STARS

Safety Research in Relation to Transient Analysis for the Reactors in Switzerland

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ABSTRACT

The explorative analysis of the first BWR RIA experiment of the ALPS program provided new insights into the complex mechanisms operational during such fuel transients. The successful participation in a benchmark in the framework of the OECD SCIP project demonstrated the potential of the FALCON code with the coupled GRSW-A fission gas model.

The TRACE code has further matured, and migration of the legacy BWR inputs to TRACE was started. At the same time, analysis of selected ROSA PWR-SBLOCA-experiments showed the good performance of this code, but also indicated that occasionally small detail (e.g. small leakage flows) may turn out crucial for successful analysis. Assessment work on condensation in U-tubes, of particular importance during reflux-condenser mode, found a strong interest from the code developers and will form an PSI in-kind contribution to CAMP.

STARS continues to develop uncertainty evaluation for best-estimate applications: The PhD-thesis on objectively deriving uncertainty characteristics of important model parameters (e.g. void, CHF) was successfully completed. Work on the application of the uncertainty evaluation methodology applied in STARS to a PWR scenario (BEMUSE phase IV) was also successfully completed with good results when compared to the international participants. In that perspective, a study related to the main steam-line break analysis of a PWR also provided useful information on possible contributors to any uncertainty evaluation of coupled analysis that arise from the cross section formalism used in the kinetic module.

A PhD study was initiated to develop a coupling between a CFD code and TRACE, and the proof-of-principle application has been implemented.

Considerable effort was spent on updating the core models, and the respective computing environment CMSYS has been upgraded respectively. The updating of the criticality safety analysis method by integrating the most modern nuclear data libraries ENDF/B-VII and JENDL-3.3 yielded a strong reduction of the keff-bias when compared to a large benchmark suite.

Finally, STARS was able to provide preliminary analysis within 24h of the request for a plant transient that happened this year. This expedite response was made possible by very experienced and knowledgeable experts using an efficient project infrastructure.

Project Goals

The mission of the **STARS project** is to maintain and further develop a comprehensive state-of-the-art best-estimate safety analysis methodology – including criticality safety – for reactor states ranging from normal operation to beyond design conditions (before core melt) and integrate the necessary tools into a consistent system. In effect, the STARS project acts as **technical support center for LWR Safety Analysis** with the following general goals:

- Conduct research necessary to further develop the high level of expertise of the project team as well as to improve the integrated state-of-the-art analysis methodologies;
- Perform independent safety analysis and related studies at the request of HSK;
- Perform studies on safety and operational issues at the request of the Swiss utilities;
- Provide general neutronic analysis incl. scientific services to the Swiss utilities.

Specific goals set for 2007 were grouped under 4 major headings representing somehow main directions of the research work of STARS, in addition to selected topics that currently are more of an exploratory character or help to extend the project infrastructure.

Goals for 2007

The main directions for 2007 are outlined below. (Some routine activities in direct support of the project infrastructure are not mentioned.)

- Enhance fuel modeling capability:
  - Initiate analysis of selected RIA and LOCA experiments from the ALPS program.
  - Continue participation in the Halden LOCA-experiments with TH and thermo-mechanical analysis, refine modeling of the relocation phenomenon and transfer insights to safety analysis; support design of the planned BWR-experiment.
  - Continue the improvements of FALCON in relation to FG-modeling.
  - Analyze selected CABRI RIA experiments (MOX and UO2) pending availability of the respective data.
- Continue research on uncertainty assessment:
  - Continue participation in CSNI/GAMA/BEMUSE Phase IV-VI (application to PWR).
  - Participation in new NSC uncertainty benchmark (UAM) phase I addressing cross-section uncertainty.
  - Participate in IAEA Uncertainty CRP (incl. task coordination).
  - Continue developing uncertainty evaluation capability for fuel behaviour analysis.
- Continue with TRACE assessment:
  - Analysis of selected tests from the ROSA program.
  - Continue assessment of condensation models.
  - Apply official release version to a simple BWR-problem.
  - Assess the generalized radiation heat transfer model using the Halden LOCA data.
- Assess capability of TRACE to analyze wave propagation problems following LOCA-events, especially in the perspective of mechanical loads on reactor internals.
- Continue development of CFD application for NPP representative geometries:
  - Complete single-phase mixing analysis capability for the KKG reactor using CFX-5.
  - Initiate PhD-study on coupling of CFD with TRACE.
- Complete pre-CHF correlation work.
- Continue participation in NURESIM:
  - Perform core physics benchmarks.
  - Perform coupled TH-neutronics analysis for the OECD/NEA PWR MSLB Benchmark.
- Continue development of Monte Carlo methodology:
  - Implementation of burnup credit for criticality safety assessment.
  - Activation of the bio-shield.
  - Perform fast fluence analysis for additional NPP.
- Develop capability for LOCA analysis for EPR.
- Explore coupling of SIMULATE-3K to TRACE / RETRAN-3D.

Work Carried Out and Results Obtained

Parametric Optimization with the FALCON Code of the Further High Temperature LOCA Test in Halden

The next LOCA test at Halden (IFA-650.7) will be the first experiment within the Halden LOCA program addressing the behaviour of commercially irradiated BWR fuel. It is planned to test a fuel segment with a pellet-averaged burn-up of 44.3 MWd/kgU . It will be subjected to a heat-up with a asymptotic peak cladding temperature of about 1150 °C. The preliminary problem statement was first discussed during a Special LOCA Meeting in Storefjel Resort Hotel, Norway, on March 13, 2007 where it was decided to perform calculations using of appropriate fuel behaviour and thermo-hydraulic codes to define the characteristics of test fuel rod design, specifically, the plenum volume and the filling gas pressure, such that after burst a maximum cladding balloon size would develop.

A comprehensive computational study has been executed in view of optimizing the fuel rod design parameters and heat-up conditions to be implemented in IFA-650.7. Most of the calculations have been performed using the FALCON fuel behaviour code [1], utilizing the results of the TRACE thermo-hydraulic code as a basis for obtaining thermal-hydraulic boundary conditions. For the sake of higher flexibility, it was found advantageous to employ the FRELAX sub-code [2] (Halden LOCA oriented thermo-hydraulic supplement of the FALCON), after it had been tuned by the results of TRACE and verified against the data of the preceding LOCA tests at Halden.

The main goals of the PSI analysis were defined to be:
- Optimizing the cladding burst strain (size and volume of the balloon) in consideration of the existing design of the LOCA test rig.
- Achieving better consistency of the test fuel rod parameters with those of commercial BWR fuel rods.
- Giving proper allowance for the uncertainty in modeling assumptions, specifically, those related to the criterion for high temperature cladding failure based on the concept of critical cumulative damage index of the FALCON code [3].

The following specific recommendations have been developed for the characteristics of both the optimal fuel rod design and the heat-up conditions:
- The initial free volume of the fuel rod should be the same as in the preceeding LOCA tests, i.e. about 20 cm^3.
- The gas filling pressure should be significantly reduced from 40 bar (25 °C) in the previous LOCA tests to 6 bar in the planned IFA-650.7.
- The target peak cladding temperature should reach 1150 °C .

The corresponding prediction with the FALCON code can be summarized as follows:
- The calculated volume of the cladding balloon formed after burst amounting to 12.6 cm^3 is large enough to ensure the onset of axial fuel relocation and the corresponding reduction of fuel stack length, according to the empiric criterion based on data from relevant experiments carried out earlier at KfK [4].
- Calculated local peak strain of cladding after burst, 55.4 %, is low enough to avoid a mechanical contact of the deformed cladding with the heater.
- The characteristic ratio of gas volume in the rod to active volume of the fuel stack is reduced more than six-fold compared to the preceding tests, bringing this parameter to a better correspondence with the real BWR fuel.

These findings have been documented in a Technical Report to be submitted to the experimental team of the Halden Project [5].

Improvement and Verification of the FALCON Code Coupled with the GRSW-A Fuel Model

Having completed the integration of the GRSW-A model into the FALCON code and respective preliminary testing [6], further validation of the advanced FALCON code against available experimental data was worked on using two datasets addressing thermal and mechanical behaviour of high burn-up LWR fuels (from both PWR and BWR) during power ramps in research reactors .

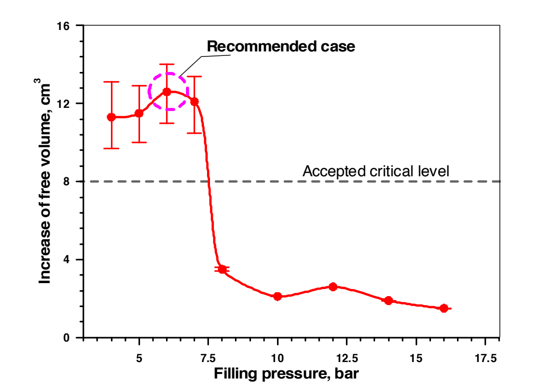

Fig. 1: Calculated characteristics of cladding strain at LOCA-stipulated burst, viz. (a) volume of the balloon as function of initial filling pressure.

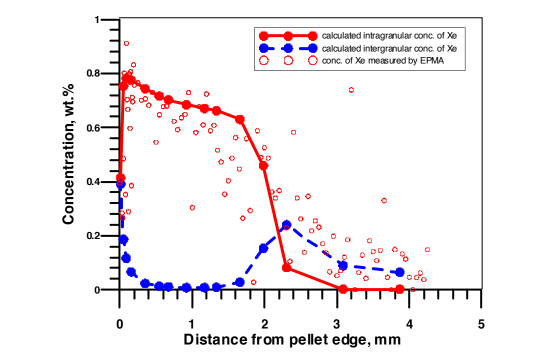

Fig. 2: Calculated Xe concentration vs. EPMA data for post-ramp matrix xenon distribution across a pellet of the REGATE fuel rod

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The analysis of the REGATE experiment with high burn-up PWR fuel aimed at a preliminary evaluation of the predictive capability for both steady-state and transient fission gas release (FGR), as well as transient cladding deformation (residual change of cladding diameter) and fission product distribution after power transient based on the experimental data that had become available in the framework of the IAEA fuel modeling project FUMEX II [7]. The analysis dealt with a segment of the full-scale segmented fuel rod base irradiated in the Gravelines-5 PWR to a pellet burn-up of 50 MWd/kgU and further submitted to a power ramp in the SILOE research reactor.
FALCON coupled with GRSW-A has demonstrated reasonable capability of predicting integral FGR in the selected rod both for base irradiation and power ramp with the same GRSW-A model used. Besides, the adequacy of modeling local FG behaviour was confirmed by a reasonable agreement with the results of EPMA for Xe and Cs (Fig. 2).
In addition, the comparison of calculation results executed with and without gaseous swelling due to pore formation confirmed that the evolution of the fuel porosity must be taken into consideration for the prediction of the cladding residual strain measured by post irradiation profilometry.
Similar conclusions were derived from the analysis of KKL fuel rods ramp test data (OECD project SCIP).

Application of FALCON Coupled with GRSW-A to Analysis of the Behaviour of Failed Fuel during a Pulse-Irradiation Test

The Japan Atomic Energy Agency (JAEA) has been conducting a comprehensive program directed at «Advanced LWR Fuel Performance and Safety» (ALPS) to promote a better understanding of fuel behaviour under accidental conditions and to provide a database for regulatory judgment. The LS-1, carried out on March 27, 2006 is the latest pulse-irradiation experiment which dealt with a fuel sample refabricated from a standard BWR fuel rod irradiated to a pellet-averaged burn-up of 69 MWd/kgU in the Leibstadt BWR (KKL) [9]. The LS-1 test was performed in the experimental capsule specially designed for simulation of Reactivity Initiated Accident (RIA) in the Nuclear Safety Research Reactor (NSRR). The LS-1 test rod was subject to a pulse-irradiation with integral energy injected of 527 J/g and a half-width of the pulse of 4.4 ms at atmospheric pressure and at room temperature.

When performing the FALCON analysis we first calculated the gas pressure balance in intra-granular bubbles by considering the bubble surface capillarity and the external hydrostatic pressure. It is seen from the multi-group calculation of the bubble-size distribution that the sharp increase of the fuel temperature leads to a dramatic bubble over-pressure. This suggests a high compressive micro-stress to be formed around bubbles. As already shown in the literature [10], [11], such over-pressure can, to a large extent, or even totally suppress the bubble coalescence due to the repulsive forces exerted on each bubble from the stress field induced by the neighbour, when the inter-bubble distance would becomes small enough for them to coalesce. Although, according to the calculation, the relaxation of bubble over-pressure happens rather quickly due to their growth, the mentioned suppression of bubble coalescence is expected to be an important factor leading to the mitigation of the intra-granular gaseous swelling and faster gas arrival at grain boundaries, which evidently may facilitate grain boundary gaseous swelling and fission gas release.
The cladding failure, early in the phase of energy injection, could result in a drastic ingress of the steam into the free volume of the rod, which then gets into contact with the surfaces of the pellet fragments causing intensive oxidation of the fuel, first of the grain boundary network area [12].

In order to further explore the impact of the two specific features in the behaviour of the failed fuel during fast power transient (LS-1), we run trial calculations with the heuristic assumptions of

1. absence of bubble coalescence throughout the transient and
2. enforced increase of the local O/U ratio in close vicinity of the grain boundaries from the initial value (2.001) up to the one of U_3O_8 (2.67).

The former of these assumptions, along with the account for the irradiation induced resolution, suggests restriction of the mechanisms under consideration for fission gas transport from the grain matrix to the boundary to mono-atomic diffusion, whereas the second one facilitates the predicted growth of boundary pores through the increase of local concentration and, therefore, diffusive fluxes of the thermal equilibrium vacancies. The outcome of this path-finding calculation is in reasonable agreement with the available data, viz. measured high quantity of fission gas released and the high extent of pellet fragmentation, which is qualitatively consistent with the predicted tendency to grain separation due to the formation of significant intergranular porosity throughout the pellet volume (Fig. 3).

Nevertheless, the calculated onset of gaseous swelling using essentially conservative assumptions is predicted to take place well after the moment of cladding failure, which, according to the calculation of cladding stress and strain conditions at the moment of failure (known from measurement), is due to purely elastic strain (Fig. 4) caused by the thermal expansion of the pellet. This suggests a minor role of gaseous swelling in the failure of the LS-1 cladding.

Thus, new insights are provided by this work, thereby emphasizing the need for further development of the model in the part of the behaviour of highly overpressurized intragranular bubbles during fast thermal transients and the kinetics of fuel oxidation in failed fuel rods.

OECD/NEA BEMUSE Programme – Phase IV Sensitivity Analysis for a Large Break LOCA in ZION Nuclear Plant

The BEMUSE (Best Estimate Methods – Uncertainty and Sensitivity Evaluation) Programme, promoted by OECD/NEA, aims at the evaluation of uncertainty methodologies applied to the predictions of best estimate (BE) system analysis codes. While the first phases of the Programme have focused on the application of BE codes and uncertainty methodologies to a LOCA in the LOFT integral test facility, the successive phases address wa Large Break LOCA in the Zion nuclear power plant, a 4-loop PWR.

The Phase IV of the BEMUSE Programme has been carried out and completed. It consists of the simulation of a Large Break LOCA in the 4-loop PWR Zion reactor and in a sensitivity analysis. A TRACE nodalization has been developed on the basis of an existing RELAP5 deck. Following the latest specifications [13], issued in July 2007,

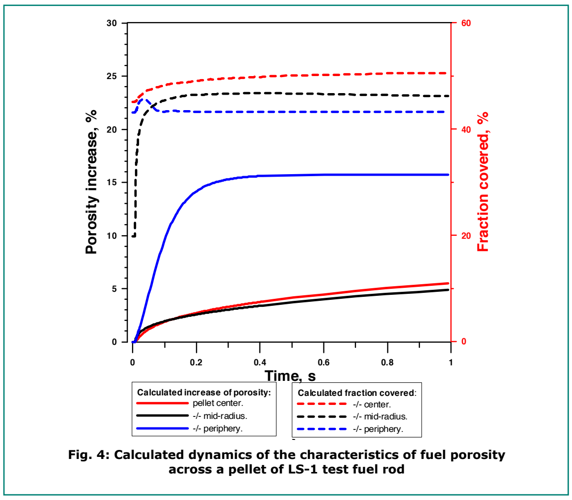

Fig. 3: Calculated dynamics of the characteristics of fuel porosity across a pellet of LS-1 test fuel rod.

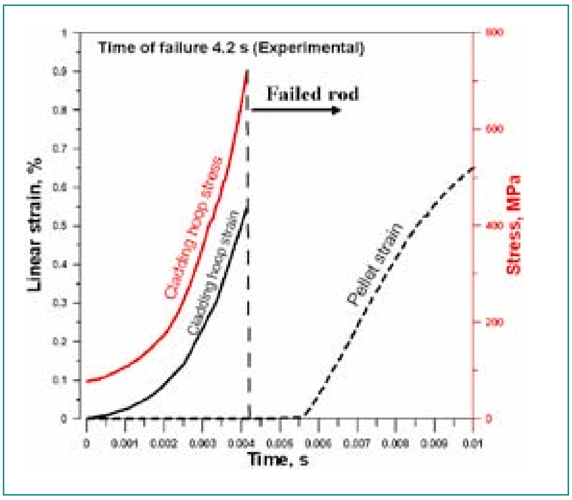

Fig. 4 :Calculated characteristics of cladding strain and pellet gaseous swelling against cladding hoop stress in LS-1 test fuel rod.

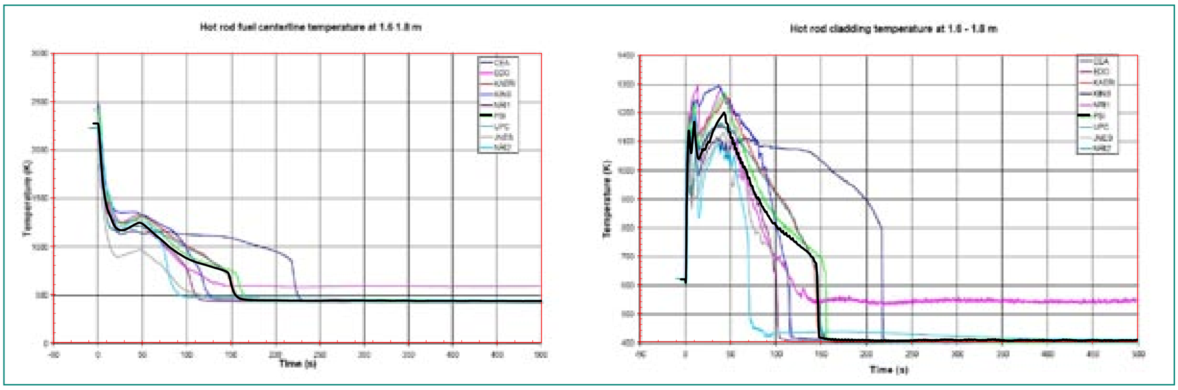

Fig. 5: Hot rod centerline (left) and cladding temperature (right) at 2/3 core height.

the core model has been expanded to include 4 radial rings. Five separate core regions (hot rod, hot assembly, hot channel, average channel and peripheral channel) are modeled.

In Fig. 5 a preliminary comparison is presented between PSI results and other participants of the BEMUSE programme. In particular, the fuel temperature and the cladding temperature are shown for the hot rod at 2/3 elevation, where the highest values are expected. The results obtained at PSI with the TRACE code are well in line with the results provided by the other participants. In addition, a sensitivity study has been performed. The influence of ten parameters on the PCT has been investigated. It has been found that the PCT is mostly influenced by a change in fuel conductivity or by changing fuel rods dimensions from cold to hot conditions. A strong influence on the PCT is found also when the gap conductivity is varied or if the maximum linear power of the hot rod is changed. The strongest influence on the reflooding time is given by the change of containment pressure evolution and by the change in decay power.

Analysis of ROSA-V SBLOCA in Vessel Experiment 6.1 Using TRACE

Recent inspections of the vessel head wall of pressurized water reactors (PWRs) have brought out the existence of significant wall degradation around the control rod drive mechanism. Axial nozzle cracking and small leakages were found in different power plants [14]. Investigations at Davis-Besse Nuclear Power station have revealed a localized large reduction of the vessel head wall thickness which could lead to a SBLOCA transient [15] initiated by small break at the upper head of the reactor pressure vessel (RPV). In this context, the OECD/NEA ROSA project conducted various tests at the ROSA Large Scale Test Facility (ROSA/LSTF) as part of the ROSA-V test program for safety research and safety assessment of LWR plants.

The OECD/NEA ROSA project aims at addressing thermal-hydraulic safety issues relevant for light water reactors through experiments making use of the ROSA/LSTF, a facility that simulates a Westinghouse design PWR with a four-loop configuration and 3423 MW_{th}. Areas, volumes and power are scaled down by a ratio of 1:48 while the elevations are kept at full height. Only two loops, sized to conserve the volume scaling (2:48), are simulated.

Test 6-1, following the findings made at Davis-Besse, simulated a RPV upper-head small break LOCA with a break size equivalent to 1.9% cold leg break [16]. The experiment assumes a total failure of the high pres-sure injection system (HPIS) and a loss of off-site power concurrent with the scram. As part of the accident management the SG relief valves are fully opened to cool down the system when the core outlet temperature reaches 623 K.

The main purpose of the study presented here is to assess the capabilities of the BE code TRACE to reproduce the physical phenomena involved in SBLOCA transients. A post test calculation of test 6-1 using TRACE (version 5.0) is presented. A previously developed nodalization of the ROSA test facility was used as starting point [17]. A full control system was developed in order to reach a correct steady-state as well as to perform the necessary actions taken during the transient. Sprays, relief valves, safety valves and corresponding control systems were included in the pressurizer. Separator components were inserted into the steam generators, thus improving the secondary-side system behaviour.

Afterwards, most of the work was focused on the nodalization of heat structures and its materials which are of main relevance during SBLOCAs. Another important point to correctly simulate this transient is the accurate nodalization of two bypasses connecting the hot leg with the downcomer (DC) and the DC with the upper head respectively. Their location is schematically displayed in Fig. 6. A more realistic nodalization of the former one led to an improvement of the core level evolution.

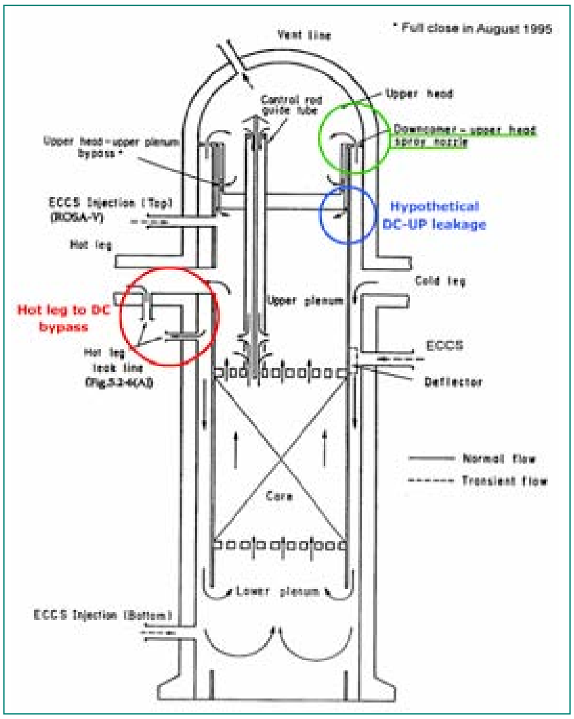

Fig. 6: Coolant flow path in RPV of the ROSA test facility. Source: JAEA

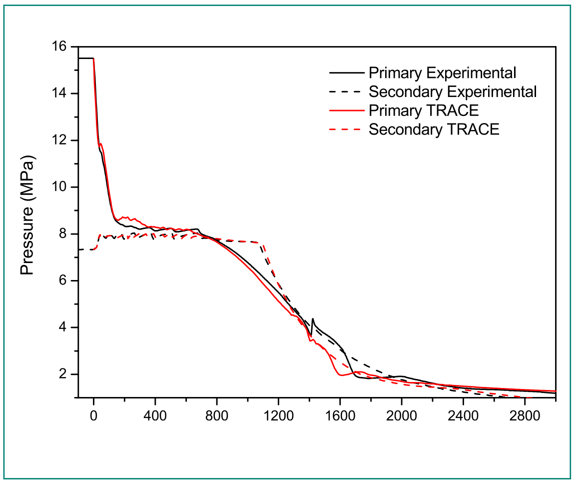

Fig. 7: Primary and secondary pressure of test 6-1.

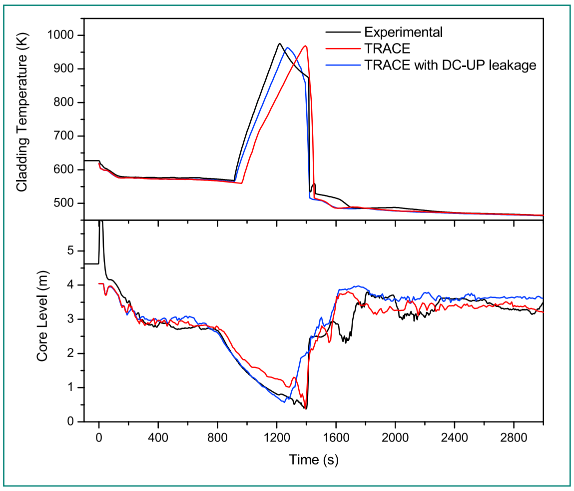

Fig. 8: Maximum cladding temperature and core level of test 6-1.

All modifications yielded a good representation of the events and phe-nomena taking place in the experiment. The cold leg and DC level evolution in the first part of the transient (500 s) brought forward interesting differences and posed a challenge for the analyst. While the level evolution in the core, upper plenum, upper head and hot legs was matched by the model, the cold leg and DC level could not be correctly simulated. What TRACE was not able to simulate was exactly the DC pressure drop which was reduced after the interruption of circulation in the experiment. This could be due to the over-estimation of the RPV heat losses, the heat transfer through the core barrel, pressure losses around the primary system or the nodalization of the bypass from the hot leg to the DC. None of these possibilities brought out variations to the issue. Following recommendations given in [18], the discrepancy could derive from a possible leak from the DC to the upper plenum through the 28 plugged bypass holes in the core support barrel, as marked in Fig. 6. In fact, both the cold leg and the DC level evolutions are correctly simulated supposing, as a first approach, a very small leakage (0.05 %) from the DC to the upper plenum. The new results obtained are shown in Fig. 7. Both the maximum cladding temperature and the core level present values which are closer to the experimental data (Fig. 8). Nonetheless, this hypothesis needs further investigations.

Validation of Film Condensation Models in TRACE

Previous work carried out at PSI with the best-estimate code TRACE has revealed, at certain operating conditions, unsatisfactory predictions of condensation in steam generators U-tubes during reflux condenser mode (film condensation) and when dealing with condensation of the steam reaching the pressurizer (direct-contact condensation). Therefore, an activity was initiated aiming at the assessment of TRACE condensation models against experiments performed with separate effect test facilities.

The condensation models in the best-estimate code TRACE have been assessed against experiments on direct-contact condensation and film condensation in vertical flows. The effect of non-condensable has been investigated as well. Only the assessment of film condensation models without non-condensable will be reported here.

Two experimental databases have been found in the open literature with regards to film condensation in U-tubes, with steam and liquid film flowing in counter-current configuration (falling liquid film in presence of upward steam flow). The first data set originates from an experimental facility built in Korea at KAIST (Korea Advanced Institute of Science and Technology) with experiments carried out in a 2.8 high U-tube having an inner diameter of 16.2 mm at 1 bar only. The second data set originates from the COTURNE facility of CEA in France. The test section consists of a tube of 4 m height and 20 mm inner diameter. Experiments over a wide pressure range (~ 6 – 60 bar) are available.

The heat transfer coefficient for film condensation is a function of the liquid film Reynolds number Re_{lq}. Therefore, all experimental results can be summarized by reporting the heat transfer coefficient against Re_{lq}. In Fig. 9 the results of the TRACE simulations are reported together with the experimental results from KAIST and CEA facilities respectively. Excellent agreement is ob-

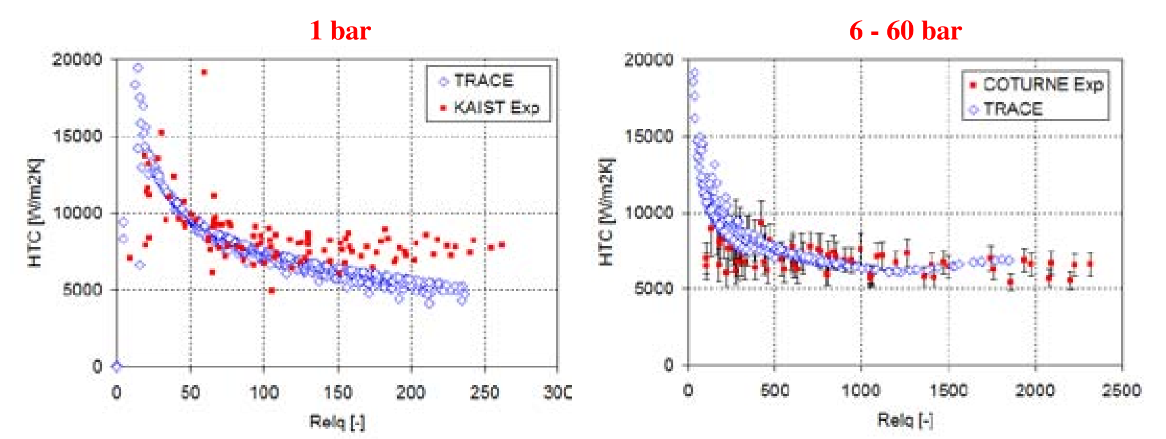

Fig. 9: Experimental and calculated heat transfer coefficients vs liquid film Reynolds number for KAIST (left) and COTURNE (right) test series.

tained in the pressure range from 6 up to 60 bar. Unsatisfactory predictions are obtained at very low pressures (1 bar, Fig. 9 left), where the heat transfer coefficient is mainly under-predicted with the TRACE model (see range of Re_{lq}. between 150 and 300).

It has to be pointed out that the KAIST experimental results obtained at atmospheric pressure lie in the same range as the heat transfer coefficients obtained at higher pressures with the COTURNE experimental set-up, i.e. no remarkable pressure dependency of the heat transfer coefficient is observed. TRACE, on the contrary, for a given liquid film Reynolds number, predicts a decreasing heat transfer coefficient with decreasing pressure, which becomes noticeable mostly at low pressures (see Fig. 10, left). The heat transfer coefficient depends on the resistance offered by the film thickness. In Fig. 10 (right) it is shown that the film thickness estimated by TRACE on the basis of the geometrical consideration $\delta = D_h(1-\alpha)/2$ that is fully consistent with the Nusselt theory, according to which the film thickness is evaluated as:

$$\delta_{Nu} = \left(\frac{3}{4}\frac{\mu_l^2}{\rho_l^2 g}\right)^{1/3} \mathrm{Re}_f^{1/3}$$

The increase of film thickness with decreasing pressure is the main reason for the strong variation of the heat transfer coefficient with pressure, in the low pressure range. The pressure dependency is due to the water properties, which strongly depend on pressure below ~10 bar. Therefore, improvements of the condensation models in the laminar film region could be obtained by correcting for the pressure dependence of the heat transfer coefficient (and/or with the estimation of the film thickness).

This work found a strong interest from the TRACE code developers and will form a PSI in-kind contribution to CAMP.

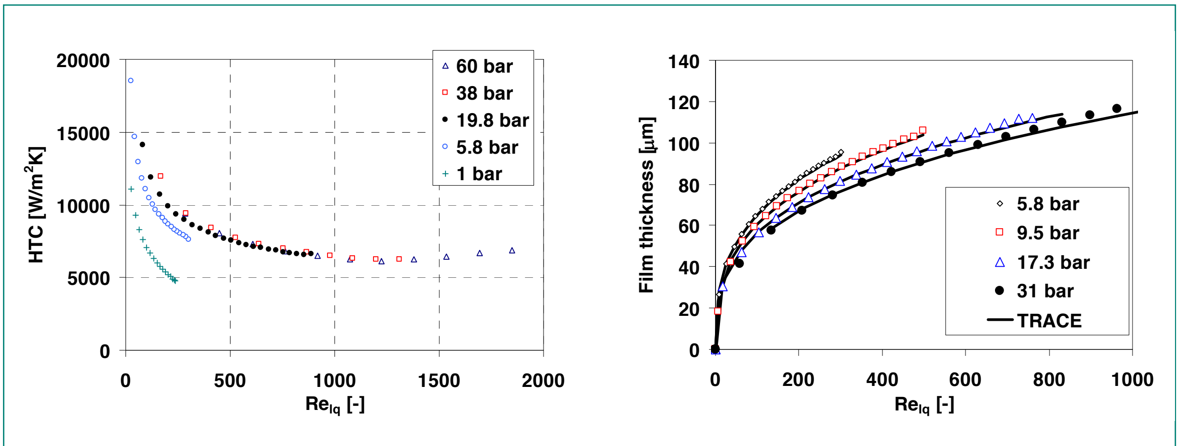

Fig. 10: Left: heat transfer coefficients as calculated by TRACE. Right: film thickness as calculated by trace (solid lines) and Nusselt theory.

A New Convective Boiling Heat Transfer Correlation

The trend towards best-estimate (BE) methodologies has been accompanied in several BE thermal-hydraulic codes, such as the latest USNRC-sponsored code TRACE, by a relinquishing of the conventional body of pre-CHF (critical heat flux) correlations dedicated to specific flow regimes and operating conditions, for a single correlation developed by Chen for saturated convective boiling. Furthermore, this two-component correlation, expressed as $h_{TP} = h_{conv} + h_{boil}$, has also been split to determine the so-called «steaming rate» – a crucial quantity in mechanistic (subcooled) void modelling.

After identifying the inadequacy of splitting Chen's correlation [19] and the root cause for the correlation deteriorating predictive capability when used beyond its developmental database [20],[21], an attempt has been made to develop a new correlation, stemming from the new appraisal of boiling heat transfer in convective flow and it suppression.

It was shown[20],[21] that for the Chen correlation, the predicted-to-measured heat transfer coefficient ratio reached a value (P/M)=2.9 at a relatively modest wall superheat, when the typical uncertainty associated with this correlation is generally considered to be about +/- 20% (i.e., twice the value obtained for the correlation developmental database). The Chen correlation was cast as,

$h_{TP} = F\, h_{D.B.} + S\, h_{F.Z.}$

where F and S are purely empirical flow parameters representing convection enhancement and boiling suppression, respectively. The basic terms $h_{D.B.}$ and $h_{F.Z.}$ represent the Dittus-Boelter and Forster-Zuber convective single-phase and pool boiling heat transfer coefficients, respectively.

Through a separate-effect approach used in the current work, functional relationships were identified and a new empirical relationship for the boiling component was developed as,

$h_{boil} = a(\Delta T_{sat} - \Delta T_{sat}^*)^n$

where the leading coefficient a, the wall superheat offset ΔT_{sat}^*, and the exponent n have been empirically determined. It could be worth noting that while most (if not all) pre-CHF two-phase heat transfer correlations do not include a wall temperature offset, the formulation is consistent with basic theories of vapour bubble nucleation.

The F factor has also been modified to be «calibrated» on a median pressure of 7 MPa – while the Chen database was developed from «low» pressures (<3 MPa).

An example of the results obtained is shown in Fig. 11. In essence, as long as the wall superheats remain relatively low, the Chen correlation performed as expected. This is ensured only by applying the correlation as recommended by its author, i.e., for the annular flow regime.

One can see that the trend to underestimate the heat transfer coefficient (or overestimate the wall superheat) as the Boiling number B_o increases (B_o being proportional to the wall heat flux) can already be seen, in the case of the Chen correlation for «low-pressure» conditions, while the new correlation prediction remains quite close to an (M/P)~1 (the dashed lines representing the +/- 20% bounds). The difference in the correlations predictive performances amplifies under «high-pressure» conditions (subcooled boiling at 13.8 MPa).

Simulation of Pressure Wave Propagation Using the TRACE Code

A variety of transients can lead to rapid and large local pressure changes that propagate through the hydraulic system, e.g. due to the fast closure of the turbine inlet valves or of the main steam isolation valves in BWRs, the propagation of pressure waves under hypothetical RIA conditions and the influence on BWR reactor internals against water hammer [22], and the expansion (depressurization) wave that forms after a Large Break LOCA [23].

We have recently analyzed the capability of the two state-of-the-art BE codes TRACE and RE-LAP5 with two experimental data sets from two-phase water hammer

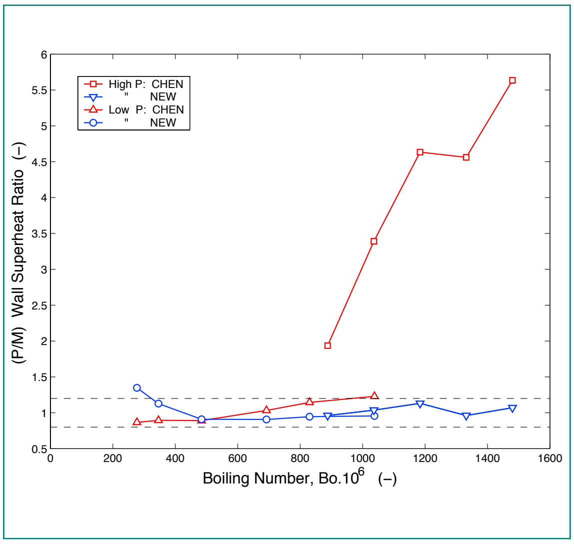

Fig. 11: Wall superheat predictions at «high» and «low» pressure under subcooled boiling conditions.

experiments at the Fraunhofer UMSICHT PPP test loop [24]. Both codes were able to model the overall behaviour of this water hammer, although code improvements were necessary [25]. The validation of the fast pressure wave propagation with the speed of sound along the pipe could only be performed in a semi-quantitative manner, because the uncertainties of modeling important two-phase effects, e.g. interfacial mass and heat transfer, as well as important fluid-structure interaction (FSI) phenomena not considered by the two codes.

The TRACE code has been assessed for linear pressure wave propagation in one- and two-dimensional cavities, i.e. a pipe or a slab, driven by a one-sided pressure boundary condition and filled with liquid water against analytical solutions. Three test cases have been studied: one-dimensional (1D) standing waves, 1D traveling pressure pulses, and two-dimensional (2D) standing waves. Standing pressure waves develop with a harmonic excitation function in the one-dimensional case and are analyzed in a «short» pipe [26]. With respect to the pressure

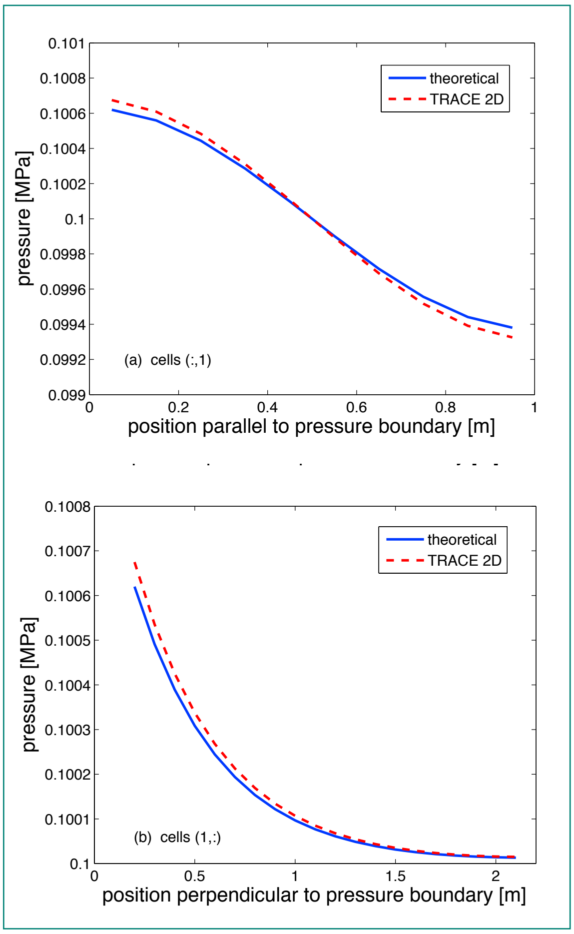

Fig. 12: Comparison of TRACE results with the theoretical ones of snapshots (a) parallel and (b) perpendicular to the pressure boundary at time of maximum pressure. Considered is a 2D standing wave with 500 Hz excitation frequency.

maxima, a generally very good agreement between the TRACE results and the analytical solution was found. At the resonance frequencies, where already very small damping has large influence, the code is tested to the extreme and shows that enforcing very small time step sizes is crucial for good results. Also for the non-linear standing waves when large amplitudes are encountered (in close neighborhood of the resonances) where the analytical linear solution diverges, TRACE yields physically consistent behaviour as the pressure amplitudes are limited by the generation of small amounts of vapor and pressure plateaus are reached.

When Gaussian shape pulses instead of harmonic boundary conditions are applied one-dimensional pressure waves are injected and theoretically propagate undisturbed through the pipe. The changes of pulse amplitude and shape are only due to numerical effects. The maximum amplitude of the pulse slightly reduces with the traveling length of the pulse, while the leading and trailing fronts behave slightly differently due to the asymmetry in the code numerical scheme. Similar to the standing waves, the accuracy of the traveling pulse solution calculated by TRACE is negatively affected for very narrow pulses with sharp fronts when the time steps are too large, while the effects of the spatial discretization are rather minor.

For 2D standing pressure waves in a slab, as before for the 1D standing waves the fluid in the cavity is harmonically driven in time, but in addition parallel to the driving boundary (x-direction) also a cosine-shape has been considered. The comparison with the analytical standing wave solution of the 2D linear wave equation shows overall good agreement of the TRACE results with the analytical solution. The frequency dependence has been analyzed and standing waves with up to three wave nodes have been considered. For low to medium frequencies the wave propagation perpendicular to the driven boundary is damped. This «skin effect» (see Fig. 12b), which does not exist in 1D wave propagation, and the transition to a harmonic shape of the wave perpendicular to the pressure boundary, with up to three wave nodes are very well represented in the TRACE calculations as also the rapid reduction of the wavelength perpendicular to the pressure boundary for yet higher frequencies. At the resonances the standing waves are considerably more dispersed/damped than in the 1D case. This can be understood by the geometrical set-up: The TRACE VESSEL component representing the slab, due to code restrictions to represent this theoretical case, is connected via TRACE PIPE components to the TRACE BREAK com-

ponents representing the pressure boundary conditions. Thus the wave propagation in the narrow region adjacent to the pressure boundary condition is one-dimensional only, while wave propagation in the TRACE VESSEL component is 2D as intended. Like in the 1D cases, reducing the numerical dispersion by refining the time-step size can compensate for these gometrical effects, while refining the spatial discretisation has only a minor effect.
This study shows that TRACE can accurately handle two-dimensional pressure wave propagation in liquid water using the VESSEL component as well as 1D wave propagation using the PIPE component.

Modelling Boron Dilution Scenarions in KKW Gösgen with CFX

In the context of single-phase mixing applications, computational fluid-dynamic (CFD) codes can be considered to have reached a satisfactory level of maturity for providing the complementary capability to system codes for accurately dealing with multidimensional flows. In the present study, boron dilution transients in the reactor pressure vessel of the Gösgen plant are simulated by means of CFD. The work is performed in the Laboratory of Thermalhydraulics (LTH) and capitalizes on the many years of experience in applying the commercial code CFX, e.g. in the framework of the FLOMIX FP5 EU project.

A nodalization of the Gösgen reactor pressure vessel has been developed, consisting of about 6.5 millions hexahedral cells and 6.7 millions nodes. A porous body formulation has been adopted for the description of the reactor core region and to model the frictional resistance of the perforated cylindrical drum in the lower plenum. Simulations of boron dilution transients were carried out with CFX-10 on 12 processors of the LTH cluster. Both SST and BSL k-ω turbulence models have been used. The results presented here have been obtained with the SST model.
In the simulated scenario, it is assumed that initially all pumps are off. The transient is initiated with the start-up of one of the pumps, leading to the introduction of a 8 m^3 plug of deborated water in the corresponding cold leg. The flow rate in this cold leg rises from 0 to 5329 Kg/s in 17 sec. The transient is run for a period of 20 s.
A snapshot of the streamlines after 12 s transient is reported in Fig. 13. The streamlines are coloured according to the deborated water concentration (1- C_B, with C_B being the boron concentration). The plug of deborated water is injected through the middle leg on the right of Fig. 14. A clear distortion of the streamlines, due to the absence of forced flow in the other cold legs, is visible in the downcomer. Snapshots of the time evolution of the deborated water plug are shown in Fig. 14.

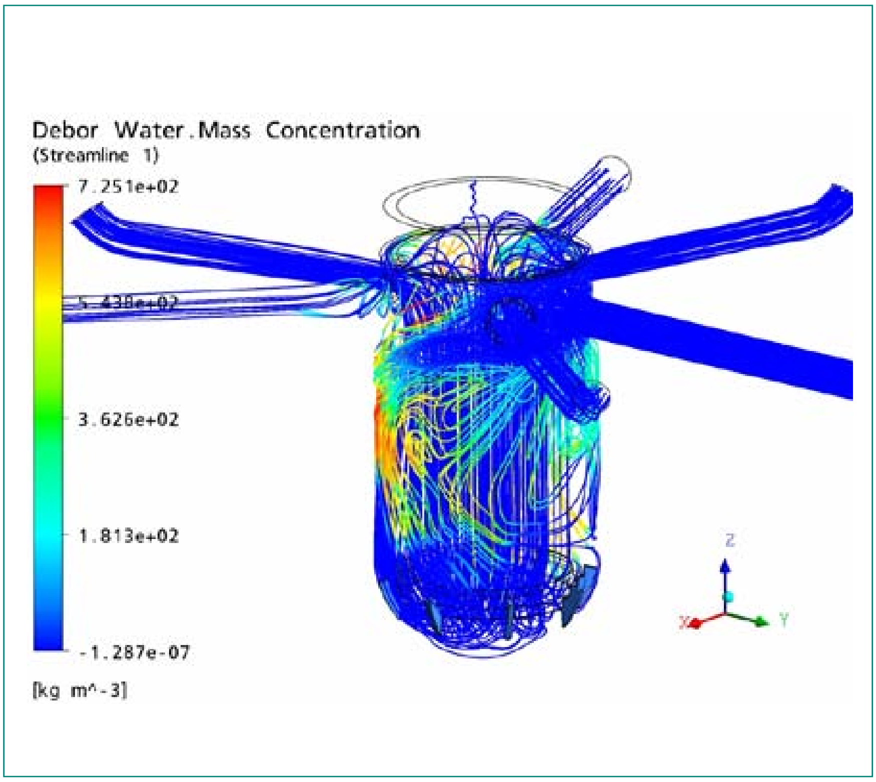

Fig. 13: Streamlines in the RPV at time t = 12 s. the colour indicates the deborated water concentration (1-C_b). frequency.

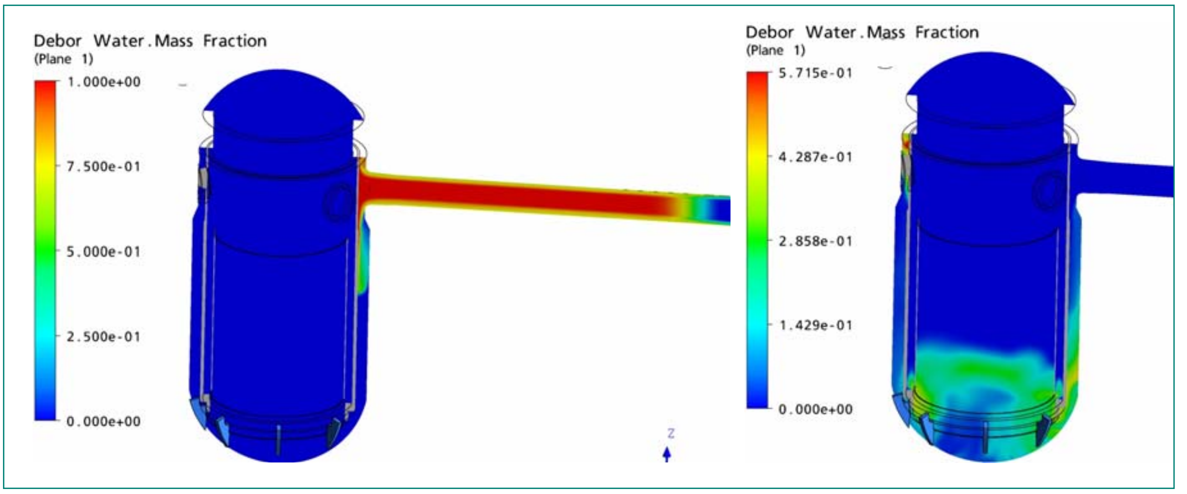

Fig. 14: Deborated water concentration at 8 s (left) and 14 s (right).

Coupling between the System Code TRACE and the CFD Code CFX

As CFD simulations require large computational resources, a coupling between CFD and system codes represents a most worthwhile endeavour for nuclear safety applications, using CFD in regions of the RPV where three-dimensional single-phase flows play an important role in the evolution of a given accident scenario (such as mixing during boron dilution or Main Steam Line Break transients) and relying on the less sophisticated and therefore computationally less demanding flow modeling available with the system codes to simulated the remaining parts of the system.

A coupling between the commercial CFD code ANSYS-CFX [28] and the BE system code TRACE [29] has been realized, using the Parallel Virtual Machines (PVM) software [30] for inter-code communication. While the TRACE source is available at PSI, the access to CFX is performed by making use of the CFX User-FORTRAN interface.

The coupled tool is verified on a simple test problem consisting of a 3 m long straight pipe having a diameter of 5 cm. The pipe is initially filled with stagnant liquid at 10 bar. At time t=0, the pipe end is opened to a lower pressure environment (9.9 bar), causing a sudden acceleration of the fluid in the pipe. As coupled problem, the first 2 m of the pipe are modeled with TRACE, while the last 1 m is modeled with CFX (mesh with 136 000 elements). In Fig. 15, the coupled solution is compared with the results of a TRACE stand-alone simulation. In the test, a flat velocity profile has been imposed at the interface between the TRACE and CFX domains. As a result, the pressure drop in the CFX domain initially deviates from linearity due to the transition to a developed turbulent velocity profile (see Fig. 15, right). The need for the velocity profile to develop causes a larger pressure drop in the coupled solution, compared to the stand-alone TRACE solution. Accordingly, a lower velocity of the fluid is estimated by the coupled tool (see Fig. 15, left).

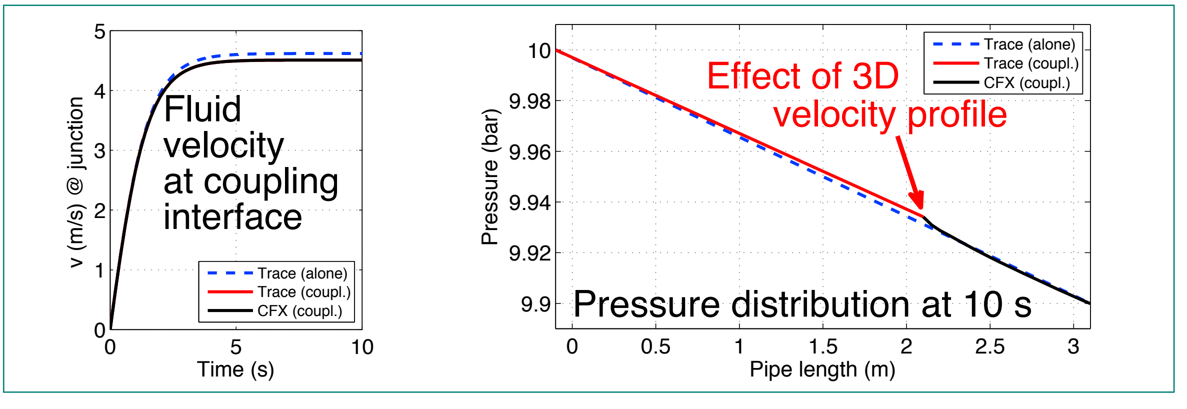

Fig. 15: Comparison between TRACE stand-alone simulation and coupled TRACE-CFX solution: fluid velocity (left) and pressure distribution (right).

Core Analysis of the Swiss Reactors

The continuous development and maintenance of an independent capability to perform 3-D core analyses of the Swiss reactors represents a central task of the STARS project. This is because the models, once developed and qualified against plant data, serve as basis for other types of analyses requiring detailed neutronic/kinetic data e.g. coupled -3D core/plant transient analyses or fast fluence assessment of reactor in-ternals.

Core analysis models must be developed and maintained for all the Swiss plants representing core configurations up to the latest cycle completed. Moreover, as the employed state-of-the-art core analysis methods – currently CASMO-4 (C4) and SIMULATE-3 (S3) – are continuously being upgraded each time a non-negligible re-qualification effort is required at each new code release. Finally, data linkage and interface tools/procedures between the core analysis models and other reactor analysis codes employed within the project need to be continuously maintained and developed. For these reasons, the PSI core management system CMSYS was developed in order to provide a framework to perform the above mentioned-activities in an efficient, accurate and consistent manner for all the Swiss plants, thereby offering a common database environment along with automated and consistent calculation procedures.

During 2007, an update of the core models for the PWR Kernkraftwerk Beznau 1 (KKB1) up to the latest completed Cycle 35 was performed. As part of this work, also an assessment of the CASMO-4E code (C4E) was performed. This code contains several model improvements compared to its predecessor C4, such as e.g. azimuthal Gd depletion, anisotropic scattering, Legendre polynomials for polar angle integration during the 2-D transport calculation, and can use the more advanced JEF-2.2 (AJ2) and ENDF-B/VI-based (E6) multi-group cross-section libraries. Therefore, in addition to the nominal C4/S3 model update, all nuclear lattice models were also analysed with the C4E code using the most recent libraries available, and an additional set of KKB1 core calculations spanning from Cycle 16 to 34 was performed. A comparison of the calculated critical boron concentration for KKB1 Cycle 16 using C4 vs. C4E and different libraries is shown on the left axis of Fig. 16 along with the measured boron concentration. On the right axis, the absolute differences against measurements are shown as function of burnup for the various libraries. Based on the J2 results, it can be seen that C4E improves slightly the core reactivity predictions compared to its predecessor C4. On the other hand, when using the latest AJ2 library, the differences against measurements are more than twice as large as previously. This is even more pronounced when comparing the results with the old E4 library against the new E6 library. With the use of the latter, the RMS differences increase from 16 ppm to 80 ppm, indicating a significant regression of the KKB1 accuracy. Based on this, the new advanced libraries have

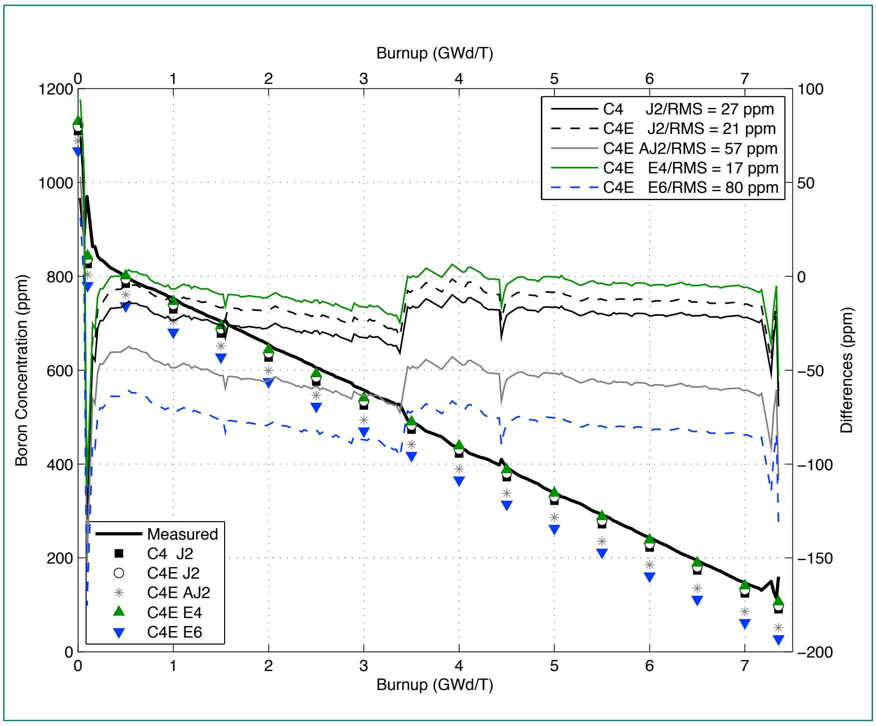

Fig. 16: Comparison of CASMO-4/SIMULATE-3 Critical Boron Concentration against Measurements for KKB1 Cycle 16 (left axis), differences (RMS) between calculated and measured boron data (right axis).

not been implemented for the CMSYS KKB1 analyses and all model updates performed during 2007 remain based on the older libraries E4 and J2 which give ex-cellent agreement with plant data.

Second, core analysis models for the KKB2 plant were for the first time implemented in CMSYS. The model development and qualification was performed for a total of 21 operating cycle starting from Cycle 12 to Cycle 32. The following results, in terms of comparing calculated against measured 3-D reaction rates, were obtained, using the advanced libraries AJ2 and E6:

- The accuracy in radial distributions is usually very good (~ 2%) while the differences in terms of the maximum nodal values are around ~ 5%-7% explaining in turn, an agreement around ~ 3% -5% in axial distributions.
- The accuracy is usually deteriorated towards EOC.
- Both libraries give approximately the same results although as can be seen for Cycle 32, a slightly better accuracy is obtained with AJ2 compared to the E6.

Related to the last observation, it must be noted that although the two advanced libraries were found to yield a regression in terms of the core reactivity accuracy (see Fig. 16), quite satisfactory overall agreement of the predicted 3-D power distributions was obtained. This seems to indicate that the use of the new libraries has a rather limited impact on the prediction of power distributions.

Finally, the update of the KKM model for Cycles 29-33 is currently being performed.

KKL Start-Up Analyses at EOC20 with SIMULATE-3K

At KKL End-of-Cycle 20 (EOC20), the reactor was shutdown for maintenance and re-started some 20 hours later. During start-up, the coolant heating rate exceeded the limit because of a too rapid heat insertion caused by the combination of a positive moderator temperature coefficient (MTC) and a non-active Residual Heat Removal system (RHR). This resulted in a rapid increase of the thermal power which was however reversed through control rod maneuvers and void formation. Hence, the primary physical reasons for this transient are related to the coolant temperature feedback mechanism combined with the heating power.

A positive MTC can be expected for conditions at EOC. Moreover, independently of the cycle burnup, two other major factors could contribute to a less negative MTC. First, the time between shut-down and reactor start-up will affect the Xenon concentration. With a large Xenon content, noting that the peak concentration is reached some 8 hours after shutdown, the MTC will become less negative because criticality will be achieved later in the withdrawal sequence. Secondly and perhaps most importantly since relevant for the KKL Cycle 20 core, is that an increased fraction of fuel assemblies with partial length rods will render the MTC less negative particularly in the upper core region due to the increased moderator-to-fuel ratio. Noting that at start-up conditions, the axial power is strongly shifted towards the top of the core, the core behaviour will be principally affected by the MTC magnitude in that core region. Concerning the heating power, it is at such conditions mainly due to decay heat power and, therefore, a correct estimate of the decay heat is required. The power increase observed during the KKL startup event is mainly determined by the control rod withdrawal and further enhanced, through the positive MTC, by the decay heat power and the RHR system that was switched off at the time when the first power peak occurred shortly after reaching criticality.

A S3K model has been set up for KKL EOC20 and steady-state calculations were performed at several operating conditions in order to verify the accuracy against the corresponding SIMULATE-3 (S3) models. As the S3K model for a given plant/cycle and operating condition is set up based on the restart data from S3, it was important to verify if the heterogeneous KKL C20 core is properly modeled with adequate initial conditions and that the correct data from S3 is employed. The comparisons were performed at EOC20 for both Cold-Zero-Power (CZP) and Hot-Full-Power (HFP) conditions and the results are shown in Table 2. At HFP, the main observation is that S3K calculates a lower k_{eff} and this is directly caused by the larger fuel temperatures compared to S3. This is confirmed by the CZP results which show identical k_{eff} between both codes because in this case, the fuel temperature is uniform over the core. The reason for the different fuel temperatures at HFP is that while S3 uses a simple interpolation procedure to determine the fuel temperature as pre-calculated function of local burnup and power density, S3K on the other hand uses an explicit (transient) fuel heat conduction model. A similar observation was made earlier while comparing S3 and CORETRAN that also estimate the fuel temperatures with an explicit heat conduction model.

Concerning the agreement in 3-D power distributions, it is seen to be very satisfactory at HFP although, as expected, better at CZP. At HFP, the small differences are probably mainly due to small local void differences, noting that the core-average void fraction is however identical which in turn confirms the similarity in thermal-hydraulic models between the codes. The differences in fuel temperatures have practically no impact on the agreement in 3-D power distribution because, as was seen with CORETRAN, larger temperatures are predicted in a consistent manner over the entire core.

To conclude, this steady-state comparison indicates that the overall agreement between S3K and S3 is very satisfactory and this is particularly valid at CZP conditions. Noting that the transient of interest is to be analyzed at low power conditions, this comparison hence provides confidence in the developed S3K model.

Since the decay power is of high relevance for the proper prediction of this transient, corresponding verification of S3K was performed as well, with very satisfactory result. The other important element concerns the Xenon concentration that is passed on to S3K from S3 before the transient. Hence, S3 was successfully checked in this respect by performing a Xenon transient.

After this testing, the analysis of the start-up sequence has begun with S3K and is still in progress.

Cross-Section and Thermal-ydraulic Modelling Effects on a PWR MSLB Analysis

The purpose of the study was to perform an assessment, as detailed as possible, of the impact that the approximations and/or simplifications to the X-S formalism and to the modelling of the flow mixing upstream and downstream of the core can have on the results of a MSLB analysis. To that aim, a 3-D core model using the CORETRAN and RETRAN-3D codes was developed for a Swiss PWR-core at EOC whose shutdown margin was reduced down to 1000 pcm through artificial modifications of the X-S interpolation model of the codes, in order to obtain a power excursion after the insertion of the control rods. The homogenized two-group XS libraries were generated using CASMO4/SIMULATE-3 along with a set of interface tools for the conversion to the appropriate format for CORETRAN (and RETRAN-3D). In this context, it was also deemed necessary to assess the applicability of the SIMULATE-3K code recently implemented in the STARS code system and therefore, a SIMULATE-3K core model was also developed.

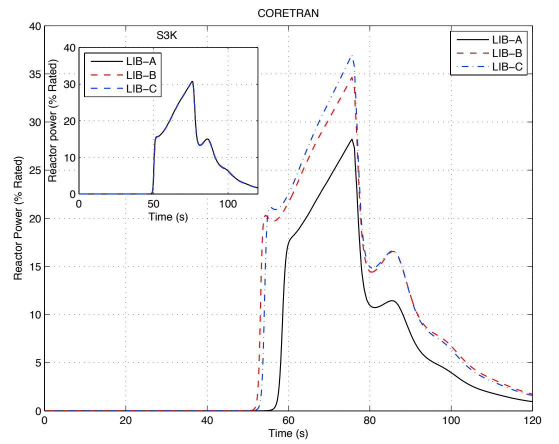

Fig. 17: CORETRAN and S3K simulations of the core power evolution during the MSLB for different boron and moderator density reference points in the X-S library.

		Hot Full Power power 3601.4 MW_{th}, core exit pressure 73.9 bar, core flow 10099 kg/s, control rod inserted 2 %			**Cold Zero Power** power 0 MW_{th}, core exit pressure 1.43 bar, core flow 3345 kg/s, control rod inserted 100 %		
		S3	Difference		S3	Difference	
			S3K-S3	(S3K-S3)/S3		S3K-S3	(S3K-S3)/S3
k_{eff} (-)		1.00814	-244 pcm		0.94624	0	
$T_{Fuel,ave}$ (K)		746.3	94.4		319.4	0	
$\alpha_{Core,ave}$ (%)		0.47	0		-0.04	0	
Power Distribution	Axial Nodal Power Peaking Factor (-)	1.148		0.5 %	2.573		0.0 %
	Radial Nodal Power Peaking Factor (-)	1.517		-0.5 %	1.871		0.1 %
	Nodal Power Peaking Factor (-)	1.802		-1.8 %	4.900		0.1 %
	Pin Power Peaking Factor (-)	2.069		0.1 %	8.228		0.1 %

Table 1 Comparison of steady-state results between S3 and S3Kat EOC20, cycle burnup 8.591 GWd/t.

To start, the 3-D core transient analysis was performed with both CORETRAN and SIMULATE-3K using specified T/H boundary conditions at the core inlet and outlet. It was found that the main differences between the two codes in terms of predicted transient power were mainly due to a smaller moderator temperature coefficient (MTC) with CORETRAN. Although both codes employed X-S libraries based on the same set of homogenised 2-group cross sections (prepared with CASMO-4), it was shown that the CORETRAN X-S model lacks an adequate treatment of coupled feedback effects, viz. the interdependency of the boron and moderator density feedback. This can lead to an under- or over- prediction of the MTC depending on the initial operating conditions assumed for the transient and the reference conditions employed during the preparation of the XS. For the selected conditions analysed here, i.e. EOC at HZP, the CORETRAN MTC was hence found to be underpredicted. To illustrate this, three different XS libraries (LIB-A, LIB-B, and LIB-C) were prepared, using different reference boron concentration and moderator temperature/density, and thereafter applied for the MSLB analyses with CORETRAN and SIMULATE-3K. As shown in Fig. 17, while S3K predicts the same transient reactor power for all three cases, non-negligible effects are seen in the CORETRAN results. An important outcome of this investigation is hence that the specific formulation of the nuclear XS parametrization may contribute considerably to the calculation uncertainty of a MSLB analysis.

As a second step, a RETRAN-3D full core/plant system model was set-up using a XS library selected appropriately based on the above study, and considerable efforts were carried out to study the T/H related effects on the MSLB analysis. Principally, the effects of coolant mixing in the lower plenum as well as the influence of the core T/H channel lumping scheme, usually employed for coupled best-estimate core/system analyses, was in this context performed.

Criticality Safety Analyses with State-of-the-Art Calculational Methods

While commercial (and research) reactors are designed to reach criticality and sustain the nuclear chain reactions over an extended period of time in order to reliably produce electricity, it is an imperative that criticality of fresh or spent fuel configurations is avoided outside of reactors. Therefore, for systems such as compact storage pools and transport casks of (spent) fuel assemblies and for all processes in the reprocessing industry, criticality safety analyses (CSA) are performed to assess their level of subcriticality under both normal and all credible abnormal conditions. Nowadays, most of the CSA work is performed by evaluating the effective neutron multiplication factor k_{eff} of the system applying a advanced neutron transport methods after thorough validation against measurements of a suitable set of critical experiments.

MCNPX is a general state-of-the-art Monte Carlo neutral-particle transport code that has been developed at the Los Alamos National Laboratory [32]. Its use offers important advantages over other codes such as the capability to model complex three-dimensional configurations and the usage of continuous-energy (or point-wise) cross section libraries. Among the evaluated nuclear data libraries available, two have recently been upda-

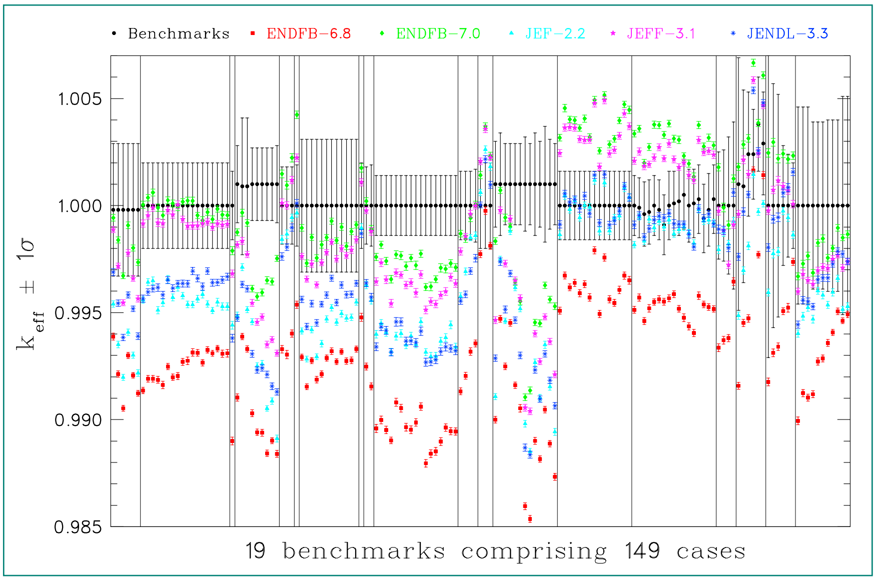

Fig. 18 Calculated and benchmark k_{eff} values with error bars representing one standard deviation (the vertical lines separate the groups of cases from the 19 benchmark configurations).

ted: the «European» JEFF-3.1 [33] in June 2006 and the «US» ENDF/B-7.0 [34] in December 2006. Together with their predecessors JEF-2.2 and ENDF/B-6.8, and with the «Japanese» library JENDL-3.3 [35], they were used in conjunction with MCNPX (version 2.5.0) to perform validation analysis based on a set of benchmarks from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [36] (ICSBEP). The benchmarks were selected based on their similarity to the designs of today's LWR compact storage pools and transport casks. The benchmark suite comprises a total of 149 different cases from 19 benchmarks [37] belonging to the category of thermal compound systems with low enriched uranium (LCT) and MOX fuel (MCT).

The effective multiplication factors (k_{eff}^{calc}) calculated with MCNPX-2.5.0 and the five continuous-energy nuclear data libraries are compared to the k_{eff}^{exp} values of the 19 benchmark configurations, in Fig. 18 (the vertical lines separate the groups of cases from the 19 different benchmark configurations). The error bars represent the uncertainties, which match one standard deviation with respect to the calculations. By using a large number of active neutron histories, the MCNPX standard deviations σ^{MC} can be kept rather small. As no confidence level is given for the benchmark uncertainties σ^{exp} of most of the ICSBEP evaluations, we as-sumed them to represent one standard deviation (similar to the few cases where the confidence level was explicitly specified).

As not all of the measured eigenvalues $k_{eff,i}^{exp}$ (i=1,...149) are exactly equal to 1.000, the calculated $k_{eff,i}^{calc}$ have been normalized to the experimental values: $k_{c,i}= k_{eff,i}^{calc}/k_{eff,i}^{exp}$. The results of a statistical evaluation of the normalized eigenvalues $k_{c,i}$ are summarized in Table 2 where the weighted average of the sample is denoted by $<k_c>$ and its standard deviation by σ'. The weights w_i were set to $1/\sigma_i^2$, where σ_i is the uncertainty of a single observation and incorporates the uncertainties from both the benchmark (σ_i^{exp}) and the calculation (σ_i^{MC}). Furthermore the minima and the maxima of the normalized eigenvalues are listed with their errors σ_i. Finally the sample standard deviation *s* is given and also the bias, i.e., the systematic difference between calculated results and experimental data, defined by b = 1.0 – $<k_c>$, is stated in the last column in units of per cent mile (pcm=10^{-5}).

In order to assess the range of applicability of MCNPX-2.5.0 in combination with the libraries and to get indications of possible deficiencies, the $k_{eff}^{calc}/k_{eff}^{exp}$ samples were analysed to detect possible trends with re-spect to experimental design parameters and spectrum related observables, but none were found.

While the weighted average $<k_c>$ of the normalized eigenvalues turns out to be slightly smaller than unity for the (somewhat) older libraries ENDF/B-6.8, JEF-2.2 and JENDL-3.3, the latest ENDF/B-7.0 and JEFF-3.1 libraries break this trend and produce very small biases of just –10 pcm and –100 pcm, respectively (cf. Table 1). Especially the largest relative error found between all measured and calculated k_{eff}–values amounts to just ~1.0% for both ENDF/B-7.0 and JEFF-3.1. Hence ENDF/B-7.0 and JEFF-3.1 are considered excellent cross section libraries that (in com-bination with MCNPX-2.5.0) yield precise k_{eff}-predictions of LCT- and MCT-systems.

Core Physics and Multi-Physics Activities within the EU 6th Framework Integrated Project NURESIM

The STARS project is participating in two sub-projects of the EU 6th framework integrated project NURESIM: «Core Physics» (SP1) and «Multi-Physics» (SP3). The former aims at the development and qualification of advanced neutronic solvers for the NURESIM platform while the latter has the integration of advanced coupling techniques for the analysis of LWR cores as primary objective. The following provides and overview of the NURESIM-related activities carried out at PSI during 2007 with regards to these sub-projects.

Cross Section Library	$\langle k_c \rangle \pm \sigma'$	Min $k_{c,i} \pm \sigma_i$	Max $k_{c,i} \pm \sigma_i$	standard dev. *s*	Bias *b* [pcm]
ENDF/B-6.8	0.9927±0.0002	0.9844±0.0019	0.9998±0.0020	0.0029	-730
ENDF/B-7.0	0.9999±0.0002	0.9901±0.0022	1.0052±0.0016	0.0030	-10
JEF-2.2	0.9962±0.0002	0.9875±0.0019	1.0026±0.0020	0.0031	-380
JEFF-3.1	0.9990±0.0002	0.9894±0.0019	1.0049±0.0016	0.0032	-100
JENDL-3.3	0.9965±0.0002	0.9874±0.0019	1.0030±0.0021	0.0031	-350

Table 2: Results from the statistical evaluation of the 149 benchmark cases from 19 experiments.

In «SP1 Core Physics», STARS is participating in the qualification of the CEA advanced deterministic solvers APOLLO-2 and CRONOS for the NURESIM PWR Core Physics Numerical benchmarks. As a first step, an APOLLO-2 computational scheme for cell calculations was developed at PSI [38]. The developed scheme uses a JEF-2-2 based 172 neutron group library, employs a 10-ring radial discretization of the fuel pellet, performs self-shielding calculations for selected isotopes at each burnup step, uses the collision probability method with reflective boundary conditions followed by a critical leakage calculation in fundamental mode for the flux calculations, solves the Bateman depletion equations for well-defined chains of nuclides and optionally, applies a special procedure to achieve Xenon equilibrium from zero burnup in the depletion calculations (as required by the benchmark specifications). To optimise and qualify the scheme, a comparison against the state-of-the-art CASMO-4E code was performed. These results in terms of k_∞ are shown in Fig. 19 for three cell models: one UOX and two MOX cells (MOX-1 and MOX-3).

The main observation is that for all types of fuel pins, the agreement between both codes remains within ± 600 pcm. This can be considered as satisfactory in the context of a code-to-code comparison between two distinct lattice solvers, noting for instance that the variation of CASMO-4E alone when using different cross-section libraries was found to be around ± 500 pcm.

The development and integration of a mesh-to-mesh interpolation tool represents an important contribution to the «Multi-Physics» sub-project. It allows for the geometric coupling between neutronics and thermal-hydraulic solvers that employ different three-dimensional non-regular meshing schemes [39]. The principle consists in identifying for each mesh of the calculation domain in one solver (the target mesh) the corresponding number of meshes in the calculation domain employed by the other solver (the source meshes) with non-zero intersection volume with the target mesh. Thereafter, the transfer of a given parameter from the source mesh to the target mesh is performed using a volumetric weighting procedure.

Secondly, a prototypical high-level Application Programming Interface (API) was developed to act as a standardized layer between the user and the various platform solvers in order to construct complex coupled calculation routes. Principally, a high-level API consists in building a chain of operational blocks, that can be manipulated through a GUI interface or specific PYTHON scripts, and with each block having a specific function e.g. solver initialisation, input specification, coupling procedure specification (e.g. interpolation tool mentioned above), steady-state calculation, time-step specification for the transient analyses. The different blocks are assembled through the high-level API which thereafter handles the interfacing and communication between the selected solvers, provided that each of these has been integrated in the platform with consistent interface protocols i.e. containing the operations called by the high-level API. Finally, the development of so-called common-input data processors was started with the objective that a single input data set is specified serving all solvers integrated into the platform: A fuel pin is defined once, and this specification can be used consistently in thermo-mechanical, neutron transport or diffusion or thermal-hydraulic solvers.

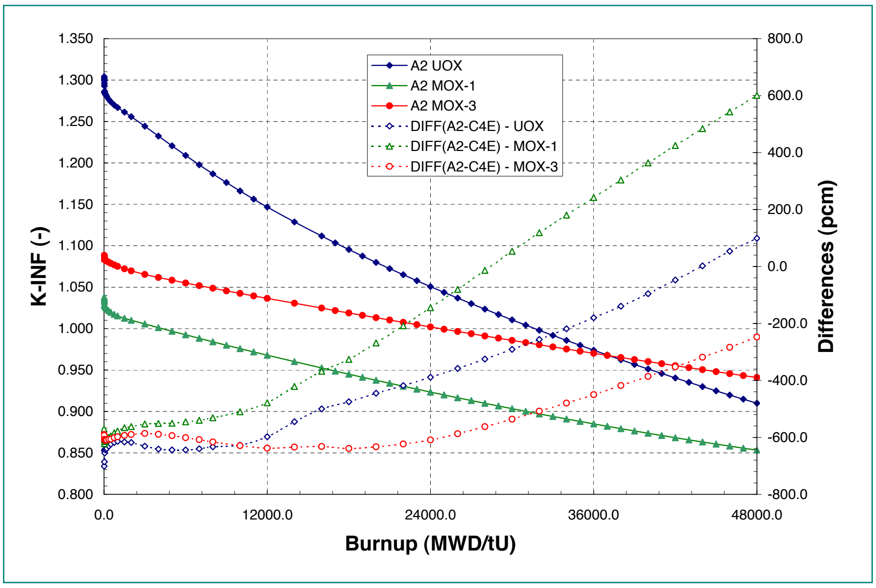

Fig. 19: APOLLO-2 and CASMO-4E results for NURESIM PWR Cell Benchmark.

Investigation of the Event «Fehlerhaftes Aktivieren von SEHR-ADS im KKL vom 6.3.07»

One of the missions of STARS is to develop and continuously maintain a set of simulations tools and methods at the state-of-the-art in order to support the Swiss Safety Authority (HSK) with a sufficient level of expertise in LWR safety. Thus, in March 2007 STARS was requested to provide the HSK with a technical review of the event «Fehlerhaftes Aktivieren von SEHR-ADS» that took place at the Leibstadt power plant on March 6, 2007 that resulted in a reactor trip. This event was initiated by a spurious activation of the Division 51 of the Automatic Depressurization System (ADS). This system in particular commands the full opening of the safety-relief valves belonging to the ADS.

Following the activation of the ADS, the reactor was automatically tripped through the RPV water level signal («Niveau 3»). The reactor trip resulted in the isolation of the steam lines, of the feedwater system (FW) and of the reactor containment building. Few minutes later, as the depleting RPV water level signal reached the «Niveau 2», the recirculation pumps were also tripped. During the depressurization phase the loss of the RPV water inventory through the Safety Relief Valves (SRVs) was compensated through the use of the High Pressure Core Spray (HPCS) injected into the upper plenum and the Reactor Core Isolation Cooling (RCIC) injected into the FW line. Later on, after the ADS relief valves were closed, the RCIC was used to control the water level in the RPV while the SRVs were used to control the reactor pressure. After the water level, the pressure and the temperature in the RPV were sufficiently low and stabilized, the recirculation pumps were started again and the Reactor Heat Removal system (RHR) was then set on the shutdown/startup cooling mode, approximately 5 hours after the beginning of the transient.

During the initial phase of the sequence, i.e. the first 6 minutes of the transient, the RPV experienced a relatively rapid depressurization and consequently, the water level measurements rose to very high levels. This increase was physically the consequence of the swelling of the fluid mixture due to steam flashing below the lower pressure measurement tap (which is used to evaluate the level), which then «lifts« liquid up to the upper regions of the downcomer. One issue was therefore to determine whether the swelling of the water level was sufficient to raise the mixture level close to the steam line intake and therefore to cause liquid spill over to the steam line. A second issue was related to the concern expressed by HSK on the risk of thermally-induced stress on the RPV wall and/or some of the internal structures due to the injection of cold water from different systems (mainly the RCIC and the CRDM cooling) and while the recirculation pumps were not in operation.

A technical review of the event was carried out through the post-analysis of the incident using the different simulation tools of STARS. Thus, using the existing and validated TRAC-BF1 model of the KKL RPV, recirculation lines and steam line systems, an analysis was performed to investigate the behaviour of the water level and examine the possibility of water carry-over into the steam line, and to estimate the evolution of the pressure gradients experienced by the RPV internals and the RPV wall during the blow-down phase of the event. It could be concluded that some liquid was entrained with the steam during the period ~ 60 to 170 seconds but of a magnitude most probably lower that shown in the calculation. A value smaller than that calculated is to be expected since the TRAC-BF1 vessel nodalization is very coarse particularly at the top of the vessel in the region of the FW lines. Fig. 20 (Left) compares the calculated RPV Narrow Range Water Level using TRAC-BF1 with the corresponding plant measurement. One can see in particular how the steam flashing resulting from the low pressure causes the water level to momentarily rise around 50 s and how this level suddenly falls as a result of both the reduction of the FW flow but also through the reduction in power due to the increase of the void in the core. One can also see how the level rises at approximately 190 s as a consequence of flashing that takes place this time inside the FW line and results in a surge of water into the RPV and therefore leads to an increase of the vessel inventory as a large fraction of the FW line is voided.

This part of the analysis benefited from a very detailed and validated TRACE model of the KKL FW system [40], in order to estimate the amount of water injected into the RPV during the depressurization of the system. One of the difficulties was to appropriately predict the FW mass flow rate during the time period when steam flashing was taking place in the line. This is shown in Fig. 20 (right) which compares the measured FW mass flow with the flows predicted with different versions of TRACE. As can be seen, a good prediction of the mass flow could be obtained before and immediately after the trip of the FW pumps, thus showing the good pump head prediction during the rundown phase following the trip. However, the predicted mass flow rate was much lower than the measurement during the steam

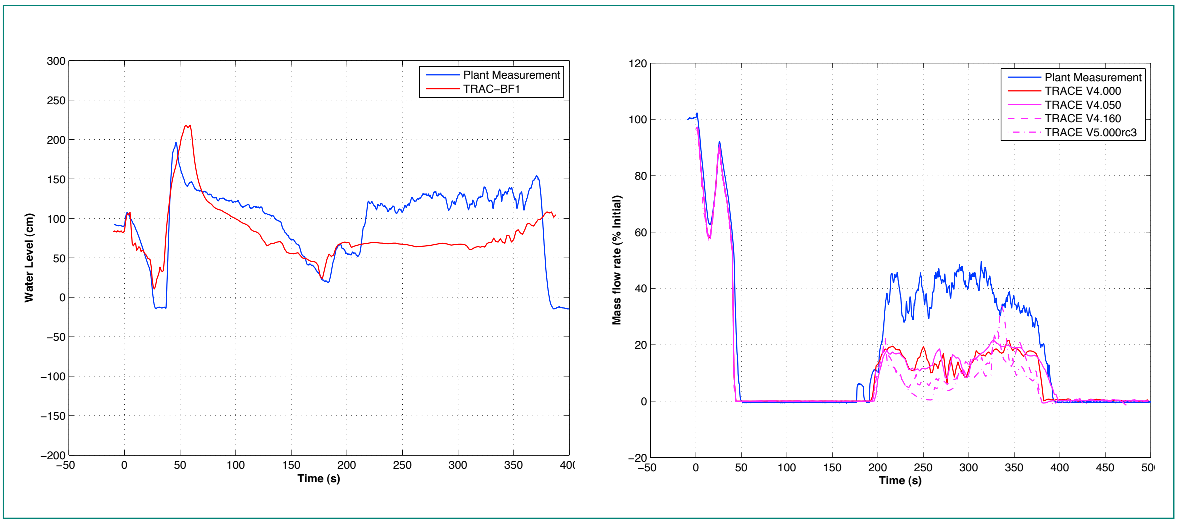

Fig. 20: Simulations of the KKL Event «Fehlerhaftes Aktivieren von SEHR-ADS». LEFT: Narrow Range Water Level in the RPV calculated with TRAC-BF1. RIGHT: FW mass flow rate calculated with different versions of TRACE.

flashing sequence, which started around 190 s. The overall amount of fluid mass injected in the RPV during the steam flashing sequence was approximately 65 tons in the TRACE calculations, whereas the measurement indicated more than 140 tons of fluid. Such a difference could significantly affect the evolution of the water mass and energy in the RPV during the pressure blow down. The reason for this discrepancy could not be investigated in details, given the limited time available.

National Cooperation

Beside the active PSI-internal collaboration within the department of Nuclear Energy and Safety (NES), STARS also enjoys substantial funding support from HSK and to a lesser degree from swissnuclear. The latter support the work based on higher-order neutronic methods, e.g. Monte Carlo analysis (ANSR), while HSK is supporting the remainder of the project.

Two doctoral students registered at EPFL's newly created Doctoral Programme in Energy are working on topics related to STARS: One student has completed his research on uncertainty analysis and its application to nuclear safety calculational methods. The second student works on the development of a new fission gas model to investigate the role of different phenomena related to high burnup and will finish early in 2008. Both PhD-studies are performed under the supervision of the head of the Laboratory for Reactor Physics and Systems Behaviour, who is professor at EPFL, with significant support from STARS experts.

International Cooperation

During 2006, STARS has participated in collaborations with the following institutions:

- Studsvik / Scandpower, Sweden / Norway / USA, which provides maintenance and support for their neutronic codes *CASMO-4, SIMULATE-3, SIMULATE-3K*.
- Electric Power Research Institute (EPRI), Palo Alto, CA, USA in relation to (a) the maintenance of the system analysis code RETRAN-3D (Computer & Simulation Inc., Idaho Falls, ID, USA), and (b) the assessment, maintenance and further development of the fuel behaviour code *FALCON* (Anatech Inc., San Diego, CA, USA).
- US-NRC through the CAMP-agreement, for TRACE assessment and development.

In the context of uncertainty analysis applied to thermal-hydraulic calculations, STARS continues to participate in the CSNI-OECD sponsored **BEMUSE** Programme.

The participation in an IAEA **CRP on uncertainty** was inactive because both of the involved collaborators left PSI. STARS is now considering to quit this project as it is largely paralleling the much more active efforts in **BEMUSE**.

The **NSC benchmark** on **Uncertainty analysis in the coupled multi-physics and multi-scale LWR modeling (UAM)** has not yet been offered for participation. One member of STARS has been elected as member of the UAM scientific board.

The work of the CSNI task group on the **Action Plan for Safety Margin** (SMAP) was completed early in 2007. It is intended that STARS continues to be involved with the follow-up activity LOSMA.

STARS also participated in several international research programs:

In the framework of the collaboration with the **OECD HALDEN** Project, a joint publication on the preliminary analysis of IFA-650.4 using TRACE and FALCON was the main achievement of 2006.

The **OECD CABRI-Waterloop** Project first provides STARS access to the CABRI RIA-experiments with UO2-fuel and the SCANAIR code. Technical exchange on the modeling of the different experiments is ongoing. During 2007, no new experimental data set became available.

The Japanese **ALPS** program provides STARS experimental data on the RIA behaviour of BWR fuel.

The **OECD PKL** and **ROSA-V** projects both provide very valuable data for the TRACE assessment. One collaborator is member of the ROSA-V project management board.

The collaboration with the German research center Rossendorf (**FZD**) was inactive during 2007, but is expected to be reactivated in the near future.

The 6th FW EU Integrated Project **NURESIM** continued during 2007 with contributions to the two subprojects «Core-Physics» and «Multi-Physics», the latter also being coordinated.

Assessment 2007 and Perspectives for 2008

Most of the goals specified for 2007 could be reached, and some work not foreseen at the time of the writing of the last report was successfully undertaken.

With the analytical support for the definition of the planned Halden LOCA-experiment IFA-650.7 using BWR fuel, the STARS project demonstrated its capability to assess fuel behaviour during LOCA, thereby at the same time shedding some light on the outcome of previous experiments. Especially, how representative experiment IFA-650.4 with its strong axial fuel relocation is for power plant conditions remains open. Hence, the unplanned design work compensated in part for the planned further modeling studies on axial fuel relocation during LOCA.

The further development of the GRSW-A fission gas model implemented in FALCON has been performed in the framework of the analysis of the first RIA-experiment using BWR-fuel from KKL (LPS-programme). As the next experiments form CABRI-WL will only become available in 2010, our work was restricted to the re-analysis of selected previous experiments using the latest version of FALCON, thereby introducing a new collaborator into this interesting topic.

The assessment of TRACE continued again with considerable effort, focusing on PWR-related problems. During the second half of 2007, the migration of the available BWR-models for the previous TRAC-BF1 code has been worked on, as TRACE has now been officially released with a frozen version. Also, work towards assessing the generalized radiation heat transfer model of TRACE using the available data from the Halden LOCA experiments could be pursued during the last quarter. Unfortunately, two collaborators heavily involved in the TRACE-work left PSI (one of them has been elected for the nuclear engineering chair at the Technical University of Munich), and the related work slowed down. New collaborators were hired, and the work on ROSA was very successfully resumed during the last quarter of 2007.

Nevertheless, further good progress was achieved in the area of uncertainty research in the field of system thermal-hydraulics: The participation BEMUSE phase-IV was completed with very good success. However, due to lack of resources, the uncertainty related work in fuel modeling did not progress during 2007. The PhD-thesis was successfully completed applying the developed methodology (objective estimation of the probability density function of code parameters determining the void prediction based on a clustering technique) to a BWR turbine trip. The collaborator has taken a Post-Doc position at Chalmers University of Technology in Sweden.

The participation in the IAEA CRP came to a halt with the leaving of the involved collaborators and will not be pursued any further. The participation in BEMUSE and the UAM benchmark will provide adequate coverage of this topic. Participation in the latter is delayed, but should being rather soon as the UAM benchmark specifications will be published soon.

The work on single-phase mixing problems in NPP geometries using CFD suffered from the retirement of the lead analyst in LTH and slowed down a bit. Yet, interesting studies in relation to the modeling assumption describing turbulence (in the framework of a lower plenum model of the KKG NPP) are now in the stage of the final analysis. The PhD with the goal of developing a coupling between a CFD and a system code has already developed the proof-in-principle, and a small experiment has been designed in collaboration with Laboratory for Thermal Hydraulics to validate the computational approach.

The work to quantitatively assess the simulation capabilities of TRACE for (de-)pressurization waves following LOCA has reached a first stage by comparing TRACE results to simple wave propagation problems that lend to analytical solutions. If this work should be refocused onto the application of CFD-methods yet needs to be decided.

The work on developing a new pre-CHF Heat Transfer correlation is in the publication phase and offers a better prediction of heat transfer in two-phase conditions.

The participation in NURESIM continued. Considerable work (beside the one reported above) was spent on developing the new proposal (NURESP) in which PSI again would coordinate multi-phyiscs. Unfortunately, this proposal did not receive funding from the EU.

While the EPR contract between STUK and PSI was finally signed later this year, the work has not yet started. It is expected to start begin of 2008, after the delivery of the first set of design data.

Updating the core models up to the latest cycle operated required significant effort (as reported above). For one plant, this work was subcontracted to a consultant.

During this year, emphasis was given to the assessment and qualification of SIMULATE-3K (S3K) as a stand-alone core dynamics solver for some selected PWR and BWR transients. For that reason, and also due to the lack of resources, the coupling of S3K with the system codes (TRACE and ev. RETRAN-3D) has been postponed to next year.

The Monte-Carlo work related to fast fluence did not proceed to the bio-shield analyses, although an outline of the challenges and requirements in terms of calculation tools has been prepared. One reason is that based on discussions with the Swiss utilities, the work program will during the coming years remain with a strong focus on fast fluence assessment for both PWR's and BWR's.. In that framework, sensitivity studies preparing uncertainty evaluations have also been conducted.

Finally, the fact that STARS personnel was able to produce first preliminary results 24 h after they were called by HSK to analyze a plant event effectively demonstrated the expertise and the adequate project infrastructure. It will be crucial for the further success to develop the young project scientists up to the senior expert level that is mandatory for STARS to continue providing excellent technical support to HSK.

Perspectives for 2008

The projected work for 2008 develops in three main domains:

- Further develop fuel modeling capability:
  - Further develop fission gas models and perform necessary validation.
  - Participate in CSNI/WGFS LOCA benchmark with analysis of IFA-650.4 / IFA-650.5.
  - Continue analysis of selected RIA and LOCA experiments from the ALPS program.
  - Continue analysis of SCIP ramp tests.
  - Establish framework for statistical fuel analysis.

- System behaviour modeling:
  - Continue migration of TRAC-BF1 BWR models and RELAP5 PWR models to TRACE, and perform testing using available plant transients.
  - Continue TRACE assessment with further analysis of ROSA experiments and in relation to condensation modeling.
  - Develop EPR models for TRACE and CFD.
  - Complete CFD-work for KKG boron dilution demonstration transient.
  - Continue with coupling PhD study.
  - Continue participation in BEMUSE-V (uncertainty evaluation) of PWR LB-LOCA in Zion Plant.
  - Initiate work on dynamic event trees and start establishing the respective tools based on TRACE (NES Seed Action 2006).

- Core behaviour modeling
  - Update Swiss core models (CMSYS).
  - Initiate migration to CASMO-5 / SIMULATE-4.
  - Couple SIMULATE-3K to TRACE / RETRAN-3D.
  - Within NURESIM, continue exploration of APOLLO-2 for application to core analysis at nodal and pin-level and perform the work necessary to achieve the BWR situation target (turbine trip transient at the core level).
  - Participate in first exercise of NSC/UAM benchmark (neutronic uncertainty in view of coupled analysis).

It is understood that a few work items might be reconsidered during 2008 in light new information becoming available.

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