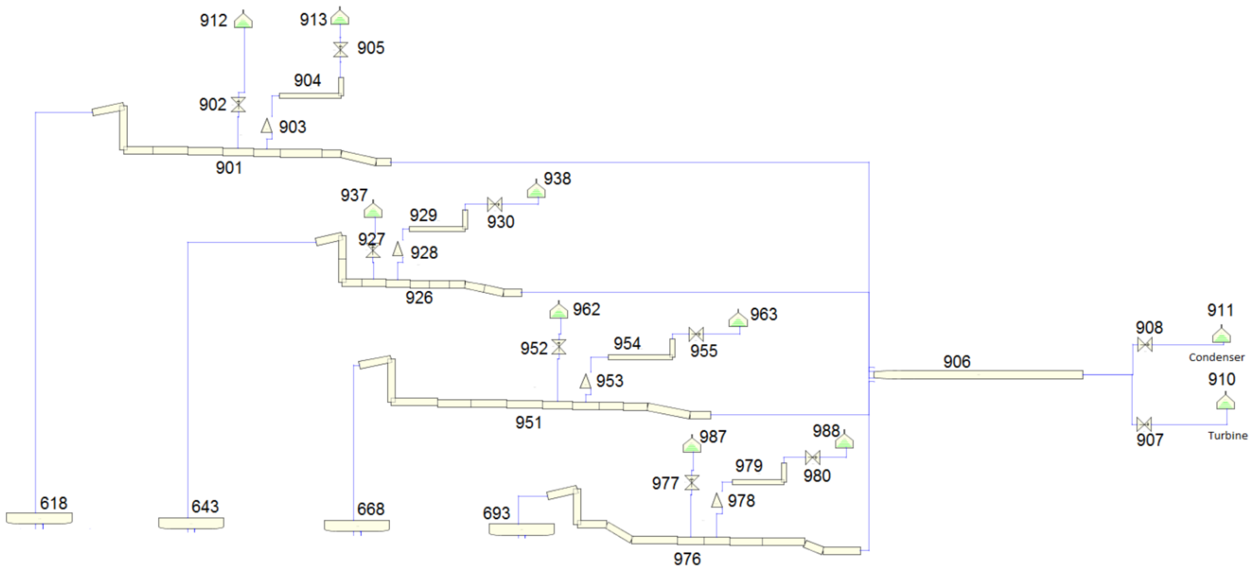

Figure 10 Steam Pipelines from SGs Domes to the Turbine

Each SG has 4 MSSVs that open in a staggered form depending on the steam pressure. In the model, this valve scheme has been simplified into a single valve. This valve has the same operational and functional characteristics as the valve group. Also, the ASDV and the CSDV have been modeled into one valve each. It should be noted that neither CSDV nor ASDV valves have safety functions. However, their control has been implemented since it will be required to simulate this particular event.

In addition, the pressure control logic and its action on ASDVs and CSDVs during cooldown mode has been implemented in the model. But the secondary system pressure control implemented in the model considers the action on the MSSV, ASDV and CSDVs valves in a simpler manner than the actual plant control [9].

2.4 Emergency Core Cooling System

The emergency core cooling system (ECCS) is one of the safety systems designed especially to maintain the fuel submerged in water. This prevents overheating and possible fuel failure, limiting the release of radioactive material. Although this system is not required during the simulated event, it is implemented in the model.

The ECCS injection into the PHTS has three phases of operation. These are namely, high, medium and low pressure injection of coolant to the core. The three stages were taken into account in the CNE model. A diagram of the high pressure ECCS model is presented in Figure 11.

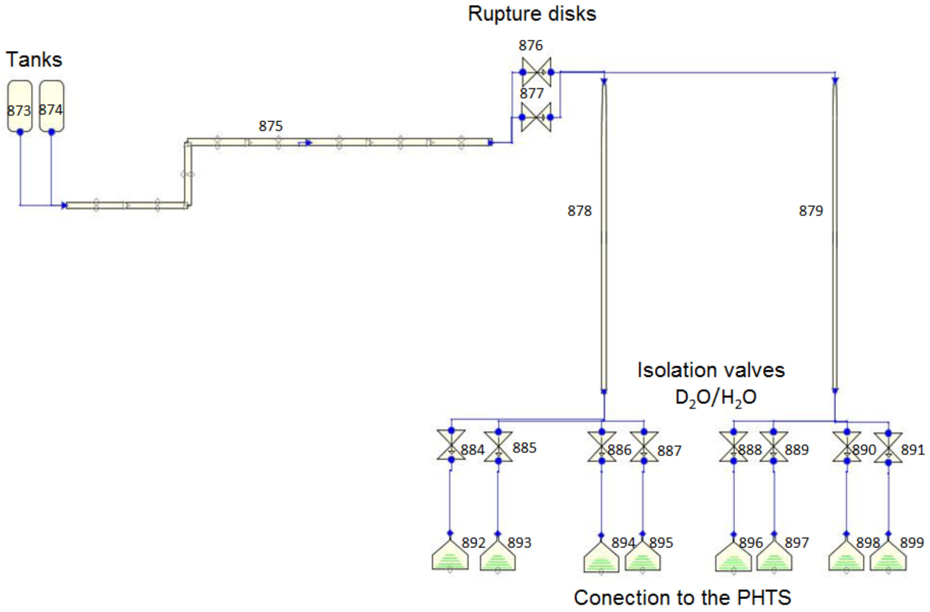

Figure 11 High Pressure ECCS Nodalized Diagram in RELAP5.

The high/medium pressure ECCS uses H_2O instead of D_2O so the RELAP5 implementation was:

- A separate model of ECC with light water was developed.
- This model injects into a RELAP5's component called time dependent volume (TMDPVOL) that follow the pressure of PHTS.
- Then, in the injection port a TMDPVOL and a RELAP5's component called time dependent junction TMDPJUN copy the pressure and mass flow of the separate model and injects heavy water to the primary system.

This modeling technique tries to maintain dynamic conditions.

The simplified low pressure ECCS model includes medium and low pressure injection valves. This model is shown in Figure 12.

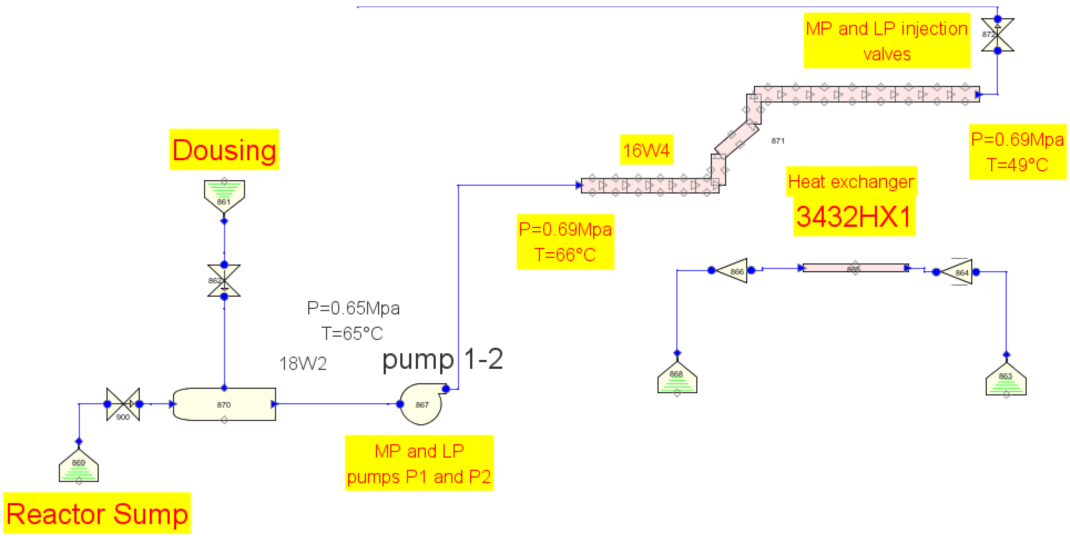

Figure 12 Diagram of the Medium and Low Pressure ECCS in RELAP5

2.5 Control System and Safety Trips

The model contains all the trip logic of shutdown system SDS N1 (safety control rods) and SDS N2 (gadolinium injection). The control logic includes most of the delays, lags and time constants consistent with the plant design. Also, in some cases, higher delay values were used as conservative boundary conditions.

The operation of all active components of the model are associated with class IV (normal power supply) and class III (diesel emergency power supply) as well as the highest delay times for starting each component. Therefore, the model can also contemplate the total loss of class IV, but for the event described in the paper, this is not required.

2.6 Other Plant Systems

The plant model contains other systems that are modeled as suitable boundary conditions or as hydraulic systems including:

A. Shutdown cooling system (SEP): this system is introduced as a boundary condition that injects and extracts coolant from the PHTS and passes it through heat exchangers.

B. Moderator system: this system is modeled with hydraulic components that represent calandria, pipes, pumps and heat exchangers. This model allows evaluation of temperature evolutions and moderator levels. It also allows simulation of interphase breaks (calandria tube).

C. Emergency water system (EWS): this system was modeled with few hydraulic components and a set of functions. The model includes injection to the SGs, the PHTS and the spray system. These related systems might be used in the very long term for a DBA or design extension conditions.

D. Dousing system: a simple model was built to study pressure increases in the reactor building and water inventory availability.

2.6.1 Pressure and Inventory Control System

The inventory control system (F&B) controls inventory in the PHTS through the pressurizer level control; it provides controlled flow to the degasser-condenser and a cooling flow to the main pumps' gland seal system.

To inject or extract heavy water from PHTS via the F&B during full power operation, the pressurizer level is used as a reference. During solid mode operation, the pressure signal of the outlet headers is used. The process control computer at the plant calculates the PHTS total mass, and then estimates the pressurizer water inventories and the pressurizer level. The model calculates the error between the estimated level and the real one and uses F&B valves to reach pressurizer level at the actual power.

3 SIMULATION OF A SPURIOUS TURBINE TRIP

On May 30th, 2007, a spurious signal produced a turbine trip at CNE. When the trip occurred, the NPP was disconnected from the national interconnected grid and a step-back to 49% of full power was automatically implemented. After 960 seconds of the event, SDS1 was triggered for SG1 low level due to a de-calibration of a feedwater valve. This latter event is not considered here.

With the turbine trip, the steam flow is interrupted and consequently, the secondary system pressure increases, reaching the ASDV opening setpoint. The reactor regulation system acts by rapidly reducing the power to approximately 49% of full power. To control the increase in pressure in the secondary system, the CSDVs are opened.

As the steam flow is reduced, the extractions used in the preheating of the feedwater are decreased and therefore the feedwater temperature drops during the event. The evolution of the feedwater temperature measured in the plant during the event was used as a boundary condition for the simulation in the model.

This work compares plant data after a spurious turbine trip to a simulation of the same transient event using the developed model. It should be noted that an equivalent comparison to that presented here is performed in Reference [8], but with CANDU reactor calculation code chains (CATHENA). Few results shown here will be a comparison between both codes since no relevant plant data is available.

3.1 Steady Sate and Model of the Spurious Turbine Trip

The simulation and evolution of an event depends on the initial plant conditions at the moment the failure occurs. Therefore, before comparing the model results with the plant measurements, some of the initial conditions must be verified with a simulation of the steady state model. When the values differ, analysis and verification of its relevance impact in the event dynamics shall be carried out.

Table 1 shows the comparison between the values obtained with the ARN model at steady state and the CNE values, together with the corresponding percentage difference.

Table 1 CNE Steady State Variables

		Plant	ARN Model	Difference (%)
Inlet headers temperature (°C)	IHD2	264.3	264.5	-0.1
	IHD4	264.1	264.7	-0.2
	IHD6	264.2	264.6	-0.1
	IHD6	264.8	264.9	0.0
Outlet header pressure (kg/cm^2)	OHD1	100.8	100.4	0.3
	OHD3	100.7	100.4	0.3
	OHD5	100.8	100.5	0.3
	OHD7	100.6	100.5	0.1

Table 1 CNE Steady State Variables (Continued)

		Plant	ARN Model	Difference (%)
Feedwater temperature (°C)	SG1	158.2	156.1	1.3
	SG2	157.9	156.1	1.2
	SG3	157.9	156.1	1.2
	SG4	157.9	156.1	1.2
SG pressure (kg/cm^2)	SG1	46.9	46.9	0.0
	SG2	46.7	46.8	-0.3
	SG3	46.8	46.9	-0.3
	SG4	46.8	46.9	-0.3
Feedwater mass flow (kg/s)	SG1	237.7	234.1	1.5
	SG2	238.7	233.2	2.3
	SG3	240.7	233.3	3.1
	SG4	239.2	233.9	2.2
Transfer power to the secondary system (MW)	SG1	504.0	500.5	0.7
	SG2	506.5	499.3	1.4
	SG3	510.6	498.9	2.3
	SG4	507.2	500.7	1.3
SG level (mm)	SG1	1089.5	1098.3	-0.8
	SG2	1083.3	1105.4	-2.0
	SG3	1097.8	1100.6	-0.3
	SG4	1102.2	1053.0	4.5
Pressurizer level (m)		9.4	10.0	-6.1
Opening of the TGVs (%)		50.2	55.3	-10.3

We see in Table 1 that the opening of TGVs is greater in the simulation than in plant measurements during steady state. This difference exists because the pre-heater stages model was simplified during the plant balance as well as the pressurizer level control model. These differences have several causes, but can easily be adjusted by changing the reference setpoint and allowing the F&B system to modify injection or extraction flow rates. On the other hand, these differences are not expected to impact the evolution of the event. The actual effect of these differences will be seen later.

3.2 Comparison of the Spurious Turbine Trip

Figure 13 shows the feedwater mass flow to one of the SGs (the other three lines are similar to the one presented here). Figure 14 shows the SG's level. It can be observed that due to the power decay, there is a sudden level drop caused by the two-phase disappearance (shrink) and decrease of the recirculation ratio. When the level drops, the feedwater mass flow also decreases. It is shown that the feedwater flows in the model are similar to plant data. In the case of SG level, the model predicts a lower reduction but then stabilizes.

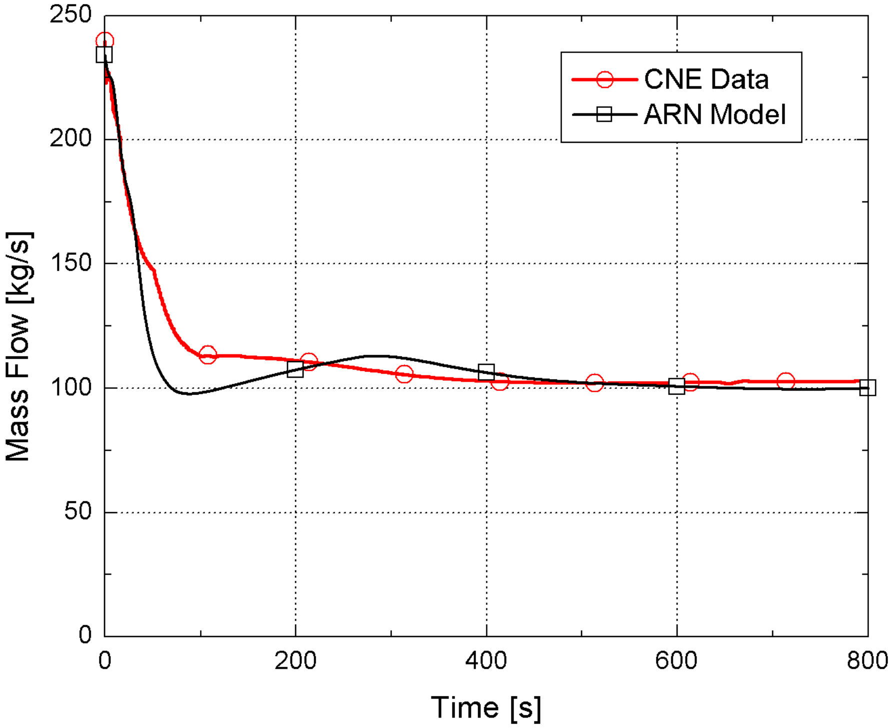

Figure 13 Comparison Between Mass Flow Rates of the ARN Plant Model and CNE Measurements for One of the Feedwater Lines

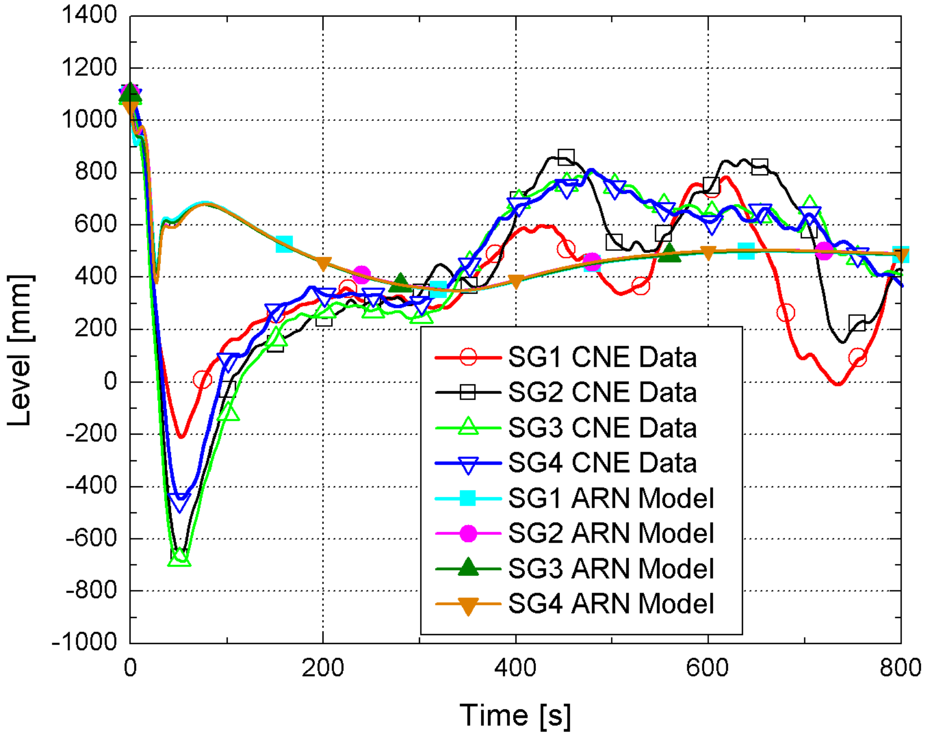

Figure 14 Levels of the 4 Steam Generators (SG1/2/3/4) Obtained from the ARN Model and the CNE Measurements

Since no relevant plant data is available, Figure 15 shows the steam mass flow to the condenser compared to the values of an operator model [8]. There is an acceptable agreement between the results. The steam mass flow includes the preheating extraction and the fraction that goes to the condenser.

Figure 16 shows the comparison between plant values and model values of the pressure in the secondary system of SG number 4. The results of the model show a higher maximum pressure. This was caused by a difference in ASDV valve opening values between the ARN model and the plant. In addition, due to the simplifications in the response times of the control logic, it is observed that the ARN model responds faster than expected.

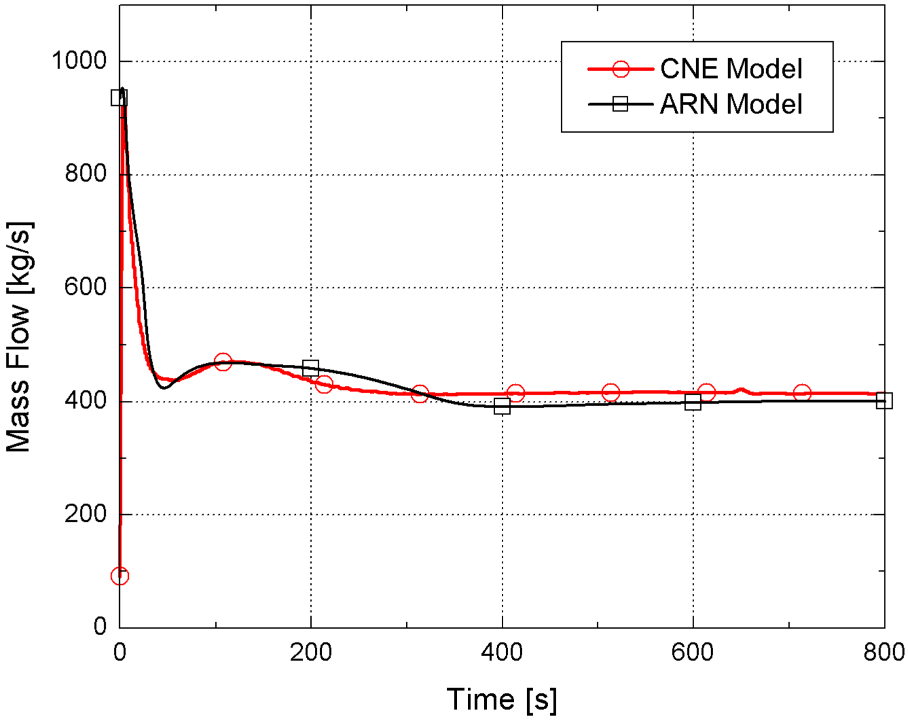

Figure 15 Steam Flow to the Condenser. For this comparison the results of the operator model were available [8] and the contributions of the flow to the condenser and the flow of preheaters were added

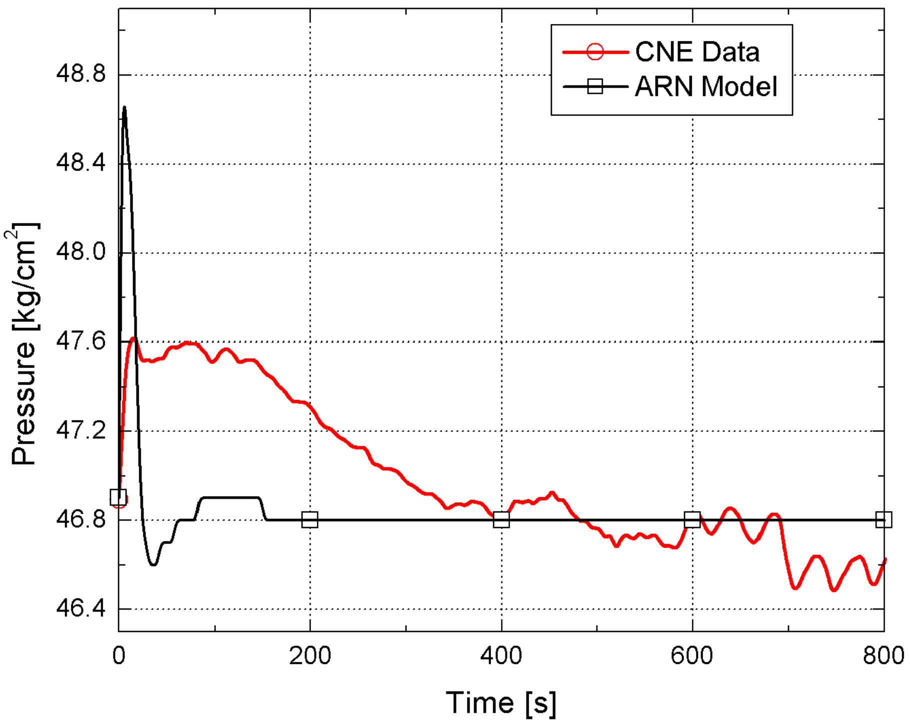

Figure 16 Comparison Between the Secondary System Pressure Measured in CNE and the Pressure Obtained by the ARN Model for SG4

When the power is reduced to 49% FP, the coolant loses part of its heat source by shrinking, decreasing the pressure and temperature, as seen in Figure 17, Figure 18 and Figure 19.

The pressure of the header HDR5 undergoes a rapid pressure decrease and then slowly recovers with the help of the pressurizer heaters in such a way that it is able to stabilize again at nominal values. The results of the model are consistent with the plant data for both pressure values and characteristic times. It can be seen that the differences are generally less than 3-4%. The trends of the temperature curves show good agreement, with differences of temperature smaller than 2ºC.

As the pressure and temperature of the PHTS decreases, a volume contraction of the coolant in the PHTS is expected. As a result, the pressurizer level should decrease depicts how its level changes. Figure 19 shows an important difference between the ARN (black curve) model and the plant data (red curve) of approximately 2 m. This is because of a difference in initial level (approximately 60 cm) between the ARN pressurizer model and the real plant value. If the curve obtained from the model is shifted to match the initial conditions, the corrected level of the pressurizer can be obtained. Although the difference between levels still exists, it should be noted that this difference does not affect the evolution of the transient. The discrepancy in pressurizer level may be due to a small temperature variation in the PHTS over the total volume (close to 120 m^3).

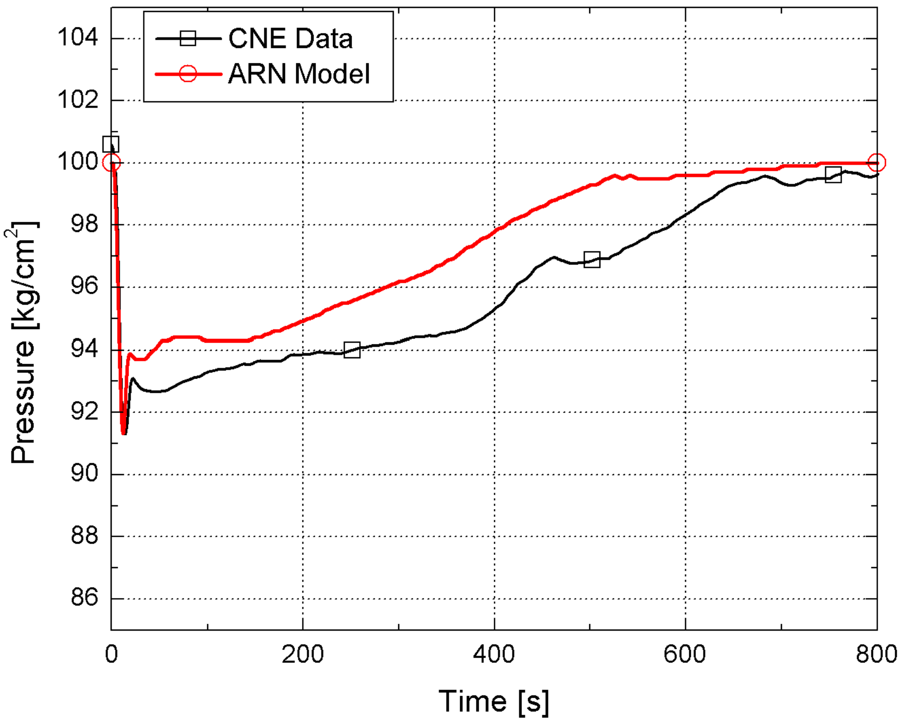

Figure 17 Comparison Between the Pressure Measured in the CNE (average) and the Pressure Obtained by the ARN Model for the HDR5

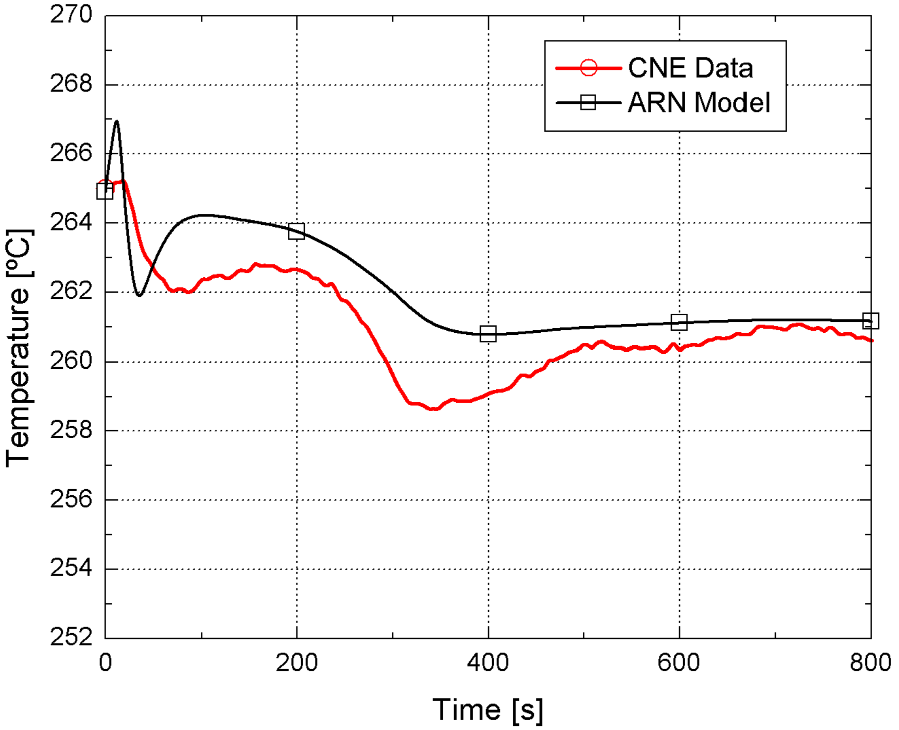

Figure 18 Comparison Between the Temperatures of the Header HDR8 and the Temperature Obtained by the ARN Model for the HDR8

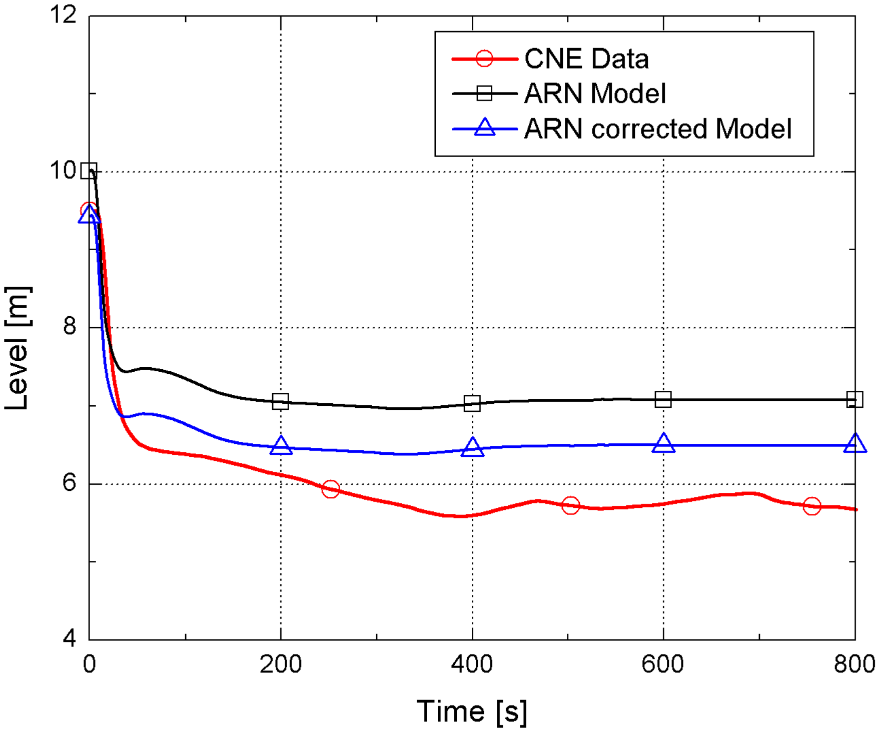

Figure 19 Comparison Between the Pressurizer Levels of CNE and the ARN Model. The resulting curve if the initial condition of the pressurizer level were changed is also shown

4 CONCLUSIONS

The ARN developed a thermal-hydraulic model with point kinetics in RELAP5 mod. 3.3 patch4 in order to simulate design basis accidents.

In this work, the results obtained from the developed model have been compared and validated with plant data before refurbishment. This comparison shows that the model acquires a steady state equivalent to the plant, while the observed deviations do not play an important role with respect to the model function. Also, the model correctly represents the steady state measured in the plant.

In order to validate the model and code, we simulated an event that occurred at the CNE in 2007 in which a spurious signal produced a turbine trip. The results obtained were analyzed and compared. The objective of this comparison was to verify that the important thermal-hydraulic plant parameters and the predictions of the model are consistent with each other.

It was observed that the results and trends were similar between the model and the plant data. In the case of the SGs' level, the model predicts a lower level drop than the plant and then stabilizes correctly. It is also observed that the ARN model responds faster than expected in relation to the pressure adjustment of the secondary system.

It was found that the results with respect to the PHTS are consistent, with differences less than 3-4% for the pressure and a maximum deviation of 2ºC for the temperature when compared with plant data. In general, the characteristic times and evolution of the PHTS are rather similar.

The evolution of some variables in regard to what has been observed at CNE has been verified, starting from equivalent and independent steady states. Also, the analysis presented in this work has contributed to the development, adjustment and improvement of the model, especially with respect to the control blocks.

5 REFERENCES

1. United States Nuclear Regulatory Commission, RELAP5/MOD3.3 Code manual, 1-8, by Information Services Laboratory Inc., Nuclear Safety Analysis Division, NUREG/CR-5535/Rev. 1, 2002.
2. NUREG/IA-0189 International Agreement Report Improvements of RELAP5/MOD3.2.2 Models for the CANDU Plant Analysis Prepared by B.D. Chung, WJ. Lee, Y.S. Bang (2000)
3. NUREG/IA-0239 International Agreement Report Development of Horizontal Off-Take Model for Application to Reactor Headers of CANDU Type Reactors Prepared by: Yong Jin Cho, Seung-Hoon Ahn, In-Goo Kim, Gyoo Dong Jeon (2010).
4. NUREG/IA-0247 International Agreement Report RELAP5 Simulation of Darlington Nuclear Generating Station Loss of Flow Event Prepared by: D. Naundorf, J. Yin, A. Petruzzi, A. Kovtonyuk (2011).
5. IAEA-TECDOC-1395. Intercomparison and Validation of computer codes for thermalhydraulic safety analysis of heavy water reactors, August 2004.
6. DM-18-63331 Heat transport system inventory control, PS-18-401.
7. https://canteach.candu.org
8. Ana L. Parrondo, Arnaldo V. Rabiti, Pablo U. Serrano, Silvana L. Calá, Hermes D. Damiani y Antoine Sablayrolles. “A complete Embalse nuclear power plant model for Cathena Code”. Argentinian association of computational mechanics, Bariloche, Argentina, September 2014

NRC FORM 335 **U.S. NUCLEAR REGULATORY COMMISSION**
(12-2010)
NRCMD 3.7

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
NUREG/IA-0489

2. TITLE AND SUBTITLE
RELAP5 Model of a CANDU-6 (Embalse) Nuclear Power Plant: Application to a Turbine Trip Event

3. DATE REPORT PUBLISHED

MONTH	YEAR
January	2019

4. FIN OR GRANT NUMBER

5. AUTHOR(S)
A.I. Lazarte, M.V. Guala

6. TYPE OF REPORT
Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)
Autoridad Regulatoria Nuclear (ARN)
Av. Del libertador 8250, CABA C1429PNB

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)
Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES
K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

As part of the Embalse Nuclear Power Plant (CANDU-6) refurbishment, the Nuclear Regulatory Authority of Argentina (ARN) has developed a full plant thermo-hydraulic model with point kinetics. Using the RELAP5 mod.3.3 Patch4 thermal-hydraulic code, the model simulates accidents within the design basis. This model is being used to perform an independent review of the safety analysis submitted by the designer and operator to the regulatory body. It would also be used as a tool for understanding plant dynamic behavior. The model developed by the ARN includes most of the hydraulic systems and control logic from the original design; the planned design changes for refurbishment are implemented in a more recent version of the model. To validate the model regarding its response to transients, a postulated initiating event (PIE) was modeled and simulated, and the results obtained were compared with recorded data of that particular event which took place in the NPP in 2007. The event consists of a spurious turbine trip leading to a rapid power reduction (stepback) to 49% of full power. This reduction was the only boundary condition imposed to the model. The purpose of this comparison is to verify consistency between important plant parameters, thermal-hydraulic model predictions and recorded data. Only the first 15 minutes of the transient were simulated. It is shown that the evolutions of some given variables are similar to those observed in the NPP. Also, the results obtained indicated that modifications should be performed to improve the model, especially regarding control components for getting better adjustment of plant data.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
Nuclear Regulatory Authority of Argentina(ARN)
Atmospheric steam discharge valves (ASDV)
Embalse Nuclear Power Plant (CNE)
Condenser steam discharge valve (CSDV)
Pressure and inventory control system (F&B)
Liquid relief valves (LRV)
Primary heat transport system (PHTS)

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page)
unclassified
(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS

NUREG/IA-0489

RELAP5 Model of a CANDU-6 (Embalse) Nuclear Power Plant: Application to a Turbine Trip Event

January 2019