STARS

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

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ABSTRACT

STARS continues to actively develop uncertainty evaluation for best-estimate applications: The PhD-thesis on objectively deriving uncertainty characteristics of important model parameters (e.g. void, CHF) is close to completion. Work on the application of the uncertainty evaluation methodology applied in STARS to a PWR scenario (BEMUSE phase IV) has started. Furthermore, work towards developing a similar methodology for fuel behavior analysis was initiated.

The analysis of the Halden LOCA-experiment IFA-650.4 successfully demonstrated the capability to analyze fuel behavior transients in an integrative manner. The on-going coupling of a modern FG-behavior model to FALCON enhances its analysis capabilities in a very attractive manner for both RIA and base irradiation.

The continued assessment of TRACE focused again on PWR-related problems. The successful analysis of selected tran-sients from the PKL- and the ROSA programs documented good performance of TRACE. Also, good results were obtained in general for a wide range of CHF experiments. However, the results obtained to date from a set of condensation experi-ments indicate that the respective TRACE models need careful review and model improvement. In terms of BWR-analysis, no progress could be made.

Considerable effort was spent on defining the (statistical) elements of a modern PSI criticality safety evaluation methodol-ogy, and the set of benchmarks was extended to include MOX configurations. The work on fast fluence evaluation was consolidated, assessing the impact of several nuclear data libraries.

Good progress was achieved with CFD-modelling of single-phase mixing in the lower plenum of a PWR, in the analysis of the UMSICHT water hammer benchmark as well as in developing a new pre-CHF HT correlation. A comparative analysis of PWR MSLB documented the quality of the thermal-hydraulic aspects of the RETRAN-model. Progress in the NURESIM related work was rather slow to date, due to the late delivery of the respective French codes.

The expertise of the STARS team is recognized internationally by the fact that two members were mandated with task co-ordination in the IAEA uncertainty CRP and one member was selected member of the scientific board of the NSC UAM benchmark.

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 2006 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.

- Continue research on uncertainty assessment:
  - Continue participation in CSNI/GAMA/BEMUSE Phase IV-VI (application to a reactor).
  - Further development and application of uncertainty methodology to simple reactor physics problem.
  - Evaluate the participation in new NSC task group on uncertainty modeling.
- Enhance fuel modeling capability:
  - Participation in the Halden LOCA-experiments (IFA-650.3, 4) with TH and thermo-mechanical analysis to assess the axial relocation phenomenon.
  - Initiate analysis of selected RIA experiments from the ALPS program.
  - Analyze selected CABRI RIA experiments (UO2 and MOX) using FALCON and SCANAIR.
- Continue development of Monte Carlo methodology:
  - Initiate implementation of burnup credit for criticality safety assessment.
  - Complete Neutron fluence calculations for KKG and perform supporting benchmarks.
  - Initiate shielding applications through analysis of selected benchmarks.
- Continue with TRACE assessment
  - Analysis of selected tests from the PKL and the ROSA programs.
  - Assessment of condensation models.
  - Assessment for BWR stability applications in combination with PARCS.
- Initiate assessment work aiming at the analysis of wave-induced loads in piping systems and the primary system.
  - Continue development of CFD application for geometries representative of nuclear reactors:
  - Apply CFD to lower plenum mixing in the KKG reactor.
- Evaluate enhancements to the core analysis methods:
  - Assessment of SIMULATE-3K.
- Continue participation in NURESIM (EU 6th FW):
  - Explore coupling capabilities of new integrated platform for safety analysis codes.
  - Develop open-core analysis for KKG.
- Develop new PhD-topics.

Work Carried out and Results Obtained

Research on Uncertainty assessment

Participation in BEMUSE (CSNI/GAMA)

BEMUSE-III was conceived as an application of the code propagation uncertainty methodologies selected by the countries and organizations taking part in the Programme to the analysis of the LOFT L2-5 LBLOCA test that was prepared during the development of BEMUSE-II following a set of specifications to be used by all participants. The objective of BEMUSE-III was to compare the different uncertainty methods and their application to a commonly defined transient analysis. In this way, it was expected to draw significant conclusions about the state-of-the-art of uncertainty analysis in the thermal-hydraulic field as applied to system codes, as well as to carry out a study of important aspects of uncertainty calculations. Such aspects, for instances, included those related to the validity of the assumptions applied by each method for the quantification of the uncertainty in the code's outputs of interest, the degree of accuracy and statistical significance of the uncertainty measures generated by the different methods, and, finally, the application and usefulness of the accompanying sensitivity analysis to identify those sources of code uncertainty with a larger impact in the variation of the code's outputs. The results of these studies and of the conclu-

sions produced by the discussions during the different meetings during BEMUSE-III and by the comparisons of the results obtained by the participants are collected in a Summary Report for BEMUSE-III, which accompanies the final BEMUSE-III report submitted by the organizers of this phase, CEA-Grenoble, to the GAMA group of the CSNI. This report has already been finalized and is expected to be officially issued in 2007.
Related to the results obtained by STARS, a detailed description can be found in the aforementioned report. A summary of the main conclusion indicates that the code uncertainty propagation methodology for thermal-hydraulic calculations used by STARS produced results of a comparable quality and consistency as those reported by all the participants, in general, and by those that used a similar approach, in particular. The methodology used currently by STARS is based on the statistical approach proposed by GRS (Germany), which makes use of tolerance intervals and non-parametric sampling methods supported by the Wilks formula, to determine the minimum number of code executions needed to produce uncertainty bands with a pre-defined level of probability coverage and confidence. STARS applied such methodology to the NRC sponsored code TRACE v4.05, based on the developments carried out in 2005, which set-up an automatic computation framework to do uncertainty analysis with TRACE. The resulting tolerance intervals for the PCT and the upper plenum pressure were reasonably well established and covered the LOFT L2-5 experimental measurements during most part of the transient. Only in the case of the second PCT peak, was the experimental value outside of the tolerance intervals. This was traced to a probable mis-prediction of the interfacial drag which, as already observe in the results of BEMUSE-II, resulted in too low vapour content during the second core dry-out, which reduced the maximum fuel clad temperature in the second peak. On the other hand, a detailed sensitivity analysis identified the most important sources of uncertainty in system description parameters and code physical models, which influenced the values of the output safety variables of interest, ie, first and second PCTs, time to complete quench of the core and time to initiate accumulator injection. The results were mainly consistent with those found by most of the participants.

The BEMUSE-IV Programme [1] has been started as a follow-up of the BEMUSE-III activities. All participants have been asked to apply their own uncertainty methodology to the simulation of a LBLOCA in the Zion nuclear power plant. The preliminary reference transient LBLOCA calculation, common to all the participants, will be the subject of BEMUSE-IV.
To minimize the impact of the nodalization on the results of the uncertainty analysis, a reference RELAP input deck as well as a TRACE input deck of the Zion nuclear power plant have been provided by the project organizers.

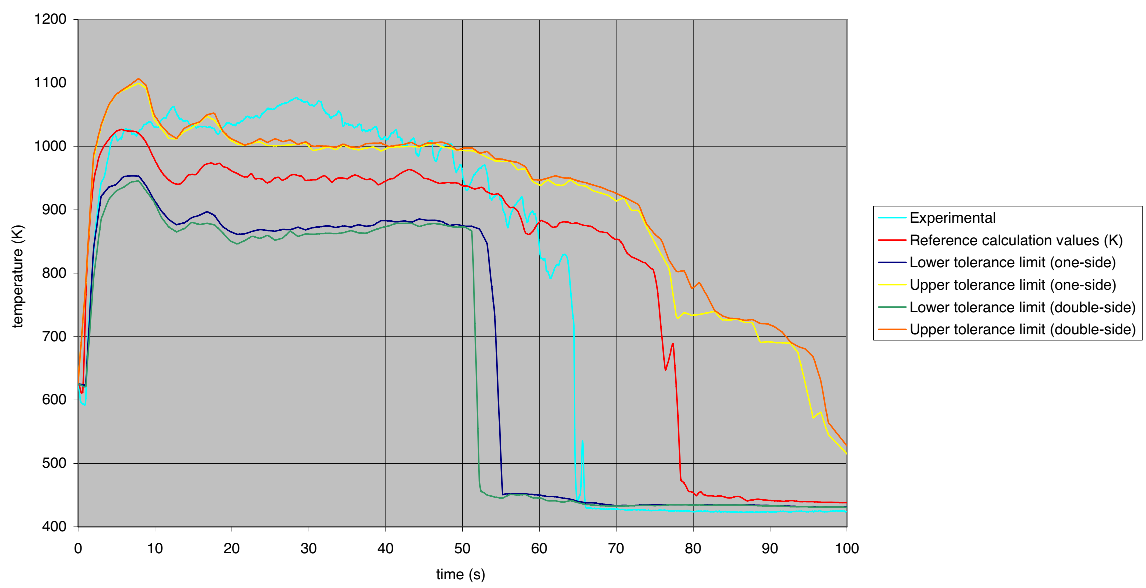

Figure 1: *Uncertainty analysis for the maximum cladding temperature: double-side 95% tolerance interval versus 5% and 95% one-side tolerance limits*

Unfortunately, it has been found that severe deviations exist between the TRACE and the RELAP input decks, both in terms of geometry parametrization and nodalization strategy. Therefore, it has been necessary to start the development of a new TRACE input deck.

Uncertainty Analysis of a Nuclear Power Plant Transient

A novel methodology that can objectively quantify the uncertainty in the predictions of physical models used in system codes, based on a statistical non-parametric treatment of model assessment data, has been developed in the framework of a STARS PhD thesis. The methodology achieves quantification of code physical model uncertainty by making use of model performance information obtained from studies of appropriate separate-effect tests addressing the physical phenomenal modeled by the physical model of interest. Uncertainties are quantified in the form of estimated probability density functions (pdfs) calculated with a newly developed non-parametric estimator. The new estimator objectively predicts the probability distribution of the model's 'error' (its uncertainty) from databases reflecting the model's accuracy, treated as a stochastically distributed variable, on the basis of available experiments. The methodology is completed by applying a novel multi-dimensional clustering technique based on the comparison of model error samples by using an extended multidimensional version of the Kruskall-Wallis test developed in the course of this PhD research. The clustering method takes into account the fact that a model's uncertainty depends on system conditions, since the model is a function of local values of system variables, eg pressure, mass flux, etc., and can give predictions for which the accuracy is affected by the regions of the physical space in which the experiments occur. The final result is an objective, rigorous and accurate manner of assigning uncertainty to code physical models, i.e. the input information needed by code uncertainty propagation methodologies used for assessing the accuracy of best estimate code results in nuclear systems analysis.

The above described methodology has been applied for evaluating the accuracy of the Critical Heat Flux (CHF) correlations implemented in the best-estimate TRACE computer code. The models studied are the Biasi and the CISE-GE correlations. The database of separate effect tests includes experimental data for maximum and minimum power at several values of pressure, as well as the relative TRACE predictions of these values. An example of results is shown in Fig. 2, where Â is the relative discrepancy between experimental and calculated values and for each value of pressure a sample of Â is available; comparing the samples from the lowest to the highest value of pressure through the clustering technique, two regions in which the Biasi correlation at maximum power is characterized by a different degree of accuracy with respect of pressure are determined; then, applying the density estimator for each region, *pdfs* are estimated quantifying the model uncertainty.

An analysis of the effect of the drift-flux model uncertainty in the prediction of the core void fraction distribution during a Nuclear Power Plant transient has been carried out with RETRAN-3D. The uncertainty in the model has been obtained by applying the methodology

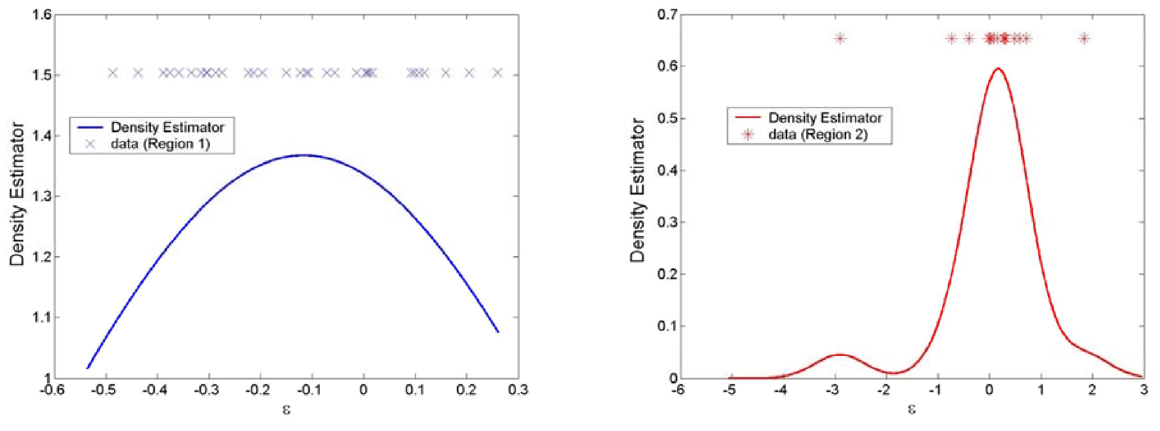

Figure 2 *Biasi Correlation at Maximum Power. Partition of the State Space defined by pressure: pdfs for Region 1 and 2. The level of significance* α *is equal to 0.1.*

described above to the void fraction predictions yielded by the RETRAN-3D Chexall-Lellouche drift flux model. The data base used for the determination of the uncertainty pdf's and the clustering of the data into regions determined by pressure and mass flux, was described in [2]. The plant transient that has been chosen for the system analysis is the Peach Bottom Turbine Trip transient that was part of an international benchmark in which STARS participated in the past [3]. A RETRAN-3D input model of the Peach Bottom plant prepared according to the benchmark specification was modified to add a one-dimensional neutronics description of the core based on cross-section data obtained from the benchmark organizers. The transient was initiated by a trip of the turbine (when the turbine stop valve is closed, the turbine bypass valve begins to open). The sudden pressure increase in the core collapsed the void fraction, increasing the liquid mass content. The resultant increase in the core density produced an insertion of positive reactivity that led to a significant power rise. Triggered by the rapid increase of neutron flux, a SCRAM occurred and the reactor was effectively shutdown in a few seconds. The analysis also considered the case in which the SCRAM failed as an example of an extreme transient, so that the power stayed about nominal value regulated by the reactivity feed-back mechanisms in the core. Based on the uncertainty quantification of the void fraction models, an uncertainty propagation calculation by using a modified version of the GRS code uncertainty propagation methodology as applied in the context of the STARS project was carried out to quantify the uncertainty in the RETRAN-3D prediction of the maximum peak power. Only the uncertainty in the void fraction was considered in the analysis. Figure 3 shows the results achieved for the two scenarios. It can be observed how the uncertainty in the

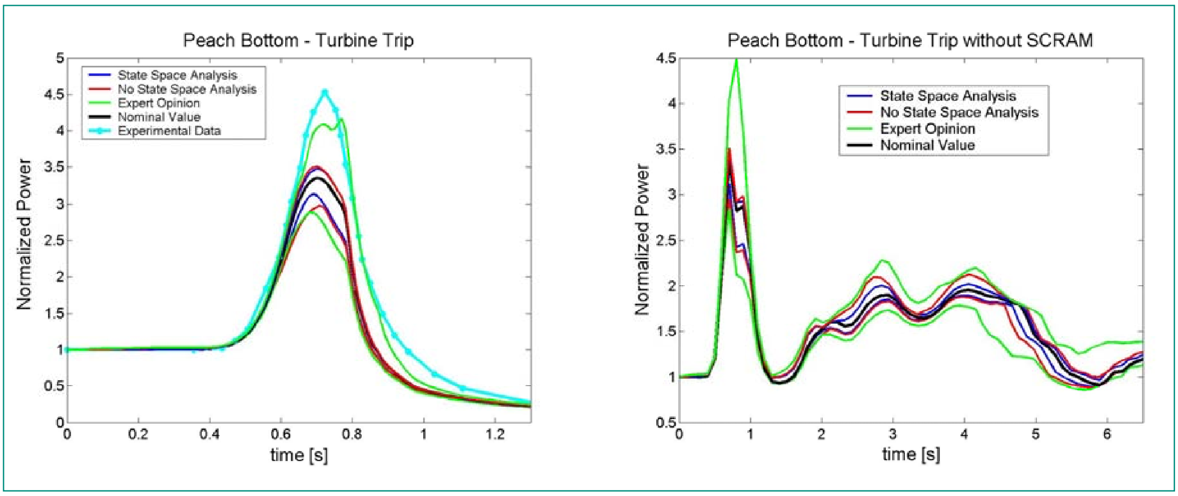

Figure 3 *RETRAN-3D results for uncertainty propagation of drift-flux model in the Peach Bottom Turbine Trip transient (left) and in the Peach Bottom Turbine Trip without SCRAM (right).*

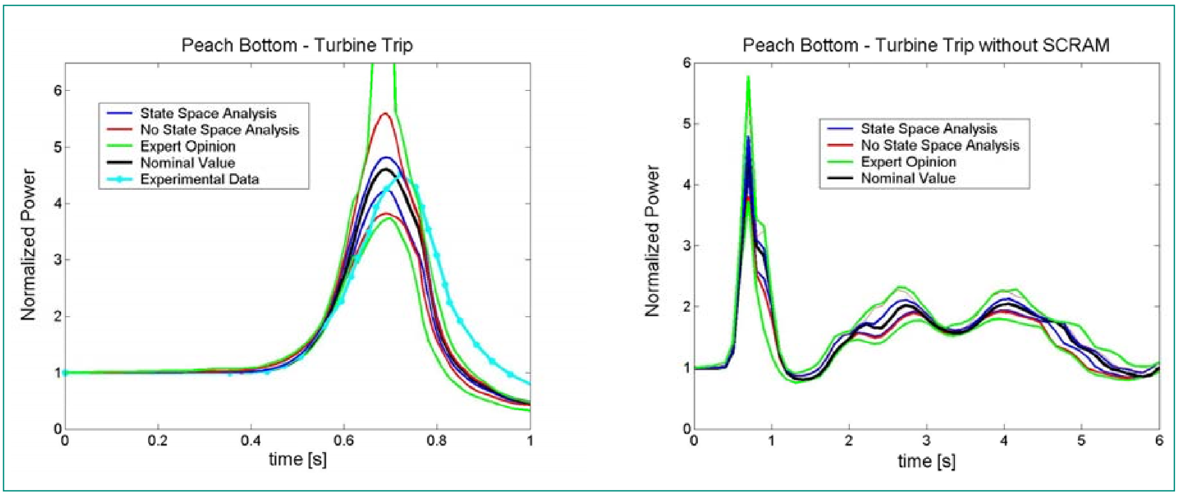

Figure 4 *RETRAN-3D results for uncertainty propagation of drift-flux model in the Peach Bottom Turbine Trip transient (left) and in the Peach Bottom Turbine Trip without SCRAM (right) with modified Cross-Sections.*

void fraction have been quantified with three different approaches, i.e. with objective uncertainty measures *(pdfs)* obtained from the novel methodology developed in the context of the PhD work, and an estimation of the Chexall-Lellouche model uncertainty on the basis of an «expert opinion» with a normal distribution and 2σ of 20%. The different quantifications can affect the uncertainty in the power, and, thus, the quantification of the drift flux model uncertainty with pdfs based on void fraction assessment studies, and also considering how the model accuracy can vary through the state space determined by pressure and mass flux, produces narrower output uncertainty bands than a typical expert opinion assessment.

As mentioned above, the reactor neutronic behaviour is modeled with one-dimensional kinetics. The discrepancy observed between the predicted and the measured power is due most probably to the cross sections used in the neutron kinetics model. In order to improve the agreement between measurement and simulation, the thermal fission cross sections have been modified so that the nominal RETRAN-3D power profile (without uncertainty in void fraction) is close to the experimental profile. The results confirm that the tolerance bands that quantify the uncertainty in the power predictions based on the application of the methodology developed is more accurate even when the possible inaccuracies in cross-section information are considered as a bias (Figure 4).

Enhance Fuel Modeling Capability

Analysis of Halden LOCA-Experiments IFA-650: Assessment of the Axial Relocation Phenomenon

The Halden LOCA experiments are designed as integral in-pile tests to study the ballooning behavior of high burnup fuel during LOCA-scenarios. Axial fuel relocation into the ballooned region, thereby locally increasing the linear heat rate, is an important question for which experimental answers are sought.

The LOCA-experiment IFA-650.3 (the first of this series performed with irradiated fuel) suffered from pre-damaged cladding and ended with an early failure at the location of the weakened cladding. Hence, this test will not support code validation in a straight-forward manner. Much before the test execution, it was selected to become the object of an International Benchmark that initially focussed on code-to-code comparisons. PSI/STARS participation in this benchmark [4] indicated that the code predictions were reasonable; sample results [5] of the second benchmark phase (thermo-mechanical calculations based on thermal-hydraulic boundary conditions as provided by GRS using ATHLET) are given in Table 1.

ORGANISATION – CODE	BURST TIME (sec)
GRS: CD	No prediction
IRSN: ICARE	356
TUEV: TRANSURANUS	305
CEA: METEOR	285
PSI: FALCON	245
MEASURED	**267**

Table 1 *Benchmark results for predictions of cladding Burst Time [5].*

The subsequent IFA-650.4 LOCA test exhibited temperatures of the fuel rod cladding and the heater sleeve (simulating the fuel rod surrounding structures) that were not observed in the previous tests and were challenging to analyze without information from Post Irradiation Examination (PIE) tests results. Thermal-hydraulics simulations with TRACE were ran with judiciously selected hypothetical boundary conditions concerning the heat sources and associated heat transfer coeffi-

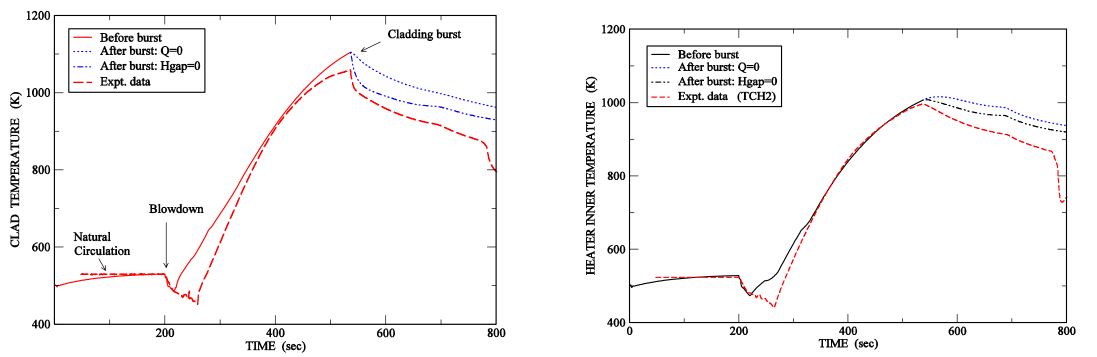

Figure 5 *TRACE predictions of cladding temperature (top)and heater temperature(bottom).*

cient. They showed unequivocally that the only explanation for the observed temperatures, as illustrated in Figure 5, is the fuel axial (downward) displacement («slump») from the top part of the (0.5 m long) fuel rod segment. Scoping calculations on internal cladding cooling due to the evacuation of «cold» gases from the top plenum during blow-down indicated a too poor effectiveness of this process to be considered as a plausible contributing factor.

Thus, the phenomenon of axial fuel relocation was becoming the main candidate for «explaining» the available thermometry data. However, since, for this experiment measurements of the cladding temperature were only available at one axial location z_0 above the balloon region (z_0/L~0.8), the extent of fuel relocation could not be determined. Only the minimum amount of fuel, corresponding to a fuel stack about 10 cm high, could be shown. This volume could easily be accommodated by the balloon size, estimated to be about 12 cm^3, based on the existing FALCON results for IFA-650.3 [6].
A rather systematic sensitivity study was undertaken in order to scope the parameters influencing cladding behavior during IFA-650.3 and to assess the robustness of the hypothesis of axial fuel relocation [7]. This was done because at the time of the study, none of the expected PIE-data for IFA-650.4 was available. (Rather late in the year and well after this study has been performed, γ-scan data of the test-rig became available, corroborating the main conclusions.)
It was found that high-temperature plasticity and appropriate temperature-dependent mechanisms of damage accumulation in the cladding were the overwhelmingly important factors in terms of the prediction of the time of failure, asking for a very accurate prediction of the cladding temperature. In addition, assumptions on the status of fuel-cladding bonding and the degree of axial mechanical constraint were found to be of relevance for the prediction of the measured cladding elongation (Figure 6). Furthermore, the fuel cracking pattern appeared also to be of importance in case pellet-cladding bonding was assumed.

The assumption of pellet-cladding bonding (subject to an appropriate choice of the cracking pattern modeling) improves the qualitative agreement with the experimental data for
- the character of cladding failure;
- the time evolution of the measured gas pressure;
- the time evolution of the measured cladding elongation.

An analytical stand-alone model was further developed in order to investigate the effects of several scenarios in relation to the axial fuel relocation phenomenon with the thermal response measured during the IFA-650.4 test. Equations for axial mass conservation and radial radiation heat transfer are introduced to predict the time evolution of the axial power profile and specific linear heat stored by the fuel during the test. The model is based on the assumption that the cladding balloon can accommodate the fuel mass relocated (or slumped) from the upper fuel stack.
The model was applied to parametrically study key relocation factors, such as the length of relocated fuel column and the filling factor in the balloon volume (Figure 7 left), by assessing their thermal effect on the fuel rod and comparing the predictions to the observed cladding and heater surface temperature histories (Figure 7 right). The model predictions are in satisfactory agreement with the thermocouple signals if the balloon filling ratio is assumed to be relatively low. Given that PIE (γ-scan) has evidenced a large reduction of the fuel

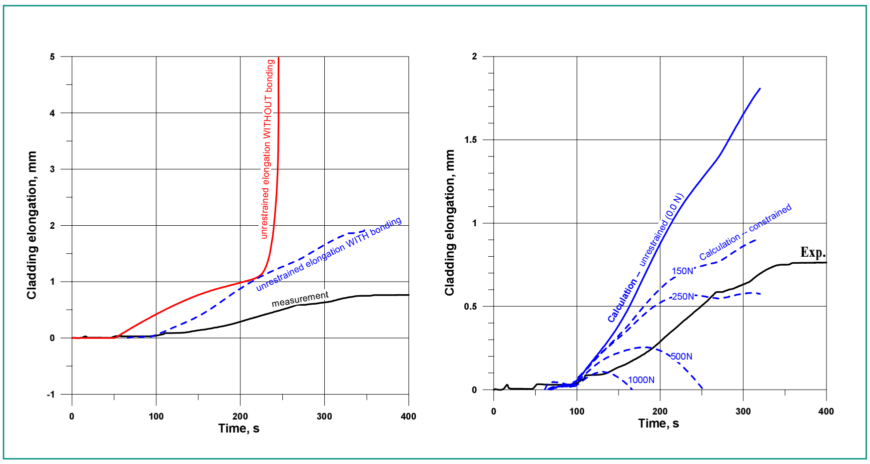

Figure 6 *Effect of pellet-cladding bonding (left) and axial mechanical constraint (right) on the predicted evolution of cladding elongation.*

stack length due to axial relocation above the balloon location, the model results indicate the possibility of fuel mass dispersion outside of the rod after cladding burst.

In summary, the computational approach adopted (including the stand-alone analytical model) allows for reasonably accurate simulation of the Halden LOCA tests and furthermore helps to identify likely ranges of certain parameters that have not been measured online (e.g. filling factor) [8]. This is of importance since the transport of the test rig from the Halden-reactor to the hot cells (to perform the destructive PIE's) introduces uncertainties in relation to the location of the fuel fragments due to the shaking.

Also, the TRACE simulations of the Halden IFA-650 tests have been fruitful in the perspective of code assessment and indicated, to date, a good performance of TRACE capability, including the new «Generalized Radiation Heat Transfer Model».

Enhance Fission Gas Release Modeling for FALCON

Recent analysis of measured fission gas release (FGR) data from modern BWR fuel pins revealed deficiencies of the related modeling in FALCON. Therefore, the initiation of work on coupling of the FALCON code with a mechanistic model for fission gas (FG) behaviour and evolution of the microstructure in Uranium Dioxide fuels [9],[10],[11] addresses the identified weakness, hopefully improving the FALCON analysis for both base load operation and thermal transients, including accident conditions.

The new FG-model covering a wide range of unsteady processes in fuel under irradiation is realised as sub-code. Specifically, it accounts for the following processes:

- Thermal and irradiation-induced point defects of crystal lattice;
- The kinetics of intra-granular FG (mono-atomic FG, non-equilibrium bubbles…);
- Inter-granular processes (gaseous porosity, gas percolation to the open surfaces…);
- Low-temperature restructuring (HBS);
- High-temperature restructuring (equi-axial grain growth);
- Evolution of as-fabricated sinterable porosity (low-temperature densification and high-temperature sintering of pores).

Currently, the primary coupling of the FALCON code with the FG-model is being worked on: All the necessary input data from FALCON is retrieved and adopted in the FG sub-code, while its output returning to the FALCON is currently limited to the FGR-related characteristics affecting gap conductivity and pressure and gaseous porosity affecting fuel thermal conductivity and swelling.

The present status of the coupling allows, however, for implementing the very important steps of numerical adaptation for robust and fast-running work of the modified code, as well as for qualitative and, in a way, preliminary quantitative verification of the results. A few test calculations of conceptual in-pile transients typical for analysis, i.e. long-term steady-state operation (Figure 8, column a) followed by either a power ramp (Figure 8, column b) or a RIA-type pulse (Figure 8, column c) were performed with the enhanced FALCON code.

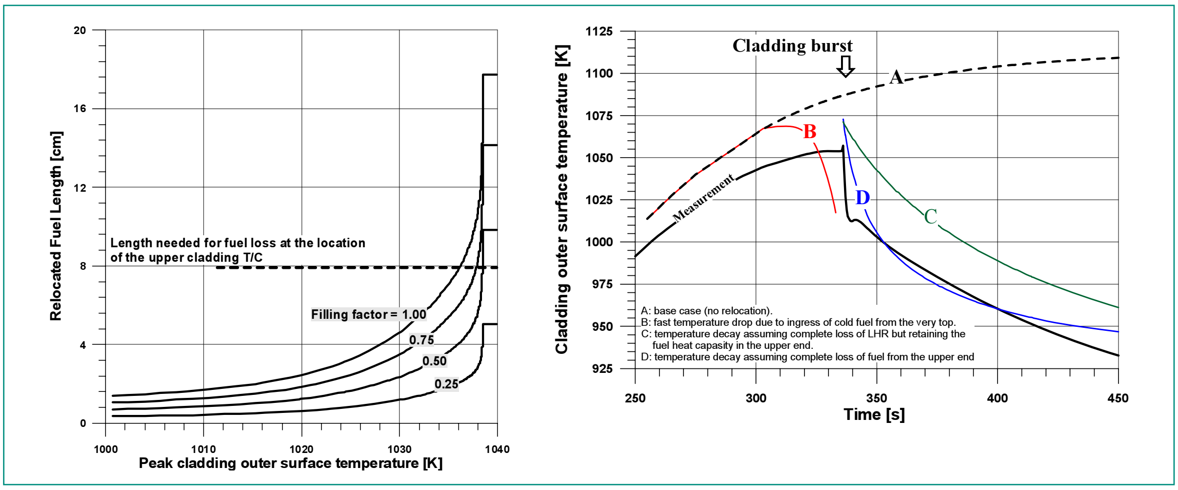

Figure 7 *Left: Calculated length of relocated fuel as a function of peak cladding temperature during the heat-up phase of Halden LOCA test IFA-650.4. Right: Calculated and measured cladding outer surface temperature at the level of upper T/C for different assumptions of axial fuel relocation.*

Currently, the results and conclusions from the ongoing work are the following:

- The FALCON code and the mechanistic FG-model are fully compatible with respect to the transfer of the necessary parameters (input/output between the models);
- Robust combination of the coupling parameters with the parameters of the numerical solver of the set of rate equations has been found, allowing for considerable speed-up. It renders the enhanced FALCON code useful for applications of practical interest;
- The results from the partly coupled code and the new FG-model are very reasonable and show that a considerable effect may be expected from the account of non-steady swelling and FGR.

Development of Monte Carlo Methodology

Criticality safety

The criticality safety research activity is aimed at developing a criticality safety evaluation (CSE) methodology for LWR compact storage pools and transport casks using modern Monte Carlo based neutron transport methods. The current approach is based on the application of the official release of the MCNPX code (2.5.0 at present) [12] and a modern standard point-wise neutron data library [13]. The approach is oriented to meet the widely accepted general requirements to establish subcriticality, such as those formulated in the ANSI/ANS-8.1–1998 [14] and ANSI/ANS–8.17-2004 [15] Standards.

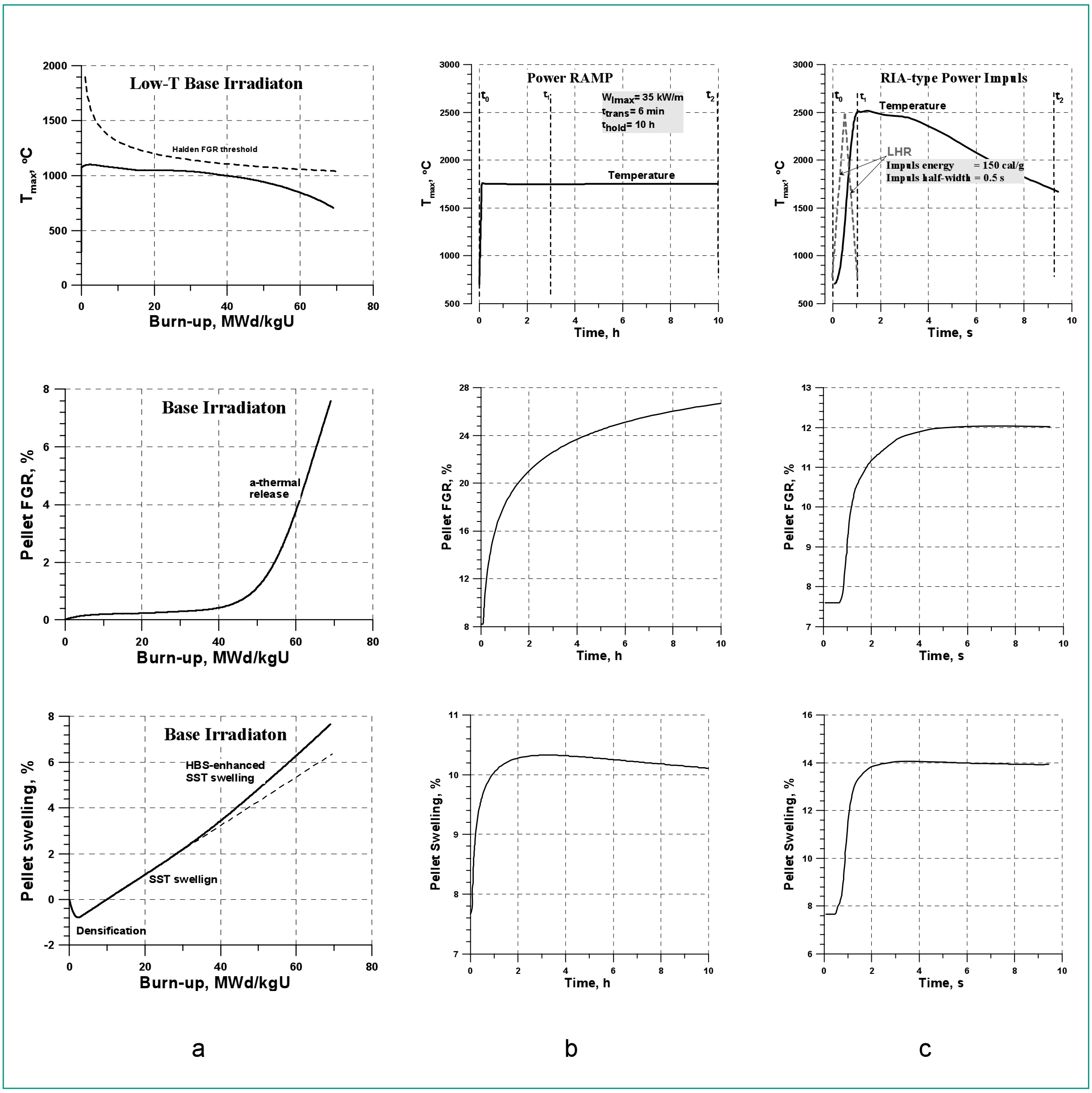

Figure 8 *Calculated maximal fuel temperature (top), pellet FGR (middle) and swelling (bottom) during conceptual base irradiation (column a), during conceptual power ramp (column b) and during conceptual RIA-type power pulse (column c).*

The assessment of standard neutron data libraries is performed on this background, based on the statistical analysis of the calculation results obtained for a set of evaluated criticality benchmark experiments. Estimates of library-related uncertainties in the calculated effective multiplication factor k_{eff} as well as the inspection of k_{eff} trends versus a set of parameters characterizing the benchmark configurations in terms of system design and benchmark physics have been produced and the definition of the area of applicability of the methodology has been derived.

The final goal of the present development is the establishment of acceptance limits for the calculated k_{eff} giving due allowance for all foreseen uncertainties with a specified confidence level and also including an additional «administrative» margin.

The assessment of the standard neutron data libraries JEF-2.2 (NEA-1616), JENDL-3.3 (NEA-1424) and ENDF/B-6.8 (NEA-1669) [13] for LWR criticality safety applications has been performed using MCNPX-2.5.0. For this purpose, benchmark calculations were performed for a suite of benchmarks from the International Handbook of Evaluated Criticality Safety Benchmark Experiments [16], selected based on their representativity to designs currently found in today's LWR compact storage pools and transport casks.

Spectrum-related characteristics of the modeled benchmark configurations have been estimated along with the k_{eff} values [17]. Subsequent analyses of trends in the calculated $k_{eff}^{calc}/k_{eff}^{bench}$ samples have been performed in order to define the range of applicability and to investigate possible cross-sections related deficiencies that would cause the calculations to underestimate the benchmark k_{eff}-values. The basic calculational results are presented in Table 2. In addition, no statistically significant spectrum-related or design-related trends in the $k_{eff}^{calc}/k_{eff}^{bench}$ values were found.

Usage of JEF-2.2 and JENDL-3.3 lead essentially to the same mean $<k_{eff}^{calc}/k_{eff}^{bench}>$ value; however, the spread of the $k_{eff}^{calc}/k_{eff}^{bench}$ distribution is slightly smaller when applying the JENDL-3.3 library. In contrast, the result for $<k_{eff}^{calc}/k_{eff}^{bench}>$ obtained with ENDF/B-6.8 is approximately 370 pcm lower than for the other libraries and on average underestimates the benchmark cases by 720 pcm.

The performed library assessments form a necessary step towards the establishment of a modern CSE methodology being developed at PSI.

Neutron Fluence Calculations for KKG and Perform Supporting Benchmarks

A methodology for an accurate analysis of the fast neutron fluence at the Reactor Pressure Vessel (RPV) of Light Water Reactors (LWR) is being developed within the STARS project [18]. Conceptually, it is based on the transfer of CASMO-4/SIMULATE-3 core-follow results (power distribution, fuel composition) into a 3D volumetric (pin-by-pin, axially distributed) fixed neutron source for ex-core neutron transport simulations using the state-of-the-art Monte Carlo MCNPX code.

An important part of the methodology development consists in the selection of the most appropriate neutron data library for posterior validation and qualification studies to evaluate the uncertainty of the fluence calculations. To date, such a pre-assessment of neutron data libraries for routine usage has been performed based on comparative analysis of the KKG reactor pressure vessel (RPV) scraping test data [19],[20] using the standard neutron data libraries JEF-2.2 (NEA-1616), JENDL-3.3 (NEA-1424) and ENDF/B-6.8 (NEA-1669) [13]. Calculations were performed with the CASMO-4/SIMULATE-3/MCNPX system of codes. The experimental scraping test data provide estimates of the fast neutron fluence (E>1MeV) at different locations of the inner wall of the reactor pressure vessel after the first 10 reactor cycles [19],[20]. The comparative calculations, displayed in Figure 9, show that the agreement between the calculated fast neutron fluence and the reference data is around ±5% using the ENDF/B-VI.8 or JEF-2.2 libraries, representing a very satisfactory result. However, using the JENDL-3.3 library leads to an overestimation of the fast neutron fluence by around

Libraty	$\left\langle \frac{k_{eff}^{calc}}{k_{eff}^{bench}} \right\rangle \pm\sigma'$	$\mathrm{Min}\left(\frac{k_{eff}^{calc}}{k_{eff}^{bench}} \right) \pm\sigma$	$\mathrm{Max}\left(\frac{k_{eff}^{calc}}{k_{eff}^{bench}} \right) \pm\sigma$	Sample standard deviation	Sample Size
JEF-2.2	0.9964±0.0002	0.9881±0.0018	1.0026±0.0020	0.0031	105
JENDL-3.3	0.9966±0.0002	0.9903±0.0018	1.0022±0.0020	0.0028	105
ENDF/B-6.8	0.9928±0.0002	0.9874±0.0018	0.9998±0.0020	0.0028	105

Table 2 *Statistical description of the calculated results.*

~10% compared to the ENDF/B-VI.8 and JEF-2.2 results.

This discrepancy problem was further investigated, and it's cause was found to be the lower neutron absorption in the water due to a lower O-16 (n, α) reaction cross-section. In this context, it should also be mentioned that JENDL-3.3 is found to be the library containing the least detailed evaluation of the high-energy neutron cross-sections (up to 20 MeV) for O-16, e.g. for (n,α) and inelastic scattering reactions. In order to corroborate this assertion, the JENDL-3.3 based calculation has been repeated exchanging the O-16 related data with the ones taken from the ENDF/B-VI.8 evaluation (see curve labeled JENDL-3.3* on Figure 9). As one can see, the new results are very close to the results obtained with the JEF-2.2 and ENDF/B-VI.8 libraries. Hence, a deficiency in the O-16 (n,α) data in the JENDL-3.3 library is shown to be likely.

Simulation of Boron Dilution Transients in a PWR Using the CFD-code CFX

Following the activity initiated in 2005 that explored the use of computational fluid dynamics (CFD) codes for applications related to nuclear systems transient analyses in collaboration with the Labor for Thermal-Hydraulics (LTH) at PSI, it was decided in 2006 to concentrate the efforts on the development of a model of the vessel of KKG that can be used with the CFD code CFX-5. The main objective was to create a model sufficiently detailed that could serve as a basis for further studies related to turbulent mixing processes in the down-comer and lower plenum of PWR vessels. Such processes become safety relevant for a certain class of transients in which masses of coolant are rapidly injected in the vessel through the cold legs, and are pushed into the core. If the mass of coolant has a low concentration of boron or low temperature, the core reactivity will increase when the coolant traverses it, which can lead to a return to power scenario, or produce large local surges of power in the core regions through which this coolant circulates. The severity of the reactivity insertion depends, to a large degree, on the mixing suffered by the original deborated coolant mass in its way from the location where it was generated in the primary side, to the core inlet. Thus, the mixing will determine the boron concentration, or the coolant temperature, at this location, which, in turn, will determine the reactivity insertion through the neutronic feedback mechanisms involved.

Accurate simulation of mixing in a three dimensional flow configuration with current system codes is difficult at best, and not feasible in most cases due to simplifications made in the flow field equations and to the numerical characteristics of their solution method. In general, they lack an appropriate treatment of turbulence, and suffer from numerical diffusion for the convective transport of solute or energy fields. For these reasons, it is important to investigate the use of CFD codes in the simulation of these kinds of system transients, so that more accurate treatments of the flow field can deliver better predictions for the mixing of the coolant masses. The developed KKG vessel model is a refined one with 6 million cells and includes a detailed description of the

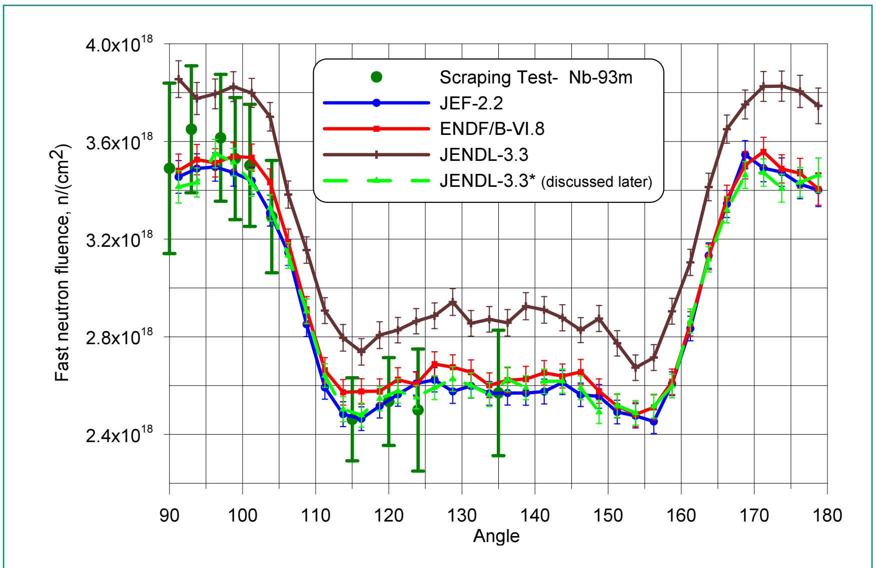

Figure 9 *Fast neutron fluence calculation results, obtained with three modern and well-established neutron data libraries.*

internals. The large number of cells permits the correct use of turbulent models near solid surface boundaries. The sieve located in the lower plenum, and the core have been modelled based on the porous medium approach, in order to reduce the complexity of the final mesh; appropriate friction loss coefficients were calculated for the sieve and for the core flow channels, with additional input from loss coefficients used in the KKG model developed for the system code RETRAN-3D.

A steady-state flow distribution, with no additional flow resistances in the model has been calculated with the new implementation of CFX-5 on the new Merlin parallel cluster configuration at PSI with 16 processors in order to check that the model has no errors in the geometry and mesh distribution. Internal flow resistances will be added to this steady state model to complete its development.

The current model may be further refined (more cells), in order to demonstrate the level of fineness at which there are no mesh effects on the results. After this is achieved, it is planned to simulate a boron dilution transient with the injection of low boron-concentration water into one of the loops from the cold-leg trap, and perform a CFD transient calculation.

Analysis of UMSICHT Water Hammer Experiments

A variety of plant transients can induce rapid and large local pressure changes that propagate through the hydraulic system at the speed of sound. For instance, in boiling water reactors (BWRs) the rapid closure of the Turbine Inlet Stop Valve or of the Main Steam Isolation Valve (MSIV) can lead to a pressurization wave entering the vessel from the main steam line, which – upon collapsing vapor – generates a positive reactivity insertion resulting in a rapid power increase [3],[21]. Another example is the depressurization wave that forms following a Loss of Coolant Accident (LOCA) and propagates from the pipe break to the reactor pressure vessel where it induces important loads onto the vessel internals. Pressure waves can also affect other parts of the nuclear system, especially pipes, where they appear and propagate as a result of water and cavitation hammers (e.g. [22]). During these types of transients, large pressure surges – challenging the mechanical component integrity – develop as a result of momentum changes in the fluid or of the formation of cavitation induced vapor pockets that collapse upon refilling with sub-cooled flow [23],[24],[25].

The assessment of the system code's capability to reliably represent pressure wave propagation is, therefore, very important, and consequently, the system transient codes TRACE and RELAP5 regularly applied in the STARS Project have been assessed by analyzing the UMSICHT PPP cavitation water hammer experiments 329 and 135. The Fraunhofer UMSICHT PPP facility consists of a 170 m long test section of 0.11 m diameter. A large pressure vessel connects both extremes of the test section and maintains test pressure and temperature at the desired values. The flow is established by a pump. Water and cavitation hammers can be produced by the fast closing valve located downstream of the pump. The piping includes two so-called bridges with related changes of the piping elevation. Interaction of the fluid flow with the pipe structures (fluid-structure interaction) are expected to be largest at these locations.

Based on a detailed multi-parameter analysis that considered the time-dependent behaviour of pressure, void

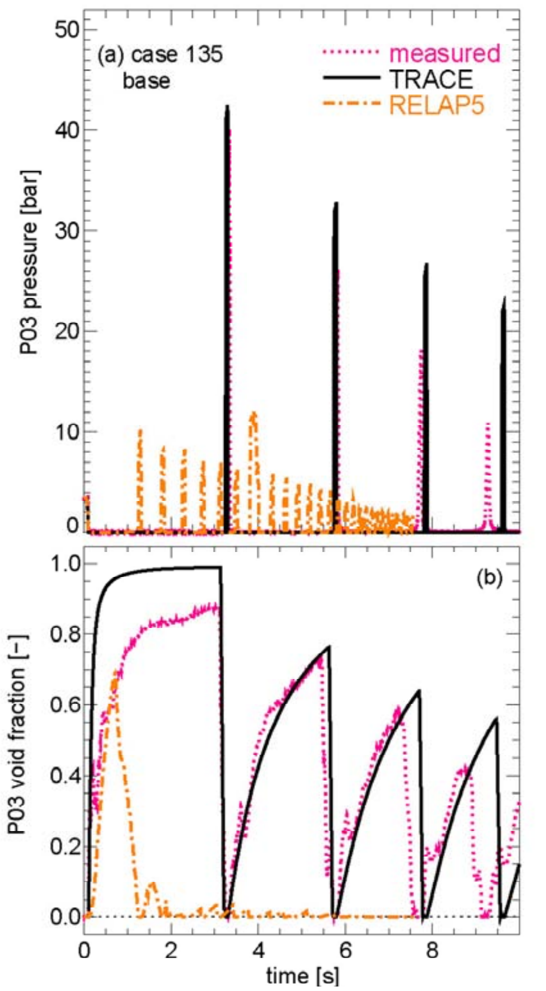

Figure 10 *Pressure (a) and void fraction (b) versus time as calculated by TRACE and RELAP5 at 0.2 m downstream the valve are compared with measured data of UMSICHT experiment 135.*

fraction and flow rate at different positions along the test pipe, the following conclusions about the quality of both codes' results of the UMSICHT PPP water hammer experiments 329 and 135 can be drawn [26],[27],[28]:

- For the lower pressure and temperature Case 135 (initially 1-4 bar and 294 K), the TRACE code was able to well calculate both the pressure and void fraction behaviour (see Figure 10). The values for maximum pressure and oscillation period are well captured. The damping of the peaks is less effective than in the experiments, most likely because of the lack of fluid-structure-interaction (FSI) modeling.
- The RELAP5 predictions for Case 135 showed local vapor generation at different positions along the pipe immediately after the valve closure. An investigation showed that the RELAP5 code fails for the combination of low pressures and high sub-cooling occurring during Case 135 [26],[28]. As a consequence, the pressure amplitudes of the first pressure peak as well as the time between the pressure peaks are considerably under-predicted (see Figure 10).
- Also for the high pressure and high temperature Case 329 (initially 10-13 bar and 420 K), the flow behaviour was well predicted by both codes. During the first seconds of the transient, the cavitation-induced generation of vapor downstream of the valve is also well predicted by both codes, as can be inferred from the comparison with the void measurement located adjacent to the valve and with the pressure measurements at different positions along the test pipe.
- However, as the accuracy in the prediction of the collapse of the cavitation-induced steam bubble is very important for the prediction of the maximum pressure value, tuning of the condensation heat transfer model parameters was needed since the original models in both codes yielded results considerably underestimating the effectiveness of the condensation process during the collapsing of the vapor bubble [26],[27],[28].
- The analysis of the results also showed that neither code was able to accurately reproduce the experimentally observed dispersion and damping of the pressure peaks. Although the calculated damping ratio appears to be close to the experimental data, further study is needed in light of the fact that neither code accounts for the effects of Fluid-Structure Interaction (FSI).
- Further investigations in relation to alternative condensation models and interfacial area density correlations applicable to highly transient conditions occurring during a cavitation process appear as interesting research topics that will lead to further improve the TRACE code.

In summary, best estimate system codes such as TRACE and RELAP5 are potentially useful tools for the analysis of cavitation water hammer and pressure wave propagation transients in one-dimensional flow geometries. Shortcomings of the interfacial transfer models in the predictions of the on-set of cavitation induced vapor bubbles and their water hammer inducing collapse have been identified.

TRACE Assessment

The transient system code TRACE is obtained from US NRC via the CAMP agreement. This code is currently only available with so-called beta-versions. Validation of the different versions is crucial before this code can be reliably applied to safety analysis problems. Occasionally, the results obtained with different code versions differ significantly in certain parameters. Hence, STARS has continued its considerable validation efforts during 2006, using experimental data sets covering several phenomenological areas.

Analysis of Selected Tests from the PKL Program

The simulation of selected PKL experiments allows the performance assessment of TRACE under low pressure conditions (similar to plant shut-down conditions) as well as for Small Break Loss Of Coolant Accident (SB-LOCA) situations.

The benchmark internal to the PKL project on calculations of test E3.1 «Loss of Residual Heat Removal (RHR) system» was revised and new experimental data on heat losses at low temperature as well as on pressure losses at low flows were supplied to the benchmark participants in April 2006. Also, a new set of boundary conditions was issued. In this respect, the TRACE input deck for the E3.1-test developed previously was modified accordingly to meet the new benchmark specifications and a new set of calculations was performed. The results obtained compare very well with the ones of the other benchmark participants [29]. TRACE (v4.160) was found to correctly reproduce all important heat and mass transfer phenomena that took place during test E3.1.

The analysis of Test F1.2 was finalized and the results of the simulations were published [30],[31]. Test F1.2 was designed as a parametric study to determine the primary coolant inventory at which onset of full reflux-condensation occurs in the test facility. This experiment allows for a performance evaluation of TRACE at various flow regimes: Single-phase natural circulation, two-phase natural circulation, transition to reflux condensation and pure reflux condensation. An example of mass flow rates calculated by TRACE is presented in Figure 11. In general, TRACE yielded a very good estimation of the main parameters and to correctly simulate the main thermal hydraulic phenomena that occurred during test F1.2.

The simulation of Run 1 and 2 of test F2.1 was also carried out. The test F2.1 was setup in a similar way as test E3.1 – Loss of Residual Heat Removal (RHR) system. For Run 1 of test F2.1, two steam generators were simulated to be in a stand-by mode as compared to one SG in reference test E3.1, thus providing higher heat removal capacity. For the Run 2 of the test F2.1, the initial primary level was set below mid-loop, close to the lower edge of the cold leg. Simulation of these cases with TRACE had provided extra confidence in the capabilities of the code to simulate the Loss-of-RHR transients and furthermore provided more insight into this complex transient due to the systematic variation of the boundary conditions complementing reference test E3.1.

Analysis of the ROSA SBLOCA Tests 6-1 and 6-2

The simulation of the OECD-NEA ROSA Project experiments allows for the assessment of the performance of the TRACE code for the analysis of various types of PWR operational and accidental transients. The tests of the ROSA Project are conducted using the Large Scale Test Facility (LSTF), which is located at the Japan Atomic Energy Agency (JAEA) in Japan. The LSTF simulates a Westinghouse-type four-loop 3423 MW_{th} PWR by a full-height and 1/48 volumetrically-scaled two-loop system [32]. Two SB-LOCA tests were executed in 2006, namely the tests 6-1 and 6-2, where the break was postulated at respectively the bottom and the upper head of the Pressure Vessel (PV), and where the high-pressure injection system was assumed to fail. The object of these tests was to better evaluate the efficiency of the accident management procedures that are based on a controlled reduction of the primary pressure using the steam generators in order to reach the pressures of both accumulator injection and low-pressure injection.

During 2006, work started with the development of a TRACE input deck describing the ROSA experimental system. The geometry part of the input deck file was converted from a TRAC-PF1 input deck provided by JAEA. Then, the heat structures of the pressure vessel were added and a new control system was included, based on the new specifications of the LSTF facility [32]. Moreover, the nodalization of the steam generators needed to be revised.

A first series of simulations of Tests 6-1 and 6-2 were executed with TRACE. The Test 6-1 actually simulated a PV upper-head break with a break size equivalent to 1.9% of the size of a double-ended cold leg break, whereas the break was assumed at the bottom of the PV for Test 6-2 and the relative break size was only 0.1%. The results were compared with the measurements provided by JAEA ([33] and [34]). These experiments actually enabled to evaluate the performance of

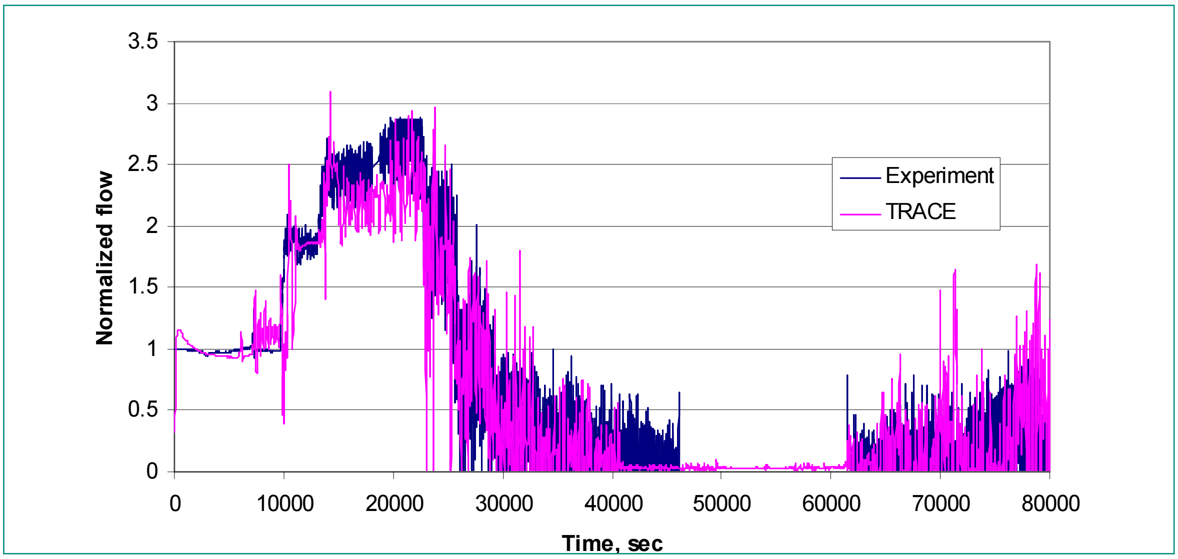

Figure 11 *Mass flow rates in one loop of the PKL facility during test F1.2.*

TRACE in simulating various important phenomena: Critical mass flow rate at the break for various flow regimes (single phase liquid, two-phase liquid/steam), single and two-phase natural circulation (after the automatic trip of the primary pumps), accumulator injection and associated condensation effects. In general, TRACE was able to reproduce reasonably well the evolution of the break mass flow rate during the first phase of the tests (before accumulator injection) also after two-phase flow conditions were reached at the break (see Figure 12).

Assessment of Condensation Models

Previous calculations with the TRACE code for the experiments performed at the PKL test facility and also earlier plant applications indicated that there are short-comings in the predictions of the condensation heat transfer for both wall and direct-contact condensation (DCC) cases. Therefore, a detailed assessment of the TRACE condensation models was initiated, directed at assessing the TRACE predictions against results obtained from Separate Effect Test (SET) facilities for both wall condensation and direct contact condensation.

An experimental database of 57 measurements on DCC for steam/water flow in a 200 mm diameter, 9 m long vertical pipe have been obtained from TOPFLOW at the Research Center Rossendorf (Germany) [35]. Experimental series are available at 10 and 20 bars. A clear underestimation of the condensation rate has been found for all 57 analyzed cases. A similar tendency was noted for the DCC experiments performed at Northwest University (NWU) for horizontal stratified flow in horizontal 6.35 cm high, 30.48 cm wide and 160.1 cm long channel and co-current steam and water flow conditions [36]. The results of the TRACE calculations clearly indicate a need to revise and further improve the interface area and/or interface heat transfer models present in the current versions of the TRACE code.

Up to date, two sets of experiments addressing wall condensation were simulated. The MIT condensation test-section [37] consists of a 2.5 m long central tube, with a inner diameter of 46 mm, in which a mixture of steam and non-condensable flows downstream, cooled by water flowing upwards in an annulus concentric to the central pipe. Experimental pressures were ranged between 1 and 4.5 bars. Up to now, 52 tests for a steam-air mixture have been analyzed with TRACE. In general, it can be concluded that the wall condensation model implemented in TRACE gives good results for low values of non-condensable mass fractions. An example (M_{air} = 0.09) is given in Figure 13 (bottom). For higher non-condensable mass fraction (M_{air} = 0.33), TRACE over-predicts condensation (Figure 13, top).

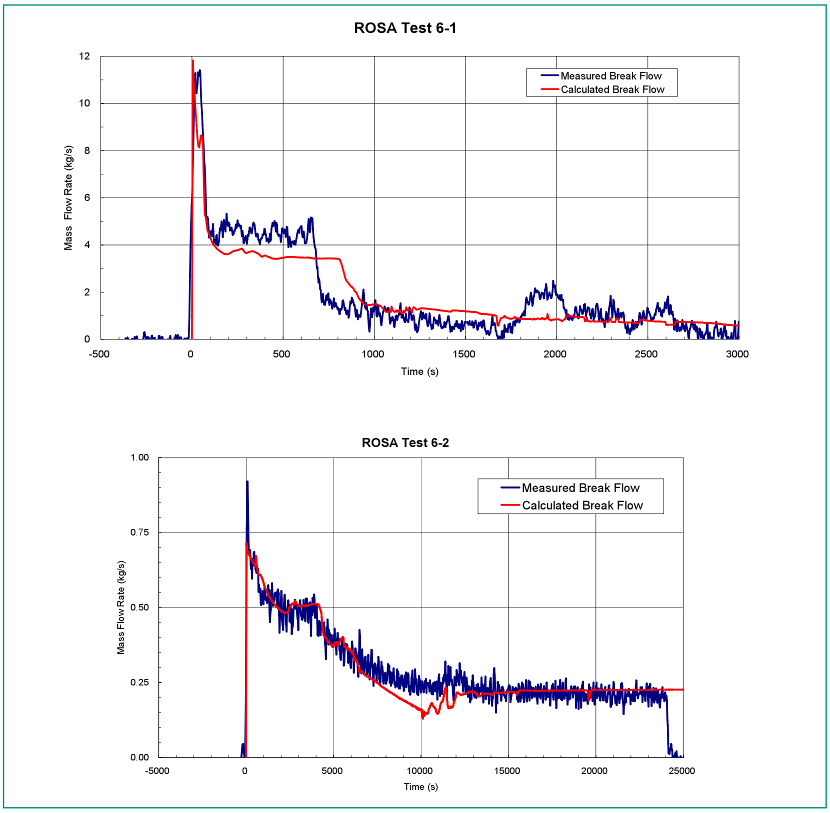

Figure 12 *Comparison of measured and calculated break mass flow rate of ROSA Tests 6-1 (top) and 6-2 (bottom), using TRACE (v5.000).*

The experiments at the Korean Institute of science and Technology (KAIST) were performed in a single actual height U-tube at various steam flows and different wall coolant temperatures, as well, the non-condensable gas (air) content was varied during the tests [38]. In total, the available KAIST database consists of 81 experiments. The common trend of the TRACE results confirms the findings of the MIT test simulations: the code provides close predictions for the wall condensation for low non-condensable content flows, and starts over-predicting the heat transfer with increasing gas content.

Hence, the results of the two series of tests on wall condensation also support the need to improve the fluid-to-wall condensation heat transfer package in TRACE.

Assessment of CHF Models

The simulation of the RIT experiments with TRACE allowed a comprehensive validation of the CHF-models for

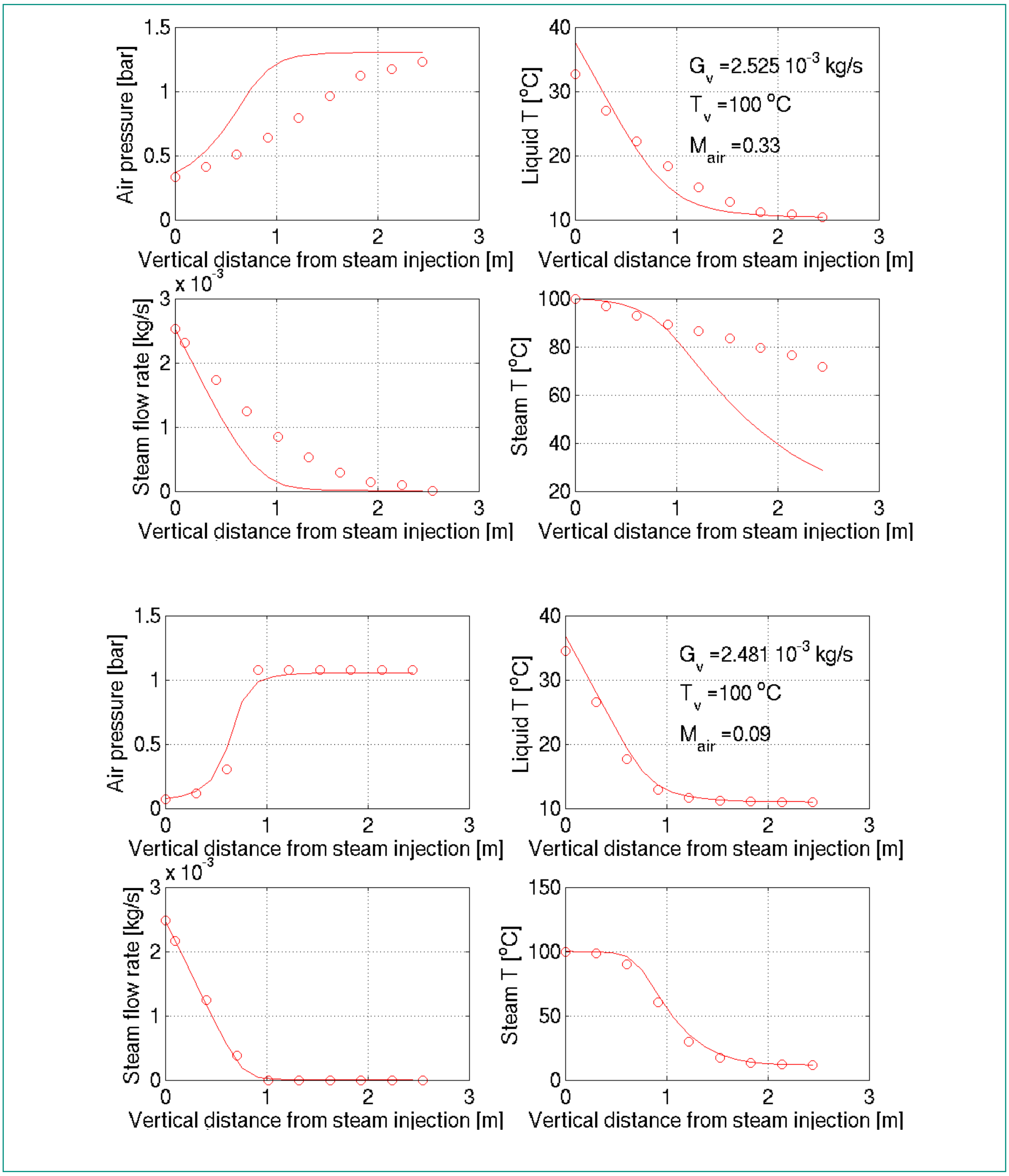

Figure 13 *Comparison between experimental data (circles) and TRACE simulations (solid lines). Two cases, respectively for low (left) and high (right) air mass-fraction are shown.*

the flow in single tubes. During 2006, the assessment of TRACE CHF model was extended to annular flow situations as well as to fuel bundle geometries as well as to low-pressure, low-flow (LPLF) conditions as found in experiments performed at KAERI.

The TRACE CHF models in the second beta-release of the code (version v4.160) were assessed against the experiments in annular tubes performed at the Royal Institute of Technology (KTH) in Stockholm, Sweden. The experimental database includes data for coolant mass fluxes between 250 and 2500 kg/m^2s and inlet subcooling values of 10 and 40 K at a pressure of 70 bar. The analysis of the performance of the standard TRACE CHF correlations shows that the CISE-GE correlation yields critical qualities (quality at CHF) closer to the experimental values at 70 bar than the Biasi correlation for annular flow conditions [39],[40],[41]. Regarding the power profile, the results of the TRACE calculations seem to be very sensitive to its shape, since, depending on the profile, different accuracies in the predictions were noted. The value of the inlet sub-cooling was also an important factor in the accuracy of TRACE CHF predictions. Thus, an increase in the inlet sub-cooling led to a clear improvement in the estimation of the critical quality with both Biasi and CISE-GE correlations.

The TRACE code CHF models were also assessed against four sets of bundle experiments. The experiments were performed at RRC Kurchatov Institute (2 sets of experiments, with 19 and 36 heated-rod bundles), V-200 facility at IPPE (Institute of Power and Physics Engineering) in Obninsk, Russia for 7 heated-rod bundle and at SKODA Large Water Loop Test Facility at NRI (Nuclear Research Institut) in Czech Republic for 19 heated-rod bundle. Figure 14 presents the combined results of the TRACE calculations that are only for Biasi CHF correlation, but the trends in the errors, as well as the magnitude of the error are similar for the CISE-GE correlation as well. The results of the assessment clearly indicate that the largest errors are obtained for low pressures, low flow and low inlet sub-cooling cases. Increase in pressure leads to a decrease in error, although the low mass flow-low inlet sub-cooling region still provides the worst results with TRACE. In general, the error trends along pressure and coolant mass flux for the TRACE-predicted CHF in rod bundles are similar to the trends observed for RIT experiments in single tubes. The simulation of the rod bundle cases as well provided additional insight on the behaviour of TRACE predictions at different inlet sub-cooling values.

The simulation of KAERI experiments for low-power low-flow conditions had resulted in a relatively good agreement with the experimental data, the error for most of the cases being +/- 40% for pressure 1-10 bar and coolant mass flux 50-250 kg/m^2s in single tubes.

Development of a New Pre-CHF HT Correlation

A single correlation, the one developed by Chen, has been implemented in TRACE to cover the several flow regimes that develop in (pre-CHF) diabatic flows. This constitutes a modification to the established methodologies[1] for the pre-CHF (Critical Heat Flux) wall-to-coolant heat transfer. While this approach allows for consistency and smooth transition between the wall heat transfer coefficients, its perceived high accuracy[2] has remained mostly uncharted for the wide range of applica-

[1] Most thermal-hydraulics codes have implemented (i) either a pre-determined flow-regime-dependent boiling curve (as in most system codes), or (ii) have left the responsibility to the code user to «build» a dedicated/specific boiling curve depending on the situation at hand (as in sub-channel codes).

[2] A component for the apparent success of the correlation, is that, statistically, the experimental database in saturated flow boiling, would tend to be more representative of the annular flow regime data (for which the Chen correlation was developed) where boiling does not provide an important contribution to the wall heat transfer. Based on this success, the correlation has been extended to the near-saturation (bubbly-flow regime) and subcooled boiling conditions. (The correlation has also been split in certain mechanistic sub-cooled boiling models).

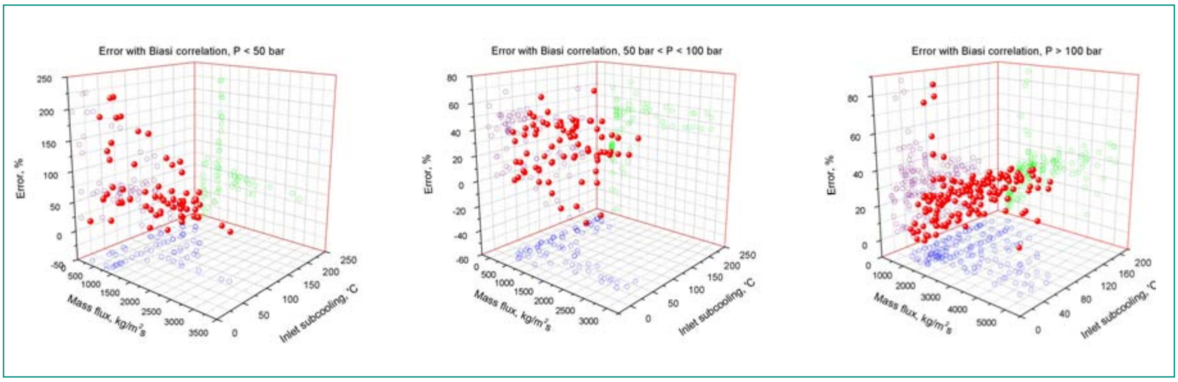

Figure 14 *Error in CHF predicted by TRACE, stratified into three pressure ranges for rod bundle experiments.*

tions expected to develop in LWR transient simulations. Previous work [42],[43],[44] identified the thermal-hydraulics conditions under which the correlation predictive capability deteriorates, due to the unanticipated growth of compensating errors, and the pitfall of splitting the correlation in mechanistic sub-cooled boiling models. The present follow up study consists in an attempt to develop a new correlation based on the segregation of the heat transfer mechanisms.

The new approach developed here is being based on identifying and quantifying the separate-effect contributions of the heat transfer mechanisms at work in diabatic (pre-CHF) two-phase flow, i.e. convection and boiling. This heat transfer segregation was based on experimental data obtained (under fixed hydrodynamic conditions) at 7 MPa.

Thus, a base-case formulation for a wide range of flow rates (1100 kg/m^2 · s to 3900 kg/m^2 · s) and qualities (0 to 0.8) at 7 MPa has been developed in the following form,

$$h_{TP} = h_{conv} + a(\Delta T_{sat} - \Delta T_{sat,o})^n$$

The convective component $\boldsymbol{h}_{\text{conv}}$ (an empirical function of the Lockhart-Martinelli parameter) is the one used in the Chen correlation, while the second component, $\boldsymbol{h}_{\text{boil}}$ replaces the original boiling term,

$$h_{boil} = Sh_{F.Z.}$$

where **S** is the «so-called» (purely empirical) «boiling suppression» factor, and ^{h}F.Z. the Foster and Zuber pool boiling heat transfer coefficient.

A unique feature of this formulation is the introduction/implementation of a wall superheat offset, $\Delta T_{\text{sat,o}}$ .It must be stated that this formulation is also consistent with heterogeneous nucleation models, not included in most, if not all, two-phase heat transfer correlation. Prediction improvement can be seen in Figure 15 for the base-case formulation at 70 bars.

The additional challenge is to develop a correlation which is valid over a wider range of thermal-hydraulic conditions. In this framework, a database for pre-CHF heat transfer has been developed. In its present size, it may represent the (bare) minimum necessary to develop basic formulations for the three «adjustable» quantities (or «free parameters»), the leading coefficient $\boldsymbol{a}$, the wall superheat offset $\Delta T_{\text{sat,o}}$ and the exponent $\boldsymbol{n}$. A certain degree of iteration remains necessary. Good results have been obtained to date, as shown for example in Figure 16 for sub-cooled boiling at 150 bars. (One can see that Thom's correlation yields excellent predictions since it is dedicated to high-pressure sub-cooled boiling.)

Thus, the domain of application of the Chen correlation is being extended, through the modification of its boiling component, in two ways: (i) first, by extending the correlation application domain over a wider range of thermal-hydraulics condition (Chen's database extended only to 30 bars), and (ii) by extending the application to the pre-annular flow regimes, when the correlation was developed solely for the (convection-controlled) annular flow regime. This limitation, recognized among others by the TRACE code developer, is being addressed through this work.

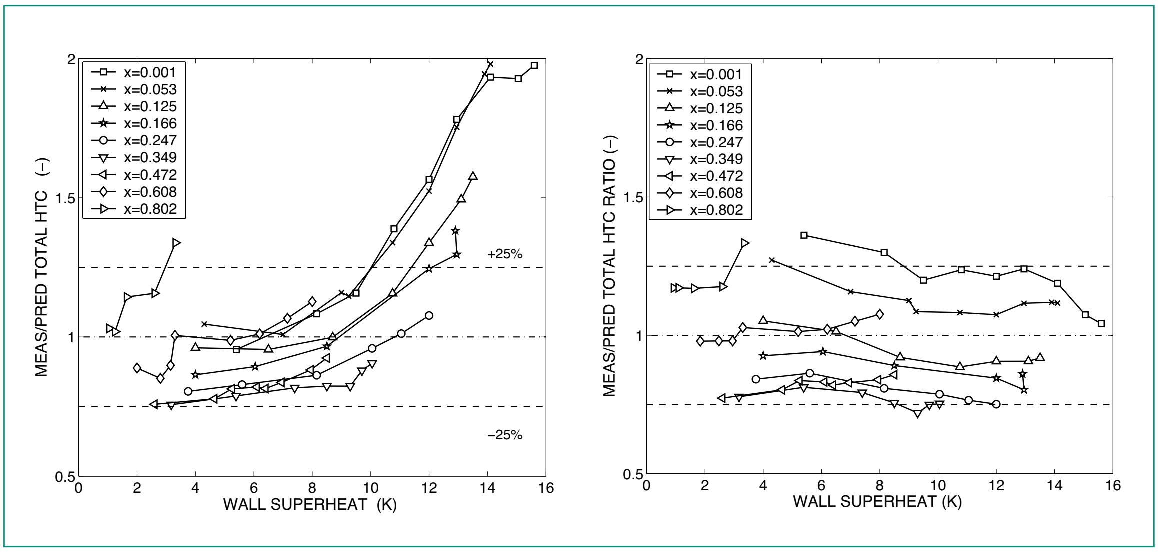

Figure 15 *Measured-to-predicted HTC ratio for CISE experimental data (70 bars): Obtained with original Chen correlation (left) and with the modified Chen correlation (right).*

Furthermore, this approach can allow the identification of functional relationships that could allow improvement of increasingly more mechanistic (detailed) two-phase flow modeling.

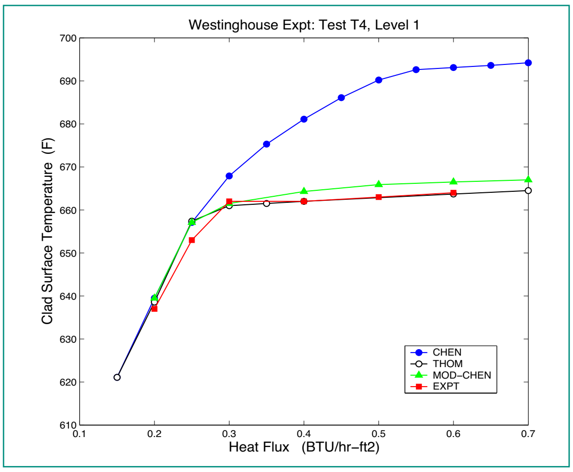

Figure 16 *Comparisons of predicted cladding temperatures against Westinghouse sub-cooled boiling data (P=152 bars).*

Core Modeling

The 3-D steady-state core analysis of the Swiss Light Water Reactors (LWR) represents a central part of the core physics activities within the STARS project. The objective is to ensure that for each plant, accurate 3-D core models up to the last completed operated cycle are developed and assessed against plant data. These core models are then intended to serve as basis for all other activities within the STARS project that require the use of 2-D/3-D neutronic data. At PSI, the steady-state core analyses are performed using the CASMO-4/SIMULATE-3 state-of-the-art codes and within the CMSYS core management system. This system was developed at PSI and serves as a central data environment where all steady-state models of the Swiss plants are developed, validated and periodically updated.

A major activity was the consolidation of the CMSYS system through the development and implementation of automatic task modules (ATMs) aimed at ensuring a secure data handling as well as efficient procedures for automatised computations. In particular, an archival structure was introduced that allows to use CMSYS as a database of reference models/calculations, each one assigned through the automatic modules, a unique identifier and stored accordingly at each model update. The CMSYS system is shown in Figure 17.

With regards to modeling and analyses within CMSYS, efforts were undertaken during 2006 to update the models for the Boiling Water Reactor (BWR) plant Kernkraftwerk Leibstadt (KKL) and for the Pressurized-Water Reactor (PWR) plant Kernkraftwerk Beznau 1 (KKB-I).

For KKL, the models for cycles 19-21 were developed and assessed against plant data. Among others, all cold critical tests including both local as well as in-sequence (global) tests were analysed in order to verify the capability of the 3-D code system to predict the cold critical reference level, recalling that this is a central quantity in the perspective of core design of subsequent operating cycles. The results obtained for the global tests using the JEF-2.2 library show that the reference level cold cri-

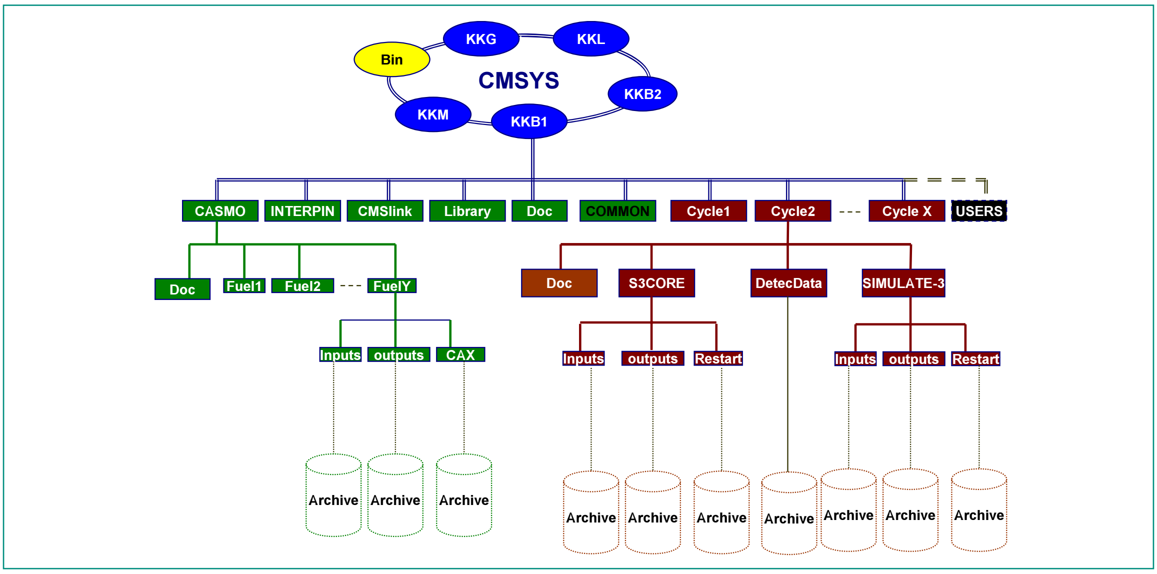

Figure 17 *PSI Core Management System for Core Analysis.*

tical tests increases from Cycle 17 to 19 and is stabilised thereafter, showing a variation of less than 200 pcm between Cycle 19 to 21, indeed a very satisfactory outcome. Although the reason for the increasing trend during Cycle 17 to 19 has not been investigated in detail, it is hypothesized to be caused by the transition from core loadings consisting of fuel assemblies with full-length fuel rods to core loadings with an increased number of fuel assemblies featuring partial-length fuel rods.

The core follow analyses of all three cycles 19–21 were performed and the preliminary results compared against TIP measurements. The differences between calculations and measurements with regards to axial, radial as well as total (nodal) power distributions are shown for each cycle and measurement in Figure 18. It is seen that the results in terms of axial power distributions are usually not as satisfactory, due to local differences (nodal), than those with regards to the assembly-average radial power distributions (which are usually very good). The RMS values are moreover seen to increase towards Middle-of-Cycle (MOC) and decrease again thereafter. Although this remains to be investigated in more details, the average total RMS difference obtained for these preliminary updated models is around 5 % which although not fully satisfactory can be considered as acceptable.

For Kernkraftwerk Beznau 1 (KKB-I), models for cycles 30-35 were developed and implemented in CMSYS. The models and analyses were performed using the two neutron data libraries ENDF-B/IV and JEF-2.2. The comparison in terms of calculated versus measured assembly-average power distributions is shown in Figure 19 where the differences between SIMULATE-3 calculations and the measured flux map values are shown for Cycle 35, noting that this is actually the currently operated cycle in KKB-I. These results illustrate that the level of accuracy achieved in the core modeling of KKB-I can be considered as very satisfactory.

KKB MSLB Analysis Using RETRAN-3D

The Main Steam Line Break (MSLB) accident represents a good example of a PWR transient for which a state-of-the-art analysis requires a coupled thermal-hydraulic/neutronic calculation. Indeed, for such an accident, reactivity is inserted into the core by the overcooling of the primary fluid, which is a direct consequence of the blow-down of the steam generator (SG) connected to the broken steam line. Before the assessment of the coupling capabilities of the RETRAN-3D/CORETRAN code system that provides the possibility to represent the core with a three-dimensional neutron kinetics model, it was necessary to first verify that RETRAN-3D is able to appropriately describe the plant-system response during a MSLB transient.

The PWR selected for the verification of RETRAN-3D is the KKB-I Nuclear Power Plant, which is based on a two-loop Westinghouse 1130 MW_{th} Reactor Coolant System, and for which a RETRAN-3D input model exists in the STARS project [45]. The MSLB analysis demons-

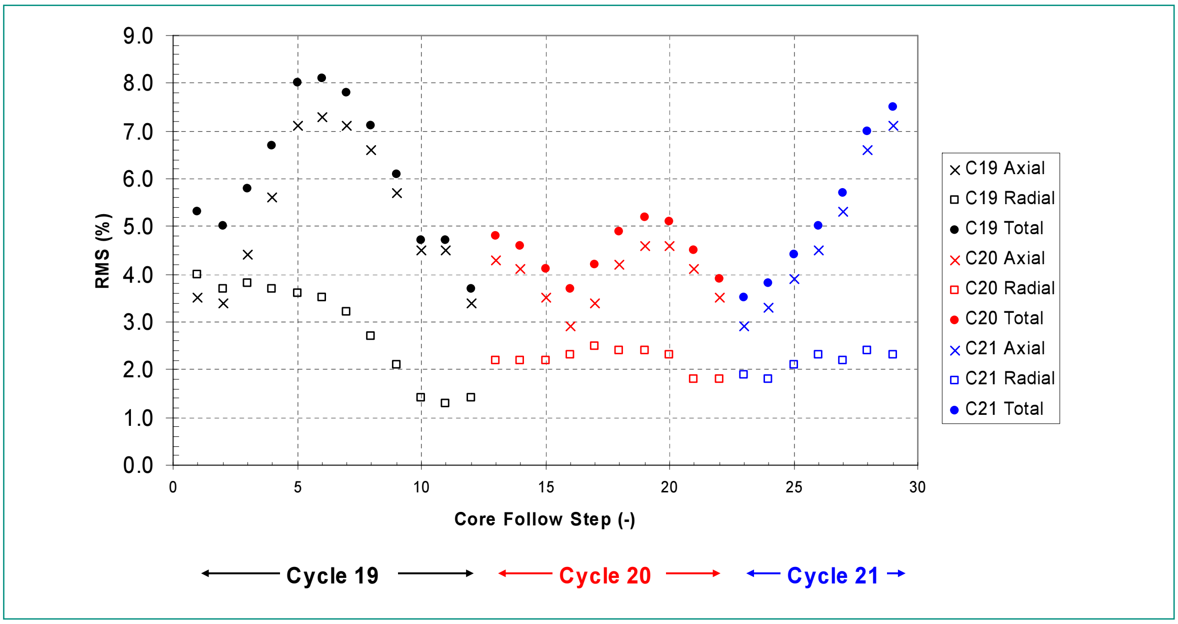

Figure 18 CMSYS Calculations versus TIP Measurements for KKL Cycles 19-21.

trated the ability of RETRAN-3D to predict the KKB-I plant system response to a MSLB accident from Hot Zero Power conditions. In comparison with the results from a similar analysis based on the vendor code (NLOOP), the RETRAN-3D model results showed only very limited differences [46]. In order to reduce errors of interpretations to a minimum and to allow for the most consistent comparison between the two codes, the plant system assumptions selected for the analysis with RETRAN-3D were kept as similar as possible to the ones specified in the reference analysis [46].

The steady-state was correctly calculated with RETRAN-3D and provided an excellent agreement with the reference data in terms of system pressures, temperatures, mass flow rates and water mass inventories. The transient analysis showed the consistency of the RETRAN-3D model, which could accurately reproduce the different stages of the MSLB transient, such as the blow down in the two SGs before the closure of the MSIVs, the thermal contraction of the primary fluid and the re-establishment of a water level in the pressurizer after the SIS injection. The model could also correctly predict some local effects that are important to appropriately characterize the plant system behaviour, like the development of a steam bubble in the upper head of the RPV which can significantly affect the system pressure evolution. One could also verify that the specific flow mixing model used in the RETRAN-3D allowed obtaining a fairly good description of the mixing phenomenon, in comparison to the results obtained in the reference analysis.

Although the overall agreement between the two code models was very good, the transient analysis results showed some discrepancies that were investigated in some detail. Most of these differences were limited in amplitude and could be actually related to some uncertainty that still existed between the specifications of the two analyses. Thus, it was shown that the small divergences observed in the primary flow rate could be related to small variations of the primary pump speed for instance (less than 2,5% over 600 s), as illustrated in Figure 20. Differences in the predictions of the water level and the pressure in the pressurizer could be related to some uncertainties with respect to the characteristics of some auxiliary systems like the accumulators, the SIS or the CVCS.

The more relevant discrepancy observed between the two analyses is related to the prediction of the cooling efficiency of the SG connected to the intact Loop A, traced back to a modeling deficiency of RETRAN-3D to account for thermal non-equilibrium in the upper part of the SG.

In summary, the RETRAN-3D model showed very satisfactory prediction capabilities, especially with respect to the main plant system parameters affecting the core response after an MSLB accident, namely the core mass flow rate, the core inlet temperature, the core inlet boron concentration and the system pressure.

	3.54 1.78	**1.93** 0.62	**-1.67** -2.77	**0.61** 0.92	**-3.17** -2.72	**0.28** 2.27
3.54 1.78	**0.28** -2.08	**1.43** 0.63	**1.68** 1.26			**-2.23** -0.28
1.93 0.62		**-0.73** -1.25		**-1.41** -1.01	**0.80** 1.82	
-1.67 -2.77	**0.59** 0.25	**0.64** 0.96	**-0.08** 0.50		**-1.05** 0.80	
0.61 0.92	**0.52** 0.93		**-2.86** -2.29	**0.68** 1.83		
-3.17 -2.72	**-1.68** -1.09	**-1.58** -0.60				
0.28 2.27						

MOX Fuel

[(SIMULATE-MEASURED)/MEASURED] (%) - JEF-2.2

[(SIMULATE-MEASURED)/MEASURED] (%) - ENDF-B/IV

Figure 19 *Differences between Calculated and Measured Radial Power Distribution for KKB-I Cycle 35.*

Participation in the NURESIM European Project

NURESIM is one of the Integrated Projects of the 6th Framework Programme offered by the European Union. The overall objective is to establish a common European numerical simulation platform based on the integration of state-of-the-art simulation methods associated with the different disciplines relevant to the steady-state and transient analysis of nuclear reactors (PWR, VVER, BWR). The disciplines were divided into 5 subprojects of which STARS contributed to two: Subproject SP1 «Core Physics» and subproject SP3 «Multi-Physics». The latter focuses on the development and the integration of advanced coupling techniques for the analysis of LWR cores using coupled neutronics and thermal-hydraulics simulation tools.

Although the long-term goal of the multi-physics activities initiated within the NURESIM project should be to combine a pin-based Monte Carlo or deterministic transport calculation with a full CFD thermal-hydraulics (2-phase) core-wide sub-channel simulation code, such a goal is far beyond the current NURESIM project. However, it is important to demonstrate a series of steps in this direction as soon as possible, in particular with regard to the development of consistent methodologies and tools to couple the different codes and solvers of the future European platform.

The underlying principle of any coupling technique is to iteratively use the results from one solver as the boundary condition of another solver until convergence of the solution of the simulated problem. In the framework of NURESIM, the coupling of separated codes/solvers inside one common simulation platform can be decomposed in four main issues:

- Ensure consistency between the input specifications and assumptions used by the different solvers,

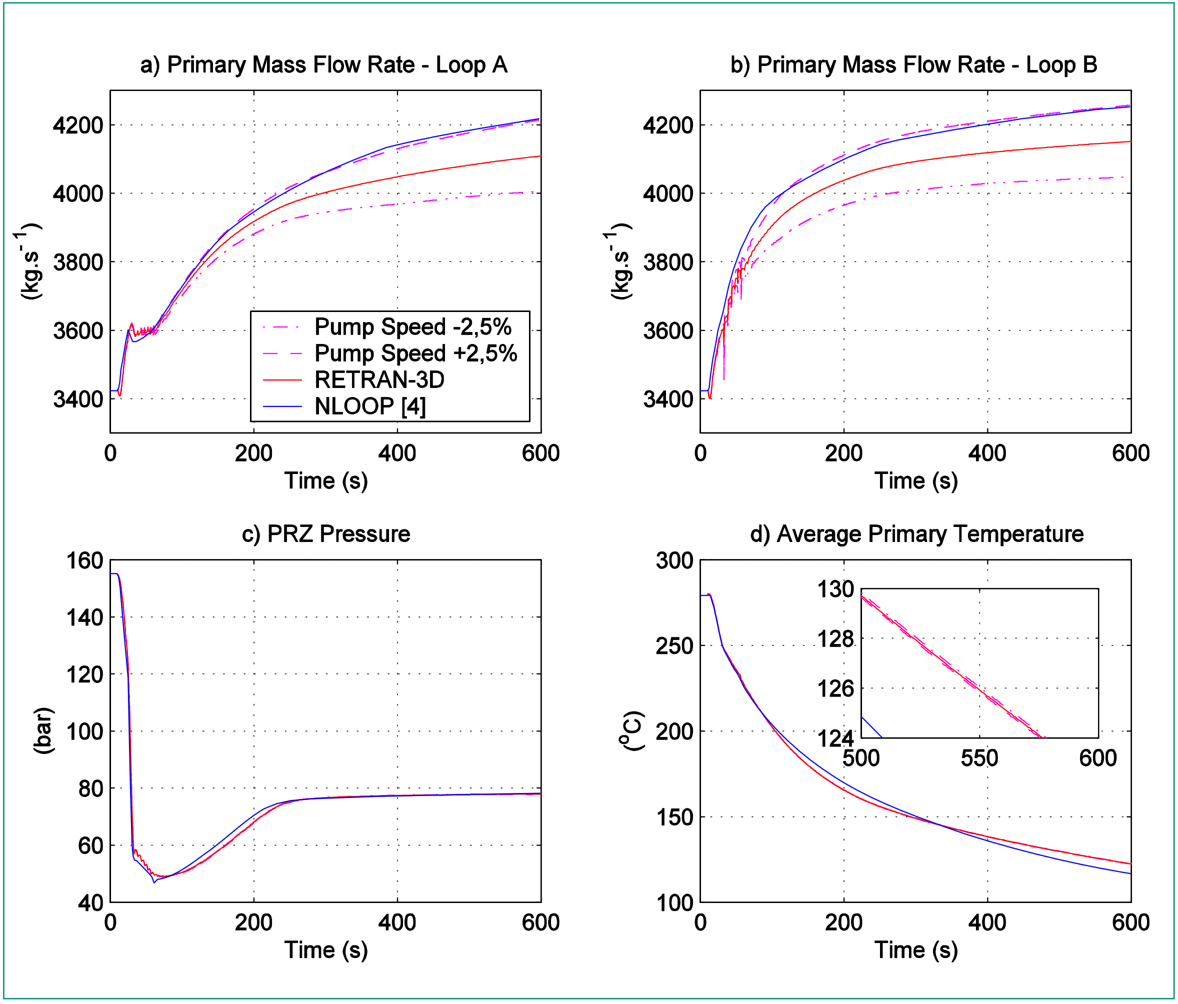

Figure 20 *TMI Core asymmetrical overcooling transient – Radial power distribution at end of transient (calculated with the CRONOS2/FLICA4 coupled component of SALOME).*

- Establish standard methods for interpolation or averaging operations from the simulation domain of one solver to another,
- Integration of the data transfer between the different solvers,
- Definition of the calculation route for steady-state and transient analyses.

Two French codes were used in order to address these issues in a practical manner. Thus, the neutron diffusion code CRONOS2 and the thermal-hydraulic code FLICA4 were coupled and integrated in the NURESIM platform, which is based on the SALOME scientific workshop software [47]. The integration work is the preliminary work that has to be made in order to allow the code/solver taking advantage of the different features of SALOME.

A multi-physics 3D interpolation/averaging tool has been developed in the SALOME platform, in order to allow two solvers using different meshing schemes to exchange result fields during a steady-state or a transient coupled analysis [48]. All the interpolation and averaging operations are based on the data structure of the platform in order to reduce to the minimum the dependency of the tool upon the two codes used and thus to ensure re-usability of the tool for other solvers that will be later integrated in the platform.

Also, a common input data pre-processing application for CRONOS2 and FLICA4 has been developed using the GUI (Graphical User Interface) and data exchange capabilities of SALOME. The concept is to provide the user with a tool that allows specifying the input data that are common to CRONOS2 and FLICA4 (e.g. geometry, initial and boundary conditions, etc.) only once during the study, in order to avoid any inconsistency between the two codes in the specifications of the coupled problem [48].

The different tools and applications described above have been successfully tested inside the SALOME plat-

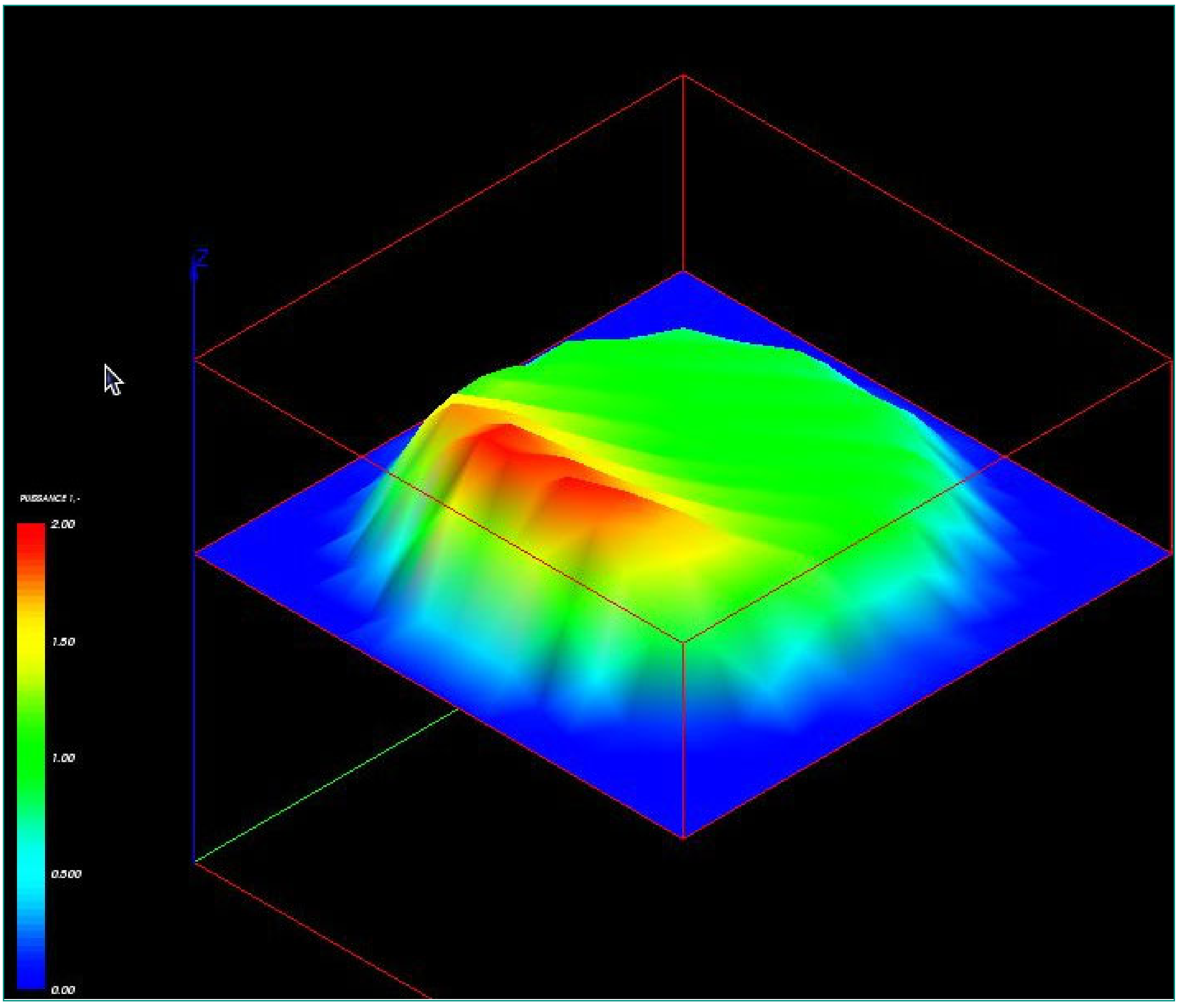

Figure 20 *KKB-I MSLB transient calculated with RETRAN-3D and compared with reference NLOOP results. Sensitivity to variations of the primary pump speed.*

form (version 3.2.1) with a coupled CRONOS2/FLICA4 analysis of a transient with specifications derived from the OECD/NEA PWR MSLB Benchmark [49]. The calculated radial power distribution following a postulated transitory decrease of the core inlet temperature for a group of 35 selected fuel assemblies located at the core periphery is displayed in Figure 21.

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 swiss*nuclear*. 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 (as described above) is completing 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. 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. Several code errors have been identified and communicated to the code development team.

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

In addition, it has started a participation in an IAEA **CRP on uncertainty.** The two Swiss representatives from STARS were selected coordinators for two tasks: One task involves an application of current uncertainty methodologies to the determination of code uncertainties in the simulation of two experimental LOCA integral tests for both a VVER and a PWR. The second task looks at developing a merged uncertainty propagation methodology, attempting to integrate the best features of the two well known methodologies from GRS (Germany) and from Univ. Pisa (Italy).

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 selected as member of the UAM scientific board.

Also, participation in the CSNI task group on the **Action Plan for Safety Margin** (SMAP) was active during 2006.

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 UO_2-fuel and the SCANAIR code. Technical exchange on the modeling of the different experiments is ongoing. During 2006, no new experimental data set became available.

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 **(FZR)** was focusing on partially supporting PhD work at U Dresden that extends the PSI reduced-order model on BWR-stability.

The 6th FW EU Integrated Project **NURESIM** continued during 2006 with contributions to the two subprojects «Core-Physics» and «Multi-Physics», the latter also being coordinated. Collaboration work with the Commissariat à L'Energie Atomique (CEA, France) and the Universidad Polytécnica de Madrid (UPM, Spain) was of special relevance to the work described in this report. It should be mentioned, though, that overall progress was rather slow to date, due to the late delivery of the respective French codes.

Assessment 2006 and Perspectives for 2007

Assessment 2006

Most of the goals specified for 2006 could be reached, and some work not foreseen at the time of the writing of the last report could be successfully undertaken.

Further good progress was achieved in the area of uncertainty research: The PhD-thesis is nearing its completion by applying the developed methodology to a plant transient. The BEMUSE phase-III participation has been completed and respective work for phase IV (application to a PWR scenario) has begun. The know-how acquired in the thermal-hydraulic area is being transferred to the fuel behavior analysis area and related work was started with establishing a data base of relevant uncertainty parameters. The expertise of the STARS team is recognized internationally by the fact that two members were given the roles of task coordination in the IAEA uncertainty CRP and one member was selected to participate in the scientific board of the NSC UAM benchmark.

The analysis of the Halden LOCA-experiment IFA-650.4 using FALCON/TRACE as well as an analytical stand-alone model successfully demonstrated the capability to analyze fuel behaviour transients in an integrative manner. Unfortunately, no significant further progress could be made in the domain of analysis of RIA experiments, as no new data became available, neither from CABRI (including MOX) nor from ALPS. However, the on-going coupling of a modern FG-behaviour model to FALCON will offer in the near future very attractive new RIA modeling capabilities beside expected better performance for the analysis of base irradiation data.

The assessment of TRACE continued again with considerable effort, focusing on PWR-related problems. The successful analysis of selected transients from the PKL- and the ROSA programs documented good performance of TRACE. Also, good results were obtained in general for a wide range of CHF experiments. However, the results obtained to date from a set condensation experiments indicate that the respective TRACE models need careful review and model improvement.

In terms of BWR-analysis, no progress could be made. Particularly, work in relation to BWR-stability problem had to be abandoned as the code could not be brought to convergence. The more general issue of BWR analysis needs to be addressed in the near future.

The work on criticality safety evaluation was consolidated and included the extension of the benchmark set to configurations with MOX. Significant effort was spent of defining the (statistical) elements of a modern PSI criticality safety evaluation methodology. Because this effort took more time than expected, implementation work towards burnup credit could not yet be initiated.

Also the work on fast fluence evaluation was more in a consolidation (and publication) phase: The impact of several nuclear data libraries was assessed – cross-section deficiencies in one nuclear data library were identified – as well as the modeling parameters key for an accurate modeling have been identified. Work towards shielding analysis did not go beyond a literature survey as the cooperating partner had to delay this work a bit due to resource problems.

The work towards CFD analysis of mixing problems in NPP geometries progressed well with the development of a «high-fidelity» model of the down-comer and lower plenum of the KKG NPP.

Good progress was also achieved in the analysis of the UMSICHT water hammer experiments, obtaining very relevant insights in the respective capabilities of the system codes (e.g. TRACE). Furthermore, work was initiated to quantitatively assess the simulation capabilities of TRACE for (de-)pressurization waves following LOCA.

The work on developing a new pre-CHF Heat Transfer correlation is nearing completion. It offers a better prediction of heat transfer in two-phase conditions and represents a possible improvement of TRACE.

Core modeling absorbed significant effort. It was paralleled with the implementation of the new modules of the CMS code system: CASMO-4E and SIMULATE-3K. Most effort was devoted to updating the core models. Because the transfer of some core data was delayed, the goal of having all core models current could not be achieved, partly also due to absence of respective personnel.

The comparative analysis of a PWR MSLB showed that the thermal-hydraulic modeling is in good shape. However, problems were identified in the cross-section preparation route that led to a study of the different approaches available in STARS. In both the methodologies available for RETRAN-3D and TRACE-PARCS, shortcomings were identified. Its impact is being investigated currently.

The participation in NURESIM generated interesting first results. The possible benefits for STARS of the modern coupling technology will be further explored during 2007. However, the goal of an open-core analysis for a PWR does no more appear to be realistic.

The very accurate analysis of PWR transients that include significant single-phase mixing in large volumes of the primary system would need a coupling of a CFD-code with a system code. A corresponding PhD topic has been developed, and the selected candidate will start early in 2007. This activity represents a natural next step after the implementation of the CFD single-phase mixing analysis.

It is worthwhile to mention that the surveillance audit of the project management system according to ISO 9001:2000 was successfully passed in July 2006.

In addition to the NES seed-action 2005 that was won by STARS collaborators, also the second NES seed action will partly support the STARS activities in the middle term as one collaborator of the STARS project together with a collaborator from the Laboratory for Energy System Analysis (LEA) won the funding (6 person-years during the coming 3 years) for the new research activity «Advanced Computational Methods for Probabilistic Dynamic Analysis in Current and Future Nuclear Systems».

Perspectives for 2007

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

- 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 behavior analysis.
- 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 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.
- 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.
- Develop capability for LOCA analysis for EPR.
- Explore coupling of SIMULATE-3K to TRACE / RETRAN-3D.

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