Annals of Nuclear Energy xxx (xxxx) 107179

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: http://ees.elsevier.com

Modelling guidelines for safety analysis of Station Black Out sequences based on experiments at the PKL test facility

J. Freixa *, V. Martinez-Quiroga, F. Reventós

Department of Physics, Universitat Politècnica de Catalunya (UPC), Spain

ARTICLE INFO

Article history:
Received 10 July 2019
Accepted 29 October 2019
Available online xxx

Keywords
Integral test facility
PWR
Station black out
PKL

ABSTRACT

After the Fukushima accident, "stress-test" activities carried out worldwide pointed out the need to study additional accident management measures to deal with prolonged Station Black Out (SBO) scenarios. Without any operator actions, a total loss of the secondary side heat sink leads to core uncovery, to core damage and ultimately to a melt-down scenario. The international NEA/OECD PKL-3 project has addressed the efficiency of possible accident management actions to re-establish core cooling by experiments at the PKL test facility. Since best estimate system codes were mainly developed to simulate LOCA scenarios, their performance and the general guidelines followed to simulate PWR power plants are called into question. In this paper, RELAP5 simulations of three SBO experiments are presented. An assessment of the code for the particular phenomenology in the experiments have been conducted. Specific guidelines on modelling and a list of the most important sources of uncertainties are provided.

1. Introduction

Station black out (SBO) accidental sequences pose a serious thread to the safety of nuclear installations. In recent years, stress tests have been carried out worldwide to study the effects and preparedness of the nuclear installations against SBO sequences (Queral et al., 2016; Tatu and Kim, 2017). Efforts have been carried out in several countries to improve the safety systems involved in such sequences (Lin et al., 2016). For instance, in Spain, emergency mobile pumps have been prepared to be able to inject water to the secondary systems (París et al., 2019).

While the focus has been put in the design and evaluation of the added systems, little attention has been taken for the assessment of thermal hydraulic system codes employed for the licensing of the newly implemented systems. Most best estimate system codes were mainly developed to address the phenomenology involved in Loss-Of-Coolant Accidents (LOCA) and small break LOCA (SBLOCA) scenarios. The licensing of the Emergency Core Cooling (ECC) systems has been based on the basis that the limiting phenomenology takes place during either LOCA or SBLOCA sequences. During LOCA and SBLOCA sequences, there are large sources of momentum in the system such as, the steam generation in the core, the break choking plane or the condensation in the U-tubes. Considering the presence of such large sources of momentum, the code developers could make several approximation in the momentum balance equation. Because the momentum is driven by external forces, the internal sources of momentum like the viscous stress can be neglected from the equation (The RELAP5 Code Development Team, 2003). In SBO sequences, the core power has been significantly reduced and the SG might be empty which means no condensation in the loops. There still can be relatively high sources of momentum such as leaks at the pump seals or openings of the pressurizer safety or relief valves. However, in most of the possible SBO sequences, the sources of momentum are much smaller than those appearing in LOCA/SBLOCA sequences and therefore, the ability of system codes to reproduce correctly the phenomenology is at stake.

In order to contribute to the efforts in evaluating the effectiveness of the measures to cope with an SBO scenario, the NEA/OECD PKL-3 project included several experiments related to SBO sequences. The NEA/OECD PKL-3 programme is addressed at investigating safety issues relevant for current pressurized water reactor (PWR) plants as well as new PWR design concepts. The main focus is given on transient tests under postulated accident scenarios and systematic parameter studies on thermal hydraulic phenomena. These issues are being investigated by means of thermal–hydraulic experiments conducted at integral test facilities (ITF). Experiments at ITF facilities consitute a vital step in the validation process of system codes for safety analysis and also in the confirmation of the expected phenomenology (Deng et al., 2019). The PKL test facility operated and owned by FRAMATOME constitutes the bulk of the experiments (Umminger et al., 2002). Safety organi-

* Corresponding author.
E-mail addresses: jordi.freixa-terradas@upc.edu (J. Freixa); victor.martinez.quiroga@upc.edu (V. Martinez-Quiroga); francesc.reventos@upc.edu (F. Reventós)

https://doi.org/10.1016/j.anucene.2019.107179
0306-4549/© 2019.

zations, research laboratories and industry from fourteen countries are supporting the PKL project.

One of the PKL test series within the NEA/OECD PKL-3 project addressed current safety issues related to beyond design basis accident transients derived from SBO situations leading to significant core heat-up (test series H2). Without any adequate accident management (AM) procedures the postulated courses of events would lead to a severe accident scenario with core damage. In the H2 experiments, the efficiency of very late initiated AM-measures is investigated. The scenarios are connected with an assessment of the performance of the core exit temperature (CET) which is used as criterion for the initiation of AM measures involving emergency operating procedures and/or severe accident management measures.

In the present work, the three experiments from the PKL H2 series have been simulated with the USNRC RELAP5 thermal hydraulic system code and the capacities of the code to reproduce the phenomenology have been assessed. In addition, the major sources of uncertainty particular of the present scenarios have been identified.

2. Station Black Out PKL Test series (H2)

The PKL test facility, located in Erlangen (Germany), is an experimental power plant facility designed to simulate pressurized water reactors under accidental conditions. The facility replicates the entire primary system and most of the secondary system (except for the turbine and condenser) of a 1300-MW PWR plant, with elevations scaled to 1:1 and volumes and power reduced by a factor of 145. The number of rods in the core and the number of U-tubes in the steam generators (SG) have been scaled by a factor of 145 as well. Unlike many experimental facilities with only two available loops (one to simulate the broken loop and the other to simulate the intact three loops), PKL simulates all four loops separately.

One of the series of experiments of the project NEA/OECD PKL-3 deals with SBO transients with very lately initiated procedures to prevent a core melt scenario. The main objective is to evaluate the efficiency of the AM measures on re-establishing core cooling. Three different experiments have been carried out: H2.1 (Schollenberger et al., 2014) H2.2 (run1) and H2.2 (run2) (Schollenberger et al., 2016).

The three possible procedures to re-establish the core cooling with very late actuation studied within the three tests are:

- Secondary side depressurization followed by a feed in the secondary side
- Primary side depressurization to induce Accumulator (ACC) actuation
- Primary side feed through an emergency Residual Heat Removal (RHR) system

The signals used to initiate the procedures are: high CET, high peak cladding temperature (PCT) and less than 50 degrees of subcooling in all 4 loops.

Another aspect that has been studied is the possible aggravation of the scenario after the injection of non-condensable gases into the reactor coolant system (RCS) which may lead to a reduction of the heat transfer area to the secondary side.

Table 1 summarizes the main boundary conditions of each test. Before the start of the test, the facility is maintained at steady state conditions with subcooled natural circulation in the primary system. The level in the SG has been reduced to mid elevation. Figs. 2–4 show the evolution of the main variables in the three experiments respectively. The tests are initiated by the following actions:

- Loss of the feedwater system in the secondary side
- Loss of the Chemical and Volumetric Control System (CVCS) in the primary system
- Shut down of the pressurizer (PZR) heaters
- Initiation of the core power coast down

Table 1
Boundary conditions of the H2 test series of the NEA/OECD PKL3 project.

Event	H2.1	H2.2 run1	H2.2 run2
Secondary side depressurization	High PZR level 2 SG available (no common header)	High PZR level 4 SG through common header	High PZR level 4 SG no common header
Primary side depressurization	High CET Area = A Valve is kept open afterwards	High CET Area = 1.43·A Valve is kept open afterwards	High CET Area = 1.43·A Closed due to low hot leg temperature or second heat-up
Accumulator	4 ACC available in cold side (26 bar)	4 ACC available in cold side (26 bar)	4 ACC available in cold side (40 bar)
	Injection of nitrogen	Injection of nitrogen avoided	Injection of nitrogen
Secondary feed	High PCT	High CET or high PCT	High CET reading or hot leg Temp below saturation + 50 K
	1 mobile pump in SG1	1 mobile pumps in SG 1 and 2	EFWS SGs 1 and in SG 4 30 min later
Primary injection	High CET	High CET	Not required (RCS closed with RC in U-Tubes)
	1 eRHR in loop 1	1 eRHR in loop 1	
Core power	No radial power profile	No radial power profile	Radial power profile

The initial response of the system is a decrease of the primary pressure due to the core power reduction and because the secondary fill levels are still high enough to extract a significant amount of heat. The level in the SGs falls due to the loss of feed so a deterioration of primary to secondary heat transfer takes place. When the secondary side is no longer able to bring the primary side temperature to the secondary side saturation conditions, the temperature at the core exit starts to rise. Because the CET rises and the primary side pressure decreases, the fluid in the Upper Plenum (UP) is soon in saturation conditions. At this point, steam flows to the Upper Head (UH) and forms a steam bubble. The formation of steam implies a primary pressure rise. The steam bubble in the UH displaces the coolant to the PZR so the fill level rises abruptly. This is a clear indication that the SGs are no longer effective to extract heat from the primary side and thus actions on the secondary relief valves are taken (these actions are slightly different in each test).

Once the primary pressure reaches a predetermined setpoint, the PZR safety valve actuates to avoid a too high pressure and therefore primary side coolant is continuously lost. Natural circulation continues in the primary side, though the steam bubble in the upper part of the reactor pressure vessel (RPV) keeps growing. At some point it spreads to the hot legs and steam flows into the U-tubes. After a short two-phase circulation peak, the accumulation of steam in the U-tubes gradually led to breaking of natural circulation in all loops. Because of the constant loss of coolant, the core is uncovered and cladding temperatures and CET rise triggering the PZR relief valve opening. From this point, each test presents a slightly different evolution due to the differences in the boundary conditions (BC). The evolution of the three experiments from this point is described separately in the following subsections.

2.1. Test H2.1

A sharp primary side depressurization induces steam condensation in the UP and UH, however, this only delays the core heat-up slightly because the area of the valve is not large enough. Depressurization continues until the pressure setpoint of the Accumulators is reached. The injection of cold water into the cold leg induces partial or total loop seal clearing. The loop seal clearing allows the pressure in the hot and cold side to equalize and therefore results in the displacement of coolant into the RPV. The core is partially quenched and the production of steam increases. This leads to a subsequent relaxation of the primary side depressurization and a cease of ACC injection. These two events repeat continuously and one can observe an intermittent injection of the ACC which is not sufficient to quench the core completely. Cladding temperatures keep rising until a high PCT signal is reached and feeding of the secondary side of SG 1 with a mobile pump is initiated. This action enables a secondary heat sink that reduces the primary pressure by condensation in the U-tubes. The primary side depressurization allows an effective discharge of the coolant from the ACC and the core is fully quenched. After a complete emptying of the ACCs, nitrogen is discharged from the ACCs and accumulated in the U-tubes of SG1 following condensation of the steam. The nitrogen deteriorates the heat transfer to the heat sink. A significant amount of steam is still being removed through the PZR valve and thus, water levels in the primary system fall again and a second core heat-up scenario takes place. When the CET is too high, primary side injection is started in the cold leg of Loop 1. With the continuous injection of coolant in the primary side and the partial heat transfer to the secondary side, the core is quenched and the long coolability is guaranteed.

2.2. Test H2.2 (run 1)

The relief valve in Test H2.2 (run1) is 43% larger than in test H2.1 and thus its effectiveness is much greater. The faster depressurization led the primary pressure to the ACC setpoint much faster and allowed a more effective discharge of the accumulators which were sufficient to completely quench the core. The valve remained stacked open throughout the rest of the experiment. The intense generation of steam in the core provoked an increase of the primary pressure so that the accumulators were not fully discharged. Nevertheless, the rather intense discharge of coolant through the PZR valve and the cease of the accumulator injection produced a second reduction of the core level rather early. This second core heat up was addressed with an effective injection in the secondary sides which enabled steam condensation and the injection of the remaining water in the Accumulators. The levels in two steam generators recovered slowly, however with no injection in the primary side and the stuck opened PZR valve, a third core uncovery was unavoidable. Again, once the high core exit temperature was detected, injection to one of the primary loops was initiated which was sufficient to rapidly quench the core and prevent further deterioration of the core conditions.

2.3. Test H2.2 (run 2)

In this experiment, in addition to the increased PZR relief valve, the Accumulators are pressurized at a higher pressure so that the injection starts earlier. These two aspects lead to a very effective cooling of the core and a rather continuous injection of the ACCs. As the penetration of coolant from the ACCs increases, the condensation induced by the injection decreases and so does the depressurization rate due to the PZR valve. As a consequence, there is a pressure build up in the primary system and the ACC injection stops. The pressure build up continues until loop seal clearing occurs, primary pressure experiences a sudden fall and most of the ACC coolant is injected. The primary pressure levels off and a limited amount of water is still available in the ACC tanks. With this condition, primary side inventory is continuously depleted and a second core heat-up sequence commences. When either a high CET or a low hot leg temperature is reached, the Emergency FeedWater System (EFWS) is started in SG 1, simultaneously the PZR relief valve is closed (the signal that initiated the event was high PCT in both the experiment and the calculations). The secondary feed induces condensation in the primary side triggering the injection of the remaining ACC water. In addition, since the PZR valve has been closed, the coolant in the PZR is displaced to the UP causing a rapid quenching of the core. It takes 30 min more to connect the EFWS to SG number 4. With feed in two SG and no leakage in the primary side, decay heat is continuously removed through the secondary side and there is no loss of coolant, which assures a long term coolability.

3. Analysis of the experiments performed with RELAP5

The main goal of thermal hydraulic system codes is to provide safety analysis for nuclear power plants (NPP). Building qualified models that correctly reproduce the events in the particular NPP is a difficult task (Freixa et al., 2016). One of the lessons learned with the use of ITFs for system codes validation is that even though a given code with a given nodalization is able to capture correctly the phenomena occurring in one test, it might not be the case in a later test with different boundary conditions. This can be true even for very similar transients like the ones addressed in the present publication. Therefore it cannot be assumed that the phenomenon will be well simulated in an NPP model. As a matter of fact, it has to be considered that the conditions to be simulated in the NPPs differ from those simulated with the ITF models, so that confidence in the system code and the associated nodalization can be built only after successful simulations of a wide range of tests. Users must not resort to simulate the different tests independently, because if done so, the robustness and the soundness of the model will be lost (Freixa and Manera, 2012). In the present work, the three experiments have been carried out with a single nodalization with only modifications in the boundary and initial conditions.

3.1. RELAP 5 nodalization of the PKL test facility

The starting point of the present analysis is a well validated RELAP5 input deck, developed at the UPC within the SETH and PKL projects. The intensive validation activities, reported in various papers and reports (Freixa et al., 2007; Reventós et al., April 2008; Freixa et al., 2009) have yielded a reliable RELAP5 nodalization of the PKL test facility. Five different PKL tests were successfully simulated with the mentioned nodalization and reported in a PhD thesis by Freixa, 2007.

The UPC PKL-R5mod3.3 nodalization (Fig. 1) has been improved for the SBO analytical exercise. It has been found that the simulation of this type of scenario requires greater detail in some parts of the primary system. SBO scenarios run for very long time, once the pressure is decreased and secondary heat sink is lost, sources of momentum are

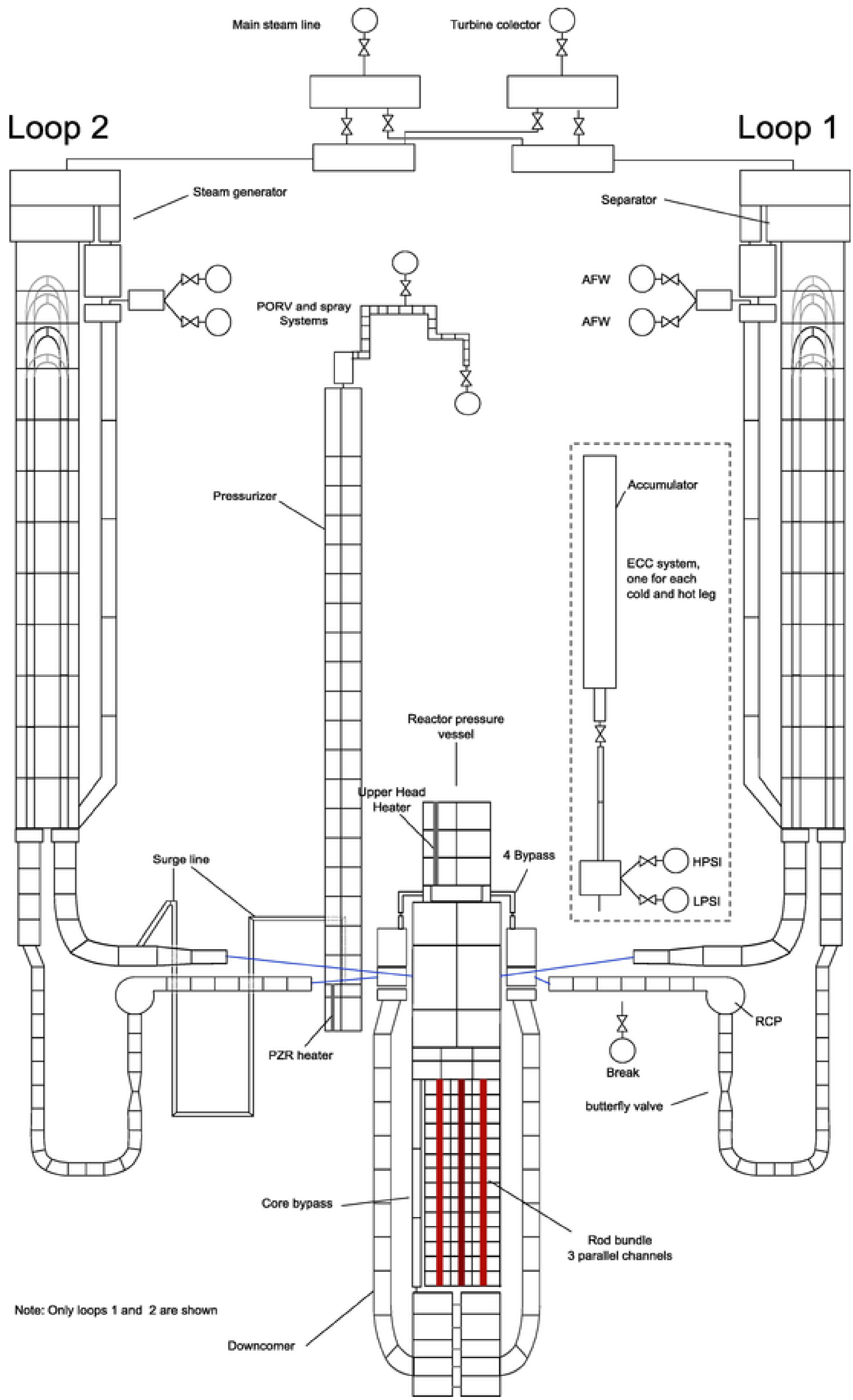
Main steam line
Turbine colector
Loop 2
Loop 1
Steam generator
Separator
AFW
AFW
PORV and spray Systems
Pressurizer
Accumulator
ECC system, one for each cold and hot leg
Reactor pressure vessel
Upper Head Heater
4 Bypass
HPSI
LPSI
Surge line
RCP
PZR heater
Break
butterfly valve
Core bypass
Rod bundle 3 parallel channels
Note: Only loops 1 and 2 are shown
Downcomer

Fig. 1.

rather small so that heat losses and convection heat transfer between the walls and the coolant become a considerable driving force.

The main features of the PKL input deck are:

- 867 volumes
- 961 junctions
- 83 heat structures (HS) with a total of 3643 mesh points

The core region (Fig. 1) has been divided into 4 channels, 3 to simulate the different radial regions of the active core, and the other one for simulating the core bypass. Core outlet is also simulated with three parallel channels in order to keep superheated temperatures distribution at the top of the core if dry out conditions are reached. Core channels are divided into 14 axial levels, and 3 HSs have been deployed to simulate radial power profiles.

The Lower plenum (LP) has been split into two channels, one for each downcomer tube connection. Axially, the LP is divided into 5 levels for preserving the real location of the flow distribution plate and the fuel end fitting plate.

In order to correctly model the CET, the guidelines defined by Freixa et al. (2015) have been followed. CET thermocouples have been modelled by the use of specific HSs. In addition, the UP has been also renodalized in order to place the thermocouples at the same level that is reported in the PKL Test Facility Description. Special attention has been given to the passive HS in the core and UP regions.

The Pressurizer is divided into 19 axial levels with two parallel channels to allow convective flows. The surge-line is defined with a pipe of 18 volumes. The Surge-line inlet connection has been adapted to the patch 4 of RELAP5mod3.3 to activate vapor entrainment and pull-through effects in the connection. The lines from the PZR to the relief and safety valves have also been modeled.

The U-tubes have been nodalized with four channels per SG, two of them reproducing the highest and lowest groups. The relative height of the U-tubes volumes is equivalent to the relative height of the SG riser nodes; hence mesh points of HSs are linked one to one with the hydrodynamic components of each of them.

3.2. Phenomenology and assessment

The following subsections analyze the simulation of the three experiments focusing on the reproduction of the most important phenomenology.

3.2.1. Post-test calculation of Test H2.1

The most relevant results of the post-test calculation of Test H2.1 are shown in Fig. 2. Table 2 shows the chronology of the most relevant events in Test H21. In general the calculation is in good agreement with the experimental data. The primary and secondary side pressures evolve very similarly to those in the test facility. The mass flow through the PZR valve is slightly over-predicted when the control of the primary side pressure starts, this can be observed in Fig. 2 third graph from the top where the integrated mass through the PZR valves is shown. Exactly the same control logic has been implemented in the RELAP5 input deck. Differences appear only when the oscillations in the PZR level are present. After that, the level stabilizes and the integrated mass flow displays the same slope.

The evolution of the PZR level is in good agreement with the experiment. The only discrepancy appears after the Accumulators are fully depleted and more mass inventory is displaced to the PZR which stabilizes at a higher level. The evolution of the first core heat-up takes place with a slight delay in time and the efficiency of the action is more limited. It takes more time for the accumulator water to reach the core. In both, the experiment and the calculation, the accumulators are not sufficient to quench the core and the amount of level provokes an intermittent discharge of the ACC water. The amount of water remaining in the ACC system is very well captured. The second rise of temperature occurs at a similar time and the start of secondary feed leads to the same response in the primary system. This response is: intense condensation, drop of the PZR level and pressure followed by the discharge of the remaining coolant in the ACCs. This phenomenon is very well reproduced. In this case the rapid discharge of the ACC water reaches the core with similar timing and intensity. The condensation in the U-tubes provokes a drop in the PZR level which is later recovered by the ACC discharge.

The second core heat up is slightly advanced in time because the amount of liquid in the primary system is lower due to a slight over prediction of the discharge through the PZR safety valve. In fact, this event takes place with the same amount of water discharged through the PZR valves. The evolution of the event is very similar, reaching a slightly higher PCT. The primary side feeding implies a rise of the PZR level. It has also been found that nitrogen from the ACC accumulates in the U-tubes of SG1.

Table 3 summarizes the phenomenology taking place in both experiments and a comment on whether the post-test calculations succeeded or not to simulate the phenomenon. Most phenomenology has been well reproduced. The major discrepancies between the calculation and the experiments are on the timing of events. The timing is greatly influenced by heat losses in each part of the facility. Those are difficult to estimate with precision and depend as well on the fluid conditions in both the inner and the outer sides. Synergies between the heat losses in the different parts play an important role as well.

3.2.2. Post-test calculation of Test H2.2 (run 1)

The evolution of Test H2.2 run2 is shown in Fig. 3 where the solid lines represent the experimental data whereas the calculation results are drawn with dashed lines. The chronology of the main events is shown in Table 4. The first part of the transient, until the opening of the PZR valve, is very well reproduced although it is noticeable that the increase of PZR level and pressure is slightly faster in the calculation (just after loosing the SGs as a heat sink). Once the high pressure setpoint is reached, the discharge through the PZR valve is overpredicted and therefore the timing of the events taking place later in the transient is conditioned. Most of the events occur with significant time anticipation (see Table 4).

The first core heat up takes place similarly as in the experiment, however, the PCT temperature is higher in the calculation because it takes a little bit longer for the CET signal to be reached. The injection of the ACC water in the calculation is not sufficiently effective to fully quench the core. The decrease of the PZR level after the opening of the PZR valve is well simulated. The second core heat up scenario takes place earlier and the cladding temperatures reach higher values. The second core heat up scenario is managed by injecting water in the secondary side which effectively induces condensation in the U-tubes and allows the remaining water in the ACC to be injected into the primary system. Even though this phenomenon takes place earlier, it is very well captured by the code. Once the accumulators are fully discharged the PZR level increases and both values stabilize at a certain level, the simulation slightly overpredicts this level.

The third core heat up scenario takes place with a significant anticipation in the calculation (700 s). It is important to notice that it occurs

UPC PKL-R5mod3.3 nodalization (only two loops out of four are shown).

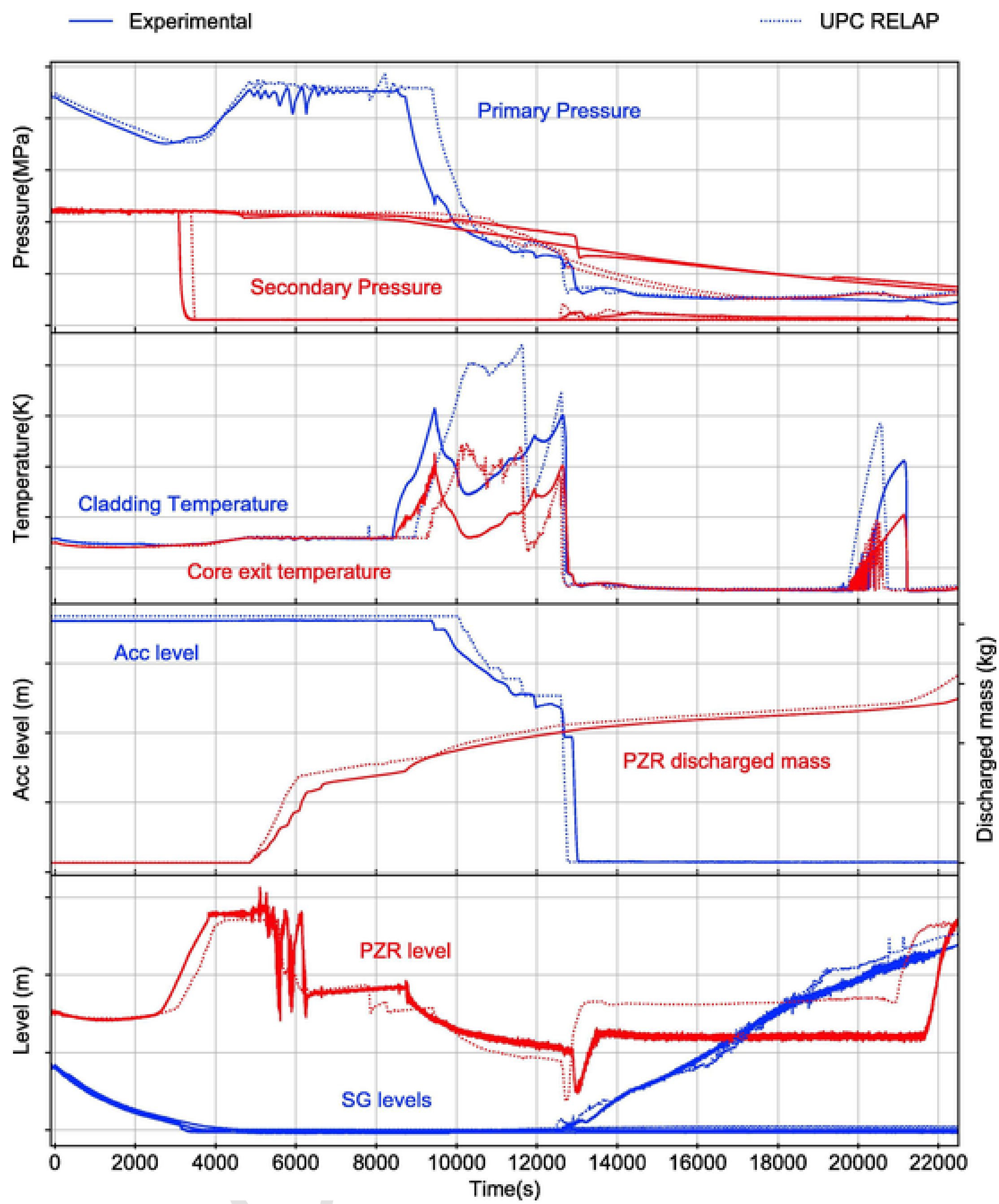

Fig. 2. Post-test calculation of Test H2.1. From top to bottom: a) primary and secondary pressures b) maximum cladding temperature c) Accumulator level and mass discharged through the PZR safety and relief valves d) SG and PZR levels.

with the same mass discharged through the PZR, which means that the balance of pressure drops and inventory in the primary system is well reproduced. This core uncovery is overcome by injecting into the primary system, this is why the PZR level and the mass discharged through the PZR valve increase abruptly at this point. The maximum temperature reached for this third heat up is again overestimated.

Table 5 summarizes the phenomenology and the ability of the model and RELAP5 to simulate it for Test H22 run1. Again, most phenomenology has been well reproduced. The major discrepancies between the calculation and the experiment is on the timing of events.

3.2.3. Post-test calculation of Test H2.2 (run 2)

The results obtained for Test H2.2 run2 are shown in Fig. 4 and Table 6 show the chronology of the most important events in the experiment. The evolution of both the simulation and the experiment are almost the same until the opening of the PZR valve. It is important to notice that as happened for Test H2.2 (run1) the increase of both level and pressure in the PZR prior to the relief valve opening is somehow more pronounced in the calculation. Afterwards, the discharge of mass is again larger than in the experiment which will lead to the anticipation of the events.

In the calculation, there is a short core uncovery at 8124 s that does not reach the safety signal to initiate accident management. The core

Table 2
Sequence of events for Test H2.1.

Event	Experiment	Post-test
Start of Test: Shutdown of feed-water. PRZ heater and CVCS	0	0
Saturation conditions at CET, rapid rise of PRZ level	2565	2792
Secondary side depressurization due to high PRZ level	3085	3390
Begin of primary-side pressure control via PRZ safety-valve	4815	4850
Begin of first core heat-up	8415	8950
Primary-side depressurization due to high CET	8740	9428
Begin of ACC injection	9375	10070
ACC injection halted by pressure supportive effect	11400	11130
Begin of secondary side feed by mobile pump in SG1 due to High PCT	12615	12782
First core heat-up ends	12700	12994
Begin of second core heat-up	20090	19922
Begin of primary-side feed by 1 emergency RHR in cold leg 1 due to high CET	21105	20650
Second core heat-up ends	21195	20937

Table 3
Assessment of RELAP5 for the phenomenology in Test H21.

Event	assessment
Rise of SG outlet temperatures as a consequence of heat transport deterioration.	Well reproduced
Saturation conditions at CET: formation of UH void and rapid rise of PZR level.	Well reproduced
ACC injection and LS clearing in all loops, core only partially quenched	Reproduced but the rise of cladding temperature is more intense
Condensation effect at the ACC injection nozzles	Well reproduced
Intermittent injection of ECC from ACC	Well reproduced
Partial quench of the core after fully opening of PZR relief valve	Well reproduced
Decay power is removed partially by the SG and the PZR valve which leads to third core uncovery	Yes but earlier
Secondary side feeding induces intense condensation in SG1 U-tubes and ensues a fully deliver of the remaining coolant in the ACCs	Well reproduced
Nitrogen is displaced with steam via UH bypass into U-tubes SG1	Well reproduced

level is recovered with a loop seal clearing. One thousand seconds later, the core is uncovered again and this time the CET reaches the setpoint triggering the opening of the PZR valve. Due to the larger discharge through the PZR valve, the mass inventory in the primary system when this first core heat up scenario takes place is lower than in the experiment. The timing is correct but not the mass inventory, in the experiment the loop seal clearing takes place at this time. The early loop seal clearing in the calculation implies a different distribution of the fluid in the system at this time. The primary depressurization is rather fast and the accumulators inject water effectively which leads to the complete quench of the core. This happens very similarly in both the calculation and the experiment, although the cladding temperature in the calculation is somewhat higher. As it happens in the experiment, the accumulators are not fully discharged.

The second core heat-up scenario happens earlier in the calculation but this time with the correct mass inventory in the system. It can be seen in the third graph of Fig. 4 that when the PZR valve is closed, the mass discharged is exactly the same. This second core heat-up is resolved by the injection in one of the SG (secondary side) and the closure of the PZR valve. This two actions simultaneously provoke the discharge of the water in the PZR directly to the UP and the core, quenching the core rapidly. This phenomenon is well simulated although it happens quite earlier.

Table 5 summarizes the phenomenology along with the assessment of the code for Test H22 run2. In the same manner as for run 1, most of the discrepancies are in the timing of the events. (See Table 7).

3.3. Guidelines for modelling and unceratainty analysis

The SBO scenarios analysed within the PKL-3 project are very long transients so that the core power has been significantly reduced; there are no break locations and only either sporadic or temporary openings of relief or safety valves take place. In this situation, sources of momentum are much smaller than in typical LOCA or SBLOCA events. Since best estimate system codes were mainly developed to simulate LOCA scenarios (in their full spectrum) the performance of system codes and the general guidelines followed to simulate PWR power plants are called into question. In addition, heat losses of the different parts of the system may play a significant role.

In order to configure the PKL model so that it can be qualified for the three experiments, the nodalizations had to be improved in several locations. An increased level of detail was necessary in the larger open areas of the system such as the pressurizer, the upper head and the lower plenum. These areas were nodalized with parallel channels in order to enable convective flows. Very precise adjustments of the pressure and heat losses were also needed.

The following is a list of the most significant sources of uncertainty for the analyzed sequences:

- **Heat losses.** In integral test facilities heat losses are more relevant than in full size PWR plants (D'Auria and Galassi, Oct. 2010). In particular, the scaling value of the heat losses in facilities build with the typical power to volume technique is (Kv) 0.5 (Martinez-Quiroga and Reventos, 2014). This means that, for PKL with a Kv of 145, heat losses are 12 times more influential than in the reference PWR. Even though the effect is much reduced, it is recommended to evaluate the influence of heat losses at PWR scale for this type of scenario.
- **UH to DC bypass.** Sensitivities on the flow resistance at the Upper head to Downcomer bypass has seen to have a strong influence in the water levels distribution of the DC and the core. This resistivity has been adjusted throughout the years with the simulation of several experiments and is well adjusted. However, it has been seen that small variations might influence significantly the results in SBO sequences so that this value should be added to uncertainty quantification analysis.
- **Entrainment and mixing in the PZR.** The entrainment and mixing in the upper part of the pressurizer will affect strongly the mass discharged through the PZR valve. The most important discrepancy presented by the calculations was the overestimation of the mass discharged through the PZR valve. Several sensitivities have been carried out to address this difference with little success. The main difference is that to keep the same pressure in the system, RELAP5 needed to discharge more coolant. This indicates that more liquid phase is being entrained to the valve. The pressurizer is a big open space with no internals and the mixing and entrainment models of system codes for large open spaces are not very accurate (D'Auria et al., 2017). This could explain the deficiencies that could end up with differences of more than 2000 s in the time to core damage.
- **ECC mixing and condensation.** Condensation models at the injection locations will affect the effectiveness of the accident management actions. Even though in the present assessment, the condensa-

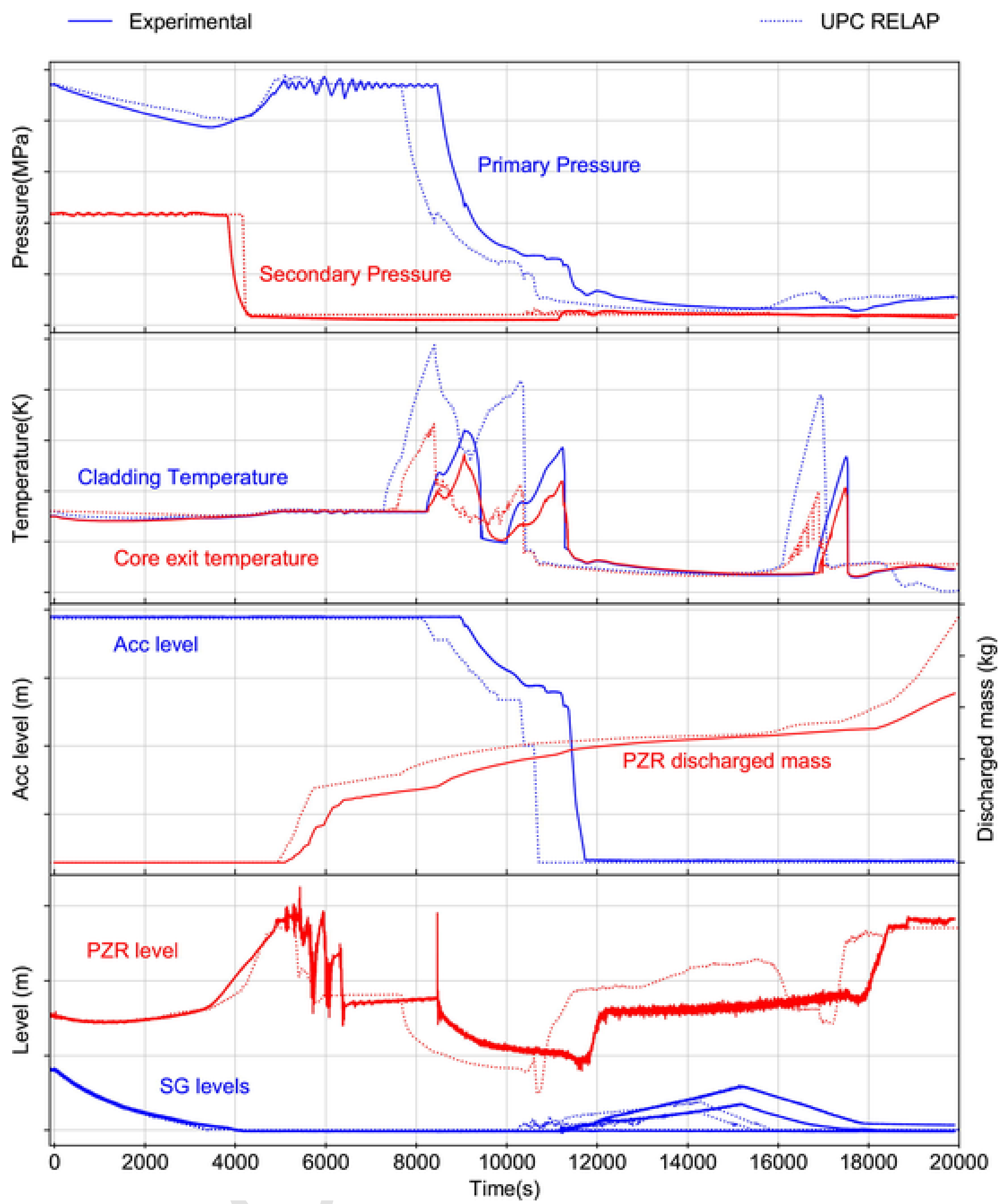

Fig. 3. Post-test calculation of Test H2.2 run 1. From top to bottom: a) primary and secondary pressures b) maximum cladding temperature c) Accumulator level and mass discharged through the PZR safety and relief valves d) SG and PZR levels.

tion models reproduced fairly well the efficiency of the ACC injection, these are a major source of uncertainty.

- **CET modelling.** CET related parameters, the present calculation relied on the modelling guidelines developed by Freixa et al. (2015) for the reproduction of the CET response. The response of the CET is very important in SBO sequences because several actions may rely on the state of the core which is indicated by the CET readings. Therefore, this is a major source of uncertainty.
- **Boundary conditions.** For the SBO sequences, it has been seen that slight modifications in the boundary conditions have a strong impact on the core heat up scenarios. The most relevant boundary conditions have been found to be:
  - SG secondary side levels
  - PZR initial water level
  - ACC conditions: pressure, level and temperature
  - Temperature of the water being fed to the SG or the primary side

4. Conclusions

A well validated input deck of the PKL test facility has been used for the simulation of SBO experiments with very lately initiated procedures to prevent core damage. The main objective of the work is the assessment of the RELAP5 code and the generation of modelling guidelines for application to PWR safety calculations. The activity is part of the Spanish participation to the NEA/OECD PKL-3 project and the CAMP International agreement.

Table 4
Sequence of events for Test H2.2 (run 1).

Event	Experiment	Post-test
Start of Test: Shutdown of feed-water. PRZ heater and CVCS	0	0
Saturation conditions at CET, rapid rise of PRZ level	3300	3900
Secondary side depressurization due to high PRZ level	3840	4150
Begin of primary-side pressure control via PRZ safety-valve	5085	4894
Begin of first core heat-up	8220	7275
Primary-side depressurization due to high CET	8465	7661
Begin of ACC injection	8975	8152
First core heat-up ends	9375	–
Begin of second core heat-up	10050	–
Begin of secondary side feed by mobile pump in SG1 due to High PCT	11135	10212
Second core heat-up ends	11295	10422
End of secondary side feed by mobile pump in SG1	15170	14297
Begin of third core heat-up	16780	16080
Begin of primary-side feed by 1 emergency RHR in cold leg 1 due to high CET	17475	16890
Third core heat-up ends	17550	17065

Table 5
Assessment of RELAP5 for the phenomenology in Test H22 run 1.

Event	Assessment
Rise of SG outlet temperatures as a consequence of heat transport deterioration.	Well reproduced
Saturation conditions at CET: formation of UH void and rapid rise of PZR level.	Well reproduced
Condensation effect at the ACC injection nozzles	Well reproduced
Loop seal clearing concurrent with the ACC injection	Well reproduced
Full quenching of the core after fully opening of PZR relief valve	The core is not fully quenched
Primary side pressure build up due to increased flashing in core, cease of ACC injection	Well reproduced
Strong condensation in U-tubes after SG feeding which induces ACC discharge	Well reproduced
Decay power is removed partially by the SG and the PZR valve which leads to third core uncovery	Yes but earlier
Injection of coolant in the primary system through RHR system in one loop induces rapid quenching of the core	Yes but earlier

The three experiments of the NEA/OECD PKL-3 project have been simulated with a single nodalization model. Each experiment presents a typical SBO sequence for a Konvoi type reactor with slight differences on the boundary conditions. The phenomenology taking place in the three experiments was well reproduced although there were some discrepancies in the timing of the events. However, in order to correctly reproduce the phenomenology involved in the transient, several improvements of the nodalization were needed mainly due to the special characteristics of SBO scenarios. In particular a very detailed nodalization was needed in the RPV, the pressurizer and the U-tubes.

With all these improvements, the simulations have reproduced well all the phenomenology of the three experiments. It is concluded that the code is suitable to simulate this type of sequences although special nodalization approaches are needed. In addition, the major sources of uncertainties have been identified.

The present work shows the complexity of simulating SBO sequences which can only be addressed with advanced best estimate system codes. In addition, the preparation of the model has to take into account the lessons learned in activities like the OECD/NEA PKL-3 project and the associated analytical activities. Non-qualified simulations may lead to large differences between the results and the reality which can be of the order of thousands of seconds for the time of core damage. Regulatory bodies should take the necessary precautions when assessing either Probabilistic or deterministic safety analysis containing SBO sequences.

Acknowledgements

This work was funded by the Spanish Safety Council CSN (Consejo de Seguridad Nuclear). This paper contains findings that were produced within the NEA/OECD-PKL-3 Project. The authors are grateful to the participants and the Management Board of the Project for their consent to this publication.

Appendix A. Supplementary data

Supplementary data associated with this article can be found, in the online version, athttps://doi.org/10.1016/j.anucene.2019.107179.

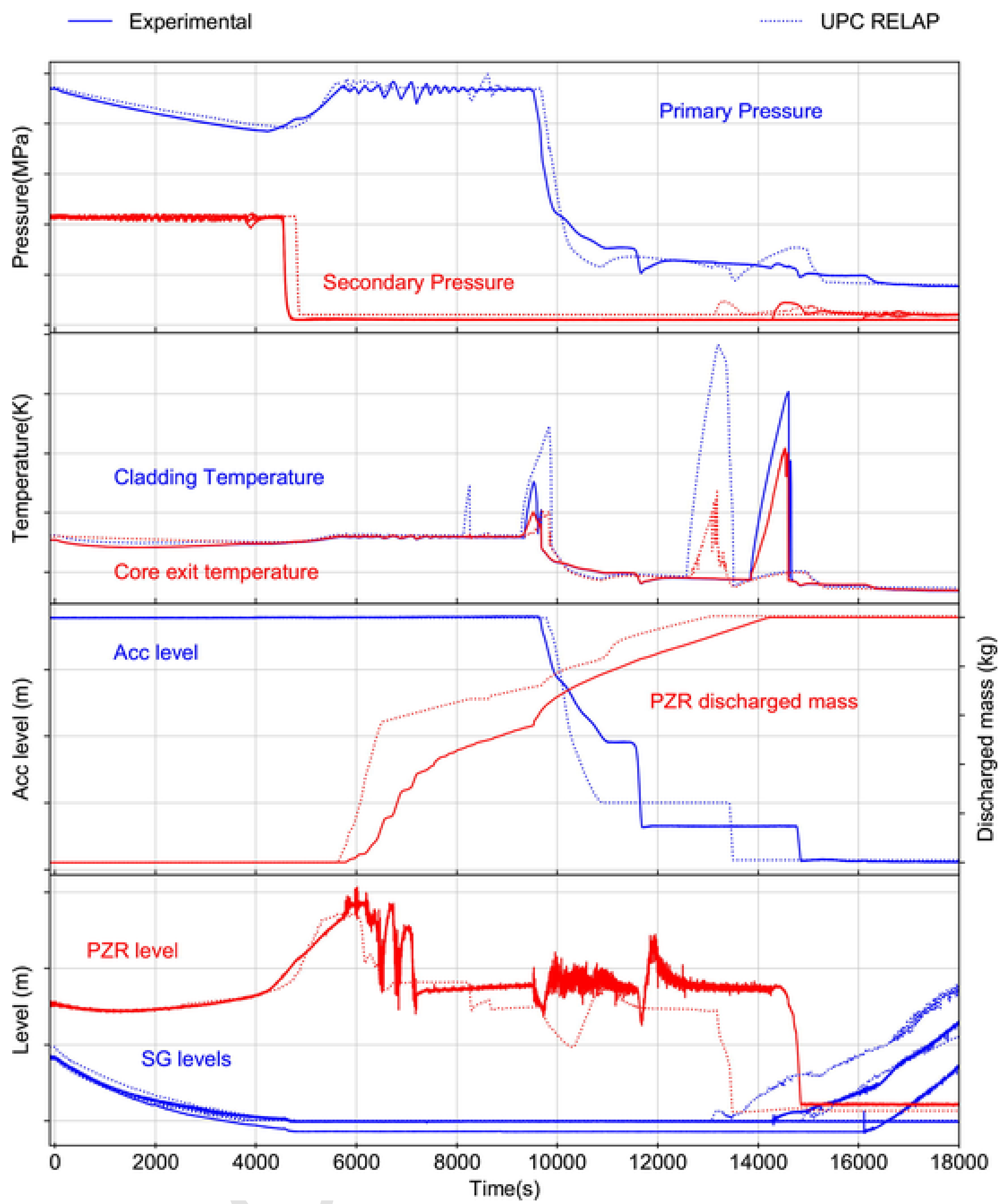

Fig. 4. Post-test calculation of Test H2.2 run 2. From top to bottom: a) primary and secondary pressures b) maximum cladding temperature c) Accumulator level and mass discharged through the PZR safety and relief valves d) SG and PZR levels.

Table 6
Sequence of events for Test H2.2 (run 2).

Event	Experiment	Post-test
Start of Test: Shutdown of feed-water. PRZ heater and CVCS	0	0
Saturation conditions at CET, rapid rise of PRZ level	3950	4428
Secondary side depressurization due to high PRZ level	4550	4778
Begin of primary-side pressure control via PRZ safety-valve	5700	5601
Begin of first core heat up	9260	9285
Primary-side depressurization due to high CET	9525	9682
Begin of ACC injection	9630	9779
First core heat-up ends	9685	9896
Begin of second core heat-up	13825	12556
Begin of secondary side feed by EFWS in SG1 due to High CET	14230	13068
Second core heat-up ends	14725	13519
Begin of secondary side feed by EFWS in SG4	16035	14868

Table 7
Assessment of RELAP5 for the phenomenology in Test H22 run 2.

Event	Assessment
Rise of SG outlet temperatures as a consequence of heat transport deterioration.	Well reproduced
Saturation conditions at CET: formation of UH void and rapid rise of PZR level.	Well reproduced
Condensation effect at the ACC injection nozzles	Yes but overestimated
Loop seal clearing concurrent with the ACC injection	Takes place earlier
Full quenching of the core after fully opening of PZR relief valve	Well reproduced
Primary side pressure build up due to increased flashing in core, cease of ACC injection followed	Phenomenon reproduced but at a different time and ACC mass inventory
Second core heat-up	Reproduced but significantly earlier
Secondary side feeding in combination to the closure of the PZR valve ensues the injection of ACC coolant and displacement of the PZR inventory to the RPV	Well reproduced
Nitrogen is displaced with steam via UH bypass into U-tubes SG1	Well reproduced

References

D'Auria, F., Aksan, N., Bestion, D., Galassi, G.M., Glaeser, H., Hassan, Y.A., Jeong, J.J., Kirillov, P.L., Morel, C., Ninokata, H., Reventós, F., Rohatgi, U.S., Schultz, R.R., Umminger, K., 2017. Thermal Hydraulics in Water-Cooled Nuclear Reactors. Woodhead Publishing Series in Energy.

D'Auria, F., Galassi, G., Oct. 2010. Scaling in nuclear reactor system thermal-hydraulics. Nucl. Eng. Des. 240 (10), 3267–3293. URL: http://linkinghub.elsevier.com/retrieve/pii/S0029549310004280.

Deng, C., Zhang, X., Yang, Y., Yang, J., 2019. Research on scaling design and applicability evaluation of integral thermal-hydraulic test facilities: a review. doi: https://doi.org/10.1016/j.anucene.2019.03.042.

Freixa, J., 2007. SBLOCA with Boron Dilution in Pressuirzed Water Reactors, Impact to the Operation and Safety thesis. Universitat PolitFcnica de Catalunya. (Ph.D. thesis).

Freixa, J., Manera, A., 2012. Remarks on consistent development of plant nodalizations: an example of application to the ROSA integral test facility. Sci. Technol. Nucl. Install. 2012. Article ID 158617. doi:10.1155/2012/158617.

Freixa, J., Martínez-Quiroga, V., Reventós, F., 2016. Qualification of a full plant nodalization for the prediction of the core exit temperature through a scaling methodology. Nucl. Eng. Des. 308, 115–132. URL: http://linkinghub.elsevier.com/retrieve/pii/S0029549316302680.

Freixa, J., Martínez-Quiroga, V., Zerkak, O., Reventós, F., 2015. Modelling guidelines for core exit temperature simulations with system codes. Nucl. Eng. Des. 286, 116–129. URL: http://linkinghub.elsevier.com/retrieve/pii/S0029549315000850.

Freixa, J., Reventós, F., Pretel, C., Batet, L., 2007. Boron transport model with physical diffusion for RELAP5. Nucl. Technol. 160, 205–215.

Freixa, J., Reventós, F., Pretel, C., Batet, L., Sol, I., 2009. SBLOCA with boron dilution in pressurized water reactors. Impact on operation and safety. Nucl. Eng. Des. 239 (4), 749–760.

Lin, H.T., Wang, J.R., Huang, K.C., Shih, C., Chiang, S.C., Liu, C.C., 2016. Station blackout mitigation strategies analysis for Maanshan PWR plant using TRACE. Ann. Nucl. Energy 89, 1–18. doi:10.1016/j.anucene.2015.11.015.

Martinez-Quiroga, V., Reventos, F., 2014. The use of system codes in scaling studies: relevant techniques for qualifying NPP nodalizations for particular scenarios. Sci. Technol. Nucl. Install. 2014, 1–13. URL: http://www.hindawi.com/journals/stni/2014/138745/.

París, C., Queral, C., Mula, J., Gómez-Magán, J., Sánchez-Perea, M., Meléndez, E., Gil, J., 2019. Quantitative risk reduction by means of recovery strategies. Reliab. Eng. Syst. Saf. 182, 13–32. doi:10.1016/j.ress.2018.09.024.

Queral, C., Mena-Rosell, L., Jimenez, G., Sanchez-Perea, M., Gomez-Magan, J., Hortal, J., 2016. Verification of SAMGs in SBO sequences with Seal LOCA. Multiple damage domains. Ann. Nucl. Energy 98, 90–111. doi:10.1016/j.anucene.2016.07.021.

Reventós, F., Freixa, J., Batet, L., Pretel, C., Luebbesmeyer, D., Spaziani, D., Macek, J., Kasahara, F., Umminger, K., April 2008. An analytical comparative exercise on the OECD-SETH PKL 2.2 experiment. Nucl. Eng. Des. 238, 1146–1154.

Schollenberger, S.P., Schoen, B., Umminger, K., 2014. Test PKL III H2.1: Station Black-out with Secondary-side Depressurization and Primary-sde Depressurization, injection from 4 ACC, mobile pump and FPCP. Tech. Rep. Technical Report FANP-PTCTP-G/2014/en/0007, Areva.

Schollenberger, S.P., Schoen, B., Umminger, K., 2016. Test PKL III H2.2: Station Black-out with Secondary-side and Primary-sde Depressurization, 4 ACC (cold side w/ (run1) or w/o (run2) N2 inflow), SG Feed from Mobile pump or EFWS. Tech. Rep. Technical Report FANP-PTCTP-G/2016/en/0021, Areva.

Tatu, A., Kim, T., 2017. Extending SBO coping capability: an improved auxiliary feedwater system. Prog. Nucl. Energy 95, 40–47. URL: http://linkinghub.elsevier.com/retrieve/pii/S0149197016302542.

The RELAP5 Code Development Team, 2003. Relap5/Mod3.3 Code Manual. Volume I: Code Structure, System Models, And Solution Methods.

Umminger, K., Mandl, R., Wegner, R., 2002. Restart of natural circulation in a PWR PKL test results and S-RELAP5 calculations. Nucl. Eng. Des. 215, 39–50.