STARS-IV

Safety Research in Relation to Transient Analysis for the Reactors in Switzerland: Know-how für Störfallanalysen

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ABSTRACT

The TRACE code developed by the US NRC was successfully applied for a number of different studies. The assessment against a wide database of CHF experiments was completed with good results. The PSI contribution to the CSNI BEMUSE-program (Phase III) focusing on the uncertainty evaluation of the LOFT L-2-5 LOCA-test was completed. In this context, TRACE performed comparatively well. Furthermore, the generalized radiation heat transfer model was tested against available Halden LOCA-tests and good agreements with the measured cladding temperatures were achieved. Less satisfactory performance is to be noted for BWR models.

The PhD study on the development of a non-parametric statistical methodology for the quantification of uncertainties in system code physical models produced a methodology that addresses the dependency of the probability density functions (PDF) on relevant system parameters. Improvements to the estimation of the PDFs have also been implemented that addresses the difficulties that arise if only a limited number of validation cases are available.

The comparative analysis for selected RIA-experiments from the CABRI-project with both the FALCON and the SCANAIR codes was continued in order to span a reasonably wide range of energy injection rates and pulse widths. Of relevance is the finding that the transient fission gas induced swelling will remain of minor importance for the parameter range expected from hypothetical RIA pulses in both PWR and BWR.

Good progress was also achieved in the area of fast fluence evaluation, an important topic related to plant aging: A Monte-Carlo based methodology has been drawn up, taking into account fission source distributions as obtained from the existing core models and was successfully tested against scraping test data from the KKG pressure vessel. In addition, the studies related to criticality safety identified criteria for selecting the number of neutron histories, given the accuracy of the experimental benchmarks available.

The capability to model single-phase mixing was demonstrated with the successful CFD-analysis of the Vattenfall mixing experiments, enabling an improved modeling of the single-phase fluid behavior in large volumes e.g. the lower plenum.

STARS obtained a certification according to ISO-9001:2000.

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 2005 were:

- Continue research on uncertainty assessment:
- Participation in BEMUSE Phase III (uncertainty evaluation for thermal hydraulic problems).
- Apply methodology to simple reactor physics problem.

Enhance fuel modeling capability:

- Analyze selected CABRI RIA experiments (UO_2 and MOX) using FALCON and ev. SCANAIR for the MOX-cases.
- Participation in the Halden LOCA-experiments (IFA-650.3) with TH and thermo-mechanical analysis.
- Statistical fuel analysis.

Continue development of Monte Carlo methodology for:

- Criticality safety and include burnup credit.
- Neutron fluence calculations for KKG.

Continue with TRACE assessment (depending on the availability of new code version):

- Analysis of selected tests from the PKL and ev. the ROSA-V program.
- for BWR applications by converting an existing TRAC-BF1 model to TRACE and benchmarking for selected plant transients.

Continue development of CFD application for geometries representative of nuclear reactors:

- Analyze mixing experiments in Vattenfall test facility.
- Apply CFD to lower plenum mixing in the KKG reactor.
- Initiate active participation in NURESIM (EU 6th FW):
- Explore coupling capabilities of new integrated platform for safety analysis codes.

Work Carried Out and Results Obtained

Only selected highlights from STARS are reported in the following sections, which provide an overview of the major part of the STARS work performed during 2005. Some further work will be addressed in the assessment chapter.

Doctoral research on uncertainty assessment

The use of best estimate computer codes to study nuclear power plant transients provides more physically based simulations than the traditional conservative approaches, but it necessitates the determination of their calculation uncertainty. Research performed in the past decade has resulted in a series of methodologies able to propagate the uncertainty in system variables and physical models to relevant code results. The quantification of these uncertainties, especially those of physical models, has been mostly based on expert opinion, which tends to be subjective and to overestimate them. Doctoral research related to uncertainty analysis in STARS is focused on the development and application of a non-parametric statistics methodology to objectively quantify the uncertainties in the physical models used by system analysis codes.

The work this year has produced a novel methodology which can yield more objective measures of code model uncertainty by using model performance information from assessment studies. A statistical non-parametric approach has been developed to quantify the uncertainty of a given physical model, as implemented in a complex computer code, by probability density functions *(pdf*s), which are obtained by applying a new non-parametric estimator based on the combination of a universal orthogonal series estimator and a kernel estimator [1]. Including the best features of each of its combined estimators, the new estimator also overcomes the numerical difficulties observed for limited number of data from the available assessment studies. The estimator thus constructed *objectively* predicts the probability distribution of the model's «error» (its uncertainty) from databases reflecting the model's accuracy, generated through code assessment studies on the basis of available experiments (see Figure 1, *left*).

By applying a newly developed multi-dimensional clustering technique based on the Kruskall-Wallis test [2], the methodology can also treat the dependency of a model's uncertainty on system conditions (see Figure 1, *right*). The data clustered as a function of selected system variables on which the code model depends, is then used to generate *pdf*s with the non-parametric estimator described above, thus quantifying the model's uncertainty in different regions of the state space. The final result is a more objective, rigorous and accurate manner of assigning uncertainty to code models, an essential step in most code uncertainty propagation methodologies developed to date.

The complete methodology has been applied to the quantification of the uncertainty in void fraction predictions for the integral test OMEGA ROD BUNDLE test No. 9 with RETRAN-3D, a double LOCA (in cold and hot legs) from PWR operating conditions. The uncertainty information associated to the RETRAN-3D drift-flux model has been added to uncertainties in other system description parameters [3], and used as input for the code uncertainty propagation methodology employed in STARS. The results are presented in Figure 2: The uncertainty in void fraction predictions, obtained from the proposed methodology (*right*), yields more narrow, realistic and objective tolerance intervals over which the code's results can vary, as compared with a subjective quantification based on expert opinion (*left*) not considering the dependence of the *pdf*s on the state-space.

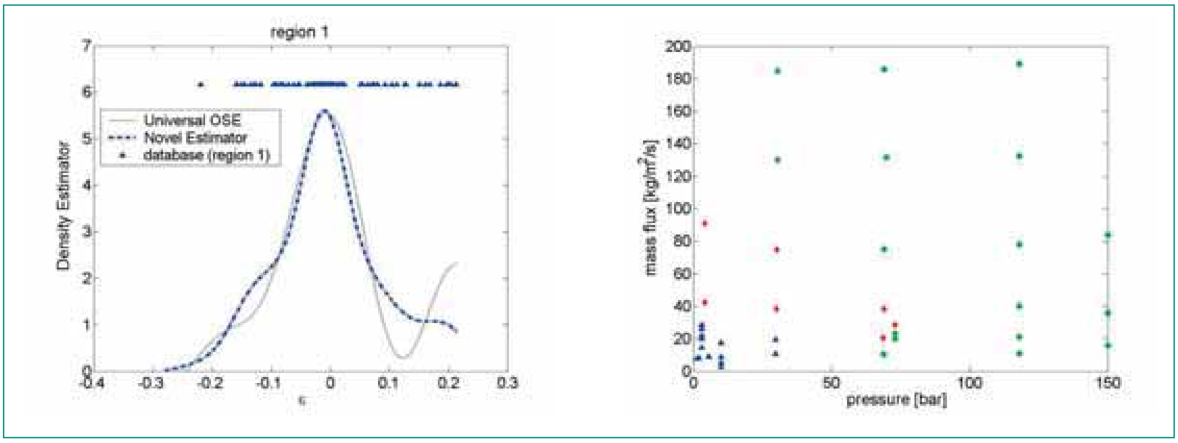

Figure 1: *Pdfs base on the non-parametric estimator for void fraction uncertainty (left); two-dimensional clustering of the void uncertainty information (right) (each point represents a set of experimental values).*

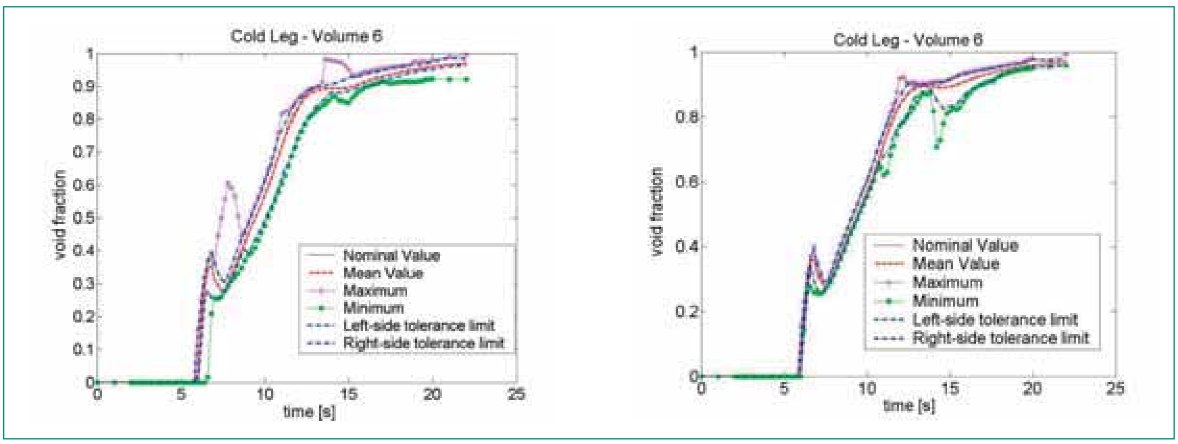

Figure 2: *Results of the application to the OMEGA ROD BUNDLE test No. 9.*
«Expert Opinion» uncertainty in drift-flux model (left); uncertainty in drift-flux model as a function of state space (right).

Participation in BEMUSE Phase III

STARS has participated in the BEMUSE International Programme [4] sponsored by the OECD since its initiation in 2003. The Programme's aim is to present and evaluate the state of the art in the field of uncertainty analysis applied to system analysis codes used in nuclear safety. BEMUSE consists of two main steps, sub-divided in three phases each, which will extend till the end of 2007. The first one, finished at the end of 2005 with the completion of Phase-III, involved the application of different code uncertainty methodologies to a loss of coolant (LOCA) transient in an integral test facility. In particular, the transient L2-5 carried out in the LOFT facility was selected for this purpose. A number of organizations from several countries, including PSI, have taken part in the first phase of the Programme, and provided results of the application of their selected uncertainty methodologies to L2-5.

The work related to BEMUSE Phase-III this year focused on the application of the code uncertainty methodology employed in STARS to the analysis of the LOFT L2-5 [5] Test transient with the system analysis code TRACEv4.05. For this purpose, a database of uncertainty values for input variables describing the L2-5 Test set-up, and for the most important models that affect the development of a large break LOCA was created based on information from the test documentation, the TRACE assessment manual, the open literature and some work on assessment of CHF and post-CHF heat transfer (also described in the present report). The uncertainty information thus compiled was used as input for the STARS uncertainty methodology, which is based on developments by McKay,et al. and GRS [6], adapted in STARS to the code TRACE. In summary, the methodology is based on the concept of sampling of the input and model uncertainty information (generally in the form of probability density functions, *pdf*s), which, after a series of code executions, generates a sample (a collection) of code results for the variables of interest, e.g. pressure, temperature, etc. The size of the sample is determined by applying the Wilks [2] formula, yielding a minimum number of execution of 93 cases for $\beta = 95\%$ and $\gamma = 95\%$.

A random sample of size 150 was generated with the help of the statistical analysis code SUSA. The 24 input variables and related code physical models considered include: Critical Heat Flux (CHF), minimum film boiling temperature, critical flow (subcooled, saturated and two-phase), single phase forced convection (liquid and vapor), interfacial drag, and nucleate boiling heat transfer. Each of the 150 sample elements consists of a combination of 24 parameter values (one per variable), which represent the uncertainty in inputs and models, and it is used to perform one TRACE execution. In total, 150 code executions were carried out in order to obtain a large enough sample of the requested output variables (93 for $\beta = 95\%$ and $\gamma = 95\%$). This was necessary because some of the code executions (~15%) failed due to numerical instabilities.

The code output variables of interest were: first and second peak clad temperature (PCT) values and timing, upper plenum pressure, time to initiate accumulator injection and time to complete the reflooding process (see Figure 3). In addition, time based tolerance intervals were computed for the hot rod temperature and the upper plenum pressure. Sensitivity measures were also calculated, so that the most important contributors to the uncertainty of the output variables of interest could be identified as the transient progressed.

All this process was carried out automatically in the framework of the uncertainty methodology developed in STARS during the BEMUSE Phase-III work for TRACE. The framework consists of three FORTRAN-95 codes which process the uncertainty contained in the input and code model sample generated by SUSA (TRACE_InpProc), and the information in the output sample for the output variables mentioned above (TRACE_OutProc, ProcessSample), so that this information can be also processed by SUSA to obtain the final uncertainty in the code's predictions. The control of all this process is automatically done by the script UncCalc_TRACE.ksh. The version of TRACE used has also been modified by the addition of a specific uncertainty calculation module to take into account the uncertainty of physical models.

Application of PSI uncertainty methodology to simple reactor physics problem

Previous efforts in the area of uncertainty in neutronic analysis have focused on the modification of the uncertainty methodology developed in STARS for thermal-hydraulic calculations and its application to depletion of highly burnt fuel. Making use of the fuel depletion capabilities of the 2D-transport code CASMO-4 [7] and of the availability of experimental measurements of isotopic inventory form the international Programme ARIANE [8], a study was undertaken to estimate the uncertainty in the isotopic inventory calculated by CASMO-4 [9]. For this application, only uncertainties in initial fuel composition, fuel depletion environment and fuel assembly geometry were considered. This year, the work focused on the extension of the methodology to include also the source of uncertainty related to the nuclear cross sections. For this purpose, CASMO-4 has been modified to process uncertainty information in the cross section data of any isotope of interest, the resulting code has been integrated in the uncertainty propagation process in a consistent way. The uncertainty in cross section data is compiled from available sources as a function of the neutron energy spectrum. Then, data tables of uncertainties for the cross section data of all the isotopes selected, are assembled in a manner that maps the energy structure of the neutronic libraries used by CASMO-4 (70 groups). These tables are read at the beginning of the computation and, during run-time, the microscopic cross-sections as a function of energy are modified with the corresponding uncertainty information in the tables before they are collapsed to compute the two-group CASMO-4 macroscopic cross-sections. During the solution of the depletion equations, or the procedures to obtain the macroscopic neutronic information, the uncertainty in cross-sections is combined with the uncertainty in the input parameters mentioned above, so that comprehensive uncertainty propagation is achieved. Application of this methodology is still not completed, but during summer student stay at PSI a depletion study for MOX sample BM1 from the ARIANE program was carried out in order to complement the previously analyzed BM5 sample. Sample BM1 is especially interesting because of the availability of measured (SIMS) radial distribution isotopic density to be compared with the CASMO-4 results. Some selected results are shown in Figure 4. These two samples will be used as a basis to apply the methodology described above.

In addition, as mentioned above, the methodology requires the input of energy dependent cross-section uncertainty data for the isotopes of interest. In this regard, this year's work also addressed this need by compiling a database of uncertainty data for several important isotopes such as U^{233}, U^{235}, U^{238}, Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Np^{237}, Am^{241}, Am^{242}, Am^{243}, Cm^{242}, Cm^{243}, Cm^{245}, Cm^{246}, Cm^{244}, Zr^{91}, and additional fission products. This database contains data from several sources, but work is still required to increase the number of data sources and to perform the necessary data processing to gain a degree of confidence in the use of these data for CASMO-4 applications with the standard libraries JEF-2.2, ENDF/B-IV and ENDF/B-VI.

Analysis of selected RIA experiments from the CABRI-Programme

The need to provide the STARS project with reliable and effective fuel modeling codes, which are used to assess the safety of the nuclear fuels under accident conditions and to provide technical assistance to HSK and

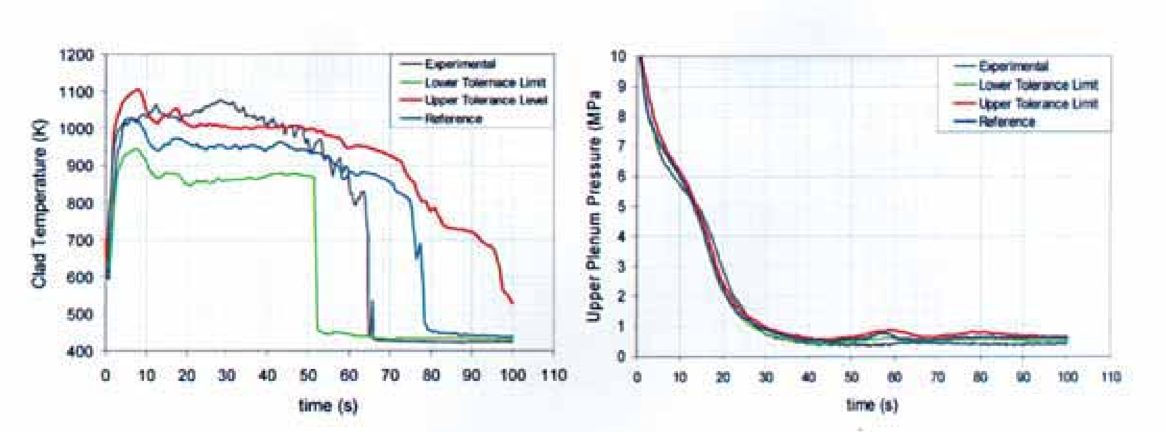

Figure 3: *Results of the application to the uncertainty methodology to TRACEv4.05 in the context of BEMUSE Phase-III for LOFT L2-5 Test. Hottest Clad Temperature (left), Upper Plenum Pressure (right).*

utilities, has driven extensive analyses targeting assessment of mainly the FALCON and SCANAIR fuel behavior codes used in STARS. Concurrent improvements have been implemented in the FALCON code that extends its versatility and capability to describe the clad deformation during RIA. Thus, a PSI-version of FALCON has been developed.

The work carried out includes:
1. a systematic comparison of SCANAIR and FALCON [10, 11] when applied to selected CABRI RIA tests [12],
2. an application of an improved version of FALCON to model the CABRI CIP0-1 test [13, 14] and
3. a sensitivity study targeting the effect of the power pulse shape on the cladding deformation in RIAs [15]. Results are briefly described next.

Table 1 shows the clad Hoop strains of five RIA CABRI tests together with the corresponding measurements (or indication of failure if applicable) [10] as predicted using the FALCON and SCANAIR codes. In general, the agreement between the two codes and the test results is satisfactory, with FALCON yielding a slight under-prediction in two tests. For the REP Na-2 test, which was characterized by a large energy injection (200 cal/g), SCANAIR largely over-predicts the clad final deformation as a result of an excessive effect of the fission gas expansion in the fuel, as modeled by this code. Similar agreements between the codes were obtained for the thermal fields as well [11].

Test	SCANAIR	FALCON	Measurement
REP Na-2	7.20/ 4.31	2.03	3.50
REP Na-4	0.56/0.46	0.43	0.40
REP Na-5	0.90/1.01	0.77	1.11
REP Na-8	1.05	0.75	Failure
REP Na-10	0.99	0.56	Failure

Table 1: *Maximum Average Clad Hoop Strains.*

Figure 5 displays the evolution of the cladding Strain Energy Density (SED) for the CABRI REP Na-5 test as predicted by FALCON when three different pulse types are considered (Gaussian, Cosecant and Pulse function) [15]. In this sensitivity analysis, the energy and pulse width have been varied from 75 to 150 cal/g and 5 to 150 ms, respectively. The analysis indicates that the effect of the pulse type is negligible and that larger SEDs are obtained when the energy injected occurs over a very short time (pulse width < 20 ms). The SED is constant for most pulse widths values and the scaling with the energy injected is fairly linear. Similar results are obtained with SCANAIR but the scaling with the energy injected is instead exponential due to the additional straining contribution of the fission gas bubble expansion in the fuel that currently is not modeled in FALCON.

Analysis of the Halden LOCA experiment IFA-650.3

The analysis of the Halden LOCA experiment IFA-650.3 was offered as a benchmark by CSNI/SEGFSM. The participants submitting solutions to the whole benchmark problem were: GRS using ATHLET-CD and ATHLET-TES-PA (two solutions), IRSN using CATHARE-ICARE, TÜV using TRANSURANUS and PSI using TRACE-FALCON.

Both the pre-test (blind) and post-test analysis of IFA-650.3 have been performed combining the application

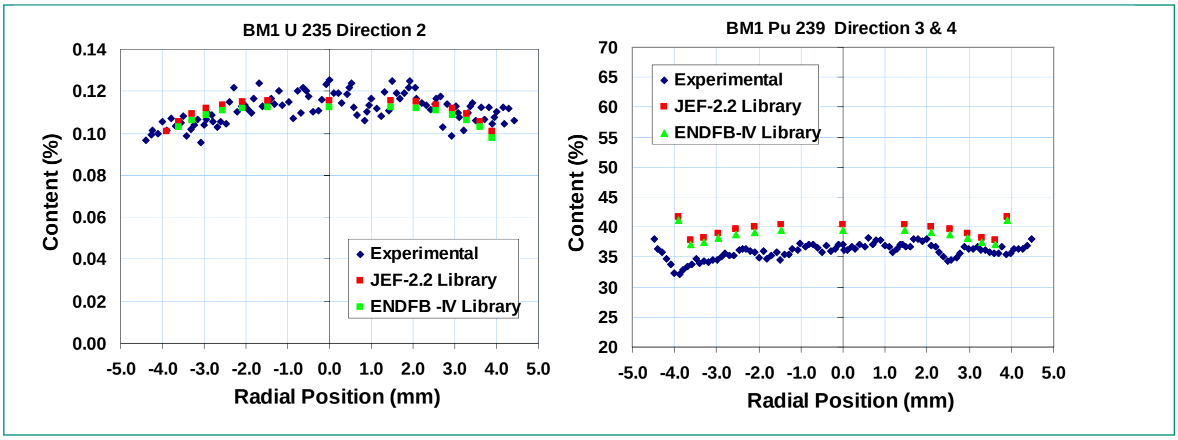

Figure 4: *Comparison of CASMO-4 predictions using JEF-2.2 and ENDF/B-IV of the radial distribution of U^{235} (left) and Pu^{239} (right) with experimental data for sample BM1.*

of the thermo-hydraulic TRACE code and the 2D-axisymetric single pin thermo-mechanical FALCON code. This methodology uses TRACE to calculate the coolant and clad outer temperatures developing in the test rig, thus providing FALCON with a thermal boundary history (clad outer temperature) in order to determine the pin mechanical deformation and failure. TRACE features a new generalized radiation transfer module which is suitable to model the heat exchanges developing between the test heater system and the fuel pin.

TRACE calculates a maximum clad outer temperature of ~ 880 °C for the blind calculation. The onset of fuel failure is recorded for a lower clad outer temperature (~ 760 °C) [16]. FALCON predicts clad ballooning (and subsequent failure) at an axial elevation z ~ 0.22 m with maximum radial deformation of ~15% and hoop stress ~ 45 MPa. Failure occurs after ~ 446 s from inception of the transient.

For the actual test [17], similar results are obtained and comparison with available experimental data is provided. Specifically, Figure 6 shows the cladding outer temperature calculated by TRACE for the two locations of the thermocouples together with the measured data. The TRACE calculations are in very good agreement with the measurements, being the cladding temperature rise after the blow-down stage and the maximum value reached during the test very closely following the recorded thermocouple signal (see Figure 6).

Figure 7 shows the final cladding deformation predicted by FALCON together with calculations from other fuel behavior codes. The final cladding strain measured in the IFA-650.3 test is also reported. FALCON predicts a maximum diametric strain of ~ 35%, located at middle height of the test rod, indicating development of cladding ballooning. FALCON results are similar to those of other codes, like ICARE (-CATHARE) and TRANSURANUS.

We note that the peak measured clad strain at failure is much lower than those predicted by the fuel behavior codes and located well below the rod middle height (at the elevation of the lower thermocouple). The explanation is found in the fact that the cladding near the lower thermocouple was weakened and in fact represented a weak spot that yielded early in the transient, causing depressurization of the rod and thus prevented the development of a balloon. This failure was not captured by any code participating in the benchmark. The test will be repeated in 2006 in order to obtain proper data on the evolution of the balloon.

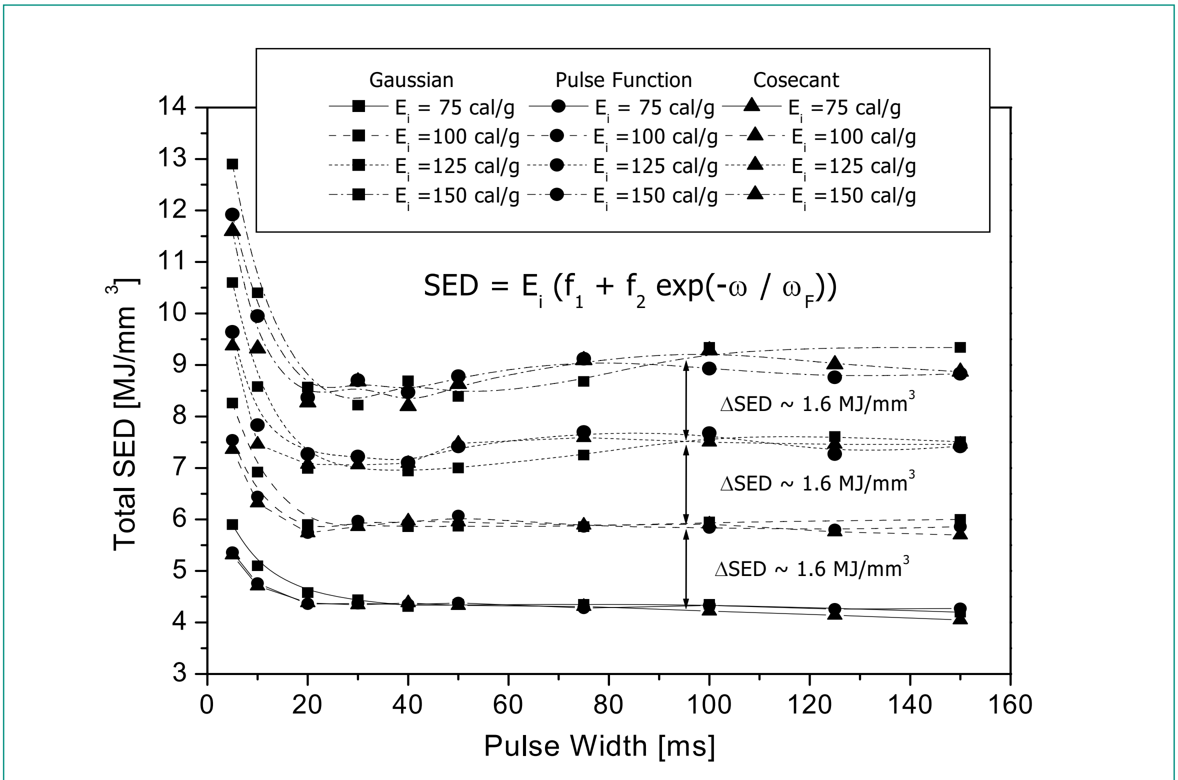

Figure 5: *Strain energy density as a function of pulse type, pulse width and energy injected for the REP Na-5 as predicted by FALCON.*

Criticality safety

The development of the criticality safety analysis methodology for LWR compact storage pools and transport casks using Monte Carlo code focused on the quantification of the calculation uncertainties mainly related to the neutron cross sections data: Estimates of the upper sub-critical (safety) limit (USL) for the effective multiplication factor k_{eff} were derived using standard neutron data libraries and is based on the statistical analysis of calculation results (k_{eff}^{calc}) for the evaluated criticality benchmark experiments [18].

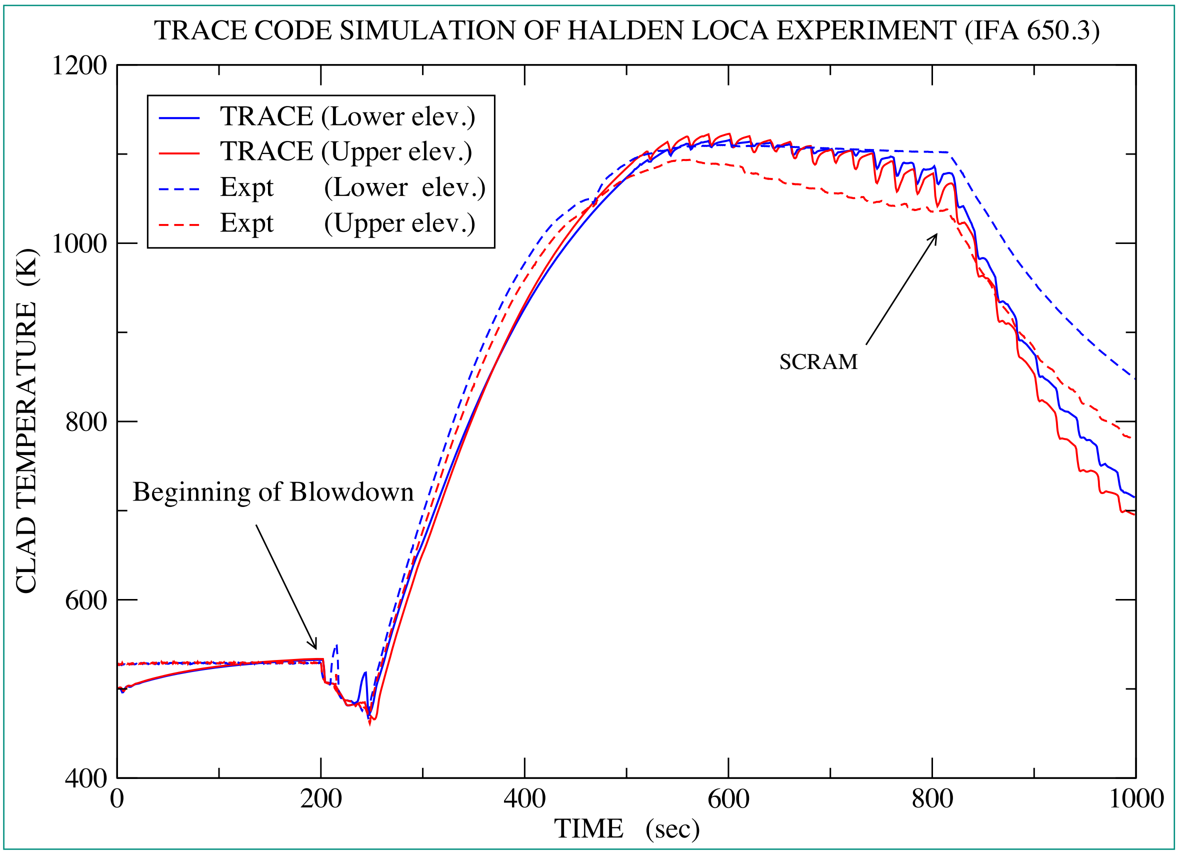

Figure 6: *Comparison of predictions obtained from TRACE and the measurements of the two cladding temperature measured for IFA-650.3. (The wiggles of the TRACE-curve represent the action of the intermittent spray that was operated during the test.)*

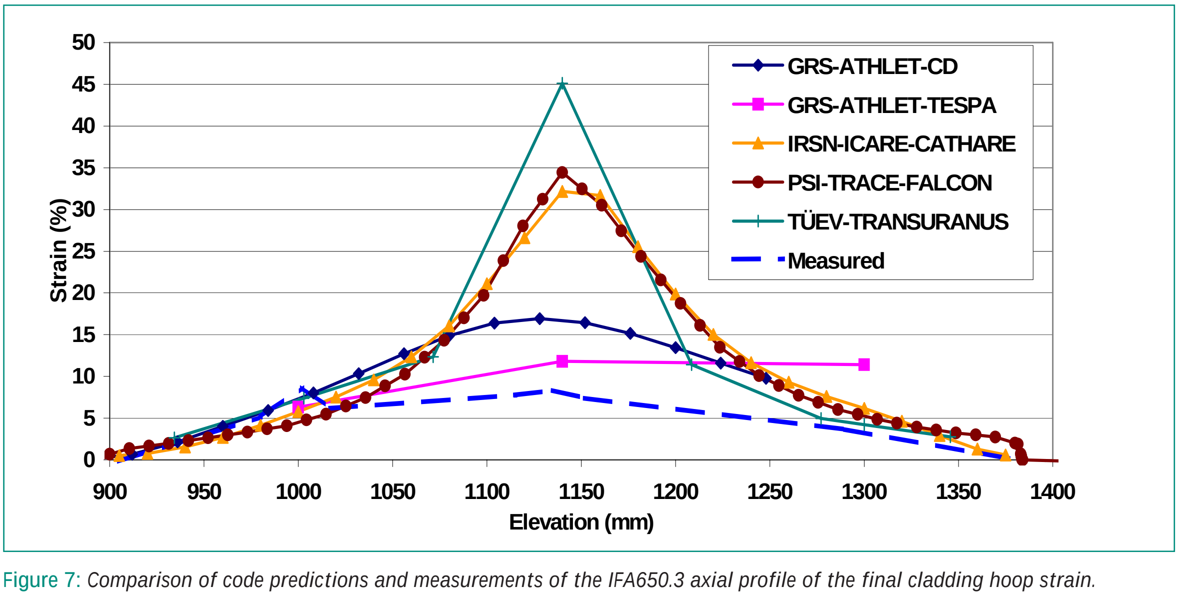

Figure 7: *Comparison of code predictions and measurements of the IFA650.3 axial profile of the final cladding hoop strain. Maximum measured strain located at elevation of ~1000 mm due to the early failure of the weakened cladding near the lower thermocouple.*

Initial studies [19] performed in the framework of the development of the PSI methodology for criticality safety assessment using MCNPX code and standard neutron data libraries were focused on a choice of statistical techniques for the estimation of the Lower Tolerance Bound and on optimization of the corresponding calculation routines. The LWR compact storage pools were selected as the first field of application of the discussed methodology and, consequently, assessment of two libraries, MCJEF22NEA.BOLIB (JEF-2.2) and FSXLIBJ33 (JENDL-3.3) [20], using the Evaluated Low-Enriched Compound Thermal Benchmark Experiments [18] has been performed with the following major steps:

- Selection and calculation of suitable benchmark configurations that are representative for LWR compact storage pool (105 configurations presently);
- Analysis of the probability distribution of the $k_{eff}^{calc}/k_{eff}^{bench}$-ratio; investigation of possible outliers and identification of possible trends found in the results; choice of applicable statistical techniques for the analysis;
- Estimation of the LTB for k_{eff}.

Findings and recommendations which were drawn from the analysis of the criticality calculations results and estimations of the k_{eff} LTB based on calculated $k_{eff}^{calc}/k_{eff}^{bench}$ samples for the considered neutron data libraries are reported in references [19, 20].

Neutron fluence calculations for KKG

A methodology for the analysis of the Swiss LWRs is currently being developed [21]. Accurate neutron transport calculations are performed (in the fixed-source mode) with the Monte-Carlo code MCNPX mentioned above and the CASMO-4/SIMULATE-3 core analysis models are used to define the fission source distributions, taking into account detailed operating history data. The validation is attempted through the analysis of selected benchmark experiments. To this aim, publicly available experiments (collected in the SINBAD International Database for Integral Shielding Experiments [22]) that are related to pressure vessel dosimetry have been reviewed, and the following benchmarks have been identified as most suitable:

- Balakovo Fast Fluence benchmark (model development already started);
- VENUS-3 Dosimetry Benchmark (model development already started);
- Robinson PWR Fast Fluence Benchmarks.

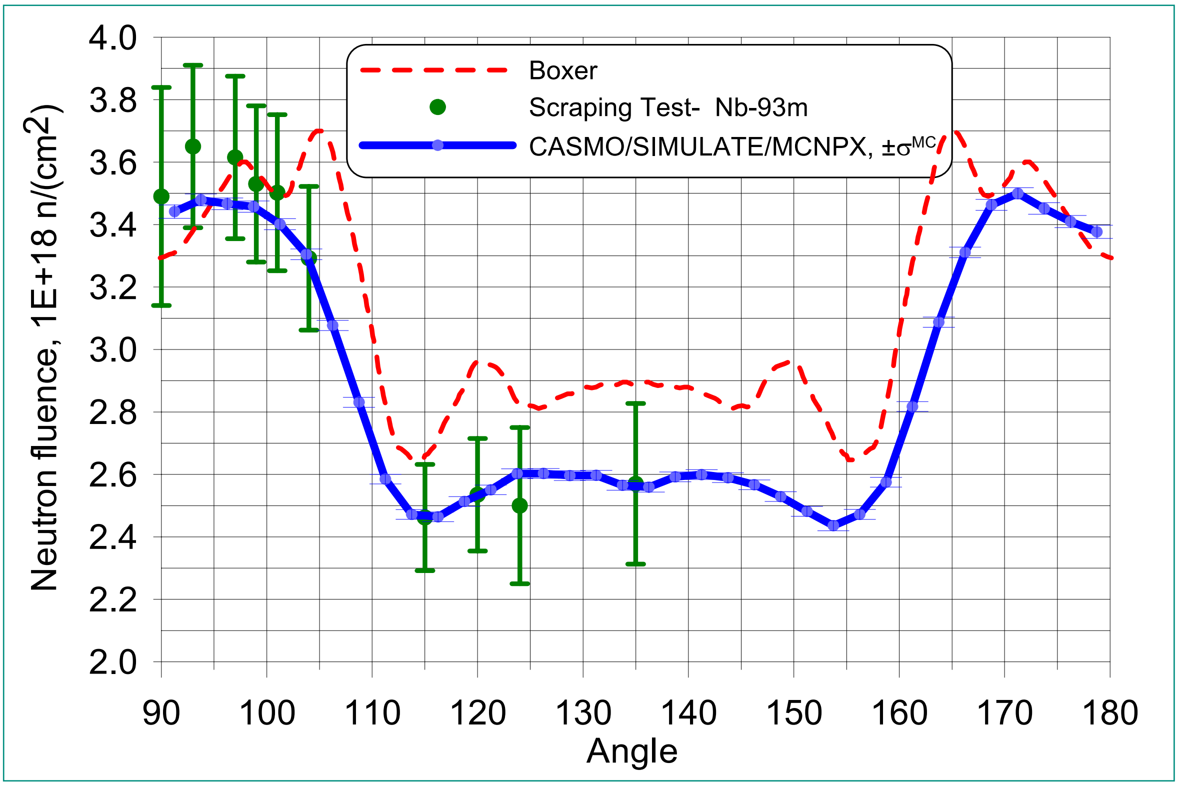

Figure 8: *Fast neutron fluence (>1 MeV) at inner surface of the KKG RPV after ten first cycles (using the JEF-2.2 nuclear data library) and comparison to the results derived from the scraping tests (Nb-93m).*

Apart from the evaluated benchmarks, very valuable scraping test data from material samples that were exposed during ten reactor cycles to the neutron flux in the KKG reactor [23] is available at PSI and represents an additional and excellent source of validation data. The significance of this data is underlined by the observation that a detailed description of the KKG operational history is available in the form of core analysis models, in contrast to rather simple descriptions available with the publicly available benchmarks.

Preliminary results of the KKG fast fluence calculations using the described methodology provide very close agreement with the experimental data (see Figure 8), indicating that quite a high level of accuracy has already been achieved. During these studies, several areas of further improvements have been identified and will be investigated next.

Analysis of selected tests from the PKL

This work is being carried out in order to assess the capabilities of the TRACE code to simulate shutdown reactor transients. The main objective is to assess the TRACE predictions of the main thermal-hydraulic phenomena which take place e.g. single-phase and two-phase natural circulation, reflux condensation, deboration and boron transport within the primary system.

A TRACE input deck was developed and verified against steady state data and two PKL experiments were analysed. Experiment E3.1 simulated the loss of residual heat removal system at $^3/_4$ loop operation conditions with the primary system closed and subsequent takeover of the decay heat removal by one steam generator which was standing by. Experiment F1.2 simulated a series of quasi-steady states at constant primary pressure of 12 bar and varying the primary coolant inventory from 100% down to about 30%. This test allows the evaluation of TRACE performance in different heat transfer modes: from single-phase natural circulation to reflux-condensation conditions.

The results of the TRACE calculations have shown that the code is, in principle, capable of predicting the thermal-hydraulic conditions under transient and quasi-steady states at low pressures. Figure 9 presents the calculated primary and secondary pressures during the PKL test E3.1: TRACE correctly predicted the primary pressure values occurring at the onset of steady-state heat removal as well as the main heat transfer phenomena observed during the experiment.

Figure 10 presents the calculated boron concentration in one of the loop seals during the PKL test F1.2. It can be noted that TRACE correctly calculates the boron transport within the primary system at various water levels in the primary system. The main heat transfer modes engaged during the test, namely single-phase

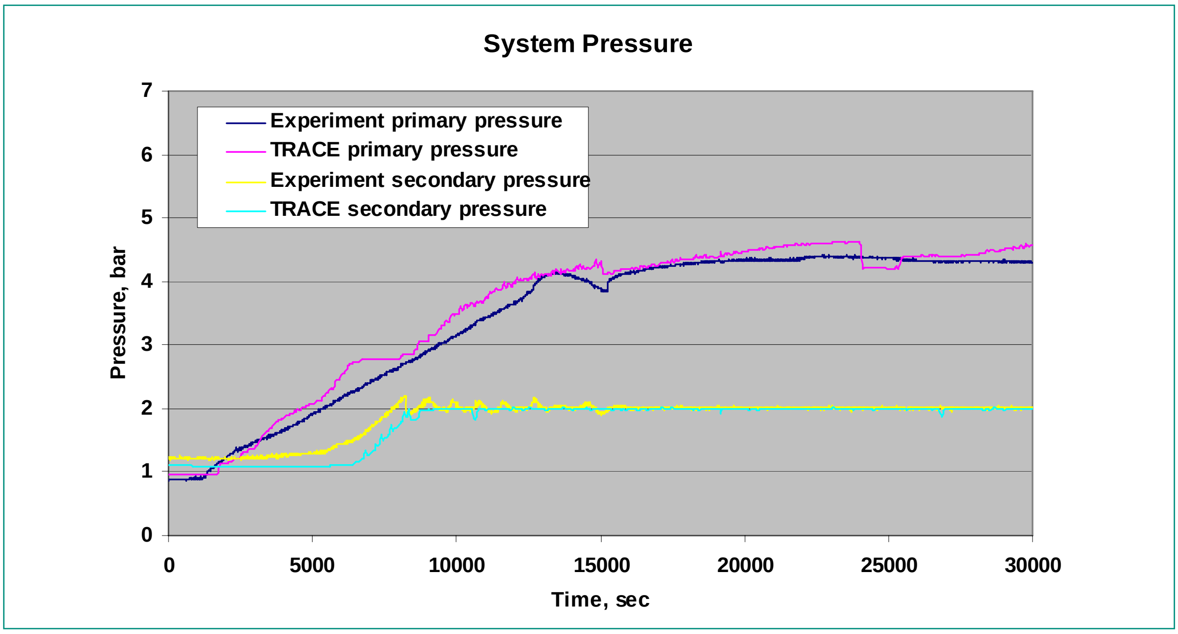

Figure 9: *Primary and secondary pressure during PKL test E3.1.*

and two-phase natural circulation, transition to reflux-condensation and pure reflux-condensation in the U-tubes as well as the return to two-phase natural circulation during the refill of the primary system are modeled correctly.

The general conclusion drawn from this work is that TRACE is correctly predicting the system behavior and the main thermal hydraulic phenomena occurring during the PKL experiments [24]: Heat transfer modes, deboration processes and other main relevant conditions are well simulated. However, it was noted that the direct condensation rates may be over-predicted with TRACE.

CHF assessment

Another contribution to the validation of the TRACE code focused on the Critical Heat Flux (CHF) and post-CHF heat transfer modes. It was noted that for certain ranges of the parameters pressure (P) – coolant mass flux (G), the error in the quality at the CHF location as predicted with TRACE may become very large, compared to the experimental results. Therefore, additional CHF correlations were added to the code in order to explore possibilities to improve the overall code predictions. In addition, it was noted that the TRACE code is not correctly predicting the film boiling heat transfer over the range of parameters investigated (P=30–200 bar and G=500–3000 kg/m^{2}s). Hence, additional film boiling correlations are needed to improve the steady state film boiling heat transfer.

Up to date, the TRACE validation against the CHF data from the Royal Institute of Technology (RIT, Stockholm) test facility were carried out for CHF and post-CHF heat transfer regimes in single tubes with uniform and non-uniform axial power distribution (featuring inlet-, middle and outlet-peaking power profiles). Additional validation was performed against experiments with annular tube geometry for uniform and non-uniform (including double-humped) axial power distribution, at a pressure of 70 bars.

Considering the whole range of parameters, TRACE was found to provide the best predictions for CHF when using the code-standard Biasi CHF correlation. However, three regions of paired values for the two parameters pressure and coolant mass flux were identified in which the critical quality as predicted by TRACE deviated 20% or more from the experimental data (see Figure 11); hence it was decided to investigate whether the use of alternative established CHF correlations would improve the overall code CHF predictions for the simulation of RIT experiments, and three CHF correlations were added to the code: Bowring, Tong-W3 and Levitan-Lantsman. It was the use of the Levitan-Lantsman correlation that led to significant improvement of TRACE predictions over the whole range of parameters (Figure 11). The calculations of the RIT experiments

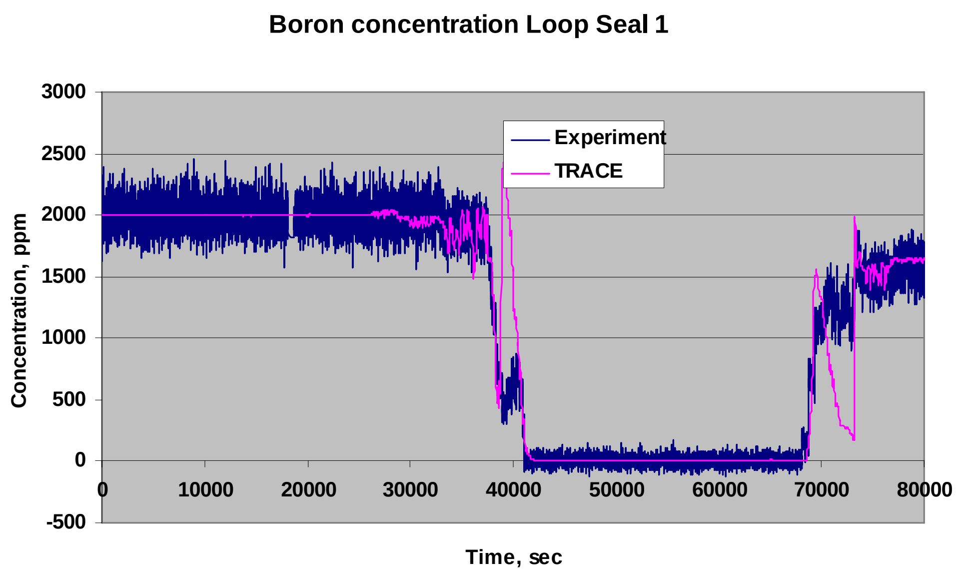

Figure 10: *Boron concentration in a loop seal during PKL Test F1.2.*

with annular geometry test sections had shown that the accuracy of the CHF predictions depends very strongly on the coolant subcooling at the inlet of the test section [25].

Film boiling heat transfer with standard TRACE correlations is not predicted correctly for the RIT experiments (Figure 12). Six additional film boiling correlations of different types were added to the code. It was found that use of the Groeneveld 5.7 film boiling correlation improves TRACE predictions, i.e. use of this correlation results in a much lower underestimation of the maximum inner wall temperature of the test section [26].

Analysis of mixing experiments in Vattenfall's test facility

The main object of the present study was the validation of the commercial Computational Fluid Dynamics (CFD) code CFX-5 for numerical modeling of the mixing aspects in the Reactor Pressure Vessels (RPV) of nuclear reactors as they occur during boron dilution transients.

Several numerical simulations of isothermal convective transport of a slug of salted water injected into the vessel initially filled with tap water have been carried out. The vessel represents the 1: 5 scale model of a Westinghouse PWR reactor (Vattenfall facility), for which experimental data is available. The problem configuration models the transport of a slug of boron-free water in the reactor filled with borated water. The experimental

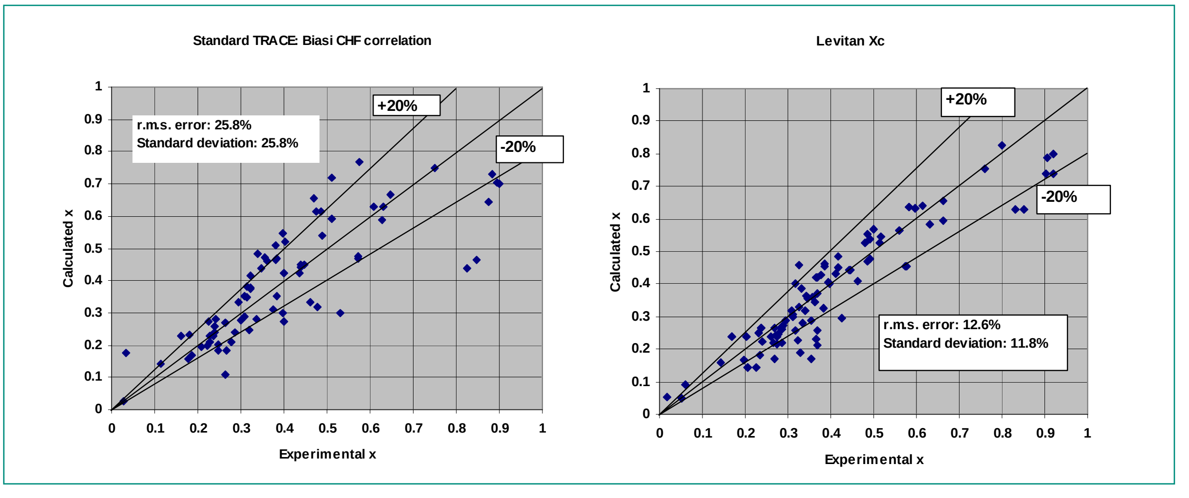

Figure 11: *TRACE predictions of the quality at CHF, using the Biasi correlation (TRACE-standard, left) and the Levitan-Lantsman correlation (right).*

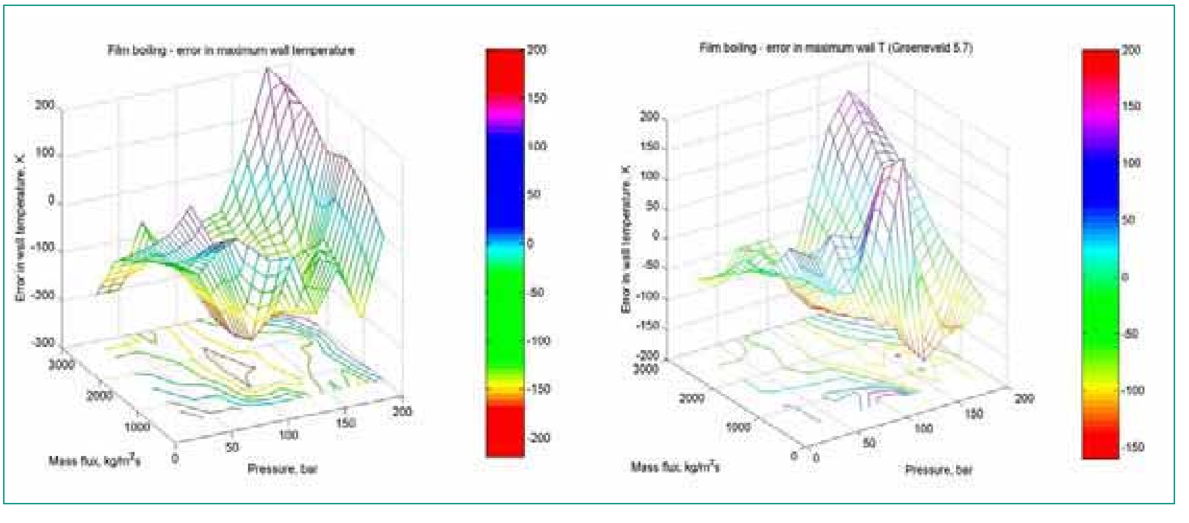

Figure 12: *Results for film boiling heat transfer: maximum errors in predicted inner wall temperature using TRACE-standard (left) and Groeneveld 5.7 (right) film boiling correlations.*

results were interpreted in such a way that salt content with volume concentration c_{salt} was understood as boron with concentration c_{boron} given by $c_{boron}(\mathbf{x}, t) = 1 - c_{salt}(\mathbf{x}, t)$.

The flow was modelled using the Reynolds-averaged equations. Both steady-state and transient regimes were analyzed. A simplified three-dimensional model of the vessel was constructed (shown in Figure 13), in which vertical columns supporting the core were not included due to lack of information on their exact dimensions and locations. The computational grid is unstructured and consists of 160 600 nodes and 434 000 cells. All simulations were performed using the CFD code CFX-5 and a 600 MHz AlphaServer workstation with a DS20 processor. The average CPU time was 2 hours for a steady-state simulation and 6 days for a transient simulation.

The flow develops a vortex pattern in the downcomer and the lower plenum. As a result, the slug of the boron-free water splits into two pockets which enter the lower plenum from opposite sides. Figure 14 displays the calculated and measured boron concentration distributions at the core inlet at the moment when the global boron concentration reaches its minimum. (From a safety point of view, this is the most interesting situation.) Thus, the experimental data confirm the splitting of the boron-free water into two pockets. Furthermore, qualitative agreement between the numerical and experimental data is observed. Also, the calculated and experimental velocity distributions of the vertical and radial velocity components in the downcomer compare well at the two elevations where measurements are available (see [27]).

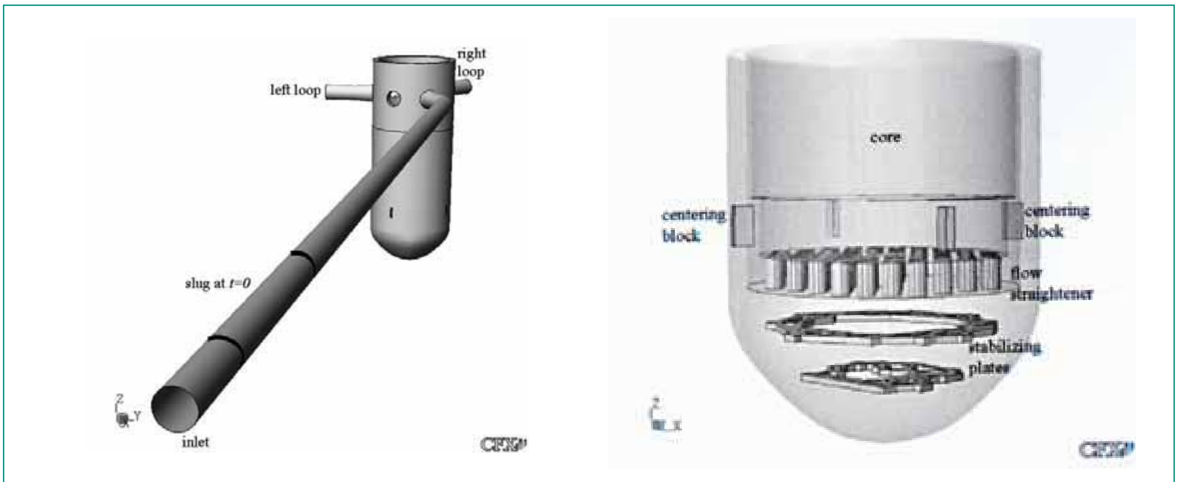

Figure 13: *Model of the Vattenfall facility: schematic of the vessel (left); schematic of the lower plenum (right).*

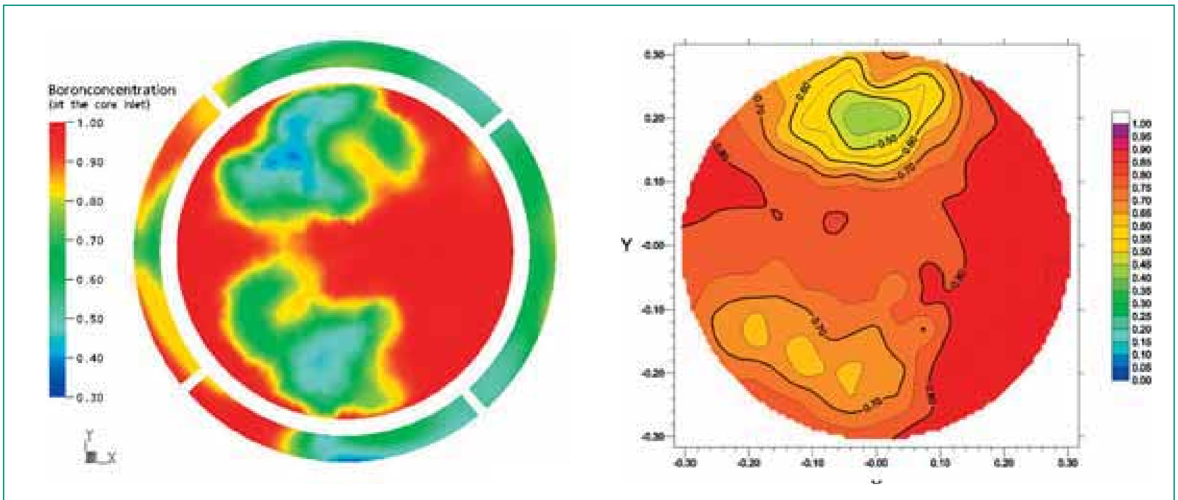

Figure 14: *Concentration of boron at the core inlet at the moment when the concentration reached its minimum: (left) Numerical solution; (right) Experimental data. Note that the experimental diagram shows the boron concentration only in the core.*

KKB main steamline break analysis

The RETRAN-3D code, which provides the possibility to represent the core of a LWR with a three-dimensional neutron kinetics model, is being used by STARS in support of the mission to develop, implement and maintain the capabilities to perform reactivity transient analysis of the Swiss Light Water Reactors using 3D-kinetics methods. As part of the continuous evaluation and improvement of the collection of input models of the Swiss LWRs developed by STARS, the RETRAN-3D input model of the Beznau-I nuclear power plant was selected and assessed for the analysis of the plant-system response during a postulated Main Steam Line Break transient. The task consisted first of adapting the existing Beznau-I model to the specificity of the Main Steam Line Break scenario and then benchmarking the results obtained with the model against a reference solution kindly provided by the Utility. The reference analysis resulted in no return to power, which allowed focusing the benchmark on the thermal-hydraulic plant-system behavior exclusively.

As it was the case in the reference analysis, the accident was postulated at the Hot Zero Power conditions. Moreover, in order to allow for a consistent benchmark, the assumptions taken for the analysis were as close as possible to those specified in the reference case.

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 in both the primary circuit and the steam generators. The transient analysis showed the consistency of the RETRAN-3D model, which could accurately reproduce the different stages of the Main Steam Line Break transient, such as the blow down in the two steam generators before the closure of the Main Steam Isolation Valves, the thermal contraction of the primary fluid and the re-establishment of a water level in the pressurizer after the safety injection. The model could also correctly predict some local effects that are important to appropriately characterize the plant system behavior, like the development of a steam dome in the upper head of the RPV (see Figure 15), which can significantly affect the system pressure evolution. One could also verify that the specific modeling assumptions assumed for flow mixing allowed obtaining a fairly good description of the temperature difference between the two primary loops, in comparison to the results obtained in the reference analysis (see Figure 16).

Although the overall agreement between the two code models was very good, the transient analysis results showed some discrepancies that were investigated more in detail. Most of these differences were very limited in amplitude and could be actually related to some uncertainty that still existed between the specifications of the two analyses. Despite these small differences, one can conclude that the RETRAN-3D model showed very satisfactory prediction capabilities, especially with respect to the main plant system parameters affecting the core response after a Main Steam Line Break acci-

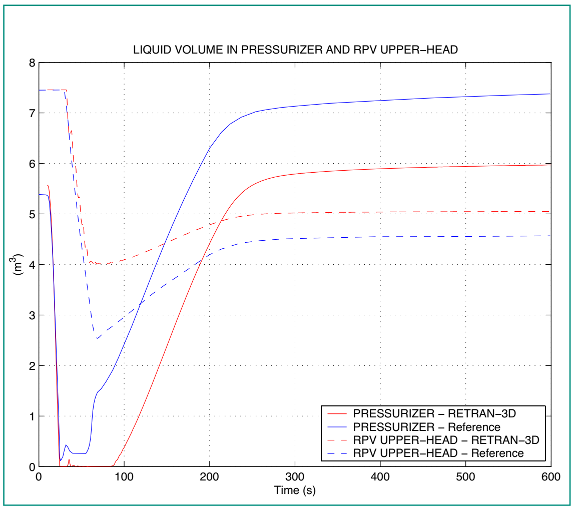

Figure 15: *Comparison between the reference and the RETRAN-3D simulation of the liquid volume in the pressurizer and the Upper Head of the RPV.*

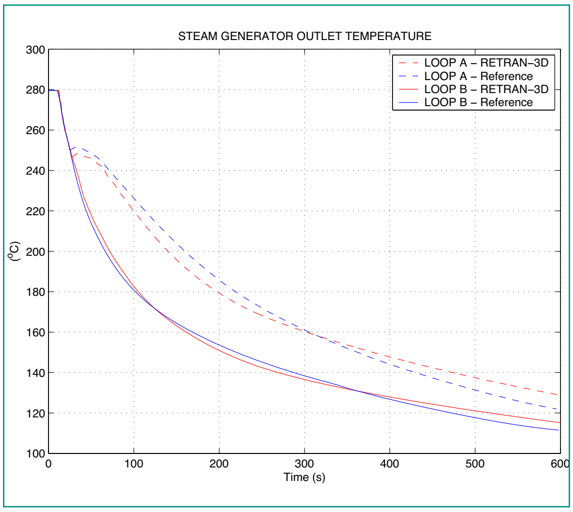

Figure 16: *Comparison between the reference and the RETRAN-3D simulation of the steam generator outlet temperatures.*

dent, namely the primary mass flow rate (Figure 17), the core inlet temperature, the core inlet boron concentration and the system pressure.

Participation in UMSICHT water hammer benchmark

Propagation of pressure waves is well known to represent a formidable challenge for modelling, particularly for two-phase conditions. For instance, the pressure waves determine the loads on reactor internals in the case of a LOCA. In order to demonstrate the applicability of the STARS codes TRACE and RELAP5 codes, experimental data of two-phase water hammer experiments performed in the context of the WAHALOADS Programme at UMSICHT Fraunhofer Institute, Germany, which was discussed during the NURETH11 water hammer benchmark workshop were analyzed. The experiment 329, initiated by a rapid closure of the main valve of the facility operating initially with an established water flow at 10 bar and 150 °C produced a water hammer with pressure wave propagation phenomena upstream and downstream of the valve.

The space-time behavior of mass flow rates and void fractions and the pressures traces at various locations of the system were analyzed. The timing of the first pressure peak occurring *downstream* the valve is well predicted by both TRACE and RELAP5 (Figure 18). First results also indicate good prediction for the timing of the first pressure peak *upstream* of the valve. The dynamics of void generation and collapse is not well captured by the two codes and is awaiting further investigation.

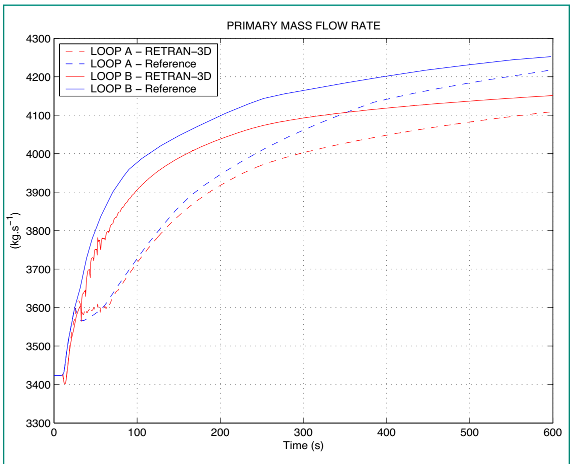

Figure 17: *Comparison between the reference and the RETRAN-3D simulation of the primary mass flow rate.*

National Cooperation

Beside the active PSI-internal collaboration within the department of Nuclear Energy and Safety (NES), STARS also maintains active collaborations at the national level mainly with HSK and the Swiss utilities. These relationships are defined with several collaboration agreements that define the work tasks. They provide substantial funding to 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 performing 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 with special consideration of the phenomena related to high burnup. These 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.

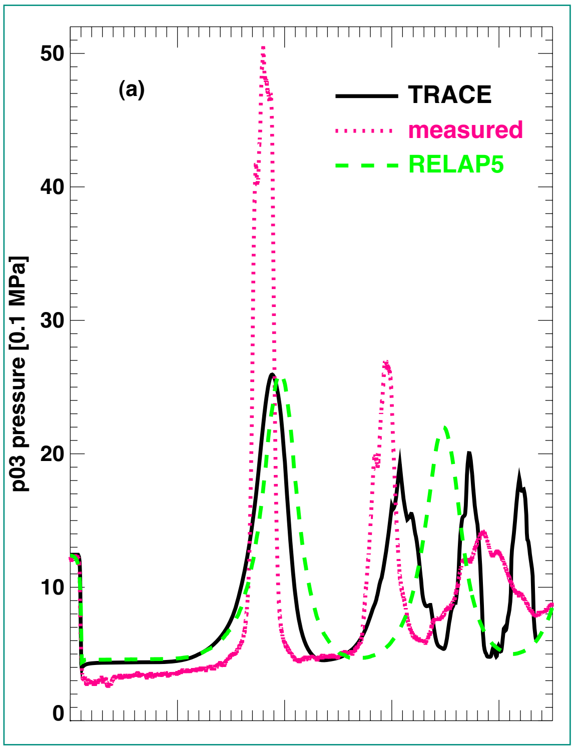

Figure 18: *Two-phase water hammer benchmark UMSICHT experiment 329: Comparison of pressure calculated by TRACE and RELAP5 at 0.2 m downstream of the valve with measured pressure.*

International Cooperation

During 2005, 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*.
- 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). During 2005, several FALCON enhancements (mainly utility features) have been developed according to the agreement with EPRI, code assessment being performed using the NFIR-burst tests as well selected Na-cooled RIA-experiments from CABRI.
- US-NRC through the CAMP-agreement, for TRACE assessment and development. Several code errors have been identified and communicated to the code development team.
- Purdue University (Prof. T. Downar) for assessing TRACE-PARCS for BWR stability analysis.

As described earlier, in the context of uncertainty analysis applied to thermal-hydraulic calculations, STARS is participating in the CSNI-OECD sponsored **BEMUSE** Programme.

Also, participation in the CSNI task group on the **Action Plan for Safety Margin** (SMAP) continued.

STARS also participated in several international research programs:

In the framework of the collaboration with the **OECD HALDEN** Project, the migration of the previously developed TRAC-BF1 model of the test rig dedicated to the LOCA-experiments (IFA-650) to the new TRACE code was completed and succesfully applied to the LOCA-benchmark offered by CSNI/SEGFSM.

The collaboration with the **OECD CABRI-Waterloop** Project first provided 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.

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

Finally, STARS participated with the TRANSURANUS code using the PSI-fission gas model in the **FUMEX-II** program that is organized by IAEA.

The collaboration with the German research center Rossendorf **(FZR)** has started with the definition of the work package for uncertainty evaluation and more recently in jointly defining the work in relation to BWR-analysis.

After considerable preparatory work, participation in the 6th FW EU Integrated Project **NURESIM** started during 2005 with contributions to the two subprojects «Core-Physics» and «Multi-Physics», the latter also being coordinated. Furthermore, STARS also participated actively in the definition of **ASTUT** that aims at BWR stability analysis.

Assessment 2005

Most of the goals defined for 2005 could be reached.

The most significant achievement during 2005 was the certification of the project management system according to ISO 9001:2000 in May 2005 with no reservations by the auditor. This just represents the tip of the support «iceberg»: Significant effort was spent in developing the project management system, a pre-requisite for the certification. Much work was also devoted to the consolidation of the existing core models; this included a reworking of the existing system of scripts automating the core analysis calculations.

An important project goal of establishing the capability for uncertainty analysis was achieved with the successful completion of the contribution to BEMUSE Phase-III that deals with the evaluation of a Large Break LOCA scenario in the LOFT test facility.

The assessment of TRACE continued with considerable effort. Based on unsatisfactory results obtained for BWR-typical problems, the planned work on the conversion of existing TRAC-BF1 inputs to TRACE inputs was deferred until TRACE has also matured for BWR applications. Indications are now that these will likely become a focus of the development activities planned in the near future by the code developer.

Assessment work for TRACE-PARCS could be organized in the framework of the collaboration with Purdue-University with the aim of BWR stability analysis. However, progress is somewhat slow as the resolution of problems with the implementation of the codes on the PSI-computing platforms is awaiting the response of the code developers.

The analysis of Halden LOCA-experiments was resumed successfully with the participation in the LOCA-benchmark organized by CSNI/SEGFSM, thereby demonstrating that TRACE offers the models needed to simulate the Halden Test Rig. Unfortunately, data on RIA-experiments with MOX have not been released as hoped; therefore, the respected work is deferred to the future.

Substantial progress was made in relation to fast fluence evaluation (as described above). Hence, state-of-the-art methods for both criticality safety assessment and fast fluence evaluation are now available within STARS.

Also, the goal of establishing CFD for single phase mixing analysis has been achieved with the successful analysis of the Vattenfall experiments. Because more studies than initially expected were needed to achieve good results, respective work towards application to a power plant is deferred to next year, even though a rough model has already been developed.

The work of fuel structure interaction initiated last year transformed into the participation in a water hammer benchmark for two phase flow conditions. It thereby addresses the need for the analysis of pressure waves that follow e.g. a LOCA-event and induce mechanical loads on the reactor structures, the topic of recent review work for HSK.

In addition, collaborators of the STARS project won the first NES seed action funding (6 person-years during the coming 3 years) for the new research activity «Atomistic Investigation of Grain Boundary Processes in UO_2».

Perspectives for 2006

The main items for 2006 are outlined below. Activities in support of the project infrastructure are not mentioned.

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:
- Analyze selected CABRI RIA experiments (UO_2 and MOX) using FALCON and SCANAIR.
- Initiate analysis of selected RIA experiments from the ALPS program.
- Participation in the Halden LOCA-experiments (IFA-650.3, 4) with TH and thermo-mechanical analysis to assess the axial relocation phenomenon.

Continue development of Monte Carlo methodology for:
- 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 for BWR stability applications in combination with PARCS.
- Assessment of condensation models.

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 6^{th} FW):
- Explore coupling capabilities of new integrated platform for safety analysis codes.
- Develop open-core analysis for KKG.

Develop new PhD-topics:
The consolidation of the existing core models was delayed by the late delivery of the respective operational data.

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