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International Agreement Report

Simulation of PKL Loss of RHRS Experiment E3.1 with RELAP5 and TRACE Codes – Application to a PWR NPP Model

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ABSTRACT

An analysis of PKL midloop test E3,1 has been performed with TRACE and RELAP5/MOD3 codes. This test, included within the OECD/PKL project, tries to analyze the phenomenology and different accident management actions after a loss of RHRS at midloop conditions with primary side closed. The comparison between the results obtained with both codes shows that, in general, the main phenomena are well reproduced. However, there are still a few phenomena that are not well predicted, like pressurizer level. The good results obtained simulating another similar test (PKL test F2.2 run2) enables to confirm that the modelling methodology is adequate for this kind of transients. This modelling methodology has been also applied to a TRACE model of a Spanish PWR, Westinghouse design, simulating a similar transient to test E3.1, obtaining similar results than in PKL models.

FOREWORD

Extensive knowledge and techniques have been produced and made available in the field of thermal-hydraulic responses during reactor transients and accidents, and major system computer codes have achieved a high degree of maturity through extensive qualification, assessment and validation processes. Best-estimate analysis methods are increasingly used in licensing, replacing the traditional conservative approaches. Such methods include an assessment of the uncertainty of their results that must be taken into account when the safety acceptance criteria for the licensing analysis are verified.

Traditional agreements between the Nuclear Regulatory Commission of the United States of America (USNRC) and the Consejo de Seguridad Nuclear of Spain (CSN) in the area of nuclear safety research have given access to CSN to the NRC-developed best estimate thermal-hydraulic codes RELAP5, TRAC-P, TRAC-B, and currently TRACE. These complex tools, suitable state-of-the-art application of current two-phase flow fluid mechanics techniques to light water nuclear power plants, allow a realistic representation and simulation of thermal-hydraulic phenomena at normal and incidental operation of NPP. Owe to the huge required resources, qualification of these codes have been performed through international cooperation programs. USNRC CAMP program (Code Applications and Maintenance Program) represents the international framework for verification and validation of NRC TH codes, allowing to: Share experience on code errors and inadequacies, cooperating in resolution of deficiencies and maintaining a single, internationally recognized code version; Share user experience on code scaling, applicability, and uncertainty studies; Share a well documented code assessment data base; Share experience on full scale power plant safety-related analyses performed with codes (analyses of operating reactors, advanced light water reactors, transients, risk-dominant sequences, and accident management and operator procedures-related studies); Maintain and improve user expertise and guidelines for code applications.

Since 1984, when the first LOFT agreement was settled down, CSN has been promoting coordinated joint efforts with Spanish organizations, such as UNESA (the association of Spanish electric energy industry) as well as universities and engineering companies, in the aim of assimilating, applying, improving and helping the international community in the validation of these TH simulation codes, within different periods of the associated national programs (e.g., CAMP-España). As a result of these actions, there is currently in Spain a good collection of productive plant models as well as a good selection of national experts in the application of TH simulation tools, with adequate TH knowledge and suitable experience on their use.

Many experimental facilities have contributed to the today's availability of a large thermal-hydraulic database (both separated and integral effect tests). However there is continued need for additional experimental work and code development and verification, in areas where no emphasis have been made along the past. On the basis of the SESAR/FAP reports Nuclear Safety Research in OECD Countries:Major Facilities and Programmes at Risk (SESAR/FAP, 2001) and its 2007 updated version Support Facilities for Existing and Advanced Reactors (SFEAR) NEA/CSNI/R(2007)6, CSNI is promoting since 2001 several collaborative international actions in the area of experimental TH research. These reports presented some findings and recommendations to the CSNI, to sustain an adequate level of research, identifying a number of experimental facilities and programmes of potential interest for present or future international collaboration within the safety community during the coming decade.

CSN, as Spanish representative in CSNI, is involved in some of these research activities, helping in this international support of facilities and in the establishment of a large network of international collaborations. In the TH framework, most of these actions are either covering not enough investigated safety issues and phenomena (e.g., boron dilution, low power and shutdown conditions), or enlarging code validation and qualification data bases incorporating new information (e.g., multi-dimensional aspects, non-condensable gas effects). In particular, CSN is currently participating in the PKL and ROSA programmes.

The PKL is an important integral test facility operated by of AREVA-NP in Erlangen (Germany), and designed to investigate thermal-hydraulic response of a four-loop Siemens designed PWR. Experiments performed during the PKL/OECD program have been focused on the issues: Boron dilution events after small-break loss of coolant accidents; Loss of residual heat removal during mid-loop operation (both with closed and open reactor coolant system.

ROSA/LSTF of Japan Atomic Energy Research Institute (JAERI) is an integral test facility designed to simulate a 1100 MWe four-loop Westinghouse-type PWR, by two loops at full-height and 1/48 volumetric scaling to better simulate thermal-hydraulic responses in large-scale components. The ROSA/OECD project has investigated issues in thermal-hydraulics analyses relevant to water reactor safety, focusing on the verification of models and simulation methods for complex phenomena that can occur during reactor transients and accidents such as: Temperature stratification and coolant mixing during ECCS coolant injection; Water hammer-like phenomena; ATWS; Natural circulation with super-heated steam; Primary cooling through SG depressurization; Pressure vessel upper-head and bottom break LOCA.

This overall CSN involvement in different international TH programmes has outlined the scope of the new period of CAMP-España activities focused on: Analysis, simulation and investigation of specific safety aspects of PKL/OECD and ROSA/OECD experiments; Analysis of applicability and/or extension of the results and knowledge acquired in these projects to the safety, operation or availability of the Spanish nuclear power plants. Both objectives are carried out by simulating experiments and plant application with the last available versions of NRC TH codes (RELAP5 and TRACE). A CAMP in-kind contribution is aimed as end result of both types of studies.

Development of these activities, technically and financially supported by CSN, is being carried out by 5 different national research groups (Technical Universities of Madrid, Valencia and Cataluña). On the whole, CSN is seeking to assure and to maintain the capability of the national groups with experience in the thermal hydraulics analysis of accidents of the Spanish nuclear power plants.

Francisco Fernández Moreno, Commissioner Consejo de Seguridad Nuclear (CSN)

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EXECUTIVE SUMMARY

The PKL III test facility models a typical 4 loop 1300 MWe pressurized water reactor of Siemens/KWU design. As a part of the OECD/PKL program, the PKL III E test series have been performed. This series includes several tests that have been carried out to investigate the thermal hydraulic phenomenology that entails the loss of RHRS system at mid-loop conditions (low power - shutdown plant state) and the different accident management actions that could be implemented by operators during the transient.

In E3.1 test, the primary inventory was fixed at mid-loop level and thus, the capability of the reflux-condensation (including the influence of non-condensable gases) as a mechanism of cooling was investigated. In this test, the RCS is closed and two steam generators are available, one of them operable.

In this report, a post-test analysis of PKL III-E3.1 using RELAP5/MOD3 and TRACE codes is presented. A description of the model inputs is given, and the comparison of measured and calculated results is discussed. Simulation results of equivalent transient conditions implemented in a Spanish PWR Westinghouse design (3-loop) plant model are also discussed in the report.

The purpose of this analysis is to contribute in the validation of TRACE and RELAP5 codes and its ability to properly simulate shutdown conditions. The main findings of the comparison of RELAP5/Mod 3.2 and TRACE 5.0 results with the PKL III E3.1 experiment are:

- Two phase reflux - condensation cooling mechanism with non-condensable gases in the primary side is well reproduced. The net heat balance obtained shows good agreement with experimental data.

- All participants of the E3.1 benchmark obtained a wrong primary mass distribution, with a very large water level in the PZR. It could be interesting to check the condensation and offtake correlations for this and other geometries (pressurizer surge line connection with the hot leg and break in the vessel head) in both codes.

In addition, the application to a mid-loop transient in a PWR Westinghouse shows similar trends that the results obtained in PKL test.

ACKNOWLEDGMENTS

This (paper/work) contains findings that were produced within the OECD-NEA (PKL/ROSA) Project. The authors are grateful to the Management Board of the (PKL/ROSA) Project for their consent to this publication, and thank the Spanish Nuclear Regulatory Body (CSN) for the technical and financial support under the agreement STN/1388/05/748.

ABBREVIATIONS

CAMP	Code Applications and Maintenance Program
CTC	Consolidated Thermohydraulic Code
CCFL	Counter-Current Flow Limit
CSN	Consejo de Seguridad Nuclear (Spanish Nuclear Regulatory Commission)
PWR	Pressurized Water Reactor
PZR	Pressurizer
RCS	Reactor Coolant System
RHRS	Residual Heat Removal System
SETH	SESAR Thermal Hydraulics
SESAR	Senior Group of Experts on Nuclear Safety Research
SG	Steam Generator

1 INTRODUCTION

The PKL III test facility simulates a typical 4 loop 1300 MWe pressurized water reactor of Siemens/KWU design. Within the PKL III-E3 test series, two tests were performed to investigate the thermohydraulic phenomenology that entails the loss of RHRS system at midloop conditions (low power - shutdown plant state).

In test E3.1, the primary inventory was at midloop level ($\sim$ 58%) and thus, the influence of noncondensables on two-phase cooling was investigated. In this test was also analyzed the capability of the reflux-condensation as a mechanism of cooling in a closed RCS considering the availability of two steam generators, one of them operable.

In this report, a post-test analysis of PKL III-E3.1 using RELAP5/MOD3 and TRACE codes is presented. A description of the model inputs is given, and the comparison of measured and calculated results is discussed. Simulation results of equivalent transient conditions implemented in the model of a W-design Spanish 3-loop plant, are also discussed in the report.

The purpose of this analysis is to contribute in the validation of TRACE code and its ability to properly simulate shutdown conditions.

2 DESCRIPTION OF PKL FACILITY AND SELECTED TEST

The PKL test facility is a large scaled-down model of a four loop PWR-KWU design with 1300 MWe (reference plant: Philippsburg 2 NPP), Figure 1. Its characteristics are:

- Elevations are scaled 1:1 while the volumes, power and mass flows are scaled 1:145.
- The reactor core is modeled by a bundle of 314 electrically heated rods with a maximum power of 2.5 MW (10% of rated power).
- Maximum operating pressure is 45 bar.
- Each steam generator has 30 U-tubes of original size and material.

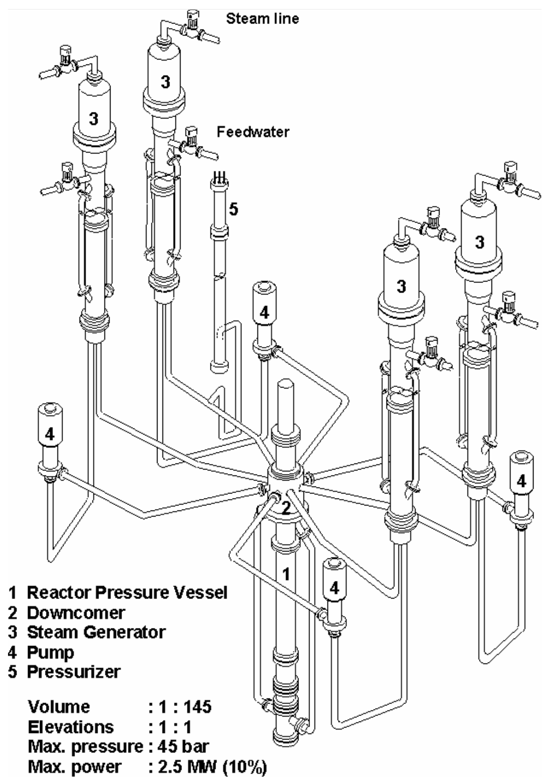

Figure 1: PKL facility

This facility allows the detailed analysis of several accident scenarios like SBLOCA, LBLOCA or loss of RHRS at midloop conditions.

PKL test E3.1 is a sequence of loss of RHRS at midloop conditions. At the beginning of the test the facility is shut down and operating at midloop conditions, Table 1. Steam generators

1 and 2 (SG1 and SG2) are full of water, while steam generators 3 and 4 (SG3 and SG4) are full of nitrogen, Table 2. SG1 is the only one which is pressure controlled with a pressure setpoint of 2 bar while SG2 is isolated. The total core power is 217 kW, which is equivalent to the 0.7% of the nominal power, in order to simulate the decay heat, plus 38 kW to take into account the heat losses of the facility. The test E3.1 has four stages, Table 3:

1. The transient starts with the loss of the RHRS. The pressure and temperature of the primary system start rising quickly after saturation temperature is reached. The steam generated in the core condensates at SG1 and SG2 U-tubes and mainly comes back to the vessel. This heat transfer mechanism between the primary and secondary side is called reflux and condensation cooling. Once the sum of heat transferred into the secondary side and primary heat losses balance core power, primary pressure and temperatures reach quasi-stationary values. During this phase of the transient the temperature and pressure in both circuits keep growing until the pressure in the secondary side reaches 2 bar.

2. When SG1 reaches 2 bar begins the second phase.

   From this point in time on, the feedwater system in SG1 starts the feeding and the steam pressure control system maintains pressure at 2bar in SG1 while SG2 is isolated from secondary system and its pressure goes on growing. From this instant on, both steam generators have different pressures which produces a inventory displacement in the primary side from SG2 towards SG1, leading to an increase of heat transfer in SG1 and a decrease in SG2.

3. After of several hours (34000 s) begins the third stage of the test, in which there are five accumulators injections, first to fourth injections are in cold leg and the last one is in hot leg 1.

4. Finally, after the last accumulators injection, RHRS becomes available, and starts to cool down RCS.

Power	$217kW = 0.7\%$ (residual heat) $+38kW$ (heat losses)
Total mass inventory	$1300\ kg$ ($\simeq 58\%$)
PZR mass inventory	$39\ kg$
PZR level	$1.04\ m$
PZR liquid temperature	$T \simeq 56\ {}^oC$
Core outlet temperature	$T \simeq 61\ {}^oC$
Primary pressure	$P = 10^5\ Pa$

Table 1: Primary side initial conditions. Test PKL E3.1

	SG1	SG2	SG3	SG4
SG level (m)	12.2	12.2	0.0	0.0
SG temperature (^oC)	66.0	66.0	33.0	33.0
FW temperature (^oC)	26.0	26.0	26.0	26.0
Operation	YES	NO	NO	NO

Table 2: Secondary side initial conditions. Test PKL E3.1

Time (s)	**Event**
0	Begin of transient
1914	Shutdown of RHRS
10139	Secondary side pressure has increased to 2 bar. Start of secondary side pressure control at 2 bar in SG1.
35009	Start of ACC injections

Table 3: Test PKL E3.1. Event sequence

3 TRACE 5.0 AND RELAP5 MODELS OF PKL FACILITY

RELAP5 model of PKL is based on an input deck provided by AREVA, operator of the facility, to the participants in the OECD/PKL project. This RELAP5 model was checked and several parameters adjusted in order to improve the results. The TRACE model of PKL was derived from the previous RELAP5 model. This model has 330 thermo-hydraulic cells, 368 SIGNAL VARIABLE, 341 CONTROL BLOCK and 15 TRIP. The vessel is modeled with both downcomer pipes as in PKL facility, Figure 2. Each hot leg is modeled with a 4 cells PIPE in order to avoid convergence problems in the calculations (the models with more cells showed instability problems). The primary side of the steam generators are modeled by three PIPE of different heights in order to simulate better the heat transfer from primary to secondary side as well as RCS flow regimes (e.g., onset and interruption of natural circulation), Figure 3. In both models, heights and volumes of each component were checked against the PKL data.

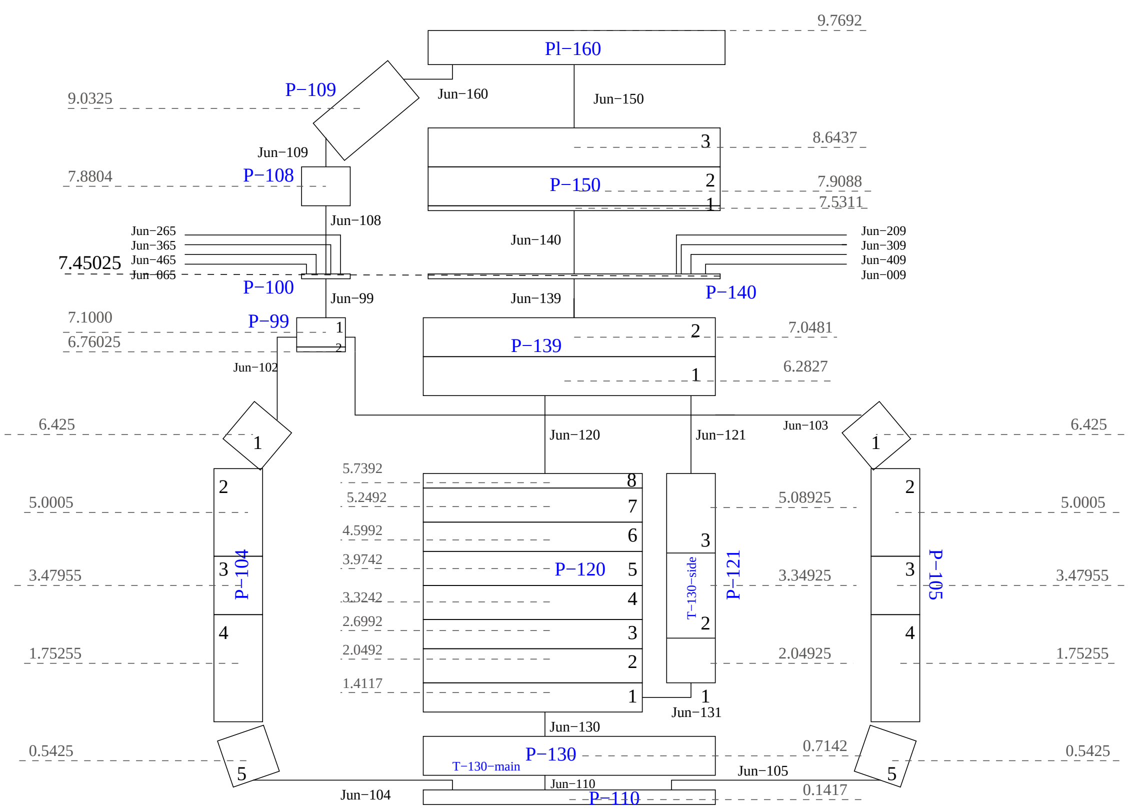

Figure 2: PKL vessel. TRACE model

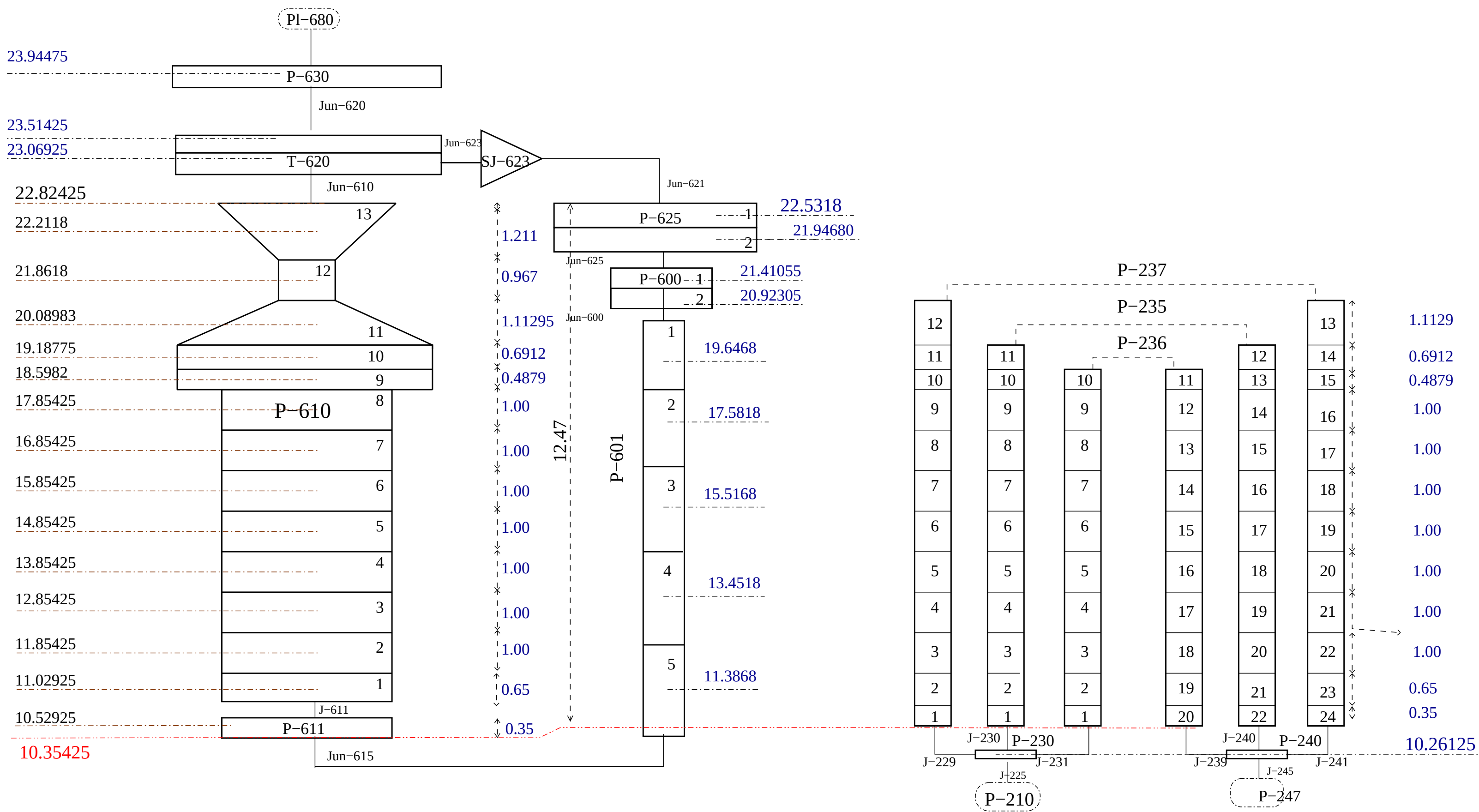

Figure 3: PKL steam generators. TRACE model

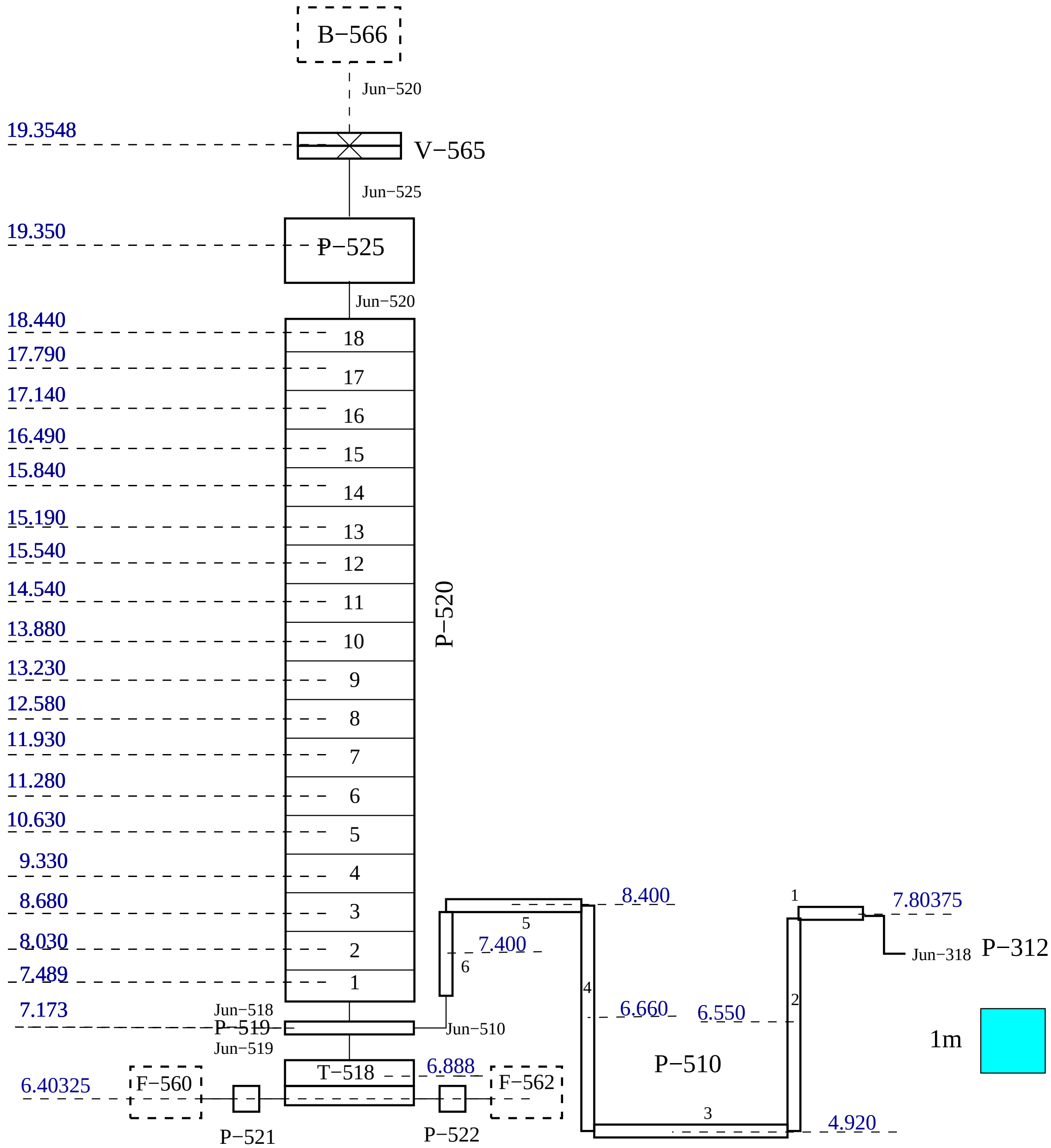

Figure 4: PKL pressurizer. TRACE model

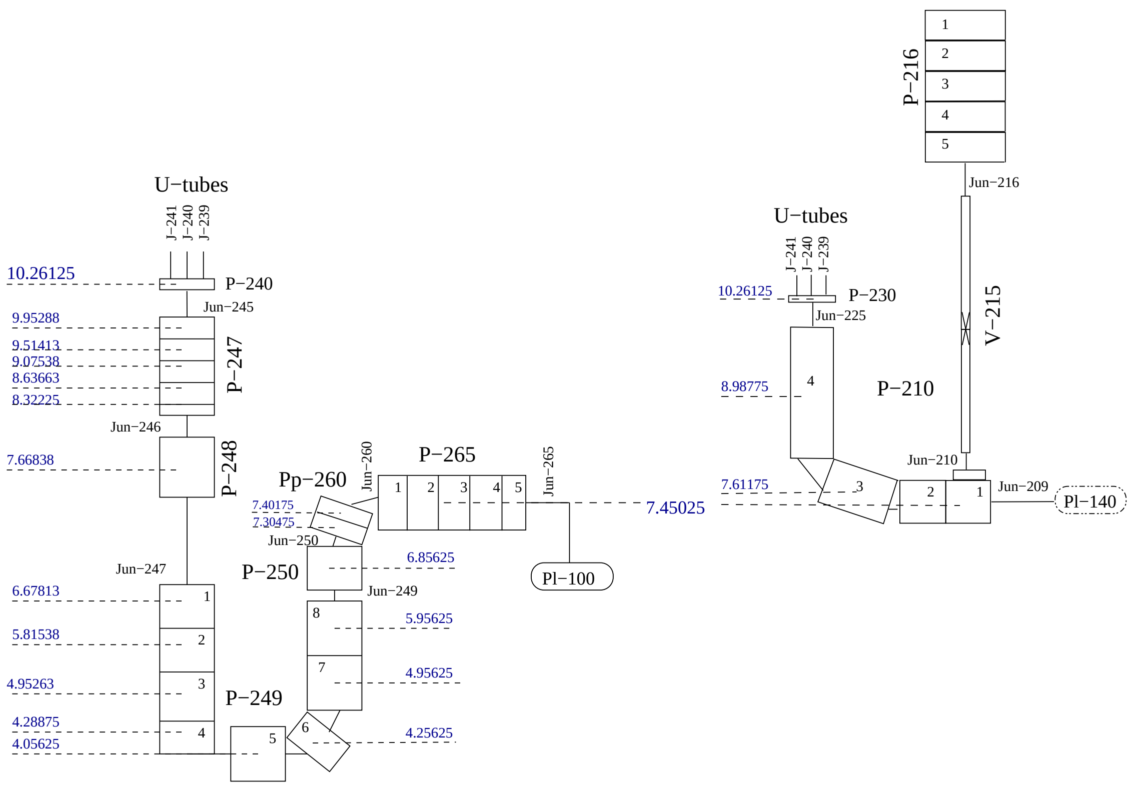

Figure 5: PKL hot leg and cold leg. TRACE model

4 COMPARISON RESULTS OF E3.1 TEST WITH TRACE AND RELAP5 CODES

In a first step, a vessel level control was included in order to obtain midloop level. Later, initial conditions were adjusted obtaining a good agreement with experimental data, Table 4. RELAP5 model was adjusted in four steps:

- First, the mass of heat structures was checked and corrected where it was necessary.
- Second, heat losses were adjusted in primary and secondary side, Table 5.
- Third, friction factors were adjusted for four different mass flows, Table 6, and the test E3.1 was simulated with this four groups of friction factors. The best result was obtained with the friction factors corresponding to the adjustment at G= 12.5 kg/s.
- Fourth, the best simulation was analyzed in detail and it was observed that the agreement of primary pressure an secondary pressures was good enough.

After this analysis, a new model was also developed for TRACE code in order to check if this code was also able to reproduced this transient. The comparison of TRACE results with respect to RELAP5 model and the experimental data is showed in this section.

Component	Initial conditions		
	E3.1	**RELAP5**	**TRACE**
Vessel			
Upper plenum pressure (Pa)	1.05E+05	1.05E+05	1.05E+05
Core exit temperature (K)	333.15	333.15	332.5
Water level	midloop	midloop	midloop
Pressurizer			
Upper plenum pressure (Pa)	1.03E+05	1.03E+05	1.05E+05
Water level (m)	1.04	1.04	1.046
Stem generators (SG1 and SG2)			
Secondary side pressure (Pa)	1.05E+05	1.05E+05	1.05E+05
Secondary side temperature (K)	339.15	339.15	338.5
Water level (m)	12.2	12.2	12.2

Table 4: Initial conditions in test PKL E3.1. Steady state values obtained with RELAP5 and TRACE codes.

Component	Heat losses (kW)		
	Experimental	**TRACE**	**RELAP5**
Downcomer vessel	1.33	1.24	1.33
Downcomer pipes	1.61	1.63	1.61
Lower plenum	1.61	1.60	1.61
Core region	5.35	5.33	5.34
Upper plenum	1.60	1.63	1.59
Upper head	0.28	0.27	0.28
Hot legs	1.51	1.20	1.51
RCP and cold legs	7.26	7.01	7.22
Loop seal	2.11	2.04	2.11
Surge line	0.12	0.12	0.13
Pressurizer	0.79	0.80	0.78
Primary	23.57	23.18	23.53
Secondary	18.43	18.54	18.39
Total	42.00	41.72	41.91

Table 5: Heat losses in TRACE and RELAP5 models. Temperature 373.74 K, loops massflow 8.5 kg/s

Component	Pressure drop - Exp. / RELAP5 (Pa)			
	G = 1.22 kg/s	G = 2.4 kg/s	G = 12.5 kg/s	G = 25 kg/s
Total Vessel	500 / 499	1420 / 1420	27050 / 27057	101150 / 101130
Hot Leg	-51 /-32	-100 / -100	-1458 / -1460	-6660 / -6676
Steam Generator	393 / 387	1484 / 1463	25750 / 25750	90910 / 90897
Loop seal	170 / 169	1020 / 1020	16000 / 16049	60000 / 60040
Cold Leg	1030 / 1029	4137 / 4126	3650 / 3665	14285 / 14293

Table 6: Adjustment of pressure drops in RELAP5 model for different mass flows.

The analysis shown in this report corresponds to the results obtained as part of our participation in an international benchmark with PKL test E3.1 in the framework of PKL/OECD project. In this benchmark the simulations were performed only until accumulators injections, from $t = 0\,s$ to $t = 30000\,s$ of the experiment and there is not attempt to compare the phase of the accumulators injections. The reason is that one of the main phenomena of interest was the reflux cooling in the steam generators.

In order to describe the results, the main thermal hydraulic variables of the simulation are clustered in two sets, one related with primary side and the other to the secondary side. The comparison of experimental and simulated data show the following results:

- Primary pressure and temperature show a good agreement with experimental data reaching the same equilibrium values, Figures 6 and 7. These results indicate that heat transfer between primary side and steam generators have been well reproduced. This hypothesis can be confirmed with the good results of steam mass flow in SG1, Figure 8.

- The mass distribution in primary side is not completely well reproduced. The main reason is that a large amount of water goes to PZR in both simulations and this quantity is lower in the experimental data, Figure 11. With respect to the vessel collapsed water level is well reproduced by RELAP5 but not by TRACE code, Figure 12. This difference could be explained taking into account the difference of the PZR level obtained by both codes, Figure 11, and the section areas in PZR and vessel. Finally, the levels in SG1 and SG2 are quite well reproduced by TRACE code but not by RELAP5 code, Figures 13 and 14.

  So, it is observed that some levels are better predicted by RELAP5 while others are better reproduced by TRACE code.

- The heat transfer from primary to secondary side causes initially a pressure increase in both steam generators filled with water, Figure 9. In this case, it is observed that SG1 pressure is well reproduced by both codes but SG2 pressure is better reproduced by TRACE than RELAP5 code.

- The analysis of heat transfer along the transient shows an interesting behavior, Figure 10 and Table 7. It is observed that both SGs have the same heat transfer capability in the first stage of the test. However, when pressure reaches 2 bar in SG1 and goes on growing in SG2, the heat transfer changes in both SG: in SG2 the heat transfer decreases from 37% to 5% of core power, whereas in SG1 increases from 37% to 77%. It also important to remark that both codes show that primary heat losses are quite important in this kind of transient in experimental facilities, near 15% at high temperature in PKL facility. In real nuclear power plants, the primary heat losses are lower because the relationship between surface and volume is lower than in experimental facilities.

In general, test E3.1 is well reproduced. However, mass distribution problems are observed, mainly related with the high water level obtained for the pressurizer.

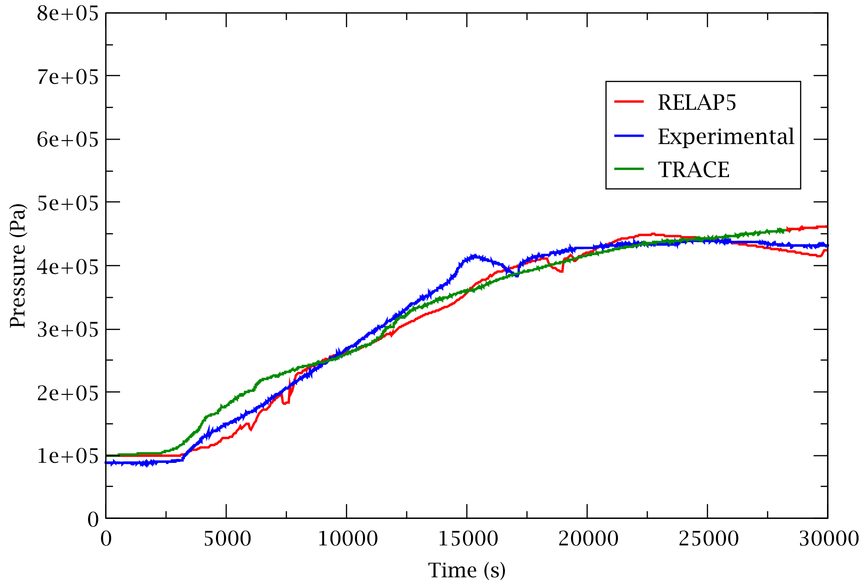

Figure 6: Pressurizer pressure. Test E3.1

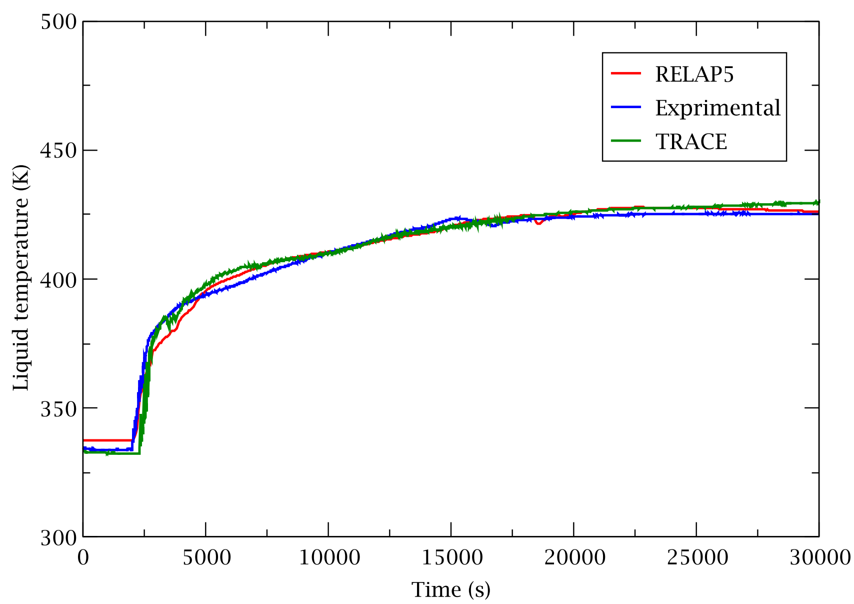

Figure 7: Core exit liquid temperature. Test E3.1

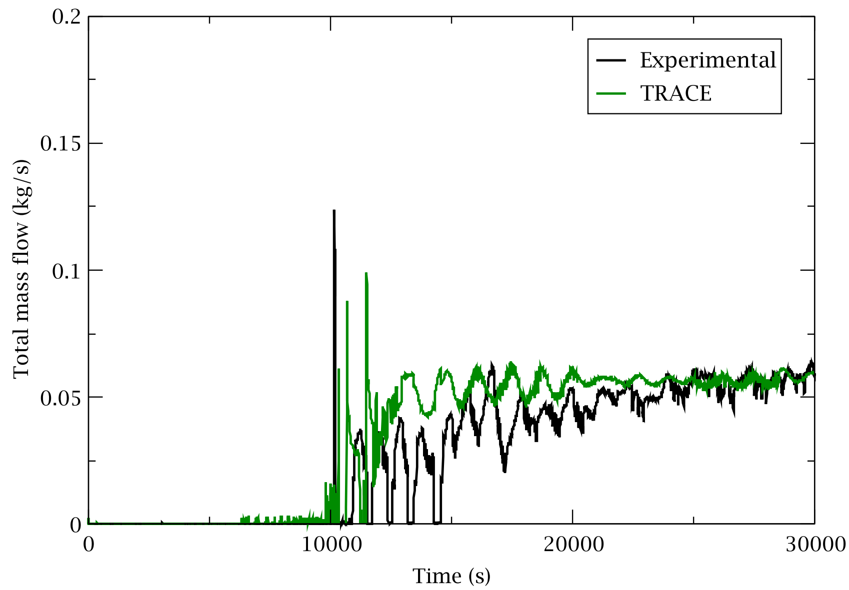

Figure 8: Steam generator 1 vapor mass flow. Test E3.1

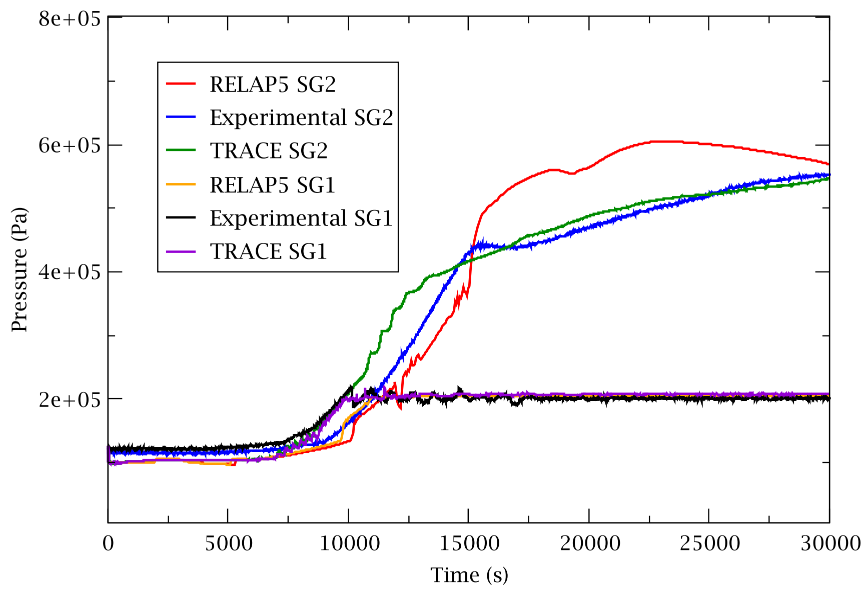

Figure 9: Steam generators pressure. Test E3.1

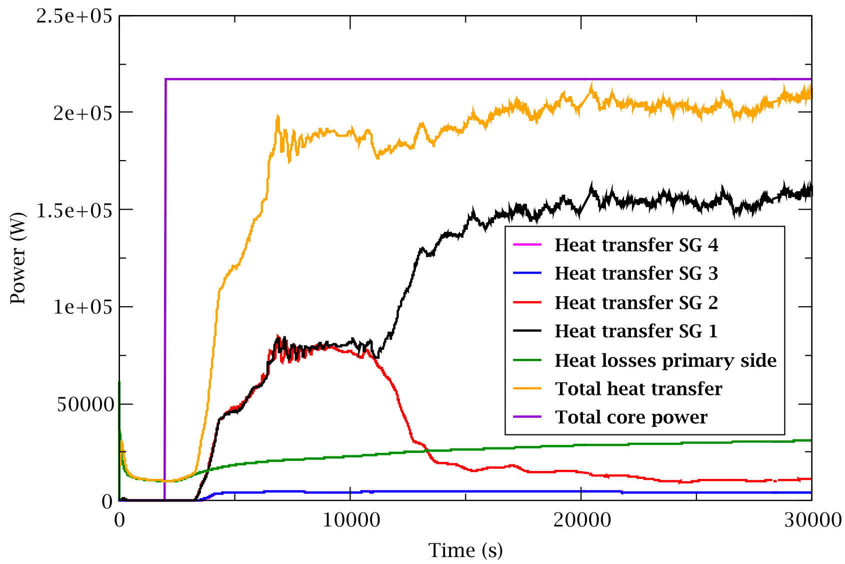

Figure 10: Heat transfer to steam generators and heat losses in primary side. Test E3.1. TRACE code

Component	**TRACE**		**RELAP5**	
	10000 s	30000 s	10000 s	30000 s
SG 1 (%)	37.0	75.0-77.0	33.9	77.0
SG 2 (%)	37.0	5.0	33.9	4.0
SG 3 (%)	2.0	2.0	2.0	2.0
SG 4 (%)	2.0	2.0	2.0	2.0
Primary heat losses (%)	10.0	14.0	13	15
Total heat transfer (%)	88.0	98.0-100.0	85.0	100
Total core power (%)	100.0	100.0	100.0	100.0

Table 7: Equilibrium heat transfer balances in the primary side before and after transient phase two for both codes. PKL E3.1

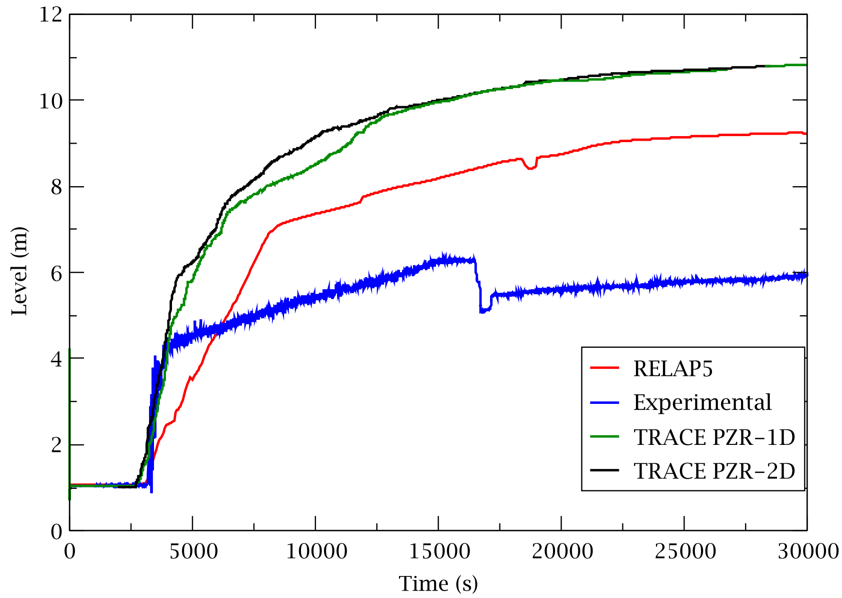

Figure 11: Pressurizer water level. Test E3.1

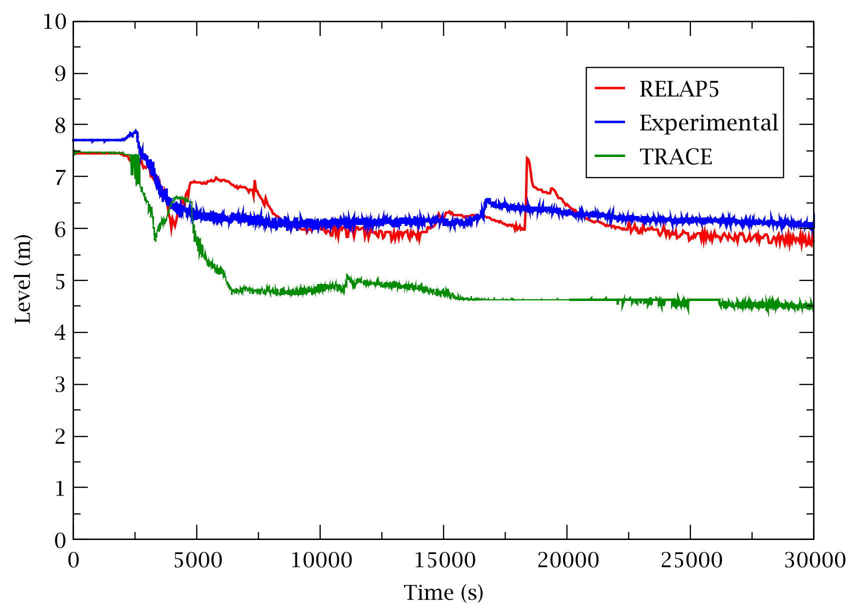

Figure 12: Vessel collapsed water level. Test E3.1

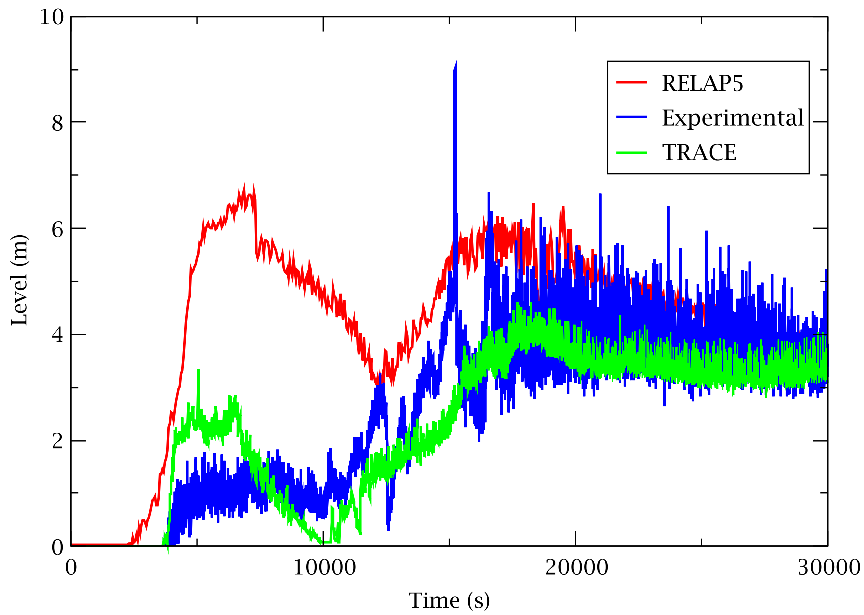

Figure 13: U-tubes water level, steam generator 1. Test E-3.1

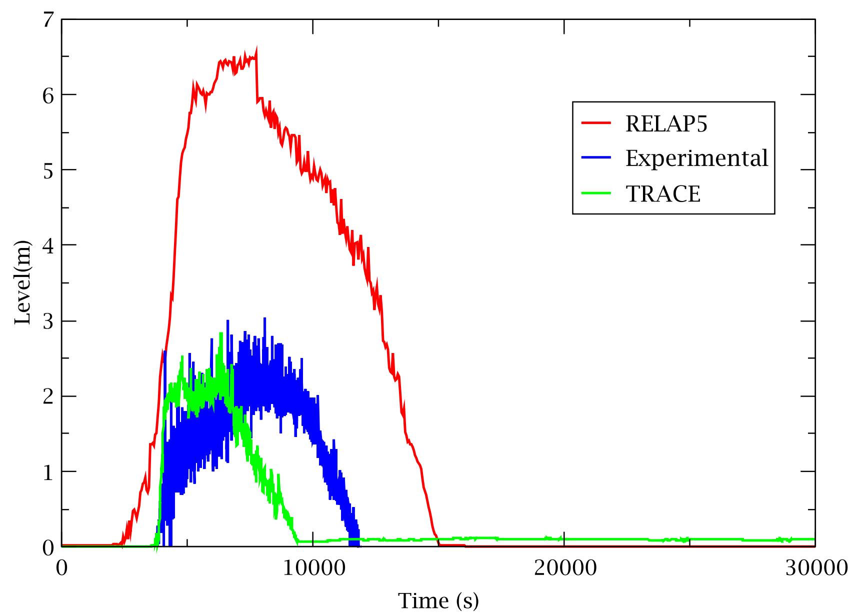

Figure 14: U-tubes water level, steam generator 2. Test E-3.1

5 SIMULATION OF LOSS OF RHRS SEQUENCES AT MIDLOOP CONDITIONS FOR ALMARAZ NPP

The development of the Almaraz NPP model for TRACE code has been framed within several national and international projects (CAMP, CTC, SETH and PKL-OECD) sponsored in Spain by the Nuclear Regulatory Body, Consejo de Seguridad Nuclear (CSN) and the electric energy industry of Spain (UNESA). In these projects, one of the most important objectives is the maintenance and development of the Spanish NPP models with the thermal-hydraulic codes that have been sponsored by NRC, such as RELAP5, TRAC-P, TRAC-M and TRACE. The capability of the TRACE code in simulating low power and shutdown sequences has been confirmed in the simulation of some PKL experiments by several groups, as it has been showed in previous section.

Almaraz NPP is a three loop PWR-W. The TRACE model of Almaraz NPP, Figure 15, is made up of 252 thermal-hydraulic components, 1 VESSEL, 52 PIPE, 71 TEE, 41 VALVE, 3 PUMP, 20 FILL, 27 BREAK, 36 HEAT STRUCTURE and 1 POWER component and also by logical and control systems signals, 685 SIGNAL VARIABLE, 1532 CONTROL BLOCK and 47 TRIP.

Regarding the primary circuit, the following components have been modeled,

- Reactor vessel, modeled by a 3D VESSEL component which includes the core region, guide tubes, support columns, core bypass, and the bypass to the vessel head via downcomer and guide tubes, Figure 16.
- The three loops, each one composed of pump and steam generator, and pressurizer in loop 2 (containing heaters, relief and safety valves and pressurizer spray system).
- Chemical and volumetric control system.
- Safety injection system and accumulators.

With reference to the secondary circuit, the following components have been modeled,

- Normal feedwater and auxiliary feedwater systems of the steam generators.
- The steam lines up to the turbine stop valves, with the relief, safety and isolating valves, and the steam dump with the eight valves.

The shutdown model of Almaraz NPP has been obtained from the model at full power operation. The following changes have been performed:

- every automatic control of the plant has been deactivated, except the level control of steam generator.

- every thermal-hydraulic volume of the model has been initialized to values of LPS conditions.

- a vessel level control has been implemented in order to reach the desired level (midloop level, vessel flange level, level in pressurizer).

- a fine nodalization in vessel has been implemented in order to reach midloop conditions in vessel as well as in cold and hot legs.

- the CCFL model has been taken into account in surge line and U-tubes.

- the offtake model on the surge line connection to the pressurizer has been considered.

An application case of the PKL test E3.1 for Almaraz NPP has been performed starting from the following conditions:

- Midloop level (mass inventory of almost 73000 kg)

- Closed primary. Initial pressure 1 bar, initial temperature $333.15K$

- Decay heat: 11 MW of thermal power

- Availability of steam generators: 1 steam generator full of water and 2 full of air (1SG).

- Secondary pressure: 1 bar

- No Auxiliary Feedwater available.

The results obtained show that the transient behavior is similar in Almaraz NPP model and PKL experimental data, Figures 17 to 20. During the transient the pressure equilibrium condition is reached after 10000 s, lower than in PKL E3.1 (T= 15000 s). In this case, it must be taken into account that in PKL test E3.1 there is a second steam generator full of water which is pressurized along the transient. This aspect may explain the difference between both times.

With respect to the equilibrium pressure, it is observed that the value of Almaraz NPP case is higher than PKL data, Table 8. This discrepancy could be explained because during the adjustment of PKL model it was observed that small differences in the hot/cold leg heights and in heat losses values gives values between 4 bar and 9 bar for equilibrium primary pressure. So, it could be expected that the differences between PKL and Almaraz NPP gives different equilibrium values. In this sense, there is also a reference of Y. Ferng and S. Ma (Investigation of system responses of the Maanshan nuclear power plant to the loss of residual heat removal during midloop operations using a RELAP5/MOD3 simulation. Nuclear Technology 116, 160-172. 1996.) which shows similar values than Almaraz NPP model for similar sequences.

Case	Maximum Pressure (bar)	Time to 2 bar (min.)	Time to 3 bar (min.)
1/3 SG Almaraz	11.5	33	48
1/4 SG PKL	4	–	–
2/4 SG PKL	–	83	165

Table 8: Simulations results for the loss of RHRS at midloop level and closed RCS. Almaraz NPP model and experimental data from PKL test E3.1

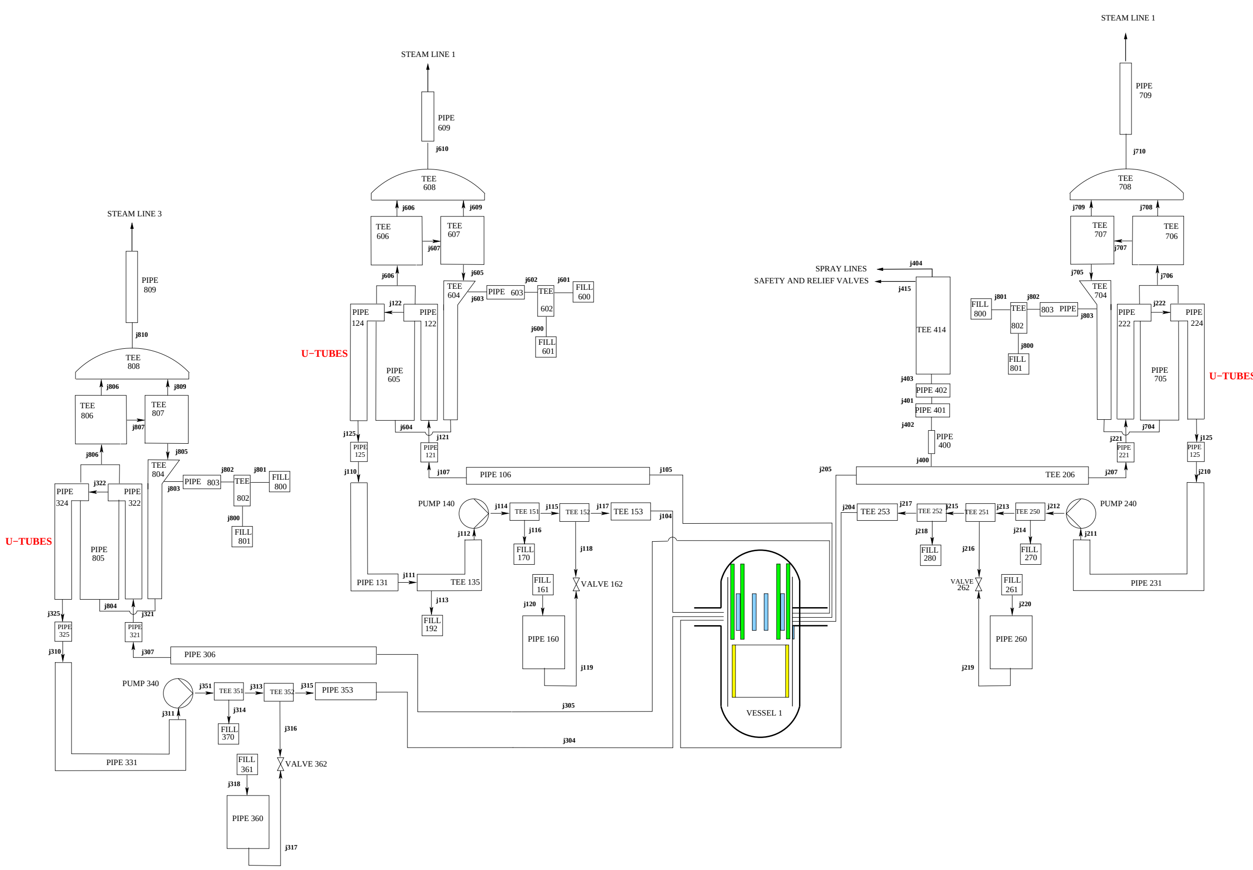

Figure 15: Nodalization of the TRACE model of Almaraz NPP

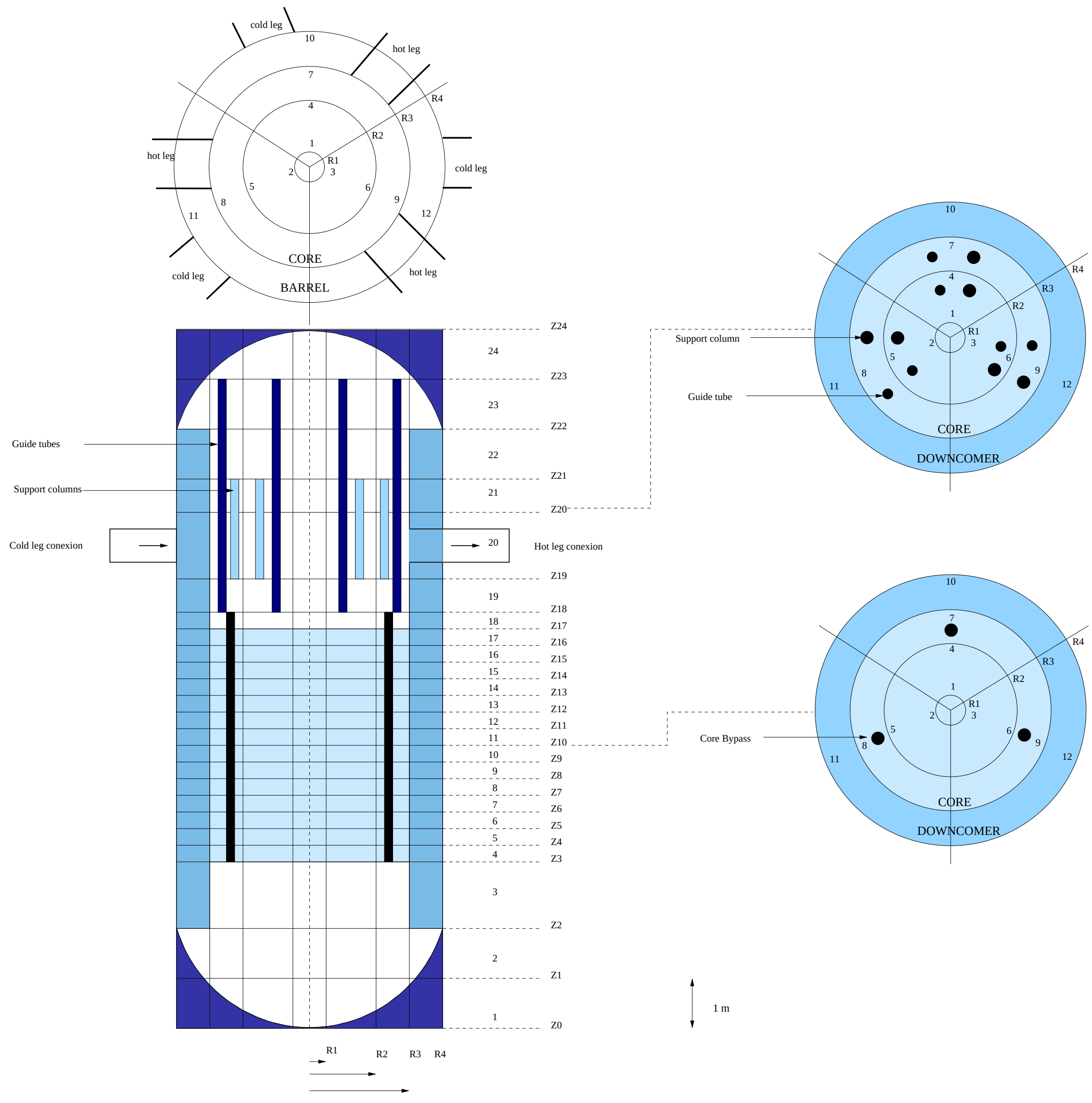

Figure 16: Nodalization of the TRACE model of 3D VESSEL

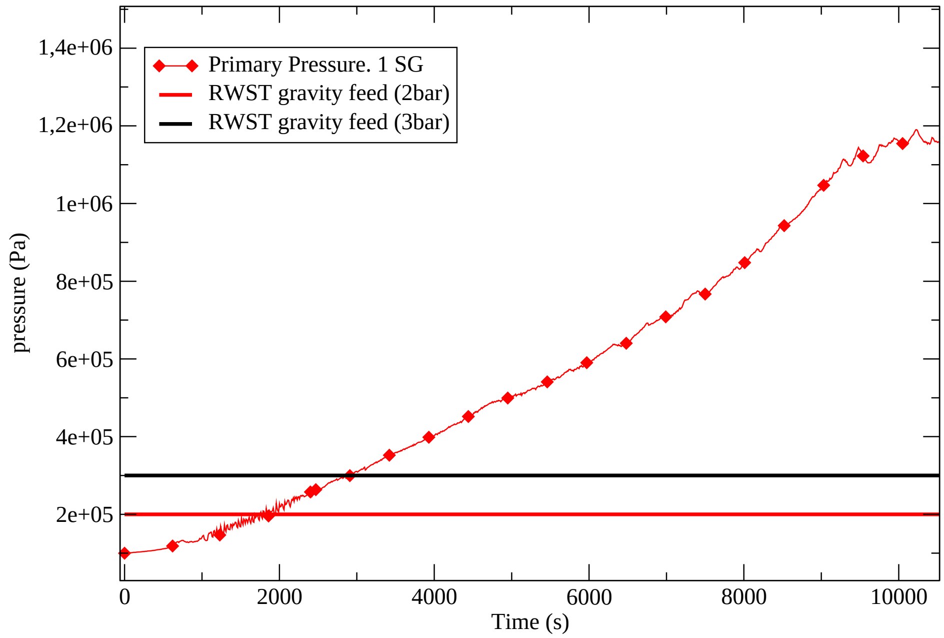

Figure 17: RCS pressure. Transients similar to PKL E3.1

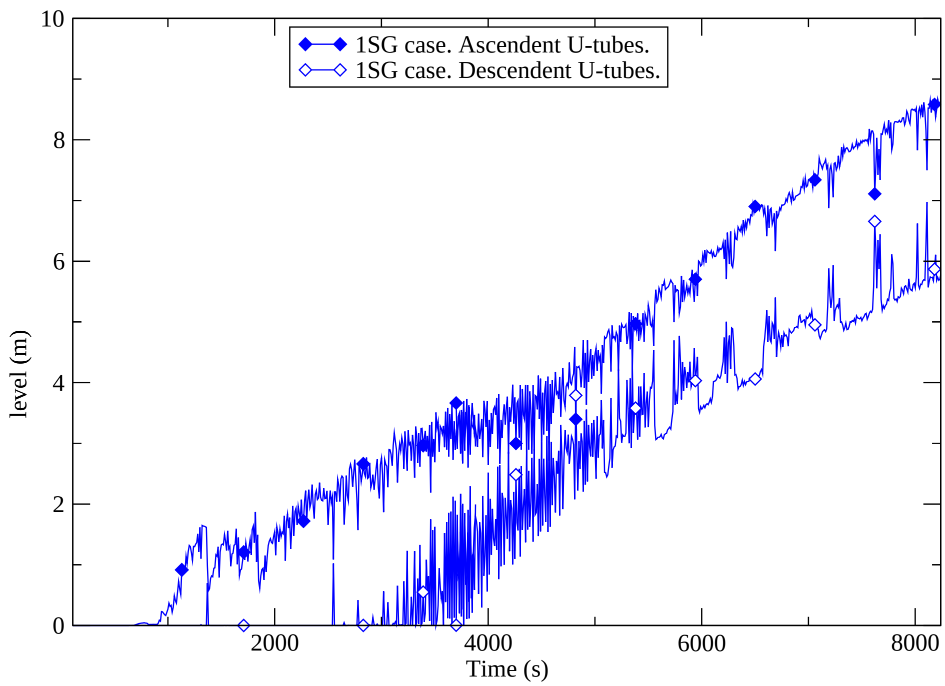

Figure 18: U-tubes level of SG operating. Transients similar to PKL E3.1

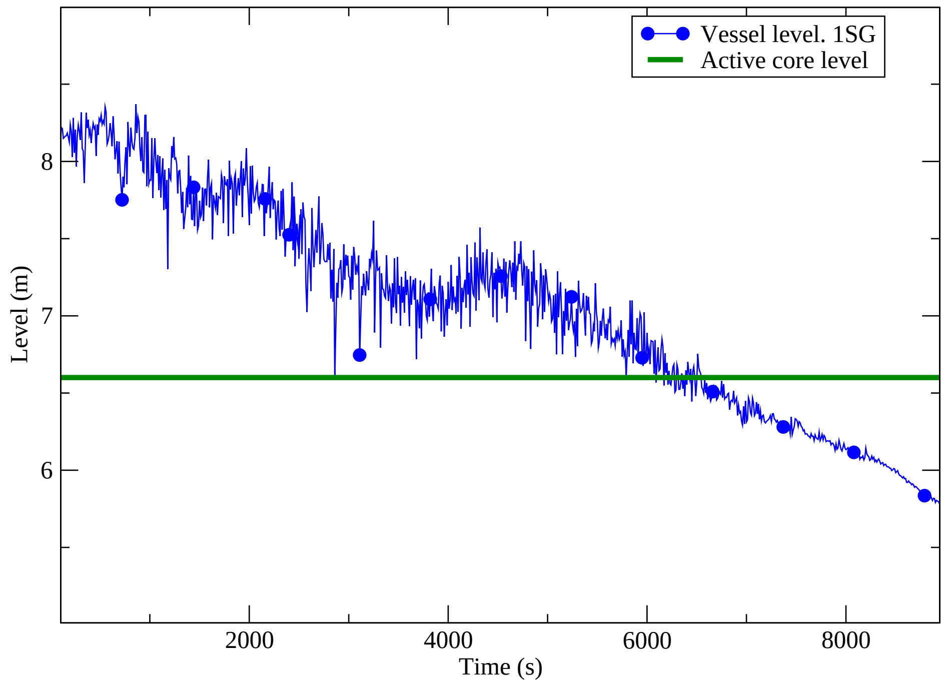

Figure 19: Vessel collapsed water level. Transients similar to PKL E3.1

6 EXECUTION STATISTICS

The simulations have been run in Pentium IV 3.4 MHz under Windows XP and AMD Opteron Dual Core Processors 180 & 1222 under Debian, both with 32 and 64 bits pre-compiled executables provided by NRC. No significant differences where found between runs executed in Windows and Debian systems, and between 32 and 64 bits code versions.

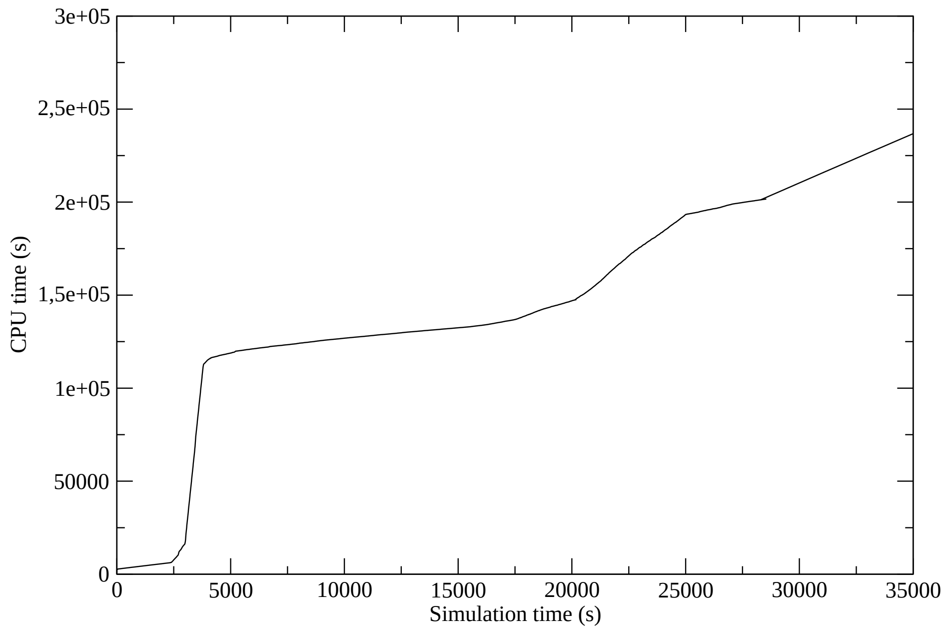

Figure 20: Execution time. PKL test E3.1. TRACE model

7 CONCLUSIONS

The PKL III test facility replicates the entire primary system and most of the secondary system of a typical 1300 MWe pressurized water reactor of Siemens / KWU design. In test E3.1, the primary inventory was at midloop level ($\sim$ 58%) and thus, the influence of noncondensables on two-phase cooling was investigated. In this test was also analyzed the capability of the reflux-condensation as a mechanism of cooling in a closed RCS considering the availability of two steam generators, one of them operable.

In this report, a post-test analysis of PKL III E3.1 using RELAP5/MOD3 and TRACE codes has been presented. The main findings of the comparison of RELAP5/Mod 3.2 and TRACE 5.0 results with the PKL III E3.1 experiment are:

- Two phase reflux - condensation cooling mechanism with non-condensables gases in the primary side is well reproduced, obtained a net heat balance quite accurate when compared with experimental data.
- All participants of the E3.1 benchmark obtained a wrong primary mass distribution, with a very large water level in the PZR. Probably, it could be interesting to check the condensation and offtake correlations for this and other geometries (pressurizer surge line connection with the hot leg and break in the vessel head).

With respect to the application to a midloop transient in a PWR Westinghouse similar results that PKL test have been obtained.