STARS

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

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ABSTRACT

During 2012, scientific support was provided to the national regulator for the assessment of BWR equilibrium cores loaded with a new fuel assembly design. This consisted in verifying safety parameters both for normal operation and under certain dynamical conditions and to compare the performance against recently operated mixed cores. Regarding research, the validation of TRACE for the Swiss BWRs was continued with emphasis on enhancing balance-of-plant (BOP) systems for operational transients. The assessment of the code itself playing a key role, analyses of experiments carried out at integral test facilities (ITF) as well as at separate-effect-test facilities (STF) were continued. For the former, a milestone was achieved by completing the evaluation of counter-part Small-Break Loss-of-Coolant Accident (SBLOCA) tests carried out at the ROSA and PKL facilities and aimed at verifying accident management procedures during core uncovery. Regarding sub-channel capabilities for Critical-Heat-Flux (CHF) predictions, the modelling with TRACE, FLICA-4 as well as the CFD STAR-CD code of the OECD/NEA benchmark on PWR PSBT bundle experiments was also finalised with focus on Departure-for-Nucleate-Boiling (DNB) analyses. Concerning core behaviour, the SERPENT Monte-Carlo (MC) code was introduced as complementary lattice code to CASMO and a first assessment was carried out with regards to the preparation of few-group homogenised nuclear data for PWR nuclear fuel assemblies as well as for reflector configurations. On the dynamic side, the validation of SIMULATE-3K (S3K) for BWR stability analysis was continued, this time with focus on regional oscillations. And to complement S3K, the PARCS 3-D kinetics solver was also introduced although efforts were so far mainly oriented towards establishing the methodology for the initialisation of core models. With regards to fuel modelling, the design and analyses of Halden LOCA tests with the FALCON code coupled to the GRSW-A model continues to be a key activity. And in that context, investigations were carried out to understand among other things, the reasons for an unexpected clad failure that occurred during the first test, revealing that a-thermal fission-gas-release (FGR) could play a larger role than initially foreseen. Also, a validation of FALCON for predicting rod failures during power ramps and caused by Pellet-Clad-Mechanical-Interactions (PCMI), including Pellet-Clad-Interaction (PCI) failures from Stress-Corrosion-Cracking (SCC), was carried out revealing that although accurate residual strain predictions could be achieved, the available models to estimate the failure probabilities deserved further attention. Regarding multi-physics, a major advance was to develop enhanced temporal coupling schemes for TRACE/S3K simulations along with the integration of an adaptive time-step algorithm. Although further verification is now necessary, preliminary simulations of major types of LWR transients showed that the new schemes allow to significantly speed-up the calculations while keeping the accuracy in key

parameters such as power peaks at the same level if not with a higher precision than the conventional explicit scheme. Finally, for uncertainty quantifications (UQ), a milestone was achieved by completing the benchmarking of a methodology developed for the propagation of cross-section uncertainties in CASMO calculations. With these techniques, the main contributors, in terms of nuclides and reactions, to calculated uncertainties could be identified. As well, a comparison of uncertainties between various fuel designs (e.g. UO2 vs. MOX) as well as reactor types (PWR vs. BWR) could be conducted.

Project goals

The STARS project aims at research related to multi-physics multi-scale state-of-the-art computational methodologies for best-estimate safety analyses of the Swiss Light-Water-Reactors (LWR) under conditions ranging from normal operation to beyond-design-basis accidents. In that framework, the main research lines are: development and validation of reference plant system/core/fuel models for the Swiss reactors, higher-order methods, coupled multi-physics methodologies and best-estimate safety analysis with uncertainty quantifications for operating as well as advanced LWR designs. On the basis of these research activities, a central mission is to provide independent scientific support to the national regulator. For 2012, the initially planned yearly objectives for the collaboration with ENSI were adapted to accommodate for recent changes in the resource situation, leading thereby to the following priorities.

Support to Licensing of new BWR Fuel design
Enhancements of TRACE Models for the Swiss BWRs
TRACE Assessment for OECD/NEA ROSA and PKL SBLOCA Tests
Establishment of Capability for PWR DNB Predictions
Developement and Assessment of Complementary Capability for Lattice Physics Calculations
Validation of S3K for Regional Oscillations and Participation to OECD/NEA stability benchmark
Assessment of Neutronic Uncertainty Quantification Methods and Participation to OECD/NEA UAM Benchmark
Analyses of Halden High-Burnup LOCA Test 1 and Design of Test 2 with Cladding Burst
Assessment of FALCON for Modelling and Analysis of Cladding Lift-Off
Validation of FALCON for PCI/PCMI Failures on the basis of OECD/DEA SCIP-II Program
Enhancements of Temporal Coupling Schemes for Multi-Physics TRACE/S3K Analyses
Benchmarking of Methodology for Cross-Section Uncertainty Propagation with CASMO Lattice Code

This report provides an overview of the status and progress achieved for selected activities conducted in relation to the above objectives.

Scientific support

During 2012, scientific support for the licensing of a new evolutionary BWR fuel assembly (FA) design was provided. Both radially and axially, this new fuel type presents an increased level of heterogeneity with regards to the nuclear as well as mechanical/structural design. To understand the impact of all these changes on the core performance and to assess the licensing analyses submitted by the vendor, STARS conducted safety evaluations of equilibrium cycles (EQC) loaded with this new fuel design. On the one hand, lattice/core physics parameters for normal steady-state operation and ranging from beginning-of-cycle (BOC) to end-of-cycle (EOC) conditions in-cluding cold as well as hot conditions, were estimated with the CASMO-5M/SIMULATE-3 codes. A part from evaluating the obtained results against those provided by the vendor, the EQC core physics characteristics were also compared to those obtained for recently operated mixed cycles (MXC). This is illustrated in the upper part of Fig. 1 where selected parameters are compared between the EQC and one of the most recent MXC cores. There, for a given parameter, the ratio between the EQC and MXC results is shown, illustrating that overall, the new design implies a rather similar core performance as previously. Some of the observed improvements are more negative void reactivity coefficients as well as isothermal temperature coefficients at BOC, while less positive isothermal temperature coefficients are obtained at EOC. Also, the EQC core provides larger thermal margins resulting from a reduced linear power density, noting that the latter allows consequently for slightly larger power peaking factors. On the dynamic side, of particular importance was to address the eventual impact of the new fuel design on the core stability behaviour. To that aim, the recently developed stability analysis methodology based on the SIMULATE-3K (S3K) code [1] was applied to evaluate the decay ratio and resonance frequency at a selected high power/low flow operating point. As shown in the lower part of Fig. 1 where the EQC results are compared to those obtained at the same operating condition for two recent mixed cores, the EQC core yields a stronger damping of the power response to a reactivity/void perturbation, indicating thus an enhanced core stability performance. Although this evaluation was carried out only for one operating point, the results are not surprising when considering the changes in mechanical and structural properties of the new fuel design, inducing in particular a higher single-phase pressure drop at the assembly inlet combined with smaller pressure drops around spacers and in the two-phase zone (assembly outlet). Concerning the resonance frequency, it is found to increase, something in-line with the shorter effective height of the active core induced by this new fuel design.

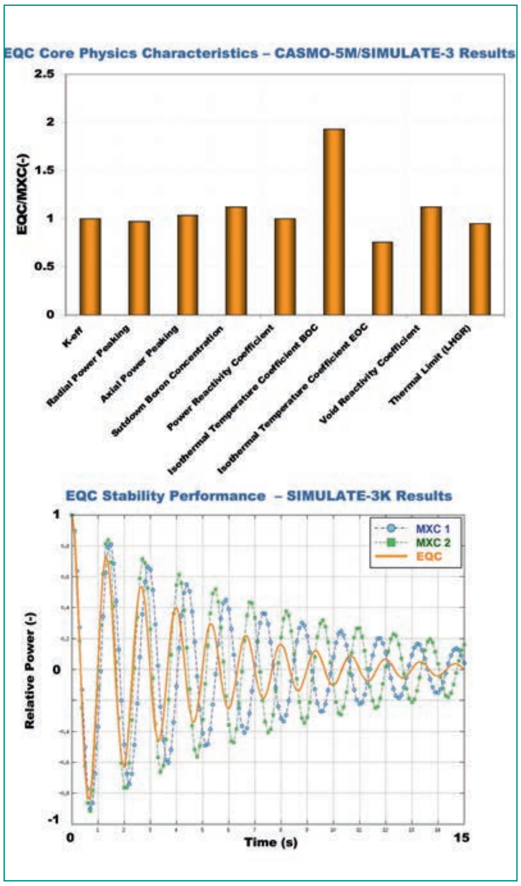
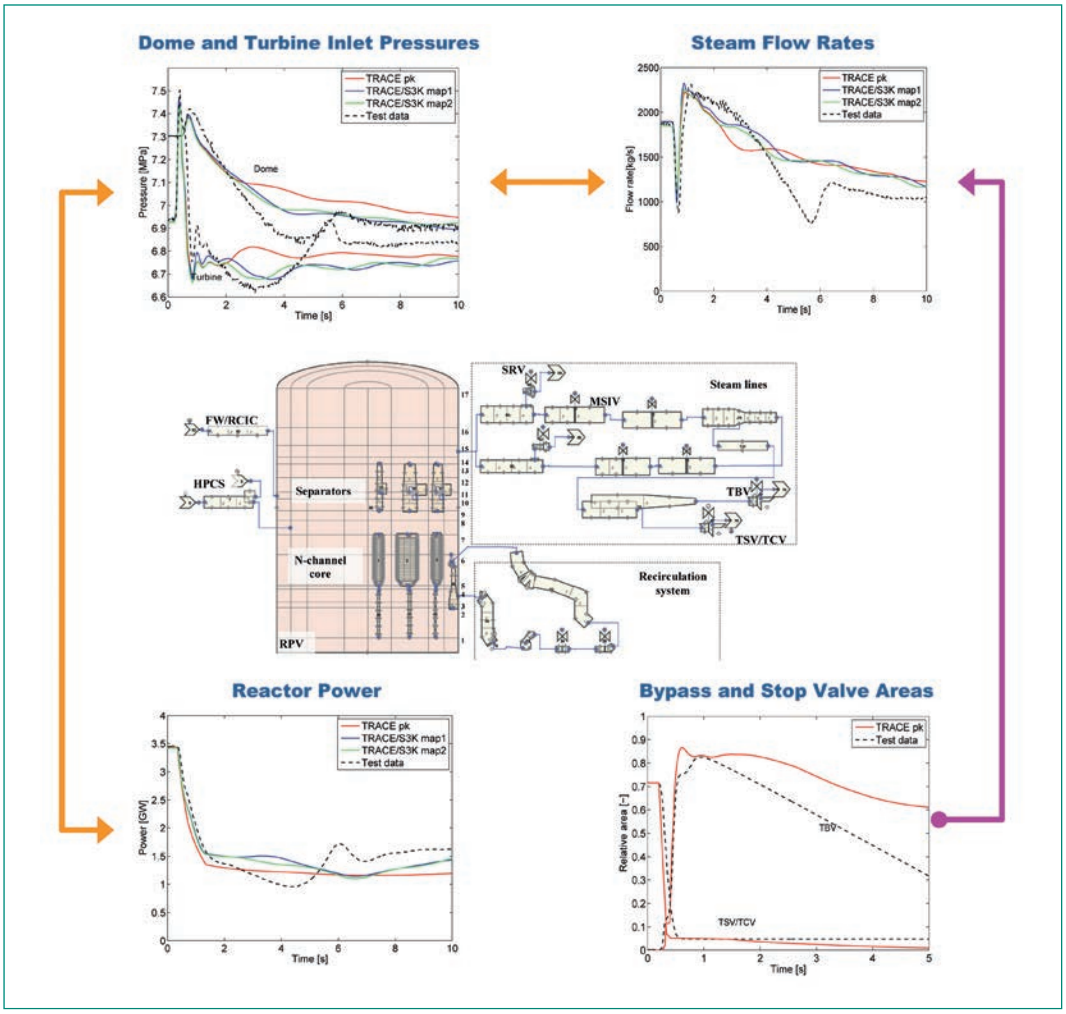


Figure 1:
Steady-state and Transient Evaluations of Equilibrium Cycle Core Loaded with new BWR Fuel Assembly Design

Validation of TRACE plant system models for the Swiss reactors

For the Swiss nuclear power plants (NPPs), simulations of the plant thermal-hydraulic behavior during transients and accidents is primarily carried out with the TRACE best-estimate system code. During 2012, the main activities in this area were focused on enhancing the performance of the TRACE KKL model for a turbine trip test carried out at the plant in 1999. To that aim, the OECD/NEA validation strategy for coupled neutronics/thermal-hydraulics simulations was adopted by decomposing the tasks into three distinct phases: 1) plant system model initialization using a point-kinetic model for the neutronics; 2) set-up and verification of the 3-D core model; 3) coupled 3-D core/plant dynamical analyses. Noting that phase 2 is implicitly tackled via the continuous valida-tion of Swiss core models, emphasis was given to address Phases 1 and 3. Especially for the former, attempts were made to update the Balance-of-Plant (BOP) models with regards to the control system in general and the actuation logic of the turbine control/stop and bypass valves in particular. As illustrated in Fig. 2, this allowed to better capture during the first second of the test, the behavior of the steam flow as well as the pressure peaks at the turbine inlet and in the reactor vessel. This is of primary importance since a steam dome pressure peak will produce a reactor power increase and even though partial rod insertion was here activated during the test, the results show that the TRACE model is able to predict this crucial phase of the transient in a satisfactory manner. However, difficulties to predict the correct long-term transient evolution appear after the initial pressure peak. Indeed, the predicted bypass valve behavior starts to deviate from measured data and this affects, via the bypass flow characteristics, the reduction of the steam flow rate and the associated increased turbine inlet pressure. For the latter, the TRACE model fails to predict the pressure recovery occurring in the time interval [3–5] seconds and moreover, the steam dome pressure increase that follows (between 5–6 s and due to pressure wave back propagation into the vessel) is not captured at all. To complement these analyses and to assess the eventual impact from the core, a coupling to S3K was as next step carried out, applying both a coarse mesh (map1) and a finer mesh (map2) mapping of the fuel assemblies with the thermal-hydraulic channels and using to that aim, the COBALT methodology currently under development [2]. As can be seen in the middle-left plot of Fig. 2, these coupled TRACE/S3K models allowed to enhance slightly the

Figure 2:
Validation of TRACE Stand-alone (with point-kinetics) and with Coupling to S3K for KKL Turbine Trip Test



prediction of the power reduction during the early phase of the transient but after the partial rod insertion, none of these models could capture sufficiently well the power evolution. Regarding the turbine inlet pressure recovery, a certain improvement is seen especially with the finer map2 model, showing thus non-negligible interactions between the predicted core behavior and the plant system response, indicating in turn that the test constitutes a rather tightly coupled core/system transient. But in that context, the results illustrate that the implementation of more detailed core neutronic methods will not allow improving the accuracy as long as the system thermal-hydraulic (T-H) code and associated BOP models are not capturing adequately the physical processes and their response to the various and inter-related complex control systems. A further review or updates of the TRACE BOP models to ensure that these are representative of actual plant systems will thus be a key priority for enhanced operational transient simulations provided that more detailed plant design data becomes available.

Assessment of TRACE code using ITF and STF Experiments

While an objective of STARS is to achieve comprehensive TRACE models of the Swiss plants, an equally important target is to continuously assess the code capabilities for simulations of the thermal-hydraulic plant behaviour during design and beyond-design basis accidents ([3], [4]). To that aim, the long-standing active involvement and participation of STARS to the OECD/NEA ROSA-2 and PKL-2 projects was continued during 2012 by finalizing the TRACE analyses of counterpart tests carried out at both facilities. These tests were designed as SBLOCA with additional system failures in order to investigate accident management procedures during core uncovery and specially, the usage of the core exit temperature (CET) to detect core heat-up and to initiate on that basis a manual secondary-side depressurization. Overall, a good agreement of the predicted integral system behaviour against plant data was obtained for both ROSA and PKL, noting that for the latter, the work was carried out with the integration of an IAEA fellowship program [5]. For the high-pressure phase of the ROSA test, difficulties were nevertheless encountered due to limitations of the choke flow as well as off-take models. Concerning detection of core heat-up, the relationship between the CET and the Peak-Clad-Temperature (PCT) predicted by TRACE for the ROSA test is shown on the left hand side of Fig. 3 noting that a CET of 623 K is taken as limit to manually trigger the secondary side depressurization. Here, two main trends could be observed. First, in the very initial phase of the transient, the measured PCT increases while the CET does not change at all and this behaviour is not captured by TRACE. Secondly, one can see that several model enhancements were necessary in order to capture the appropriate PCT/CET relationship. Most influential was found to be the modelling of the non-uniform distributions at the upper core plate of the flow cross-section and heat structures area for the selected radial nodalization of the RPV (right hand side of Fig. 3), as this will in addition to the predicted vapour mass velocity, affect the heat transfer to the passive structures and influence thereby the predicted CET. Finally, it must be mentioned that the TRACE ROSA nodalization results from several years of assessment covering a total of seven different small and intermediate break LOCA tests. In that framework, a systematic methodology to track the evolution of the TRACE model was elaborated in order to apply a consistent approach for all tests and to validate thus, the same underlying physical models under various conditions. This was done in order to ensure that such ITF assessment serves its principal purpose, namely to derive proper expertise and guidelines for a) the development of system code models for real nuclear power plants; b) the application of such models to a wide range of postulated accidents.

Regarding the assessment of the TRACE physical models and numerical methods using STF experiments, a study was conducted in 2012 to verify the code capabilities for void and level swell predictions during blowdown conditions [7]. Such conditions, leading to flashing with immediate void formation accompanied by a level swelling of the liquid-vapour free interface and a subsequent level shrink, can be encountered during BWR steam line breaks, fast depressurization transients or LOCA scenarios. Also, level swelling might occur during boil-off conditions, for instance as a consequence of loss-of-cooling capacity in e.g. a spent fuel pool [8]. To assess TRACE, the so-called «Level Swell» tests conducted by General Electric (GE) to investigate void formation and level swell and shrink as well as critical flow during blowdown, were selected. The «Small Vessel» tests were first analysed and for one of these, the reference results, i.e. obtained with a TRACE axial nodalization consistent with the pressure line taping distances, are shown on the upper plot of Fig. 4. As can be seen, a tendency of TRACE to underpredict the void fraction is observed, particularly for the lower nodes. Several sensitivity analyses were carried in

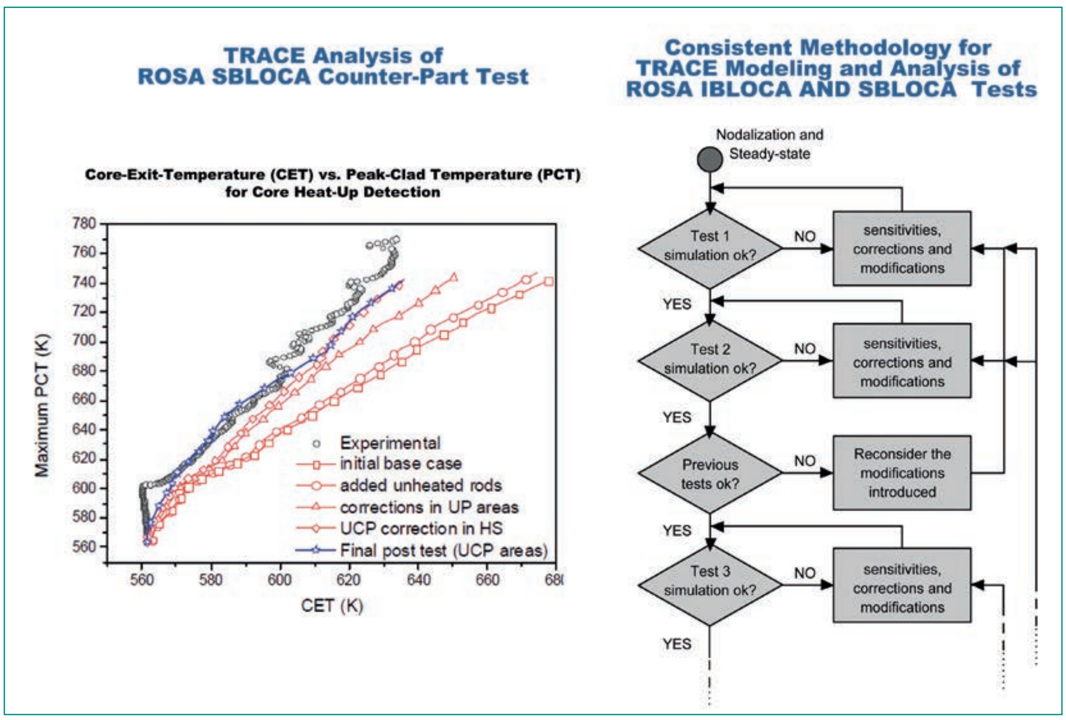
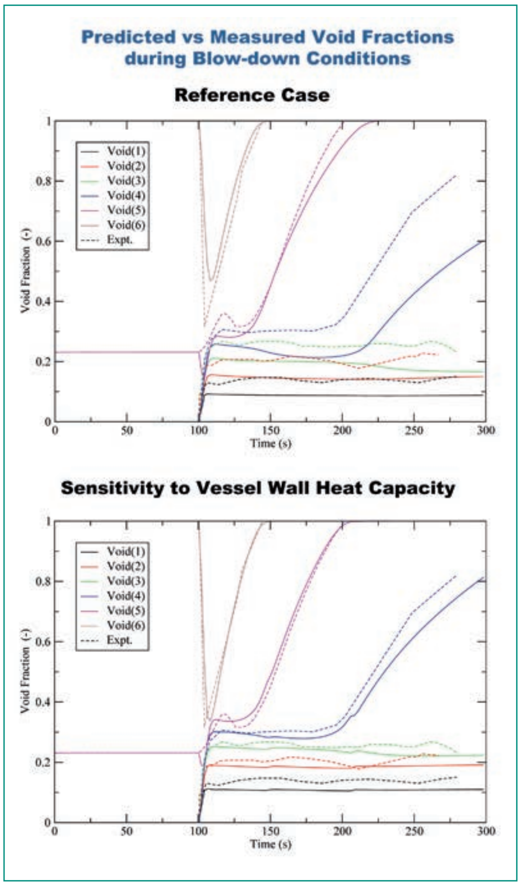


Figure 3:
Assessment of TRACE for SBLOCA Counterpart Tests at ROSA and PKL Integral Effect Test Facilities

Figure 4:
Figure 4: Analysis with TRACE of Level-Swell Tests



an attempt to understand this behaviour. The most relevant finding was that adding a heat structure to represent the vessel wall heat capacity and the initially stored energy would allow to better model heat transfer mechanisms during the transient and improve thereby significantly the void predictions (see lower plot of Fig. 4). Consequently, additional assessment has now been initiated to confirm the TRACE capabilities for other types of tests (e.g. larger vessels, faster depressurisation rates).

Sub-channel modelling and analyses

One objective of STARS is to gradually establish through a coupling with the 3-D core simulators, the capability for critical-heat-flux (CHF) predictions during normal operation as well as under transient/accident conditions. As first step in that direction, the project participated in recent years to the OECD/NRC PWR subchannel and bundle tests (PSBT) Benchmark aimed at assessing the capabilities of T-H codes to predict void distributions and departure from nucleate boiling (DNB) on the basis of experimental data measured in a full-scale prototypical PWR rod bundles at the NUPEC test facility ([9], [10], [11]). In this context, participation to the first phase of the benchmark on void predictions during steady-state and transient conditions for both single-channel and bundle geometries, was conducted with the TRACE system code, the FLICA-4 sub-channel code and the STAR-CD computational fluid dynamics (CFD) code. For the latter, emphasis was given to assess in collaboration with the code developers, the implementation of a new generation boiling model. An overview of the results obtained for single-channel tests is provided on the left hand side of Fig. 5. Principally, it was found that for all investigated geometries, no significant discrepancy against measured data was obtained when considering the cross-section averaged void fractions. However, a tendency for slight over-prediction in the low void fraction range could nevertheless be observed, pointing thus to the need for further enhancements of this new boiling model. Concerning Phase 2, aimed at steady-state and transient DNB predictions for bundle geometries, the analyses were carried out with the 3-D two-phase flow FLICA-4 code which is based on a 4-equation drift-flux approach combined with a turbulent mixing model. The obtained steady-state results are shown in the upper-right plot of Fig. 5, indicating thus an overall good performance of FLICA-4. A tendency for under-predicting the DNB power is nevertheless observed although this mainly indicates that the code ensures a certain level conservatism. In the same context, the various DNB correlations available in the code, including the W3 correlation and the Groeneveld «Look-up» table (GLT) were also assessed (see lower-right part of Fig. 5). This mainly showed that the W3 correlation a) would tend to produce higher DNB powers than the GLT approach; b) would be more limited in terms of range of applicability since some cases could not be analysed. But all in all, this benchmark has certainly provided a first confirmation that FLICA-4 could be a suitable candidate for DNB predictions of the Swiss PWRs although further validation remains necessary, specially in order to address the difficulty of the code to predict thermal mixing around spacers.

Lattice physics with Monte-Carlo codes

One principal purpose of lattice physics codes is to produce the few-group homogenised nuclear data, hereinafter referred to as XS data, for the downstream 3-D steady-state/transient core simulators. In that framework, the intention of STARS remains to use principally the deterministic 2-D lattice transport code CASMO-4 (C4) as well as its successor CASMO-5M (C5), not only for production calculations for the Swiss reactors but also for various types of design and optimization ([12], [13]) or method development/assessment [14] studies. However, Monte-Carlo (MC) stochastic transport codes are nowadays also being developed for this purpose. These codes usually rely on continuous-energy neutron data libraries and offer greater geometrical modelling flexibility, reducing thereby the need for simplifications and/or approximations inherent to deterministic methods. In that context, the SERPENT MC code, under development by the Finnish VTT research institute, was specifically designed for XS data preparation. Therefore, it was considered appropriate to introduce it in STARS as a potentially complementary tool to C4/C5 and a preliminary assessment of its capabilities for XS data preparation was initiated during 2012. To start, a comparison against C5 as well as MCNPX reference solutions was carried out for a PWR nuclear fuel lattice representative of designs employed in the Swiss reactors [15]. Without considering burnup at this stage, the results illustrated in the upper part of Fig. 6 show rather small differences both for crosssections as well as assembly-discontinuity-factors (ADFs). As can be further seen, the agreement only slightly deteriorates when reducing the number of neutron histories to ~ 1^{05}. This is important since a major drawback of MC codes for production calculations might indeed be prohibitive computational costs. Secondly, the assessment for reflector segments was also initiated with as central objective to also verify the performance of the methodology employed by deterministic code systems such as CASMO/SIMULATE and as function of reflector design. Therefore, conventional GII baffle/barrel (BB) reflector concepts as well as advanced GIII/III+ heavy reflector (HR) designs were included in the study. For the latter, opportunity was also taken to assess through SERPENT, the impact of using a homogeneous modelling approach (HR4), as necessary in C5M, versus a reference explicit heterogeneous representation (HRR). To summarise, it was found that both codes predict a similar reflector saving (around 4500 pcm) of GIII designs compared to GII concepts. Concerning the XS data, an agreement within 10% was obtained for both diffusion coefficients as well as ADFs although for the latter, a tendency for larger differences was observed for the GII design. For the cross-sections, the comparison is illustrated on the lower-part of Fig. 6 showing thus a good agreement for the GII design while a deterioration is seen for the GIII design. Specially, the fast-to-thermal removal cross-section is seen to be underpredicted by C5 and this effect becomes more pronounced if a comparison to an

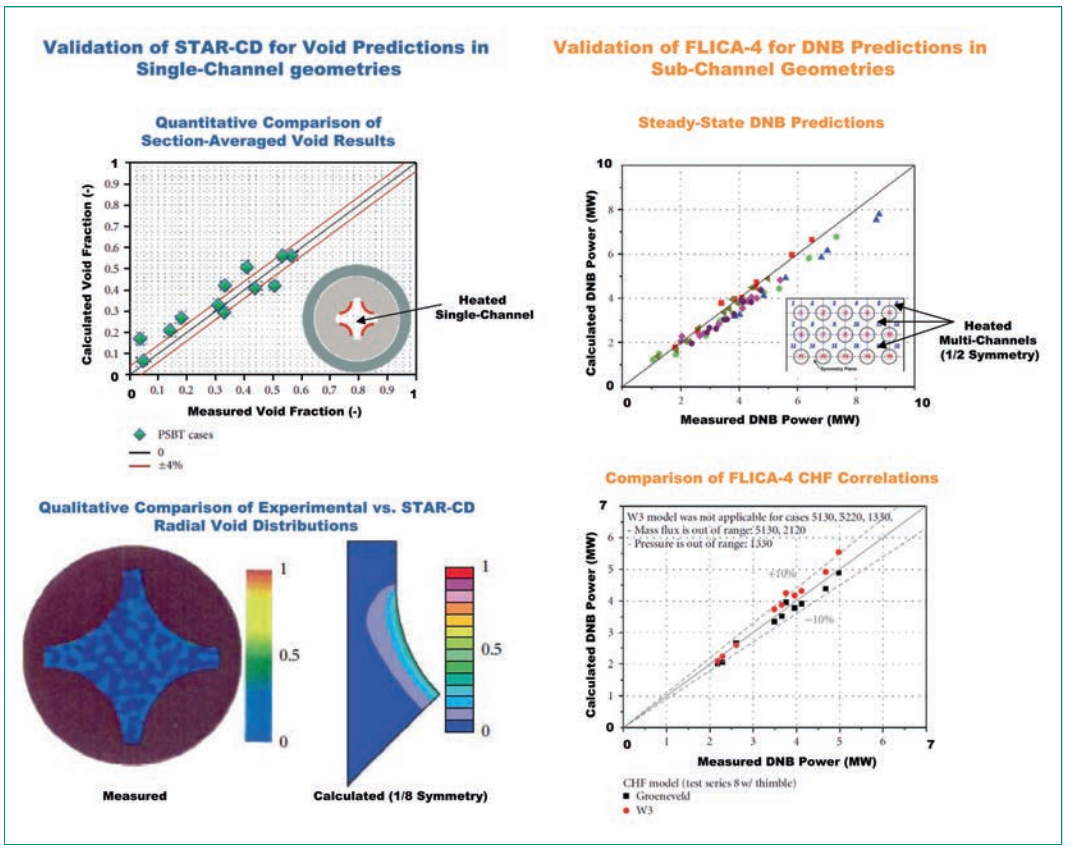


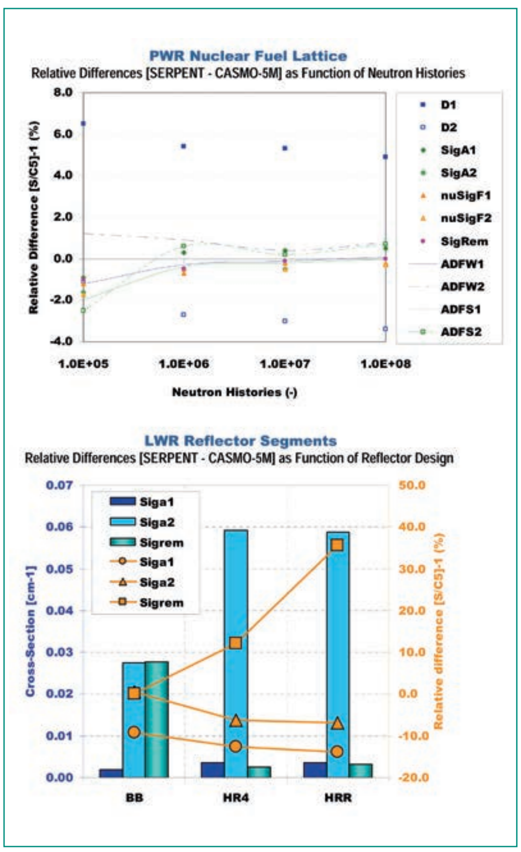
Figure 5:
Modeling and Analysis of OECD/NEA PWR PSBT Void and DNB Benchmark with CFD (STAR-CD) and Sub-Channel (FLICA-4) Codes

explicit heterogeneous model is made, indicating that for advanced GIII designs, the homogeneous approach used by C5M will tend to overpredict the backscattering of neutrons into the active core zone. Finally, it must be underlined that for all nuclear data and configurations, the statistical error of SERPENT would be less than 0.5% and would converge rapidly i.e. even if a much smaller number of histories than a finer reference case would be applied.

Validation of BWR stability methodology for out-of-phase oscillations

The new PSI BWR stability analysis methodology based on S3K and developed in recent years [1] has so far not been validated for regional instability i.e. for out-of-phase oscillations. This type of instability mode constitutes a greater safety concern since it can not be easily detected solely by global average neutron flux monitors and it is also known to be more challenging for system code simulations because of the more complex underlying dynamical behaviour. For this reason, it was considered appropriate to start during 2012, the validation of the S3K methodology for regional stability and to that aim, one of the KKL cycle 07 stability tests was selected. During this specific test, not only an out-of-phase oscillation mode but also an azimuthal rotation of the symmetry line, were observed. Hence, using the same methodology as validated so far for global stability, i.e. without any modifications neither to the computational route nor to the modelling options, the analysis of this test was carried out. All in all, it was found that S3K would predict rather well the dynamical reactor behaviour including the excitation of the out-of-phase mode with rotation and oscillation of the symmetry line. Furthermore, a detailed analysis of the predicted LPRM signals showed that the two out-of-phase oscillation modes, associated to the first and second azimuthal neutronic modes, were simultaneously excited with oscillations growing to a limit cycle with a ~ 40% amplitude around the steady-state value. In addition, the S3K results indicated that it is the superposition of these two modes that will make the symmetry line oscillate or rotate, depending on the dominance of one of these two modes. With only one dominant mode, the superposition of the two azimuthal modes will produce an oscillatory behaviour around the axis of the dominant mode. On the other hand, with a comparable strength between the two modes, a superposition of the two azimuthal modes will result in a rotational behaviour of the symmetry line with no favourite direction. A summary of these findings is given in Fig. 7. On the upper-left plot, the symmetry lines for the two azimuthal modes are illustrated while the upper-right plot shows the calculated neutron fluxes at selected LPRM locations for a 300 s time interval of the test. On the lower plots, snapshots of the 2-D power distribution and its evolution as function of the LPRM dominance ratios are shown. As can be seen, in a first stage, the symmetry line is located around the NE-SW direction but as the dominance ratio approaches one, a rotational pattern of the symmetry line gradually develops. Hence both qualitatively and quantitatively, the S3K methodology is able to reproduce the complex core behaviour that occurred during this test, noting however that the resonance frequency was found to be slightly over-predicted.

Figure 6: *Assessment of SERPENT for LWR Nuclear Data Preparation*



Establishment of complementary capability for 3-D reactor kinetics

Although S3K will remain the principal solver for either stand-alone coupled neutronics/thermal-hydraulics 3-D core simulations or as kinetic solver for coupled core/plant transient simulation with TRACE, it was considered adequate and useful to establish a complementary 3-D kinetics capability based on the PARCS code. Among other things,

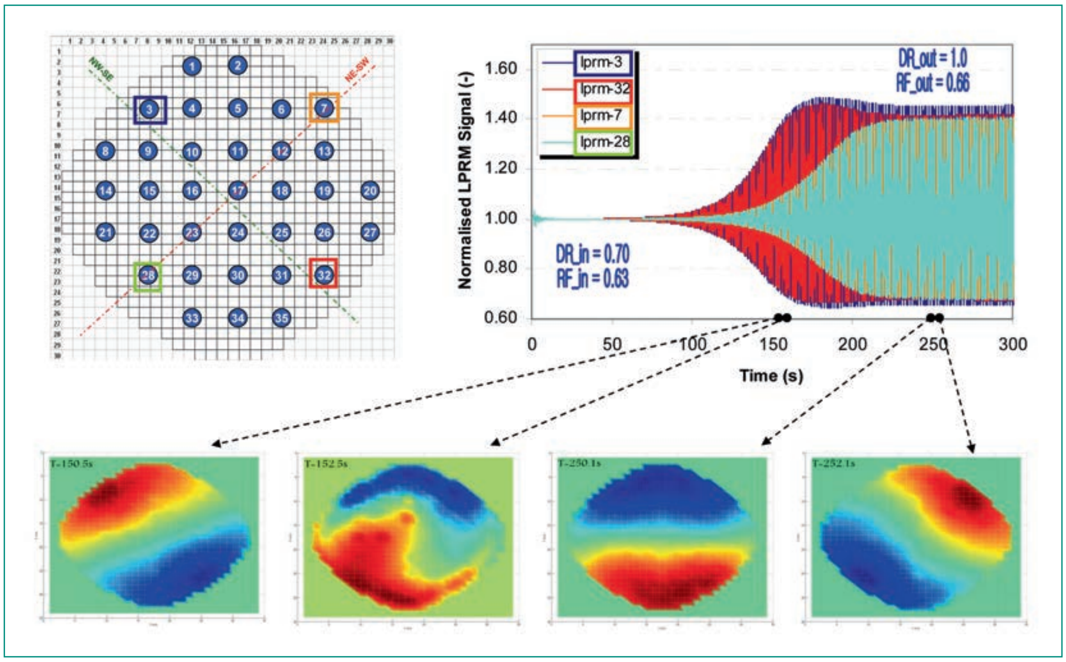
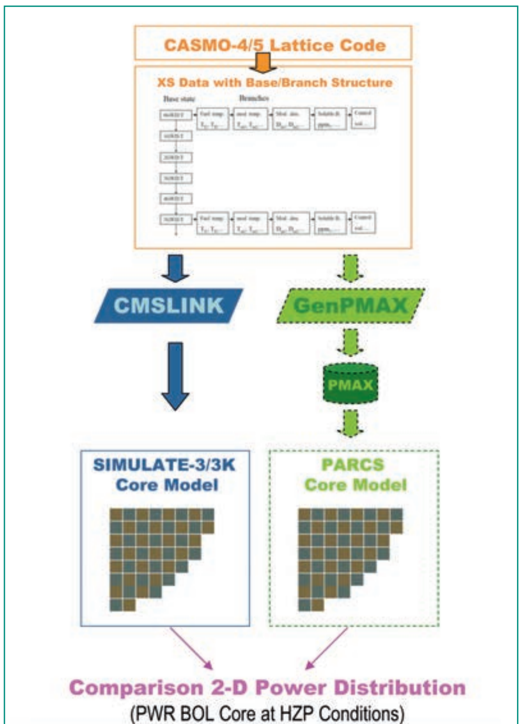


Figure 7:
Figure 7: Modeling and Analysis with SIMULATE-3K of KKL Cycle 07 Regional Stability Test

PARCS is currently the standard kinetic solver available in TRACE and also includes different neutronic models compared to S3K (e.g. multi-group diffusion) providing thereby the opportunity to benchmark, when considered as appropriate, S3K or coupled TRACE/S3K results. For this reason, a semester/master project was initiated during 2012 with as first objective to develop a methodology to initialise the PARCS core models using as basis, the same few-group XS data produced by the upstream CASMO-4/5 transport calculations [16]. In that framework, the GenPMAX module designed for the transfer to PARCS of XS data produced by lattice codes such as e.g. CASMO, was implemented and tested. To start, a verification of the compatibility between the PMAX library produced by GenPMAX and the CASMO results was conducted. This was confirmed to be the case when using a simplified structure for the XS data matrix produced by CASMO via base depletion and branch calculations. However, when applying the standard base/branch structure used for the Swiss core models and referred to as the S3C matrix, several deficiencies were identified. For instance, very large reactivity errors (around 2000 pcm) were encountered for certain combinations of state variables. This was found to be caused by the inability of GenPMAX to map the XS data when more than two instantaneous variables are simultaneously perturbed, something that will constitute a limitation of PARCS when simulating transients with non-negligible effects from coupled feedback terms. Large discrepancies were also found when the Xe/Sm contribution to the absorption cross-section is treated separately. A third limitation is that the number of burnup steps in branch calculations must be the same for different histories, something that is not necessarily the case with the S3C structure. All of these points were therefore notified to the code developers which have initiated efforts to enhance GenPMAX for applications with standard CASMO LWR calculations, something that was hence not the case up to now. In parallel to this, it was nevertheless considered necessary to verify the PARCS results when using as basis, at least simplified XS data structures. To that aim and selecting a PWR core model as starting point, simplified CASMO models were thus used to initialise both a SIMULATE-3 and a PARCS 3-D core model. At this stage, only comparisons for a BOL fresh-fuel core at hot-zero-power (HZP) were made in order to eliminate the impact from burnup as well as well as from thermal-hydraulics. The results, illustrated on the lower part of Fig. 8, show a rather good agreement in terms of 2-D radial power distribution except at the core periphery where different reflector methodologies might however be the primary cause. But overall, this level of agreement provides confidence that the GenPMAX/PARCS methodology should also produce correct results for more complex configurations once the above mentioned updates have been correctly implemented. Before that, further assessment will be necessary specially for BWR analyses.

Modelling and analysis of fuel rod behaviour during LOCA

A central activity of STARS with regards to fuel behavior is to improve the modeling of complex thermo-mechanical phenomena for base irradiation as well as for major types of transient/accidents such as Reactivity-Initiated-Accidents [17] or LOCAs and involving primarily UO2 but also other types of fuel design such as MOX ([18], [19]). Specially for enhanced LOCA fuel safety criteria, the project has been and continues to be highly active in experimental tests such as those conducted at the JAEA NSSR facility [20] but also and specially at the Halden reactor. For the latter, activities in recent years were focused around the design and analyses, using FALCON coupled to GRSW-A, of two LOCA tests with high burnup (HBU) fuel samples from the KKL reactor. The first test, namely IFA-650.12, was successfully conducted following the PSI design specifications but a clad rupture caused by an unforeseen level of rod internal pressure occurred during the cool down test phase. Hence, during 2012, a major activity was to investigate the reasons for this behavior and on that basis, to attempt drawing the necessary lessons for the design of the second test, IFA-650.13, which was also completed during the year. The investigations were focused on assessing the hypothesis of a-thermal fission gas release (FGR) from HBU fuel during the test as this could constitute a safety-significant finding if experimentally confirmed and theoretically justified. First, an assumption was made that the measured gas pressure could perhaps be explained by a high fraction of the free rod volume distribution along the high-temperature fuel stack. However, this assumption was ruled out based on a FALCON calculation showing that such level of pressure could not be reached even under the most conservative assumption, namely when assuming that the entire free volume would solely consist of the voids within the active fuel stack. Secondly, an alternative method was designed and used to estimate the FGR during the test in question. This method can be referred to as «quasi-puncturing» because the principle is the same as in corresponding experimental techniques widely used in hot laboratories. Basically, the method consists in using available data for gas pressure and free volume at two time points, namely at the beginning of the heat-up phase and just before the cladding rupture. The free volume at the latter time-point was inferred from the measured cladding deformation after the test and additional assumptions were made regarding the probable gas temperature distribution in the free volume based on available thermometry data. This alternative method resulted in an estimated FGR amount quite close to the one obtained from a FALCON-based numerical fitting of the calculated pressure dynamics to the measured one, assuming different values of the a-thermal FGR in the rod. In fact, both methods converged to a FGR quantity amounting to around 60 cc@STP. As next step, a base irradiation of the full-scale mother rod used for refabrication of the tested sample was carried out with FALCON coupled with the GRSW-A model. The objective was to evaluate the distribution of the retained gas in the fuel sections subject to the LOCA tests. Specifically, the gas retained by the large inter-granular pores formed in the pellet centre and rim during the base irradiation as well as the gas released by the HBS in the pellet rim but trapped in the closed gap due to the tight pellet-cladding bonding, were considered as likely contributors to FGR during the



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7	1,826	2,113	1,877	1,447	0,982	0,667	0,282	G-
	1,829	2,091	1,865	1,442	0,984	0,670	0,297	
	-0,2%	1,1%	0,7%	0,4%	-0,2%	-0,4%	-5,4%	
8	2,113	1,945	1,849	1,464	0,995	0,568	0,243	H-
	2,091	1,937	1,835	1,447	0,992	0,576	0,263	
	1,1%	0,4%	0,8%	1,1%	0,3%	-1,4%	-8,2%	
9	1,877	1,849	1,709	1,248	0,885	0,487	I-	
	1,865	1,835	1,694	1,242	0,882	0,502		
	0,7%	0,8%	0,9%	0,5%	0,4%	-3,1%		
10	1,447	1,464	1,248	1,054	0,635	0,305	J-	
	1,442	1,447	1,242	1,043	0,637	0,323		
	0,4%	1,1%	0,5%	1,1%	-0,3%	-6,0%		
11	0,982	0,995	0,885	0,635	0,364	K-		
	0,984	0,992	0,882	0,637	0,373			
	-0,2%	0,3%	0,4%	-0,3%	-2,4%			
12	0,667	0,568	0,487	0,305	L-			
	0,670	0,576	0,502	0,323				
	-0,4%	-1,4%	-3,1%	-6,0%				
13	0,282	0,243	M-					
	0,297	0,263						
	-5,4%	-8,2%						

Figure 8: *Establishment and Verification of Methodology for PARCS Core Modeling of the Swiss Reactors*

LOCA tests. The values of these parameters calculated for both tested sections after base irradiation, are presented in Fig. 9. As can be seen in that figure, the estimated total quantity of retained gas available for transient a-thermal FGR was found to be comparable to the filling gas for the IFA-650.12 test. Hence, its release could have been the cause for the drastic up-swing of the measured pressure during the test. Obviously, the impact of such phenomenon on fuel behavior during LOCAs might thus be significant. Therefore, the extension of models available for predictive analyses of effects from pellet-cladding bonding, fragmentation and transient a-thermal FGR, was recently proposed as a PhD project in order to complement further the design and analyses of Halden LOCA tests.

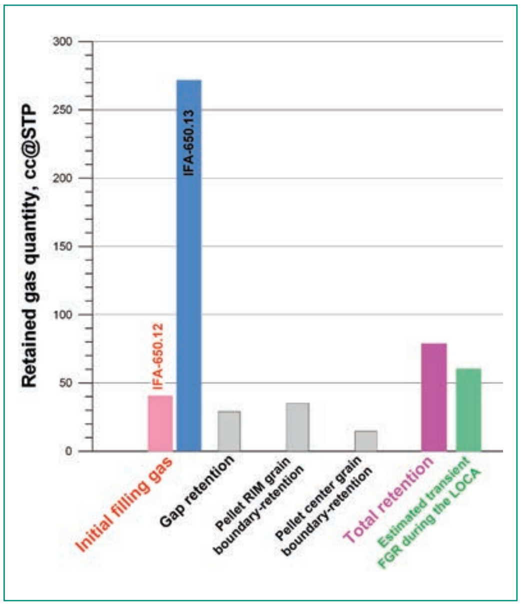

Figure 9:
Calculated Characteristics of Gas Retention in Segment C of KKL Fuel Rod AEB072-E4 versus Filling Gas Quantity in Re-Fabricated Rods used for the Halden LOCA Tests

Assessment of fuel performance codes for PCI/PCMI failures

During reactor start-up and/or operation manoeuvres involving power ramps, nuclear fuel rods are vulnerable to failures from Pellet-Clad-Mechanical-Interactions (PCMI) caused by pellet expansion combined with cladding stresses. Three main failure modes can take place, namely Pellet-Clad-Interaction (PCI) resulting from Stress-Corrosion-Cracking (SCC), Hydrogen embrittlement and delayed-Hydrogen-Cracking. Therefore, the OECD/NEA SCIP-II program was recently launched to address among other things, PCI/PCMI failure mechanisms during various types of power ramps (e.g. single ramps as well as stepwise ramps with stress-relaxation from power holding). The STARS project is participating to this program with the objective to enhance FALCON for PCI/PCMI modelling and specially to assess the code's capabilities for clad failure predictions. During 2012, analyses with FALCON coupled to GRSW-A were thus conducted for 8 BWR rods subject to various types of ramps (see upper table of Fig. 10). On this basis, a validation of the capabilities to predict residual strains after the ramps was first carried out. This is illustrated in the lower-left plot of Fig. 10 where the FALCON results are compared to experimental data for one of the BWR rod (xM3). Here, one can note the significant improvement obtained

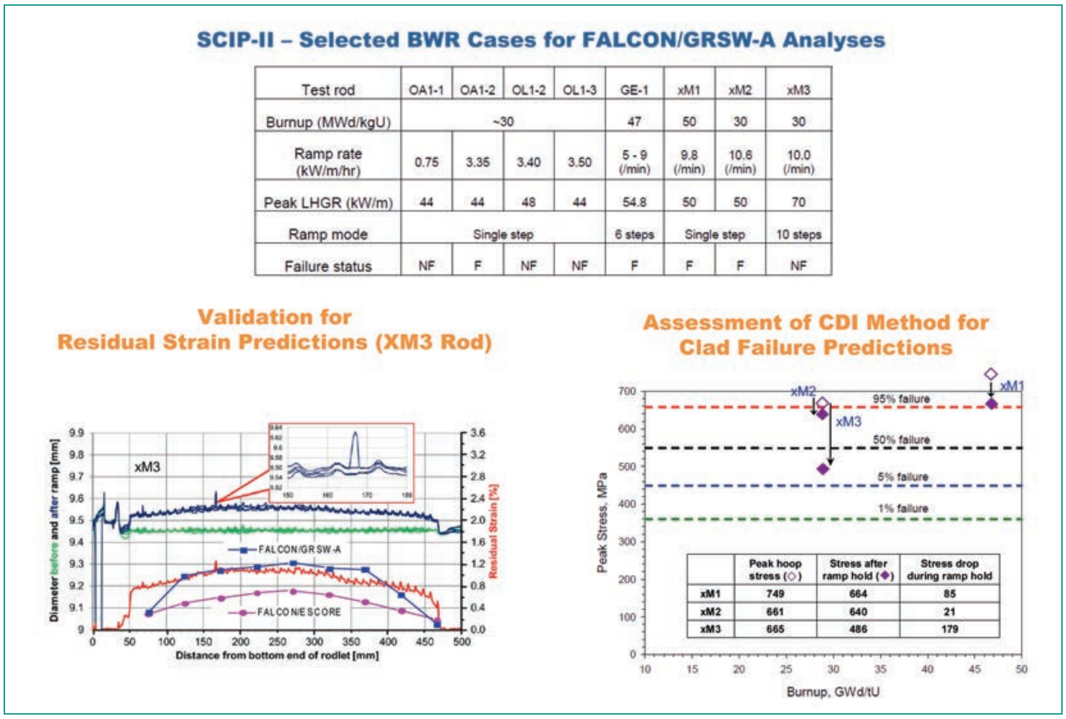

Figure 10:
Modeling and Analysis with FALCON/GRSW-A of SCIP-II BWR Ramps (Results for xM Rod Series)

with the GRSW-A model compared to the nominal FALCON ESCORE approach. Thereafter, the Cumulative-Damage-Index (CDI) method available in FALCON and established by EPRI to estimate failure probability as function of peak hoop stress was applied. As can be seen in the lower-right plot of Fig. 10 showing the obtained results for the xM rod series, the CDI method predicts a very high failure probability for all three rods while this was not the case experimentally since only the xM1 and xM2 rods failed. This indicates that the CDI method did not capture the correct failure probability pattern between the various rods. When considering that the three rods were subject to different types of ramps, an attempt was made to include the clad relaxation from power holding on the CDI predictions. More precisely, by taking into account the stress drop during ramp hold instead of solely applying the peak hoop stress in the CDI correlation, an improved and correct discrimination of failure versus non-failure was achieved with a reduction of the failure probability for the xM3 rod to around 5% (see lower-right plot of Fig. 10). Although this finding must be considered with care since this implies a deviation from the nominal range of application of the CDI method, it certainly illustrates that the latter might not account for all relevant effects from the ramp dynamics when estimating failure probabilities. For this reason, these validation efforts are now being continued using not only SCIP-II data but also other experimental tests available from e.g. the IAEA FUMEX program [21] in order to understand better the capabilities of the CDI method and to indentify eventual areas of enhancements for applicability to a wider range of power ramps.

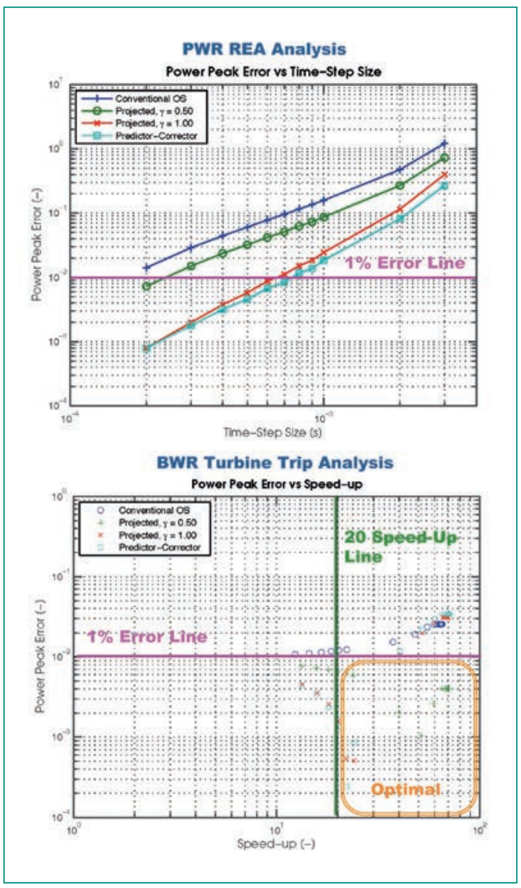

Figure 11: *Evaluation of Enhanced Temporal Coupling Schemes for TRACE/S3K Analyses*

Multi-physics and coupling methodologies

One central mission of STARS is to develop multi-scale and multi-physics computational methodologies to improve the space-time resolution of models applied for transient/accident safety evaluations. During 2012, emphasis was given to develop and assess enhanced temporal coupling schemes for TRACE/S3K (TS3K) simulations ([22], [23]) and as part of the participation to the EU NURISP project ([24], [25]). For TS3K, the particular intention was to establish complementary schemes that would allow for more efficient simulations when large detailed core models are necessary or when very long transients are to be analysed. The nominal TS3K temporal scheme is indeed based on an explicit operator-splitting (OS) method which requires very small time-step sizes in order to ensure numerical stability and convergence of the coupled simulation. To overcome this, two enhanced temporal coupling schemes were established, namely 1) a time-projected power (TPP) method; 2) a predictor-corrector approach to advance the thermal-hydraulic solution (PCTH). For both schemes, the objective was to achieve a given accuracy target using larger time-step sizes than with the explicit OS method. Also, an adaptive time-step algorithm (ATS) applicable to all coupling schemes was developed. The ATS is based on adapting the time-step sizes by tracking the fastest dynamical scales of the main physical variables (e.g. power, T-H feedback quantities) and to optimise thereby, the trade-off between accuracy and CPU cost. To assess these developments, both a neutronic driven transient such as the PWR Rod-Ejection-Accident (REA) and a T-H driven transient such as the BWR turbine trip, were analysed. For the REA case, some of the achieved results are illustrated on the top part of Fig. 11. There, the power peak error as function of time-step size is compared between the TPP approach (using two variants of the weighting factor γ applied to the temporal projection of the power, the PCTH scheme and the conventional OS method. Clearly,

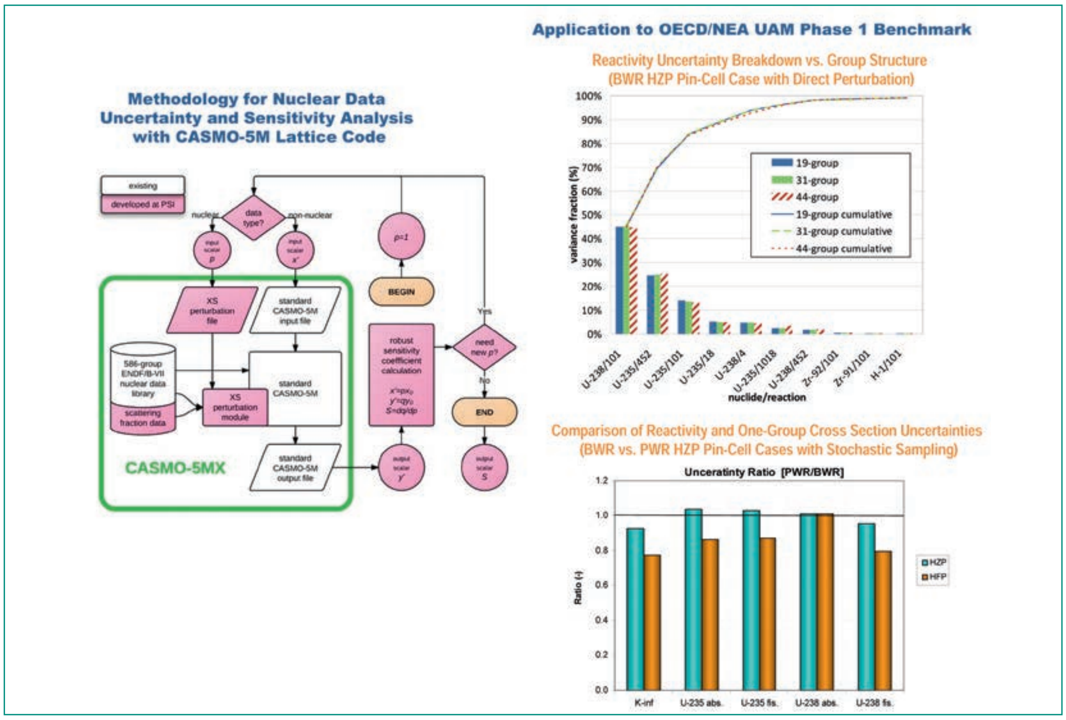

Figure 12:
Development and Assessment of Nuclear Data UQ Methodology for CASMO-5M Code

one can observe that the time-step needed to achieve an error below 1% by the PCTH is more than three times smaller than with the OS. Regarding the turbine trip transient, some of the results obtained with the various schemes and with implementation of the ATS, are illustrated in the lower part of Fig. 11. There, the relation between power peak error and computational speed-up (achieved through a variation of a «safety factor» in the ATS that limits the maximum fractional change of any state variable between two consecutive time-steps) is shown along with the 1% error and 20 times speed-up lines, noting that the lower right quadrant constitutes thus the optimal region. A first observation is that even with the slowest speed-up, the conventional OS method fails in reducing the power peak error below 1%. In contrast, with the ATS applied to the TPP or PTCH schemes, a speed-up factor of 20 can be achieved while maintaining a high accuracy. Therefore, when combined together, the new schemes and the ATS allow reducing the error to levels where numerical diffusion becomes non-negligible, if not predominant. However, along that these new approaches will be assessed for other types of transients, further studies will also be needed to address, among other things, if an optimal safety-factor for the ATS algorithm can be estimated. Also, studies will be conducted regarding the development of alternative coupling approaches and/or parallelization capabilities.

Uncertainty analysis

As STARS aims at developing safety-related computational methodologies using best-estimate codes, one important complementary activity is to establish advanced and reliable methods to quantify the uncertainties associated with the simulation results. In that framework, a capability for uncertainty quantification (UQ) with respect to nuclear data was in recent years developed for the CASMO-5M lattice transport code (see left part of Fig. 12). The guiding principle was to implement two complementary non-intrusive UQ techniques in a special PSI code version referred to as CASMO-5MX: 1) direct perturbation (DP) and 2) stochastic sampling (SS). During 2012, a benchmarking of these techniques for UQ related to cross-sections was completed through participation to the OECD/NEA UAM Phase 1 benchmark. More precisely, solutions for BWR and PWR pin-cell as well as assembly models at both HZP and hot-full-power (HFP) Beginning-of-Life (BOL) conditions were finalised. Among other things, uncertainty estimations of the reactivity (k-inf) as well as 1-group collapsed cross-sections for various nuclides and reactions were estimated with both the DP and the SS techniques, indicating a close agreement between both methods as well as to independent solutions provided by other participants. Also, a breakdown of the k-inf uncertainty allowed to assess the most influential parameters and to study in that context, the impact of the perturbation group structure. This is illustrated in the upper-right part of Fig. 12 showing for the BWR HZP pin-cell case, a ranking of the variance contributions as well as the cumulative variance aimed at delimiting the most influential parameters (e.g. representing more than 99% of the total k-inf uncertainty). Principally, for UO2 fuel, it could thus be observed that the most important contributors to k-inf uncertainty are ^{238}U capture, ^{235}U neutrons per fission and ^{235}U capture. And similarly, uncertainty breakdown showed that ^{238}U inelastic scattering would account for well over 50% of the 1-group ^{238}U absorption cross-section. Based on these types of investigations, the estimated uncertainties could also be compared between different fuel designs (e.g. UO_2 vs. MOX) or LWR type (BWR vs. PWR) as illustrated in the lower-right part of Fig. 12 where for a given quantity, the ratio between the PWR versus the BWR uncertainty is shown. As can be seen, the BWR k-inf uncertainty tends to be larger, especially at HFP conditions when spectrum hardening increases substantially the contribution from ^{238}U fission cross-section uncertainties. As next step, UQ for fuel depletion will be aimed at, including thus propagation of fission yield and decay data uncertainties. As well, the relative effect of nuclear data versus design/geometrical uncertainties on depletion uncertainties will in that framework also be studied.

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 laboratories at PSI, among which the Laboratory for Thermal-Hydraulics (LTH), the Laboratory for Energy Systems Analysis (LEA) and the Laboratory for Nuclear Materials (LNM) can be mentioned. Finally, the project is also collaborating with the Swiss federal polytechnic institutes ETHZ/EPFL for the elaboration and supervision of relevant MSc and/or PhD theses as well as for the realisation of courses for the Nuclear Engineering Master Program including the «Nuclear Computation Laboratory» course on reactor simulations.

International Cooperation

At the international level, the project collaborates with international organisations (OECD/NEA, IAEA) principally as part of working/expert groups as well as through international research programs and benchmarks. The project also collaborates with other research organisations, on the one hand through e.g. EU 7th FP NURISP project and on the other hand, through bilateral cooperation e.g. GRS, CEA, KTH, Michigan University. An active cooperation with the Finnish regulatory body STUK as well as with the AREVA plant vendor is also carried out for safety evaluations related to the GIII/GIII+ EPR and Kerena reactors respectively. Finally, close cooperation with code developers and/or providers is necessary and conducted principally with US NRC (TRACE), Studsvik Scandpower (CASMO/SIMULATE-3/SIMULATE-3K) and EPRI/ANATECH (FALCON).

Assessment 2012 and Perspectives for 2013

During 2012, despite a shift in priorities due to changes in resource situation, several key objectives could be achieved. Among other things, scientific support to ENSI could be provided and on the research side, progress was reached in many activities with two main characteristics. On the one hand, participation to several OECD/NEA benchmark programs and/or benchmark phases was finalised. On the other hand, along the development and validation of methodologies using reference codes for the Swiss reactors, efforts were initiated to integrate complementary computation methods in order to further strengthen the project capabilities for comprehensive state-of-the-art steady-state and safety analyses. Some deviations to the initially planned targets were nevertheless encountered. For instance, the development of uncertainty quantification methods related to physical models of thermal-hydraulic system codes did not materialize in concrete progress due to the lack of resources. Also, an assessment of the BWR core stability analysis methodology for the OECD/NEA Oskarshamn stability benchmark was not started because of a delay by the organizers to provide final specifications. Finally, the validation of fuel performance codes for the modelling of clad lift-off phenomena at high-burnup has not yet been launched because higher priority was given to the interpretation and design of LOCA experimental tests. All these activities remain key targets for the next phase of the STARS collaboration with ENSI which will start in 2013 and for which the scientific support and research programs are currently being finalised. And in line with these programs, the main objectives for 2013 are as follows.

Plant System and Thermal-Hydraulics	Modelling with TRACE of KKL Fast Run-Up of new Recirculation Pumps
	Updates of KKG TRACE Model for SGTR Accident Analyses
	Analysis of OECD/NEA PREMIUM Phase-3 with TRACE plus CIRCE UQ Methodology
	Validation of STAR-CCM+ for PKL-2/PKL-3 Rocom Tests
	Development and Testing of STAR-CCM+ CFD Mesh for KKG Vessel
Core Behavior and Reactor Physics	Support to Licensing of new KKL Core Loading
	Development and Testing of CMSYS/FLICA Methodology for PWR DNBR Calculations
	Assessment of Hybrid Monte-Carlo/Deterministic Scheme for Enhanced LWR Reflector Modelling
	Validation of S3K for OECD/NEA Oskarshamn Stability Benchmark
	Development of Methodology for Nuclear Data Uncertainty Propagation in CASMO-5M Depletion Calculation
Fuel Modelling and Thermo-Mechanics	Validation of FALCON for Halden LOCA Test 2 and Design of Test 3
	Completion of FALCON Assessment for Modelling of Cladding Lift-Off at High Burnup
	Development and Validation of Reference Methodology for Base Irradiation of Swiss Fuel Rod Designs
	Continued Validation of FALCON for PCI/PCMI Fuel Rod Failures
Multi-Physics	Consolidated Verification of COBALT Methodology for TRACE/S3K Analyses
	Enhancements of TRACE/S3K Coupling Scheme for Heterogeneous Feedback Distributions
	Participation to OECD/NEA UAM Phase 2 for Fuel performance, Assembly Depletion and Bundle Thermal-Hydraulics

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