SARNET

Network of Excellence for a Sustainable Integration of European Research on Severe Accident Phenomenology

SIXTH FRAMEWORK PROGRAMME

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

G. Bandini *(ENEA, Bologna)*, **M. Buck** *(IKE, Stuttgart)*, **W. Hering** *(FZK, Karlsruhe)*,
L. Godin-Jacqmin, G. Ratel *(CEA, Cadarache)*, **P. Matejovic, M. Barnak** *(IVS, Trnava)*,
G. Paitz *(BUTE, Budapest)*, **A. Stefanova** *(INRNE, Sofia)*,
N. Trégourès, G. Guillard *(IRSN, Cadarache)*, **V. Koundy** *(IRSN, Fontaney-aux-Roses)*

ERMSAR 2008

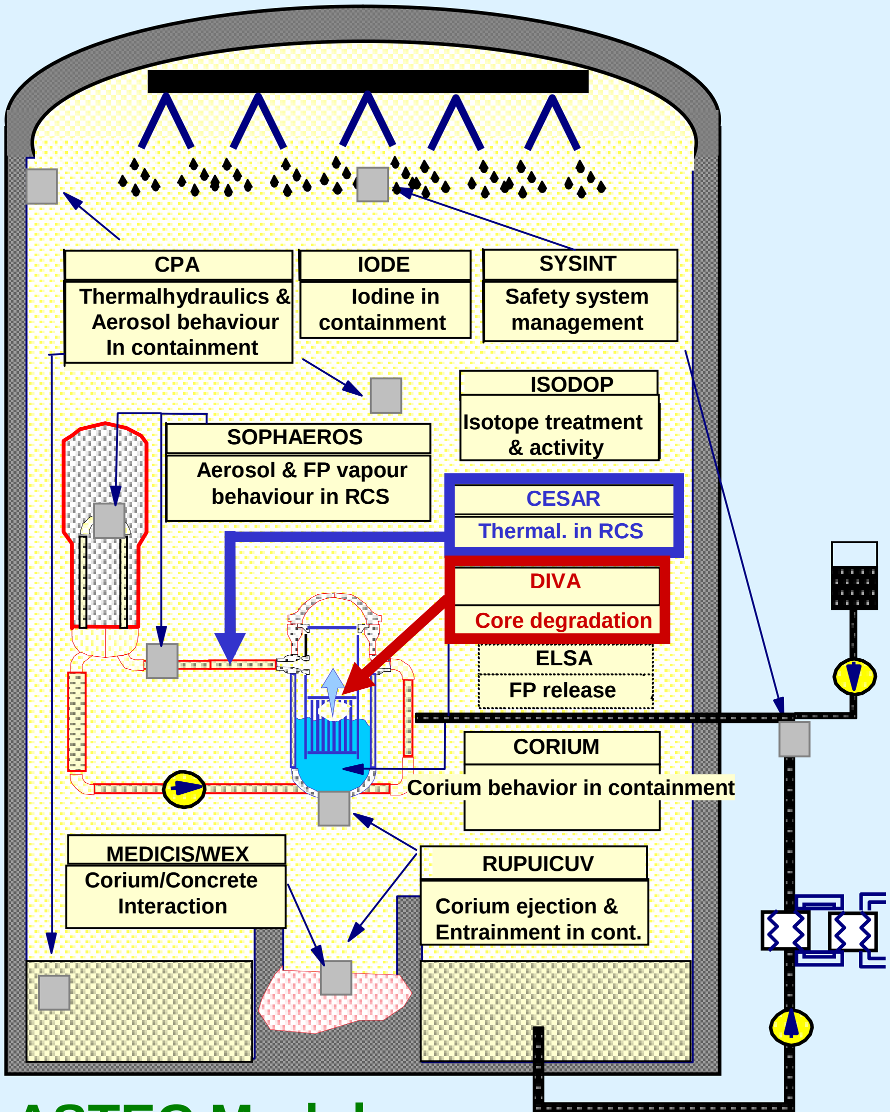
3rd European Review Meeting on Severe Accident Research

September 23 – September 25, 2008 – Nesseber (Bulgaria)

SUMMARY

- The ASTEC Code System
- CESAR and DIVA Modules Validation Tasks
  - RCS Thermal-Hydraulics (CESAR)
    - Experiments on BETHSY, PACTEL, PMK2 Facilities
  - Core Degradation (DIVA)
    - Experiments on CORA, QUENCH, Phébus, FARO, LIVE, OLHF, FOREVER Facilities
  - CESAR and DIVA Coupling
    - Experiment on LOFT Facility and TMI-2 Accident
- Conclusions

The ASTEC System Code



ASTEC Modules

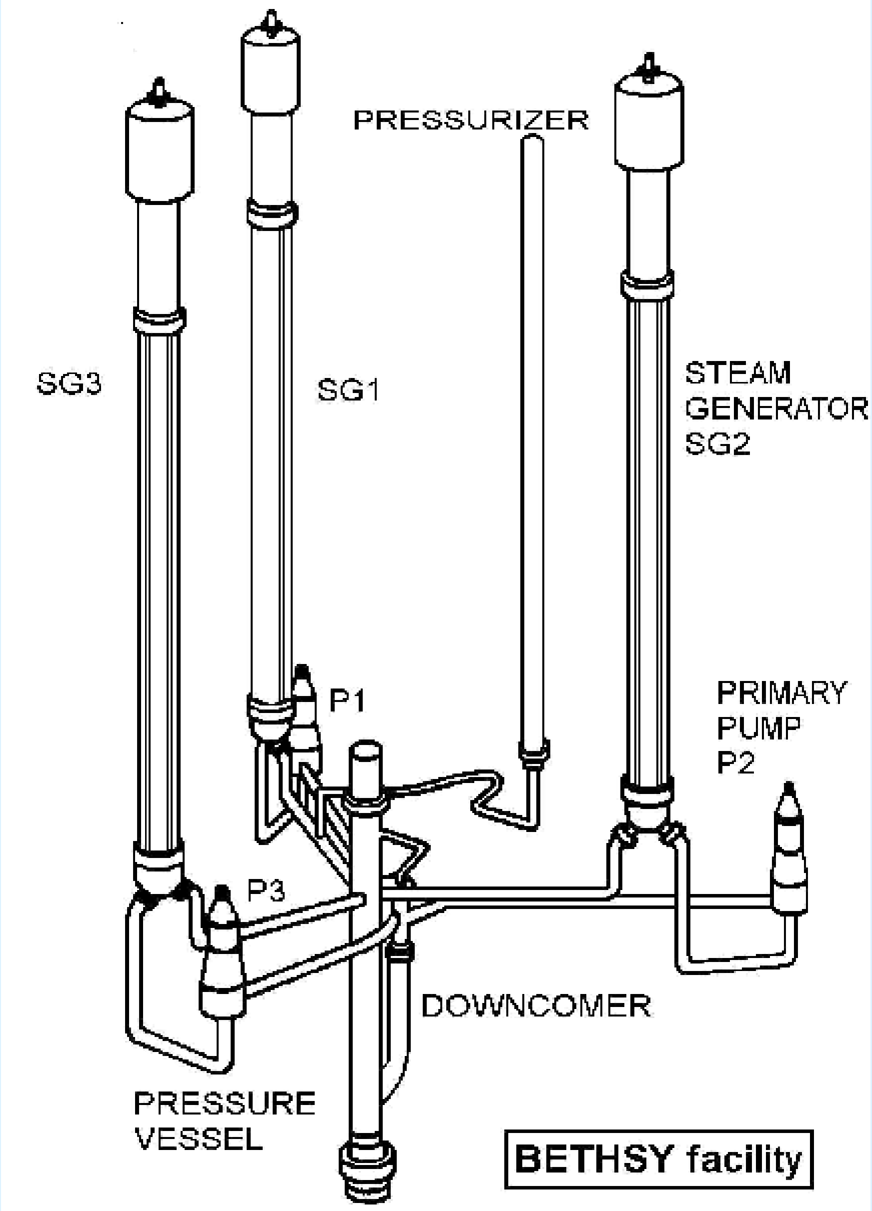
- **Reference integral code of the European Accident Research Network Project (SARNET) in the Framework Program of the European Commission**
- **Jointly developed by IRSN and GRS to address severe accident sequences in nuclear power plants**
- **Currently used by 28 European Organizations as well as the BARC of India, the AECL of Canada and the Russian Kurchatov Institute**
- **11 coupled modules dealing with different phenomena occurring during a severe accident**
- **CESAR module for reactor system thermal-hydraulics**
- **DIVA module for core degradation**

CESAR and DIVA Module Validation Tasks

- ASTEC V1.3R2 code delivered in December 2007
- Suitability and capability of new improved model implemented

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

RCS T-H: BETHSY 9.1b Experiment (1)



- The BETHSY facility at CEA/ Grenoble is a scale down model of a French 900 MWe PWR
- Overall scaling factor (power, volumes, mass flowrate) is close to 1/100
- More than 80 tests have been conducted in the BETHSY facility

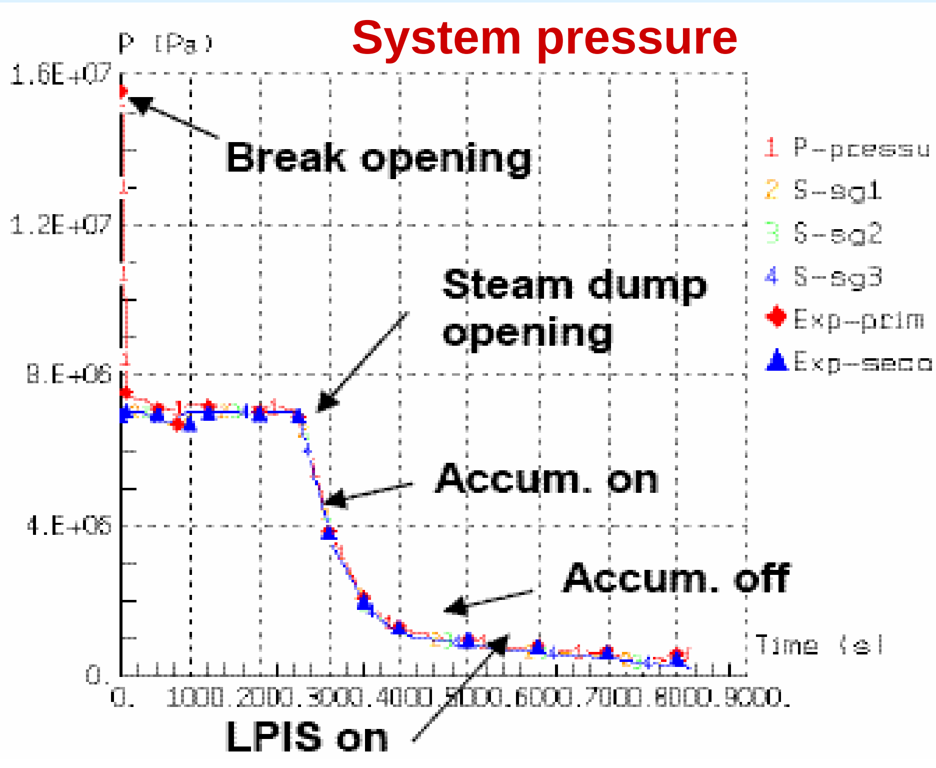
BETHSY 9.1 experiment:

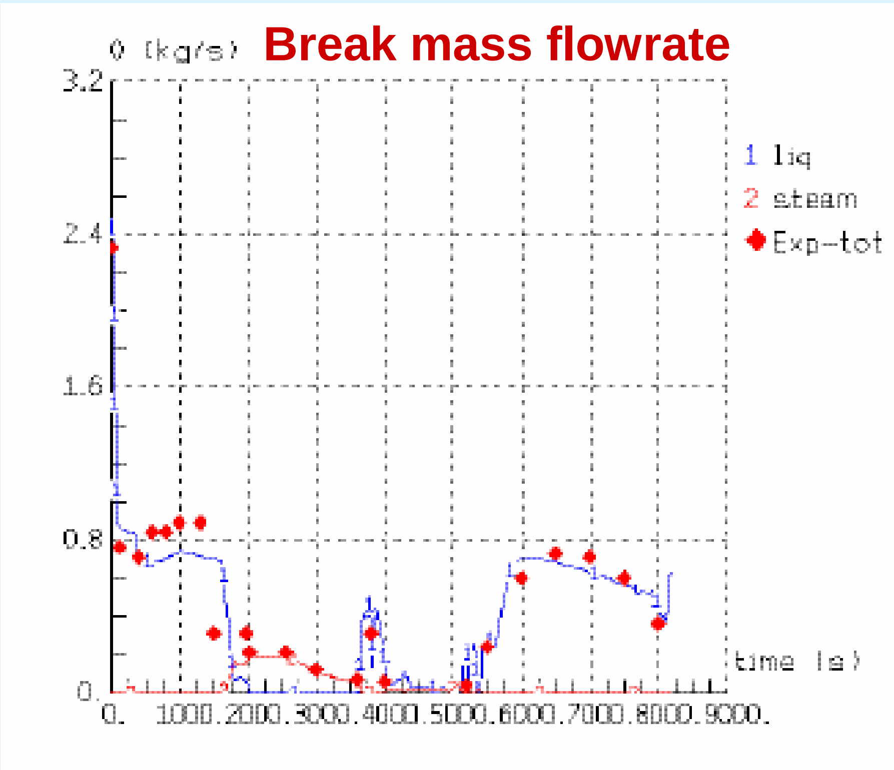
2" cold leg break with unavailability of HPIS results in:

- Primary system depressurization
- Large core uncovery and heatup
- Delayed recovery procedure by further system depressurization and accumulator and LPI system injection

RCS T-H: BETHSY 9.1b Experiment (2)

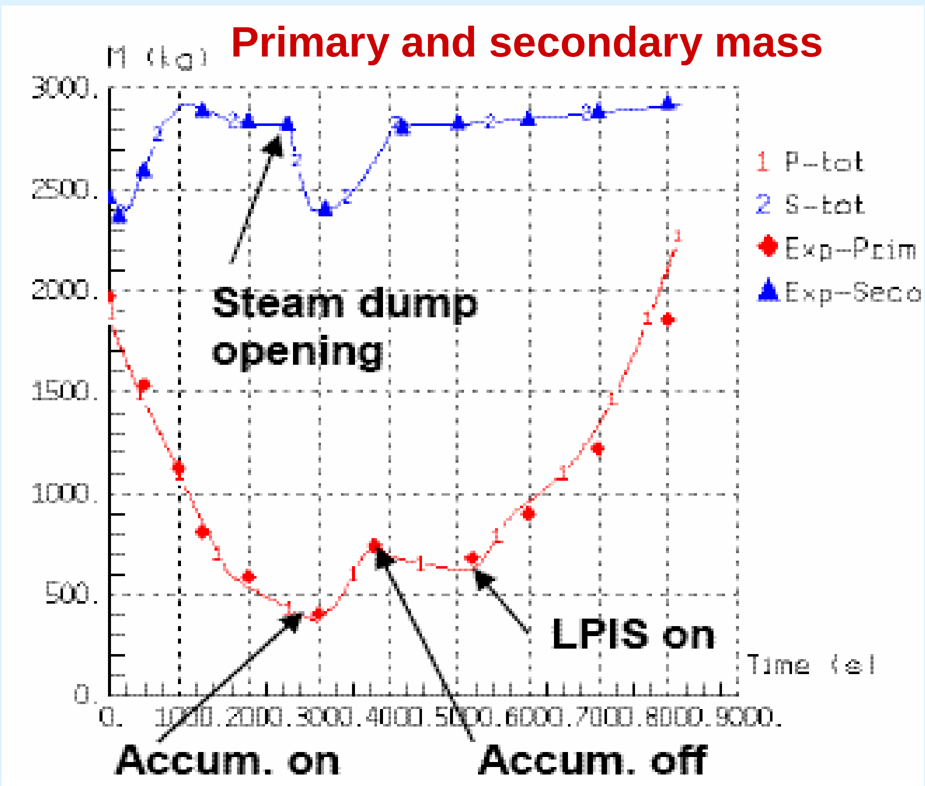
- Different BETHSY tests have been analyzed by IRSN for CESAR validation
- In Test 9.1b analysis, thermal-hydraulics of primary and secondary systems is very well simulated
- Initial primary system depressurization and depressurization after steam dump opening is well reproduced
- Time evolution of break mass flow rate during primary system draining and following safety injection system actuation is captured

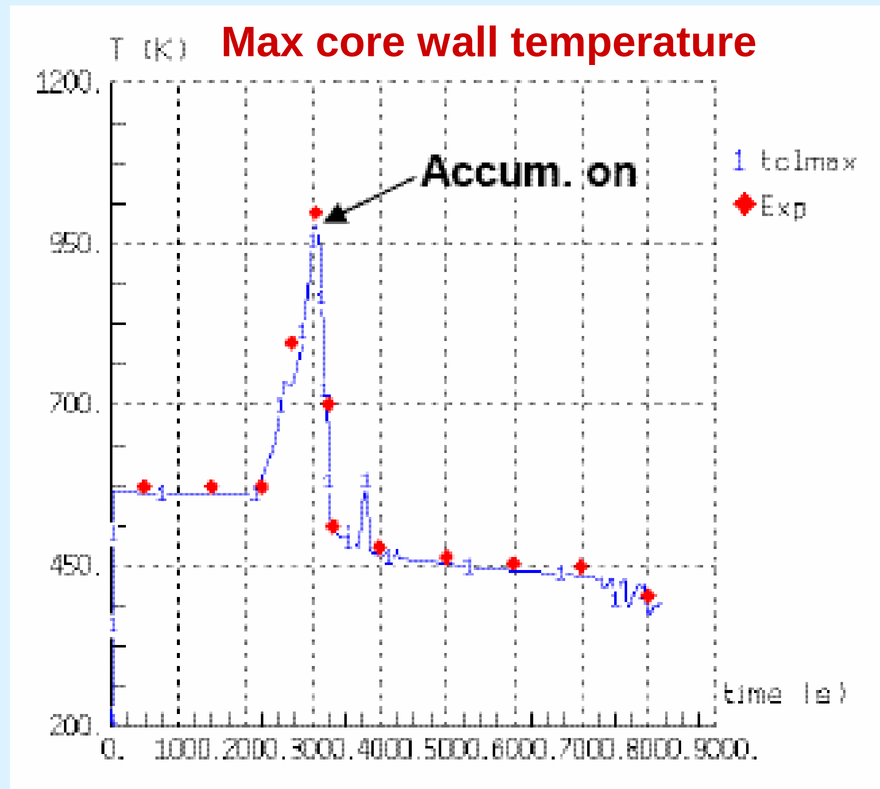




RCS T-H: BETHSY 9.1b Experiment (3)

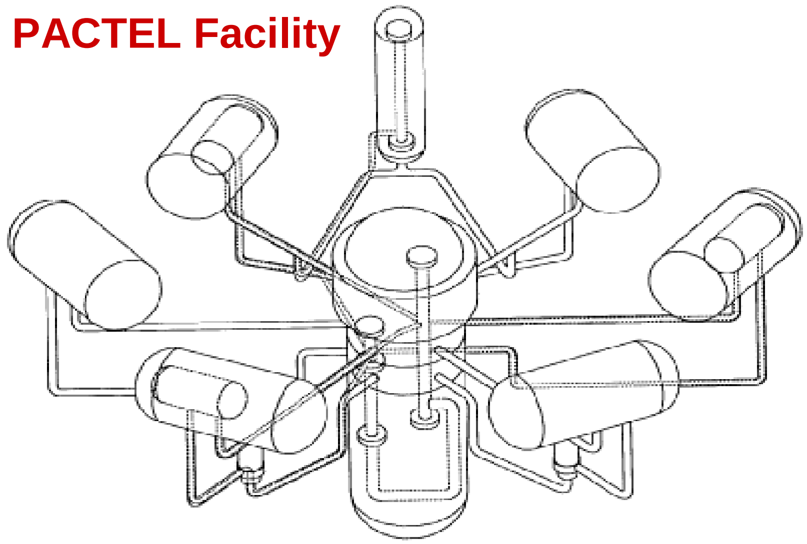
- Time evolution of primary mass inventory during primary circuit draining and refilling is well reproduced by CESAR
- Secondary mass controlled by feedwater injection fit very well the measured value
- General trend of core uncovery is well predicted à fuel rod simulator heatup and cooldown after accumulator injection is very well simulated





RCS T-H: PACTEL ISP-33 and T2.1 Tests

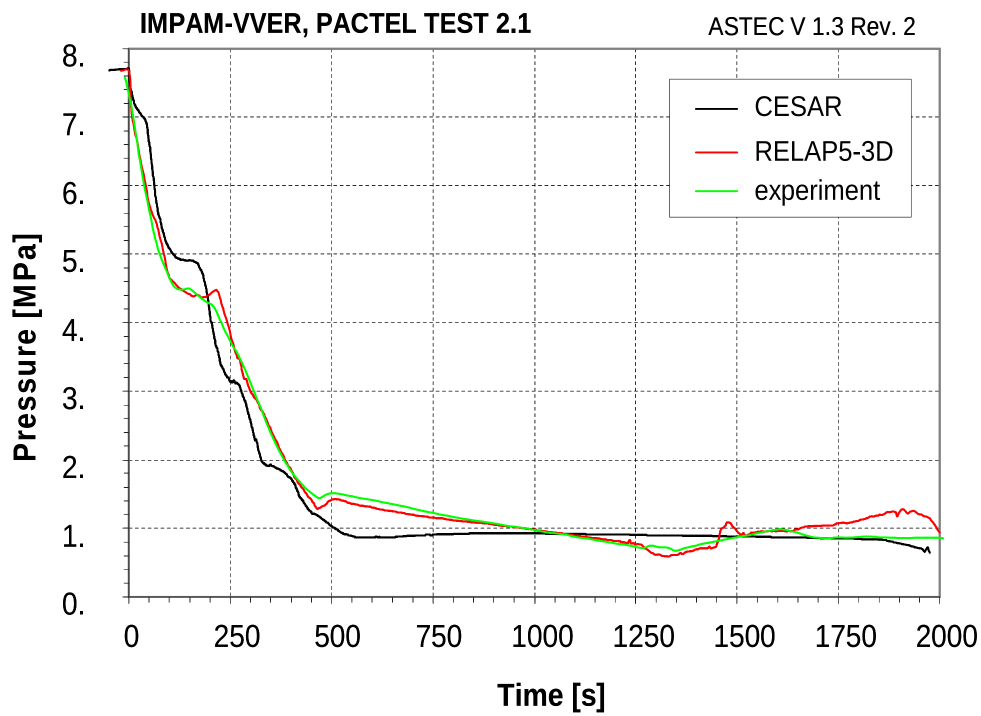
PACTEL Facility



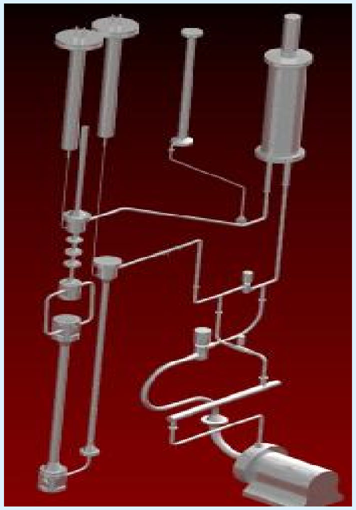
- The PACTEL Facility constructed in Finland is a scale down model (1:305) of a 6-loop VVER-440 reactor
- Designed to study thermal-hydraulics characteristics of VVER-440 reactors during LOCA's and operational transients

- Test ISP-33 and T2.1 analyzed by IVS
- Different modes of natural circulation investigated in ISP-33 experiment – main phenomena and overall system behavior well predicted by CESAR
- Test T2.1 represented a SB LOCA scenario without availability of HPI – main phenomena such as depressurization of primary system, accumulator injection and overall system behavior well predicted

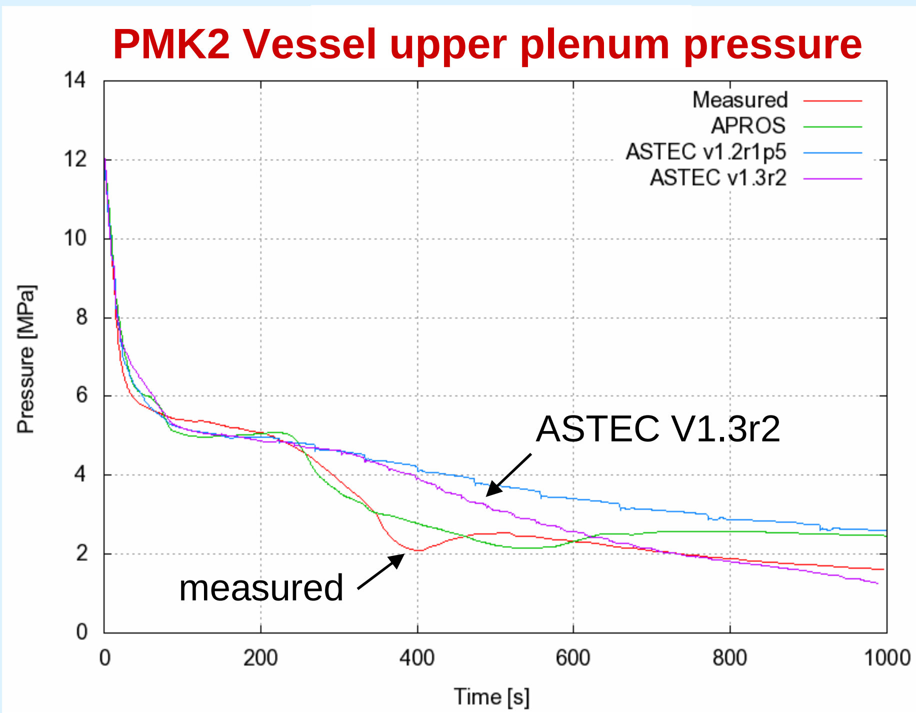
Test T2.1: Primary pressure

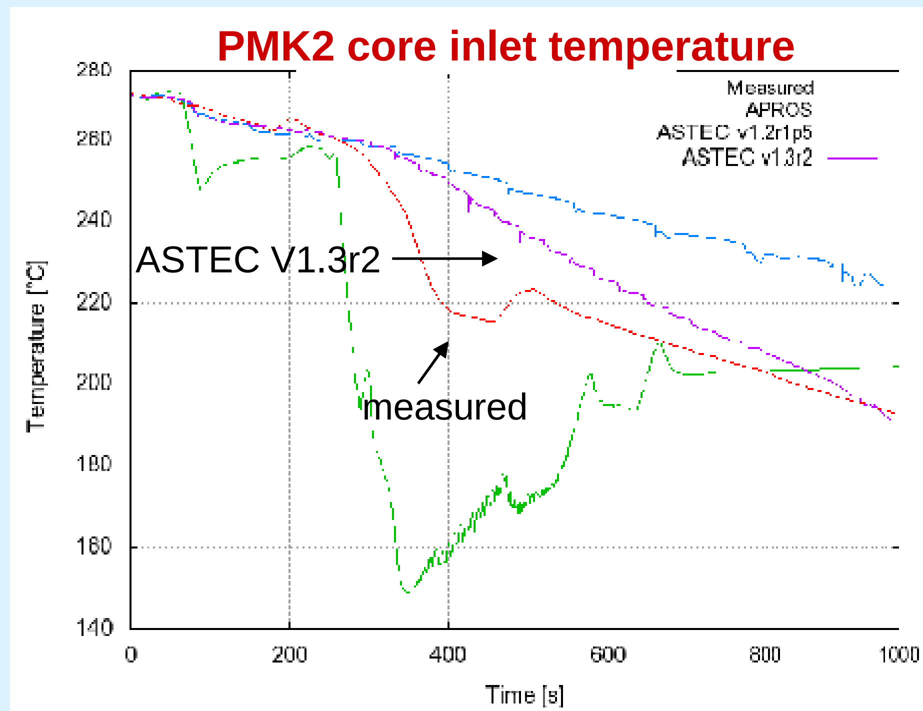


RCS T-H: PMK2 SBLOCA Experiment

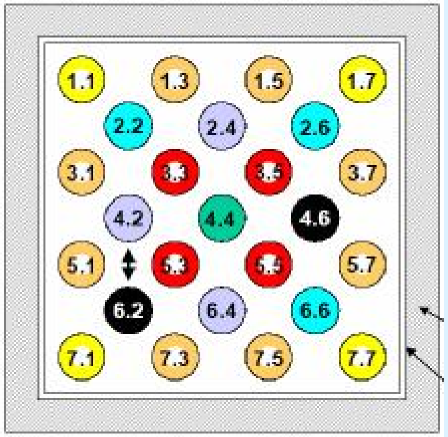


- The PMK2 facility at KFKI AEKI (Hungary) is a scale down model (1:2070) of a VVER-440 reactor
- The 3 mm SBLOCA transient with HPIS and hydro-accumulator injection was selected by BUTE for ASTEC validation
- Good agreement during initial rapid depressurization before hydro-accumulator injection
- Main deviation between results and measurements after hydro-accumulator injection



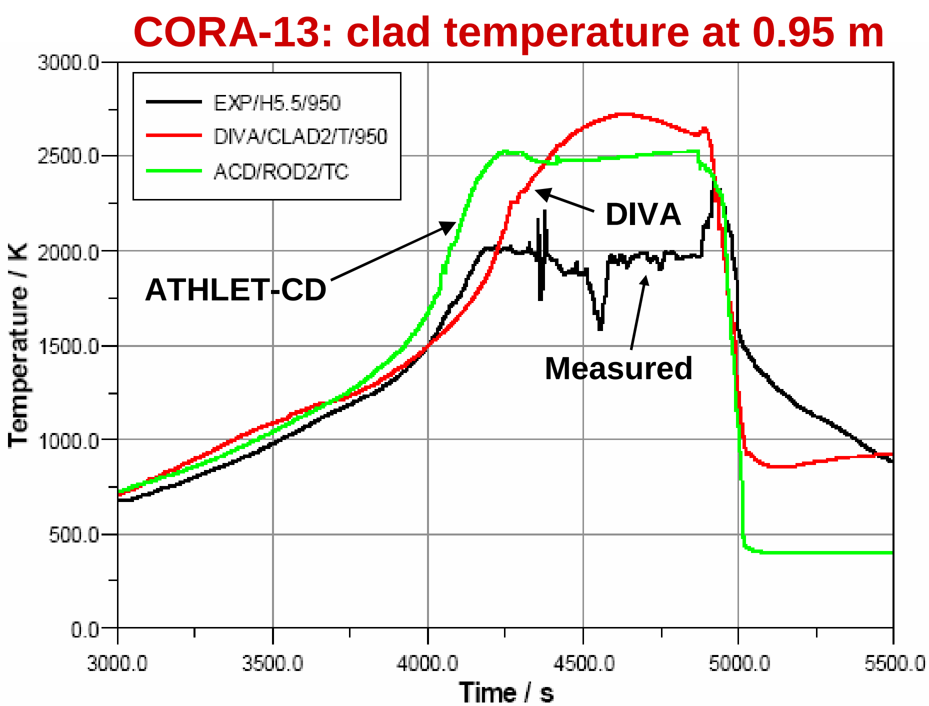


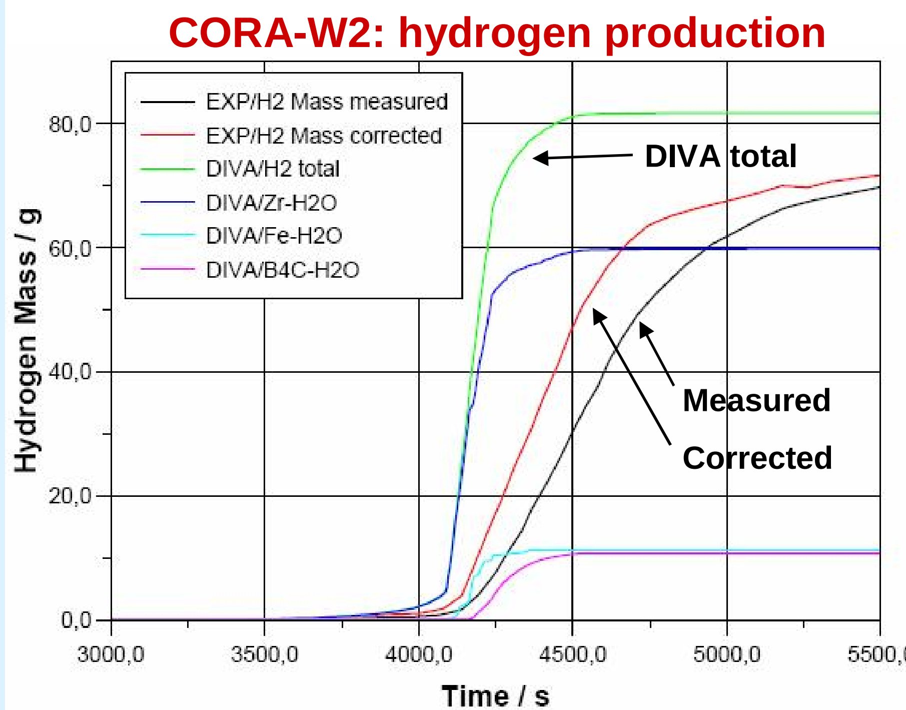
C.D.: CORA-13 and –W2 Experiments



CORA-13 Bundle

- **Application of ASTEC to CORA-13 bundle degradation experiment (western PWR type) and CORA-W2 (VVER type) performed by IKE**
- **Assessment of specific models of DIVA for oxidation, bundle degradation and quenching**
- **CORA-13: Bundle temperature, bundle degradation and hydrogen before quenching well predicted – hydrogen peak during quenching not reproduced**
- **CORA-W2: B_4C control rod oxidation model available – total hydrogen production slightly overestimated**

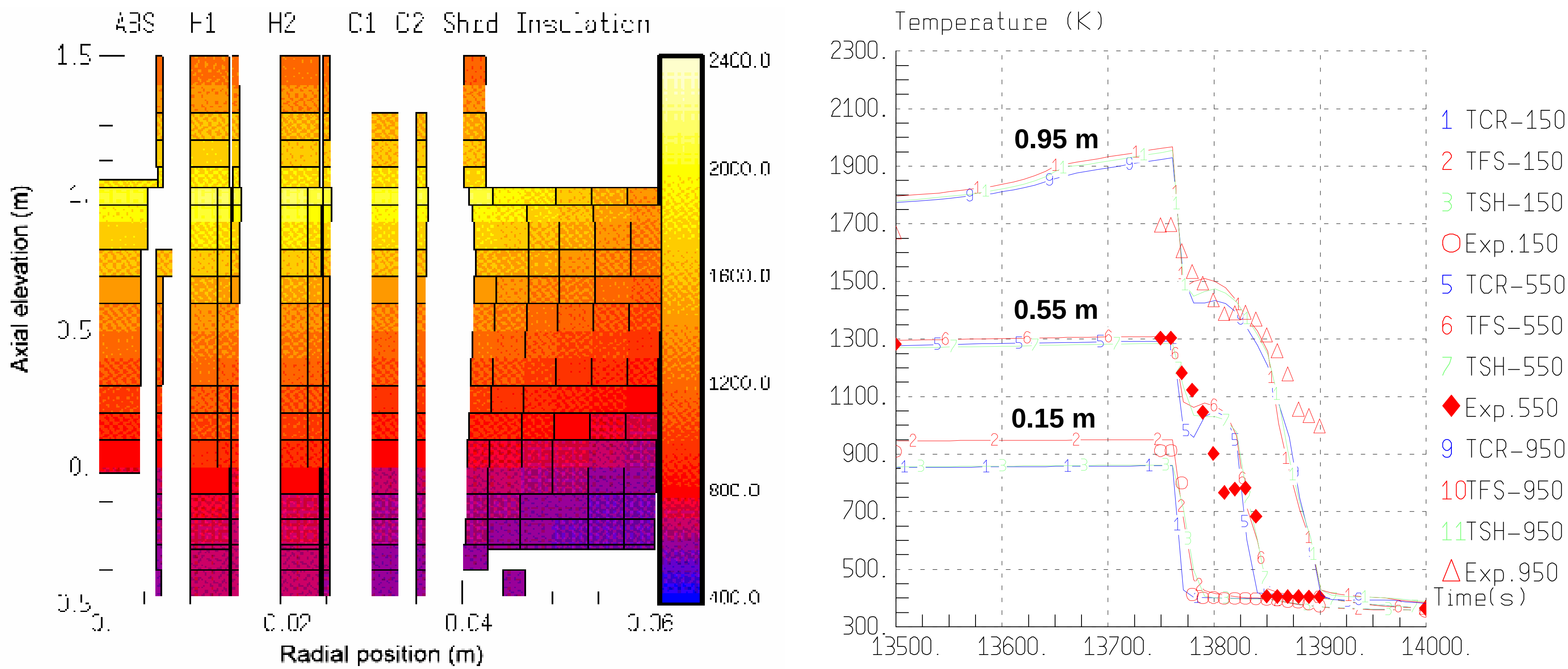




C.D.: QUENCH-13 Experiment

- Assessment of DIVA module on QUENH-13 experiment performed by FZK
- Quenching of an intact bundle initiated by fast water injection – central absorber rod
- Bundle temperature prior to reflood agree well with measured values
- Some deviations found during reflood due to simplified reflood model compared to thermal-hydraulic codes like CATHARE, ATHLET or RELAP5

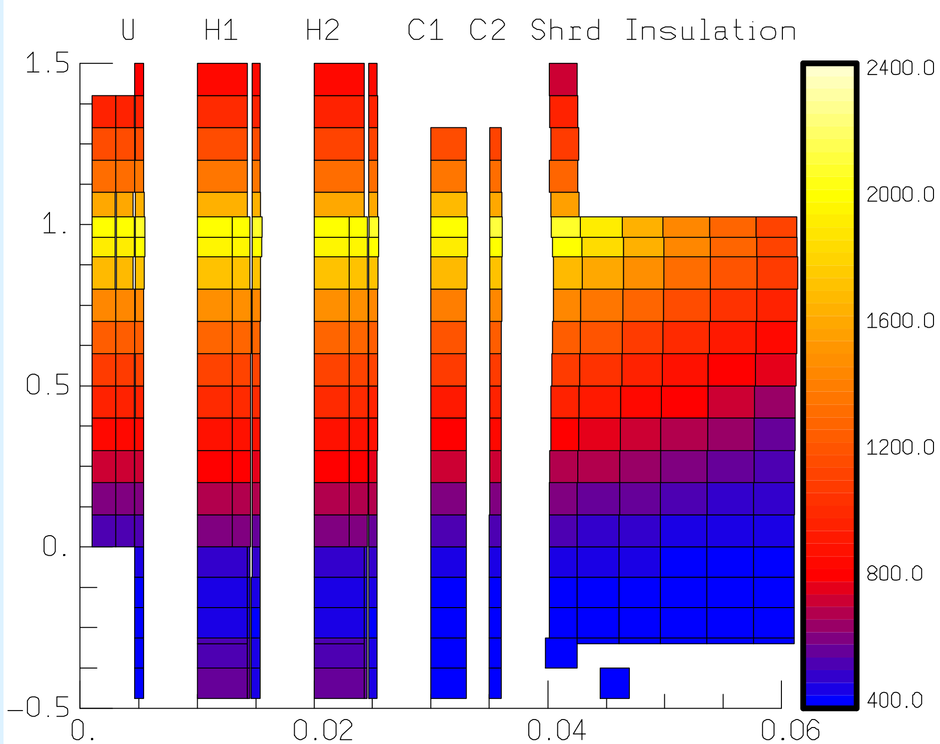
QUENCH-13: Bundle temperatures and comparison with measurements

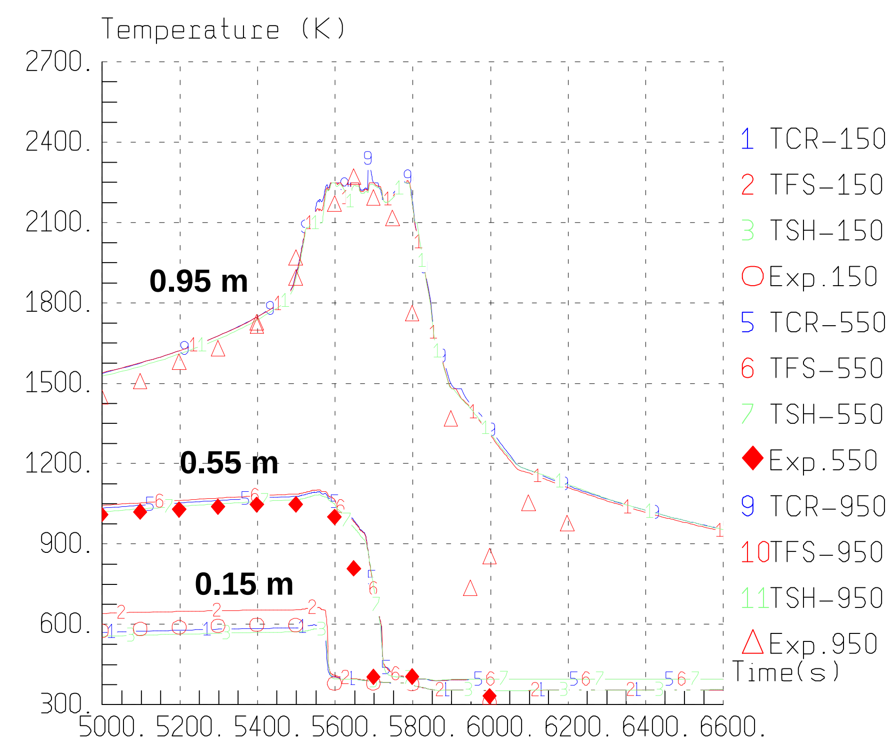


C.D.: QUENCH-11 Experiment (1)

- QUENCH-11 experiment has been analyzed by FZK, INRNE and ENEA
- Initial boil-off phase starting with bundle dry-out - followed by bundle heatup and degradation - terminated with a very low reflood rate
- **FZK Analysis:** Some difficulties in simulating the experiment, but taking into account the experimental uncertainties, the discrepancies found can be judge as acceptable

QUENCH-11: Bundle temperatures during reflooding phase (FZK analysis)

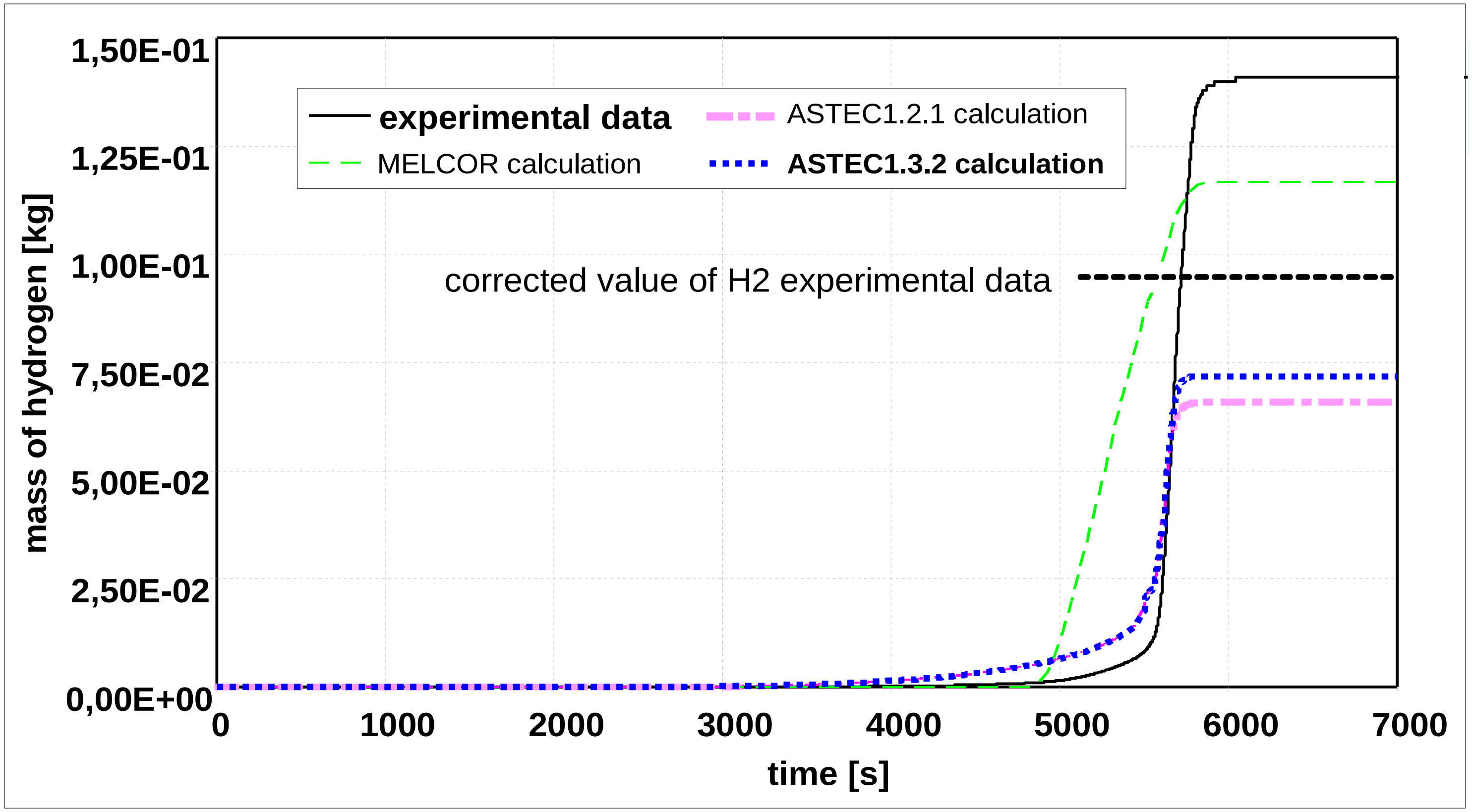




C.D.: QUENCH-11 Experiment (2)

INRNE Analysis (comparison with MELCOR):

- **In general the results correspond to the measured data, except for the increasing level during quenching due to lack of shroud failure simulation, which is reflected in the underestimation of total hydrogen generation**
- **The new V1.3R2 version calculates slightly better the hydrogen generation compared to previous code version due to improvement of oxidation modeling**



QUENCH-11

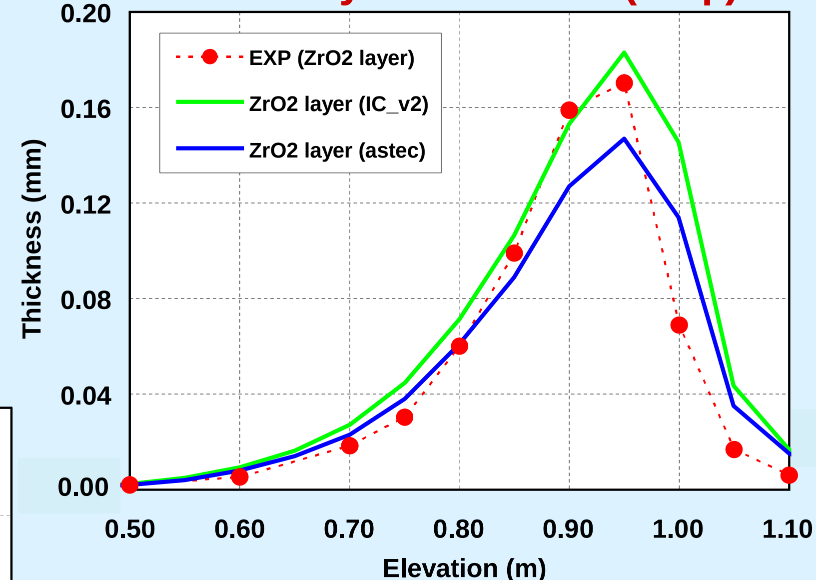
Total mass of hydrogen produced (INRNE analysis)

C.D.: QUENCH-11 Experiment (3)

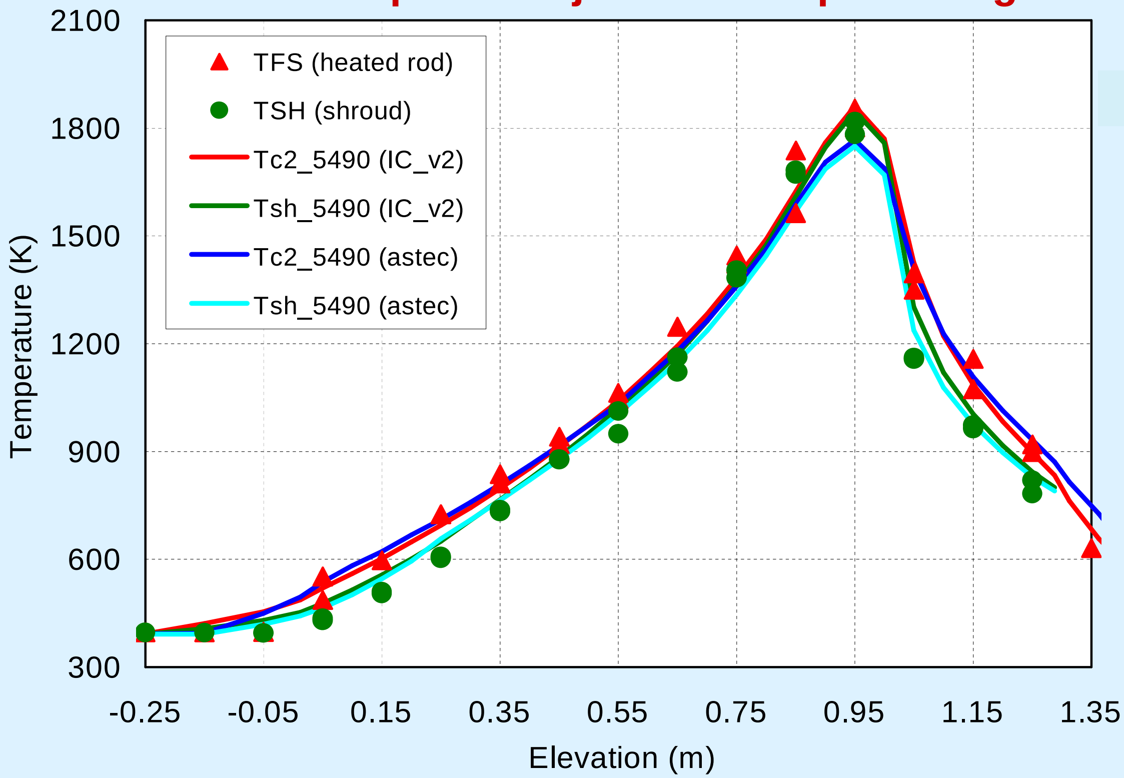
ENEA Analysis (comparison with I/C):

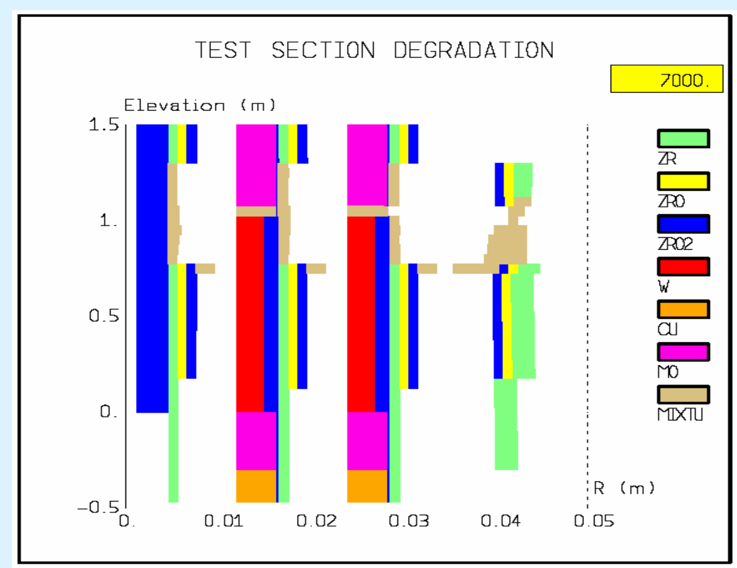
- **The results confirm the good capability of ASTEC to simulate the bundle behavior during the boil-off phase**
- **Major uncertainties observed during the quenching phase**

Oxide layer thickness (b. q.)



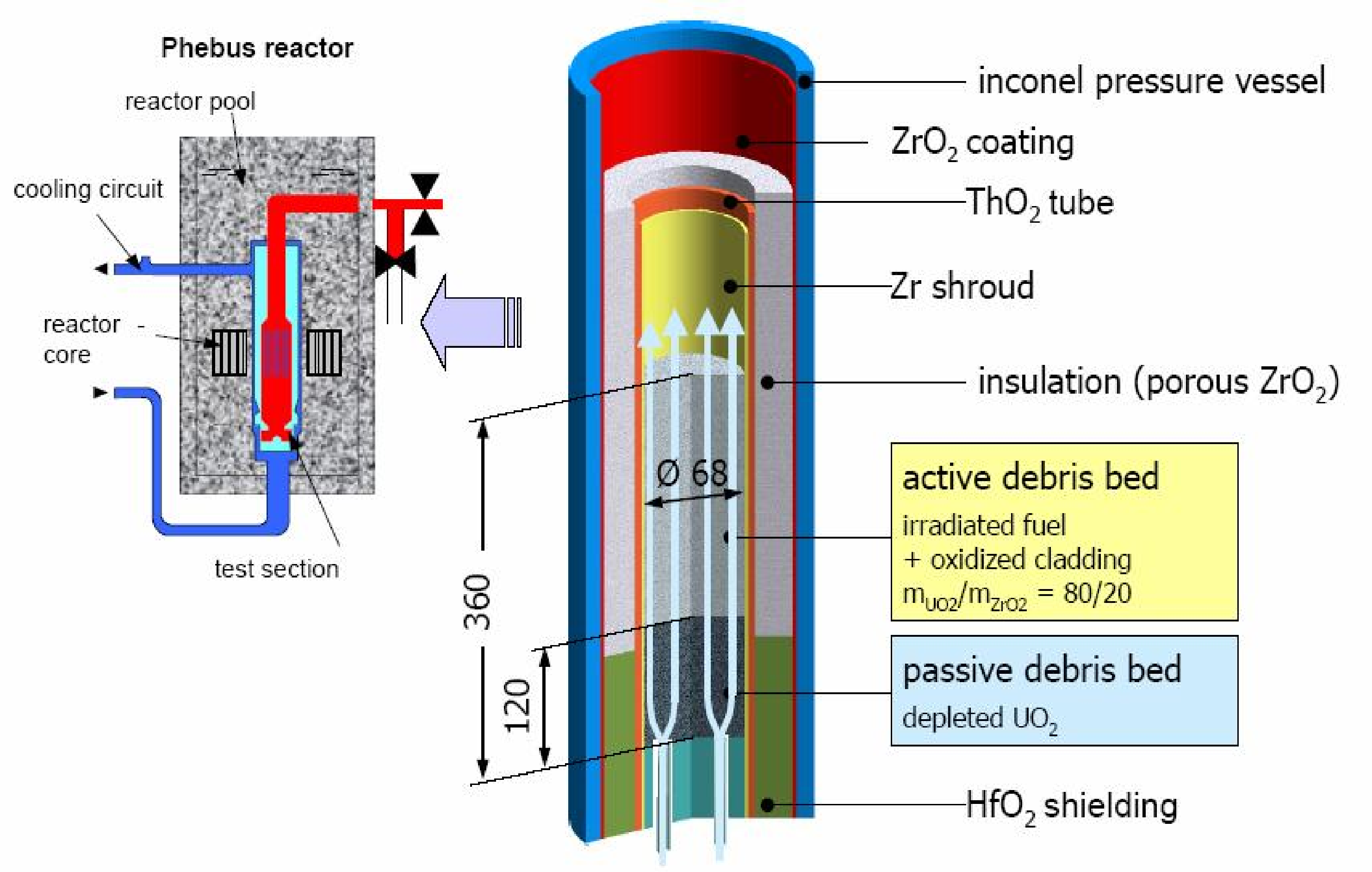
Clad temperature just before quenching





C. D.: Phébus FPT-4 Experiment (1)

- **DIVA calculation of the French Phébus FPT-4 experiment by IKE – checking of DEBRIS and MAGMA preliminary models (late phase degradation)**
- **Validation of heat transfer and melting in a highly degraded core, molten pool behavior and fission product release**

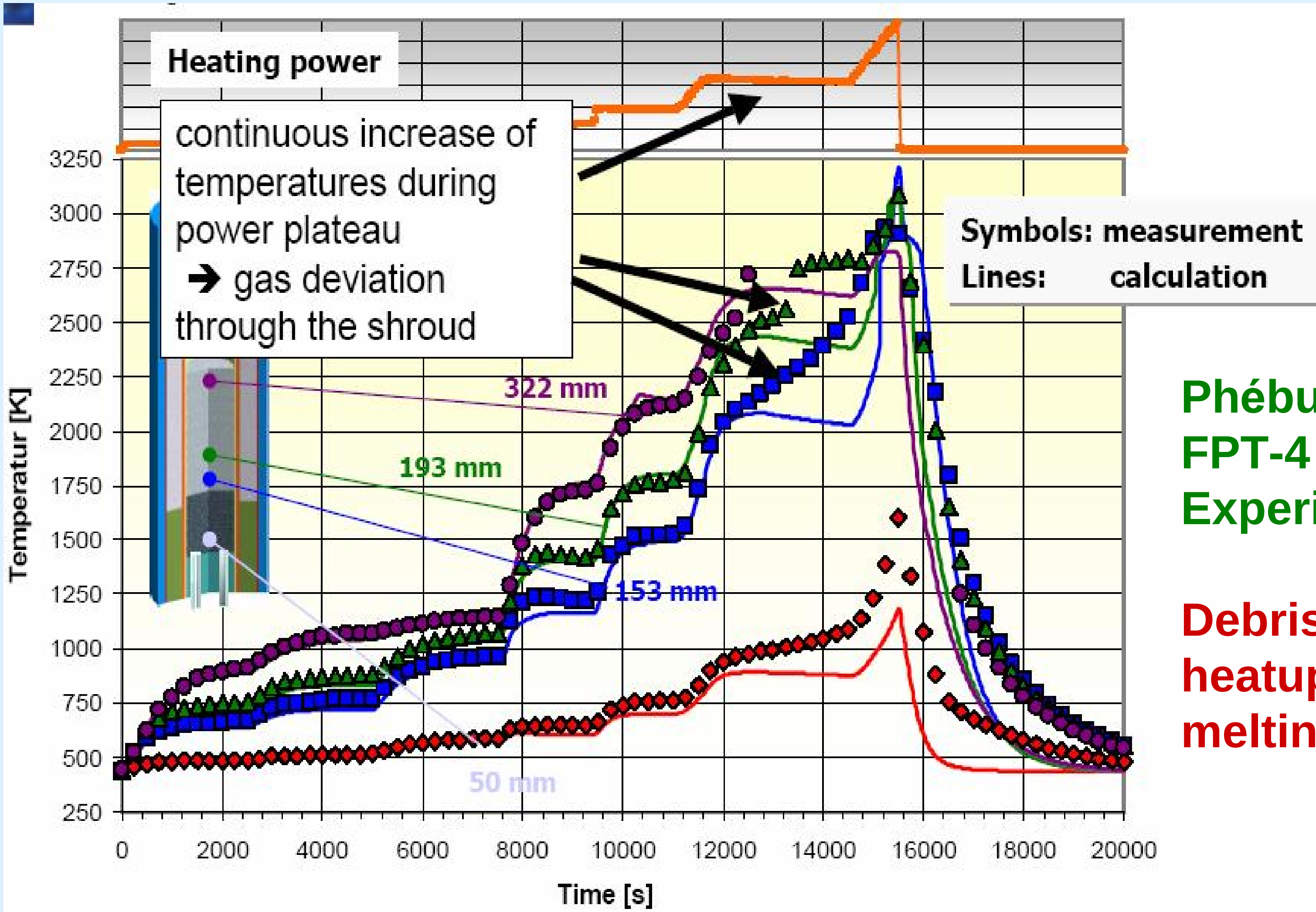


Phébus FPT-4 Experiment:

Debris bed heatup and melting

C. D.: Phébus FPT-4 Experiment (2)

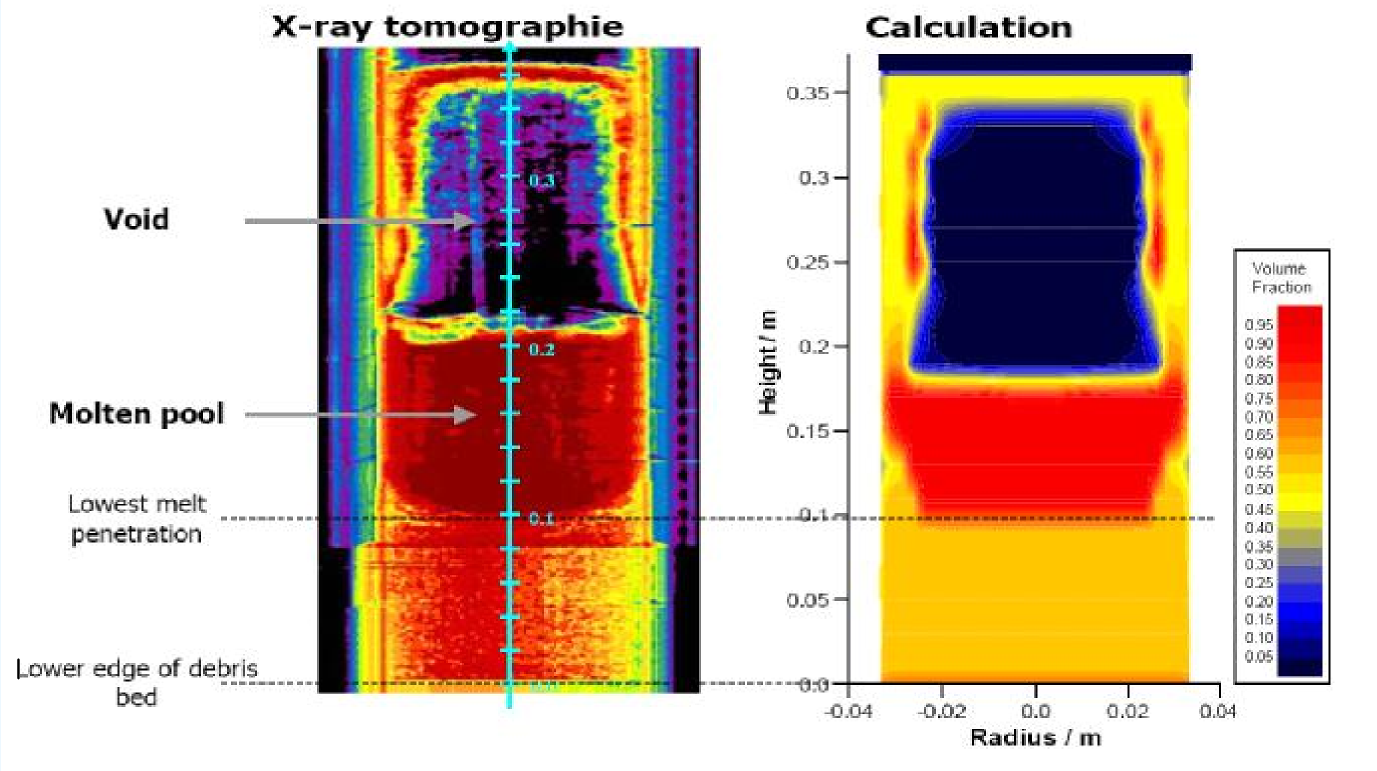
- Results for temperatures in good agreement with experiment, new calculations match also temperature increase in upper part (underestimated in last phase)



Phébus FPT-4 Experiment:

Debris bed heatup and melting

C. D.: Phébus FPT-4 Experiment (3)



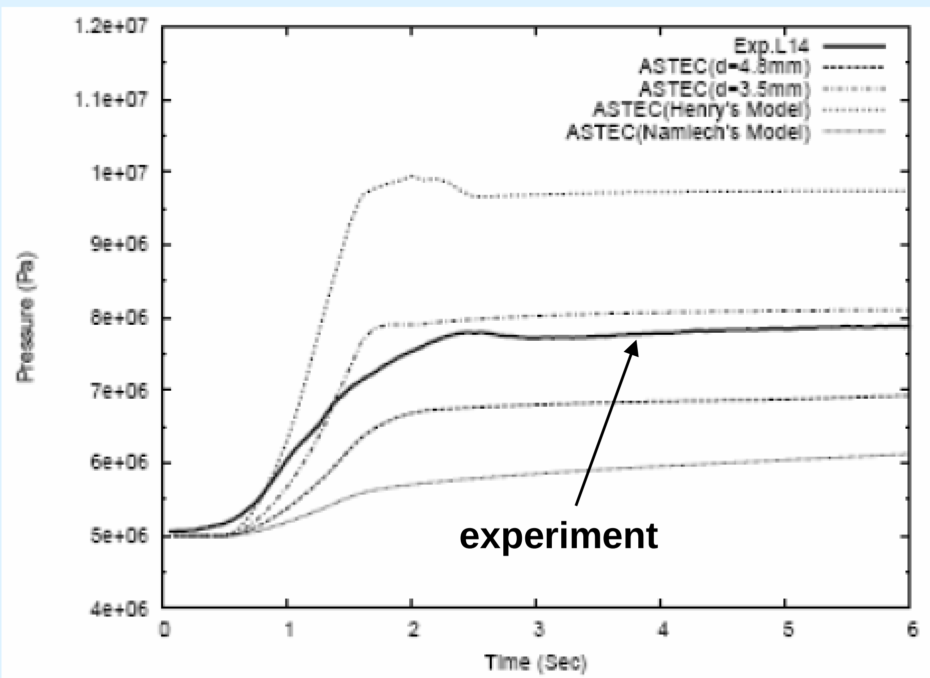
- **Final configuration (molten pool and crust position, molten mass) match well with the experiment**

- **Simulation of fission product release from the debris bed show reasonable agreement with experiment:**
  - **Good agreement for high volatile species (Cs, I, noble gases)**
  - **For some elements, especially Pd and Sr the release is over- respectively under-estimated by an order of magnitude**
  - **calculated release rates of major actinides (U, Pu and Am) exceeds the measured values à possible re-deposition inside the debris bed is not taken into account**

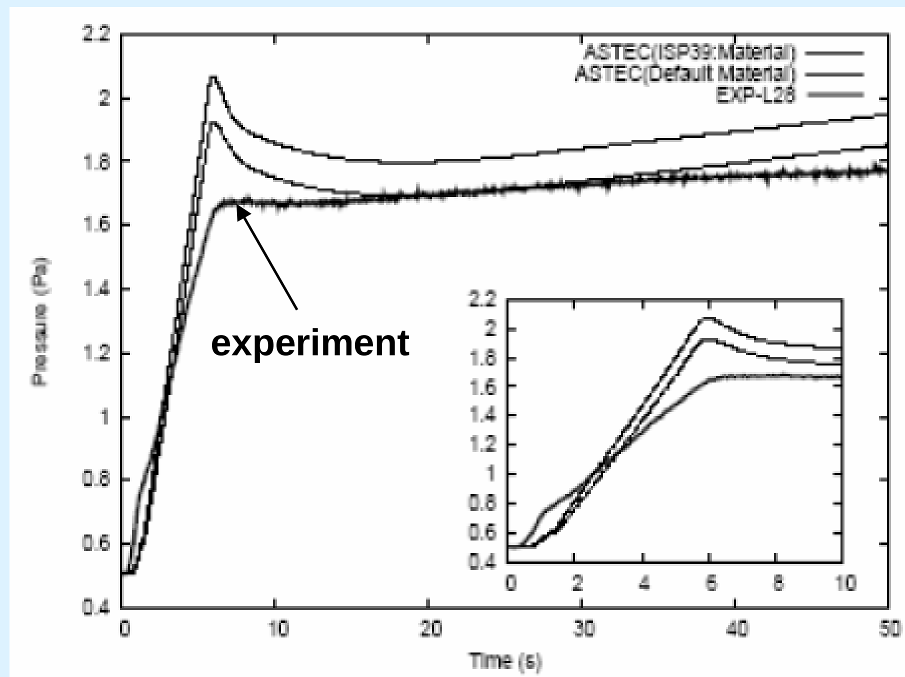
C. D.: FARO L14 and L28 Experiments

- **The FARO L14 and L28 performed by JRC-Ispra represent the interaction of a corium melt (80wt% UO2, 20wt% ZrO2) poured by gravity into a water pool**
- **The FARO tests have been analyzed by IRSN with previous V1.2 code version (same conclusions could be drawn for V1.3)**
- **The debris masses and the initial pressure rise are reasonably well predicted**
- **Main discrepancies with experimental data have been explained by sensitivity analyses on model parameters and corium properties were uncertainties exist**

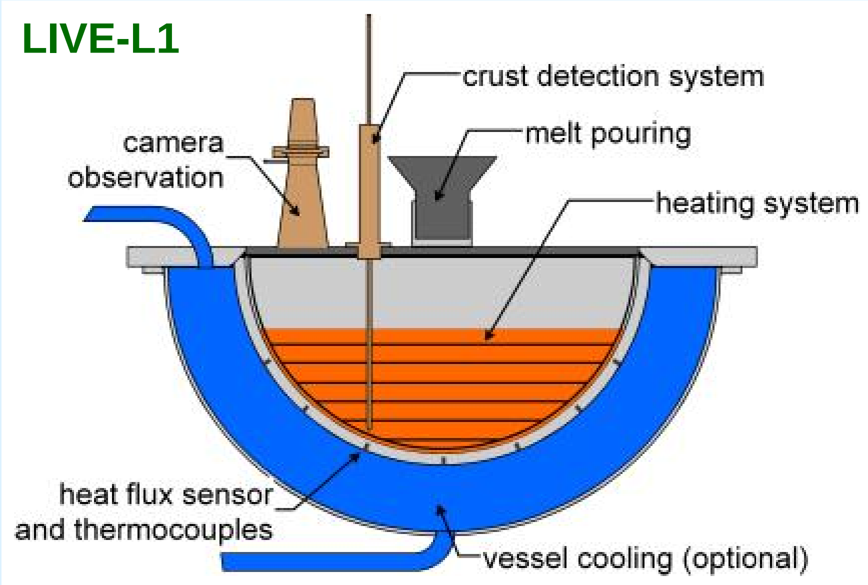
Pressure history in FARO L14 (ASTEC V1.2)



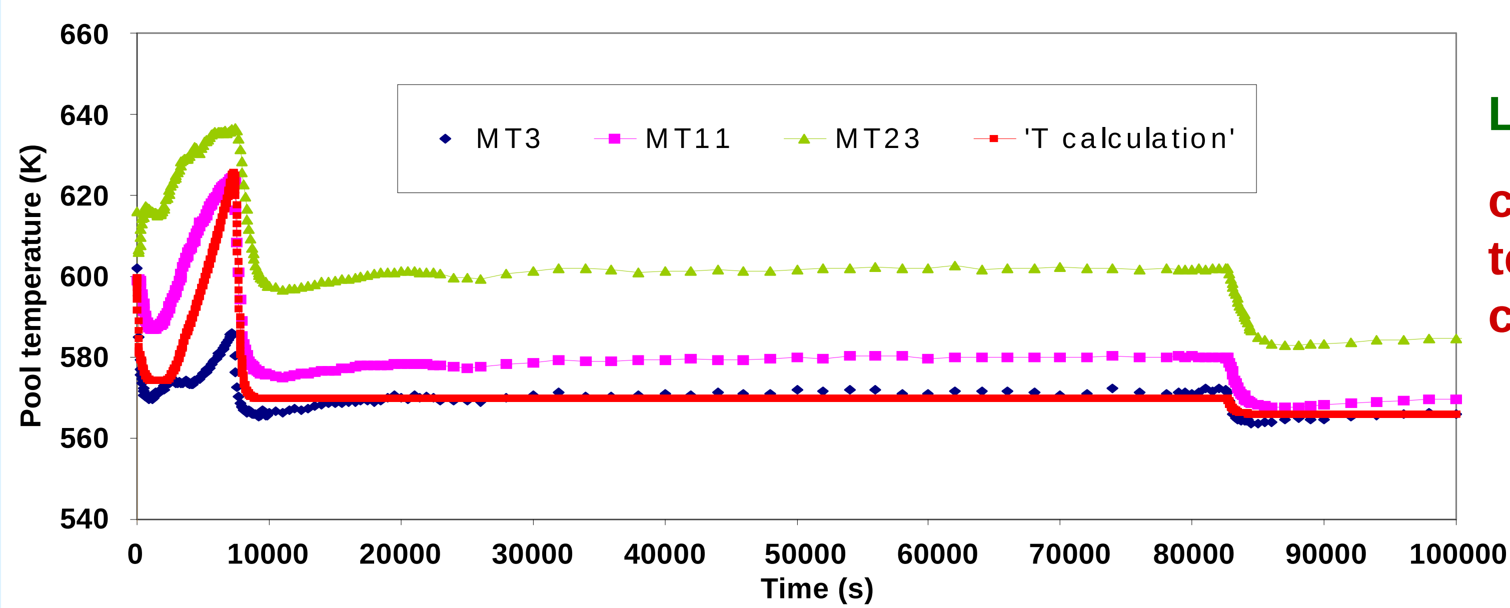
Pressure history in FARO L28 (ASTEC V1.2)



C.D.: LIVE-L1 Experiment (1)



- Conducted at FZK/Karlsruhe to investigate core melt behavior in the lower plenum with external vessel cooling
- LIVE-L1 analysis has been performed by CEA for DIVA late phase model validation
- Molten pool behavior and crust formation evaluation (one-layer pool modeling in DIVA)

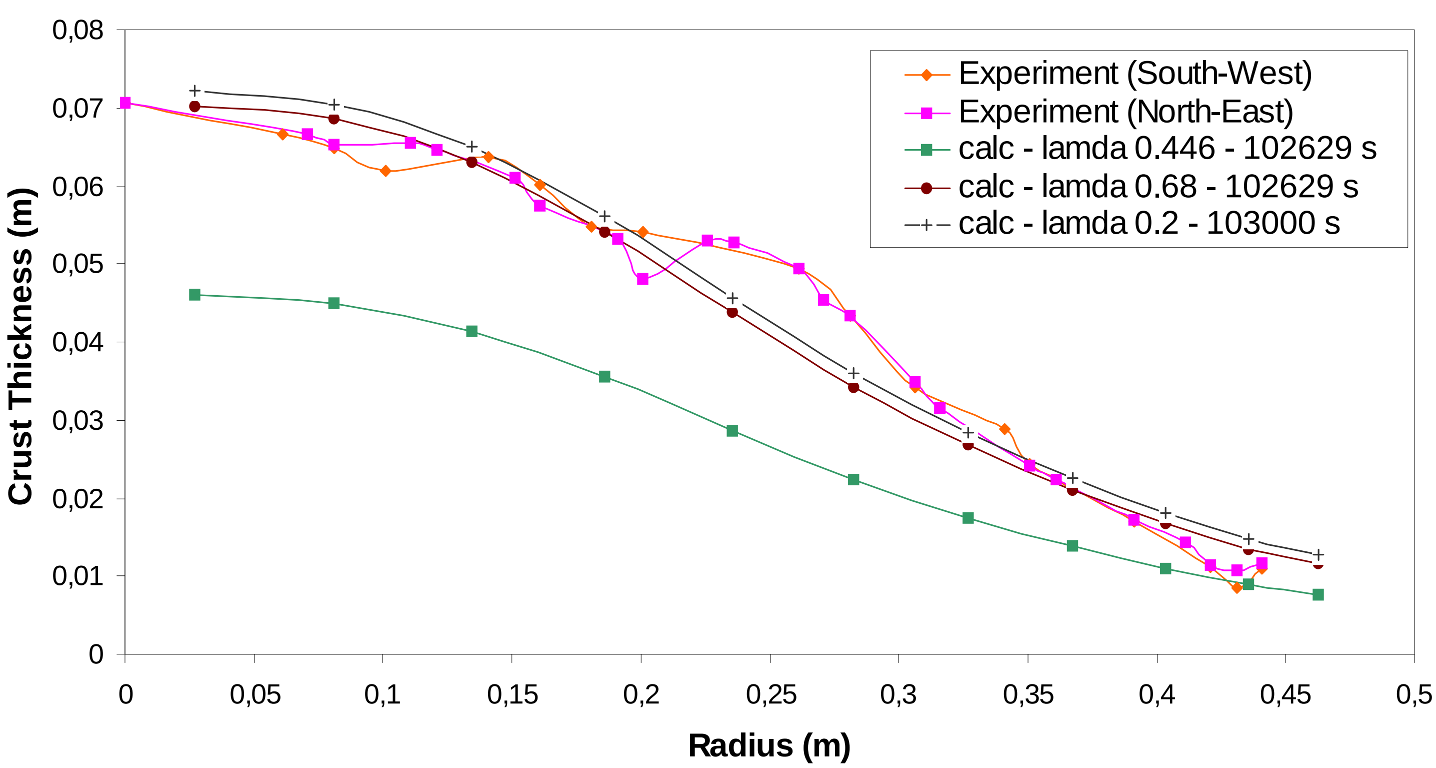


LIVE-L1:

central pool temperature comparison

C.D.: LIVE-L1 Experiment (2)

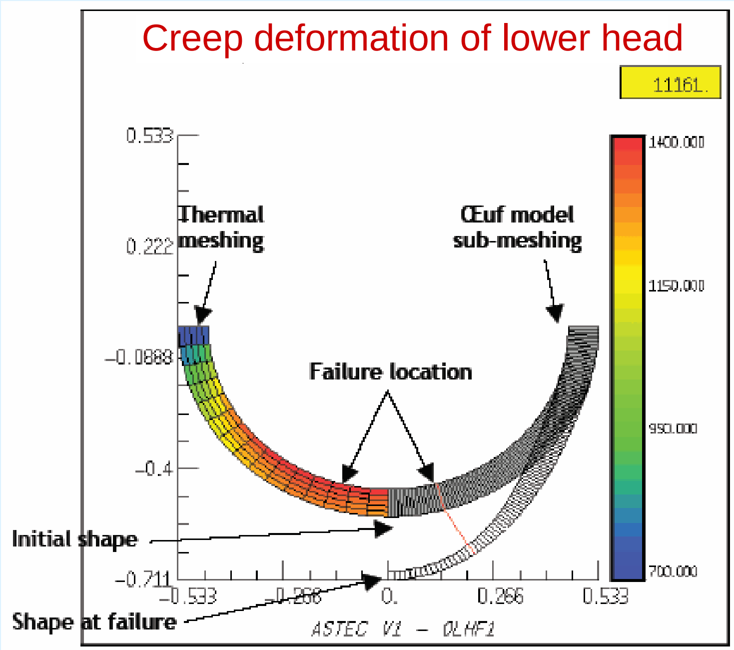
- Globally the experimental trend is obtained even if:
  - Corium temperature is 10 to 20 K lower than the measured value
  - Vessel wall temperatures are slightly higher than the measured values
- Even with the single layer molten pool approach, the crust thickness profiles compare fairly well with the post-test experimental values – The crust thickness is sensitive to the crust thermal conductivity model



LIVE-L1:

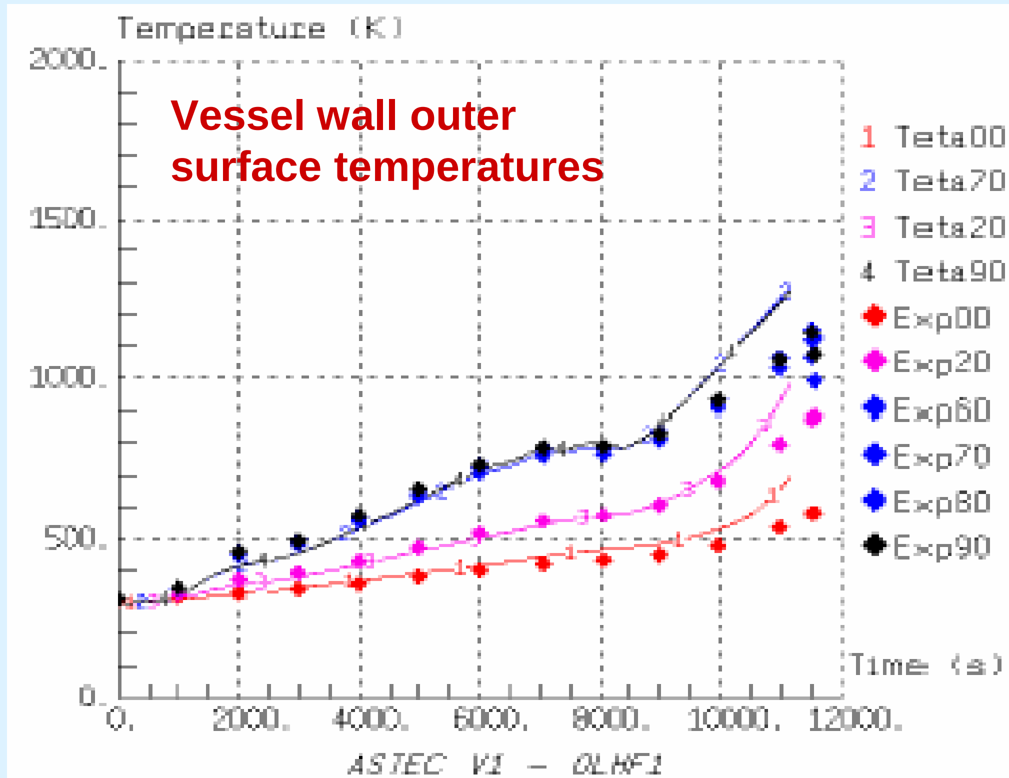
Molten pool crust thickness comparison

C.D.: OLHF-1 Experiment

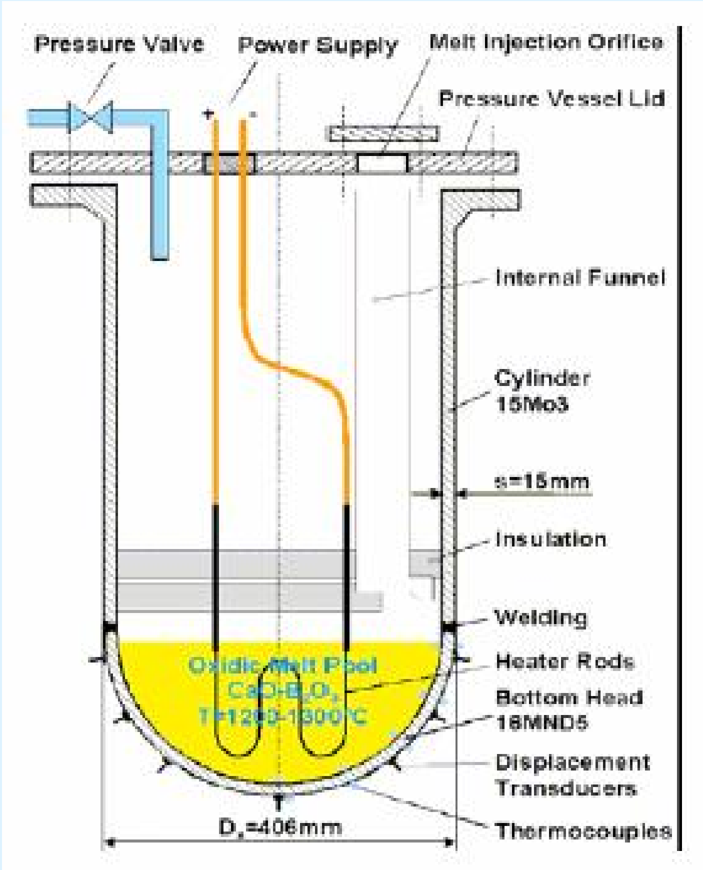


- The first OECD Lower Head Failure (OLHF-1) experiment has been analyzed by IRSN to validate the DIVA vessel rupture model (OEUF)
- Simulation of thermal / mechanical loads of reactor pressure vessel on a 1:5 scale model
- Uniform heating by induction and constant pressure of 12.4 MPa

- The global mechanical behavior is well represented
- The OEUF model is able to correctly predict failure time and location
- External vessel wall temperatures are overestimated after creep initiation likely due to uncertainties on external heat transfer coefficients

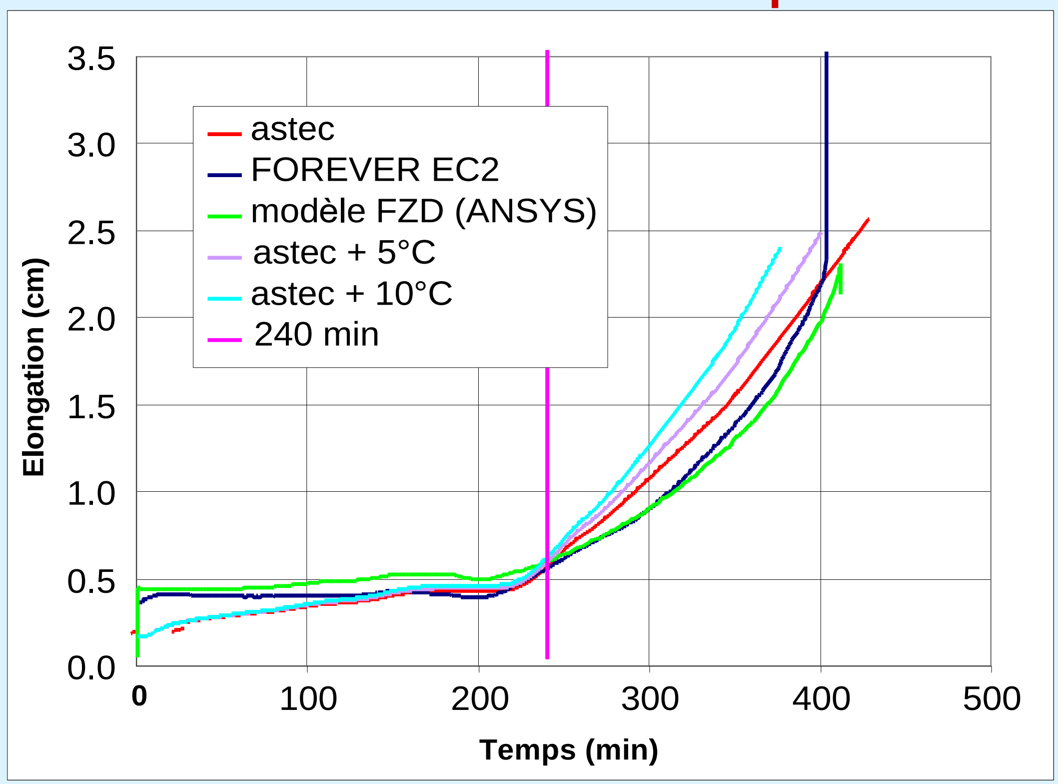


C.D.: FOREVER EC2 Experiment



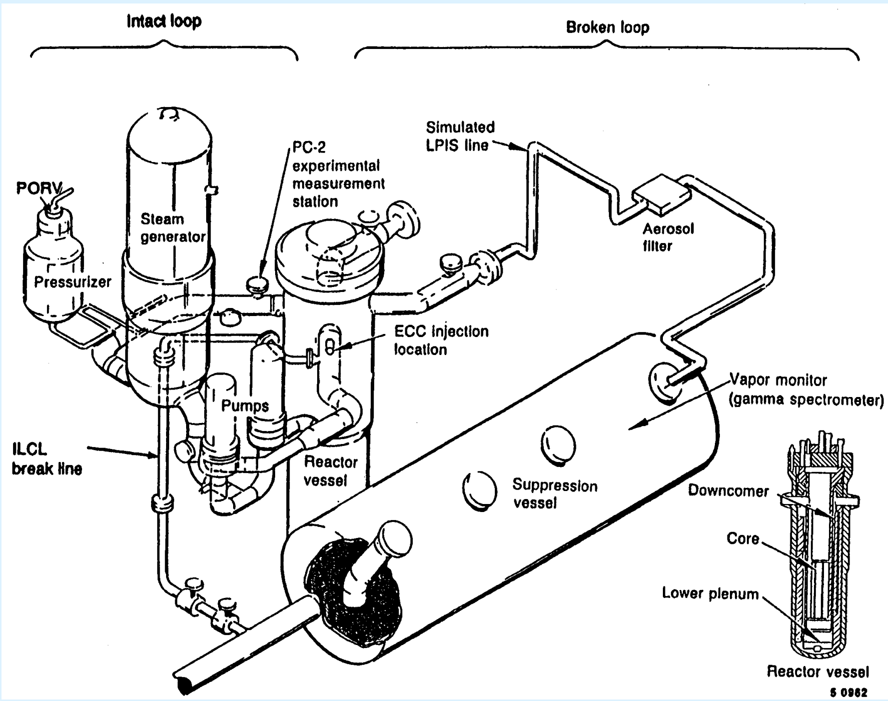
- Simulation by IRSN of the lower head failure experiment performed at RIT/Stockholm
- Molten oxide mixture poured at 1200 K simulated the corium behavior in the lower head
- Electric heating and pressure at 2.5 MPa

FOREVER EC2: Lower head displacement

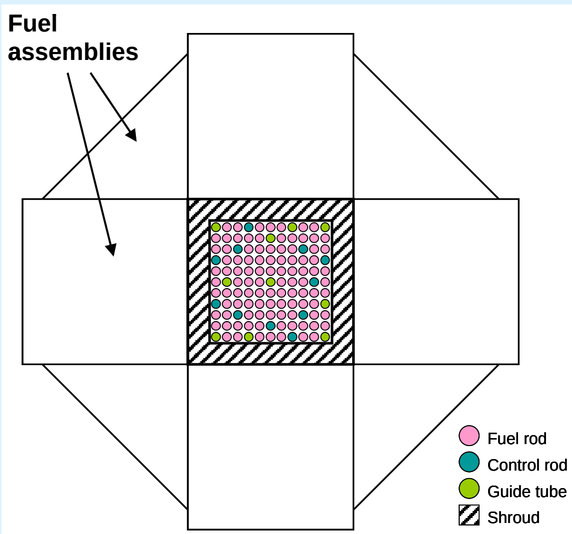


- Global behavior well predicted
- Rupture time and location and lower head displacement are coherent and close to test measurements and the values calculated by the ANSYS code

Coupling: LOFT LP-FP-2 Test (1)

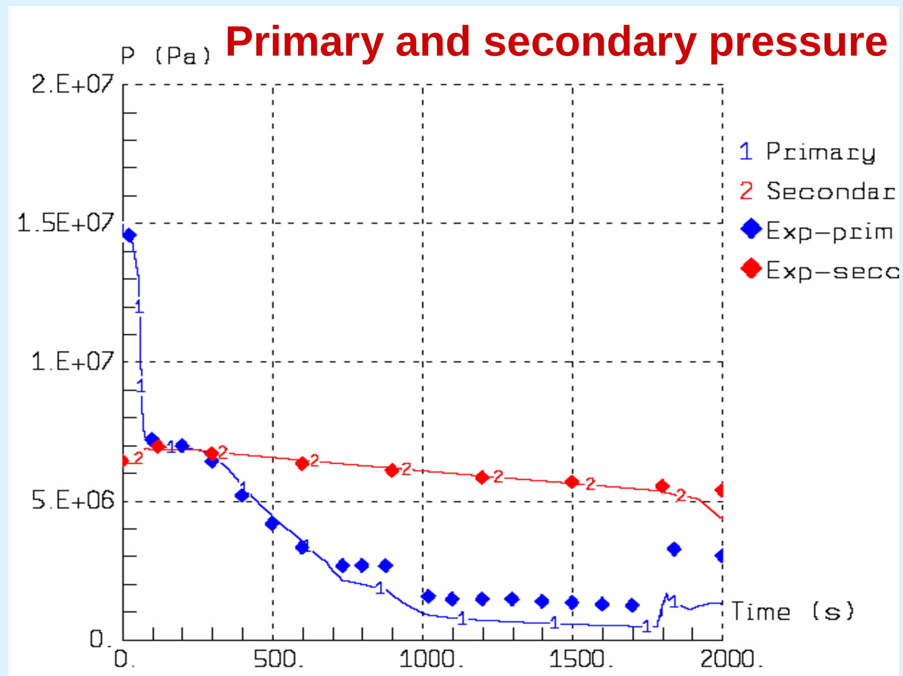


LOFT Facility at INEL (USA)



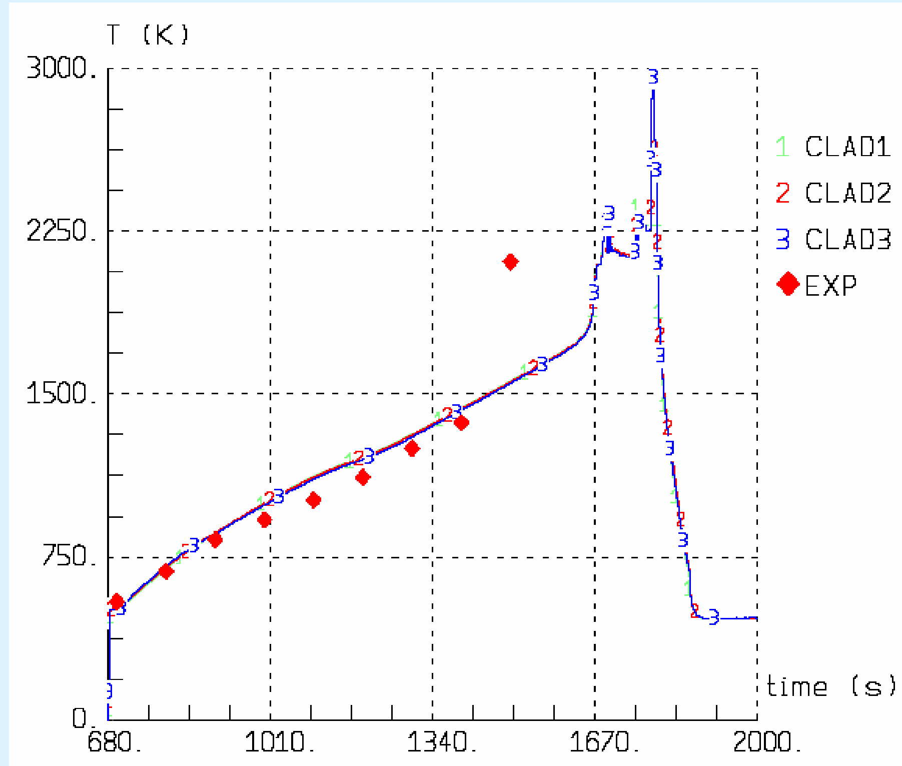
Central Fuel Module Arrangement Inside the Reactor Core

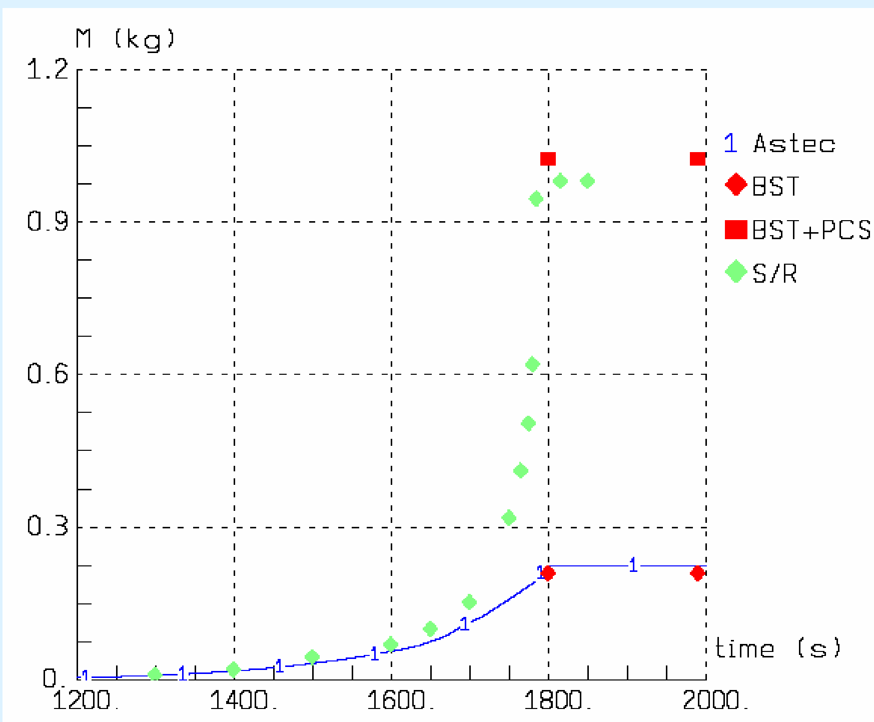
- **The LOFT LP-FP-2 test has been analyzed by ENEA for CESAR and DIVA coupling validation**
- **Loss of coolant accident scenario in PWRs that resulted in severe core damage – Terminated by core reflood**
- **Provided information on fuel rod behavior, hydrogen generation and fission product release and transport**



Coupling: LOFT LP-FP-2 Test (2)

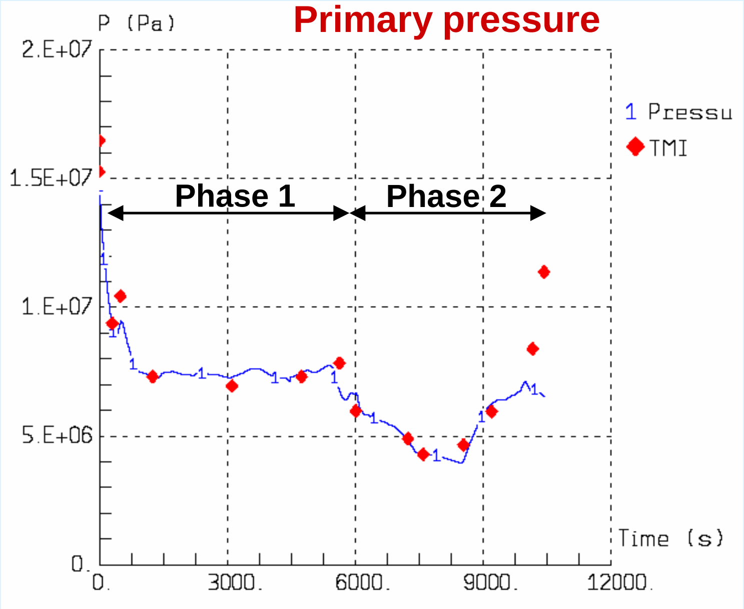
- The thermal-hydraulic behavior of the system is reasonably well predicted by ASTEC
- Onset of core uncovery and CFM rod heatup is very well reproduced but the onset of temperature escalation in the upper part of the CFM is significantly delayed

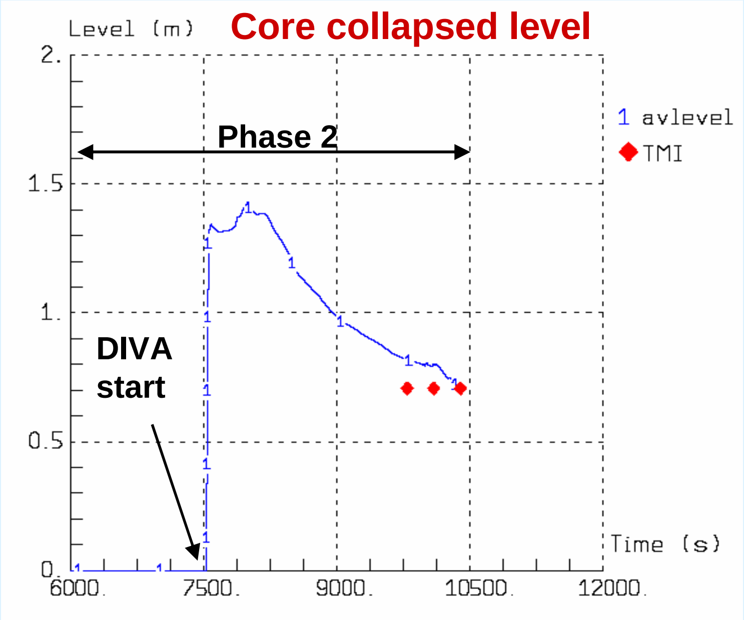




- Mass of hydrogen produced before reflood is very well predicted
- High temperature excursions with extended core degradation and enhanced hydrogen release observed in the test during reflood was not reproduced due to the lack of adequate modeling

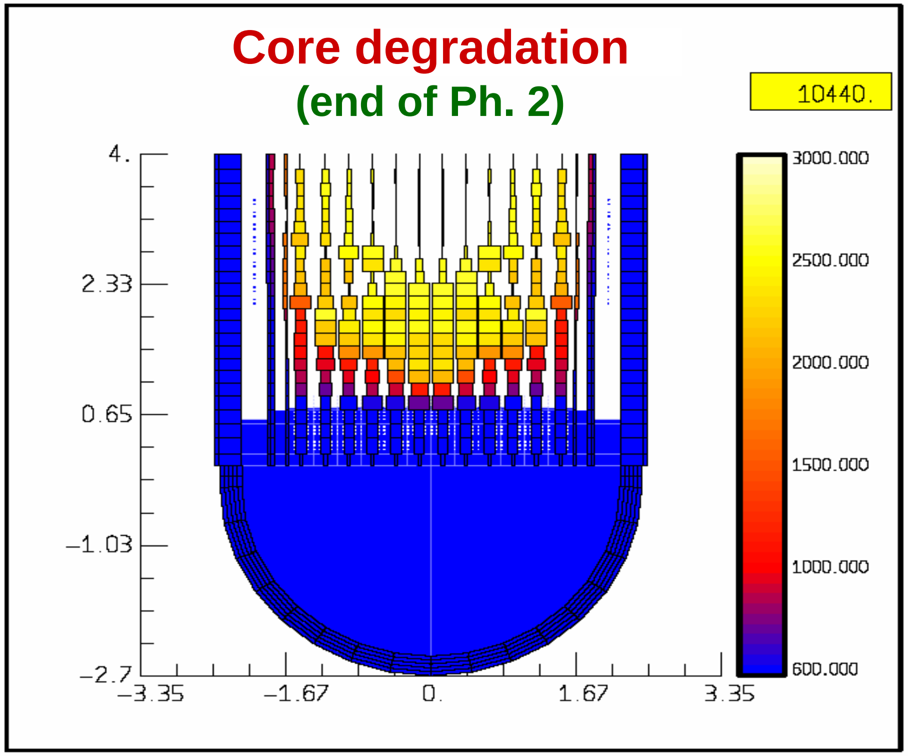
Coupling: TMI-2 Accident (1)

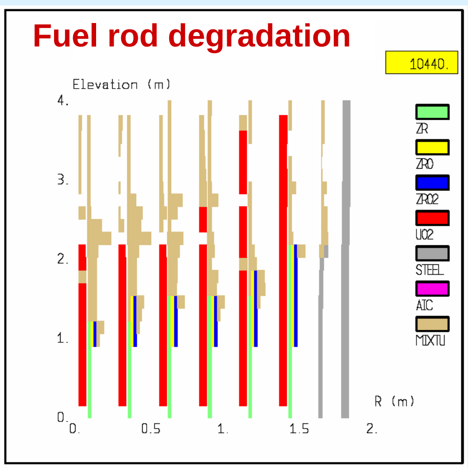


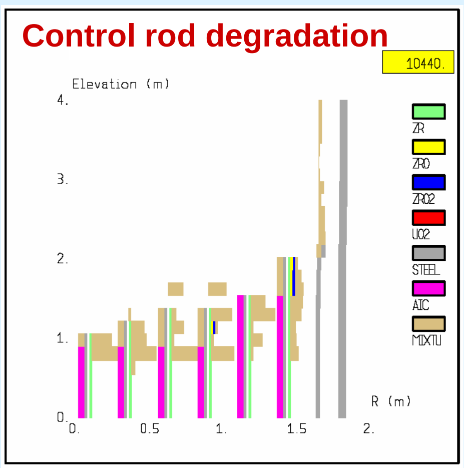


- **First two Phases of the TMI-2 accident have been analyzed by ENEA**
  - **Phase 1: loss of coolant through the PORV until primary pump stop**
  - **Phase 2: core uncovery, heatup and melting before core reflood**
- **TMI-2 provides a unique opportunity to assess code capabilities to simulate a severe accident in a full scale reactor**
- **Thermal-hydraulics in the primary system is very well predicted by the code during both Phase 1 and 2**
- **Pressurizer level behavior is very well simulated**
- **The residual water level in the core at the end of Phase 2 is in good agreement with TMI-2 observations (according to bottom crust location in central core ring)**

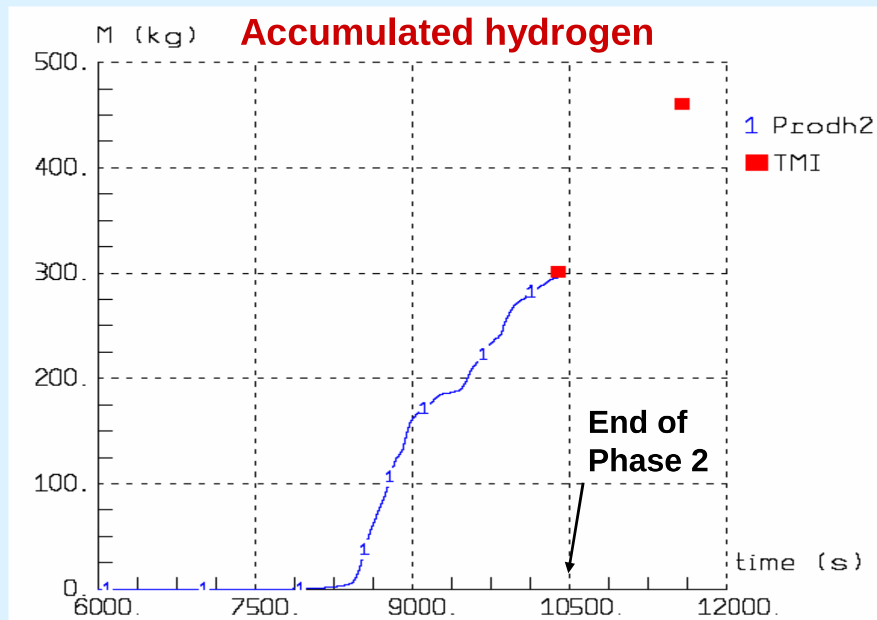
Coupling: TMI-2 Accident (2)







- Core degradation and molten mass calculated at the end of Phase 2 are consistent with hypothesized TMI-2 core degradation scenario
- Accumulated hydrogen at end of Phase 2 is very well predicted
- Extension of TMI-2 accident analysis to Ph. 3 and 4 (core reflood and core relocation) is foreseen with ASTEC V2 (improved debris and magma models)



CONCLUSIONS (1)

CIRCUIT THERMAL-HYDRAULICS:

- **The robustness of the thermal-hydraulic module CESAR is highly improved with respect to previous code versions**
- **Good results have been obtained on the integral test LOFT LP-FP-2 and on two PACTEL experiments in VVER geometry covering various thermal-hydraulic flow regimes**
- **Good results are confirmed by the intensive validation work against BETHSY integral tests as shown for 9.1b test**
- **TMI-2 accident analysis has confirmed the capability of CESAR to simulate primary circuit thermal-hydraulics in a real plant application**
- **A discrepancy remains on PMK2 SBLOCA transient – investigation of hydro-accumulator behavior is under way to clarify the difference**
- **Outside the SARNET framework, good results have also been obtained in IRSN large validation work on separate-effect tests (COSI, COTURNE, REBECA, SUPER MOBY DICK experiments in CEA), on steam generator experiments (PATRICIA GV in CEA), on PWR transient pressurizer data, and benchmarks with CATHARE V2.5 on PWR 1300 plant applications**

CONCLUSIONS (2)

CORE DEGRADATION:

- Many partners efforts focused on DIVA core degradation module validation
- Good results have been obtained for early-phase models, oxidation and hydrogen production before quenching in all calculated experiments (CORA, QUENCH and LOFT LP-FP-2) and TMI-2 analysis
- Improvements on the thermal-hydraulic phase of reflooding for quasi-intact bundle have been observed on CORA and QUENCH experiments with respect to previous code versions – But total amount of hydrogen under reflooding remains highly underestimated in CORA-13 and LOFT LP-FP-2 experiments
- Results for late phase models can be considered as good for:
  - Debris bed melting (Phébus FPT-4)
  - Corium fragmentation at slumping in lower plenum (FARO)
  - Corium behavior in the vessel lower head (LIVE-L1)
  - Vessel lower head mechanics and failure (OLHF1 and FOREVER EC2)
- Reasonable agreement for fission product release in analyses of FPT-4 (from debris - molten pool) and LOFT (before quenching) experiments

CONCLUSIONS (3)

- Main modeling weakness remains the reflooding of a degraded core, like most other codes
- Implementation of improved debris bed and magma models is in progress by merging DIVA with ICARE2 in the next ASTEC V2 release à more realistic simulation of the late phase phenomena up to the failure of the lower head
- Further validation of these models on experiments such as LOFT LP-FP-2 and on TMI-2