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International Agreement Report

Simulation of PKL Loss of RHRS Experiment F2.2 Run 2 with RELAP5 and TRACE Codes – Application to a PWR NPP Model

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ABSTRACT

An analysis of PKL midloop test F2.2 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 E3.1) 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 F2.2 run2, 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 PKL/OECD program the PKL III-F2 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 test F2.2 run2, the primary inventory was fixed at midloop level and thus, the capability of the reflux-condensation (including the influence of noncondensables gases) as a mechanism of cooling was investigated. In this test the RCS is closed and two steam generators are available, both of them operable. The subsequent operation of the other two SGs, which were empty of water and filled with nitrogen at the beginning of the transient, is used to recover a decay heat sink.

In this report, a post-test analysis of PKL III-F2.2 run2 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 F2.2 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.

- A wrong primary mass distribution has been obtained, 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.

- The main phenomena of AFW injection phase have been also well reproduced.

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. As a part of the PKL/OECD program the PKL III-F2 test series have been performed. This series includes several tests that have been performed to investigate the thermohydraulic phenomenology that entails the loss of RHRS system at midloop conditions (low power - shutdown plant state) and the different accident management actions that could be implemented by operators during the transient.

In test F2.2 run2, the primary inventory was at midloop level and thus, the capability of the reflux-condensation, including the influence of noncondensables gases, as a mechanism of cooling was investigated. In this test the RCS is closed and two steam generators are available, both of them operable. The subsequent operation of the other two SGs, which were empty of water and filled with nitrogen at the beginning of the transient, is used to recover a decay heat sink.

In this report, a post-test analysis of PKL III-F2.2 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 main 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.

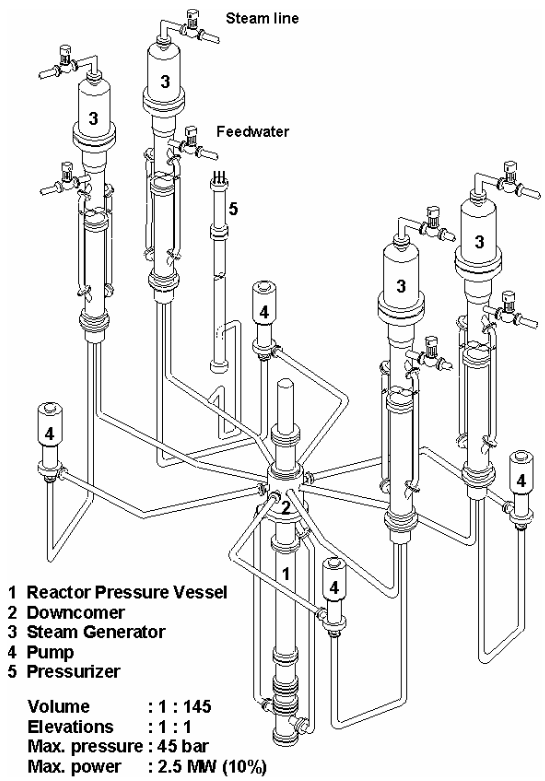

Figure 1: PKL facility

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

Run 2 of PKL test F2.2 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 and SG2 are pressure controlled with a pressure setpoint of 2 bar. The total core power is 219 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 F2.2 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/SG2 reach 2 bar begins the second phase. From this point in time on, steam pressure control system maintains pressure at 2bar in these two SGs; SG level control system also starts to feed both SG. An additional heat sink, implemented through the later actuation of SG3/SG4 level control system, feeds these SGs until the level setpoint is reached, and through subsequent actuation of SG3/SG4 pressure control system, controlling also in 2 bar, allows to reach a new and lower quasi-stationary primary pressure.

3. Two similar injection sequences, using the fuel pool cooling system (FPCS), were initiated for makeup of reactor coolant system inventory. This emergency measure allows to recover primary inventory.

4. After last FPCS injection, RHRS becomes available, and starts to cool down RCS.

Only the first and second phases are simulated in this work because the main interest is to analyze reflux-condensation mechanism and compare it with similar conditions in a realistic case in an actual NPP.

Power	$219kW = 0.7\% + 38kW$
Total mass inventory	1300 kg
PZR mass inventory	39 kg
PZR level	1.0 m
PZR temperature	$T \simeq 56\ C$
Core outlet temperature	$T \simeq 61,5\ C$
Primary pressure	$P = 0,95 \cdot 10^5\ Pa$

Table 1: Primary side initial conditions. Test PKL F2.2 run 2

	SG1	SG2	SG3	SG4
SG level	12.2 m	12.2 m	0.0 m	0.0 m
SG temperature	$33 - 47\ C$	$33 - 47\ C$	$33 - 47\ C$	$33 - 47\ C$
FW temperature	26.0 C	26.0 C	26.0 C	26.0 C
Operation	YES	YES	YES	YES

Table 2: Secondary side initial conditions. Test PKL F2.2 run 2

Time	**Event**
0 s	Shutdown of RHRS
8961 s	SG1 pressure control activated
10447 s	SG2 pressure control activated
10981 s	begin of feedwater feeding in SG1
11825 s	begin of feedwater feeding in SG2
18543 s	begin of feedwater feeding in SG3
22783 s	begin of feedwater feeding in SG4
30563 s	Begin of FPCS injections
37818 s	RHRS started
39547 s	end of the test

Table 3: Test PKL F2.2 run 2. 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.

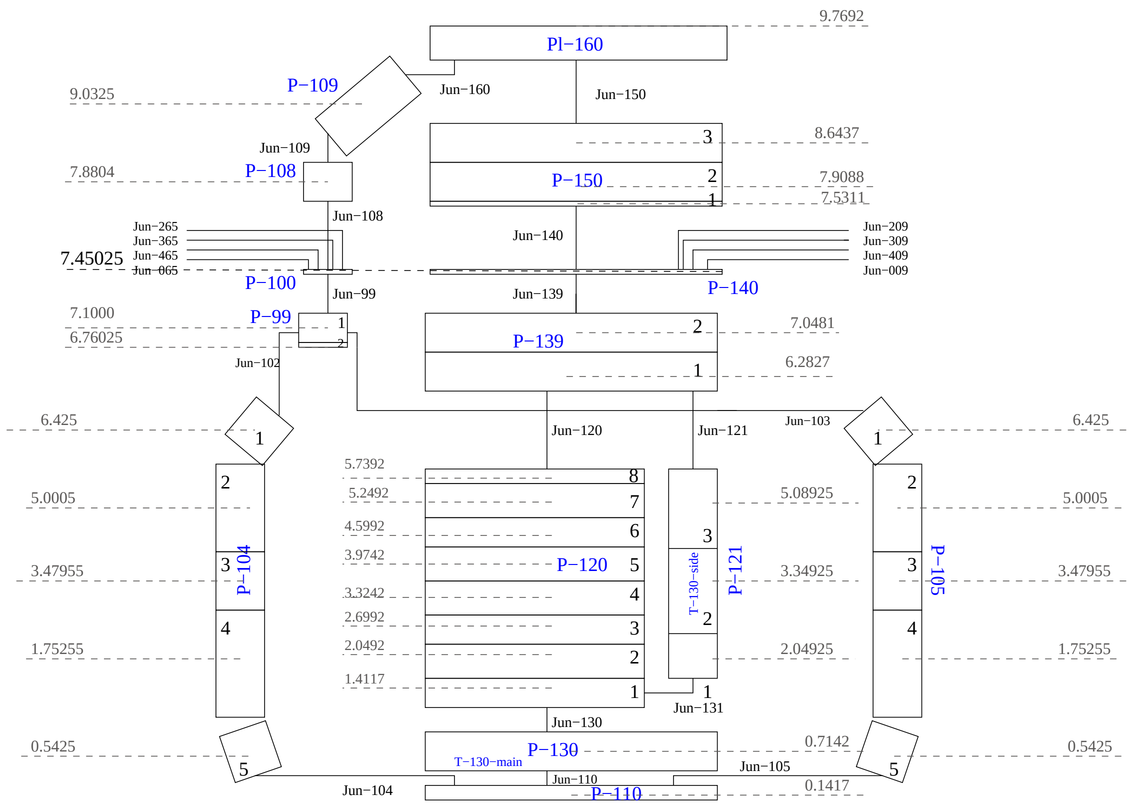

Figure 2: PKL vessel. TRACE model

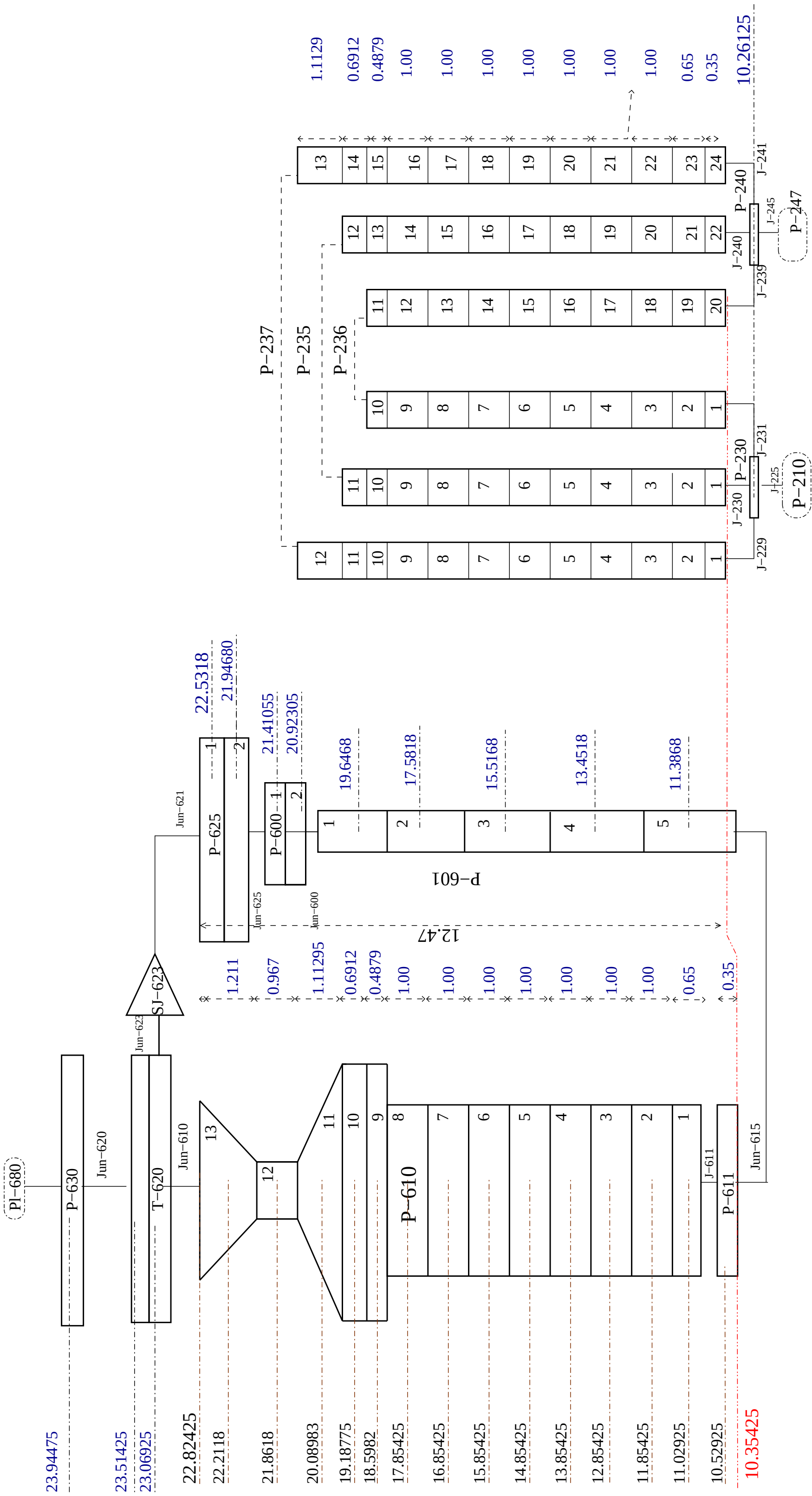

Figure 3: PKL steam generators. TRACE model

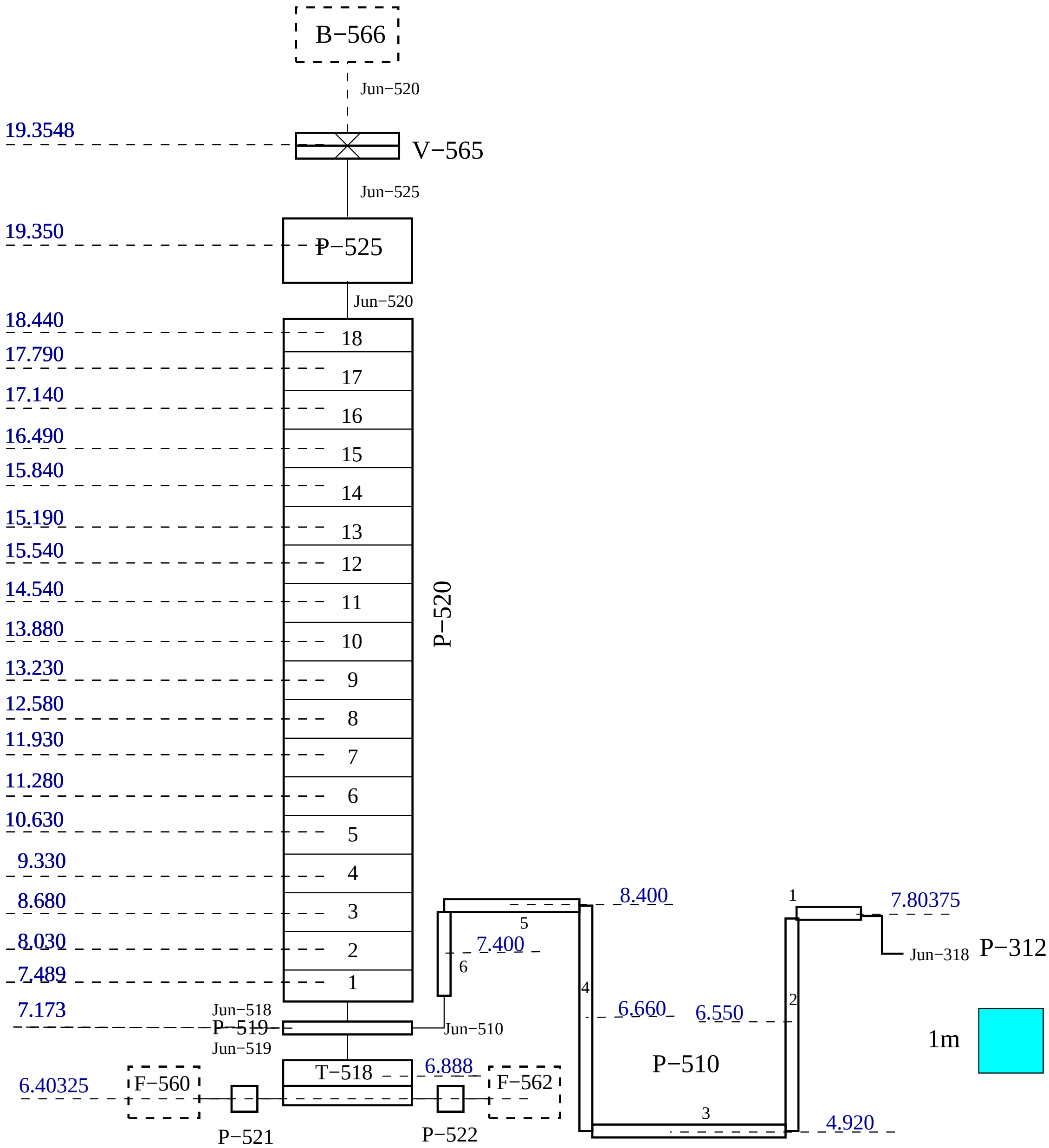

Figure 4: PKL pressurizer. TRACE model

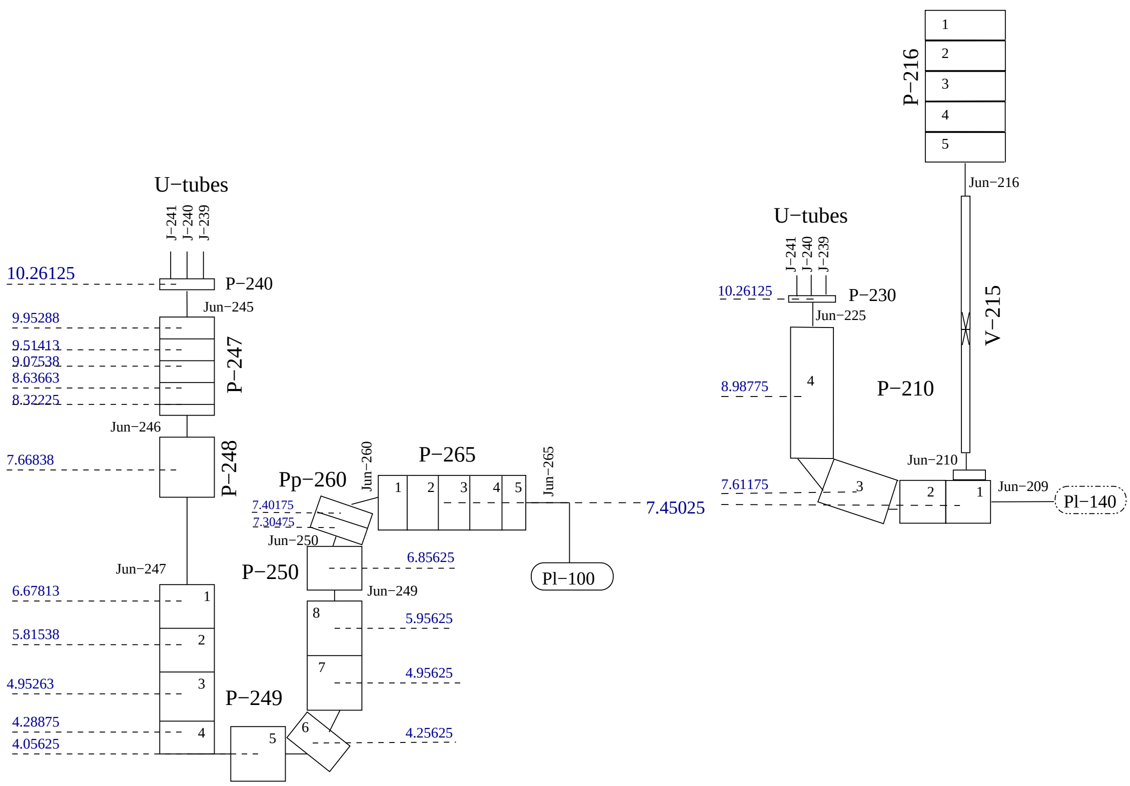

Figure 5: PKL hot leg and cold leg. TRACE model

4 COMPARISON RESULTS OF F2.2 RUN 2 TEST WITH TRACE AND RELAP5 CODES

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 slightly different equilibrium values with two SG filled of water, Figures 6 and 7. However, the simulated and experimental values are quite similar in the last stage of the transient, when there is water in all SG. These results indicate that heat transfer between primary side and steam generators have been well reproduced, mainly in the last stage of the transient.

- 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 they are larger than the experimental data, Figure 8. With respect to collapsed vessel level, it is observed that the results of RELAP5 are better than TRACE code ones, Figure 9.

- The heat transfer from primary to secondary side causes a pressure increase in all steam generators when they are filled with water, Figures 10 and 11. The level in all SG is well reproduced by both codes, Figure 11. However, the simulation with RELAP5 shows a peak in the secondary pressure at the beginning of feedwater injection in SG3 and SG4, Figure 10. Apart from this, the secondary pressures are well reproduced by TRACE and RELAP5 codes.

- The analysis of heat transfer in this transient shows an interesting behavior, Figure 12 and Table 4. It is observed that both SGs filled of water have the same heat transfer capability in the first stage of the test. When SG3 and SG4 are partially filled with water the heat transfer from primary to secondary grows up and primary pressure decreases. The comparison of heat transfer results, Table 4, shows that the SGs that are partially filled with water have the same heat transfer capability than the SGs with higher level. Also it is observed that the primary heat losses are higher in RELAP5 simulation.

In general, test F2.2 run 2 is well reproduced. However, mass distribution problems are observed, mainly related with the high water level obtained for the pressurizer.

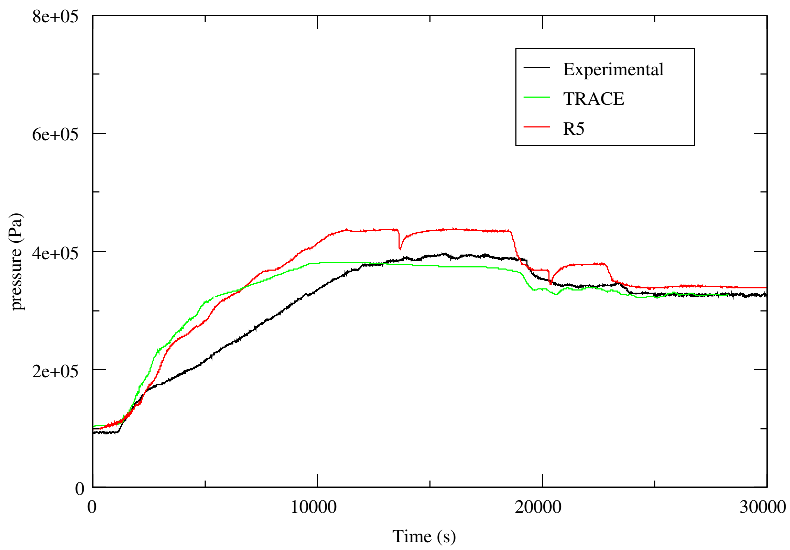

Figure 6: Reactor vessel pressure in the upper plenum. Test F2.2

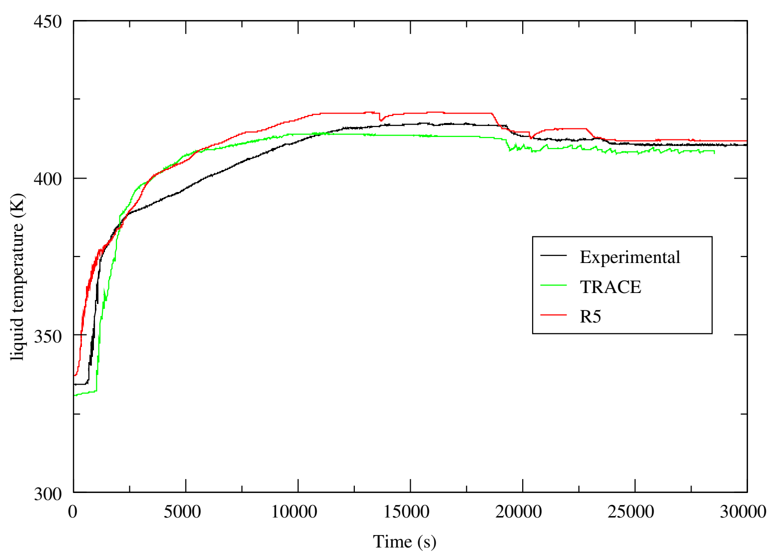

Figure 7: Liquid temperature in the upper plenum of the reactor vessel. Test F2.2

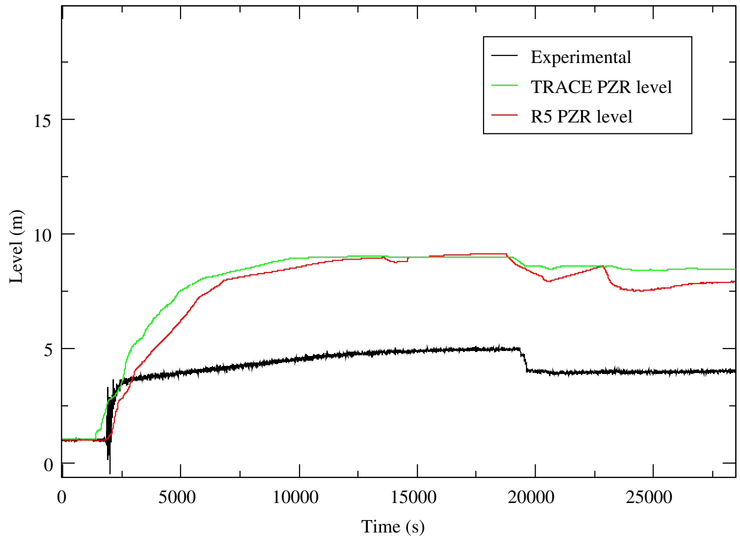

Figure 8: Pressurizer level. Test F2.2 run2

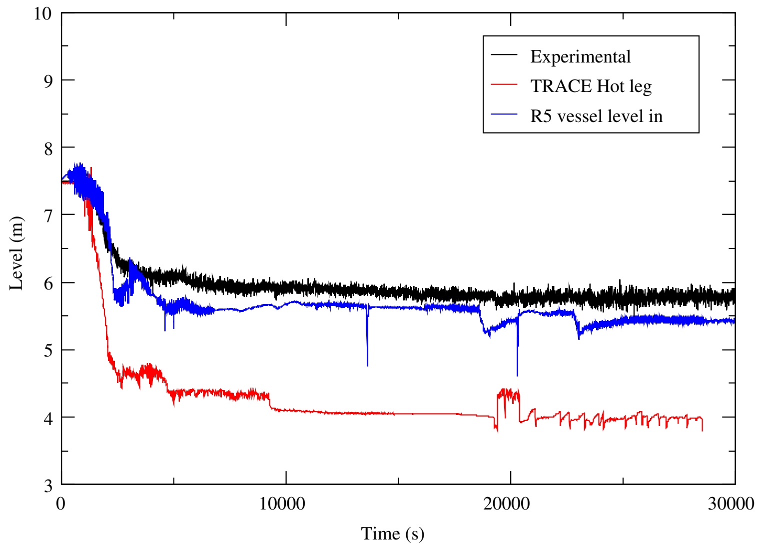

Figure 9: Vessel level. Test F2.2 run2

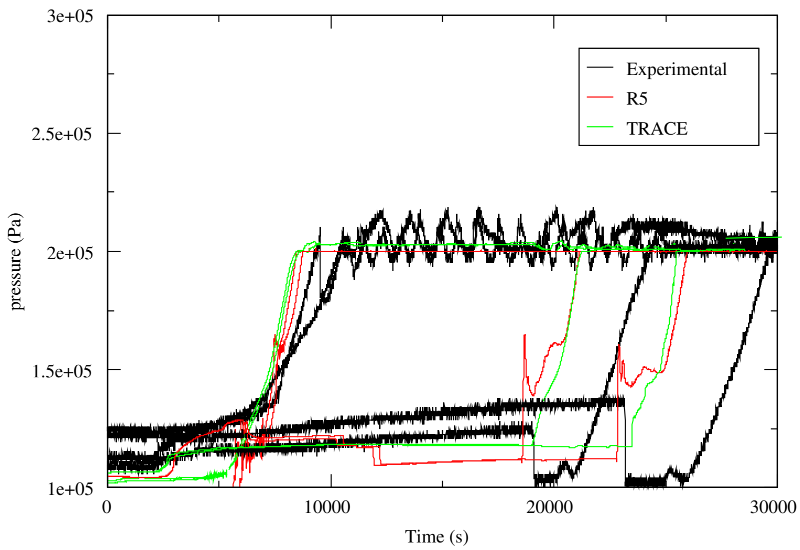

Figure 10: Steam generators pressures. Test F2.2

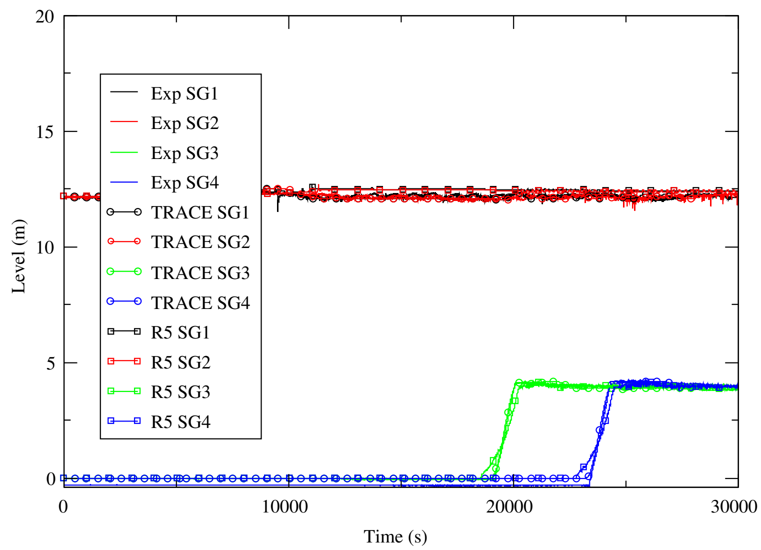

Figure 11: Steam generators water level. Test F2.2

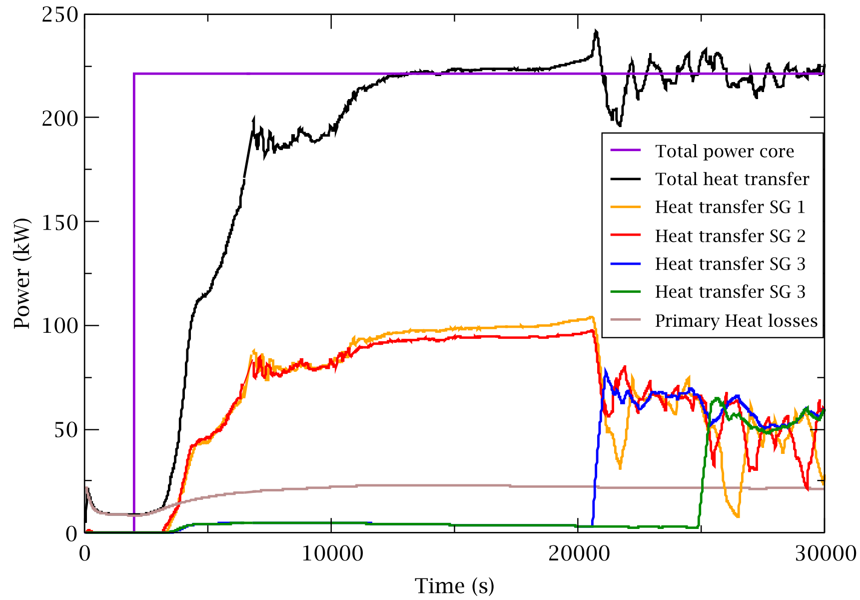

Figure 12: Heat transfer to steam generators and heat losses in primary side. Test F2.2. TRACE code

	TRACE			RELAP5		
	17500 s	22000	27000	17500 s	22000	27000
SG 1	43.0%	29.3%	22.5%	42.0%	28.3%	21.5%
SG 2	43.0%	29.3%	22.5%	42.0%	28.3%	21.5%
SG 3	2.0%	29.3%	22.5%	1%	28.3%	21.5%
SG 4	2.0%	2.0%	22.5%	1%	1%	21.5%
Primary heat losses	10.0%	10.0%	10.0%	14.0%	14.0%	14.0%
Total heat transfer	100%	100.0%	100.0%	100%	100.0%	100.0%
Total core power	100.0%	100.0%	100.0%	100%	100.0%	100.0%

Table 4: Heat transfer balance in primary side. PKL F2.2. TRACE and RELAP5 codes.

5 SIMULATION OF LOSS OF RHRS SEQUENCES AT MIDLOOP CONDITIONS IN 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 13, 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 via guide tubes, Figure 14.
- 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 F2.2 run 2 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: 2 steam generators full of water and 1 full of air (2SG).
- Secondary pressure: 1 bar
- Auxiliary Feedwater available for SG3 when equilibrium primary pressure conditions are reached (13000 s.).

The case with 3 steam generators full of water (3SG) it is also simulated in order to check the final steady state in the application case described above.

The results obtained show that the transient behavior is similar in Almaraz NPP model and PKL experimental data, Figures 15 to 17. There are two stages in the transient:

- In the first part of the transient the pressure equilibrium condition is reached after 13000 s, similar to PKL F2.2 run 2 (T= 15000 s).
- In the second part (T>13000 s) the auxiliary feedwater is available for SG3 and a new stationary state is reached after $\sim$ 500 s (T= 13500 s), Figures 15 to 17. The pressure decreases from 5.5 bars to 3.5 bars, obtaining better cooling conditions (lower temperatures) and lower pressures.

  Concerning the gravity feed from the RWST, the maximum pressure for gravity feed is conditioned by the specific plant configuration, but this pressure is generally between 2 and 3 bar. Therefore, this result also indicates that if AFW is injected soon in the empty SG it is possible to increase the available time to inject with gravity feed from RWST.

So, it is observed that the Almaraz NPP results are similar to PKL F 2.2 run 2. In fact, the comparison of the equilibrium pressure values of Almaraz NPP model with respect to PKL data also show a similar order of magnitude, Table 5.

Case	Maximum Pressure (bar)	Time to 2 bar (min.)	Time to 3 bar (min.)
2/3 SG Almaraz	5.6	33	62
3/3 SG Almaraz	3.8	45	85
2/4 SG PKL	4	85	150
3/4 SG PKL	3	–	–
4/4 SG PKL	2.7	–	–

Table 5: Simulations results for the loss of RHRS at midloop level and closed RCS. Almaraz NPP model and experimental data from PKL test F2.2 run2

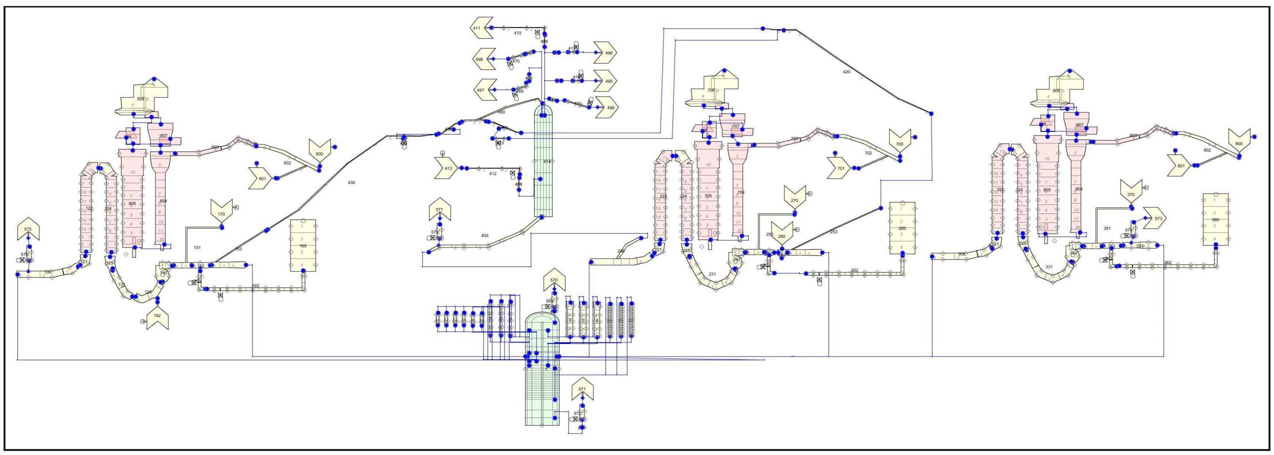

Figure 13: TRACE model of Almaraz NPP (SNAP mask with mean components)

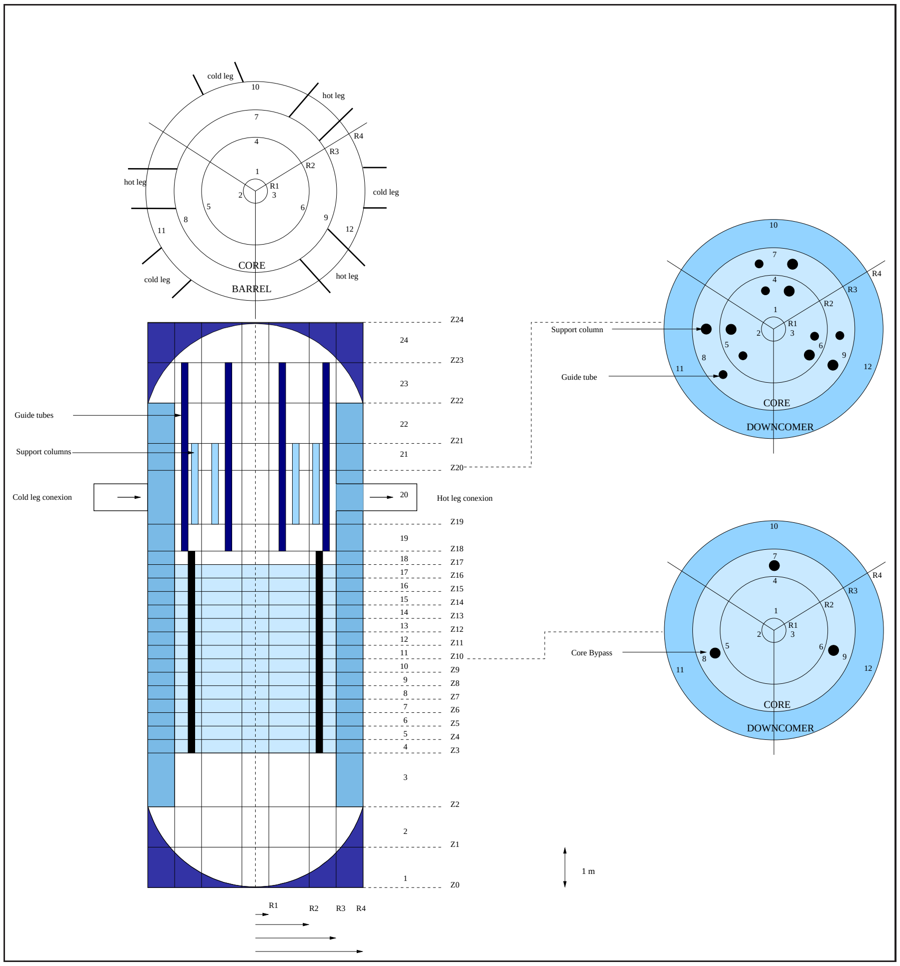

Figure 14: Nodalization of the TRACE model of 3D VESSEL

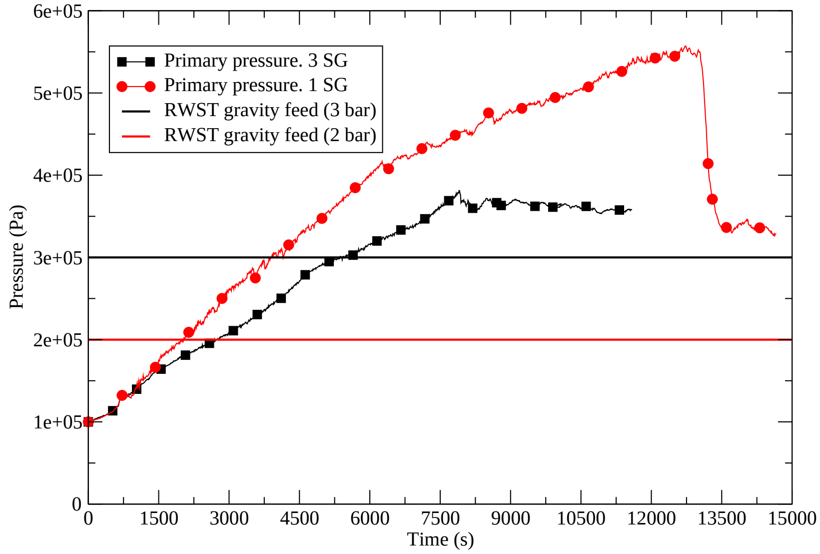

Figure 15: RCS pressure. Transients similar to PKL test F2.2 run 2

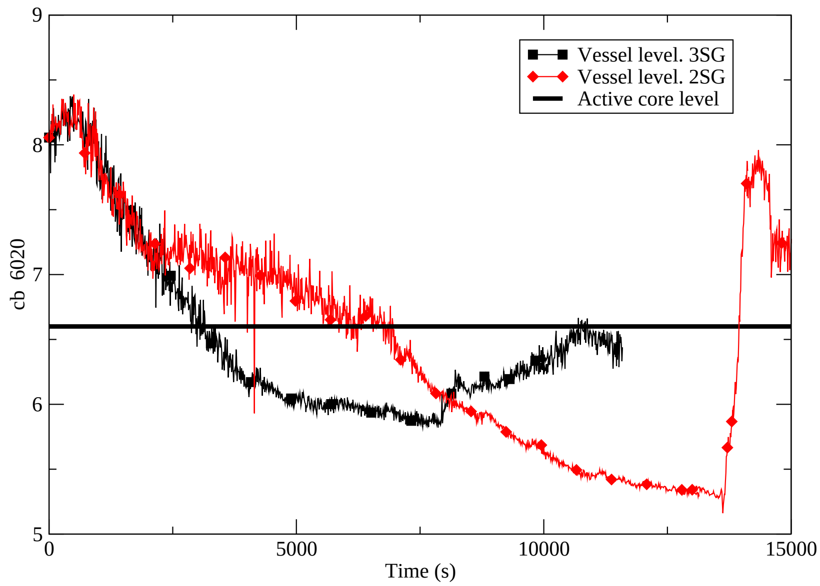

Figure 16: Collapsed vessel level. Transients similar to PKL test F2.2 run 2

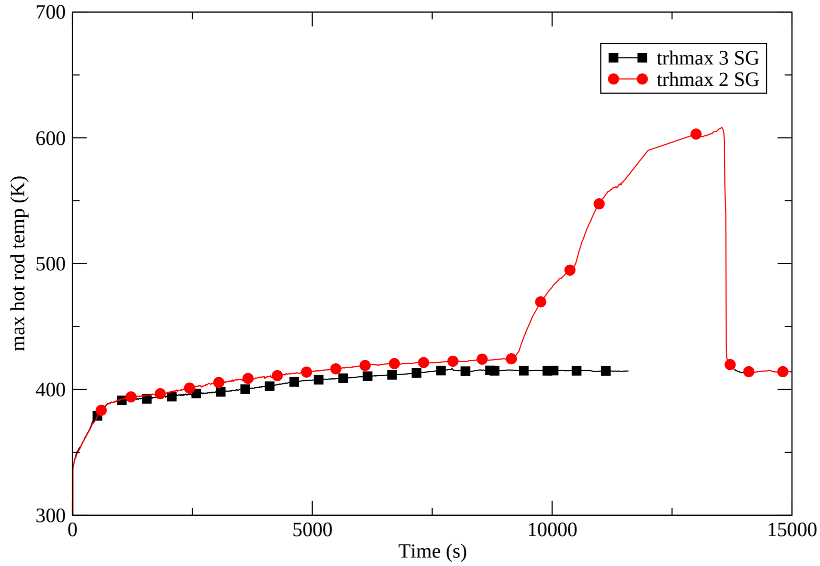

Figure 17: Maximum rod temperature. Transients similar PKL test F2.2 run 2

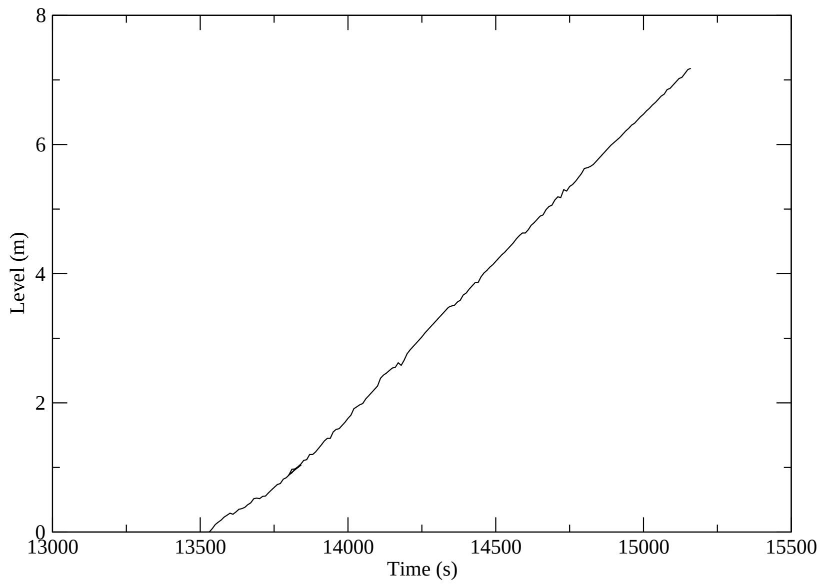

Figure 18: Level of SG3. Transient equivalent to PKL test F2.2 run 2

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.

7 CONCLUSIONS

The PKL III test facility simulates a typical 1300 MWe pressurized water reactor of Siemens / KWU design. In test F2.2 run2, the primary inventory was at midloop level and thus, the influence of noncondensables gases on reflux-condensation cooling mechanism was investigated. In this test was also analyzed the impact of injecting AFW in the two SG initially empty.

In this report, a post-test analysis of PKL test F2.2 run 2 using RELAP5/MOD3 and TRACE codes has been presented. A description of the model inputs are given, and the comparison of measured and calculated results is discussed.

The main findings of the comparison of RELAP5/Mod 3.2 and TRACE 5.0 results with the PKL III F2.2 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.

- A wrong primary mass distribution has been obtained, with a very large water level in the PZR. Probably, it could be interesting to check the heat transfer condensation correlations and offtake correlations for this and other geometries (pressurizer surge line connection with the hot leg and break in the vessel head).

- The main phenomena of AFW injection phase have been also well reproduced.

With respect to the application to a midloop transient in a PWR Westinghouse similar results that in PKL test F2.2 run2 have been obtained.