NUREG/IA-0538

International Agreement Report

Natural Circulation Assessment of a PWR Loss of Off-site Power with RELAP5/MOD 3.2

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ABSTRACT

The work presented corresponds to the assessment of a PWR Loss of Off-Site Power with RELAP5 [1]. The RELAP5 input deck has been obtained from the existing RELAP5 input deck with some modifications. The calculated results were compared against the plant data. The results show that RELAP5 is capable of reproducing the phenomena with a good level of satisfaction. However, some discrepancies between the predicted variables and the plant data suggest further investigation of the RELAP model.

The simulations were prepared by IDOM Nuclear Services. A Spanish NPP provided the input model, the plant data, and transient available data and associated information.

The report is focused on the transient calculations and the modifications and conclusions obtained from them. The purpose of this work is to detect possible errors in the existing plant model and provide the modifications for the input model.

The work to which this report refers, falls within the framework of the CAMP/SPAIN Project. The objectives being those referred to in the CAMP Program itself, in particular, achievement of the capacity required to apply calculation tools supporting the quantitative analysis of transients and operation procedures, as well as joint participation in the international efforts aimed to validate and experience the use of appropriate calculation codes.

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EXECUTIVE SUMMARY

This report represents an assessment or application calculations submitted to fulfil the agreement for cooperation in thermal-hydraulic analysis between the Consejo de Seguridad Nuclear (CSN) and the U.S. Nuclear Regulatory Commission (NRC) in the form of the contribution to the NRC's Code Assessment and Management Program (CAMP) the main purpose of which is to validate the RELAP code.

The report presents the use of RELAP5/Mod 3.2 code to reproduce a transient occurred at a Spanish NPP, as a part of the CAMP program. The input deck has been obtained by the existing RELAP5 input deck, where, some modifications are introduced to better represent the plant behavior. The simulation results are compared against the plant data.

The transient to be simulated consists in a Loss of Off-Site Power, producing the reactor and turbine trips. Several actions were taken by the time of the event, including the operability of certain systems with operator manual actions that affected the evolution of the parameters analyzed.

The results show that RELAP5 is capable of reproducing the phenomena with a good level of acceptance. However, some discrepancies between the predicted variables and the plant data suggest further investigation of the RELAP model.

Concerning the transient calculation statistics, the simulation is run on an Intel® Xeon® CPU E3-1226 v3 @ 3.30 GHz processor. The 6000 s transient calculation required 376 CPU seconds with a 0.06 ratio of CPU time to real time.

The benchmarking with other LOOP studies resulted in the impossibility to correlate due to the differences between events.

The work to which this report refers falls within the framework of the CAMP/SPAIN Project. The objectives being those referred to in the CAMP Program itself, in particular, achievement of the capacity required to apply calculation tools supporting the quantitative analysis of transients and operation procedures, as well as joint participation in the international efforts aimed to validate and experience the use of appropriate calculation codes.

ACKNOWLEDGMENTS

The authors acknowledge the support from the Spanish Nuclear Power Plant and CSN within CAMP program for the help provided to perform the assessment. The authors also acknowledge IDOM for the financial support in the calculations.

ABBREVIATIONS AND ACRONYMS

AFW	Auxiliary Feed Water
CAMP	Code Assessment and Management Program
CPU	Central Processing Unit
CSN	Consejo Seguridad Nuclear
CVCS	Chemical and Volume Control System
I&C	Instrumentation and Control
LSTF	Large Scale Test Facility
LOOP	Loss of Off-site Power
MSR	Moisture Separator Reheater
NCFM	Natural Circulation Flow Map
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PORV	Pressurizer Operated Relief Valve
PWR	Pressurized Water Reactor
RELAP	Reactor Excursion and Leak Analysis Program
RCP	Reactor Coolant Pump
SG	Steam Generator
SIS	Safety Injection System
US NRC	United States Nuclear Regulatory Commission

1 INTRODUCTION

The RELAP5 thermal-hydraulic safety analysis code is a best-estimate system code developed by the US NRC for light water reactors thermal-hydraulic analysis. The first RELAP5 version was released in 1979 and has evolved continuously, through the development of new models and improvement of existing ones. The RELAP5 computer code is one of the most used system codes in the international community.

This report represents an assessment or application calculations submitted to fulfil the agreement for cooperation in thermal-hydraulic analysis between the Consejo de Seguridad Nuclear (CSN) and the U.S. Nuclear Regulatory Commission (NRC) in the form of the contribution to the NRC's Code Assessment and Management Program (CAMP) the main purpose of which is to validate the RELAP code.

This work presents the assessment of the Loss of Off-Site Power with RELAP5 of a Spanish NPP. The RELAP5 input deck has been obtained by the existing RELAP5 input deck with some modifications.

First, a description of the plant is done to understand the different systems and equipment inside the reference plant. Following this part, a detailed plant nodalization of the model in RELAP5 is shown, with the corresponding auxiliary and safety systems.

Then, a comparison between plant data and RELAP5 data is done for the most important parameters, in order to show the capability of the model to reproduce this type of transient. Finally, a sensibility analysis of the decay heat model is presented.

2 PLANT DESCRIPTION

The current relevant characteristics of the Nuclear Power Plant are summarized in the following table (see Table 2-1):

Table 2-1 Main Characteristics of the NPP

Number of Units	1
Design NSSS	Westinghouse
Coolant Loops	3
Thermal Power (MWth)	2940
Gross Electric Power (MWe)	1087
Cycle Length (months)	18

The Nuclear Steam Supply System (NSSS) consists of a reactor and three closed coolant loops. Each primary loop is composed by a hot leg, steam generator, intermediate leg, reactor coolant pump and a cold leg that connects to the reactor. An electrical heater pressurizer is connected through a surge line to the hot legs of one loop. Chemical and Volume Control System (CVCS) and the Residual Heat Removal System (RHRS) are also connected to the coolant loops.

The rated power of the reactor is 2940 MWth, electrical output being 1087 MWe. The reactor core is made up of Zircaloy-cladding slightly enriched uranium dioxide fuel rods. The core is housed in a reactor vessel in which pressurized light water flows, acting as moderator and coolant.

3 PLANT NODALIZATION

3.1 RELAP 5 Input Model

The simulation with RELAP5 was performed with a model that has grown over the years with different plant modifications. The base model consists of 541 control volumes, 4615 connections and 12 heat structures. The thermo-hydraulic and I&C simulation includes:

The primary circuit (as seen in Figure 3-1):

- Reactor vessel with a downcomer and two-channel core.
- Detailed primary loops and RCPs.
- Pressurizer with surge line, heaters, spray lines, two pressurizer operated relief valves (PORVs) and three pressurizer safety valves.
- Auxiliary systems: CVCS (Chemical and volume control system)
- Safeguard Systems: Safety Injection (High Pressure, Low pressure and Accumulators)
- Primary side of the Steam Generators consists of inlet and outlet plenums, tube sheet and the U-tube bundle represented by a single pipe.

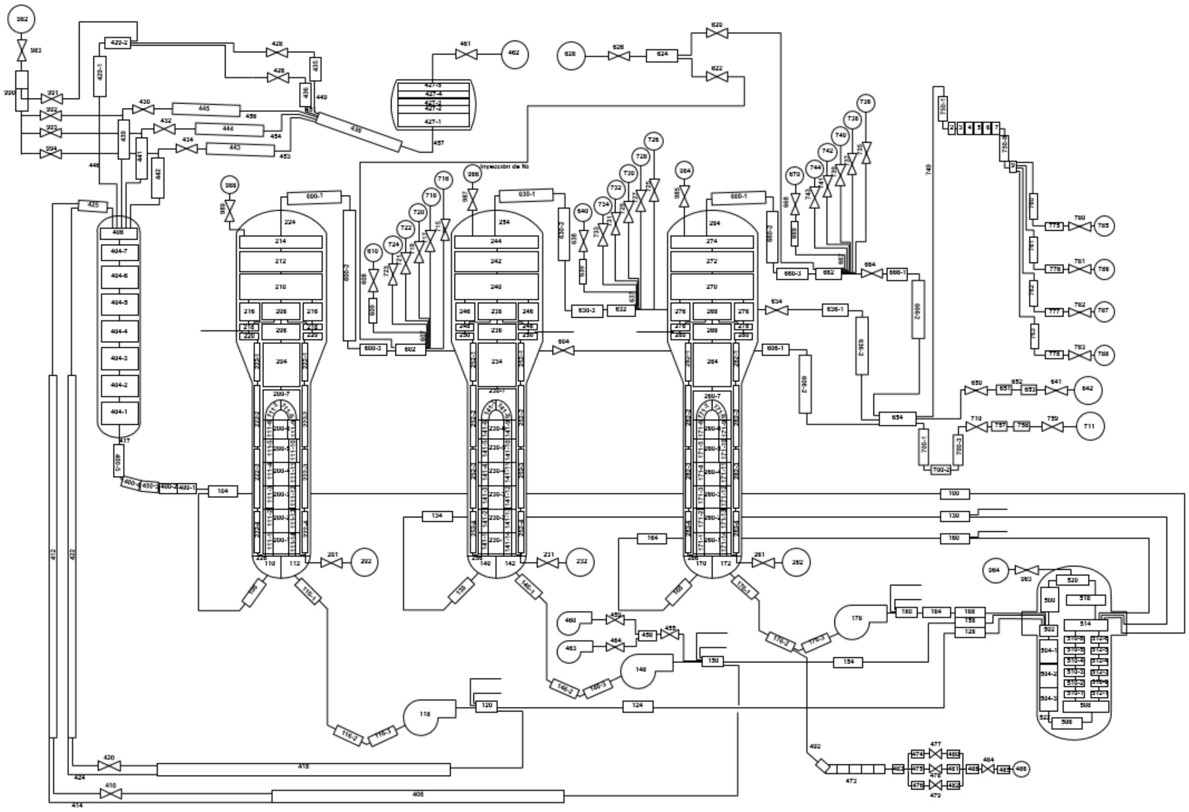

Figure 3-1 Primary Circuit

The secondary side (as seen in Figure 3-2):

- Steam Generators secondary side.
- Main Feedwater system.
- Main Steam System (with relief and safety valves).
- Main turbine and turbine bypass system.
- Safeguard Systems: Auxiliary Feedwater system.

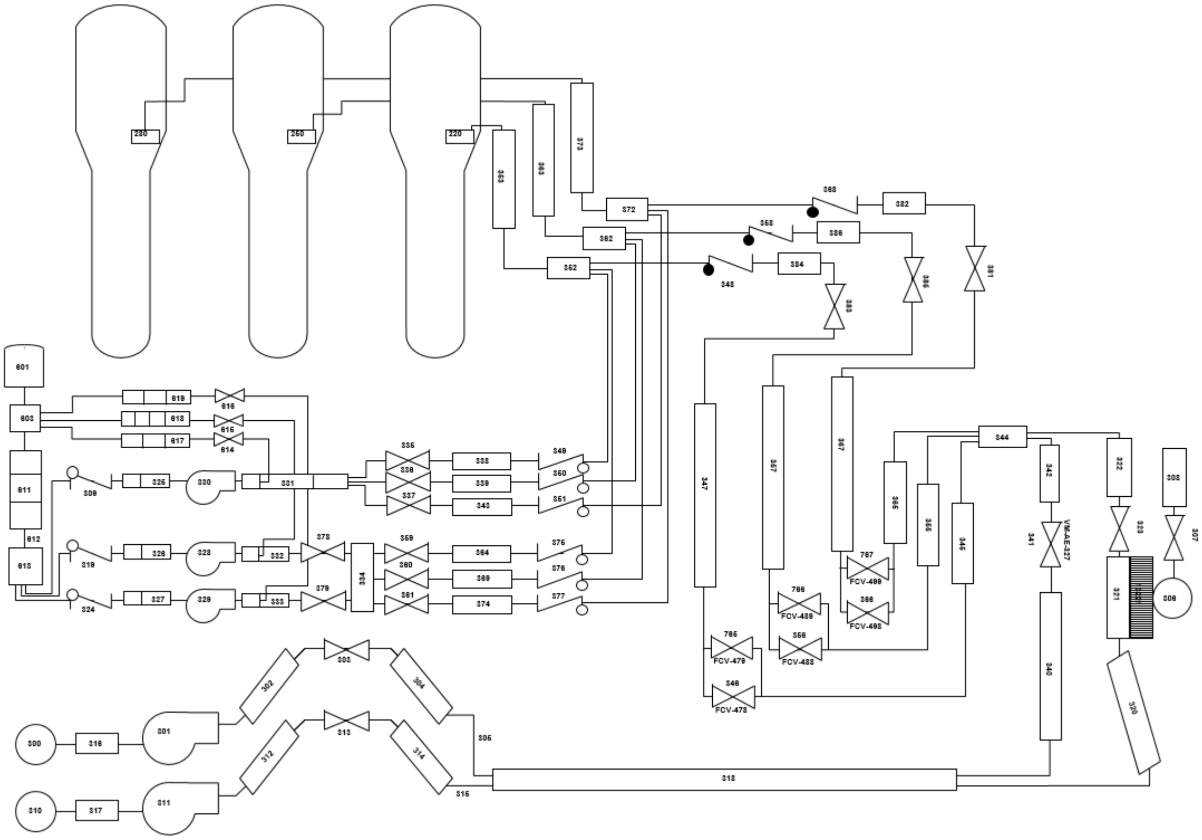

Figure 3-2 Secondary Circuit

The auxiliary feedwater turbopump nodalization (as seen in Figure 3-3).

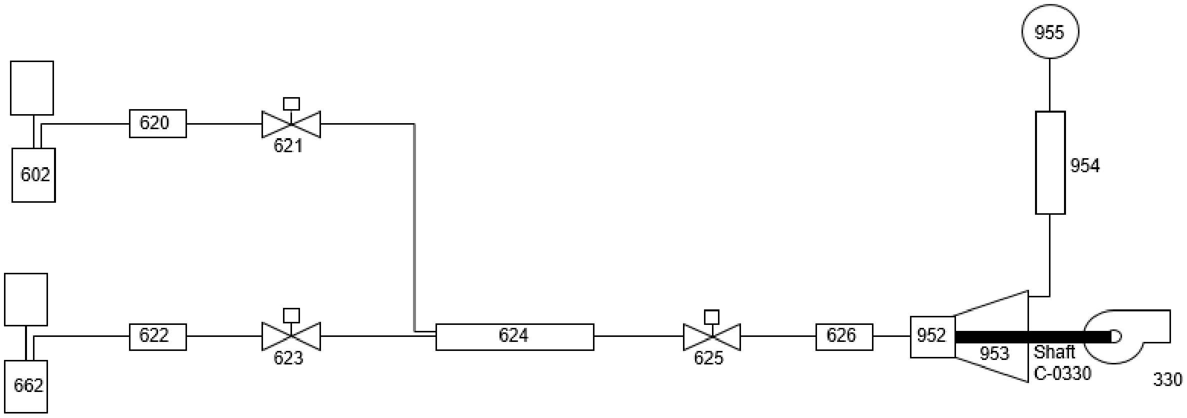

Figure 3-3 Auxiliary Feed Water System

The safety injection system includes (as seen in Figure 3-4):

- High Pressure Injection System
- Low Pressure Injection System
- Accumulators
- Sumps
- Refueling Water Storage Tank

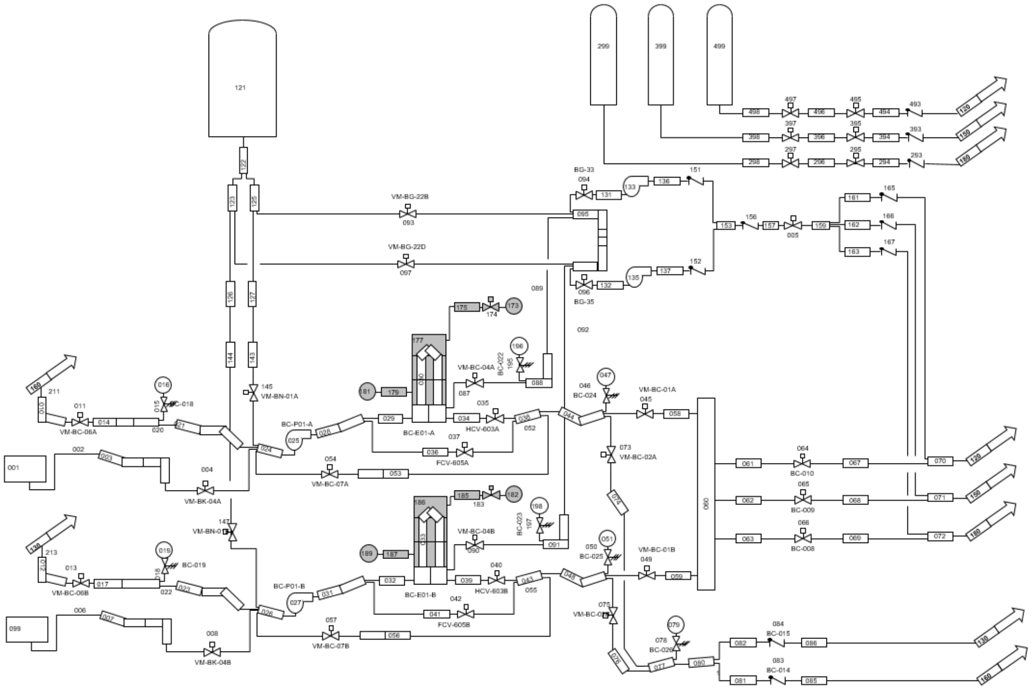

Figure 3-4 Safety Injection System

The CVCS system includes (as seen in Figure 3-5):

- Charging pump
- Discharge lines
- Boric acid tank
- Demineralized water tank

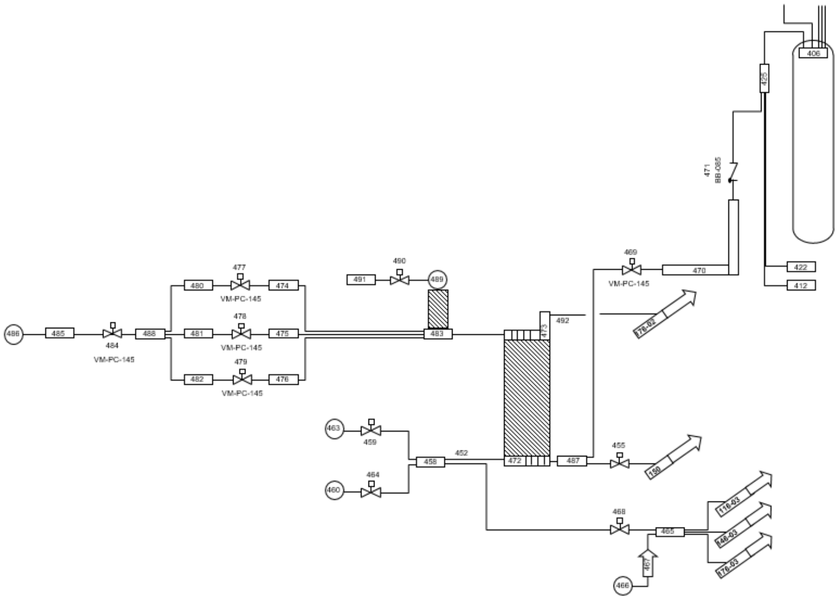

Figure 3-5 Control Volume Control System

4 PLANT TRANSIENT

4.1 Plant Transient Description

U.S NRC defined the loss of offsite power as an unavailability of all normal electrical equipment with most of the safety electrical equipment powered by diesel generators. The corrective actions are the reactor trip and residual heat removal using emergency core cooling system and auxiliary feedwater system.

The transient analysed occurred when the plant was operating at 100% power. The loss of offsite power produced the coast-down of reactor pumps and the start-up of the diesel generator. Then, a reactor trip occurred on RCP low-low speed, followed by the turbine trip. Steam dump to the condenser is fully open and normally feedwater is isolated. SGs are fed by means of the auxiliary feedwater.

During the first instants of the transient, the pressure in the primary system decreased due to the reactor trip. Then, it is increased due to the PZR heaters intervention and the charging pump. Pressure is established by means of the auxiliary sprays because the PZR sprays were not available due to the trip of RCPs.

The pressure of the secondary system was reduced by means of the steam dump. This procedure is done by the plant operators to start again the RCPs. The transient is analysed until the start-up of the first RCP.

4.2 Initial and Boundary Conditions

The comparison of the initial and boundary conditions between the model of RELAP5 and the reference plant is done in to see the acceptability of these conditions. The following table (see Table 4-1) shows initial and boundary conditions at the beginning of the simulation.

Table 4-1 Initial and Boundary Conditions of the Reference Plant and RELAP5 Model

Parameter (unit)	Reference PWR	RELAP/MOD3.2
	Value	
Core power (MW)	2938.6	2941.0
Pressurizer pressure (MPA)	15.712	15.7
Pressurizer level (%)	57.5	57.0
Average RCS temperature 1 (K)	581.14	580.95
Average RCS temperature 2 (K)	580.08	580.92
Average RCS temperature 3 (K)	581.11	580.95
Cold leg temperature 1 (K)	563.63	563.08
Cold leg temperature 2 (K)	562.02	563.01
Cold leg temperature 3 (K)	562.57	563.08
Hot leg temperature 1 (K)	598.65	598.82
Hot leg temperature 2 (K)	598.15	598.82
Hot leg temperature 3 (K)	599.65	598.82
Steam Generator pressure 1 (MPa)	6.386	6.42

Steam Generator pressure 2 (MPa)	6.372	6.42
Steam Generator pressure 3 (MPa)	6.343	6.42
Steam flow 1 (kg/s)	551.32	541.94
Steam flow 2 (kg/s)	511.85	539.25
Steam flow 3 (kg/s)	540.14	538.37
Main feedwater temperature (K)	497.15	498.15

The initial and boundary conditions have been obtained from the I&C department directly from the NPP plant process computer. The output of this data was printed every second, which means that the resolution is suitable for the assessment.

It is important to point out that some data needed to analyse the transient, is not available. The mass flow in the three legs of the primary is not available, but it is approximated by means of the RCP coast-down. The data from the auxiliary sprays in the pressurizer is not available, so an approximation of the mass flow is done to match with the pressurizer pressure data.

4.3 Analysis of the Model for the Natural Circulation

An analysis of the natural circulation is done to proof that the plant is capable of extracting the heat with natural circulation. It will be important to analyze the natural circulation map of the model [2]. Measured values of core inlet flow rate (G, kg/s), core power (P, MW), primary system fluid mass inventory (RM, kg), and net volume of the primary system (V= const., m3) have been used for setting up the natural circulation flow map (NCFM). The diagram G/P versus RM/V has been preferred for the NCFM over other possible choices including non- dimensional quantities. The point of established quasi-steady natural circulation is compared to the experimental Natural Circulation Map. The model has good agreement with the experimental data, in particular with the experiment LSTF at 5% (see Figure 4-1).

Once more, the NCFM results in a very helpful tool to judge the natural circulation performance of different reactor types. Considering the characteristics and the way the flow map has been performed, other applications may regard support on scaling analysis, design of natural circulation system, and validation of computational tools.

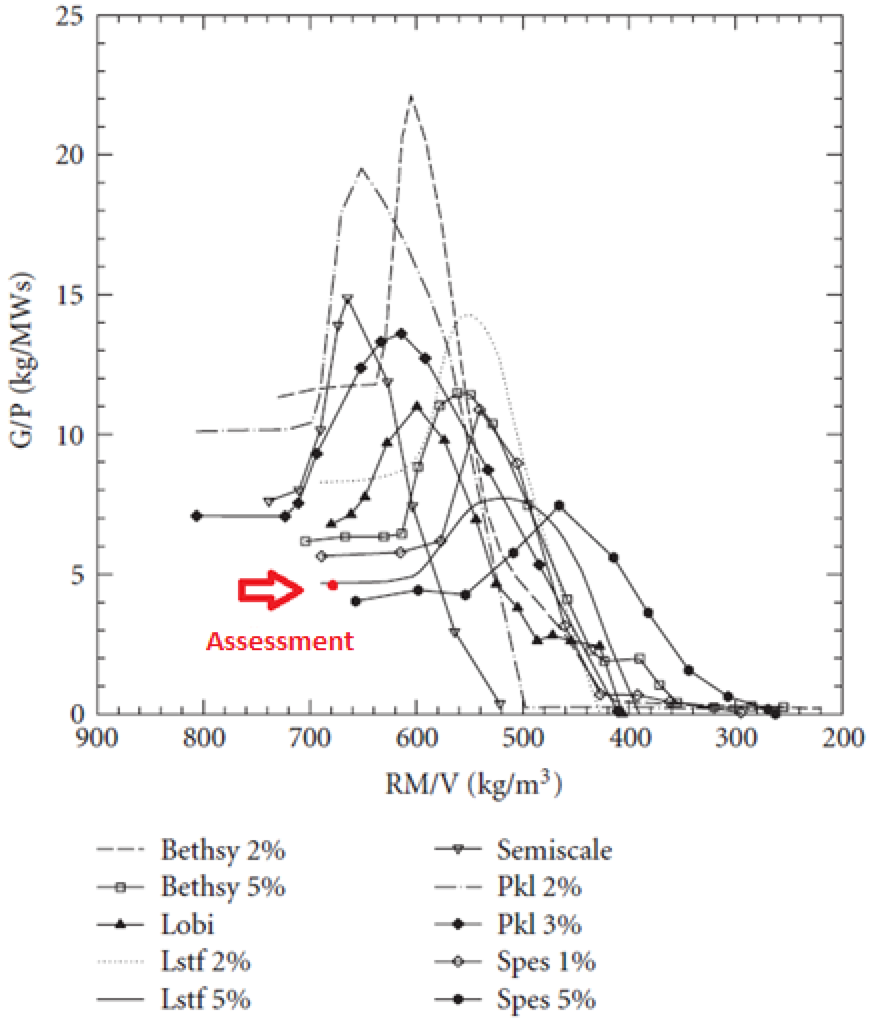

Figure 4-1 Natural Circulation Map

5 RESULTS

5.1 RELAP5/MOD3.2 Calculations

The RELAP5/MOD3.2 calculations data is compared against the plant data as shown in Figure 5-1 to Figure 5-14. In Figure 5-15, only the simulation results are presented due to the unavailability of plant data for those parameters.

The transient calculation is conducted by means of a previous restart calculation, starting with the steady state of the NPP. An intermediate calculation is performed with the steady-state option. This allows to reset the problem time to zero at the start of the transient calculation. The transient initial conditions were explained before. The time of the transient was 6000 s, where the first 3071 s is in steady state conditions.

The initiating event is a LOOP, which is implemented by a manual action in the input model that produces the LOOP at 3071 s. The manual actions of the operators are implemented in the following way:

- Regulation of the mass flow of the AFW system by creating 3 tables to apply a time dependent factor to the mass flow rate to simulate the operator manual control of the AFW makeup.
- Regulation of the turbopump by creating a table for the control logic of the velocity of the turbine to reproduce the regulation of the velocity done by the operation in the plant.
- Regulation of the valves that control the flow to the MSRs.
- Cooling of the secondary system by selecting the pressure mode and regulating the pressure by means of the collector pressure to reproduce the cooling down done by the operator of the primary to start the RCPs.
- Regulation of the auxiliary spray valves by creating a table for the control logic of the valves of the system to simulate the manual action of the operator to reduce the pressure of the primary system.

Some modifications are implemented to the input model because it has been seen that they represent better the thermohydraulic state of the NPP:

- Change in the regulation of the MFW valves closure during a reactor trip: the timing of the closure of the MFW valves was later than in the real plant, thus, a modification in the control system is been added in order to have a correct level in the SGs.
- Change in the calculation of the wide range level of SGs: the control variable of the wide range level of SGS was not completely correct, an adjustment is been done to match the plant data.
- Change in the seals flows of the RCPs: the mass flow injected to the seals was different in the model than in the plant data, an adjustment is been done to adjust this mismatch.
- Change in the control of the charge flow: the controller, which regulates the valve of the charging pump, did not have the correct parameters to match the plant data, thus, the main parameters of the controller are adjusted to have the correct behavior.
- Change in the decay heat power model: it has been noticed that the decay heat model ANS79-1 did not match correctly the results of the plant data; it is substituted for ANS79-3 decay heat model which gives better results as it can be seen in the §5.3 .

5.2 Results Discussion

The results are shown in Figure 5-1 to Figure 5-14, they show the calculation results compared to the associated plant instrumentation for the most important variables. In Figure 5-15, only the simulation results are presented due to the unavailability of plant data for those parameters. There are some parameters that the associated calculations completely match, because they are implemented from plant data, for example the auxiliary feedwater flow to the three steam generators or the pressure of the three steam generators.

In the Figure 5-1, the Fission Power level is shown. As it can be seen, the plant data and the RELAP5 calculation match perfectly. At 3071 s, scram of the reactor is produced.

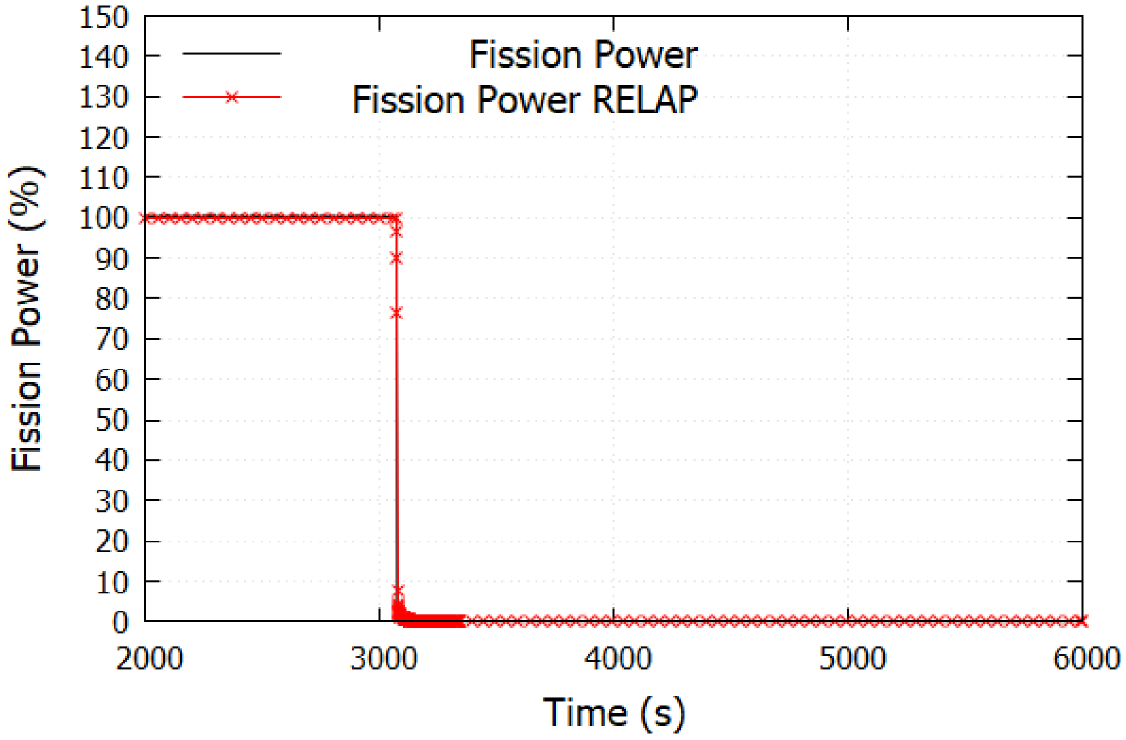

Figure 5-1 Fission Power Level of Data Recorded and RELAP

In the following figure (Figure 5-2) it can be seen the pressurizer level of the plant data and the RELAP5 calculation. The plotting lines are similar between each other, the little difference, it is because of the regulation of the auxiliary spray system, which data was not available, so, an approximation is done. Some differences are present in the PZR level because the charging flow was difficult to reproduce, due to no available data of the flow error controller.

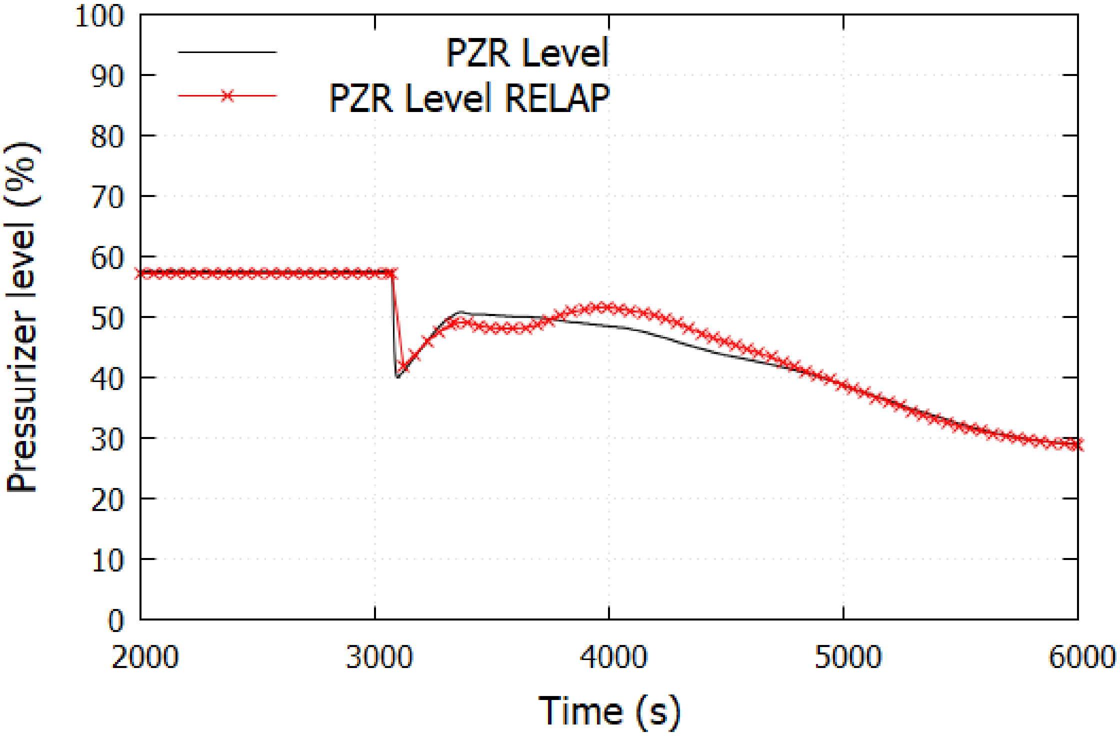

Figure 5-2 Pressurizer Level of Data Recorded and RELAP

In the Figure 5-3, the pressurizer pressure is presented. The main differences are produced due to the regulation of the auxiliary spray system, which is manually actuated by the operator, as it can be seen in the Figure 5-4. The plant data of this variable is not available, so an approximation is used to match the simulation with the plant data. Nevertheless, the simulation and the plant data match with an acceptable level of error.

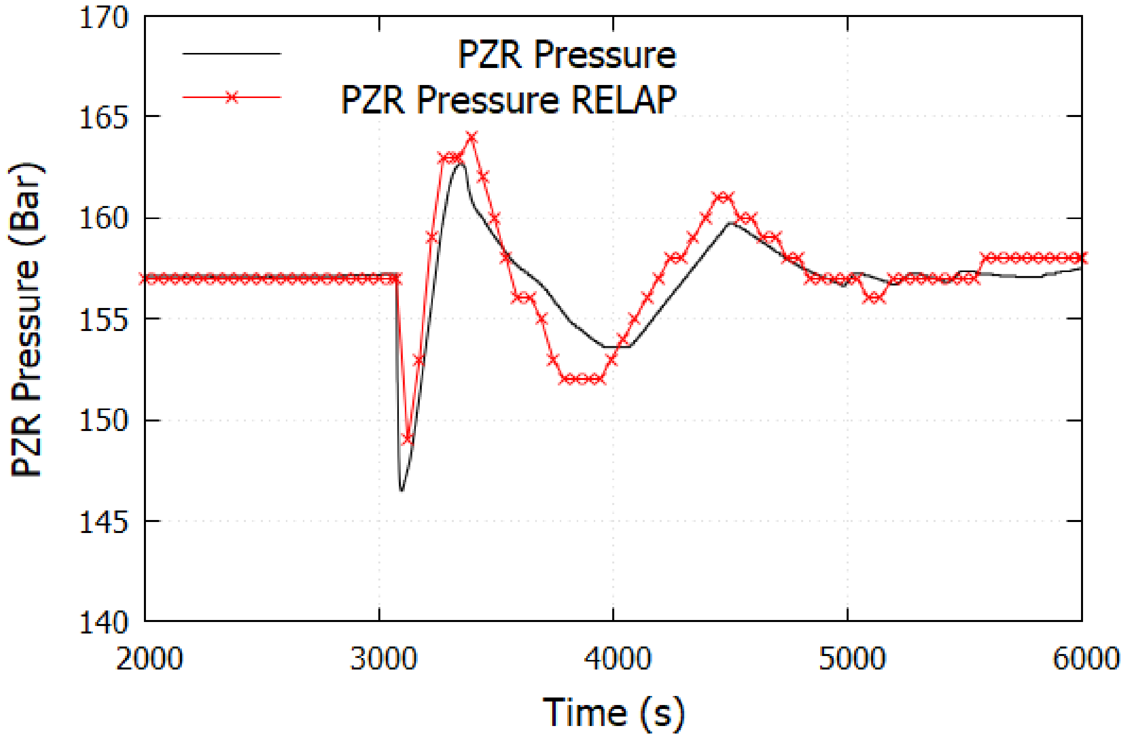

Figure 5-3 Pressurizer Pressure of Data Recorded and RELAP

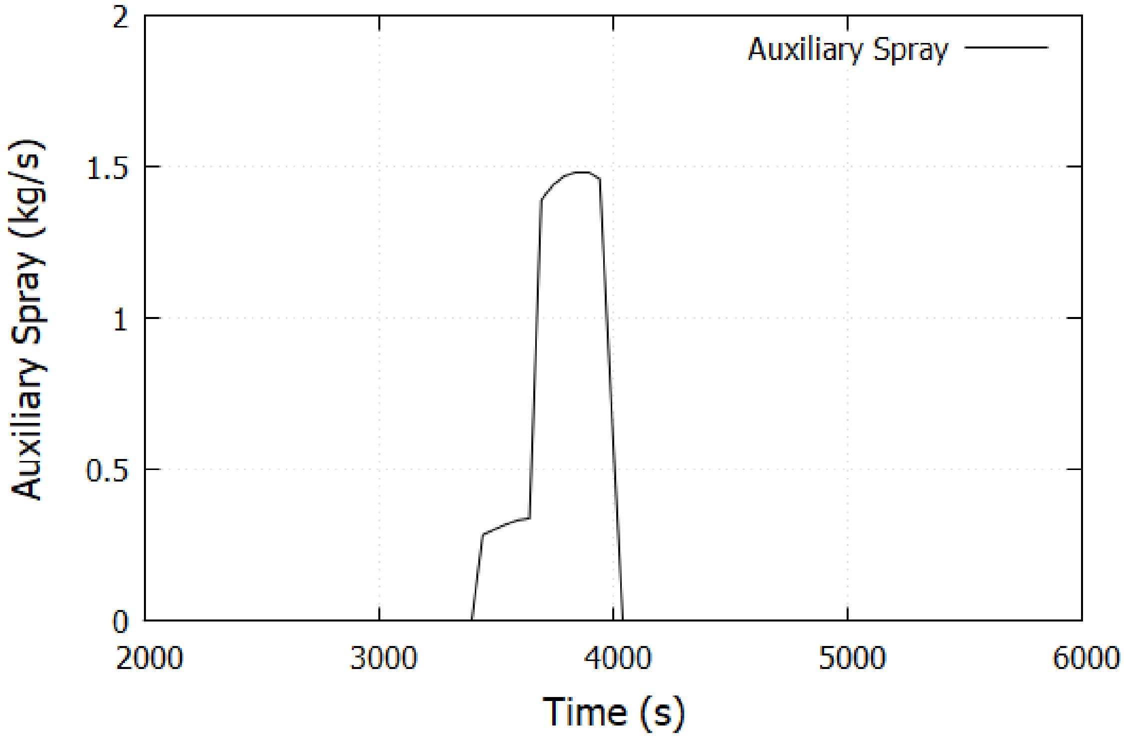

Figure 5-4 Auxiliary Sprays Mass Flow Rate in RELAP

In the Figure 5-5 to Figure 5-7, the hot leg and cold leg temperatures for the different loops of the plant are plotted. In the three cases, the temperatures match with an acceptable error.

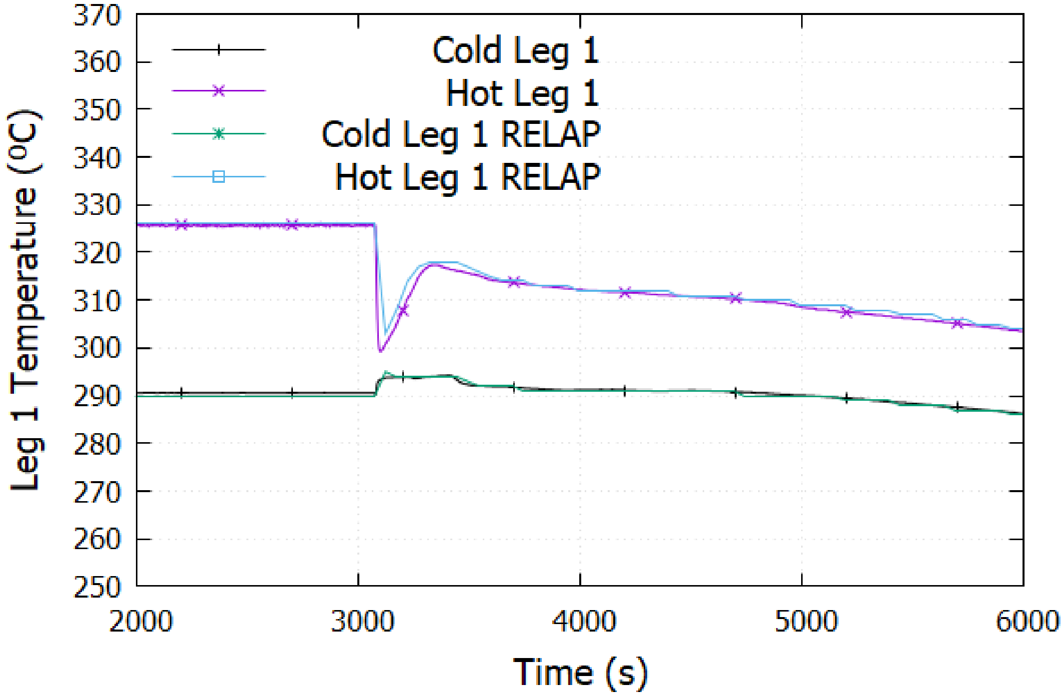

Figure 5-5 Hot/Cold Leg 1 Temperature of Data Recorded and RELAP

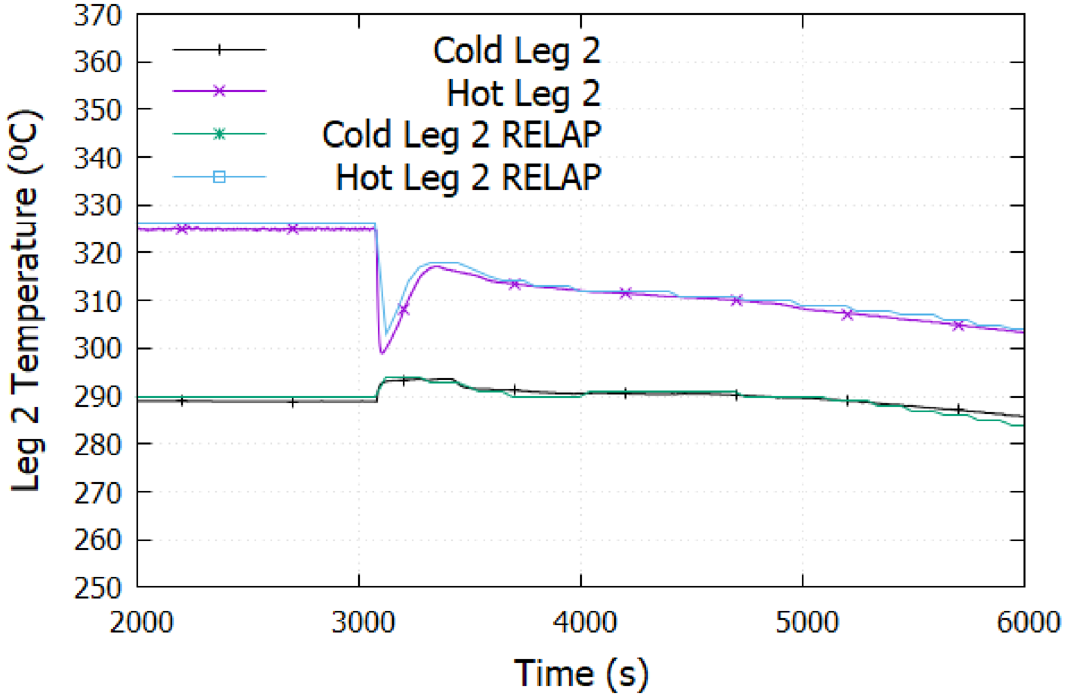

Figure 5-6 Hot/Cold Leg 2 Temperature of Data Recorded and RELAP

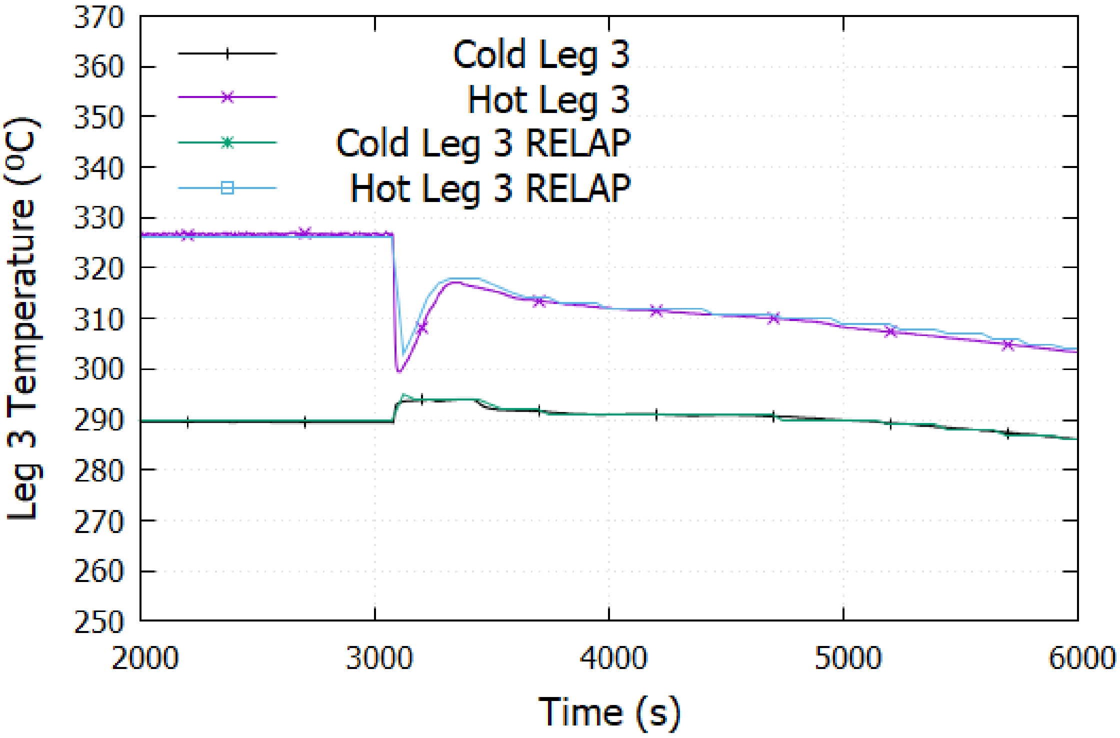

Figure 5-7 Hot/Cold Leg 3 Temperature of Data Recorded and RELAP

In the Figure 5-8, the simulation data is implemented based on the plant data. The cooling of the secondary system is done by an operator manual action by setting the pressure mode of the controller and regulating the pressure of the collector. This manual action is implemented by regulating the setpoint of the steam dump.

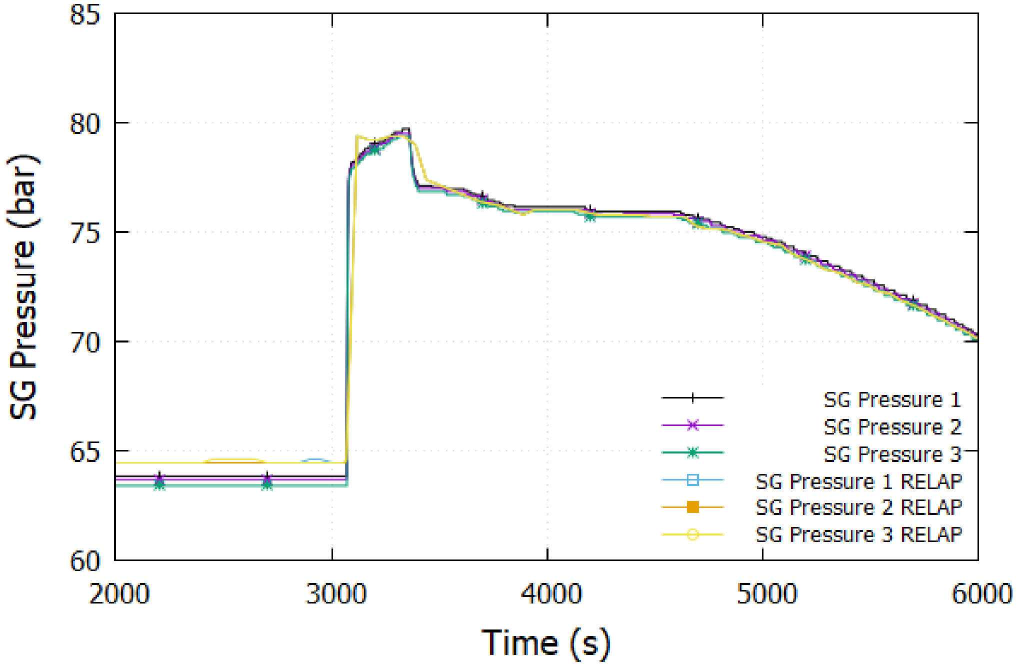

Figure 5-8 Steam Generators Pressure of Data Recorded and RELAP

In the Figure 5-9, the wide range level of the three SGs is shown. As it can be seen, it matches with an acceptable error the values of the plant data. The main reason is that the AFW mass flow is implemented in the simulation based on the plant data. This happens because the regulation of the AFW is a manual operation during the transient.

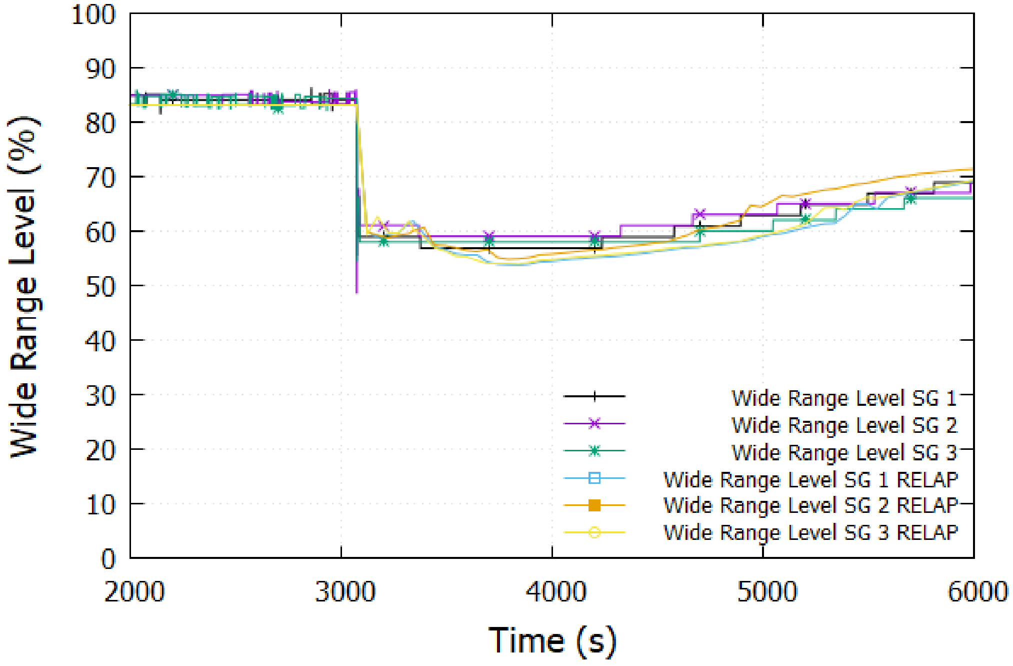

Figure 5-9 Wide Range Level of SGs of Data Recorded and RELAP

The Narrow Range Level is shown in the Figure 5-10. The simulation data matches perfectly the values of the plant data.

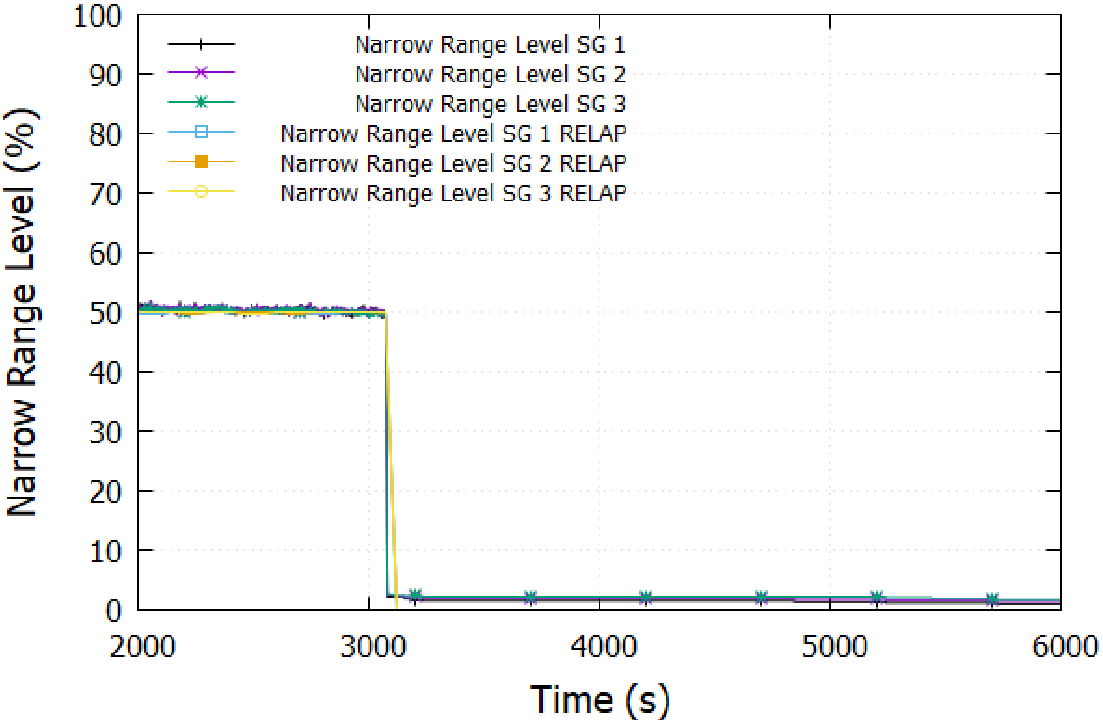

Figure 5-10 Narrow Range Level of SGs of Data Recorded and RELAP

In the Figure 5-11, the AFW flow is shown. During the transient, the plant data and RELAP data match perfectly because the regulation of the AFW is a manual action from the operator. The RELAP data is adjusted following the actions done by the operator to reproduce the correct transient. The peak around 4000 s in the three SGs of RELAP it is because of the difficulties in the regulation of the turbopump, but it has no impact on the results.

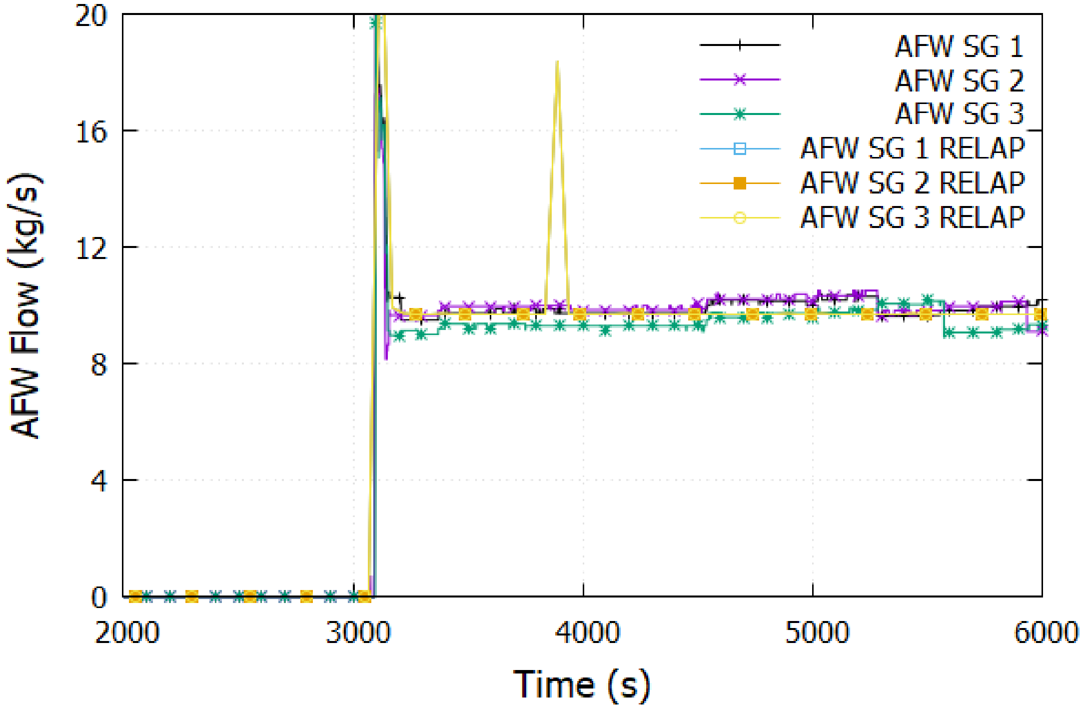

Figure 5-11 AFW Flow to SGs of Data Recorded and RELAP

In the Figure 5-12, the mass flow of the charging pump is shown. Without having recent data from the charging flow controller, a modification of the initial controller is done in order to reproduce the correct mass flow injected.

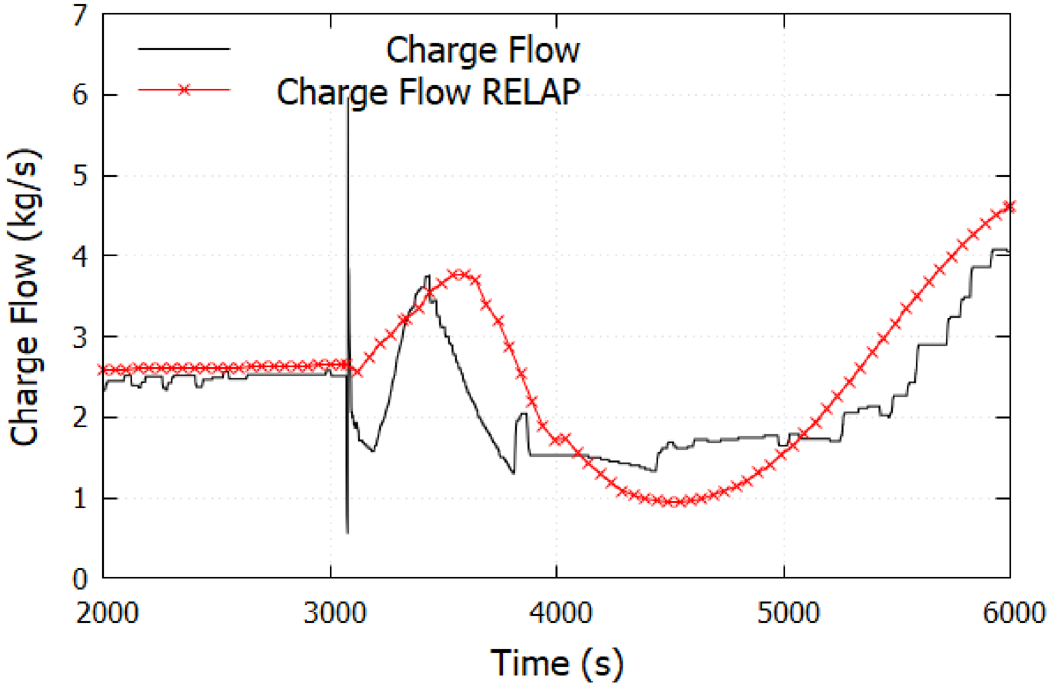

Figure 5-12 Charge Flow of Data Recorded and RELAP

The seals flow is adjusted by modifying the corresponding valve to match the mass flow rate of the plant data. Small differences are appreciated in the Figure 5-13, which are almost negligible and acceptable.

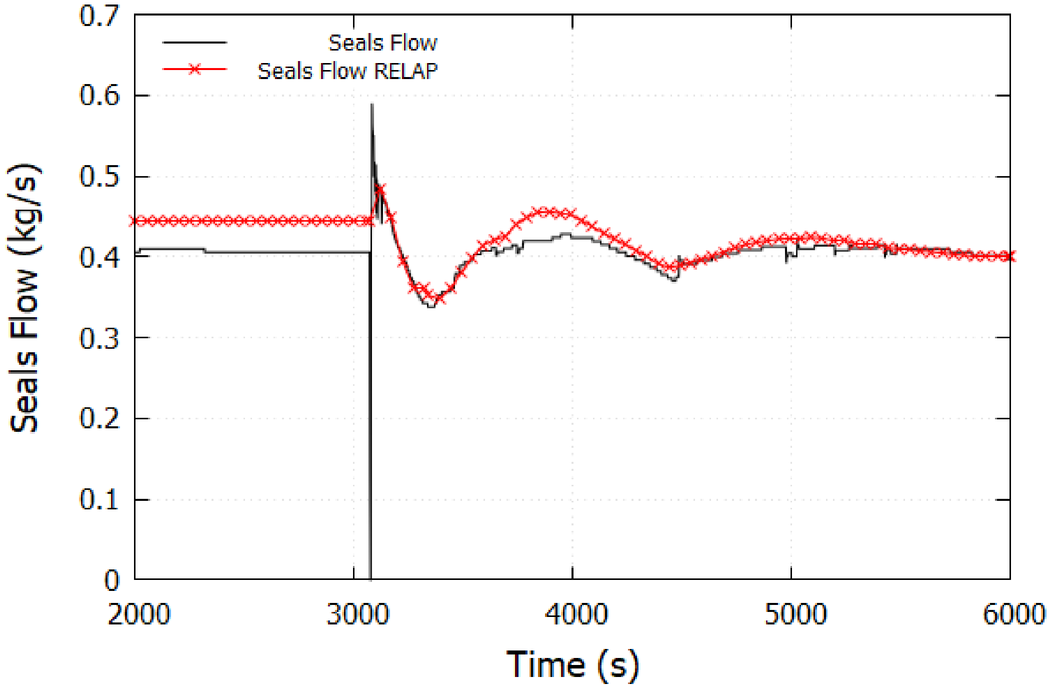

Figure 5-13 Seals Flow of Data Recorded and RELAP

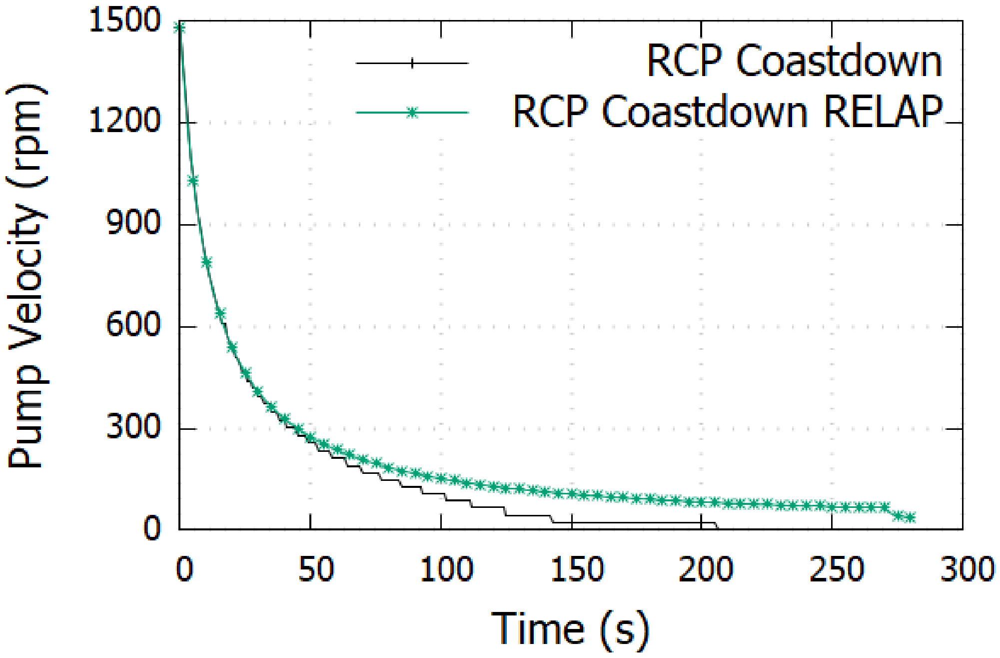

Figure 5-14 RCP Coast Down of Data Recorded and RELAP

The comparison of the RCP coast-down between plant data and the RELAP model (Figure 5-14) is shown during first 200 seconds of the coast-down. The differences in the velocities are negligible, so there is not impact on the main parameters of the transient. The analysis of the sensibility of the coast-down is not relevant. For the primary mass flow, no results of plant data are available (Figure 5-15), only RELAP data is presented.

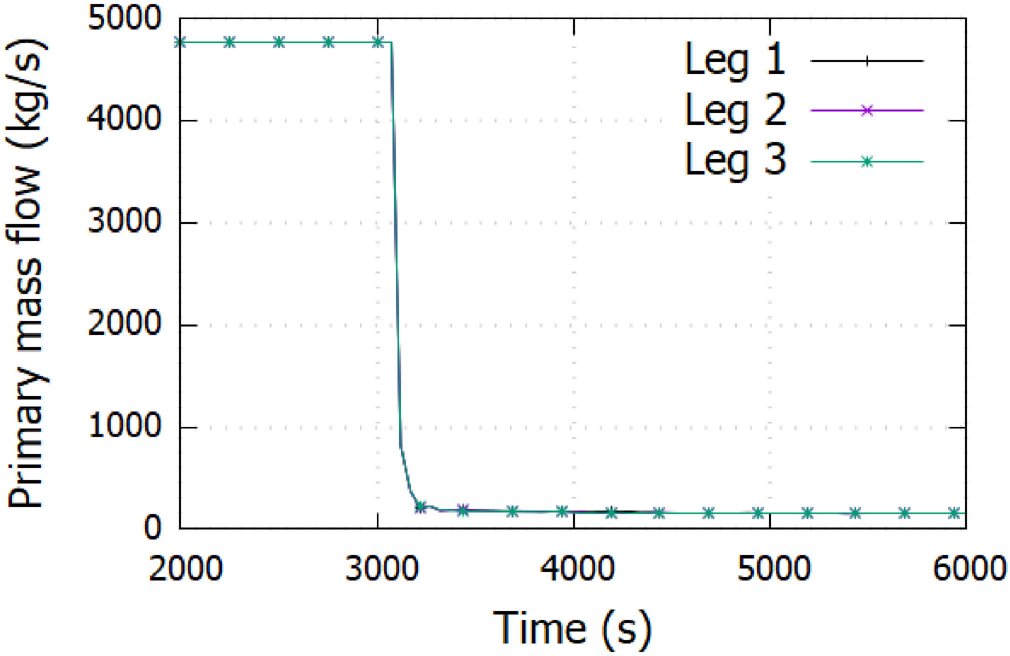

Figure 5-15 Primary Mass Flow of RELAP Model

5.3 Sensibility Analysis

In the last chapter, a sensibility analysis of different parameters will be done to see how it affects to the most important parameters of the model and analyze which model is more suitable for this transient.

5.3.1 Decay Heat Power Model

The decay heat power model is selected because after running several tests of the transients with different decay heat model, it has been observed that the decay heat power has a significant impact in different parameters, such as temperatures of the hot leg or the level in the SGs.

Initially, the model used in the input model provided was ANS79-1, but after some analysis, it has been concluded that ANS79-3 was more suitable for this particular case. In the following figures, an analysis of the different parameters depending on the decay heat model is done.

In the Figure 5-16 and Figure 5-17, the decay heat power and the temperature in the third leg for different fission product data are presented. Where the ANS73 specifies the Proposed 1973 ANS Standard data, ANS79-1 specifies the 1979 Standard data for U-235, and ANS79-3 specifies the 1979 ANS Standard data for the three isotopes, U-235, U-238 and Np-239. The one used in the model is the ANS79-3 because it represents better the relevant parameters of plant data. This model, it is characterized by having three isotopes with 23 different decay groups of fission products for each of them.

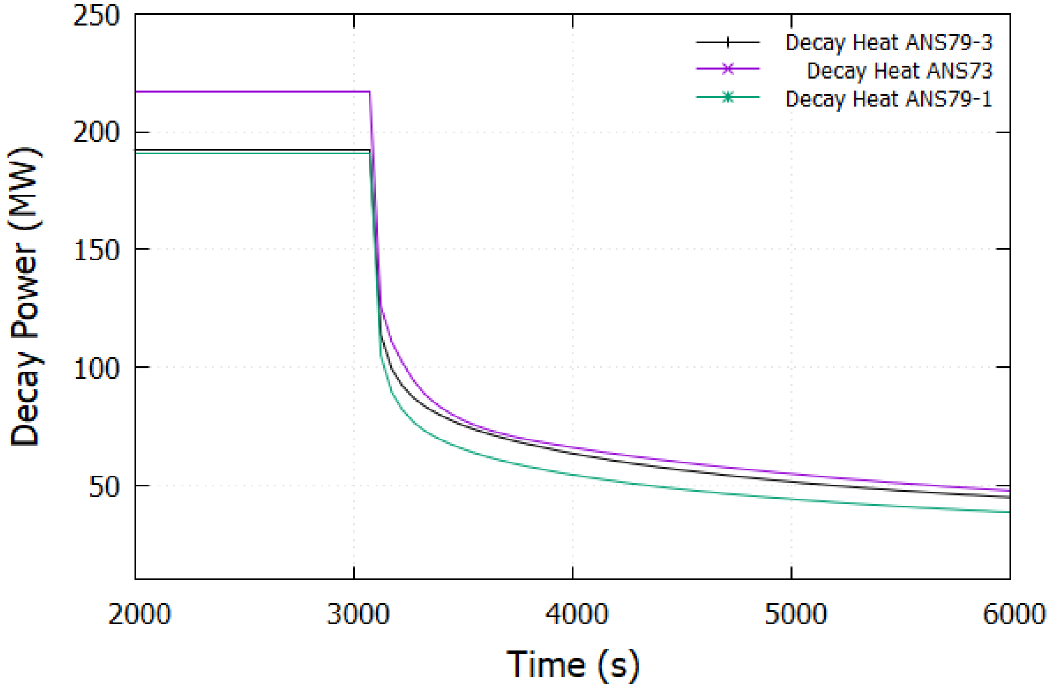

Figure 5-16 Decay Heat for Different Fission Product Data in RELAP

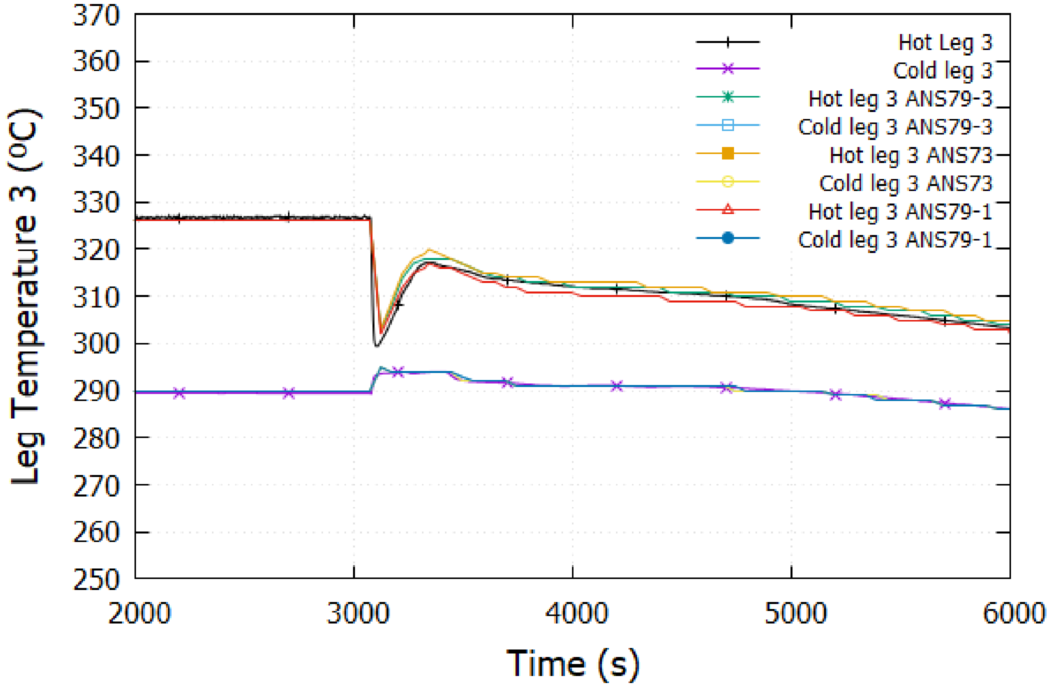

Figure 5-17 Leg Temperature 3 for Different Fission Product Data in RELAP

In the Figure 5-18 and Figure 5-19, the pressurizer pressure and the pressurizer level for the old decay heat model (ANS79-1) and the new decay heat model (ANS79-3) in comparison with planta data is shown. By observing the results, it can be concluded that the model ANS79-3 represents better results in respect to the plant data. In the following figures, the main differences will be shown.

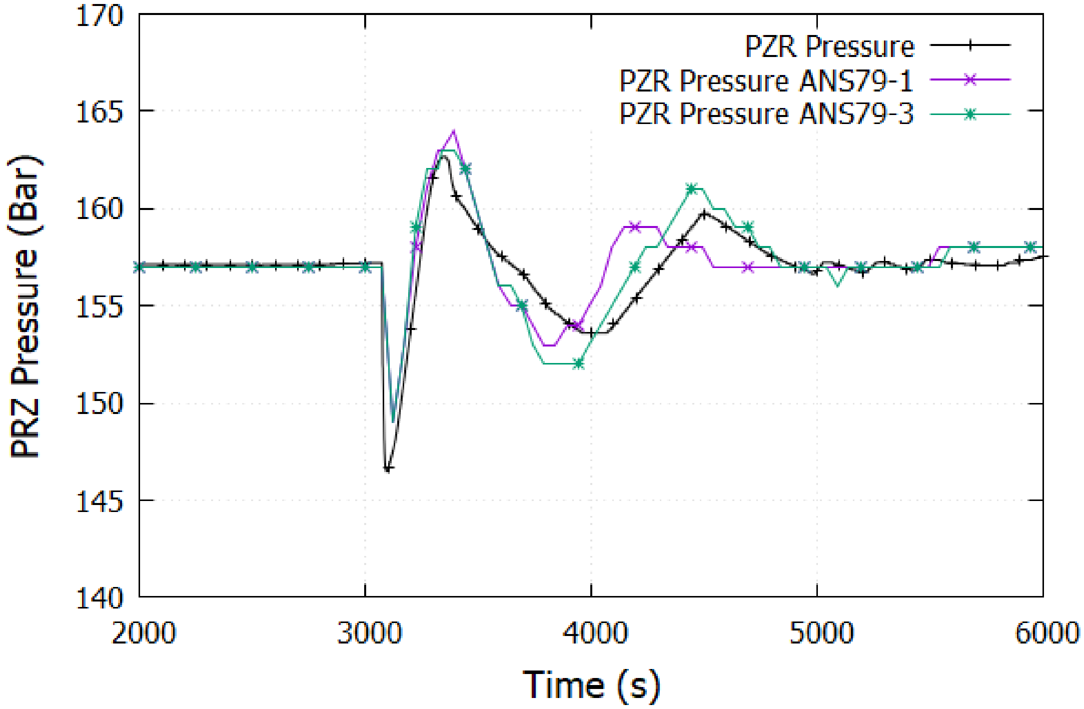

Figure 5-18 Pressurizer Pressure for Different Decay Heat Model in RELAP and Plant Data

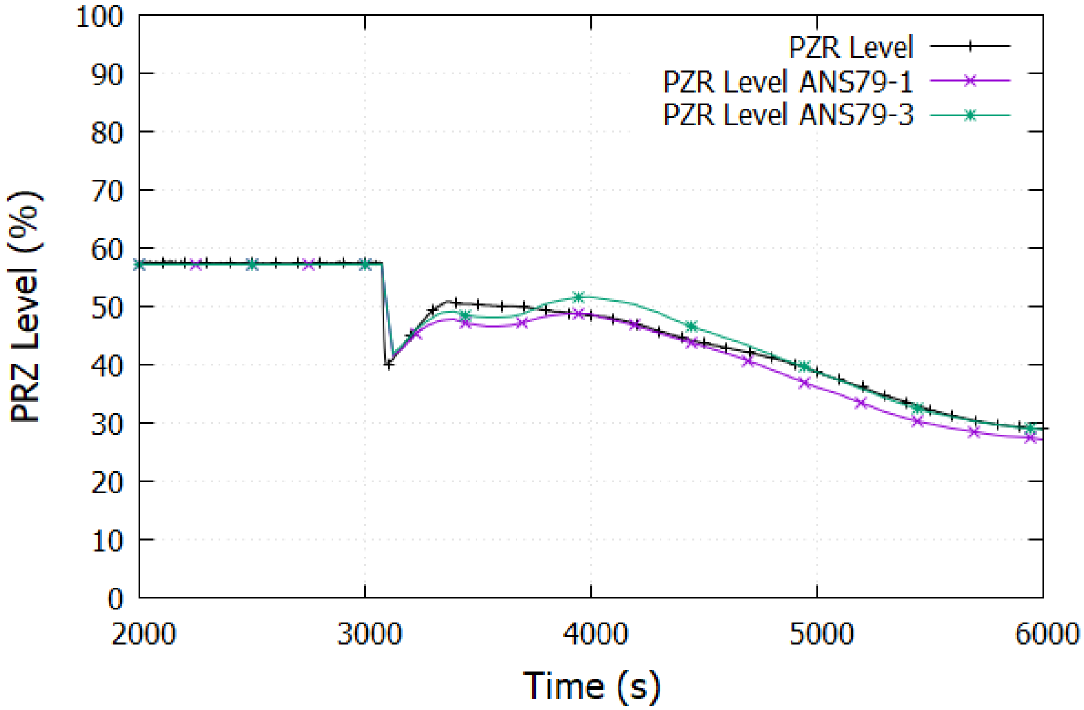

Figure 5-19 Pressurizer Level for Different Decay Heat Model in RELAP and Plant Data

The comparison of the wide and narrow range level is done only with one SG because the three of them present similar results (Figure 5-20 and Figure 5-21). As it can be observed in Figure 5-16, the decay heat power is lower in the ANS79-1 model, thus, the power transferred to the steam generators is lower than in the case of ANS79-3.

This affects the level of the water inside the SGs, because different flow regimes will occur and the amount of water that evaporates into vapor will be different. The ANS79-3 model approximates with higher quality the plant results.

After analyzing all the parameters for different decay heat power models, apparently, ANS79-3 offers greater representation for the simulation of the natural circulation in the primary system in this particular NPP.

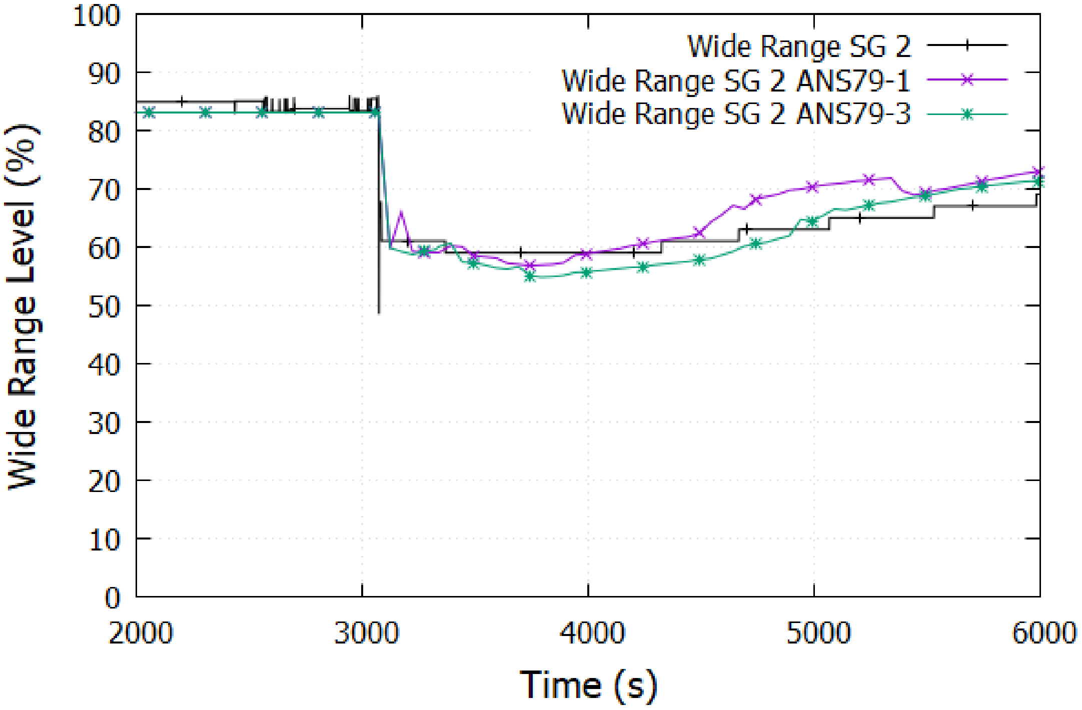

Figure 5-20 Wide Range Level for Different Decay Heat Model in RELAP and Plant Data

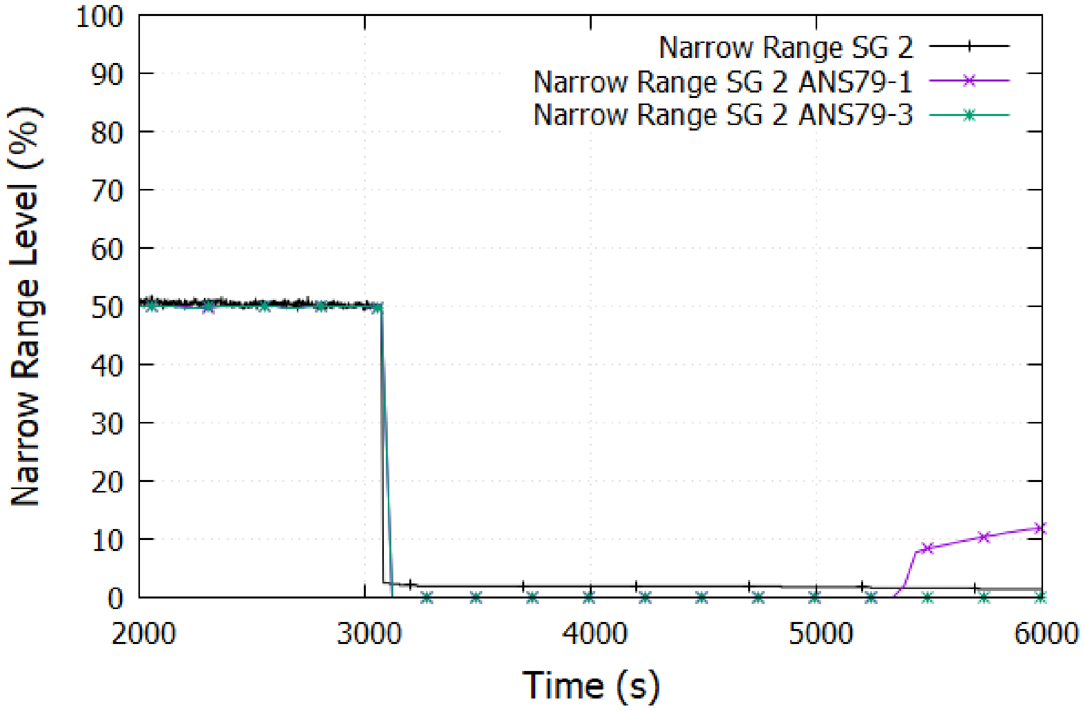

Figure 5-21 Narrow Range Level for Different Decay Heat Model in RELAP and Plant Data

6 RUN STATISTICS

The calculations were performed using Intel® Xeon® CPU E3-1226 v3 @ 3.30 GHz processor. The operating system is Windows 10 Pro.

Table 6-1 shows the run statistics for the codes RELAP5/MOD3.2 calculations.

Table 6-1 Run Statistics

Code	Transient Time [s]	CPU Time [s]	CPU/Transient Time	# of Time Steps
RELAP5/MOD3.2	6000.0	376.484	0.06275	120026

7 BENCHMARK WITH OTHER LOOP ANALYSES

Other LOOP transient analyses have been performed historically for Spanish NPP. In 2011 the NUREG/IA-0243 [3] was published presenting a validation of plant response using the plant model developed in TRACE code, simulating a real event occurred in 1993. Although this event is significantly different to the occurrence analyzed in the present report, both models in TRACE and RELAP show consistency with the relevant plant data.

8 CONCLUSIONS

The Loss of Off-site Power simulation was calculated using RELAP5/MOD3.2. The calculation results were compared to the plant data. In general, the agreement of RELAP calculations with plant data is acceptable. The different phases of the transient are well predicted by the model. The assessment of the model set goodness has resulted limited due to the fact of the lack of some plant data like the auxiliary spray valves flow, which was necessary to run several tests, to estimate this parameter for matching with the plant data performance.

The results of this study show the capability of RELAP5 to reproduce this type of transient that can occur in a NPP, it is important to point out that the results are really congruent with the plant data and all the phenomena are correctly simulated. The adopted code and the modelling techniques are adequate to predict the natural circulation phenomenon and the user effect is sufficiently limited.

The sensibility studies helped to discover how the decay heat power model impacts to the most important parameters of the simulation. For this particular model, at first sight, the ANS79-3 is more suitable for this particular model.

It can be concluded that the model and the code RELAP5 can perfectly reproduce natural circulation in a NPP. For future analysis, where the turbopump is actuating, it is recommended that some improvements in the control of the turbopump should be done to represent more accurately the reality.

During this analysis, the processing of the plant data has been a challenge due to the large amount of information. On the other hand, the correct identification and implementation of all the manual actions occurred in the transient required multiple tests, due to the lack of detailed information in the report describing the transient occurred in the plant.

9 REFERENCES

[1] USNRC, 2010. RELAP5/MOD3.2 code manual. Information Systems Laboratories, Inc., Rockville, Maryland, Idaho Falls, Idaho.

[2] M. Cherubini,W. Giannotti, D. Araneo, and F. D’Auria. Use of the Natural Circulation FlowMap for Natural Circulation Systems Evaluation. Hindawi Publishing Corporation Science and Technology of Nuclear Installations Volume 2008, Article ID 479673

[3] O. Lozano, C-P. Chiang, C. Llopis, L. Batet, F. Reventós. 2011. NUREG/IA-0243. 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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10. SUPPLEMENTARY NOTES

K. Tien, Project Manager

11. ABSTRACT (200 words or less)

The work presented corresponds to the assessment of a PWR Loss of Off-Site Power with RELAP5 [1]. The RELAP5 input deck has been obtained from the existing RELAP5 input deck with some modifications. The calculated results were compared against the plant data. The results show that RELAP5 is capable of reproducing the phenomena with a good level of satisfaction. However, some discrepancies between the predicted variables and the plant data suggest further investigation of the RELAP model.The simulations were prepared by IDOM Nuclear Services. A Spanish NPP provided the input model, the plant data, and transient available data and associated information.The report is focused on the transient calculations and the modifications and conclusions obtained from them. The purpose of this work is to detect possible errors in the existing plant model and provide the modifications for the input model.The work to which this report refers, falls within the framework of the CAMP/SPAIN Project. The objectives being those referred to in the CAMP Program itself, in particular, achievement of the capacity required to apply calculation tools supporting the quantitative analysis of transients and operation procedures, as well as joint participation in the international efforts aimed to validate and experience the use of appropriate calculation codes.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.) Loss of Off-Site Power (LOOP) Large Scale Test Facility (LSTF) Natural Circulation Flow Map (NCFM) Auxiliary Feed Water (AFW) Safety Injection System (SIS)	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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