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Research Article

CATHARE Assessment of PACTEL LOCA Experiments with Accident Management

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This paper summarizes the analysis results of three PACTEL experiments, carried out with the advanced thermal-hydraulic system computer CATHARE 2 code as a part of the second work package WP2 (analytical work) of the EC project "Improved Accident Management of VVER nuclear power plants" (IMPAM-VVER). The three LOCA experiments, conducted on the Finnish test facility PACTEL (VVER-440 model), represent 7.4% cold leg breaks with combination of secondary bleed and primary bleed and feed and different actuation modes of the passive safety injection. The code was used for both defining and analyzing the experiments, and to assess its capabilities in predicting the associated complex VVER-related phenomena. The code results are in reasonable agreement with the measurements, and the important physical phenomena are well predicted, although still further improvement and validation might be necessary.

1. Introduction

This study was carried out in the framework of the EC project "Improved Accident Management of VVER nuclear power plants" (IMPAM-VVER) with participation of Finland, France, Germany, Hungary, Czech Republic, Slovakia, and Bulgaria. The objective of the project was to gain experimental and analytical results in order to improve the safety management practices and provide information for both utilities and safety authorities. In some VVER small break LOCA scenarios, it has been found out that there may be problems to depressurize the primary system in order to allow the emergency core coolant injection from the low-pressure system. The main objective of this project was to investigate which means and criteria for starting depressurization measures, like feed and bleed, would be the most efficient. Also it had to assesse the capability of computer codes like APROS, ATHLET, CATHARE and RELAP to predict the associated complex VVER-related phenomena. The experiments have been performed on the Finnish test facility PACTEL and Hungarian rig PMK-2. This paper presents the modeling and the results of CATHARE calculations, compared to the three PACTEL experiments. More details and results are provided in [1].

2. Description of the PACTEL Experimental Facility

The PACTEL experimental facility (Figure 1) was designed to model the thermal-hydraulic behavior of VVER-440-type pressurized water reactors (PWR). These reactors have several unique features that differ from other PWR designs. PACTEL simulates all the major components and systems of the reference VVER-440, making it a realistic tool to examine a broad range of postulated accidents and operational transients [2].

PACTEL is a volumetrically scaled (1 : 305) facility including core, cold and hot legs, steam generators, main coolant pumps, pressurizer, high- and low-pressure emergency core cooling systems, and hydro-accumulators. The maximum operating pressures on the primary and secondary sides are 8 MPa and 4.6 MPa, respectively. The corresponding values in VVER-440 are 12.3 MPa and 4.6 MPa. The reactor vessel is simulated with separate downcomer and core sections. The core itself consists of 144 full-length, electrically heated fuel rod simulators with a heated length of 2.42 m. The axial power distribution is a chopped cosine with a peaking factor of 1.4. The maximum total core power output is 1 MW, 20% of scaled full power. The fuel rod pitch (12.2 mm) and diameter (9.1 mm) are identical to those of the reference reactor. The rods are divided into three roughly triangular-shaped parallel channels representing the intersection of the corners of three hexagonal VVER rod bundles.

Component heights and relative elevations correspond to those of the full-scale reactor to match the natural circulation gravitational heads in the reference system. The hot and cold leg elevations of the reference plant have been maintained, including the loop seals. To preserve flow regime transitions in the horizontal sections of the loop seals under two-phase flow conditions, the Froude number has been applied to select the diameter and length of the hot and cold legs. Three coolant loops with double capacity steam generators are used to model six loops of the reference power plant. The steam generators (SG) have vertical primary collectors and horizontal heat exchanging tubes. The external and internal SG tube diameters are 16 mm and 13 mm as in real NPP. The scaled heat transfer area of the tubes is preserved. Secondary side steam production is vented through control valves directly to the atmosphere.

PACTEL experimental facility

Figure 1: PACTEL experimental facility

3. PACTEL Modeling by CATHARE

The calculations have been performed with the system thermal-hydraulic code CATHARE 2, version V1.3L_1.

The input data deck has been prepared on the basis of the CATHARE nodalization [3] of the PACTEL facility, which was used for ISP-33 analysis [4]. The input model has been modified in order to correspond to the PACTEL state [2] of the experiments.

The main modifications are as follows:

(i) the full-length steam generators have been replaced by the model of Large Diameter SGs with shorter heat exchange tubes but with real SG collectors,

(ii) main coolant pumps have been added,

(iii) ECCS has been modeled (Hydro-Accumulators and LPSI pump).

The three real loops of PACTEL are modeled *separately* because of some differences in the lengths, elevations, and so forth.

The *core vessel* (Figure 2) is modeled by an *average core channel* with 11 axial meshes and weight 144 and a *bypass* with 11 axial meshes. The model of the upper plenum consists of a volume with 2 core and bypass inlet junctions and 3 outlet junctions (hot legs).

The *pressurizer* presents a volume with an external wall and 3 internal walls, modeling the heaters. For the modeling of the *steam generator* a *multitube* approach is applied. The heat exchange tubes of every SG, primary side, are presented as *9 axial elements*, located at different horizontal elevations. Every axial element is divided in *10 meshes*. The *pressurizer* is connected in the hot leg of *Loop 1*. The *break* is located in the *cold leg* of *Loop 3* close to the reactor vessel.

As a whole, the primary side contains 92 junctions, 1 tee element, 10 volumes, and 40 axial elements with 539 segments. Figure 3 illustrates the modeling of the primary circuit of PACTEL.

The *secondary side* of the SG is presented by *recirculation* model. Every one of the secondary circuits comprises 4 junctions, 1 volume for the steam dome and 2 axial elements, modeling the SG liquid pool and the steam line with 22 segments.

The *heat losses* to the environment are modeled based on the information of the previous PACTEL configuration with some corrections taking into account the PACTEL heat losses test [5] (e.g., 9.5 kW per RCP etc.).

In the junction between the core, and upper plenum the CATHARE *Kutateladze model for CCFL* has been applied. The CATHARE CCFL operator allows the user to specify the parameters M, C, E, and X in the flooding equation:

$$\left[J_G^* Bo^{E/2}\right]^X + M\left[J_L^* Bo^{E/2}\right]^X = C, \qquad (1)$$

where Bo is the Bound number and J_G^* and J_L^* are the dimensionless superficial velocity of gas and liquid, respectively.

The peak cladding temperature is very sensitive to CCFL model and plays an important role in the considered scenario of the transient.

4. Results of the CATHARE Calculations and Comparison with the Experiments

4.1. Test T2.1 Analysis. The test T2.1 represents a *7.4% (7.8 mm) cold leg break with secondary bleed and primary*

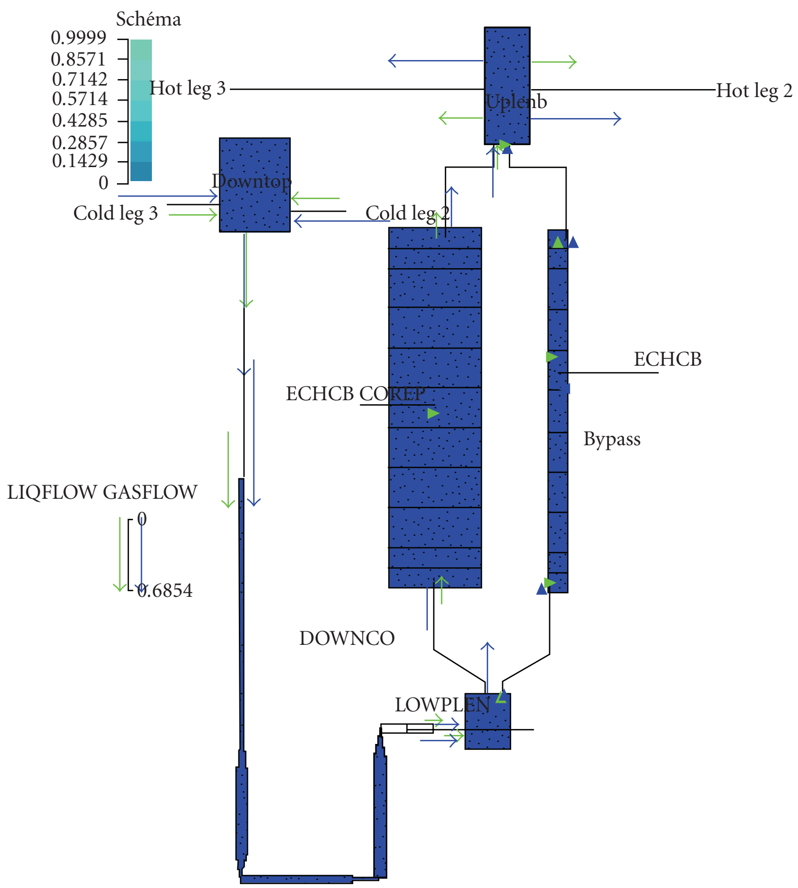

Figure 2: CATHARE core model of PACTEL.

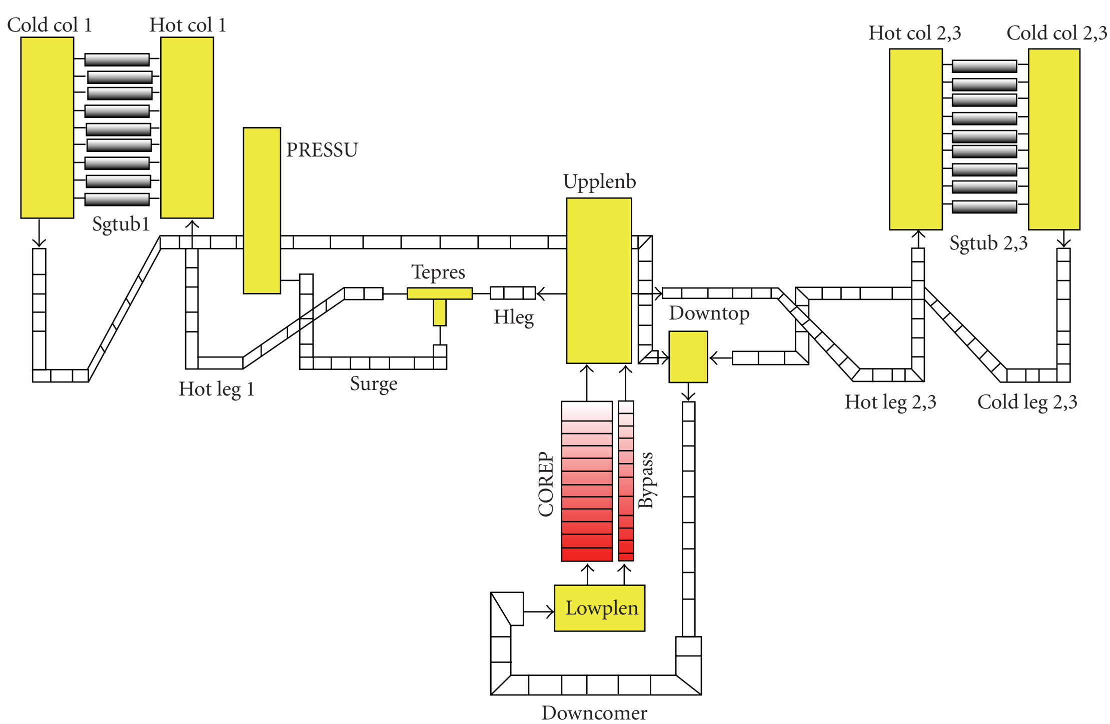

Figure 3: CATHARE nodalization of the PACTEL primary circuit.

bleed and feed. The bleed and feed occur if predefined core heat-up takes place. Regarding the ECCS configuration, the hydro-accumulators and the LPSI pump are available [6].

> *The objective of Test T2.1 was to investigate whether the primary pressure can be reduced to the LPSI delivery pressure without high-pressure injection in LOCA scenario.*

The initial primary pressure in the experiment was close to the maximum operating pressure of the facility and the lower-maximum power was compensated by decreasing the primary mass flow so that temperature distribution in the initial phase in the facility is as close as possible to the nominal temperature distribution in the plant.

The main conditions of the test are the following:

(i) the test is started from nominal conditions of the loop by opening the break in cold leg and initiating simultaneously:

   (a) scram,
   (b) steam line and feedwater isolation,
   (c) pump coast down;

(ii) injection of 1 accumulator to upper plenum starts, and 2 accumulators to downcomer;

(iii) secondary bleed starts at $T_{wall} > 350^{\circ}C$;

(iv) primary bleed starts if $T_{wall} > 400^{\circ}C$;

(v) LPSI starts at $P < 0.7$ MPa;

(vi) test is terminated if $T_{wall} > 450^{\circ}C$.

The sequence of the main events of the pretest and posttest calculations and comparison with the measured parameters are provided in the Table 1.

Regarding the boundary conditions, the posttest calculations are based on the specification and measurements of test T2.1. In the posttest analysis also some modification of the singularity in the hydroaccumulator line modeling has been introduced in order to get better timing in the prediction of the maximal fuel cladding temperature, although even in the pretest calculations the timing and the amplitude of the core heat up were predicted quite well.

Due to the coolant leakage, a rapid primary pressure drop takes place (Figure 4). The pressure calculations are in good agreement with the experiment. Some overprediction can be observed during the HA injection phase.

The primary pressure decrease below 5.5 MPa (after 50 seconds) leads to Hydro-Accumulators injection (Figure 5), which lasts until 450 seconds. In the calculations, the HA injection is relatively well predicted, but is slightly longer compared to the experiment.

Intensive core boiling takes place (Figure 6). The core liquid mass is going down and steam mass is increasing. The liquid flow in the core, downcomer and loops is stagnating around zero.

After the emptying of the HA, the further decrease of the primary mass inventory leads to core uncover, and core heat up starts at t = 940 seconds in the test, at t = 920 seconds in the posttest calculations (Figure 7), and at 900 seconds in the pretest. The timing and the temperature peaks are very well predicted by the calculations.

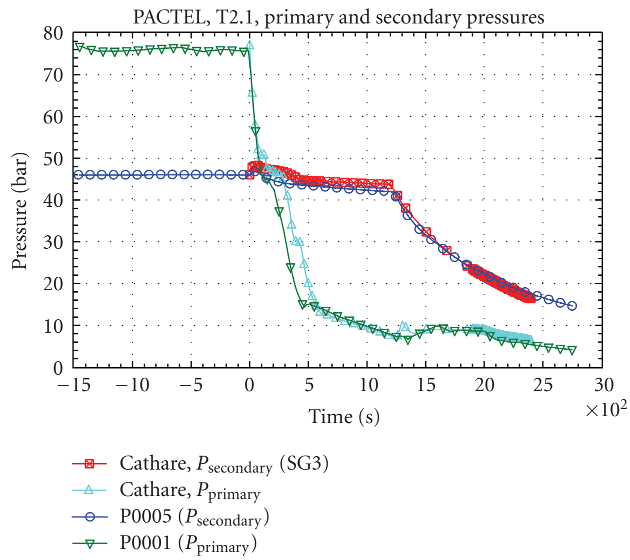

FIGURE 4: Primary and secondary pressures.

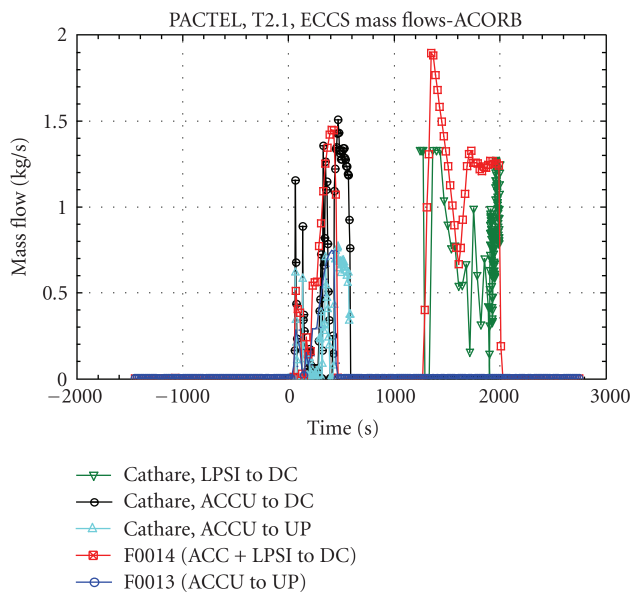

FIGURE 5: ECCS mass flows.

According to the scenario when the maximal cladding temperature exceeds 350°C, the operator starts the secondary bleed by opening the steam dump device to the atmosphere BRU-A (t = 1230 seconds experiment, t = 1202 posttest, t = 1166 seconds pretest). The calculated secondary pressure is decreasing in good agreement with the experiment (Figure 4).

The further decrease of the primary pressure leads very soon after the operator intervention to LPSI pump injection, (Figure 5, t = 1250 seconds test, t = 1244 seconds posttest

Table 1: Test T2.1: Timing of the main events, CATHARE versus experiment

Event	Time [S] Exp.	Time [S] Pretest	Time [S] Posttest	Comment
Start of calculation	−1500	−5000	−5000	Stabilization
Opening break valve	0	0	0	7.4% cold leg break (∅ 7.8 mm)
Reactor scram	0	0	0	
Pumps coast-down	0	0	0	Coast-down linear (0–150 s)
Isolation of feedwater and steam lines	0	0	0	Closing time is 3 s and 10 s, respectively
Pressurizer heaters off		18.9	18.9	Level in PRZ < 2.7 m
ACCU injection initiated to Downcomer to Upper plenum	50	60 61	60 61	Primary pressure < 5.5 MPa
END of HA injection to UP to DC	450	582 586.4	582 586	HA empty
Increase of fuel cladding temperature start	940	900	920	Core uncovery and heat-up start
Secondary bleed	1230	1166	1202	Cladding temperature > 350°C,
Primary bleed	—	—	—	Cladding temperature > 400°C,
LPSI start	1250	1205	1244	Primary pressure < 0.7 MPa
Maximal fuel cladding temperature	1261	1215	1267	Tclad, exp = 379°C Tclad, calc = 394°C posttest Tclad, calc = 391°C pretest
LPSI pump switched off	2000		2000	
End of test	2750	3600	2400	

and $t = 1205$ seconds pretest). With the LPSI, the core heat up is stopped, core quenching occurs, and the $T_{\max}$ is going down (Figure 7). So $T_{\max}$ does not reach the criterion for primary bleed (400°C), neither in the test nor in the posttest and pretest calculations. A stable cool down of the reactor vessel and primary circuit is achieved without primary bleed. It should be noted that the threshold for LPSI (0.7 MPa) could be reached even without operator intervention.

4.2. Test T2.3 Analysis. The test T2.3 is similar to test T2.1, but the pressure set-point for hydroaccumulators injection is lower: *3.5 MPa instead of 5.5 MPa* and the *water volume is increased*.

> *The objective of Test T2.3 was to investigate whether the primary pressure can be reduced to the LPSI pressure without high-pressure injection in LOCA scenario and if delayed hydroaccumulators injection with lower pressure set-point is more favorable for core cooling (to reach LPSI before core overheating occurs).*

Until 500 seconds, the primary pressure is very well predicted (Figure 8). Between 500 seconds and 900 seconds, the pressure drop is faster in the experiment than in the calculations. Probably this is related to the start of HA injection and stronger condensation in the experiment than in the calculations in the upper plenum. The cold water penetration into the core is quite sensitive to the CCFL modeling in CATHARE (strong dependence on the geometry and corresponding relationships).

The primary pressure decrease below 35 bars (after 422 seconds in the test and 441 seconds in the calculations) leads to Hydro-Accumulators injection.

Intensive core boiling takes place. In the calculations a small core uncovery occurs between 287 seconds and 441 seconds, which is not observed in the experiment. The core liquid mass is going down and steam mass is increasing. The liquid flow in the core and in the down comer is stagnating around zero. The maximal fuel cladding temperature (Figure 9) remains below the threshold values to begin operator actions of secondary and then primary bleed (350°C and 400°C, resp.).

The further decrease of the primary pressure leads to LPSI pump actuation. It should be noted that the threshold for LPSI (0.7 MPa) has been reached without operator intervention. A stable cool down of the reactor vessel and primary circuit is achieved (Figures 9 and 10) without

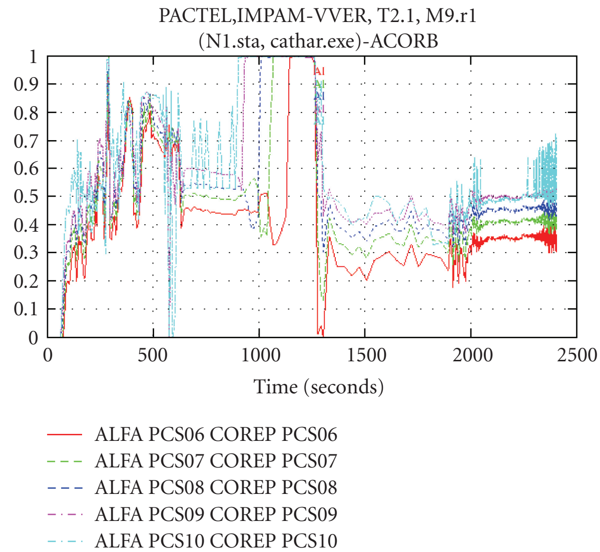

Figure 6: Void fractions (core upper half part).

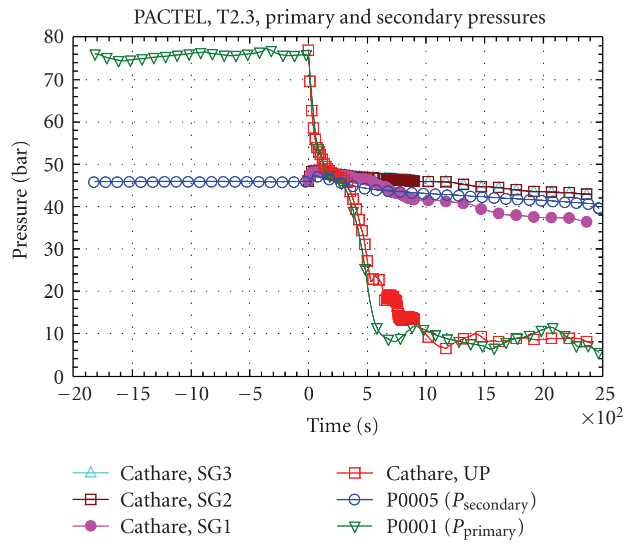

Figure 8: Primary and secondary pressures.

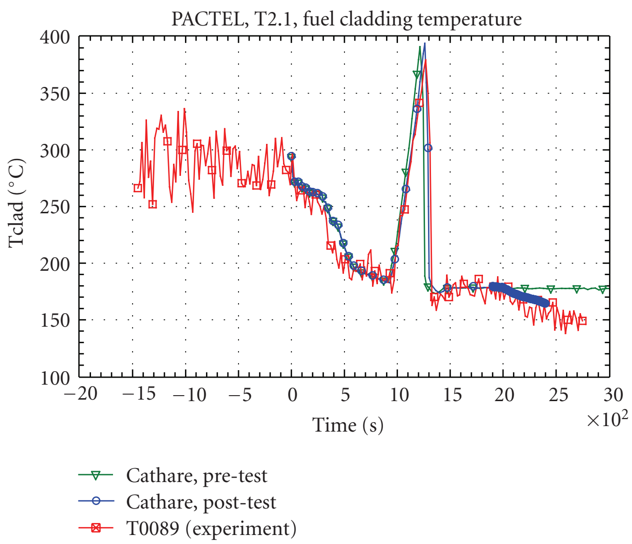

Figure 7: Fuel cladding temperature.

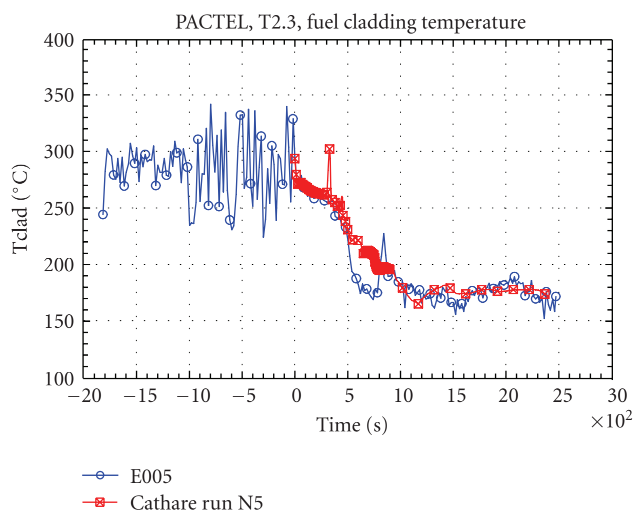

Figure 9: Fuel cladding temperature.

secondary and primary bleed. The delayed HA injection (reduced HA pressure set-point) had a favorable effect on the core cooling: no overheating occurred.

4.3. Test T3.2 Analysis. The test T3.2 is similar to test T2.1, but *secondary bleed is not actuated even if there are conditions to start it*. The primary bleed and feed occur if predefined core heat-up takes place. Based on the experience from the earlier tests the temperature criterion to start the primary bleed was *increased*.

> *The objective of Test T3.2 was to investigate the effect of low-pressure injection in the conditions in which the secondary side pressure remains high.*

The main conditions of the test are similar to T2.1 with exception of the following:

(i) no secondary bleed starts even if $T_{wall} > 350°C$;

(ii) primary bleed starts if $T_{wall} > 500°C$;

(iii) test is terminated if $T_{wall} > 550°C$.

The calculated primary and secondary pressures are in good agreement with the measurements (Figure 11). Small overprediction of the primary pressure can be observed during the HA injection period. It is due probably to the modeling of HA and some underestimation by CATHARE of the condensation effects.

The primary pressure decrease below 55 bars leads to Hydro-Accumulators injection (Figure 12), which is well predicted by the pretest and posttest calculations.

With the LPSI, start the break flow is increasing again (Figure 13). The comparison of the calculated and measured break flows shows a good agreement.

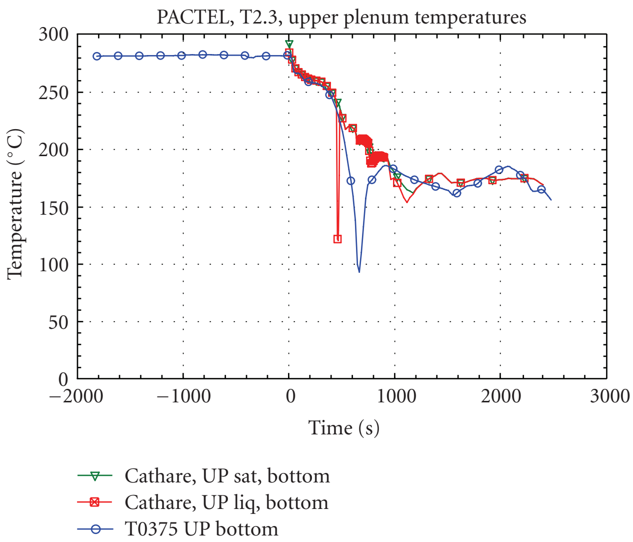

Figure 10: Upper plenum temperatures.

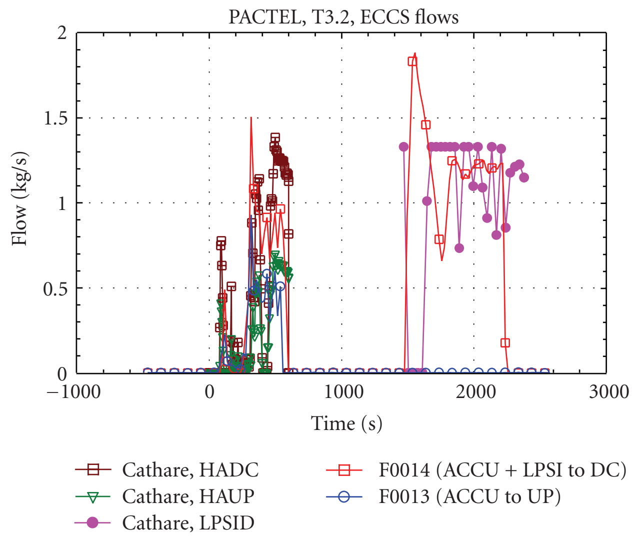

Figure 12: ECCS mass flows.

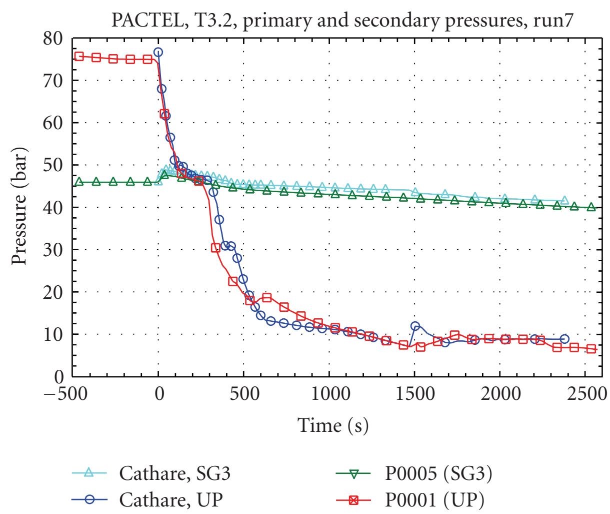

Figure 11: Primary and secondary pressures.

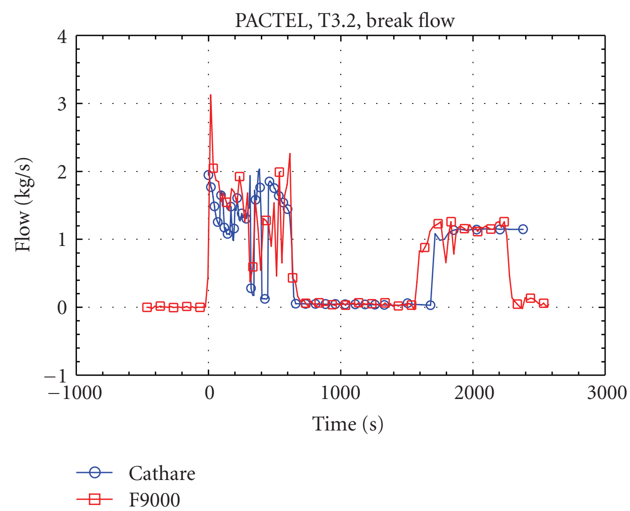

Figure 13: Break flow.

After the emptying of the HA, the further decrease of the primary mass inventory leads to core uncover, and core heat up starts at $t = 1082$ seconds in the test, at $t = 900$ seconds in the pretest calculations, and 1055 seconds in the posttest (Figure 14). The maximal fuel cladding temperature is achieved at 1476 seconds in the experiment and 1470 seconds in the posttest. So the posttest results have been largely improved in the timing and amplitude and a very good agreement can be observed. It should be pointed out that $T_{\max}$ is an extremely sensitive parameter.

The further decrease of the primary pressure ($P1 < 0.7$ MPa) leads to LPSI pump injection, which is well predicted in the posttest calculations (Figure 12). With the LPSI, the core heat up is stopped, core quenching occurs, and the cladding temperature is going down (Figure 14). So $T_{\max}$ does not reach the criterion for primary bleed (500°C), neither in the test nor in the posttest and pretest calculations. A stable cool down of the reactor vessel and primary circuit is achieved without primary bleed. The threshold for LPSI was reached without operator intervention.

5. Conclusions

The main objective of the IMPAM-VVER project was to investigate experimentally and analytically the means and criteria in case of SB LOCA to depressurize the primary circuit to the value of the LPSI pump head without high-pressure injection before core heat up takes place. The available measures for cooldown and pressure reduction are the hydroaccumulator injection and operator actions of secondary bleed and primary feed and bleed.

Correct definition of the initial and boundary condition of the tests is important for the proper code predictions. Global parameters as pressures, mass inventory, and so forth. are less sensitive compared to fuel cladding temperature,

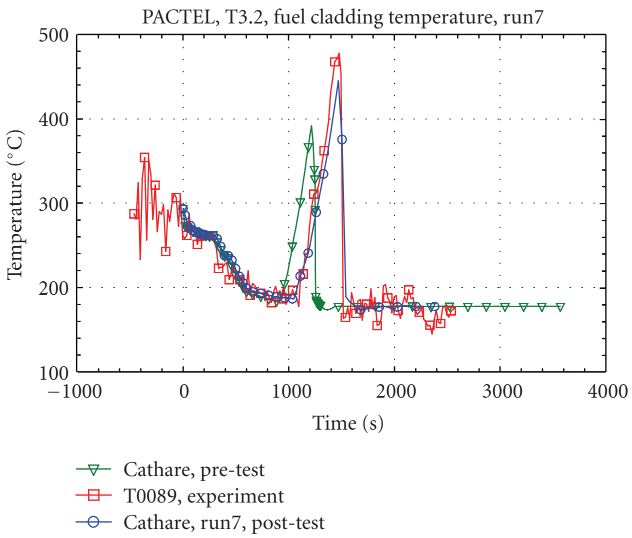

Figure 14: Fuel cladding temperature.

which is a key criterion in the safety studies and in the test scenarios.

The investigated break size of 7.4% is close to the spectrum of intermediate break LOCAs. Because of the relatively big break size, it was observed relatively fast that primary pressure decrease and the value of LPSI pump head (0.7 MPa) were reached even without operator actions in the code calculations as in the tests.

In the code calculations of test T2.1 and T3.2 (higher HA pressure set-point actuation), boiling crisis and core heat up took place, but the maximal heater rod wall temperatures did not exceed the predefined criteria for primary bleed. Timing and value of T_{max} are well computed by CATHARE code.

With the start of LPSI, the core heat up is stopped, core quenching occurs and the maximal cladding temperature starts to decrease. So T_{max} does not reach the criterion for primary bleed, neither in the tests nor in the calculations.

Test T2.3 was carried out with delayed HA injection (reduced HA pressure set-point). This measure had a favorable effect on the core cooling: no overheating occurred. T_{max} remained below 350°C. This effect was reproduced well by CATHARE code calculations. No secondary and no primary bleed took place.

Acknowledgments

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