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EUROPEAN VALIDATION OF THE INTEGRAL CODE ASTEC
(EVITA)

Contract FIKS–CT-1999–00010
(Cost-shared action)

EVITA Final Synthesis Report

(ASTEC)

Co-ordinator: GRS (H.-J. Allelein)

Remarks about the picture on the front page:

The picture of the front page shows the sculpture 'Czech Musicians' (sometimes called 'Prague Spring') by Anna Chromy. This sculpture is at Senovazne square where the EVITA Prague meeting took place in the building behind. The building is the seat of the State Office for Nuclear Safety who kindly provided the meeting room with background.

Originally the meeting was planned in Vltava hotel in Rez. Reason for the change of the meeting place was the great flood in August 2002 that destroyed the hotel restaurant and meeting rooms (they have been reconstructed and are available at present).

If you look closely to the musicians you will find that they play their instruments blindfolded. This seemed to be a symbolical hint about the EVITA partners at their task of applying ASTEC and was the reason why this picture was chosen for the front page of the EVITA reports.

In spite of their original blindness the EVITA partners have learned to play as an orchestra with a respectable sound.

GRS	**EURATOM RESEARCH FRAMEWORK PROGRAMME 1998-2002** **NUCLEAR FISSION** Operational safety of existing installations	

EVITA

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Dissemination level: PU

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1 Introduction

The European Validation of the Integral Code ASTEC (EVITA) involves 20 partners from eight European countries plus JRC. It started in February 2000 and ended in July 2003. The main objective is to distribute the severe accident integral code ASTEC (Accident Source Term Evaluation Code), jointly developed by "Institut de Radioprotection et de Sûreté Nucléaire" (IRSN, France) and "Gesellschaft für Anlagen- und Reaktorsicherheit" (GRS, Germany), to European partners in order to apply the validation strategy issued from the VASA project within the 4th European Community Framework Programme (H.-J. Allelein et al., 2002).

Severe accident management (SAM) measures are currently being developed and implemented at Nuclear Power Plants (NPP) worldwide in order to prevent or to mitigate severe accidents. This needs a deep understanding of processes leading to severe accidents and of phenomena related to them. As greater account of severe accident measures is taken in the regulation of plants, there will be the need to show a higher degree of validation of codes and a better understanding of uncertainties and their impact on plant evaluations.

To fulfil the objectives above described, systems of computer codes, so-called "integral" codes, are being developed to simulate the scenario of a hypothetical severe accident in a light water reactor, from the initial event until the possible radiological release of fission products out of the containment. They couple the predominant physical phenomena that occur in the different reactor zones and simulate the actuation of safety systems by procedures and by operators. In order to study a great number of scenarios, a compromise must be found between precision of models and calculation time. This search of compromise is a real challenge for such integral codes. Such codes have been developed in the United States (MAAP4, MELCOR) and are used worldwide.

In France and in Germany, IRSN and GRS decided to co-operate in development and validation of a new integral code ASTEC. The needs for such a code are source term determination studies, probabilistic safety assessment level 2 (PSA-2) studies, accident management studies, as well as detailed analyses of particular experiments to improve the understanding of the phenomenology.

As the great number of users has significantly increased especially the level of MELCOR, IRSN and GRS - learning from this - opened the ASTEC use for extended validation and generic application to a wide spectrum of European organisations, first realised in EVITA. The objective was also to get an evaluation of the code capabilities, especially its user-friendliness.

Following the complementary dualism of risk- and phenomenon-oriented validation strategies, experiments and severe accident plant sequences have been selected for the ASTEC validation and application process:

- Validation on high-quality experiments such as OECD International Standard Problems (ISP),
- Plant applications on different types of NPP (PWR, VVER) with activation of safety systems (spray, venting, etc....).

Each case included a comparison with internationally used codes that should represent the State of the Art in modelling: detailed codes for validation, integral codes for plant applications.

Both recent and actual ASTEC versions V0 and V1 were installed successfully on the partners' platforms and after extensive portability checks used for the validation and application efforts.

The EVITA evaluation of ASTEC code capabilities and the corresponding feedback towards the code development are an important step towards the intention to provide end-users (like utilities, vendors and licensing authorities) with a well-validated European integral code for the simulation of severe accidents in NPPs.

The extended final EVITA report is released in two parts: the activities related to ASTEC V0 in Part 1, the ones related to ASTEC V1 in Part 2. The work performed in WP1 (Upstream reflections of the learning acquired from VASA) is related to the entire activities. The major outcomes of WP1, the EVITA validation and application matrices, are the basis for the ASTEC V0 and V1 activities. Nevertheless WP1 is only reported in Part 1 of the extended EVITA final report in order to avoid duplications. The results from validation effort for modules like SOPHAEROS and CPA, which remained nearly unchanged switching from version V0 to version V1, appear only in one part of this final report.

This summary version of the EVITA final report is mainly focused on the actual version ASTEC V1, which is subject of the beginning SARNET activities.

2 WP1: Validation and plant application matrix

A review of the current approaches in European countries to validate severe accident codes was performed in the VASA project (4^{th} FwP). One of the main outcomes was the strong need for a common European integral severe accident code, which should be validated in greater depth than presently required. This request for code validation arises from its broader use for the regulation of plants and especially for severe accident management (SAM) strategies. These strategies are based not only on the development of plant specific hardware changes but also on the development of SAM procedures. Consequently, this means that apart from the main use of an integral code for PSAs with risk-related investigations and source term determination, the evaluation of such SAM-procedures should be added to the objectives of an integral code. As a consequence, in accordance with this new trend, besides the commonly adopted ”phenomena-oriented“ validation strategy, a risk-oriented one setting priorities for validation steps is also necessary. Applied to the usual stepwise validation procedure (separate-effect tests, coupled-effect tests and integral tests), such a strategy will consist in building ”validation matrices“ as seen by end-users, that is selecting experiments with respect to the question: which mitigation procedure can be validated by this experiment? Adopting such a strategy will also require a certain level of modularity in the code, and a greater involvement of uncertainty studies.

It is concluded that different validation procedures for mechanistic and integral codes are not fruitful. The essential point is to reach an adequate level of detail in modelling and validation, depending on the need to resolve the issues addressed by risk, technical devices and accident management measures for prevention and mitigation.

Following this objective, a guideline for the ASTEC validation process was established based on the selection of the most relevant risk sequences and phenomena that is suitable for specific end-user needs as well as for research requirements. The previous validation exercises performed with ASTEC were considered in the selection of the experiments and plant specific sequences, in order to ensure a complete validation process. ASTEC validation by IRSN and GRS has been supported by a large set of French, German and international experiments, which cover most aspects of severe accident phenomenology. The validation matrix comprises a “minimum“ set which allows the first applications to reactor-cases to be tackled with an acceptable degree of confidence. Most results agreed satisfactorily with experimental measurements. Besides, the application on the Phébus FPT1 experiment, coupling modules for the primary circuit and for the containment, showed a global agreement with the experimental results and the functionality of the code for an integral calculation.

The EVITA matrix of ASTEC validation is presented in Table 1. ASTEC modules have been validated with the help of experiments or alternative codes that should simulate sequence behaviour and evolution. The experiments have been selected bearing in mind that the code should be able to evaluate the sequence taking into account mitigation procedures.

As for application to plant sequences, ASTEC has been validated with the help of the comparison between ASTEC results and alternative code results that simulate sequences behaviour and evolution. The selected sequences for plant specific validation involve the core damage progress to vessel failure, the physical and chemical processes during severe accident phenomenology, the containment loads and failure modes, and the fission product release. The sequences are representative for different types of reactors. During the application of ASTEC as integral code, the SYSINT module has been analysed and partially validated for the management of event and system interactions along this selected sequence.

The EVITA matrix of ASTEC plant applications is shown in Table 2. It assesses the completeness of the process and establishes the relationship between the ASTEC modules and the accident management strategies. The full range analysis of the phenomena which might occur during the severe reactor accident development and analysis of the severe accident mitigation measures should be included in the selected sequence, with the plant specific characteristics.

Table 1: EVITA matrix of code validation

Partner	Phenomena	ASTEC module	Code for comparison	Experiment
ARCS	Molten corium concrete interaction (MCCI)	WECHSL / WEX	Older WECHSL versions	BETA 1.8, MACE M0, SURC4 (ISP24)
B Tech.	T/h in containment	CPA-THY		VANAM M4
CEA	Core t/h in boil-off conditions	DIVA	CATHARE	3 PERICLES boil-off tests
FZK	H_2 production during core reflooding	DIVA	SCDAP/RELAP5	QUENCH-04
GRS	T/h in containment and spray system	CPA-THY	COCOSYS	NUPEC M7-1 (ISP35)
IKE	Early phase of core degradation in PWR and VVER	DIVA	KESS and ATHLET-CD	CORA 13 (ISP 31), CORA W2 (ISP36) w/o quench phase
IRSN	H_2 production during core reflooding	DIVA	SCDAP/RELAP5 from FZK, ATHLET-CD and KESS from IKE	QUENCH 01 and 05
	T/h and aerosol behaviour in containment	CPA-THY/AFP	COCOSYS calculation by GRS	VANAM M2*
IVS	RCS VVER t/h	CESAR	RELAP and published calculations with CATHARE and ATHLET	PACTEL experiment (ISP33)
JRC-IE	Fission product deposition and resuspension in the primary circuit	SOPHAEROS	VICTORIA 2.0	SD 04, 05, 07, 08; SR09-11 (ISP40)
				TUBA TT28 and TT31
	Iodine behaviour in the containment	IODE	IMPAIR	ISP 41 follow-up intermediate phase
	Integral behaviour	ASTEC	Existing calculations	Phébus FPT1 (ISP46)
NRI	Fission product transport in VVER circuits (deposition, resuspension)	SOPHAEROS	MELCOR 1.8.5	PSAERO and HORIZON
	Iodine behaviour in containment	IODE	Old IODE version, codes used in ISP41	ISP41 RTF experiment
RUB	RCS t/h, core reflooding, H_2 release (+ FP release and transport)	CESAR-DIVA	ATHLET-CD and published calculations with other codes	LOFT-LP-FP2 w/o quench phase
VEIKI	Fission product deposition and resuspension in the primary circuit	SOPHAEROS	VICTORIA and ISP 40 results	STORM SD and SR11 (ISP40)
	T/h in VVER containment, behaviour of suppression system	CPA	CONTAIN	EREC No. 5

Table 2: EVITA matrix of plant applications

Partner	French PWR 900	Westing-house PWR 1000	EPR	KONVOI 1300	VVER 1000	VVER 440/213	VVER 440/230
CSN		SBLOCA *(MELCOR)*					
ENEA	TMLB + variants TMLB *(MELCOR 1.8.5)*						
FANP GmbH			SBLOCA with CPA *(COCOSYS)*				
EA		SBO *(MELCOR)*					
EdF	H3 *(MAAP4)*						
FANP Paris	TMLB *(MAAP4)*						
GRS				MB and LBLOCA SBLOCA *(ATHLET, MELCOR 1.8.4)*			
IRSN	Complete sequence "H2"						
IVS						SBO *(MAAP4-VVER, RELAP5-3D)*	
NRI					SBO SBLOCA, SBO *(MELCOR 1.8.5)*		
VEIKI						MBLOCA with CPA *(MELCOR 1.8.5)*	
UJD						SBO SBO *(MELCOR 1.8.3)*	SBLOCA *(MELCOR 1.8.3)*
VUJE						SBO SBO, SBLOCA MBLOCA, LBLOCA *(MAAP4/VVER, RELAP5, S/R5, MELCOR 1.8.3)*	LBLOCA *(MAAP4/ VVER)*

In blue fonts: work with ASTEC V0 - In black fonts: work performed with ASTEC V1 - *(in parentheses: codes for comparison)*

3 WP2: ASTEC V1 code release and user's support

The Work-package N°2 includes the release of two successive versions of ASTEC code (V0.3 first and then V1.0) and support in its use to all EVITA partners. Two partners contribute to this WP2: IRSN, task leader, and GRS.

3.1 ASTEC V0 version features

The release of the version ASTEC V0.3 was performed in October 2000. The main limitation of this version was the lack of any module able to simulate the front-end phase of RCS thermalhydraulics. This implied to begin any plant calculation at time of beginning of core uncovery, and also to need a separate calculation with a thermalhydraulics code to yield initial conditions for RCS and core.

This version included the following modules:

- VULCAIN 7.1 block 2: RCS thermalhydraulics and core degradation, up to vessel lower head failure.
- ELSA 1.1 rev.2: FP release from fuel rods in intact geometry.
- SOPHAEROS 1.1 rev.2: FP vapour and aerosol transport in the RCS.
- RUPUICUV 1.1: DCH (Direct Containment Heating).
- CORIUM: heat transfers between corium ejected by DCH and containment gas.
- WECHSL 3.5: molten-core concrete interaction (MCCI) in the cavity.
- CPA: thermalhydraulics and aerosol and FP behaviour inside the containment.
- IODE 5.0: iodine behaviour in the containment (sump and gas phase).
- ISODOP: decay of FP and actinide isotopes.
- SYSINT: management of engineered safety features (spray, safety injection system...).

The computer interface was based on the SIGAL software (see the § 3.2) and the code release was made on a CD-ROM.

Two "patch versions" were released later on to all partners, including some error corrections issued from the maintenance activities:

– The first "patch" was delivered on a CD-ROM on the 14th November 2000,

– another version for PC was delivered the 26th January 2001. This version was compiled with the new compiler Lahey F95.

Three users' training courses were organized by IRSN and GRS in November 2000.

3.2 Main ASTEC V1 model evolutions with respect to ASTEC V0.3

The release of the version ASTEC V1.0 was performed in June 2002. This new version (Fig. 1) overcame an important ASTEC V0 limitation by adding a new module CESAR for simulation of the front-end phase of RCS thermal hydraulics. Another important evolution came from the replacement of the VULCAIN module by a new module DIVA for core degradation with a more flexible code structure (notably in order to simulate experimental facilities) and more detailed models, the models of which were mostly derived (with some simplifications) from the IRSN mechanistic code ICARE2. Both modules thus replaced the VULCAIN module:

- CESAR 1.0 for RCS two-phase thermal hydraulics, based on a five-equations system with drift. This module computes alone the front-end sequence, and, after beginning of core uncovery (and switch to DIVA module for calculation of core degradation), thermal hydraulics in the RCS loops and vessel upper plenum.
- DIVA 1.0 for core degradation up to vessel lower head failure. After beginning of core uncovery, this module computes (in place of CESAR module) thermal hydraulics in the vessel below the upper core plate. The late-phase degradation simulates corium pool formation and evolution, corium slump into the lower plenum, corium behaviour in the lower plenum and vessel melt-through.

As concerning the other modules, the main model evolutions with respect to ASTEC V0.3 were the following ones:

- ELSA 2.0 able to compute FP release also in degraded core geometry. When the core geometry becomes degraded, volatile FPs are released by U-Zr-O mixtures during their candling along the rods. Release from debris beds is modelled in a similar semi-empirical approach than from intact fuel rods. Models of release of SIC (Silver-Indium-Cadmium) materials from control rods were also improved.
- WEX 3.1 with some slight improvements with respect to WECHSL. New corium-concrete heat transfers were fitted on some BETA and ACE experiments and allowed to give closer results on plant applications to CORCON ones than WECHSL.
- RUPUICUV included as a new user option a model of corium entrainment and kinetics adapted to the “open” cavity geometry of Western PWRs, i.e. with annular space between cavity and containment.
- CPA with a simplified model for the behaviour of Siemens-type recombiners and with the updated combustion model DECOR 2.0.
- COVI for modelling the instantaneous virtual combustion of H_2 and CO in the containment (adiabatic total combustion, no feedback on thermal hydraulics) and determination of the risk of deflagration or detonation (computation of inflammability limits of the Shapiro diagram). At each given time, it provides the temperature and pressure peaks which would be obtained by instantaneous burning of the remaining H_2 and CO at this time.
- IODE 5.1 with new models of organic iodine formation in liquid (Taylor-Liger) and gas (Funke) phase, and improvement of model of radiolysis in liquid phase.
- The MDB (**M**aterial **D**ata **B**ank) new library gathers properties of all materials involved in a Light Water Reactor, including isotopes and various other properties like iodine chemical reactions. In this ASTEC V1.0 version, MDB was only used in ISODOP, SOPHAEROS and IODE modules. The other modules SOPHAEROS, ISODOP, SYSINT did not include major evolutions.

ASTEC V1.0 has yet some limitations in the frame of reactor applications:

- The robustness of the CESAR module had to be improved,
- the DIVA models of corium behaviour in the vessel lower plenum had to be further tested and developed (absence of any mechanical failure model of vessel lower head),

– the simulation of reflooding of an intact core was possible but without any specific reflooding model in DIVA,

– the full reflooding of primary circuit was not yet tested.

The computer interface, based on the SIGAL software, included: automatic data checkers for pre-processing, on-line visualisation tools (including a specific development for visualisation of RCS), and post-processing of the graphical files with the TIC tool.

The delivery was done through a CD-ROM that includes:

- The code source of the reference version,
- The SIGAL tools,
- The executable programs for PC Windows, SUN, DEC, PC-Linux,
- The delivery test-cases:
    - Validation input decks in stand-alone mode (with their result files): around 30 cases on separate-effect experiments, and 1 application to a Phébus.FP experiment,
    - Reactor calculations using CESAR and DIVA modules in a stand-alone mode, except 2 cases with CESAR-DIVA coupling (LOFT-LP-FP2 experiment and LFW sequence on PWR 900). No calculation of a full reactor sequence with coupling of all modules was ready at time of delivery of ASTEC V1.0.
- The documentation: theoretical manuals of most of the modules (plus the ICARE2 documentation to complete the DIVA description), on-line user manuals, list of most usual questions by users, some advice for the code use, collection of the slides of the users' training course.

Before the end of EVITA project in August 2003, only one update version was released in January 2003: ASTEC V1.0 rev1. This release only included robustness improvements of CESAR-DIVA modules. It was necessary to allow partners to perform the first CESAR-DIVA coupled plant applications in WP5 for the various types of reactors (PWR and VVER).

The users' training course was organized by IRSN in June 2002 in Cadarache Center. All EVITA partners sent representatives to attend this training (24 trainees). This course focused on use of the 2 new modules CESAR and DIVA. A short overview of the new version ASTEC V1.0 was presented. Then after a brief list of phenomena treated by the two new modules, some skill exercises of progressively increasing complexity were performed directly by trainees on the PC computers. At the end of the course, the new tools designed by IRSN to help the users were presented with some examples of applications.

After that release, "hot line" assistance was available at IRSN to EVITA users (e-mail: maintenance.astec@irsn.fr). The Web site was updated by IRSN and GRS.

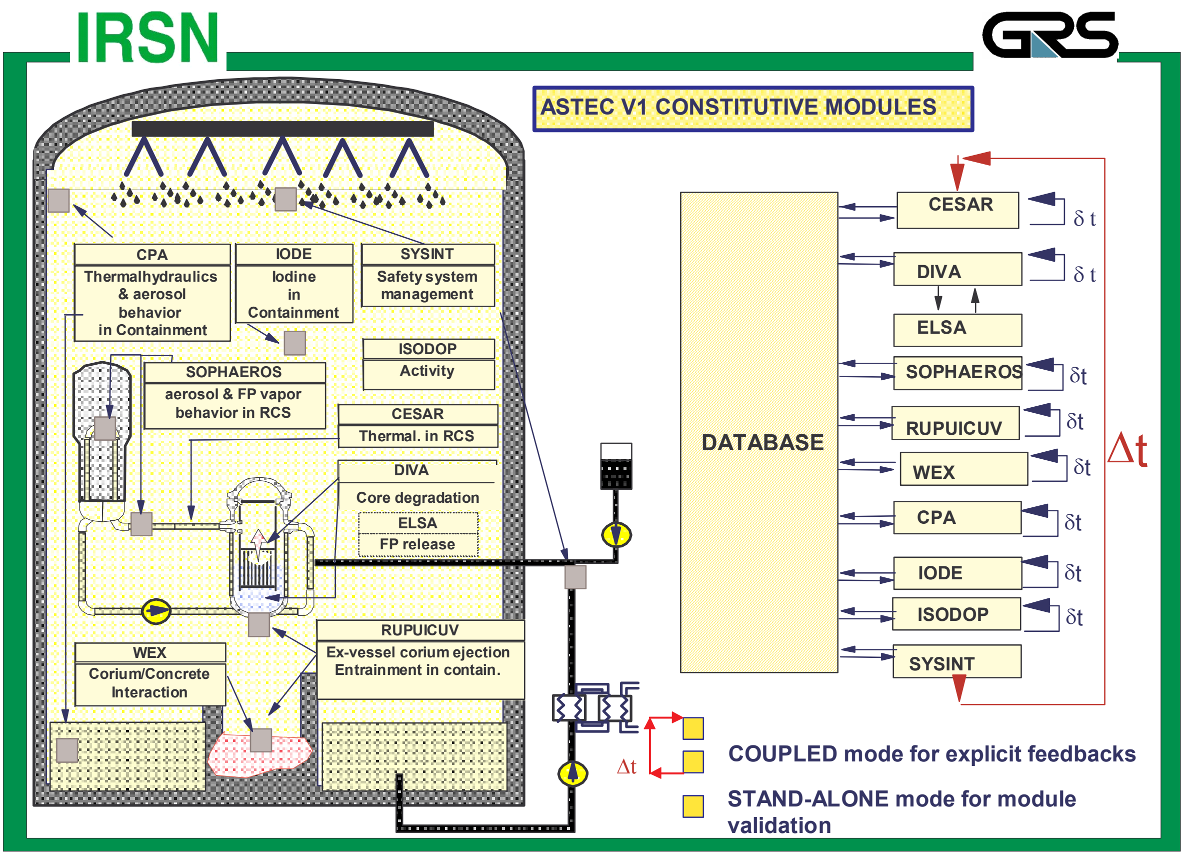


Fig.1: List and role of ASTEC V1.0 modules

4 WP3: Testing of ASTEC code on different platforms phase B – ASTEC V1.0_p1

In the EVITA first stage, the installation of the ASTEC V0.3 version and its portability were checked to be correct on the different partners' computer platforms (SUN and DEC workstations, PC with diverse operating systems and HP). The portability was checked on one validation case per module and on one reactor case combining several modules. The highest obtained discrepancies ranged from 1% to 10 % but limited to only one module and for a limited time period: this concerned the modules VULCAIN, WECHSL, RUPUICUV and CPA but was judged acceptable. It could be concluded that the ASTEC code is portable to the partners' different computer platforms and the user should not fear portability effects. One exception was the Hewlett Packard (HP) workstation where the code installation was not completely successful.

In the EVITA second stage, the ASTEC V1.0 rev1 version was used to check the portability on different computer platforms. Their variety included SUN, DEC, IBM workstations and various types of PC under some Windows type operating systems (WinNT/98/2000/XP) and Linux.

One validation case for each code module and one reactor case combining CESAR and DIVA have been selected for testing. Both ASCII text results and selected graphical output have been compared. This comparison resulted in following findings:

- The portability of these modules was judged as excellent: no or negligible differences have been found in modules CORIUM, DIVA including ELSA, IODE, ISODOP, RUPUICUV, SOPHAEROS, WEX.
- Sometimes very small and always physically tolerable portability effects have been found in CESAR and CPA.
- The most significant differences among the results on different platforms could be observed for the coupling of CESAR-DIVA. Their more detailed investigation, especially their time evolution, showed that these differences are mostly physically tolerable and smaller than could be expected for this type of code.
- Except the IBM workstation, the tests did not indicate any specific portability problems. However, attention should be paid to CESAR-DIVA portability in the future when situations with core degradation up to vessel bottom head failure should be used as test cases.
- For IBM workstation, the CESAR and CPA modules caused problems and failed and WEX results are too different from other platforms.

5 WP4: ASTEC validation

In the first part of EVITA, some validation tasks have been performed with the 1st version ASTEC V0.3: they focused on WECHSL, SOPHAEROS, CPA and IODE module. The WECHSL results are not presented here since this module has been replaced in ASTEC V1. For the other validation efforts, the differences of the concerned modules between the versions ASTEC V0.3 and V1.0 are not significant and the validation conclusions can be kept for ASTEC V1. These conclusions are indicated if necessary in the following chapters.

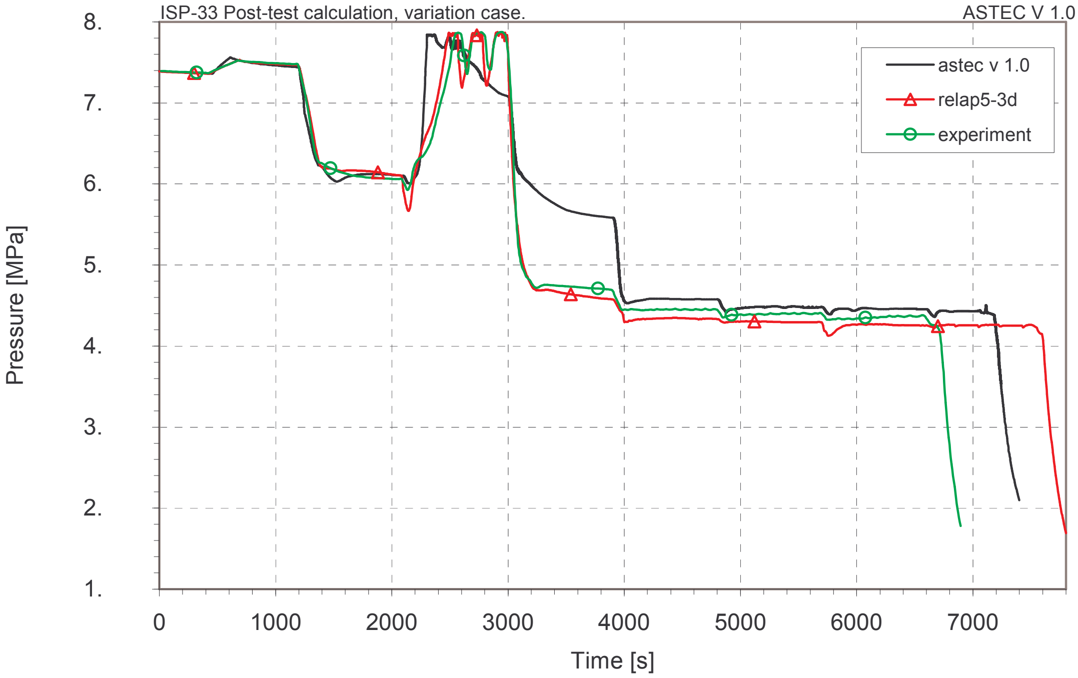
5.1 CESAR module on RCS thermalhydraulics

Analysis of ISP-33 stepwise reduced primary inventory experiment in PACTEL facility using CESAR module

The application to ISP-33 PACTEL test on the thermal hydraulics of VVER-440 units showed that the overall behaviour of the system (course of the primary pressure and temperature) was well predicted. The crucial phenomenon of the experiment - start of core heat-up in dependence of the primary inventory - was well matched, too. Some limitations were found in the underestimation of the mass-flow-rate in primary system under degraded primary inventory (two-phase flow natural circulation). Furthermore, the numerical stability and robustness of the code (especially stability of the axial element) should be increased.

Comparing to the results of other codes used in ISP-33, CESAR results seemed to be on an "average" level. One of the most important findings of the ISP-33 experiment was that in the case of boiler-condenser mode, the ability of the horizontal steam generator to retain significant amount of condensate has a tendency to shorten the time available before the core heat-up starts. Nearly all of the ISP calculations did not reflect this effect correctly and the start of core heat-up was predicted one draining later than in the experiment: CESAR predicted properly (although with certain time delay) the start of core heat-up after 7th draining (see Fig. 2). This was due to proper prediction of water level in reactor vessel in the final stage of the CESAR calculation.

Fig.2 : Pressure in pressurizer.



LOFT LP-FP-2 calculation with ASTEC V1 and comparison with ATHLET-CD)

The CESAR and DIVA modules were applied in a coupled mode to the LOFT LP-FP-2 experiment that is one of the largest severe fuel damage experiments and serves as an important benchmark between smaller scale tests and the TMI-2 accident.

The primary pressure was underestimated after the opening of the break; otherwise the pressure evolved qualitatively similarly to the test (Fig.3). Until the experimental heat-up starts, the calculated results agreed well with the measurements. However, the simulated clad temperatures rose too late in the lower part of the bundle and too early in the upper part in comparison to the experiment. The simulation of the temperature escalation due to the Zircaloy oxidation reaction occurring in the upper bundle part was deficient. Particularly for the mid-bundle part ASTEC tended to over-predict the heat-up rate.

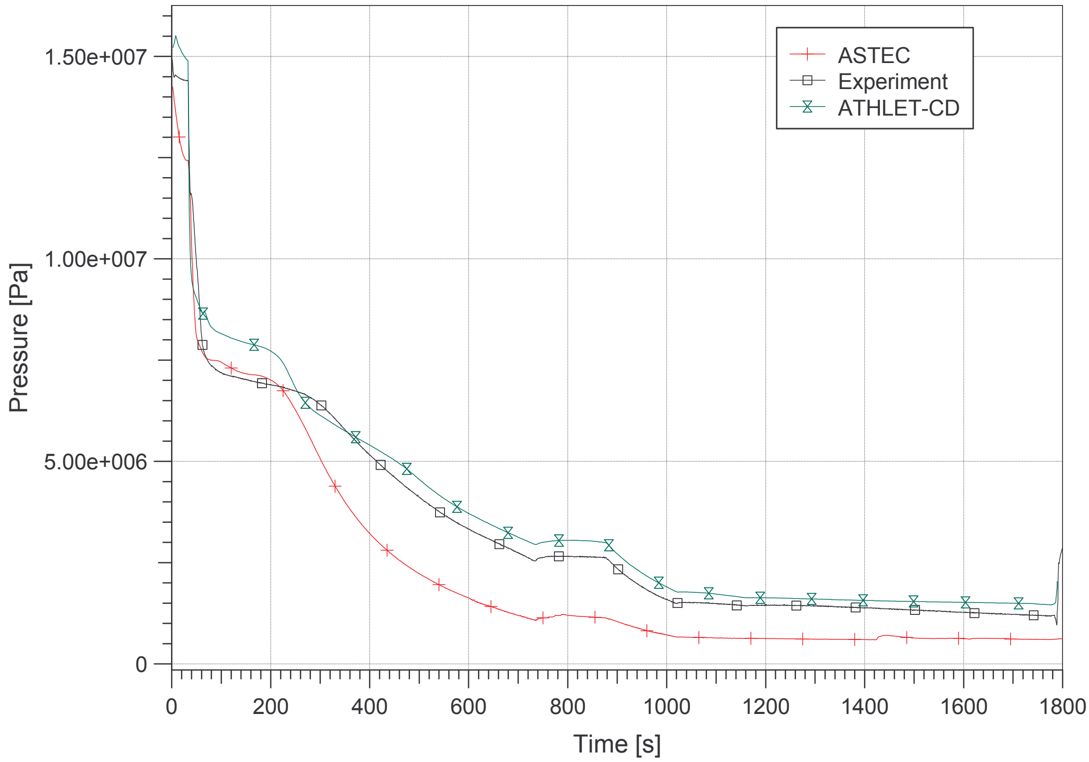


Fig.3 Primary pressure

The hydrogen production was calculated to 272 g, in the order of magnitude close to the supposed 205 g ± 11 g (exclusive of the amount of hydrogen stored in the system). As concerning the fission product release (calculated including the ELSA module), ASTEC gave acceptable results with respect of measurements for volatile fission products (see Fig. 4).

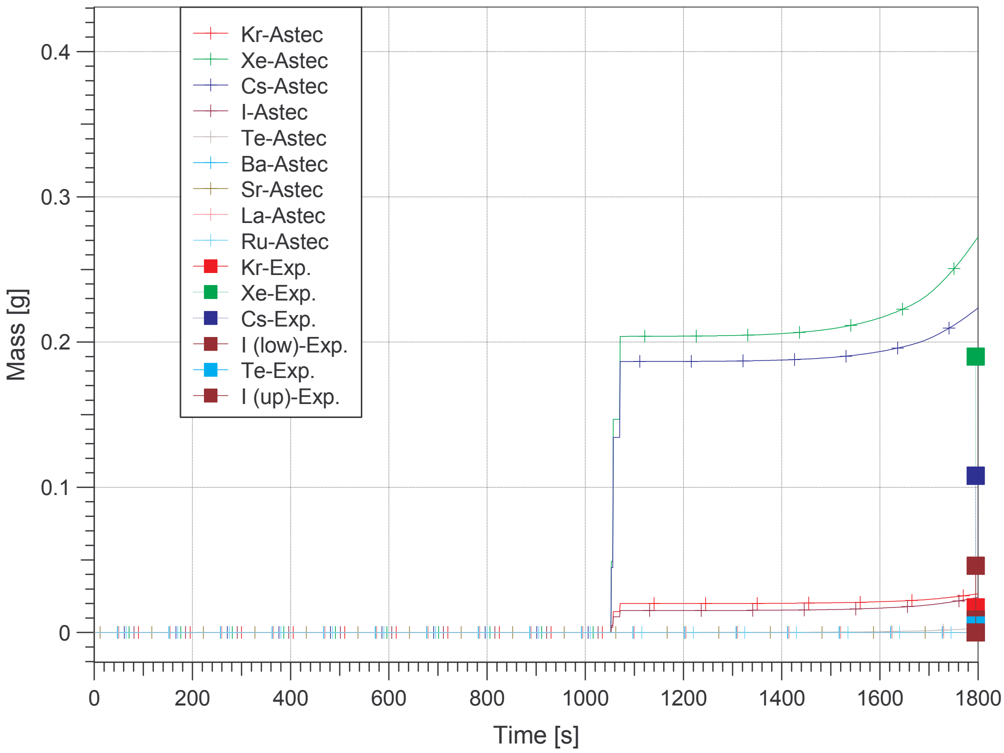


Fig.4: Fission product release in LOFT LP-FP-2

5.2 DIVA module on core degradation

Validation of DIVA module against PERICLES tests

The validation of DIVA in the dewatering phase in an intact core was checked on PERICLES 1D boil-off experiments at low or medium pressures. It showed that a reliable correlation for nucleate boiling was lacking (heat transfer in the wet region was underestimated). Deeper analysis showed that a value of 2.000 W/m^2 for the nucleate boiling regime could predict with sufficient accuracy the experimental collapsed level versus time (see Fig.5). Then, using this value, satisfactory results were obtained in terms of vaporization rate in the wet region and prediction of the axial void fraction (see Fig.6). Nevertheless, improvement of heat transfer in the dry region by modelling a droplet field seems desirable. The over-prediction of cladding and steam temperatures that was obtained here at temperatures of 600 - 700 °C raises the question to what extend such deviations could probably lead at severe accident temperatures.

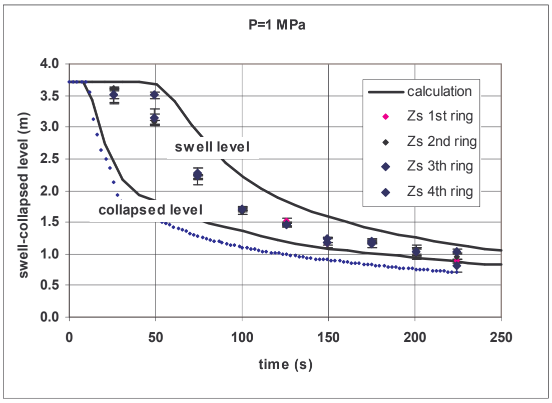


Fig.5: PERICLES experimental and computed time history of the collapsed and swell level

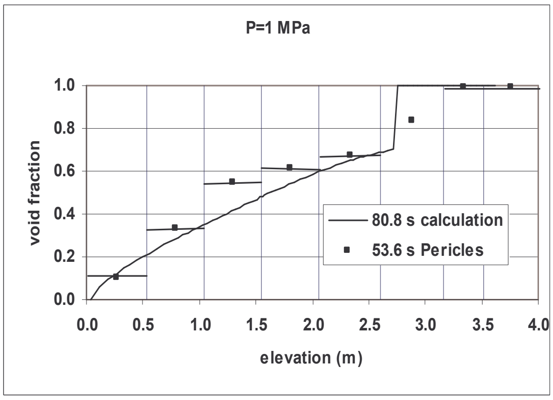


Fig.6: Comparison between experimental and computed axial void fraction profiles at the beginning of PERICLES experiment

Validation of the DIVA module on the experiments CORA-13 and CORA-W2 and comparison with ATHLET-CD

DIVA was also applied to the CORA-13 (ISP31) and CORA-W2 (ISP36, for VVER type fuel element) experiments on core meltdown early phase. Satisfactory results were obtained for the major phenomena observed in these experiments. The remaining differences between code predictions and experimental results were in the same range as those in the comparison calculations with ATHLET-CD. Further, both codes compared favorably with the results given for other codes used in these ISP.

- CORA-13: the onset of the temperature escalation in bundle upper part and of downward propagation of the escalation front were correctly predicted. The calculated integral hydrogen production was close to the experimental value up to the onset of quenching (Fig.7). Both codes however predicted higher oxidation rates and shorter duration of significant hydrogen production. The behaviour during the quenching phase, especially the oxidation peak in the experiment during flooding, was not well reproduced by the codes. Both codes predicted too fast quenching. The overall results showed only a small sensitivity to the choice of the oxidation model (differences in hydrogen production remain within a range of 15 %).

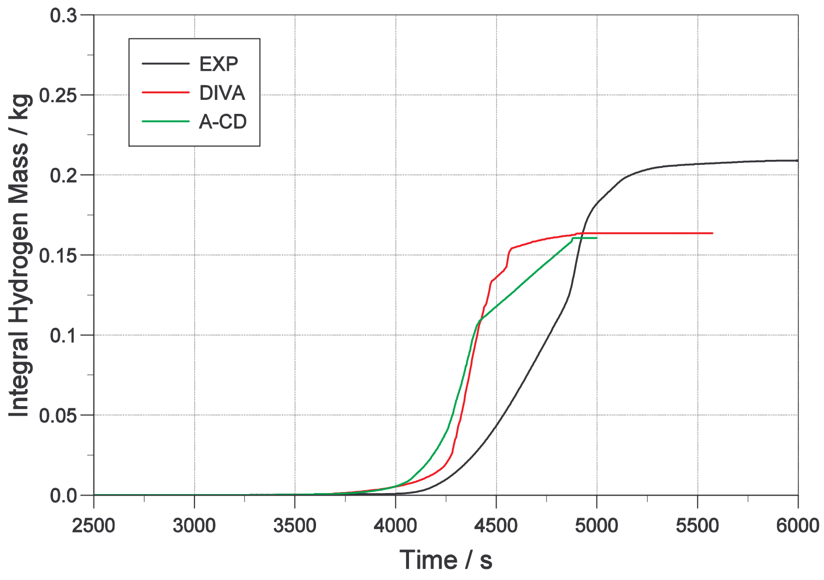


Fig.7: Comparison of hydrogen production as calculated by DIVA and ATHLET-CD (A-CD) with measurements from experiment CORA-13 (EXP)

- CORA-W2: heat-up and temperature escalation at upper bundle elevations were correctly predicted (see Fig. 8). Model parameter settings favoring earlier melt relocation and later melt refreezing were required to obtain the experimentally observed temperature escalation in bundle lower part. The DIVA results then were close to ATHLET-CD and in fair agreement with the experimental measurements. ATHLET-CD predicted stronger bundle degradation and more relocated melt. The final bundle degradation state as calculated by DIVA was in good agreement with results of post-test analyses concerning absorber and cladding material relocation. Consideration of B_4C oxidation by activation of the new preliminary model in DIVA gave consistent results. The impact of the B_4C oxidation on the integral results was only small in the DIVA calculations. The total hydrogen production was very close to the experimental results .

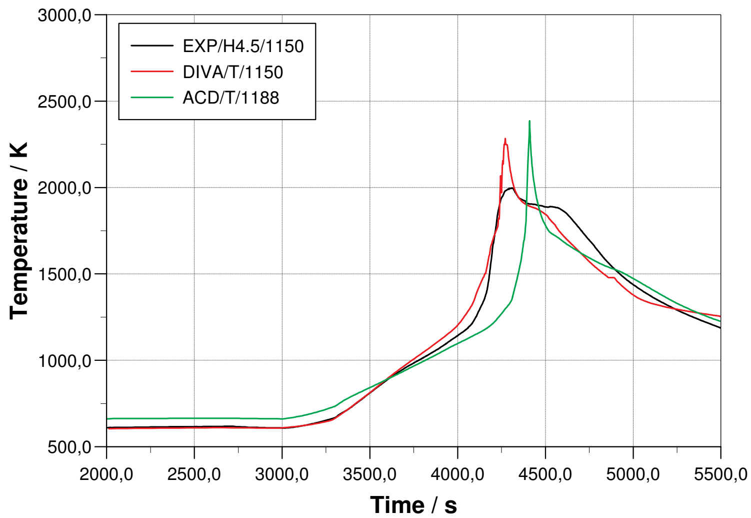


Fig.8: Comparison of temperatures calculated by DIVA and ATHLET-CD (ACD) with experimental measurements CORA-W2 (EXP) at bundle elevation 1150mm

One of the most important accident management measures to control severe accidents in PWR is the core quenching by injection of cold water. DIVA was thus applied to different QUENCH experiments.

Simulation of the QUENCH-05 experiment with DIVA module

QUENCH-05 studied cool-down behaviour of pre-oxidized cladding at high temperature (2.000 K) by injecting cold steam from the bottom. For heat-up and pre-oxidation phases, the calculated temperatures were in very good agreement with the experimental data. For oxidation phase, the temperature escalation in bundle upper levels was not very well captured, except when slightly increasing the steam/argon injection temperature (from 580 to 604 K). For quenching phase, the calculated temperature evolution during the quenching was in good agreement with the data. Hydrogen production was underestimated with 19-21 g, while the experimental value is 27 g. Using the PRATER correlation and again, as above, slightly increasing the steam/argon injection temperature allowed to reproduce the data (26 g of hydrogen with DIVA) (see Fig. 9 and 10). The comparison with ATHLET-CD results showed that DIVA tended to slightly underestimate both bundle heat-up and production of hydrogen mass while ATHLET-CD tended to overestimate them. But globally the discrepancies of both codes remained rather small with respect to the experimental data.

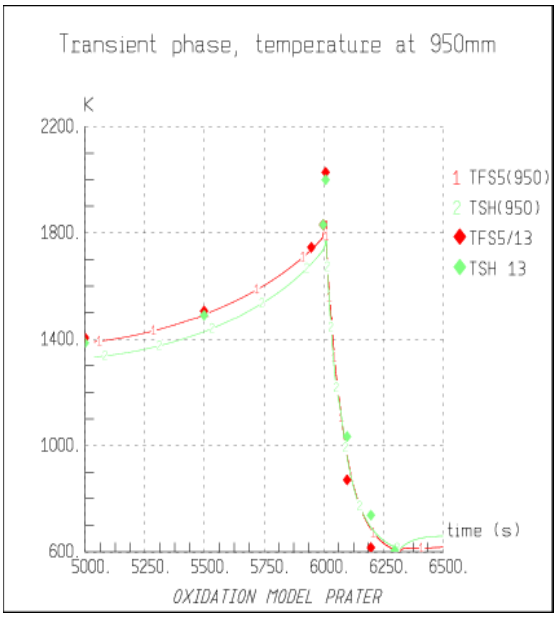


Fig.9: Temperature of heated rod simulator 5 and shroud during the transient phase at z=950mm using the PRATER oxidation correlation: comparison DIVA / experiment

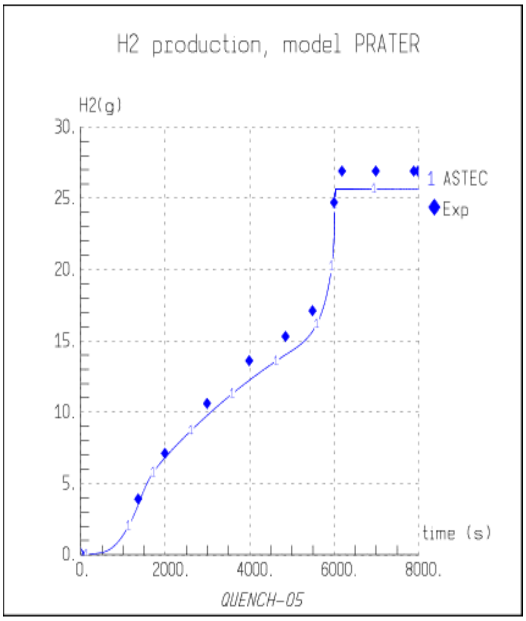


Fig.10: Hydrogen production using a 604K temperature of the steam/argon injection and the PRATER correlation: comparison DIVA / experiment

DIVA validation on QUENCH experiments

The preliminary application to QUENCH-06 (ISP45) experiment with reflooding by water injection showed an under-prediction of pre-quench temperatures (possibly caused by the radial heat losses through the shroud, the convective heat loss or the resistivity of the wire materials) and thus a large under-estimation of hydrogen mass (11 to 18 g to compare with 38 g: see Fig.11).

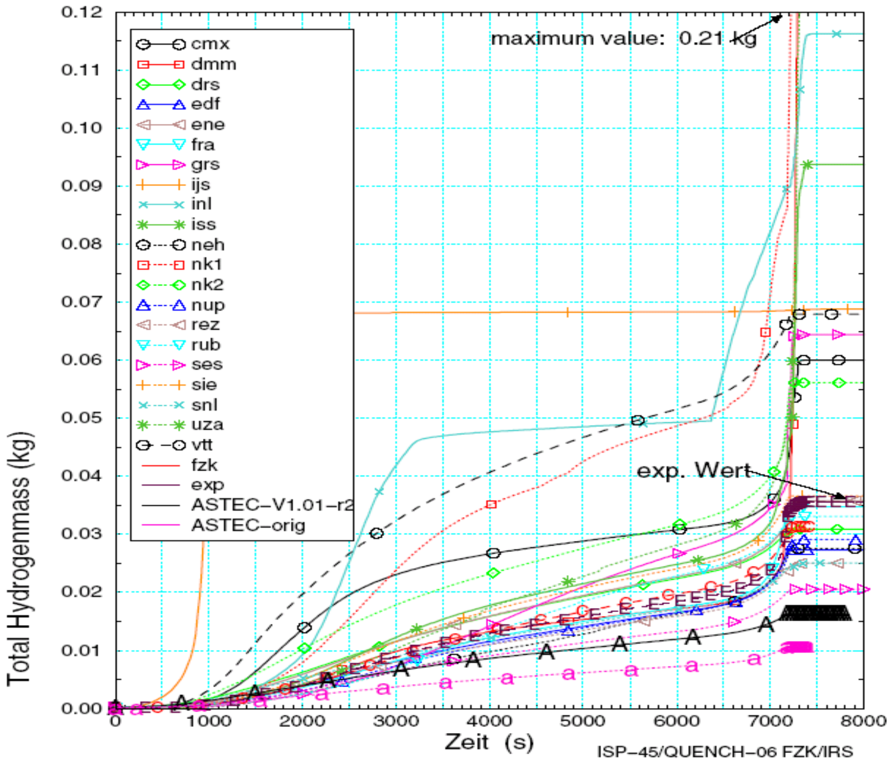


Figure11 Comparison of first ASTEC V1.01q calculations (a): original version, (A) adapted power input.

Status of reflooding models

It must be recalled that no specific modelling of reflooding of intact rods by water is included in ASTEC V1.0. It means that it is possible to compute the water arrival into vessel but the heat exchange model between hot walls and water is not accurate: a nucleate boiling heat exchange coefficient is computed without accounting for any critical heat flux which could limit this heat transfer. Thus a simple model was developed by IRSN to be included in the next version ASTEC V1.1. Preliminary testing of this new model has been performed on some PERICLES reflooding experiments at low pressure (3 bar). It showed that the quench front evolution was well reproduced using a constant coefficient of 2.000 W/m^2K for the nucleate boiling regime.

5.3 SOPHAEROS module on FP and aerosol transport in RCS

Analyses of the STORM test series using the SOPHAEROS module

The applications to STORM SD experiments confirmed the conclusions made from internal IRSN validation, i.e. satisfactory results for deposition compared with the international standards with a typical deviation of 20 % (see Fig. 12).

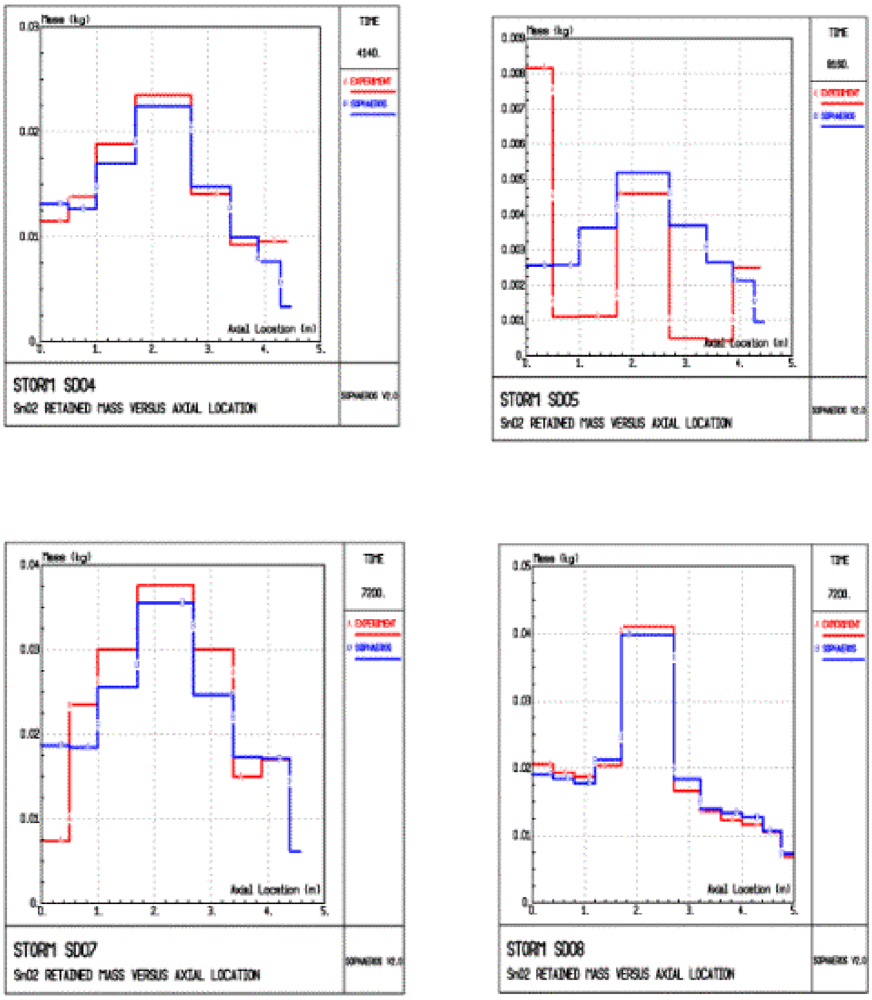


Fig.12: Calculated mass distribution at the end of the re-suspension phase compared with experimental data of the STORM tests SD Series

The applications to STORM SR-09 and SR-11 (ISP30) experiments showed discrepancies on mechanical resuspension especially at high carrier gas velocity (above 60 m/s). A good agreement of the final retained mass in these two tests (see Fig.13) could be only obtained by using artificially for the same material two different factors for the cohesive force coefficient. It is thus obvious that a more mechanistic approach of the re-suspension model should be implemented in the code (like the approach of Biasi, JRC/Ispra, currently under integration into ASTEC by IRSN).

Further examinations concerning the uncertainties in the thermal-hydraulic conditions are still necessary. The main reasons are that the gas temperature profile as well as the inlet gas temperature in the test pipe could not be directly measured. Therefore in all the calculations, the thermal-hydraulic conditions have to be pre-calculated.

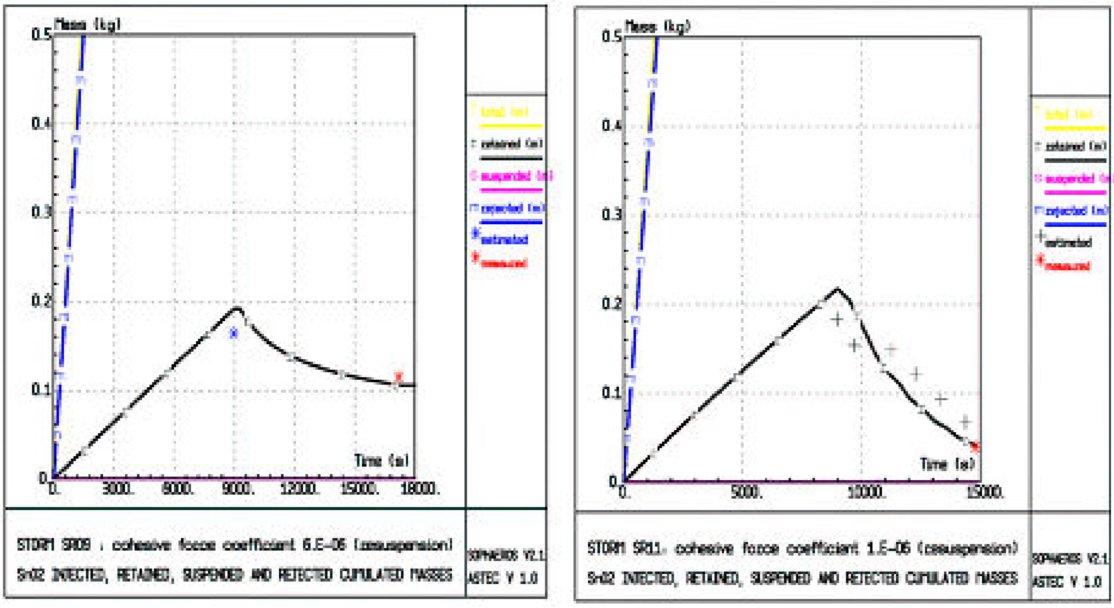


Fig.13: Calculated deposited mass compared with experimental data of the STORM tests SR09 (left) and SR11 (right)

Analyses of two TUBA tests using SOPHAEROS module

The applications to TUBA thermophoresis TT28 (turbulent case) and TT31 (laminar case) experiments showed an overestimation of deposits but it was concluded that, with respect to international standards, the agreement between the Talbot interpolation model used in SOPHAEROS and the TUBA data was acceptable (see Fig. 14 and 15).

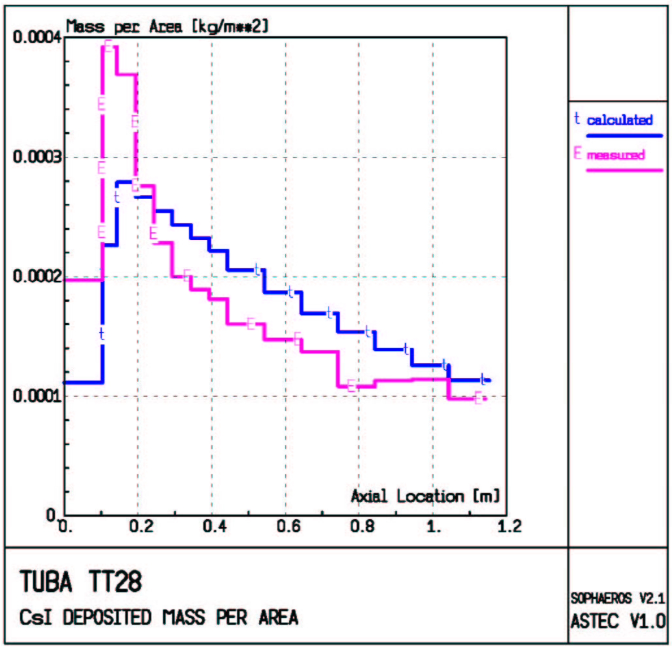


Fig.14: Calculated and measured deposited mass in the heat exchanger (TUBA TT28)

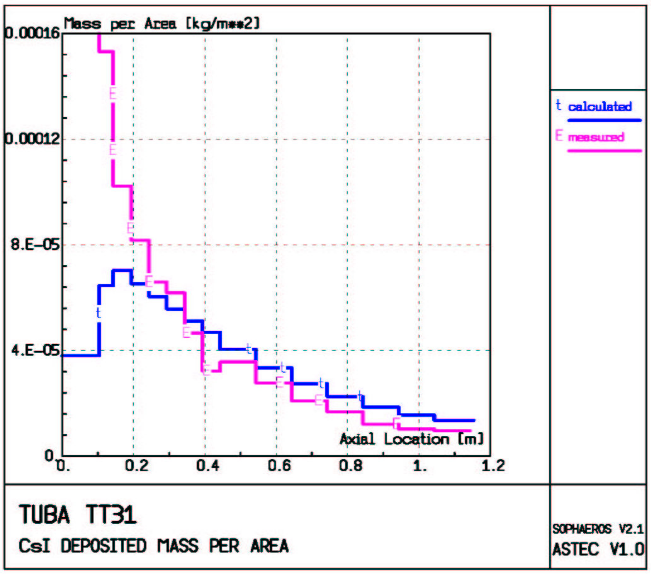


Fig.15: Calculated and measured deposited mass in the heat exchanger (TUBA TT31)

Validation of SOPHAEROS module on selected HORIZON and PSAERO experiments

The applications to HORIZON experiments on a scaled-down model of a VVER-440 steam generator and to PSAERO experiments on a straight part of steam generator tube concluded on the need to improve the models of aerosol turbulent impaction and mechanical resuspension. The resuspension model in SOPHAEROS led to under-prediction (Fig.16). It was recommended to include the turbulent deposition model developed in the SGTR 5th Framework Programme project into SOPHAEROS as an option. Another conclusion was the overestimation of bend deposition at low and at high gas velocities, but bend deposition did not much influence the results of the HORIZON experiments.

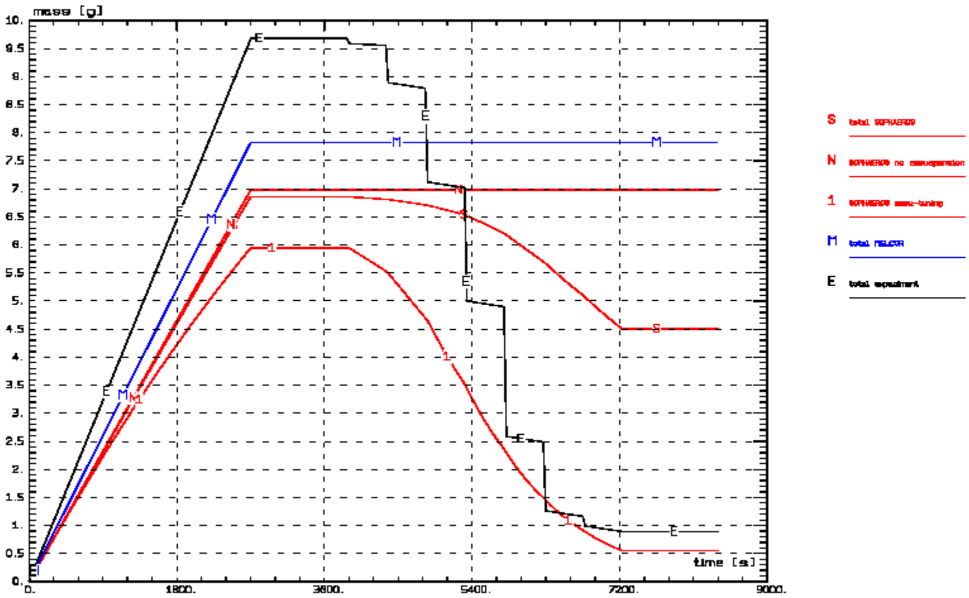


Fig.16

PSAERO experiment 3

Aerosol deposited mass

SOPHAEROS V2.1

MELCOR 1.8.5

Status of SOPHAEROS models

Except for bend deposition, the partners agreed that the aerosol behaviour at low gas velocities, especially at laminar flow, including settling, laminar deposition and thermophoresis is sufficiently described in codes like SOPHAEROS as indicated by the EVITA results. But there is still a lack of validation of deposition in complex structures such as steam generator or upper plenum (see ARTIST program at PSI, Swiss).

5.4 CPA module on containment behaviour

The applications to EREC T5 and VANAM M2*/M3 experiments showed a good agreement with measured data for thermal hydraulics behaviour in a multi-compartment containment.

CPA validation on BCEQ (EREC) T5 blow down test

The CPA module was applied to the EREC T5 experiment simulating a large break LOCA in geometry and conditions of a VVER-440 containment with bubble condenser towers. Using the DRASYS zone CPA model for the bubbler condenser, a good agreement (often better than with CONTAIN code in the Phare 2.13/95 project) was found in particular on pressure peaks and time histories.

The calculated pressure difference on the tray (Fig.17) was higher than the measured value but after the initial pressure oscillation the calculated pressure difference corresponds to the measurement. The agreement was also good on the check valve, gas temperature in most compartments (both codes overestimated the temperature in the gasroom of the tray and in the air trap), and water heat-up in tray. The measured flow rates between the SG box and the localisation tower are very different depending on type and location. The calculated values of CPA and CONTAIN are similar.

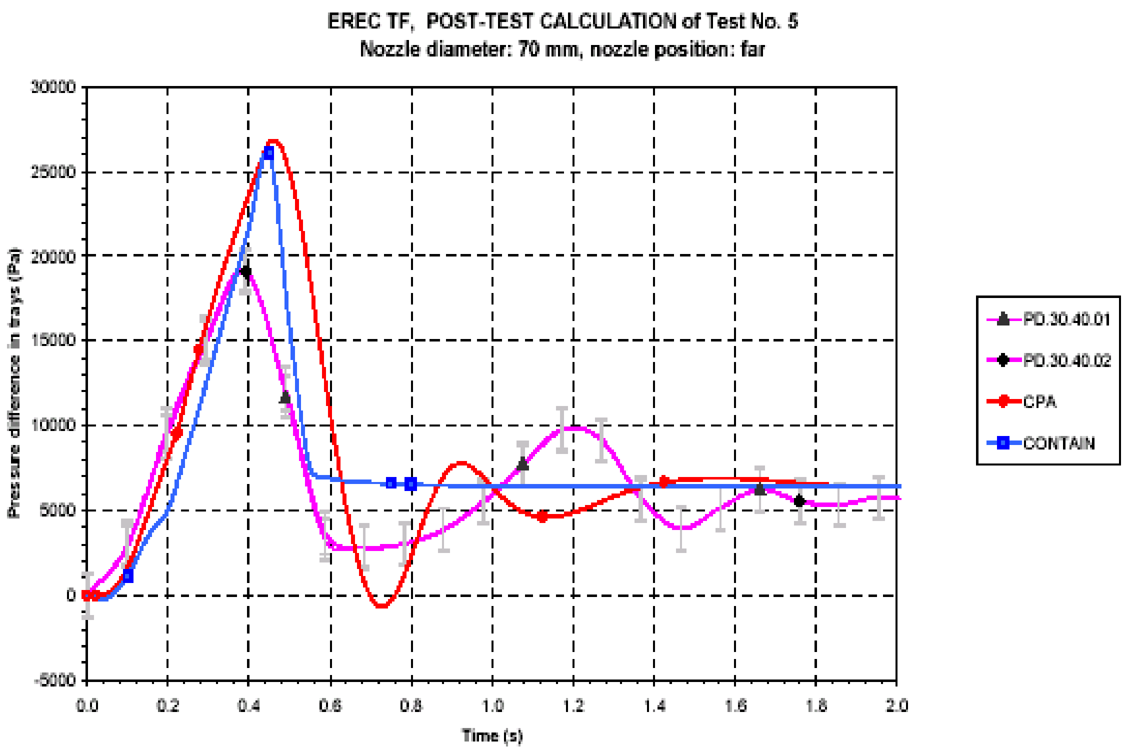


Fig.17: Pressure difference in trays

Simulation of the VANAM M2* experiment with CPA module

The CPA module was applied to the VANAM M2* experiment conducted in the 626m^3 Battelle Model Containment aiming at providing relevant experimental data for behaviour of insoluble and hygroscopic aerosol material in atmosphere containing steam and fog droplets, in a multi-compartment containment. This experiment was performed with an insoluble aerosol material (SnO_2).

The CPA pressure evolution was in agreement with experimental data.

The Fig.18 shows the evolution of the aerosol concentration in the containment gas atmosphere of the dome zone (R9). The two peaks correspond to the two aerosol injection phases: first in a “dry” steam-air atmosphere (saturated or slightly superheated conditions) and later on in a “wet” atmosphere (supersaturated conditions, fog formation). The calculated aerosol concentrations are here in a good agreement with the measurement (this was not right in lateral zones R10 and R11 where final values are too high). The calculation was done with a characteristic length equal to the wall height of the containment.

Comparison results between ASTEC and COCOSYS (performed calculations with the latter code by GRS) show a good agreement for the thermal-hydraulics evolution and for the aerosol depletion, although COCOSYS tends to calculate a slightly higher volume condensation rate and thus more water on the aerosol particles and slightly faster depletion.

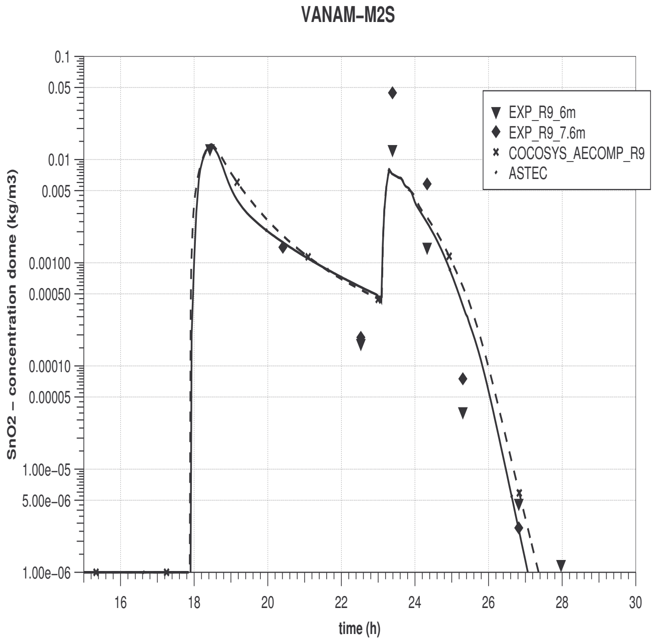


Fig.18: Aerosol concentration in dome zone R9 (experiment, ASTEC and COCOSYS)

CPA validation on NUPEC M7-1

The objective of the experiment NUPEC M7–1 (blind ISP35) was to study the distribution of hydrogen (simulated by helium) in a multi-compartment containment vessel under the influence of a spray system. The test facility represents a ¼ scaled Japanese PWR with a volume size of 1.312m^3 and 25 compartments.

CPA gives reasonable results compared to the experiments and nearly the same agreement like the COCOSYS calculation. The pressure decreases faster in both calculations than in the experiment (Fig. 19 for ASTEC results). The minimum is slightly lower and reached earlier. The following pressure increase is higher in the calculations. This pressure behaviour can be attributed to the distribution of the spray water between the atmosphere and the vessel walls, which is determined by the user. In the calculation too much heat and steam is taken from the atmosphere by the spray water (a too strong pressure drop) and too much heat goes to the structures. This leads to more energy stored in the structure and higher structure temperatures.

During the spray phase, the calculated helium concentrations are well in agreement with the measured concentrations (Fig.20 for ASTEC) in both calculations.

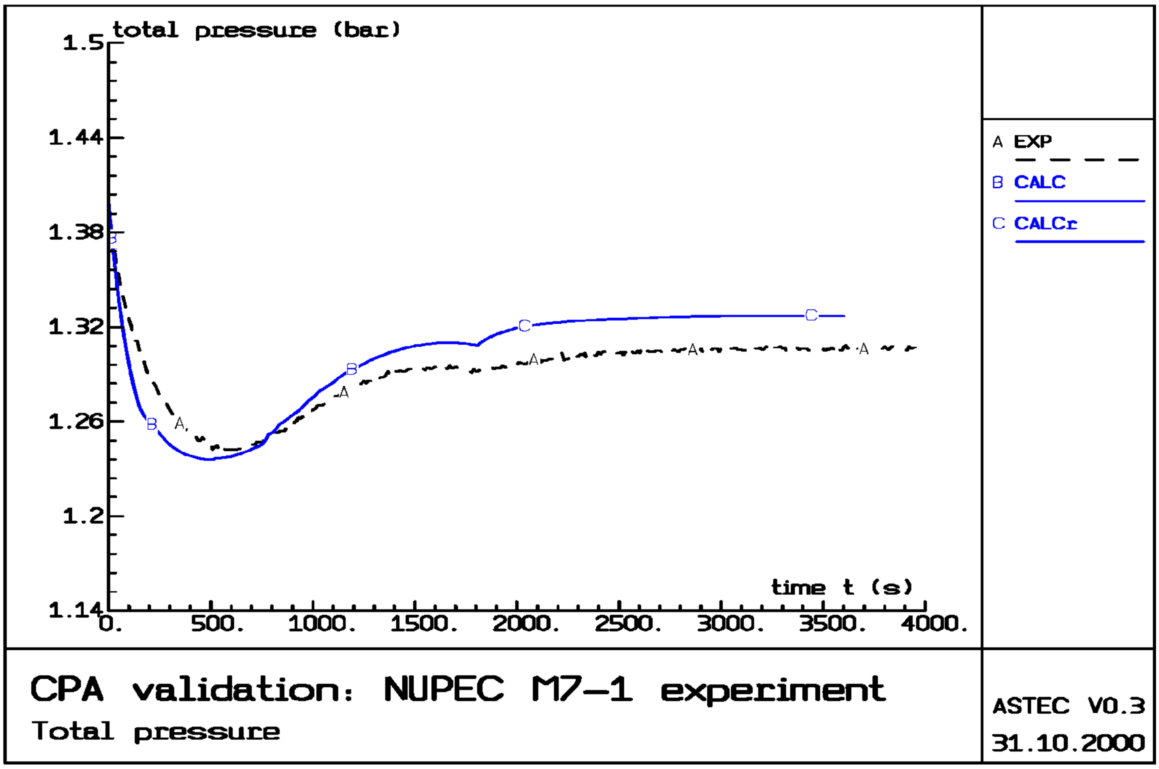


Fig.19: CPA total pressure

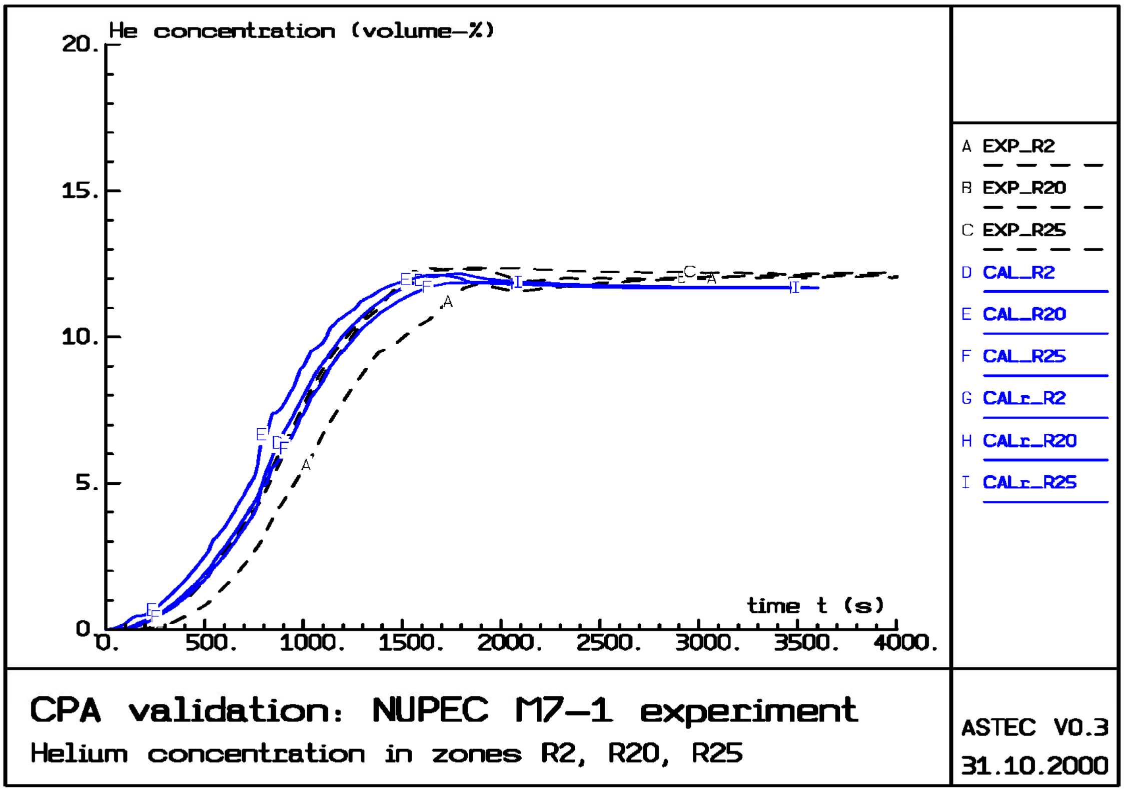


Fig.20: ASTEC – Helium concentration in zones R2 (bottom), R20, R25 (top of dome)

Status of CPA module validation

GRS reported about some general conclusions from its internal experience:

- Depletion of aerosols due to condensation on particles is influenced by the relative humidity and the balance between wall and volume condensation. This is in turn influenced by the choice of the condensation parameters: a characteristic length of 5 cm was recommended by GRS for convective heat transfer since it is an empirical fit to experiments and takes into account the complex geometry of a containment.
- The non-equilibrium model zone model shows a slight tendency to overestimate superheating (but the VANAM M3 experiment is not a good basis to help to conclude because the humidity measurement in this experiment was not reliable enough).
- The ISP 44 on KAEVER experiments showed that important effects like volume condensation, solubility, Kelvin-effect (although the model robustness has to be improved) and hygroscopic effect are well described by CPA.
- The H_2-combustion DECOR model describes well the peak pressure and the pressure difference between compartments, but with a limitation for applications with flame speed below ~300 m/s and with needs of CPU reduction and robustness increase.
- For spray system, a first level of validation was obtained on NUPEC M7.1, CARAIDAS and CSE experiments, but further validation effort on other experiments is necessary.

5.5 IODE module on iodine behaviour in the containment

Applications of the ASTEC V0.3 IODE module to the 1st phase of the OECD ISP41

The version IODE 4.2 of ASTEC V0.3 was applied by NRI to the RTF iodine experiment of the 1st phase of the OECD ISP41. It showed that the experimental results could only be fitted by changing the default values of sump radiolysis coefficients. But IRSN confirmed that the default recommended values were derived from IRSN validation on a wide range of small and medium-scale experiments: these default values are only valid for a given range of parameters, but not for T=25°C like in ISP41 which is not a severe accident temperature for common PWR containments. Anyway, IRSN use of the sump radiolysis improved models which were implemented into the IODE version of ASTEC V1.0, based on a mechanistic approach with a dependence on pH and temperature, gave better overall results on this ISP41 1st phase.

Benchmarking the IODE module against the results of the OECD International Standard Problem No. 41 Follow-up (Phase 1)

The first OECD ISP 41 exercise, an iodine-chemistry code comparison study, was based on a simple RTF test and showed that all the participating iodine-chemistry codes had the capability to reproduce the iodine behaviour for a *narrow* range of severe-accident conditions. However, in view of the increasing demand for nuclear-safety analyses to move from conservative towards best estimates, iodine-chemistry codes are required to demonstrate that they can provide accurate predictions of the iodine volatility for a *wide* range of conditions. Hence, the second step of the ISP 41 exercise (ISP 41 Follow-up Phase 1 or ISP 41-F) consisted of a set of parametric studies to examine the sensitivity of the code output to boundary conditions such as pH, dose rate, initial iodine concentration and the presence of organic impurities, painted surfaces and silver.

In agreement with other empirical and mechanistic iodine-chemistry codes used in the ISP41-F exercise and in line with our current understanding of iodine chemistry under severe-accident conditions, the IODE 5.1 module of the integral ASTEC code (version V1.0) predicts a pronounced impact of the solution pH and the silver-iodine reactions on the gas-phase chemistry with the lowest iodine volatility for a basic pH in the presence of silver. Not surprisingly, the effect of changes in dose rate and initial iodine concentration was found to depend on the sump pH. This parametric study (10 calculations) also demonstrated that IODE reproduces the trends for the volatile-iodine behaviour (see ASTEC results in Fig. 21) observed with the other codes used in the ISP41-F exercise. The agreement is less satisfactory for the liquid-phase chemistry and for the amount of iodine adsorbed on liquid- and gas-phase surfaces, in particular for the test cases at low pH.

While it is likely that these discrepancies are caused by important changes in the underlying modelling in IODE 5.1, which affect the radiochemistry of iodine, a detailed quantitative analysis of the reasons for the observed differences would be called for. Due to the unavailability of data on the iodine speciation from the ISP41-F this analysis could, however, not be performed.

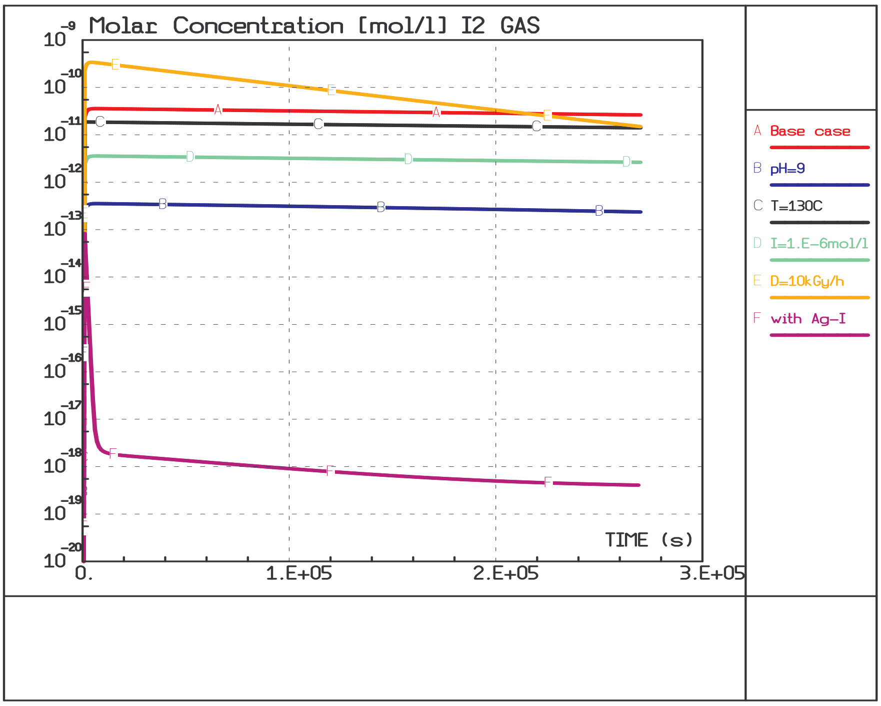


Fig.21: IODE predictions of the gaseous I_2-concentration as a function of boundary conditions. The base-case calculation was performed for T = 90 °C, $I_0 = 10^{-5}$mol/l, D = 1 kGy/h, pH = 5 and without Ag-I reactions

Status of IODE module validation

IRSN concluded that there is a need of further validation of the most recent models, especially organic iodide formation in the gas phase and in the liquid phase. Other validation needs concern especially I_2-adsorption and desorption on surfaces between gas phase and liquid phase and transport of iodine through a multi-compartment containment (ThAI experiments under performance).

5.6 WEX module for MCCI

The WEX module was applied to the BETA 1.8, MACE M0 and SURC4.

The results on BETA 1.8 show that WEX predicts well the final cavity shape. The axial erosion is dominant compared to the radial erosion (Fig.22). The final axial ablation depth is slightly underestimated at the end of the heating phase (500s). The gas release flow is well predicted. Especially the calculated values of the burnable gases H_2 and CO are in good agreement with the experiment. The values for steam and CO_2 are under-predicted. Both the metal and the oxide temperatures show a qualitative good behaviour. During the interaction the temperatures decrease at a high rate because the heat transfer between molten corium and concrete was very efficient. However the temperatures are over-predicted.

The specificities of the MACE M0 (complex experimental behaviour) and SURC4 (adding some Zirconium during the experiment) experiments could not be predicted in details by the models.

The overall behaviour of the code on these 3 experiemnts was reasonable. In most cases the vertical erosion is calculated quite well. In addition to that, mostly the temperatures are predicted well. To obtain better results in detail, some model adjustments should be performed in the future and basic data should be improved. It is expected that the new MCCI module MEDICIS becoming part of the next ASTEC version related within SARNET overcome most of the identified difficulties.

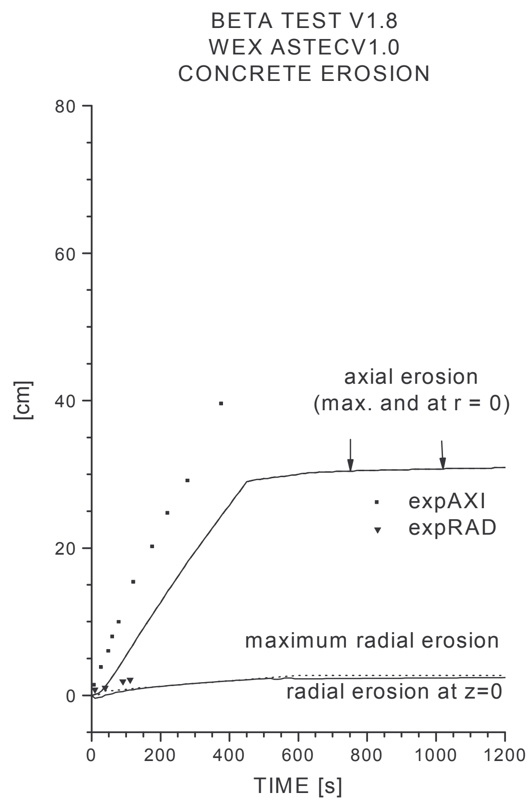


Fig.22: Comparison of BETA 1.8 axial and radial erosion as calculated with WEX and as measured in the experiment

5.7 Applications to the Phébus integral experiment FPT1

ASTEC V0 applications to the Phébus integral experiment FPT1

These ASTEC V0 applications showed the code capability to perform a calculation of an integral test with a reasonable CPU running time while coupling most of modules (VULCAIN, ELSA, SOPHAEROS, CPA and IODE). Using only standard models and parameters (without tuning), the calculation yielded the general qualitative experimental trends, and for some phenomena a satisfactory quantitative agreement with the measurements. But some discrepancies appeared, mainly on FP release and transport behaviour, part of which have been also identified in ASTEC V1 application (see below for more details).

ASTEC V1 application to the Phébus integral experiment FPT1

The results of the ASTEC V1.0 application on the Phébus FPT1 in the frame of the ISP46 exercise showed the code capability to perform a calculation of an integral test. Globally, experimental results have been satisfactorily reproduced:

- Acceptable agreement for bundle degradation: fuel temperature evolution, hydrogen production, and final state of degradation (using a relocation temperature of 2.450 K for UO_2 and ZrO_2 derived from the IRSN interpretation of Phébus FP experiments with ICARE2 mechanistic code).

- Correct prediction of kinetics and final release of volatile fission products but discrepancies on semi-volatile release (Mo, Ba) and under-estimation of release of structural materials (Fig.23).

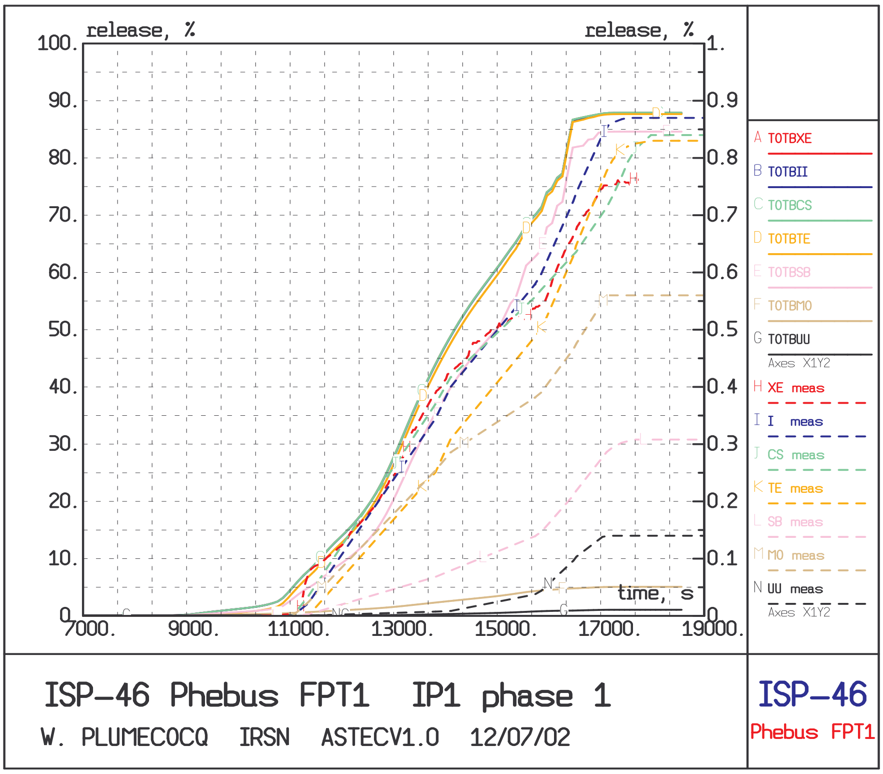


Fig.23: Fission product release from the core

- Good agreement on general fission products retention in reactor coolant system, but - as all other codes - a too low deposition in the upper plenum and a too high deposition in steam generators. No iodine species in gaseous form at circuit exit was predicted during the oxidation phase as observed in the test. There was agreement on thermal hydraulics conditions in the containment. Under-estimation of the aerosol source for the containment calculation but correct repartition factor of the deposited aerosol mass. A very good agreement of the deposition by diffusiophoresis was obtained with a stand-alone CPA-calculation by injecting the experimental source of aerosols (Fig.24). Agreement of I_2 concentration in the gas phase (up to the washing phase which could still not be modelled in the version used), while the CH_3I-concentration was lower by a factor of ten (probably due to the lack of account for dose rate in the gas phase). Trapping of iodine by silver in the aqueous phase was as well reproduced as the low concentrations in the gas phase.

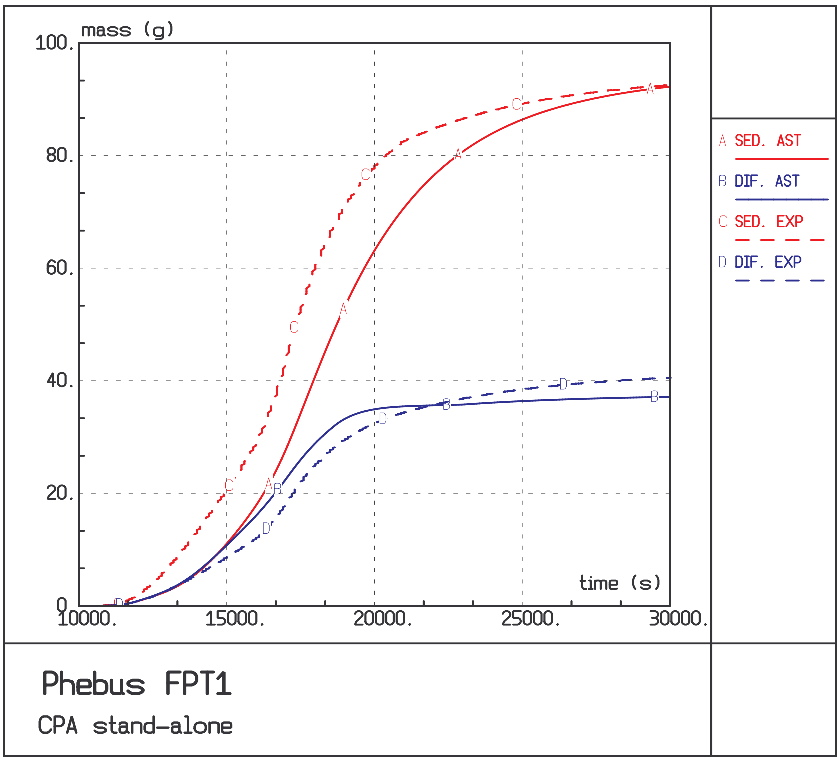


Fig.24: Evolution of the deposited mass of aerosols by sedimentation and diffusiophoresis in the containment (CPA stand-alone calculation)

The main limitations were:

- Need of clarification of the too high temperature of gas at the bundle exit.
- No model for release of structural materials (except for Ag-In-Cd); need of review of modelling of release of semi-volatile fission products (Mo, Ba) and Ag-In-Cd;
- The primary/containment source term coupling interface: fission products under gaseous form and especially iodine, as calculated at the primary circuit outlet, are transformed and ‘seen’ by the containment module in aerosol form.
- No model of washing phase: iodine species are not drained by water along walls. (Included in the next version ASTEC V1.1).
- No calculation of dose rate in the gas phase in the present version (user input) although it is involved in some reactions of the IODE module.

6 WP5: ASTEC plant applications

Summary of ASTEC V0 plant applications in the first EVITA stage

Several applications have been performed on various types of reactors: Konvoi 1300, French PWR 900, VVER-440/V-213 or V-230, and VVER-1000. They covered mainly LOCAs sequences but also a SBO one in a VVER-1000 and a TMLB one in a PWR 900.

ASTEC V0 applications showed that the code was able to calculate LOCA scenarios without difficulties for PWR and also for VVER. Most VVER features could be simulated correctly with the code, including for instance the VVER-440 bubble condenser towers. However the IRSN experience showed that there were difficulties in calculating station blackout events where some water remains in the primary circuit, which cannot be represented correctly by VULCAIN module.

As for accident management measures, the code allowed to correctly represent most existing safety systems: primary circuit depressurisation, water injection systems, containment spray, and passive heat removal systems. One important missing model was linked to situations of late reflooding of a degraded core, which was not possible with VULCAIN models.

The main limitation for sequence calculations was obviously the absence of accident front-end primary system calculation, which leads to difficulties when starting the core degradation calculation and describing correctly its initial conditions. Another limitation was related to modelling insufficiencies during the in-vessel late phase, mainly high-temperature fuel behaviour and formation of ceramic mixtures.

The lack of flexibility of VULCAIN was observed to model VVER geometry, particularly for VVER-440: fuel shrouds and fuel followers in tandem design.

IRSN and GRS indicated that the VULCAIN lack of flexibility, first to integrate new models such for instance developments for new generation reactors, and secondly for an easy validation against experiments (the module was initially designed to cover only plant applications) was an important reason to replace it by two new modules in ASTEC V1.

ASTEC V1 plant applications in the second EVITA stage

The progress made with the switch from ASTEC version V0 to version V1 made it possible to start performing several different plant sequence calculations from the very beginning of the accident, i.e. the front-end phase. The main outcomes of this extensive activity are summarized in the following chapters.

6.1 French PWR 900MW$_e$

For the French PWR 900MW$_e$, two scenarios were calculated with ASTEC V1 (CESAR-DIVA) and the in-vessel results were compared with MAAP4 results:

- Total loss of steam generator feedwater without any active or passive injection system available,
- Total loss of electric power (Station Blackout).

Loss of SG feedwater in a French PWR 900MWe

The calculation of the first sequence demonstrated the ability of the new ASTEC modules CESAR and DIVA to compute in principle the behaviour of the vessel and the circuits during a severe accident from the initial event until vessel rupture. The main physical phenomena were computed altogether: thermal hydraulics, chemistry, vessel mechanics, and core degradation. This was the first reactor case computed with the two modules in coupled mode up to vessel lower head failure. Fig.25 shows the modelling of the plant.

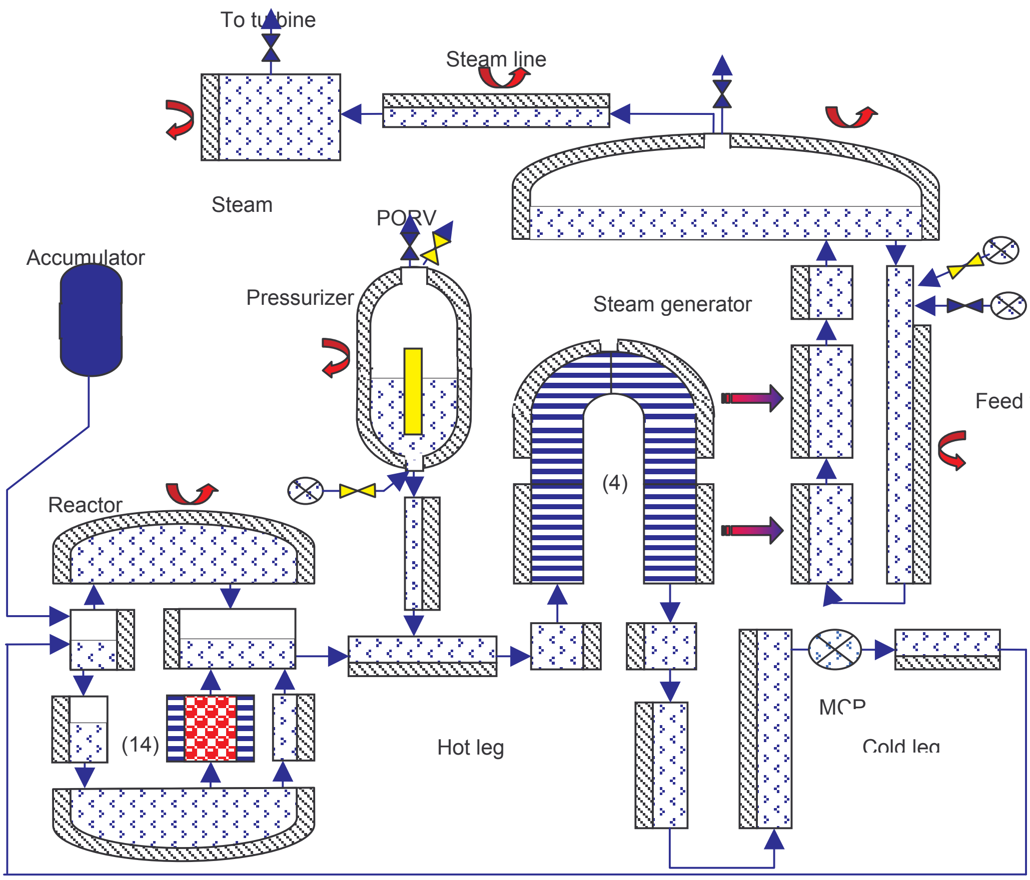


Fig.25: CESAR-DIVA modeling of the PWR 900 plant

A comparison of steady state and front-end thermal-hydraulics was also performed with CATHARE2 on this sequence (see Fig.26). The pressure curves of ASTEC and CATHARE showed quite similar shapes. The main differences concerned a sooner increase of pressure with ASTEC than in CATHARE after the steam generators are dried (due to a too low quantity of water in the secondary side of steam generator in ASTEC), oscillations due to opening and closing of the power operated relieve valve (PORV) -to smooth the curve like in CATHARE, it would be necessary to compute the section implicitly-, and pressure increase again with ASTEC after PORV full opening by the operator, which seems not physical (probably linked with modeling of heat exchanges between walls and fluids).

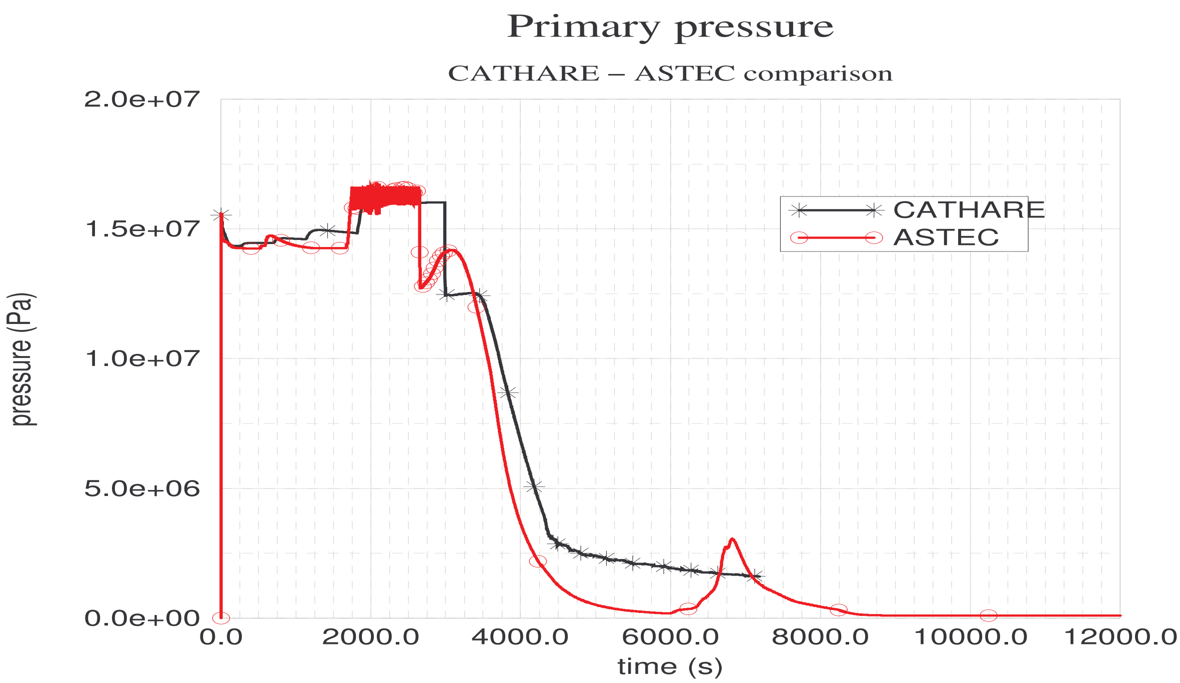


Fig.26: Comparison between ASTEC and CATHARE results on the front-end phase of the H2 sequence

French PWR 900 with total loss of feedwater, ASTEC V1 comparison with MAAP4

The ASTEC V1 results were behaving in the same way than MAAP4, both regarding timing and relevant values, where discrepancies lie within a 20 % range (see the Table 3 below and the Fig.27 on RCS pressure). The differences that can be observed at this stage of the comparison were mainly explained through the ASTEC model assumptions of corium candling for the rods degradation, without formation of a fully tight molten pool like in MAAP4, and the absence of power release through fission products in DIVA calculation.

Table 3: Transient time events

Events	ASTEC Time (s)	MAAP Time (s)
Secondary feed water pumps off	0	0
Scram	12,8	9,2
Steam Generators dried out	1.610	2.282
PORV fully opened on 330°C criteria	2.652	2.879
Primary pumps off	2.655	2.885
Start of core uncovery	3.420	3.432
Onset of oxidation	3.602	4.100
First material slump into lower plenum	4.274	7.448
Vessel failure	11.334	10.189

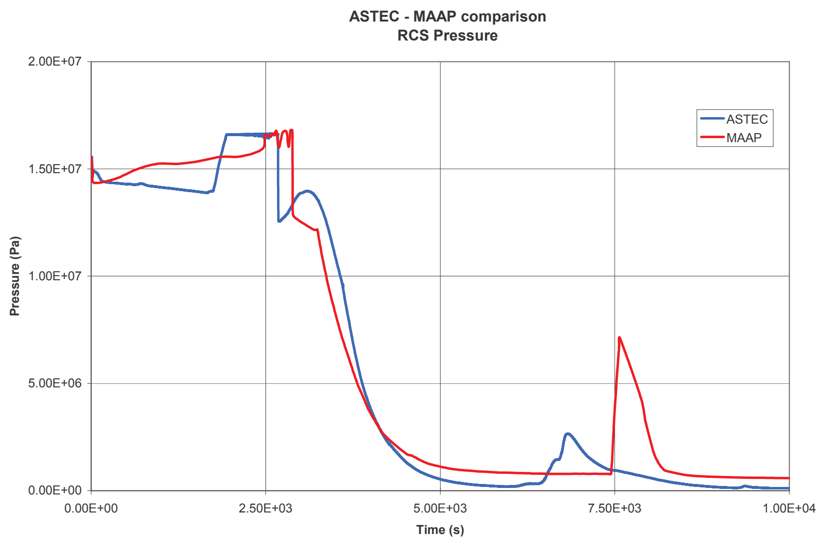


Fig.27: Comparison of RCS pressure between ASTEC V1 and MAAP4 calculations

Loss of electric power in a French PWR 900MWe / Comparison MAAP-ASTEC V1

Results of the comparison on thermal hydraulics with MAAP4 were rather good during the first three hours of this accident: pressure and mass of water in primary and secondary circuit were close (Fig. 28 and 29). But ASTEC yielded a much smaller final hydrogen mass than MAAP4: 200 kg instead of 600 kg with MAAP4. This seemed to be caused by a much later accumulator discharge with MAAP4 than with ASTEC.

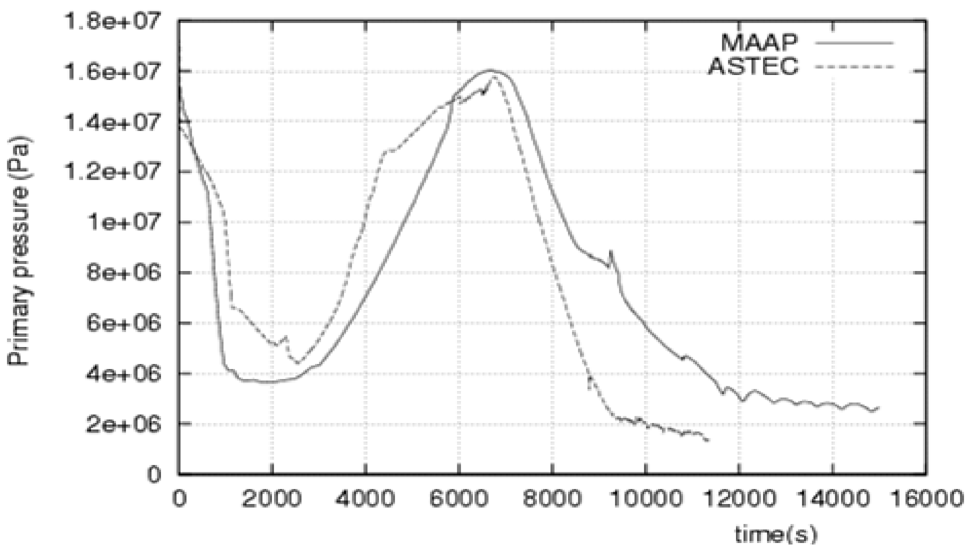


Fig. 28: Comparison of primary pressure between ASTEC V1 and MAAP4 on a loss of electric power sequence for a French PWR 900

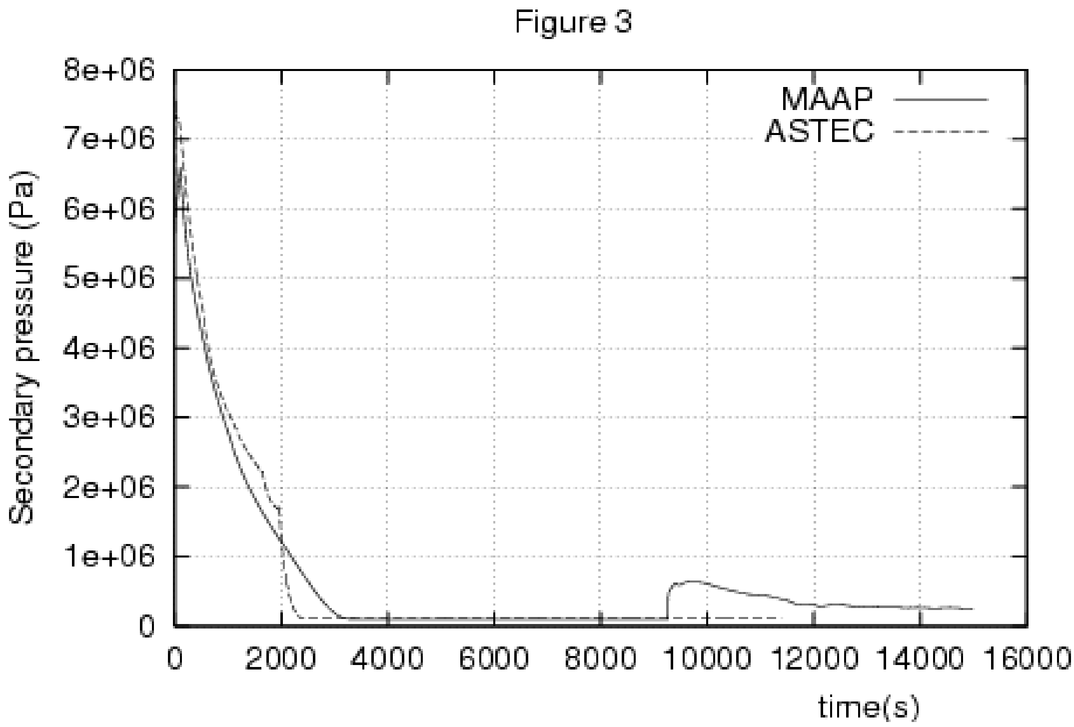


Fig. 29: Comparison of secondary pressure between ASTEC V1 and MAAP4 on a loss of electric power sequence for a French PWR 900

The calculation of this sequence stopped at about 11.400 s due to a numerical problem in ASTEC without reaching vessel failure. This shows the necessity to improve the code robustness for further work with ASTEC.

In the future, within SARNET, both sequence calculations have to be investigated in more detail: for the first one reflooding has to be calculated until the envisaged end, the first step for the second sequence is to calculate reactor pressure vessel failure precisely. It is true that ASTEC has not reached the level of maturity of MAAP. But EDF is interested in ASTEC developments and will go on the ASTEC evaluation in the frame of SARNET project.

Preparation of a comparison on a high pressure sequence for a French PWR 900 between ASTEC V1 and MELCOR 1.8.5

A MELCOR 1.8.5 input deck has been built to represent the geometry of a French PWR 900MWe and the scenario of a TMLB accident, in a very similar way to the ASTEC input deck in the file H2.dat delivered by IRSN in EVITA along with the ASTEC V1 version. The Fig.30 presents the MELCOR 1.8.5 modelling of the primary circuit for the calculation of this sequence TMLB.

This MELCOR input deck is ready now to be used, beyond EVITA, for benchmarks with the future ASTEC V1 versions.

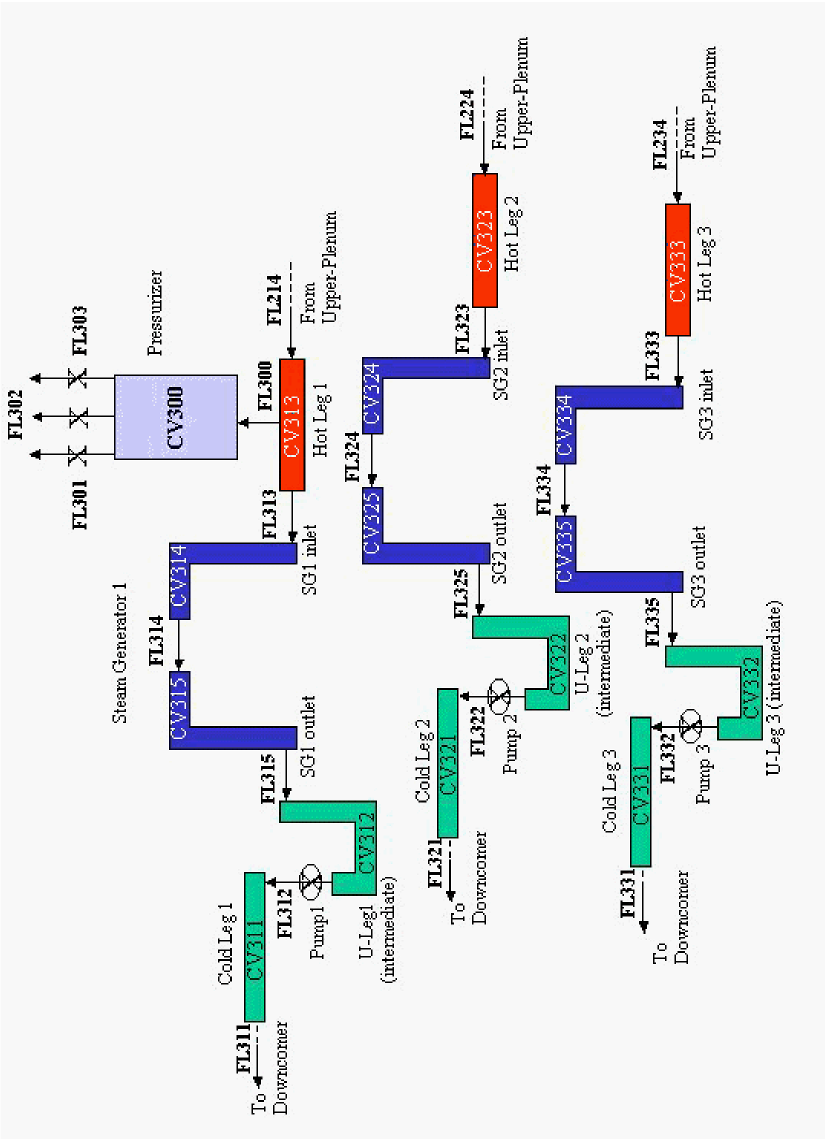


Fig.30: French PWR900 MELCOR primary circuit modeling (3 loops)

6.2 Westinghouse PWR 1000

For the Westinghouse PWR 1000MW_e, two scenarios were calculated with ASTEC V1 (CESAR and DIVA) and the in-vessel results were compared with MELCOR 1.8.4:

- Station blackout
- Small break LOCA (1") in the cold leg with no safety injection systems available.

In both applications, the calculations stopped normally when the temperature criterion was reached for lower head vessel failure. It was not possible to activate the ELSA module due to execution errors. Consequently the release of fission products inside the core has not been evaluated.

Westinghouse PWR 1000 Station-Blackout sequence analysis and comparison with MELCOR 1.8.4

The steady-state results showed good agreement with the power plant actual characteristics. The model of the secondary side should be refined to adjust the secondary flow rate to the real one.

At the start of the station blackout sequence, the agreement between ASTEC and MELCOR was good. The safety valves started and stopped oscillating at close times. But the discrepancy observed in the pressure and the liquid mass inside the pressurizer was not well understood as initial conditions in MELCOR and ASTEC models are the same.

The Table 4 presents the timing of key events with both codes. Core uncovery started nearly at the same time in both calculations but core degradation exhibited large differences between both codes (in ASTEC, start of fuel melting about 3.000 s later but the reactor vessel break occurs at about 5.000 s sooner than in MELCOR). The in-vessel hydrogen generation did not differ much (Fig.31): with ASTEC about 450 kg and with MELCOR about 590 kg were calculated. The fraction of oxidized Zirconium was much higher in ASTEC (64 %) than in MELCOR (41 %).

Table 4:Timing of key events for SBO scenario on West.1000 (comparison ASTEC-MELCOR).

Event		**Time**			
		ASTEC		**MELCOR**	
		sec.	**hours**	**sec.**	**hours**
Event initiation		0	0	0	0
Start of SG PORV cycling		335	0,09	--(Not available)	--
End of SG PORV cycling		2.565	0,71	2.700	0,75
Start of PZR PORV cycling		2.605	0,72	3.200	0,89
Start of core uncovery		5.594	1,55	5.952	1,65
Start of PZR SRV cycling		5.959	1,66	4.503	1,25
Rapid oxidation		7.640	2,12	7.250	2,01
Total core uncovery		7.850	2,18	7.200	2,00
Gap release	Ring 1	7.873	2,19	6.972	1,94
	Ring 2	7.915	2,20	--	--
	Ring 3	10.032	2,79	--	--
Fuel Melting	Ring 1	10.862	3,02	7.000	1,94
	Ring 2	10.892	3,03	7.200	2,00
	Ring 3	10.862	3,02	7.200	2,00
Guide tubes dislocated	Ring 1	10.419	2,89	--	--
	Ring 2	10.438	2,90	--	--
	Ring 3	10.471	2,91	--	--
Claddings dislocated	Ring 1	10.419	2,89	--	--
	Ring 2	10.439	2,90	--	--
	Ring 3	10.475	2,91	--	--
Vessel Breach (VB)		14.982	4,16	20.300	5,64

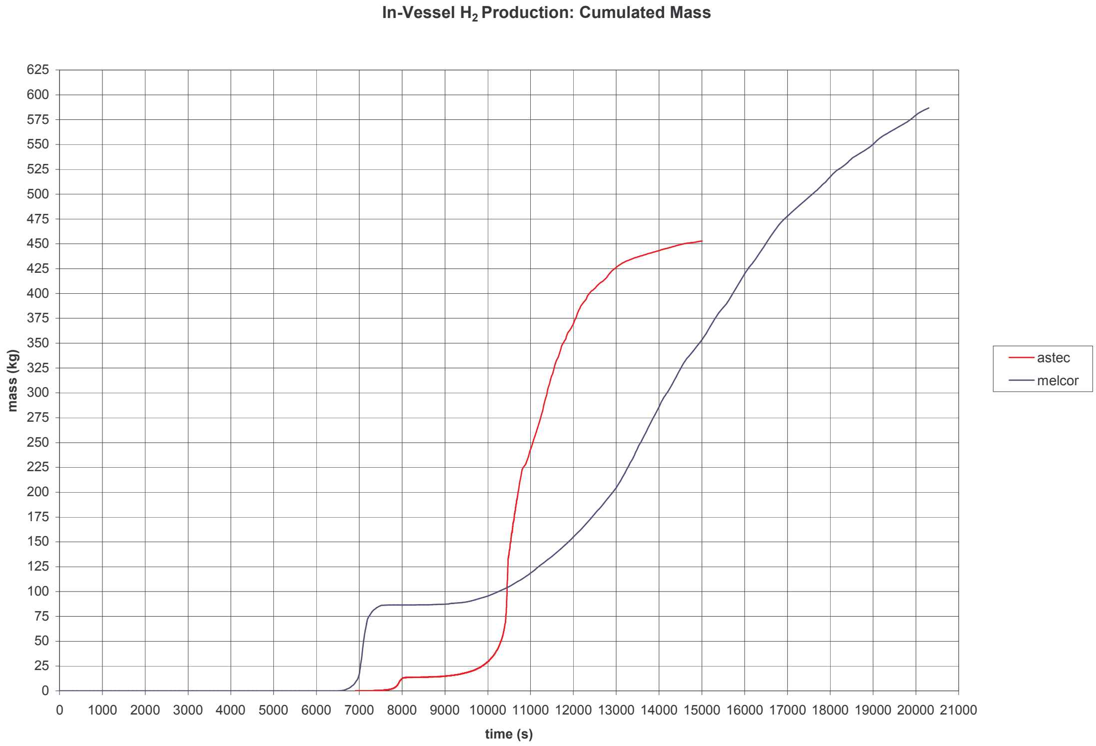


Fig.31: Comparison between ASTEC and MELCOR on hydrogen production (cumulated mass) for the SBO sequence in a West.1000

Analysis of 1" LOCA accident in a PWR-1000MWe with CESAR/DIVA

For the LOCA sequence, the ASTEC calculation gave a very delayed start of core degradation, probably linked with the simplified representation used here for the secondary circuit. But in compensation the degradation was much faster in ASTEC, which led to close times for vessel rupture (9.400 s for ASTEC and 8.300 s for MELCOR). The Table 5 presents the timing of the main events.

So a more detailed analysis of the plant model and of the steady state is needed. All the core degradation phenomena were simulated suitably: fuel temperature peak matched with MELCOR value (Fig.32), the hydrogen production escalation was quite similar in both codes, although in ASTEC the total hydrogen mass released was smaller (180 kg in ASTEC vs. 280 kg in MELCOR).

Table 5: Timing of the main degradation processes for the SBLOCA sequence on the West.1000. Comparison of ASTEC and MELCOR simulations.

	ASTEC (case 1)	ASTEC (case 2)	MELCOR
Beginning LOCA	0s	0s	0s
SCRAM	275,5s	275,5s	275s
Start DIVA	11.200s	6.000s	-
Beginning of core uncovering			3.400s
Cladding oxidation	11.660s	7.000s	4.300s
Ag-In-Cd melting	12.350s	~7.800s	
Cladding oxidation runaway	12.500s	8.300s	4.800s
UO_2 melting	~12.500s	~8.500s	
Corium slump	12.470s	7.890s	6.100-8.060s
Complete core uncovering	13.390s	7.830s	6.800s
Vessel rupture	14.080s	9.360s	8.300s

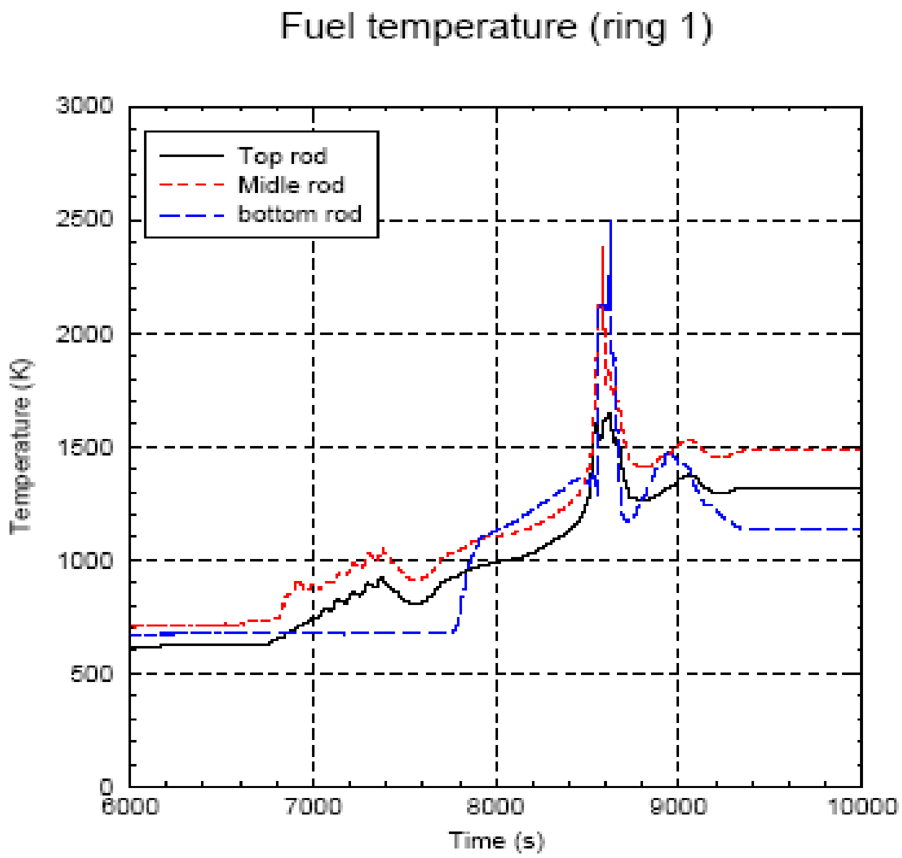


Fig.32: . Fuel temperature in central ring (ASTEC

These two above comparisons between ASTEC V1 and MELCOR were only preliminary and limited to the RCS and vessel behaviour. In the steps beyond EVITA, the efforts will be focused on a more detailed comparison of the in-vessel part of the accident sequence and on an extension of calculations up to the containment failure.

6.3 KONVOI 1300MW$_e$

MBLOCA in a German PWR 1300 with MELCOR comparison

For the German KONVOI 1300MW$_e$ PWR type, two sequences were calculated:

- Small break LOCA (5 % leak)
- Medium break LOCA (200 cm^2 leak in the hot leg of the pressurizer loop).

The SBLOCA calculations were performed only for front-end thermal hydraulics using the ASTEC V1 module CESAR. CESAR results were compared with ATHLET ones. This SBLOCA sequence is of interest as German PWR have hot leg emergency core cooling injection, e.g. in contrary to French PWR with cold leg injection.

Some CESAR results were quite similar to ATHLET ones such as primary pressure. But the cladding temperatures are much higher because the whole core dries out, which is different from ATHLET results. Some reasons are a larger mass loss in the blow-down phase and a different water distribution in the primary circuit in CESAR calculation. Another reason is a much lower injection rate of accumulators. These thermal hydraulic results underline the necessity of further intensive benchmarking or validation. On the basis of the presented comparison, it seems that the models describing critical discharge, relative motion of gas and liquid phase (counter current flow) and the drift model should be checked once more carefully. Particularly the drift model should be further investigated. It seems that in the present version CESAR has difficulties to simulate hot leg emergency injection. This application underlined also some remaining problems of convergence when doing relatively small changes in parameters of integration procedure and model options.

For the MBLOCA sequence, the ASTEC V1 modules CESAR, DIVA, ELSA and CPA were activated in a coupled mode. The catalytic recombination of hydrogen and carbon monoxide was taken into account in the CPA module. The SOPHAEROS module was not activated because the retention of fission products and aerosols in the loops is neglected, as the break is located near to the reactor pressure vessel in the hot leg.

This calculation with the updated version demonstrated that ASTEC V1 has actually the ability to calculate a severe accident sequence from the start of the event until vessel rupture. The main physical phenomena for the primary circuit and the secondary circuit as thermal hydraulics, chemistry, mechanics and core degradation, and behaviour of water, steam, hydrogen and fission products in a 25-nodes containment can be computed as well.

These ASTEC results were compared with results from a MELCOR 1.8.4 calculation:

- The agreement on timing of main events between both codes was qualitatively acceptable: first slump of corium at 5.100 s with ASTEC and 6.300 s with MELCOR; reactor pressure vessel failure at 13.350 s with ASTEC and 12.400 s with MELCOR (see Fig.33).
- The amount of H_2 in-vessel production compared with calculations with the former ASTEC version is now in a better agreement with MELCOR, but yet lower: 450 kg with ASTEC and 640 kg with MELCOR.

- The trends of the curve of the ASTEC containment pressure evolution were in a good accordance with the MELCOR results (see Fig.34).
- The treatment of aerosol release from the core led to code instabilities and could not be conducted further than until 5.800 s.

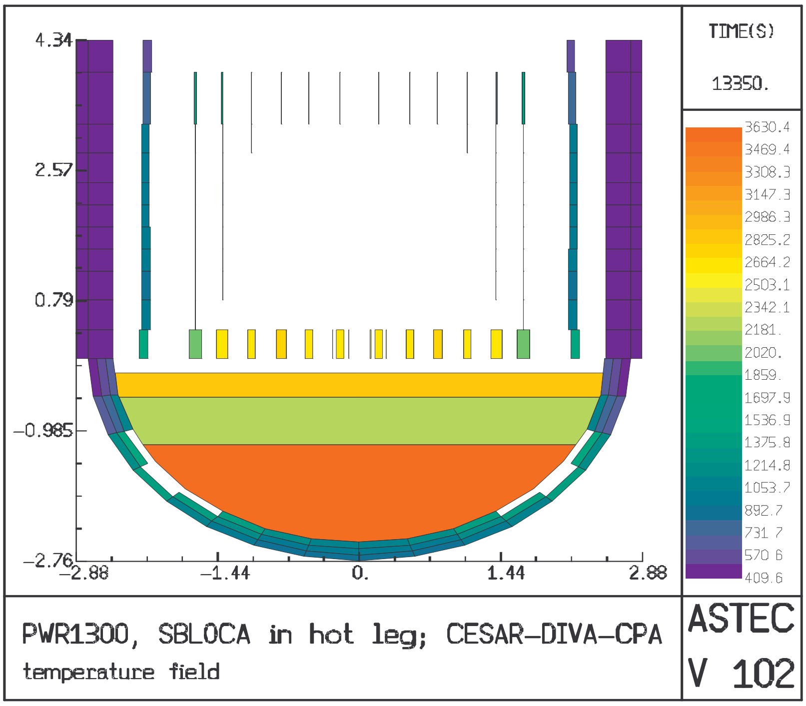


Fig.33: Corium in the lower plenum shortly before vessel rupture

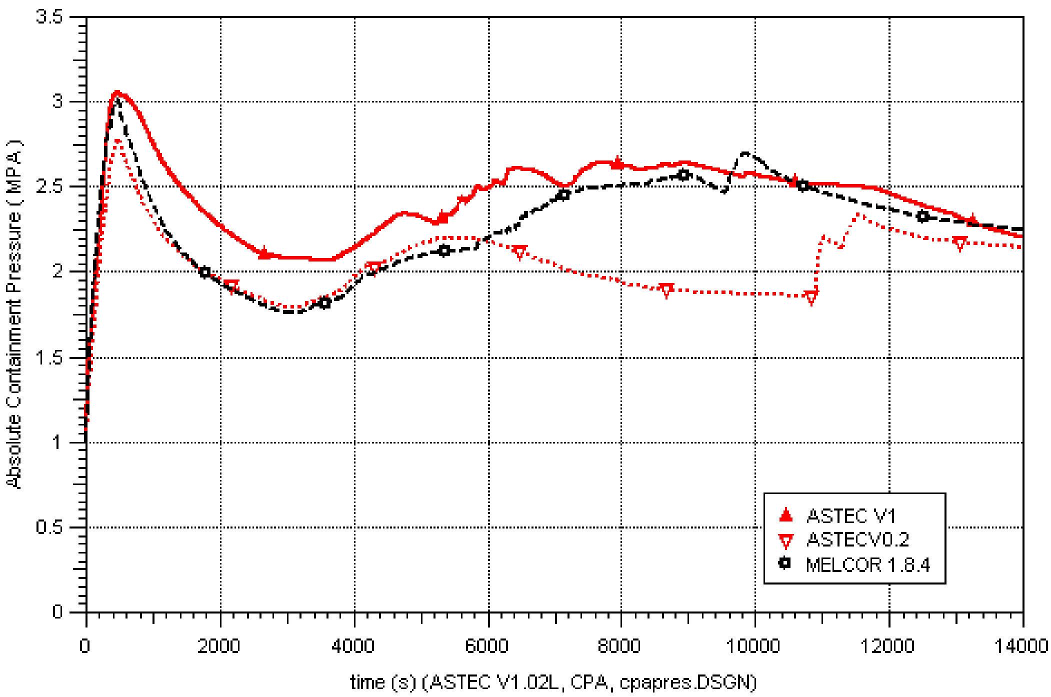


Fig.34: Comparison of the evolution of the containment pressure for ASTEC V1rev2L, ASTEC V0 and MELCOR

The calculations were performed on PC-linux and PC-windows. The results showed some portability effects. The results on PC-linux seemed more reliable than the PC-windows ones.

Significant points for the further work of ASTEC code development should be the reduction of the portability effect that was observed, further increase of the robustness of the code and extension to complete fission products calculations including aerosols and iodine chemistry.

Both ASTEC V1 KONVOI sequence calculations showed the fast running capability of ASTEC V1. The Fig.35 shows an example for the on-line and off-line information provided by the GRS visualisation tool ATLAS actually under coupling to the next ASTEC V1 version.

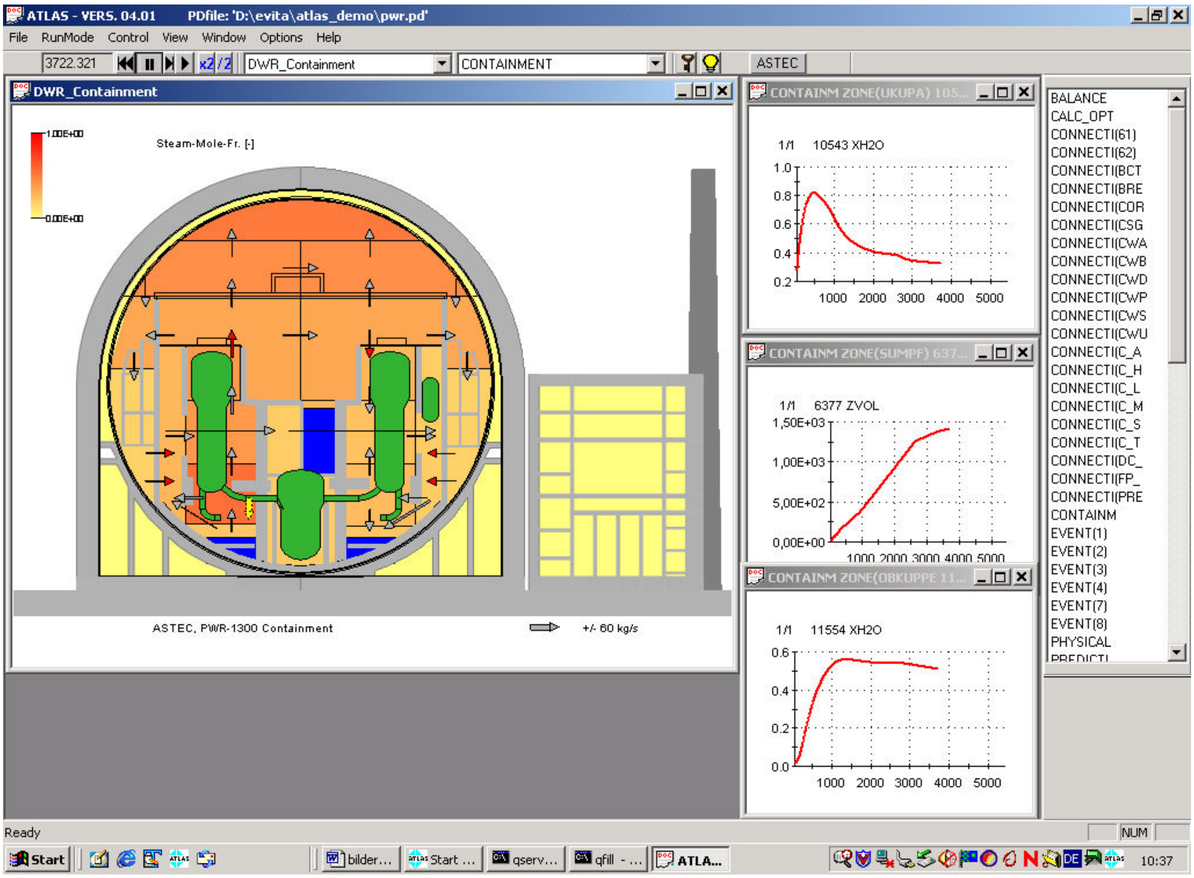


Fig.35: Overview over the steam mole fraction in the compartments of the containment shown by the GRS-tool ATLAS.

6.4 European Pressurized water Reactor (EPR)

2"SBLOCA analysis for EPR containment with ASTEC V1 CPA and comparison with COCOSYS results

The CPA module of the integral code ASTEC V1.0 was applied as a stand-alone code for containment analysis with respect to thermal hydraulic and aerosol behaviour after a Small Break Loss of Coolant Accident (SBLOCA). The results were compared with results obtained for corresponding input data by the containment code system COCOSYS V2.0 (see Fig. 36 to 38).

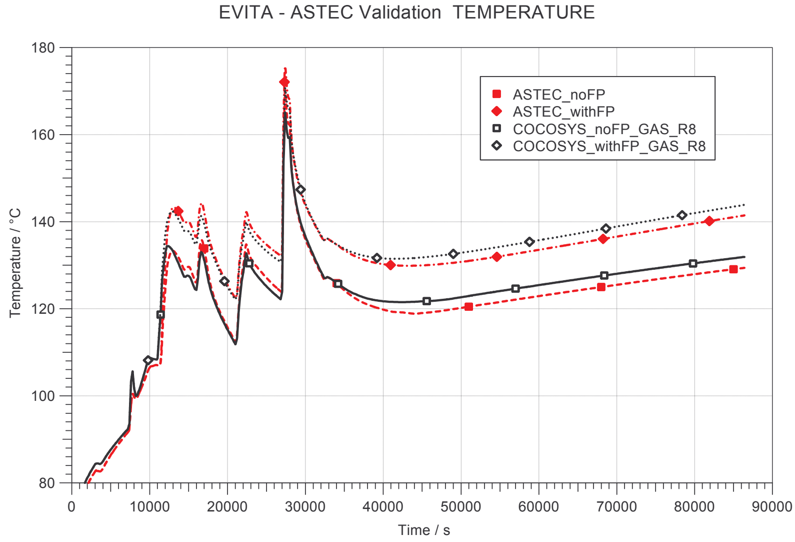


Fig. 36: Gas temperature in R8 - ASTEC-COCOSYS comparison

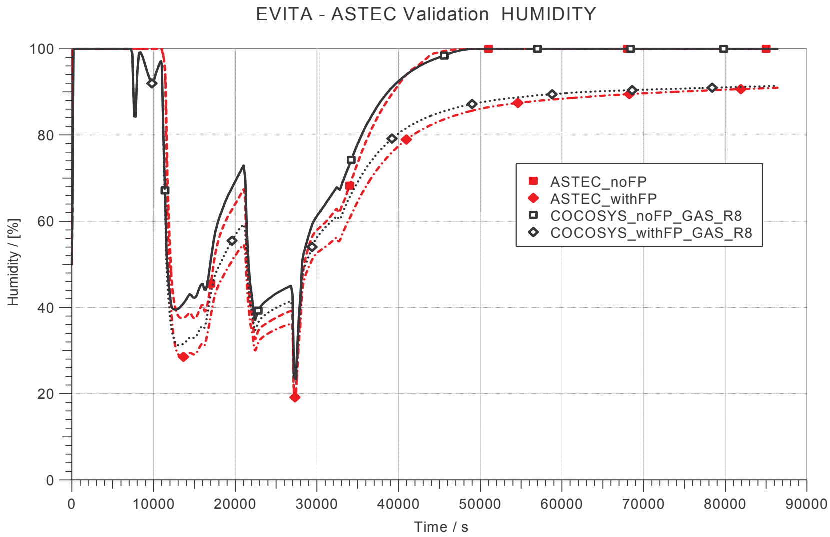


Fig. 37: Humidity in R8 - ASTEC-COCOSYS comparison

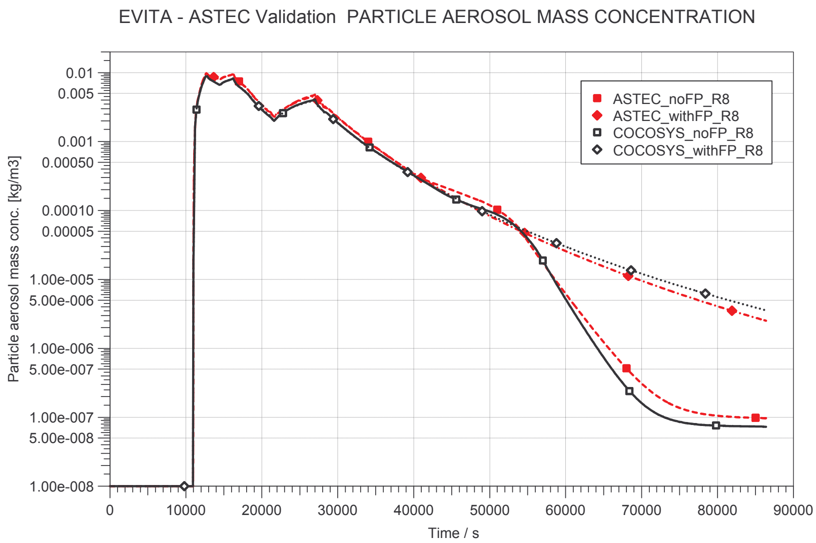


Fig. 38: Aerosol concentration in R8 - ASTEC-COCOSYS comparison

This 2"-SBLOCA calculation included the in-vessel and ex-vessel phase (steam and aerosol release) for long term (1 day) on the basis of a coarse 14 zones containment mode. The steam, water and hydrogen masses and referring energy releases into the containment during the in-vessel phase were taken from a MAAP 4 calculation.

The comparison of the ASTEC module CPA with reference to the code COCOSYS led to the following conclusions:

- Concerning pressure, temperature of atmosphere (Fig. 36) and sump, humidity (Fig. 37), sump level and fog mass, the results for comparable input data were close all over the problem time.
- The global gas flow field in the containment was almost similar.
- Without consideration of the volatile FP decay heat release, CPA and COCOSYS results are slightly different: in ASTEC, lower final containment pressure by 0.25 bar and temperature values by 3-4 °C. These differences have to be understood as they may influence the long-term containment phase.
- The analysis with CPA and COCOSYS underlined the strong influence of volatile FP decay heat release upon the long-term pressure and temperatures (more than 1bar higher when accounting for this decay heat). This underlines also the need in future studies to activate the FIPHOST models for fission product transport in CPA and to couple CPA with the respective ASTEC module (ISODOP) that evaluates decay heat in different reactor zones.
- Concerning aerosol deposition, the so far obtained acceptable accordance of all cases got lost after the time of 30.000 s. Then, significant differences on aerosol deposition were observed between the two codes: constant values with ASTEC and decreasing values with COCOSYS. The main reason seems to be the absence of a model for washing of aerosol deposits on wall by spray water and condensate in the used ASTEC version.
- Concerning the aerosol mass concentration, the values and tendencies were in good agreement (Fig. 38).

6.5 WWER-1000/320

VVER-1000 S1B and TMLB sequences analysis with ASTEC V1.0 and MELCOR 1.8.5 codes

For the Russian WWER-1000/230 type, two sequences were calculated:

- Small break LOCA in the cold leg (70 mm) without emergency core cooling,
- Station blackout.

The ASTEC V1 modules CESAR, DIVA (incl. ELSA), SOPHAEROS and CPA were used in a coupled mode and the results were compared to MELCOR 1.8.5 results.

<u>In SBLOCA case</u>, the ASTEC calculation stopped at 17.800 s without reaching complete core uncovery (reached in MELCOR at about 15.000 s).

The small and slow injection from hydro-accumulators in ASTEC was found to be the main reason for large differences between the ASTEC and MELCOR timing of events: cladding oxidation started in MELCOR about 3 h later than in ASTEC. The injection in ASTEC was much slower than in MELCOR and moreover the injected water could not reach the core while oscillations of the water level above the core mid-plane were observed in MELCOR due to oscillating accumulator injection. But using isothermal expansion of gas in the accumulators in ASTEC, the deviations of results between both codes for the initial period were acceptable.

The primary and secondary pressure histories were similar for both codes (Fig.39).

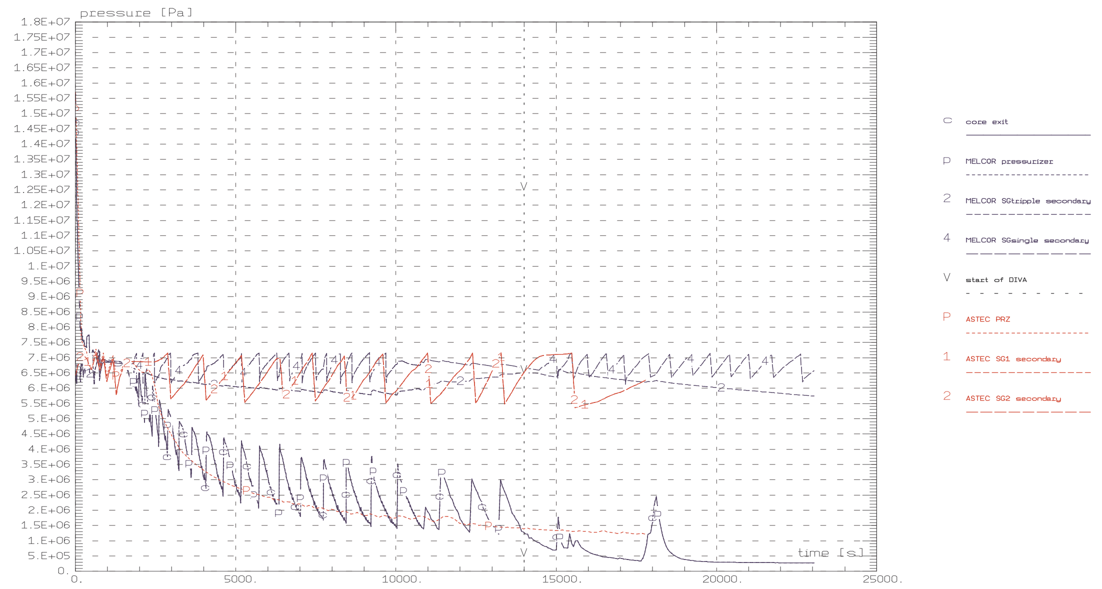


Fig.39

VVER-1000/320 Balakovo ... S1B sequence

Pressure

ASTEC V1.0_p1
MELCOR 1.8.5

The agreement on containment pressure was good between both codes (Fig.40).

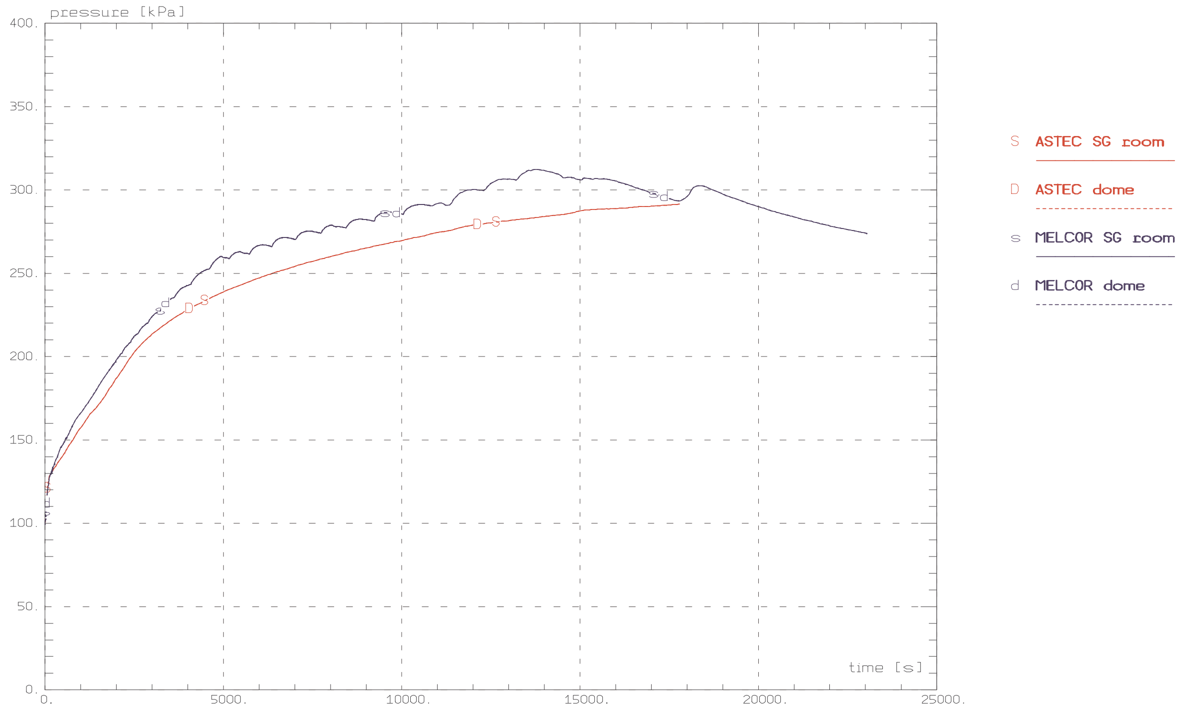


Fig.40

VVER-1000/320 Balakovo ... S1B sequence

Containment pressure

ASTEC V1.0_p1
MELCOR 1.8.5

<u>For the station blackout sequence</u>, the core was completely uncovered in ASTEC calculation at 8.800 s, in MELCOR one at 7.300 s. Though the primary and secondary pressures were similarly calculated with both codes, core uncovery was reached faster in MELCOR due to higher water level and mass in the pressurizer that led to an earlier loss of water. The agreement in fission product release from fuel was good between the codes except for shift in ASTEC to later time connected with the delay of the whole heat-up phase.

These results were very preliminary as for the core degradation phase some problems occur in DIVA preventing to reach reactor pressure vessel failure and in SOPHAEROS (problems of geometry generation).

6.6 WWER-440

For the Russian type WWER-440/V213, three sequences were calculated with ASTEC V1:

- Station blackout,
- Small break LOCA (100 mm) in the cold leg without emergency core cooling injection and without spray operation,
- Medium break LOCA (100mm) in the cold leg but focused on the containment behaviour only.

The station blackout sequence was calculated by 3 EVITA partners: IVS on the front-end thermal hydraulics only using CESAR, VUJE and UJD calculated until reactor pressure vessel failure with CESAR/DIVA, one of them without containment thermal hydraulics, the other with containment thermal hydraulics using CPA. These ASTEC V1 results were compared with RELAP5-3D, MAAP4/VVER and MELCOR ones.

Analysis of VVER-440/V213 station blackout with ASTEC V1

The 3 modules CESAR, DIVA and CPA were used. The Fig.41 and 42 present respectively the selected nodalisation of the vessel and the loop N°1 with pressurizer.

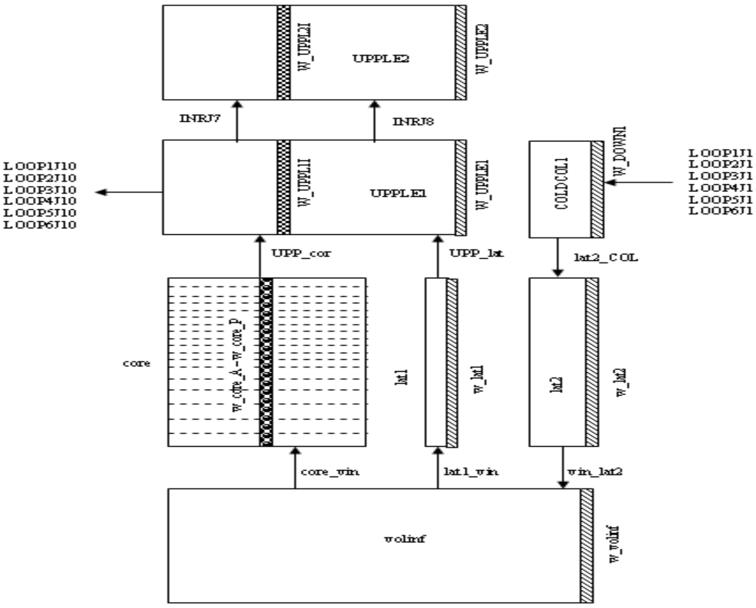


Fig. 41: CESAR nodalisation of the VVER-440/V213 reactor vessel

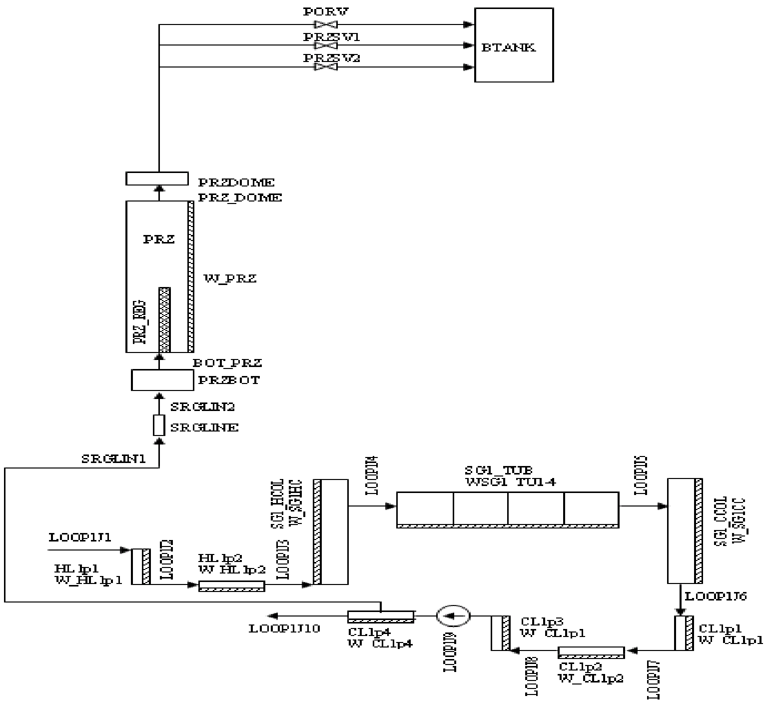


Fig. 42: CESAR nodalisation of the VVER-440/V213 loop N°1 with pressurizer

Comparison of the ASTEC V1.0 results with the results obtained by RELAP5-3D (front-end phase only) and MAAP4-VVER codes (till RPV failure) is given in Table 7. In general, the course of the main parameters during the front-end phase of the accident agrees well with the prediction of RELAP5-3D code. From the qualitative point of view, overall behaviour of the primary and secondary system was predicted similarly by both codes. However, regarding the time margin to core damage, the RELAP5 prediction is slightly more optimistic. This may be caused e.g. due to use in RELAP5 of "refined" SG model, consisting of 17 layers of SG tubes.

The MAAP4/VVER prediction of time margin to core damage is more pessimistic. This may be caused by more conservative decay curve and by simplified front-end thermal-hydraulics used in MAAP code. The natural circulation counter-current flow-path for hot gases in the hot leg and SG tubes was deactivated in MAAP calculation, since it is believed that the hot leg loop seal geometry of VVER-440 would prevent the gas circulation.

In spite of the absence of counter-current gas flows in the hot leg, MAAP4/VVER predicts that the hot leg between the vessel and surge line is heated sufficiently prior to vessel failure so that creep failure could occur. The gas flow from the core to the pressurizer induced by cyclic opening of PRZ relief valve is sufficient to heat the piping significantly, and could lead to induced rupture. This would alter significantly the subsequent accident progression.

Special attention was paid on hydrogen production during the core degradation phase and hydrogen release into containment. The hydrogen production started in ASTEC at about 32.400 s compared to 34.550 s and 28.450 s calculated with RELAP and MAAP, respectively. Regarding the H_2 production, MAAP predicted about 360 kg of hydrogen before core support plate failure. ASTEC prediction at time of the code failure was only slightly higher, about 390 kg of hydrogen.

Table 7: Comparison of RELAP5-3D, MAAP4-VVER and ASTEC V1.0 results of SBO on a VVER-440/V213

Event	Timing [h]		
	RELAP5-3D	MAAP4-VVER	ASTEC V1.0
First opening of PRZ relieve valve	3,69	4,41	5,49
Break of membrane in quench tank	6,06	5,14	6,78
SG dryout (SG water level < 0,2m)	~ 8,4	~ 4,4	4,5 – 5,0
Start of core uncovery *	~ 9,2	~ 6,6	~ 7,6
Start of core heat-up	~ 9,5	~ 6,7	~ 7,9
Start of hydrogen generation	~ 9,6	~ 7,9	~ 9,0
Hot leg creep failure**	-	~ 9,2	-
Core support plate failure	-	10,4	-
Vessel failure	-	10,8	-

*Collapsed water level in reactor vessel < 7m.

Analysis of the Transient Blackout sequence with CESAR1.0 module (ASTEC V1) for VVER-440/V213 Unit and comparison with the codes RELAP5-3D & MAAP4/VVER

As for the comparison with RELAP5-3D, the first analysis of the calculations showed that, from the qualitative point of view, the overall behaviour of the reactor coolant system was predicted similarly by CESAR and RELAP5-3D. The slightly sooner start of core heat-up in CESAR calculation in comparison with RELAP was mainly a consequence of the very detailed nodalization scheme of the lower part of the steam generator tubes that is used in RELAP. It is expected that a more detailed CESAR nodalization in vertical direction of the steam generator tubes would lead to a still better agreement with RELAP.

The following 4 cases of SG CESAR nodalisation have been tested:

- Case-A: SG tubes modelled only as 1 volume subdivided into 4 cells - ”axial nodes” in horizontal direction,
- Case-B: SG tubes modelled by 3 volumes in vertical direction subdivided into 4 cells in horizontal direction.
- Case-C: 3 levels in vertical direction, each one containing 4 volumes in horizontal direction (12 control volumes together).
- Case-D: SG tubes modelled by 3 volumes in vertical direction without dividing into ”cells” in horizontal direction.

The nodalization of the horizontal SG tubes in vertical direction (Case-B, -C & -D), in comparison with Case-A (1 volume), significantly influences RCS response (pressure evolution, timing of 1st opening of PRZ RV and total uncovery of the SG tubes) only during the beginning phase when there is water on secondary side of SG. Timing of later events (start of core uncovery & start of core heat-up) is very similar.

The model Case-B has the best agreement with reference code RELAP5-3D prediction from the point of view of crucial parameters evolution (pressure, temperature, speed of secondary water evaporation...) and timing of main events (see Table 8 & Fig.43). The ”axial modules” are used in Cases-B (in Cases-C, -D not). Such arrangement enables a more complete consideration of transport phenomena, and in particular, transport of linear momentum.

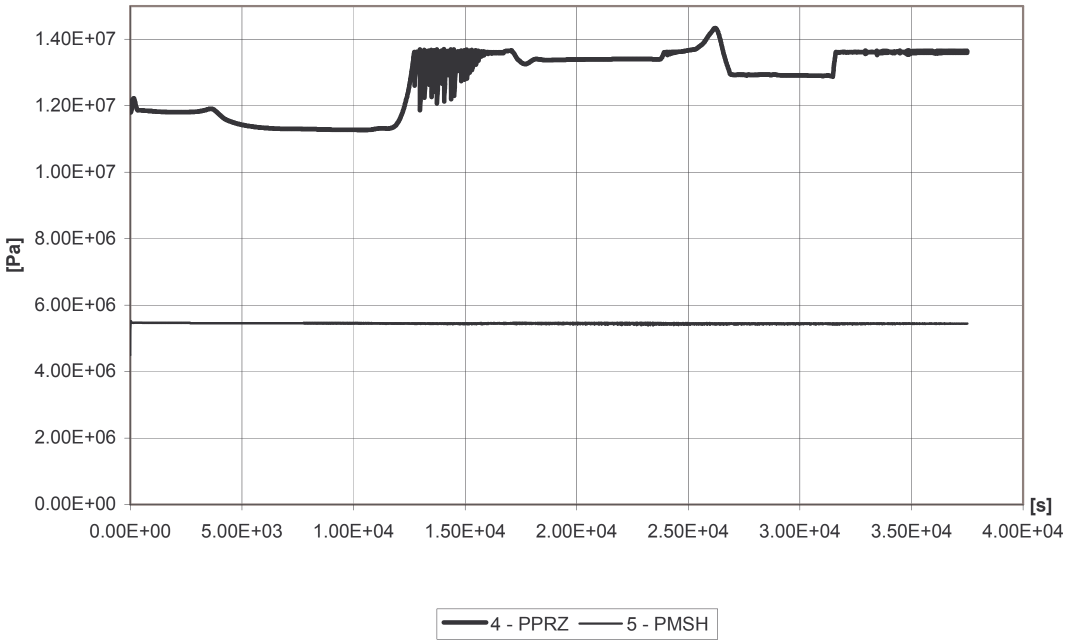
The start of core heat–up in CESAR calculations for the different cases A to D is ≈5.000s sooner in comparison with RELAP5. This time shift (speeding-up) is a consequence of ”rough” nodalization of lower part of SG tubes (2 volumes) in comparison with very detailed RELAP5 nodalization

From the qualitative point of view, overall behaviour of the RCS was predicted similarly by both codes ASTEC and RELAP5-3D. The slightly sooner start of core heat-up in CESAR calculation in comparison with RELAP5 is mainly a consequence of the very detailed nodalization scheme of lower part of SG tubes which is used in RELAP5 model. It can be expected that a more detailed CESAR nodalization (5-6 levels) of SG tubes in vertical direction would lead to a still better agreement with RELAP5.

Finally, the simpler model of SG tubes (Case-D, decreasing of CPU time) can be recommended for analysis of medium & large break LOCAs where the secondary side response is not so important: SG tubes modelled by 3 volumes in vertical direction without dividing into ”cells” in horizontal direction.

Table 8: Primary and secondary circuit response of VVER-440/V213 – CESAR comparison with RELAP5-3D and MAAP4

Event	**CESAR Case B**	**CESAR Case D**	**RELAP5-3D**	**MAAP4**
Reactor trip	0 s	0 s	0 s	0 s
Minimum PRZ pressure after reactor scram (Corresponding time)	11,3MPa (11670s)	11,05MPa (12250s)	11,5MPa (3810s)	11,2MPa (10550s)
1st opening of PRZ Relief Valve	12660s	15050s	13280s	15800s
Total uncovery of SG tubes	≈ 24900s	≈ 22900s	≈ 30200s	≈ 15500s
Start of core uncovery	≈ 25100s	≈ 25500s	≈ 33100s	≈ 23650s
Start of core walls temperature increase	29000s	29450s	34200s	23700s
Maximum core wall temperature ≥ 800°C	30100s	30400s	35950s	24750s
Total core uncovery	≈ 32900s	≈ 31500s	≈ 39000s	27400s
CPU time (PC, 450MHz)	≈ 7,0h	≈ 5,5h	> 15h	≈15min.



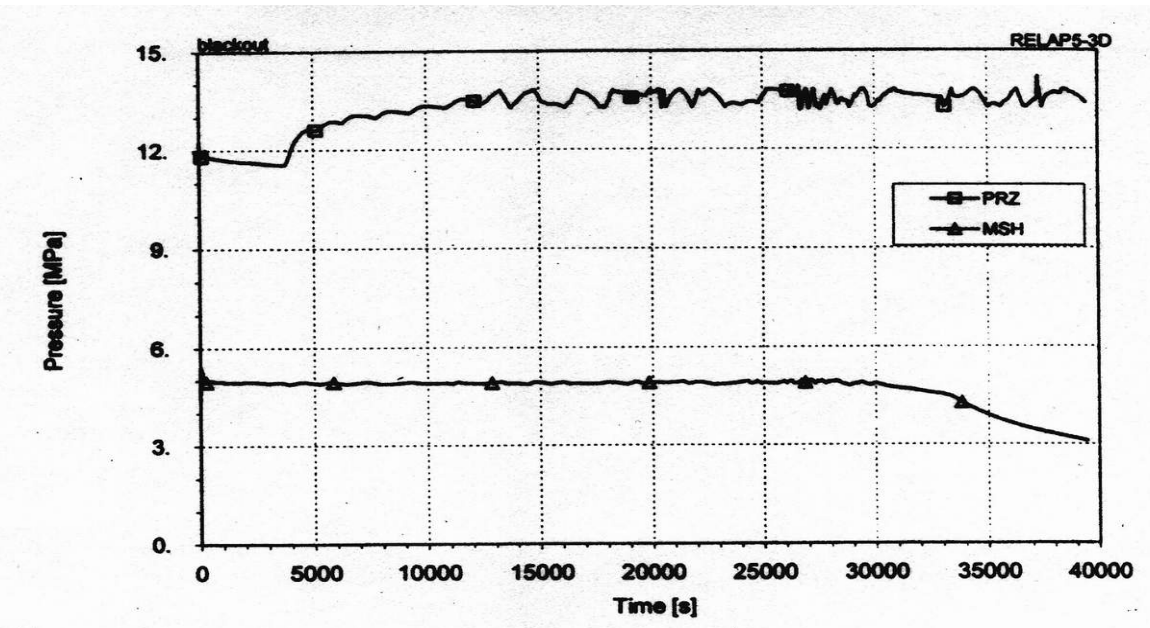


Fig. 43: Pressurizer (P-PRZ) & Secondary pressures (P-MSH) as calculated with CESAR Case-B (Figure above) and RELAP5-3D (Figure below)

ASTEC V1 application to a SBO scenario on a VVER-440/V-213 and comparison with MELCOR 1.8.3

The general behaviour of the system and the trends of calculated parameters in ASTEC (CESAR/DIVA) and MELCOR were in good agreement (see on the Fig. 44 the comparison of pressure in the pressurizer and of the water mass in the RCS). The quantitative discrepancies were mainly caused by differences in input decks (e.g. not identical initial water inventories, not identical power profiles, etc.) and different features of the individual models. For instance, the initial water inventory of the secondary circuit in ASTEC input deck was substantially higher comparing to MELCOR data. The ASTEC data corresponded to the real situation in the plant. The model of SGs is not very well prepared in MELCOR input deck and it was necessary to decrease the initial water mass on the secondary side, artificially.

Due to lower water inventory, the SGs in MELCOR analyses are depleted earlier (by about 5.000s). It leads to earlier heat up and pressurization of the primary circuit followed by earlier cycling of the PORV and loss of primary system inventory. The first core uncovery occurs and although the operator action allows injection from HAs, the rate of re-flood is not sufficient to prevent some cladding oxidation followed by hydrogen production in MELCOR.

After the injection of cold water from HAs, the primary system is cooled down for more than 3h in both analyses. Later, continuing decay heat generation heats up the water in the core, which is vaporised leading to permanent core uncovery. After permanent core uncovery, the fuel rods in the upper regions of the core are heated up and oxidation of cladding starts.

About 1h later, the ASTEC calculation failed due to numerical problems. Despite of various input deck modifications (e.g. decreasing the minimum time step in DIVA module, etc.), it was impossible to prolong the calculation.

Some strange responses of some ASTEC models were observed. The value of primary pressure was influenced by the number of hydro-accumulators defined in the scenario even before they start to inject water to the primary system. Correctness of the valve operation defined through hysteresis functions (i.e. opening and closing pressures are defined) seemed questionable because maximum and minimum zone pressures regulated by such valves seem to be oscillating within a range higher than defined by the hysteresis function.

As to H_2 production, by the end of ASTEC calculation, about 153kg of H_2 is produced. The amount of H_2 produced in MELCOR analyses by this time is very similar.

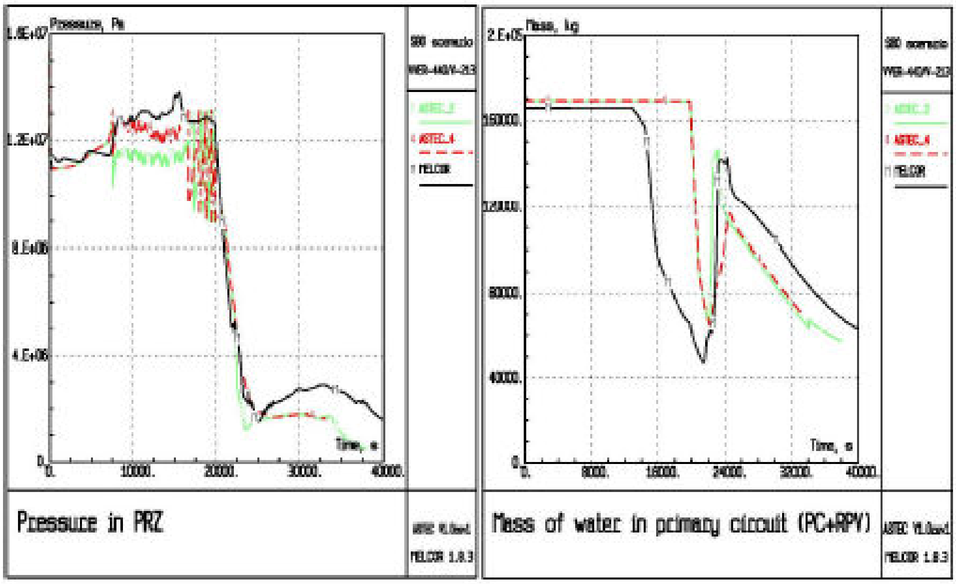


Fig. 44: Comparison of pressure in the pressurizer (left) and of the water mass in the RCS (right) on a SBO for VVER-440/V-213 between ASTEC and MELCOR 1.8.3

VVER-440/V-213 calculation for 100mm CL break / Stand-alone CPA calculation and comparison with MELCOR

In addition, a comparison of CPA module stand-alone application to the containment behaviour in a MBLOCA sequence was done with MELCOR 1.8.5 results. The CPA blowdown source input was based on the MELCOR primary system thermal hydraulics calculation. The containment models of both codes were identical, with the exception of the model of the bubbler condenser trays (detailed DRASYS model in CPA and a flow path solution with SPARC model in MELCOR).

From the performed analysis, the following conclusions could be drawn:

- The DRASYS bubbler condenser model condenses steam more effectively than the MELCOR model.
- The results showed good agreement between both codes in pressure (Fig.45) and temperatures (Fig.46), however there is a balance of differences in heat transfer to walls and on pool surfaces and in the bubbler condenser model.
- Water condensed on walls seemed to be missing from the mass balance in CPA.

A much higher CPU time was needed for the stand-alone CPA calculation with the full input and for 33.800 s sequence time than MELCOR one which was very rapid (about 3 min). The main reason for this difference was the more detailed bubbler condenser model and the limiting mass flow through certain junctions in CPA, where the time step was reduced. Some investigations should continue in the future on these CPU aspects.

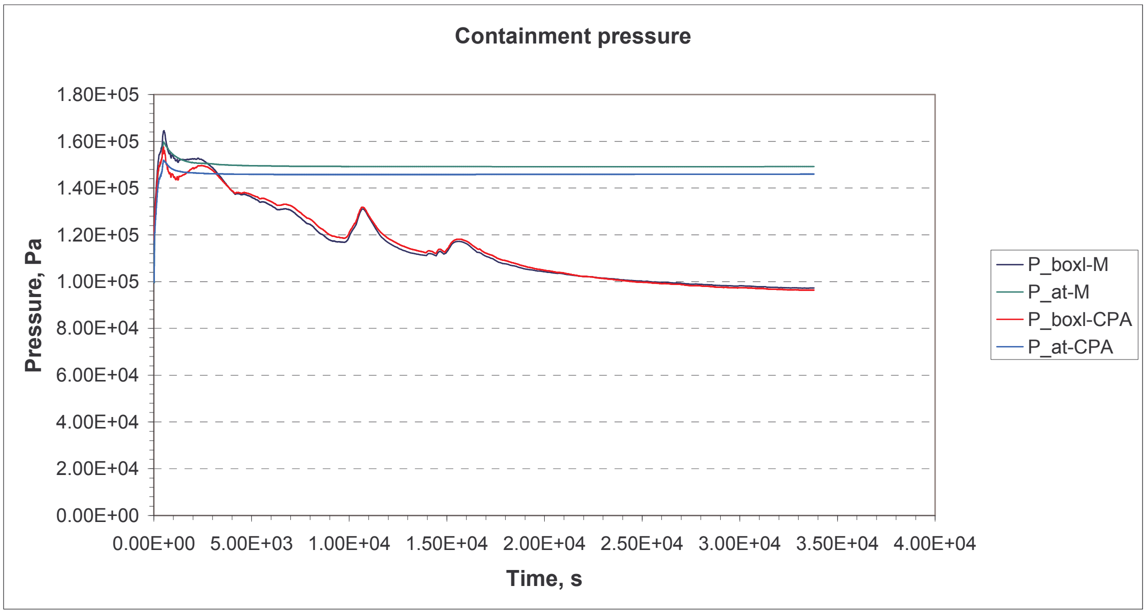


Fig.45: Comparison of containment pressure between CPA and MELCOR (in zones SG box and air trap)

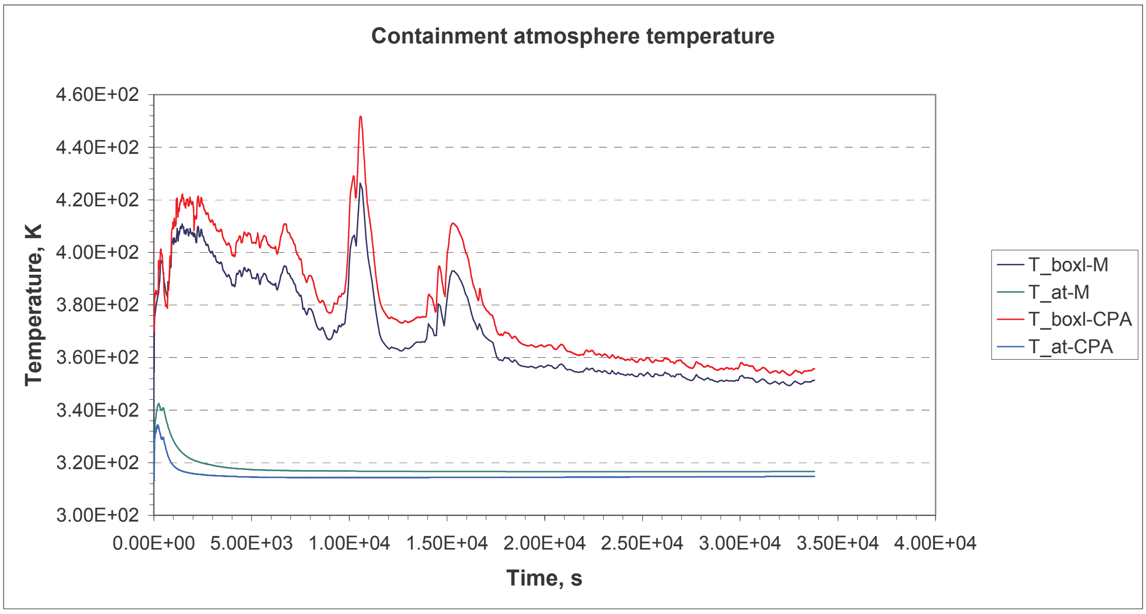


Fig.46: Comparison of containment atmosphere temperatures between CPA and MELCOR (in zones SG box and air trap)

ASTEC V1 application to a SBLOCA scenario on a VVER-440/V-230 and comparison with MELCOR 1.8.3

For the Russian WWER-440/V230 type, a small break (20 mm diameter in cold leg) LOCA sequence was calculated with the ASTEC modules CESAR and DIVA. The results were compared with MELCOR 1.8.3 ones. The ASTEC calculation did not reach total core uncovery as the minimum time step was reached in DIVA. Due to the short calculation time, one can only say that the pressure in the pressurizer as also the mass of water in the primary circuit agreed quite well (Fig. 47). In contrary pressure and water mass in the steam generator showed larger deviations.

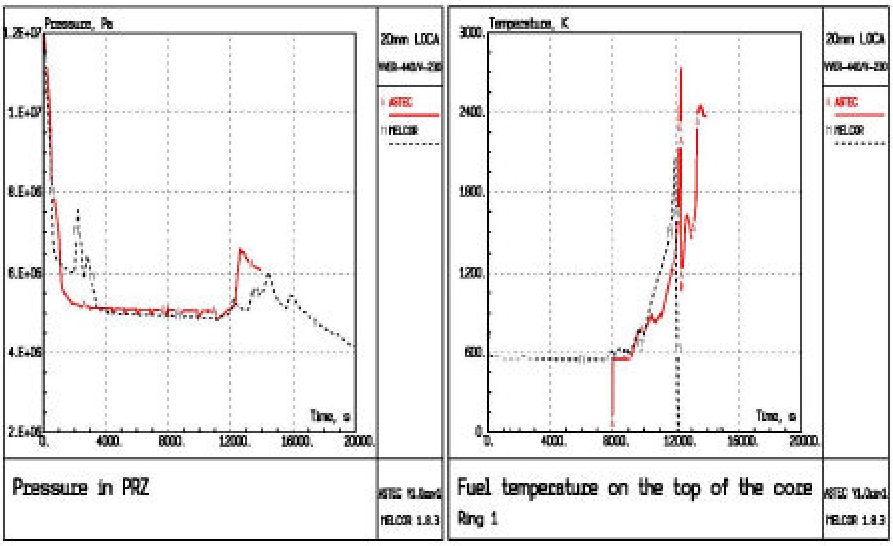


Fig. 47: Comparison of pressure in the pressurizer (left) and of the water mass in the RCS (right) on a SBLOCA for VVER-440/V-230 between ASTEC and MELCOR 1.8.3

7 WP6: Conclusions, further recommendations and benefit

One of the partners' conclusions was that the level of ASTEC models was near the state-of-the-art in most domains. Of course understanding and thus adequate modelling is still missing - like in all other codes - in some domains: reflooding of a degraded core, long-term molten core concrete interaction (MCCI), iodine behaviour in containment, etc. And obviously recommendations were made to continue efforts on validation in specific domains and on plant applications. The new version V1 allows simulating complete scenarios including the front-end-phase. Both developing organisations, GRS and IRSN, which will continue to assure the code maintenance beyond EVITA, ensure that the EVITA outcomes will be respected for the future ASTEC development. Some of the above mentioned improvements are already foreseen, as well as the extension to Boiling Water Reactors.

Some specific needs of new models or model improvements were underlined along with the plant applications: for instance new pump models, improvements of hydro-accumulator models, and implementation of core support plate models.

Further recommendations are given by the EVITA partners to improve:

- Management of input decks: improved tools for automatic check, standard input decks for generic plant applications on each type of reactor, consistency of input data (especially for CESAR-DIVA modules),
- Visualization / post-processing (remark: the GRS visualization tool ATLAS will be coupled to the next version),
    - Possibility to extract plots at the end of calculation without restarting the whole calculation (this is possible with MELCOR and MAAP4 codes),
    - Necessity of the possibility to decouple the on-line visualisation from the calculations.
- Users' support, including:
    - Training courses not only for beginners but also for experienced users, either focused on specific parts of the accident or to specific modelling,
    - Wish for faster response of the ASTEC maintenance team,
    - More complete and detailed code documentation: more detailed input/output data description, examples of inputs, users guidelines with recommended default values and range of model parameters,
    - Need for Users' guidelines in the future.
    - Organisation of a widespread Users' Club in the future, favouring closer links between users and dissemination of news about the anomalies that are met, which may enable solution of “simple” problems by users without discussion with the maintenance team.

The plant applications with the first version of ASTEC V1 showed that the code is still not so robust that a sequence could be calculated up to its normal end. Only 4 sequences reached the vessel lower head failure, 2 performed by IRSN and GRS and the 2 others on Westinghouse 1000 by EA and CSN. There were strong requests to increase the robustness of the coupling of the two new modules CESAR-DIVA that calculate the circuit thermal hydraulics and the core degradation.

Discussions took place about the difficulty of the benchmarking exercises on plant applications:

- Importance (and difficulty) to define quasi-identical input decks for the different codes to represent the reactor characteristics,
- Propagation of discrepancies when applied to an integral experiment (it was illustrated in the ISP46 on Phébus FPT1 experiment) or to a plant application: for instance, the front-end and thus the initial thermal hydraulic conditions in the reactor coolant system before core degradation should be close enough before comparing precisely the core degradation results. And so on…
- Interest for future benchmarks of a stepwise approach, starting with as simple as possible accident scenarios,
- Interest for future benchmarks of comparison between modules of integral codes using fixed initial and boundary conditions in reactor-cases.

A proposal of methodology for comparison code-to-code was identified (but not applied due to lack of time): identify for each code the dominant model parameters and their variation range, define the calculation matrices (necessary to limit the number of calculations…); identify the variables to be compared (and time periods), perform sensitivity calculations and post-processing of the common variables, analysis of discrepancies and variations, and synthesis. The main difficulty is that models parameters are code-dependent. Another solution could be to identify a set of user-parameters that lead for instance to the maximum or the minimum hydrogen production, or to the fastest core degradation or the slowest one.

With respect to computing time, the EVITA users - researchers, licensing authorities and industry - accepted the definition elaborated in the VASA project as target, that a full sequence calculation (incl. post-processing) should not need more than 12 h. ASTEC performance is in most applications around this value, even if some improvements must still be done to increase the calculation speed.

The progress made since the beginning of the project, where only a preliminary code version was available, became evident. The actual version allows in principle to simulate the entire sequence of events during a severe accident. Besides, a first level of validation was attained successfully within the project. The next priorities are now clearly identified: code consolidation in order to increase the robustness, extension of all plant applications beyond the vessel lower head failure and coupling with fission product modules, and continuous improvements of users' tools.

As EVITA has very successfully made the first step into the intention to provide end-users (like utilities, vendors and licensing authorities) with a well validated European integral code for the simulation of severe accidents in NPPs, the EVITA partners strongly recommend to continue validation, benchmarking and application of ASTEC. This work is continued in SARNET (Severe Accident Research Network) in the 6th Framework Programme, where ASTEC plays the key role as the European reference integral code.