Progress of ASTEC Validation on Circuit Thermal-Hydraulics and Core Degradation

G. Bandini[1], M. Buck[2], W. Hering[3], L. Godin-Jacqmin[4], G. Ratel[4],
P. Matejovic[5], M. Barnak[5], G. Paitz[6], A. Stefanova[7],
N. Trégourès[8], G. Guillard[8], V. Koundy[9]

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1) ENEA, Bologna (IT)
2) IKE, Stuttgart (DE)
3) FZK, Karlsruhe (DE)
4) CEA, Cadarache (FR)
5) IVS, Trnava (SK)
6) BUTE, Budapest (HU)
7) INRNE, Sofia (BG)
8) IRSN, Cadarache (FR)
9) IRSN, Fontenay-aux-Roses (FR)

Summary

Within the SARNET project, the severe accident code ASTEC is under validation against experiments and analytical results from other integral and/or mechanistic codes. Many applications concerning circuit thermal-hydraulics and core degradation have been performed on various integral and separate-effect experiments for the CESAR thermal-hydraulic module (experiments: BETHSY 9.1 b, PACTEL ISP 33 and T2.1, PMK2, LOFT LP-FP-2) and the DIVA core degradation module (experiments: CORA-13 and -W2, QUENCH-11 and -13, LOFT LP-FP-2, Phébus FPT-4, FARO L14 and L28, LIVE-L1, OLHF-1, FOREVER EC2) for validation purposes. Besides, the TMI-2 accident has been analyzed using the CESAR and DIVA modules for validation.

The suitability and capability of new or improved models implemented in successive code versions up to ASTEC V1.3R2 delivered in December 2007 have been evaluated. The emphasis of the new or improved models in CESAR concern reflooding of an intact core, condensation in the pressurizer, sub-critical break flow correlation, and a new pressurizer spray model. Improvements of DIVA mainly concern the models of the corium behavior in the lower head and of lower head mechanical failure.

The paper will present and discuss the progress in ASTEC validation with respect to circuit thermal-hydraulics and core degradation, mainly focussed on the use of the version ASTEC V1.3R2. Good results have been obtained with CESAR on the integral LOFT LP-FP-2 test and on the two PACTEL experiments which cover various thermal-hydraulic flow regimes. These good results have been confirmed by the intensive validation work done on BETHSY integral tests. The results are good for early-phase DIVA models of core heatup, oxidation and hydrogen production (before any quenching phase) on different CORA, QUENCH and LOFT LP-FP-2 experiments. For the late-phase DIVA models, the results can be considered as good regarding debris bed melting (Phébus FPT4), corium fragmentation at slump in vessel lower plenum (FARO), molten pool behaviour in lower plenum (LIVE-L1), and vessel lower head mechanics (OLHF-1 and FOREVER EC2). Furthermore, the first two phases of the TMI-2 accident before core reflood are very well calculated by ASTEC. Major remaining weaknesses are found in the late phase degradation and the reflooding of a degraded core. Implementation of improved debris bed and magma models is in progress for the ASTEC V2 version, which will allow a more realistic simulation of late phase phenomena up to the failure of the lower head.

A. INTRODUCTION

ASTEC is an integral code jointly developed by IRSN (FR) and GRS (DE) to assess the whole sequence of a severe accident in nuclear power plants, from the initiating event up to fission product release and behaviour in the containment. The code consists of several coupled modules, each one of them dealing with different severe accident phenomena. Among them, the CESAR module, which computes the two-phase thermal-hydraulics in primary and secondary systems, is coupled to the DIVA module able to calculate core degradation, melt relocation and corium behaviour in the lower head up to vessel failure. Most DIVA models are issued from the ICARE2 IRSN mechanistic code for core degradation [1], except some fast-running models that were specifically developed (core gas 2D thermal-hydraulics and corium behaviour in the lower plenum).

Many partners of the SARNET project are involved in ASTEC code validation against experiments. This paper deals with the main work performed so far within the ASTEC topic of SARNET for CESAR and DIVA module validation.

B. PROGRESS OF ASTEC CODE VALIDATION

The progress of ASTEC code validation against experiments regarding CESAR and DIVA modules and their coupling is presented in this section. The validation tasks performed by SARNET partners are summed up in Table 1 below.

Table 1: CESAR and DIVA module validation tasks

Module	Phenomena	Partner	Experiment
CESAR	RCS thermal-hydraulics	IRSN/Cad.	BETHSY 9.1b
		IVS	PACTEL ISP 33 and T2.1
		BUTE	PMK2 - SBLOCA
DIVA	Core degradation	IKE	CORA-13 and -W2, Phébus FPT4
	Core reflooding	FZK	QUENCH-11 and -13
	Core reflooding	INRNE, ENEA	QUENCH-11
	Corium fragmentation	IRSN	FARO L14 and L28
	Corium in lover head	CEA	LIVE-L1
	Lower head mechanics	IRSN	OLHF-1, FOREVER EC2
CESAR + DIVA	RCS thermal-hydraulics and core degradation	ENEA	LOFT LP-FP-2, TMI-2

B.1 RCS Thermal-Hydraulics

The suitability and capability of new or improved models implemented in successive code versions up to ASTEC V1.3R2 delivered in December 2007 have been evaluated. The emphasis of the new or improved models in CESAR concern reflooding of an intact core, condensation in the pressurizer, sub-critical break flow correlation, and a new pressurizer spray model. The main results of CESAR validation work on different experiments are presented below.

BETHSY 9.1b experiment

The BETHSY facility (in CEA) is a scaled down model of a 900 MWe Framatome PWR. The elevation scaling factor of the facility is 1:1 in order to preserve gravitational heads, while the overall scaling factor applied to volumes, mass flow rates and power level is close to 1:100. The BETHSY primary system has three identical loops; each one is equipped with a main coolant pump, capable of delivering up to the nominal flow rate, and a U-tube steam generator. Primary and secondary engineered safety systems are simulated. This includes high and low pressure injection systems, accumulators, pressurizer spray and relief circuits, auxiliary feedwater system and steam dump to the atmosphere. The facility is designed to operate over the full range of primary (0.1 MPa to 17.2 MPa) and secondary (0.1 to 8 MPa) pressures and corresponding fluid temperatures. The core power is limited to 10% of the nominal value, i.e. 3 MW for the 428 electrically heated rods which simulate the core.

The test 9.1b (OECD ISP N°27), consists in a 2" cold leg break, while the High Pressure Injection System (HPIS) is assumed to be unavailable. This transient leads to a large core uncovery and fuel heat-up, requiring the implementation of the, so called, Ultimate Procedure U1 (UP) which aims at recovering the primary mass inventory by all means. The Ultimate Procedure should be applied as soon as the unavailability of HPIS is known. In the presently studied scenario, the actuation of the procedure is delayed and, therefore, the steam dumps to atmosphere are fully opened only when the core outlet temperature rises significantly higher than the saturation temperature. This action allows the primary circuit to depressurize down to the accumulator injection threshold, then to Low Pressure Injection System (LPIS) actuation. The end of the test is reached as soon as a safe state of the primary coolant circuit is recovered, i.e. when the conditions required for the stable operation of the Residual Heat Removal System (RHRS) are attained.

The initial primary system depressurization is well simulated by CESAR IRSN calculation as illustrated in Fig 1 (the solid lines are the result of CESAR calculation while the dots are experimental data). After primary pump coastdown at about 350 s, the mass flow rate reduces but with a 200s time delay with respect to the experiment, resulting in the same time delay in the drastic reduction of break flow after important cold leg draining (Fig 2). The depressurization of both primary and secondary systems after UP initiation and steam dump opening is well simulated by CESAR. The primary pressure is slightly underestimated by CESAR (about 1 bar) towards the end of the transient after LPIS startup. The time evolution of primary mass inventory during primary circuit draining and refilling is well reproduced by CESAR (Fig 3). The secondary mass, which is controlled by auxiliary feedwater injection, fits very well the measured value. The general trend of core uncovery is also well predicted by CESAR. The core collapsed level is slightly over predicted after accumulator injection. This difference of about 30 cm, which remains practically constant until complete core refilling takes place, could be explained by the larger accumulator discharge or by discrepancy in the prediction of water boil-off from the core [2]. The fuel rod simulator heatup and cooldown after accumulator injection is very well reproduced by CESAR as illustrated in Fig 4.

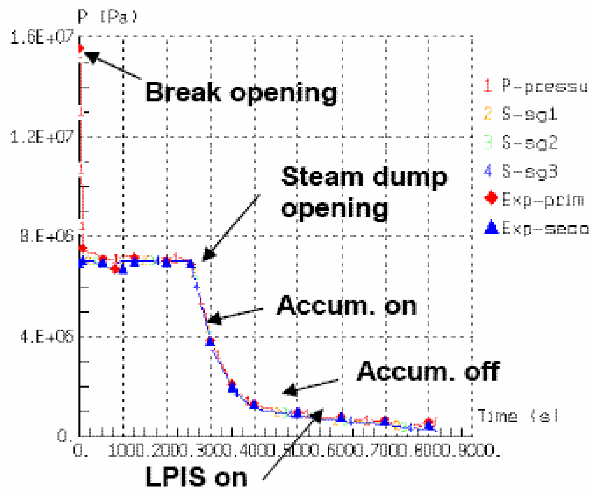

Fig.1: BETHSY primary - secondary pressure

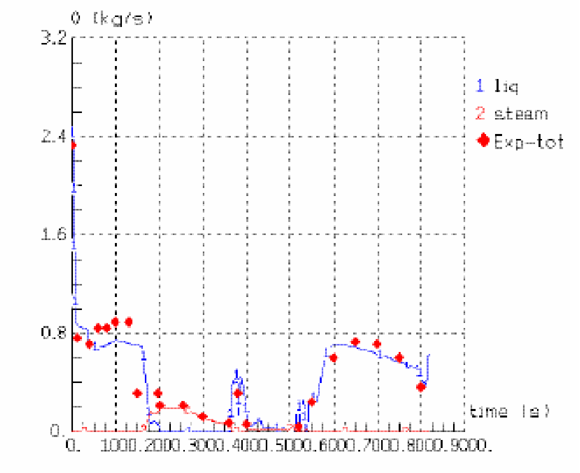

Fig.2: BETHSY break mass flow rate

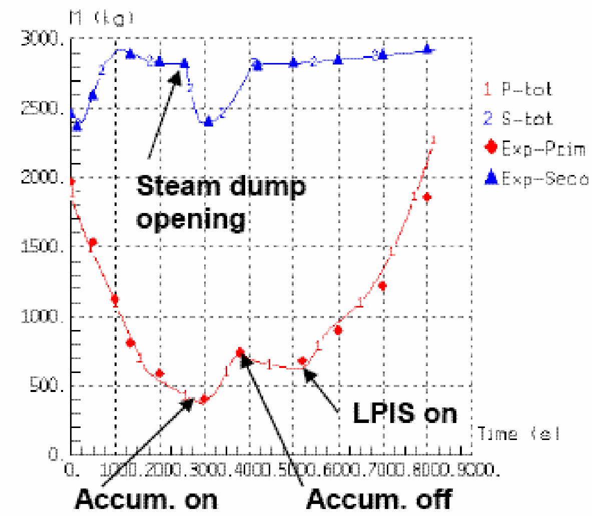

Fig.3: BETHSY primary-secondary coolant mass

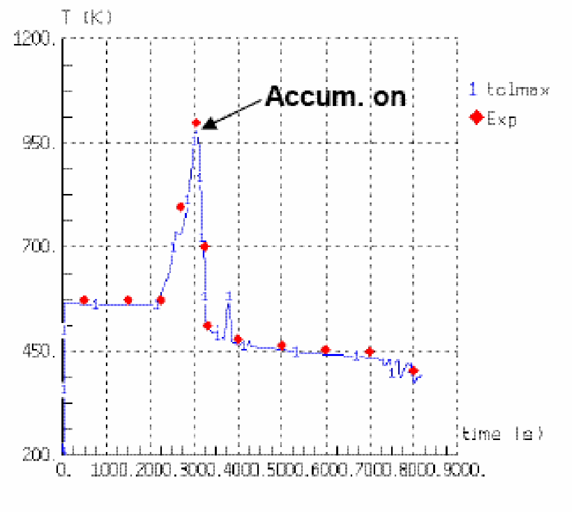

Fig.4: Maximum core wall temperature

PACTEL ISP 33 and T2.1 experiments

Since the beginning of the ASTEC development significant attention was paid to applicability of the code to VVER reactors (e.g. [3], [4]). The PACTEL facility was designed and constructed in Finland to study experimentally thermal hydraulic characteristics of VVER-440 reactors during LOCA's and operational transients. The facility represents a volumetrically scaled model (1:305) of the 6-loop VVER-440/V213 reactor with three separate double capacity loops and full length, electrically heated fuel rod simulators. The elevations are preserved in full height. In the frame of the SARNET project, the 2 experiments ISP-33 and T2.1 were analysed several times by IVS (e.g. [5]) using different ASTEC versions.

The main goal of the ISP-33 experiment was to study different modes of natural circulation in complicated VVER-440 geometry (loop seals are on both hot and cold legs). Primary inventory was stepwise decreased until the core heat-up started while secondary heat sink was preserved. Main phenomena, transition from single to two-phase natural circulation and overall system behaviour were well predicted by the CESAR module.

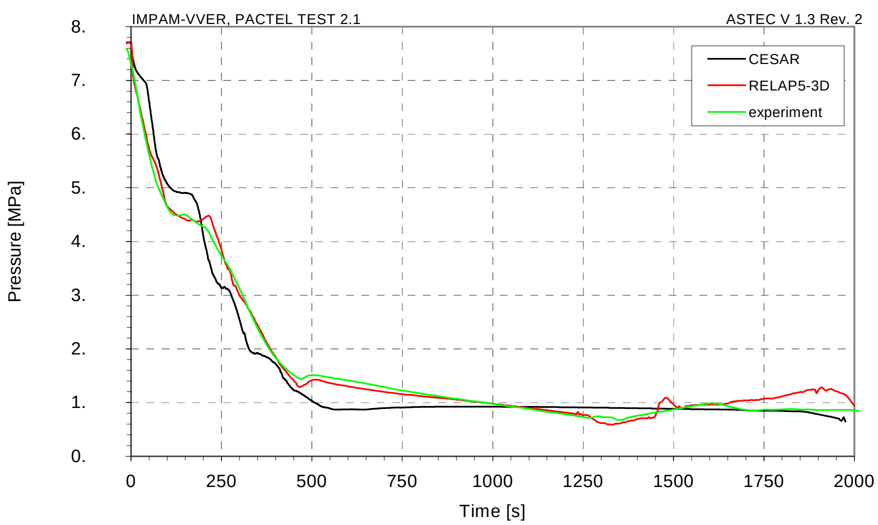

Fig. 5: Analysis of PACTEL T2.1 experiment
- the course of primary pressure

The main objective of the IMPAM-VVER project (5th Framework Programme EU) was to investigate the means and criteria for starting the depressurisation measures in VVER-440 small break LOCA scenarios in order to enable injection from low-pressure pump. Another objective was to assess the applicability of computer codes used in support analyses for emergency operating procedures to predict important phenomena during depressurisation of primary or secondary system. For this purpose a series of experiments was performed on PACTEL facility. One of these experiments denoted as T2.1 represented SB LOCA scenario without availability of high-pressure injection. When the core heat-up takes place, the secondary bleed was initiated in order to decrease primary pressure below discharge head of LP injection. The objective of this test was to investigate whether the primary pressure can be reduced via secondary and/or primary bleed to the value of LP pump head in SB LOCA scenarios before the core heat-up takes place. Again, main phenomena, such as depressurisation of the primary system (Fig.5), accumulator injection and overall system behaviour were well predicted by the CESAR module.

PMK2 SBLOCA experiment

The PMK2 facility – a scaled down model (1:2070) of the PAKS NPP (VVER-440/213) – originally was designed for investigating loss of coolant accidents (LOCA) and to supply experimental data for computer code validations in KFKI AEKI (Hungary). The 3mm (7.4%) SBLOCA transient with HPIS and hydro-accumulator injection was selected by BUTE for ASTEC validation. The experiment and the calculation started from steady-state conditions corresponding to normal operation at 100% power of a VVER-440 unit. In the early phase of the transient – during the rapid depressurization before the start of hydro-accumulator injection – the measured and calculated pressures (Fig. 6) and core inlet temperature (Fig. 7) show good agreement. The main deviations of results and measurements start when the primary pressure falls below the hydro-accumulators pressure. During hydro-accumulator injection, which started at about 40 s and increased noticeably after about 250 s, pressure in the primary circuit is higher in the ASTEC calculations than in the experiment. This higher primary pressure after 300 s resulted in lower hydro-accumulator flow rate. Investigation of the difference of the hydro-accumulator behavior is under way.

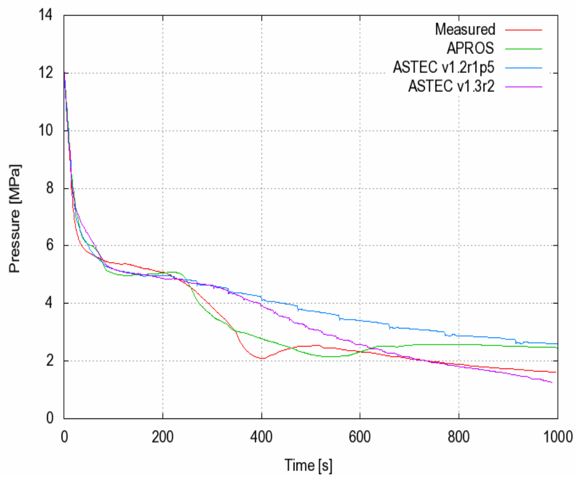

Fig. 6: PMK2 Vessel upper plenum pressure

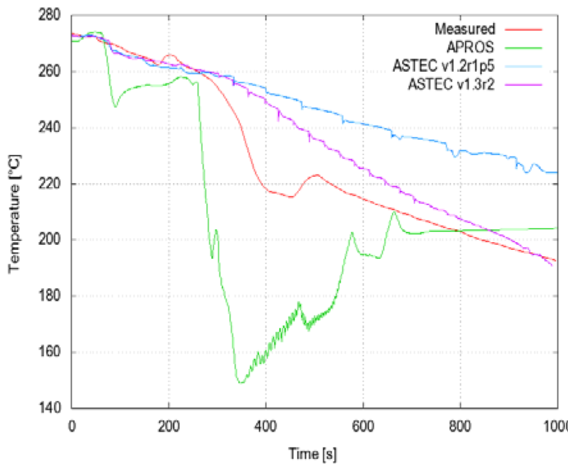

Fig. 7: PMK2 core inlet temperature

B.2 Core Degradation

The suitability and capability of new or improved models implemented in successive code versions up to ASTEC V1.3R2 delivered in December 2007 have been evaluated. Improvements of DIVA in the last code release mainly concern the models of the corium behaviour in the lower head and of lower head mechanical failure. The main results of DIVA validation work is presented below.

Phébus FPT-4 experiment

The calculations with the DIVA module on Phébus FPT-4 involved the DEBRIS and MAGMA preliminary models and aimed at the validation of heat transfer and melting in a highly degraded core, molten pool behaviour as well as fission product release from highly degraded structures. Results of calculation concerning temperature development, amount of molten material and final material distribution (Fig. 8) match well with the experimental data. Simulation of fission product release from the debris bed shows a reasonable agreement with the experiment. Good agreement was obtained for high volatile species (Cs, I, noble gases). For some elements, especially Pd and Sr, the release is over- respectively under-estimated by an order of magnitude. Concerning major actinides (U, Pu and Am), the calculated release fractions exceed the experimental values determined from filter deposition. However, they show the right tendency, considering possible re-deposition inside the debris bed which is not treated in the model.

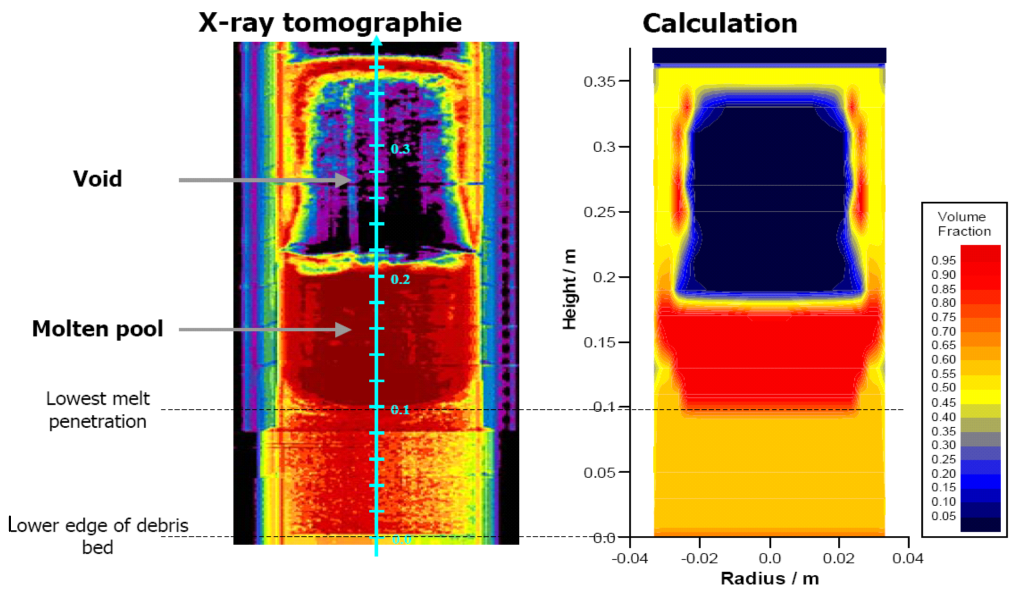

Fig. 8: Phébus FPT4 material distribution after the test

CORA-13 and CORA–W2 experiments

The results of the application of ASTEC to the CORA bundle degradation experiment CORA-13 (bundle typical for western PWRs) and CORA-W2 (bundle typical for VVERs) are discussed in this section. In the calculations for the CORA-13 experiment, the assessment of specific models in the DIVA module for oxidation, bundle degradation and quenching stood in the foreground. ASTEC results were compared with results of the calculations with the ATHLET-CD code. Special emphasis was on the behaviour during the quenching phase. Good agreement of DIVA and ATHLET-CD results with the experimental temperature development up to quenching was obtained (Fig. 9), reproducing experimental characteristics and timing. Sensitivity studies revealed that the maximum temperatures as well as melted and relocated mass calculated by DIVA are significantly influenced by the chosen oxide shell failure criterion. The calculated total hydrogen production up to the quench phase is close to experimental value, although production rates are higher and peak earlier compared to measurements. The bundle degradation results of DIVA correspond qualitatively well to experimental end state concerning amount and axial distribution of relocated materials. The thermal-hydraulic modelling in DIVA has been improved in ASTEC V1.3 compared to previous versions. It now allows a more realistic description of the quench phase, but the hydrogen production peak observed in the experiment during quenching is not reproduced. From reactor applications it was observed that the chosen oxide shell failure criterion can have a significant influence on the degradation and hydrogen production results. Sensitivity studies carried out for the CORA-13 experiment with this respect showed that an earlier failure of the oxide shell yields earlier relocation of metallic melt from hot to cold regions, thus less dissolution of fuel by molten metals, lower maximum temperatures and less hydrogen production. The results varied by 20 to 30% concerning total molten mass and hydrogen produced. At present, it is not possible to give a recommendation from these results for the shell failure criterion, in view of the experimental uncertainties.

Calculations for the CORA-W2 experiment were performed with the previous V1.2 code version. The calculations reproduced main experimental results concerning the thermal behaviour of the bundle, although the temperatures were somewhat overestimated in the upper part and underestimated in the lower part of the bundle. In ASTEC V1.2 also the modelling for oxidation of B_4C control rod material was already available. The total hydrogen production amounted to 80 g, which slightly overestimated the experimental value (Fig. 10).

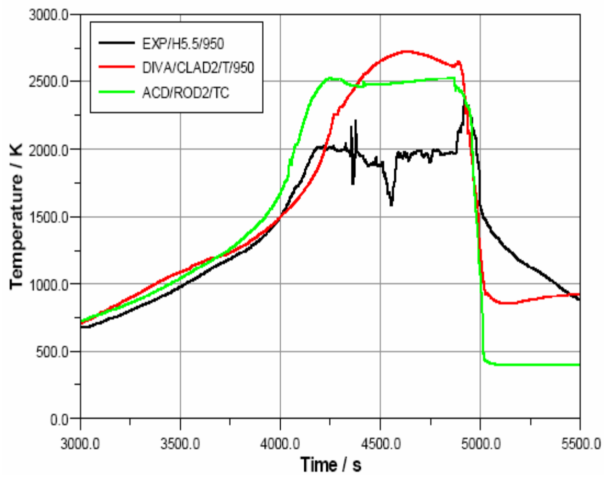

Fig. 9: Clad temperature at 0.95 m (CORA-13)

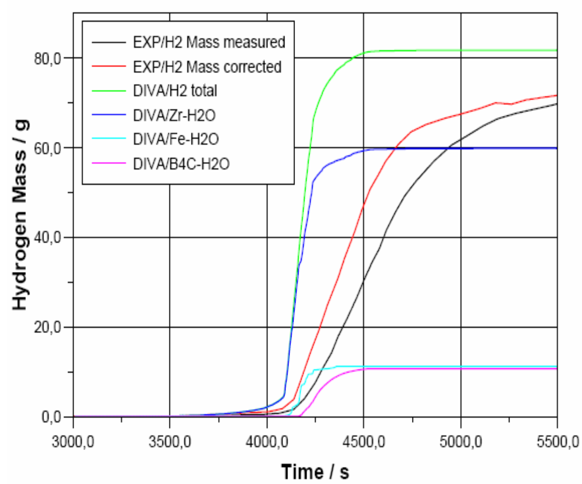

Fig. 10: Hydrogen production (CORA-W2)

QUENCH-11 and QUENCH-13 experiments

Degraded core reflood is characterized by strong interaction of the fluid with core components. With increasing temperatures, chemical reactions between materials and steam as well as between various materials take place, and material relocations change the initial geometry of the cooling channels. To assess the capability of DIVA in ASTEC V1.3, QUENCH experiments [6] with intact (QUENCH-13) and with slightly damaged geometry (QUENCH-11) were investigated by FZK. The QUENCH test section is simulated using 21 axial nodes, a fluid channel, and six components (see Fig. 11, left): at central bundle position either an absorber rod (ABS, QUENCH-13) or an unheated rod (UNH, QUENCH-11), two heater rods (H1, H2), two corner rods (C1, C2), and the one shroud with its fibre insulation.

In QUENCH-13 the reflood was initiated by fast water injection, a pressure driven accumulator injection, which is used to fill instantaneously the lower plenum volumes. Except for a damaged central absorber rod (see Fig. 11, left), the fuel rod bundle is intact. The temperatures prior to reflood agree pretty to well the measured values (see Fig 11, right), except for 950 mm, where only one TC remains. During reflood some deviations are found, which are attributed to the simplified reflood model, compared to pure thermal-hydraulic codes such as CATHARE, ATHLET, or RELAP5, in the DIVA part of ASTEC. On one side the sharp temperature drop due to the injected water is reproduced quite well, but on the other side the long term behaviour shows a faster cool-down. The reasons of this deviation are presently unclear and may arise from water losses into the shroud which are not considered here.

To assess the efficiency of degraded core reflood [7], a very low mass flow rate (18 g/s) was used in QUENCH-11. The test started with bundle dry-out, followed by a heat-up phase to 2300 K (see Fig.12), and was terminated with a very low reflood rate, showing that oxidation leads to adverse effects such as temperature increase and bundle damage. ASTEC had some difficulties in simulating the experiment but taking into account the experimental uncertainties the discrepancies can be judged as acceptable.

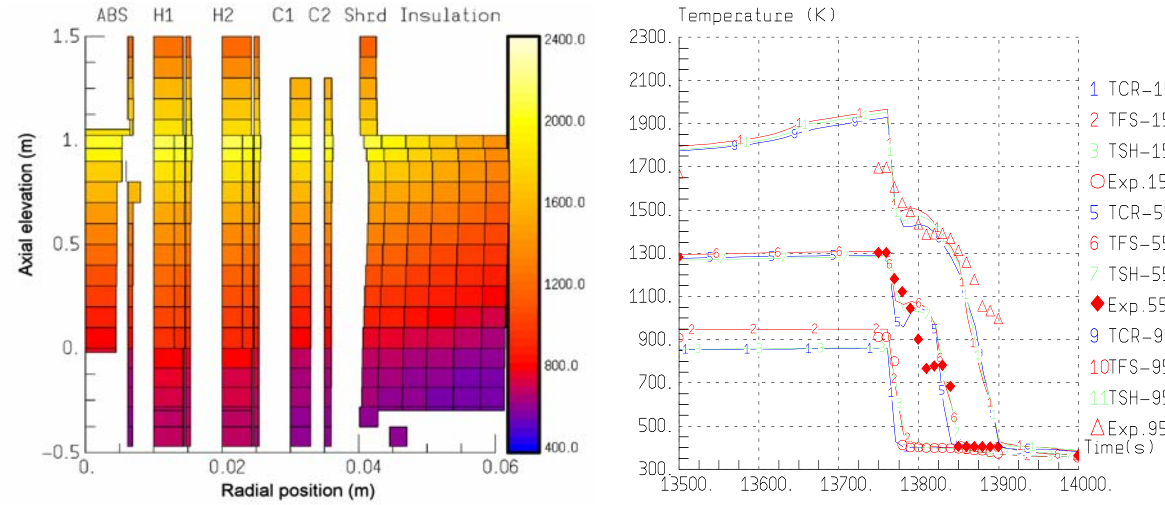

Fig. 11: Geometry and temperatures of QUENCH-13 prior to reflood (left) and calculated and measured (symbols) temperatures during reflood phase at 150, 550, and 950 mm

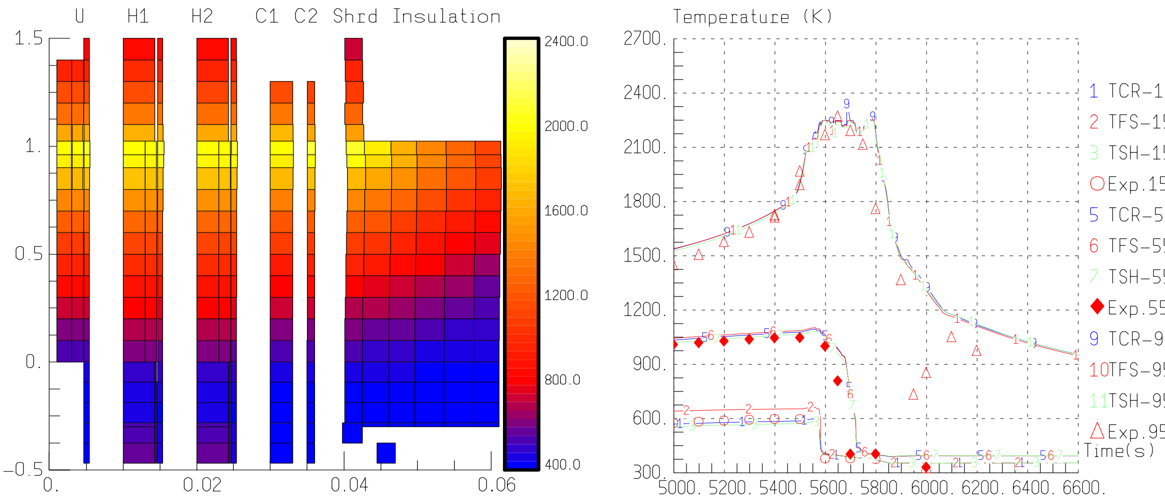

Fig. 12: Geometry and temperatures of QUENCH-11 prior to reflood (left) and calculated and measured (symbols) temperatures during reflood phase at 150, 550, and 950 mm

Even if the present thermal-hydraulic model is rather simple compared to system codes like CATHARE, ATHLET, or RELAP5, the results for bottom flooding are acceptable. The deviations observed are due to uncertain mass flow rates in the experiment and code limitations, but such situations are not typical for nuclear power plants. For intact fuel elements and normal bottom flooding the results are satisfactory.

Applications of ASTECV1.2R1 and ASTECV1.3R2 for simulation of QUENCH-11 experiment were also performed by INRNE. The comparison of ASTEC calculated results with experimental data during the boil-off test show good agreement, as it is the case for the MELCOR INRNE results. In general the results correspond to the measured data, except for the increasing water level during the quench phase due to lack of shroud failure simulation, which is reflected too in the underestimation of total hydrogen production (Fig. 13). The new

ASTEC V1.3R2 version calculates slightly better the hydrogen generation compared to the V1.2.R1 version due to improvement of oxidation modelling.

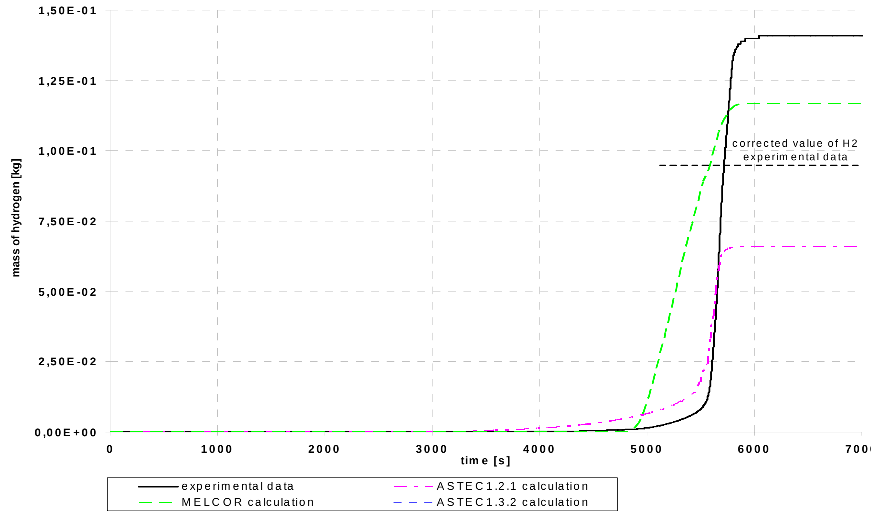

Fig. 13: Total mass of hydrogen produced in QUENCH-11

Results from QUENCH-11 experiment analysis performed by ENEA confirm the good capability of ASTEC to simulate the bundle behaviour during the boil-off phase. Major uncertainties are observed during the quenching phase. Better agreement for material relocation and hydrogen generation during quenching is obtained reducing down to 0.001 m/s the material relocation velocity. The ASTEC results well compare to the results of calculation performed with ICARE/CATHARE V2.1 code.

FARO L14 and L28 experiments

The FARO L14 and L28 experiments, performed by JRC-Ispra (Italy), represent the interaction of a corium melt (80wt% UO_2, 20wt% ZrO_2) poured by gravity into a water pool at saturation temperature at pressure 5.1 MPa. The FARO experiments were analyzed by IRSN with the previous V1.2 version of ASTEC (the same conclusions could be drawn for V1.3). The prime concern of validation against FARO L14 and L28 experiments was the prediction of debris masses and initial pressure rise: they are reasonably well predicted by the code. A large discrepancy was observed with the cake formation in L28 test that can be improved upon with further studies on particle agglomeration and decantation process. Some sensitivity studies were performed on model parameters or on corium material properties where uncertainties exist. They could explain the main discrepancies observed with experimental data.

LIVE-L1 experiment

The LIVE-L1 experiment was performed at FZK/Karlsruhe to investigate the core melt behaviour in the lower plenum of a reactor pressure vessel mainly with external cooling of the vessel [8], [9]. A first analysis of this experiment has been performed by CEA using the ASTEC V1.3 code, specifically the stand-alone DIVA module with imposed boundary conditions to represent the external cooling of the LIVE test vessel. In fact, this was simulated using boundary conditions for the temperature and the heat transfer coefficients at various positions along the vessel wall, derived from the test measurements. The following results were obtained from the DIVA calculation and compared with the experimental data: the temperatures of the inner and outer vessel wall surface, the temperature distribution in the molten pool, and the crust thickness. The ASTEC lower plenum model does not nodalize the homogeneous oxide layer using several axial sub-layers. For this first DIVA analysis, the molten pool was treated as a single layer and the total power was homogeneously distributed in the layer. Due to this modelling, it was not possible to reproduce the temperature distribution in the pool since the code calculates one temperature per layer. This calculated temperature is compared in Fig. 14 with the central pool measurements from bottom to top MT3, MT11 and MT23. Globally the experimental trend is obtained even if the calculated temperature is 10K to 20K lower than the measured temperatures after the beginning of the external water cooling of the vessel. The crust thickness along the vessel wall is not directly provided as a standard DIVA output but the basis for its calculation is given in [10]. Because it depends on the heat transfer coefficients between the molten pool and the vessel wall at various elevations, the calculated value of the crust thicknesses varies with the pool and vessel wall nodalization modelling. However, even with the single layer molten pool approach, the crust thickness profiles compares fairly well with the post-test experimental values (Fig. 15). The crust thickness is sensitive to the crust thermal resistance at steady-state calculated by ASTEC and to the crust thermal conductivity model. An improvement of the crust modelling should be planned in the next ASTEC versions.

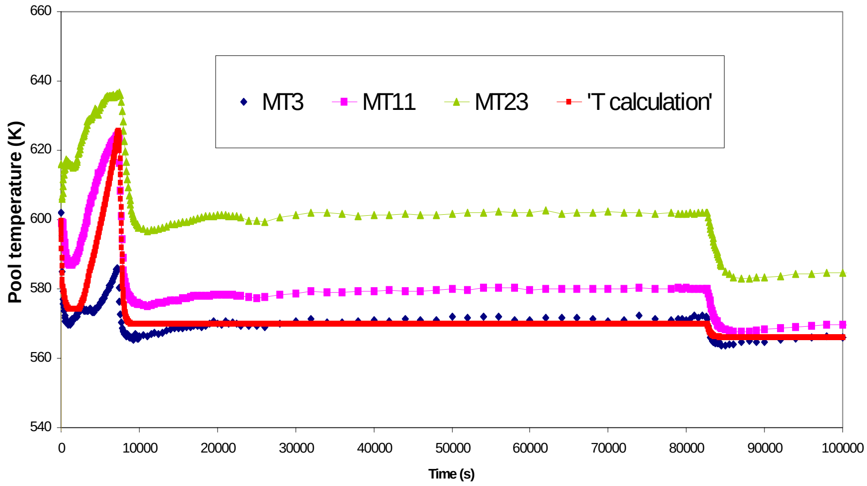

Fig. 14: LIVE-L1 melt temperature comparison

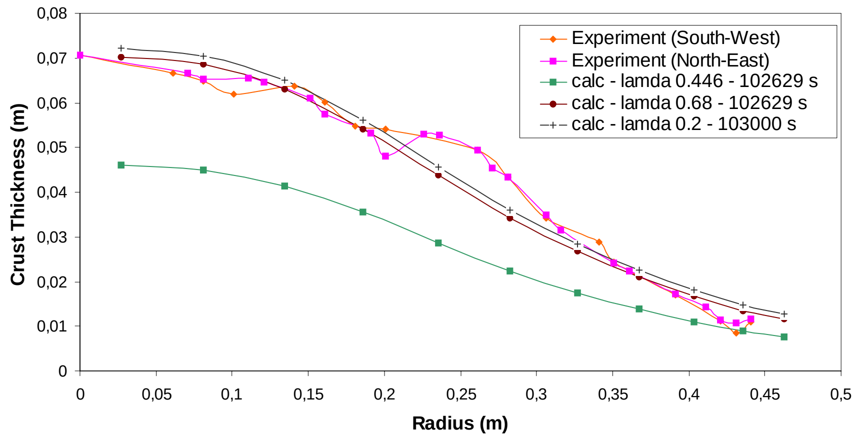

Fig. 15: LIVE-L1 crust thickness comparison

CEA has tested and improved the applicability of the CESAR module to calculate vessel wall external cooling (see the ERMSAR-08 paper [11]). The main next step is thus to perform a coupled DIVA-CESAR application to the LIVE-L1 experiment.

OLHF-1 and FOREVER EC2 experiments

The first OLHF-1 (OECD Lower Head Failure Project) experiment is used for validating the new DIVA vessel rupture 'OEUF' model [12]. This experiment simulates the thermal/mechanical loads of reactor pressure vessel on a 1:5 scale model. During the test, a constant pressure of 12.4 MPa is maintained inside the vessel and the vessel is uniformly heated at the inner surface by induction at a constant rate of 12 K/min. In the DIVA model, temperature boundary conditions vs. time are imposed at the inner surface. At the vessel outer surface, a constant heat transfer coefficient and a constant ambient temperature is assumed. The pressure condition inside the vessel is maintained similar to experiment. The global behaviour of OLHF-1 experiment was well represented. External wall temperatures (see Fig. 16) are overestimated after creep initiation likely due to uncertainties on external heat transfer coefficient. Nevertheless, the ‘OEUF’ model is able to predict reasonably well the rupture location and time (with a discrepancy lower than 3%).

The ‘OEUF’ DIVA model has been also used to simulate the lower head failure FOREVER EC2 experiment performed at RIT/Stockholm. In this experiment, a molten oxide mixture poured at a temperature of 1200 K simulated the corium behaviour in the lower head. Electric heating is provided into the oxidic pool to maintain the temperature of the corium simulant at the desired value. The internal pressure is constant at 2.5 MPa. The results of ASTEC code are compared to experimental data and the results of calculation with the Finite Elements code ANSYS. The ASTEC code is able to describe the global behaviour of the FOREVER EC2 experiment. Despite the simplicity of the ASTEC modelling, the rupture time and location as well as the lower head displacement (Fig. 17) calculated by ASTEC are coherent and close to the experimental measurements and the values calculated by the ANSYS code.

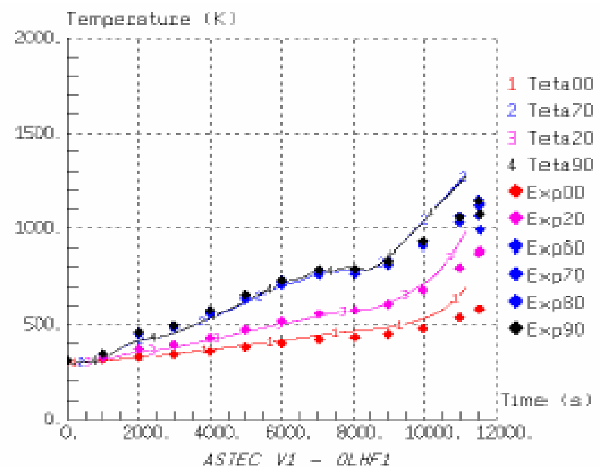

Fig. 16: Vessel wall outer surface temperature (OLHF-1 experiment)

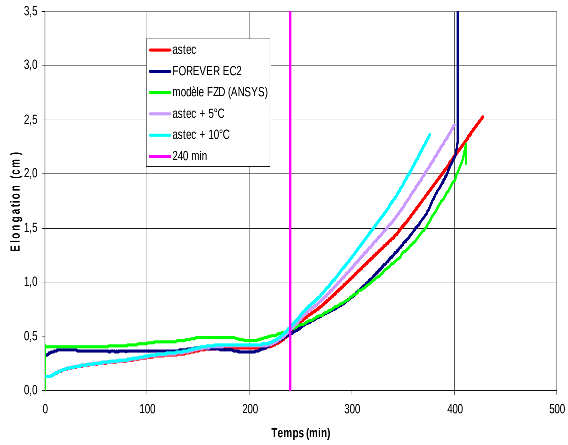

Fig. 17: Lower head displacement (FOREVER EC2 experiment)

B.3 RCS Thermal-Hydraulics and Core Degradation Coupling (CESAR + DIVA)

The main results of CESAR and DIVA coupling validation work on LOFT LP-FP-2 experiment and TMI-2 accident are presented below.

LOFT LP-FP-2 Experiment

The LP-FP-2 experiment was the second fission product release and transport test performed in the Loss-of Fluid Test (LOFT) facility at INEL under the sponsorship of OECD. The objectives of the test were to provide information on fuel rod behaviour, hydrogen generation, and fission product release and transport during a loss-of-coolant accident scenario that resulted in severe core damage. The initial conditions of the experiment represented typical commercial PWR operating conditions. The simulated accident scenario was a pipe break in the low pressure injection system line, which represents a potential pathway for the release of primary coolant from the reactor vessel to the containment. The transient was terminated by core reflood.

Although the hydraulic separation between the Central Fuel Module (CFM) and the surrounding driver core zone could not be simulated with ASTEC, the thermal-hydraulic behaviour of the circuits was reasonably well predicted by the code. Primary system pressure was underestimated after start of DIVA likely due to under prediction of heat transfer from hot vessel lower plenum structures. The onset of core uncovery and heat was very well reproduced by ASTEC (Fig. 18), but the onset of temperature escalation in the upper part of the CFM was delayed. The total mass of hydrogen produced before reflooding was very well predicted by the code (Fig. 19). In spite of the high CFM temperatures reached, the fission product release fractions calculated by ASTEC before reflood are lower than expected but, however, in reasonable agreement with the test estimations.

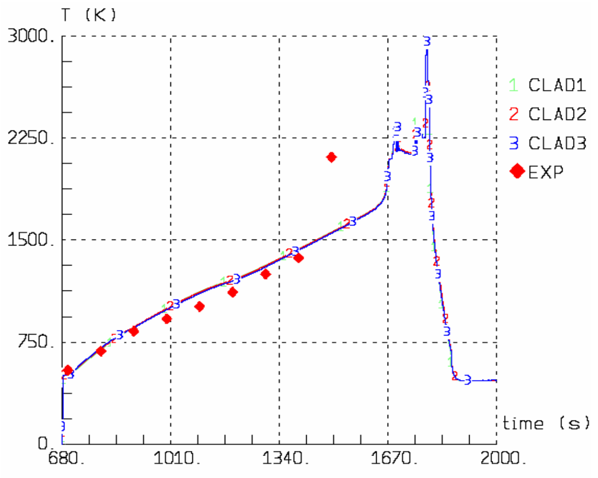

Fig.18: LOFT CFM clad temperature (1.067 m)

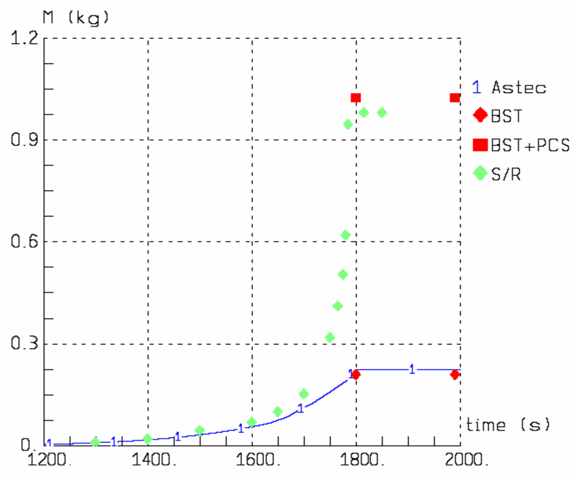

Fig. 19: LOFT total mass of hydrogen produced

High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test during reflood were not reproduced by ASTEC due to lack of adequate modelling.

TMI-2 Accident

The TMI-2 accident provides a unique opportunity to assess the capability of codes to simulate a severe accident on a full scale nuclear power plant. The first two phases of TMI-2 accident have been calculated with ASTEC. The Phase 1 of the transient was characterized by loss of primary coolant through the PORV until shutdown of all primary pumps. The Phase 2 started with core uncovery and involved core heatup and melting until core reflood was initiated by restart of one primary pump.

The overall primary system behaviour was well predicted by ASTEC during both Phase 1 and 2. The primary pressure history was well reproduced (Fig. 20); it was just slightly underestimated towards the end of Phase 2. Pressurizer level behaviour, which played a key role in the accident evolution, was very well simulated. Hot leg gas temperature increase following core uncovery and heatup in Phase 2 was reasonably well predicted by ASTEC. The residual water level in the core at the end of Phase 2 is in good agreement with TMI-2 observations, as inferred from bottom crust location in the central core ring.

The best simulation of TMI-2 core melt progression in Phase 2 was obtained using standard clad failure criteria (T > 2300 K and oxide shell thickness < 300 µm), BEST-FIT correlation (recommended in ICARE2 code) for zircaloy oxidation, and lowering the UO_2-ZrO_2 ceramic melting temperature down to 2550 K, according to the interpretation of some Phébus FP tests. Core degradation and total molten mass calculated by ASTEC at the end Phase 2 (Fig. 21) are consistent with the TMI-2 core configuration hypothesized just before molten core relocation in the lower plenum of the vessel. The total mass of hydrogen produced in Phase 2 (300 kg) was very well predicted by ASTEC.

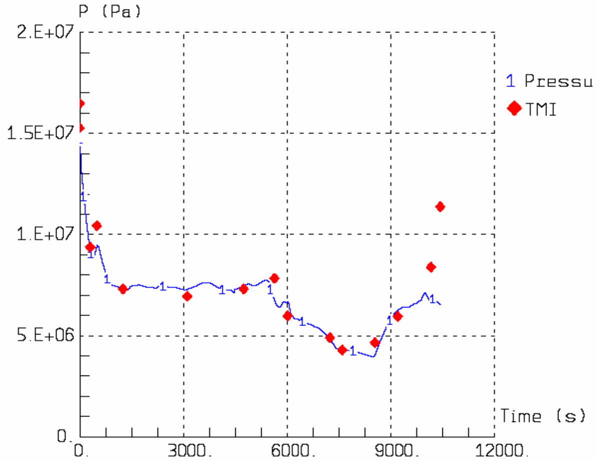

Fig. 20: TMI2 primary system pressure

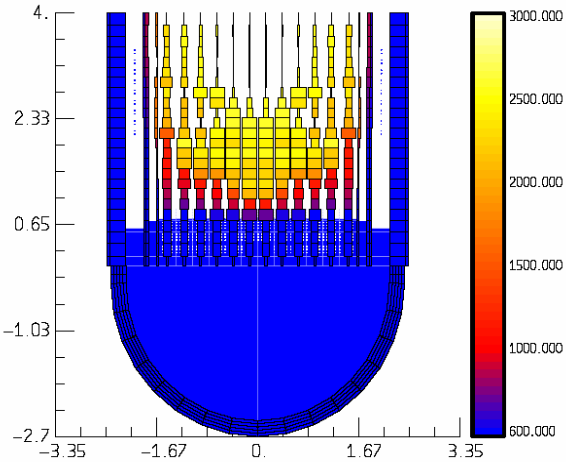

Fig. 21: TMI2 core degradation as calculated by ASTEC (end of Phase 2)

Extension of the TMI-2 accident analysis to Phase 3 and 4 including core reflood and molten material relocation into the lower head of the vessel is foreseen with the ASTEC V2 code version under development provided with ICARE2 debris-magma model for late phase simulation and improved models for degraded core reflood.

C. CONCLUSIONS

The main conclusions of ASTEC validation work on circuit thermal-hydraulic and core degradation are summarized below.

Circuit thermal-hydraulics

The robustness of the thermal-hydraulic module CESAR was highly improved with respect to previous code versions. Good results have been obtained on the integral test LOFT LP-FP-2 and on the two PACTEL experiments in geometry of VVER circuits. This is particularly important since LOFT is an integral test, thus near the reactor conditions, and since the PACTEL facility cover various thermal-hydraulic flow regimes. These good results have been confirmed by the results of the very intensive validation work against BETHSY CEA integral tests, as indicated by the results of 9.1b test analysis illustrated in this paper. Furthermore, the analysis of the first two phases of the TMI-2 accident has confirmed the good capability of CESAR to simulate primary circuit thermal-hydraulics in a real plant application. A discrepancy remains on PMK2 SBLOCA transient after hydro-accumulator injection. Investigation is under way to clarify the origin of this difference.

It is worth mentioning that, outside of the SARNET framework, good results were also obtained in the IRSN large validation work on separate-effect-tests (COSI, COTURNE, REBECA, SUPER MOBY DICK experiments in CEA), on steam generator experiments (PATRICIA GV in CEA), on PWR transient pressurizer data, as well as many benchmarks with CATHARE V2.5 on PWR 1300 plant applications.

Core degradation

Many partners' efforts focused on DIVA core degradation module validation. Good results have been obtained for early-phase models of core heat-up, oxidation and hydrogen total production before any quenching phase for all calculated experiments (CORA, QUENCH and LOFT LP-FP-2) and in the analysis of TMI-2 accident. Improvements on the thermal-hydraulic phase of reflooding of quasi-intact bundles have been observed on QUENCH and CORA experiments with respect to previous code versions. Core degradation and hydrogen production during quenching in QUENCH-11 experiment are improved using a much reduced corium candling velocity. However, the total hydrogen mass produced under reflooding remains highly underestimated in CORA-13 and LOFT LP-FP-2 experiments. For late-phase models, the results can be considered as good for debris bed melting (Phébus FPT-4), corium fragmentation at slump in vessel lower plenum (FARO experiments), corium behaviour in the vessel lower head (LIVE-L1), and vessel lower head mechanics and failure (OLHF1 and FOREVER EC2). Further validation work is foreseen on KTH SIMECO experiments. Furthermore, a reasonable agreement was obtained for fission product release from a degraded geometry such as debris bed and molten pool in Phébus FPT-4. The same conclusion can be drawn from the application to LOFT LP-FP-2 for the transient phase before bundle reflooding.

Finally, the main modeling weakness remains the reflooding of a degraded core, like most other codes. Implementation of improved debris bed and magma models is in progress by merging DIVA with ICARE2 code in the next ASTEC V2 release, which will allow a more realistic simulation of late phase phenomena up to the failure of the lower head. These models will be further validated on experiments such as LOFT LP-FP-2 and on TMI-2.

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