STARS

Safety Research in relation to Transient Analysis of the Reactors in Switzerland

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ABSTRACT

During 2015, progress was achieved with regards to most of the objectives. On the plant behaviour side, the TRACE modelling capability and the associated assessment was enlarged to PWR Steam Generator Tube-Rupture and Station-Black-Out scenarios as well as to BWR LB-LOCA simulations. The verification of TRACE for Critical Heat Flux calculations was also continued with a widening of the validation basis to transient experiments. Significant progress was also achieved with regards to CFD methods, both concerning validation for mixing and boron dilution as well as towards the development of Swiss plant specific models. Finally, for uncertainty quantifications related to physical models, the development and application of a novel global sensitivity analysis method to TRACE reflood models for LOCA analyses was performed.

For core physics, the transition to SIMULATE-5 was launched and a first application of a nuclear data uncertainty quantification methodology to Swiss core analyses was performed. Regarding higher-order transport full core pin-by-pin solvers, a new promising method to enhance super homogenisation techniques was proposed. Also, the direct core integral neutron transport based nTRACER code was further assessed, including a first application to 3-D cycle depletion. Concerning the SIMULATE-3K core dynamics solver, a first validation against the unique RIA SPERT experiments was launched, showing a very satisfactory performance but also strong effects from nuclear data.

Concerning fuel modelling, significant progress was achieved in the modelling of cladding lift-off phenomena through FALCON validation against Halden experiments. Also, a new method to account for fission gas trapping due to fuel-clad bonding was developed, indicating rather substantial enhancements when comparing fission gas release calculations against measurements. Finally, the co-development of Falcon V1 was continued and significant steps towards full implementation of the GRWS-A model into the official code version were undertaken.

Finally, for multi-physics, the first validation of TRACE/S3K for BWR stability analyses was conducted, indicating the need to carefully verify spatial-temporal convergence. Regarding full core LOCA analyses, a first study for a Swiss BWR plant was carried out, including significant new developments of the COBALT Loop aimed at acting as platform for integral transient analyses of the Swiss reactors. Finally, a new method to initialize 3-D core transient simulations from sub-critical core conditions was developed and verified on the basis of a NURESAFE PWR Main Steam Line-Break benchmark.

Table 1:
Goals 2015

Plant System and Thermal-Hydraulics	Completion of plant system On-Calls upon request and TRACE model upgrades for each Swiss Plant
	Modelling and analysis with TRACE of PKL-3 station black-out experiment H2.2 run 2
	Assessment of thermal-hydraulic solvers and critical-heat-flux models for BWR transients
	Refinements and assessment of KKG CFD model for transient applications
	Development of a TRACE PWR plant system model with parametrized model of CFD based coolant mixing matrices
Core Behaviour and Reactor Physics	Periodic model updates and core licensing verifications for all the Swiss Reactors
	Establishment of CPR methodology for BWR core analyses
	Modelling and validation of S3K against RIA experiments with nuclear data uncertainty quantification
	Assessment of nTRACER for PWR cycle depletion
	Nuclear data uncertainty propagation methodology for PWR core depletion analyses
Fuel Modelling and Thermo-Mechanics	Review and assessment of new fuel code licensing application
	Modelling and analysis of PWR Halden clad lift-off tests
	Establishment of FMSYS for reference steady-state methodology and models for Swiss fuel rod designs
	Development and application of Falcon uncertainty and sensitivity analysis methodology to UAM and RIA benchmarks
	Coupling of GRSW-A model with Falcon V1 code and validation for steady-state benchmark cases
Multi-Physics	Full Core BWR LOCA simulations for core-wide estimations of fuel ballooning, ruptures and dispersal
	Consolidation of COBALT methodology and TRACE/S3K assessment for BWR transients
	Establishment of methodology for coupled S3K/Falcon hot rod transient evaluations
	Coupled TRACE/S3K modelling and analyses of PWR MSLB benchmark

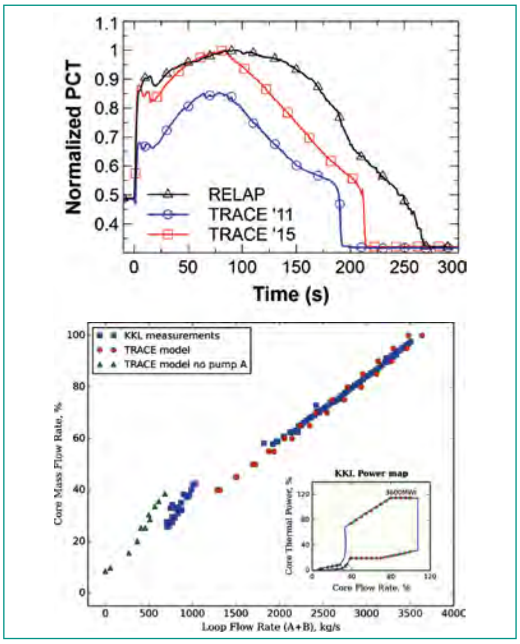


Figure 1:
Top: Benchmarking of revised KKB TRACE model PCTs estimate for 45 cm Cold Leg break LOCA; Bottom: Verification of revised KKL TRACE steady-state controller at various operating points.

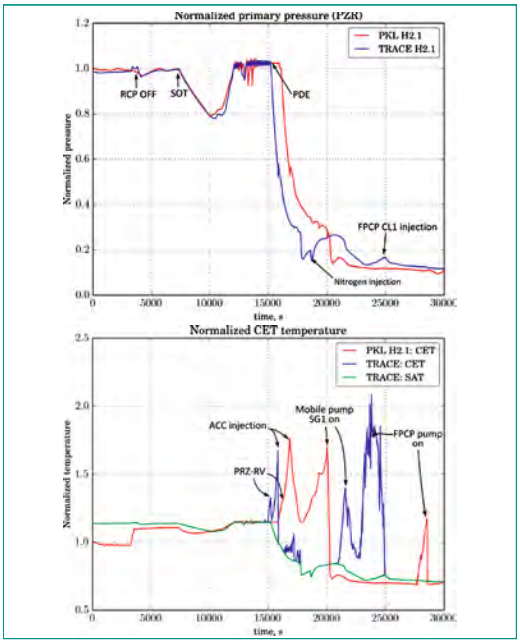


Figure 2:
TRACE assessment for PKL-3 SBO Test H2.1: Pressure (top) and Core Exit Temperature (bottom)

Project goals

The STARS collaboration with ENSI aims at scientific support and research related to multi-physics multi-scale modelling and simulations of Light-Water-Reactors (LWR) with emphasis on best-estimate safety analyses with uncertainty quantifications for the Swiss reactors. Within this framework, the objectives for 2015 are listed in Table 1.

TRACE modelling and analyses for the Swiss reactors

During 2015, the fleet of TRACE models for the Swiss reactors have been further updated. First, the KKB model has been significantly revised as part of the execution of a Steam Generator Tube Rupture (SGTR) audit analysis for ENSI, including a complete overhaul of the control system, point kinetics model, updated pressurizer (PZR) and SG nodalizations and resulting in several sensitivity studies to determine impact of analysis assumptions (single failure, initial power, decay heat) on SG levels and eventual releases. The KKB model revision has been verified for non-regression based on a Loss-Of-Coolant Accident (LOCA) analysis from 2011. As shown in Fig. 1 (top), the new model resulted in Peak Cladding Temperatures (PCT) in better agreement with a reference RELAP5 analysis, thanks in particular to a corrected decay heat control in the new model. Also, the KKL model has been improved for more realistic response to fast core power variation events. The model proved apt for precise initialization at various points of the powerflow map (Fig. 1, bottom), thanks to a more robust control system. Similar revision to the KKM model has been initiated, with first assessment at nominal conditions, using a point kinetics model to capture the core behaviour.

TRACE modelling and analysis of PKL-3 ITF experiments

The TRACE code and nodalization practices have been further assessed during 2015 on the basis of Station-Blackout (SBO) tests performed as part of the OECD/NEA PKL-3 Project. One objective of the SBO test series H2 is to assess the Core Exit Temperature (CET) signal as a criterion for accident management (AM) action to prevent core damage, with studies of procedure variants to explore efficiency and safety margins. All scenarios assumed failure of all safety injection systems and steam generator (SG) feeds. The TRACE model captured all AM actions and confirmed key tests results; Phase A: Initial primary pressure decrease and recovery due to competition between core decay heat and combination of SG water boil-off and facility heat losses (Fig. 2, top), thus affecting Secondary-side Depressurization (SDE) actuation time (high pressurizer water level criterion); Phase B: Core heat-up and Primary-side Depressurization (PDE) actuation based on CET criterion, resulting in accumulators (ACC) injection and temporary core refill; Phase C: AM action to feed the SG(s) using mobile pump(s). As shown in Fig. 2 for both primary pressure and CET evolutions during Test H2.1, the model accurately captured the timing of Phases A and B until PDE, whereas divergences in PDE actuation time and CET later evolution calls for further analysis of coolant distribution in the vessel and effect of nitrogen from ACCs.

Assessment of TRACE CHF models

While assessments of TRACE for stationary critical heat flux (CHF) conditions has been extended in 2015 to the Stewart-Groeneveld's and the Tadeka single tube tests, the work has been extended to transients and more complex geometries, starting with the Studsvik FIX-II transient tests, relevant to BWR conditions (P=70 bars, G=830–1160 kg/m^{2}s) and performed on a full-length 6x6 rod assembly. Code versions v5.0RC3 and v5.0Patch4 have been assessed for ten tests reflecting various power-increase and/or flow-decrease boundary conditions. As shown in Fig. 3 (top), the TRACE predictions appeared realistic when considering the wall temperature peak, with discrepancies ranging between –133 K and 32 K for both code versions (Fig. 3, bottom), whereas significant adjustment to the CHF model multiplier (CHFM) was necessary to capture the temperature descent down to saturation temperature during the power-to-flow ratio reduction phase. Also, the 40 K wall temperature offset prior to start of transient confirms the need to improve the two-phase pre-CHF heat transfer model in TRACE. Nevertheless, in these FIX-II tests, the fast evolving boundary conditions prompted entry in post-CHF regime thus over-shadowing the initial pre-CHF model error. These results show the complex interplay between boundary conditions and heat transfer regime changes and calls for better results interpretation tools, such as Global Sensitivity Analysis.

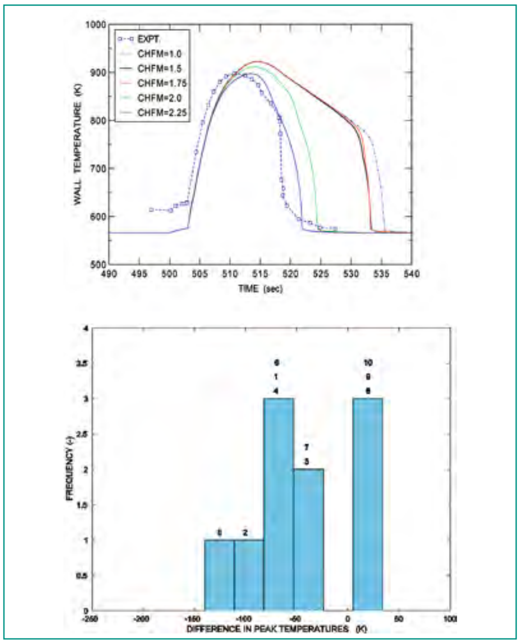


Figure 3:
Top: Assessment of TRACE v5.0Patch4 for FIX-II transient tests; Impact of CHF multiplier for Test T6291-9;
Bottom: Distribution of wall peak temperature prediction error for ten tests.

Development of CFD models for the Swiss reactors

The development of Computational Fluid Dynamics (CFD) models for the Swiss reactors was continued in 2015 with emphasis on the KKG reactor. To optimize the calculation scheme regarding meshing and convergence, a study to support a separate fluence related project and aimed at providing information on the detailed flow behaviour and heat transfer characteristics in the core bypass region was used as test case. The flow characteristics in the vessel can indeed be of importance for accurate local thermal and structural analysis or further estimations of activation of vessel steel structures. A CFD model of the core bypass region was created using OpenFOAM (see Fig. 4). A detailed analysis showed that the flow in the core bypass region is unsteady, with low-frequency vortex shedding taking place downstream of the core formers. This was confirmed by additional analyses using different meshes, multiple turbulence models and finally by an independent CFD simulation using STAR-CCM+. Two approaches were proposed to model the unsteady flow behaviour in the coupled conjugate heat transfer solution. The results yielded similar heat transfer coefficients and temperature estimates in the steel structures of the core shroud, core formers and core barrel. On its own, the study showed that heat transfer from the core to the bypass through convection and conduction has a negligible impact on the core bypass temperature. Simulations using approximated gamma heat sources in the steel structures have confirmed that direct gamma heating in the steel structures and in the core bypass coolant is more important.

Validation of CFD solvers for RPV mixing

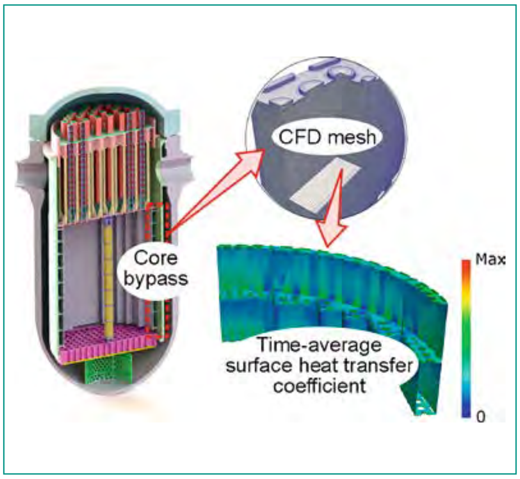
Test data from the Juliette EPR pressure vessel mock-up facility were used in 2015 to assess capabilities of CFD in simulating inherent boron dilution following a small break LOCA. The challenge was to model simultaneously the transport of two scalar fields (boron concentration and temperature). In this work, only the temperature was modelled as an «active» field, i.e. affecting the momentum equations (buoyancy force in natural circulation regime). Results obtained with the Unsteady Reynolds-Averaged Navier-Stokes (URANS) of STAR-CCM+ showed good model capabilities in capturing important features of the experiment mixing process (Fig. 5). A parametric study was performed for two parameters with high uncertainty, namely the turbulent Schmidt number (TSN, unknown in the Juliette system) and the time of injection of borated water from the Emergency Core Cooling System (ECCS) that was not recorded during the test. The former is relevant to turbulent diffusion of boron in water while the latter affects premixing in the cold legs thus potentially hindering the motion of the unborated slug when entering the vessel downcomer. Quantitatively speaking, the model confirmed large impact from the test specification uncertainties and from the TSN. Nevertheless, useful conclusions could be drawn: First, for a same TSN, the space average boron concentration at the core inlet region is low when the ECCS injection is delayed, meaning that premixing in the affected cold-leg decreases the risk of return to criticality; Second, for cases with a same ECCS injection time, the space average boron concentration at the core inlet is low when the TSN is large.

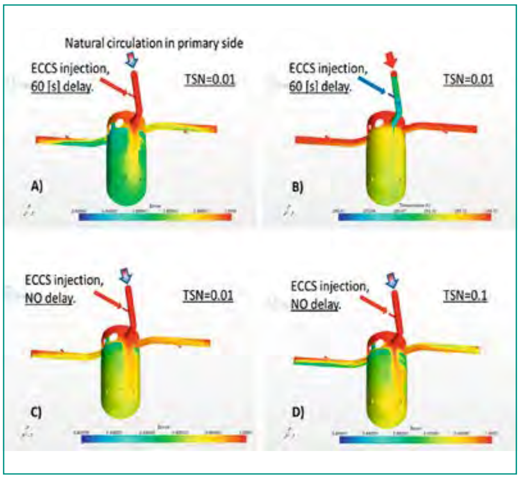
Global sensitivity analysis for TRACE analyses of reflood experiments

The analysis of the TRACE reflood models were focused during 2015 on quantifying how model parameters interact and affect relevant transient simulation outputs. As shown in Fig. 6, Global Sensitivity Analysis (GSA) has been combined with Functional Data Analysis (FDA). The latter helped capturing dominant features (modes) in the variability of time-dependent outputs due to variations in the model parameters (a). A time wraping transform separated the convoluted phase-amplitude variations, out of which few main variation modes were extracted using the Karhunen-Loève transform (b). After reduction of the problem size down to ten most influental parameters using Morris screening method (c), GSA was then applied to obtain a set of global sensitivity measures, the Sobol' indices, quantifying contributions of inputs variances to the variance of each different output mode (d). The work focused on the main- and total-effect indices that quantify

Figure 4 (left): *OpenFOAM Simulation of the KKG Core Bypass Region*

Figure 5 (right): *STAR-CCM+ results at 200 [s] of the Juliette Test 3. Boron concentration (A) and normalized temperature (B) fields for TSN=0.01 and delayed ECCS injection; Boron concentration field for (C) TSN=0.01 and (D) TSN=0.1 without ECCS injection delay.*





the effects of a single parameter variation either alone or in combination with all other parameters, respectively. The Sobol' indices showed that Dispersed Flow Film Boiling (DFFB) parameters and the spacer grid heat transfer enhancement model were most influential to the analysed FEBA test. Variance decomposition for the 1st mode, associated with the temperature ramping, resulted in very close main- and total-effect indices (d), thus confirming weak interactions between the influential parameters. Actually, only four parameters related to DFFB and spacer grid models proved to contribute to 87 % of the variance in temperature ramping. When considering the 2nd variation mode, only a third of the variation was due to the main-effects, indicating that parameters interactions contribute to most of the variation in the temperature descent prediction.

Steady-state core analysis methods for the Swiss reactors

For steady-state analyses of the Swiss reactors, the development and validation of state-of-the-art core models is conducted within the in-house CMSYS calculation system based on the CASMO/SIMULATE codes. During 2015, CMSYS was on the one hand used for independent verifications of Swiss reload licensing submittals for all plants. Related to this, the establishment of a CPR methodology was started for one of the BWR plants and despite the high complexity of implementation, the first test cases showed a satisfactory performance. On the other hand, research aimed at continuously developing new capabilities was conducted in several areas. Among other things, the assessment of SIMULATE-5 for PWRs was launched for a Swiss plant as well as for the open MIT BEAVRS benchmark and for an in-house EPR model, the two latter being aimed at complementing the verification and validation (V&V) basis. A trend for lower critical boron predictions was observed for all reactors although for BEAVRS, part of this could be attributed to a high sensitivity upon assumed irradiation history (Fig. 7, left). A second activity to be mentioned as milestone is that steps were undertaken to enlarge CMSYS with a capability for integrated core analyses with nuclear data (ND) uncertainty quantification (UQ). Concretely, for one of the Swiss PWR reactors, the SHARK-X methodology was integrated in CMSYS and applied to estimate uncertainties in all quantities of interest (QoI) for cycle depletion as well as for reload licensing related analyses. The right hand side of Fig. 7 illustrates preliminary uncertainty results (shown as 1 sigma bars around the mean value) on the Moderator Temperature Coefficient as function of burnup and operating conditions for a selected cycle. To ensure proper statistics on all QoIs, detailed convergence studies were performed to determine an optimal sample size,

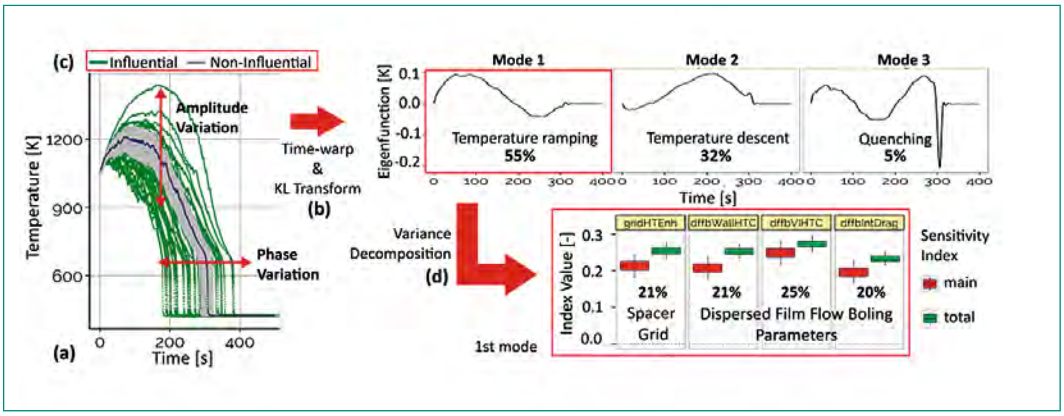


Figure 6: *Global Sensitivity Analysis of 10 most-influential reflood model parameters in TRACE for FEBA reflood tests simulations.*

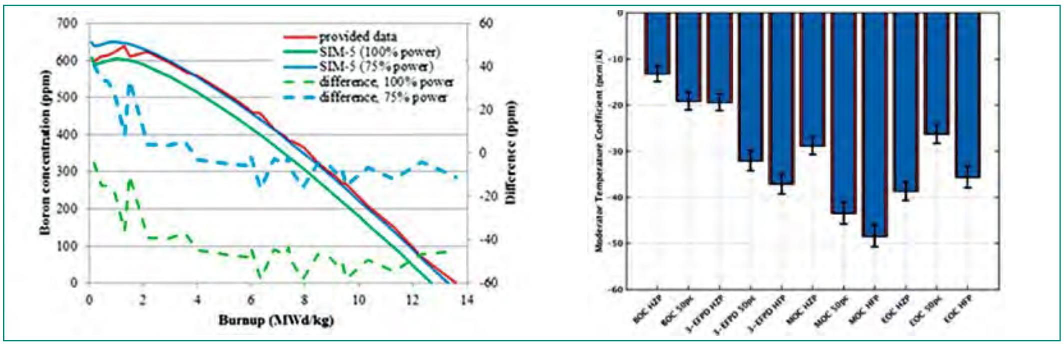


Figure 7: *Validation of SIMULATE-5 for BEAVRS Benchmark (Left) and Preliminary Results for MTC of Core Analyses with Nuclear Data Uncertainty Quantification for a Swiss Operated Cycle (Right)*

showing that 300 samples would be a minimum. Considering that each sample represents a complete model (from all lattices to all cycles), which for this relatively small PWR plant amount to ca. 10 GB data, it can easily be understood that a full deployment of such core UQ methodology constitutes a large challenge, not only in terms of computational resources but also regarding data storage and handling.

Simulate-3K validation against spert RIA experiments

The SIMULATE-3K (S3K) code acts as the primary 3-D core dynamics solver in STARS. For Reactivity-Initiated-Accidents (RIAs), the U.S Special Power Excursion Reactor Tests III (SPERT-III) tests constitute unique experiments to validate time-dependent 3-D neutron kinetics solvers. For this reason, one of the SPERT tests, used also as part of the OECD/NEA benchmark, was selected in the perspective of conducting a validation of S3K with nuclear data (ND) uncertainty quantification (UQ) for all SPERT experiments. The modelling and analysis of the UAM test 43 was thus conducted in 2015 in collaboration with Studsvik. Results of the unperturbed case (Reference) obtained with three distinct S3K models based on different CASMO-5/ENDF-B versions show that a change of code version has practically no impact on the results, while using E7.0 library provides a closer agreement to the experimental data compared to the E7.1 (Fig. 8). This clearly indicates a non negligible sensitivity upon the employed ND library. As next step, the SHARK-X methodology for the propagation of ND uncertainties in CASMO-5 was integrated in the S3K analysis of this specific SPERT test. Preliminary uncertainties in k-eff, static rod worths as well as transient power along with fuel temperatures and enthalpies have on this basis been estimated. Further investigations are now on-going in order to evaluate the impact from various variance-covariance matrices (VCM) sources as well as to enlarge the analyses with UQ to other SPERT tests. Once this will be completed, the plan will be to use these SPERT experiments as basis for further developments of SHARK-X relevant for transients and RIAs with focus on self-shielding effects as well as for treatment of kinetics data uncertainties.

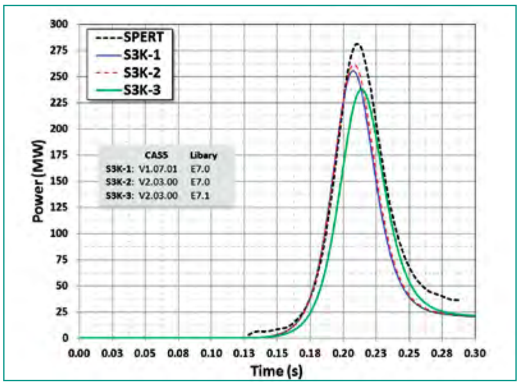
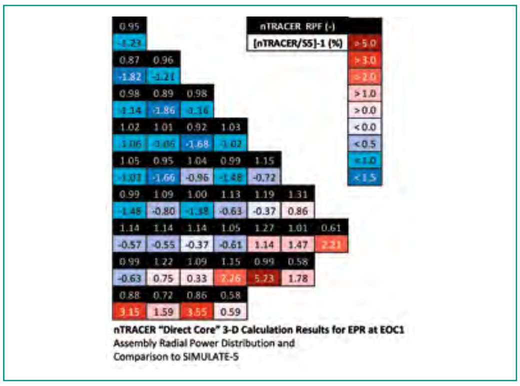


Figure 8 (top-left): *SPERT Test 43 Reactor Power – S3K Versions vs. Measurements*



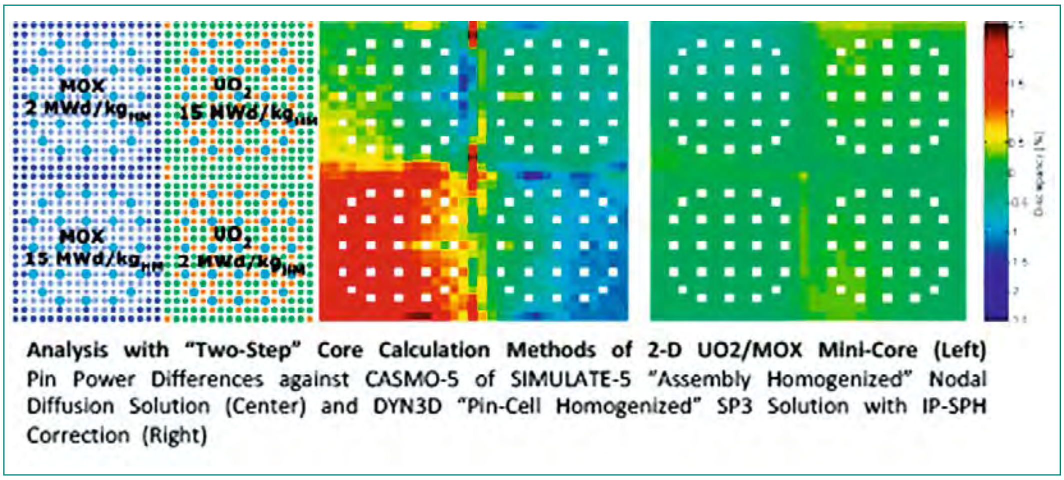


Figure 9 (top-right and bottom): *Development and Verification of Full Core Pin-By-Pin Methodologies with Higher-Order Transport on self-shielding effects as well as for treatment of kinetics data uncertainties.*

Full core 3-D pin-by-pin higher-order transport methods

Although SIMULATE-5 is intended to become the new standard code for Swiss core analyses, research towards establishing complementary 3-D core simulation methods with pin (or sub-pin) resolution was continued during 2015. Regarding «two-step» core calculation approaches with prior preparation of cross-section data, further development of Super Homogenization (SPH) techniques for 3-D core higher-order transport solvers with pin-cell homogenization was continued. Among other things, it was found that the SPH factors dependence on the assembly surface flux is approximately quadratic. On this basis, a new method was proposed in which the SPH factors are parameterized with respect of the assembly interface flux spectrum. First verifications using DYN3D with SP3 showed that the maximum pin power discrepancies against a reference transport solution could be reduced from an initial 2–3 % level to below 1.5 % and would through this, provide a clearly enhanced performance compared to assembly homogenized diffusion based methods (Fig. 9, bottom). Parallel to this, significant progress was also achieved regarding the establishment of a one-step «Direct Core» calculation approach, i.e. with on-the-fly cross-section evaluations, based on the nTRACER code. Among other things, an in-depth benchmarking of nTRACER vs CASMO-5/SIMULATE-5 following a gradual systematic approach from pin-cell to 3-D core models was carried out and a complete cycle depletion for an EPR core model could be realised in collaboration with Seoul University. The motivation for using an EPR core is simply because its heavy reflector design constitutes a challenge for deterministic core analysis methods for the treatment of the flux boundary conditions. A preliminary assessment of the nTRACER full core pin-by-pin results against corresponding SIMULATE-5 analyses tends to confirm this, even if the overall agreement can be considered as rather satisfactory for a first calculation case (Fig. 9, top-right).

Shark-X methodolgy and nuclear data uncertainties

The development of the SHARK-X methodology for propagation of nuclear data (ND) uncertainties in CASMO-5 and downstream steady-state, core dynamics and coupled core/plant transient analyses was continued during 2015. Primarily focus was given to refine the methodology for fission yields (FY) uncertainties and for the treatment of associated correlations via variance-covariance matrices (VCM). An important highlight of 2015 to mention in relation to this research is that the development and further improvements of the TENDL library, based on the nuclear reaction code TALYS and increasingly used worldwide for a continuously growing spectrum of applications, was integrated to the core behaviour research group attached to STARS. Through this new expertise, the research will no longer be limited solely to the usage of ND and VCM libraries but also to their development and improvements via a close and direct link to validation activities of high relevance for Swiss safety applications. As example, a Bayesian Monte-Carlo Method (BMC) combining theoretical FY calculations using the GEF code with an evaluated library of independent FYs to act as reference data was elaborated for four fission systems: $^{235, 238}$U and $^{239, 241}$Pu. The proposed BMC method basically constitutes a new and unique approach to provide and complement existing FY libraries (e.g. ENDFB/VII, JEFF, JENDL) with key information that has been lacking so far, namely FY correlation matrices for each fission system as well as «cross-correlations» between the fission systems (Fig. 10, left). To illustrate the relevance of

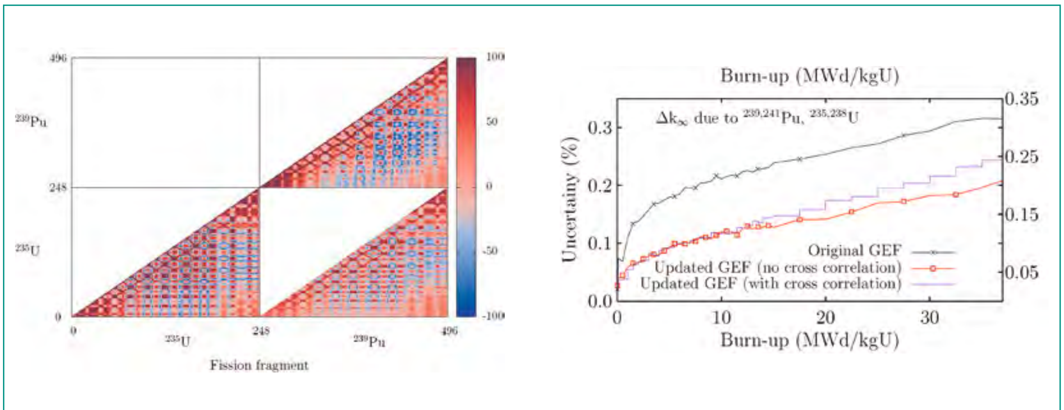


Figure 10:
Left: correlations and cross-correlations between FYs for the thermal neutron-induced fission of U and Pu.
Right: effect of FY correlations on keff uncertainties.

this, the BMC updated FYs with associated VCMs were applied for an uncertainty quantification of PWR pin-cell depletion (Fig. 10, right), confirming a substantial reduction of the uncertainties when accounting for the correlations although with a slight uncertainty increase when also accounting for cross-correlations between systems.

Validation of Falcon for Halden clad lift-off experiments

Several overpressure («lift-off») tests conducted within the Halden Reactor Project (HRP) were modelled and analyzed during 2015 with FALCON MOD01 coupled to the inhouse GRSW-A model for fission gas release (FGR) and swelling. The major goal of these experiments and calculations is to reveal the crucial processes and specific features of the high-burnup fuel behavior and, eventually, to explore critical conditions for rod failures caused by very high internal gas pressures. The following conclusions were drawn from the IFA-610.10 test analysis (BWR). Based on the analysis of the calculated and measured centerline- and normalized temperatures (see Fig. 11): (1) The critical overpressure for geometric-gap opening («lift-off») depends on the level of power; (2) for a Linear Heat Generation Rate (LHGR) of 11 kW/m, the geometric-gap opening happens at an overpressure of 250 bar; (3) for an overpressure of 150 bar, the «lift-off» is expected at an LHGR smaller than 7 kW/m; (4) after geometric-gap opening, there are indications for a secondary gap closure after power/overpressure dips and rises, presumably caused by a fuel fragment relocation into the open-gap space. A partial pellet-cladding bonding could facilitate this relocation. The analysis of the cladding elongation revealed that (1) the extent of pellet-cladding bonding is essentially lower than in all PWR cases considered so far; (2) an average fractional bonding of about 33 % can be estimated; (3) a reduction of the thermomechanical impact of bonding can be seen at an overpressure increase from 150 to 250 bar. The IFA-610.1/3/5 test analyses (PWR) based on the data for fuel temperature and cladding elongation indicate a strong bonding between fuel and cladding in the segments used in all the considered PWR samples. No effects of geometric-gap opening could be seen for overpressure levels as high as 300 bar. The fuel fragments remained adherent to the cladding inner surface throughout the test, regardless of over-pressurization level, power history and cladding temperature. The thermal feedback effect of the imposed overpressure seems to be caused mainly by: (1) fuel property degradation with burnup and (2) fragment rearrangement due to hypothesized formation of circumferential cracks. The cladding elongation was most likely controlled by the fuel swelling, due to strong bonding between the pellets and cladding. A fractional bonding in the PWR samples is estimated at 70–80 %, which is a factor of 1.5–2 higher than in the BWR test. All in all, considerable effects of bonding and fuel cracking/relocation were identified. It is concluded that the identified effects are out of the current traditional licensing analysis methods.

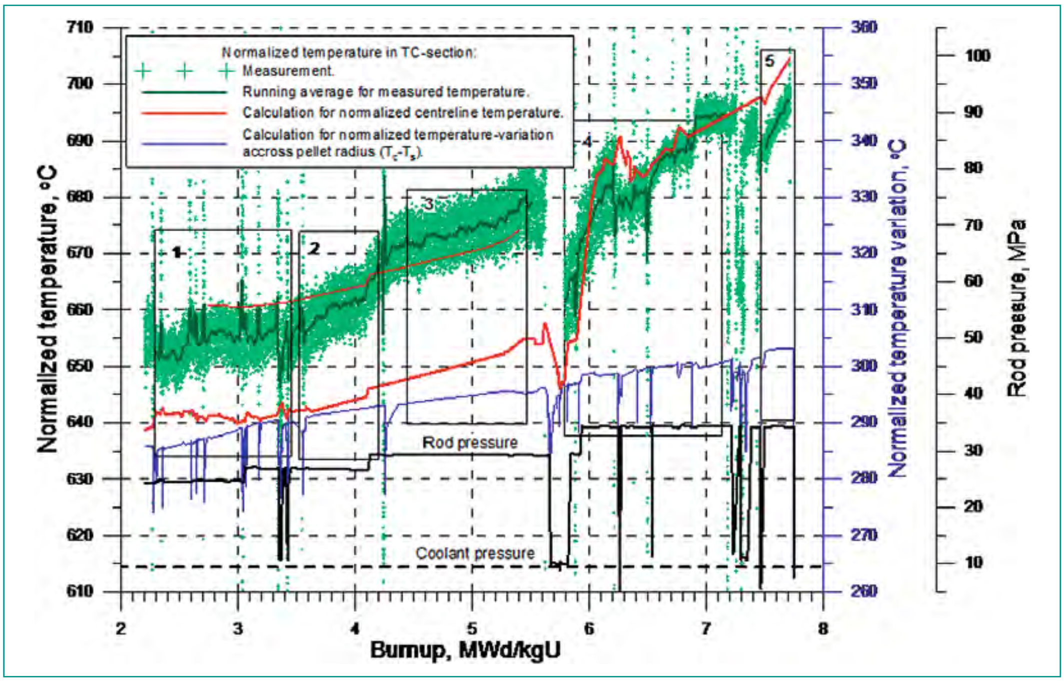


Figure 11: *Calculated and measured temperature against internal rod pressure as function of burn-up increase during IFA-610.10 test*

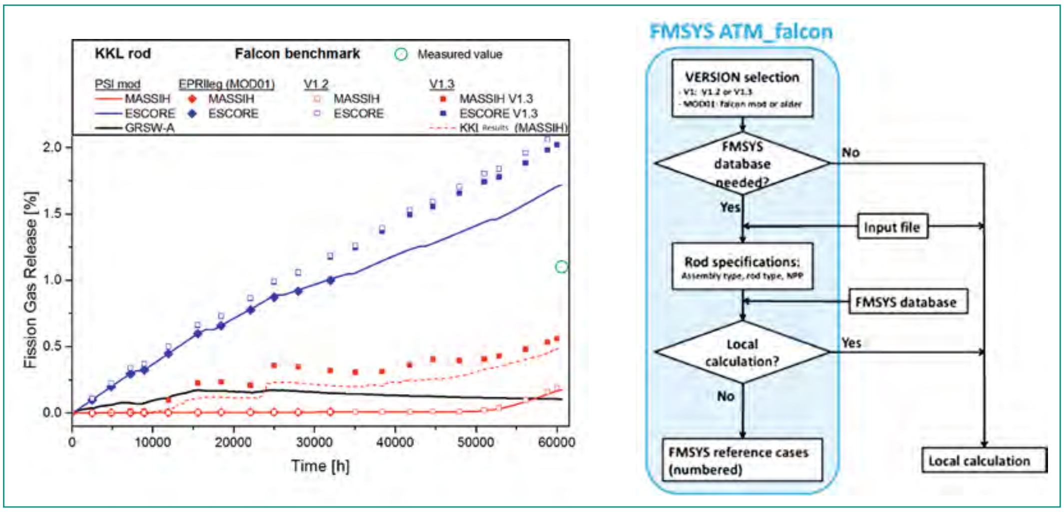


Figure 12:
Results of FGR Calculations for Reference Benchmark Case of Falcon V1; Development Activities (Left) and Development of associated FMSYS Platform (Right)

Falcon-V1 and FMSYS development

Starting from 2008, EPRI has modernized the FALCON MOD01 legacy code and structured it into a Python environment providing a graphical user interface (GUI) with HDF5 I/O support as well as extensive post-processing functionalities. During 2014, PSI/STARS became co-developer of this redesigned Falcon V1 code and in this framework, activities were continued during 2015 towards the integration of the in-house GRWS-A model for fission gas release (FGR) and swelling. As part of this, a systematic review of all Falcon code versions was carried out, starting from the legacy FALCON MOD01 version used so far at PSI (PSI MOD) as well as the corresponding EPRI legacy version (EPRILeg) up to the latest Falcon V.1.3 version which is planned to accommodate the integration of GRWS-A. The EPRI and PSI FALCON legacy code versions showed differences even if the GRSW-A model was not activated. Therefore, in a first step, efforts were undertaken to ensure a harmonization and bring both versions closer to each other. Thereafter, a regression test system was devised which guarantees the quality of the joint source code modernization and development process. To complement this, a selected KKL rod with available FGR measured data was adopted as benchmark case to verify in a systematic manner, the impact of FGR models versus code version progression (Fig. 12, left). Parallel to the Falcon V1 development, the establishment of an in-house Fuel Management System (FMSYS) to serve as platform for the maintenance and validation of reference FALCON MOD01 and Falcon V1 methodologies for the Swiss reactors, was continued (Fig. 12, right). Primary focus was given to start developing databases aimed at storing material and geometrical design data for the various Swiss assembly types and to continue the development of Application Programming Interfaces (API), including versioning capabilities, to manage the computations as function of code versions, rod modelling and meshing (with links to the fuel rod design databases) and calculation schemes (e.g. physical models and numerical methods).

Development of models for fragmentation, relocation and dispersal

Base irradiations with FALCON coupled to the GRSW-A fission gas release (FGR) model were performed on a number of high-burnup KKL BWR fuel rods (6 and 7 cycles) that were part of a fuel performance program between KKL, PSI, and Westinghouse. For all rods, the calculation over-predicts the measurement. These rods came from different fuel assemblies. The FGR measurement for pairs of rods with similar enrichment and discharge burnup but from different regions within the FA differed by up to a factor of five. On the other hand, rods that came from the same region had better agreement. Therefore the rods were segregated into three regions. The scatter in FGR measurements by rod puncture is explained with a hypothesis for fission gas trapping depending on the region. It is formulated with the help of existing EPMA data which showed significant azimuthal burnup asymmetry for the rods around the moderator channel. By contrast almost no asymmetry was observed for rods in the interior. It is hypothesized, that the trapping of fission gas depends on the degree of burnup asymmetry. The trapping itself is facilitated by the development of a permanent partial pellet-cladding bond layer, which acts as obstacle to FGR. Therefore, largest trapping is assumed for the rods around the moderator channel and smallest trapping for those in the interior, since the latter ones are exposed to a more homogeneous thermal neu-

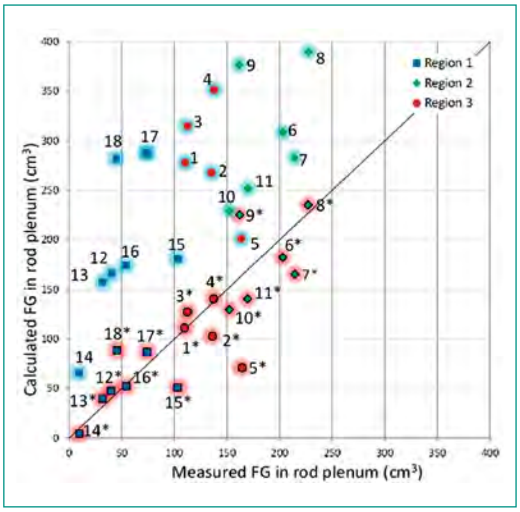


Figure 13:
Measured versus calculated fission gas release in the rod plenum without (blue shaded) and with (red shaded) new model for fission gas trapping assumption

tron flux. Based on these assumptions, a much better agreement between measurement and calculation was achieved (see Fig. 13). The hypothesis of fission gas trapping would provide a reasonable explanation for the observed scatter in the rod puncture data on high-burnup BWR fuel rods. This would imply that a strong link between FGR and rod position in a BWR FA exists. In the case of a LOCA and clad ballooning, the fuel cladding bonding will break and the trapped fission gas is released which may promote a rod rupture.

BWR stability analyses with TRACE/S3K coupled core/plant system code

Regarding coupled core/plant transient analyses with TRACE/S3K (TS3K), one of the main activities during 2015 was to start validating the code system for stability analyses on the basis of the OECD/NEA Oskarshamn-2 (O-2) stability benchmark. Although TS3K is not intended to replace S3K for Swiss BWR stability analyses, the complex non-linear and tightly coupled neutronics/thermal-hydraulics involved in such transients provide an optimal framework to test the capabilities of multi-physics coupled code systems for general purpose BWR safety analyses. And in this context, one main challenge that was encountered is that to reduce numerical diffusion, TRACE requires a non-uniform axial nodalization while such is not applicable with S3K. Therefore, using a «Base Model» TS3K with 25 axial nodes, measured transient behavior could not be reproduced (Fig. 14, top). For this reason, a systematic study was launched using different time-space discretization schemes in order to identify an optimized methodology to minimize numerical diffusion and therefore to simulate correctly the O-2 stability event. Eventually, the results showed that with a refined model in space and time, the TS3K model could capture the entire behavior of the transient, even if it was found that the simulations during the phase when power oscillations start to take place would be highly sensitive to the pump boundary conditions (Fig. 14, bottom). An in-depth comparison of TS3K against TRACE/PARCS was also carried out. A tendency of the latter to predict much higher power oscillation amplitudes compared to TS3K was in this context observed. Now although refinements of the TS3K model for further space-time convergence analyses are planned, the results obtained so far clearly represent a milestone in advancing the TS3K capabilities, noting that this analysis indeed constitutes the first reported assessment of this code system for BWR stability.

Method enhancements for multi-physics PWR MSLB simulations

As part of a participation to the EU 7th Framework NURESAFE project, a second main multi-physics activity during 2015 was the analysis of a PWR Main Steam Line Break (MSLB) transient. One primary objective for STARS was to use this NURESAFE MSLB benchmark as opportunity to widen the assessment basis of TRACE/S3K (TS3K). To this aim, CASMO/SIMULATE/S3K models were developed for the specified UO2/MOX core in order to be later coupled with an in-house TRACE model of the Zion plant. However, a main challenge was that the transient was specified to take place at hot-zero-power conditions with a -2$ core sub-criticality level, something outside of the design range of state-of-the-art transient core simulators such as S3K. Therefore, focus was shifted to assess the S3K capabilities to properly simulate such scenario before proceeding with a coupling to TRACE. In this context, a first approach was to adjust the initial control pattern and trigger an immediate partial SCRAM in order to reach a -2$ sub-criticality before start of the overcooling transient. With this model, referred to as SUB_{Crit}(p-scram) in Fig. 15, it turned out that S3K would predict a too early return to power and an overestimated reactor integral power. For this reason, a new method was implemented in the code to adjust the production matrix via weighting upon a given reactivity bias. This new approach, referred to as SUB_{Crit}(w-prm) in Fig. 15, was found to have a very strong impact on the predicted reactor power response. And despite that it had practically no effect on the coolant inlet temperature at the bottom-of-active fuel (BAF), the re-distribution of the core power was substantially altered with an increase by up to 10 % of the power in the hot assemblies.

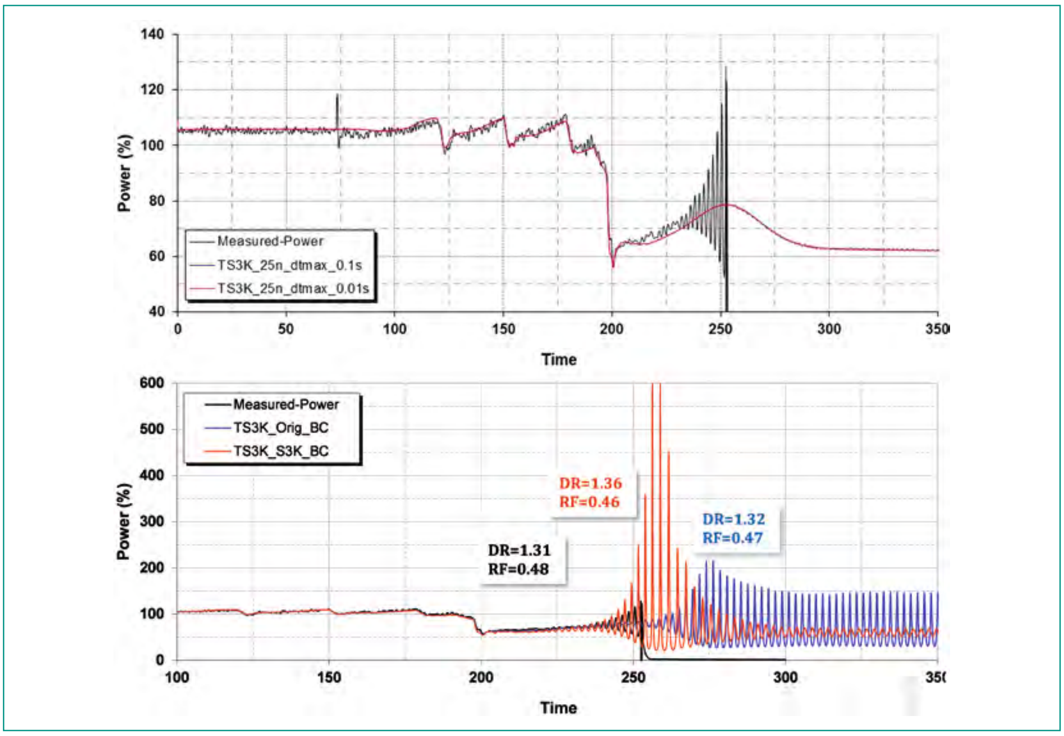


Figure 14:
Transient Analysis of O-2 Event using TS3K: Base Model (Top), Refined Model with Pump Speed BC Sensitivity (Bottom)

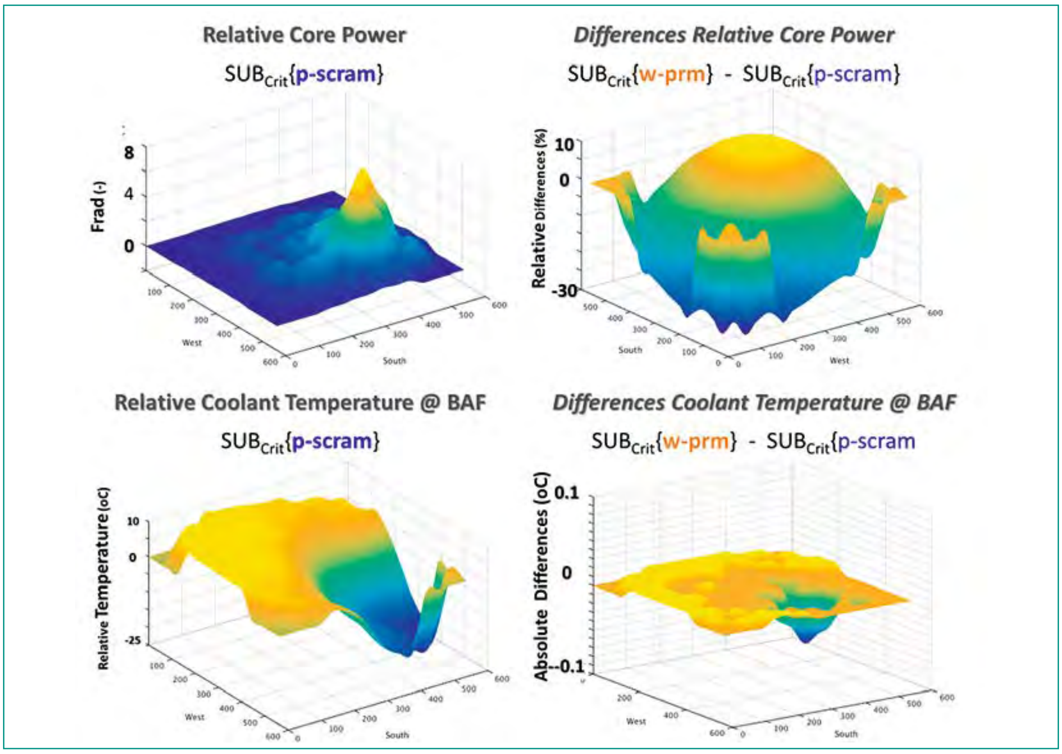


Figure 15:
Effect of new S3K Sub-critical Core Initialization Method on Solution to NURESAFE PWR MSLB Benchmark

Advanced methodology for transients and full core LOCA analyses

As part of a collaboration between ENSI, PSI/STARS and the U.S. NRC, it was agreed to conduct for a Swiss BWR/6 plant and using a recent real operated cycle as basis, a full core LOCA (FCL) analysis aimed at core-wide estimations of all the individual fuel rod responses. Such FCL analysis constitutes one of the most cross-cutting multi-physics activities since it requires establishing a computational route that integrates coupling procedures between codes and models in all areas i.e. core physics, plant system thermal-hydraulics (T-H) and fuel thermo-mechanics (T-M). During 2015, focus was therefore given to set-up such computational route via further developments of the in-house COBALT Loop. The latter refers to a methodology centred around TRACE/S3K (TS3K) to perform coupled core/plant safety analyses of the Swiss reactors with model initialisation from upstream reference validated methodologies and with downstream coupling to sub-channel and fuel behaviour codes for transient hot-assembly and hot rod analyses respectively (Fig. 16). On the one hand, a first version of the COBALT.coreFALCON module was developed in order to provide operating histories based on validated core models to FALCON base irradiation (BI) models. As part of this, a special Bundle Operating History Reconstruction (BOHR) algorithm, for which the development was also launched in 2015 for a wide range of target applications, was integrated as subroutine. On the other hand, a preliminary COBALT.falconTRACE module to initialise the TRACE heat structures from the FALCON BI fuel models as well as a COBALT.traceFALCON module to transfer TRACE predicted peak clad temperatures (PCT) to FALCON transient analysis (TA) models, were also developed. With the first versions of all these modules, preliminary FCL analyses were carried out for the selected BWR plant and core. Among other things, this involved to setup a core 3-D assembly and pin power/burnup bining algorithm in order to determine clusters of rods representative of the whole core. On the T-H side, the existing TRACE model of the plant was enlarged to LOCA analyses and various accident scenarios were investigated in order to determine a bounding case. Finally, FALCON BI and TA analyses were conducted with coupling to the PSI in-house GRWS-A model for fission gas release and swelling. On this basis, the core-wide fuel rod responses are currently being assessed prior to finalising this first study. As next steps, several refinements of the methodology are planned in order to enhance current approximations even if these further developments might be undertaken and thus shifted to corresponding analyses for PWR plants.

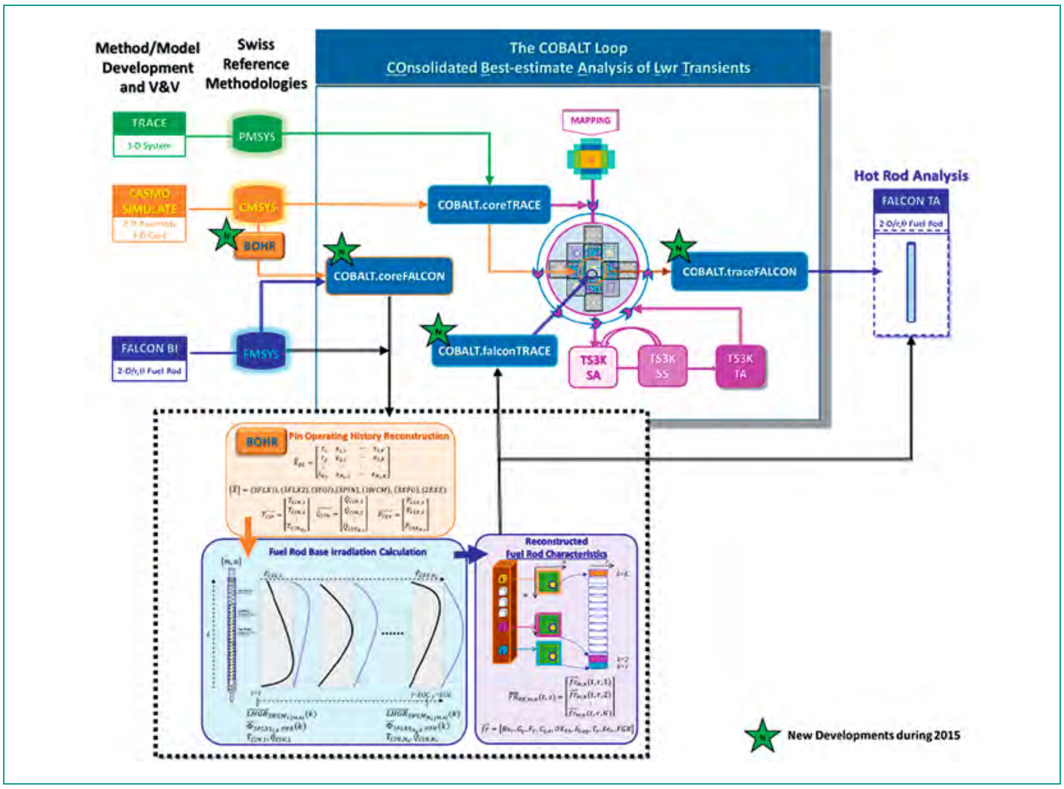
National Cooperation

To carry out its research and scientific support activities, the STARS project collaborates with ENSI as well as with swissnuclear and NAGRA for operational and waste management issues. The project also collaborates with other PSI laboratories as well as with the Swiss federal polytechnic institutes ETHZ/EPFL for the elaboration and supervision of MSc and/or PhD theses as well as for the realisation of courses for the Nuclear Engineering Master Program including «Special Topics in Reactor Physics» and the «Nuclear Computations Laboratory» course on reactor simulations.

International Cooperation

At the international level, the project collaborates with international organisations (OECD/NEA, IAEA) as part of working/expert groups as well as through international research programs. The project also

Figure 16:
New Developments of the PSI COBALT Loop and Application to BWR Full Core LOCA Analyses



collaborates with the Finnish regulatory body STUK as well as with other technical safety organisations of the ETSON network and with NURESAFE project partners. Also, cooperation with other research and expert organisations as well as industry partners was continued, including primarily Seoul National University, EPRI and Studsvik in relation to code development and validation.

Assessment

At the end of 2015, the current phase of the ENSI/STARS project will be completed. The primary outcome for the last three years can be summarised as follows.

On the side of plant system modelling, expertise was enlarged to additional Design-Basis-Accidents. In particular, Steam-Generator-Tube-Rupture (SGTR) and Station Black-Out (SBO) scenarios were added to the spectrum of events modelled with TRACE, both via validation against experiments from integral-test-facilities as well as through scientific support to ENSI. For separate-test-effects, the TRACE validation was continued with focus on the code performance for pre- and post CHF regimes, including a widening of the V&V basis to include not only stationary but also transient experiments. To complement the capabilities for CHF analyses, the COBRA-TF sub-channel solver was integrated in the STARS suite of codes and a first validation against BWR bundle experiments from an OECD/NEA/NRC benchmark was conducted. Finally, concerning CFD methods, new steps towards their usage for Swiss safety applications were undertaken, including 1) an extended validation of STAR-CCM+ for coolant mixing under natural circulation conditions as well as for boron dilution; 2) the introduction of OpenFOAM as complementary open-source CFD solver with a first promising benchmarking against ROCOM mixing tests; 3) the development with OpenFOAM of a Swiss PWR plant specific vessel model using as test case, a study of coolant flow and temperature patterns in core bypass.

For neutronics, reactor physics and core physics, the transition to CASMO-5 combined with the most recent nuclear data libraries was completed and the code was integrated as reference lattice solver for the CMSYS modelling of the Swiss reactors. Combined with this, preliminary steps towards a transition to the next-generation 3-D core simulator SIMULATE-5 were also undertaken with emphasis at this stage on PWR models. To complement this, research on higher-order transport full core pin-by-pin solvers was intensified. On the one hand, a deployment and first assessments of nTRACER as potential «Direct Core» PWR static/transient solver with the on-the-fly cross-section homogenisation was carried out. On the other hand, the static/transient DYN3D solver was also integrated in order to extend the capabilities with core simulation methods based on pin-cell homogenisation and to support research aimed at improving super homogenisation techniques for such codes. Regarding core dynamics, the development and validation of a systematic S3K methodology for Swiss BWR stability analyses was completed, including the development of new promising method to treat bypass voided conditions as well as a successful benchmarking of the S3K methodology through participation to the Oskarshamn-2 stability benchmark. The applicability of S3K with core T-H boundary conditions to new transient types such as BWR fast pump run-up was also assessed and a first validation against experimental RIA tests was conducted.

Concerning fuel modelling, the analyses with FALCON of LOCA experiments at the Halden reactor was continued, including successful modelling aided experimental test design. A PhD thesis was also launched to develop models for fuel fragmentation, relocation and dispersal during LOCAs and as part of the latter, a new method to account for fission gas trapping due to fuel-clad bonding was also established. With regards to high burnup fuel behaviour, increased focus was also given to the modelling of clad lift-off phenomena, including the development of advanced methodologies to study the impact of central void formation during irradiation at very high temperatures. For PCI/PCMI related fuel failures, selected BWR and PWR power ramp experiments from the SCIP-II program were studied with FALCON but no convincing results could so far be obtained concerning the available stress/strain based models aimed at predicting fuel rod ruptures. Finally, an important milestone is that PSI/STARS became co-developper of the Falcon code with as first objective to integrate in the official re-designed code version, the in-house GRWS-A model for fission gas release and swelling. The long-term advantage is that by being part of the code development team, all further in-house model developments will be closely linked to enhancements of the Falcon state-of-the-art code.

For multi-physics, the main new developments achieved during the last three years were focused around the coupled core/plant TRACE/S3K code system. For the numerical methods, an external coupling scheme based on using TRACE for system thermal-hydraulics only and S3K for both core neutronics and thermal-hydraulics was elaborated in order to complement the existing internal coupling approach. For the physics modelling, studies were launched to investigate method enhancements that would allow to treat strong reactivity feedback effects caused by very heterogeneous spatial void and boron distributions expected during e.g. complex scenarios such as ATWS and currently neglected in most state-of-the-art code systems. For assessment and validation, a first validation of TS3K for BWR stability analysis could be successfully completed on the basis of the OECD/NEA Oskarshamn stability benchmark. Also, the application of TRACE/S3K was enlarged to BWR flow transients as well as to full core LOCA analyses. And in relation to TS3K, an important achievement is that the COBALT Loop was established as integrated platform to couple core physics, plant T-H and fuel modelling codes and models for transient analyses of the Swiss reactors. One of the largest and unique potential of COBALT is that it opens perspectives for complementing the transient analyses with an integrated and systematic approach to propagate uncertainties from all the considered physics. Also, through participation to the NURESAFE project, a new method to initialize S3K (and later TS3K) transient analyses from subcritical core conditions was developed and verified for PWR MSLB analyses. So far, such method was beyond the capabilities of the code, limiting thereby the options to investigate e.g. return to criticality during accident management procedures.

And in relation to uncertainty and sensitivity analyses (UQ/SA), this is perhaps the area that characterises mostly where a strong intensification of the STARS research activities was undertaken with as main outcome to place the project at the front edge of on-going international developments. On the thermal-hydraulic side, novel approaches for global sensitivity analyses related to physical models were developed and applied to both TRACE reflooding models in the context of the OECD/NEA PREMIUM benchmark and COBRA-TF nucleate boiling models in the framework of the OECD/NEA UAM benchmark. For nuclear data uncertainties, a strong consolidation of the SHARK-X methodology was achieved, especially regarding fission yield treatment with an assessment basis spanning from real Swiss irradiated samples to neutronics cases of the OECD/NEA UAM benchmark. Also, a first-of-a-kind integration and testing of such advanced methodologies for real core licensing related modelling and simulations was conducted. Moreover, the recent attachment to these activities of the in-house TENDL nuclear data library will provide large opportunities for further strengthening of the expertise and capabilities in this area. Finally, also for fuel modelling, a coupling of FALCON with the CEA URANIE UQ/SA platform was established and a first assessment was made for a base irradiation case of the OECD/NEA UAM benchmark, including testing of a Sobol based global sensitivity analysis approach.

Most of the above activities remain key targets for the next phase of the STARS collaboration with ENSI which will start in 2016 and for which the scientific support as well as research program scope have recently been agreed.

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